May 8, 1989

Docket Nos. 50-266
and 50-301

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Mr. C. W. Fay, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room 308
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

SUBJECT: AMENDMENT NOS. 120 AND 123 TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27 (TACS 69349/69350)

The Commission has issued the enclosed Amendment Nos. 120 and 123 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments revise the Technical Specifications in response to your application dated August 26, 1988, as supplemented October 28, November 30, and December 23, 1988; and as modified January 17, 1988 (sic).

These amendments revise the provisions in the Point Beach Nuclear Plant, Unit Nos. 1 and 2, Technical Specifications (TS's) relating to the design and operation of the Point Beach fuel cycle with upgraded core features and at higher core power peaking factors (F_Q and F-delta H) than are currently permitted by the plant TS.

Specifically, the amendments incorporate higher core power peaking factors which allow the use of a low-low leakage loading pattern (L4P) fuel management strategy and will result in decreased neutron fluence to the reactor vessel. This fluence reduction will help address reactor vessel irradiation damage issues such as pressurized thermal shock, low upper shelf material toughness and pressure-temperature restrictions on heatup and cooldown. The higher core power peaking factors allow additional fluence reduction measures, such as the use of peripheral power suppression assemblies, to be pursued.

In addition to the increase in core power peaking factors, the changes and reanalyses supporting them permit the use of an upgraded fuel product features package. The upgraded fuel product features include: removable top nozzles, integral fuel burnable absorbers, axial blankets, extended burnup geometry, and inclusion of a debris filter bottom nozzle. The reactor core description is modified to reflect these changes. Further, this amendment allows the removal of the fuel assembly thimble plugging devices and the elimination of the third line segment of the K(z) curve.

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1/1

CP-1
ac

Copies of the Safety Evaluation and of the Notice of Issuance are also enclosed. The notice has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

Warren H. Swenson, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 120 to DPR-24
2. Amendment No. 123 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	SRXB	OGC-WF1
Surname:	*PKreutzer	*WSwenson/tg	*JHannon	*WHodges	*STurk
Date:	04/28/89	04/28/89	04/28/89	05/01/89	05/02/89

Mr. C. W. Fay
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Units 1 and 2

cc:

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Office of Executive Director
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Glen Ellyn, Illinois 60137

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
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Two Rivers, Wisconsin 54241

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
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 August 26, 1988, as supplemented October 28, November 30, and December 23, 1988; and as modified January 17, 1988 (sic) 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. 123 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective November 1, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn

Timothy G. Colburn, Acting Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 8, 1989

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
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 August 26, 1988, as supplemented October 28, November 30, and December 23, 1988; and as modified January 17, 1988 (sic) 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. 120 , 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 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn

Timothy G. Colburn, Acting Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 8, 1989

ATTACHMENT TO LICENSE AMENDMENT NOS. 120 * AND 123 **

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 marginal lines indicating the area of change.

REMOVE	INSERT
15.2.1-1	15.2.1-1
15.2.1-2	15.2.1-2
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
15.3.1-10	15.3.1-10
15.3.1-19	15.3.1-19
15.3.3-8	15.3.3-8
15.3.3-9	15.3.3-9
-	15.3.3-10
15.3.10-2	15.3.10-2
15.3.10-11	15.3.10-11
15.3.10-12	15.3.10-12
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	Figure 15.3.10-4
15.5.3-1	15.5.3-1
15.5.3-2	15.5.3-2

*For Unit 1, the amendment is effective immediately, with the TS changes to be implemented within 20 days.

**For Unit 2, The amendment is effective on November 1, 1989.

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. 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.

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.

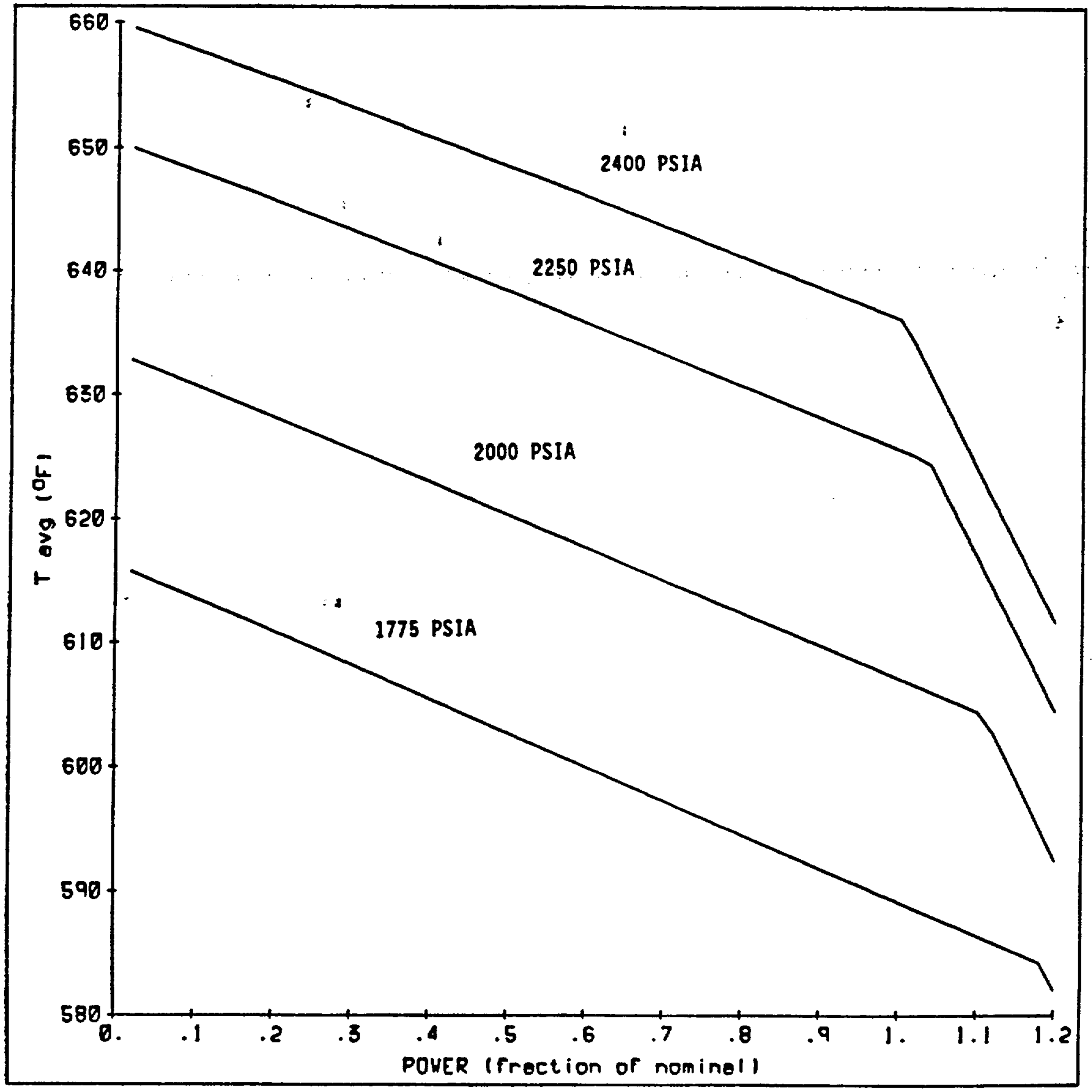
Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore 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 an appropriate margin to DNB for all operating conditions.

The curves of Figure 15.2.1.1 are applicable for a core of 14 x 14 OFA. The curves also apply to the reinsertion of previously-depleted 14 x 14 standard fuel assemblies into an OFA core. The use of these assemblies is justified by a cycle-specific reload analysis. The WRB-1 correlation is used to generate these curves. Uncertainties in plant parameters and DNB correlation predictions are statistically convoluted to obtain a DNBR uncertainty factor. This DNBR uncertainty factor 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.

Figure 15.2.1-1

REACTOR CORE SAFETY LIMITS
POINT BEACH UNITS 1 AND 2



Unit 1 - Amendment No. ~~86~~, 120
Unit 2 - Amendment No. ~~21~~,~~90~~, 123

(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 $\Delta T \left(\frac{1}{1+\tau_3 S}\right)$

$$\leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3 (P-P') - f(\Delta I)\right)$$

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	$\leq$	573.9°F
P	=	pressurizer pressure, psig
P'	=	2235 psig
K_1	$\leq$	1.30
K_2	=	0.0200
K_3	=	0.000791
τ_1	=	25 sec
τ_2	=	3 sec
τ_3	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
	=	0 sec for Sostman or equivalent RTD

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 $q_t - q_b$ within -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 setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

(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.

(5) Overpower $\Delta T \left(\frac{1}{1+\tau_3 S}\right)$

$$\leq \Delta T_o \left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S + 1}\right)\left(\frac{1}{1+\tau_4 S}\right) T - K_6 \left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right] - f(\Delta I)\right]$$

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	$\leq$	573.9°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_5	=	10 sec
		f (ΔI) as defined in (4) above,
τ_3	=	2 sec for Rosemont or equivalent RTD
		0 sec for Sostman or equivalent RTD
τ_4	=	2 sec for Rosemont or equivalent RTD
		0 sec for Sostman or equivalent RTD

(6) Undervoltage - $\geq$75 percent 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$55.0 HZ

 (b) Low voltage set point $\geq$75 percent of normal voltage.

Unit 1 - Amendment No. ~~3~~,~~28~~,~~86~~,~~90~~,~~94~~, 120
Unit 2 - Amendment No. ~~32~~,~~90~~,~~91~~,~~98~~, 123

power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figure 15.2.1-1. If axial peaks are greater than design, as indicated by the 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 include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints will not exceed the core safety limits as shown in Figure 15.2.1-1.

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 the 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

Basis:

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 500 gpd in either steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative for Point Beach Nuclear Plant.

Continued power operation for limited time periods with the reactor coolant's specific activity greater than 1.0 microcurie/gram Dose Equivalent I-131, but within the allowable limit shown on Figure 15.3.1-5, accommodates possible iodine spiking phenomenon which may occur following changes in thermal power. Operation with specific activity levels exceeding 1.0 microcurie/gram Dose Equivalent I-131 but within the limits shown on Figure 15.3.1-5 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to less than 500°F normally prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

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 below 578°F.

2. Reactor Coolant System (RCS) pressurizer 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 181,800 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.

Unit 1 - Amendment No. ~~44~~,~~87~~,~~86~~, 120
Unit 2 - Amendment No. ~~49~~,~~90~~,123

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.*

Time After Shutdown	Decay Heat % of Rated Power
1 min.	3.6
30 min.	1.55
1 hour	1.25
8 hours	0.7
48 hours	0.4

*Based on ANS 5.1-1979, "Decay Heat Power in Light-Water Reactors"

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safety system components in order to effect repairs.

Failure to complete safety injection system repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition. When the failures involve the residual heat removal system, in order to insure redundant means of decay heat removal, the reactor system may remain in a condition with reactor coolant temperatures between 500 and 350°F so that the reactor coolant loops and associated steam generators may be utilized for redundant decay heat removal. However, when the remaining RHR loop must be relied upon for redundant decay heat removal capability, reactor coolant temperatures shall be maintained between 350°F and 140°F.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.(2)

The operability of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core; (2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, spray additive tank, containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1); (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area greater than 3 ft^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO); and (4) long term subcriticality is maintained following a steamline break assuming ARI-1 and fuel failure is precluded.

The containment cooling function is provided by two independent systems: (a) fan coolers and (b) containment spray which, with sodium hydroxide addition, provides the iodine removal function. During normal power operation, only three of the four fan coolers are required to remove heat lost from equipment and piping within the containment.[3] In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: (1) four fan coolers, (2) two containment spray pumps, (3) two fan coolers plus one containment spray pump.[4] Sodium hydroxide addition via one spray pump reduces airborne iodine activity sufficiently to limit off-site doses to acceptable values. One of the four fan coolers is permitted to be inoperable for up to 48 hours during power operation.

The component cooling system is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit either following a loss-of-coolant accident, or during normal plant shutdown. If during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected.[5]

A total of six service water pumps are installed, only three of which are required to operate during the injection and recirculation phases of a postulated loss-of-coolant accident,[6] in one unit together with a hot shutdown condition in the other unit.

References

(1) FSAR Section 3.2.1

(2) FSAR Section 6.2

(3) FSAR Section 6.3.2

(4) FSAR Section 6.3

(5) FSAR Section 9.3.2

(6) FSAR Section 9.6.2

B. Power Distribution Limits

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

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

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

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

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. 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.

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.B.2. The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be 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 $F^N_{\Delta H} \leq 1.70/1.08$. 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.

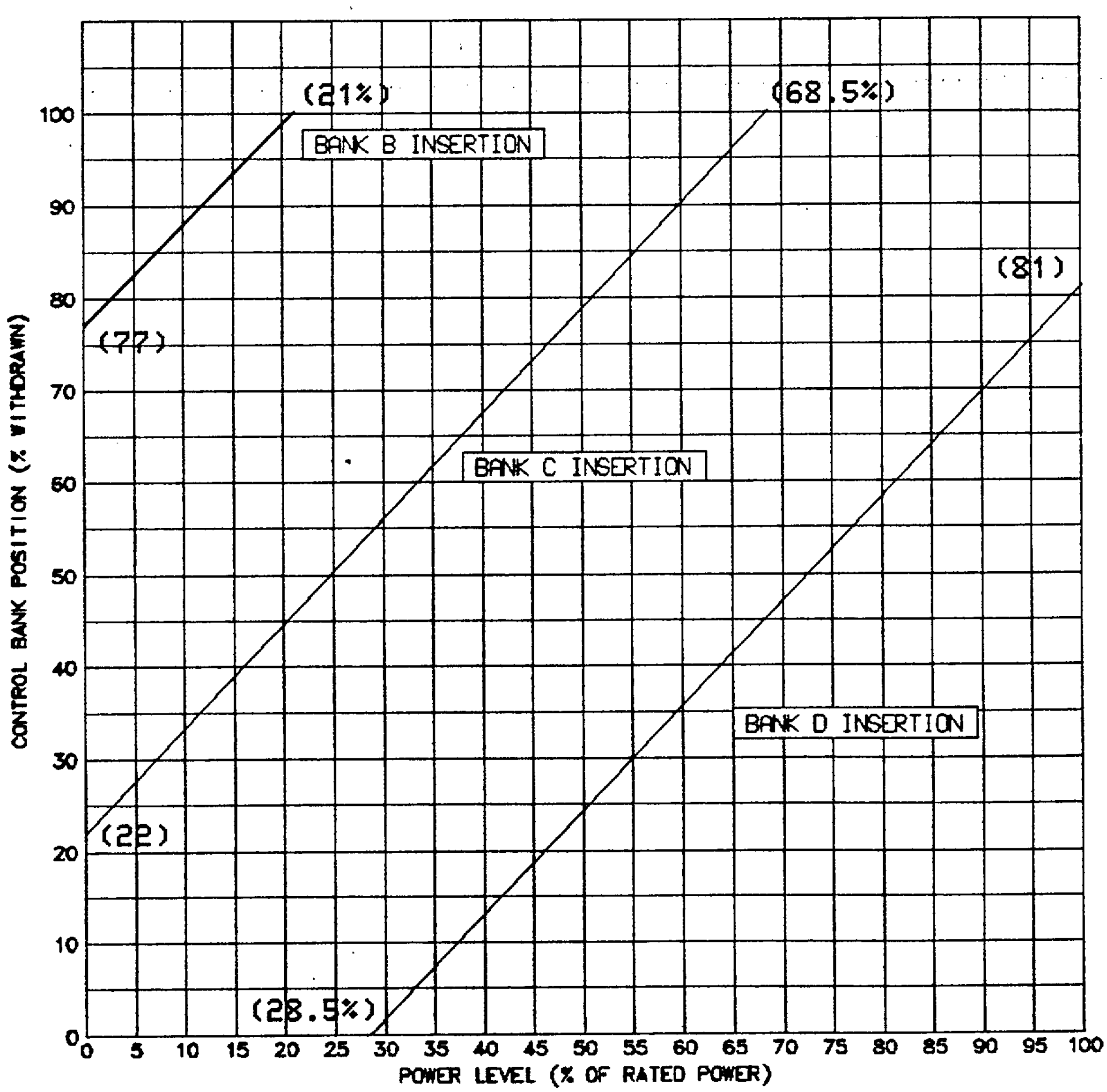
Axial Power Distribution

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. 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 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.

FIGURE 15.3.10-1

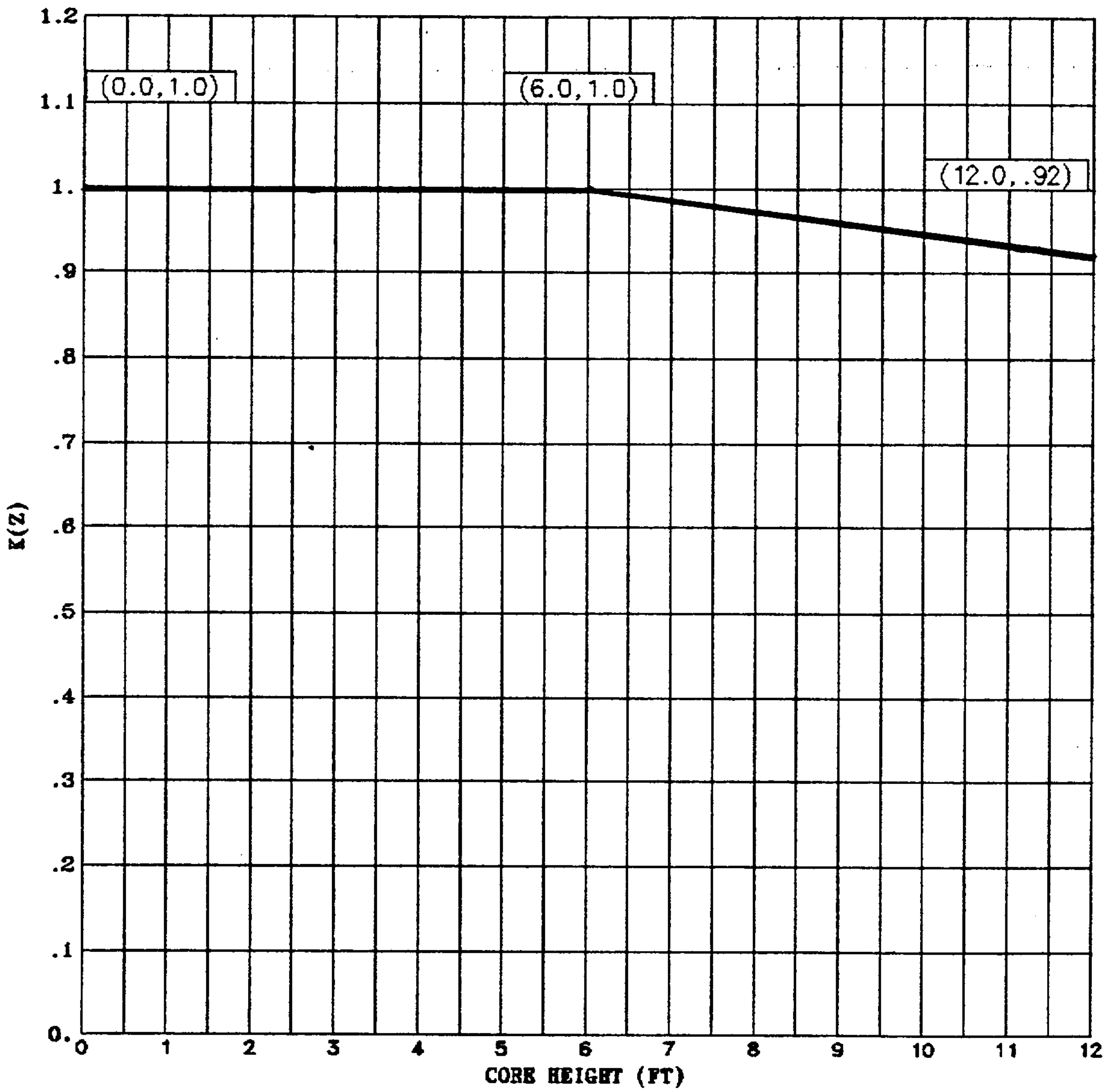
CONTROL BANK INSERTION LIMITS
POINT BEACH UNITS 1 AND 2



Unit 1 - Amendment No. ~~25,49,86,88~~, 120
Unit 2 - Amendment No. ~~30,55,90,93~~, 123

FIGURE 15.3.10-3

POINT BEACH UNITS 1 AND 2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



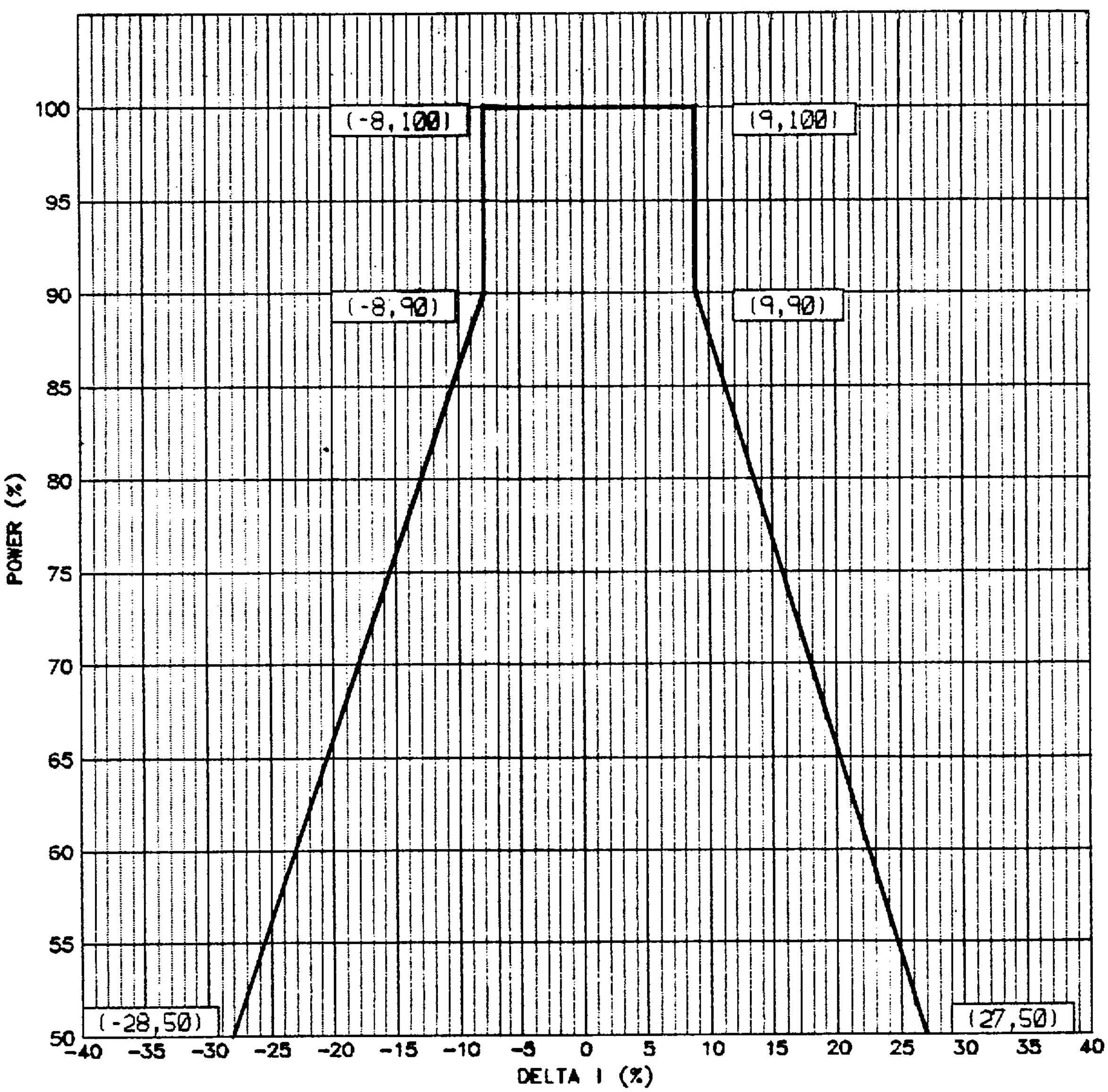
Unit 1 - Amendment No. ~~14~~,~~22~~,~~86~~,120
Unit 2 - Amendment No. ~~18~~,~~29~~,~~90~~, 123

FIGURE 15.3.10-4

FLUX DIFFERENCE

OPERATING ENVELOPE

POINT BEACH UNITS 1 AND 2



Unit 1 - Amendment No. ~~86~~, 120
Unit 2 - Amendment No. ~~90~~, 123

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, limited substitutions of fuel rods by filler rods consisting of Zircaloy 4 or stainless steel, or by vacancies, may be made to replace damaged fuel rods if justified by cycle specific reload analysis.

2. Core

A reactor core is a core loading pattern containing any combination of 14x14 OFA and 14x14 upgraded OFA fuel assemblies. The core may also contain previously depleted 14x14 standard fuel assemblies. The use of previously depleted 14x14 standard fuel assemblies will be justified by a cycle specific reload analysis.

3. Burnable absorber and/or water displacer rods are incorporated for reactivity and/or power distribution control. The burnable absorber rods consist of borated pyrex glass clad with stainless steel[4]. The water displacer rods are empty burnable absorber rods containing no pyrex glass. Another type of burnable absorber may consist of a thin coating of zirconium diboride on the radial surface of selected fuel rod pellets.

4. 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.

5. Neutron source assemblies are used to provide a required minimum count rate during startup operations. The core contains at least two such assemblies, each containing four source rodlets comprised of a mixture of antimony and beryllium.

6. Peripheral power suppression assemblies (PPSA) are used to reduce neutron fluence at the welds in the beltline region of the reactor vessel. Peripheral fuel assemblies may contain PPSAs, which utilize part-length hafnium absorber rods in the assembly guide tubes.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NOS. 120 AND 123 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 August 26, 1988 (Ref. 1), as supplemented by letters dated October 28 (Ref. 2), November 30 (Ref. 4), December 23, 1988 (Ref. 27), and as modified January 17, 1988 (sic) (Ref. 28), the Wisconsin Electric Power Company (the licensee) made application to change the Technical Specifications of the Point Beach Nuclear Plant, Units 1 and 2. The proposed changes would permit the design and operation of future Point Beach Nuclear Plant (PBNP) reactor cores with enhanced Optimized Fuel Assembly (OFA) fuel and at higher core power peaking factors than are allowed by current Technical Specifications. The higher power peaking factors will allow the use of a low-low leakage loading pattern (L4P) fuel management strategy which will result in a decrease in the fluence accumulation rate to the reactor pressure vessel. Additional core design features included in the licensee's submittals are (1) use of Peripheral Power Suppression Assemblies (PPSA), (2) removal of fuel assembly thimble plugging devices, and (3) elimination of the third line segment of the K(Z) curve in Technical Specification Figure 15.3.10-3. The use of PPSA's by the licensee is part of the L4P fuel management strategy.

The enhanced 14x14 OFA fuel design incorporates the following features: (1) Removable Top Nozzles (RTN), (2) Integral Fuel Burnable Absorbers (IFBA), (3) axial blankets, (4) Debris Filter Bottom Nozzles (DFBN), and (5) extended burnup geometry. These fuel features, with the exception of the DFBN, are a subset of the Westinghouse Vantage 5 fuel design. The bottom nozzle of the PBNP fuel will differ from the Vantage 5 bottom nozzle in that it will be fabricated from stainless steel rather than Inconel and the size and pattern of the flow holes have been changed. The DFBN will, however, meet all other design requirements. The Vantage 5 fuel design was generically approved with conditions (Ref. 3).

The licensee has submitted a revised large-break LOCA analysis (Ref. 4) as part of the resolution of the Upper Plenum Injection (UPI) issue using the best estimate W COBRA/TRAC model (Ref. 5). This model has been reviewed and approved by the NRC (Ref. 6). The licensee also submitted a reanalysis of the small-break LOCA using the approved NOTRUMP code (Ref. 7). Based on generic small-break LOCA studies and results of analyses for the Northern States Power Prairie Island plants, the licensee analyzed only the 4-inch cold-leg break.

The steam generator tube rupture (SGTR) event has been reanalyzed using the same methodology that was previously used in the PBNP Final Safety Analysis Report (FSAR). However, some revised input assumptions have been used.

The licensee's analysis includes the use of a revised methodology, the revised thermal design procedure (RTDP). The RTDP (Ref. 8) has been reviewed and approved by the staff (Ref. 9). All other analysis methodologies for the non-LOCA transients, except for the input changes noted for the SGTR event, are the same as those currently used in the PBNP FSAR analyses.

2.0 EVALUATION

2.1 Design Features and Parameters

Future PBNP cores will contain OFA, enhanced OFA, and previously depleted Low-Parasitic (LOPAR) fuel assemblies that are also known as standard (STD) fuel assemblies. The STD fuel assemblies will be bounded by the OFA fuel assemblies because the STD fuel assemblies will be previously irradiated fuel that will operate at lower power than the OFA fuel. The licensee will justify the use of previously irradiated STD fuel by cycle-specific reload analyses. All of the fuel designs have a 14x14 geometry with 179 fuel rods and 17 guide tubes and an instrumentation thimble. The upgraded OFA will include a number of Vantage 5 features: (1) Removable Top Nozzles (RTN), (2) Integral Fuel Burnable Absorbers (IFBA), (3) axial blankets, and (4) extended burnup geometry. In addition, the upgraded OFA will include a Debris Filter Bottom Nozzle (DFBN). The licensee states that these OFA upgrade features may not all be used together in upgraded OFA fuel but that the upgrade features used will be bounded by the reference analyses that have been submitted.

For its plant life extension (PLEX) program the licensee proposes to introduce a low-low leakage loading pattern (L4P) fuel management strategy. The L4P PBNP reactor cores will use a loading pattern that includes low power peripheral fuel assemblies and Peripheral Power Suppression Assemblies (PPSA's) (Ref. 28). The PPSA's are specially designed fuel assemblies that will be inserted on the core periphery to further reduce the fluence accumulation rate at specific reactor vessel welds. The PPSA's will use the neutron absorber hafnium in the thimble tubes of the fuel assemblies. The hafnium will be a part-length design similar in mechanical design to the present Westinghouse hafnium Rod Cluster Control Assembly (RCCA) that is used in some Westinghouse plants. In addition to its reduced fluence accumulation rate to the reactor vessel, the L4P provides PBNP with improved fuel utilization.

PBNP currently uses thimble plugging devices in some fuel assemblies to minimize core bypass flow through fuel assembly thimble tubes. The licensee states that the analysis will support the removal of these thimble plugging devices.

The current and proposed PBNP design parameters are as follows:

	Current	Proposed
Fuel Type (Westinghouse)	STD, OFA	STD, OFA, upgraded OFA
Core Power (MWt)	1518.5	1518.5
Average linear heat generation rate (kW/ft)	5.7	5.7
System Pressure (psia)	2000 (or 2250)	2000 (or 2250)
Core Inlet Temperature (°F)	545.3	545.3
Enthalpy rise hot channel peaking factor limit (F-Delta H)	1.58	1.70
Total Peaking Factor Limit (F_Q)	2.21	2.50
Total thermal design flow (gpm)	178,000	178,000
Steam generator uniform tube plugging levels (%)		13% (Unit 1) 14% (unit 2)

NOTE: The LOCA and SGTR analyses used a 25% uniform tube plugging level and the associated reduction in thermal design flow.

The proposed design provides for the removal of the third line segment of the Technical Specification K(Z) curve. This K(Z) curve is used to provide the required axial variation of the total peaking factor with core height such that at any core height the peaking factor limit will always be equal to or less than 2.50 (the F_Q limit). The removal of the third line segment of the K(Z) curve is supported by the small-break LOCA analysis.

The staff has reviewed the design features and parameters proposed for future PBNP cores and concludes that they are acceptable because they are typical of the types of changes previously reviewed and approved for other plants and because they lead to improvements in fuel utilization, fuel performance (for example, DFBN for the reduction of the passage of flow-entrained debris into the fuel assembly), and a reduction in the fluence accumulation rate to the reactor vessel.

2.2 Fuel Rod Design

The increased power peaking factors affect the fuel rod design through increases in the steady-state fuel rod power histories and through the fuel rod transient duty. The licensee states that the fuel rod design criteria for the most limiting fuel rod design will be considered for PBNP including all combinations of Westinghouse STD, OFA, and upgraded OFA fuel. The fuel rod design criteria affected by this more severe fuel duty are the rod internal pressure, cladding stress and strain, and cladding surface temperature.

For the fuel rod internal pressure, Westinghouse uses an NRC-approved design limit that the internal fuel rod pressure of the lead fuel rod will be limited to a value below that which could cause (1) the diametrical gap to increase due to outward cladding creep during steady-state operation, and (2) extensive DNB propagation to occur. This fuel rod internal pressure limit is a function of system pressure. The design limit used for cladding stress is that the volume average effective stress is less than the Zircaloy 0.2% offset yield strength for Condition I and Condition II modes of operation, including the effects of temperature and irradiation. The design limit for cladding strain during steady-state operation is that the total plastic tensile creep and uniform cylindrical fuel pellet expansion due to fuel swelling and thermal expansion are less than 1 percent from the irradiated condition. For Condition II events the design limit for cladding strain is that the total tensile strain due to uniform cylindrical pellet thermal expansion during a transient is less than 1 percent from the pre-transient value. The design limit applied to Zircaloy cladding corrosion during steady-state and Condition II transients is to preclude a condition of accelerated oxidation. The controlling factor for Westinghouse reactors is the oxide-to-cladding interface temperature.

The fuel performance results for the PBNP are obtained using the approved PAD3.3 (Ref. 12) and PAD3.4 (Ref. 13) codes. The fuel rod design analysis is based on a best estimate plus uncertainty basis. The total uncertainty is based on a statistical convolution of the applicable individual uncertainties. Appropriate power histories which define limiting duty for each of the fuel rod design criteria are used. The most limiting values of core inlet temperature and flow rate are used in the evaluations. The most limiting value of the system pressure (2000 or 2250 psia) for each of the fuel rod design criteria is also used.

In addition to the fuel rod design criteria discussed above, the PBNP fuel will incorporate design changes to allow for extended burnup operation. These changes are primarily concerned with the axial growth of fuel rods.

The staff has reviewed the fuel rod design for future reactor cores for PBNP and concludes that it is acceptable because (1) approved codes are used, (2) all applicable criteria are evaluated, and (3) the results for the increased power peaking factors and increased fuel duty are acceptable.

2.3 Nuclear Design

The licensee evaluated a reference core design that included the upgraded PBNP core features. A low-low leakage loading pattern (L4P) fuel management strategy was used. A cycle length of 10,500 MWd/MTU was obtained through the use of 28 fresh fuel assemblies. Sixteen of the fresh assemblies were enriched to 4.0 w/o uranium-235. Twelve of the assemblies were enriched to 3.8 w/o uranium-235 and included 24 IFBA fuel rods per assembly for a total of 288 IFBA rods. The burnable poison coating of the IFBA rods was 96 inches in length and centered about the midplane. All fuel assemblies, except the center assembly, contain axial blankets at the top and bottom of each fuel rod. The twelve assemblies on the core flats each contain a PPSA, with hafnium in the lower 6 feet of the guide tubes.

The analysis was performed using the approved Westinghouse reload safety evaluation methodology (Ref. 14) and approved codes. Because of the heterogeneous nature of this reference core design, a three dimensional core nodal model (Ref. 15) was used. The Relaxed Axial Offset Control (RAOC) was performed with an approved methodology (Ref. 16).

The results of an analysis of the reference core design showed that the key safety parameters were insensitive to fuel type and primarily affected by the loading pattern. The results also indicated that future PBNP cores would require changes to the Technical Specifications for (1) an increase in the total power peaking factor limit (F_Q), (2) an increase in the enthalpy rise hot channel factor limit (F-Delta H), (3) and a change to the allowable flux difference operating envelope (RAOC delta flux band). The analysis assumed that the third segment of the Technical Specification K(Z) curve was removed (this will be confirmed later in our review of the small-break LOCA and large-break LOCA analyses). The licensee also changed the power-dependent rod insertion Technical Specification limits to ensure that the RAOC delta-flux difference band is conservative for future PBNP cores.

The staff has reviewed the nuclear design of the reference core and concludes that it is acceptable because (1) approved codes and methodologies have been used, (2) acceptable reactor core parameters have been obtained, and (3) appropriate changes to the Technical Specifications have been determined.

2.4 Thermal Hydraulic Design

The licensee performed a thermal hydraulic analysis for the upgraded core features of PBNP. The analysis was performed for a nuclear enthalpy rise hot channel factor F-Delta H of 1.70 and for removal of thimble plugs. The increase in F-Delta H is the result of the L4P fuel management strategy. The increase in F-Delta H and the removal of the thimble plugs are accommodated by using the Departure from Nucleate Boiling Ratio (DNBR) design margin available in the safety analysis DNBR.

The current thermal-hydraulic analysis of OFA fuel is based on the Improved Thermal Design Procedure (ITDP) (Ref. 17) and the Westinghouse WRB-1 critical heat flux correlation (Ref. 18). The analysis of the upgraded core features for future PBNP reloads is based on the Revised Thermal Design Procedure (RTDP) (Ref. 8) and the WRB-1 critical heat flux correlation. However, for some transient events the Standard Thermal Design Procedure (STDP) is used with the W-3 critical heat flux correlation. The RTDP methodology removes some of the conservatism in the ITDP methodology by combining directly both system uncertainties and Departure from Nucleate Boiling (DNB) correlation uncertainty. The RTDP methodology safety analysis DNBR is 1.33 for both a typical cell and a thimble cell. In addition this safety analysis DNBR includes 8.6 percent DNBR margin. The THINC IV code was used to perform the thermal-hydraulic calculations (Refs. 19 and 20).

The upgraded OFA fuel is hydraulically identical to the OFA fuel and no transition core penalty is required. The use of STD fuel requires a small DNBR penalty on all the fuel. A rod bow penalty of less than 3% on DNBR is used in accordance with References 21, 22, and 23. This rod bow penalty is the maximum rod bow penalty for 14x14 OFA fuel at an assembly average burnup of 24,000 MWd/MTU. No rod bow penalty is taken for burnups greater than 24,000 MWd/MTU because credit is taken for the decrease in F-Delta H with burnup.

The axial blankets and the increased allowable F_Q affect the axial power distribution and, therefore, the DNBR. For events not protected by the Overtemperature Delta-T (OTDT) trip function, a limiting axial power distribution was used in the DNBR analyses. For these events, cycle-specific limiting axial power shapes will be evaluated and compared with this limiting axial power distribution for future PBNP reloads.

The licensee plans to remove the thimble plugs in addition to implementing upgraded core features. This removal of the thimble plugs results in an increase in the bypass flow from 4.5% to 6.5%. There is a slight increase in the core flow rate which does not impact any mechanical design criteria. The removal of the thimble plugs results in a small decrease in DNBR margin. The licensee also evaluated the effect of thimble plug removal on fuel assembly hydraulic lift forces, fuel rod fretting wear, and control rod wear. For these three areas, the licensee concluded that there were no significant effects caused by the thimble plug removal. The licensee has accounted for the increased bypass flow in both non-LOCA and LOCA safety analyses.

The staff has reviewed the thermal hydraulic design of the reference core and concludes that it is acceptable because (1) approved codes and methodologies have been used, (2) DNBR penalties resulting from the increase in peaking factor and removal of thimble plugs are offset by the present DNBR margin and the additional margin provided by the RTDP methodology, (3) rod bow penalty and any transition core effects are offset by DNBR margin available in the safety limit DNBR, and (4) all of the current thermal-hydraulic design criteria are satisfied.

2.5 Reactor Pressure Vessel Internals System Evaluation

The licensee evaluated the effect of removing the thimble plugs on the reactor pressure vessel internals system design requirements. Thimble plug removal leads to a reduction in core hydraulic resistance and to an increase in the portion of the bypass flow passing through the fuel assembly. The licensee's evaluations used operating, geometric and hydraulic characteristics of the PBNP with 14x14 OFA fuel and thimble plugs removed. System pressures of 2000 and 2250 psia were considered. The increased bypass flow in the fuel assembly and the reduction in core hydraulic resistance affects fluid system pressure drops, core bypass flow, baffle gap jetting momentum flux, closure head fluid temperature, internals component lift forces, and control rod drop times. The licensee determined that the core pressure drop would decrease by less than 10%, the total core bypass flow would be bounded by a value of 6.5%, the baffle gap jetting momentum flux is not adversely affected, the closure head fluid temperature is unaffected, the internals component lift forces are not adversely impacted (in fact they are reduced somewhat), and the control rod drop times are not adversely impacted.

The staff has reviewed the licensee's evaluation of the reactor pressure vessel internals with respect to the thimble plug removal and concurs with the licensee's assessments.

2.6 Non-LOCA Accidents

The licensee has evaluated the impact of the upgraded core features on the non-LOCA events presented in Chapter 14 of the PBNP FSAR. The licensee has used the approved reload core design methodology of Reference 14 and approved design codes.

In the present PBNP cores, thimble plugs are installed in all fuel assemblies which are not under control rod locations or do not contain neutron sources or burnable poisons. The removal of the plugs has two primary effects. It increases the total core bypass flow and it reduces the core pressure drop somewhat. The events that have been reanalyzed have incorporated these effects. For the steamline break event and the mass and energy release to containment event, the licensee concludes that either the impact on the analyses is not significant or the conclusions of the previous analyses remain valid. Therefore, these two events were not reanalyzed by the licensee.

The licensee used the Revised Thermal Design Procedure (RTDP) in the analysis of a number of events. This extension of the ITDP methodology uses a safety analysis DNBR limit of 1.33 for both typical and thimble cells. The licensee used the Standard Thermal Design Procedure (STDP) for those events which did not use the RTDP methodology.

The removable top nozzles (RTN) and debris filter bottom nozzles (DFBN) were designed to preserve core flow areas and loss coefficients. Therefore, no parameters important to the non-LOCA safety analyses were affected. The effects of integral fuel burnable absorbers (IFBA), axial blankets of natural uranium on the ends of the upgraded OFA fuel rods, and extended burnup are taken into account in the reload design process. The effects of the L4P fuel management strategy and PPSA's (part length hafnium absorbers) in the core locations on the flats of the core periphery are also taken into account in the reload design process. These result in an increase in F-Delta H to 1.70 and F_Q to 2.50 which impact the safety analysis.

The licensee has also made a number of other design changes for future PBNP reloads. These changes include control rod power dependent insertion limits, the elimination of the third line segment of the Technical Specification K(Z) curve, and a revised flux difference operating envelope. The change to the rod insertion limits impacts (1) shutdown margin, (2) trip reactivity, (3) power distribution limits, (4) ejected and dropped rod worths, (5) post ejected rod peaking factors, and (6) differential rod worths. The licensee states that nuclear design calculations with the proposed rod insertion limits ensure that the values for shutdown margin, trip reactivity, dropped rod worths, and differential rod worths assumed in the non-LOCA safety analyses are valid. The effects of new rod insertion limits have been included in the power distribution limits and rod ejection parameters for the affected events that were reanalyzed. In addition, the shutdown margin and power distribution assumptions used in the steamline break analysis remain valid with the proposed rod insertion limits. The proposed changes to the K(Z) curve and the flux difference operating envelope could impact the power distribution assumptions used for the non-LOCA analyses. The licensee states that nuclear design analyses show that the power distribution assumptions of the non-LOCA analyses are ensured by the proposed K(Z) curve and the proposed flux difference operating envelope.

The effect of the proposed power-dependent F-Delta H limit does not directly affect the system transient response of the PBNP. This is because the PBNP system response is determined with a point kinetics system code which does not directly use F-Delta H as an input quantity. Instead, the F-Delta H power dependent limit is used to determine DNBR for those events for which DNB is the acceptance criterion, once the plant's systems response has been determined. The licensee splits the DNBR limited events into two categories. The first category includes those events in which the power-dependent value of F-Delta H is indirectly taken into account by the core limits. The second category includes those events which directly assumes the power-dependent value of F-Delta H in the analysis.

For events in the first category, the licensee used new Overtemperature Delta-T (OTDT) and Overpower Delta-T (OPDT) setpoint equations which include the revised F-Delta H limit of 1.70. Events which require either the OPDT or the OTDT trip functions were reanalyzed. These FSAR events are:

FSAR Section	Event
14.1.2	Uncontrolled RCCA Withdrawal at Power
14.1.6	Reduction in Feedwater Enthalpy Incident
14.1.7	Excessive Load Increase Incident
14.1.9	Loss of External Electrical Load

The uncontrolled rod cluster control assembly (RCCA) withdrawal at power event was reanalyzed at various power levels and reactivity insertion rates, for both minimum and maximum reactivity feedback cases. This transient is terminated by a reactor trip on either High Neutron Flux or Overtemperature Delta-T trip functions. The results of the reanalysis indicate that DNBR never falls below the safety analysis DNBR value. The DNB design basis has, therefore, been met.

The reduction in feedwater enthalpy event is bounded by the excessive load increase event and was not reanalyzed. The excessive load increase event was reanalyzed for both beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions, with and without automatic rod control. The results show that, for all cases, the reactor does not trip but reaches a new equilibrium state. The DNBR value remains above the DNBR safety analysis value. The DNB design basis has, therefore, been met for this event.

The loss of external electrical load event was reanalyzed at both BOC and EOC conditions, with and without pressurizer control and with the reactor in manual control. The reanalysis of this event shows that the DNBR remains above its safety analysis limit value. In addition, the peak reactor coolant system pressure and secondary side pressure remain below 2500 and 1100 psia, respectively, that is, below the design pressure values. The DNBR value remains above the safety analysis DNBR design limit for all of the cases analyzed. The system pressure and DNB design bases have, therefore, been met for this event.

For events in the second category, the increased value for F-Delta H was used in the analysis of the following FSAR events:

FSAR Section	Event
14.1.1	Uncontrolled RCCA Withdrawal from Subcritical Condition
14.1.3	RCCA Drop
14.1.5	Startup of an Inactive Reactor Coolant Loop
14.1.8	Loss of Reactor Coolant Flow

In general, an increase in F-Delta H results in a decrease in DNBR for a given set of thermal-hydraulic conditions.

The uncontrolled RCCA withdrawal from subcritical conditions event was analyzed assuming the most limiting axial and radial power shapes associated with having the two highest combined worth sequential control rod banks in their highest worth position. The maximum withdrawal speed of 45 inches/minute is assumed in the analysis for a reactivity insertion rate of 100 pcm/second. The results of the reanalysis indicate that DNBR remains above the safety analysis DNBR limit. Therefore, the DNB design basis for this event has been met.

The RCCA drop event consists of two separate events. These events are (1) a rod drop event, and (2) a misaligned rod event. The analysis of the rod drop event was performed with an unapproved Westinghouse rod drop methodology (Ref. 24). The staff requested that an analysis be performed with an approved methodology. The licensee submitted an analysis based on the methodology currently used for PBNP (Ref. 28). A number of dropped rod cases were evaluated with respect to the DNBR design basis and acceptable results were obtained. The misaligned rod event results in an increase in the radial heat flux hot channel factor. However, the safety limit DNBR design limit is met. The DNB design basis has, therefore, been met for the misaligned and dropped rod events.

The startup of an inactive reactor coolant loop event was analyzed at a reactor core power of 10% of full rated power. The event is a reactivity excursion which causes an increase in core heat flux. The transient is a relatively mild event with DNBR remaining above the safety analysis DNBR design limit. The DNB design basis has, therefore, been met for this event.

The loss of reactor coolant flow event consists of two separate events: (1) the two-pump coastdown event, and (2) the one-pump coast down event. The loss of reactor coolant flow event caused by a locked rotor is reviewed elsewhere in this Safety Evaluation. The results of the analysis indicate that the safety analysis DNBR design limit has been met for these two flow coastdown events. The DNB design basis has, therefore, been met for this event.

In addition, the licensee considered the effect of the increase in F-Delta H on the steamline break accident which is discussed in PBNP FSAR Section 14.2.5. The licensee performed the analysis of this event at hot, zero power conditions with the most reactive control rod stuck in its fully withdrawn position. The increase in the power-dependent F-Delta H limit results in an increase in the stuck rod power peaking factor at zero power, with a resulting decrease in DNBR. The licensee states that its analysis shows that the safety analysis DNBR design limit is met. The licensee also concludes that the mass and energy release to containment event is not impacted by the increase in F-Delta H because the primary to secondary heat transfer characteristics of the event are not affected. The licensee concludes that this event is not impacted by the increase in F-Delta H.

The licensee analyzed two events affected by the increase in F_Q. These events are:

FSAR Section	Event
14.1.8	Locked Rotor
14.2.6	Rod Ejection

The locked rotor event is classified as an accident. The results of the analysis show that the maximum reactor coolant system pressure (RCS) is 2744 psia, the maximum cladding temperature is 2166°F, the amount of zirconium-water reaction is 1.30% by weight, and less than 86% of the fuel rods in the core undergo DNB. Because RCS pressure remains below the faulted condition pressure and because the core remains in place with no consequential loss of core cooling capability, the locked rotor event, therefore, meets all applicable safety criteria.

The rod ejection event is also classified as an accident. For all of the cases analyzed, the maximum fuel stored energy is less than 200 cal/gm, and the maximum fuel melt at the hot spot is less than 10%. The analysis of the rod ejection event, including the effect of an increased peaking factor F_Q, shows that the applicable criteria for this event have been met.

The boron dilution event was reanalyzed by the licensee although it was not directly affected by the upgraded core features. Dilution events were analyzed for refueling, startup, and power operation. The results obtained show that it would take at least 30.1 minutes before the loss of shutdown margin for the refueling dilution event. About 18.8 minutes would be available for operator action before the reactor would become critical for the startup dilution event. At least 16.2 minutes would be available for operator action for a boron dilution event during power operation. The results show that, for all cases, sufficient time is available for the operator to determine the cause and take corrective action before the required shutdown margin is lost.

The loss of normal feedwater event was reanalyzed by the licensee even though it was not directly affected by the upgraded core features. The reactor is protected by a reactor trip either on low-low water level in either steam generator or on steam flow-feedwater flow mismatch coincident with low water level in either steam generator and by the auxiliary feedwater system. The results of the analysis show that the auxiliary feedwater system provides sufficient flow to the two steam generators to maintain heat transfer capability to prevent water relief from the reactor coolant system relief or safety valves. Therefore, the loss of normal feedwater event does not lead to any adverse core conditions.

The loss of all AC power to the station auxiliaries was reanalyzed by the licensee even though it was not directly affected by the upgraded core features. The assumptions used in the analysis are similar to those for the loss of normal feedwater event except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip. The results of the analysis show that the natural circulation flow that is available is sufficient to provide adequate decay heat removal following reactor trip and reactor coolant pump coastdown. No water relief occurs for this event through the pressurizer relief or safety valves. Therefore, the loss of AC power to the station auxiliaries event does not result in adverse core conditions.

The licensee reevaluated fuel handling accidents. The following conservative assumptions were made: (1) all fuel rods in an assembly are assumed to be damaged, (2) the assembly power is assumed to be 1.8 times the core average assembly power, (3) fission products released from the assembly consist of 3.6% halogens (as I-131) and 30% noble gases (as Kr-85) (these values are based on a conservative axial power distribution of 1.87 peak to average, corresponding to a peak linear assembly power of 15.6 kW/ft), (4) of the halogens released, only 0.01 escape from the spent fuel pool surface to the environment, and (5) of the noble gases released, 100% escape the spent fuel pool surface to the environment. The 2-hour site boundary thyroid dose is estimated to be 17.5 rem, based on the above assumptions. This dose is much less than the 10 CFR Part 100 guideline value of 300 rem. The integrated whole body dose for distances beyond the site boundaries is less than 10 rem, which is less than the 10 CFR guideline value.

The staff has evaluated the licensee's evaluation and analysis of the non-LOCA events, using the revised safety analysis assumptions associated with the upgraded core features, and concludes that they are acceptable because (1) approved methodologies and computer codes have been used, and (2) all applicable safety criteria have been met.

2.7 Large-Break and Small-Break LOCA Analyses

2.7.1 Large-Break LOCA Analysis

The Point Beach Nuclear Plant (PBNP) is a Westinghouse-designed two-loop plant equipped with a low-pressure upper plenum injection (UPI) system as part of the emergency core cooling system (ECCS). The previous PBNP ECCS evaluation model assumed that the UPI water fell directly into the lower plenum without interaction with the core, and could therefore be treated as if it were a cold-leg injection plant. In support of the proposed TS change to increase the power peaking factors, the licensee in a letter dated November 30, 1988 (Ref. 4) provided a new large-break LOCA (LBLOCA) analysis described in Addendum 2 to WCAP-10924-P, Revision 1, Volume 2. This LBLOCA analysis uses a new Westinghouse ECCS evaluation model developed for application to the two-loop UPI plants. This new ECCS model, described in Westinghouse topical report WCAP-10924-P (Ref. 5), uses a best-estimate thermal-hydraulic code WCOBRA/TRAC and the approach described in SECY 83-472 (Ref. 25). In using the SECY 83-472 approach, an estimate of the 95th percentile peak cladding temperature (PCT) is calculated using a best-estimate code and accounting for the uncertainties associated with the code and application. Another calculation is also required to determine the "Appendix K PCT" by applying all the required features set forth in Appendix K to 10 CFR Part 50. The "Appendix K PCT" must then be shown to be greater than the 95th percentile PCT and remains below the 2200°F acceptance criterion. WCAP-10924-P has been reviewed and approved by NRC for referencing in the licensing calculations (Ref. 6), and has been used by Northern States Power Company for application to the Prairie Island (PI) unit which was the lead plant in using the methodology of WCAP-10924-P.

Addendum 2 to Volume II of WCAP-10924-P provides the Point Beach plant-specific analysis to demonstrate that the method of analysis complies with the SECY 83-472 guidelines and Appendix K requirements, and that the acceptance criteria of 10 CFR Part 50.46 are not violated with the proposed higher peaking factors, i.e., the enthalpy rise factor, F-Delta H, of 1.70 and the total peaking factor, F_Q, of 2.50. The analysis follows the same procedure described in Volume II of WCAP-10924-P which was done with the data of the lead plant, Prairie Island.

In the PBNP plant modeling, the primary and secondary loop models are the same as the PI lead plant model. The reactor vessel model follows the same approach and details of the PI unit in using the four-channel core model, but accounts for the differences in the reactor internals between the two plants. The major differences in the reactor internals are in the upper plenum configuration.

For example, the PBNP unit has free-standing mixers above some of the open holes on the upper core plate whereas the PI units have no free-standing mixer; and the PBNP has a flat upper support plate compared to the inverted top hat upper support plate for the PI units. Also the core barrel-baffle arrangements are such that, during steady state, the barrel-baffle flow is an upflow in the PBNP unit compared to a downflow in the PI units. These differences are reflected in the plant modeling. In addition, since the PBNP upper plenum configuration is different from the lead plant, a sensitivity study is required, as specified in the staff safety evaluation report for WCAP-10924-P, to determine the upper plenum structure under which the hot assembly and hot rod would be placed to obtain the highest PCT. This sensitivity study, described in Section 5-3 of Addendum 2, is performed using a three-channel core configuration as was done for the lead plant. The result of the sensitivity study justifies the locations of the hot assembly and hot rod in the PBNP reactor vessel model. For the four-channel core model, the outer low power channel uses a conservatively high power factor to represent the flatter radial power profile expected for the PBNP reload designs. The use of flatter radial power profile is conservative because a sensitivity study has shown that it will result in poorer core cooling and therefore higher PCT.

In accordance with the methodology of WCAP-10924-P, the PCT's are calculated for both the blowdown and reflood peaks. The calculations are made for the realistic nominal condition, superbounded condition, and with Appendix K requirements. The 95th percentile PCT's at the blowdown and reflood peaks are obtained from the superbounded PCT's plus the respective code and application uncertainties.

In order to perform the superbounded calculation, conservative bounding values and assumptions of some plant parameters and models are used in the calculation. Sensitivity studies would be necessary to determine the directions of conservatism for the parameter uncertainties or assumptions, i.e., the directions to place the uncertainties and conservative assumptions that would result in higher PCT. The licensee asserted that the sensitivity studies performed in Volume II of WCAP-10924-P for the PI lead plant are bounding for the PBNP unit because the PI units have higher core power to ECCS flow ratio and therefore a greater PCT sensitivity. In addition, only the direction of conservatism, instead of the magnitudes of the PCT sensitivity, of the parameters and assumptions are used in placing the conservative bounding values and conditions for the superbounded calculations. The staff agrees with this observation that the PI sensitivity study results are applicable to the PBNP superbounded PCT calculation.

With regard to the Appendix K PCT calculation, the staff, in the evaluation report for acceptance of WCAP-10924-P for licensing application to Westinghouse two-loop UPI plants, required that the UPI-licensees apply for exemptions to Items I.D.3 and I.D.5 of Appendix K to 10 CFR Part 50. These exemptions are necessary because Item I.D.3, which requires the use of a carryover fraction to calculate the reflood core exit fluid flow, and Item I.D.5, which sets specific requirements for refill and reflood heat transfer calculation, were intended for the conventional cold-leg injection plants and are not applicable to the UPI plants. The licensee, in its November 30, 1988 letter, requested an exemption to these two requirements and the exemption request has been granted (Ref. 26).

The analysis was performed with the proposed enthalpy rise factor and total peaking factor of 1.70 and 2.50, respectively, and assuming a full core of 14x14 optimized fuel assemblies (OFA) and a steam generator tube plugging level of 25 percent. In addition, since PBNP units are licensed to operate at the nominal reactor system pressures of 2250 and 2000 psia, the nominal pressure of 2250 psia was used in the analysis. This is because the sensitivity study indicated that the operating pressure of 2250 psia produced the highest PCT, and therefore the analysis using 2250 psia would be a bounding analysis. The results show a reflood PCT of 2023°F for the Appendix K calculation. This PCT is higher than the 95th percentile PCT of 1932°F and 1892°F, respectively, for the blowdown and reflood peaks, and below the 2200°F acceptance criterion. In addition, the Appendix K calculation results shown in Table 6-4 of Addendum 2 indicate that both maximum local cladding oxidation and total hydrogen generation are below the acceptance criteria of 17 percent and 1 percent, respectively.

Since the analysis assumed a full core of 14x14 OFA fuel, this is inconsistent with the actual fuel loading of a transitional mixed core of standard, OFA and upgraded OFA fuel assemblies, and an adjustment for the calculated PCT may be needed to account for the neglect of the effect of the hydrodynamic mismatch among the different fuel designs. However, since both (1) the OFA and the upgraded OFA fuel designs have the same hydrodynamic characteristics, and (2) the standard fuel has higher flow resistance (but is not the limiting fuel assembly), and would therefore increase flow into the more limiting OFA fuel types, the analysis assumption of a full core of 14x14 OFA fuel bounds the mixed core effects.

2.7.2 Small-Break LOCA Analysis

The licensee has performed a reanalysis of small-break LOCA (SBLOCA) using the approved method of WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code." The analysis assumed a full core of 14x14 OFA fuel and 25 percent steam generator tube plugging, and used the proposed peaking factors of 1.70 and 2.50, respectively, for the enthalpy rise factor and total peaking factor. In addition, the third segment of the K(Z) curve was not used, consistent with the proposed TS change. The analysis was performed for a 4-inch cold-leg break. Use of this break size and location as the limiting case was based on a previous generic Westinghouse two-loop plant analysis and Prairie Island SBLOCA analysis using the NOTRUMP code. The RCS pressures of both 2250 and 2000 psia were analyzed, the analyses demonstrated the 2000 psia case was limiting. The analysis result of the 2000 psia case shows a PCT of 809°F, far below the 2200°F acceptance criterion. Therefore, there is no concern that a SBLOCA would result in violation of the ECCS acceptance criteria of 10 CFR 50.46.

2.7.3 Staff Position on PBNP LOCA Analyses

The staff has reviewed both LBLOCA and SBLOCA analyses in support of the PBNP technical specification changes for increased peaking factors, and concludes that the ECCS acceptance criteria set forth in Section B of 10 CFR 50.46 have been complied with.

2.8 Steam Generator Tube Rupture Accident Analysis

The licensee reanalyzed the steam generator tube rupture (SGTR) event for PBNP using the same methodology as in the existing SGTR analysis with two key changes in the assumptions. These are (1) the increased peaking factors proposed by the licensee for PBNP and (2) that both safety injection pumps will run for 30 minutes to reflect a change in the safety injection (SI) termination portion of the SGTR recovery procedures. The reanalysis generates maximum radiological doses of 2.13 rem to the thyroid and 0.059 rem to the whole body. Although slightly higher than doses cited in the existing FSAR for the SGTR Accident Analysis (0.700 rem to the thyroid and 0.200 to the whole body), the doses calculated for the reanalysis of the SGTR accident remain a small fraction of the 10 CFR Part 100 exposure guidelines and are therefore acceptable. Because the methodology used in this reanalysis is the same as the staff-approved methodology used in the existing analysis, the methodology used remains acceptable.

Departure from nucleate boiling (DNB) is not approached in the design basis SGTR analysis. Thus, the increase in peaking factors maintains acceptable results in the SGTR scenario.

2.9 <u>Technical Specifications</u>

(1) Specification 15.2.1.1 and Basis - This Specification and Basis were rewritten to eliminate reference to the transition core safety limits. The changes to pages 15.2.1-1 and 15.2.1-2 are acceptable because future PBNP cores will not require transition core penalties, as discussed in the new PBNP safety analysis.

(2) Figures 15.2.1-1 and 15.2.1-2 are replaced with a revised Figure 15.2.1-1. This is acceptable because a figure related to transition cores is removed and a new figure which corresponds to the new PBNP safety analysis is included.

(3) Specification 15.2:3.1.B.3 - The removal of the asterisk and footnote in this specification is acceptable because the safety analysis was performed at a bounding value of the pressure.

(4) Specifications 15.2.3.1.B.4 and 15.2.3.1.B.5 - These specifications have been revised to reflect the new setpoints used in the PBNP safety analysis. The changes are, therefore, acceptable.

(5) Specification 15.2.3 Basis - The references to the transition core are removed from page 15.2.3-6. This is acceptable because transition core penalties will no longer be required for the PBNP.

(6) Specification 15.3.1.C Basis - The assumed steady-state primary-to-secondary steam generator leakage rate is changed to make it consistent with the more conservative value in Specification 15.3.1.D.4. The new steady-state leakage rate is used as an input to the steam generator tube rupture event. This change is, therefore, acceptable.

(7) Specifications 15.3.1.G.1 and 15.3.1.G.2 - The change to Specification 15.3.1.G.1 is made to reflect the value used in the PBNP safety analysis. The change to Specification 15.3.1.G.2 is made to reflect the location where the pressure indication is taken. The footnote is removed to reflect the fact that the safety analysis was performed at a bounding value of the pressure. These changes are, therefore, acceptable.

(8) Specification 15.3.3 Basis - An addition is made to the Basis to describe the basis for the RWST minimum volume and minimum boron concentration. This change, is therefore, acceptable.

(9) Specification 15.3.10.B.1.a - This specification was changed to reflect the value of F_Q equal to 2.50 and the value of F-Delta H equal to 1.70 that were used in the PBNP safety analysis. These changes are, therefore, acceptable.

(10) Specification 15.3.10 Basis - The Basis to Specification 15.3.10 is changed to reflect the new values of F_Q and F-Delta H. A clarification was also made to the basis of the Hot Channel Factor Normalized Operating Envelope and its use in the safety analysis. These changes are, therefore, acceptable.

(11) Specification 15.3.10 - Figure 15.3.10-1 on control bank insertion limits was revised to obtain a wider delta-flux band. This change is acceptable because it is used in the PBNP safety analysis.

(12) Specification 15.3.10 - Figure 15.3.10-3 on Hot Channel Factor Normalized Axial Operating Envelope (K(Z) curve) is revised. This revised figure is acceptable because it results in an acceptable small-break LOCA analysis as well as other safety analyses.

(13) Specification 15.3.10 - Figure 15.3.10-4 on Flux Difference Operating Envelope (delta-I band) is revised. This change is acceptable because adherence to the delta-I band limits will ensure that the power distribution limits of the safety analysis will be enforced.

(14) Specifications 15.5.3.A.2, 15.5.3.A.3, 15.5.3.A.4, and 15.5.3.A.5 - Specification 15.5.3.A.2 has been revised and the current Specifications 15.5.3.A.3, 15.5.3.A.4 and 15.5.3.A.5 have been deleted to revise the description of the reactor core by eliminating reference to a transition core. This is acceptable because transition cores will no longer be used in future PBNP reloads.

(15) IFBA Description Addition - Specification 15.5.3.A.6 is deleted and Specification 15.5.3.A.3 in the Reactor Design Features section of the Technical Specifications is inserted to describe the integral fuel burnable absorbers (IFBA's). This change is acceptable because it is one of the upgraded core features in future PBNP reloads.

(16) Specification 15.5.3.A.7 has been renumbered to 15.5.3.A.4. This change is acceptable because it is an administrative change.

(17) Water Displacer/Neutron Source Description Addition - Specification 15.5.3.A is revised to include a description of water displacer rods. Specification 15.5.3.A.5 is added to describe the neutron source assemblies. The addition of the water displacer rods is acceptable because these water displacer rods have been previously used at PBNP. The addition of the description of the neutron source assemblies is acceptable because it provides a necessary description of a core feature.

(18) Peripheral Power Suppression Assemblies Description Addition - Specification 15.5.3.A.6 is added to describe the peripheral power suppression assemblies (PPSA's) to be used in future PBNP reloads. The addition of the description of the PPSA's is acceptable because it provides a necessary description of a core feature.

3.0 FINDINGS

The staff has reviewed the request by the Wisconsin Electric Power Company to operate the Point Beach Nuclear Plants, Units 1 and 2, with upgraded core features, including an increased total power peaking factor (F_Q) of 2.50 and an increased enthalpy rise hot channel factor (F-Delta H) of 1.70. Based on this review, the staff concluded that appropriate material was submitted and that normal operation and the transients and accidents that were evaluated and reanalyzed are acceptable. The Technical Specifications submitted for this license amendment suitably reflect the necessary modifications for the operation of future PBNP reloads.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 28, 1989 (54 FR 12696). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of the human environment.

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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: May 8, 1989

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