JAN 1 5 1975

TBAbernathy
JRBuchanan

DISTRIBUTION:
Docket (2)
NRC PDR (2)
Local PDR
ORB-3 Reading
Attorney, OELD
OI&E (3)
NDube
BJones
JMcGough
CParrish
JWetmore
KRGoller/TJCarter
BScharf (15)
EP Licensing Assistant
CHebron
DEisenhut
ACRS (16)

Docket Nos. 50-266
and 50-301

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

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 5 and 9 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units Nos. 1 and 2. The amendments include Changes Nos. 10 and 15 to the Technical Specifications and are in accordance with your application dated July 11, 1975.

This amendment will modify the reactor coolant system pressure temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT}).

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

Sincerely,

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

Enclosures:
1. Amendments Nos. 5 and 9
2. Safety Evaluation
3. Federal Register Notice

GL

OFFICE	RL:ORB-3	RL:ORB-3	OELD	RL:ORB-3		
x27872 SURNAME	CParrish:esp	JWetmore		GLear		
DATE	12/30/75	12/19/75	1/12/76	12/13/75		

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

Wisconsin Michigan Power Company
Wisconsin Electric Power Company

cc:

Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
Barr Building
910 17th Street, N. W.
Washington, D. C. 20006

Mr. Arthur M. Fish
Document Department
University of Wisconsin -
Stevens Point Library
Stevens Point, Wisconsin 54481

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

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 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. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 15th day of January, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

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

OFFICE➤	RL:ORB-3	RL:ORB-3	OELD	RL:ORB-3		
x27872 SURNAME➤	CParrish	JWetmore		GLear		
DATE➤	12/31/75	12/19/75	~~12/ /75~~ 1-12-76	1/13/75		

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

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 5 and 9 to Facility Operating Licenses Nos. DPR-24 and 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 Units Nos. 1 and 2, located in the town of Two Creeks, Manitowoc County, Wisconsin.

The amendment modifies the reactor coolant system pressure temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT}).

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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated July 11, 1975, (2) Amendments Nos. 5 and 9 to Licenses Nos. DPR-24 and DPR-27, with Changes Nos. 10 and 15

OFFICE➤						
SURNAME➤						
DATE➤						

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

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 5
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated July 11, 1975, 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 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 with the Technical Specifications, as revised by issued changes thereto through Change No. 10."

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Change No. 10 to the
Technical Specifications

Date of Issuance: January 15, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 5

CHANGE NO. 10 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NO. 50-266

Replace pages 15.3.1-3, 15.3.1-4 through 15.3.1-8 and 15.3.1-17 and 15.3.1-18, and Figure 15.3.1-1 with the attached revised pages. (No change made on page 15.3.1-3.)

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]

Reference

(1) FSAR Section 14.1.6

(2) FSAR Section 7.2.3

B. Pressure/Temperature Limits

Specification:

1. The Reactor Coolant System temperature and pressure shall be limited in accordance with the limit lines shown in Figure 15.3.1-1 (Unit 1) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

   a. A maximum heatup of 100°F in any one hour, 10

   b. A maximum cooldown of 100°F in any one hour, and

   c. An average temperature change of ≤10°F per hour during inservice leak and hydrostatic testing operations.

2. The secondary side of the steam generator will not be pressurized above 200 psig if the temperature of the steam generator vessel shell is below 70°F.

3. The pressurizer temperature shall be limited to:

   a. A maximum heatup and cooldown of 200°F in any one hour, and 10

   b. A maximum spray water temperature differential between the pressurizer and spray fluid of not greater than 320°F.

4. The reactor vessel material irradiation surveillance specimens shall be removed and examined in accordance with a program compatible with Appendix H to 10 CFR Part 50 to determine changes in material properties. The results of these examinations shall be used to update Figure 15.3.1-1. The updated figure for 10
Figure 15.3.1-1 shall be provided to the Commission at least sixty (60) days before the calculated exposure of the applicable reactor vessel exceeds the exposure for which the figures apply.

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 FFDSAR. 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.

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 10
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 produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

During cooldown the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stress at the outside wall.

The heatup and cooldown curve is a composite curve which was prepared by determining the most conservative case with either the inside or outside wall controlling for any heatup or cooldown rate up to 100°F in any one hour. 10

In developing these curves, an initial unirradiated RT_{NDT} of 0°F for Unit 1 was used. As a result of fast neutron irradiation, there will be an increase in the RT_{NDT} with nuclear operation. The maximum integrated fast neutron exposure of the vessel is computed to be 3.8×10^{19} neutrons/cm^2 for 40 years of operation at 1520 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 RT_{NDT} shift is the one quarter shell thickness neutron exposure. This is assumed to be 2/3 of the fluence seen at the inner vessel wall. The RT_{NDT} shift due to irradiation is predicted from the "max" for 550°F curve in FFDSAR Figure 4.2-7 for Unit 1.

The heatup and cooldown curves presented in Figure 15.3.1-1 (Unit 1) 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 operation exposure indicated on the figures.

Further shifts in RT_{NDT} as a result of neutron irradiation due to operation beyond the limits indicated on the present curves may be again predicted using FFDSAR Figure 4.7-2 (Unit 1), unless measurements on the irradiation specimens, as mentioned below, show actual RT_{NDT} shifts above and to the left of the predicted curve. If this occurs, a new curve having the same slope as the original shall be constructed such that it is above and to the left of all the applicable data points. The applicable units heatup and cooldown curve, in that case, must also be recalculated. 10

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, compatible with 10CFR50 App.H, 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 transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. As mentioned above, if the actual RT_{NDT} shift exceeds the predicted shift, new prediction curves and heatup and cooldown curves must be developed.

The pressure-temperature limit lines shown on Figure 15.3.1-1 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum 10 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.

References

(1) FSAR, Section 4.1.5

(2) FSAR, Section 4.2.5

F. MINIMUM CONDITIONS FOR CRITICALITY

Specification:

1. Except during low power physics tests, the reactor shall not be made critical unless the moderator temperature coefficient is negative.

2. In no case shall the reactor be made critical (other than for the purpose of low level physics tests) to the left of the Reactor Core 10
Criticality curve presented in figures 15.3.1-1 for Unit 1.

3. When the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least $1\% \frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.[1] [2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the initial fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[1][2] Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the Reactor 10
Core Criticality Curve provides assurance that a proper relationship between
reactor coolant pressure and temperature will be maintained during system
heatup and pressurization. Heatup to this temperature will be accomplished
by operating the reactor coolant pumps. However, as provided in 10 CFR Part 50 10
Appendix G Section IV.A.2.c. the reactor core may be taken critical below
this curve for the purpose of low level physics tests.

If the specified shutdown margin is maintained (Section 15.3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-9

(3) FSAR Figure 3.2.1-10

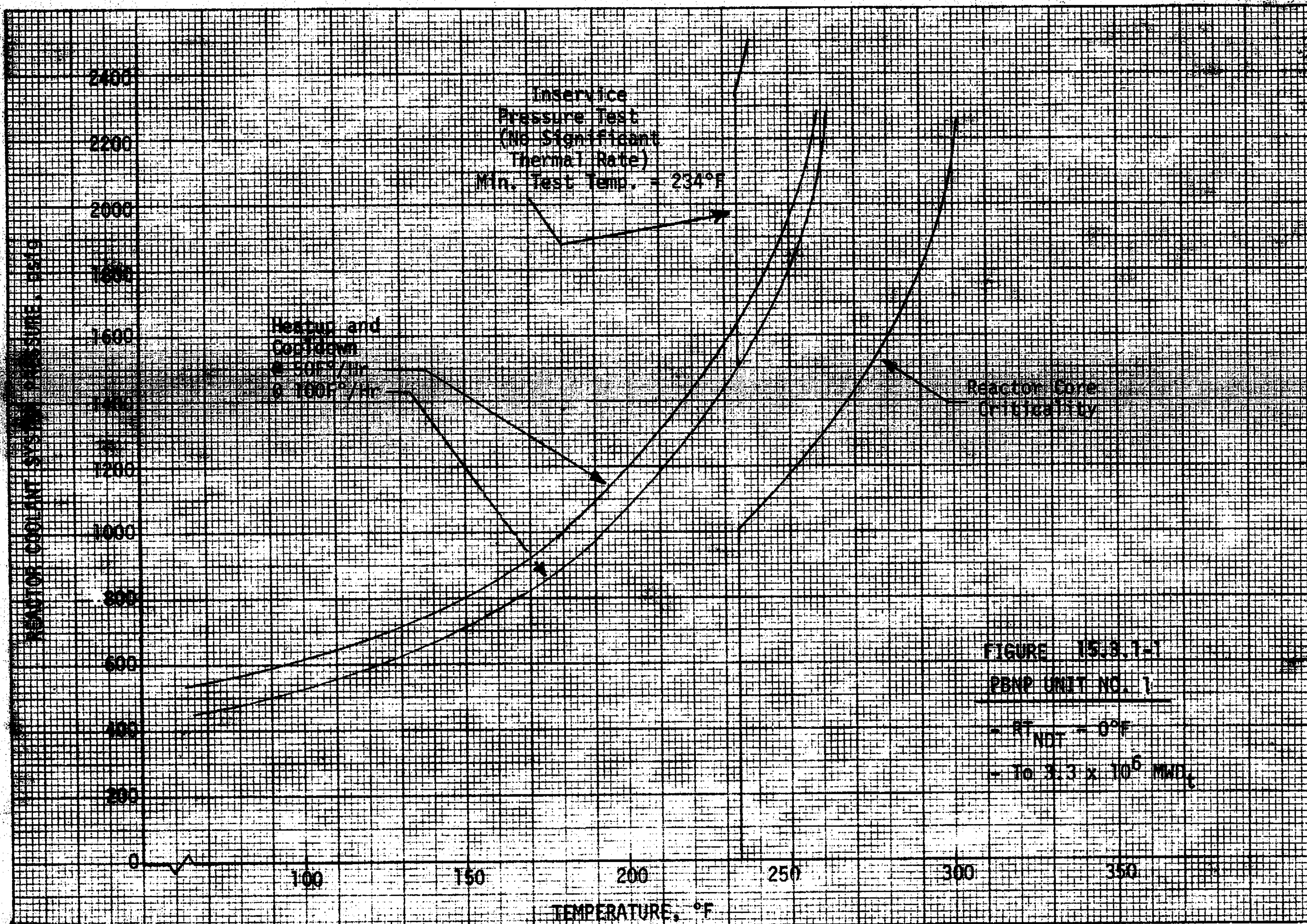

FIGURE 15.3.1-1
PBNP UNIT NO. 1
- RT_{NDT} = 0°F
- To 3.3 x 10^6 MWD_t

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

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. 9
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company and Wisconsin Michigan Power Company (the licensees) dated July 11, 1975, 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 and Paragraph 3.B. of Facility 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, 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. 15."

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Change No. 15 to the
Technical Specifications

Date of Issuance: January 15, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 9

CHANGE NO. 15 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NO. 50-301

Replace pages 15.3.1-3, 15.3.1-4 through 15.3.1-8 and 15.3.1-17 and 15.3.1-18, and Figure 15.3.1-2 with the attached revised pages. (No change made on page 15.3.1-3.)

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]

Reference

(1) FSAR Section 14.1.6

(2) FSAR Section 7.2.3

B. <u>Pressure/Temperature Limits</u>

<u>Specification:</u>

1. The Reactor Coolant System temperature and pressure shall be limited in accordance with the limit lines shown in 15.3.1-2 (Unit 2) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:
   a. A maximum heatup of 100°F in any one hour, 15
   b. A maximum cooldown of 100°F in any one hour, and
   c. An average temperature change of $\leq$10°F per hour during inservice leak and hydrostatic testing operations.
2. The secondary side of the steam generator will not be pressurized above 200 psig if the temperature of the steam generator vessel shell is below 70°F.
3. The pressurizer temperature shall be limited to:
   a. A maximum heatup and cooldown of 200°F in any one hour, and 15
   b. A maximum spray water temperature differential between the pressurizer and spray fluid of not greater than 320°F.
4. The reactor vessel material irradiation surveillance specimens shall be removed and examined in accordance with a program compatible with Appendix H to 10 CFR Part 50 to determine changes in material properties. The results of these examinations shall be used to update Figure 15.3.1-2. The updated figure for
15.3.1-2 shall be provided to the Commission 15
at least sixty (60) days before the calculated exposure of the applicable reactor vessel exceeds the exposure for which the figures apply.

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 FFDSAR. 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.

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. 15

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 produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

During cooldown the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stress at the outside wall.

The heatup and cooldown curve is a composite curve which was prepared by determining the most conservative case with either the inside or outside wall controlling for any heatup or cooldown rate up to 100°F in any one hour.

In developing these curves, an initial unirradiated RT_{NDT} of 15
40° for Unit 2 was used. As a result of fast neutron irradiation, there will be an increase in the RT_{NDT} with nuclear operation. The maximum integrated fast neutron exposure of the vessel is computed to be 3.8×10^{19} neutrons/cm^2 for 40 years of operation at 1520 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 RT_{NDT} shift is the one quarter shell thickness neutron exposure. This is assumed to be 2/3 of the fluence seen at the inner vessel wall. The RT_{NDT} shift due to irradiation is predicted from the "max" for 550°F curve in FFDSAR Figure 4.2-7A for Unit 2.

The heatup and cooldown curves presented in Figure 15.3.1-2 (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 operation exposure indicated on the figures.

Further shifts in RT_{NDT} as a result of neutron irradiation due to operation beyond the limits indicated on the present curves may be again predicted using FFDSAR Figure 4.7-2A (Unit 2), unless measurements on the irradiation specimens, as mentioned below, show actual RT_{NDT} shifts above and to the left of the predicted curve. If this occurs, a new curve having the same slope as the original shall be constructed such that it is above and to the left of all the 15
applicable data points. The applicable units heatup and cooldown curve, in that case, must also be recalculated.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, compatible with 10CFR50 App.H, 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 transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. As mentioned above, if the actual RT_{NDT} shift exceeds the predicted shift, new prediction curves and heatup and cooldown curves must be developed.

The pressure-temperature limit lines shown on Figure 15.3.1-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum 15 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.

References

(1) FSAR, Section 4.1.5

(2) FSAR, Section 4.2.5

F. MINIMUM CONDITIONS FOR CRITICALITY

Specification:

1. Except during low power physics tests, the reactor shall not be made critical unless the moderator temperature coefficient is negative.

2. In no case shall the reactor be made critical (other than for the purpose of low level physics tests) to the left of the Reactor Core Criticality curve presented in figure 15.3.1-2 for Unit 2. 15

3. When the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.(1) (2) The moderator coefficient at low temperatures will be most positive at the beginning of life of the initial fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.(1)(2) Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the Reactor 15
Core Criticality Curve provides assurance that a proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. However, as provided in 10 CFR Part 50 15
Appendix G Section IV.A.2.c. the reactor core may be taken critical below this curve for the purpose of low level physics tests.

If the specified shutdown margin is maintained (Section 15.3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-9

(3) FSAR Figure 3.2.1-10

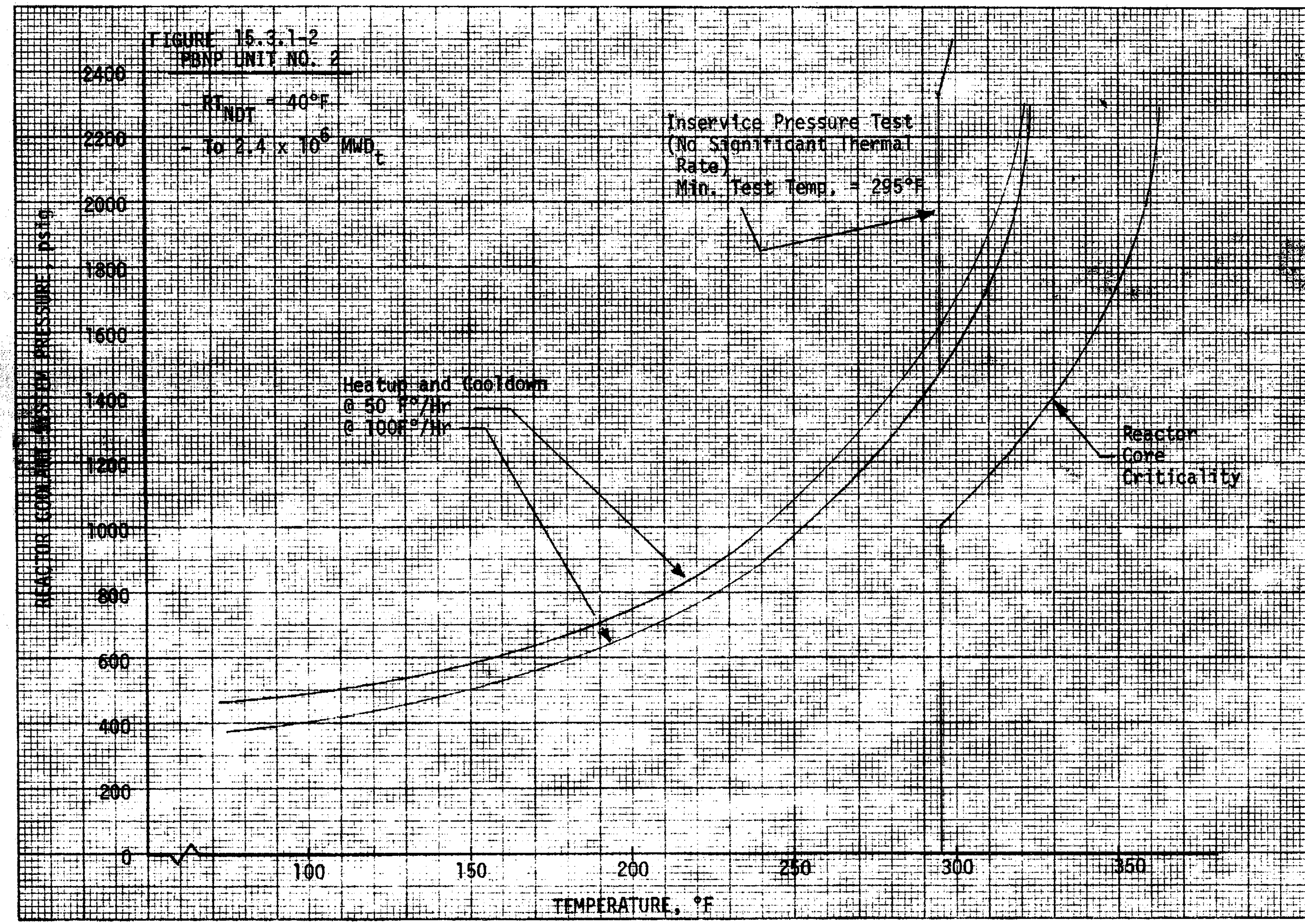
FIGURE 15.3.1-2
PBNP UNIT NO. 2
- RT_{NDT} = 40°F
- To 2.4×10^6 MWD_t
Inservice Pressure Test
(No Significant Thermal
Rate)
Min. Test Temp. = 295°F
Heatup and Cooldown
@ 50 F°/Hr
@ 100F°/Hr
Reactor
Core
Criticality
REACTOR COOLANT SYSTEM PRESSURE, psig
TEMPERATURE, °F
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
100
150
200
250
300
350

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 5 AND 9 TO LICENSE NOS. DPR-24 AND DPR-27

(CHANGE NOS. 10 AND 15 TO THE TECHNICAL SPECIFICATIONS)

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

Introduction

By letter dated July 11, 1975 Wisconsin Electric Power Company (WEPCO) requested changes to the Technical Specifications appended to Facility Operating Licenses DPR-24 and DPR-27, for Point Beach Nuclear Plant, Units 1 and 2. The requested changes would modify the reactor coolant system pressure temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT})[1].

Discussion

Title 10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material and the design service conditions. However, the metallurgical properties vary over the lifetime of the reactor vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature (RT_{NDT}) to increase or shift with time. The practical results of the RT_{NDT} shift

---

[1] RT_{NDT} is the temperature associated with the transition from ductile to brittle fracture mode of failure.

is that, for any given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT}, the stresses in the reactor vessel are maintained within acceptable limits.

The magnitude of the shift in RT_{NDT} is proportional to the integrated amount of neutron irradiation experienced by the reactor vessel. For Point Beach Units 1 and 2 the predicted RT_{NDT} shift is given in the Final Facility Description and Safety Analysis Report (FFDSAR). In addition a reactor vessel material surveillance program is established to check the validity of the predicted increases in RT_{NDT}. Surveillance specimens are periodically removed from the reactor vessel for testing and analysis. The results of the tests and analysis are compared with the predicted shifts in RT_{NDT}, then the pressure-temperature limits are revised accordingly.

Analyses of the first reactor vessel material surveillance specimens for Point Beach, Units 1 and 2 were completed and the results were submitted by reports dated June 15, 1973 for Unit 1 and June 10, 1975 for Unit 2. The proposed changes to the reactor coolant system pressure-temperature limits evaluated below account for the observed shifts in RT_{NDT}.

Evaluation

The proposed changes to the reactor coolant system pressure-temperature limits involve limits for heatup and cooldown, hydrostatic testing, and reactor core criticality. The proposed limits are based on experimental data, predicted increases in RT_{NDT}, and upgraded analytical techniques. Each aspect of the proposed changes is discussed below:

(1) Revised pressure-temperature limits for heatup and cooldown operations:

The current Technical Specifications limit heatup and cooldown rates to a maximum of $100^{o}F/hr$ with lower rates imposed for certain temperature ranges. The accompanying pressure-temperature limits were developed before the plants were licensed using techniques which have since been improved. The proposed changes would modify the effective temperature and pressure ranges but would retain the heatup and cooldown rate limit of $100^{o}F/hr$.

The proposed heatup and cooldown rates and accompanying pressure-temperature limits are based on updated techniques which are spelled out in the Summer 1972 Addenda to the ASME Boiler and Pressure Vessel Code, Section III. The techniques are presently required in Appendix G to 10 CFR Part 50. The techniques involve the use of the RT_{NDT} (i.e., reference RT_{NDT} plus any radiation induced shift) which is in turn used to index the material to a reference stress intensity factor curve which appears in Appendix G of the ASME Code. The curve in the Code provides the stress intensity factor for the material and this stress intensity factor serves as an allowable upper limit in the analysis. Actual stress intensity factors are then determined by combining the effects of pressure stress and thermal gradient stress. The allowable stress intensity factor must then be larger than the actual stress intensity factors with appropriate safety margins included. The analysis then consists of determining limiting conditions within these guidelines for various heatup and cooldown conditions within these guidelines for various heatup and cooldown rates.

The value of RT_{NDT} which is used to construct the pressure-temperature curves is a plant dependent parameter. The results of the Unit 1 reactor vessel surveillance specimen tests indicated that the shifts in RT_{NDT} were within the FFDSAR predicted values. Therefore the predicted values were used to revise the Unit 1 pressure-temperature curves (Figure 15.3.1-1 for Unit 1). However, the results of the Unit 2 reactor vessel surveillance specimen tests indicated shifts in RT_{NDT} which were greater than predicted in the FFDSAR. This was caused by the relatively high residual element content (0.25% copper and 0.014% phosphorus) of the Unit 2 reactor vessel weld metal. Consequently, the measured shifts in RT_{NDT} were used to revise the Unit 2 pressure-temperature curves (Figure 15.3.1-2) for Unit 2 because they yielded more conservative limits. Also, the predicted shifts for Unit 2 in the FFDSAR were adjusted accordingly.

We have reviewed the reactor vessel surveillance specimen test data, the analytical techniques, and the resulting pressure-temperature curves. We have concluded that the proposed changes effectively account for shifts in RT_{NDT}, and are compatible with Appendix G to 10 CFR Part 50, and therefore, are acceptable.

(2) Definition of pressure-temperature limits for reactor coolant system hydrostatic testing:

The pressure-temperature limits defined for hydrostatic leak tests (Figure 15.3.1-1 for Unit 1 and Figure 15.3.1-2 for Unit 2) were determined in a similar manner as the limits discussed above and in accordance with the ASME Code. In this case, however, there was no need for consideration of the thermal gradient stress intensity factor because no significant metal heatup or cooldown occurs during the hydrostatic test. We have reviewed the proposed limits and conclude that they meet the requirements of Appendix G to 10 CFR Part 50; and therefore are acceptable.

(3) Pressure-temperature limits for reactor core criticality:

The proposed limits (Figure 15.3.1-1 for Unit 1 and Figure 15.3.1-2 for Unit 2) were derived using methods described earlier with additional safety factors required by Appendix G to 10 CFR Part 50. These additional factors include the temperature for criticality (equal to that of hydrostatic test) and an additional 40°F temperature increase beyond the operating pressure-temperature limits discussed earlier. We have reviewed the basic test data, the analytical techniques and the resultant limits and conclude that they meet the requirements of Appendix G to 10 CFR Part 50; and therefore are acceptable.

(4) Administrative change to Technical Specification 15.3.1.F.2:

The existing Technical Specification 15.3.1.F.2 specifies a core criticality limit based on outdated calculational techniques. The proposed change would reference the revised Figures 15.3.1-1 and 2 for the reactor core criticality limit. This limit was evaluated above. The proposed change to Technical Specification 15.3.1.F.2 is administrative in nature and only serves to make the Technical Specifications consistent; and therefore is acceptable.

Summary

WEPCO has conducted tests on their first capsule irradiation specimens from Point Beach Units 1 and 2. Based on the results from the test program, WEPCO has proposed revised reactor vessel heatup and cooldown rates and accompanying pressure-temperature limits. We have reviewed the test results, analytical techniques, and proposed limits and conclude that all of the proposed changes 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) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: January 15, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 5 and 9 to Facility Operating Licenses Nos. DPR-24 and 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 Units Nos. 1 and 2, located in the town of Two Creeks, Manitowoc County, Wisconsin.

The amendment modifies the reactor coolant system pressure temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT}).

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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated July 11, 1975, (2) Amendments Nos. 5 and 9 to Licenses Nos. DPR-24 and DPR-27, with Changes Nos. 10 and 15 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 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. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 15th day of January, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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