October 27, 1993

Docket Nos. 50-266
and 50-301

DISTRIBUTION

Docket File	GHill(4)
NRC & Local PDRs	SVAthavale
PD3-3 Reading	CGrimes
JRoe	ACRS(10)
JZwolinski	OPA
JHannon	OC/LFDCB
MRushbrook	DHagan
ATGody, Jr.	EGreenman,RIII
OGC-OWF	

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

Dear Mr. Link:

SUBJECT: AMENDMENT NOS. 142 AND 146TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 (TAC NOS. M86783 AND M86784)

The Commission has issued the enclosed Amendment Nos. 142 and 146 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 June 11, 1993, and the supplemental information provided in your letter dated October 19, 1993.

The amendments modify Technical Specification Section 15.3.1.G, "Operation Limitations," Specification 3, to reduce the reactor coolant system raw measured total flow rate limit by 2,600 gallons per minute (gpm), change the overtemperature and overpower setpoints, and change the Reactor Core Safety Limits for Unit 2.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by Allen G. Hansen

for Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 142 to DPR-24
2. Amendment No. 146 to DPR-27
3. Safety Evaluation

cc w/enclosures:
See next page

LA:PDIII-3:DRPW	PM:PDIII-3:DRPW	D:PDIII-3:DRPW	OGC-OWF
MRushbrook	ATGody, Jr.	JHannon	EHoller
10/22/93	10/25/93	10/27/93	10/25/93

290042

DF01

Mr. Robert E. Link
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

Ernest L. Blake, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mr. Gregory J. Maxfield, Manager
Point Beach Nuclear Plant
Wisconsin Electric Power Company
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Chairman
Public Service Commission
of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
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. 142
License No. DPR-24

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

   A. The applications for amendment by Wisconsin Electric Power Company (the licensee) dated June 11, 1993, as supplemented by letter dated October 19, 1993, 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 applications, 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. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 142, 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 upon issuance. The Technical Specifications are to be implemented within 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Allen G. Hansen for

Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of issuance: October 27, 1993

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
License No. DPR-27

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

   A. The applications for amendment by Wisconsin Electric Power Company (the licensee) dated June 11, 1993, as supplemented by letter dated October 19, 1993, 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 applications, 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. 146 , 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 upon issuance. The Technical Specifications are to be implemented within 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of issuance: October 27, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 142 AND 146

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
15.2.1-1	15.2.1-1
15.2.1-2	15.2.1-2
Figure 15.2.1-1	Figure 15.2.1-1
--------	Figure 15.2.1-2
15.2.3-2	15.2.3-2
15.2.3-3	15.2.3-3
15.2.3-6	15.2.3-6
15.2.3-7	15.2.3-7
15.3.1-19	15.3.1-19

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

Basis:

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant temperature and pressure have been related to DNB.

This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability at a 95 percent confidence level that DNB will not occur during steady state operation, normal operational transients, and anticipated transients and is an appropriate margin to DNB for all operating conditions.

The curves of Figure 15.2.1-1 and 15.2.1-2 are applicable for a core of 14 x 14 OFA. The curves also apply to the reinsertion of previously-depleted 14 x 14 standard fuel assemblies into an OFA core. The use of these assemblies is justified by a cycle-specific reload analysis. The WRB-1 correlation is used to generate these curves. Uncertainties in plant parameters and DNB correlation predictions are statistically convoluted to obtain a DNBR uncertainty factor. This DNBR uncertainty factor establishes a value of design limit DNBR. This value of design limit DNBR is shown to be met in plant safety analyses, using values of input parameters considered at their nominal values.

Figure 15.2.1-1
REACTOR CORE SAFETY LIMITS
POINT BEACH UNIT 1

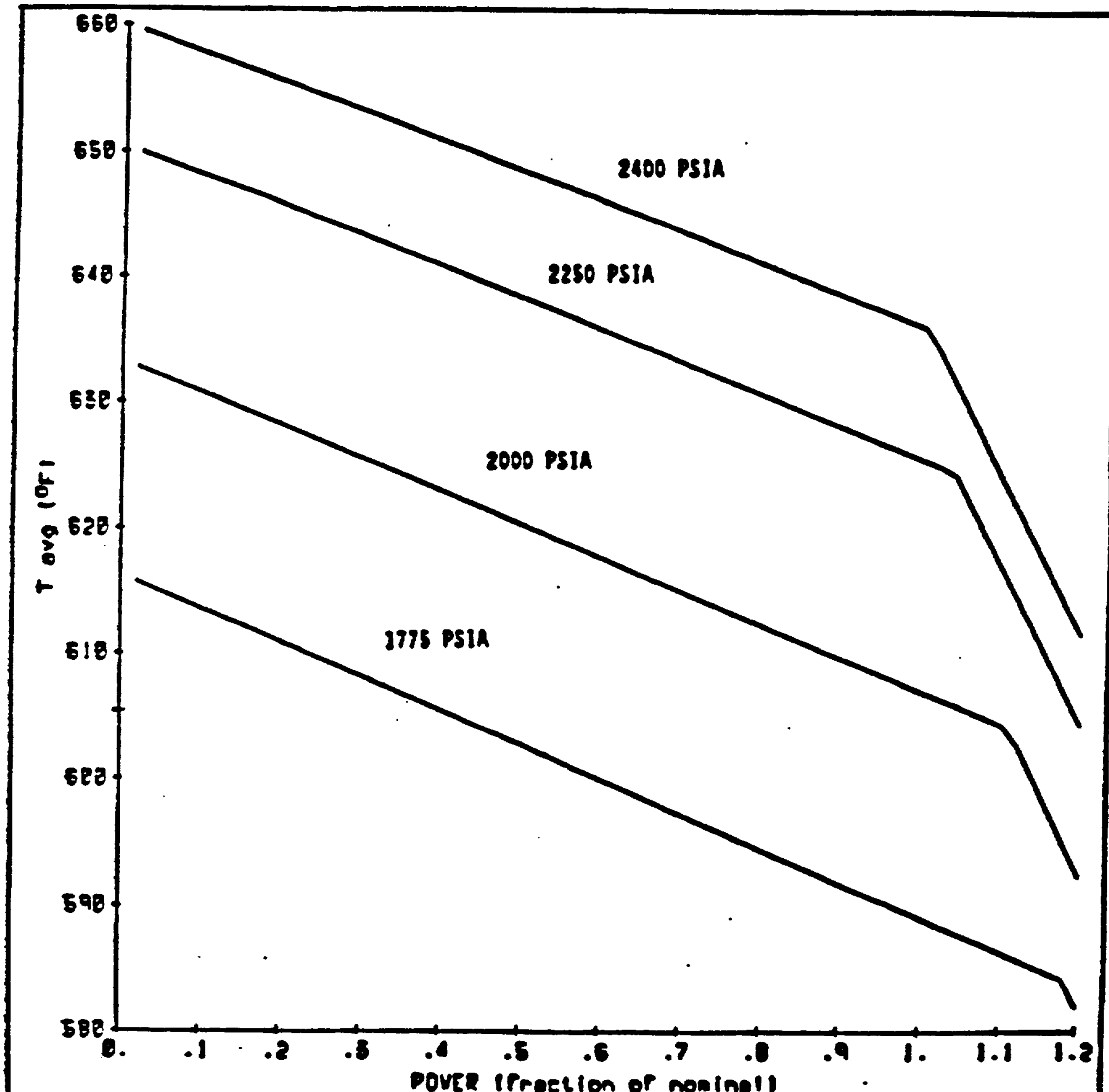
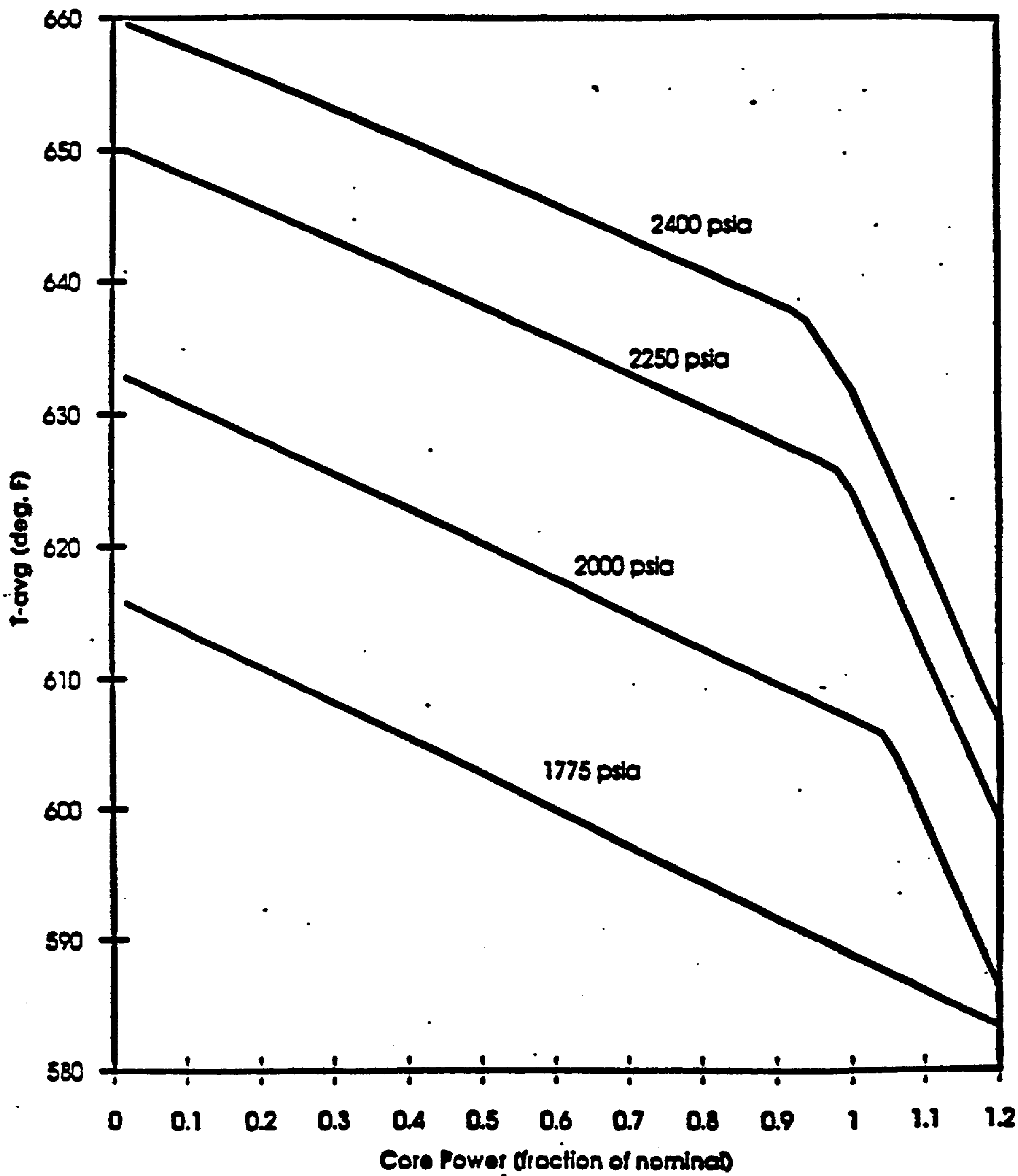


Figure 15.2.1-2
REACTOR CORE SAFETY LIMITS
POINT BEACH UNIT 2



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

(4) Overtemperature

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

$$\leq \Delta T_0\ (K_1 - K_2 (T(\frac{1}{1+\tau_4 S}) - T')(\frac{1+\tau_1 S}{1+\tau_2 S} + K_3 (P-P') - f(\Delta I))$$

where

ΔT_0	=	indicated ΔT at rated power, ^0F
T	=	average temperature, ^0F
T'	$\leq$	573.9^{0}F (Unit 1)
T'	$\leq$	570.0^{0}F (Unit 2)
P	=	pressurizer pressure, psig
P'	=	2235 psig
K_1	$\leq$	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 $f(\Delta I)$ 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, $f(\Delta I) = 0$.

(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.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

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

$$\leq \Delta T_0\left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

where

ΔT_0	=	indicated ΔT at rated power, ^oF
T	=	average temperature, ^oF
T'	$\leq$	573.9^{o}F (Unit 1)
T'	$\leq$	570.0^{o}F (Unit 2)
K_4	$\leq$	1.089 of rated power
K_5	=	0.0262 for increasing T
	=	0.0 for decreasing T
K_6	=	0.00123 for $T \geq T'$
	=	0.0 for $T < T'$
τ_5	=	10 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

(6) Undervoltage - ≥75 percent of normal voltage

(7) Indicated reactor coolant flow per loop - ≥90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point ≥55.0 HZ

(b) Low voltage set point ≥75 percent of normal voltage.

power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced[6][7].

The overpower, overtemperature and pressurizer pressure system setpoints include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints will not exceed the core safety limits as shown in Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident[4].

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis[8]. The low loop flow signal is caused by a condition of less than 90 percent flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency setpoint, 55.0 HZ, which assures a trip signal before the pump inertia is reduced to an unacceptable value. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified setpoint allows adequate operating instrument error[2] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[9]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed. Specifications 15.2.3.2.A(1) and 15.2.3.2.C have ±1% tolerance to allow for a 2% deadband of the P10 bistable which is used to set the limit of both items. The difference between the nominal and maximum allowed value (or minimum allowed value) is to account for "as measured" rack drift effects.

Sustained operation with only one pump will not be permitted above 3.5 percent power. If a pump is lost while operating between 3.5 percent and 50 percent power, an orderly and immediate reduction in power level to below 3.5 percent is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.

References

(1) FSAR 14.1.1
(2) FSAR, Page 14-5
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.3
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11

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:
   ≥2205 psig during operation at 2250 psia, or
   ≥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 ≥ 179,200 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 179,200 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (175,400 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.

Unit 1 - Amendment No. ~~44~~,~~87~~,~~86~~,~~120~~, 142
Unit 2 - Amendment No. ~~49~~,~~90~~,~~123~~,146

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 142 AND 146 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 June 11, 1993 (Reference 1), the 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 (PBNP), Units 1 and 2, respectively. The amendment proposed modification to Technical Specifications (TS), Sections 15.3.1.G.3 and 15.2.3.1.B(4) and (5). The proposed TS revision will reduce the Reactor Coolant System (RCS) total flow rate by 2600 gallons per minute (gpm), and change the T′ temperature limit associated with the overtemperature and overpower delta-T setpoint functions, from 573.9 °F to 570 °F in Unit 2.

As part of the justification to support the decrease in RCS flow rate limit, the licensee's submittal included a reference to a justification for continued operation (JCO) regarding the structural integrity of systems and components for operation of Point Beach at a reduced RCS T_{avg} of 570 °F. The licensee concluded in their submittal that the reduced RCS flow rate and reduced T′ setpoint as they relate to changes in primary coolant system temperatures do not significantly affect the structural integrity of the reactor coolant pressure boundary.

In response to questions by the NRC, the JCO was provided by the licensee. Conference calls, between WEPCO, Westinghouse and NRC, were held on October 6 and October 7, 1993, regarding questions on the JCO. On October 15, 1993, a meeting was held at the NRC headquarters, where WEPCO and Westinghouse presented their response to the staff's questions in a draft revision of the JCO. The details of the meeting are documented in a meeting summary dated October 21, 1993. By letter dated October 19, 1993 (Reference 2), WEPCO submitted clarifying information regarding the proposed amendment, including a copy of the revised JCO.

2.0 BACKGROUND

2.1 Current License Condition

The current license condition as stated in the TSs is applicable for both Units 1 and 2 as follows:

(1) TS 15.2.2, "Safety Limits, Reactor Core" specifies the reactor core safety limits that are used to maintain the integrity of the fuel cladding. The specification states that the combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits in Figure 15.2.1-1.

(2) TS 15.2.3.1.B(4), Section 15.2.3, "Limiting Safety System Settings, Protective Instrumentation," is the overtemperature ΔT core limit protection setpoint function. TS 15.2.3.1.B(5) is the overpower ΔT core limit protection setpoint function. These functions provide setpoints that prevent exceeding the reactor core safety limits shown in Figure 15.2.1-1.

(3) TS 15.3.1.G, "Operational Limitations," specifies the RCS operational limitations for DNB (Departure from Nucleate Boiling)-related parameters. TS 15.3.1.G.3 specifies that reactor coolant system raw measured total flow rate must be equal to or greater than 181,000 gpm.

2.2 Proposed Changes

The proposed change to reduce the RCS measured total flow rate by 2,600 gpm (1.4 %) for Unit 2 requires a change to the Reactor Core Safety Limits graph which in turn causes the Overtemperature and Overpower ΔT setpoints to be changed. The licensee proposed TS changes which will revise the reactor core safety limits figure, overtemperature ΔT setpoint, overpower ΔT setpoint, and the minimum RCS flow rate for Unit 2.

(1) A new figure is being added to TS 15.2.1, "Safety Limit, Reactor Core," which is applicable to Unit 2. The title of the existing figure is being modified to indicate it is applicable to only PBNP, Unit 1.

(2) TS 15.2.3.1.B(4) and (5) is being modified as follows:
$T' \leq$ 573.9 °F (Unit 1)
$T' \leq$ 570.0 °F (Unit 2)

(3) TS Section 15.3.1.G, "Operational Limitations," is being modified to provide Reactor Coolant System flow limits specific to each unit as follows:

Reactor Coolant System raw measured Total Flow Rate:

a. ≥ 181,800 gpm (Unit 1)
b. ≥ 179,200 gpm (Unit 2)

3.0 EVALUATION

3.1 TRANSIENT ANALYSIS:

The licensee proposed to reduce the RCS measured total flow rate limit for Unit 2 by 2,600 gpm. The reduction in flow rate was evaluated by Westinghouse. The evaluation covered the (1) Non-Loss of Coolant Accident (Non-LOCA) transient analyses, (2) Loss of Coolant Accident (LOCA) analysis, and (3) Steam Generator Tube Rupture (SGTR) analysis.

3.1.1 Non-LOCA Transient Analyses Evaluation

The analyses by Westinghouse used NRC-approved methodologies and included Departure from Nucleate Boiling Ratio (DNBR) design margin. One percent of the DNBR design margin was allocated to offset the reduction in DNBR margin that would occur at the reduced RCS flow rates. This one percent reduction in DNBR design margin justifies up to a 2,600 gpm reduction in the RCS total flow rate limit. This reduction in RCS flow limit required a change to the Reactor Core Safety Limits for Unit 2.

The change to the Reactor Core Safety Limits required a change to the Overtemperature and Overpower ΔT setpoints for Unit 2. The T′ term of these setpoint functions were reduced from 573.9 °F to 570 °F. The setpoint change is based on WCAP-8745-A (Ref. 3). The reduction of these setpoints provides the appropriate protection against DNB in the core for all of the licensing basis accidents described in the FSAR for Point Beach, Unit 2.

The FSAR Section 14.1.8, "Loss of Reactor Coolant Flow," transient analyses were reanalyzed for the lower RCS flow condition with acceptable DNB results.

An evaluation of the Point Beach FSAR non-LOCA accident analyses that contain non-DNB acceptance criteria was also performed. All acceptance criteria were found to be met with the lower RCS flow. For the FSAR Section 14.1.8, "Loss of Reactor Coolant Flow," a normal operating pressure of 2,000 psia was used in the analysis to satisfy the RCS pressure limit criteria for this transient.

We have found that the results of the evaluation of the Non-LOCA transient analyses are acceptable as NRC-approved methodologies were used and the results were bounded by the FSAR acceptance criteria.

3.1.2 LOCA Evaluation

3.1.2.1 The Small Break LOCA is presented in Point Beach FSAR Section 14.3.1. As part of the increased peaking factor change in a previous Unit 2 Amendment (No. 123), a Small Break LOCA analysis was made which supported a RCS flow rate limit as low as 174,000 gpm. This is less than the 179,200 gpm being proposed in the TS change request. Since the Small Break LOCA analysis is not affected by the reduction in the RCS flow limit, we find it to be acceptable.

3.1.2.2 The Large Break LOCA analysis is presented in Point Beach FSAR Section 14.3.2. The evaluation for the Large Break LOCA was performed by Westinghouse. Their evaluation indicated that the approximately 1.5% reduction in RCS flow was well within the allowed variance for this parameter; and, because there was very little, if any, impact on the transient results, the Large Break LOCA results do not change. Since compliance with 10 CFR 50.46 is maintained, we find the results of the Large Break LOCA analysis to be acceptable.

3.1.3 SGTR Evaluation

The Steam Generator Tube Rupture analysis is described in the Point Beach FSAR Section 14.2.4. This analysis is not affected by the RCS flow rate limit reduction because the flow rate used in the analysis is lower than the proposed RCS flow rate limit. The RCS pressure and temperature could also affect this analysis. The analysis is based on an RCS pressure of 2,250 psia and average temperature of 573.9 °F. However, Unit 2 is operated at 2,000 psia and 570 °F. The lower pressure would result in a slightly lower mass release and the lower temperature would result in a slightly higher mass release in this analysis. The analysis performed by Westinghouse determined that the pressure effect is greater than the temperature effect and that the off-site radiation doses for the FSAR Section 14.2.4 SGTR analysis remain applicable for Unit 2. Therefore, we find the results of the SGTR analysis to be acceptable.

3.2 SYSTEM AND COMPONENT INTEGRITY EVALUATIONS:

Additionally, the effects of reduced RCS flow were assessed for the system and component integrity evaluations. In October 1992 Westinghouse provided an assessment to the licensee in the form of a JCO regarding operation of Point Beach at a reduced RCS temperature, based on reduced T_{hot} operation programs at "other" Westinghouse plants, and using engineering judgement to extrapolate those results to Point Beach. On October 19, 1993, the licensee submitted clarifying information, including a revision of the JCO which provided plant specific information regarding reduced T_{avg} operation of Point Beach, Units 1 and 2.

The October 19, 1993, submittal (Reference 2) indicated that both Units 1 and 2 have been operating at reduced T_{avg} of 570 °F for the past 21 years. As such, the licensee evaluated the effects of the reduced RCS average temperature on the structural and pressure boundary integrity of the piping systems, components, and their supports. Further, the licensee utilized the results of their Transient and Fatigue Cycle Monitoring Program from 1986 and 1987 to establish which components in the plant are most susceptible to fatigue over plant life. Components evaluated include the control rod drive mechanisms (CRDM), pressurizer, reactor vessel and internals, steam generators and reactor coolant pumps.

3.2.1 Reactor Vessel and Internals

For the reactor vessel and internals, the original fatigue analyses were reviewed by Westinghouse, and the limiting components were identified to be the reactor vessel closure studs and the safety injection nozzles. The maximum cumulative usage factor reported in the reactor design stress report is 0.79 for the safety injection nozzles on 40-years life of plant operation. The maximum stresses at critical locations were also evaluated for the reduced T_{avg} condition. Westinghouse indicated that the increase in stress was insignificant (less than seven percent for the core barrel outlet nozzle). Considering the conservatism in the analysis, as stated by the licensee, 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, we conclude that operation at the reduced T_{avg} has negligible impact on the original stress and fatigue analyses of the reactor vessel and internals.

3.2.2 Reactor Coolant Loop Piping and Supports

The Point Beach piping, including the primary loops and Class 1 auxiliary piping systems, was originally designed to the ASME/ANSI B31.1 Power Piping Code. Fatigue analyses were not required by the Code. Westinghouse indicated that static LOCA loads were used for the original piping analyses and that these loads are not affected by the reduced T_{avg}. Westinghouse also indicated that the analysis of the pressurizer surge line, performed in response to NRC Bulletin 88-11 regarding the thermal stratification issue, included the impact of the reduced T_{avg} operation. Therefore, the original analyses for the piping, including the pressurizer surge line, primary loops and Class 1 auxiliary systems, and pipe supports remain unchanged for the operation of the Units at a T_{avg} of 570 °F.

3.2.3 Control Rod Drive Mechanisms

The Point Beach Units used the same model of Control Rod Drive Mechanisms (CRDM) as one of the other Westinghouse plants that was specifically analyzed for reduced T_{hot} operation. Westinghouse indicated that the Point Beach reduced T_{avg} is equivalent to a T_{hot} reduction from 602 °F to about 598.9 °F. The similar CRDMs of the "other" plant was analyzed for a T_{hot} reduction from 610 °F to 595 °F, which is bounding for the Point Beach operation. Therefore, we conclude that the Point Beach CRDMs will not be affected at a reduced T_{avg} of 570 °F.

3.2.4 Pressurizer

Westinghouse evaluated the Series 84 Point Beach pressurizer by comparing the design basis thermal analysis with the T_{avg} of 570 °F operation. The evaluation consisted of reviewing the original stress analysis for the most limiting locations (the Spray Nozzle and the

Upper Head to Shell Junction). The licensee stated that the pressurizer spray nozzle was the most limiting component from a fatigue perspective, with the fatigue usage factor predicted to not exceed approximately 0.85 at the end of the current license. The review indicated that the design bases of stress and fatigue analyses for the Series 84 pressurizer were based on a delta-T of 125 °F which envelopes the Point Beach operating condition with a delta-T of 112.4 °F at the reduced T_{avg} operation. Based on our review, we conclude that operation at an RCS T_{avg} of 570 °F will not have an adverse impact on the structural integrity of the pressurizer.

3.2.5 Steam Generator

The Series 44 Steam Generators in the Point Beach Units were not specifically analyzed for Point Beach design parameters, but had been previously analyzed by Westinghouse using design parameters consistent with other Westinghouse plants. The key design input parameters such as T_{avg} and pressure differential at the interface of the primary and the secondary systems for the previously analyzed Series 44 Unit and the Point Beach Series 44 Unit, were summarized in the JCO. The comparison shows that the input conditions of the previous analysis are more severe than the operation of Point Beach at a T_{avg} of 570 °F. Based on the above review, Westinghouse determined that stresses and fatigue usage factors for the Point Beach Units are within the Code allowable limits. Based on our review, we concur with Westinghouse's conclusion that the steam generators are acceptable for operation at the reduced T_{avg} of 570 °F.

3.2.6 Reactor Coolant Pumps

Westinghouse evaluated the adequacy of the Reactor Coolant Pumps by comparing the operating parameters of Point Beach, Units 1 and 2 at the reduced T_{avg} of 570 °F, with those used in the design analysis of the same Model 93 RCP for the other Westinghouse plant. The comparison shows that the delta-T and corresponding design pressure and thermal transients of the design basis analysis envelop those for Point Beach, Units 1 and 2 at the reduced T_{avg} condition. Based on the above review, we agree with Westinghouse's conclusion that the structural integrity of the RCPs is not adversely impacted by the reduced T_{avg} operating conditions.

4.0 EVALUATION OF TECHNICAL SPECIFICATIONS

The TSs were changed as a result of the proposed revisions to modify TS Section 15.3.1.G, "Operation Limitations," Specification 3, to reduce the reactor coolant system (RCS) total flow rate limit by 2,600 gallons per minute (gpm), change the overtemperature and overpower setpoints, and change the Reactor Core Safety Limits for Unit 2.

(1) A new figure was added to TS 15.2.1, "Safety Limit, Reactor Core," which is applicable to Unit 2. The title of the existing figure was modified to indicate it is applicable to only PBNP, Unit 1. TS 15.2.1 was modified to read:

"1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2."

The associated basis was also changed to reflect the revision to TS 15.2.1. The proposed basis revision is as follows:

"The curves of Figure 15.2.1-1 and 15.2.1-2 are applicable for 14 x 14 OFA. The curves also apply to the reinsertion of previously-depleted 14 x 14 standard fuel assemblies into an OFA core."

(2) TS 15.2.3.1.B(4) and (5) modified as follows:

"T′ $\leq$ 573.9 °F (Unit 1)
T′ $\leq$ 570.0 °F (Unit 2)"

The associated basis was also changed to reflect the revision to TS 15.2.3.1.B(4) and (5). The proposed basis revision is as follows:

"With normal axial power distribution, the reactor trip limit, with allowance for errors[2] is always below the core safety limit as shown on Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the overtemperature ΔT setpoint is automatically reduced[6][7].

The overpower, overtemperature, and pressurizer pressure system setpoints include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints will not exceed the core safety limits as shown in Figure 15.2.1-1 for Unit 1 and Figure 15.2.1-2 for Unit 2."

(3) TS Section 15.3.1.G, "Operational Limitations," was modified to provide Reactor Coolant System flow limits specific to each unit as follows:

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

a. $\geq$ 181,800 gpm (Unit 1)
b. $\geq$ 179,200 gpm (Unit 2)"

The associated basis was also changed to reflect the revision to TS 15.2.1.G.3. The proposed basis revision is as follows:

"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 179,200 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (175,400 gpm). The raw measured flow is based upon the use of normalized elbow tap differential which is calibrated against a precision flow calorimetric at the beginning of each cycle."

We find the above changes to be acceptable as discussed in the evaluation made in Section 3.0.

5.0 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.

6.0 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 or change an inspection or surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (58 FR 43940). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

7.0 CONCLUSION

The staff agrees with the licensee's conclusion that component and system stress and fatigue for operation at reduced reactor coolant temperatures will not exceed Code allowable limits over the current licensed life of the facility. The staff also agrees with the licensee that their JCO is sufficient to justify operation of Unit 2 until December 31, 1996 (which coincides with the planned replacement of the Unit 2 steam generators).

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: H. Balukjian
Chen-Ih Wu

Date: October 27, 1993

REFERENCES

1. Letter, B. Link, Wisconsin Electric Power Company (WEPCo), to USNRC, dated, June 11, 1993.

2. Letter, Bob Link, WEPCo, to USNRC, dated October 19, 1993.

3. WCAP-8745-A, "Design Basis of Over-Temperature-delta-T and Over-Temperature-delta-T Trip Functions," Ellenberger, F. L., et al., dated September 1986.