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

May 8, 2001

Mr. Mark Reddemann
Site Vice President
Kewaunee and Point Beach Nuclear Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INDIVIDUAL ROD POSITION INDICATION SYSTEM (TAC NOS. MB0671 AND MB0672)

Dear Mr. Reddemann:

The Commission has issued the enclosed Amendment No. 200 to Facility Operating License No. DPR-24 and Amendment No. 205 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 20, 2000, as supplemented February 6 and May 3, 2001.

These amendments incorporate changes to the TSs to increase the allowable deviation in individual rod position indication. By the February 6, 2001, supplemental letter, the licensee withdrew portions of the original application that dealt with operation at greater than 85-percent power. The licensee plans to submit those portions that deal with operation at greater than 85-percent power as a separate amendment request at a later time.

During the NRC staff's review of the initial amendment application, which also included an application for withholding of proprietary information, the staff encountered several issues related to the contents and timeliness of your submittals. The resolution of these issues necessitated numerous teleconferences with your staff to clarify the application and proprietary information. Your final supplemental letter was submitted within days of your requested approval date. The time and effort the staff dedicated to the resolution of these issues impacted the schedule for issuance of these amendments.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Beth A. Wetzel

Beth A. Wetzel, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 200 to DPR-24
2. Amendment No. 205 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Beth A. Wetzel, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures: 1. Amendment No. 200 to DPR-24
2. Amendment No. 205 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

PUBLIC	OGC	MChatterton
PDIII-1 Reading	ACRS	RCaruso
CCraig	WBeckner	
BWetzel	GHill(4)	
RBouling	RLanksbury, RGN-III	

*Previously Concurred

OFFICE	PDIII-1/PM	PDIII-1/LA	SRXB/SC*	OGC*	PDIII-1/SC
NAME	BWetzel BAW	RBouling RB	RCaruso	RHoefling	CCraig CC
DATE	5/8/01	5/8/01	5/7/01	5/8/01	5/8/01

DOCUMENT NAME: G:\PDIII-1\PTBEACH\AMDb0672.wpd
OFFICIAL RECORD COPY

cc:

Mr. John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, DC 20037-1128

Mr. Richard R. Grigg
President and Chief Operating Officer
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, WI 53201

Site Licensing Manager
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

Mr. Ken Duveneck
Town Chairman
Town of Two Creeks
13017 State Highway 42
Mishicot, WI 54228

Chairman
Public Service Commission
of Wisconsin
P.O. Box 7854
Madison, WI 53707-7854

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
6612 Nuclear Road
Two Rivers, WI 54241

Ms. Sarah Jenkins
Electric Division
Public Service Commission of Wisconsin
P.O. Box 7854
Madison, WI 53707-7854

Michael D. Wadley
Chief Nuclear Officer
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Nuclear Asset Manager
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, WI 53201

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 200
License No. DPR-24

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee) dated November 20, 2000, as supplemented February 6 and May 3, 2001, 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. 200 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: May 8, 2001

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 205
License No. DPR-27

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee) dated November 20, 2000, as supplemented February 6 and May 3, 2001, 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. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 205 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: May 8, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 200

TO FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 205

TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
15.3.10-1	15.3.10-1
15.3.10-2	15.3.10-2
15.3.10-3	15.3.10-3
15.3.10-4	15.3.10-4
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	15.3.10-16
15.3.10-17	15.3.10-17
15.3.10-18	15.3.10-18

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. SHUTDOWN MARGIN

1. 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. If the shutdown margin is less than the applicable value of Figure 15.3.10-2, within 15 minutes initiate boration to restore the shutdown margin.

2. A shutdown margin of at least 1% Δk/k shall be maintained when the reactor coolant temperature is less than 350°F. If the shutdown margin is less than this limit, within 15 minutes initiate boration to restore the shutdown margin.

B. ROD OPERABILITY AND BANK ALIGNMENT LIMITS

NOTE: One hour is allowed following rod motion prior to verifying rod operability and bank alignment limits.

1. During power and low power operation, all shutdown and control rods shall be operable and positioned within the allowed rod misalignment between the individual indicated rod positions and the bank demand position as follows;

   i) For operation ≤ 85 percent of rated power, the allowed indicated misalignment between the bank demand position and the individual indicated rod position shall be ≤ ±24 steps.
   ii) For operation > 85 percent of rated power, the allowed indicated misalignment between the bank demand position and the individual indicated rod position shall be ≤ ±12 steps.

   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 declared inoperable until it has been verified that it will step in or would drop upon demand.

   a. Rod Operability Requirements

(1) If one rod is determined to be untrippable, perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2;
<u>OR</u>
(b) Within one hour restore the shutdown margin by boration;
<u>OR</u>
(c) Within six hours be in hot shutdown.

(2) If sustained power operation with an untrippable rod is desired, perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; <u>OR</u> within one hour restore the shutdown margin by boration;
<u>AND</u>
(b) Within six hours, adjust the insertion limits to reflect the worth of the untrippable rod.

(c) If the above actions and associated completion times are not met, be in hot shutdown within six hours.

(3) If more than one rod is determined to be untrippable, perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; <u>OR</u> within one hour restore the shutdown margin by boration;
<u>AND</u>
(b) Within six hours be in hot shutdown.

b. <u>Rod Bank Alignment Limits</u>

(1) If it has been determined that one rod is not within alignment limits, and the indicated misalignment is not being caused by malfunctioning rod position indication, within one hour restore the rod to within alignment limits; <u>OR</u> perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; <u>OR</u> within one hour restore the shutdown margin by boration;
<u>AND</u>

(b) Within eight hours reduce thermal power to ≤ 75 percent of rated thermal power;
<u>AND</u>

(c) Verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2 once per twelve hours;
<u>AND</u>

(d) Within 72 hours verify that measured values of FQ(Z) are within limits;
<u>AND</u>

(e) Within 72 hours verify that FNΔH is within limits;

(f) If the above actions and associated completion times are not met, be in hot shutdown within the following six hours.

(g) In order to subsequently increase thermal power above 75 percent of rated thermal power with the existing rod misalignment, perform an analysis to determine the hot channel factors and the resulting allowable power level in accordance with TS 15.3.10.E.

(2) If it has been determined that more than one rod is not within alignment limits and the misalignments are not being caused by malfunctioning rod position indication, perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; <u>OR</u> within one hour restore the shutdown margin by boration;
<u>AND</u>

(b) Be in hot shutdown within six hours.

C. <u>ROD POSITION INDICATION</u>

NOTE: Separate entry into TS 15.3.10.C.1.a, b, or c is allowed for each inoperable rod position indicator and each bank of demand position indication.

1. During power operation ≥ 10 percent of rated thermal power, the rod position indication system and the bank demand position indication system shall be operable.

a. If one or more rod position indicators (RPI) are determined to be inoperable, perform the following actions:

(1) Within eight hours verify the position of the rods with inoperable RPIs by using movable incore detectors;
<u>AND</u>

(2) Once per shift check the position of the rods with inoperable RPIs by using excore detectors, or thermocouples, or movable incore detectors;

(3) If the above actions and associated completion times are not met, perform the actions in accordance with TS 15.3.10.B.1.b.

b. If one or more rods with inoperable RPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position, perform the following actions:

(1) Within four hours check the position of the rods with inoperable RPIs by using excore detectors, or thermocouples, or movable incore detectors;

(2) If the above action and associated completion time is not met, perform the actions in accordance with TS 15.3.10.B.1.b.

c. If bank demand position indication, for one or more banks, is determined to be inoperable, perform the following actions:

(1) Once per shift verify that all RPIs for the affected banks are operable;
<u>AND</u>
(2) Once per shift verify that the most withdrawn rod and the least withdrawn rod of the affected banks are within the allowed rod misalignment in accordance with TS 15.3.10.B.1.

(3) If the above actions and associated completion times are not met, perform the actions in accordance with TS 15.3.10.B.1.b.

D. <u>BANK INSERTION LIMITS</u>

NOTE: One hour is allowed following rod motion prior to verifying bank insertion limits.

1. When the reactor is critical, the shutdown banks shall be fully withdrawn. Fully withdrawn is defined as a bank position equal to or greater than 225 steps. This definition is applicable to shutdown and control banks.

If this condition is not met, perform the following actions:

a. Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; <u>OR</u> within one hour restore the shutdown margin by boration;

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.

The specifications of 15.3.10 ensure that (1) acceptable power distribution limits are maintained, (2) the minimum shutdown margin is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. Operability of the control rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits. Permitted control rod misalignments (as indicated by the RPI System within one hour after control rod motion) are; a) ±12 steps of the bank demand position (if power level is greater than 85 percent of rated power), and b) ±24 steps of the bank demand position (if the power level is less than or equal to 85 percent of rated power). For power levels less than or equal to 85 percent of rated power, the peaking factor margin does not have to be verified on an explicit basis. This is due to the rate of peaking factor margin increase (due to the peaking factor limit increasing) as the power level decreases being greater than the peaking factor margin loss (due to the increased control rod misalignment). This effect is described in WCAP-15432 Rev. 1. These limits are applicable to all shutdown and control rods (of all banks) over the range of 0 to 230 steps withdrawn inclusive.

The comparison of bank demand position and RPI System may take place at any time up to one hour after rod motion, at any power level. This allows up to one hour of thermal soak time to allow the control rod drive shaft to reach a thermal equilibrium and thus present a consistent position indication. A similar time period (up to one hour after rod motion) is allowed for comparison of the bank insertion limits and the RPI System. This comparison is sufficient to verify that the control rods are above the insertion limits and thus assures the presence of sufficient shutdown margin to satisfy the assumptions of the safety analyses.

The action statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Actual misalignment of a rod requires measurement of peaking factors (to confirm acceptability) or a restriction in thermal power; either of these restrictions provides assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

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 24 steps because the concomitant increase in power density will normally be less than 1% for a 24 step 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 LVDTs below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

Power Distribution

During power operation, the global power distribution is limited by TS 15.3.10.E.2, "Axial Flux Difference," and TS 15.3.10.E.3, "Quadrant Power Tilt," which are directly and continuously measured process variables. These specifications, along with TS 15.3.10.D, "Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

As a result of the increased peaking factors allowed by the new 422V+ fuel, a new column was added to TS 15.3.10.E.1.a. The full power $F^N_{\Delta H}$ peaking factor design limit (radial peaking factor) for 422V+ fuel will increase to 1.77 from the 1.70 value for the OFA fuel. The maximum $F_Q(Z)$ peaking factor limit (total peaking factor) for 422V+ fuel will increase to 2.60 from the 2.50 value for the OFA fuel. The OFA fuel design will retain the current $F^N_{\Delta H}$ and $F_Q(Z)$ peaking factors of 1.70 and 2.50, respectively. In addition, the K(Z) envelope for the new 422V+ fuel was modified and a new TS figure 15.3.10-3a was developed and inserted in the Technical Specifications. The K(Z) envelope in TS Figure 15.3.10-3 remains for the OFA fuel.

The purpose of the limits on the values of $F_Q(Z)$, the height dependent heat flux hot channel factor, is to limit the local peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

The purpose of the limits on $F^N_{\Delta H}$, the nuclear enthalpy rise hot channel factor, is to ensure that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$, 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 $F^N_{\Delta H}$.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least monthly. However, during power operation, the global power distribution is monitored by TS 15.3.10.E.2, "Axial Flux Difference," and TS 15.3.10.E.3, "Quadrant Power Tilt," which address directly and continuously measured process variables.

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 24 steps from the bank demand position (operation at greater than 85 percent of rated power), nor more than 36 steps (operation at less than or equal to 85 percent of rated power). An indicated misalignment limit of 12 steps precludes a rod misalignment of greater than 24 steps with consideration of instrumentation error; 24 steps indicated misalignment corresponds to 36 steps with instrumentation error.

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

3. 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 four conditions are observed, these hot channel factor limits are met. In Specification 15.3.10.E.1.a, F_Q is arbitrarily limited for $p \leq 0.5$.

The upper bound envelope F_Q (defined in 15.3.10.E) times the normalized peaking factor axial dependence of Figure 15.3.10-3 for OFA and Upgraded OFA Fuel and Figure 15.3.10-3a for 422V+ Fuel (consistent with the Technical Specifications on power distribution control as given in Section 15.3.10) was used in the large and small break LOCA analyses. The envelope was determined based on allowable power density distributions at full power restricted to axial flux difference (ΔI) values consistent with those in Specification 15.3.10.E.2.

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 taken into account. 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 $F^N_{\Delta H} \leq 1.70/1.08$ for OFA and Upgraded OFA fuel and 1.77/1.08 for 422V+ fuel. The logic behind the larger uncertainty in this case is as follows:

(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) 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 $F^N_{\Delta H}$.

(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.

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.

The measured hot channel factors are increased as follows:

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

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

<u>Axial Power Distribution</u>

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the $F_Q(Z)$ upper bound envelope of F_Q^{LIMIT} times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD monitor alarm. The computer determines the AFD for each of the operable excore channels and provides a computer alarm if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and the reactor power is greater than 50 percent of Rated Power.

Quadrant Tilt

The quadrant tilt limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, specifications associated with axial flux difference, quadrant tilt, and control rod insertion limits provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

The excore detectors are somewhat insensitive to disturbances near the core center or on the major axes. 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 on the 45° axes.

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.

Physics Tests Exceptions

The primary purpose of the at-power and low power physics tests is to permit relaxations of existing specifications to allow performance of instrumentation calibration tests and special physics tests. The at-power specification allows selected control rods and shutdown rods to be positions outside their specified alignment and insertion limits to conduct physics tested at power. The power level is limited to ≤85 percent of rated thermal power and the power range neutron flux trip setpoint is set at maximum of 90 percent of rated thermal power. Operation with thermal power ≤85 percent of rated thermal power during physics tests provides an acceptable thermal margin when one or more of the applicable specifications is not being met. The Power Range Neutron Flux - High trip setpoint is reduced so that a similar margin exists between the steady-state condition and the trip setpoint that exists during normal operation at rated thermal power.

The low power specification allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits to conduct physics tests at low power. If power exceeds two percent, as indicated by nuclear instrumentation, during the performance of low power physics tests, the only acceptable action is to open the reactor trip breakers to prevent operation of the reactor beyond its design limits. Immediately opening the reactor trip breakers will shut down the reactor and prevent operation of the reactor outside of its design limits. If the RCS lowest loop average temperature falls below the minimum temperature for criticality, the temperature should be restored within 15 minutes because operation with the reactor critical and temperature below the minimum temperature for criticality could violate the assumptions for accidents analyzed in the safety analyses. If the temperature cannot be restored within 15 minutes, the plant must be made subcritical within an additional 15 minutes. This action will place the plant in a safe condition in an orderly manner without challenging plant systems.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 200 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 205 TO FACILITY OPERATING LICENSE NO. DPR-27

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By application dated November 20, 2000, as supplemented February 6, and May 3, 2001, the Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for Point Beach Nuclear Plant, Units 1 and 2. The proposed changes would revise the TSs to increase the allowable deviation in individual rod position indication (IRPI). Specifically, the proposed amendments would change the allowable deviation in IRPI from the bank demand position and add a note to allow for a 1-hour thermal soak, following rod motion, prior to verifying limits.

By the February 6, 2001, supplemental letter, the licensee withdrew portions of the original application that dealt with operation at greater than 85-percent power. The licensee plans to submit those portions that deal with operation at greater than 85-percent power as a separate amendment request at a later time. The May 3, 2001, supplemental letter provided additional clarifying information that was within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards consideration determination. A revision to the supporting Westinghouse topical report was also submitted by the licensee's letter dated May 3, 2001.

2.0 BACKGROUND

The TSs for Westinghouse reactors typically require the position of all control rods as indicated by position indicators (actual position) to be in agreement with the group step counter demand positions within ± 12 steps. A step is 5/8 inch. The ± 12 requirement reflects the accident analysis assumption that the rods can be misaligned by 24 steps, which consists of an indicated 12-step misalignment and a 12-step uncertainty. There has been a long history of violations of the ± 12 step requirement, particularly in the shutdown modes and during power ascension. The difficulty lies in the characteristics of the analog system, which has a nonlinear, steady-state response, and a time-dependent response, which is the result of temperature dependence. The licensee's experience with the IRPI system shows that indicated misalignment is often greater than ± 12 steps. The root cause of this phenomenon is the analog rod position indication variation with temperature, which occurs most often after a recent power level change.

The proposed changes are based on an evaluation performed by Westinghouse and documented in Topical Report WCAP-15432, Revision 2, "Conditional Extension of the Rod Misalignment Technical Specification for Point Beach Units 1 and 2," dated May 2001. Westinghouse performed the evaluations of the effects of increasing the allowed control rod indicated misalignment from ± 12 steps to an indicated misalignment of up to ± 24 steps when the core power is less than or equal to 85 percent of rated power and ± 12 steps above 85 percent of rated power. WCAP-15432, Revision 2, has been reviewed by the NRC staff and will be discussed in the evaluation section below.

3.0 EVALUATION

Since the IRPI system is designed to an accuracy of 12 steps, in order to guarantee a rod misalignment of less than ± 24 steps (12 steps misalignment + 12 steps IRPI uncertainty), the IRPI readings must be no larger than 12 steps. In order to justify changing the misalignment to ± 24 steps, evaluations were performed for misalignments of up to 36 steps (24 steps indicated and 12 steps uncertainty). The TS limits on peaking factors F_q and $F_{\Delta H}$ increase as the power level decreases. The increase in the limit for F_q and $F_{\Delta H}$ was used to accommodate the larger than ± 12 step misalignment at the reduced power levels.

The principal tool used in the analysis was the PHOENIX-P/ANC code system (Topical Report WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," dated June 1988, and Topical Report WCAP-10965-P-A, "A Westinghouse Advanced Nodal Computer Code," dated December 1985.) The calculations were performed by Westinghouse and documented in WCAP-15432, Revision 2, as part of the submittal.

To justify the increase in allowable rod misalignment at a reduced power level, the effects of the additional misalignment on normal operating peaking factors and on safety analysis inputs were evaluated. To perform the analysis of the possible rod misalignments, Westinghouse used one ANC model of Point Beach, Unit 1, and one ANC model of Point Beach, Unit 2. The first model was the currently operating Unit 1 Cycle 26 and represents the current Point Beach licensing basis for fuel and peaking factor limits. The second model was intended to represent a future cycle. As stated in the licensee's May 3, 2001, supplemental letter, "Future core loading patterns will be evaluated and confirmation of the rod misalignment analysis will be validated on a cycle specific basis."

The number and type of rod misalignments analyzed were limited to those permitted by the failure mode and effects analysis performed for the rod control system. The evaluation was limited to single failures within the rod control system logic cabinets, power cabinets, and control rod drive mechanisms. A key assumption in the analysis of the feasible failures was that the current Point Beach licensing basis requires the consideration of a single failure only. The evaluation concluded that six categories of failure mechanisms warranted investigation. The categories are described in WCAP-15432. The cases analyzed involved single and multiple rod misalignments in a single group in either the insertion or withdrawal directions, as well as cases involving all rods in a group misaligned from the group step demand position.

The limits for $F_{\Delta H}$ and F_Q, as specified in the TSs, increase as power level decreases. At 85-percent power, $F_{\Delta H}$ has increased 4.5 percent and F_Q has increased 17.6 percent from the 100-percent power limits. The NRC staff reviewed the results of the misalignment analysis as presented in WCAP-15432, Revision 2. The increases in $F_{\Delta H}$ and F_Q for control rod misalignments of 24 steps for power levels at and below 85 percent of rated power are proprietary values that are less than 4.0 percent and 10.0 percent respectively. Since the peaking factor limits increase more than the changes due to the increased misalignment, the proposed rod misalignment TS limit of ± 24 steps is acceptable for power levels at and below 85 percent of rated power.

The safety analyses parameters were reviewed and it was found that the following parameters are expected to be affected by the increase in the rod misalignment: rod insertion allowance; ejected rod $F_q(z)$; and ejected rod worth (delta Rho_{EJ}). The licensee's analyses show that the maximum effect on the rod insertion allowance will occur upon the misalignment of all the rods at the rod insertion limit in the inserted direction. The analysis showed that the reduction in available shutdown margin was covered by available margin.

The licensee also analyzed rod ejection accidents, including potential misalignment of individual rods, groups and entire banks of rods. The subsequent effects on $F_q(z)$ and delta Rho_{EJ} were determined. Results of the analysis indicated the maximum increases in $F_q(z)$ and in delta Rho_{EJ} for the increased misalignment. The actual calculated values are proprietary. These proprietary values have been conservatively increased. The new proprietary values will be used in the safety evaluation for future cycles. The staff finds this approach acceptable.

TSs 15.3.10-B and 15.3.10-D have been modified to allow up to 1 hour after control rod motion to verify control rod position. This time period is based on the time necessary to allow the control rod drive shaft to reach thermal equilibrium. For purposes of invoking this allowance, a substantial rod movement is required. Allowing 1 hour of thermal soak following an accident occurring during this time is small. In addition, generally, the rods will not be misaligned, only indicated to be misaligned. This 1-hour thermal soak allowance is similar to that previously approved by the NRC staff for other plants and is acceptable.

In summary, control rod misalignments of up to 36 steps (24 steps indicated + 12 steps uncertainty) have been evaluated for their impact on peaking factors and reactivity worth. The results of the analyses showed that the incremental increases in the peaking factors were within the increase in the peaking factor limits for operation at powers 85 percent and below. The change in reactivity worth was also shown to be within the margin available. Thus, it has been shown that the increase in peaking factors will be accommodated at or below 85 percent of rated power and the change to the TSs to allow misalignment of up to 24 steps is acceptable. The 1-hour thermal soak following substantial rod motion is also acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (66 FR 9386). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: M. Chatterton

Date: May 8, 2001