October 22, 1985

Docket Nos. 50-266
and 50-301

DISTRIBUTION:
Docket File
NRC PDR
L PDR
ORB#3 Rdg
HThompson
PMKreutzer-3
TColburn
OELD
SECY
LHarmon
RDiggs
LTremper
OPA, CMiles
ACRS-10
MVirgilio
WJones
TBarnhart-8
JPartlow
EJordan
Gray File +4

Mr. C. W. Fay, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room 308
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment Nos. 98 and 102 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated June 17, 1985.

These amendments revise the Point Beach Unit 1 and 2 reactor vessel surveillance capsule removal schedules in Tables 15.3.1-1 and 15.3.1-2, respectively, and the related bases. Two typographical errors have been corrected in your submittal. The word "load" on line 2 of page 15.3.1-7 was misspelled. Also, incorrect reference was made to Table 15.3.1-3 in the last paragraph of page 15.3.1-8. The correct reference is Table 15.3.1-2.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/s/

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 98 to DPR-24
2. Amendment No. 102 to DPR-27
3. Safety Evaluation

cc w/enclosures:
See next page

8511070174 851022
PDR ADOCK 05000266
P PDR

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD/OR:DL
PMKreutzer	TColburn:dd	EJButcher		GCLainas
10/9/85	10/9/85	10/11/85	10/16/85	10/18/85

Mr. C. W. Fay
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Units 1 and 2

cc:
Mr. Bruce Churchill, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Mr. James J. Zach, Manager
Point Beach Nuclear Plant
Wisconsin Electric Power Company
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Chairman
Public Service Commission
of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Resident Inspector's Office
U.S. Nuclear Regulatory Commission
6612 Nuclear Road
Two Rivers, Wisconsin 54241

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated June 17, 1985 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 98 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Edward J. Butcher

Edward J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 22, 1985

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated June 17, 1985 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Edward J. Butcher

Edward J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 22, 1985

ATTACHMENT TO LICENSE AMENDMENTS NOS. 98 AND 102
TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27
DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.1-5	15.3.1-5
15.3.1-7	15.3.1-7
15.3.1-8	15.3.1-8
15.3.1-8a	15.3.1-8a
Table 15.3.1-1 (Unit 1 only)	15.3.1-1 (Unit 1 only)
Table 15.3.1-2 (Unit 2 only)	15.3.1-2 (Unit 2 only)

Basis:

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.

The ASME Code, Section III, Non-mandatory Appendix G contains procedures for the development of heatup and cooldown curves for protection against non-ductile failure. The ASME Code requires that a 1/4 wall thickness flaw, either on the inside or outside depending upon the location of concern, be assumed to exist in the structure. As the Code of Federal Regulations, Title 10, Chapter 50, Appendix G invokes the ASME Code, Appendix G, the ASME Code procedures are utilized in developing the heatup and cooldown limitation curves.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup neutron exposure of the vessel is computed to be 3.5×10^{19} neutrons/cm^2 for 40 years of operation at 1518 MWt and 80 percent load factor.[2] This is the exposure expected at the inner reactor vessel wall. However, the neutron fluence used to predict the ΔRT_{NDT} shift is the one-quarter shell thickness neutron exposure. The relationship between fluence at the vessel ID wall and the fluence at the one-quarter and three-quarter shell thickness locations has been calculated and is presented in References 3 and 4 as a function of Effective Full Power Years. These curves are used to determine the fluence at the location of interest when the heatup and cooldown curves are to be revised.

Once the fluence is determined, the temperature shift used in revising the heatup and cooldown curves is obtained from the temperature versus fluence curves (the 0.25% Copper Base, 0.20% Weld line for Unit 1 and the 0.30% Copper base, 0.25% Weld line for Unit 2) also contained in References 3 and 4. These curves are used because they are based upon a substantial amount of experimental data and represent the results of the chemical analysis of the weld metal in the reactor vessels.

The heatup and cooldown curves presented in Figures 15.3.1-1 and 15.3.1-2 (Unit 1) and 15.3.1-3 and 15.3.1-4 (Unit 2) were calculated based on the above information and the methods of ASME Code Section III (1974 Edition) Appendix G, "Protection Against Nonductile Failure", and are applicable up to the operational exposure indicated on the figures. Corrections for possible instrumentation inaccuracies have been incorporated into these curves. The temperature correction is made by adding the temperature error (24°F) to the required temperature and the pressure correction is made by subtracting the pressure error (64 psi) from the required pressure. These corrections adjust the curves in the conservative direction.

The actual temperature shift of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are identified by a specified lead factor, the measured temperature shift for a sample is an excellent indicator of the effects of power operation on the adjacent section of the reactor vessel. If the experimental temperature shift (at the 30 ft-lb level) does not substantiate the predicted shift, new prediction curves and heatup and cooldown curves must be developed.

The pressure-temperature limit lines shown on Figures 15.3.1-1 (Unit 1) and 15.3.1-3 (Unit 2) for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

The temperature requirements for the steam generator correspond with the measured NDT for the shell.

The reactor vessel materials surveillance capsule removal schedules are presented in Table 15.3.1-1 for Unit 1 and Table 15.3.1-2 for Unit 2. These schedules have been developed based upon the requirements of the Code of Federal Regulations, Title 10, Chapter 50, Appendix H and with consideration of ASTM Standard E-185-82. When the capsule lead factors are considered, the scheduled removal dates will provide materials data representative of about 10%, 20%, 50%, 90% and 110% of the actual reactor vessel exposure anticipated during the vessel life.

References

(1) FSAR, Section 4.1.5

(2) Westinghouse Electric Corporation, WCAP-10638

(3) Westinghouse Electric Corporation, WCAP-8743

(4) Westinghouse Electric Corporation, WCAP-8738

TABLE 15.3.1-1

POINT BEACH NUCLEAR PLANT, UNIT NO. 1
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Letter	Approximate Removal Date*
V	September 1972 (actual)
S	December 1975 (actual)
R	October 1977 (actual)
T	March 1984 (actual)
P	Spring 1994
N	Standby

*The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

TABLE 15.3.1-2

POINT BEACH NUCLEAR PLANT, UNIT NO. 2
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Letter	Approximate Removal Date*
V	November 1974 (actual)
T	March 1977 (actual)
R	April 1979 (actual)
P	Fall 1989
S	Fall 1995
N	Standby

*The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 98 AND 102 TO

FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

Introduction

In a letter from C. W. Fay to H. R. Denton dated June 17, 1985 the Wisconsin Electric Power Company (the licensee) requested amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2 (PBNP), respectively. The changes would revise the capsule removal schedule in the PBNP reactor vessel material surveillance program. The PBNP reactor vessel material surveillance program must meet the requirements of Appendix H, 10 CFR 50, which became effective on July 26, 1983. Appendix H, 10 CFR 50 requires that the surveillance program meet the requirements of ASTM E 185-82 to the extent practical.

Evaluation

ASTM E 185-82 requires that the PBNP reactor vessel surveillance program have a minimum of five capsules and the last capsule should be withdrawn when the capsule neutron fluence is between one and two times the peak vessel end-of-life value. Other capsules are to be withdrawn at earlier times in the vessel's life in order to determine the extent of neutron irradiation damage to the PBNP reactor vessel beltline materials.

The change in schedule requested would extend the removal dates for the final capsule in Unit 1 to a period corresponding to 110% of the peak vessel end-of-life neutron fluence and would extend the removal dates for the last two capsules in Unit 2 to the periods corresponding to 90% and 110% of the peak end-of-life neutron fluence. Capsules have been previously withdrawn from the PBNP reactor vessels at earlier times to determine the extent of neutron irradiation damage.

The staff has determined that the proposed reactor vessel surveillance capsule withdrawal schedules meet the requirements of ASTM E 185-82 as incorporated by reference into 10 CFR Part 50 Appendix H and are, therefore, acceptable.

ENVIRONMENTAL CONSIDERATION

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

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

Date: October 22, 1985

Principal Contributor:
B. Elliot
T. Colburn