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MAR 2 2 1976

Docket No. 50-301

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

Gentlemen:

In response to your requests dated December 22, 1975, January 29, 1976 and March 5, 1976, the Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit 2.

The amendment consists of changes that will revise the Technical Specifications to modify the fuel residence time limit and allow an increase in normal operating reactor coolant system pressure. The core power distribution limits would be modified and new operating limits established for parameters related to Departure from Nucleate Boiling (DNB) to allow operation of Unit 2 in core Cycle 3.

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 21
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

| | | | | | | |
|---|---|---|---|---|---|---|
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| DATE ➤ | 3/17/76 | 3/17/76 | 3/18/76 | 3/22/76 | 3/22/76 | 3/18/76 |

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 21 TO LICENSE NO. DPR-27

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH NUCLEAR PLANT UNIT 2

DOCKET NO. 50-301

## Introduction

By letter dated December 22, 1975, Wisconsin Electric Power Company (WEPCO) proposed changes to the Technical Specifications of Facility Operating License DPR-27 for Point Beach Unit 2. Supplemental information supporting the requested changes was supplied by WEPCO by letters dated January 29 and March 5, 1976. The proposed changes, as modified by the staff and concurred in by the licensee, would allow operation of Unit 2 in core Cycle 3 by: (1) modifying the fuel residence time limit, (2) increasing the normal operating reactor coolant system pressure, (3) altering the core power distribution limits, and (4) adding new operating limits on DNB related parameters.

## Discussion

The Point Beach Unit 2 core Cycle 3 reloading will consist of replacing 33 Region 2 and 3 Region 3 assemblies with 36 Region 5 assemblies. The mechanical, thermal-hydraulic and chemical design of the new Region 5 assemblies is essentially the same as the Regions 3 and 4 fuel which will remain in the core during Cycle 3. WEPCO states that core Cycle 3 will be operated at an increased reactor coolant system pressure of 2250 psia (the previous cycle was operated at 2000 psia) to preclude rod internal pressure exceeding system pressure for Region 4 fuel.

The principle effect of the increased pressure on other fuel is that some Region 3 fuel could experience clad flattening before the end of core Cycle 3. It is the staff's position that fuel elements that could experience clad flattening must be operated such that a peak clad temperature (PCT) of 1800°F vs 2200°F as specified in 10 CFR Part 50, Section 50.46(b)(1)) would not be exceeded in the event of a loss of coolant accident (LOCA). Consequently, WEPCO has reanalyzed the worst case LOCA for Point Beach Unit 2, and has proposed changes to the core power distribution limits to ensure that Region 3 fuel PCT would not exceed 1800°F. Our evaluation of this and other changes related to operation of Unit 2 during core Cycle 3 follows.

Evaluation

1. Accident Analyses

The LOCA analyses for Point Beach, Unit 2 were performed and reported in a previous safety analysis (Reference 2). These analyses were reviewed by the NRC staff and found to be in conformance with the requirements of 10 CFR Part 50, Section 50.46. These analyses considered rod bowing effects, however, they did not include clad flattening which is predicted to occur in Region 3 fuel rods for burnups higher than 7190 MWD/MTU during core Cycle 3. The licensee has included consideration of clad flattening in the LOCA analyses for this cycle by using a peak clad temperature of 1800°F. This limit is met by restricting the total peaking factor for flattened rods to $F^T_Q = 2.25$. Based on our review, we find this to be acceptable. In addition, the licensee has performed power distribution calculations to demonstrate that the constant axial offset control procedures employed at Unit 2 (+6, -9 $\Delta I$ control band) are adequate to ensure that the total peaking factors can be maintained below the LOCA limit discussed above. These calculations consisted of simulations of power maneuvers under a variety of conditions, including extremes in offset maintained, power swings considered and the time of life assumed. These calculations show that the power peak ($F_Q x P$) in both the Region 3 fuel and other regions of the core can be maintained below the value used in the LOCA analysis. We have reviewed and approved the methods used in these calculations (Reference 7). Moreover, we have reviewed the results of the calculations performed specifically for this reload and find that the expected peaking factors are below the limiting values; and thus, are acceptable.

The safety analyses applicable to operation during core Cycle 3 are based on previous Cycle 2 safety analyses (Reference 3) and on the analyses reported in the Final Facility Description and Safety Analysis Report (FFSDAR) (Reference 4). The proposed operation at 2250 psia is acceptable to the staff, since raising the operating pressure does not have adverse effects on the accident analyses (DNB heat flux increase with increasing system pressure).

The previous analyses were performed with a pitch reduction factor which results in a 3.2 percent margin in DNBR to allow for rod-to-rod bowing. Recently, however, it was found that this penalty is inadequate. New data on 15x15 rod bundles with burnup up to 27,000 MWD/MTU have shown that the present model (Reference 5) under-estimates the extent of rod bowing. We believe the data is also applicable to 14x14 rod bundles of the type used at Point Beach, Unit 2. The data indicates that a penalty of 3.6 percent in DNBR should be applied to the Point Beach Unit 2 design. In addition, it is the staff's position that an additional 2 percent penalty should be imposed because our review of the Westinghouse analysis of the data has not been completed. In the staff's judgement this additional 2% margin is well within expected variations in rod bow effects and provides reasonable assurance of conservative operation until the staff completes its review. Therefore, the previously mentioned 3.2 margin in DNBR, when subtracted from the 5.6 percent penalty, leaves 2.4 percent penalty which is equivalent to 1.4 percent in heat flux reduction. To achieve this reduction, WEPCO has proposed to limit Operation of Point Beach, Unit 2 core Cycle 3 to a radial peaking factor of: $F^N \Delta H = 1.55 [1 + 0.2 (1-P)]$. Based on our review we find this to be acceptable.

Most of the core parameters, determined for Cycle 3, fall within the range of values used in the previously submitted accident analyses and therefore the existing safety analyses for Cycle 1 and 2 continue to apply to Cycle 3. Nonetheless, to ensure that the DNB design basis axial shapes used in these referenced safety analyses are conservative for Cycle 3 operation with a +6, -9% $\Delta I$ control band, the licensee evaluated the DNBR using a range of axial power distributions that represent the extremes expected during both normal and abnormal operation. The DNBR's calculated using these shapes (under both loss of flow and overpower conditions) were compared to those calculated assuming the design basis axial shapes. This comparison demonstrates the conservatism of the design basis shapes and the continued applicability of DNB analyses performed using these distributions. Thus the DNB analyses for core Cycle 3 are acceptable.

All other parameters of importance to the safety analyses were calculated and compared to the values used in the reference safety analyses. This comparison indicated that the rod ejection analyses performed for the previous cycles were not conservative for Cycle 3 because, for the beginning of cycle (BOC) and hot full power (HFP) condition, the maximum ejected rod worth is greater for Cycle 3 than the corresponding value for Cycle 2. Also the minimum $\beta$eff (delayed neutron fraction) at this condition was 0.0050 for Cycle 3 as compared to 0.0007 for the previous cycle. Because of these differences the rod ejection accident was re-analyzed by the licensee using a standard Westinghouse procedure (Reference 6). The analysis was performed for beginning and end of cycle conditions, and assumed a conservatively high initial fuel average temperature. The results of the analysis indicate no fuel melting and an acceptable value of peak fuel enthalpy. Based on these results, we have concluded that the rod ejection accident analysis is acceptable.

Finally, the main steam line break accident was re-analyzed using a revised injection curve for high head safety injection. This curve is based on experimental data and includes 5 percent safety margin. The analysis indicated that for the credible break the core does not return to critical and the minimum DNBR is always greater than 1.3. These results are acceptable to the staff.

2. Startup Tests

The Cycle 3 physics startup tests for Point Beach, Unit 2 were reviewed to check that: 1) all necessary tests would be performed, and 2) the acceptance criteria are reasonable. The startup tests will check the fuel loading and verify the calculational methods used to determine power distributions, shutdown margin and control rod worths. Core flux maps at various power levels will be taken and evaluated to verify power distribution predictions. This data will also be used in establishing the excore/incore calibration. The test proposed to verify shutdown margin and control rod worths consists of determining the differential and integral rod worths for control banks D and C. Based on our review, it is our position that the physics startup test program is acceptable only if the following conditions are met: (1) the acceptance criterion for control rod group worth tests is that the measured worth of each bank is within 10% of the predicted worth for that bank, (2) if either bank D or C measured worth is outside this range, the worth of banks B&A also must be measured, and (3) if either bank B or A measured worth is outside this range, the shutdown margin must be evaluated. These requirements have been discussed with and concurred in by the licensee.

3. Technical Specifications

The licensee has proposed plant operating parameters that reflect plant operation at 2250 psia during core Cycle 3. These changes account for the flattening of fuel rods which is predicted to occur in Region 3 fuel for burnups in excess of 7190 MWD/MTU. Our evaluation of the specific Technical Specification changes follows:

(a) Fuel Residence Time (Technical Specification 15.2.1.2)

WEPCO has included a fuel residence time limit for Cycle 3 of 9000 EFPH. We find this change to be acceptable, because we have concluded that the licensee has properly evaluated the impact of this residence time on the Cycle 3 fuel.

(b) Overtemperature ΔT and Pressurizer Low Pressure Trip Setpoint

The pressurizer low pressure trip setpoint and the overtemperature ΔT settings are specified in Technical Specification 15.2.3.1.B(3) and (4) respectively. Point Beach Unit 2, has been operated in the past at a system pressure of 2250 psia and nominal average temperature of 581.3°F. As a consequence of a subsequent fuel densification review by the staff, the Point Beach, Unit 2, operating pressure was restricted during previous Cycle 2 operation to 2000 psia and nominal average temperature of 572.9°F. For Cycle 2 operation, the licensee modified the Technical Specifications, making the overtemperature ΔT trip limits more restrictive, and lowering the pressurizer low pressure trip setpoint in consideration of the effects of reduced system operating pressure. The overtemperature $\Delta T$ trip limits were made more restrictive by modifying the constants and nominal pressure setpoints in the overtemperature $\Delta T$ trip. The licensee now has proposed to operate the plant for Cycle 3 at a system pressure of 2250 psia. In this matter, the nominal system pressure was increased from 2000 to 2250 psia while all other constants and system parameters (average temperature) remained identical to the Cycle 2 values. The staff has reviewed the proposed Cycle 3 Technical Specifications and has concluded that since the overtemperature $\Delta T$ trip setpoint for Cycle 2 was more restrictive than the originally (licensed) approved value, operation as proposed for Cycle 3 will be more conservative and therefore, the proposed modification to the Technical Specification 15.2.3.1.B(4) is acceptable. In addition, the pressurizer low pressure trip setpoint, Technical Specification 15.2.3.1.B(3), has been changed back to the value (1865 psig) it was before the system operating pressure was reduced to 2000 psia. Based on previous safety evaluations of operation at 2250 psia, made by the staff, this proposed change is also acceptable.

(c) Core Power Distribution Limits

The core power distribution limits are specified in Technical Specification 15.3.10.B. As discussed earlier, the licensee has proposed an operating limit on $F^N_{\Delta H}$ of 1.55 at full power, to accommodate possible effects of rod bowing. Based on our review discussed earlier we find this change to be acceptable. In addition, we have reviewed the total peaking factors ($F_Q{}^T$), applicable to both the periods before and after fuel rod flattening in Region 3. Based on the results of the worst case LOCA that was re-analyzed by WEPCO, which shows that a peak clad temperature of 1800°F would not be exceeded, we have concluded that the proposed total peaking factors are also acceptable.

(d) Operating Limits on DNB Related Parameters

At the request of the staff, the licensee proposed a new specification (Technical Specification 15.3.1.A.4) that requires reactor coolant system average temperature (TAVG), pressurizer pressure, and core flow be maintained within the range of values assumed as initial conditions in the safety analysis. We have concluded that this specification will provide additional assurance that the validity of the safety analysis will be maintained; and thus, the proposed change is acceptable.

## Summary

The safety analyses applicable to operation of Point Beach Unit 2 during core Cycle 3 are based on: previous Cycle 2 safety analyses and those reported in the FFDSAR; additional LOCA analyses which account for flattened clad in Region 3 fuel; and additional analyses of rod ejection accidents. The proposed operation at 2250 psia is acceptable to the staff because raising the operating pressure will have no adverse effects on the accident analyses. In addition, the proposed changes in the Technical Specifications, as modified by the staff and concurred in by the licensee, will provide assurance that core Cycle 3 will be operated within the design basis; and therefore, the proposed changes in the Technical Specifications are acceptable.

## Environmental Finding

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

## Conclusion

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

Dated: March 22, 1976

## REFERENCES


1. Letters to B. C. Rusche (NRC) from S. Burstein (WEPCO), dated: December 22, 1975, January 29, 1976 and March 5, 1976.
2. Letters to B. C. Rusche (NRC) from S. Burstein (WEPCO), dated June 24, 1975, November 5, 1975 and November 26, 1975.
3. Letter to E. Case (AEC) from S. Burstein (WEPCO), dated: August 30, 1974.
4. Final Facility Description and Safety Analysis Report Point Beach Nuclear Power Plant, Units 1 and 2.
5. Sheppard, K. D., Cerni S., Reavis, J. R., Au Evaluation of Fuel Rod Bowing, WCAP-8346, May 1974.
6. Risher, D. H. Jr., An evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactor Using Spatial Kinetics Method, WCAP 7588, Revisional, December 1971.
7. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plant, NUREG -75/087, September 1975. (Branch Technical Position CTBP4.3-1).

Wisconsin Michigan Power Company
Wisconsin Electric Power Company

MAR 2 2 1976

cc:

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. DPR-27

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

   A. The applications for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated December 22, 1975, January 29, 1976 and March 5, 1976, 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Date of Issuance: March 22, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 21

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NO. 50-301

Replace pages 15.2.1-1 through 15.2.1-3, 15.2.3-2, Figure 15.2.1-1, 15.3.1-2, 15.3.1-3, 15.3.10-2, 15.3.10-2a and 15.3.10-13 with the attached revised pages.

## 15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 15.2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

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

Objective:

To maintain the integrity of the fuel cladding.

Specification:

1. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 15.2.1-1. The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.
2. The fuel residence time of cycle 3 for Unit No. 2 shall be limited to 9,000 effective full power hours (EFPH) under design operating conditions, with a primary system pressure of 2,250 psia.

Basis:

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot regions of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is vary large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation 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, 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 minimum value of the DNB ratio, DNBR, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNB ratio of 1.30 corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 15.2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNB ratio is no less than 1.30. The area of safe operation is below these lines. The safety limits curves have been revised to allow for heat flux peaking effects due to fuel densification and flattened fuel cladding sections.

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 $P > 1.0$.

An additional rod bow penalty is applied for the Point Beach Unit 2 core Cycle 3 to limit the radial peaking factor $F_{\Delta H}$, to a more conservative value of 1.55 instead of 1.58. This additional penalty is based on new data (plus appropriate conservatisms) which shows that the bowing model in WCAP-8386, "An Evaluation of Fuel Rod Bowing" underestimates the extent of fuel rod bowing.

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 Cycle 3 is limited to 9,000 EFPH. The residence time of 9,000 EFPH is based on predicted cycle length with some allowance for stretch. Cycle 3 will operate up to 5,446 EFPH with no clad collapse. Beyond 5,446 EFPH an assumption of clad flattening is presently required for Region 3. An allowance in specification 15.3.10.B.1.a has been included for the assumed clad flattening in Region 3 beyond 5,446 EFPH.

(3) Low pressurizer pressure - $\geq$ 1865 psig.

(4) Overtemperature $\Delta T$

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

where

$\Delta T_o$ = indicated $\Delta T$ at rated power, °F

$T$ = average temperature, °F

$T'$ = 572.9°F

$P$ = pressurizer pressure, psig

$P'$ = 2235 psig

$K_1$ = 1.11

$K_2$ = 0.0158

$K_3$ = 0.000852

$\tau_1$ = 25 sec

$\tau_2$ 3 sec

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrumented 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 -12, + 5 percent, $f(\Delta I) = 0$.

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

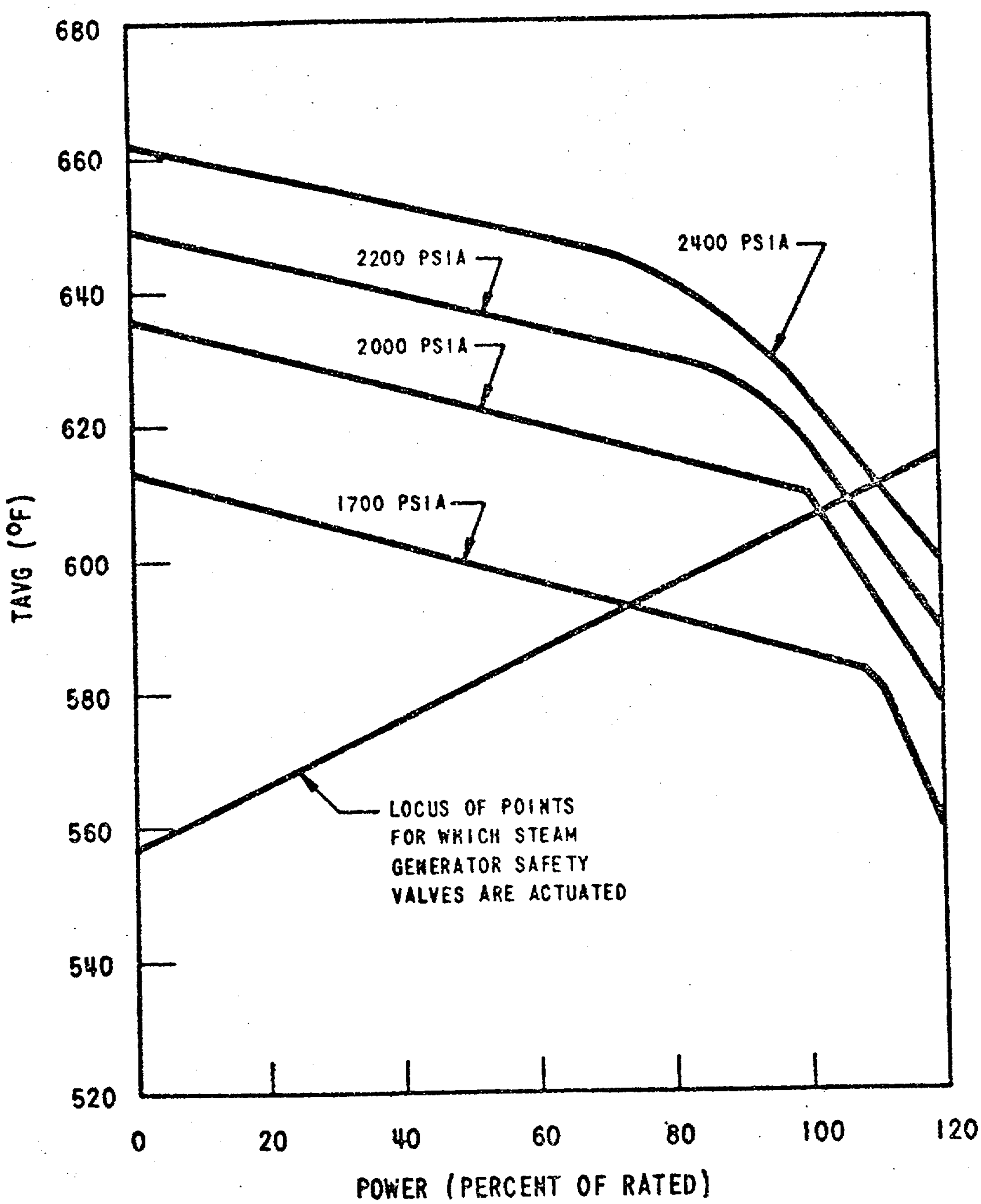


Figure 15.2.1-1 Core DNB Safety Limits
Point Beach Unit 2

3. Safety Valves

a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.

b. Both pressurizer safety valves shall be operable whenever the reactor is citical.

4. OPERATIONAL LIMITATIONS

a. The following DNB related parameters shall be maintained within the limits shown:

1) Reactor Coolant System $T_{AVG} \leq$ 578°F

2) Pressurizer Pressure $\geq$ 2220* psia during operation at 2250 psia.

3) Reactor Coolant System Total Flow Rate $\geq$ 178,000 gpm.

* Limit not applicable during either a thermal power ramp increase in excess of 5% rated thermal power per minute or a thermal power step increase in excess of 10% rated thermal power.

Basis:

When the boron concentration of the reactor coolant system is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of little concern because of the low pressurizer volume and because of the pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Part 1 of the specification requires that a sufficient number of reactor coolant pumps be operating to provide core cooling in the event that a loss of flow occurs. The flow provided in each case will keep DNBR well above 1.30 as discussed in FFDSAR Section 14.1.9. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Heat transfer analyses (1) show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only: hence, the specified upper limit of 1% rated power without operating pumps provides a substantial safety factor.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs., per hr. of saturated steam at set point. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization. Part 1 c(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 10% and 50% of rated power. Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0 which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.(2)

Although the operational limitations above require reactor coolant system total flow be maintained above a minimum rate, no direct means of measuring absolute flow during operation exists. However, during initial unit startup reactor coolant flow was measured and correlated to Core $\Delta T$. Therefore monitoring of $\Delta T$ may be used to verify the above minimum flow requirement is met. If a change in steady state full power $\Delta T$ greater than $3^{\circ}F$ is observed then actual flow measurements will be taken.


Reference

(1) FSAR Section 14.1.6

(2) FSAR Section 7.2.3

3. The part-length rods shall be fully withdrawn from the core, except for physics.

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 if 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. Power Distribution Limits

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

| | |
|---|---|
| $F_Q(z) \leq \left(\frac{2.32}{P}\right) \times k(z)$ | for $P > 0.5$ and for cycle burnups $<$ 7190 MWD/MTU. |
| $F_Q(z) \leq \left(\frac{2.25}{P}\right) \times k(z)$ | for $P > 0.5$, for Region 3 only, and for cycle burnups $\geq$ 7190 MWD/MTU. |
| $F_Q(z) \leq \left(\frac{2.32}{P}\right) \times k(z)$ | for $P > 0.5$ for Region 4 and 5, and for cycle burnups $\geq$ 7190 MWD/MTU. |
| $F_Q(z) \leq (4.64) \times k(z)$ | for $P \geq 0.5$ and for cycle burnups $<$ 7190 MWD/MTU. |
| $F_Q(z) \leq (4.50) \times k(z)$ | for $P \leq 0.5$, for Region 3 only, and for cycle burnups $\geq$ 7190 MWD/MTU. |
| $F_Q(z) \leq (4.64) \times k(z)$ | for $P \leq 0.5$, for Region 4 and 5, and for cycle burnups $\geq$ 7190 MWD/MTU. |

$F^N_{\Delta H} \leq 1.55 \times (1 + 0.2\,(1-P))$

where P is the fraction of full power at which the core is operating, K(Z) is the function in Figure 15.3.10-3 and Z is the core height location of $F_Q$.

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

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

In the specified limit of $F^N_{\Delta H}$ there is 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (i.e., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting $F_Q$, (b) the operator has a direct influence on $F_Q$ through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in $F_Q$ by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month of 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 procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference ($\Delta I$) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = $\Delta I$/fractional power).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 21 to Facility Operating License No. DPR-27 issued to Wisconsin Electric Power Company and Wisconsin Michigan Power Company which revised Technical Specifications for operation of the Point Beach Nuclear Plant Unit No. 2, located in the Town of Two Creeks, Manitowac County, Wisconsin. The amendment is effective as of its date of issuance.

The amendment will revise the Technical Specifications to modify the fuel residence time limit and allow an increase in normal operating reactor coolant system pressure. The core power distribution limits would be modified and new operating limits established for parameters related to Departure from Nucleate Boiling (DNB) to allow operation of Unit 2 in core Cycle 3.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 5, 1976 (41 F.R. 5354). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated December 22, 1975, January 29, 1976 and March 5, 1976, (2) Amendment No. 21 to License No. DPR-27, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Document Department, University of Wisconsin - Stevens Point Library, Stevens Point, Wisconsin.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of March, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-301

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FOR THE NUCLEAR REGULATORY COMMISSION

/S/

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

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| SURNAME ➤ | CParrish:kmf | JWetmore | GLear | KETCHEN | KRGoller | |
| DATE ➤ | 3/ 17 /76 | 3/ 17 /76 | 3/ 18 /76 | 3/ 22 /76 | 3/ /76 | |

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