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Docket Nos. 50-266
and 50-301

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

Dear Mr. Burstein:

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. DPR-24 and Amendment No. 55 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 19, 1979 and modified by letter dated February 3, 1981.

These amendments remove rod bow penalties and requirements related to control rod misalignment and position indication. They also make administrative changes to various parts of section 15.3.10 of the Technical Specifications.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 49 to DPR-24
2. Amendment No. 55 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD	CPB
SURNAME	PMKreutzer	IColburn/pn	RAClark	TMNovak		WJohnston
DATE	4/23/81	4/13/81	4/?/81	4/?/81	4/29/81	5/4/81

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

DISTRIBUTION:
Docket File
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PMKreutzer

Docket No. 50-266 and 50-301

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: WISCONSIN ELECTRIC POWER COMPANY, Point Beach Nuclear Plant, Unit Nos. 1 and 2.

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: ~~Amendment Nos. 49 and 55.~~
Referenced documents have been provided PDR.

Division of Licensing, ORB#3
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#3:DL					
SURNAME →	PMKreutzer/pn					
DATE →	5/5/81					

NRC FORM 102 7—79

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

May 4, 1981

Docket Nos. 50-266
and 50-301

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

Dear Mr. Burstein:

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. DPR-24 and Amendment No. 55 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 19, 1979 and modified by letter dated February 3, 1981.

These amendments remove rod bow penalties and requirements related to control rod misalignment and position indication. They also make administrative changes to various parts of section 15.3.10 of the Technical Specifications.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 49 to DPR-24
2. Amendment No. 55 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

Wisconsin Electric Power Company

cc:
Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
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Mr. Glenn A. Reed, Manager
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Point Beach Nuclear Plant
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Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Ms. Kathleen M. Falk
General Counsel
Wisconsin's Environmental Decade
302 E. Washington Avenue
Madison, Wisconsin 53703

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U.S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 S. Dearborn Street
Chicago, Illinois 60604

cc w/enclosure(s) and incoming
dtd: 12/19/79, 2/3/81

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

Mr. William Guldemond
USNRC Resident Inspectors Office
6612 Nuclear Road
Two Rivers, Wisconsin 54241

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 49
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated December 19, 1979 and modified by letter dated February 3, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49 , are hereby incorporated in the license. The licensee 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: May 4, 1981

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated December 19, 1979 and modified by letter dated February 3, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   (B) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 55 , are hereby incorporated in the license. The licensee 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 4, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.1-4	15.1-4
15.2.1-3	15.2.1-3
15.3.10-1	15.3.10-1
15.3.10-2	15.3.10-2
15.3.10-3	15.3.10-3
15.3.10-3a	-
15.3.10-4	15.3.10-4
15.3.10-5	15.3.10-5
15.3.10-6	15.3.10-6
15.3.10-7	15.3.10-7
15.3.10-8	15.3.10-8
15.3.10-8a	-
15.3.10-9	15.3.10-9
15.3.10-10	15.3.10-10
15.3.10-11	15.3.10-11
15.3.10-12	15.3.10-12
15.3.10-13	15.3.10-13
15.3.10-14	15.3.10-14
15.3.10-15	15.3.10-15
-	15.3.10-16
FIGURE 15.3.10-1	FIGURE 15.3.10-1

2) Cold Shutdown

The reactor is in the cold shutdown condition when the reactor has a shutdown margin of at least 1% $\Delta k/k$ and reactor coolant temperature is $\leq 200°F$.

3) Refueling Shutdown

The reactor is in the refueling shutdown condition when the reactor is subcritical by at least 10% $\Delta k/k$ and T_{avg} is $\leq 140°F$. A refueling shutdown refers to a shutdown to move fuel to and from the reactor core.

4) Shutdown Margin

Shutdown margin is the instantaneous amount of reactivity by which the reactor core would be subcritical if all withdrawn control rods were tripped into the core but the highest worth withdrawn RCCA remains fully withdrawn. If the reactor is shut down from a power condition, the hot shutdown temperature should be assumed. In other cases, no change in temperature should be assumed.

h. Power Operation

The reactor is in power operating condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates greater than 2% of rated power.

i. Refueling Operation

Refueling operation is any operation involving movement of core components (those that could affect the reactivity of the core) within the containment when the vessel head is unbolted or removed.

j. Rated Power

Rated power is here defined as a steady state reactor core output of 1518.5 MWT.

k. Thermal Power

Thermal power is defined as the total core heat transferred from the fuel to the coolant.

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. These curves are based on an $F^N_{\Delta H}$ of 1.58, cosine axial flux shape, and a DNB analysis as described in Section 4.3 of WCAP-8050, "Fuel Densification, Point Beach Nuclear Plant Unit 1 Cycle 2" (including the effects of fuel densification and flattened cladding).

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:

$F^N_{\Delta H} = 1.58 \{1 + 0.2 (1-P)\}$ where P is a fraction of rated power when $P < 1.0$. $F^N_{\Delta H} = 1.58$ when $P \geq 1.0$.

The effects of rod bow have been included in the determination of a conservative value for $F^N_{\Delta H}$. Rod bow effects of up to 14.9% DNBR are offset by credits available from the design limit DNBR, pitch reduction, design thermal diffusion coefficient and the fuel densification power spike, which were previously approved.*

The hot channel factors are also sufficiently large to account for the degree of malpositioning of full-lenth rods that is allowed before the reactor trip setpoints 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.

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* Memorandum from D. F. Ross and D. G. Eisenhut, USNRC, to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977.

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and to core power distribution limits.

Objective

To insure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical rod cluster control assembly (RCCA) ejection, and (3) an acceptable core power distribution during power operation.

Specification

A. Bank Insertion Limits

1. When the reactor is critical, except for physics tests and control rod exercises, the shutdown banks shall be fully withdrawn.

2. When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. Exceptions to the insertion limit are permitted for physics tests and control rod exercises.

3. The shutdown margin shall exceed the applicable value as shown in Figure 15.3.10-2 under all steady-state operating conditions from 350°F to full power. An exception to the stuck RCCA component of the shutdown margin requirement is permitted for physics tests.

4. Except for physics tests a shutdown margin of at least 1% $\Delta k/k$ shall be maintained when the reactor coolant temperature is less than 350°F.

5. When the reactor is in the hot shutdown condition or during any approach to criticality, except for physics tests, the critical rod position shall not be lower than the insertion limit for zero power. That is, if the control rods were withdrawn in normal sequence with no other reactivity change, the reactor would not be critical until the control banks were above the insertion limit.

B. Power Distribution Limits

1. a. Except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \le \frac{(2.32)}{P} \times K(Z)$ for $P > .5$

$F_Q(Z) \le 4.64 \times K(Z)$ for $P \le .5$

$F^N_{\Delta H} \le 1.58 \times \{1 + 0.2\ (1-P)\}$

Where P is the fraction of full power at which the core is operating, K(Z) is the function in Figure 15.3.10-3 and Z is the core height location of F_Q.

b. Following a refueling shutdown prior to exceeding 90% of rated power and at effective full power monthly intervals thereafter, power distribution maps using the moveable incore detector system shall be made to confirm that the hot channel factor limits are satisfied. The measured hot channel factors shall be increased in the following way:

(1) The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

(2) The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error.

c. If a measured hot channel factor exceeds the full power limit of Specification 15.3.10.B.1.a, the reactor power and power range high setpoints shall be reduced until those limits are met. If subsequent flux mapping cannot, within 24 hours, demonstrate that the full power hot channel factor limits are met, the overpower and overtemperature ΔT trip setpoints shall be similarly reduced and reactor power limited such that Specification 15.3.10.B.1.a above is met.

2. a. The target flux difference as defined in the basis shall be measured at least quarterly. A target flux difference update value shall be determined monthly by measurement, or by linear interpolation between the last measured value and 0% at end of cycle life (that is when the boron concentration in the coolant is zero ppm), or by extrapolation of the last three measured points. The target flux difference and its associated alarm setpoints need not be updated if the update value for full power target flux difference is within ±0.5% of the presently employed full power target flux difference value.

b. Except for physics testing, excore detector calibration (including recovery), or as modified below, the indicated axial flux difference shall be maintained within a range of +6 and -9 percent of the target flux difference. This is defined as the target band.

c. At a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent of rated power.

d. At a power level no greater than 90 percent of rated power,

(1) The indicated axial flux difference may deviate from its +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1% and +1% for each 2% of

rated power below 90%. If the cumulative time exceeds one hour in any 24 hour period, then the reactor power shall be reduced immediately to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55% of rated power.

(2) A power increase to a level greater than 90% of rated power is contingent upon the indicated axial flux difference being within its target band.

e. At a power level no greater than 50 percent of rated power,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50% of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period. One half of the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90% of rated power.

f. Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 15.3.10.B.2.c or the flux difference-time requirement of 15.3.10.B.2.d(1). If the alarms are temporarily out-of-service, the axial flux difference shall be noted and conformance with the limits assessed every hour for the first 24 hours, and half-hourly thereafter.

3. Except for physics tests, whenever the indicated quadrant power tilt exceeds 2% the tilt condition shall be eliminated within two hours or the following actions shall be taken:

   a. Reduce core power level and the power range high flux setpoint two percent of rated values for every percent of indicated quadrant power tilt.

   b. If the tilt is not corrected within 24 hours, but the hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission. Return to full power is permitted, providing the hot channel factors are not exceeded.

   c. If the design hot channel factors for rated power are exceeded or not determined within 24 hours, the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip set-points shall be reduced by the equivalent of 2% power for every percent of quadrant power tilt.

   d. The excore nuclear instrumentation system serves as the primary quadrant power tilt alarm. If the alarm is not functional for two hours, backup methods of assuring that the quadrant power tilt is acceptable shall be used. These methods include hand calculations, incore thermocouples using either a computer or manual calculations or incore detectors.

   e. When one power range channel is inoperable and thermal power is greater than 75% of rated thermal power, the quadrant power tilt shall be confirmed as acceptable by use of the movable incore detectors at least once per 12 hours.

C. <u>Inoperable Rod Cluster Control Assembly (RCCA)</u>

   1. An RCCA shall be considered inoperable if one or more of the following occurs:

a. The RCCA does not drop upon removal of stationary gripper coil voltage.

b. The RCCA does not step in properly when the proper voltage sequences are applied to the control rod drive mechanism coils. It shall then be assumed inoperable until it has been tested to verify that it does drop.

c. If the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, and the rod position indicator channel shows a misalignment of 15 inches. The RCCA shall be assumed inoperable until it has been tested to verify that it does step properly.

d. If the bank demand position is between 215 steps and 30 steps, and the rod position indicator channel shows a misalignment of 7.5 inches. The RCCA shall be assumed inoperable until it has been tested to verify that it does step properly.

2. Specification 15.3.10.C.1.b can be modified by the following:

a. If an RCCA does not step in upon demand, up to six hours is allowed to determine whether the problem with stepping is an electrical problem. If the problem cannot be resolved within six hours, the RCCA shall be assumed inoperable until it has been verified that it will step in or would drop upon demand.

b. If more than one RCCA does not step in, apparently due to electrical problems, the situation shall be rectified or clearly defined that it is an electrical problem and the RCCAs are capable of dropping upon demand or an orderly shutdown shall commence within six hours.

3. No more than one inoperable RCCA shall be permitted during sustained power operation.

4. When it has been determined that an RCCA does not drop on removal of stationary gripper coil voltage, the shutdown margin shall be maintained by boration as necessary to compensate for the withdrawn worth of the inoperable RCCA. If sustained power operation is anticipated, the insertion limit shall be adjusted to reflect the worth of the inoperable RCCA.

D. Misaligned or Dropped RCCA

1. If the rod position indicator channel is functional and the associated RCCA is more than 7.5 inches indicated out of alignment with its bank and cannot be aligned when the bank is between 215 steps and 30 steps, then unless the hot channel factors are shown to be within design limits as specified in Section 15.3.10.B-1 within eight (8) hours, power shall be reduced to less than 75% of rated power. When the bank position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the allowable indicated misalignment is 15 inches.

2. To increase power above 75% with an RCCA more than 7.5 inches indicated out of alignment with its bank when the bank position is between 215 steps and 30 steps, an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 15.3.10.B. When the bank position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the allowable indicated misalignment is 15 inches.

3. If it is determined that the apparent misalignment or dropped RCCA indication was caused by rod position indicator channel failure, sustained power operation may be continued if the following conditions are met:

   a. For operation between 10% power and rated power, the position of the RCCA(s) with the failed rod position indicator channel(s) will be checked indirectly by core instrumentation (excore detectors, and/or thermocouples, and/or moveable incore detectors) every shift and after associated bank motion exceeding 24 steps in one direction.

   b. For operation below 10% of rated power, no special monitoring is required.

E. RCCA Drop Times

1. At operating temperature and full flow, the drop time of each RCCA shall be no greater than 1.8 seconds from the loss of stationary gripper coil voltage to dashpot entry.

Basis

Insertion Limits and Shutdown Margin

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration.

During power operation, the shutdown banks are fully withdrawn and control of reactor power is by the control banks. The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time and assume the highest worth control rod remains fully withdrawn. A 10% margin in reactivity worth of the control rods is included to assure meeting the assumptions used in the accident analysis. So a reactor trip occurring during power operation will put the reactor into the hot shutdown condition. In addition, the insertion limits provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection and provide for acceptable nuclear peaking factors. The specified control rod insertion limits take into account the effects of fuel densification. The rods are withdrawn in the sequence of A, B, C, D with overlap between banks. The overlap between successive control banks is provided to compensate for the low differential rod worth near the top and bottom of the core.

When the insertion limits are observed and the control rod banks are above the solid lines shown on Figure 15.3.10-1, the shutdown requirement is met. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Figure 15.3.10-2 shows the shutdown margin equivalent to 2.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses assume 1% or greater reactivity shutdown margin. Shutdown margin calculations include the effects of axial power distribution. One may assume no change in core poisoning due to xenon, samarium or soluble boron.

Part length rod insertion is not permitted, thus eliminating certain adverse power shapes which might occur during power operation. The part length rods have been removed from the core.

Power Distribution

Design criteria have been chosen which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must not be less than 1.30 in normal operation or in short-term transients.

In addition to the above, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss of coolant accident analysis based upon the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident.

To aid in specifying the limits on power distribution, the following hot channel factors are defined:

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. Imposed limits pertain to the maximum $F_Q(Z)$ in the core.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically, the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a fuel rod to the average fuel rod power. Imposed limits pertain to the maximum $F^N_{\Delta H}$ in the core, that is the fuel rod with the highest integrated power. It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat flux is obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided the following conditions are observed, the hot channel factor limits will be met:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position, when the bank demand position is between 30 steps and 215 steps. 22.5 inches misalignment is allowed when the bank position is less than or equal to 30 steps, or, when the bank position is greater than or equal to 215 steps, due to the small worth and consequential effects of an individual rod misalignment.

2. Control rod banks are sequenced with overlapping banks as described in Figure 15.3.10-1.

3. The full-length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation of $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factor limits are met. In Specification 15.3.10.B.1.a, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 15.3.10-3 consistent with the Technical Specifications on power distribution control as given in Section 15.3.10 was used in the LOCA analysis. The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.58/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affect F_Q, (b) while the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system. The $F^N_{\Delta H}$ limits in Specification 15.3.10.B.1.a take into account the effects of rod bow. This is further explained in the Basis on page 15.2.1-3.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based upon measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

Axial Power Distribution

The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/fractional power).

The full power target flux difference is defined as that indicated flux difference of the core in the following condition: equilibrium xenon (little or no oscillation) and with the full-length rod control rod bank more than 190 steps withdrawn (i.e., the normal full power position). Values for all other core power levels are obtained by multiplying the full power value by the factional power. At zero power the target flux difference is 0%. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of +6 and -9 percent ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides three methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during reduced power operation. This is because xenon distribution control at reduced power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at reduced power. Strict control of the flux difference is not possible during certain physics tests or during required periodic excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations. This is acceptable due to the increased core monitoring performed as part of the tests and low probability of a significant accident occurring during these operations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band. However, to simplify the specification for operation up to 90% of full power, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This insures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band.

For normal operation and anticipated transients, the core is protected from overpower and minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a pre-condition; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant Tilt

The excore detectors are somewhat insensitive to disturbances near the core center such as misaligned inner control rods. It is therefore possible that a five percent tilt might actually be present in the core when the excore detectors respond with a two percent indicated quadrant tilt. On the other hand, they are overly responsive to disturbances near the periphery.

Tilt restrictions are not applicable during the startup and initial testing of a reload core which may have an inherent tilt. During this time sufficient testing is performed at reduced power to verify that the hot channel factor limits are met and the nuclear channels are properly aligned.

The excore detectors are normally aligned indicating no quadrant power tilt because they are used to alarm on a rapidly developing tilt. Tilts which develop slowly are more accurately and readily discerned by incore measurements.

The excore detectors serve as the prime indication of a quadrant power tilt. If a channel fails, is out-of-service for testing, or is unreliable, two hours is a short time with respect to the probability of an unsafe quadrant power tilt developing. Two hours gives the operating personnel sufficient time to have the problem investigated and/or put into operation one of several possible alternative methods of determining tilt.

Inoperable RCCA

An inoperable rod imposes additional demands on the operators. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden.

From operating experience to date, an RCCA which steps in properly will drop when a trip signal occurs because the only force acting to drive the rod in is gravity. When it has been determined that a rod does not drop, extra shutdown margin is gained by boration or by adjusting the insertion limit to account for the worth of the inoperable control rod.

Further experience indicates that control rods which do not step are usually affected by electrical problems. That is, normally the problem is in the rod control cabinets. If operability cannot be restored, the RCCA will be declared inoperable and corrective actions can be taken to compensate for the associated reduction in shutdown margin. If there is more than one RCCA affected, an orderly shutdown would be started. Such an evolution would have to be performed in a deliberate manner without undue pressure on the operating personnel because of the unusual techniques to be used to accommodate the reactivity changes associated with the shutdown.

Misaligned RCCAS

The various control rod banks (shutdown banks and control banks, A, B, C, and D) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Direct information on rod position indication is provided by two methods: A digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position. The rod position indicator channel has a demonstrated accuracy of 5% of span (±7.2 inches). Therefore, an analysis has been performed to show that a misalignment of 15 inches cannot cause design hot channel factors to be exceeded. A single fully misaligned RCCA, that is, an RCCA 12 feet out of alignment with its bank, does not result in exceeding core limits in steady-state operation at power levels less than or equal to rated power. In other words, a single dropped RCCA is allowable from a core power distribution viewpoint. If the misalignment condition cannot be readily corrected, the specified reduction in power to 75% will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The eight (8) hour permissible limit on rod misalignment at rated power is short with respect to the probability of an independent accident.

Because the rod position indicator system may have a 7.5 inch error when a misalignment of 15 inches is occurring, the Specification allows only a 7.5 inch indicated misalignment. However, when the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the consequences of a misalignment are much less severe. The differential worth of an individual RCCA is less, and the resultant purturbation on power distributions is less than when the bank is in its high differential worth region. At the top and bottom of the core, an indicated 15 inch misalignment may be representing an actual misalignment of 22.5 inches.

The failure of an LVDT in itself does not reduce the shutdown capability of the rods, but it does reduce the operator's capability for determining the position of that rod by direct means. The operator has available to him the excore detector recordings, incore thermocouple readings and periodic incore flux traces for indirectly determining rod position and flux tilts should the rod with the inoperable LVDT become malpositioned. The excore and incore instrumentation will not necessarily recognize a misalignment of 15 inches because the concommitant increase in power density will normally be less than 1% for a 15 inch misalignment. The excore and incore instrumentation will, however, detect any rod misalignment which is sufficient to cause a significant increase in hot channel factors and/or any significant loss in shutdown capability. The increased surveillance of the core if one or more rod position indicator channels is out-of-service serves to guard against any significant loss in shutdown margin or margin to core thermal limits.

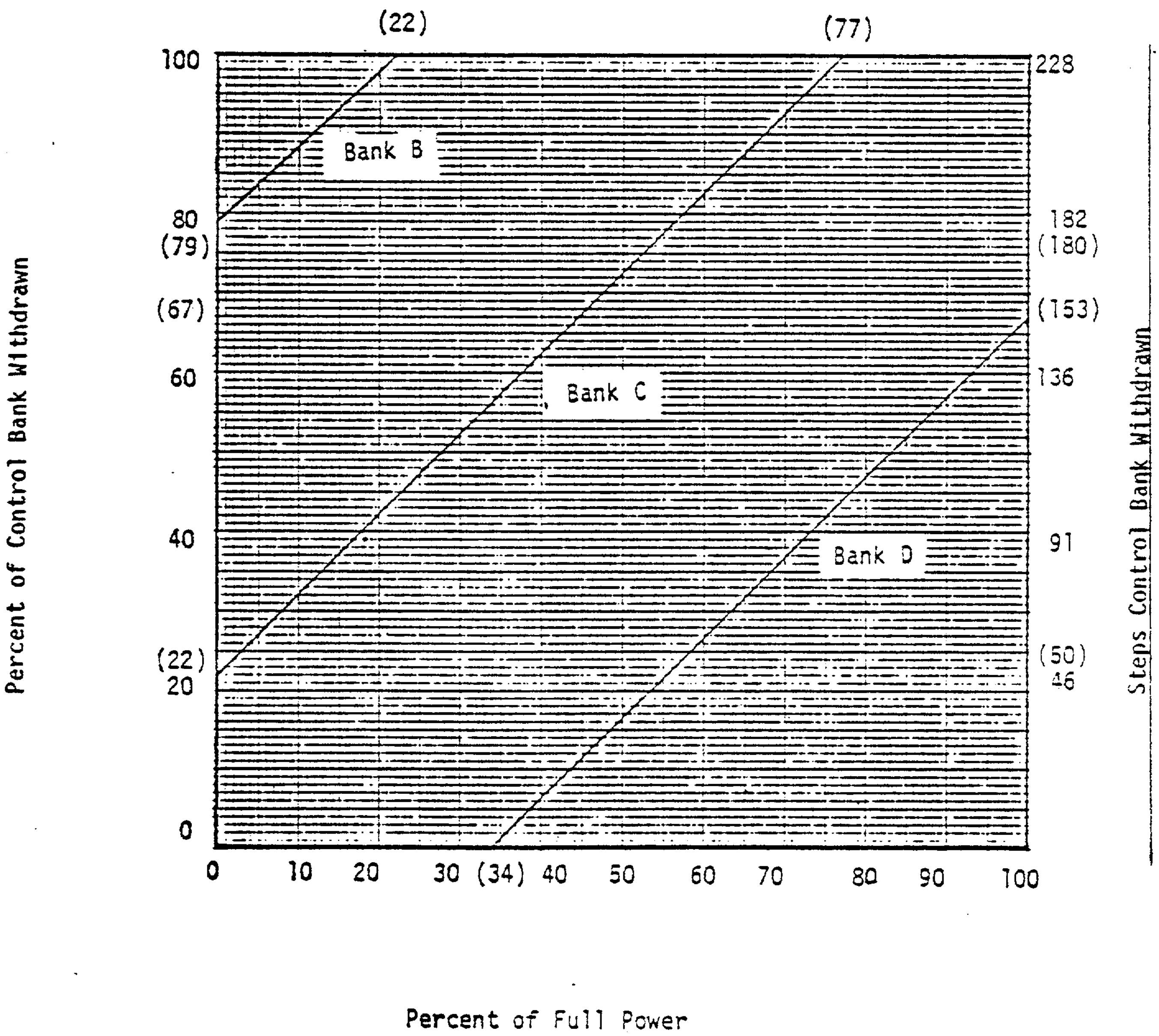
The history of malpositioned RCCA's indicates that in nearly all such cases, the malpositioning occurred during bank movement. Checking rod position after bank motion exceeds 24 steps will verify that the RCCA with the inoperable LVDT is moving properly with its bank and the bank step counter. Malpositioning of an RCCA in a stationary bank is very rare, and if it does occur, it is usually gross slippage which will be seen by external detectors. Should it go undetected, the time between the rod position checks performed every shift is short with respect to the probability of occurrence of another independent undetected situation which would further reduce the shutdown capability of the rods.

Any combination of misaligned rods below 10% rated power will not exceed the design limits. For this reason, it is not necessary to check the position of rods with inoperable LVDT's below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

FIGURE 15.3.10-1

CONTROL BANK INSERTION LIMITS

POINT BEACH UNITS 1 AND 2



Unit 1 - Amendment No. ~~25~~, 49
Unit 2 - Amendment No. ~~30~~, 55

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

Introduction:

In a letter dated December 19, 1979 and supplemented by letter dated February 3, 1981 Wisconsin Electric Power Company (WEPCO) requested changes to the Technical Specifications of the Point Beach Nuclear Plant Unit Nos. 1 and 2. These changes would remove the rod bow penalty and requirements related to rod cluster control assembly (RCCA) misalignment and position indication. They would also make various administrative changes to section 15.3.10 of the Technical Specifications.

Discussion and Evaluation:

The rod bow penalty currently in effect for Point Beach Units 1 and 2 was proposed under Technical Specification Change Request Number 38 dated January 6, 1977 and was approved by amendments dated May 4, 1977. The penalty is to offset the effects of a bowed rod on critical heat flux and is calculated as a function of region-average fuel burnup and is expressed as the following value:

Burnup (MWd/MtU)	Reduction in $F_{\Delta H}$ (%)
0-15,000	0-2 ramp
15,000-24,000	4
>24,000	6

Subsequent to NRC approval of the aformentioned $F_{\Delta H}$ penalty, Westinghouse submitted test results on the effects of a bowed rod on critical heat flux for Westinghouse PWR's. These results showed a significant reduction in the presupposed DNBR penalty on the basis of a new small gap (85% closure) test. The NRC then approved, for Westinghouse applications, the use of a less conservative reduction-in-DNBR versus gap-closure model.

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WEPCO has requested elimination of the $F_{\Delta H}$ penalty because of (a) the proposed use of the less conservative reduction-in-DNBR versus gap-closure model and (b) the application of generic thermal margin credits that are available to offset DNBR reductions due to fuel rod bowing. The NRC has generically approved the new DNBR model, and we find WEPCO's request to apply the model to the Point Beach analyses to be acceptable.

In regard to using thermal margin credits to offset the residual $F_{\Delta H}$ penalty that remains after application of the new DNBR model, the staff has made an independent calculation to determine the magnitude of margin required. This calculation was performed by way of the generic methodology for interim rod bowing analyses. Specifically, the approved Westinghouse rod bow magnitude correlation was used in conjunction with the new DNBR model. The resulting margin needed to offset the reduction in DNBR was found to be zero until a burnup of 8660 MWd/MtU, whereupon the required margin monotonically increases to the following values at a burnup of 33,000 MWd/MtU:

(1) 12.5% for all loops in service and
(2) 14.9% for loss-of-flow accident analyses.

WEPCO has identified a total of 18.1% DNBR margin credits that are available from the following sources:

1. 4.8% from using 1.30 DNBR limit in analysis rather than allowed 1.24 design limit.
2. 3.3% from pitch reduction.
3. 3.0% from using 0.019 thermal diffusion coefficient in analysis rather than allowed 0.038 value.
4. 7.0% from new densification model that eliminates power spike effect on DNB.

These margin credits have been previously approved for the Point Beach type of fuel design, and WEPCO has stated that these credits are to be used solely for this application. Additionally, the Basis to the Technical Specifications is being revised to reflect the basis for discontinuing rod bow penalty calculations, which is that sufficient generic thermal margin credits be maintained to offset the rod bow penalty.

Based on the above evaluation, we find that the combination of the new reduction-in-DNBR versus gap-closure model with the generic thermal margin credits is sufficiently large to completely eliminate the rod bow $F_{\Delta H}$ penalty. Therefore, we agree with the WEPCO proposal to delete the $F_{\Delta H}$ penalty from the Technical Specifications.

In regard to the second proposed change, the results of our evaluation of the proposed Technical Specification changes related to inoperable and misaligned control rods (Sections 15.3.10.C.1.c and d, 15.3.10.C.2.a and b, and 15.3.10.D.1 and 2) are in agreement with those of the generic resolution of control rod position indicating system requirements for Westinghouse

PWRs, and are therefore acceptable.

The proposed administrative changes to Section 15.3.10 of the Technical Specifications were submitted to reorganize and clarify this section of the Technical Specifications including updating of terminology and removal of references to equipment no longer applicable to the Point Beach Nuclear Plant facility such as references to part length control rods which have been removed from the core as per previous approval. We have reviewed the administrative changes to this section and find that they do not change the meaning or intent of the Technical Specifications and are therefore acceptable.

Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 4, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 49 to Facility Operating License No. DPR-24, and Amendment No. 55 to Facility Operating License No. DPR-27 issued to Wisconsin Electric Power Company (the licensee), which revised Technical Specifications for operation of Point Beach Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendments are effective as of the date of issuance.

The amendments remove rod bow penalties and requirements related to control rod misalignment and position indication. They also make administrative changes to various parts of section 15.3.10 of the Technical Specifications.

The application for the amendments complies 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 amendments.

For further details with respect to this action, see (1) the application for amendments dated December 19, 1979 as revised by letter dated February 3, 1981, (2) Amendment Nos. 49 and 55 to License Nos. DPR-24 and DPR-27, and (3) 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. 20555, and at the Joseph Mann Library, 1516 16th Street, Two Rivers, Wisconsin 54241. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of May, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing