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

October 12, 1999

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, MN 55401

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: REVISION OF REACTOR PRESSURE VESSEL PRESSURE-TEMPERATURE LIMIT CURVES AND REMOVAL OF STANDBY LIQUID CONTROL RELIEF VALVE SETPOINT (TAC NO. MA4532)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 31, 1998, as supplemented May 17, 1999.

The amendment revises the TS reactor pressure vessel (RPV) pressure-temperature (P-T) limit curves, deletes completed RPV sample surveillance requirements, deletes the requirement to withdraw a specimen at the next refueling outage, removes the standby liquid control system relief valve setpoint, and makes associated administrative changes.

Following discussion between the NRC and Northern States Power Company (NSP) staffs during telephone conferences on February 18 and March 10, 1999, the NRC staff issued a request for additional information dated March 24, 1999, which NSP responded to by letter dated May 17, 1999. In the May 17, 1999, letter, NSP made the following new commitment: The Updated Safety Analysis Report will be revised to summarize the results of the surveillance capsule data obtained by irradiating the capsule to beyond end-of-life RPV exposure. It will also state that the next surveillance capsule will be removed during the 2003 refueling outage unless the results of the Integrated Surveillance Program Focus Group determines that removal is unnecessary.

DFO1

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A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/s/

Carl F. Lyon, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 106 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

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Monticello Nuclear Generating Plant

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.106
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 December 31, 1998, as supplemented May 17, 1999, 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. 106, 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 its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 12, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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 vertical lines indicating the areas of change.

REMOVE	INSERT
v	v
94	94
122	122
133	133
134	134
135	135
136	136
146	146

LIST OF FIGURES

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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	b. Explode one of two primer assemblies manufactured in the same batch to verify proper function. Then install, as a replacement, the second primer assembly in the explosion valve of the system tested for operation.

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.

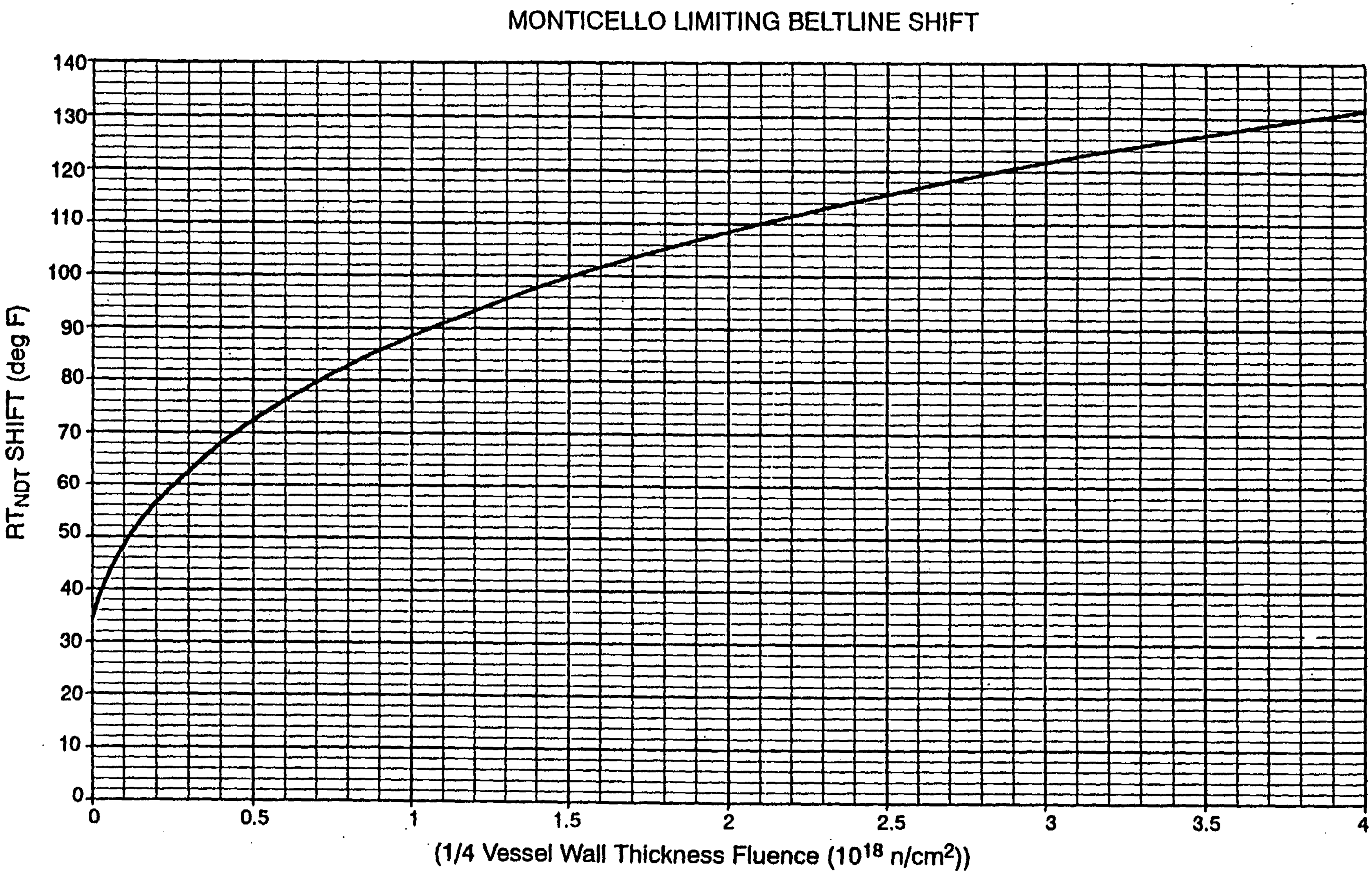

Figure 3.6.1 Core Beltline Operating Limits Curve Adjustment vs. Fluence

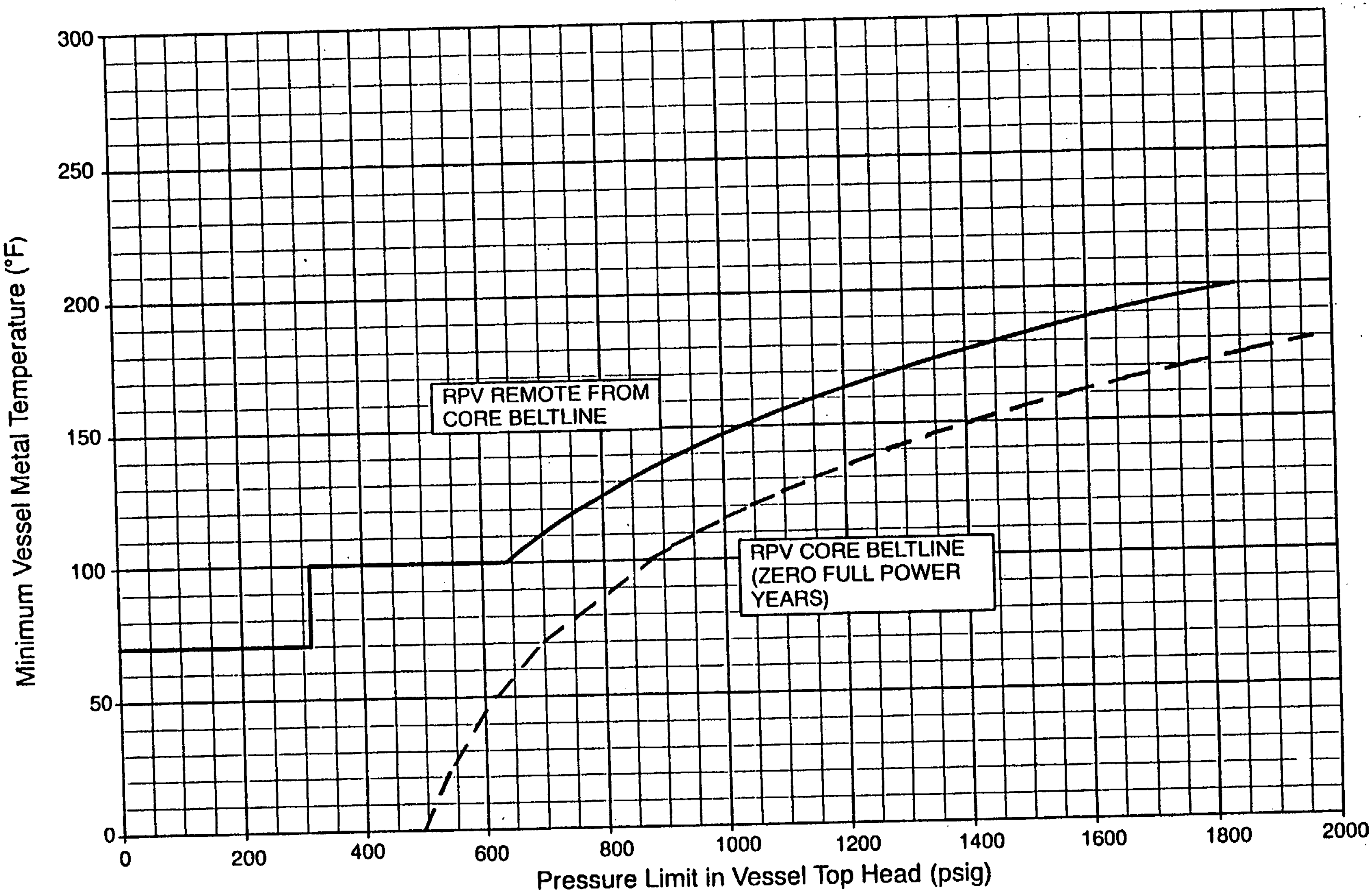

Figure 3.6.2 Minimum Temperature vs. Pressure for Pressure Tests

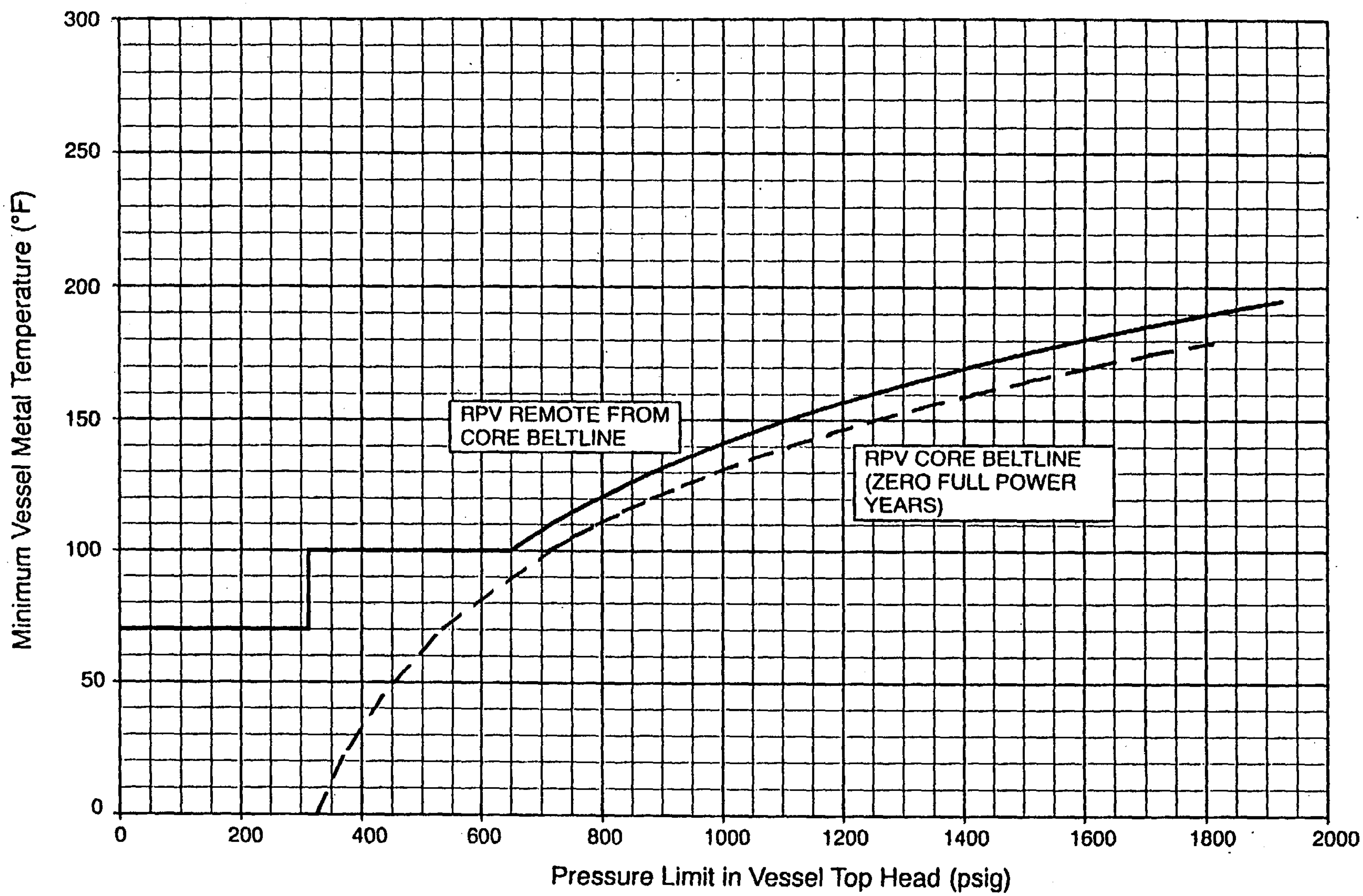

Figure 3.6.3 Minimum Temperature vs. Pressure Mechanical Heatup or Cooldown Without the Core Critical

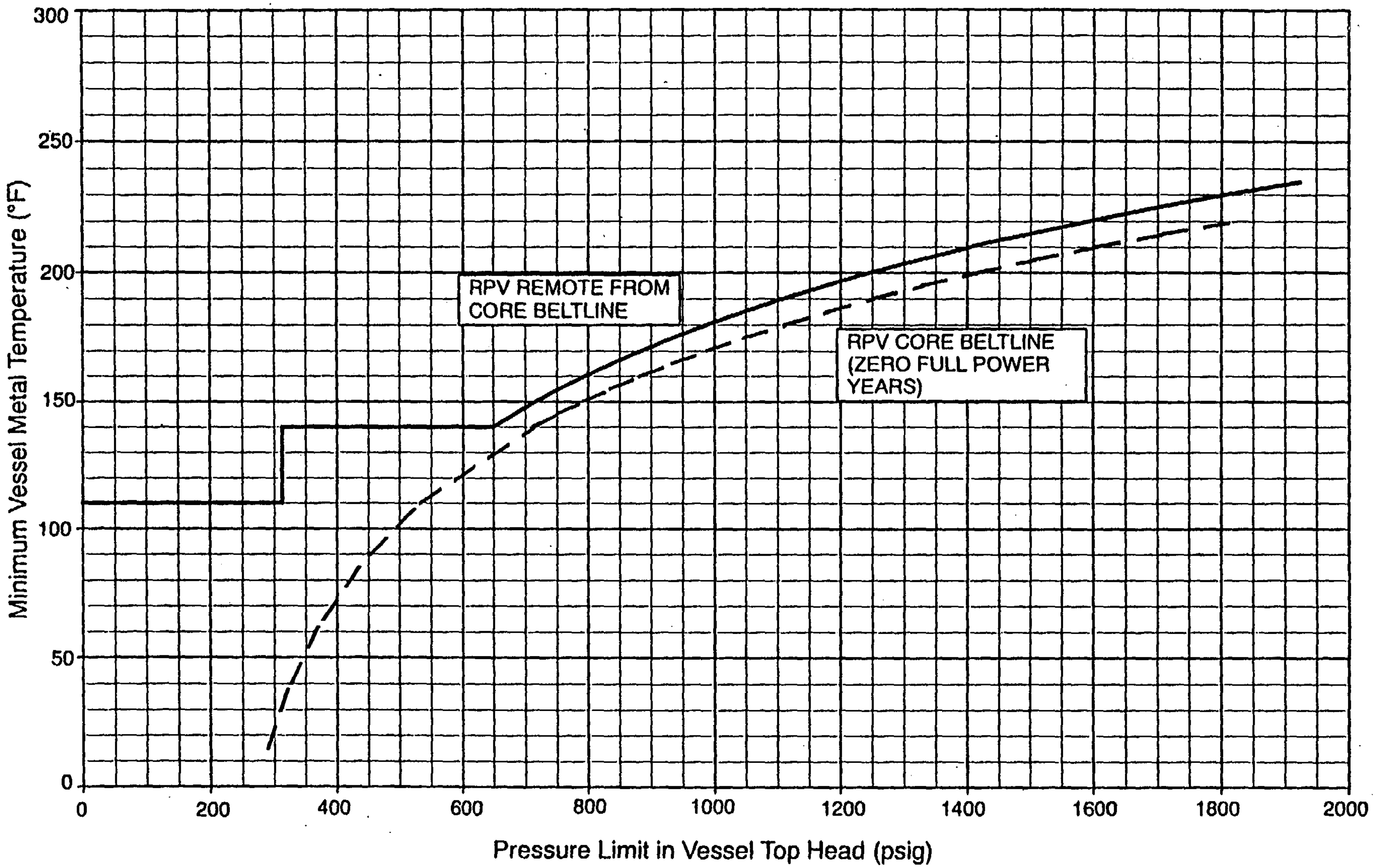

Figure 3.6.4 Minimum Temperature vs. Pressure for Critical Core Operation

Bases 3.6/4.6 (Continued):

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 originally 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 with a noncritical core), 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. Two sets of specimens were contained in the first surveillance capsule which was removed from the vessel in 1981. One set of specimens was tested at this time. The second set was later inserted into a new capsule, and installed in the Prairie Island Nuclear Generating Plant RPV for accelerated irradiation. This capsule was removed and tested in 1996. NSP performed calculations per the requirements of Regulatory Guide 1.99, Rev. 2, Position 2.1 to develop new pressure/temperature (P-T) curves. Results of Charpy V-notch impact tests for the two sets of data and from 1997 non-irradiated material test data were used in developing the revised Figures 3.6.1, 3.6.2, 3.6.3, and 3.6.4. 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By application dated December 31, 1998, as supplemented May 17, 1999, the Northern States Power Company (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). The proposed amendment would revise the TS reactor pressure vessel (RPV) pressure-temperature (P-T) limit curves, delete completed RPV sample surveillance requirements, delete the requirement to withdraw a specimen at next refueling outage, remove the standby liquid control system relief valve setpoint, and make associated administrative changes. The May 17, 1999, letter provided clarifying information that was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

Following discussion between the NRC and Northern States Power Company (NSP) staffs during telephone conferences on February 18 and March 10, 1999, the NRC staff issued a request for additional information dated March 24, 1999, to which NSP responded by letter dated May 17, 1999. NSP also committed to modify the reactor pressure vessel (RPV) surveillance program in the MNGP Updated Safety Analysis Report (USAR) to withdraw the next MNGP surveillance capsule in 2003 pending resolution of an initiative by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) to develop an Integrated Surveillance Program for boiling water reactors (BWRs).

These submittals were made in accordance with regulations in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50), Appendices G and H, which govern the development of RPV P-T curves and the RPV surveillance programs, respectively.

2.0 EVALUATION

2.1 Reactor Pressure Vessel Pressure-Temperature Limits

In the early 1980s, NSP withdrew its first RPV surveillance capsule from the MNGP vessel. This capsule contained two sets of Charpy V-notch specimens, sufficient to generate two complete Charpy curves. One set was tested and the results were reported in surveillance capsule report BCL-585-84-2, Revision 1, dated November 1984, along with the results of the capsule's dosimeter wire analysis. The dosimeter wire analysis demonstrated that the capsule had been subjected to a low lead factor of approximately 0.3 (meaning the capsule acquired about 30 percent of the fluence associated with the peak vessel wall location). NSP decided to re-encapsulate the remaining set of Charpy specimens and insert them for additional irradiation in the Prairie Island Nuclear Generating Plant (Prairie Island) RPV. These specimens were withdrawn from Prairie Island in 1996 and tested, and the results were reported in surveillance capsule report SIR-97-033, Revision 2, dated October 1998 (transmitted by letter from M. Hammer (NSP) to NRC dated December 21, 1998), along with a reanalysis of the original 1984 Charpy data.

NSP concluded that the results from both the original surveillance capsule from MNGP and from the supplemental capsule irradiation in Prairie Island were applicable to the evaluation of the MNGP vessel. A summary of this information is shown in attached Tables 1 and 2 (taken from Tables 3-1 and 3-2 of SIR-97-033, Revision 2) for the surveillance plate (heat C2220) and the surveillance weld, respectively. Note that the shifts in the 30 ft-lb transition temperature can only be given for the MNGP surveillance plate, since the unirradiated Charpy curve for the MNGP surveillance weld material was unavailable. NSP has recently taken action to look for and acquire the unirradiated Charpy curve for plate heat C2220, which is based on testing of this material at Oak Ridge National Laboratory (ORNL).

The NRC staff has raised this issue of "lack of unirradiated baseline surveillance data" and its resolution on an industry-wide basis. It was the impetus for the BWRVIP action to develop an Integrated Surveillance Program for BWRs. The ORNL data was also used to reevaluate the plate's initial nil-ductility reference temperature (IRT_{ndt}).

In its most recent analysis of the MNGP P-T limit curve methodology (conducted in 1998 to support the MNGP power uprate) prior to its December 31, 1998, submittal, the licensee determined that RPV plate I-15 (heat C2220-2) was the limiting material for the MNGP vessel. Since the MNGP surveillance plate is representative of RPV plate I-15, NSP assessed whether the new unirradiated and irradiated test data indicated the need to modify the MNGP P-T limit curve methodology. Based on the methodology found in Regulatory Guide 1.99, Revision 2 (RG 1.99, Rev. 2), for the evaluation of radiation embrittlement in RPV steels, a material's adjusted reference temperature (ART), which for the purpose of the P-T limits evaluation the licensee calculated at a depth 1/4 of the way through the vessel wall (the 1/4T location), can be determined from the following equations:

$$ART = IRT_{ndt} + \Delta RT_{ndt} + \text{Margin}$$

where ΔRT_{ndt} = the irradiation induced shift in RT_{ndt} = CF * FF

with

CF = the Chemistry Factor, a function of the copper and nickel content of the material or as interpreted from the evaluation of available surveillance data

FF = the Fluence Factor, a function of the RPV material's neutron fluence at the 1/4T depth

and Margin = 34 °F (as assumed by NSP for all MNGP materials)

In the NRC staff's safety evaluation, dated September 16, 1998, of the MNGP power uprate amendment, it was documented that the IRT_{ndt} for plate I-15 was 14 °F, its CF was 125.3 °F (based on a chemistry of 0.17 wt% Cu and 0.58 wt% Ni), and its FF was 0.745. This resulted in an end-of-license ART of

$$ART = 14\ °F + (125.3\ °F * 0.745) + 34\ °F = 141\ °F$$

Using the newly acquired unirradiated and irradiated surveillance data, NSP concluded that the IRT_{ndt} value for plate I-15 was 27 °F, that the chemical composition of the plate was 0.17 wt% Cu and 0.65 wt% Ni, and that the plate's CF based on an evaluation of the available surveillance data was 130.8 °F. This resulted in (with no change proposed for the margin and a small modification in the FF to 0.730) an end-of-license ART of

$$ART = 27\ °F + (130.8\ °F * 0.730) + 34\ °F = 156.5\ °F$$

Since these results demonstrated that the previous analysis would be nonconservative at end-of-license (and for any fluence level up to end-of-license), the licensee submitted its December 31, 1998, application to modify the MNGP license by incorporating figures into the P-T limit methodology in the TS that are based on the more conservative material property values above. It should also be noted that since the licensee's evaluation assigned the same initial properties, chemistries, and fluences to the other MNGP lower intermediate shell course plate (I-14, heat C2220-1), that plate was calculated to have the same 1/4T end-of-license ART as plate I-15 and is therefore an equally limiting material.

NSP also identified passages in MNGP TS 4.6.B.2. and TS 4.6.B.3. which referred only to the historical testing of the first MNGP surveillance capsule. NSP concluded that these passages were no longer relevant for inclusion in the TS and proposed to eliminate them as part of this amendment.

Finally, in a separate action not involving the proposed TS change, NSP committed to incorporate into the MNGP USAR a withdrawal date of 2003 for the next MNGP surveillance capsule. This date may be changed later pending resolution of the initiative by the BWRVIP to develop an Integrated Surveillance Program for BWRs.

2.2 Staff Evaluation and Conclusions

The NRC staff independently evaluated the new unirradiated and irradiated surveillance material test data for plate heat C2220 as it applies to either MNGP plate I-14 or I-15. Regarding the licensee's modification of the IRT_{ndt} value for heat C2220 to 27 °F from 14 °F based on the unirradiated data from ORNL, the staff concurs with NSP's assessment. The staff accepts the use of 27 °F as the IRT_{ndt}. In addition, the use of this measured IRT_{ndt} value permits the margin term to be established as no greater than 34 °F based on the use of RG 1.99, Rev. 2, methodology.

The NRC staff then evaluated the licensee's proposal to use the data from the surveillance material irradiated in Prairie Island for assessing the MNGP vessel. While the licensee's action to take measures to address the low lead factor problem associated with the first capsule that was withdrawn from the MNGP vessel in the early 1980s is commendable, it also raises certain technical concerns. In report SIR-97-003, Revision 2, NSP noted that the second capsule (from

Prairie Island), "saw accelerated fluence (lead factor >10)...." The NRC staff's calculation based on the flux and fluence information provided in this report supported this statement and estimated that the lead factor would be about 18.

In the development of surveillance programs for RPVs, the Commission has recognized in its regulations (10 CFR Part 50, Appendix H) the use of American Society for Testing and Materials Standard Practice E185 (ASTM E185). Since at least the 1966 edition of this standard, an emphasis has been placed on maintaining the irradiation history of surveillance capsules such that their irradiation temperature, neutron flux, neutron spectrum, and maximum neutron fluence are similar to the conditions experienced by the vessel, while still permitting the capsule fluence to lead the vessel so that the testing results are predictive. For the RPV "wall" capsules required by ASTM E185, the upper limit recommended for the lead factor has ranged from 3 to 5. Such lead factors permit end-of-license fluences to be achieved in a reasonable amount of time while not causing the irradiation history of the capsule to be so different from that of the RPV that the potential for significant flux effects on the irradiation damage response must be evaluated. However, in considering a capsule irradiated with a lead factor of 18, the NRC staff is concerned that some flux effect may occur, particularly since the capsule is essentially being irradiated in an environment (including temperature history and neutron spectrum) characteristic of a pressurized water reactor rather than that characteristic of a BWR. For these reasons, and without additional data from the MNGP surveillance program to verify that the embrittlement behavior projected by the capsule pulled from Prairie Island is appropriate or conservative for assessing the embrittlement of the MNGP RPV, the NRC staff utilized the RG 1.99, Rev. 2, CF Tables in its evaluation. The tables were used in the absence of demonstrated credible surveillance data.

Plate heat C2220 was determined to have a best-estimate copper content of 0.17 wt% and a best-estimate nickel content of 0.65 wt%, which equates to a CF of 128.3 °F. Using the same IRT_{ndt}, margin, and fluence factor used in the licensee's analysis, the NRC staff calculated the end-of-license ART for the limiting MNGP plate as

$$ART = 27\ °F + (128.3\ °F * 0.730) + 34\ °F = 154.7\ °F$$

This analysis will also be recorded in the NRC's Reactor Vessel Integrity Database (RVID) and the basis for its determination documented there. The data supplied by the licensee based on their testing of the surveillance capsule from Prairie Island will also be included in the RVID should the staff's questions regarding its use be addressed in the future.

Note that the results of the staff's analysis show that the CF and end-of-license ART values calculated by the licensee are conservative (higher than the staff's value). Therefore, although the staff does not concur with the basis of the licensee's analysis (i.e., the use of the Prairie Island surveillance data) the staff finds the results of the licensee's analysis to be acceptable. Furthermore, since the results of the licensee's analysis are acceptable, the TS changes proposed by licensee based on their analysis are also acceptable, since they are equivalent or conservative to the analysis results independently calculated by the staff, and their implementation is consistent with the requirement of 10 CFR Part 50, Appendix G.

The NRC staff also concurs with the licensee's proposal to remove historical references to the testing activities of the first MNGP surveillance capsule from the TSs. This information has been acquired and reported to the NRC as required under 10 CFR Part 50, Appendix H. The historical reference in the TSs provides no added value.

Based on the information provided by NSP in its submittals and the results of the NRC staff's independent analysis, the staff concludes that the proposed changes are acceptable. This finding was made since the requirements in the TSs will continue to comply with 10 CFR Part 50, Appendix G. The commitment by NSP to incorporate a 2003 withdrawal date for the next MNGP surveillance capsule into the facility's USAR is also acceptable to the staff since the licensee continues to comply with 10 CFR Part 50, Appendix H.

2.3 Standby Liquid Control Relief Valve Setpoint

The licensee proposes to delete TS 4.4.A.2.c, which specifies the Standby Liquid Control (SBLC) relief valve setpoint. The setpoint of the SBLC system relief valves is governed by the provisions of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section XI, as required by TS 3.15. The testing required by TS 4.4.A.2.c is enveloped by the current testing performed by MNGP's Inservice Test (IST) Program. The IST Program implements an edition of ASME Code Section XI that has been approved in 10 CFR 50.55a. Any modification to the setpoint is controlled by the plant's configuration control process, which would ensure the requirements of ASME Code Section XI are invoked as required by TS 3.15. The IST Program required by TS 4.15 ensures the SBLC relief valves would be properly tested for operability. Therefore, the change is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (64 *FR* 6706). Accordingly, the amendment meets 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 the amendment.

5.0 CONCLUSION

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

Attachment: Tables 1 and 2

Principal Contributors: M. Mitchell
F. Lyon

Date: October 12, 1999

Table 1
Summary of Estimated Shift per Regulatory Guide 1.99, Rev. 2 and Measured Shift from the Charpy V-Notch Test Data for the Base Metal (Plate)

Condition	Fluence (f) n/cm^2 (1) E> 1 MeV	f / 10^{19} n/cm^2 E> 1 MeV	Fluence Factor FF (2)	Chemistry Factor, °F From Table 2-2	Reg Guide 1.99 Rev. 2 (3) Predicted Shift °F	Temperature @30 ft-lbs, °F From CVGraph Analysis	Measured Shift °F Capsule - Unirradiated	Measured-Predicted Shift, °F
Unirradiated	0	0	--	128.3	--	27	--	--
1st Capsule	2.93E+17	0.0293	0.2166	128.3	27.8	56	29	1.2
2nd Capsule	3.33E+18	0.333	0.6974	128.3	89.5	118	91	1.5
				1st to 2nd capsule shift	61.7	62	62	

Notes
1. Fluence values for the 1st and 2nd capsules are from References 4 and 5, respectively.
2. Fluence factor = FF = $f^{(0.28-0.10\log f)}$, where f is the fluence at the point of interest.
3. Predicted shift ΔRT_{NDT} = (CF) * (FF), where CF is the chemistry factor [3].

Table 2
Summary of Estimated Shift per Regulatory Guide 1.99, Rev. 2 and Measured Shift from the Charpy V-Notch Test Data for the Surveillance Weld Metal

Condition	Fluence (f) n/cm^2 (1) E> 1 MeV	f / 10^{19} n/cm^2 E> 1 MeV	Fluence Factor FF (2)	Chemistry Factor, °F (Note 4)	Reg Guide 1.99 Rev. 2 (3) Predicted Shift °F	Temperature @30 ft-lbs, °F From CVGraph Analysis	Measured Shift °F Capsule - Unirradiated
Unirradiated	0	0	--	82	--	Unknown	--
1st Capsule	2.93E+17	0.0293	0.2166	82	17.8	-65	Unknown
2nd Capsule	3.26E+18	0.326	0.6918	82	56.7	-0.1	Unknown
				1st to 2nd capsule shift	39.0	64.9	

Notes
1. Fluence values for the 1st and 2nd capsules are from References 4 and 5, respectively.
2. Fluence factor = FF = $f^{(0.28-0.10\log f)}$, where f is the fluence at the point of interest.
3. Predicted shift ΔRT_{NDT} = (CF)*(FF), where CF is the chemistry factor [3].
4. Chemistry factor is from Reference 3 for a composition of Cu=0.06% and Ni=0.95% (highest measured values from Reference 4).

Attachment