MAR 6 1975

Docket No. 50-266

dispatched 3/12.

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 4 to Facility License No. DPR-24 for the Point Beach Nuclear Plant Unit No. 1. This amendment includes Change No. 9 to the Technical Specifications, and is in response to your request dated November 15, 1974.

The amendment permits the Point Beach Nuclear Plant Unit No. 1 to operate core cycle 3 to 18,000 effective full power hours and deletes certain technical specifications related to steam generator leak detection that are not applicable to core cycle 3 operation.

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

Sincerely,

Original Signed

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

Enclosures:
1. Amendment No. 4
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

OP (1)

See attached Concurrence Sheet for previous Concurrence.

OFFICE➤	ORB #3	ORB #3	OELD	L:AD/ORs	D:DRL	
SURNAME➤	PErickson:kmf SATeets	GLear GL	WILCHINS	KRGoller KRG	AGiambusso	
DATE➤	2/25/75	2/26/75	2/27/75	2/28/75	3/4/75	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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bcc: JRBuchanan
TBAbernathy

OFFICE	ORB#3	ORB#3 pbr	ORB#3	OELD EK	L:AD/ORs	D:DRL
SURNAME	SATeets:kmf	PErickson:kmf	GLear	Ketchen	KRGoller	AGiambusso
DATE	2/ 6 /75	2/ 6 /75	2/ /75	2/ 14 /75	2/ /75	2/ /75

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 cc: w/enclosures

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OFFICE➤						
SURNAME➤						
DATE➤						

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

WISCONSIN ELECTRIC POWER COMPANY
AND
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated November 15, 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 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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and 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, as revised, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 9."

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

FOR THE NUCLEAR REGULATORY COMMISSION

A Giambusso

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 9 to the
Technical Specifications

Date of Issuance: MAR 6 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 4

CHANGE NO. 9 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

The Technical Specifications are changed as follows:

Replace Pages 15.2.1-1, 15.2.1-3, 15.3.1-12 and 15.3.1-14a

Delete Figure 15.3.1-3

Delete Page 15.3.1-14b

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 18,000 effective full power hours (EFPH) under design operating conditions, with a primary system pressure of 2000 psia. 9

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 18,000 EFPH to assure no clad flattening without prior review by the Regulatory staff. The residence time of 18,000 EFPH is based on predicted minimum time to clad flattening for an operating pressure of 2,000 psi. Beyond a residence time of 18,000 EFPH 9
for cycle 3, an assumption of clad flattening is presently required. The basis for the calculation of clad flattening time is given in WCAP 8377, "Revised Clad Flattening Model".

5. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

6. When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means are available to detect leakage. 9

7. Secondary coolant gas radioactivity shall be monitored continuously by the air ejector gas monitor.

   Secondary coolant gross radioactivity shall be measured weekly. If the air ejector monitor is not operating, the secondary coolant gross radioactivity shall be measured daily to evaluate steam generator leak tightness.

leakage within the containment will be conducted to enhance early detection of problems and to assure best on-line reliability.

If leakage is to another system, it will be detected by the plant radiation monitors and/or water inventory control.

Steam generator tube leakage limits are based upon offsite dose considerations as limited by 10 CFR Part 20 in the event of an overpower transient with the presence of collapsed rods and 10 CFR Part 100 limits in the event of a steam line break or rod ejection accident.

The evaluation of the overpower transient assumed:

a. Five percent of the core iodine inventory is present in the fuel rod gaps.
b. The overpower transient is assumed to fail all flattened rods in the core, and all iodine in the gaps of those rods are immediately released to the coolant.
c. The coolant activity is assumed to leak to the secondary side at a constant rate as given in Figure 15.3.1-3.
d. A retention factor of ten is applied to iodine releases. It is assumed for this analysis that the relief valves remain open for 2 hours following the transient.
e. No activity is released after 2 hours.
f. $X/Q - 3.0 \times 10^{-4}$ sec/m^3.
g. The 2-hour site boundary dose limit is 1.5 Rem thyroid as per 10 CFR Part 20. 9

Continuous monitoring of steam generator tube leakage is accomplished by either the Air Ejector Radiation Monitor or the Steam Generator Blowdown Radiation Monitor in combination with periodic surveillance of the primary coolant activity. Backup monitoring can be accomplished by sampling secondary coolant gross activity.

<u>References</u>

FFDSAR Section 6.5, 11.2.3

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

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING SUPPORTING AMENDMENT NO. 4 TO LICENSE NO. DPR-24 CHANGE NO. 9 TO THE TECHNICAL SPECIFICATIONS

WISCONSIN ELECTRIC POWER COMPANY
AND
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-266

Introduction

In their letter dated November 15, 1974, Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) requested a change to the Technical Specifications appended to Facility Operating License No. DPR-24 for the Point Beach Nuclear Plant Unit No. 1. The licensees requested that the present cycle 3 fuel residence time of 6000 Effective Full Power Hours (EFPH), with the primary pressure limited to 2000 psia, be changed to 18,000 EFPH with primary pressure limited to 2250 psia. They also requested that the specifications on steam generator leakage that were established for cycle 2 operation be deleted as they are not applicable to cycle 3 operation.

Evaluation

A. Minimum Time to Clad Flattening (Fuel Residence Time)

We have reviewed the Wisconsin Electric Power Company and Wisconsin Michigan Power Company request to remove the current operational restriction on fuel cycle 3. Our review of cycle 3 operation to 18,000 EFPH is based on the revised clad flattening model (WCAP 8377)(1), surveillance information obtained from other reactors having similar fuel and the potential for clad flattening of the most limiting region (Region 4B) of fuel cycle 3.

The revised clad flattening model (WCAP 8377) predicts the initial flattening time and the flattened rod frequency of occurrence as a function of time for a given fuel region. A statistical analysis is

AMERICAN REVOLUTION BICENTENNIAL 1776-1976 employed in the model which combines the initial ovality distribution and the fuel rod power census (number of rods at each power level). The report (WCAP 8377) provides an acceptable basis for calculating a conservative fuel residence time for pressurized rods containing relatively stable fuel.

Specific confirmatory information on the revised clad flattening model (WCAP 8377) is available from surveillance of other reactors having similar fuel, e.g., R. E. Ginna and H. B. Robinson. Flattened rods have not been observed in H. B. Robinson (Regions 2 & 3) for exposures of 19,000 EFPH (14,000 EFPH predicted). Region 4A of R. E. Ginna has been exposed for 19,400 EFPH (14,000 EFPH predicted) with no observed flattened rods. A few flattened rods (0.7% in Region 2 and 0.05% in Region 3) have been observed in Point Beach Unit 1 for exposures of 21,000 EFPH (14,000 EFPH predicted for Region 2 and 16,000 EFPH predicted for Region 3). It should be noted that the initial rod internal pressure is higher in Region 4B of Point Beach Unit 1 than in the reactor regions mentioned above. Consequently, the higher initial internal pressure increases the fuel rods resistance to clad flattening.

Based on our review of the licensees proposed operation and the revised clad flattening model (WCAP 8377), we have determined that operation of core cycle 3 to 18,000 EFPH can be accomplished without a significant decrease in any safety margin and without any increase in the probability or consequences of an accident.

B. Change in Limit on Primary System Pressure

The proposed change in the limit on primary system pressure from 2000 psia to 2250 psia is not consistent with Technical Specification 15.2.3.1.B(4) which assumes a pressurizer pressure of 1985 psig (2000 psia) in the calculation of the over temperature ΔT limit. For the above reason and the licensees stated intention to continue to operate Unit 1 at 2000 psia, we have retained 2000 psia as the Technical Specification limit on primary system pressure. The 2000 psia limit on primary system pressure is more conservative than a 2250 psia limit as lower pressure further extends the predicted time to clad collapse for all of the fuel.

C. Steam Generator Leakage

The Technical Specifications on steam generator leakage (specification 15.3.1D - item 7 and figure 15.3.1-3) may be deleted as they were specifically developed for cycle 2 only and were not intended to be applied to cycle 3 operations. This change to the steam generator leakage specification is therefore editorial only and has no safety significance.

D. Summary

We have therefore concluded that the proposed operation of Point Beach Unit 1 to 18,000 EFPH in core cycle 3 is acceptable because clad flattening will not occur during this period of operation. We have also determined that the Technical Specifications on steam generator leakage may be deleted as proposed because they are not applicable to cycle 3 operation. Conversly, we have not changed the Technical Specification limit on primary system pressure as proposed by the licensee because of the licensees stated intention to operate within the existing limit and the fact that the proposed limit was not consistent with other Technical Specifications.

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: MAR 6 1975

BIBLIOGRAPHY

1. Westinghouse Electric Corporation, "Revised Clad Flattening Model", WCAP 8377, July 1974.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-266

WISCONSIN ELECTRIC POWER COMPANY
AND
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. 4 to Facility Operating License No. DPR-24, issued to Wisconsin Electric Power Company and the Wisconsin Michigan Power Company (the licensees) which revised Technical Specifications for operation of the Point Beach Nuclear Power 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 operation of Point Beach Nuclear Plant Unit No. 1 in cycle 3 for 18,000 Effective Full Power Hours and deletes certain technical specifications related to steam generator leak detection that are not applicable to core cycle 3 operation.

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. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this license amendment see (1) Amendment No. 4 to Facility License No. DPR-24 with Change No. 9, (2) the related Safety Evaluation, and (3) information submitted by the licensee in a letter dated November 15, 1974, which are available for public inspection at the Commission's Public Document Room, 1717 H Street,

N. W., Washington, D. C. and at the University of Wisconsin - Stevens Point Library, Stevens Point, Wisconsin. A copy of items (1) and (2) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing, Office of Nuclear Reactor Regulation.

Dated at Bethesda, Maryland, this 6th day of March, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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