December 24, 1975

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 Amendment No. 14 to Facility Operating License No. DPR-24 and Amendment No. 18 to Facility Operating License No. DPR-27 for Point Beach Nuclear Generating Units 1 and 2. These amendments include Changes Nos. 19 and 24 to the Technical Specifications and are in response to your requests dated September 6, 1974, June 24, 1975, and October 6, 1975, and Supplements dated December 6, 1974, May 7, November 5 and 26, and December 15 and 18, 1975.

These amendments: (1) incorporate operating limits in the Technical Specifications for the facilities based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50, and (2) modify certain Unit 1 operating limits to reflect the results of the cycle 4 core performance analysis.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the Point Beach Nuclear Generating Units 1 and 2 in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

See Shelf

OFFICE ➤						
SURNAME ➤				Q		
DATE ➤						

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and 50-301

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ATTN: Mr. Sol Burstein
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These amendments: (1) incorporate operating limits in the Technical Specifications for the facilities based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50, and (2) modify certain Unit 1 operating limits to reflect the results of the cycle 4 core performance analysis.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the Point Beach Nuclear Generating Units 1 and 2 in proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

You will note that the Technical Specifications for Unit 1 have an additional rod bow penalty which has been applied to the radial peaking factor, $F_{\Delta H}$. In addition, the Technical Specifications require removing AC power from the accumulator isolation valves (MOV-841 A&B) for both Units 1 and 2 to meet the single failure criterion. Moreover, we require that your ECCS Emergency Operating Procedures be modified, as specified in the enclosed Safety Evaluation by March 1, 1976. These measures were discussed with and agreed to by your staff in telephone conversations of November 14, and December 5 and 12, 1975.

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

Sincerely,

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

Enclosures:
1. Amendment No. 14
2. Amendment No. 18
3. Negative Declaration
4. Environmental Impact Appraisal
5. Safety Evaluation
6. Federal Register Notice

cc w/enclosures:
See next page

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Refer to memos from Stello to Goller of 12/15 and 12/16/75

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Wisconsin Michigan Power Company
Wisconsin Electric Power Company

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Public Service Commission
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Hill Farms State Office Building
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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. 18
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated September 6, 1974, June 24, 1975, and October 6, 1975, and Supplements dated December 6, 1974, May 7, 1975, November 5, 1975, November 26, 1975, and December 15, 1975, 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 1;

   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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B. of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 24".

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 24 to the
Technical Specifications

Date of Issuance: December 24, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 18

CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NO. 50-301

Replace page 15.3.3-1, page 15.3.3-2, pages 15.3.10-1 through 15.3.10-13, and Figure 15.3.10-3 with the attached revised pages. Add page 15.3.3-2a, page 15.3.10-14, and page 15.3.10-15.

15.3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, AND CONTAINMENT SPRAY

Applicability:

Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray.

Objective:

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident.

Specification:

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made critical, except for low temperature physics tests, unless the following conditions associated with that reactor are met:

   a. The refueling water tank contains not less than 275,000 gal. of water with a boron concentration of at least 2000 ppm.

   b. Each accumulator is pressurized to at least 700 psig and contains at least 1100 ft^3 but no more than 1136 ft^3 of water with a boron concentration of at least 2000 ppm. 24
   Neither accumulator may be isolated.

   c. Two safety injection pumps are operable.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

f. The isolation valves in the discharge header of the high head safety injection system are in the open position.

g. All valves, interlocks and piping associated with the above components and required to function during accident conditions, are operable.

h. During conditions of operation with reactor coolant system pressure in excess of 1000 psig the source of AC power shall be removed from the accumulator isolation valves MOV-841 A & 24
B at the motor control center and the valves shall be open.

i. Power may be restored to MOV-841 A & B for the purpose of valve testing or maintenance providing the testing and maintenance is completed and power is removed within 4 hours.

2. During power operation, the requirements of 15.3.3.A-1 may be modified to allow one of each of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 15.3.3.A-1 within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 15.3.3.A-1 are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition.

a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. The other safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours. The other residual heat removal pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

c. One residual heat exchanger may be out of service for a period of no more than 48 hours.

d. Any valve in the system, required to function during accident conditions, may be inoperable provided repairs are completed within 24 hours. Prior to initiating repairs, all valves in the system that provide the duplicate function shall be tested to demonstrate operability.

e. One accumulator may be isolated for a period of up to one hour to permit a check valve leakage test.

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

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

Objective

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

Specification

A. Control Bank Insertion Limits

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

2. When the reactor is critical, the control rods shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1 and the shutdown margin with allowance for a stuck rod shall exceed the applicable value shown on Figure 15.3.10-2 under all steady-state operating conditions from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn and assuming no changes in xenon, boron, or part-length rod position. Exceptions to the insertion limit and stuck rod requirements only are permitted for physics tests and control rod exercises.

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3. The part-length rods shall be fully withdrawn from the core, except for physics testing.

4. When the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved if the control rods were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

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:

$F_Q(Z) \leq \left(\frac{2.32}{P}\right) \times K(Z)$ for $P > .5$ 24

$F_Q(Z) \leq 4.64 \times K(Z)$ for $P \leq .5$ 24

$F_{\Delta H}^{N} \leq 1.58 \times (1 + 0.2\ (1-P))$

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 core loading prior to exceeding 90% of rated power and at effective full power monthly intervals thereafter, power distribution maps using the movable incore detector system shall be made to confirm that 24 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, reactor power and power range high setpoint shall be reduced until the limits in B.1.a 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 B.1.a above is met.

2. a. The target flux difference as defined in the basis shall be measured at least quarterly and updated monthly. It may be updated by measurement, or by linear interpolation between

the last measured value and 0% at end of cycle life, or by extrapolation of the last three measured points. The target flux difference varies with power level in a linear fashion with 0% flux difference at 0% power.

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. 24

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, then the reactor power shall be 24
   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 hours 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.3.c or the flux difference-time requirement of 15.3.10.B.3.d(1). If the alarms are temporarily out of service, the axial flux difference shall be logged, 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 ratio exceeds 1.02, the tilt condition shall be eliminated 24
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 power tilt ratio exceeding 1.0.

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 as an abnormal occurrence to the Nuclear Regulatory Commission.

c. If the design hot channel factors for rated power are exceeded or not determined the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% power for every 1% quadrant tilt.

C. Inoperable Control Rods

1. A control rod shall be considered inoperable if the following occurs:

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

   b. The rod does not step in properly. It shall be assumed inoperable until it has been tested to verify that it does drop.

   c. The rod is shown by the rod position indicator channel to be misaligned by more than 15 inches. It shall be assumed inoperable until it has been tested to verify that it does step in properly or that it does drop.

2. No more than one inoperable control rod shall be permitted during sustained power operation.

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

D. Misaligned or Dropped Control Rod

1. If the rod position indicator channel is functional and the associated part-length or full-length control rod is more than 15 inches out of alignment with its bank and cannot be aligned, 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.

2. To increase power above 75% with a part-length or full length control rod more than 15 inches out of alignment with its bank 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.

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

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

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

E. Rod Drop Times

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

Basis

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 groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition. 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. The rods are withdrawn in the sequence of A, B, C, D with overlap between banks and a 10% margin in reactivity worth of the control rods to assure meeting the assumptions used in the accident analysis. In addition, they 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 solid lines shown on Figure 15.3.10-1 meet the shutdown requirement. 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. Early in core life, less shutdown margin is required, and 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 are based on 1% reactivity shutdown margin.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

The overlap between successive control banks is provided to compensate for the low differential rod worth near the top and bottom of the core. Positioning of the part-length rods is governed by the requirement to maintain the axial power shape within specified limits or to accept an automatic cutback of the overpower ΔT and overtemperature ΔT setpoints (see Specification 15.2.3).

Part-length rod insertion is not permitted, thus eliminating certain adverse power shapes which might occur during power operation. Part-length rod insertion for the purpose of physics testing is allowed because of increased surveillance.

The various control rod banks (shutdown rods, control banks A, B, C, D, and part-length rods) 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, a 15 inch indicated misalignment of 15 inches cannot cause design hot channel factors to be exceeded, and complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at rated power. 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. 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 core 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 rods indicates that in nearly all the cases when the rods have been malpositioned, the malpositioning occurred when the bank was moving. The checking of the rod position after bank motion exceeding 24 steps will verify that the rod with the inoperable LVDT is moving properly with its bank and according to the bank step counter. Malpositioning of a rod in a bank which is not moving is very rare, and, if it does occur, it is usually gross slippage or complete rod dropping which will be seen by external detectors. Should it go undetected, the checking of the rod position every shift is short with respect to the probability 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, the position of the rods with inoperable LVDT's need not be checked below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

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, a control rod 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 margin is gained by boration or by adjusting the insertion limit to account for the worth of the inoperable control rod.

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. 24

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 maximum 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.

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.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat flux are 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 $F_{\Delta H}^N$.

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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.
2. Control rod banks are sequenced with overlapping banks as described in Figure 15.3.10-1.
3. The full-length and part-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. 24

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 1996°F corresponding to a 204°F margin to the 2200°F limit. 24

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 movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is 10 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.58/1.10$. 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 affecting F_Q, (b) 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 movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of 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. 2

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 that indicated flux difference of the core in the following condition; equilibrium xenon (little or no oscillation) with part-length rods withdrawn from the core 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 fractional power. 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 part power operation. This is because xenon distribution control at 24
part 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 part 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 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 bank 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. 24

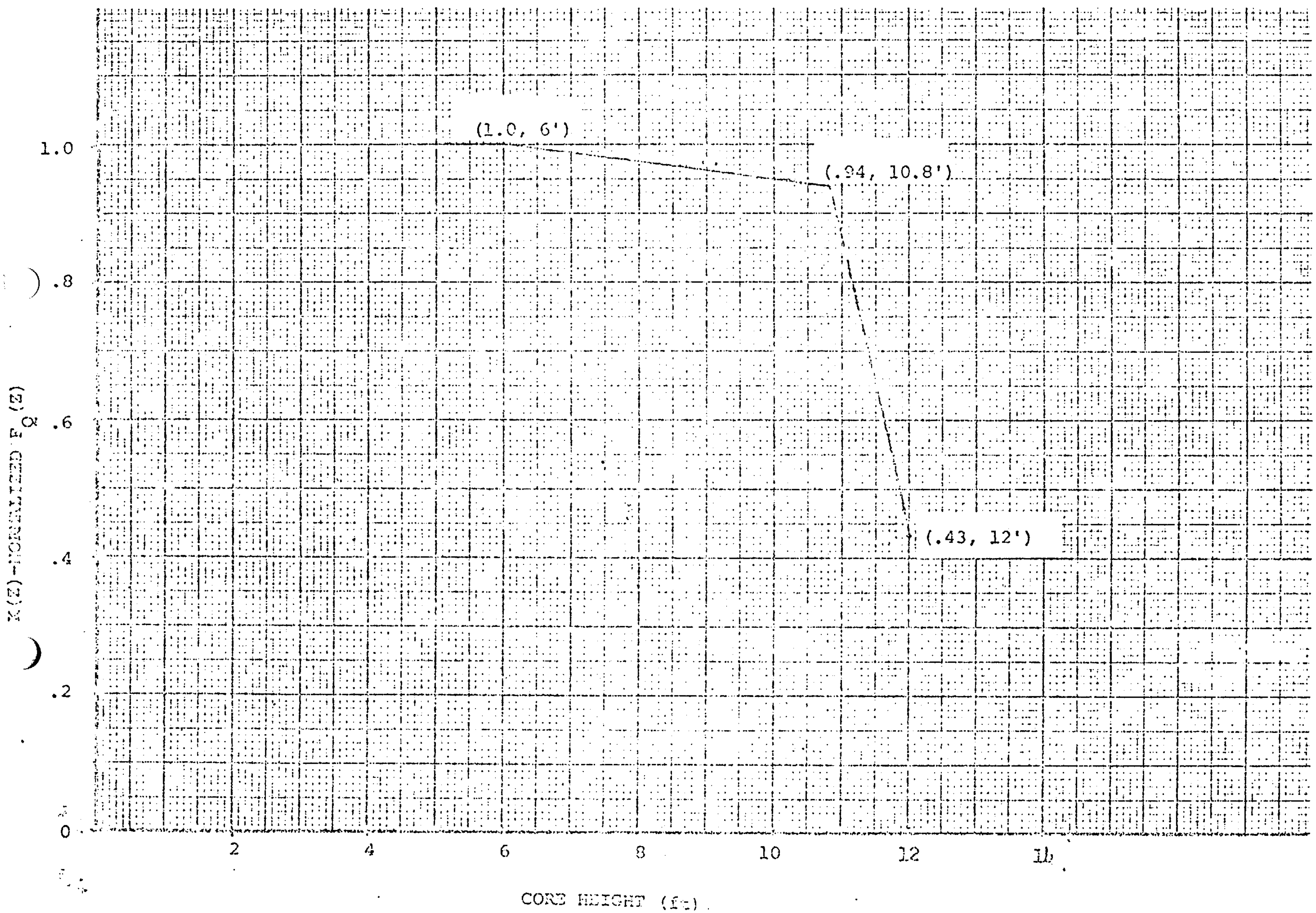
For normal operation and anticipated transients, the core is protected from overpower and minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition, however, operator error and equipment malfunctions are separately assumed to lead to the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

A two percent quadrant tilt allows that a five (5) percent tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods and an error allowance. No increase in F_Q occurs with tilts up to five percent because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The 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 24
the hot channel factor limits are met and the nuclear channels are properly aligned.

FIGURE 15.3.10-3

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE



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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING SUPPORTING AMENDMENT NOS. 14 AND 18 TO DPR-24 AND DPR-27 CHANGE NOS. 19 AND 24 TO THE TECHNICAL SPECIFICATIONS WISCONSIN ELECTRIC POWER COMPANY AND WISCONSIN MICHIGAN POWER COMPANY POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letter dated September 6, 1974, Wisconsin Electric Power Company and Wisconsin Michigan Power Company (licensees in the above captioned dockets) requested an amendment to Facility Operating Licenses DPR-24 and DPR-27 for the Point Beach Nuclear Plant Units 1 and 2. The licensees provided supplemental information by letters dated December 6, 1974 and May 7, 1975 in response to requests from the staff of the Nuclear Regulatory Commission (the Commission). By letter dated June 24, 1975, the licensees submitted a reevaluation of the proposed amendment in response to the Commission's December 27, 1974 Order for Modification of License. At the request of the NRC staff, the licensees supplemented their reevaluation with letters dated November 26, 1975 and December 15, 1975.

The proposed changes would revise the limiting conditions for operation of the Point Beach Units 1 and 2 as a result of implementing the "Acceptance Criteria for the Emergency Core Cooling System for Light Water Nuclear Power Reactors" (ECCS) as specified in Section 50.46 of 10 CFR Part 50. The licensees are presently permitted to operate Point Beach Units 1 and 2 at power levels up to a total of 3,036 megawatts thermal. The proposed change is being made in conjunction with a partial refueling of Unit 1.

2. Environmental Impacts of Proposed Action

The proposed change to incorporate the ECCS Acceptance Criteria would not result in an increase or decrease in power levels of Point Beach Units 1 and 2. The restrictions on heat generation rates will require careful control of fuel operating history; however, there should be no

AMERICAN REVOLUTION BICENTENNIAL 1776-1976 reduction in total burnup resulting from the revised ECCS evaluation methods.

In the absence of any significant change in power levels, there would be no change in cooling water requirements. Further, there would be no change in radioactive effluents or thermal effluents from normal operation or post accident conditions.

No environmental impacts are expected other than those described in the Commission's Final Environmental Statement for the Point Beach Nuclear Plant, issued May 1972. The Commission's calculated releases of radioactive effluents, both gaseous and liquid, are based on expected release rates from the total quantity of nuclear fuel within the reactor units. The proposed action would not affect the total quantity of fuel used at Point Beach. No increases in radiation doses to man or other biota are expected. It is not anticipated that the issuance of this change to the Appendix A Technical Specifications would affect the cost-benefit balance nor would it require changes in the Environmental Technical Specifications in Appendix B of the licenses.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than those impacts described in the Final Environmental Statement, issued May 1972. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action needs to be prepared and that a negative declaration to this effect is appropriate.

DATE: DEC 1 3 1975

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE NOS. DPR-24 AND DPR-27

POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to Facility Operating License Nos. DPR-24 and DPR-27 for Point Beach Nuclear Plant Units 1 and 2 in Manitowoc County, Wisconsin. These changes would authorize the licensees, Wisconsin Electric Power Company and Wisconsin Michigan Power Company, to operate the Point Beach Nuclear Plant Units 1 and 2 with certain revisions to the present limiting conditions for operation specified in Appendix A of the referenced licenses. These revisions would result from the implementation of the Acceptance Criteria For the Emergency Core Cooling Systems For Light Water Nuclear Power Reactors (ECCS) as specified in Section 50.46 of 10 CFR Part 50. The proposed change would be made in conjunction with a partial refueling of Unit 1. No revisions to the Environmental Technical Specifications (Appendix B) were requested in connection with this proposed change.

The Commission's Division of Reactor Licensing has appraised the expected environmental impact of the proposed changes. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than those impacts described in the Commission's Final Environmental Statement, issued May 1972, concerning the operation of Point Beach Units 1 and 2.

The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Documents Department, Library, University of Wisconsin--Stevens Point, Stevens Point, Wisconsin 54481.

Dated at Rockville, Maryland, this 16th day of December 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Gordon K. Dicker

Gordon K. Dicker, Chief
Environmental Projects Branch 2
Division of Reactor Licensing

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NOS. 14 AND 18 TO LICENSES DPR-24/27

(CHANGE NOS. 19 AND 24 TO THE TECHNICAL SPECIFICATIONS)

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNITS 1/2

DOCKETS NOS. 50-266/301

Introduction

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR §50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading, "the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, §50.46". The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results.

Wisconsin Electric Power Company (WEPCO) has requested a license amendment which will allow Point Beach Unit 1 operation following reload for core Cycle 4. This license amendment request included analyses of the applicability of previously performed safety analyses and proposed Technical Specification changes based on the Unit 1 core configuration for Cycle 4.

As required by our Order of December 27, 1974, WEPCO has also submitted an ECCS reevaluation and related Technical Specifications. The ECCS reevaluation applies also to Point Beach Unit 2 which initiated core Cycle 2 operation in December 1974. Since there are no significant differences between the core configurations for Unit 2 Cycle 2 and Unit 1 Cycle 4, the ECCS reevaluation and specifically related Technical Specifications apply to both Units 1 and 2.

The first part of this safety evaluation, "Unit 1 Core Cycle 4 Reload", discusses and evaluates the requested action regarding the Point Beach Unit 1 core Cycle 4 reload. The second part of this safety evaluation, "Emergency Core Cooling System", discusses and evaluates the ECCS reevaluation and related Technical Specifications which are applicable to both Units 1 and 2.

PART I: UNIT 1 CORE CYCLE 4 RELOAD

A. Introduction

By letter dated October 6, 1975, Wisconsin Electric Power Company (WEPCO) proposed changes to the Technical Specifications of Facility Operating License DPR-24 for Point Beach Unit 1. Supplemental information related to the requested changes was supplied by WEPCO with letters dated October 22, November 26, and December 15 and 18, 1975. To allow Unit 1 operation in core Cycle 4, WEPCO requested: (1) changes to the Unit 1 control rod insertion limits, (2) changes to clad flattening limitations, and (3) changes to the overtemperature ΔT and pressurizer low pressure limiting safety system settings to reflect a proposed increase in reactor coolant system operating pressure to 2250 psia.

B. Discussion

The Point Beach Unit 1 core Cycle 4 loading includes 32 new fuel assemblies (Region 7) and one twice burned assembly from Region 4 plus 11 assemblies from Unit 2 (7 from Region 1 and 4 from Region 2). The mechanical design of the new Region 7 assemblies is essentially the same as the Regions 5 and 6 fuel which will remain in the core during Cycle 4.

1. Control Rod Insertion Limits

Control of the operating reactor is provided by neutron absorbing control rods and soluble boric acid in the reactor coolant. The more boric acid contained in the reactor coolant the less the control rods need to be inserted to provide reactor control. The proposed control rod insertion limits are the result of analyses performed for the Unit 1 Cycle 4 core configuration to insure: (1) an adequate shutdown margin is maintained throughout cycle life, (2) hot channel factors are maintained below design limits, (3) acceptable consequences of rod ejection accident, and (4) acceptable consequences of rod misalignment. The maintenance of adequate shutdown margin at the end of core life is the consideration which typically defines the control rod insertion limits.

2. Minimum Time to Clad Flattening

Point Beach Unit 1 has been operating at a reduced primary pressure of 2000 psia in core Cycle 3. Reduced primary pressure was initiated in order to lengthen the predicted time to clad flattening by reducing the pressure differential across the fuel cladding and thus reducing the clad creep rate. The presently specified Unit 1 fuel residence limit of 18,000 EFPH is the analytically determined minimum time to clad flattening for Unit 1 core Cycle 3, using a previously approved model and assuming continued reactor operation at 2000 psia.

Westinghouse has revised the clad flattening model and has submitted reports WCAP-8377[1] and WCAP-8381[2] which describe the revised model. The revised model as described in the referenced reports has been approved for licensing actions and was used in support of Point Beach Unit 2 License Amendment No. 13.[3] The revised model as described in License Amendment No. 13 predicts longer times to clad flattening. Since the predicted time to clad flattening for Unit 1 now exceeds the expected life of the Unit 1 fuel assemblies, there is no longer an advantage for operation at reduced pressure. Therefore, WEPCO has stated that they plan to return Unit 1 to 2250 psia primary system pressure following reload for core Cycle 4.

3. Overtemperature ΔT and Pressurizer Low Pressure Trip Setpoint

The core protection system operates by defining a region of permissible operation in terms of power, pressure, temperature, coolant flow and axial power distribution. This allowable operating region with regard to coolant temperature difference across the reactor core is determined by the equations which define the overtemperature ΔT reactor trips. The overtemperature ΔT reactor trip protects the core against nucleate boiling, excessive hot channel exit quality, and hot channel boiling for any combination of power, pressure, temperature, and axial core power distribution.

WEPCO, in order to resume reactor operation at 2250 psia, has proposed modifying the overtemperature ΔT reactor trip expression and has proposed that the pressurizer low pressure trip setpoint be returned to its pre-Cycle 3 value, which is consistent with the new overtemperature ΔT setpoint expression.

4. Additional Rod Bow Penalty

Recent data on Westinghouse 15 x 15 fuel assemblies, which is generally applicable to 14 x 14 fuel assemblies of the type used at Point Beach, indicates that the bowing model in WCAP-8586, "An Evaluation of Fuel Rod Bowing" underestimates the extent of fuel rod bowing. Consequently, the staff has applied an additional penalty in radial peaking factor, $F_{\Delta H}$, to Point Beach Unit 1, core Cycle 4.

C. Evaluation

1. Control Rod Insertion Limits

Calculations of the core kinetics parameters indicate the values for Cycle 4 fall within the limits based upon previously submitted accident analyses, except the most negative Doppler coefficient. This becomes only slightly more negative than the current limit, and has a negligible effect on the accident analysis. Therefore previously submitted analyses of accidents affected by these parameters remain acceptable for Cycle 4.

The revised control rod insertion limits (Figure 15.3.10-1) and calculated shutdown margin for Cycle 4 indicate more than required shutdown margins will be maintained throughout cycle life. This includes a 10% uncertainty allowance in calculations of control rod worths. Startup measurements of control bank worths will confirm the validity of the calculation of bank worths and hence the calculation of shutdown margins. However, the Cycle 4 hot full power Beginning of Cycle (BOC) and End of Cycle (EOC) maximum ejected rod worths are greater than the corresponding Cycle 3 values. In addition, the minimum BOC Delayed Neutron Fraction (B_{eff}) was found to be .0059 for Cycle 4 vs .0064 for previous cycles. But, reanalysis of these rod ejection accidents using NRC approved Westinghouse procedures[4] indicates no centerline fuel melting and a peak enthalpy of 143 cal/gm for the worst case. These are acceptable results; and therefore, the proposed control rod insertion limits are acceptable.

2. Minimum Time to Clad Flattening

WEPCO has recalculated the minimum time to clad flattening using the approved model described in WCAP-8377[1] and WCAP-8381[2]. WEPCO has determined this time to be 30,000 EFPH for Unit 1, Regions 5, 6 & 7 fuel assemblies, assuming reactor operation at 2250 psia. However, the Unit 2, Region 2 fuel assemblies, which will be used in Unit 1 Cycle 4, are calculated to have a minimum time to clad flattening of 22,020 Effective Full Power Hours (EFPH), and thus these assemblies are limiting. These calculations were also performed using the approved model described in WCAP-8377[1] and WCAP-8381[2].

Therefore, based on the calculated minimum time to clad flattening for the limiting Unit 2 Region 2 assemblies, we have concluded that a fuel residence time limit of 22,020 EFPH for Point Beach Unit 1 core Cycle 4 is acceptable. Technical Specification 15.2.1.2 incorporates this requirement.

3. Overtemperature ΔT and Pressurizer Low Pressure Trip Setpoint

The pressurizer low pressure trip setpoint and the overtemperature ΔT settings are specified in Technical Specification 15.2.3.1.B(3) and (4) respectively. Point Beach Unit 1, has been operated in the past at a system pressure of 2250 psia and nominal average temperature of 581.3°F. As a consequence of a subsequent fuel densification review by the staff, the Point Beach, Unit 1, operating pressure was restricted during previous Cycle 3 operation to 2000 psia and nominal average temperature of 572.9°F. For Cycle 3 operation, the licensee modified the Technical Specifications, making the overtemperature ΔT trip limits more restrictive, and lowering the pressurizer low pressure trip setpoint in consideration of the effects of reduced system operating pressure. The overtemperature ΔT trip limits were made more restrictive by modifying the constants and nominal pressure setpoints in the overtemperature ΔT trip. The licensee now has proposed to operate the plant for Cycle 4 at a system pressure of 2250 psia. In this matter, the nominal system pressure was increased from 2000 to 2250 psia while all other constants and system parameters (average temperature) remained identical to the Cycle 3 values. The staff has reviewed the proposed Cycle 4 Technical Specifications and has concluded that since the overtemperature ΔT trip setpoint for Cycle 3 was more restrictive than the originally (licensed) approved value, operation as proposed for Cycle 4 will be more conservative and therefore, the proposed modification to the Technical Specification 15.2.3.1.B(4) is acceptable.

In addition, the pressurizer low pressure trip setpoint, Technical Specification 15.2.3.1.B(5), has been changed back to the value (1865 psig) it was before the system operating pressure was reduced to 2000 psia. Based on previous safety evaluations of operation at 2250 psia, made by the staff, this proposed change is also acceptable.

4. Additional Rod Bow Penalty

The safety analyses applicable to operation during Cycle 4 are based on previous Cycle 3 safety analyses[5] and those reported in the Final Facility Description and Safety Analysis Report (FFDSAR)[6]. These analyses were, however, performed with a pitch reduction factor which results in a 3.2 percent margin in DNBR to allow for rod-to-rod bowing. Recent discussions with Westinghouse indicate that this penalty is inadequate. New data on 15 x 15 rod bundles with up to 27,000 MWd/MTu burnup show that the bowing model presented in WCAP-8346, "An Evaluation of Fuel Rod Bowing," underestimates the extent of rod bowing. The 15 x 15 bowing data indicate that a penalty of approximately 3.6 percent in DNBR should be applied to the Point Beach design to account for rod bowing during Cycle 4. We will require that a total penalty of 5.6 percent in DNBR (including Point Beach design pitch reduction penalty be used to account for rod bowing. A suitably conservative value of 5.6 percent was chosen instead of the 3.6 percent penalty because the review of the Westinghouse approach for 15 x 15 geometry has not been completed. Once the review is complete the 5.6 percent penalty may be modified to conform to the data

As stated previously, the Point Beach Unit 1 core design offers approximately 3.2 percent margin in DNBR due to pitch reduction in the analyses. The remaining 2.4 percent of the 5.6 percent penalty is equivalent to a 1.4 percent heat flux penalty. To achieve a 1.4 percent heat flux reduction it will be necessary to limit operation of the Point Beach Unit 1 Cycle 4 core to a radial peaking factor, $F_{\Delta H}$, of 1.55 rather than 1.58. With this limitation, operation of the Point Beach Unit 1 plant with the Cycle 4 core is acceptable. Technical Specification 15.3.10.B.1 has been modified accordingly and the licensee has concurred with this modification.

D. Summary

The safety analyses applicable to operation during core Cycle 4 are based on previous Cycle 3 safety analyses and those reported in the FFDSAR, and additional analyses of rod ejection accidents. The proposed operation at 2250 psia is acceptable to the staff, since raising of the operating pressure will have no adverse effects on the accident anlayses; DNB heat flux increases with increasing system pressure. The analyses previously reported in References 5 and 6 were reviewed and approved by the staff and, since the effects of the Cycle 4 reload on the design basis and postulated accidents can be conservatively accommodated with the previous analyses, with additional modifications made by the staff, operation in core Cycle 4 is acceptable.

PART II: EMERGENCY CORE COOLING SYSTEM

A. Introduction

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR §50.46, "Acceptance Criteria and Emergency Core Cooling System for Light Water Nuclear Power Reactors". One of the requirements of the Order was that the licensee shall submit a reevaluation of the ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, §50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in the Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Wisconsin Electric Power Company (WEPCO) submitted an ECCS reevaluation and related Technical Specifications, by letter dated June 24, 1975. This reevaluation, complied with previous submittals dated September 6, 1974, December 6, 1974, and May 7, 1975, is applicable to both Point Beach Units 1 and 2. In addition, WEPCO submitted additional information regarding ECCS cooling performance by letters dated November 5, 1975, November 26, 1975, and December 15, 1975.

B. Discussion

The Order for Modification of License issued December 27, 1974[7], stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report (SER) of Point Beach Units 1 and 2 dated December 27, 1974.

The background of the staff review of the Westinghouse ECCS models and their application to Point Beach is described in the staff SER for this facility dated December 27, 1974 (the December 27, 1974, SER) issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974[8] and the November 1974[9] Supplement to the Status Report which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also described the various changes required in the earlier Westinghouse evaluation model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model.

The Point Beach ECCS evaluation which is covered by this safety evaluation properly conforms to the accepted model. The June 24, 1975 submittal contained: (1) analyses of sufficient break sizes and location to verify that the worst break condition had been considered and (2) documentation, by reference to submitted Westinghouse Topical Reports, of the ECCS model modifications described in our December 27, 1974 SER.

C. Evaluation

We have reviewed the evaluation of ECCS performance submitted by WEPCO for the Point Beach Nuclear Generating Units 1 and 2 and concluded that the evaluation has been performed wholly in conformance with the requirements of Appendix K to 10 CFR Part 50. Therefore, operation of the reactor would meet the requirements of 10 CFR §50.46 provided that (1) the reactor is operated in accordance with the proposed Technical Specifications as modified by subsequent NRC review, and (2) the Emergency Operating Procedures are modified as described in this evaluation. Specific areas of review are discussed below:

1. ECCS Reanalysis

The licensee submitted Loss-of-Coolant Accident (LOCA) analyses, by letter dated June 24, 1975, that addressed small ruptured pipes and major reactor coolant system pipe ruptures. The small break LOCA incorporated a previous September 6, 1974 submittal.

A three break spectrum, specific for Point Beach, was submitted and an applicable generic plant sensitivity study was used in conformity with the break spectrum requirements of 10 CFR 50.46(a). The analyses submitted were performed with an acceptable evaluation model which is wholly in conformance with 10 CFR Part 50, Appendix K.

The analyses identified the worst break size as the 0.4 double-ended cold leg guillotine with a calculated peak clad temperature of 1996°F; this is within the acceptable limit of 2200°F specified in 10 CFR 50.46(b). In addition, the calculated maximum local metal/water reaction of 3.2% and total core wide metal/water reaction of less than 0.3% are well below the allowable limits of 17% and 1%, respectively. These results are for region 3 cycle 1 fuel in Unit 2 which was identified as the limiting fuel for both Units 1 and 2.

These analyses assumed that there was a coincident loss of offsite power at the initiation of the LOCA, which would result in pump coastdown. A sensitivity analysis was cited for the limiting LOCA with no loss of offsite power. The results showed that the peak clad temperature would be increased 25°F which would still result in a peak temperature significantly below the acceptable limit.

The licensee indicated that rod bowing would produce a maximum power spike of 4.7% along the hot rod. This power spike was accounted for in the LOCA analyses, by letter dated December 15, 1975, and the results indicate that no additional allowance on power peak is required.

Since analyses were presented only for two loop operation, the reactor will not be allowed to operate at greater than 10% power with one idle loop. This requirement is reflected in existing Technical Specification 15.3.1.A.1.C(1).

2. <u>ECCS Containment Pressure Evaluation</u>

The ECCS containment pressure calculations for Point Beach were done using the Westinghouse ECCS evaluation model. The NRC staff reviewed Westinghouse's model and published a Status Report on October 15, 1974(8), which was amended November 13, 1974(9). We concluded that Westinghouse's containment pressure model was acceptable for ECCS evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for the Point Beach plants on December 6, 1974. WEPCO has reevaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for ECCS analysis. This evaluation was based on measurements within the containment and from as-built drawings to which additional margin was added. The containment heat removal systems were assumed to operate at their maximum capacities and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Point Beach plants is conservative and therefore the calculated containment pressure is in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

3. Single Failure Criterion

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most limiting single failure of ECCS equipment has occurred. The worst single failure which would minimize the ECC available to cool the core and provide maximum containment cooling was identified by Westinghouse as the loss of a low pressure ECCS pump. The staff concluded in Ref. 8 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

A review of the Point Beach P&ID's indicated that the spurious actuation of the electrically operated accumulator isolation valves (841 A/B) would violate the LOCA analysis assumption that both accumulators are available. To preclude this adverse condition, the staff requires that these values be aligned in the open position and A.C. power removed at the motor control center, when the reactor is at elevated pressures. With the licensee's concurrence, we have modified Technical Specification 15.3.3.1 accordingly, and thus we have concluded that the single failure criterion is satisfied.

The emergency operating procedures were reviewed to verify their consistency with the ECCS description in the FFDSAR. The licensee has agreed to modify these procedures by incorporating cautionary notes (for the operator) that relate to switchover times and procedures for the RHR pumps.

4. Boric Acid Concentration During Long Term Cooling

By letter dated November 5, 1975, WEPCO submitted the procedures for post-LOCA long term cooling in order to prevent excessive concentration of boric acid in the reactor vessel. The procedures were augmented by a May 7, 1975 submittal containing the results of an analysis of the mechanisms that would lead to the concentration of the boric acid solution injected into the vessel.

According to these procedures boric acid solution is injected into the reactor vessel by the Low Pressure Safety Injection (LPSI) system and into the cold legs by the High Pressure Safety Injection (HPSI) system. It was recommended, however, by the licensee that, for small breaks, boric acid solution should be injected simultaneously into the cold legs and into the reactor vessel. For the large breaks, when larger quantities of borated water are needed, the injection will be provided to the reactor vessel only by the LPSI system. The licensee claims, that even with the hot leg break, sufficient thermal stirring will be provided to mix the injected liquid with the boric acid solution in the core and hence to prevent boric acid buildup.

Both LPSI and HPSI systems have two independent injection trains, each of which is able to provide enough boric acid solution to replace the boiloff and assure sufficient flow through the core.

In the recirculation mode the LPSI pumps draw the solution from the containment sump and deliver it either directly to the reactor vessel or to the suction side of the HPSI pumps. This arrangement permits both the high and the low pressure injection systems to meet the single active and passive failure criteria.

The staff has reviewed the proposed procedures and has come to the conclusion that the system can be operated in a satisfactory manner during the long term, post-LOCA cooling. The staff recommends, however, that for large breaks the direct injection of boric acid solution into the reactor vessel by the LPSI pumps should be supplemented by a simultaneous cold leg injection. The licensee agrees, that the simultaneous cold leg injection can be accomplished by the HPSI pumps after the containment spraying is discontinued and the containment spray pumps are shutoff. It was determined that without cold leg injection the concentration of boric acid in the core region can be maintained below the solubility limits for a sufficiently long period of time (14 hours) to make this mode of operation possible.

It is the staff's position that the emergency operating procedures must be revised to require either of the following approaches, fourteen hours after a LOCA:

(a) Simultaneous vessel and cold leg injections

(b) Alternate cold and vessel injections with the time period between them sufficiently short to prevent high buildup of boric acid in the core region.

With these procedural modifications we have concluded that the solubility of the boric acid will be maintained and thus, the long term cooling provisions are acceptable.

5. Technical Specifications

The performance evaluation of the ECCS is based on certain assumptions that will be incorporated in the Technical Specifications. A summary of the required specifications is presented below:

1. The core power distribution limits are specified in Technical Specification 15.3.10.B. These include an overall peaking factor (F_Q) of 2.32 based on full power operation at 1518 MWt.

2. The reactor is limited to operation with primary coolant pumps in service, as specified in existing Technical Specification 15.3.1.A.1.C(1).

3. A.C. power must be removed from the accumulator isolation valves (MOV-841 A&B) with the valves in their proper orientation during reactor operation at elevated pressures, as specified in Technical Specifications 15.3.3.1.h and i.

In consideration of our evaluation presented above we have concluded that the reevaluation of the ECCS cooling performance for Point Beach Units 1 and 2, and the proposed Technical Specifications, as modified by the staff and concurred in by the licensee, are acceptable.

D. Summary

Based on our review, we have determined that: (1) the LOCA analyses that were performed are wholly in conformance with the requirements of Appendix K to 10 CFR Part 50, (2) the ECCS cooling performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR §50.46, (3) ECCS cooling performance will be adequate despite any postulated failure of a single component, (4) adequate systems exist to provide long term cooling to the reactor vessel. However, the emergency operating procedures must be modified. The licensee has agreed to modify the emergency operating procedures to incorporate the staff's requirements. Therefore, we have concluded that the Emergency Core Cooling System Analysis is acceptable.

PART III: CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Dated: December 24, 1975

References

(1) George, R. A., Lee, Y. C., and Eng, G. H., "Revised Clad Flattening Model," Westinghouse Electric Corporation, WCAP-8377, (Proprietary) July 1974.

(2) George, R. A., Lee, Y. C., and Eng, G. H., "Revised Clad Flattening Model," Westinghouse Electric Corporation, WCAP-8381, July 1974.

(3) Amendment No. 13 to Facility Operating License DPR-27 for Point Beach Unit 2 (Docket No. 50-301), dated October 6, 1975.

(4) Risher, D. H. Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods", WCAP-7588, Revision 1, December 1971.

(5) Letter to J. F. O'Leary (AEC) from S. Burstein (WEPCO) dated May 1, 1974.

(6) Final Facility Description and Safety Analysis Report - Point Beach Unit 1 and 2.

(7) "Order for Modification of License", letter to Wisconsin Electric Power Company from George Lear, December 27, 1974.

(8) "Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Co., ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K," October 15, 1974.

(9) "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Co., ECCS Evaluation Model Conformance to 10 CFR Part 50, Appendix K," November 13, 1974.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 14 and 18 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.

These amendments: (1) incorporate operating limits in the Technical Specifications for the facilities based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50, and (2) modify certain Unit 1 operating limits to reflect the results of the cycle 4 core performance analysis.

The application for the amendment 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 amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with item (1) above was published in the FEDERAL REGISTER on August 7, 1975 (40 F.R. 33290) and Notice of Proposed Issuance of Amendment to Facility Operating License in connection with items (2) and (3) above was published in the FEDERAL REGISTER on November 21, 1975 (40 F.R. 54311). No request for a hearing or petition for leave to intervene was filed following notices of the proposed actions.

For further details with respect to this action, see: (1) the applications for amendment dated September 6, 1974, June 26, 1975, and October 6, 1975, and supplements dated December 6, 1974, May 7, November 5, November 26, and December 15 and 18, 1975, (2) Amendments Nos. 14 and 18 to Licenses Nos. DPR-24 and DPR-27 with Changes Nos. 19 and 24, (3) the Commission's concurrently issued related Safety Evaluation, and (4) the Commission's Negative Declaration dated December 16, 1975, (which is also being published in the FEDERAL REGISTER) and associated Environmental Impact Appraisal. 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 Document Department, University of Wisconsin - Stevens Point Library, Stevens Point, Wisconsin. A single copy of items (2), (3) and (4) may be obtained upon request addressed to the Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 24th day of December, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald M. Elliott

Donald M. Elliott, Acting Chief
Operating Reactors Branch #3
Division of Reactor Licensing