January 16, 1997

Mr. Richard R. Grigg, President &
Chief Operating Officer
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONTROL ROD AND POWER DISTRIBUTION LIMITS (TAC NOS. M94782 AND M94783)

Dear Mr. Grigg:

The Commission has issued the enclosed Amendment Nos. 171 and 175 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments revise the Technical Specifications (TS) in response to your application dated February 8, 1996, as supplemented August 15, 1996, December 2, 1996, December 19, 1996, and January 6, 1997.

These amendments revise TS Section 15.3.10, "Control Rod and Power Distribution Limits," and the associated Bases to improve the clarity of this section. Additionally, requirements to perform core power distribution and shutdown margin testing frequency and applicability statements are added to TS Table 15.4.1-2, "Minimum Frequencies for Equipment and Sampling Tests." The changes were made with the guidance contained in Westinghouse Owner's Group Improved Standard TS, NUREG 1431, Revision 0.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,

Original signed by
Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

Enclosures:
1. Amendment No.171 to DPR-24
2. Amendment No.175 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

<u>DISTRIBUTION</u>: See next page

DOCUMENT NAME: G:\WPDOCS\PTBEACH\PTB94782.AMD *See previous concurrence

OFFICE	PM:PD31	E	LA:PD31	E	*BC:SRXB		*BC:TSB		OGC		D:PD31
NAME	LGundrum:lg		CJamerson		TCollins		CGrimes				JHannon
DATE	1/6/97		1/3/97		12/26/96		1/02/97		1/9/97		1/16/97

OFFCIAL RECORD COPY

230023

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DATE	12/23/96		12/26/96		1/2/97		1/ /97		1/ /97		1 / /97	

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

January 16, 1997

Mr. Richard R. Grigg, President &
Chief Operating Officer
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONTROL ROD AND POWER DISTRIBUTION LIMITS (TAC NOS. M94782 AND M94783)

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Mr. Richard R. Grigg
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated February 8, 1996, as supplemented August 15, December 2, December 19, 1996, and January 6, 1997, 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. 171 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: January 16, 1997

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 175
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated February 8, 1996, as supplemented August 15, December 2, December 19, 1996, and January 6, 1997, 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. 175 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Linda L. Gundrum

Linda L. Gundrum, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of issuance: January 16, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 171 AND 175

TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area 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
15.3.10-5	15.3.10-5
15.3.10-6	15.3.10-6
15.3.10-7	15.3.10-7
15.3.10-8	15.3.10-8
15.3.10-9	15.3.10-9
15.3.10-10	15.3.10-10
15.3.10-11	15.3.10-11
15.3.10-12	15.3.10-12
15.3.10-13	15.3.10-13
15.3.10-14	15.3.10-14
15.3.10-15	15.3.10-15
15.3.10-16	15.3.10-16
--	15.3.10-17
--	15.3.10-18
Table 15.4.1-1 (Page 6 of 6)	Table 15.4.1-1 (Page 6 of 6)
Table 15.4.1-2 (Page 1 of 4)	Table 15.4.1-2 (Page 1 of 5)
Table 15.4.1-2 (Page 2 of 4)	Table 15.4.1-2 (Page 2 of 5)
Table 15.4.1-2 (Page 3 of 4)	Table 15.4.1-2 (Page 3 of 5)
Table 15.4.1-2 (Page 4 of 4)	Table 15.4.1-2 (Page 4 of 5)
--	Table 15.4.1-2 (Page 5 of 5)

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^{O}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^{O}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

1. During power and low power operation, all shutdown and control rods shall be operable, with all individual indicated rod positions within twelve steps of their bank demand position, except when the bank demand position is ≤30 steps or ≥215 steps. In this case, all individual indicated rod positions shall be within 24 steps of their bank demand position.

   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;
AND

(d) Within 72 hours verify that measured values of $F_Q(Z)$ are within limits;
AND

(e) Within 72 hours verify that $F^N_{\Delta H}$ 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; OR within one hour restore the shutdown margin by boration;
AND

(b) Be in hot shutdown within six hours.

C. ROD POSITION INDICATION

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;
AND

(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 ≤12 steps apart, except when the bank demand position is ≤30 steps or ≥215 steps. In this case, once per shift verify that the most withdrawn rod and the least withdrawn rod of the affected banks are ≤24 steps apart;

(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>

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;

AND

b. Within six hours fully withdraw the shutdown banks.

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

2. When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. 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; OR within one hour restore the shutdown margin by boration;

AND

b. Within six hours restore the control banks to within limits.

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

E. POWER DISTRIBUTION LIMITS

1. Hot Channel Factors

a. The hot channel factors defined in the basis shall meet the following limits:

$$F_Q(Z) \leq \frac{(2.50)}{P} \times K(Z) \qquad \textit{for } P > 0.5$$

$$F_Q(Z) \leq 5.00 \times K(Z) \qquad \textit{for } P \leq 0.5$$

$$F^N_{\Delta H} < 1.70 \times [1 + 0.3\,(1-P)]$$

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. If $F_Q(Z)$ exceeds the limit of Specification 15.3.10.E.1.a, within fifteen minutes reduce thermal power until $F_Q(Z)$ limits are satisfied;

(1) After thermal power has been reduced in accordance with Specification 15.3.10.E.1.b, perform the following actions:

(a) Within eight hours reduce the full power Power Range Neutron Flux - High trip setpoints by an amount equivalent to the power reduction required in Specification 15.3.10.E.1.b;
<u>AND</u>

(b) Within 72 hours reduce Overpower and Overtemperature ΔT trip setpoints by an amount equivalent to the power reduction required in Specification 15.3.10.E.1.b;
<u>AND</u>

(c) Verify that $F_Q(Z)$ will be within limits for the increased power level prior to increasing any setpoints that have been reduced and thermal power above the limit specified in Specification 15.3.10.E.1.b.

(d) If the above actions and associated completion times are not met, be in low power operation within the following six hours.

c. If $F^N_{\Delta H}$ exceeds the limit of Specification 15.3.10.E.1.a. within four hours reduce thermal power to restore $F^N_{\Delta H}$ to within limits <u>OR</u> perform the following actions:

(1) Within four hours reduce thermal power to ≤50 percent rated thermal power;
<u>AND</u>

(2) Within eight hours reduce the full power Power Range Neutron Flux - High trip setpoints to ≤58 percent rated thermal power.

In addition to the above actions, the following actions shall also be performed during the subsequent power escalation if $F^N_{\Delta H}$ had exceeded the limit of Specification 15.3.10.E.1.a:

(3) Verify that $F^N_{\Delta H}$ is within limits within 24 hours;
<u>AND</u>

(4) Verify that $F^N_{\Delta H}$ is within limits prior to thermal power exceeding 50 percent of rated thermal power;
<u>AND</u>

(5) Verify that $F^N_{\Delta H}$ is within limits prior to thermal power exceeding 75 percent of rated thermal power;
<u>AND</u>

(6) Verify that $F^N_{\Delta H}$ is within limits within 24 hours after reaching ≥95 percent of rated thermal power.

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

2. <u>Axial Flux Difference</u>

NOTE: The axial flux difference shall be considered outside limits when two or more operable excore channels indicate that axial flux difference is outside limits.

a. During power operation with thermal power ≥50 percent of rated thermal power, the axial flux difference shall be maintained within the limits specified in Figure 15.3.10-4.

(1) If the axial flux difference is not within limits, within 15 minutes restore to within limits. If this action and associated completion time is not met, perform the following actions:

(a) Reduce thermal power until the axial flux difference is within limits;
<u>OR</u>
(b) Within three hours reduce thermal power to ≤50 percent of rated thermal power.

b. If it is necessary to restrict thermal power to ≤50 percent of rated thermal power, within the next four hours reduce the Power Range Neutron Flux - High Trip setpoints to ≤55 percent.

c. If the alarms used to monitor the axial flux difference are rendered inoperable, verify that the axial flux difference is within limits for each operable excore channel once within one hour and every hour thereafter.

3. <u>Quadrant Power Tilt</u>

a. During power operation with thermal power greater than 50 percent of rated thermal power, the indicated quadrant power tilt shall not exceed 2 percent. If this condition is not met, perform the following actions:

(1) Within two hours, reduce thermal power ≥2 percent from rated thermal power for each 1 percent of indicated quadrant power tilt;
<u>AND</u>
(2) Within 24 hours and once per seven days thereafter, verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the limits of Specification 15.3.10.E.1.a;
<u>AND</u>

(3) Upon completion of Specification 15.3.10.E.3.a(2), calibrate the excore detectors. This action shall be completed prior to increasing thermal power above the limit imposed by Specification 15.3.10.E.3.a(1);
<u>AND</u>

(4) Verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the limits of Specification 15.3.10.E.1.a within 24 hours after reaching rated thermal power, or within 48 hours after increasing thermal power above the limit imposed by Specification 15.3.10.E.3.a(1).

(5) If the above actions and associated completion times are not met, within the following four hours reduce thermal power to ≤50 percent of rated thermal power.

b. If no quadrant power tilt alarms are available, within twelve hours and every twelve hours thereafter, verify that quadrant power tilt is within limits by performing calculations.

c. When one power range channel is inoperable and thermal power is greater than 75% of rated thermal power, within twelve hours and every twelve hours thereafter, verify that quadrant power tilt is within limits by use of the movable incore detectors.

F. <u>AT-POWER PHYSICS TESTS EXCEPTIONS</u>

1. During the performance of at-power physics tests, the requirements of:

Specification 15.3.10.B, "Rod Operability and Bank Alignment Limits"
Specification 15.3.10.D, "Bank Insertion Limits"
Specification 15.3.10.E.2. "Axial Flux Difference"
Specification 15.3.10.E.3. "Quadrant Power Tilt"

are suspended, provided:

a. Thermal power is maintained ≤85 percent of rated thermal power:
<u>AND</u>

b. Power Range Neutron Flux - High Trip setpoints are set at a maximum setting of 90 percent of rated thermal power;

2. Within 8 hours prior to the initiation of physics tests, verify that Power Range Neutron Flux - High Trip setpoints are ≤90 percent of rated thermal power.

3. If the shutdown margin is not within the limits of Specification 15.3.10.A.1, within 15 minutes initiate boration to restore the shutdown margin, AND within one hour suspend physics tests exceptions.

4. If thermal power exceeds 85 percent of rated thermal power, within one hour reduce thermal power to ≤85 percent of rated thermal power, OR within one hour suspend physics tests exceptions.

5. If the Power Range Neutron Flux - High Trip setpoints are greater than 90 percent of rated thermal power, within one hour restore the Power Range Neutron Flux - High Trip setpoints to ≤90 percent of rated thermal power, OR within one hour suspend physics tests exceptions.

6. Every hour, while at-power physics tests are in progress, verify that thermal power is ≤85 percent of rated thermal power.

7. At least once every 12 hours, verify $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits.

G. LOW POWER PHYSICS TESTS EXCEPTIONS

1. During the performance of low power physics tests, the requirements of:

   Specification 15.3.10.B, "Rod Operability and Bank Alignment Limits"
   Specification 15.3.10.D, "Bank Insertion Limits"
   Specification 15.3.10.E, "Power Distribution Limits"

   are suspended, provided the lowest RCS loop average temperature is greater than the minimum temperature for criticality.

2. If the shutdown margin is not within the limits of Specification 15.3.10.A, within 15 minutes initiate boration to restore the shutdown margin, AND within one hour suspend physics tests exceptions.

3. If power is not within limits, open the rector trip breakers immediately.

4. If lowest RCS loop average temperature is less than the minimum temperature for criticality, within 15 minutes restore lowest RCS loop average temperature to within limits, OR within 30 minutes be subcritical.

H. RCCA DROP TIMES

1. With RCS temperature greater than the minimum temperature for criticality and with both reactor coolant pumps running, the drop time of each RCCA shall be no greater than 2.2 seconds from the loss of stationary gripper coil voltage to dashpot entry. If this condition is not met, perform the following actions:

   a. If the reactor is critical, declare the rod untrippable;
   OR
   b. If the reactor is subcritical, maintain the reactor subcritical.

Basis

Insertion Limits and Shutdown Margin

During power operation, the shutdown banks are fully withdrawn. Fully withdrawn is defined as a bank demand position equal to or greater than 225 steps. Evaluation has shown that positioning control rods at 225 steps, or greater, has a negligible effect on core power distributions and peaking factors. Due to the low reactivity worth in this region of the core and the fact that, at 225 steps, control rods are only inserted one step into the active fuel region of the core, positioning rods at this position or higher has minimal effect. This position is varied, based on a predetermined schedule, in order to minimize wear of the RCCA's from the guide cards.

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

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

If the shutdown margin requirements are not met, boration must be initiated promptly. Fifteen minutes is an adequate period of time for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until shutdown margin requirements are met.

Rod Operability Requirements and Bank Alignment Limits

The operability (e.g. trippability) of the shutdown and control rods is an initial assumption in all safety analyses that take credit for rod insertion upon reactor trip. Maximum rod misalignment is also an initial assumption in the safety analyses that directly affect core power distributions and assumptions of available shutdown margin. A rod cluster control assembly (RCCA) shall be considered operable if the RCCA drops upon removal of stationary gripper coil voltage.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution. This will also cause a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and operability are related to core operation in design power peaking limits and the core design requirement of a minimum shutdown margin.

From operating experience to date, an RCCA which steps in properly will drop when a trip signal occurs because the only force acting to drive the rod in is gravity. When it has been determined that a rod does not drop, the shutdown margin calculation will need to include the worth of the inoperable control rod. Further experience indicates that control rods which do not step are usually affected by electrical problems. That is, normally the problem is in the rod control cabinets.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of one or two groups that are moved in staggered fashion, but always within one step of each other. Each unit has four control banks and two shutdown banks.

When one or more rods are determined to be untrippable, there is a possibility that the required shutdown margin may be adversely affected. Under these conditions, it is important to determine the shutdown margin, and if it is less than the required value, initiate boration until the required shutdown margin is restored. The one-hour time limit is adequate for determining the shutdown margin and, if necessary, for restoring the shutdown margin by boration. In this situation, shutdown margin verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

If the untrippable rods cannot be restored to an operable condition, the plant must be placed in a condition where the LCO requirements are not applicable. To achieve this status, the unit must be placed in hot shutdown within six hours. This allows this plant condition to be reached in an orderly manner, without challenging any plant systems.

Limits on control rod alignment have been established and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and shutdown margin limits are preserved.

If the misalignment condition cannot be readily corrected, thermal power will be adjusted so that hot channel factors are maintained, and so that the requirements on shutdown margin and ejected rod worth are preserved. Continued operation of the reactor with a misaligned control rod is allowed if $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, axial flux difference limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping.

Upon detection of a potential problem concerning one or more rods, a maximum of six hours is provided for troubleshooting activities. Immediately upon determining that one or more rods is inoperable, the applicable actions in TS 15.3.10.B shall be performed. If after six hours, an operability determination has not yet been made, the rod(s) shall be declared inoperable and the applicable actions in TS 15.3.10.B shall be performed.

Rod Position Indication

During power operation at greater than ten percent of rated thermal power, the rod position indication system and the bank demand position indication system are required to be operable. These systems are required to be operable because the position of rods must be determined in order to ensure that rod alignment and insertion limits are being satisfied. Rod position accuracy is essential during power operations. Power peaking, ejected rod worth, or shutdown margin limits may be violated in the event of a design basis accident with rods operating, undetected, outside of their required limits.

The various control rod banks (shutdown banks and control banks, A, B, C, and D) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Direct information on rod position indication is provided by two methods: A digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position. The rod position indicator channel has a demonstrated accuracy of 5% of span (±11.5 steps). Therefore, an analysis has been performed to show that a misalignment of 24 steps cannot cause design hot channel factors to be exceeded. A single fully misaligned RCCA, that is, an RCCA 230 steps out of alignment with its bank, does not result in exceeding core limits in steady-state operation at power levels less than or equal to rated power. In other words, a single dropped RCCA is allowable from a core power distribution viewpoint. If the misalignment condition cannot be readily corrected, the specified reduction in power to 75% will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The eight (8) hour permissible limit on rod misalignment at rated power is short with respect to the probability of an independent accident.

Because the rod position indicator system may have a 12 step error when a misalignment of 24 steps is occurring, the Specification allows only an indicated misalignment of 12 steps. However, when the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the consequences of a misalignment are much less severe. The differential worth of an individual RCCA is less, and the resultant perturbation on power distributions is less than when the bank is in its high differential worth region. At the top and bottom of the core, an indicated 24 step misalignment may be representing an actual misalignment of 36 steps.

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 LVDT's 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.

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, when the bank demand position is between 30 steps and 215 steps. A misalignment of 36 steps is allowed when the bank position is less than or equal to 30 steps, or, when the bank position is greater than or equal to 215 steps, due to the small worth and consequential effects of an individual rod misalignment.

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

3. 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 $p \leq 0.5$.

An upper bound envelope of 2.50 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 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^{O}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 $F^N_{\Delta H} \leq 1.70/1.08$. 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.

Axial Power Distribution

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^{O} 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 positioned outside their specified alignment and insertion limits to conduct physics tests 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.

NOTES USED IN TABLE 15.4.1-1 (continued)

(10) When used for the Overpressure Mitigating System, each PORV shall be demonstrated operable by:
a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

(11) Performance of a channel functional test is required, excluding valve operation.

(12) Shiftly check is required when the reactor coolant system is not open to the atmosphere and the reactor coolant system temperature is less than the minimum temperature for the in-service pressure test as specified in TS Figure 15.3.1-1.

(13) An AFW flow path to each steam generator shall be demonstrated operable, following each cold shutdown of greater than 30 days, prior to entering power operation by verifying AFW flow to each steam generator.

(14) Calibration is to be a verification of response to a source.

(15) Sample gas for calibration at 2% and 6%.

(16) A check of one pressure channel per steam generator is required whenever the steam generator could be pressurized.

(17) Includes test of logic for reactor trip on low-low level, automatic actuation logic for auxiliary feedwater pumps, and test of logic for feedwater isolation on high steam generator level.

(18) Rod positions must be logged at least once per hour, after a load change >10% or after >30 inches of control rod motion if the on-line computer is inoperable.

(19) The daily heat balance is a gain adjustment performed to match Nuclear Instrumentation System indicated power level with reactor thermal output.

(20) To confirm that hot channel factor limits are being satisfied, the requirements of TS 15.3.10.E must be met.

(21) Check required only when the overpressure mitigation system is in operation.

(22) Not required during period of cold and refueling shutdowns, but must be performed prior to reactor criticality if it has not been performed during previous surveillance period.

(23) Each train tested at least every 62 days on a staggered basis.

(24) Neutron detectors excluded from calibration.

TABLE 15.4.1-2
MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency
1.	Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week[7]
		Tritium activity	Monthly
		Radiochemical $\bar{E}$ Determination	Semiannually [2,10]
		Isotopic Analysis for Dose Equivalent I-131 Concentration	Every two weeks[1]
		Isotopic Analysis for Iodine including I-131, I-133, and I-135	a.) Once per 4 hours whenever the specific activity exceeds 1.0μCi/gram Dose Equivalent I-131 or 100/$\bar{E}$ μCi/gram.[6] b.) One sample between 2 and 6 hours following a thermal power change exceeding 15% of rated power in a one-hour period.
		Chloride Concentration	5/week[8]
		Diss. Oxygen Conc.	5/week[6]
		Fluoride Conc.	Weekly
2.	Reactor Coolant Boron	Boron Concentration	Twice/week
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly[6]
4.	Boric Acid Tanks	Boron Concentration	Twice/week and after each BAST concentration change when they are being relied upon as a source of borated water.
5.	Spray Additive Tank	NaOH Concentration	Monthly
6.	Accumulator	Boron Concentration	Monthly

		Test	Frequency
7.	Spent Fuel Pit	a) Boron Concentration b) Water Level Verification	Monthly Weekly
8.	Secondary Coolant	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly[6]
		Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.2 μCi/cc [6]
9.	Control Rods	a) Rod drop times of all full length rods [3]	Each refueling or after maintenance that could affect proper functioning [4]
		b) Rodworth measurement	Following each refueling shutdown prior to commencing power operation
10.	Control Rod	Partial movement of all rods	Every 2 weeks [18]
11.	Pressurizer Safety Valves	Set point	Every five years [11]
12.	Main Steam Safety Valves	Set Point	Every five years [11]
13.	Containment Isolation Trip	Functioning	Each refueling shutdown
14.	Refueling System Interlocks	Functioning	Each refueling shutdown
15.	Service Water System	Functioning	Each refueling shutdown
16.	Primary System Leakage	Evaluate	Monthly [6]
17.	Diesel Fuel Supply	Fuel inventory	Daily
18.	Turbine Stop and Governor Valves	Functioning	Annually [6]
19.	Low Pressure Turbine Rotor Inspection [5]	Visual and magnetic particle or liquid penetrant	Every five years
20.	Boric Acid System	Storage Tank and piping temperatures ≥ temperature required by Table 15.3.2-1	Daily[19]

		Test	Frequency
21.	PORV Block Valves	a. Complete Valve Cycle b. Open position check	Quarterly (13) Every 72 hours (14)
22.	Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Each refueling cycle
23.	Containment Purge Supply and Exhaust Isolation Valves	Verify valves are locked closed	Monthly (9)
24.	Reactor Trip Breakers	a. Verify independent operability of automatic shunt and undervoltage trip functions.	Monthly (9)
		b. Verify independent operability of manual trip to shunt and undervoltage trip functions.	Each refueling shutdown
25.	Reactor Trip Bypass Breakers	a. Verify operability of the undervoltage trip function.	Prior to breaker use
		b. Verify operability of the shunt trip functions.	Each refueling shutdown
		c. Verify operability of the manual trip to undervoltage trip functions.	Each refueling shutdown
26.	120 VAC Vital Instr. Bus Power	Verify Energized(12)	Shiftly
27.	Power Operated Relief Valves (PORVs), PORV Solenoid Air Control Valves, and Air System Check	Operate(16)	Each shutdown(15)
28.	Atmospheric Steam Dumps	Complete valve cycle	Quarterly
29.	Crossover Steam Dump System	Verify operability of each steam dump valve.	Quarterly

30.	Pressurizer Heaters	Verify that 100 KW of heaters are available.	Quarterly
31.	CVCS Charging Pumps	Verify operability of pumps.(17)	Quarterly
32.	Potential Dilution in Progress Alarm	Verify operability of alarm.	Prior to placing plant in cold shutdown.
33.	Core Power Distribution	Perform power distribution maps using movable incore detector system to confirm hot channel factors.	Monthly (20)
34.	Shutdown Margin	Perform shutdown margin calculation.	Daily (21)

(1) Required only during periods of power operation.
(2) $\bar{E}$ determination will be started when the gross activity analysis of a filtered sample indicates ≥10μCi/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10μCi/cc.
(3) Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot condition, but cold drop tests need not be timed.
(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.
(5) As accessible without disassembly of rotor.
(6) Not required during periods of refueling shutdown.
(7) At least once per week during periods of refueling shutdown.
8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.
9) Not required during periods of cold or refueling shutdown, but must be performed prior to exceeding 200°F if it has not been performed during the previous surveillance period.
0) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer.
1) An approximately equal number of valves shall be tested each refueling outage such that all valves will be tested within a five year period. If any valve fails its tests, an additional number of valves equal to the number originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested.
) The specified buses shall be determined energized in the required manner at least once per shift by verifying correct static transfer switch alignment and indicated voltage on the buses.
) Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat leakage.

Only applicable when the overpressure mitigation system is in service.

Required to be performed only if conditions will be established, as defined in Specification 15.3.15, where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 171 AND 175 TO

FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated February 8, 1996, as supplemented August 15, December 2, December 19, 1996, and January 6, 1997, Wisconsin Electric Power Company (WEPCO, the licensee) requested amendments to the Technical Specification (TS) appended to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant (PBNP), Unit Nos. 1 and 2. The proposed amendments would change Section 15.3.10, "Control Rod and Power Distribution Limits," to improve the clarity of this section. The licensee also proposed a surveillance change to Section 15.4.1. The above referenced changes are carried out in accordance with the guidance provided in NUREG-1431, "The Westinghouse Owner's Group Improved Standard Technical Specifications (ISTS)," Revision 0. The staff reviewed the proposed amendment for consistency with NUREG-1431, Revision 1. References to the licensee's submittal for NUREG-1431 relate to Revision 0 whereas references to NUREG-1431 in staff conclusions relate to Revision 1.

Specification 15.3.10, "Control Rod and Power Distribution Limits," ensures core subcriticality following a reactor trip; it places a limit on possible reactivity insertion due to a hypothetical rod cluster control assembly (RCCA) ejection, and it also ensures an acceptable core power distribution exists during normal power operations. The letters dated August 15, December 2, December 19, 1996, and January 6, 1997, provided clarifying information and updated TS pages that were within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The current shutdown margin TS at the Point Beach Nuclear Plant does not stipulate a time limit at which time operators would start to borate as necessary to restore the shutdown margin to the value specified in the TS. The change requested by WEPCO would add an action statement to the TS requiring the operators to initiate boration within 15 minutes to restore shutdown margin requirements. This is consistent with NUREG-1431, and the staff finds this administrative change acceptable.

The safety analyses associated with a reactor trip assumes operability of shutdown and control rods. If one or more of the rods were determined to be inoperable, the possibility exists that the shutdown margin may be affected. If the shutdown margin requirements are not met, boration is initiated.

However, at the present time, there is no time limit imposed on how long it should take to recover the shutdown margin. The change requested by the licensee would put a 1-hour time limit to accomplish/recover the shutdown margin. The time to recover the shutdown margin did not previously appear in the PBNP TS. However, the proposed 1-hour limit is consistent with NUREG-1431 and is more restrictive than the existing TS. The staff finds this change acceptable.

Rod misalignment is also part of the safety analyses associated with power distributions and shutdown margin. Should a rod misalignment occur, the operators are required to restore the misaligned rod to within the alignment limits within 1-hour. If this is not possible, the thermal power is reduced to less than or equal to 75 percent power within 8 hours. This time period is consistent with the existing TS for PBNP.

The $F_Q(z)$ and $F^N_{\Delta H}$ must be verified within their limit during continued operation with a misaligned rod. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, axial flux difference limits, and quadrant power tilts limits are not preserved. Consequently, peaking factor limits may not be preserved. As a result, $F_Q(z)$ and $F^N_{\Delta H}$ must be verified by incore mapping. These proposed revisions are consistent with the requirements of NUREG-1431 and will impose stricter requirements than those imposed by the existing TS.

The licensee also made changes (added additional requirements) to the control rod position indication system to make the specifications consistent with those of NUREG-1431.

The licensee proposed changes associated with bank insertion limits. The licensee proposed that the operator be required to verify that the shutdown margin is within limits within the span of an hour or initiate boration to restore the shutdown margin within 1-hour. The licensee also proposed that the shutdown banks be fully withdrawn within 2 hours. If these actions cannot be performed, the operator will put the reactor in hot standby within the following 6 hours. The staff finds these changes acceptable since they enhance the safety function at PBNP.

Similarly, the licensee proposes to add requirements to the TS concerning operator actions should the control banks insertion limits be violated. The proposed changes would require that the operator restore the control banks to within limits within 2 hours. These changes are more restrictive than the existing TS and therefore are acceptable.

The licensee also proposed changes (added actions) when monitoring the heat flux hot channel factor and the nuclear enthalpy hot channel factor. The proposed additional actions would delineate specific operator action that must be performed, including power reductions and readjustments of any reactor protection system setpoints. The proposed changes are more prescriptive and will enhance the clarity of the TS. An additional change was required to modify the Note 20 for TS Table 15.4.1-1 to reflect the correct TS reference as 15.3.10.E instead of 15.3.10.B. The proposed additional actions are consistent with NUREG-1431. The staff finds the proposed changes acceptable.

Current TS are in place at PBNP to prevent the limits on axial flux difference from being exceeded. Operator actions are being rewritten to more clearly define required operator actions.

Changes are proposed regarding the quadrant power tilt. If the quadrant power tilt exceeds 2 percent with thermal power greater than or equal to 50 percent of rated power, the licensee proposed to reduce power greater than or equal to 2 percent from rated thermal power for each 1 percent of indicated power tilt, reevaluate safety analyses and confirm results remain valid for duration of operation under this condition, and within 24 hours and once per 7 days thereafter, verify $F_Q(z)$ and $F^N_{\Delta H}$ are with TS 15.3.10.E.1.a limits. This is consistent with the current TS at PBNP and additional actions are being added to the TS in accordance with the guidance suggested in NUREG-1431. The limitations imposed on the heat flux hot channel factor and the enthalpy rise hot channel factor will ensure that the core power distributions are maintained within design limits. An administrative change was made to relocate the monthly surveillance requirements for the hot channel factors to TS Table 15.4.1-2. The proposed changes are acceptable to the staff.

During physics testing, current PBNP TS do not identify the plant conditions that must be maintained during at-power or low-power physics testing. The licensee is proposing to add two sections to TS Section 15.3.10 to specify which conditions must be maintained, and which requirements may be suspended during the performance of these tests, and what actions should be taken should any of these required plant conditions be violated. These additions to the TS will enhance the TS by providing guidance to plant operators in an area where none currently exists. The licensee also added to TS Section 15.4.1-2 a daily surveillance of shutdown margin during the low power physics testing. The proposed changes are consistent with the guidance contained in NUREG-1431. The staff finds these changes acceptable.

The licensee has revised TS 15.3.10.H to identify the actual plant conditions when the actual drop times for the RCCAs are performed. Typically, RCCA drop times must be no greater than 2.2 seconds. The licensee is revising the TS to require actions be taken should a drop exceed 2.2 seconds. These actions are identical to those currently being performed at PBNP. The staff finds these revisions acceptable.

Finally, the licensee requests a change to the bases of Section 15.3.10 to reflect the modifications made to the TS and incorporate the information associated with the new proposed additions. The added background information is consistent with bases information in NUREG-1431. The staff agrees with the requested change.

3.0 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.

4.0 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 and change a surveillance requirement. The 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 (61 FR 10398). 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 staff 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: Anthony Attard

Date: January 16, 1997

DATED: January 16, 1997

AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-24 - POINT BEACH UNIT NO. 1
AMENDMENT NO. 175 TO FACILITY OPERATING LICENSE NO. DPR-27 - POINT BEACH UNIT NO. 2

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