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Dockets 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. 24 and 28 to Facility Operating Licenses Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications and are in accordance with your application dated October 1, 1976.

These amendments consist of changes in the Technical Specifications that 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}), and will add a new specification showing the reactor vessel surveillance capsule withdrawal schedules.

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

Sincerely,

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

Enclosures:
1. Amendment No. 24 to License DPR-24
2. Amendment No. 28 to License DPR-27
3. Safety Evaluation
4. Federal Register Notice

cc:
See next page

OFFICE	ORB#3	ORB#3	DOR	OELD	ORB#3
SURNAME	CParrish	JWetmore:acr	LShao	Ketchen	GLear
DATE	2/8/77	2/8/77	2/11/77	2/22/77	2/24/77

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

Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. Norman Clapp, Chairman
Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

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

Wisconsin Electric Power Company
ATTN: Mr. Glen Reed
Manager, Nuclear Power Division
Point Beach Nuclear Plant
231 West Michigan Street
Milwaukee, Wisconsin 53201

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Revion V Office
ATTN: EIS COORDINATOR
230 S. Dearborn Street
Chicago, Illinois 60604

Walter L. Meyer
Town Chairman
Town of Two Creeks, Wisconsin
Route 3, Two Rivers, Wisconsin 54241

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. 24
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 October 1, 1976 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   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 the 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. 24, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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 Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 24

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-24

DOCKET NO. 50-266

Replace pages 15.3.1-4 through 15.3.1-8 with the attached revised pages. The attached Figures 15.3.1-1 thru 15.3.1-4 replace the existing Figures 15.3.1-1, 15.3.1-1a, 15-3.1-2, 15.3.1-2a and 15.3.1-3. Add pages 15.3.1-8a and Tables 15.3.1-1 and 15.3.1-2.

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 and 15.3.1-2 (Unit 1) and 15.3.1-3 and 15.3.1-4 (Unit 2) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

   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

   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 the schedules presented in Table 15.3.1-1 (Unit 1) and 15.3.1-2 (Unit 2) to determine changes in material properties. The results of these examinations shall be considered in the evaluation of the prediction method to be used to update Figures 15.3.1-1, 15.3.1-2, 15.3.1-3, and 15.3.1-4. Revised figures 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 strees 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 nonductile 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 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 curves are composite curves which are 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 30°F for Unit 1 and 30°F for Unit 2 was used. These values are based upon a review of the materials test data obtained from either the pressure vessel vendor test certificates, or the pre-irradiation data (core region beltline materials) developed by Westinghouse and as reported in references 3 and 4. The material property tests performed by the reactor vessel vendor were to applicable codes in effect at the time the reactor vessel was ordered. In the instances where drop weight data (NDTT) and transverse Charpy V-notch data were not obtained, an estimate of these data was made. These data were estimated by the methods of the "U. S. NRC Regulatory Standard Review Plan, Section 5.3.2, Pressure Temperature Limits". 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 3.9 x 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 (10F °)to the required temperature and the pressure correction is made by subtracting the pressure error (30 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-185-73. When the capsule lead factors are considered, the scheduled removal dates will provide materials data representative of about 10%, 20%, 40%, 60%, and 80% of the actual reactor vessel exposure anticipated during the vessel life.

References

(1) FSAR, Section 4.1.5

(2) Westinghouse Electric Corporation, WCAP-8739

(3) Westinghouse Electric Corporation, WCAP-8743

(4) Westinghouse Electric Corporation, WCAP-8738.

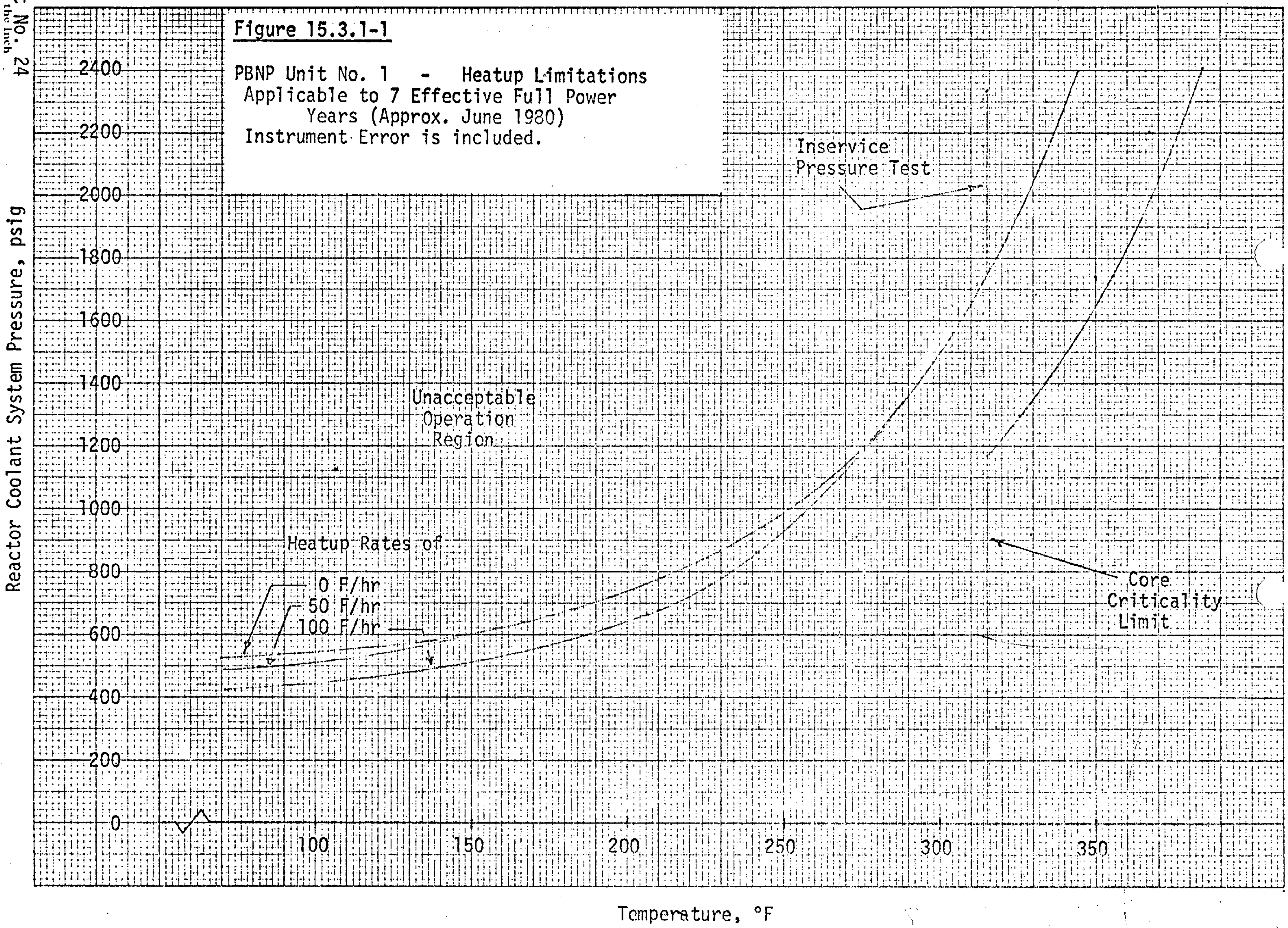

Figure 15.3.1-1

PBNP Unit No. 1 - Heatup Limitations
Applicable to 7 Effective Full Power
Years (Approx. June 1980)
Instrument Error is included.

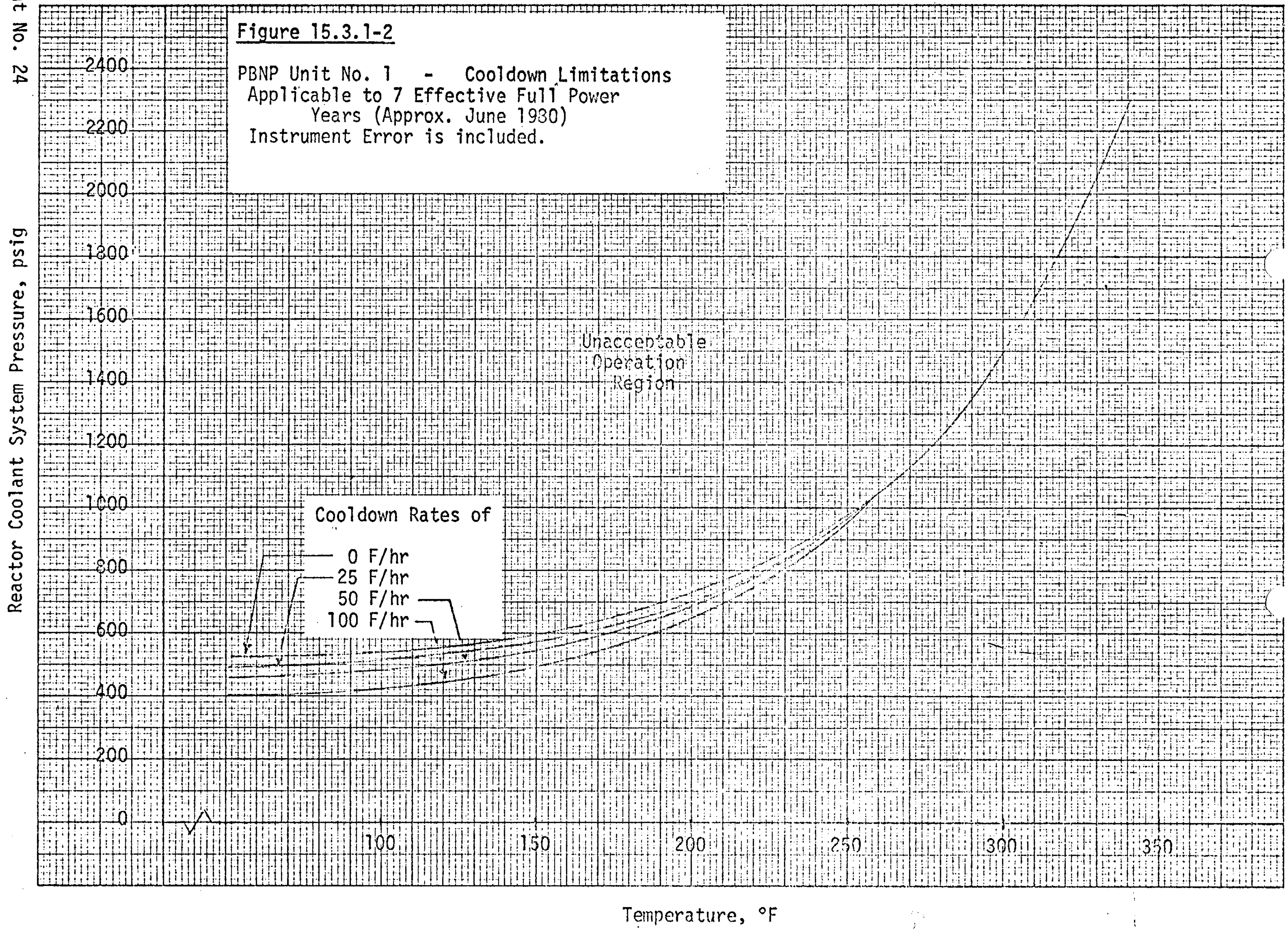
Figure 15.3.1-2
PBNP Unit No. 1 - Cooldown Limitations
Applicable to 7 Effective Full Power
Years (Approx. June 1980)
Instrument Error is included.
Unacceptable
Operation
Region
Cooldown Rates of
0 F/hr
25 F/hr
50 F/hr
100 F/hr
Reactor Coolant System Pressure, psig
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
100
150
200
250
300
350
Temperature, °F

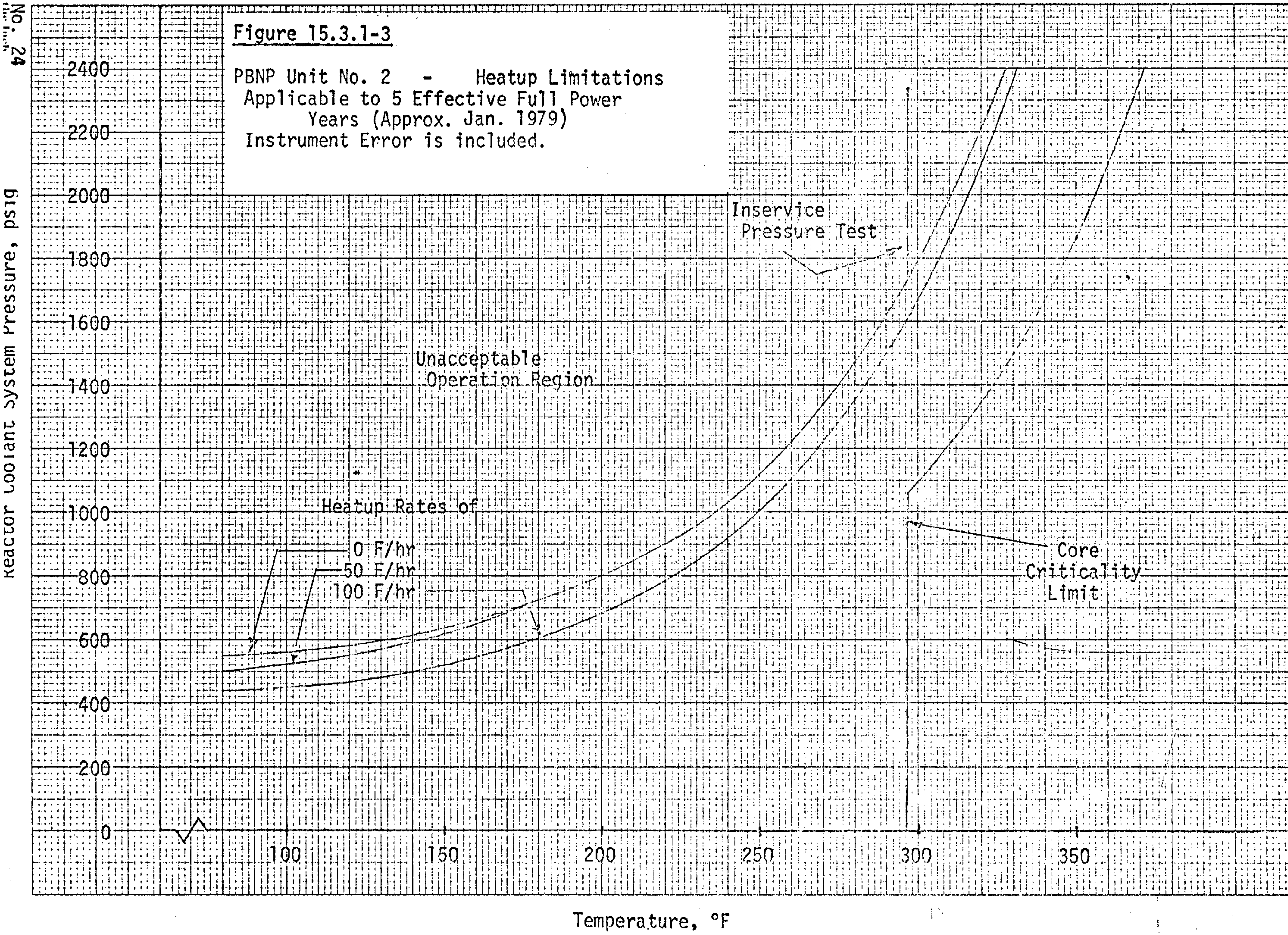

Figure 15.3.1-3

PBNP Unit No. 2 - Heatup Limitations
Applicable to 5 Effective Full Power
Years (Approx. Jan. 1979)
Instrument Error is included.

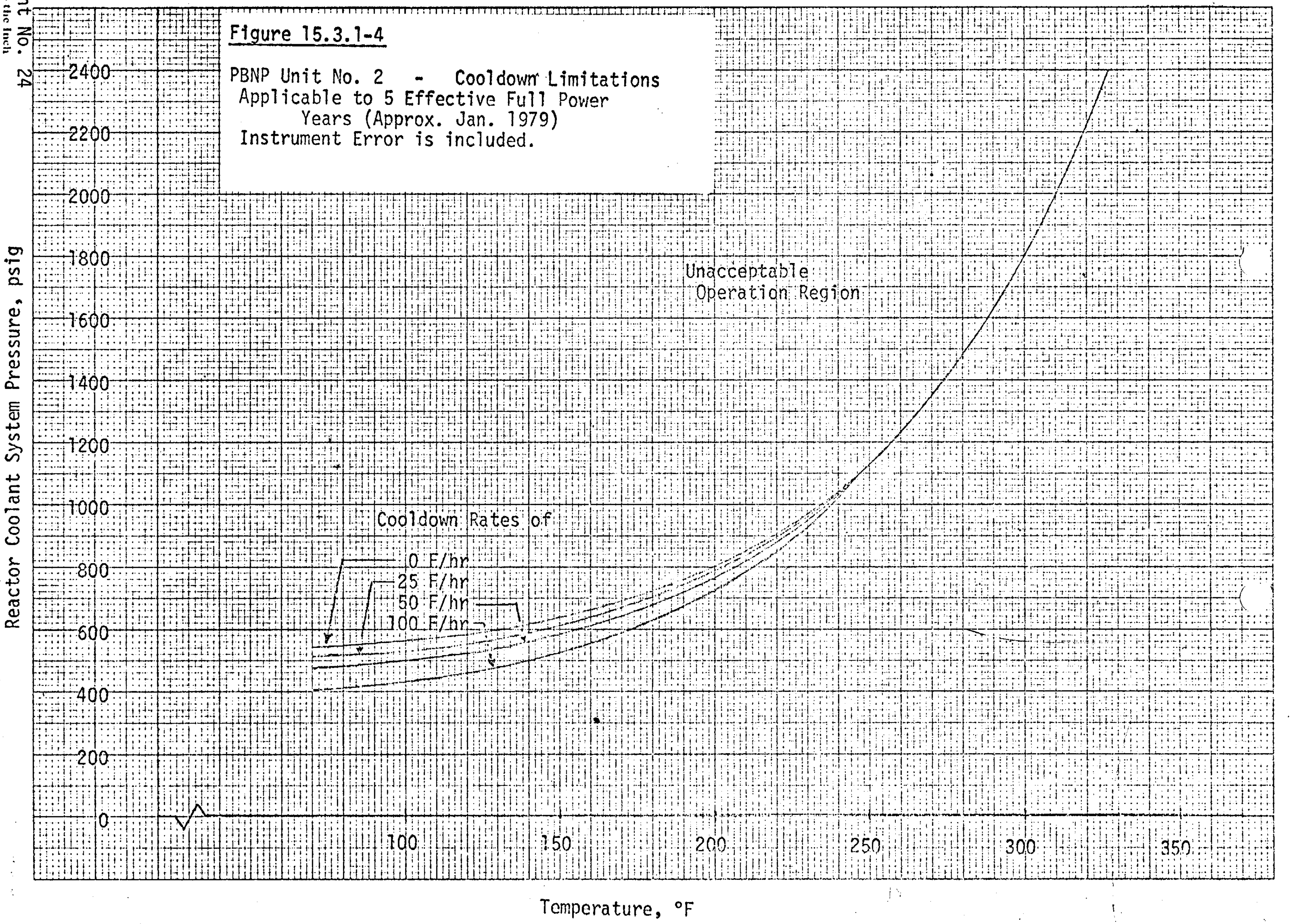

Figure 15.3.1-4

PBNP Unit No. 2 - Cooldown Limitations
Applicable to 5 Effective Full Power Years (Approx. Jan. 1979)
Instrument Error is included.

TABLE 15.3.1 - 1

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

Capsule Letter	Approximate Removal Date*
V	Sept. 1972 (actual)
S	Dec. 1975 (actual)
R	Dec. 1977
T	Nov. 1985
P	Nov. 1989
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	Nov. 1974 (actual)
T	Mar. 1977
R	Mar. 1979
P	Mar. 1986
S	Mar. 1993
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

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. 28
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 October 1, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   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. 28, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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 Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NO. 50-301

Replace pages 15.3.1-4 thru 15.3.1-8 with the attached revised pages (delete pages 15.3.1-4a and 15.3.1-4b). The attached Figures 15.3.1-1 thru 15.3.1-4 replace existing Figures 15.3.1-1, 15.3.1-1a, 15.3.1-2, 15.3.1-2a and 15.3.1-3. Add page 15.3.1-8a and Tables 15.3.1-1 and 15.3.1-2.

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 and 15.3.1-2 (Unit 1) and 15.3.1-3 and 15.3.1-4 (Unit 2) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

   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

   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 the schedules presented in Table 15.3.1-1 (Unit 1) and 15.3.1-2 (Unit 2) to determine changes in material properties. The results of these examinations shall be considered in the evaluation of the prediction method to be used to update Figures 15.3.1-1, 15.3.1-2, 15.3.1-3, and 15.3.1-4. Revised figures 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 strees 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 nonductile 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 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 curves are composite curves which are 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 30°F for Unit 1 and 30°F for Unit 2 was used. These values are based upon a review of the materials test data obtained from either the pressure vessel vendor test certificates, or the pre-irradiation data (core region beltline materials) developed by Westinghouse and as reported in references 3 and 4. The material property tests performed by the reactor vessel vendor were to applicable codes in effect at the time the reactor vessel was ordered. In the instances where drop weight data (NDTT) and transverse Charpy V-notch data were not obtained, an estimate of these data was made. These data were estimated by the methods of the "U. S. NRC Regulatory Standard Review Plan, Section 5.3.2, Pressure Temperature Limits". 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 3.9×10^{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 (10F °)to the required temperature and the pressure correction is made by subtracting the pressure error (30 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-185-73. When the capsule lead factors are considered, the scheduled removal dates will provide materials data representative of about 10%, 20%, 40%, 60%, and 80% of the actual reactor vessel exposure anticipated during the vessel life.

References

(1) FSAR, Section 4.1.5

(2) Westinghouse Electric Corporation, WCAP-8739

(3) Westinghouse Electric Corporation, WCAP-8743

(4) Westinghouse Electric Corporation, WCAP-8738.

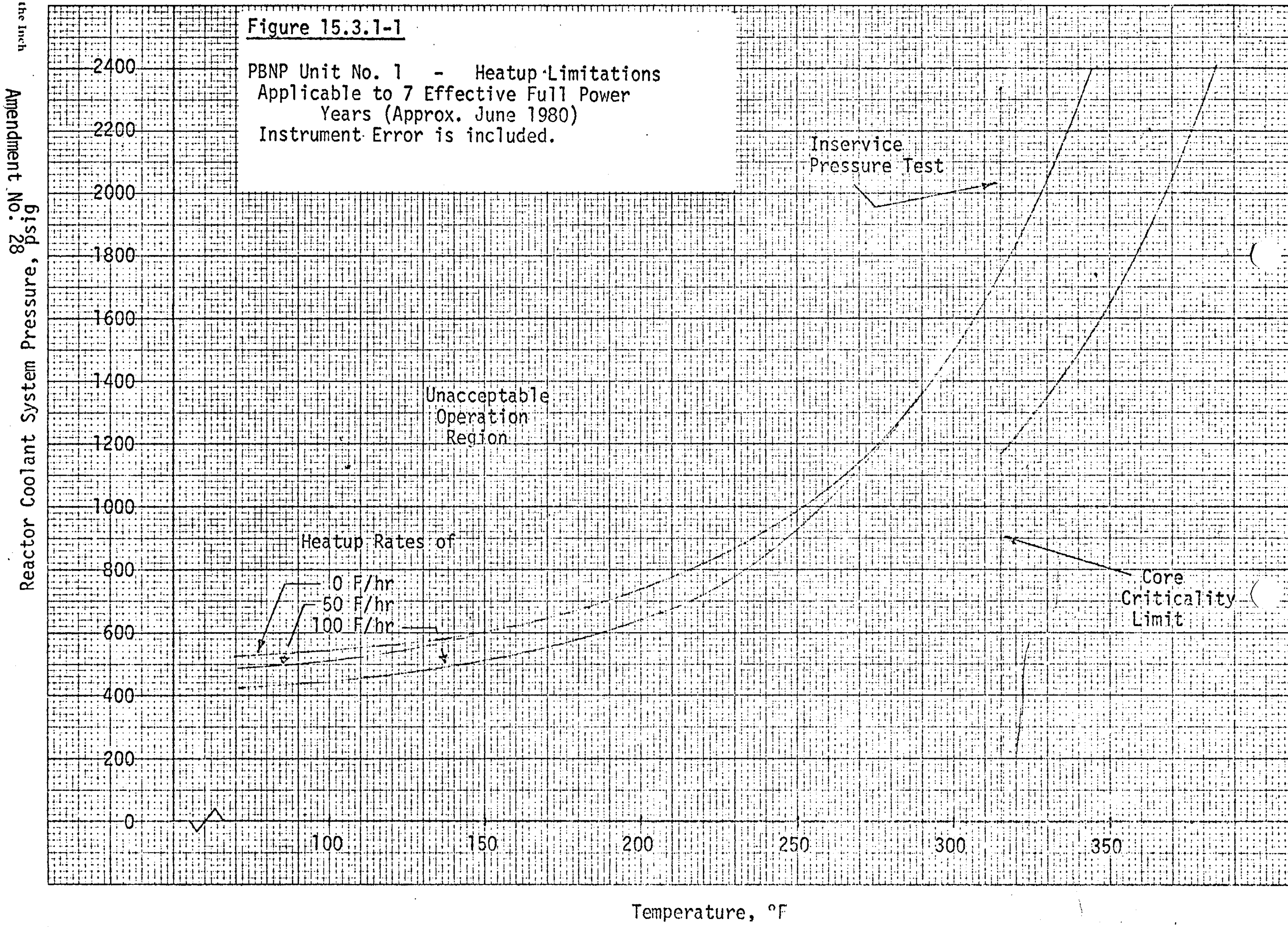
Figure 15.3.1-1
PBNP Unit No. 1 - Heatup Limitations Applicable to 7 Effective Full Power Years (Approx. June 1980) Instrument Error is included.
Inservice Pressure Test
Unacceptable Operation Region
Heatup Rates of
0 F/hr
50 F/hr
100 F/hr
Core Criticality Limit
Reactor Coolant System Pressure, psig
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
100
150
200
250
300
350
Temperature, °F

Amendment No. 28

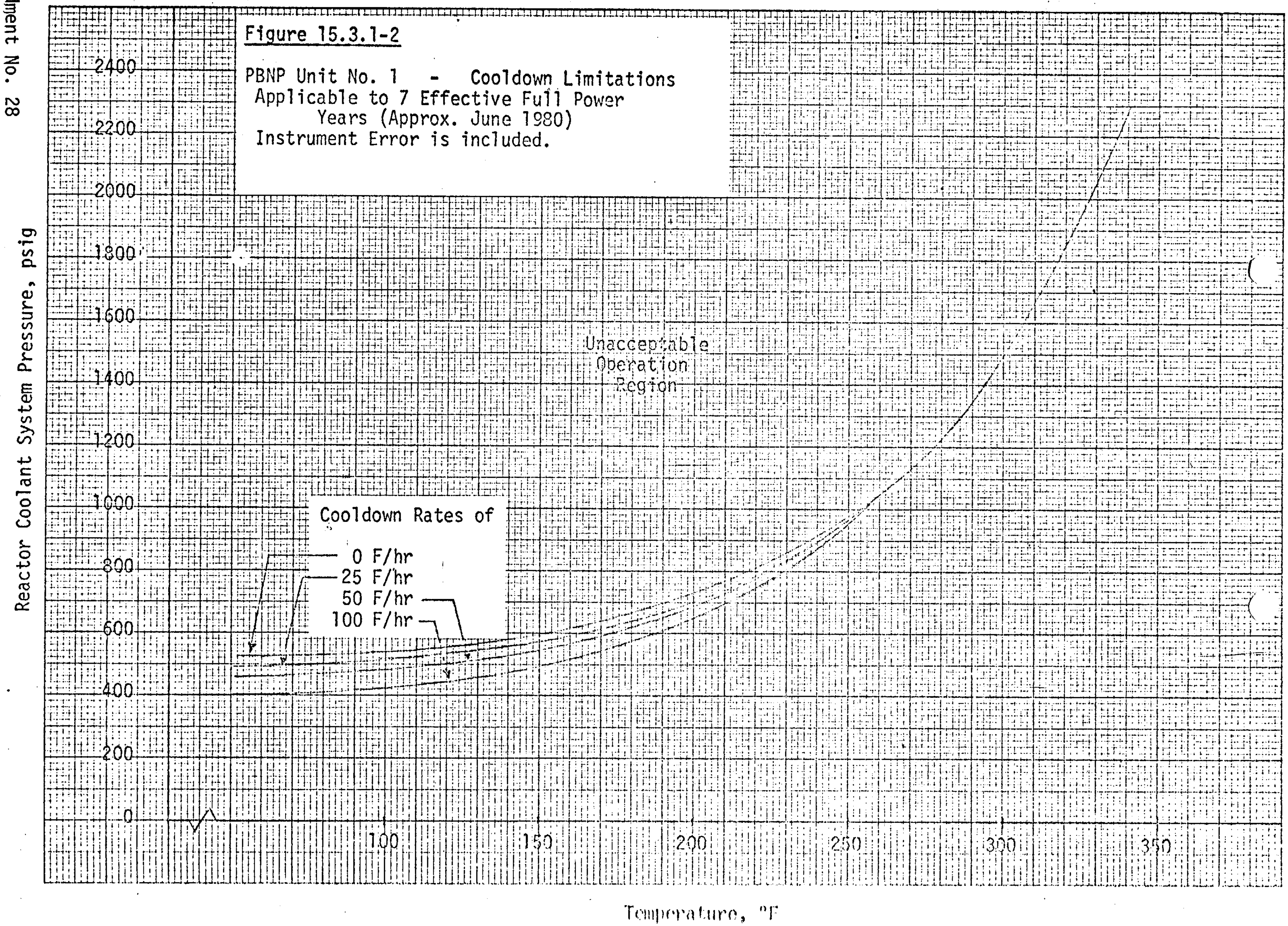

Figure 15.3.1-2

PBNP Unit No. 1 - Cooldown Limitations
Applicable to 7 Effective Full Power Years (Approx. June 1980)
Instrument Error is included.

Amendment No. 28

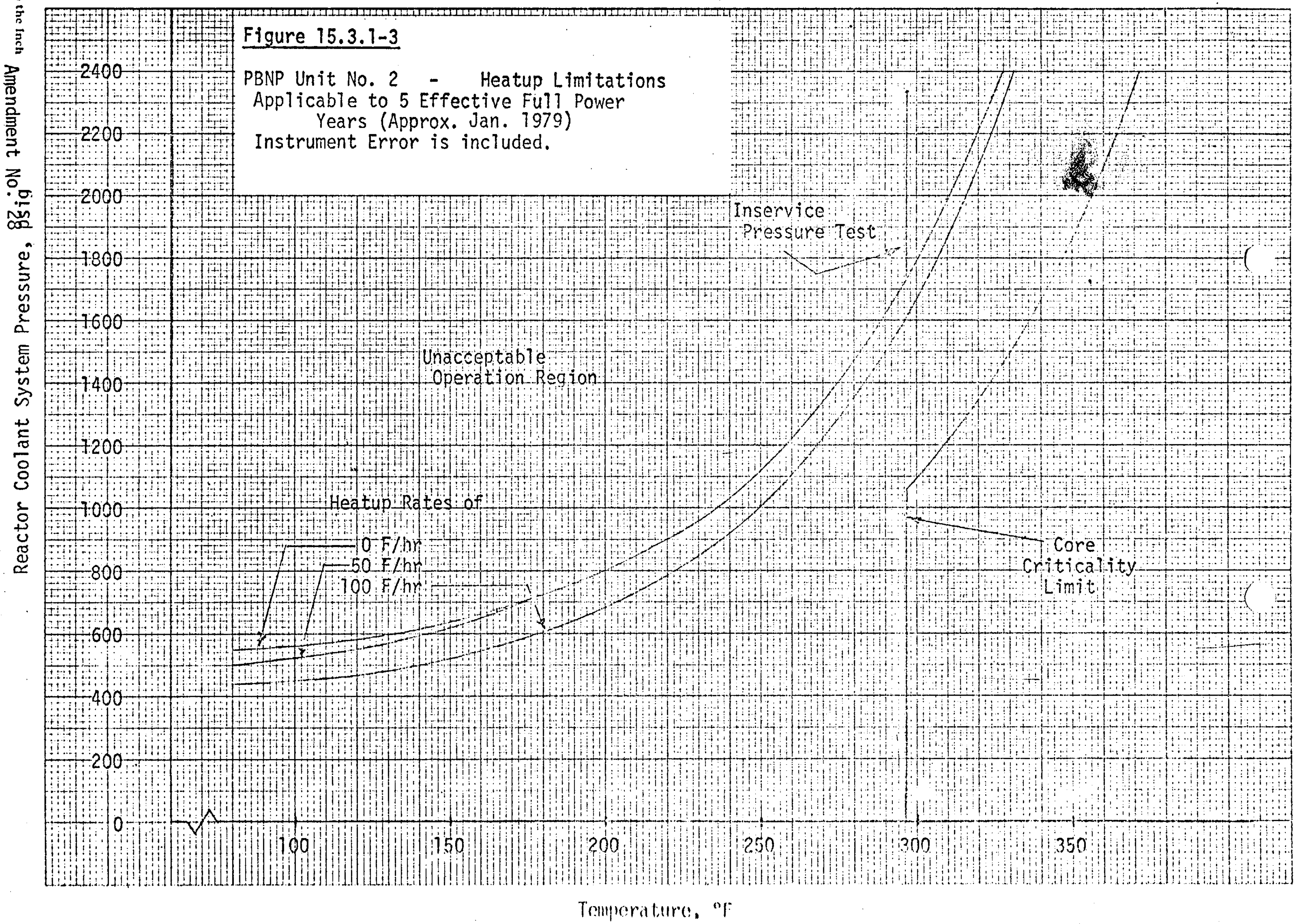

Figure 15.3.1-3

PBNP Unit No. 2 - Heatup Limitations Applicable to 5 Effective Full Power Years (Approx. Jan. 1979) Instrument Error is included.

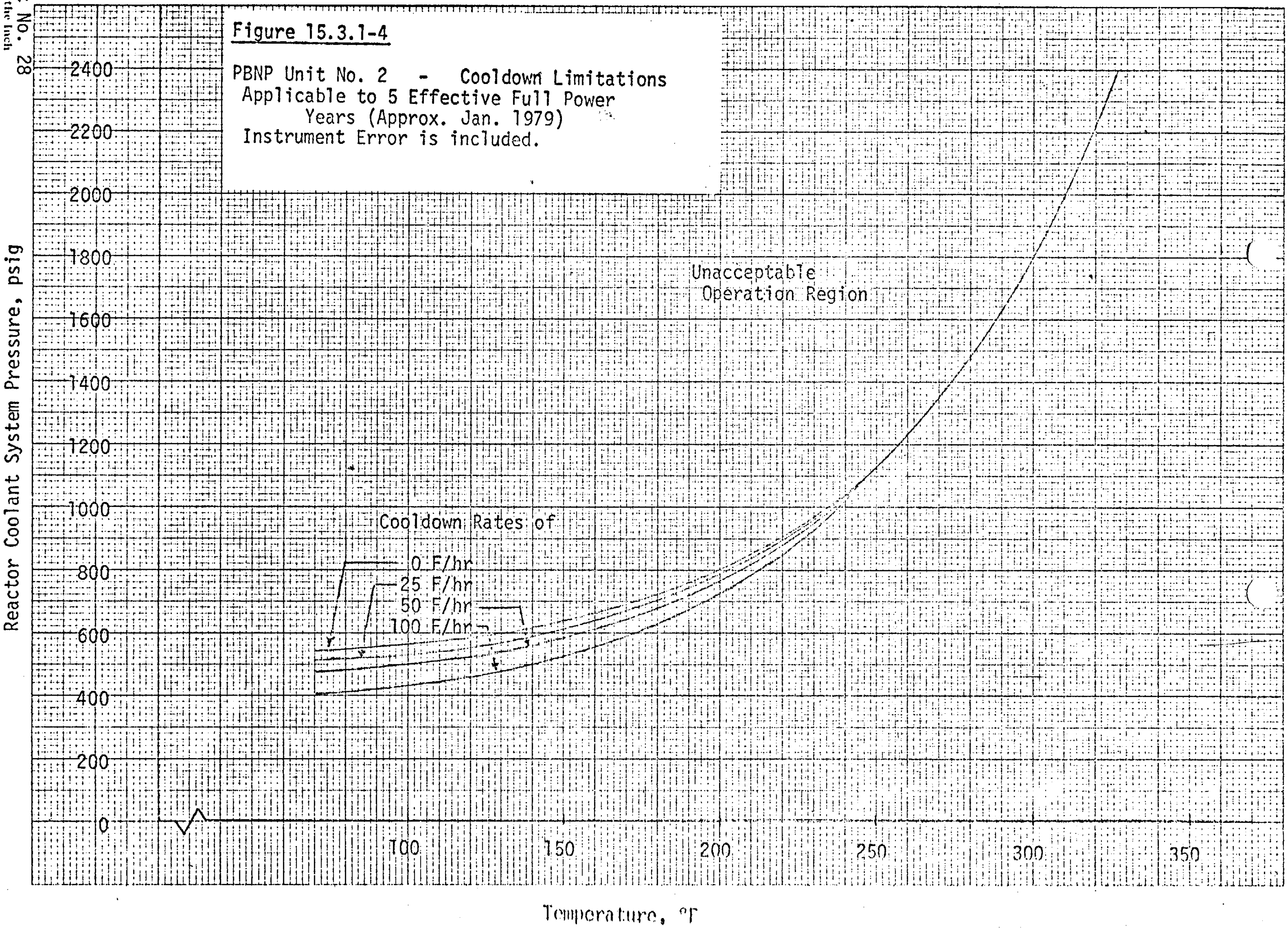

Figure 15.3.1-4

PBNP Unit No. 2 - Cooldown Limitations
Applicable to 5 Effective Full Power Years (Approx. Jan. 1979)
Instrument Error is included.

TABLE 15.3.1 - 1

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

Capsule Letter	Approximate Removal Date*
V	Sept. 1972 (actual)
S	Dec. 1975 (actual)
R	Dec. 1977
T	Nov. 1985
P	Nov. 1989
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	Nov. 1974 (actual)
T	Mar. 1977
R	Mar. 1979
P	Mar. 1986
S	Mar. 1993
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

SUPPORTING AMENDMENTS NOS. 24 AND 28 TO FACILITY LICENSES DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

WISCONSIN MICHIGAN POWER COMPANY

POINT BEACH UNITS NOS. 1 AND 2

DOCKETS NOS. 50-266 AND 50-301

Introduction

By letter dated October 1, 1976, Wisconsin Electric Power Company (WEPCO) requested changes to the Technical Specifications appended to Facility Operating Licenses DPR-24 and DPR-27, for Point Beach Units Nos. 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/], and would add a new specification showing the reactor vessel surveillance capsule withdrawal schedules.

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 a ductile to a brittle fracture mode of failure of a metal test specimen.

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. 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.

Evaluation

The current reactor coolant system pressure-temperature limits were established to cover periods of reactor operation up to about 5.9 EFPY (Effective Full Power Years) for Point Beach Unit No. 1 and about 4.3 EFPY for Unit No. 2. The proposed revised limits would allow reactor operation up to 7 EFPY for Unit No. 1 and up to 5 EFPY for Unit No. 2. The proposed limits were calculated using the methods presented in Appendix G to ASME Code, Section III and the data obtained from the material surveillance programs. The limiting material for both vessels is weld material. The weld material for the Unit No. 1 vessel has 0.20% copper and the Unit No. 2 vessel weld has 0.25% copper. Data from the material surveillance programs show that the Unit No. 1 weld material at a fluence of 3.58×10^{18} n/cm^2 has a shift in RT_{NDT} of 110°F at 30 ft-lbs and 120°F at 50 ft-lbs and that the Unit No. 2 weld material has a shift in RT_{NDT} of 165°F at the 30 ft-lb level for a fluence of 4.7×10^{18} n/cm^2. Since the upper shelf Charpy energy for Unit No. 2 material fell below 50 ft-lbs, no RT_{NDT} shift at this level could be calculated. Fluence on the inner vessel wall is calculated to be $3.9 \times 10^{19} \times$ n/cm^2 for 40 years of operation at 1518 Mwt and 80% load factor.

Based on our review of the proposed changes to the reactor coolant system pressure-temperature limitations, we have concluded that they conform to Appendix G to 10 CFR Part 50; and thus, are acceptable.

Also, based on our review of the proposed addition of the reactor vessel surveillance capsule withdrawal schedules, we have concluded that they conform to 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"; and thus, are acceptable.

Environmental Considerations

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 impact statement, negative declaration and 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: February 24, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY
WISCONSIN MICHIGAN POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 24 and 28 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 amendments are effective as of the date of issuance.

These amendments consist of changes to the Technical Specifications that 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}), and will add a new specification showing the reactor vessel surveillance capsule withdrawal schedule.

The application for the amendments 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendments dated October 1, 1976, (2) Amendment No. 24 to License No. DPR-24, (3) Amendment No. 28 to License No. DPR-27, and (4) 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 University of Wisconsin - Stevens Point Library, ATTN: Mr. Arthur M. Fish, Stevens Point, Wisconsin 54481.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 24 day of February 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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