

March 20, 1996

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Mr. Robert E. Link, Vice President
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Wisconsin Electric Power Company
231 West Michigan Street, Room P379
Milwaukee, WI 53201

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SUBJECT: AMENDMENT NOS. 168 AND 172 TO FACILITY OPERATING LICENSE
NOS. DPR-24 AND DPR-27, POINT BEACH NUCLEAR PLANT,
UNITS 1 AND 2 (TACS M89550 AND M89551)

Dear Mr. Link:

The Commission has issued the enclosed Amendment Nos. 168 and 172 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2. The amendments revise the Technical Specifications (TSs) in response to your application dated May 26, 1994, as supplemented January 5, April 25 and October 12, 1995, and February 2 and March 1, 1996.

The amendments revise the TSs by extending the operation of both units with the current heatup and cooldown limit curves to 23.6 effective full power years. The basis for TS Section 15.3.1.B, "Pressure/Temperature Limits," is also revised to reflect the methodology for the curve compilation.

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

Sincerely,

Original Signed By:

Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-266
and 50-301

- Enclosures: 1. Amendment No. 168 to DPR-24
2. Amendment No. 172 to DPR-27
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

March 20, 1996

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Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room P379
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cc w/encls: See next page

Mr. Robert E. Link, Vice President
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Unit Nos. 1 and 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
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 May 26, 1994, as supplemented January 5, April 25 and October 12, 1995, and February 2 and March 1, 1996, 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.

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

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: March 20, 1996



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN ELECTRIC POWER COMPANY
DOCKET NO. 50-301
POINT BEACH NUCLEAR PLANT, UNIT NO. 2
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
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 May 26, 1994, as supplemented January 5, April 25 and October 12, 1995, and February 2 and March 1, 1996, 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. 172, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately upon issuance. The Technical Specifications are to be implemented within 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Allen G. Hansen, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: March 20, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 168 AND 172
TO FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27
DOCKET NOS. 50-266 AND 50-301

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE

TS 15.3.1-7

TS 15.3.1-8

TS Figure 15.3.1-1

TS Figure 15.3.1-2

INSERT

TS 15.3.1-7

TS 15.3.1-8

TS Figure 15.3.1-1

TS Figure 15.3.1-2

of the vessel is computed to be 2.5×10^{19} neutrons/cm² for 40 years of operation at 1518 Mwt and 80 percent load factor.⁽²⁾ This maximum fluence 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 is as presented in Regulatory Guide 1.99 Revision 2, "Radiation Damage to Reactor Vessel Materials."

(Reference 6)

Once the fluence is determined, the adjusted reference temperature used in revising the heatup and cooldown curves is obtained by utilizing the method in Section 1.1 of Regulatory Guide 1.99 Revision 2 (Reference 6) for the limiting weld material of both Unit 1 and Unit 2.

The heatup and cooldown curves presented in Figure 15.3.1-1 and 15.3.1-2 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.

The regulations governing the pressure-temperature limits (10 CFR 50 - Appendix G and ASME Code Section III - Appendix G) do not require additional margins for instrumentation uncertainties be added to the heatup and cooldown curves. This is because the inclusion of instrumentation uncertainties, in addition to other conservatisms in the methods for calculating the pressure temperature limits, is not necessary to protect the vessel from damage.

Unit 1 - Amendment No. ~~24, 53, 98, 125, 168~~
Unit 2 - Amendment No. ~~57, 59, 102, 129, 172~~

15.3.1-7

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 Figure 15.3.1-1 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 320F°. 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 have been developed based upon the requirements of the Code of Federal Regulations, Title 10, Part 50, Appendix H, and with consideration of ASTM Standard E-185-82. When the capsule lead factors are considered, the scheduled removal dates accommodate the weld data needs of all the participants in the Babcock and Wilcox Master Integrated Reactor Vessel Surveillance Program. Additionally, the schedule will provide plate/forging material data as well as fluence data corresponding to the expiration of the current licenses and of any future license extensions.

References

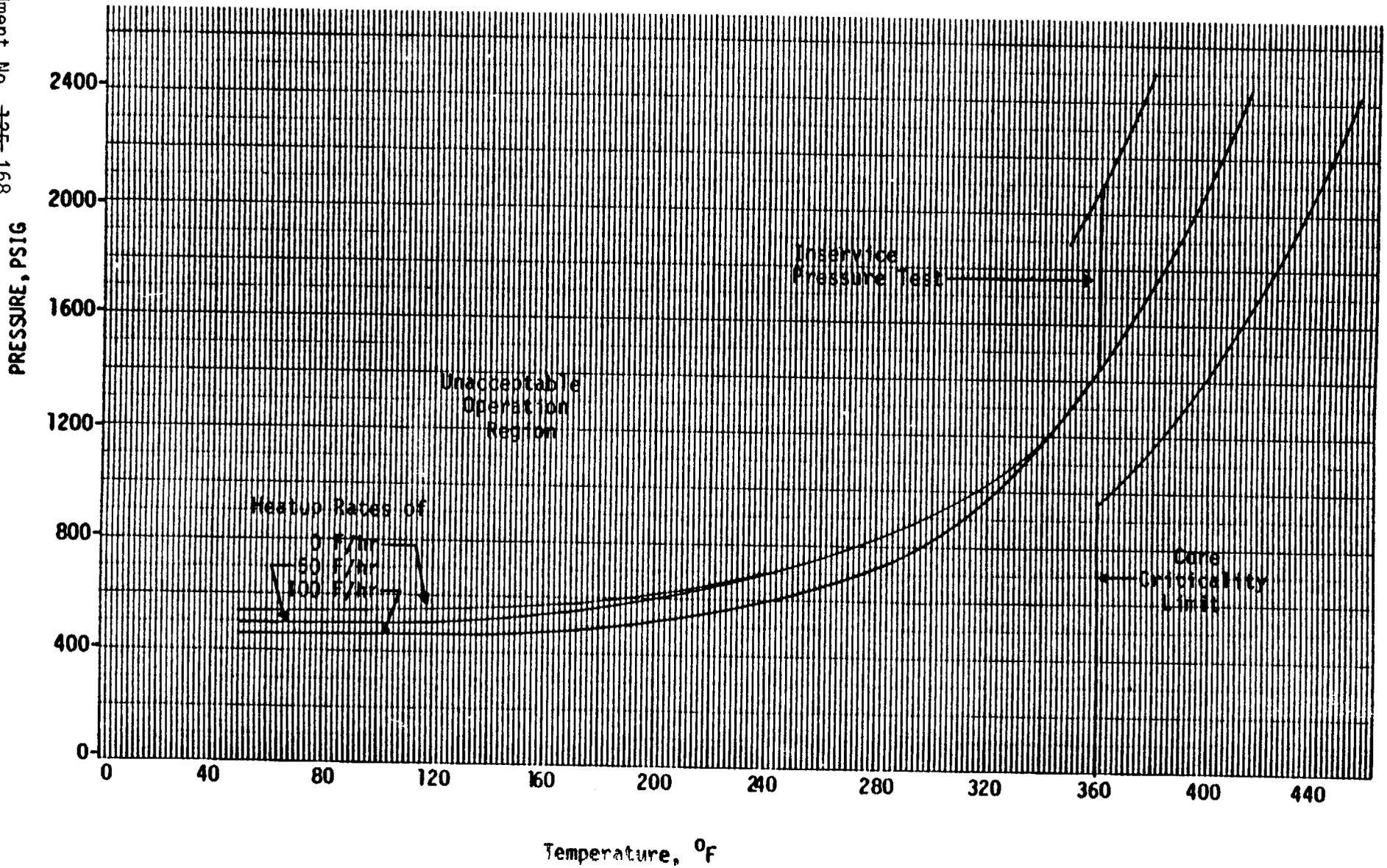
- (1) FSAR, Section 4.1.5
- (2) Westinghouse Electric Corporation, WCAP-12794, Rev. 2/12795, Rev. 2
- (3) Westinghouse Electric Corporation, WCAP-8743
- (4) Westinghouse Electric Corporation, WCAP-8738
- (5) Babcock & Wilcox, BAW 1803
- (6) Regulatory Guide 1.99, Revision 2

Unit 1 - Amendment No. ~~24,98,125,131,168~~ 15.3.1-8

Unit 2 - Amendment No. ~~57,102,128,129,135,172~~

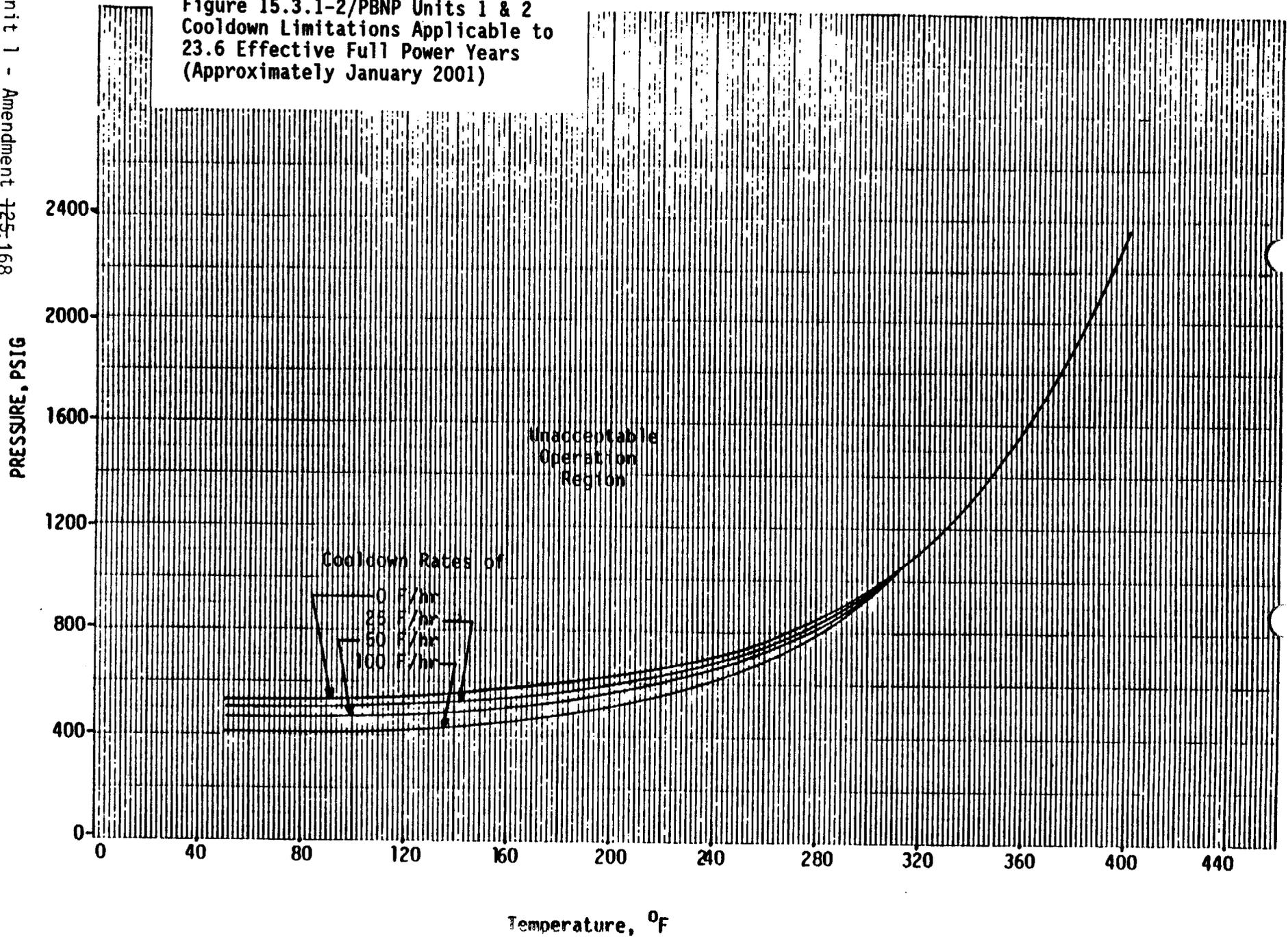
Unit 1 - Amendment No. 125, 168
Unit 2 - Amendment No. 129, 172

Figure 15.3.1-1/PBNP Units 1 & 2
Heatup Limitations Applicable to
23.6 Effective Full Power Years
(Approximately January 2001)



Unit 1 - Amendment 125, 168
Unit 2 - Amendment 129, 172

Figure 15.3.1-2/PBNP Units 1 & 2
Cooldown Limitations Applicable to
23.6 Effective Full Power Years
(Approximately January 2001)





UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 168 AND 172 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

1.0 INTRODUCTION

By letter dated May 26, 1994, as supplemented January 5, April 25 and October 12, 1995, and February 2 and March 1, 1996, the Wisconsin Electric Power Company (the licensee) submitted a request for revision to the Point Beach Nuclear Plant, Units 1 and 2, Technical Specifications (TSs). The requested amendments would extend the applicability of the pressure-temperature (P-T) limits curves in the TSs from 18.1 effective full power years (EFPY) to 23.6 EFPY. The licensee also proposed that the curves not be changed, stating that the curves have enough margins for reactor operation up to 23.6 EFPY based on neutron fluence reduction. The supplemental submittals provided additional information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Background

To protect reactor vessels from brittle fracture, the NRC requires licensees to use P-T limits for the operation of the reactor coolant system to limit loads applied to the reactor vessel and the rate of vessel material embrittlement. The P-T limits are constructed using an adjusted reference temperature (ART) of the vessel material and applied loads to limit pressures and temperatures during normal operation (in accordance with Appendix G to 10 CFR Part 50).

The ART is a measure of the embrittlement of reactor vessel materials caused by neutron irradiation. Regulatory Guide 1.99, Revision 2, defines the ART as the sum of the initial nil-ductility transition reference temperature (RT_{ndt}) of the material, the increase in RT_{ndt} caused by neutron irradiation, and a margin to account for uncertainties in the calculation. The increase in RT_{ndt} is calculated from the product of a chemistry factor and a neutron fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the vessel material. Therefore, the rate of material embrittlement increases as the neutron fluence, copper content, and nickel content increase.

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2.2 Fluence Determinations

The Point Beach units are each equipped with a thermal shield. The dosimetry surveillance capsules which were previously analyzed were located behind the thermal shield with respect to the direction of neutron propagation. A cavity dosimetry program was implemented for both units using fast neutron flux which has traversed the thermal shield and the pressure vessel. Measurements obtained from these dosimetry programs were subject to the underestimation of the (Evaluated Nuclear Data File) ENDF/B-IV cross sections. ENDF/B-IV based cross sections had an error in the inelastic scattering of iron which resulted in an underestimation of the fast neutron flux, whenever such flux traversed a significant thickness of iron such as the thermal shield or the pressure vessel. This discrepancy has been corrected in the sets of cross sections based on ENDF/B-VI.

WCAP-12795, Revision 3, documents a reestimation of the capsule dosimetry and the cavity dosimetry data for Unit 2 using cross sections based on ENDF/B-VI. In addition, the calculations in WCAP-12795 use a benchmarked version of the DOT code, use the P_3 scattering approximation and the S_8 quadrature approximation. The neutron sources were estimated on a pin-wise basis and accounted for the plutonium buildup, including the spectral effects. The assumptions and methods are consistent with accepted industry practice and state of the art, and are therefore acceptable.

The pressure vessel critical element with respect to pressurized thermal shock is the peripheral weld SA-1484. The licensee now estimates that the $E > 1.0$ MeV fluence to the SA-1484 weld will not reach the 2.05×10^{19} n/cm² level until after 23.6 EFPYs. Estimates based on the results of the surveillance capsules and the reactor cavity measurements support this conclusion. Therefore, the staff finds it acceptable.

The licensee did not submit a revision to WCAP-12794, Revision 2, which deals with the Unit 1 dosimetry data. However, the P-T curves are the same for both units. In addition, the same measures have been implemented in both units regarding fluence reduction. Therefore, it is reasonable to conclude that the P-T curves are applicable for Unit 1 as well. In addition, the licensee stated that in the last refueling outage neutron dosimetry was removed from Unit 1, and the measurements were analyzed using ENDF/B-VI cross sections. The licensee noted in the March 1, 1996, submittal that the dosimetry analysis confirmed that the fast neutron fluence level of 2.05×10^{19} neutrons/cm² (n/cm²) for the Unit 1 SA-1101 weld will not be reached until after 23.6 EFPYs (about January 2001). The staff finds this acceptable. The dosimetry data should be kept available for staff audit.

2.3 Adjusted Reference Temperature

In 1989, the licensee implemented a low-low leakage pattern core with hafnium inserts in the guide tubes of peripheral assemblies to reduce neutron fluence in each unit. With the low leakage cores and hafnium inserts, the neutron flux has been reduced compared to previous core loading patterns. Consequently, the rate of irradiation embrittlement of reactor vessel materials has been reduced over what was projected in the current P-T limits,

which the NRC approved on January 10, 1990. For the current limits, the licensee calculated a limiting (maximum) ART of 258.4°F based on a fluence of 2.05E19 n/cm² at the 1/4 location of the vessel beltline thickness (the 1/4T location). The ART was calculated using the material data of weld SA-1484 in the Unit 2 reactor vessel.

In a discussion with the licensee on October 31, 1994, the licensee indicated that for the proposed amendment, the fluence of 2.05E19 n/cm² was at the inside surface instead of at the 1/4T location of the vessel wall. This is a conservative assumption because the fluence at the inside surface is greater than the fluence at the 1/4T location. The fluence at the 1/4T location is the value used in the P-T curve calculation since the calculation assumes a 1/4T deep flaw.

For the proposed amendment, the staff recalculated the ART for each beltline material in the Unit 1 and Unit 2 reactor vessels. The ARTs were compared to the limiting ART in the current P-T limits to verify whether the current P-T curves have sufficient margins. The staff used the material data that the licensee submitted under Generic Letter (GL) 92-01, which the NRC issued on March 6, 1992. GL 92-01 requested licensees to submit information on reactor vessel materials. In response to GL 92-01, Wisconsin Electric submitted reactor vessel material data for both Point Beach units on June 25, 1992, with supplements on July 30, 1992, May 21, 1993, May 2, 1994, and June 27, 1994.

Using the material data in Wisconsin Electric's response to GL 92-01 and the fluence of 2.05E19 n/cm², the staff calculated a limiting ART of 240.3°F at the 1/4T location based on material data of weld SA-1101 in the Unit 1 reactor vessel. (It should be noted that the material used in the current P-T curves, weld SA-1484 in the Unit 2 reactor vessel, is no longer limiting because Wisconsin Electric has updated the material data since the current P-T curves were approved in 1990.) For the proposed amendment, the staff calculated the highest ART based on the material data of Unit 1 weld SA-1101. Therefore, weld SA-1101 is the limiting material. The staff's calculated ART of 240.3°F is less than the ART of 258.4°F in the current P-T limits. This shows that the current P-T limits have a sufficient margin (18.1°F) and the curves do not need to be revised for the proposed amendment.

2.4 Summary

The staff concludes that the current P-T limit curves for heatup, cooldown, criticality, and inservice pressure test are valid up to 23.6 EFPY. The P-T limits were calculated in accordance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99, Revision 2. Hence, the proposed changes to the P-T limits may be incorporated into the Point Beach, Units 1 and 2, Technical Specifications.

2.5 Technical Specification and Bases Changes

The length of time and date of applicability for Figures 15.3.1-1 and 15.3.1-2 are being changed to 23.6 EFPY and January 2001. These changes are consistent with the licensee's submittal and with the staff's evaluation, and are therefore acceptable. The bases for TS 15.3.1 are also being modified. The

computed maximum integrated fast neutron exposure of the vessel is being changed to $2.5E19$ n/cm² for 40 years of operation. In addition, editorial changes are proposed. The staff agrees with the licensee that these changes are consistent with the TS changes.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent 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 (59 FR 37093). 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.

5.0 CONCLUSION

The staff has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Lois
J. Tsao

Date: March 20, 1996