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Dockets Nos. 50-266
and 50-301

Wisconsin Electric Power Company
Wisconsin Michigan Power Company
ATTN: Mr. Sol Burstein
Executive Vice President
231 West Michigan Street
Milwaukee, Wisconsin 53201

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 28 and 32 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications and are in accordance with your applications dated July 8, 1977 (as supplemented by letter dated September 20, 1977) and July 8, 1977.

These amendments: (1) modify the coefficients of the overpower and overtemperature ΔT setpoint equations, (2) permit limited removal of individual fuel rods suspected of leaking, and (3) delete the description of Part Length Rod Control Cluster assemblies from the Technical Specifications of Point Beach Nuclear Plant Units Nos. 1 and 2.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

Original si[illegible]

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. to License DPR-24
2. Amendment No. to License DPR-27
3. Safety Evaluation
4. Federal Register Notice

Const. 1

cc w/enclosures: *SEE PREVIOUS YELLOW FOR CONCURRENCES
See next page

GD

OFFICE➤		ORB #3	ORB #3	OELD	ORB #3	DOR
SURNAME➤		*CParrish	*PWagnwr:acr	*	GLear *	VStello
DATE➤		10/31/77	10/31/77	11/2/77	11/ 2/77	11/ /77

cc:

Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
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Mr. Norman Clapp, Chairman
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of Wisconsin
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Chief, Energy Systems Analysis Branch (AW-459)
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Walter L. Meyer
Town Chairman
Town of Two Creeks, Wisconsin
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 28
License No. DPR-24

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

   A. The applications for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated July 8, 1977 (as supplemented by letter dated September 20, 1977) and July 8, 1977, comply 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 28, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NO. 50-266

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change.

Remove	Replace
15.2.3-2	15.2.3-2
15.2.3-3	15.2.3-3
15.5.3-1	15.5.3-1
15.5.3-2	15.5.3-2

(3) Low pressurizer pressure - $\geq$1865 psig.

(4) Overtemperature ΔT

$$\leq \Delta T_{\circ} \; [K_1 - K_2(T-T') \; \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3 \; (P-P') - f(\Delta I)]$$

where

$\Delta T_{\circ}$ = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 574.2 °F

P = pressurizer pressure, psig

P' = 2235 psig

$K_1 \leq$ 1.117

K_2 = 0.0150

K_3 = 0.000791

τ_1 = 25 sec

τ_2 = 3 sec

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, +9 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceed +9 percent the ΔT trip set point shall be automatically reduced by an equivalent of two 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 two percent of rated power.

[1.B. (5)] Overpower ΔT

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

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 574.2

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_3 = 10 sec

$f(\Delta I)$ as defined in (4) above,

(6) Undervoltage - $\geq$ 75% of normal 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

15.5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Emergency Core Cooling Systems.

Objective

To define those design features which are essential in providing for safe system operation

Specifications

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly nominally contains 179 fuel rods.[1] Where safety limits are not violated, individual fuel rods suspected of leaking may be replaced with an inert rod or the assembly left with a water hole to prevent possible reinsertion of leaking fuel rods. No more than one fuel rod may be replaced in any single assembly and no more than six (6) such modified assemblies may reside in the core at any time.
2. The average enrichment of the initial core is a nominal 2.90 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.40 weight percent of U-235.[2]
3. Standard reload fuel will be similar in design to the initial core.

4. Burnable poison rods are incorporated in the initial core. There are 704 poison rods in the form of 8, 12 and 16 rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borated pyrex glass clad with stainless steel.[4]

5. There are 33 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain a 142 inch length of silver-indium-cadmium alloy clad with the stainless steel.

6. Up to ten (10) grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. <u>Reactor Coolant System</u>

1. The design of the Reactor Coolant System complies with the code requirements.[6]

2. All high pressure piping, components of the Reactor Coolant System and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g, acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

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

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. DPR-27

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

   A. The applications for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated July 8, 1977 (as supplemented by letter dated September 20, 1977) and July 8, 1977, comply 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) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 32

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NO. 50-301

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change. Add page 15.5.3-2A.

Remove	Replace
15.2.3-2	15.2.3-2
15.2.3-3	15.2.3-3
15.5.3-1	15.5.3-1
15.5.3-2	15.5.3-2
—	15.5.3-2A

(3) Low pressurizer pressure - $\geq$1865 psig.

(4) Overtemperature ΔT

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

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 574.2 °F

P = pressurizer pressure, psig

P' = 2235 psig

$K_1 \leq$ 1.117

K_2 = 0.0150

K_3 = 0.000791

τ_1 = 25 sec

τ_2 = 3 sec

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, +9 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceed +9 percent the ΔT trip set point shall be automatically reduced by an equivalent of two 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 two percent of rated power.

[1.B. (5)] Overpower ΔT

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

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 574.2

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_3 = 10 sec

$f(\Delta I)$ as defined in (4) above,

(6) Undervoltage - $\geq$ 75% of normal 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

15.5.3 REACTOR

Applicability

Applies to the reactor core, Reactor Coolant System, and Emergency Core Cooling Systems.

Objective

To define those design features which are essential in providing for safe system operation

Specifications

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly nominally contains 179 fuel rods.[(1)] Where safety limits are not violated, individual fuel rods suspected of leaking may be replaced with an inert rod or the assembly left with a water hole to prevent possible reinsertion of leaking fuel rods. No more than one fuel rod may be replaced in any single assembly and no more than six (6) such modified assemblies may reside in the core at any time.

2. The average enrichment of the initial core is a nominal 2.90 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.40 weight percent of U-235.[(2)]

3. Standard reload fuel will be similar in design to the initial core.

Amendment No. 32

4. Burnable poison rods are incorporated in the initial core. There are 704 poison rods in the form of 8, 12 and 16 rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borated pyrex glass clad with stainless steel.[4]

5. There are 33 full-length RCC assemblies in the reactor core. The full-length RCC assemblies contain a 142 inch length of silver-indium-cadmium alloy clad with the stainless steel.

6. Up to ten (10) grams of enriched fissionable material may be used either in the core, or available on the plant site, in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the Reactor Coolant System complies with the code requirements.[6]

2. All high pressure piping, components of the Reactor Coolant System and their supporting structures are designed to Class I requirements, and have been designed to withstand:

   a. The design seismic ground acceleration, 0.06g, acting in the horizontal and 0.04g acting in the vertical planes simultaneously, with stresses maintained within code allowable working stresses.

b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the Reactor Coolant System, at rated operating conditions, is 6040 cubic feet.

References

(1) FSAR Section 3.2.3

(2) FSAR Section 3.2.1

(3) FSAR Section 3.2.1

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-9

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 28 AND 32 TO FACILITY LICENSES DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH UNITS NOS. 1 AND 2

DOCKETS NOS. 50-266 AND 50-301

Introduction

By the applications dated July 8, 1977(1) (as supplemented by letter dated September 20, 1977(2)) and July 8, 1977(3), Wisconsin Electric Power Company (the licensee) proposed to change the Technical Specifications for the Point Beach Nuclear Plant Units Nos. 1 and 2. The proposed change would permit the licensee to replace 48 of 121 fuel assemblies in the reactor core of Point Beach Unit No. 1 constituting refueling of the core for the sixth cycle of operation. This refueling consists of the replacement of 48 burned fuel assemblies by 32 fresh assemblies and sixteen previously exposed fuel assemblies. One Region 7 assembly which resided in the core during Cycle 4 will be loaded in the core for Cycle 6 operation with one fuel rod removed.

Analyses performed for the Cycle 6 reload core design were based on the following assumptions:

1. Cycle 5 operation is terminated after 9050 (+200, -450) MTD/MTU.

2. Cycle 6 operation is in adherence to plant operating limitations given in the Technical Specifications.

3. Cycle 6 is to be operated at a reactor coolant systems pressure of 2250 psia.

The licensee has proposed the following changes to the Technical Specifications for Units Nos. 1 and 2:

1. Revision of coefficients of overpower and overtemperature ΔT setpoint equations to permit operation at 2250 or 2000 psia system pressure using a single set of coefficients.

2. Permit limited removal of individual fuel rods suspected of leaking. The suspected fuel rod would be replaced with an inert rod or the assembly left with a water hole.

3. Delete description of part length Rod Control Cluster assemblies to reflect removal of the part length control rods from the reactor core.

Evaluation

Fuel Mechanical Design

The mechanical design of the Region 9 fuel assemblies is the same as Region 8 except for revision of the assembly holddown springs. The holddown springs have been modified to provide additional holddown force hence providing greater confidence that the assembly will not be displaced by flow forces. This modification is considered acceptable by the NRC Staff. The mechanical design of Region 6 through 8 fuel assemblies has been previously accepted by the NRC staff and has not been re-reviewed for this reload.

The fuel rod design for Region 9 is mechanically the same as Region 8 except for a reduction of the helium backfull pressure by 20 psi. The fuel rod design for Region 9 is mechanically the same as Region 8 except for a reduction of the helium backfill pressure by 20 psi. The resultant fractional change of the backfill pressure is insignificant with respect to clad collapse considerations. The reduction of the backfill pressure results in a small decrease of the licensee predicted fuel rod internal pressure of the exposed fuel rod. The change has been made to allow potential future operation at 2000 psia system pressure. Other aspects of the fuel rod mechanical design have been previously accepted by the NRC Staff and have not been re-reviewed for this reload.

One Region 7 fuel assembly (G07), core location G-2, which will reside in the core during Cycle 6 sustained damage to a single fuel rod while in the core during Cycle 4. The damaged fuel rod has been removed and replaced with an inert rod. In order to perform this procedure a slot was machined in the assembly bottom nozzle. The licensee has assessed the stresses associated with the mechanical slot and the effects of localized cross flow. The licensee has also evaluated the dynamic performance of the fuel rods in the vicinity of the machined slot. The inert rod and capture mechanism were designed by Exxon Nuclear. Similar designs have been successfully utilized in BWR's. The inert rod is predicted to reside in a low power, non-limiting, region of the core. On these bases the modification of bundle G07 is considered acceptable.

Future limited use of inert rods to replace damaged fuel rods is envisioned by the licensee. The program is to be limited in scope (less than or equal to 6 inert rods). The program is considered acceptable based on the analyses described above. Future use of inert rods is to be restricted to use in low power non-limiting, core locations. The Commission is to be notifed of future replacement or removal of damaged fuel rods.

Clad collapse is not predicted to occur during Cycle 6. Supporting analysis was performed using the Westinghouse revised clad flattening model[4]. To preclude rod internal pressure from exceeding nominal system pressure in thirteen of the seventeen Region 6 assemblies which will reside in the core during Cycle 6, the reactor coolant system is to be operated at a pressure of 2250 psia. This is in conformance with the staff's previously approved design criteria. The staff has recently approved a design criterion that permits internal W fuel pressure to exceed system pressure. On these bases the staff finds the predicted fuel rod internal pressure for Cycle 6 acceptable. In addition, calculations were performed by the licensee using a revised Westinghouse model[5] which considers enhanced fission gas release. This revised model is currently under staff review.

Nuclear Design

Nominal design parameters are consistent with previous cycles. The core is to be operated at 100% of rated power, 1518 MW_t (nuclear), 2250 psia system pressure and an average linear heat generation rate of 5.70 kw/ft.

The licensee has stated that kinetics parameters for Cycle 6 with the exception of the delayed neutron fraction (βeff), are within the bounds of the values used in previous safety analyses; based on the Cycle 6 fuel inventory, this statement is acceptable. The minimum delayed neutron fraction used in the Cycle 6 analysis is less than the value used in previous analysis; this change impacts the ejected rod safety analysis which was reanalyzed (addressed below) and has neglibible impact on other transients and accidents.

The licensee predicted net rod worth and shutdown requirements provide ample shutdown margin for transients and accidents other than the steam line break accident. These values are consistent with values predicted for previous cycles. The licensee has not explicitly predicted shutdown margin in excess of the requirements for the current steam line break analysis. Hence, no explicit margin exists to accommodate potential errors in the predicted power defect (doppler, moderator, voids). However, the licensee's analysis of shutdown worth includes allowance for a 10 percent uncertainty in the total worth of the control rods, with the highest worth rod withdrawn. The steam line break accident analysis is addressed below.

Continued operation using the Westinghouse Constant Axial Offset Control Strategy(6) during Cycle 6 with a +6%, -9% flux difference target band has been supported by recomputation of anticipated axial power distributions and corresponding axial peaking factors for a subset of anticipated load follow operations.

The control strategy, flux difference target band, power dependent bank insertion limits, core power outputs and operating inlet temperature were not changed. The Cycle 6 fuel inventory is not dissimilar from that of Cycle 5. Hence anticipated axial power distributions and concomitant axial power peaking factors as a function of core height are not expected by the staff to be substantially different from values calculated for the previous cycles and in turn support the scope of the licensee's computations. It is noted that the control analyses are based on generic radial unrodded and rodded peaking factors of 1.435 and 1.58 respectively. Operation such that achieved radial power peaking factors are less than or equal to those assumed in the analysis is to be confirmed using monthly incore maps. Operation using the control strategy and incore power distribution mapping will insure that limiting values of the peak to average linear heatrate, F_q=2.32, and the peak to average enthalpy rise, $F_{\Delta H}$=1.58, are not violated.

The NRC staff concludes that the nuclear design is acceptable.

Accident Analysis

The licensee has reviewed all postulated accidents which were reported in the Final Facility Description and Safety Analysis Report (FFDSAR) and states that those transients and accidents which were found to be potentially affected were reanalyzed. These were identified by the licensee to be the ejected rod accident and control rod withdrawal from power conditions. The licensee states that all other transients and accidents are bounded by the Reference Analyses* and on this basis we accept their conclusion that those transients and accidents need not be reanalyzed.

*Reference analyses are defined as those transient and accident analyses which appear in the FFDSAR and/or have been updated by subsequent submittals.

The rod withdrawal incident was reanalyzed because the reactivity insertion rate (trip reactivity function) for the scrammed rods is slower for Cycle 6 than the previous cycles. The licensee did not present the detailed results of this analysis but has stated that the incident was bounded by the applicable design basis limits of the Reference analyses. The staff accepts the licensees assertions based on the following:

1. Reference analyses were performed using extrema values of kinetics parameters. Although expectation values of Cycle 6 kinetics parameters have not been provided, based on the Cycle 6 fuel inventory there is no reason to believe that the extrema values will be exceeded during Cycle 6.

2. Initial state points (nuclear power, heat flux, moderator temperature and pressure) are the same as in the reference analyses.

3. Trip state points have not been changed for Cycle 6 operation.

4. Hence based on 1, 2 and 3 above the predicted gross core behavior during the transient or accident should be readily bounded.

5. For those transients which terminate rapidly relative to the fuel pin thermal time constant*, or during which the relative power distribution remains relatively unperturbed**, the Technical Specification on peak to average enthalpy rise, $F_{\Delta H}$, bounds local peaking. Hence the local behavior is readily bounded by the reference analysis.

6. For those transients which terminate slowly relative to the fuel pin thermal time constant and during which the relative power distribution is perturbed, cycle specific values of local peaking are considered. Predicted values of local peaking and DNBR have not been provided by the licensee. The licensee has stated that the predicted minimum DNBR (at the peak pin) is greater than 1.3. Specific values of the predicted minimum DNBR (as long as they are greater than 1.3) at or near the hot rods for this subset of transients is not considered significant.

---

* e.g. high worth rod withdrawal at power

**e.g. loss of feedwater

The ejected rod accident was reanalyzed because post rod ejection peaking factors were predicted to be higher and values of the delayed neutron fraction smaller in Cycle 6 than the values used in previous ejected rod safety analyses. The rod ejection accident analysis was performed at beginning and end of cycle, hot full power, and hot zero power. Predicted results of the analysis demonstrate ample margin to the design criteria that the maximum fuel enthalpy not exceed 280 cal/gm. The NRC Staff finds this acceptable.

The steamline break (SLB) accident was not reanalyzed for Cycle 6. The licensee has stated that Cycle 6 expectation values are within the bounds of the input parameters of previously performed reference SLB analysis. The licensee states that minimum predicted DNBR is greater than 1.30. SLB analysis methods are currently being generically reviewed by the NRC staff.

Pending completion of the review of SLB analyses methods, contined operation is judged acceptable based on the following analysis. The hypothetical steamline break is a design bases event for which limited clad failure is permitted. Staff scoping calculations show that approximately as much as 16% of the fuel rods could be failed without exceeding the site boundary dose rate limits. The relative power density predicted during the course of steamline break with all control rods, except the most reactive rod, inserted is highly non-uniform. The predicted minimum DNBR during the transient would occur near the region of the stuck rod and be restricted to a small region of the core. Even if departure from nucleate boiling were to occur, and even if clad failure were to occur, the staff concludes that less than 16% of the fuel rods would fail and hence site boundary dose rate limits would not be violated.

The staff has under consideration the need to reassess the Westinghouse evaluation model for 2 loop units. The application of such reassessment to the Point Beach plant will be based upon the results of any such reassessment. Preliminary evaluation of the potential effect of such assessment demonstrates that facility operation at Point Beach could continue without endangering the health and safety of the public.

Technical Specifications (TS)

1. T.S. 15.2.3

   Revision of overpower and overtemperature ΔT setpoint equations will permit operation with a single set of coefficients to the setpoint equations at a reactor coolant system pressure of either 2000 or

2250 psia. The revised setpoints* will result in reactor trip at values of process variables equal to or more restrictive than the values which now initiate reactor trip and hence are considered conservative. The use of a single set of coefficients will decrease potential errors associated with "dialing in" coefficient values and is considered desirable. Therefore, we find this change to be acceptable.

2. T.S. 15.5.3

   The removal of individual fuel rods suspected of leaking and their replacement with an inert rod or leaving a water hole, is acceptable subject to the constraints outlined in the mechanical design section of this report.

3. T.S. 15.5.3

   The deleted description of the part length Rod Control Cluster assemblies is acceptable since the deletion reflects removal of the part length rods from the reactor core.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 2, 1977

*The value of $\tau 3$ = 3 sec proposed for use in the overpower ΔT setpoint submitted by the licensee is in error. The correct value is 10 sec. The licensee has agreed to the correction.

REFERENCES

1. Letter from C. S. McNeer (WEPCO) to E. G. Case (NRC), dated July 8, 1977.
2. Letter from Sol Burstein (WEPCO) to E. G. Case (NRC), dated September 20, 1977.
3. Letter from C. S. McNeer (WEPCO) to E. G. Case (NRC), dated July 8, 1977.
4. George, R. A., et al., "Revised Clad Flattening Model," WCAP-8377.
5. Coye, T. E., et. al., "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations", WCAP 8720.
6. Morita, A., et. al., "Topical Report Power Distribution Control and Load Following Procedures," WCAP-8385.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 28 and 32 to Facility Operating Licenses Nos. DPR-24 and DPR-27 issued to Wisconsin Electric Power Company and Wisconsin Michigan Power Company, which revised Technical Specifications for operation of the Point Beach Nuclear Plant Units Nos. 1 and 2, located in the town of Two Creeks, Manitowoc County, Wisconsin. The amendments are effective as of the date of issuance.

These amendments: (1) modify the coefficients of the overpower and overtemperature ΔT setpoint equations, (2) permit limited removal of individual fuel rods suspected of leaking, and (3) delete the description of Part Length Rod Control Cluster assemblies from the Technical Specifications of Point Beach Nuclear Plant Units Nos. 1 and 2.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendment.

For further details with respect to this action, see (1) the applications for amendments dated July 8, 1977 (as supplemented by letter dated September 20, 1977) and July 8, 1977, (2) Amendment No. 28 to License No. DPR-24, (3) Amendment No. 32 to License No. DPR-27, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street N. W., Washington, D. C. and at the University of Wisconsin - Stevens Point Library, ATTN: Mr. Arthur M. Fish, Stevens Point, Wisconsin 54481. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 2 day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors