November 2, 198_

Docket No. 50-263

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DHagan	BGrimes
MJVirgilio	

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-22: MISCELLANEOUS CHANGES (TACS 69146/71517/72861)

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated March 31, 1989.

The amendment would:

(1) revise the reactor vessel pressure vs. temperature limit curves to meet the staff positions of Regulatory Guide 1.99, Revision 2;

(2) add a new requirement for augmented inservice inspection of piping susceptible to intergranular stress corrosion cracking; and

(3) revise the requirements for Type A containment integrated leak rate testing to permit the use of the mass point method.

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by

William O. Long, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 72 to License No. DPR-22
2. Safety Evaluation
3. Notice of Issuance

OFOI 1/1 CP-1

cc w/enclosures:
See next page

*See previous concurrence

*LA/PD31:DRSP	*PM/PD31:DRSP	*BC:EMCB	*(A)D/PD31:DRSP	*OGC
RIngram	WLong	CYCheng	JThoma	
9/15/89	9/15/89	10/3/89	10/4/89	10/10/89

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RIngram	WLong	CYCheng	JThoma	BMB
9/15/89	9/15/89	10/3/89	10/4/89	10 9/61

subject to prior issuance of EA

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

November 2, 1989

Docket No. 50-263

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(3) revise the requirements for Type A containment integrated leak rate testing to permit the use of the mass point method.

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William O. Long

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Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

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3. Notice of Issuance

cc w/enclosures:
See next page

Mr. T. M. Parker, Manager
Northern States Power Company

Monticello Nuclear Generating Plant

cc:

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Shaw, Pittman, Potts and
Trowbridge
2300 N Street, NW
Washington, D. C. 20037

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Box 1200
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
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Minneapolis, Minnesota 55440

O. J. Arlien, Auditor
Wright County Board of
Commissioners
10 NW Second Street
Buffalo, Minnesota 55313

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated March 31, 1989, 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 2.C.2 of Facility Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 72, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John O. Thoma

John O. Thoma, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 72

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

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

REMOVE	INSERT
v	v
122	122
133	133
134	134
135	135
136	136
145	145
146	146
159	159
-	229ff (new page)

LIST OF FIGURES

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing, the reactor vessel shell temperatures specified in 4.6.B.1, except for the reactor vessel bottom head, shall be at or above the temperatures shown on the two curves of Figure 3.6.2, where the dashed curve, "RPV Core Beltline," is increased by the core beltline temperature adjustment from Figure 3.6.1. The reactor vessel bottom head temperature shall be at or above the temperatures shown on the solid curve of Figure 3.6.2, "RPV Remote from Core Beltline," with no adjustment from Figure 3.6.1.

2. During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

4.0 SURVEILLANCE REQUIREMENTS

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing when the vessel pressure is above 312 psig, the following temperatures shall be recorded at least every 15 minutes.

   a. Reactor vessel shell adjacent to shell flange.

   b. Reactor vessel bottom head.

   c. Reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region.

2. Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM E 185-66. Samples shall be withdrawn at one fourth and three fourths service life. Analysis of the first sample shall include a quantitative determination of the material chemistries. (Note: Analysis of the first sample has been completed. The Figure 3.6.1 core beltline temperature adjustment curve reflects the chemistry data obtained).

3. Neutron flux wires shall be installed in the reactor vessel adjacent to the reactor vessel wall at the core midplane level. The wires shall be removed and tested during the first refueling outage to experimentally verify the calculated value of neutron fluence at one fourth of the beltline shell thickness that is used to determine the NDTT shift from Figure 3.6.1.

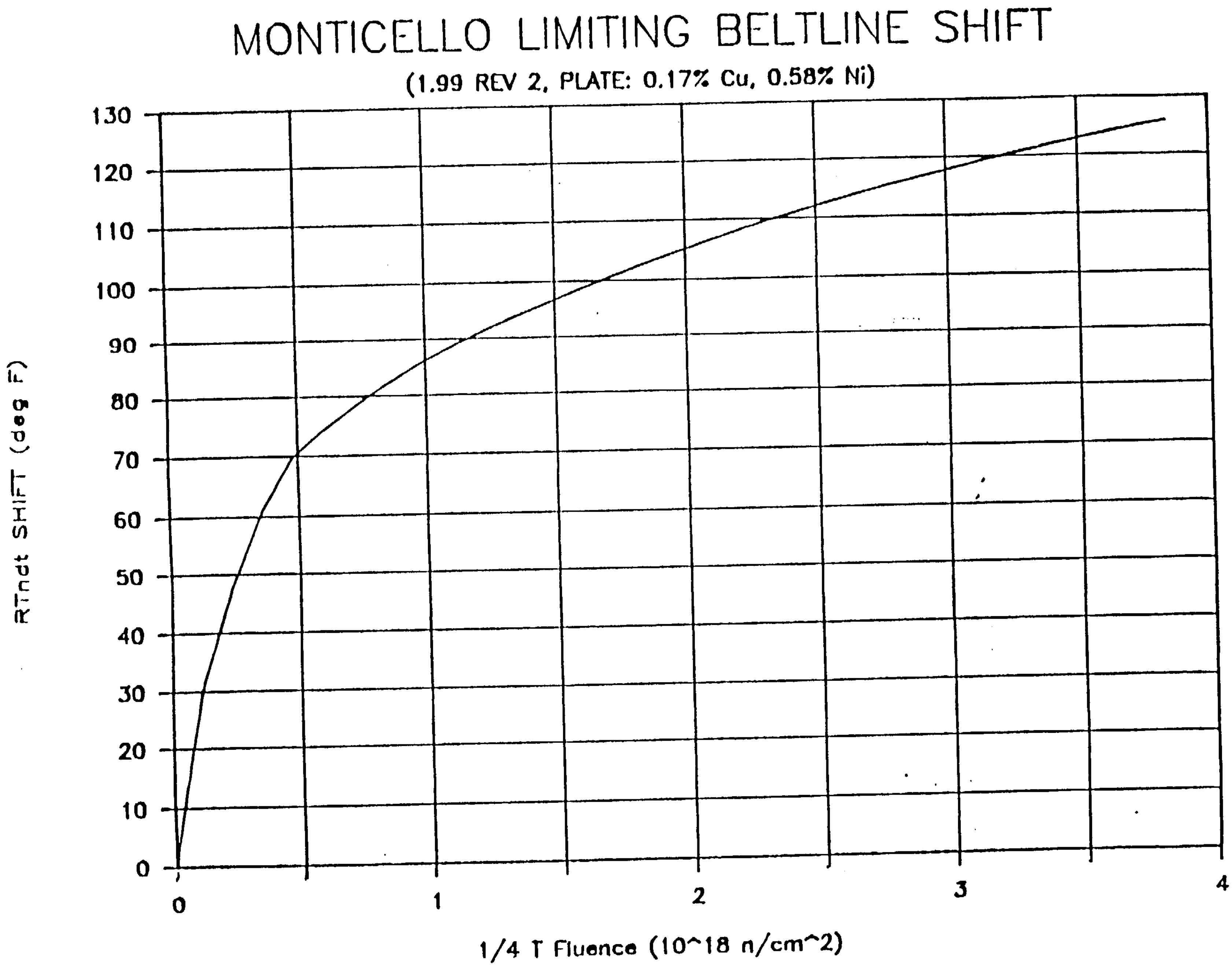

Figure 3.6.1 Core Beltline Operating Limits Curve Adjustment vs. Fluence

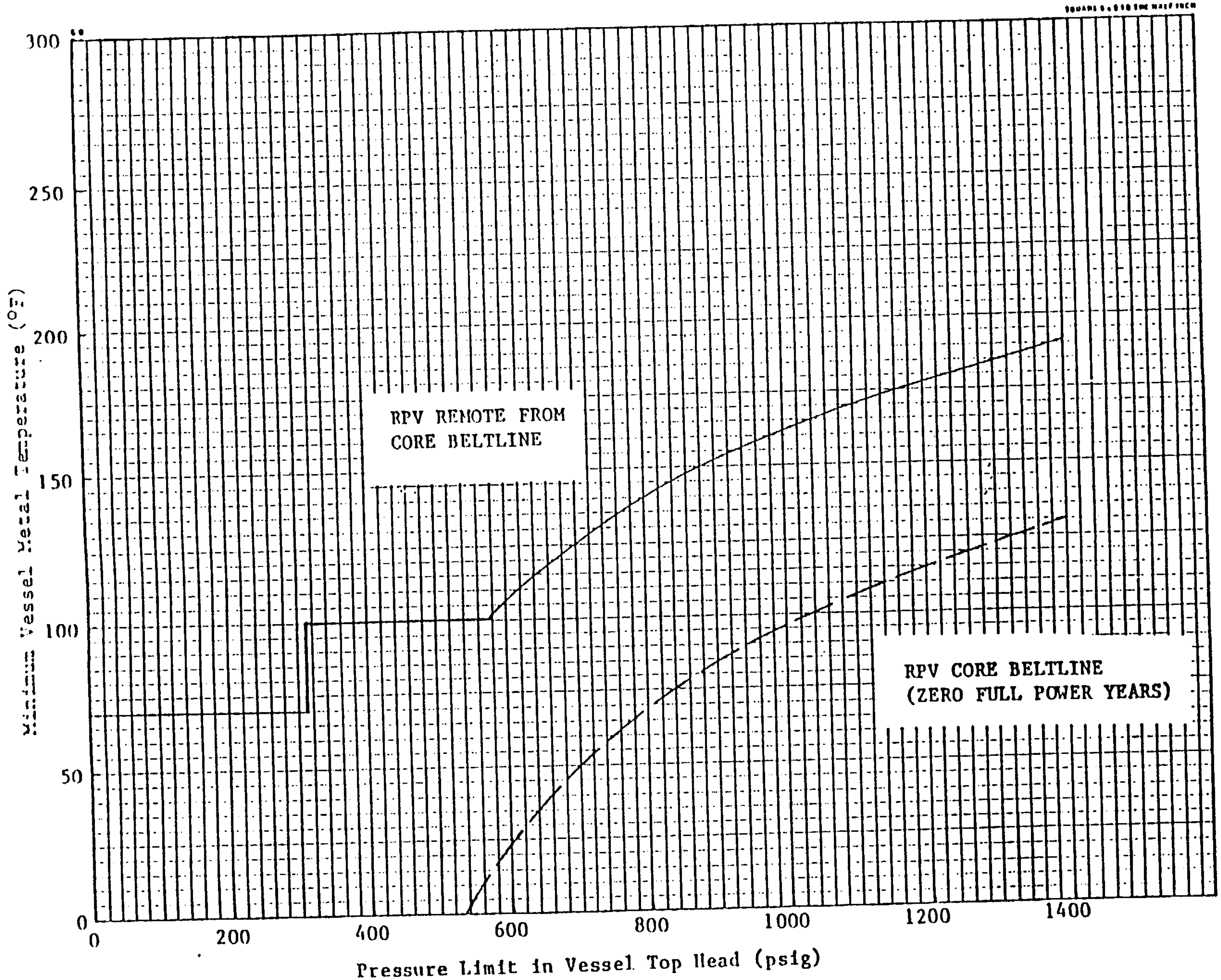

Figure 3.6.2 Minimum Temperature vs. Pressure for Pressure Tests

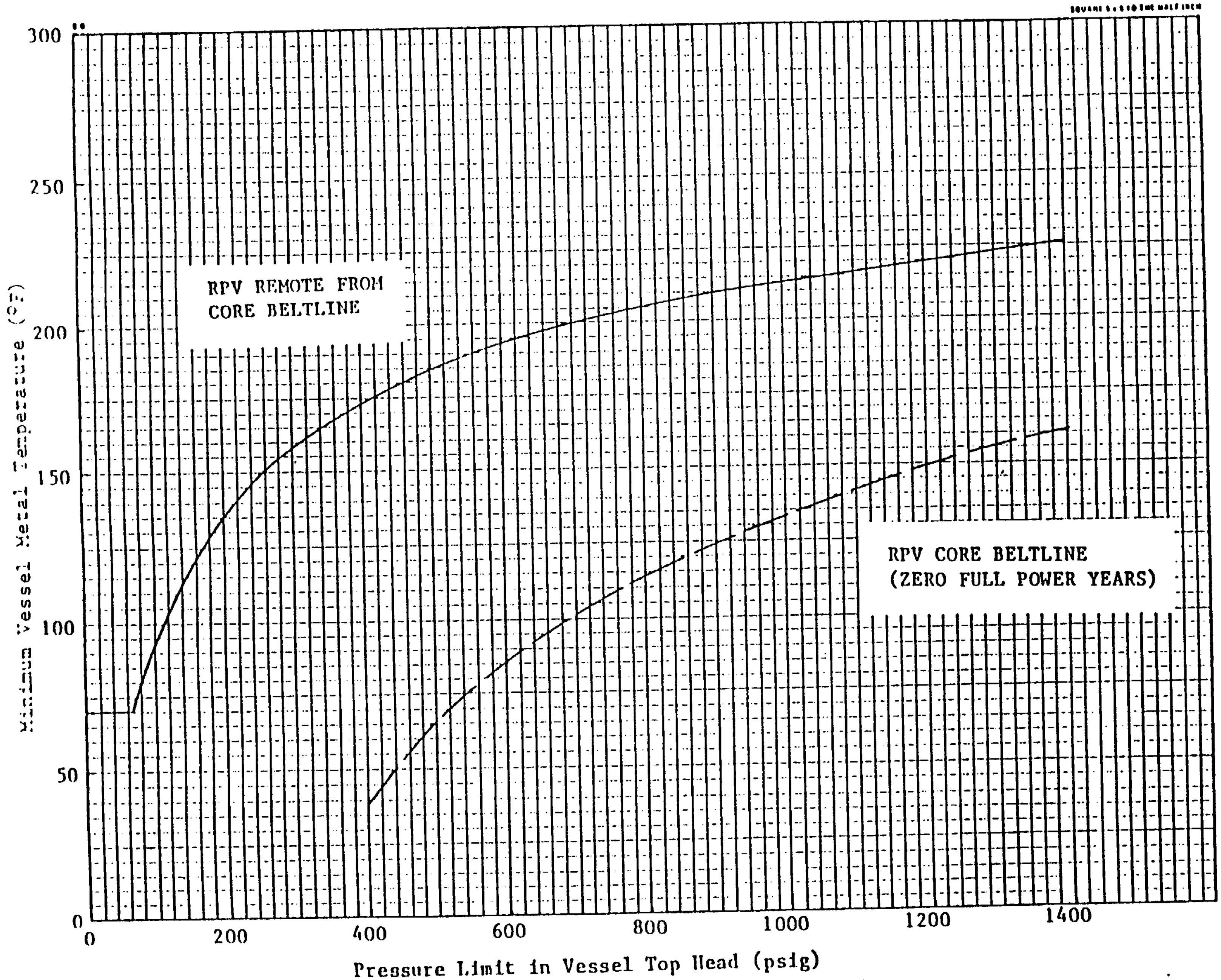

Figure 3.6.3 Minimum Temperature vs. Pressure for Mechanical Heatup or Cooldown Without the Core Critical

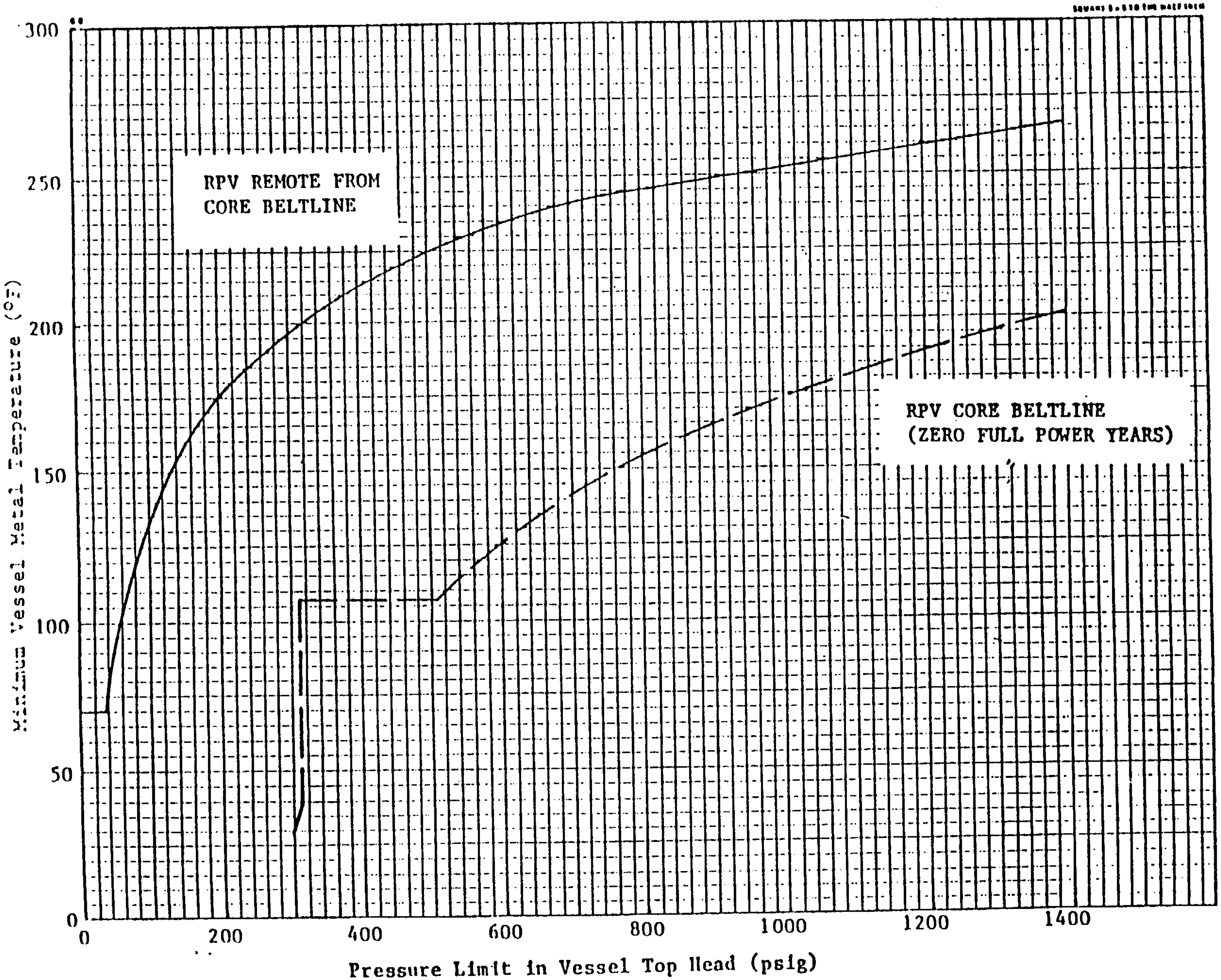

Figure 3.6.4 Minimum Temperature vs. Pressure for Core Operation

Amendment No. 72

Bases 3.6 and 4.6:

A. Reactor Coolant Heatup and Cooldown

The vessel has been analyzed for stresses caused by thermal and pressure transients. Heating and cooling transients throughout plant life at uniform rates of 100°F per hour were considered in the temperature range of 100 to 546°F and were shown to be within the requirements for stress intensity and fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code.

During reactor operation, the temperature of the coolant in an idle recirculation loop is expected to remain at reactor coolant temperature unless it is valved out of service. Requiring the coolant temperature in an idle loop to be within 50°F of the reactor coolant temperature before the pump is started assures that the change in coolant temperature at the reactor vessel nozzles and bottom head region are within the conditions analyzed for the reactor vessel thermal and pressure transients.

During hydrostatic pressure testing, a coolant heatup or cooldown of 20°F in any one-hour period has a negligible effect on the reactor operating limits of Figure 3.6.2.

B. Reactor Vessel Temperature and Pressure

Operating limits on the reactor vessel pressure and temperature during normal heatup and cooldown and during inservice hydrostatic testing were established using 10 CFR Part 50, Appendix G, May 1983 and Appendix G of the Summer 1976 or later Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These operating limits assure that a large postulated surface flaw, having a depth of 0.24 inches at the flange-to-vessel junction and one-quarter of the material thickness at all other reactor vessel locations and discontinuity regions can be safety accommodated. For the purpose of setting these operating limits the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with requirements of the Code to which this vessel was designed and manufactured (1965 Edition including Summer 1966 Addenda).

A General Electric Company procedure, designed to evaluate fracture toughness requirements for older plants where information may be incomplete, was used to estimate RT_{NDT} values on an equivalent basis to the new requirements for plants which have construction permits after August 15, 1973.

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the reactor pressure vessel. Two types of information are needed in this analysis: 1) A relationship between the changes in fracture toughness of the reactor pressure vessel steel and the neutron fluence (integrated neutron flux), and 2) A measure of the neutron fluence at the point of interest in the reactor pressure vessel wall.

The relationship of predicted adjustment of reference temperature versus fluence and the copper and nickel content of the core beltline materials given in Regulatory Guide 1.99, Revision 2, was used to define the core beltline temperature adjustment versus fluence shown on Figure 3.6.1.

A relationship between full power years of operation and neutron fluence has been experimentally determined for the reactor vessel. The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown following nuclear shutdown), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which were installed adjacent to the vessel wall were removed to experimentally determine the neutron fluence versus full power years of operation. This experimental result was updated by testing additional dosimetry removed with the first surveillance capsule.

Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. An analysis and report will be submitted to the Commission on all such surveillance specimens removed from the reactor vessel in accordance with 10 CFR 50, Appendix H, including information obtained on the level of integrated fast neutron irradiation received by the specimens and actual vessel material.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

b. When Primary Containment Integrity is required, leakage rates shall be limited to:

1. An overall integrated leakage rate of less than or equal to La, 1.2 percent by weight of the containment air per 24 hours at Pa, 42 psig.

2. A combined leakage rate of less than or equal to 0.6La for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to Pa.

3. Less than or equal to 11.5 scf per hour for any one main steam isolation valve when tested at 25 psi.

With the measured overall integrated primary containment leakage rate exceeding 0.75La, or the measured combined leakage rate for all penetrations and valves, except main steam isolation valves, subject to Type B and C testing exceeding 0.6La, or the measured leak rate exceeding 11.5 scf per hour for any one main steam isolation valve, restore leakage rates to less than or equal to these values prior to increasing reactor coolant system temperature above 212°F or, alternatively, restore measured leakage rates to within these limits within one hour or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

b. The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods, and provisions of 10 CFR Part 50:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at 40±10 month intervals* during shutdown at ≥Pa during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If any periodic Type A test fails to meet 0.75La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75La, at which time the above test schedule may be resumed.

3. All Type A test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

*The second test of the second 10-year service period may be conducted during the 1989 refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	2. Welds in austenitic stainless steel piping four inches or larger in diameter containing reactor coolant at a temperature above 200 degrees F during power operation, including reactor vessel attachments and appurtenances, shall be included in an augmented inspection program meeting the requirements of Generic Letter 88-01.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated March 31, 1989, Northern States Power Company (the licensee) requested an amendment to the Technical Specifications appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment would:

(1) revise the reactor vessel pressure vs. temperature (P/T) curves for consistency with Revision 2 of Regulatory Guide (RG) 1.99;

(2) add requirements for augmented inservice inspection (ISI) of piping susceptible to intergranular stress corrosion cracking (IGSCC); and

(3) revise the requirements for the periodic Type A containment integrated leak rate test (CILRT) to permit the use of the mass point test method.

A discussion of the proposed changes and the NRC staff's evaluation and findings relative to each are addressed in Section 2 of this Safety Evaluation.

2.0 DISCUSSION AND EVALUATION

2.1.1 Revised Pressure/Temperature Limits

Pressure/temperature limits are included in facility technical specifications for the purpose of precluding conditions conducive to brittle fracture of reactor coolant system (RCS) materials. The fracture toughness of RCS materials is a function of the material chemistry and decreases as irradiation accumulates. Revision 2 of Regulatory Guide 1.99, issued May 1988, specifies a more accurate and more conservative means than previously used for predicting the effects of irradiation damage on RCS materials. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations" requested licensees of operating reactors to reanalyze their P/T limits using the revised criteria. In response to Generic Letter 88-11, the licensee applied to revise the P/T limits in the Monticello Nuclear Generating Plant Technical Specifications, Section 3.6. The NRC staff evaluated the proposed changes using the following NRC regulations and guidance: Appendices

G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11. Appendices G and H of 10 CFR Part 50 define specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. SRP Section 5.3.2 describes an acceptable method for constructing the P/T limits. Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline. These tests define the extent of vessel embrittlement at the time of surveillance specimen capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE).

All surveillance capsules contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal. The licensee has removed one surveillance capsule. The results from capsule 117C 3991 G-1 were reported in a Battelle-Columbus Laboratories Report BCL-585-84-2, Revision 1.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 140 degrees F. for the limiting plate material (I-15) at 32 effective full power years (EFPY) at 1/4T (T=reactor vessel thickness) in the Monticello beltline. The ART was calculated using Section 1 of RG 1.99, Rev. 2, because only one surveillance capsule has been withdrawn from the Monticello reactor pressure vessel. The NRC staff performed a similar calculation and verified the licensee's ART to be conservative (see Table 1). Substituting the ART of 140 degrees F. into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 degrees F for normal operation and by 90 degrees F. for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 10 degrees F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Based on data from surveillance capsule 117C 3991 G-1 withdrawn at 7.63 EFPY, the lowest measured irradiated Charpy USE of the material tested is 109 ft-lb for the beltline plate I-15. To estimate the USE at 32 EFPY, the staff subtracted the fluence to which the surveillance capsule was exposed, 2.9E17 n/cm^2, from the fluence at 32 EFPY, 5.1E18 n/cm^2. The staff then referred to Figure 2 of RG 1.99, Rev. 2, for the predicted decrease in USE and calculated that the USE, at 32 EFPY, for the I-15 beltline plate material would be 85 ft-lb. This value is greater than 50 ft-lb and, therefore, is acceptable.

In addition to revising the P/T limit curves to meet RG 1.99, Rev. 2 criteria, the licensee also proposes that, during pressure tests, the reactor vessel bottom head temperature be monitored separately from the beltline region. This would facilitate pressure testing by reducing the amount of non-nuclear heatup required for pressure testing. In order for this to be acceptable, it is necessary that operators have the capability to monitor the beltline region temperature separately. The licensee has advised the staff that this capability is provided by redundant resistance temperature detectors (RTDs).

The staff concludes that the proposed P/T limits for the RCS for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Monticello Technical Specifications.

2.1.2 Augmented Inservice Inspection for IGSCC

Generic Letter 88-01 requested licensees to describe their plans for replacement, inspection, repair and leakage detection of piping susceptible to intergranular stress corrosion cracking (ISGCC). Among the items specified to be included in the licensees' responses to Generic Letter 88-01 is an application to change the Technical Specifications to include a requirement that the Inservice Inspection Program for piping covered by the scope of the Generic Letter be in conformance with the staff positions on schedule, methods and personnel, and sample expansion. The licensee's March 31, 1989 application proposed to invoke NUREG-0313, Revision 2, as a reference for the augmented ISI requirements. In a letter dated September 27, 1989, the licensee revised the application to cite Generic Letter 88-01 as the reference for the new augmented ISI requirements. The use of Generic letter 88-01 as the reference is consistent with the Model Technical Specifications provided in Generic Letter 88-01 and is acceptable.

(Note: The NRC staff will issue a separate evaluation of the licensee's complete response to Generic Letter 88-01 in the near future. This change serves only to implement that portion of the generic letter relating to Technical Specification requirements for augmented ISI.)

2.1.3 CILRT Test Method

The proposed amendment would change Technical Specification 4.7.A.2.b to delete the requirement that the test method for the CILRT be in accordance with the 1972 revision of ANSI N45.4. ANSI N45.4-1972 specifies use of either the total time or point-to-point method of containment integrated leak rate testing. An exemption was issued to the licensee on October 21, 1988 to permit use of the superior "mass point" method pending revision of 10 CFR Part 50, Appendix J.

Appendix J has since been revised and now permits use of the mass point method (when used for a period of at least 24 hours) as an alternative to the total time and point-to-point methods specified by ANSI N45.4-1972. The amendment would bring the Technical Specifications into consistency with the revised 10 CFR Part 50, Appendix J and is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on September 28, 1989. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (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: W. Long
J. Tsao (Section 2.1.1)

Dated: November 2, 1989

TABLE 1

The NRC Staff Calculated Adjusted Reference Temperature for the Limiting Reactor Beltline Material at Monticello Nuclear Generating Plant

Limiting Beltline Material		Plate material
Code No.		I-15
Copper Content		0.17%
Nickel Content		0.58%
Initial Reference Temperature		14 degrees F
Reactor Vessel Beltline Thickness (in.)		5.06
Chemistry Factor Used in Calculation		125.3
Neutron Fluence (n/cm^2) at 32 EFPY		
	At ID	0.51E19
	At 1/4T	0.38E19
	At 3/4T	0.21E19
Fluence Factor		
	At ID	0.812
	At 1/4T	0.732
	At 3/4T	0.575
Margin		34 at 1/4T
ART at 1/4T at 32 EFPY:		140 degrees F.

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The United States Nuclear Regulatory Commission (the Commission) has issued Amendment No. 72 to Facility Operating License No. DPR-22, issued to the Northern States Power Company (the licensee), which revised the Technical Specifications for operation of the Monticello Nuclear Generating Plant, located in Wright County, Minnesota. The amendment is effective as of the date of issuance.

The amendment (1) revises the reactor vessel pressure vs. temperature curves for consistency with Revision 2 of Regulatory Guide 1.99; (2) adds requirements for augmented inservice inspection of piping susceptible to intergranular stress corrosion cracking; and (3) revises the requirements for the periodic Type A containment integrated leak rate test to permit the use of the mass point test method approved by the Commission in a change to 10 CFR Part 50, Appendix J, as published in the FEDERAL REGISTER on November 15, 1988 (53 FR 45891).

The licensee's application for the amendment dated March 31, 1989, 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.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on June 1, 1989 (54 FR 23553). No request for hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to this action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see (1) the application for the amendment dated March 31, 1989, (2) Amendment No. 72 to License No. DPR-22, (3) the Commission's related Safety Evaluation and (4) the Environmental Assessment dated October 11, 1989 (54 FR 71701). All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC, and at the Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects - III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 2nd day of November 2, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

William O. Long

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