February 8, 2000

Mr. Michael B. Sellman
Senior Vice President and
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: DESIGN AND OPERATION OF FUEL CYCLES WITH UPGRADED WESTINGHOUSE FUEL (TAC NOS. MA5939 AND MA5940)

Dear Mr. Sellman:

The Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. DPR-24 and Amendment No. 198 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, respectively. The amendments revise the Technical Specifications (TSs) in response to your application dated June 22, 1999, as supplemented December 17, 1999.

These amendments reflect changes to the TSs related to the design and operation of the Point Beach fuel cycle in order to incorporate the Westinghouse 422V+ fuel assemblies into reactor cores.

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

Sincerely,
/RA/
Gregory P. Hatchett, Interim Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures: 1. Amendment No. 193 to DPR-24
2. Amendment No. 198 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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

February 8, 2000

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Senior Vice President and
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: DESIGN AND OPERATION OF FUEL CYCLES WITH UPGRADED WESTINGHOUSE FUEL (TAC NOS. MA5939 AND MA5940)

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Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

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cc w/encls: See next page

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
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 June 22, 1999, as supplemented December 17, 1999, 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. 193 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of issuance: February 8, 2000

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
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 June 22, 1999, as supplemented December 17, 1999, 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. 198 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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

Attachment: Changes to the Technical Specifications

Date of issuance: February 8, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 193

TO FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 198

TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

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

REMOVE	INSERT
Page 15.2.1-1	Page 15.2.1-1
Page 15.2.1-2	Page 15.2.1-2
Figure 15.2.1-1	Figure 15.2.1-1
Figure 15.2.1-2	Figure 15.2.1-2
Page 15.2.3-1	Page 15.2.3-1
Page 15.2.3-2	Page 15.2.3-2
Page 15.2.3-3	Page 15.2.3-3
Page 15.2.3-3a	Page 15.2.3-3a
Page 15.2.3-5	Page 15.2.3-5
Page 15.2.3-6	Page 15.2.3-6
Page 15.2.3-7	Page 15.2.3-7
Page 15.3.1-19	Page 15.3.1-19
Table 15.3.5-1 (page 1 of 2)	Table 15.3.5-1 (page 1 of 2)
Page 15.3.10-5	Page 15.3.10-5
Page 15.3.10-14 thru Page 15.3.10-18	Page 15.3.10-14 thru Page 15.3.10-18
Figure 15.3.10-3	Figure 15.3.10-3
-	Figure 15.3.10-3A
Page 15.5.3-1	Page 15.5.3-1
Page 15.5.3-3	Page 15.5.3-3
Page 15.5.4-1	Page 15.5.4-1

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 or Figure 15.2.1-2 as applicable for Units 1 and 2. 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.

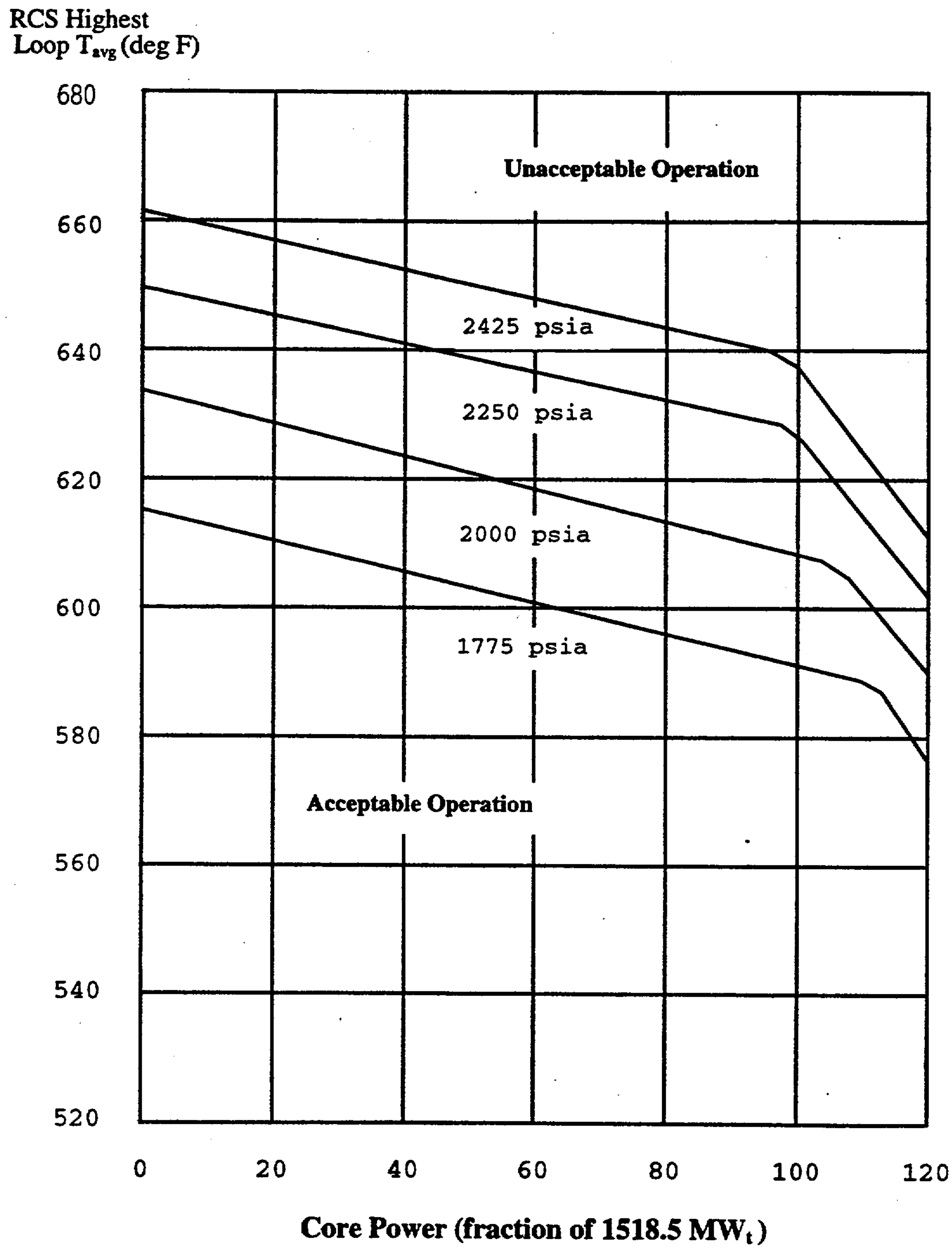
Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore 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 percent probability at a 95 percent 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 family of curves in Figure 15.2.1-1 is applicable to a core with any combination of 14 x 14 OFA and 14 x 14 upgraded OFA fuel assemblies. The family of curves in Figure 15.2.1-2 is applicable to any combination of 422V+ fuel assemblies, burned 14 x 14 OFA fuel assemblies, and burned 14 x 14 Upgraded OFA fuel assemblies, or a full core of 422V+ fuel assemblies. 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*
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2
REACTOR CORE SAFETY LIMITS

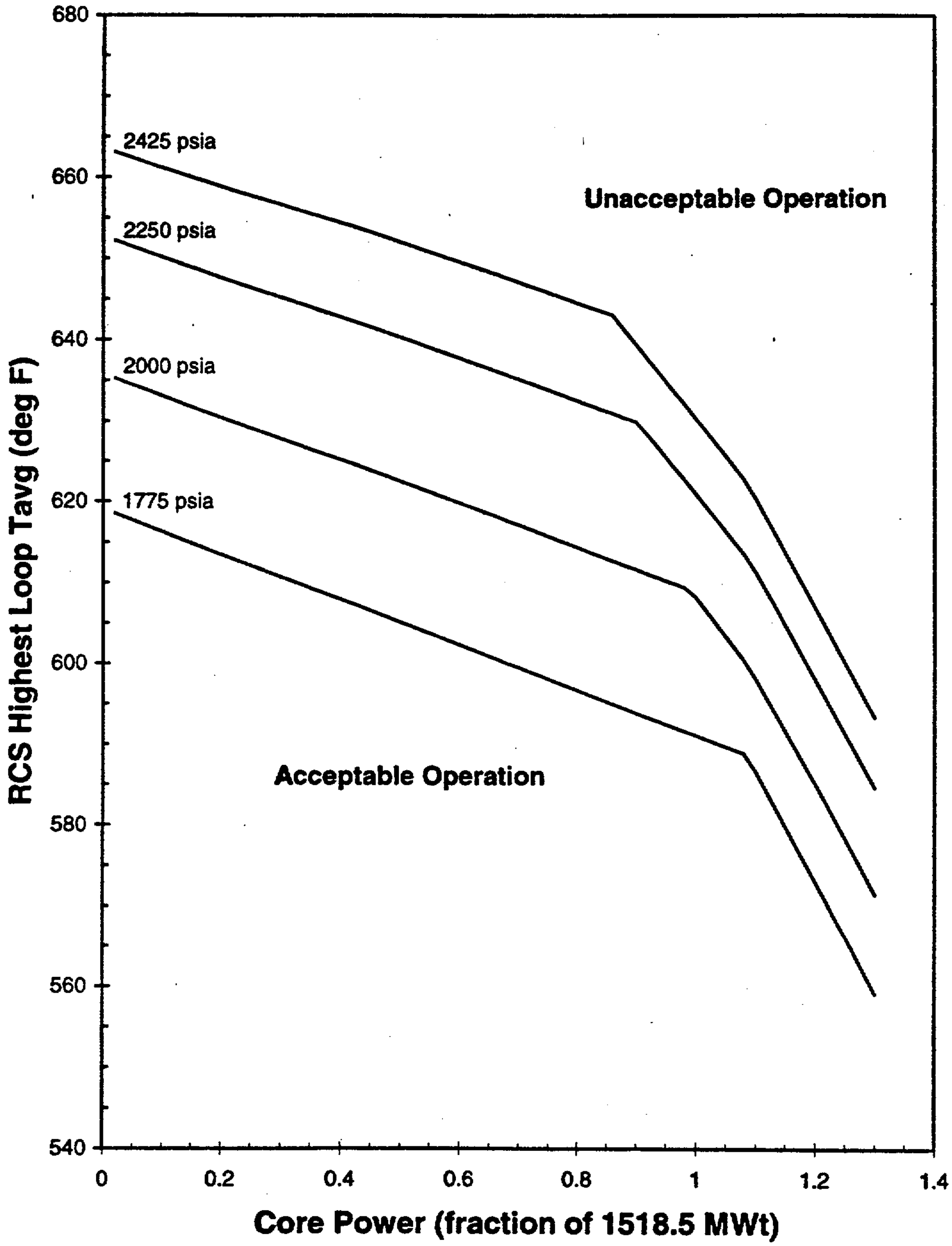


*	This figure applies to core reloads with any combination of OFA and Upgraded OFA fuel assemblies.

Unit 1 - Amendment No. ~~173~~, 193
Unit 2 - Amendment No. ~~177~~, 198

Figure 15.2.1-2*

POINT BEACH NUCLEAR PLANT UNITS 1 AND 2
REACTOR CORE SAFETY LIMITS



* This figure applies to core reloads with any combination of 422V+ fuel assemblies, burned OFA and burned Upgraded OFA fuel assemblies, or a full core of 422V+ fuel assemblies.

15.2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability:

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

Objective:
To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification:
1. Protective instrumentation for reactor trip settings shall be as follows:

    A. Startup protection

        (1) High flux, source range - within span of source range instrumentation.

        (2) High flux, intermediate range - ≤40% of rated power.

        (3) High flux, power range (low setpoint) - ≤25% of rated power.

    B. Core limit protection

        (1) High flux, power range (high setpoint) - ≤108% of rated power.

        (2) High pressurizer pressure - ≤2385 psig for operation at 2250 psia primary system pressure
        ≤2210 psig for operation at 2000 psia primary system pressure and cores not containing 422V+ fuel assemblies

(3) Low pressurizer pressure - $\geq$1905 psig for operation at 2250 psia primary system pressure
$\geq$1800 psig for operation at 2000 psia primary system pressure and cores not containing 422V+ fuel assemblies

(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 (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	≤	569.0°F (for cores containing 422V+ fuel assemblies)
T'	≤	572.9°F (for cores not containing 422V+ fuel assemblies)
P	=	pressurizer pressure, psig
P'	=	2235 psig (for 2250 psia operation)
P'	=	1985 psig (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	1.16 (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_1	≤	1.19 (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	1.14 (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	0.0149 (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_2	=	0.025 (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	0.022 (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	0.00072 (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_3	=	0.0013 (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	0.001 (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
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, $f(\Delta I) = 0$ for cores not containing 422V+ fuel assemblies; for $q_t - q_b$ within –12, +5 percent, $f(\Delta I) = 0$ for cores containing 422V+ fuel assemblies.

(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 for cores not containing 422V+ fuel assemblies and reduced by an equivalent of 2.12 percent of rated power for cores containing 422V+ fuel assemblies.

(c) for cores not containing 422V+ fuel assemblies, 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; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of q_t - q_b exceeds –12 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.

(5) Overpower

$$\Delta T \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_5 S}{\tau_5 S + 1}\right)\left(\frac{1}{1+\tau_4 S}\right) T - K_6 [T\left(\frac{1}{1+\tau_4 S}\right) - T']]$$

where (values are applicable to operation at both 2000 psia and 2250 psia)

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	≤	569.0°F (for cores containing 422V+ fuel assemblies)
T'	≤	572.9°F (for cores not containing 422V+ fuel assemblies)
K_4	≤	1.10 of rated power (for cores containing 422V+ fuel assemblies)
K_4	≤	1.09 of rated power (for cores not containing 422V+ fuel assemblies)
K_5	=	0.0262 for increasing T
	=	0.0 for decreasing T
K_6	=	0.00103 for $T \geq T'$ (for cores containing 422V+ fuel assemblies)
K_6	=	0.00123 for $T \geq T'$ (for cores not containing 422V+ fuel assemblies)
	=	0.0 for $T < T'$
τ_5	=	10 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

(6) Undervoltage - ≥3120V

(7) Indicated reactor coolant flow per loop ≥90 percent of normal indicated loop flow

(8) Reactor coolant pump motor breaker open
   (a) Low frequency set point ≥55.0 HZ
   (b) Low voltage set point ≥3120V

C. Other reactor trips:

(1) High pressurizer water level - ≤95% of span

(2) Low-low steam generator water level -
≥20% of narrow range instrument span

(3) Steam-Feedwater Flow Mismatch Trip - ≤1.0×10^6 lb/hr

(4) Turbine Trip (Not a protection circuit)

(5) Safety Injection Signal

(6) Manual Trip

Basis

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.[1]

The high flux low power reactor trip provides redundant protection in the power range for a power excursion beginning from low power. This trip insures that a more restrictive trip point is used for this case than for an excursion beginning from near full power.[1]

The overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[3]

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial 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 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 for OFA and Upgraded OFA fuel include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints for 422V+ fuel do not include the effect of reduced system pressure operation; therefore, cores containing 422V+ fuel must be operated at 2250 psia. The setpoints will not exceed the core safety limits as shown in Figures 15.2.1-1 (for OFA and Upgraded OFA fuel only cores) and 15.2.1-2 (for cores containing 422V+ fuel).

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 setpoint 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 percent 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, 55.0 HZ, 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 setpoint 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 setpoint 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 setpoint 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 ±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 power operation is not be permitted with only one reactor coolant pump. If a pump is lost while operating below 50 percent power, an orderly shutdown 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 14.0

[3] FSAR 14.2.6

[4] FSAR 14.3.1

[5] FSAR 14.1.2

[6] FSAR 7.2, 7.7

[7] FSAR 3.2.1

[8] FSAR 14.1.8

[9] FSAR 14.1.10 and 14.1.11

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 ≥558.1°F and ≤574.0°F for cores containing 422V+ fuel assemblies. T_{avg} shall be maintained ≥557°F and ≤573.9°F for cores not containing 422V+ fuel assemblies.

2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:
≥2205 psig during operation at 2250 psia, or
≥1955 psig during operation at 2000 psia for cores not containing 422V+ fuel assemblies.

3. Reactor Coolant System raw measured Total Flow Rate shall be maintained ≥182,400 gpm for cores containing 422V+ fuel assemblies, or ≥181,800 gpm for cores not containing 422V+ fuel assemblies.

Basis:

The reactor coolant system total flow rate of 182,400 gpm for cores containing 422V+ fuel assemblies is based on an assumed measurement uncertainty of 2.4 percent over thermal design flow (178,000 gpm). The reactor coolant system total flow rate of 181,800 gpm for cores not containing 422V+ fuel assemblies 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 calorimetric at the beginning of each cycle.

TABLE 15.3.5-1
(PAGE 1 OF 2)
ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

NO.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1	High Containment Pressure (Hi)	Safety Injection*	≤ 6 psig
2	High Containment Pressure (Hi-Hi)	a. Containment Spray b. Steam Line Isolation of Both Lines	≤ 30 psig ≤ 20 psig
3	Pressurizer Low Pressure	Safety Injection*	≥ 1715 psig
4	Low Steam Line Pressure	Safety Injection* Lead Time Constant Lag Time Constant	≥ 500 psig ≥ 12 seconds ≤ 2 seconds
5	High Steam Flow in a Steam Line Coincident with Safety Injection and Low T_{AVG}	Steam Line Isolation of Affected Line	≤ d/p corresponding to 0.66×10^6 lb/hr at 1005 psig ≥ 540°F
6	High-high Steam Flow in a Steam Line Coincident with Safety Injection	Steam Line Isolation of Affected Line	≤ d/p corresponding to 4×10^6 lb/hr at 806 psig
7	Low-low Steam Generator Water Level	Auxiliary Feedwater Initiation	≥ 20% of narrow range instrument
8	Undervoltage on 4 KV Busses	Auxiliary Feedwater Initiation	≥ 3120V

* Initiates also containment isolation, feedwater line isolation and starting of all containment fans.

AND

b. Within two hours fully withdraw the shutdown banks.

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

2. When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. If this condition is not met, perform the following actions:

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

AND

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

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

E. POWER DISTRIBUTION LIMITS

1. Hot Channel Factors

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

	For OFA and Upgraded OFA Fuel	For 422V+ Fuel
for $P > 0.5$	$F_Q(Z) \leq (2.50)/P \times K(Z)$	$F_Q(Z) \leq (2.60)/P \times K(Z)$
for $P \leq 0.5$	$F_Q(Z) \leq 5.00 \times K(Z)$	$F_Q(Z) \leq 5.20 \times K(Z)$
	$F^N_{\Delta H} < 1.70 \times [1 + 0.3\ (1-P)]$	$F^N_{\Delta H} < 1.77 \times [1 + 0.3\ (1-P)]$

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 or Figure 15.3.10-3a, as applicable, and Z is the core height location of F_Q.

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

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

Power Distribution

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

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

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

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

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

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

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a fuel rod to the average fuel rod power. Imposed limits pertain to the maximum $F^N_{\Delta H}$ in the core, that is the fuel rod with the highest integrated power. It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat flux is obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

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

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

It has been determined that, provided the following conditions are observed, the hot channel factor limits will be met:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 24 steps from the bank demand position, when the bank demand position is between 30 steps and 215 steps. A misalignment of 36 steps is allowed when the bank position is less than or equal to 30 steps, or, when the bank position is greater than or equal to 215 steps, due to the small worth and consequential effects of an individual rod misalignment.

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

3. Control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation of $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above four conditions are observed, these hot channel factor limits are met. In Specification 15.3.10.E.1.a, F_Q is arbitrarily limited for $p \leq 0.5$.

The upper bound envelope F_Q (defined in 15.3.10.E) times the normalized peaking factor axial dependence of Figure 15.3.10-3 for OFA and Upgraded OFA fuel and Figure 15.3.10.3a for 422V+ fuel (consistent with the Technical Specifications on power distribution control as given in Section 15.3.10) was used in the large and small break LOCA analyses. The envelope was determined based on allowable power density distributions at full power restricted to axial flux difference (ΔI) values consistent with those in Specification 15.3.10.E.2.

The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be taken into account. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.70/1.08$ for OFA and Upgraded OFA and 1.77/1.08 for 422V+ fuel. The logic behind the larger uncertainty in this case is as follows:

(a) Normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q.

(b) While the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$.

(c) An error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control; but compensation for $F^N_{\Delta H}$ is less readily available.

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

The measured hot channel factors are increased as follows:

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

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

Axial Power Distribution

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

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

Quadrant Tilt

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

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

The excore detectors are somewhat insensitive to disturbances near the core center or on the major axes. It is therefore possible that a five percent tilt might actually be present in the core when the excore detectors respond with a two percent indicated quadrant tilt. On the other hand, they are overly responsive to disturbances near the periphery on the 45° axes.

Tilt restrictions are not applicable during the startup and initial testing of a reload core which may have an inherent tilt. During this time sufficient testing is performed at reduced power to verify that the hot channel factor limits are met and the nuclear channels are properly aligned. The excore detectors are normally aligned indicating no quadrant power tilt because they are used to alarm on a rapidly developing tilt. Tilts which develop slowly are more accurately and readily discerned by incore measurements. The excore detectors serve as the prime indication of a quadrant power tilt. If a channel fails, is out-of-service for testing, or is unreliable, two hours is a short time with respect to the probability of an unsafe quadrant power tilt developing. Two hours gives the operating personnel sufficient time to have the problem investigated and/or put into operation one of several possible alternative methods of determining tilt.

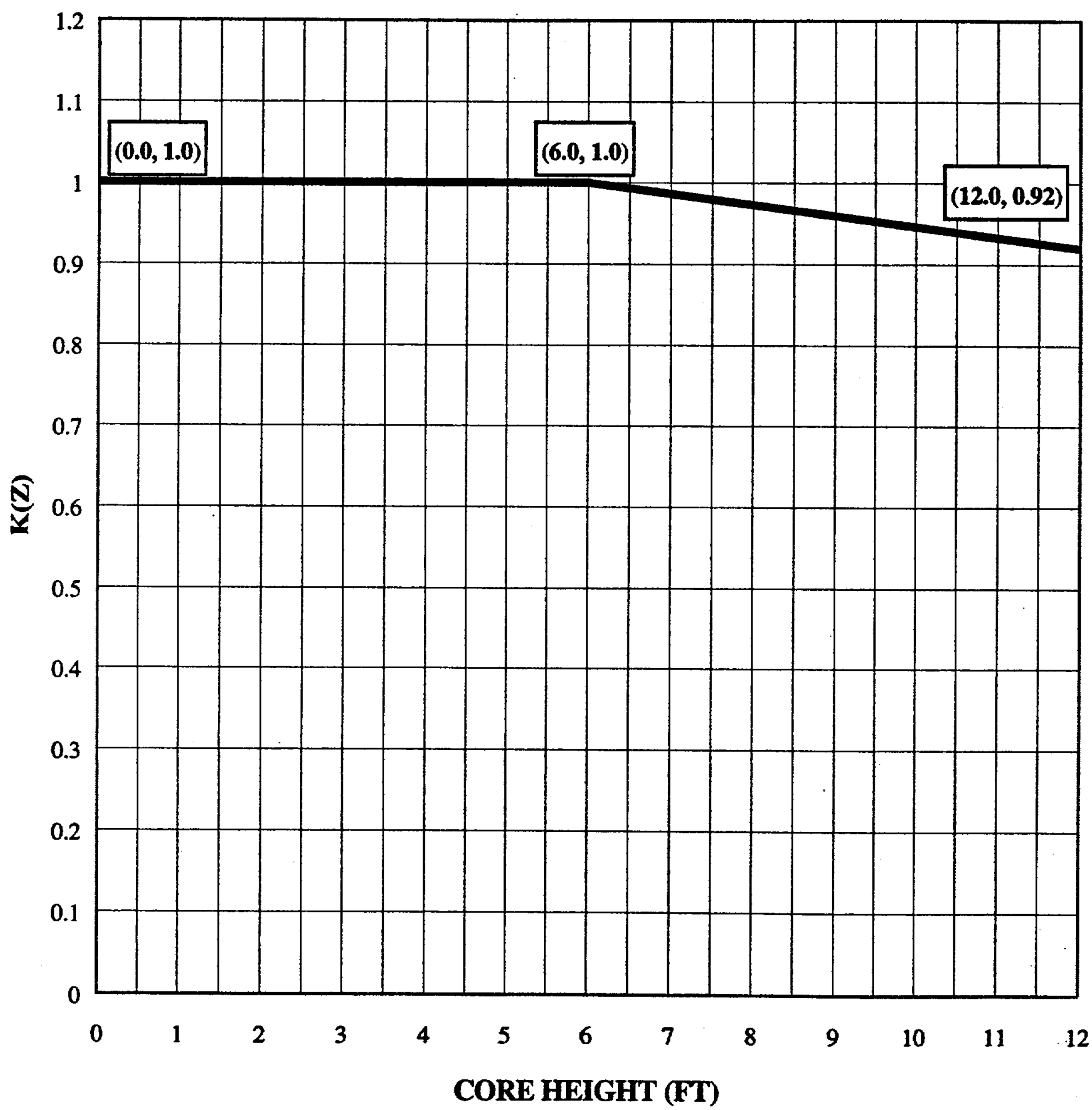
Physics Tests Exceptions

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

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

FIGURE 15.3.10-3

POINT BEACH UNITS 1 AND 2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE FOR OFA
AND UPGRADED OFA FUEL



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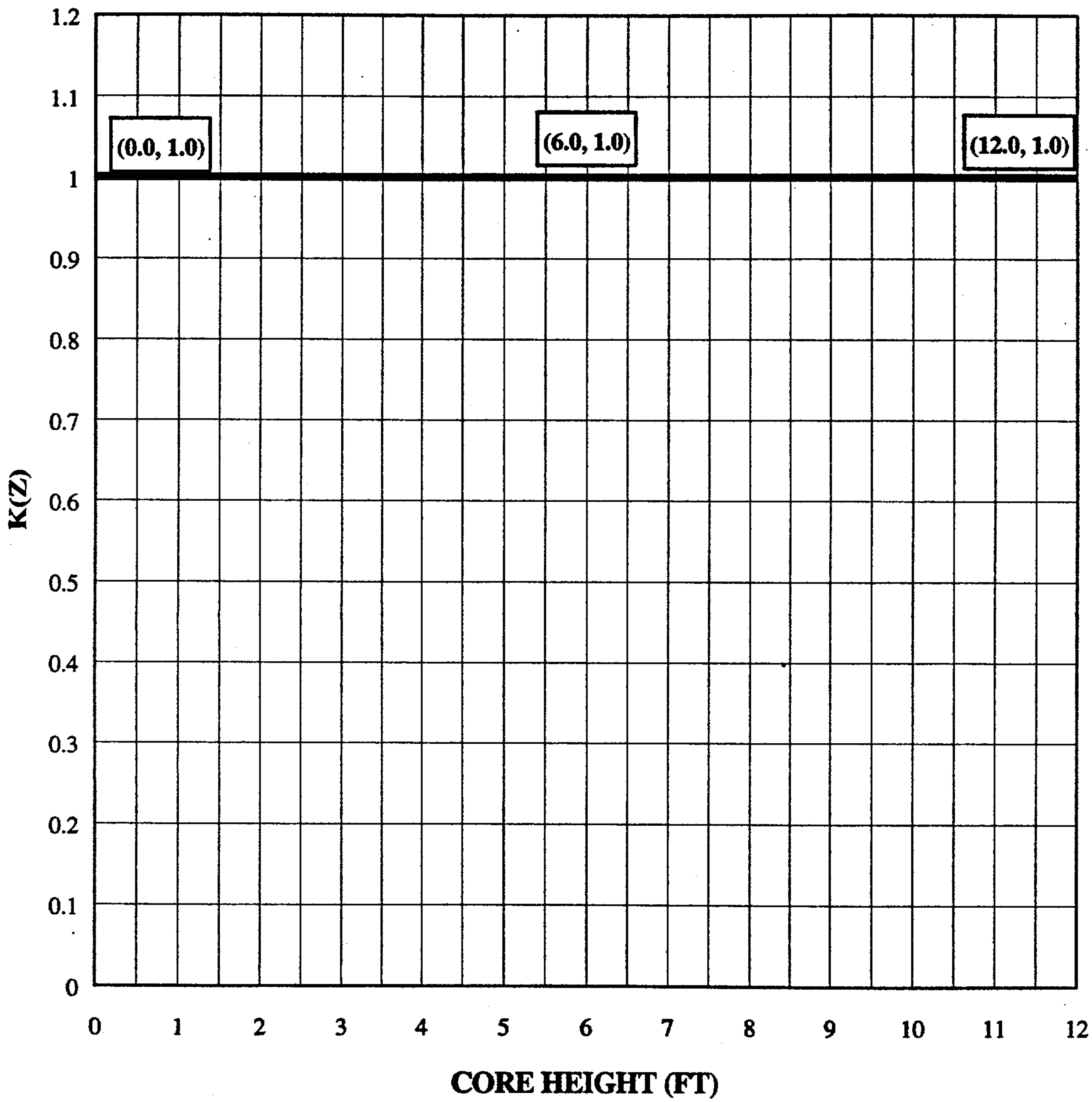
Unit 2 – Amendment No. ~~123~~, 198

FIGURE 15.3.10-3A

POINT BEACH UNITS 1 AND 2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE FOR 422V+ FUEL



Unit 1 – Amendment No. 193

Unit 2 – Amendment No. 198

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 or ZIRLO™ 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, ZIRLO™ , 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, or any combination of 422V+ and burned 14x14 OFA or burned 14x14 upgraded OFA fuel assemblies. The use of these fuel assemblies will be justified by a cycle specific reload analysis.

b. The maximum potential seismic ground acceleration, 0.12g, acting in the horizontal and 0.08g acting in the vertical planes simultaneously with no loss of function.

3. The nominal Reactor Coolant System volume (both liquid and steam) at rated operating conditions and zero percent steam generator tube plugging is:

   Unit 1 - 6500 ft^3

   Unit 2 - 6643 ft^3

References

(1) FSAR Section 3.2

(2) Deleted

(3) Deleted

(4) FSAR Section 3.2

(5) Deleted

(6) FSAR Table 4.1-9

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 5.0 w/o U-235, with or without axial blanket loadings. Each assembly with a fuel loading greater than 4.6 w/o U-235 must contain Integral Fuel Burnable Absorber (IFBA) rods in accordance with Figure 15.5.4-1 for the spent fuel pool. Fresh fuel assemblies with the maximum enrichment of up to 5.0 weight percent U235 and a minimum of 32 1.25X IFBA rods can utilize all available new fuel vault storage cells. 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 2100 ppm boron whenever there are spent fuel assemblies in the storage pool.
4. 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193

TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By application dated June 22, 1999, as supplemented December 17, 1999, the Wisconsin Electric Power Company (the licensee) requested changes to the Technical Specifications (TSs) for Point Beach Nuclear Plant (PBNP), Units 1 and 2. The proposed changes would permit reload core design and operation of future PBNP core operating cycles with the upgraded 0.422-inch outer diameter, 14x14, Vantage + (422V+) fuel design replacing the current optimized fuel assembly (OFA) fuel design, and with operation at higher core power peaking factors beginning with Unit 2, Cycle 25 and Unit 1, Cycle 27. The December 17, 1999, letter provided clarifying information that was within the scope of the original Federal Register notice and did not affect the staff's initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Design Features And Parameters

Compared to the OFA and upgraded OFA assemblies currently in use at PBNP, the 422V+ fuel assemblies covered by this amendment request would have increased fuel rod diameter, increased power peaking factors, an increased radial peaking factor limit, a low pressure-drop mixing vane mid-grid design, and larger instrumentation tubing. In addition, the 422V+ fuel is clad with ZIRLO, while the current fuel cladding is Zircaloy. Nonetheless, the 422V+ fuel is both mechanically and hydraulically compatible with the OFA fuel currently in use at PBNP.

The licensee proposes to begin using 422V+ fuel during reload transition cycles. During the reload transition cycles, the PBNP cores will contain combinations of partially burned OFA, burned upgraded OFA, and 422V+ fuel assemblies. Attachment 2 of the WE, June 22, 1999, application discusses the fuel design changes and describes the reference safety analyses that were performed to bound core conditions containing any combination of the above fuel, including a full core of 422V+ fuel. Future reload, cycle-specific evaluations will be performed utilizing the Westinghouse standard reload methodology (Reference 6) and will verify that the applicable safety limits do not exceed the reference analyses in this license amendment request or currently in the Final Safety Analysis Report (FSAR) (Reference 7). Note that most analyses were performed accounting for a future power uprate from the current licensed power of 1518.5 MWt to 1650 MWt, although this power uprate is not requested at this time.

The current and proposed PBNP key safety parameter changes are:

	Current	Bounding Analysis
Fuel type (Westinghouse)	OFA, upgrade OFA	OFA, upgrade OFA, 422V+
Reactor Core Power (MWt)	1518.5	1650
Coolant System Pressure (psia)	2000 or 2250	2250
Ave. Coolant Temperature, HFP (F)	573.9	558.1 to 574.0
Normal Operation F-delta H	1.70	1.77 (1.70 for OFA)
Normal Operation $F_Q(Z)$	2.50	2.60 (2.50 for OFA)

(F-delta H = full power peaking factor limit; $F_Q(Z)$ = maximum peaking factor limit; and HFP (F) = hot full power)

We have reviewed the design features and key safety parameter changes proposed for the future PBNP cores and conclude that they are acceptable since they are similar to improved 17x17 designs, and have been selected consistent with the NRC-approved Westinghouse fuel criteria evaluation process (FCEP) (Reference 8).

2.2 Fuel Rod Design

The increased fuel rod diameter and the increased power peaking factors affect the fuel rod performance through increases in the steady-state fuel rod power history and the fuel rod transient duty. The licensee states that the fuel rod performance for all fuel types has been shown to satisfy the NRC Standard Review Plan (Reference 9) fuel rod design bases. The design bases for the 422V+ fuel are described in Reference 2. Since the fuel rod design evaluations for the 422V+ fuel were performed with NRC-approved models and design criteria methods and used the approved PAD 3.4 code (Reference 10), the staff concludes that these evaluations are acceptable through the transition cycles to a full core of 422V+ fuel.

2.3 Nuclear Design

The licensee states that multiple cycle core models were established to cover the transition to a full 422V+ fuel core. Typical loading patterns were developed based on projected energy requirements for nominal 18-month cycles, considering coastdown from hot full power (HFP) conditions. The evaluation concentrated both on the initial transition cycle to capture the predominant transition effect and on the equilibrium full core 422V+ cycle. The increased radial peaking factor limit allows an extension of the low leakage fuel management scheme by placing additional burned fuel on the core periphery. Since the analyses were performed with the approved Westinghouse reload safety evaluation (RSE) methodology using approved codes and since acceptable core parameters were obtained from the evaluation and the required TS changes were determined and justified, the staff finds the analyses acceptable.

2.4 Thermal Hydraulic Design

The licensee states that extensive prototype hydraulic testing was performed due to the larger diameter fuel rod, the new low pressure drop mixing vane mid-grid design, and the larger instrumentation tubes, and further states that the OFA and the 422V+ fuel designs are hydraulically compatible. With the increase in system pressure (to 2250), the analyses can support the increase in the power peaking factor because Westinghouse performed bounding analyses for reload transition cores and for full 422V+ core loadings. Therefore, the departure from nucleate boiling (DNB) analysis of cores containing both OFA and 422V+ fuel has been shown to be valid using the approved Westinghouse FCEP approach (Reference 9) with the WRB-1 DNB correlation (References 11 and 12), the revised thermal design procedure (RTDP) methodology (References 4 and 5), and the improved THINC-IV model (Reference 13). The licensee states that the approved W-3 correlation and the approved standard thermal design procedure (STDP) method may still be used when conditions are outside of the range of the WRB-1 correlation and beyond the statistical variance of the RTDP methods.

The licensee also states that the use of the WRB-1 correlation with a 95-percent probability at a 95-percent confidence level (95/95) correlation limit, the departure from nucleate boiling ratio (DNBR) of 1.17 is appropriate for both the OFA and 422V+ fuel, using the FCEP approach. Currently the design limit DNBR values for OFA are 1.22/1.21 for typical/thimble cells, respectively. With the proposed PBNP plant-specific RTDP uncertainties, the fuel upgrade analysis application uses design limit DNBR values of 1.24/1.23 for both OFA and 422V+ fuel for typical/thimble cells. For the DNB safety analyses, the design limit DNBR is conservatively increased to provide margin to rod bow, transition mixed fuel core, and other penalties to provide design and operational flexibility. The safety analyses DNB limit used for the OFA fuel is 1.32/1.32 and for the 422V+ fuel the safety analysis limit is 1.36/1.36 (typical/thimble). For the proposed fuel upgrade DNB analysis, the maximum F-delta H is 1.77 for 422V+ and 1.70 for OFA fuel. Since approved methods and correlations are used with conservative application of the RTDP uncertainties, the staff finds this approach acceptable.

2.5 Non-LOCA Accidents

The licensee has evaluated the impact of the upgraded 422V+ reload core features on the non-LOCA events in Chapter 14 of the PBNP FSAR and has provided a draft of the required FSAR updates. The licensee has used the approved RSE methodology with other approved methodologies and design codes, as appropriate. The licensee stated that it was determined that the current licensing basis analysis remains bounding for the non-LOCA systems and components. A new fuel handling accident analysis was performed using the 422V+ fuel source term and it was determined that the offsite dose acceptance criteria are met based on delaying fuel movement until 161 hours after the reactor goes subcritical. Because the current licensing basis analysis is bounding, the proposed amendment is acceptable with respect to these considerations.

2.6 Technical Specifications And Other Licensing Uses

The proposed TS and TS Bases changes are consistent with the evaluation and analyses reviewed, and are required to incorporate the use of the 422V+ fuel into PBNP reload cores. These include an increase in the minimum boron concentration of the spent fuel pool from 1800 to 2000 parts per million (ppm) to be consistent with the required minimum primary coolant system boron concentration for refueling activities. Restrictions were added on fuel to be stored in the new fuel vault, resulting from new analyses performed by Westinghouse. The marked-up TS pages reflecting the proposed changes were provided in Attachment 6 and are described and justified in Attachment 1 of the licensee's submittal of June 22, 1999 (Reference 1). Based on the staff review, these changes are acceptable because Westinghouse used more conservative assumptions and the proposed TS conditions are more restrictive than the current TS.

The NRC safety evaluation report (SER) (Reference 14) for the Westinghouse generic licensing report on VANTAGE+ fuel, WCAP-12610 P-A (Reference 2), approves the licensing topical report for leading rod-average burnup levels up to 60,000 MWD/MTU. The staff concludes that this limit for the 422V+ fuel is acceptable for PBNP reload cores and that the licensee has appropriately evaluated the limiting fuel and core conditions for licensing analyses. The licensee has shown that WCAP-12610 P-A is bounding for the PBNP. Accordingly, the proposed amendment is acceptable in this respect.

As a result, the staff has concluded that the methodologies, processes, codes, evaluations, and analyses discussed in the licensee's application dated June 22, 1999, as supplemented December 17, 1999, are acceptable. Therefore, the staff concludes that the design changes for use of the 422V+ fuel are acceptable for inclusion in licensing documentation, including the PBNP Updated FSAR and TSs.

2.7 Best Estimate (BE) Large-Break LOCA (LBLOCA) and Small-Break LOCA (SBLOCA) Analyses

2.7.1 Best Estimate Large-Break LOCA Analysis

In their June 22, 1999, application (Attachment 2, Section 9) the licensee discusses LBLOCA reanalyses to reflect the presence of ZIRLO fuel in the PBNP core, the adaptation of the PBNP BE LOCA model used to perform the reanalyses, the process implemented to determine the adaptation, and the LBLOCA results. The licensee used the Westinghouse 2-Loop Upper Plenum Injection (UPI) version (discussed in WCAP-14449 P-A, dated October 1999), of the NRC-approved Westinghouse best estimate LBLOCA analysis methodology to perform the PBNP LBLOCA analyses discussed in its submittal. PBNP is a UPI plant. The staff concludes that the UPI version of the methodology may be applied to PBNP because PBNP is of the type of plant for which the approved methodology was developed.

2.7.2 Small-Break LOCA Analysis

In their June 22, 1999, application (Attachment 2, Section 14.3.1), the licensee discusses PBNP SBLOCA. The licensee performed PBNP SBLOCA analyses using the latest version of the Westinghouse NOTRUMP SBLOCA analysis methodology described in WCAP-10054 P-A, addendum 2, Revision 1, July 1997. This methodology was approved for application to all current conventional Westinghouse plant designs, including UPI plants. We conclude that this version of the NOTRUMP SBLOCA methodology may be applied to PBNP because PBNP is of the type of plant for which the approved methodology was developed.

2.7.3 LOCA Analysis Input Values

In its December 17, 1999, supplemental letter, the licensee stated that Wisconsin Electric and Westinghouse have ongoing processes which assure that the analysis input values for peak-cladding-temperature-sensitive parameters conservatively bound the as-operated plant values for those parameters. With this provision by the licensee, we conclude that the Westinghouse LBLOCA and SBLOCA analyses using the methodologies described in WCAP-14449 P-A and WCAP-10054, Revision 2, addendum 1, apply to PBNP, Units 1 and 2.

2.7.4 ZIRLO Fuel

The NRC SER generic licensing report for VANTAGE+ (ZIRLO) fuel, WCAP-12610 P-A, states that there is sufficient similarity between ZIRLO fuel and co-resident Zircaloy fuel such that when they have like features (geometry), no mixed core penalty need be applied to either fuel in LOCA analyses. However, in this PBNP case, the co-resident fuel and ZIRLO do not have like geometry. Therefore, the licensee performed sensitivity studies identifying that the ZIRLO-clad fuel was limiting for both LBLOCA and SBLOCA analyses. We conclude that the licensee has appropriately identified ZIRLO as the limiting fuel for PBNP licensing analyses.

2.7.5 PBNP LOCA Analyses

The sections of the licensee's June 22, 1999, application discussed above also describe LBLOCA and SBLOCA licensing analyses performed by the licensee to establish new analyses of record for licensing uses, such as showing compliance with the requirements of 10 CFR 50.46, establishing PBNP TSs and surveillance requirements, and for reference in complying with the reporting requirements of the governing regulations.

2.7.6 Technical Specifications And Other Licensing Uses

The NRC staff concludes that the LOCA methodologies and processes discussed in the licensee's application dated June 22, 1999, as supplemented December 17, 1999, are acceptable. In addition, the staff concludes that design changes for use of the 422V+ fuel are acceptable for inclusion in licensing documentation, including PBNP Updated FSAR, TSs, and core operation limits report.

2.7.7 Other Changes

The licensee is also removing footnotes to TS Section 15.2.1.1 that no longer apply at PBNP and is renumbering TS Section 15.2.3.1 to 15.2.3.1.C as the "C" was inadvertently deleted in the past and is being recaptured to properly identify this section of the TS. The conditions to which these TSs applied no longer exist at PBNP or are editorial in nature, and, therefore, are acceptable.

2.8 Staff Conclusion

The staff concludes that the methodologies and processes discussed in Section 2 of this safety evaluation are acceptable for licensing use at PBNP, they will continue to be acceptable for that use as long as they are not changed, and are acceptable for inclusion in licensing documentation including the PBNP Updated FSAR, TSs, and core operation limits report.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 40910). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

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

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Date: February 8, 2000

6.0 REFERENCES

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9) "Standard Review Plan - 4.2 - Fuel System Design," NUREG-0800, Revision 2, July 1981.

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14) Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report'," July 1, 1991.