MAR 29 1974

Docket No. 50-266

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

Change No. 6
License No. DPR-24

Gentlemen:

By letter dated February 22, 1974, you proposed a change to the Technical Specifications of Facility Operating License No. DPR-24 for the Point Beach Nuclear Plant Unit No. 1. The proposed change would allow an increase in authorized effective full power hours (EFPH) from 8000 EFPH to 9000 EFPH for Unit No. 1 Cycle 2. Subsequent discussions with your staff confirmed that an increase of 150 EFPH would allow operation of Unit No. 1 to the scheduled refueling shutdown in early April 1974. We have, therefore, accomplished our evaluation with respect to the required increase of 150 EFPH and have modified our action accordingly.

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 Attachment A.

We are also hereby correcting an administrative error contained in Change No. 4 to the Technical Specifications issued May 22, 1973, as follows:

Page 15.3.1-12, Item 7 - the reference to Figure 15.3.1-1 - this should read "Figure 15.3.1-3".

Also, the Figure 15.3.1-1 entitled "Maximum Steam Generator Leak Rate VS Time" should be removed from the Technical Specifications and the enclosed corrected Figure 15.3.1-3 inserted in the correct position within the Technical Specifications.

Sincerely,

Original signed by
Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Attachment A - Change No. 6 to the Technical Specifications
2. Figure 15.3.1-3
3. Safety Evaluation

cc w/enclosures:
Mr. Bruce W. Churchill, Esquire
Shaw, Pittman, Potts, Trowbridge and Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

Myron M. Cherry, Esquire
One IBM Plaza
Cambridge, Massachusetts 02127

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808 Hamilton Street
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Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
1 N. Wacker Drive
Chicago, Illinois 60606

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PCollins, L:OLB
ACRS (16)
RO (3)
OGC

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SURNAME➤	PBErickson:dc	SATeets	RJSchemel	DJSkovholt		
DATE➤	3/26/74	3/26/74	3/26/74	3/[illegible]/74		

ATTACHMENT A

CHANGE NO. 6 TO THE TECHNICAL SPECIFICATIONS

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-266

Delete Pages 15.2.1-1, 15.2.1-3, 15.3.1-14a and replace with the attached revised pages. Add the attached new page numbered 15.3.1-14b.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

CHANGE NO. 6 TO THE TECHNICAL SPECIFICATIONS

DOCKET NO. 50-266

By letter dated February 22, 1974, Wisconsin Michigan and Wisconsin Electric Power Company proposed a change in the Technical Specifications of Facility Operating License No. DPR-24 to allow an increase in effective full power hours (EFPH) of operation for Point Beach Unit No. 1 from 8000 to 9000 EFPH. Subsequent discussions with Wisconsin Michigan and Wisconsin Electric Power Company staff confirmed that an increase in authorized EFPH of 150 EFPH would allow operation of Unit No. 1 to the scheduled refueling shutdown in early April 1974. This review is, therefore, concerned only with the required increase to 8150 EFPH.

Our concerns relative to the proposed change are (1) will the potential increase in the number of collapsed fuel rods effect the fuel clad temperature such that the required minimum DNB ratio of 1.3 cannot be maintained during anticipated transients, and (2) are limits on steam generator leakage restrictive enough such that there would be no increase in potential offsite exposures in the event of an over power transient or a steam line break.

DNB Requirements

The analysis provided by the licensee in January 1973 "Fuel Densification, Point Beach Nuclear Plant, Unit 1 - Cycle 2" shows that the required minimum DNB ratio of 1.3 is maintained during anticipated transients with an assumed flattened section in every fuel rod. The predicted number of flattened sections in fuel rods at 8150 EFPH is a small fraction of the number assumed in the licensees analysis of January 1973. We, therefore, have concluded that the DNB ratio of 1.3 will be maintained as required at 8150 EFPH of operation.

Steam Generator Leakage Limits

Proposed changes to the steam generator leakage limits adequately compensate for any increase in the number of flattened fuel rod sections caused by an increase in EFPH.

OFFICE➤						
SURNAME➤						
DATE➤						

New leakage limits are based on straight line extrapolation of the limits as specified in Figure 15.3.1-3 of the Technical Specifications. Straight line extrapolation of the leakage limit vs. EFPH plot results in lower leakage limits than would a best fit extrapolation to the limit curve. The increase in EFPH, therefore, will result in no increase in the potential release of fission products in the event of a steam line break or a rod ejection accident as the small increase in potential fuel rod flattening is compensated for by a larger decrease in steam generator leakage rate limit.

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 in the manner proposed.

/S/

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

Original signed by
Robert J. Schemel

Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Date: MAR 29 1974

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Also, the Figure 15.3.1-1 entitled "Maximum Steam Generator Leak Rate VS Time" should be removed from the Technical Specifications and the enclosed corrected Figure 15.3.1-3 inserted in the correct position within the Technical Specifications.

If there are other references to "Figure 15.3.1-1 - Maximum Steam Generator Leak Rate VS Time" that have not been noted herein, please make the proper correction.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Attachment A - Change No. 6 to the Technical Specifications
2. Figure 15.3.1-3
3. Safety Evaluation

cc w/enclosures:
Mr. Bruce W. Churchill, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

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DATE ➤	3/25/74	3/25/74	3/ /74	3/ /74		

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

March 29, 1974

Docket No. 50-266

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

Change No. 6
License No. DPR-24

Gentlemen:

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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 Attachment A.

We are also hereby correcting an administrative error contained in Change No. 4 to the Technical Specifications issued May 22, 1973, as follows:

Page 15.3.1-12, Item 7 - the reference to Figure 15.3.1-1 - this should read "Figure 15.3.1-3".

Also, the Figure 15.3.1-1 entitled "Maximum Steam Generator Leak Rate VS Time" should be removed from the Technical Specifications and the enclosed corrected Figure 15.3.1-3 inserted in the correct position within the Technical Specifications.

Sincerely,

Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
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Enclosures:
1. Attachment A - Change No. 6 to the Technical Specifications
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Chicago, Illinois 60606

ATTACHMENT A

CHANGE NO. 6 TO THE TECHNICAL SPECIFICATIONS

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-266

Delete Pages 15.2.1-1, 15.2.1-3, 15.3.1-14a and replace with the attached revised pages. Add the attached new page numbered 15.3.1-14b.

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 for Unit 1, Cycle 2, shall be presently limited to 8,150 effective full power hours (EFPH) under design operating conditions. The Licensee may propose to operate the core in excess of 8,150 EFPH by providing an analysis which includes the effect of further clad flattening or a change in operating conditions. Any such analysis, if proposed, shall be approved by the Regulatory staff prior to operation in excess of 8,150 EFPH.

Additional peaking fact s to account for local peaking e 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, a 1.55 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 2 is limited to 8,150 EFPH to assure adequate consideration by the Regulatory Staff of the number of fuel rods with flattened cladding. Prior to 8,150 EFPH, the licensee may provide the additional analyses required for operation beyond 8,150 EFPH. 6

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 20 in the event of an overpower transient with the presence of collapsed rods and 10 CFR 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 20.

The curve in Figure 15.3.1-3 is a linear extrapolation between 8,000 EFPH and 9,000 EFPH. The end point (0.186 GPM) at 9,000 EFPH is obtained by multiplying the allowed leakage at 8,000 EFPH (0.22 GPM) by the ratio 6 of the predicted number of flattened fuel rods at 8,000 EFPH and 9,000 EFPH. This extrapolation is conservative since a best estimate extrapolation of the curve results in an allowed leakage of 0.194 GPM at 9,000 EFPH. 6

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.

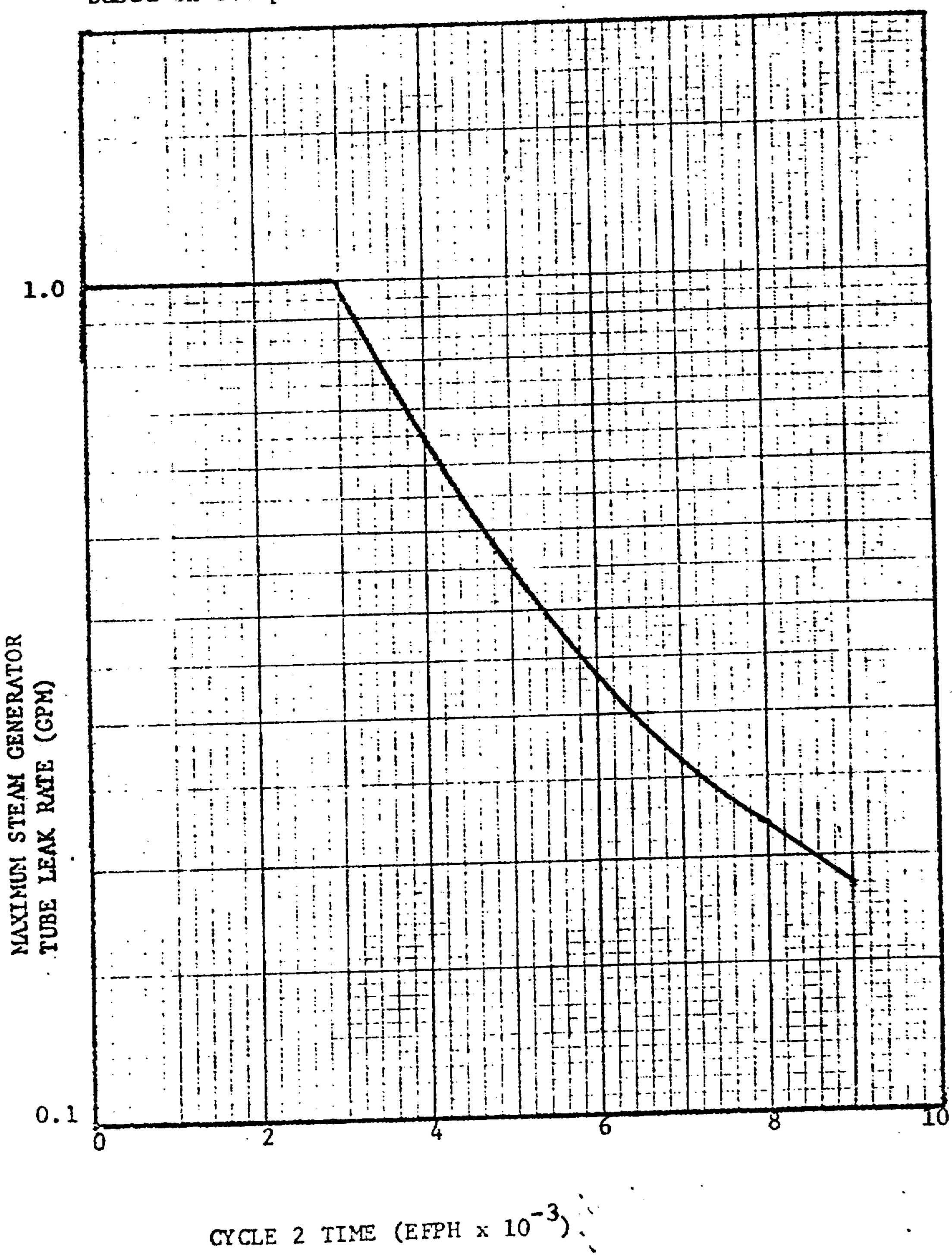
References

FFDSAR Section 6.5, 11.2.3

FIGURE 15.3.1-3

MAXIMUM STEAM GENERATOR LEAK RATE VS. TIME

Based on overpower transient 10 CFR 20 dose limits of 1.5 Rem thyroid



Change No. 6
FIGURE 15.3.1-3
Date: 3/29/74

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

WISCONSIN MICHIGAN AND WISCONSIN ELECTRIC POWER COMPANY

CHANGE NO. 6 TO THE TECHNICAL SPECIFICATIONS

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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 in the manner proposed.

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

Robert J Schemel
Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing

Date: March 29, 1974