DEC 04 1973

Docket No. 50-301

Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein, Vice President
231 West Michigan Street
Milwaukee, Wisconsin 52301

Change No. 8
License No. DPR-27

Gentlemen:

By letter dated September 24, 1973, you proposed revisions to the Technical Specifications attached as Appendix A to Facility Operating License DPR-27 designated as Change No. 8.

The requested change allows operation of the Point Beach Nuclear Plant Unit 2 at a reduced reactor coolant system pressure in order to achieve a longer second fuel cycle. In support of this you submitted a report entitled "Fuel Densification, Point Beach Nuclear Plant Unit No. 2, Low Pressure Analysis" dated June 1973.

We have reviewed your report and have determined that the effects of low pressure operation on DNB and the cladding creep rate have been appropriately considered and that Cycle 1 of Unit 2 can be extended to 13,000 EFPH provided operation at reduced pressure is initiated prior to 6500 EFPH and appropriate changes to the Limiting Safety System Settings in the Technical Specifications are made. Our Safety Evaluation is attached as Enclosure 1.

Changes to the Limiting Safety System Settings will be made in Section 2.1 to limit fuel residence time to 13,000 EFPH; to Section 2.3.1-B to reduce the low pressurizer pressure trip setting, and to revise the Overtemperature ΔT trip setting, and to revise the Overpower ΔT trip setting; all in accordance with revised Figure 15.2.1-1.

We conclude that the changes do not involve significant hazard considerations and there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly,

LB

OFFICE➤						
SURNAME➤						
DATE➤						

pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License DPR-27 are hereby changed as described in the Wisconsin Electric Power Company letter of October 12, 1973, and as set forth in the revised pages which are enclosed.

The revised Technical Specifications covering operation of Unit 2 at a lower primary system pressure shall become effective at the discretion of the Wisconsin Electric Power Company, but prior to attaining a fuel exposure of 6500 EFPH for Cycle 1. The implementation of this change involving resetting of the protection system instrumentation shall be at reduced power and in accordance with a procedure approved by the Manager of Nuclear Power's supervisory staff. Until such time as these revised Technical Specifications are implemented, the current Technical Specifications shall be deemed to apply to Unit 2.

Sincerely,

D. J. Skovholt, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages

cc:
John K. Babbitt, Vice President
and General Manager
Wisconsin Michigan Power Company
807 South Oneida Street
Appleton, Wisconsin 54911

						AD/ORs
						DSkovholt
						12/5/73
OFFICE➤	PWR-2 KK	OR-1 PBE	OR-1	AD/PWRs	OGC	RO K. Kniel
SURNAME➤	KKniel:ng	PErickson	RSchemel	RCDeYoung		for SBryan
DATE➤	11/27/73	11/28/73	11/28/73	12/5/73	12/5/73	11/28/73

cc (continued)
Gerald Charnoff, Esquire
Shaw, Pittman, Potts & Trowbridge
910 17th Street, NW
Washington, D. C. 20006

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 4, 1973

Docket No. 50-301

Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein, Vice President
231 West Michigan Street
Milwaukee, Wisconsin 52301

Change No. 8
License No. DPR-27

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By letter dated September 24, 1973, you proposed revisions to the Technical Specifications attached as Appendix A to Facility Operating License DPR-27 designated as Change No. 8.

The requested change allows operation of the Point Beach Nuclear Plant Unit 2 at a reduced reactor coolant system pressure in order to achieve a longer second fuel cycle. In support of this you submitted a report entitled "Fuel Densification, Point Beach Nuclear Plant Unit No. 2, Low Pressure Analysis" dated June 1973.

We have reviewed your report and have determined that the effects of low pressure operation on DNB and the cladding creep rate have been appropriately considered and that Cycle 1 of Unit 2 can be extended to 13,000 EFPH provided operation at reduced pressure is initiated prior to 6500 EFPH and appropriate changes to the Limiting Safety System Settings in the Technical Specifications are made. Our Safety Evaluation is attached as Enclosure 1.

Changes to the Limiting Safety System Settings will be made in Section 2.1 to limit fuel residence time to 13,000 EFPH; to Section 2.3.1-B to reduce the low pressurizer pressure trip setting, and to revise the Overtemperature ΔT trip setting, and to revise the Overpower ΔT trip setting; all in accordance with revised Figure 15.2.1-1.

We conclude that the changes do not involve significant hazard considerations and there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License DPR-27 are hereby changed as described in the Wisconsin Electric Power Company letter of October 12, 1973, and as set forth in the revised pages which are enclosed.

The revised Technical Specifications covering operation of Unit 2 at a lower primary system pressure shall become effective at the discretion of the Wisconsin Electric Power Company, but prior to attaining a fuel exposure of 6500 EFPH for Cycle 1. The implementation of this change involving resetting of the protection system instrumentation shall be at reduced power and in accordance with a procedure approved by the Manager of Nuclear Power's supervisory staff. Until such time as these revised Technical Specifications are implemented, the current Technical Specifications shall be deemed to apply to Unit 2.

Sincerely,

D. J. Skovholt, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages

cc:
John K. Babbitt, Vice President
and General Manager
Wisconsin Michigan Power Company
807 South Oneida Street
Appleton, Wisconsin 54911

cc (continued)
Gerald Charnoff, Esquire
Shaw, Pittman, Potts & Trowbridge
910 17th Street, NW
Washington, D. C. 20006

ENCLOSURE 1

POINT BEACH NUCLEAR PLANT UNIT 2

SAFETY EVALUATION FOR REDUCED PRIMARY SYSTEM PRESSURE OPERATION

We have reviewed the report "Fuel Densification - Point Beach Nuclear Plant Unit 2 - Low Pressure Analysis," WCAP 8151 dated June 1973, which was submitted by the Wisconsin Electric Power Company in support of a proposed change to the Technical Specifications for Facility Operating License DPR-27 Point Beach Nuclear Plant Unit 2. The proposed change, designated as Change No. 8, concerns those changes to the Technical Specifications necessary for operation of Unit 2 for the remainder of Cycle 1 at a reduced primary system pressure of 2000 psia.

On the basis of our review we have determined that the areas requiring assessment were minimum DNB ratio, nuclear design considerations, and time dependent creep collapse of fuel element cladding.

The DNBR analysis was performed using methods as described and previously reviewed and accepted by the staff for both Point Beach Units 1 and 2 (Docket Nos. 50-266 and 50-301). The minimum value of the DNB ratio during normal operation and anticipated transients is limited to a value of 1.30. A reduction in the core inlet temperature of the reactor coolant from 552.5°F to 545°F is required for the reduction in reactor coolant system pressure from 2250 psia to 2000 psia in order that the calculated DNBR is greater than 1.30 for power operation up to 100 percent of rated power, including anticipated transients. We find this acceptable and conclude that the DNB margins are similar to those at 2250 psia pressure because the reactor coolant temperatures have been reduced accordingly. The revised Core DNB Safety Limits are presented in Figure 15.2.1-1 of WCAP 8151.

The reduction in average coolant temperature has a minor effect on the core nuclear properties and we conclude that any such effects have been properly accounted for.

The reduction in primary system operating pressure reduces the pressure differential across the fuel cladding and thus reduces the clad creep rate. Using a Westinghouse time dependent creep collapse model which we have previously reviewed for Point Beach Unit 2 and found acceptable, you have calculated that Cycle 1 fuel exposure can be extended to a minimum of 13,000 EFPH without clad flattening. We have independently calculated the effect of reduced system pressure on clad time-to-collapse for Unit 2 Cycle 1 and conclude that your result is conservative and acceptable provided the depressurization occurs as planned before 6500 EFPH.

In summary we have determined that the effects of fuel densification and reduced primary coolant system pressure at rated power for Unit 2 Cycle 1 have been adequately analyzed and that the Cycle 1 operation can be extended to 13,000 EFPH. We conclude that appropriate changes to the Limiting Safety System Settings in the Technical Specifications which reflect the Core DNB Safety Limits presented in Figure 15.2.1-1 of the report WCAP 8151, will assure that the consequences of reactor transients and accidents are not significantly different from those previously judged acceptable by the staff.

Karl Kniel

Karl Kniel, Chief
Pressurized Water Reactors
Branch No. 2
Directorate of Licensing

ENCLOSURE 2

CHANGE NO. 8 TO

TECHNICAL SPECIFICATIONS FOR

FACILITY OPERATING LICENSE NO. DPR-27

FOR POINT BEACH NUCLEAR PLANT UNIT 2

WISCONSIN ELECTRIC POWER COMPANY

WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-301

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. 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.

2. The fuel residence time for Unit 2, Cycle 1, shall be presently limited to 13,000 effective full power hours (EFPH) under design operating conditions, provided the primary system pressure is reduced to 2000 psia by 6500 EFPH. The Licensee may propose to operate the core in excess of 13,000 EFPH by providing an analysis which includes the effect of clad flattening or a change in operating conditions. Any such analysis, if proposed, shall be approved by the Regulatory staff prior to operation in excess of 13,000 EFPH. 8

This combination of hot channel factors is higher than that calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 15.3.10-1. Somewhat worse hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 15.3.10-1 insure that the DNB ratio is always greater at part power than at full power. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of the curves shown in Figure 15.2.1-1.

Figure 15.2.1-1 also includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression: 6

$F^N_{\Delta H} = 1.58\ [1 + 0.2\ (1-P)]$ where P is the fraction of rated power.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of part-length rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30.

The fuel residence time for Unit 2, Cycle 1 is limited to 13,000 EFPH to assure no fuel clad flattening without prior review by the Regulatory 8
staff. The residence time of 13,000 EFPH is based on predicted minimum time to clad flattening for an operating pressure of 2250 psi up to mid-cycle, or 6500 EFPH, and 2000 psi thereafter. Beyond a residence time of 13,000 EFPH for the present core, an assumption of clad flattening is presently required. Prior to 13,000 EFPH, the Licensee may provide the additional analysis required for operation beyond 13,000 EFPH.

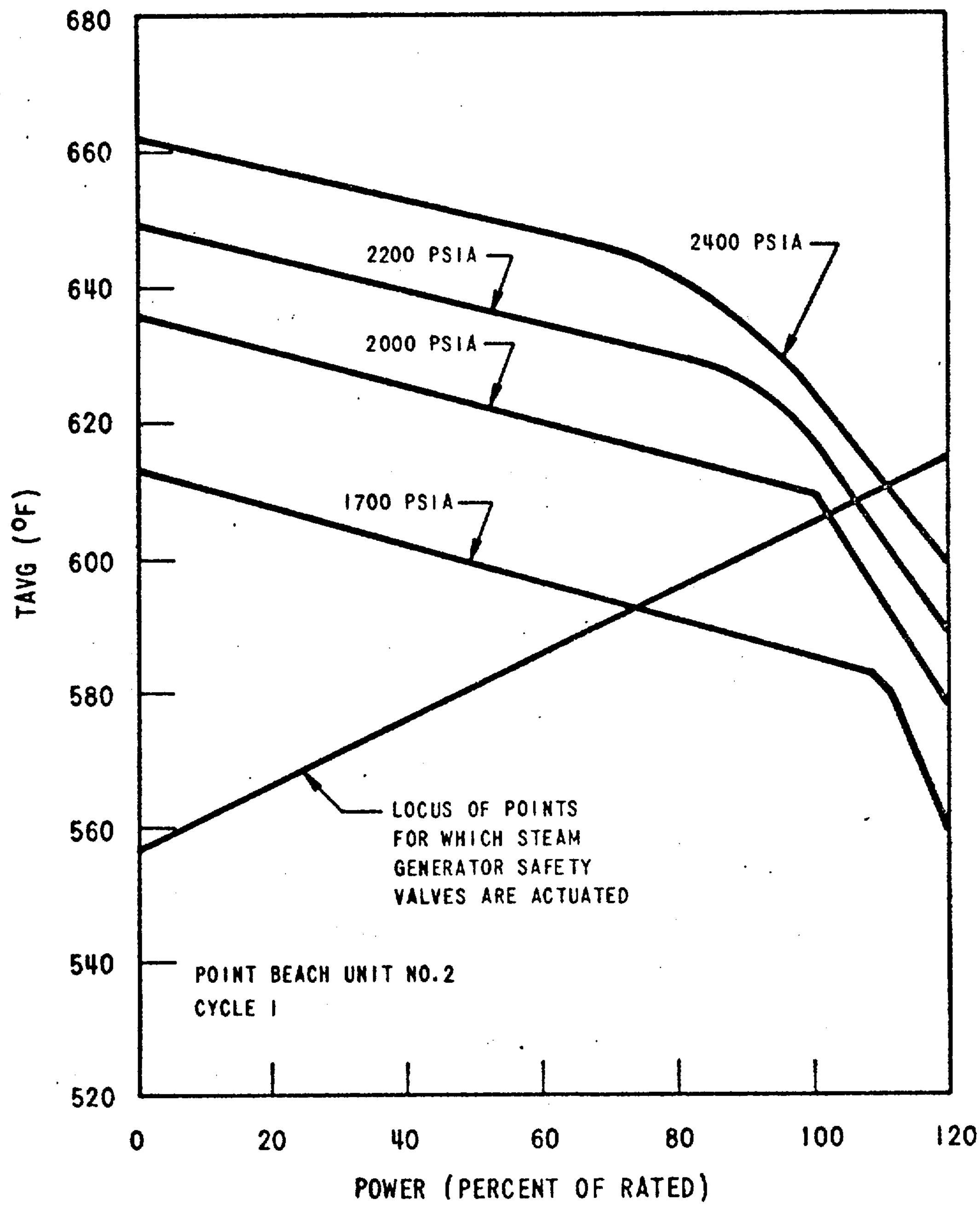

Figure 15.2.1-1 Core DNB Safety Limits

(3) Low pressurizer pressure - $\geq$ 1710 psig. 8

(4) Overtemperature ΔT

$$\leq \Delta T_o \left[K_1 - K_2 (T-T') \frac{1 + \tau 1^S}{(1 + \tau 2^S)} + K_3 (P-P') - f(\Delta I)\right]$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 572.9°F 8

P = pressurizer pressure, psig

P' = 1985 psig

K_1 = 1.11

K_2 = 0.0158 8

K_3 = 0.000852

τ_1 = 25 sec

τ_2 3 sec

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 instrumented 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 -12, + 5 percent, f (ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent the ΔT trip set point shall be automatically reduced by an equivalent 6 of two percent of rated power.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -12 6
percent the ΔT trip setpoint shall be automatically reduced
by an equivalent of two percent of rated power. 8

[1.B.(5)] Overpower T

$$\leq \Delta T_o \left[K_4 - K_5 \frac{\tau_3 S}{\tau_3 S + 1} T - K_6 (T - T') - f(\Delta I)\right]$$

where

ΔT_o = indicated T at rated power, °F

T = average temperature, °F

T' = 572.9°F 8

K_4 $\leq$ 1.08 of rated power

K_5 = 0.0262 for increasing T

= 0.0 for decreasing T 6

K_6 = 0.0 for $T<T'$

= 0.0012 for $T \geq T'$

τ_3 = 10 sec 8

f (ΔI) as defined in (4) above,

(6) Undervoltage - $\geq$75% of nromal voltage

(7) Low indicated reactor coolant flow per loop -
$\geq$90% of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point $\geq$57.5 cps

(b) Low voltage set point $\geq$75% 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. If axial peaks are greater than design, as indicated by 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 have been revised to include effect of reduced system pressure operation (including the effects of fuel densification). The revised setpoints as given above will not exceed the revised core safety limits as shown in Figure 15.2.1-1. 8

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. 6

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 of a sudden loss of pwoer to one or both reactor coolant pumps. The set point 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% 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 set-point, 57.5 cps, 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 set point 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 set point 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 set point above which these trips are unblocked assures their availability in the power range where needed.

Sustained operation with only one pump will not be permitted above 10% power. If a pump is lost while operating between 10% and 50% power, an orderly and immediate reduction in power level to below 10% 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) FASR 14.1.1

(2) FSAR, page 14-3

(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