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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
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 March 14, 1983, as modified September 6, 1983 and July 13, 1984, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective upon completion of the next refueling which ends approximately May 30, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 5, 1984

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
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 March 14, 1983, as modified September 6, 1983 and July 13, 1984, 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 90, 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 the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 5, 1984

ATTACHMENT TO LICENSE AMENDMENT NOS. 86 AND 90

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

DOCKET NOS. 50-266 and 50-301

Revise Appendix A as follows:

Remove pages	Insert pages
15.1-4	15.1-4
15.2.1-1	15.2.1-1
15.2.1-2	15.2.1-2
15.2.1-3	15.2.1-3
Figure 15.2.1-1	Figure 15.2.1-1
-	Figure 15.2.1-2
15.2.3-2	15.2.3-2
15.2.3-3	15.2.3-3
15.2.3-6	15.2.3-6
15.2.3-7	15.2.3-7
15.3.1-17	15.3.1-17
15.3.1-18	15.3.1-18
15.3.1-19	15.3.1-19
15.3.6-1	15.3.6-1
15.3.6-2	15.3.6-2
15.3.8.4	15.3.8.4
15.3.10-2	15.3.10-2
15.3.10-3	15.3.10-3
15.3.10-4	15.3.10-4
15.3.10-7	15.3.10-7
15.3.10-9	15.3.10-9
15.3.10-11	15.3.10-11
15.3.10-12	15.3.10-12
15.3.10-13	15.3.10-13
Figure 15.3.10-1	Figure 15.3.10-1
Figure 15.3.10-3	Figure 15.3.10-3
-	Figure 15.3.10-4
Table 15.4.1-1 (1 of 4)	Table 15.4.1-1 (1 of 4)
15.5.3-1	15.5.3-1
15.5.3-2	15.5.3-2
15.5.4-1	15.5.4-1

2) Cold Shutdown

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

3) Refueling Shutdown

The reactor is in the refueling shutdown condition when the reactor is subcritical by at least 5% $\Delta k/k$ and T_{avg} is $\leq 140°F$. A refueling shutdown refers to a shutdown to move fuel to and from the reactor core

4) Shutdown Margin

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

h. Power Operation

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

i. Refueling Operation

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

j. Rated Power

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

k. Thermal Power

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

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1 for a transition core* and 15.2.1-2 for the full optimized fuel assembly (OFA) core. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

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*Transition core is defined as being any core loading pattern consisting of standard and OFA 14x14 fuel assemblies, but not including cores consisting of standard assemblies and OFA demonstration assemblies only.

Basis:

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant temperature and pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95% probability at a 95% confidence level that DNB will not occur during steady state operation, normal operational transients, and anticipated transients and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 15.2.1-1 are applicable to transition cores. The W-3 correlation is used to generate these curves. The DNBR limit of this correlation is shown to be met in plant safety analyses using values of input parameters with uncertainties considered at fixed conservative values.

The curves of Figure 15.2.1-2 are applicable for a full core of 14x14 OFA. The WRB-1 correlation is used to generate these curves. Uncertainties in plant parameters are statistically convoluted to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a value of design limit DNBR. This value of design limit DNBR is shown to be met in plant safety analyses, using values of input parameters considered at their nominal values.

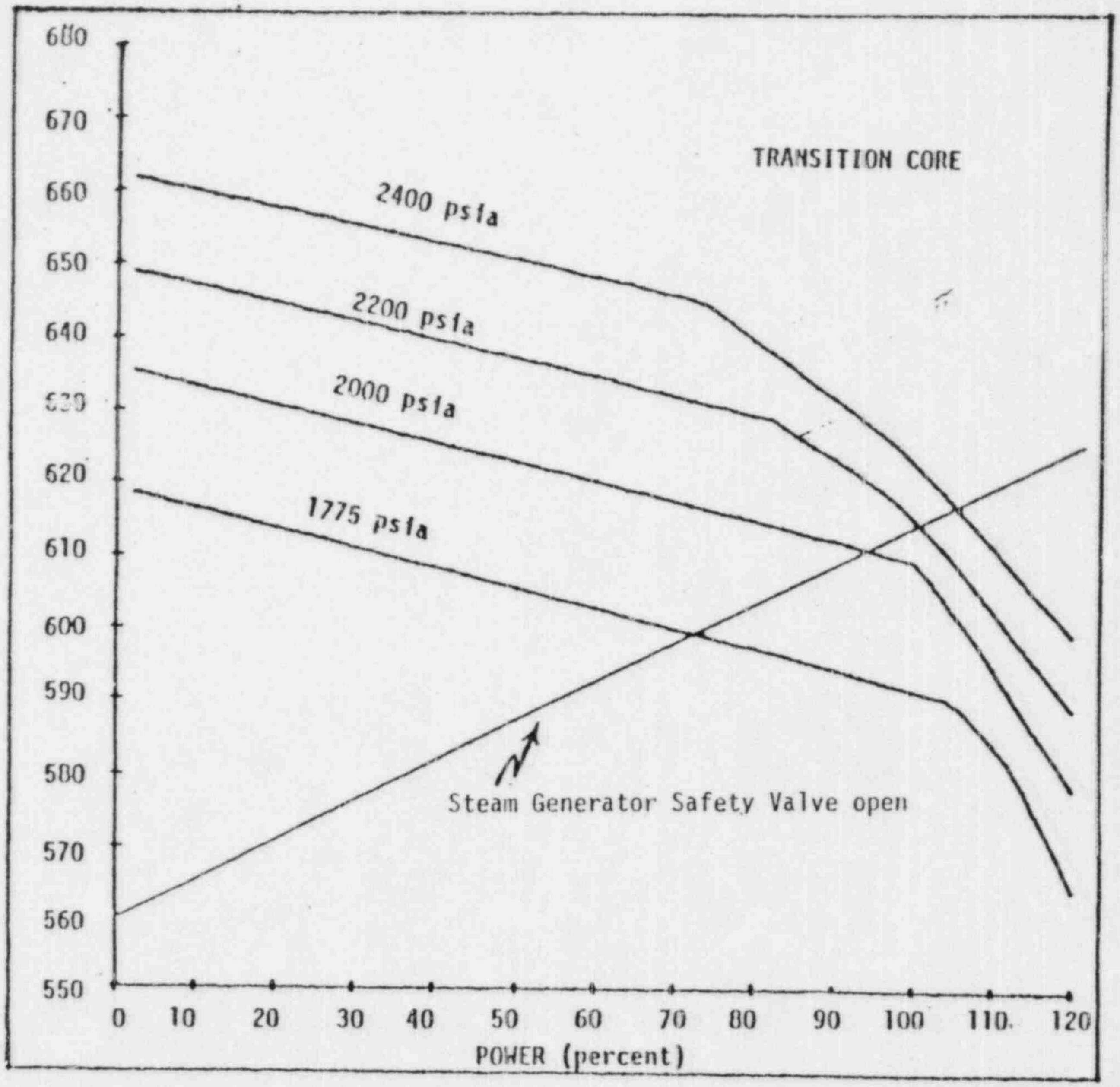
These curves represent the loci of points of thermal power, Reactor Coolant System pressure and average temperature for which the calculated DNBR is no less than the design limit-DNBR or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Appropriate rod bow penalties have been included in the generation of these curves. The effects of fuel densification and possible clad flattening have also been taken into account.

An allowance is included in these curves for an increase in $F_{\Delta H}^{N}$ at reduced power based on the expression:

$$F_{\Delta H}^{N} = \text{Full Power } F_{\Delta H}^{N} \ [1 + 0.3\ (1 - P)]$$

where P is a fraction of rated thermal power.

The hot channel factors are sufficiently large to account for the degree of malpositioning of full-length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached. The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than the design limit DNBR.



T_{avg} - (°F)

Figure 15.2.1-1

CORE DNB SAFETY LIMITS (TRANSITION CORE)
POINT BEACH UNITS 1 & 2

Unit 1 - Amendment No. 86
Unit 2 - Amendment No. 21, 90

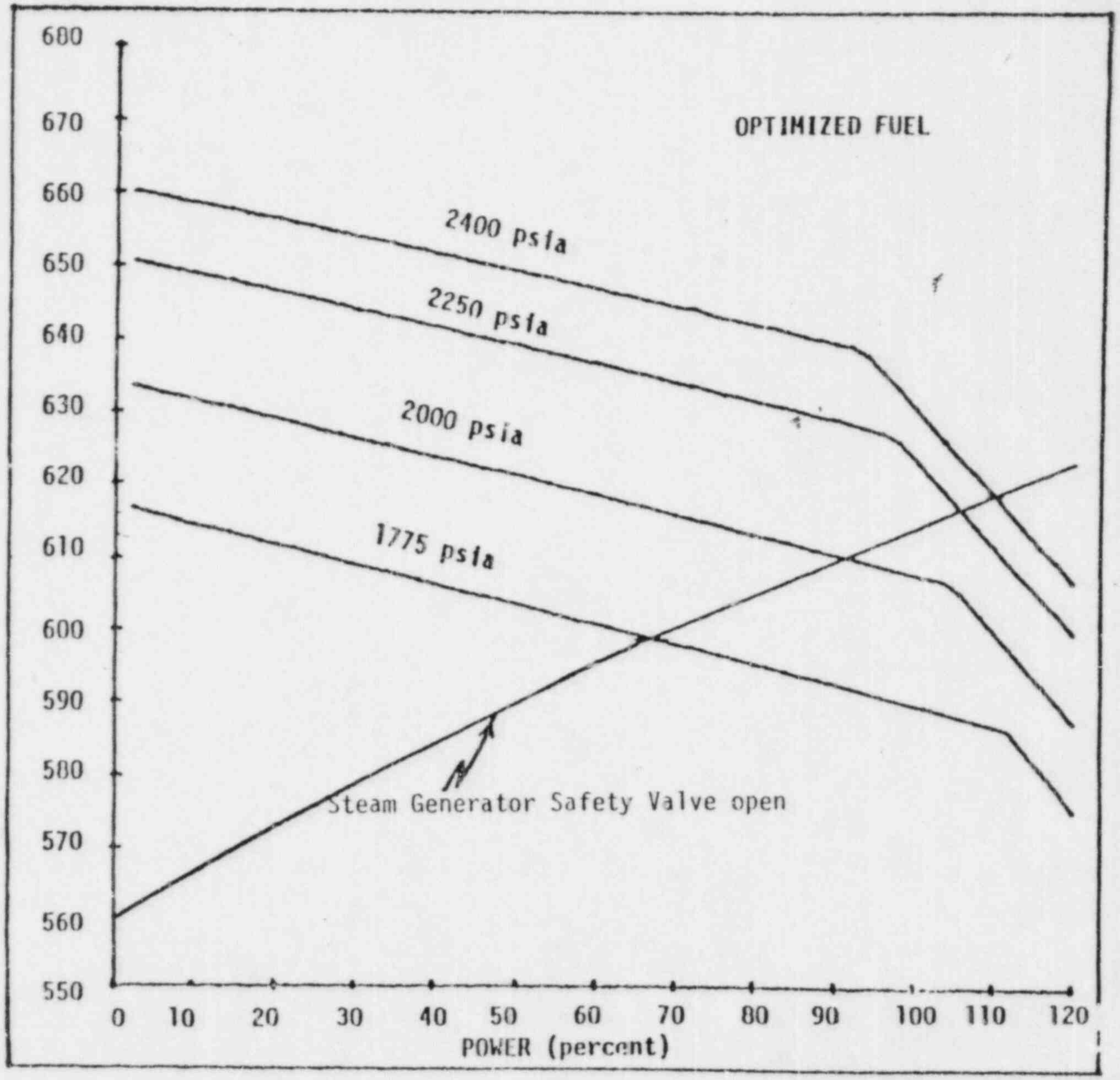


Figure 15.2.1-2

CORE DNB SAFETY LIMITS (OPTIMIZED FUEL)

POINT BEACH UNITS 1 & 2

Unit 1 - Amendment No. 86

Unit 2 - Amendment No. 90

*(3) Low pressurizer pressure - $\geq$1865 psig for operation at 2250 psia primary system pressure
$\geq$1790 psig for operation at 2000 psia primary system pressure

(4) Overtemperature ΔT

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

where

ΔTo = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 574.2°F

P = pressurizer pressure, psig

P' = 2235 psig

*K_1 $\leq$1.117 for operation at 2250 psia primary system pressure
$\leq$1.30 for operation at 2000 psia primary system pressure

K_2 = 0.0150

K_3 = 0.000791

$\tau 1$ = 25 sec

$\tau 2$ = 3 sec

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ with -17, +5 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent the ΔT trip set point shall be automatically reduced by an equivalent of 2.0 percent of rated power.

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*Appropriate safety analyses shall be performed prior to shifting operation from one primary system pressure to the other.

(c) for each percent that the magnitude of $q_t - q_b$ exceeds -17 percent the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

[1.B (5)] Overpower ΔT

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

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T' = 574.2°F

K_4 $\leq$ 1.089 of rated power

K_5 = 0.0262 for increasing T

= 0.0 for decreasing T

K_6 = 0.00123 for $T \geq T'$

= 0.0 for $T < T'$

τ_3 = 10 sec

f (ΔI) as defined in (4) above,

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

(7) Indicated reactor coolant flow per loop - $\geq$90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open

(a) Low frequency set point $\geq$ 57.5 cps

(b) Low voltage set point $\geq$ 75% of normal voltage power distribution, the reactor trip limit, with allowance for errors,[2] is always below the core safety limit as shown on Figures 15.2.1-1 and 15.2.1-2. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6][7]

The overpower, overtemperature and pressurizer pressure system setpoints have been revised to include effect of reduced system pressure operation (including the effects of fuel densification). The revised setpoints as given above will not exceed the revised core safety limits as shown in Figures 15.2.1-1 and 15.2.1-2.

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[4]

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis.[8] The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency setpoint, 57.5 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error[2] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[9]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed. Specifications 15.2.3.2.A(1) and 15.2.3.2.C have a ± 1% tolerance to allow for a 2% deadband of the P10 bistable which is used to set the limit of both items. The difference between the nominal and maximum allowed value (or minimum allowed value) is to account for "as measured" rack drift effects.

Sustained operating with only one pump will not be permitted above 10% power. If a pump is lost while operating between 10% and 50% power, an orderly and immediate reduction in power level to below 10% is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.

References

(1) FSAR 14.1.1
(2) FSAR, Page 14-3
(3) FSAR 14.2.6
(4) FSAR 14.3.1
(5) FSAR 14.1.2
(6) FSAR 7.2, 7.3
(7) FSAR 3.2.1
(8) FSAR 14.1.9
(9) FSAR 14.1.11

F. MINIMUM CONDITIONS FOR CRITICALITY

Specification:

1. Except during low power physics tests, the reactor shall not be made critical when the moderator temperature coefficient is more positive than 5 pcm/°F.

2. Reactor power shall not exceed 70 percent of Rated Power if the moderator temperature coefficient is positive.

3. In no case shall be reactor be made critical (other than for the purpose of low level physics tests) to the left of the reactor core criticality curve presented in Figures 15.3.1-1 for Unit 1 and 15.3.1-3 for Unit 2.

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis:

During the early part of the fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below 70 percent of rated thermal power.(1)(2) The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative when $\geq$ 70 percent of rated thermal power. Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

The limitations of the moderator temperature coefficient are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the Reactor Core Criticality Curve provides assurance that a proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. However, as provided in 10 CFR Part 50 Appendix G Section IV.A.2.c, the reactor core may be taken critical below this curve for the purpose of low level physics tests.

If the specified shutdown margin is maintained (Section 15.3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-9

(3) FSAR Figure 3.2.1-10

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown during Rated Power operation:

1. T_{AVG} shall be maintained at or below 578°F.

*2. Reactor coolant system pressure shall be maintained:
   $\geq$ 2205 psig during operation at 2250 psia, or
   $\geq$ 1955 psig during operation at 2000 psia.

3. Reactor Coolant System raw measured Total Flow Rate

   $\geq$ 181,800 gpm (See Basis).

Basis:

The reactor coolant system total flow rate of 181,300 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimeter at the beginning of each cycle.

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*Appropriate safety analyses shall be performed prior to shifting operation from one primary system pressure to the other.

15.3.6 CONTAINMENT SYSTEM

Applicability:

Applies to the integrity of reactor containment.

Objective:

To define the operating status of the reactor containment for plant operation.

Specification:

A. Containment Integrity

a) The containment integrity (as defined in 15.1) shall not be violated when a nuclear core is installed in the reactor unless the reactor is in the cold shutdown condition.

b) The containment integrity shall not be violated when the reactor vessel head is removed unless the reactor is in the refueling shutdown condition.

c) Positive reactivity changes shall not be made by rod drive motion when the containment integrity is not intact except for the testing of one bank of rods at a time, rod disconnecting, and rod reconnecting provided the reactor is initially subcritical by at least 5% $\Delta k/k$.

d) Positive reactivity changes shall not be made by boron dilution when the containment integrity is not intact unless the boron concentration in the reactor is maintained $\geq$ 1800 ppm.

B. Internal Pressure

If the internal pressure exceeds 3 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor rendered subcritical.

C. Containment Purge Supply and Exhaust Valves

The containment purge supply and exhaust valves shall be locked closed and may not be opened unless the reactor is in the cold shutdown or refueling shutdown condition.

Basis:

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if the Reactor Coolant System ruptures.

The shutdown conditions of the reactor are selected based on the type of activities that are being carried out. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality under any occurrence. During refueling the reactor is subcritical by [illegible] $\Delta k/k$. Positive reactivity changes for the purpose of rod assembly testing will not result in criticality because no control bank worth exceeds 3%. Positive reactivity changes by boron dilution may be required or small concentration fluctuations may occur during preparation for, recovery from, or during refueling but maintaining the boron concentration greater than 1800 ppm precludes criticality under these circumstances. 1800 ppm is a nominal value that ensures 5% shutdown for typical reload cores. Should continuous dilution occur, the time intervals for this incident are discussed in Section 14.1.5 of the FSAR.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 6 psig.(1) The containment is designed to withstand an internal vacuum of 2.0 psig.(2)

The containment purge supply and exhaust valves are required to be locked closed during plant operations since these valves have not been demonstrated capable of closing from the full open position during a design basis loss-of-coolant subcritical approximately by 5% $\Delta k/k$ in the cold condition with all rods inserted.[2] Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. Part A6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

During the refueling operation a substantial number of station personnel and perhaps some regulatory people will be in the containment. The requirements are to prevent an unsafe amount of radioactivity from escaping to the environment in the case of a refueling accident, and also to allow safe avenues of escape for the personnel inside the containment as required by the Wisconsin Department of Industry, Labor and Human Relations. To provide for these requirements, the personnel locks (both doors) are open for the normal refueling operations with a third temporary door which opens outward installed across the outside end of the personnel lock.[3] This hollow metal third door is equipped with weather stripping and an automatic door closer to minimize the exchange of inside air with the outside atmosphere under the very small differential pressures expected while in the refueling condition. Upon sounding of the containment evacuation alarm, all personnel will exit through the temporary door(s) and then all personnel lock doors shall be closed. As soon as possible, the fuel transfer gate valve shall be closed to back up the 30 foot water seal to prevent escape of fission products.

The spent fuel storage pool at the Point Beach Nuclear Plant consists of a single pool with a four foot thick reinforced concrete divider wall which separates the pool into a north half and south half. The divider wall is notched to a point sixteen feet above the pool floor to allow transfer of assemblies from one half of the pool to the other.

B. Power Distribution Limits

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

$F_Q(Z) \le \left(\frac{2.21}{P}\right) \times K(Z)$ for $P > 0.5$

$F_Q(Z) \le 4.42 \times K(Z)$ for $P \le 0.5$

$F^N_{\Delta H} < 1.58 \times [1 + 0.3\ (1-P)]$

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

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

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

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

2. a. The indicated axial flux difference (AFD) shall be maintained within the allowed operational space defined by Figure 15.3.10-4 except during physics tests. The physics test exemption applies provided that the thermal power is less than or equal to 85% of Rated Power and the limits of Specification 15.3.10.B.1.a are satisfied. During suspension of the specification, the thermal power shall be determined to be less than or equal to 85% of rated thermal power at least once per hour. In addition, the surveillance requirements of 15.3.10.B.1.b shall be performed at least once per 12 hours.

b. If the indicated AFD deviates from the Figure 15.3.10-4 limits, the AFD shall be restored to within the Figure 15.3.10-4 limits within 15 minutes. If this cannot be accomplished, then reactor power shall be reduced until the AFD is within the envelope or the power level is less than 50% of Rated Power. Normally the rate of power reduction is 15% per hour. Once AFD has been returned to and maintained within the operating envelope, power level is no longer restricted. If it is necessary to reduce power to 50%, the Power Range Neutron Flux-High Trip Setpoints shall be reduced to less than or equal to 55 percent within the next 4 hours.

c. A power increase to a level greater than 50% of Rated Power is contingent upon the indicated AFD being within the Figure 15.3.10-4 limits.

d. Alarms shall normally be used to indicate non-conformance with the flux difference requirements of 15.3.10.B.2.a and 15.3.10.B.2.b. If the alarms are totally out of service, the AFD shall be noted and conformance with the limits assessed every hour for the first 24 hours, and half-hourly thereafter.

e. The indicated AFD shall be considered outside of its limits when at least 2 operable excore channels are indicating the AFD to be outside the limits.

D. Misaligned or Dropped RCCA

1. If the rod position indicator channel is functional and the associated RCCA is more than 7.5 inches indicated out of alignment with its bank demand position and cannot be aligned when the bank demand position is between 215 steps and 30 steps, then unless the hot channel factors are shown to be within design limits as specified in Section 15.3.10.B-1 within eight (8) hours, power shall be reduced to less than 75% of Rated Power. When the bank demand position is greater than or equal to 215 steps, or less than or equal to 30 steps, the allowable indicated misalignment is 15 inches between the rod position indicator and the bank demand position.
2. To increase power above 75% full power with an RCCA more than 7.5 inches indicated out of alignment with its bank demand position when the bank demand position is between 215 steps and 30 steps, an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 15.3.10.B. When the bank demand position is greater than or equal to 215 steps, or less than or equal to 30 steps, the allowable indicated misalignment is 15 inches between the rod position indication and the bank demand position.
3. If it is determined that the apparent misalignment or dropped RCCA indication was caused by rod position indicator channel failure, sustained power operation may be continued if the following conditions are met:
   a. For operation between 10% power and Rated Power, the position of the RCCA(s) with the failed rod position indicator channel(s) will be checked indirectly by core instrumentation (excore detectors, and/or thermocouples, and/or movable incore detectors) every shift and after associated bank motion exceeding 24 steps in one direction.
   b. For operation below 10% of Rated Power, no special monitoring is required.

E. RCCA Drop Times

1. At operating temperature and full flow, the drop time of each RCCA shall be no greater than 2.2 seconds from the loss of stationary gripper coil voltage to dashpot entry.

of axial power distribution. One may assume no change in core poisoning due to xenon, samarium or soluble boron.

Power Distribution

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

In addition to the above, the peak linear power density must not exceed the limiting kw/ft values which result from the large break loss of coolant accident analysis based upon the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution, the following hot channel factors are defined:

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

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

An upper bound envelope of 2.21 times the normalized peaking factor axial dependence of Figure 15.3.10-3 consistent with the Technical Specifications on power distribution control as given in Section 15.3.10 was used in the LOCA analysis. The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.58/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily 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 $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

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

Axial Power Distribution

The 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.

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 one minute average of each of the operable excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 operable excore channels are outside the AFD limits and the thermal power is greater than 50 percent of Rated Power.

Quadrant Tilt

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.

Figure 15.3.10-1

CONTROL BANK INSERTION LIMITS
POINT BEACH UNIT 1 AND 2

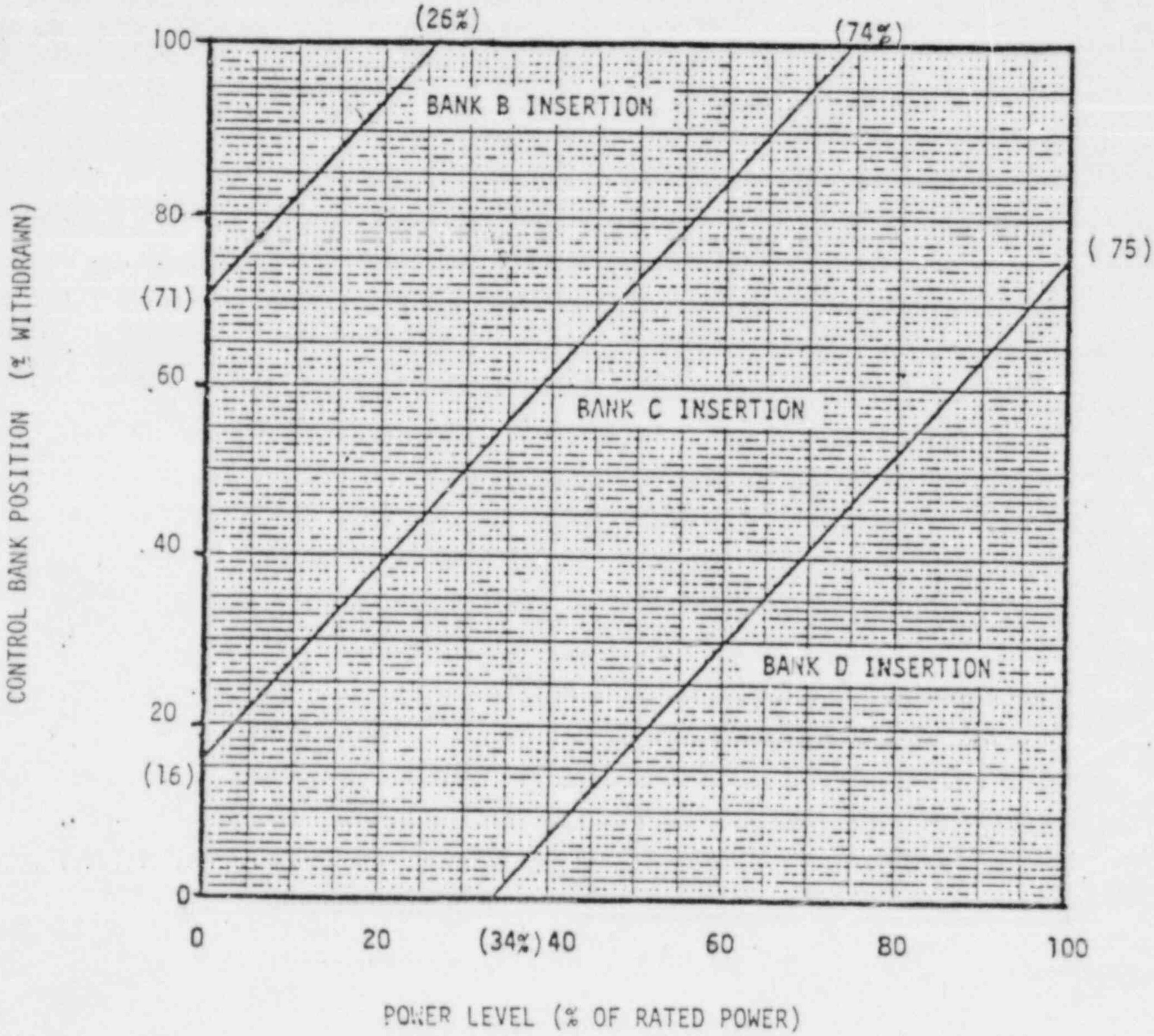

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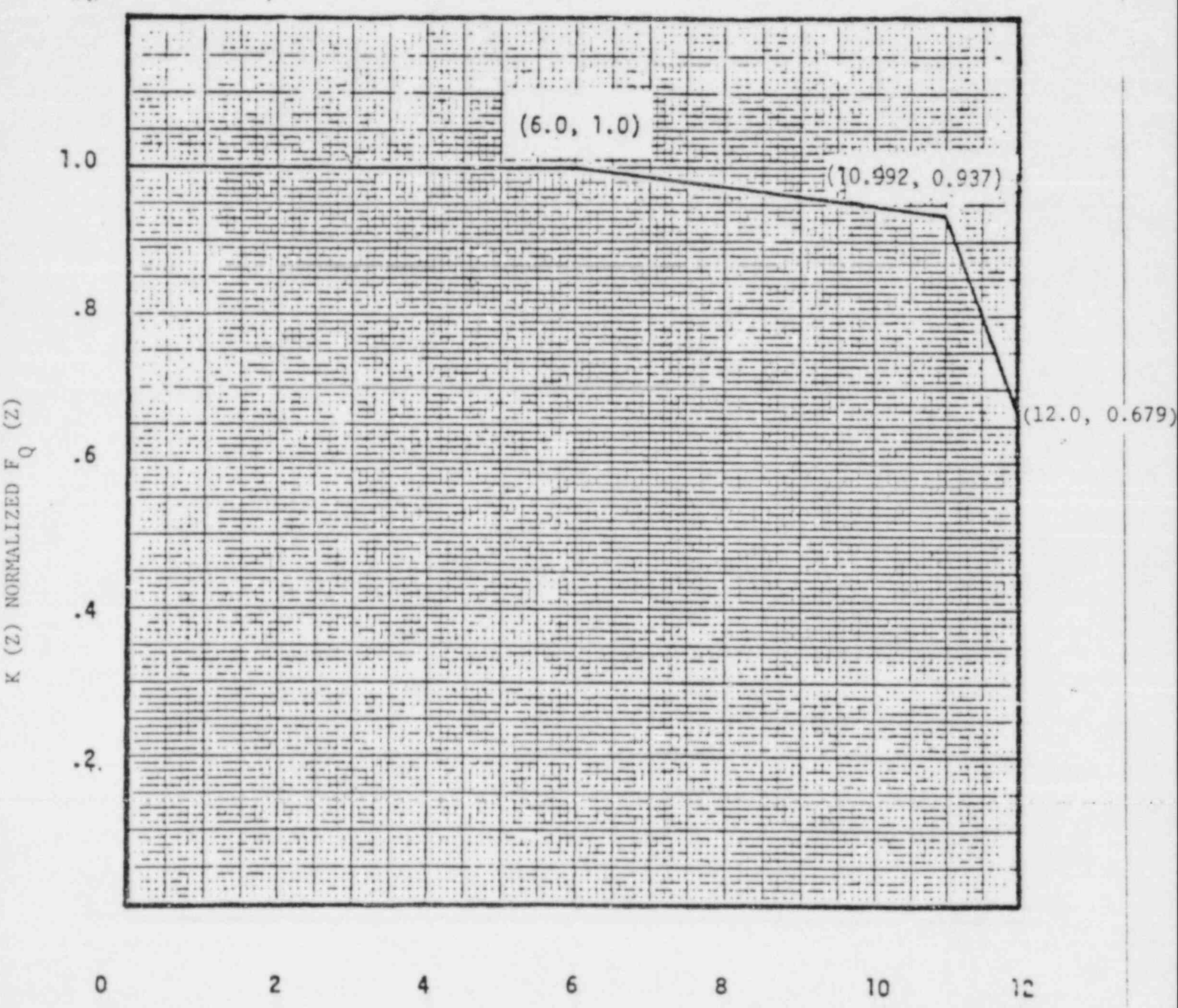
FIGURE 15.3.10-3

POINT BEACH UNITS 1 & 2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



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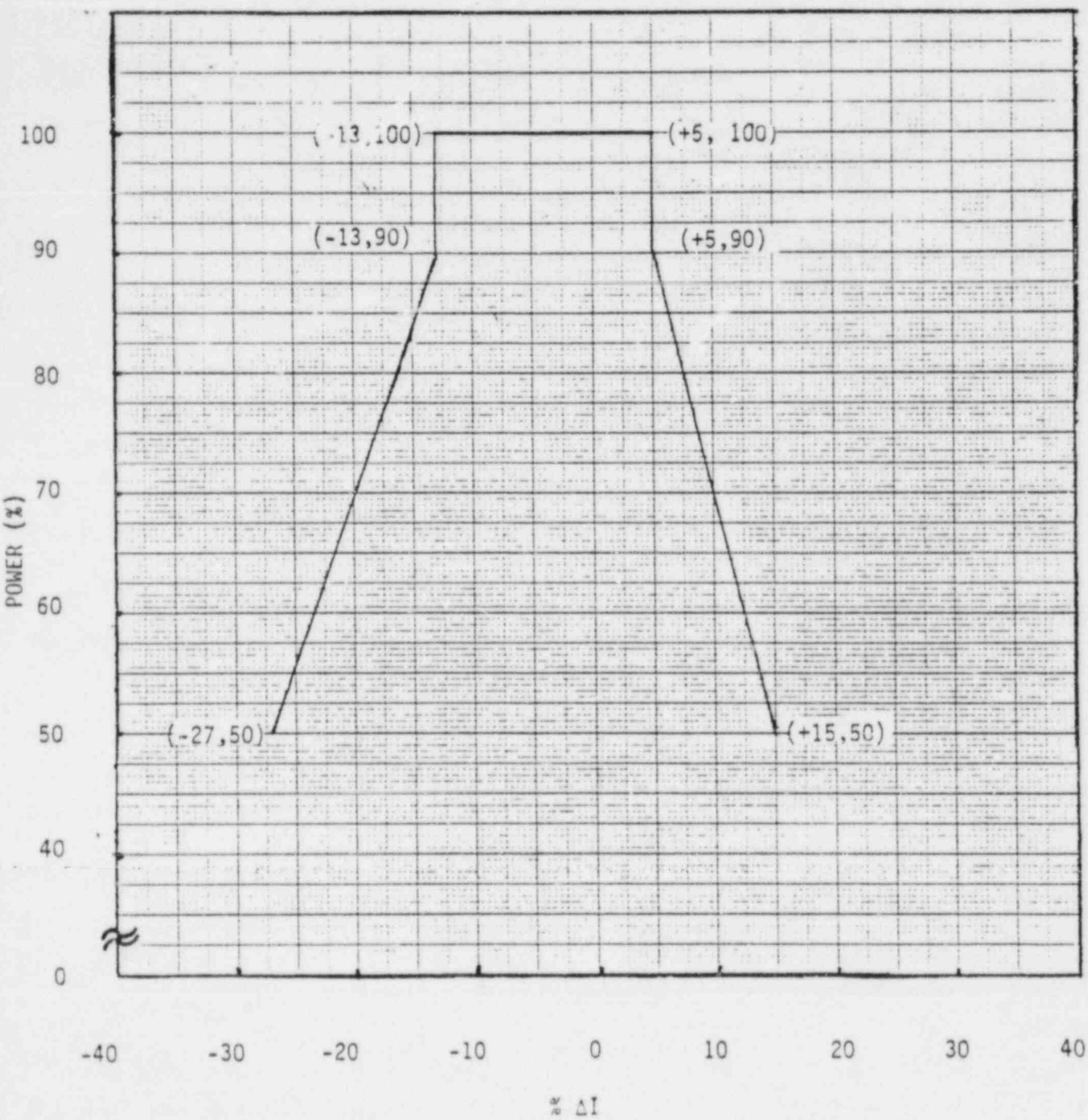
Unit 2 - Amendment No. 18, 79, 90

FIGURE 15.3.10-4

FLUX DIFFERENCE
OPERATING ENVELOPE

POINT BEACH UNITS 1 & 2



Unit 1 - Amendment No. 86
Unit 2 - Amendment No. 90

Unit 1 - Amendment No. 43, 47, 86
Unit 2 - Amendment No. 48, 52, 90

TABLE 15.4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S(1)** M*(3)**(4)	D (1)** Q*(3)**(4)	B/W (2)**	(1) Heat Balance (2) Signal to ΔT; bistable action (permissive, rod stop, trips) (3) Upper and lower chambers for axial off-set (4) Compare incore to excore axial flux difference. Recalibrate if the absolute difference is greater than or equal to 3 percent.
2.	Nuclear Intermediate Range	S (1)**	N.A.	P (2)	(1) Once/shift when in service (2) Log level; bistable action (permissive, rod stop, trips)
3.	Nuclear Source Range	S (1)	N.A.	P (2)	(1) Once/shift when in service (2) Bistable action (alarm, trips)
4.	Reactor Coolant Temperature	S	R	B/W (1)** (2)	(1) Overtemperature-ΔT (2) Overpower - ΔT
5.	Reactor Coolant Flow	S**	R	M**	
6.	Pressurizer Water Level	S**	R	M**	
7.	Pressurizer Pressure	S**	R	M**	
8.	4 Kv Voltage	N.A.	R	M**	Reactor protection circuits only
9.	Analog Rod Position	S (1)**	R	M**	(1) With step counters

* By means of the movable in-core detector system.

**Not required during periods of refueling shutdown, but must be performed prior to starting up if it has not been performed during the previous surveillance period. Tests of permissive and low power trip bistable setpoints which cannot be done during power operations shall be conducted prior to startup if not done in the previous two weeks.

15.5.3 REACTOR

Applicability

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

Objective

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

Specifications

A. Reactor Core

1. General

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

2. Standard Cores

   Standard reactor cores consisting entirely of standard design fuel, contain approximately 48 metric tons of slightly enriched uranium. Standard design fuel assemblies are essentially the same as those contained in the initial cores[1].

3. Transition Cores

The transition cores are defined as being any core loading pattern consisting of standard and OFA 14x14 fuel assemblies. Use of OFA demonstration assemblies in cores of standard design fuel does not constitute a transition core. The initial transition reactor core contains approximately 47 metric tons of uranium in the form of slightly enriched uranium dioxide pellets.

4. The average reload region enrichment of the initial transition core is a nominal 3.20 weight percent of U-235.

5. The transition reload fuel will be similar in design to the initial core standard fuel.

6. Burnable poison rods are incorporated for reactivity and/or power distribution control. The burnable poison rods consist of borated pyrex glass clad with stainless steel.[4]

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

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

B. Reactor Coolant System

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

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

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

15.5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The new fuel storage and spent fuel pool structures are designed to withstand the anticipated earthquake loadings as Class I structures. The spent fuel pool has a stainless steel liner to ensure against loss of water.

2. The new and spent fuel storage racks are designed so that it is impossible to store assemblies in other than the prescribed storage locations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure $K_{eff} < 0.95$ with the storage pool filled with unborated water and with the fuel loading in the assemblies limited to 44.8 grams of U-235 per axial centimeter of standard fuel assemblies and 39.4 grams of U-235 per axial centimer of OFA fuel assemblies. An inspection area shall allow rotation of fuel assemblies for visual inspection, but shall not be used for storage.

3. The spent fuel storage pool shall be filled with borated water at a concentration of at least 1800 ppm boron whenever there are spent fuel assemblies in the storage pool.

4. Except for the two storage locations adjacent to the designated slot for the spent fuel storage rack neutron absorbing material surveillance specimen irradiation, spent fuel assembly storage locations immediately adjacent to the spent fuel pool perimeter or divider walls shall not be occupied by fuel assemblies which have been subcritical for less than one year.