MAY 23 1974

Docket No. 50-266

Wisconsin Michigan and Wisconsin
Electric Power Company
ATTN: Mr. Sol Burstein
Senior Vice President
213 West Michigan Street
Milwaukee, Wisconsin 53201

Gentlemen:

By letter dated May 1, 1974, you proposed a change to the Technical Specifications of Facility Operating License No. DPR-24 for the Point Beach Nuclear Unit No. 1. The proposed change provides requirements for Cycle 3 operation of Unit No. 1. We have, as discussed with your staff, modified your proposed change to meet Regulatory requirements.

Based on our evaluation of the proposed change as modified, we have concluded that it does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of License No. DPR-24 are changed as shown in Amendment No. 3.

We note you may make application to operate Cycle 3 in excess of the initially authorized period. In this respect, we wish to advise you that you should provide your analysis at least 90 days prior to your need for approval to allow sufficient time to schedule and accomplish our review.

Sincerely,

Original signed by
Dennis L. Ziemann for

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Called Licensee 5/23/74
4:42 PM that Approval given

Enclosures:
1. Amendment No. 3
2. Federal Register Notice
3. Safety Evaluation

OFFICE cc: on next page

DPPk

MAY 23 1974

cc w/enclosures:
Mr. Bruce W. Churchill, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th ~~Street~~, N. W.
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Environmental Protection Agency
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Chicago, Illinois 60606

Manitowoc Public Library
808 Hamilton Street
Manitowoc, Wisconsin 54220

Mr. William F. Eich, Chairman
Public Service Commission
of Wisconsin
Hill Farms State Office Building
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WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-24

1. The Atomic Energy Commission ("the Commission") has found that:

   A. The application for amendment by Wisconsin Michigan and Wisconsin Electric Power Company ("the licensee") dated May 1, 1974, 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 license, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, paragraph 3.B of Facility License No. DPR-24 is hereby amended to read as follows:

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SURNAME ➤						
DATE ➤						

"B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B attached to Facility Operating License No. DPR-24 are revised as indicated in the attachment to this license amendment. The Technical Specifications, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

3. This license amendment is effective as of the date of its issuance.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann for

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 8 to Appendix A
Technical Specifications

Date of Issuance: MAY 23 1974

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 3

CHANGE NO. 8 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

Delete pages 15.2.1-1, 15.2.1-3, 15.2.3-2, 15.2.3-3, 15.2.3-6, 15.2.3-7 15.3.10-1, 15.3.10-2, 15.3.10-5, 15.3.10-6, 15.3.10-7, 15.3.10-8, 15.3.10-10, and 15.3.10-11 and replace with the attached revised pages. Also delete Figure 15.2.1-1 and Figure 15.3.10-1 and replace with the attached revised figures.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-266

WISCONSIN ELECTRIC AND WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF FACILITY LICENSE AMENDMENT

Notice is hereby given that the U. S. Atomic Energy Commission ("the Commission") has issued Amendment No. 3 to Facility Operating License No. DPR-24 issued to Wisconsin Electric and Wisconsin Michigan Power Company which revised Technical Specifications for operation of the Point Beach Nuclear Plant Unit No. 1, located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendment is effective as of its date of issuance.

The amendment permits changes to the Technical Specifications to permit Cycle 3 operation at a reduced system pressure.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act, as amended ("the Act"), and the Commission's rules and regulations and the Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendment.

For further details with respect to this action, see (1) the application for amendment dated May 1, 1974, (2) Amendment No. 3 to License No. DPR-24 and Change No. 8, and (3) the Commission's related Safety

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DATE ➤						

Evaluation. All of these are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Manitowoc Public Library, 808 Hamilton Street, Manitowoc, Wisconsin.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

r Dated at Bethesda, Maryland, this MAY 23 1974

FOR THE ATOMIC ENERGY COMMISSION

/s/

Robert A. Purple, Chief
Operating Reactors Branch #1
Directorate of Licensing

OFFICE ➤						
SURNAME ➤						
DATE ➤						

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. DPR-24

(CHANGE NO. 8 TO APPENDIX A OF TECHNICAL SPECIFICATIONS)

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR UNIT NO. 1

DOCKET NO. 50-266

Introduction

By letter dated May 1, 1974, Wisconsin Michigan and Wisconsin Electric Power Company proposed a change to the Technical Specifications of Facility Operating License No. DPR-24 to provide specifications applicable to Cycle 3 operation of Unit No. 1 and provide for reducing the primary system pressure to 200 psia to reduce the potential for fuel rod flattening.

On the basis of our review, we have determined that areas requiring assessment were reduced pressure operation and Cycle 3 exposure.

Evaluation

1. Reduced Pressure Operation

   The technical justification for operation of Unit No. 1 at a reduced pressure of 2000 psia is based on the analysis provided in support of low pressure operation of Unit No. 3 Cycle 1 (WCAP-8150). The low pressurizer pressure, overtemperature T, and overpower T setpoints proposed for Unit No. 1 Cycle 3 are the same for Unit No. 2 Cycle 1, with the exception of the inputs to overtemperature and overpower T setpoints which are based on flux difference between the top half and bottom half of the core (axial flux distribution).

   The axial distribution is dependent on burnup history in the core. The curve in Figure 1 in the licensee's submittal provides the upper bounds to the peak local power distribution as a function of axial

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offset for Cycle 3 operation of Unit No. 1. These results, as calculated by Westinghouse, are based on the fuel loading for Cycle 3. From our evaluation of these proposed axial flux limits, we conclude that axial flux distribution is conservatively considered in establishing overtemperature and overpower Δ T setpoints and operating limits for Cycle 3.

Therefore, with the above exception which we have determined to be acceptable, the proposed technical specifications for Cycle 3 of Unit No. 1 that are important to operation at 2000 psia are identical to the specifications for Unit No. 2 Cycle 1 operation at 2000 psia. The Commission has found these specifications acceptable for 2000 psi operation of Unit No. 2 (Change No. 8 dated December 4, 1973) and also finds these specifications acceptable for Unit No. 1 Cycle 3 because the units are identical with respect to thermal and hydraulic considerations and nuclear core safety evaluation parameters.

The licensee proposed a limit on insertion of part-length rods to be based on reactor power level. Per our discussion with the licensee, the proposed insertion limit has been modified to be identical to the limit now in effect for Unit No. 2 Cycle 1. The Unit No. 2 limit previously evaluated and accepted by the Commission is more conservative, is applicable, and is also found acceptable for operation of Unit No. 1 Cycle 3.

Accidents have been evaluated for Unit No. 1 Cycle 3 by the licensee. The consequences of these accidents are no greater than those previously reviewed and accepted by the Commission.

2. Cycle 3 Exposure

Westinghouse Report WCAP-8050, "Fuel Densification Point Beach Nuclear Plant 1 - Cycle 2," provides analysis to support licensee's statement that no clad collapse would occur during 5000 EFPH of Cycle 3 operation with primary system pressure of 2250 psia. Primary system pressure will be reduced to 2000 psia for Cycle 3 and the licensee concludes that this reduction in pressure will extend the time to collapse for the most limiting assemblies to 6000 EFPH. Based on WCAP-8050, only 3 assemblies (Region 4B) have a potential for collapse at 5000 EFPH at 2250 psi and we have therefore concluded, based on licensee's calculation presentation for Cycle 3, that the reduced pressure would reduce the potential for collapse and allow operation to 6000 EFPH with no collapse in Region 4B. In addition, since all fuel is prepressurized, 77 out of the total 121 assemblies are new and no region other than Region 4B is predicted to collapse at 6000 EFPH even at 2250 psia, we have concluded that no collapse will occur for the proposed operation of Unit No. 1 Cycle 3.

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Conclusion

We have concluded that the proposed change, as modified, does not involve a significant hazards consideration because it does not involve a safety consideration of a type or magnitude not previously considered, it does not potentially increase the probability or consequences of an accident previously considered, and does not potentially decrease the margins of safety during normal plant operation, anticipated operational occurrences, or postulated accidents previously considered. We also conclude that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner proposed.

/s/

Peter B. Erickson
Operating Reactors Branch #1
Directorate of Licensing

/s/

Robert A. Purple, Chief
Operating Reactors Branch #1
Directorate of Licensing

Date: MAY 23 1974

OFFICE ➤						
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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 23, 1974

Docket No. 50-266

Wisconsin Michigan and Wisconsin
Electric Power Company
ATTN: Mr. Sol Burstein
Senior Vice President
213 West Michigan Street
Milwaukee, Wisconsin 53201

Gentlemen:

By letter dated May 1, 1974, you proposed a change to the Technical Specifications of Facility Operating License No. DPR-24 for the Point Beach Nuclear Unit No. 1. The proposed change provides requirements for Cycle 3 operation of Unit No. 1. We have, as discussed with your staff, modified your proposed change to meet Regulatory requirements.

Based on our evaluation of the proposed change as modified, we have concluded that it does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the manner proposed. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of License No. DPR-24 are changed as shown in Amendment No. 3.

We note you may make application to operate Cycle 3 in excess of the initially authorized period. In this respect, we wish to advise you that you should provide your analysis at least 90 days prior to your need for approval to allow sufficient time to schedule and accomplish our review.

Sincerely,

Dennis L. Ziemann

for/ Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 3
2. Federal Register Notice
3. Safety Evaluation

cc: on next page

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Shaw, Pittman, Potts, Trowbridge
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Manitowoc Public Library
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Manitowoc, Wisconsin 54220

Mr. William F. Eich, Chairman
Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-24

1. The Atomic Energy Commission ("the Commission") has found that:

   A. The application for amendment by Wisconsin Michigan and Wisconsin Electric Power Company ("the licensee") dated May 1, 1974, 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 license, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, paragraph 3.B of Facility License No. DPR-24 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendices A and B attached to Facility Operating License No. DPR-24 are revised as indicated in the attachment to this license amendment. The Technical Specifications, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

3. This license amendment is effective as of the date of its issuance.

FOR THE ATOMIC ENERGY COMMISSION

Dennis L. Ziemann

for Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 8 to Appendix A
Technical Specifications

Date of Issuance: May 23, 1974

ATTACHMENT TO LICENSE AMENDMENT NO. 3

CHANGE NO. 8 TO APPENDIX A OF TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

Delete pages 15.2.1-1, 15.2.1-3, 15.2.3-2, 15.2.3-3, 15.2.3-6, 15.2.3-7 15.3.10-1, 15.3.10-2, 15.3.10-5, 15.3.10-6, 15.3.10-7, 15.3.10-8, 15.3.10-10, and 15.3.10-11 and replace with the attached revised pages. Also delete Figure 15.2.1-1 and Figure 15.3.10-1 and replace with the attached revised figures.

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applied 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.

2. Unit 1, Cycle 3 shall be limited to 6,000 effective full power hours (EFPH) under design operating conditions, with a primary system pressure of 2000 psia.

Change No. 8
Date: 5/23/74

Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length as well as a penalty to account for rod bowing, have been included in the calculation of the curves shown in Figure 15.2.

These curves are based on an $F^N_{\Delta H}$ of 1.58, cosine axial flux shape, and a DNB analysis as described in Section 4.3 of WCAP-8050 "Fuel Densification, Point Beach Nuclear Plant Unit 1 Cycle 2," (including the effects of fuel densification and flattened cladding).

Figure 15.2.1-1 also includes an allowance for an increase in the enthalpy rise hot channel factor at reduced power based on the expression:

$F^N_{\Delta H}$ = 1.58 [1 + 0.2 (1-p)] where P is a fraction of rated power when $P \leq 1.0$. $F^N_{\Delta H}$ = 1.58 when $P > 1.0$.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of full-length rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than 1.30.

The fuel residence time for Unit 1, Cycle 3 is limited to 6,000 EFPH to assure no clad flattening without prior review by the Regulatory Staff. The residence time of 6,000 EFPH is based on predicted minimum time to clad flattening for an operating pressure of 2,000 psi. Beyond a residence time of 6,000 EFPH for cycle 3, an assumption of clad flattening is presently required. Prior to 6,000 EFPH, the licensee may provide the additional analyses required for operation beyond 6,000 EFPH.

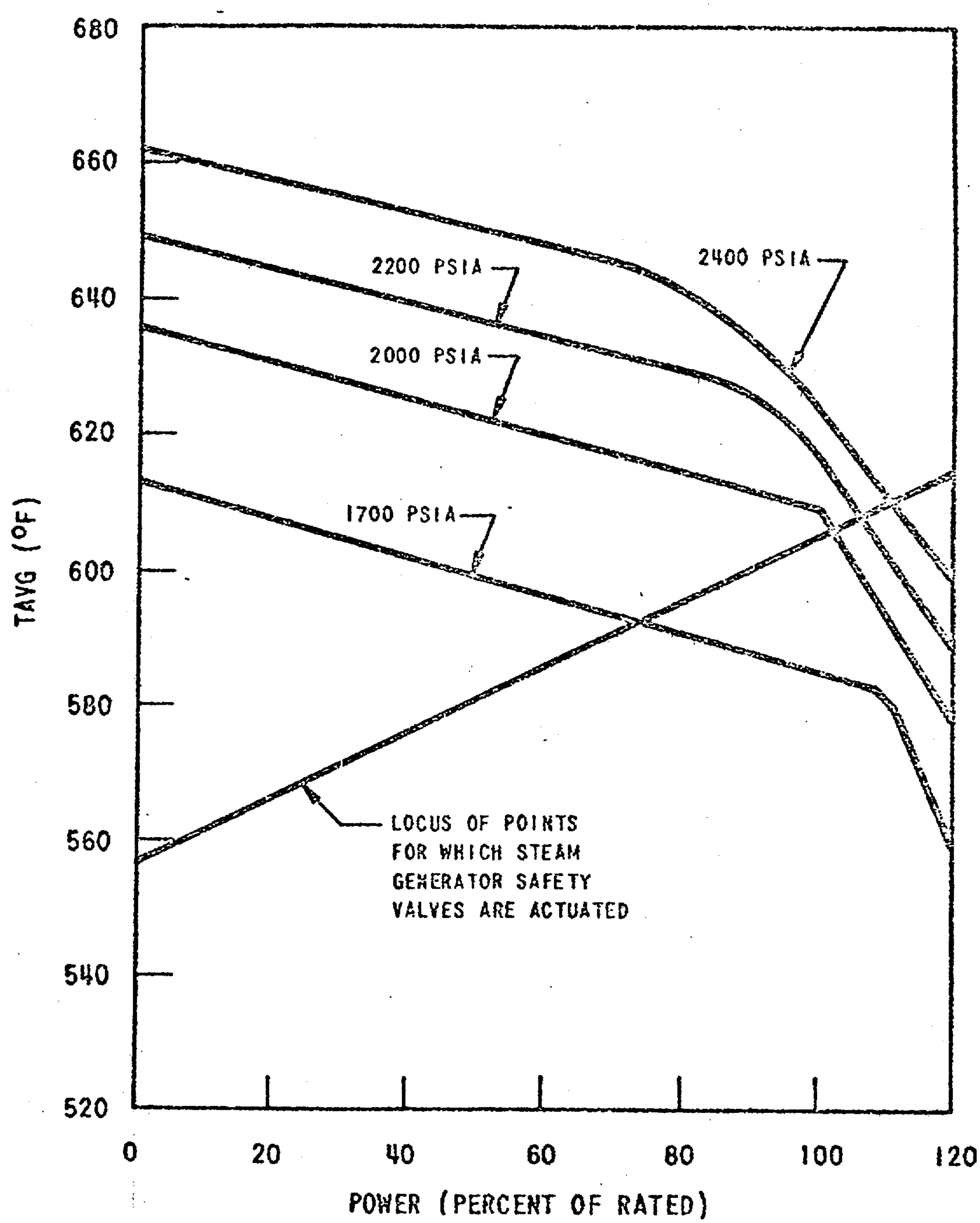


Figure 15.2.1-1 Core DNB Safety Limits
Point Beach 1 - Cycle 3

Change No. 8
Date: 5/23/74

(3) Low pressurizer pressure - $\geq$ 1710 psig.

(4) Overtemperature ΔT

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

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	=	572.9 °F
P	=	pressurizer pressure, psig
P'	=	1985 psig
K_1	$\leq$	1.11
K_2	=	0.0158
K_3	=	0.000852
τ_1	=	25 sec
τ_2	=	3 sec

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

(a) for $q_t - q_b$ within -17, +9 percent, $f(\Delta I) = 0$.

(b) for each percent that the magnitude of $q_t - q_b$ exceed +9 percent the ΔT trip set point shall be automatically reduced by an equivalent of four 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 2.5 percent of rated power.

[1.B. (5)] Overpower ΔT

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

where

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	=	572.9°F
K_4	$\leq$	1.08 of rated power
K_5	=	0.0262 for increasing T
	=	0.0 for decreasing T
K_6	=	0.0012 for $T \geq T'$
	=	0.0 for $T < T$
τ_3	=	10 sec
		f (ΔI) as defined in (4) above,

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

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

(8) Reactor coolant pump motor breaker open

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

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

Change No. 8
Date: 5/23/74

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 difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[6] [7]

The overpower, overtemperature and pressurizer pressure system setpoints have been revised to include effect of reduced system pressure operation (including the effects of fuel densification). The revised setpoints as given above will not exceed the revised core safety limits as shown in 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 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 of a sudden loss of pwoer to one or both reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis.[8] The low loop flow signal is caused by a condition of less than 90% flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency set-point, 57.5 cps, which assures a trip signal before the pump inertia is reduced to an unacceptable value.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows adequate operating instrument error [2] and transient overshoot in level before the reactor trips.

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

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their availability in the power range where needed.

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

References

(1) FASR 14.1.1

(2) FSAR, page 14-3

(3) FSAR 14.2.6

(4) FSAR 14.3.1

(5) FSAR 14.1.2

(6) FSAR 7.2, 7.3

(7) FSAR 3.2.1

(8) FSAR 14.1.9

(9) FSAR 14.1.11

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification:

A. Control Bank Insertion Limits

1. When the reactor is critical, except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

2. When the reactor is critical, the control rods shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1 and the shutdown margin with allowance for a stuck rod shall exceed the applicable value shown on Figure 15.3.10-2 under all steady state operating conditions from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn and assuming no changes in xenon, boron, or part length rod position. Exceptions to the insertion limit and stuck rod requirements only are permitted for physics tests and control rod exercises.

3. The part length rods shall not be more than 70% inserted.

4. When the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod position at which criticality would be achieved in the control rods were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

B. <u>Power Distribution Limits</u>

1. At all times the hot channel factors defined in the basis must meet the following limits:

   a. $F_Q^N \leq 2.52\ [1 + 0.2\ (1-P)]$ in the indicated flux difference

      $F_{\Delta H}^N \leq 1.58\ [1 + 0.2\ (1-P)]$

   where P is the fraction of full power at which the core is operating ($P \leq 1.0$)

   b. If peaking factors exceed the limits of Section B.1.a, the reactor power and high neutron flux trip setpoint shall be reduced by 1 percent ofr every percent excess over $F_{\Delta H}^N$ or F_Q^N, whichever is limiting. If the peaking factors cannot be corrected within 1 day, the overpower ΔT and overtemperature ΔT setpoints shall be similarly reduced.

   c. The permissible fraction of full power, P, at which the reactor can be operated up to the level of 1518.5 MWt, shall be determined by

$$P = \frac{15.6}{5.7 \times 1.02 \times 1.024 \times 1.007 \times Fq}$$

3. If it be determined that the apparent misalignment or dropped rod indication was caused by Rod Position Indicator Channel failure, sustained power operation can be continued if the following conditions are met:

   a. For operation between 10% power and rated power, the position of the rod(s) with the failed Rod Position Indicator Channel(s) will be checked indirectly by core instrumentation (excore detectors, and/or thermocouples, and/or moveable incore detectors) every shift or after associated bank motion exceeding 24 steps, whichever comes sooner.

   b. For operation below 10% of rated power, no special monitoring is required.

E. Rod Drop Times

1. At operating temperature and full flow, the drop time of each control rod shall be no greater than 1.8 seconds from the loss of stationary gripper coil voltage to dashpot entry.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occuring during power operation will put the reactor into the hot shutdown condition. The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time and assume the highest worth control rod remains fully withdrawn. The rods are withdrawn in the sequence of A, B, C, D with overlap between banks and a 10% margin in reactivity worth of the control rods to assure meeting the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection, and provide for acceptable nuclear peaking factors. The solid lines shown on Figure 15.3.10-1 meet the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is b based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 15.3.10-2 shows the shutdown margin equivalent to 2.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

The overlap between successive control banks is provided to compensate for the low differential rod worth near the top and bottom of the core. Positioning of the part-length rods is governed by the requirement to maintain the axial power shape within specified limits or to accept an automatic cut-back of the overpower ΔT and overtemperature ΔT set points (see Specification 15.2.3).

Part length rod insertion has been limited to eliminate certain adverse power shapes.

The various control rod banks (shutdown rods, control banks A, B, C, D, and part-length rods) are each to be moved as a bank, that is, with all rods in the bank within one step (5/8-inch) of the bank position. Direct information on rod position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position. The Rod Position Indicator channel has a demonstrated accuracy of 5% of span (7.2 inches). Therefore, a 15-inch indicated misalignment of a rod from its bank is necessarily a true misalignment. Misalignment of 15 inches cannot cause design hot channel factors to be exceeded, and complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at rated power. If the misalignment condition cannot be readily corrected, the specified reduction in power to 75% will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The 8-hour permissible limit on rod misalignment at rated power is short with respect to the probability of an independent accident. The failure of an LVDT in itself does not reduce the shutdown capability of the rods, but it does reduce the operator's capability for determining the position of that rod by direct means. The operator has available to him the core detector recordings, incore thermocouple readings and periodic incore flux traces for indirectly determining rod position and flux tilts should the rod with the inoperable LVDT become malpositioned. The excore and incore instrumentation will not necessarily recognize a misalignment of 15 inches because the concommitant increase in power density will normally be less than 1% for a 15-inch misalignment. The excore and incore instrumentation will, however, detect any rod misalignment which is sufficient to cause a significant increase in hot channel factors and/or any significant loss in shutdown capability. The increased surveillance of the core if one or more Rod Position Indicator Channels is out of service serves to guard against any significant loss in shutdown margin or margin to core thermal limits. The history of malpositioned rods indicates that in nearly all the cases when the rods have been malpositioned, the malpositioning occurred when the bank was moving. The checking of the rod position after bank motion exceeding 24 steps will verify that the rod with the inoperable LVDT is moving properly with its bank and according to the bank step counter. Malpositioning of a rod in a bank which is not moving is very rare, and, if it does occur, it is usually gross slippage or complete rod dropping which will be seen by external detectors. Should it go undetected, the checking of the rod position every shift is short with respect to the probability of another independent undetected situation which would further reduce the shutdown capability of the rods. Any combination of misaligned rods below 10% rated power will not exceed the design limits. For this reason, the position of the rods with inoperable LVDT's need not be checked below 10% power; plus, the incore instrumentation is not effective for determining rod position until the power level is above approximately 5%.

An inoperable rod imposes additional demands on the operators, the permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden. From operating experience to date, a control rod which steps "in" properly will drop when a trip signal occurs because the only force acting to drive the rod in is gravity. When it has been determined that a rod does not drop, extra margin is gained by boration or by adjusting the insertion limit to account for the worth of the inoperable control rod.

It has been determined by analysis that the design ~~limits~~ on peak local power density on minimum DNBR at full power and LOCA are met, provided:

$$F_Q^N \leq 2.52 \text{ and } F_{\Delta H}^N \leq 1.58$$

These quantities are measurable although there is not normally a requirement to do so. Instead it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met at full power; these conditions are as follows.

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.
2. Control rod banks are sequenced with overlapping banks as shown in Figure 15.3.10-1.
3. The control bank insertion limits and part-length rod insertion limits are not violated.
4. Axial power distribution guide lines, which are given in terms of flux difference control 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 power between the top and bottom halves of the core. Calculation of core peaking factors under a variety of operating conditions have been correlated with axial offset. The correlation shows that an F_Q^N of 2.52 and allowed DNB shapes, including the effects of fuel densification, are not exceeded if the axial offset (flux difference) is maintained between -20 and +12%. The specified limits of -17 and +9% allow for a 3% error in

the axial offset. In order to take credit for operation at the bounding value of the correlation in the permitted range of the axial offset, surveillance of the axial peaking factor, F_Z, is specified. Otherwise the specification leads to a 4% penalty in power.

For operation at a fraction, P, of full power the design limits are met, provided, 4

$$F_Q^N \leq 2.52 \; [1 + 0.2(1-P)] \text{ in the indicated flux difference range of } +9 \text{ to } -17$$

and $F_{\Delta H}^N \leq 1.58 \; [1 + 0.2 \; (1-P)]$

The permitted relaxation allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

For normal operation and anticipated transients the core is protected from exceeding 18.1 kw/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the flux difference is an explicit input to the protection system.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditons require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_Q^N there is a 5% allowance for uncertainties[1] which means that normal operation of the core within the defined conditions 4
and procedures is expected to result in $F_Q^N \leq$ 2.52/1.05 even on a worst

Change No. 8
Date: 5/23/74

FIGURE 15.3.10-3

POWER SPIKE FACTOR VERSUS CORE HEIGHT
POINT BEACH UNIT 1 - CYCLE 3

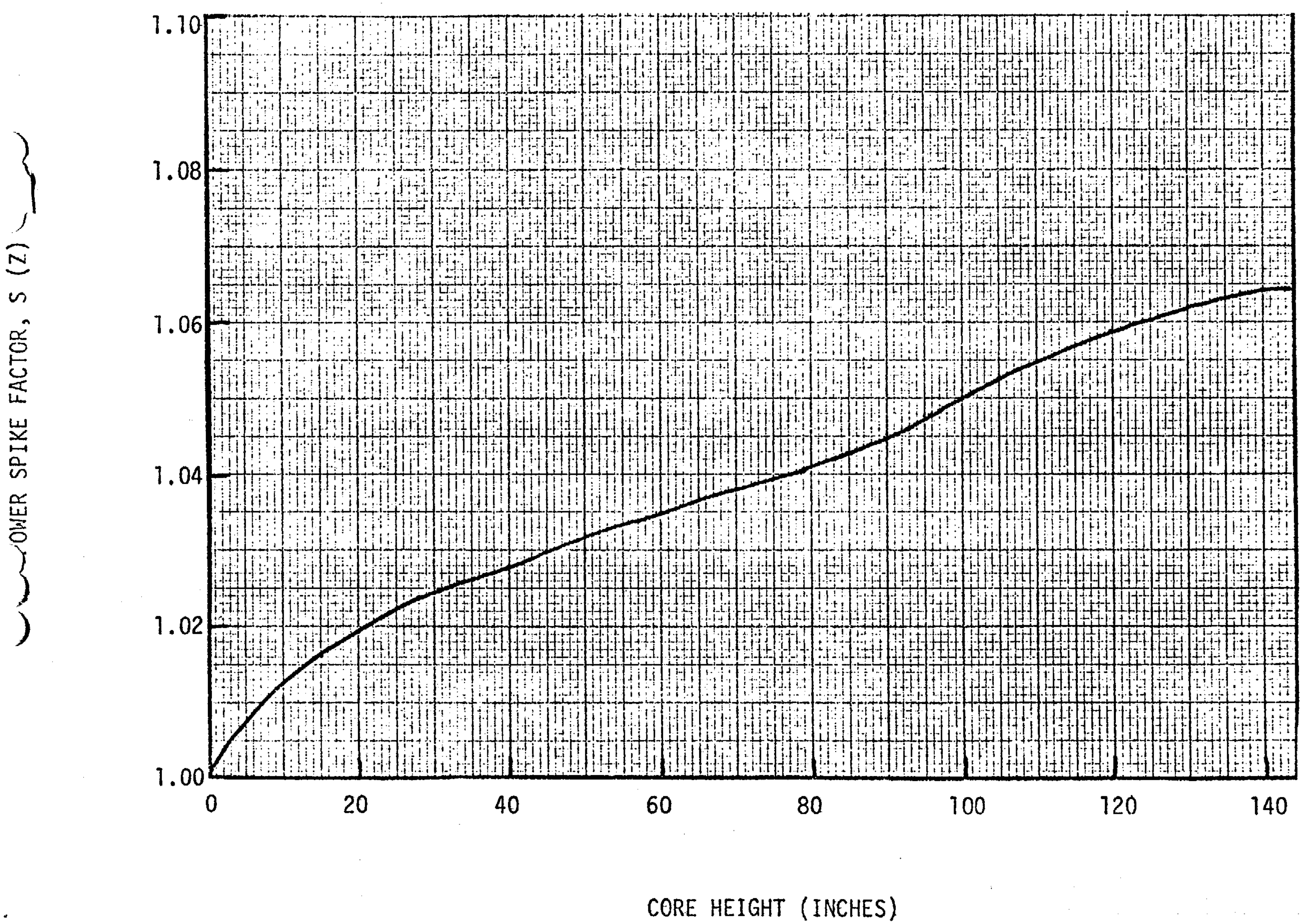
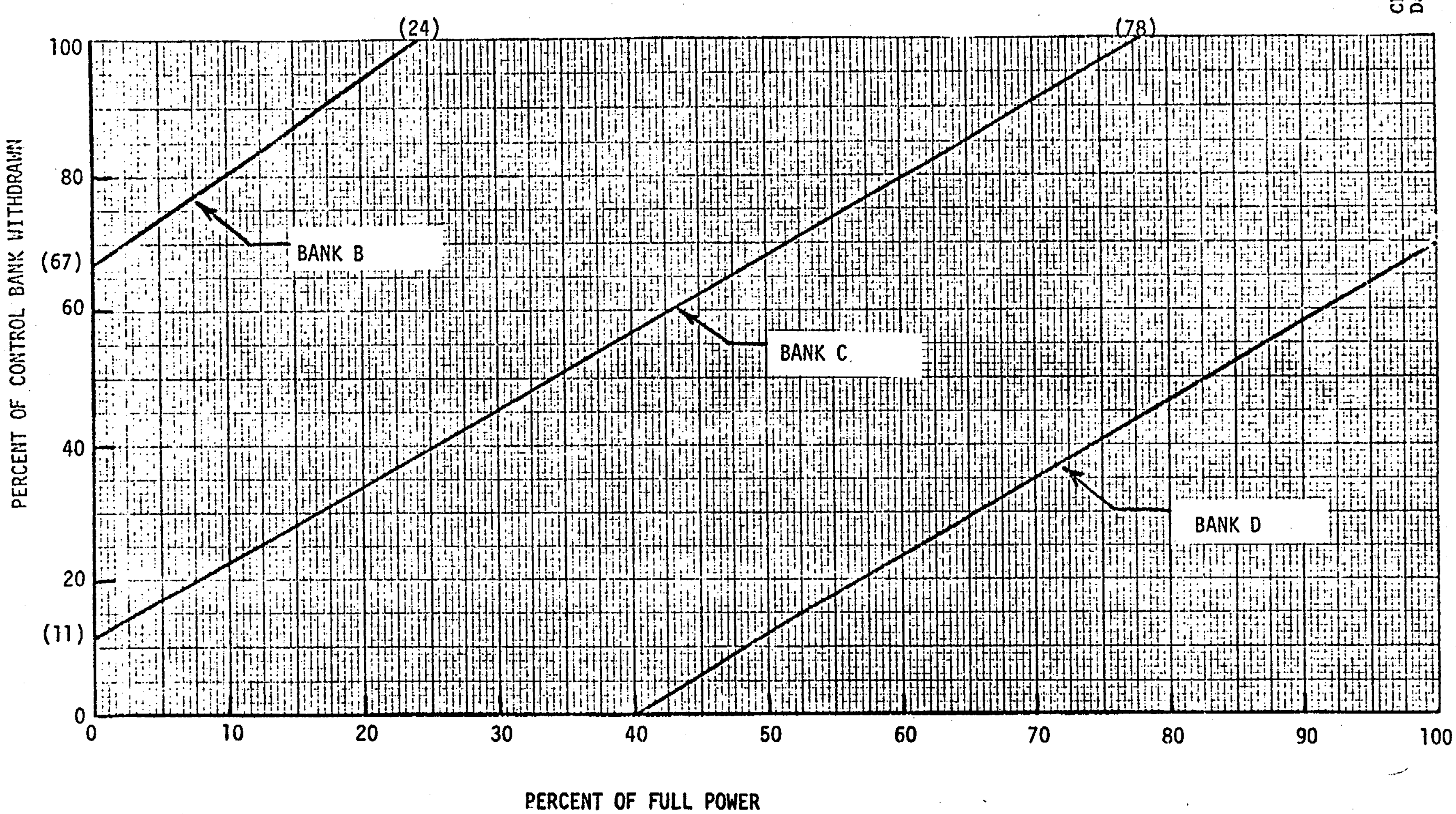


FIGURE 15.3.10-1

FULL LENGTH ROD INSERTION LIMITS
POINT BEACH UNIT 1 - CYCLE 3



UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-266

WISCONSIN ELECTRIC AND WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF FACILITY LICENSE AMENDMENT

Notice is hereby given that the U. S. Atomic Energy Commission ("the Commission") has issued Amendment No. 3 to Facility Operating License No. DPR-24 issued to Wisconsin Electric and Wisconsin Michigan Power Company which revised Technical Specifications for operation of the Point Beach Nuclear Plant Unit No. 1, located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendment is effective as of its date of issuance.

The amendment permits changes to the Technical Specifications to permit Cycle 3 operation at a reduced system pressure.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act, as amended ("the Act"), and the Commission's rules and regulations and the Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendment.

For further details with respect to this action, see (1) the application for amendment dated May 1, 1974, (2) Amendment No. 3 to License No. DPR-24 and Change No. 8, and (3) the Commission's related Safety

Evaluation. All of these are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Manitowoc Public Library, 808 Hamilton Street, Manitowoc, Wisconsin.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this 23rd day of May 1974.

FOR THE ATOMIC ENERGY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Directorate of Licensing

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. DPR-24

(CHANGE NO. 8 TO APPENDIX A OF TECHNICAL SPECIFICATIONS)

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR UNIT NO. 1

DOCKET NO. 50-266

Introduction

By letter dated May 1, 1974, Wisconsin Michigan and Wisconsin Electric Power Company proposed a change to the Technical Specifications of Facility Operating License No. DPR-24 to provide specifications applicable to Cycle 3 operation of Unit No. 1 and provide for reducing the primary system pressure to 200 psia to reduce the potential for fuel rod flattening.

On the basis of our review, we have determined that areas requiring assessment were reduced pressure operation and Cycle 3 exposure.

Evaluation

1. Reduced Pressure Operation

The technical justification for operation of Unit No. 1 at a reduced pressure of 2000 psia is based on the analysis provided in support of low pressure operation of Unit No. 2 Cycle 1 (WCAP-8150). The low pressurizer pressure, overtemperature Δ T, and overpower Δ T setpoints proposed for Unit No. 1 Cycle 3 are the same for Unit No. 2 Cycle 1, with the exception of the inputs to overtemperature and overpower T setpoints which are based on flux difference between the top half and bottom half of the core (axial flux distribution).

The axial distribution is dependent on burnup history in the core. The curve in Figure 1 in the licensee's submittal provides the upper bounds to the peak local power distribution as a function of axial offset for Cycle 3 operation of Unit No. 1. These results, as calculated by Westinghouse, are based on the fuel loading for Cycle 3. From our evaluation of these proposed axial flux limits, we conclude that axial flux distribution is conservatively considered in establishing overtemperature and overpower ΔT setpoints and operating limits for Cycle 3.

Therefore, with the above exception which we have determined to be acceptable, the proposed technical specifications for Cycle 3 of Unit No. 1 that are important to operation at 2000 psia are identical to the specifications for Unit No. 2 Cycle 1 operation at 2000 psia. The Commission has found these specifications acceptable for 2000 psi operation of Unit No. 2 (Change No. 8 dated December 4, 1973) and also finds these specifications acceptable for Unit No. 1 Cycle 3 because the units are identical with respect to thermal and hydraulic considerations and nuclear core safety evaluation parameters.

The licensee proposed a limit on insertion of part length rods to be based on reactor power level. Per our discussion with the licensee, the proposed insertion limit has been modified to be identical to the limit now in effect for Unit No. 2 Cycle 1. The Unit No. 2 limit previously evaluated and accepted by the Commission is more conservative, is applicable, and is also found acceptable for operation of Unit No. 1 Cycle 3.

Accidents have been evaluated for Unit No. 1 Cycle 3 by the licensee. The consequences of these accidents are no greater than those previously reviewed and accepted by the Commission.

2. Cycle 3 Exposure

Westinghouse Report WCAP-8050, "Fuel Densification Point Beach Nuclear Plant 1 - Cycle 2," provides analysis to support licensee's statement that no clad collapse would occur during 5000 EFPH of Cycle 3 operation with primary system pressure of 2250 psia. Primary system pressure will be reduced to 2000 psia for Cycle 3 and the licensee concludes that this reduction in pressure will extend the time to collapse for the most limiting assemblies to 6000 EFPH. Based on WCAP-8050, only 3 assemblies (Region 4B) have a potential for collapse at 5000 EFPH at 2250 psi and we have therefore concluded, based on licensee's calculation presentation for Cycle 3, that the reduced pressure would reduce the potential for collapse and allow operation to 6000 EFPH with no collapse in Region 4B. In addition, since all fuel is prepressurized, 77 out of the total 121 assemblies are new and no region other than Region 4B is predicted to collapse at 6000 EFPH even at 2250 psia, we have concluded that no collapse will occur for the proposed operation of Unit No. 1 Cycle 3.

Conclusion

We have concluded that the proposed change, as modified, does not involve a significant hazards consideration because it does not involve a safety consideration of a type or magnitude not previously considered, it does not potentially increase the probability or consequences of an accident previously considered, and does not potentially decrease the margins of safety during normal plant operation, anticipated operational occurrences, or postulated accidents previously considered. We also conclude that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner proposed.

Peter B. Erickson

Peter B. Erickson
Operating Reactors Branch #1
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Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Directorate of Licensing

Date: May 23, 1974