Xcel Energy®

Monticello Nuclear Generating Plant
2807 W County Road 75
Monticello, MN 55362

WITHOLD ENCLOSURE 10 FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390 AND 10 CFR 9.17

January 20, 2012

L-MT-12-002
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License No. DPR-22

<u>License Amendment Request: Revise and Relocate Pressure Temperature Curves to a Pressure Temperature Limits Report</u>

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, proposes to revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) to add a Pressure and Temperature Limits Report (PTLR) incorporating new pressure-temperature (P-T) limit curves.

The conditions of U.S. Nuclear Regulatory Commission (NRC) Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," were applied during development of the proposed P-T curves. The TS changes were developed in accordance with the NRC approved guidance of Technical Specification Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in Improved Standard Technical Specification 5.6.6, Reactor Coolant System PTLR." The analytical methodology used in development of the proposed P-T curves and the PTLR, is the NRC approved, Structural Integrity Associates (SIA) methodology report No. SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors."

The proposed P-T curves were developed including the effects of operation throughout the twenty-year extended period of operation (54 Effective Full Power Years) in accordance with the Monticello Renewed Facility Operating License. The P-T curves also consider the effects of operation at the Extended Power Uprate power level of 2004 MWth and operation in the MELLLA and MELLLA+ [1] operating domains. The

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1. MELLLA stands for <u>M</u>aximum <u>E</u>xtended <u>L</u>oad <u>L</u>ine <u>L</u>imit <u>A</u>nalysis. MELLLA+ expands the MELLLA operating domain.

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P-T curves were developed including these considerations, but are not dependent on approval of these amendments, as these conditions conservatively provide additional margin and bound MELLLA operation at the current licensed power level.

This license amendment request (LAR) also addresses two non-conservative TS conditions. The guidance of NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," is being applied until these conditions are resolved with approval of this LAR. Corrective actions have been taken for the interim period between identification and resolution to ensure conservative operation.

- The first non-conservative TS condition discovered is that with incorporation of recent surveillance specimen capsule results TS Figure 3.4.9-1, "Core Beltline Operating Limits Versus Fluence," becomes non-conservative in approximately 2026. This figure is part of the current MNGP specific methodology (not part of the SIA methodology) and will be removed with approval of this amendment, eliminating this condition.

- The second non-conservative TS condition is that sufficient fluence had accumulated for the recirculation inlet nozzles to enter the beltline region. These nozzles are now a limiting component under low temperature / low pressure conditions. Procedural controls have been put in place to prevent entry into this region during hydrostatic testing.

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Enclosure 2 provides a marked-up copy of the TS pages showing the proposed changes. Enclosure 3 provides a marked-up copy of the TS Bases pages, for information, indicating the proposed changes. Enclosure 4 provides the proposed MNGP PTLR report.

Enclosure 5 provides a copy of the calculation for the adjusted reference temperatures (ART) and reference temperature shifts for the reactor pressure vessel (RPV) components. Enclosure 6 provides a copy of a calculation for the finite element stress analysis of the RPV feedwater nozzle. Enclosure 7 provides a copy of a calculation for the finite element stress analysis of the RPV recirculation inlet nozzle. Enclosure 8 provides a copy of the new P-T curves calculation. Enclosure 10 contains information that is proprietary to SIA Incorporated and the Electric Power Research Institute (EPRI). Accordingly, this proprietary information should be withheld from public disclosure in accordance with 10 CFR 2.390(a)4 and 10 CFR 9.17(a)4. Enclosure 9 provides the affidavits from SIA and EPRI supporting treatment of this calculation in Enclosure 10 as proprietary. Enclosure 10 provides the proprietary copy of the ART and reference temperature shifts for RPV components calculation.

Approval of this license amendment is requested by January 31, 2013. To allow sufficient time for the extensive operator training and procedural changes required, and not impact preparation for the 2013 Refueling Outage, NSPM proposes to implement this amendment within 180 days after start-up from the 2013 Refueling Outage.

The MNGP Plant Operations Review Committee has reviewed this application. In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Minnesota Official.

Should you have questions regarding this letter, please contact Mr. Richard Loeffler at (763) 295-1247.

<u>Summary of Commitments</u>

This letter proposes no new commitments and does not revise any existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on January 20, 2012.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (10)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Minnesota Department of Commerce
Resident Inspector, Monticello, USNRC

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE TECHNICAL SPECIFICATIONS TO INCLUDE A PRESSURE TEMPERATURE LIMITS REPORT

DESCRIPTION OF CHANGES

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DESCRIPTION OF CHANGES

LICENSE AMENDMENT REQUEST
REVISE THE TECHNICAL SPECIFICATIONS TO INCLUDE A PRESSURE TEMPERATURE LIMITS REPORT

1.0 SUMMARY

Pursuant to 10 CFR 50.90, Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, proposes to revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications (TS) to add a Pressure and Temperature Limits Report (PTLR) and to incorporate new pressure-temperature (P-T) limits developed applying the U.S. Nuclear Regulatory Commission (NRC) approved Structural Integrity Associates (SIA) methodology.

SIA Licensing Topical Report (LTR) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," (Reference 1) was utilized to generate the new MNGP P-T curves. The guidance of NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," (Reference 2) was applied during P-T curve development. Also, Technical Specification Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in Improved Standard Technical Specification 5.6.6, Reactor Coolant System PTLR," (Reference 3), which has received NRC approval, was followed in development of the proposed TS changes.

The proposed P-T curves include the effects of operation throughout the twenty-year extended operation period (54 Effective Full Power Years (EFPY)) in accordance with the Renewed Facility Operating License. The P-T curves also consider operation at the Extended Power Uprate (EPU) power level (2004 MWth), with operation in the MELLLA and MELLLA+ [(1)] operating domains at EPU conditions. The P-T curves were developed considering and including these conditions, but are not dependent on approval of these amendments, as these considerations conservatively provide additional margin, and bound MELLLA operation at the current licensed power level.

Two non-conservative TS conditions are also addressed herein. Corrective actions have been taken for the interim period between identification and resolution to ensure conservative operation.

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1. MELLLA stands for Maximum Extended Load Line Limit Analysis. MELLLA+ expands the MELLLA operating domain.

2.0 DETAILED DESCRIPTION

The NRC has established requirements in 10 CFR 50, Appendix G, "Fracture Toughness Requirements," in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Appendix G requires that the P-T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code were used to generate the P-T limits. Also, 10 CFR 50 Appendix G requires that applicable surveillance data from reactor pressure vessel (RPV) material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

P-T curves were developed for MNGP applying the methodology of the 2004 Edition of the ASME Code, Section XI, Appendix G; 10 CFR 50 Appendix G; and the Boiling Water Reactors Owners Group (BWROG) Licensing Topical Report (LTR) (Reference 1). The new P-T curves apply at the fluence levels associated with EPU including MELLLA+ operation(2) through the twenty-year renewed operating license period. A full set of P-T curves was developed for all plant conditions at 54 effective full power years (EFPY), including curves for the following conditions: hydrostatic pressure testing (Curve A), plant operation – core not critical (Curve B), and plant operation – core critical (Curve C).

Operational challenges are presented by performance of pressure / hydrostatic testing near 212°F,(3) as required by the 54 EFPY curve. To avoid this region until necessary, additional pressure / hydrostatic testing curves (Curve A) have been developed for application at two intermediate fluence levels, 36 and 40 EFPY.

Pressure-temperature limits are discussed in Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits." This license amendment request (LAR) modifies the TS by adding a definition of the PTLR, revises the Limiting Condition for Operation (LCO) and surveillance requirements in Specification 3.4.9 by removing references to the P-T curves (and temperature values) and replaces them with references to the PTLR, and adds the PTLR as a report in the Administrative Controls section of the TS.

The requirements of NRC Generic Letter 96-03 were applied in development of this LAR and the PTLR. Also, the guidance of NRC approved TSTF traveler TSTF-419-A

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2. Section 5.2 of this enclosure also discusses impact on fluence calculations of operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain at EPU conditions.
3. The 54 EFPY pressure / hydrostatic test curve will require testing near 212°F. While the 54 EFPY curve is conservative for lower EFPYs, curves developed for lower, intermediate EFPYs provide margin which does not require testing near 212°F.

was applied as modified by the NRC letter approving the TSTF-419 (Reference 4) and considering recent NRC determinations.

The following two non-conservative TS conditions were identified during review and incorporation of recent surveillance specimen capsule results and subsequent analyses. These conditions are addressed in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," until these conditions are resolved with the approval of this LAR.

1. Incorporation of recent surveillance specimen results have resulted in current TS Figure 3.4.9-1, "Core Beltline Operating Limits Versus Fluence," curve becoming non-conservative in approximately 2026. (This curve is part of the current MNGP P-T methodology and will be eliminated with adoption of the SIA methodology via this amendment.)

2. Determination that the recirculation inlet (N2) nozzles are in the beltline region(4) of the vessel and are a limiting component for low temperature / low pressure conditions.

Also, while not a non-conservative condition, it was also identified during the analyses that the upper-intermediate plates have entered the beltline region of the vessel.

3.0 PROPOSED CHANGES

The proposed change includes the following TS modifications related to adopting a PTLR and developing new P-T curves in accordance with the methodology of SIR-05-044-A.

1) Add a definition in Section 1.0 for the Pressure and Temperature Limits Report. The wording for this definition is consistent with that in TSTF-419-A.

2) Revise the LCO for Specification 3.4.9, "RCS Pressure and Temperature (P-T) Limits," to refer to the PTLR for the P-T limits.

3) Surveillance requirements SR 3.4.9.1 and SR 3.4.9.2 each contain a note specific to application of the current MNGP methodology for the P-T curves.

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4. The reactor beltline refers to the vessel region (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage (1.0×10^{17} n/cm^2 in accordance with Regulatory Guide 1.99) to be considered in selection of the most limiting material with regard to radiation damage.

Following adoption of the BWROG SIR P-T methodology, this note is not applicable, and is being removed.[5]

Specifically, the note is being removed from:

- SR 3.4.9.1, second note, and
- SR 3.4.9.2, first note.

4) Revise the reference to the figures (P-T curves) in surveillance requirements SR 3.4.9.1 and SR 3.4.9.2 to refer to the PTLR.

5) Revise the reference to the temperature limits in surveillance requirements SR 3.4.9.1, SR 3.4.9.3, SR 3.4.9.4, SR 3.4.9.5 and SR 3.4.9.6 to refer to the PTLR.

6) Remove the present P-T curves, i.e., Figure 3.4.9-1 through Figure 3.4.9-4.

7) Add a new Specification 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to Subsection 5.6, "Reporting Requirements," in Section 5.0, "Administrative Controls." The format and content is according to TSTF-419-A. This new specification:

   - includes the individual TSs that address reactor coolant system P-T limits,
   - references the NRC approved SIR topical report which documents the PTLR methodology, and
   - requires the PTLR and any revisions or supplements to be submitted to the NRC.

A mark-up of the revised pages and TS changes associated with adopting a PTLR are provided in Enclosure 2. A mark-up of the proposed draft TS Bases changes is provided in Enclosure 3, for information only. The changes to the TS Bases pages will be incorporated upon approval of this LAR in accordance with Specification 5.5.9, "Technical Specifications (TS) Bases Control Program."

4.0 TECHNICAL ANALYSIS

10 CFR 50, Appendix G requires licensees to establish limits on the pressure and temperature of the RCPB in order to protect against brittle failure. These limits are defined by P-T curves for normal operations (including heatup and cooldown operations of the RCS, normal operation of the RCS with the reactor being in a critical condition

5. The current plant-specific P-T curves are adjusted in accordance with an NRC approved MNGP core beltline operating limits versus fluence curve. This curve is being removed.

and anticipated operational occurrences) and during pressure testing conditions (i.e., inservice leak rate testing and / or hydrostatic testing conditions).

Historically, utilities have submitted LARs to update their P-T curves. Processing LARs has caused both the NRC and licensees to expend resources that could otherwise be devoted to other activities. The objective of the SIA LTR is to provide a generically approved method for utilities to generate P-T curves.

GL 96-03 allows plants to relocate their P-T curves and the associated numerical limits (such as heatup / cooldown rates) from the plant TS to a PTLR – a licensee-controlled document. As stated in the generic letter, during development of the improved Standard Technical Specifications (STS), a change was proposed to relocate the P-T limits contained in the plant TS to a PTLR. As one of the improvements to the STS, the NRC staff agreed with the industry that the P-T curves could be relocated outside the plant TS to a PTLR so that licensees could maintain these limits efficiently.

TSTF-419-A and the associated LTRs provide the ability for BWR licensees to relocate their P-T curves and the associated numerical values (such as heatup/cooldown rates) from the facility TS to a PTLR, a licensee-controlled document, using the guidelines in GL 96-03. The transmittal letter for the NRC Safety Evaluation Report (SER), dated February 6, 2007 (Reference 5), states "the NRC staff has found that SIR-05-044 is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified and under the limitations delineated in the TR [Technical Report] and in the enclosed final SE."

The proposed MNGP PTLR is based on the methodology and template provided in SIR-05-044-A. The purpose of the MNGP PTLR is to present operating limits related to Reactor Coolant System (RCS) pressure versus temperature limits during heatup, cooldown, and hydrostatic/class 1 leak testing. The curves, which have been prepared using NRC approved methodology, will allow system pressurization at lower temperatures thus saving critical path time and provide improved work environment conditions for the inspectors during leak testing inspections.

PART I – DEVELOPMENT OF THE PRESSURE / TEMPERATURE LIMIT CURVES

To apply the PTLR option, the method used to develop the P-T curves and associated limits must be NRC approved. Also, the associated LTR is required to be referenced in the specification for the PTLR program in the plant TSs. The SIA LTR provides one of the current NRC-approved BWROG fracture mechanics methodologies for generating P-T curves / limits.

The results of the MNGP surveillance capsule removed from the reactor vessel in 2007 were considered in accordance with the guidance of Regulatory Guide 1.190 and are reflected in the fluence calculations for development of P-T limits.

As discussed in the following sections, the new P-T curves apply at the fluence levels associated with MNGP EPU through the twenty-year renewed operating license period. A full set of P-T curves was developed for all plant conditions at 54 EFPY, including curves for the following conditions:

- hydrostatic pressure testing (Curve A),
- plant operation – core not critical (Curve B), and
- plant operation – core critical (Curve C).

For these P-T curves, a 1.3 multiplication factor was applied to the fluence values over the full assumed RPV lifetime (i.e., 0 EFPY through 54 EFPY) to account for potential fluence variation during *future cycles* of operation.

A consequence of the application of this multiplication factor is that it results in overly conservative hydrostatic / pressure test curves with test temperatures approaching 212°F(6) for EFPYs substantially below 54 EFPY. This condition presents operational and personnel safety challenges. NSPM has removed this 1.3 multiplication factor for the calculation of hydrostatic / pressure test curves (Curve A) between 0 EFPY and an assumed conservative EPU implementation value determined to be 33.4 EFPY. Two additional pressure / hydrostatic testing curves (Curve A) have been developed for application for two intermediate fluence levels, 36 and 40 EFPY. This is discussed in Section 4.4.

P-T Curve Axes Inversion for MNGP

While most plants use P-T curves, MNGP uses the inverse of P-T curves, i.e., T-P curves for operation. The MNGP operators have been trained to operate ABOVE these curves. Use of the inverse of the P-T curves has no effect on the validity of the curves to protect the RPV from fracture. To avoid operator confusion and prevent error-likely situations, MNGP will continue to use the T-P curve format.(7)

4.1 Development of the P-T Curves In Accordance with the SIA Methodology

One of the prerequisites for the PTLR option is that the method used to develop the P-T curves and associated limits be NRC approved, and that the associated LTR for such approval be referenced in the specification for the PTLR program in the plant TSs. The SIA LTR provides one of the current NRC-approved BWROG fracture mechanics methodologies for generating P-T curves / limits and allows

6. The 54 EFPY pressure / hydrostatic test curve will require testing near 212°F. While the 54 EFPY curve is conservative for lower EFPYs, curves developed for lower, intermediate EFPYs provide margin which does not require testing near 212°F.
7. To avoid confusion throughout this LAR, with the exception of the depiction of the actual curves, the P-T curve nomenclature will be applied herein.

BWR plants to adopt the PTLR option in accordance with TSTF-419-A and GL 96-03.

As discussed in the NRC's SER approving the SIA LTR, the licensing topical report has three sections and two appendices, the content of which is summarized below.

- Section 1.0 describes the background and purpose for the LTR.

- Section 2.0 of the SIA LTR provides the fracture mechanics methodology and its basis for developing P-T limits. Attachment 1 of GL 96-03 provides seven technical criteria that contents of a methodology should conform to, to develop P-T limits and to be acceptable by the NRC staff.

- Section 3.0 of the SIA LTR provides a step-by-step procedure for calculating P-T limit curves. This section indicates that typically three reactor pressure vessel (RPV) regions are evaluated with respect to P-T limits: (1) the beltline region; (2) the bottom head region; and (3) the non-beltline region.

  The MNGP P-T curves consider these three regions of the vessel; beltline, bottom head, and non-beltline (feedwater nozzle / and upper intermediate plates[8]). The MNGP PTLR is based on the methodology and template provided in the SIA LTR and the MNGP specific calculations detailed below.

- Appendix A of the LTR provides guidance for evaluating surveillance data.

- Appendix B provides a template for development of an acceptable PTLR.

The NRC staff evaluation of the contents of the BWROG SIA methodology against the seven criteria of GL 96-03 is provided in Section 3.1 of the SER.

The following calculations performed for the MNGP by SIA support development of the PTLR and are included in the following enclosures for information. Design inputs and assumptions utilized for the fluence determination and in development of the proposed MNGP P-T curves are described in the applicable calculations below.

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8. Approximately 18 inches of the upper intermediate plates will have entered the beltline region beginning at 12 EFPY and projected through 54 EFPY. However, the ART of the other beltline materials is significantly higher than the ART of the upper vessel plates. Since the ART of the upper vessel plates is bounded by the ART of the other components, the entry of the upper plates into the beltline region has no effect on the P-T curves.

Encl.	Title	NSPM Number	SIA Number
5	Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts	CA 11-003	1000847.301
6	Finite Element Analysis of Monticello RPV Feedwater Nozzle	CA 11-004	1000847.302
7	Finite Element Analysis of RPV Recirculation Inlet Nozzle	CA 11-020	1000720.301
8	Revised P-T Curves	CA 11-005	1000847.303
10	Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts (Proprietary Version)	CA 11-003	1000847.301

4.2 EPU and MELLLA+ Fluence Calculations

In November 2008, NSPM submitted a LAR applying for an EPU for the MNGP to 2004 MWth (Reference 6). Subsequently, in January 2010, NSPM submitted a LAR requesting application of the GE Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain[9] to the EPU conditions (Reference 7). A request for concurrent review of the MELLLA+ LAR together with the EPU LAR was submitted in October 2009 (Reference 8), and authorized by the NRC in November 2009 (Reference 9). Subsequent to development of the EPU[10] and MELLLA+ license amendment requests, NSPM decided to pursue development of new P-T curves applying the NRC approved SIA methodology contained in SIR-05-044-A. The P-T curves developed will apply at EPU and at EPU within the MELLLA+ operating domain and bound current licensed thermal power operation.

9. MELLLA+ increases the flow control range providing additional operating flexibility.
10. The EPU LAR indicated that revised P-T curves, developed in accordance with GE methods, would be submitted in the future as a separate LAR.

The EPU and MELLLA+ amendment reviews have been delayed due to the issues described in SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," (Reference 10). At this time, it is difficult for NSPM to predict what the regulatory guidance updates will be, or the timeframe for update completion, and the subsequent analyses (and their duration) that will have to be performed by NSPM and submitted for NRC review and approval.

In preparation for the above LARs, fluence calculations were performed in accordance with the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 11). The General Electric – Hitachi (GEH) fluence methodology, NEDC-32983P-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," (Reference 12) was used to perform the fluence calculations. The GEH fluence methodology is NRC approved and Regulatory Guide 1.190 compliant.

These calculations determined the projected fluence accumulation for the reactor vessel, considering EPU power level (2004 MWth) and EPU operation in the MELLLA+ operating domain. The calculations were performed for the assumed fluence at the end of the renewed facility operating license period (i.e., 54 EFPY) on September 8, 2030. A multiplication factor of 1.3 was applied by GEH to the fluence values to account for potential variation in future cycles of operation. EPU operation was assumed to begin at 28.82 EFPY (with exception of the later developed 36 and 40 EFPY hydrostatic / pressure testing curves – where EPU operation was postulated to begin at 33.4 EFPY[11] [12]). The neutron source distribution used in the calculations is based on the EPU equilibrium core, not on an extrapolation of the original licensed thermal power source distribution.

The fluence values for the components under EPU and EPU with MELLLA+ were compared to determine the adjusted reference temperature (ART) and related P-T limits / curves were compared and the most conservative values applied. The fluence associated with EPU operation was determined to be more conservative (higher fluence values) then that for EPU operation in the MELLLA+ domain, and hence, the EPU fluence values were applied to determine the P-T limits / curves in the PTLR. The table below provides a comparison of the fluence results for these two conditions from the EPU (Reference 13) and MELLLA+ fluence

11. The P-T curves for plant operation – core not critical (Curve B), core critical (Curve C), and 54 EFPY hydrostatic / pressure test curve (Curve A), were developed applying a 1.3 multiplication factor to the fluence from 0 to 54 EFPY. The 36 and 40 EFPY hydrostatic / pressure testing curves (Curve A), were developed applying the 1.3 multiplication factor to the fluence above 33.4 EFPY.
12. 33.4 EFPY is the EFPY conservatively calculated up to the 2011 Refueling Outage.

analyses, based on the location of the component from the BAF (bottom of active fuel).

Table A – Comparison of EPU and MELLLA+ Fluences

Area/Component	Fluence in 10^{18} n/cm^2		Elevation from BAF (inches)
	EPU	EPU with MELLLA+	
Upper/Intermediate Shell Plates I-12 / I-13	0.406	0.395	158.6
Lower/Intermediate Shell Plates I-14 / I-15	6.43	6.39	N/A*
Lower Shell Plates I-16 / I-17	4.46	4.43	27.1
Limiting Weld – Beltline	6.43	6.39	N/A*
Bounding N-2 Nozzle	1.01	0.955	-3.2

* Since these components are contained completely in the beltline region. The highest fluence value in the beltline or the peak fluence is used in the evaluation.

The P-T curves discussed herein were developed to apply to current MELLLA operation, at the current power level, and to EPU / MELLLA+ operation, through the 20-year term of the Renewed Facility Operating License.(13)

In summary, projected fluence calculations were performed for the EPU power level (2004 MWth) in accordance with the Regulatory Guide 1.190 compliant, NRC approved GEH fluence methodology. The fluence associated with EPU operation (not including MELLLA+) was determined to be more limiting and was applied to determine the ART and related P-T limits / curves. These calculations were performed for the assumed fluence at the end of the renewed facility operating license period, i.e., 54 EFPY.

13. This LAR supersedes the P-T limits discussion in the EPU and MELLLA+ LARs (and associated RAI responses), since this LAR was developed later and addresses additional items, as discussed herein. If the EPU / MELLLA+ LARs are approved subsequent to this LAR, this PTLR LAR regarding P-T curve design and licensing bases applies.

4.3 Monticello 300 Degree Surveillance Capsule Results

In 2007, the MNGP surveillance capsule located at the 300 degree reactor vessel azimuth was removed from the RPV and sent out for testing in accordance with the Boiling Water Reactor Vessel Improvement Program (BWRVIP) Integrated Surveillance Program (ISP), of which the MNGP is an active member. The results are required in accordance with the ISP and Regulatory Guide 1.190 to be included in the fluence calculations for development of P-T limits. Because the EPU fluence calculation was completed prior to removal and analysis of the 300 degree surveillance capsule, the capsule results were not included in the EPU fluence evaluation.

To address this condition, NSPM had GEH evaluate the EPU fluence results to determine the effect on the results if the 300 degree surveillance capsule data had been included. It was determined that the 300 degree surveillance capsule fluence data was within the uncertainty range of the EPU fluence calculation. Recalculation of the fluence explicitly including 300 degree surveillance capsule data was not required. In summary, the EPU fluence calculation results (Reference 13) were applied and no adjustment to the EPU fluence was necessary to account for the surveillance capsule data.

4.4 Development of Hydrostatic / Pressure Test Curves for Various Fluence Levels

As discussed in Section 4.2, the NRC approved Regulatory Guide 1.190 compliant GEH fluence methodology (NEDC-32983P-A) was used to determine the EPU fluence values. A multiplication factor of 1.3 was then applied to the fluence values by GEH to account for potential variation in future cycles of operation.

During review of the MNGP EPU fluence report (Reference 13), it was identified that for locations with an accumulated fluence near the lower bound of 1.0×10^{17} n/cm^2, application of this factor was overly conservative. Also, the 1.3 multiplication factor was applied over the full assumed RPV lifetime (i.e., 0 EFPY through 54 EFPY), although it was established to account for potential fluence variation during *future cycles* of operation. Therefore, application of the multiplication factor for fluence calculations up to the present is unnecessary since the fluence accumulation on the vessel is monitored every cycle and is known up to the current cycle.

A consequence of the application of this multiplication factor is that it results in overly conservative hydrostatic / pressure test curves with test temperatures approaching 212°F for EFPYs substantially below 54 EFPY. RPV test temperatures near 212°F result in increased personnel safety concerns. Additionally, it takes longer to achieve test conditions and to decrease temperature and demobilize from test conditions post-test. These additional preparations will result in longer outage durations, accumulation of additional personnel dose and increased risk to the plant and plant personnel with no corresponding safety benefit.

Accordingly, NSPM examined removal of the 1.3 multiplication factor for the calculation of hydrostatic / pressure test curves (Curve A) between 0 EFPY and an assumed conservative EPU implementation value determined to be 33.4 EFPY. This value is the calculated EFPY for MNGP up to April 2011 (the 2011 Refueling Outage). Intermediate EFPY hydrostatic / pressure test curves were developed for 36 EFPY and 40 EFPY.(14) The intermediate 36 EFPY and 40 EFPY hydrostatic / pressure test curves were calculated up to 33.4 EFPY without application of the 1.3 multiplication factor and beyond 33.4 EFPY with application of the factor. Even with the 1.3 multiplication factor for past operation removed, the fluence values are still conservative because prior operation is bounded by the pre-EPU flux. Note also that these fluence values are still calculated in accordance with NRC Regulatory Guide 1.190. The resulting fluence values calculated for 36, 40 and 54 EPFY are:

Table B – Hydrostatic / Pressure Test Curve Development

Area/Component	Component Fluence in 10^{18} n/cm^2		
	36 EFPY	40 EFPY	54 EFPY(14)
Upper/Intermediate Shell Plates I-12 / I-13	0.197	0.230	0.406
Lower/Intermediate Shell Plates I-14 / I-15	2.77	3.36	6.43
Lower Shell Plates I-16 / I-17	1.85	2.28	4.46
Limiting Weld - Beltline	2.77	3.36	6.43
Bounding N-2 Nozzle	0.427	0.523	1.01

In summary, the P-T curves for plant operation – core not critical (Curve B), core critical (Curve C), and the 54 EFPY hydrostatic / pressure test curve (Curve A), were developed applying a 1.3 multiplication factor to the fluence over the full assumed RPV lifetime (0 to 54 EFPY). Due to hydrostatic test performance

14. The 54 EFPY fluence for hydrostatic / pressure test curve (Curve A) was calculated applying the 1.3 factor from 0 to 54 EFPY. EPU operation for the 54 EFPY hydrostatic / pressure test curve (Curve A) was assumed to start at 28.82 EFPY.

operational and safety considerations, 36 and 40 EFPY hydrostatic / pressure testing curves (Curve A), were developed where the 1.3 multiplication factor was applied only to the fluence above 33.4 EFPY. Application of this multiplication factor for fluence calculations up to the present is unnecessary since the fluence accumulation on the vessel is monitored and is known up to the current cycle.

4.5 Response to NRC RAIs Identified From Other Licensee PTLR Submittals

The NRC has identified a number of questions during review of other PTLR licensee submittals (see References 14, 15 and 16). The following responses are provided to proactively address likely questions determined to be applicable to the MNGP.

a) Small diameter, drill-hole type, instrument nozzles in the RPV beltline.

MNGP has no instrument or additional forged nozzles in the beltline region of the RPV. Therefore, no finite element analysis was performed for these nozzles.

b) Temperature and Pressure Instruments Uncertainty and the Pressure Head For Column of Water in the RPV

This is discussed in Enclosure 8, "Revised P-T Curves Calculation," Section 3.0, "Assumptions / Design Inputs" (SIA No. 1000847.303). The instrument uncertainty assumed in the analysis for pressure is 0 psig. The instrument uncertainty assumed in the analysis for temperature is 0°F. The instrument uncertainty is assumed to be zero since the temperature and pressure monitoring are procedurally controlled and margin is placed on these limits for monitoring vessel temperature and pressure conditions. Procedural controls will continue to include sufficient margin with the introduction of the PTLR.

The pressure head to account for the column of water in the RPV is 27.4 psig.

PART II – RESULTS OF THE PRESSURE / TEMPERATURE LIMIT ANALYSES

4.6 Recent Surveillance Specimen Results in Non-Conservative TS Figure 3.4.9-1 Beyond 2025

The BWRVIP ISP Data Source Book (BWRVIP-135) is periodically updated to include results of new surveillance capsule data. BWRVIP-135 indicates that the chemistry factor associated with MNGP plate heat number C2220 has increased from 130.8°F to 180°F (required margin decreased from 34°F to 17°F) due to the inclusion of the MNGP 300 degree surveillance capsule results. The newly determined ART for the end-of-life fluence increases from the current value of 157°F to 175°F.

Calculational results incorporating these changes indicate that current TS Figure 3.4.9-1, "Core Beltline Operating Limits Curve Adjustment Versus Fluence," will become non-conservative in approximately 2025 at the current rate of fluence accumulation. This condition is being treated as a non-conservative TS in accordance with NRC Administrative Letter 98-10.

Figure 3.4.9-1 is part of the current MNGP specific P-T methodology. This figure allows the core beltline operating limits to be adjusted for fluence under the current MNGP methodology. This adjustment capability is NOT part of the BWROG SIA methodology. Figure 3.4.9-1 will be eliminated following adoption by the MNGP of the BWROG SIA methodology with the approval of this amendment – precluding this condition in the future.

Results validate that in the interim, the operating curves currently used in the plant operating procedures are conservative. This is due to the fluence used to determine the operating curves being much higher then that currently accumulated by the RPV. Since approximately 13 years will pass before TS Figure 3.4.9-1 becomes non-conservative, ample time is provided for correction, and hence this condition is not an immediate concern.

4.7 Determination that Recirculation Inlet Nozzles are a Limiting Beltline Component

It was identified, and confirmed during development of the proposed P-T curves, that the recirculation inlet (N2) nozzles were the limiting components for the reactor vessel beltline region at low temperature / low pressure conditions. Previously it had been assumed that the recirculation inlet nozzles were bounded by other evaluated components and they were not considered in P-T curve development. This condition is being treated as a non-conservative TS in accordance with NRC Administrative Letter 98-10.

Although the current P-T curves are non-conservative with respect to low temperature / low pressure conditions for the recirculation inlet nozzles, this is acceptable in the interim until corrected for the following reasons. First, the MNGP operates well within the limits of the P-T curves since the plant operates on the saturation curve during normal operation and during heat-up / cool-down evolutions. The only time the P-T limits could be challenged is during RPV hydrostatic / leak testing.

Second, in the interim until the new P-T curves are approved, the curves in operating procedures have been adjusted to reflect incorporation of the recirculation inlet nozzles and conservative limits established. Test data for ten-years of hydrostatic / leak tests were reviewed and it was determined that the interim operating limits (which include the recirculation inlet nozzles) have not been violated during test performance.

Third, procedures for performing hydrostatic / leak testing provide additional assurance that operating limits will not be violated by defining the minimum allowable temperature for pressurization at test conditions and procedural steps to control test conditions above that point.

In summary, during development of the proposed P-T curves, the recirculation inlet (N2) nozzles were determined to be limiting components for the reactor vessel beltline region at low temperature / low pressure conditions. This is a non-conservative TS condition. P-T limits could be challenged during hydrostatic testing. Procedural controls were put in place to provide operator guidance precluding test performance in this operating region of the hydrostatic pressure test curves.

4.8 Addition of Upper-Intermediate Plates to the Beltline Region

During preparation of the new P-T limit curves, it was identified that the upper-intermediate plates were calculated to have absorbed sufficient fluence to be considered in the beltline region of the reactor vessel. Previously, only the lower and lower-intermediate plates, and the recirculation inlet nozzles, were considered to be in the vessel beltline region.

Accordingly, these were the only areas evaluated for changes in ART which determines P-T limits.(15) A review of chemistry and fluence values indicated that the ART for the limiting plates, i.e., the lower-intermediate plates, is approximately 100°F higher than the ART for the upper-intermediate plates. Because the ART of the lower-intermediate plates is limiting, i.e., higher than the upper-intermediate plates, this discovery has no effect on the P-T curves and will not have an effect on future P-T curves. Because the upper-intermediate plates, however, are now included within the vessel beltline, they were evaluated in accordance with the guidance of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2 (Reference 17), and hence included in the PTLR.

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15. Linear interpolation identified that the upper-intermediate plates entered into the beltline region at approximately 12 EFPY. MNGP at the 2011 RFO was at approximately 33.4 EFPY.

In summary, the upper-intermediate plates were determine to have absorbed sufficient fluence to be considered in the beltline region but the ART of the lower-intermediate plates remains limiting, and this condition has no effect on the proposed present and future P-T curves. The upper-intermediate plates, because they are now included within the vessel beltline, are required to be evaluated now and in the future and will be included in the PTLR.

4.9 RPV Flange and Adjacent Shell Temperature Limit

The current TS indicate an RPV flange and adjacent shell temperature limit of greater than or equal to 70°F. The method used in development of the PTLR, in accordance with the SIA LTR, determines that the RPV flange and adjacent shell temperatures has a limiting value of the flange RT_{NDT} or 60°F, whichever is greater. Sixty-degrees Fahrenheit (60°F) was determined to be the new temperature limit.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Northern States Power Company – Minnesota (NSPM) requests an amendment to the facility Renewed Operating License DPR-22, for the Monticello Nuclear Generating Plant (MNGP). It is proposed to revise the MNGP Technical Specifications (TS) to incorporate a Pressure and Temperature Limits Report (PTLR). The proposed changes would replace the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P-T) limit curves with references to the PTLR. Relocation of the P-T limit curves to the PTLR is consistent with the guidance provided in NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." These new curves have been developed applying the analytical methodology described in Structural Integrity Associates (SIA) Report SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," which has received NRC approval. The new P-T curves are valid for Extended Power Uprate (EPU) operation through the renewed operating license period.

NSPM has evaluated the proposed amendment in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the MNGP in accordance with the proposed amendment presents no significant hazards. NSPM's evaluation against each of the criteria in 10 CFR 50.92 follows.

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes modify the MNGP TS by replacing references to existing reactor vessel heatup and cooldown rate limits and P-T limit curves with references to the PTLR. The proposed license amendment requests adoption of the NRC approved methodology of SIA report SIR-05-044-A for preparation of MNGP P-T limit curves. The MNGP PTLR was developed based on the methodology and template provided within the SIA report.

   10 CFR 50 Appendix G establishes requirements to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Implementing the NRC approved SIA methodology for calculating P-T limit curves and relocating those curves to the PTLR provide an equivalent level of assurance that RCPB integrity will be maintained, as required by 10 CFR 50 Appendix G.

   Additionally, 10 CFR Part 50, Appendix H, provides the NRC criteria for design and implementation of reactor pressure vessel (RPV) material surveillance programs for operating lightwater reactors. Implementing this NRC approved SIA methodology does not reduce the ability to protect the RCPB as specified in Appendix G, nor will this change increase the probability of malfunction of plant equipment, or the failure of plant structures, systems, or components. Incorporation of the new methodology for calculating P-T curves, and the relocation of the P-T curves from the TS to the PTLR provides an equivalent level of assurance that the RCPB is capable of performing its intended safety functions.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The change in methodology for calculating P-T limits and the relocation of those limits to the PTLR do not alter or involve any design basis accident initiators. RCPB integrity will continue to be maintained in accordance with 10 CFR 50 Appendix G, and the assumed accident performance of plant structures, systems and components will not be affected. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), and the installed equipment is not being operated in a new or different manner.

Accordingly, no new failure modes are introduced which could introduce the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes do not affect the function of the RCPB or its response during plant transients. Calculating the MNGP P-T limits using the NRC approved SIA methodology, ensures adequate margins of safety relating to RCPB integrity are maintained. The proposed changes do not alter the manner in which the Limiting Conditions for Operation P-T limits for the RCPB are determined. There are no changes to the setpoints at which protective actions are initiated, and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NSPM has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

5.2 <u>Applicable Regulatory Requirements</u>

The NRC has established requirements in 10 CFR 50, Appendix G, "Fracture Toughness Requirements," in order to protect the integrity of the RCPB in nuclear power plants. Appendix G requires that the P-T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code were used to generate the P-T limits. Also, Appendix G requires that applicable surveillance data from reactor pressure vessel (RPV) material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in the TSs which includes limiting conditions for operation (LCO's), surveillance requirements and administrative controls. Previously the plant-specific P-T limits had been incorporated into the TS and controls were placed on operation and testing by the associated specification. This proposed change revises the TS to relocate to a licensee controlled document the P-T limit curves in accordance with the guidance of Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 2) and TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (Reference 3).

MNGP was designed before the publishing of the 70 General Design Criteria for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission (AEC) for public comment in July 1967, and constructed prior to the 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. As such, the MNGP was not licensed to the Appendix A, General Design Criteria (GDC).

MNGP USAR, Section 1.2, lists the principal design criteria (PDCs) for the design, construction and operation of the plant. USAR Appendix E provides a plant comparative evaluation to the 70 proposed AEC design criteria. It was concluded that the plant conforms to the intent of the 70 proposed AEC GDCs. The applicable PDCs, July 1967 - 70 AEC GDCs, and applicable current 10 CFR 50, Appendix A GDC are discussed below.

- <u>PDC 1.2.11 -- Class I Equipment and Structures</u>

  Class I structures, systems and components are those whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant under normal or accident conditions and to the removal of decay and sensible heat from the reactor.

- <u>AEC 70 GDC 33 -- Reactor Coolant Pressure Boundary Capability</u>

  The reactor coolant pressure boundary shall be capable of accommodating without rupture and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

- GDC 14 -- Reactor coolant pressure boundary.

  The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 15 -- Reactor coolant system design.

  The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- AEC 70 GDC 35 -- Reactor Coolant Boundary Brittle Fracture Prevention

  Under conditions where reactor coolant pressure boundary system components constructed of Ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120° F above the nil ductility transition (NDT) temperature of the component material if the resulting energy is expected to be absorbed within the elastic strain energy range.

- AEC 70 GDC 34 -- Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevent

  The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties if materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loading, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

- GDC 31 – Fracture prevention of reactor coolant pressure boundary.

  The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- AEC 70 GDC 36 -- Reactor Coolant Pressure Boundary Surveillance

  Criteria 36 - Reactor Coolant Pressure Boundary (Category A) Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leak tight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

- GDC 32 – Inspection of reactor coolant pressure boundary.

  Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

NSPM has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. It was concluded that the proposed TS changes will continue to assure that the design requirements and acceptance criteria of MNGP pressure / temperature reload limit analyses are met. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this TS change, is unaffected.

5.3 Precedent

The NRC has approved similar license amendments to relocate the P-T limit curves to a PTLR. Recent examples for BWR plants include:

- Oyster Creek Nuclear Generating Station, License Amendment No. 269 issued September 30, 2008 (Reference 18).
- James A. Fitzpatrick Nuclear Power Plant, License Amendment No. 292 issued October 3, 2008 (Reference 19).
- Nine Mile Point Nuclear Station, Unit No. 1, License Amendment No. 204 issued January 21, 2010 (Reference 20).

6.0 ENVIRONMENTAL EVALUATION

NSPM has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility or component located within the restricted area, as defined in 10 CFR 20, nor would it change an inspection or surveillance requirement in such a way that it does not meet the following criteria. The proposed amendment does not involve (i) a significant hazards consideration, or (ii) authorize a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, the NSPM concludes pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. SIR-05-044-A, “Pressure-Temperature Limits Report Methodology for Boiling Water Reactors,” dated April 2007.

2. U.S. NRC Generic Letter 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” dated January 31, 1996.

3. TSTF-419-A, “Revise PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS [Reactor Coolant System] PTLR.”

4. NRC letter from W. Becker to A. Pietrangelo (NEI), dated March 21, 2002.” [Letter documents approval of three travelers – including TSTF-419.]

5. Letter from H. K. Nieh (NRC) to R. C. Bunt (Southern Nuclear Operating Company), "Final Safety Evaluation for the Boiling Water Reactor Owners' Group (BWROG) Structural Integrity Associates Topical Report (TR) SIR-05-044, "Pressure Temperature Report Methodology for Boiling Water Reactors" (TAC No. MC9694)", dated February 6, 2007. (ADAMS Accession No. ML053560336)

6. NSPM to NRC letter, “License Amendment Request: Extended Power Uprate (TAC MD9990),” letter number L-MT-08-052, dated November 5, 2008.

7. NSPM to NRC letter, “License Amendment Request: Maximum Extended Load Line Limit Analysis Plus,” letter number L-MT-10-003, dated January 21, 2010.

8. NSPM to NRC letter, “Xcel Energy Request for NRC Concurrent Review of Monticello Nuclear Generating Plant Maximum Extended Load Line Limit Analysis Plus (MELLLA+) License Amendment Request (LAR) with Extended Power Uprate (EPU) LAR Review Delay (L-MT-09-100),” dated October 28, 2009.

9. NRC to NSPM letter, “Monticello Nuclear Generating Plant - Linking of the Proposed Extended Power Uprate Amendment and the MELLLA+ Amendment (TAC NOS. MD9990 and ME2449),” dated November 23, 2009. (ADAMS Accession No. ML093160816)

10. U.S. NRC SECY-11-0014, “Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents,” dated January 31, 2011.

11. U.S. NRC, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

12. GE Hitachi Report NEDC-32983P-A, Revision 2, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," dated January 2006.

13. Monticello Nuclear Generating Plant Calculation 2011-039, "Monticello Neutron Flux and Fluence Evaluation for Extended Power Uprate," Revision 0.

14. AmerGen Energy Company, LLC to NRC letter, "Response to Request for Additional Information Concerning Technical Specification Change Request No. 348 - Relocation of Pressure and Temperature (P-T) Curves to the Pressure and Temperature Limits Report (PTLR)," dated June 30 2008.

15. Entergy to NRC letter, "James A. FitzPatrick Nuclear Power Plant - Response to Request for Additional Information Regarding Relocation of Pressure and Temperature, Curves to the Pressure and Temperature Limits Report (TAC No. MD8556)," dated July 2, 2008.

16. CENG to NRC letter, "License Amendment Request Pursuant to 10 CFR 50.90: Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report - Response to NRC Request for Additional Information (TAC No. ME0817)," dated December 17, 2009.

17. U.S. NRC, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

18. NRC to AmerGen Energy Company, LLC (C. Pardee) letter, "Oyster Creek Nuclear Generating Station – Issuance of Amendment Re: Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report (TAC No. MD8253)," dated September 30, 2008. (ADAMS Accession No. ML082390685)

19. NRC to Entergy Nuclear Operations, LLC letter, "James A. Fitzpatrick Nuclear Power Plant – Issuance of Amendment Re: Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report Consistent With TSTF-419-A (TAC No. MD8556)," dated October 3, 2008. (ADAMS Accession No. ML082630385).

20. NRC to Nine Mile Point Nuclear Station, LLC letter, "Nine Mile Point Nuclear Station, Unit No.1 – Issuance of Amendment Regarding Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report (TAC No. ME0817)," dated January 21, 2010. (ADAMS Accession No. ML093370002)

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE TECHNICAL SPECIFICATIONS TO INCLUDE A PRESSURE TEMPERATURE LIMITS REPORT

MARKED-UP TECHNICAL SPECIFICATION PAGES

(9 pages follow)

1.1 Definitions

OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.5.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1775 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from initiation of any RPS channel trip to the de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within ~~limits~~ the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Required Action A.2 shall be completed if this Condition is entered. -------------------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. ------------NOTE------------ Required Action C.2 shall be completed if this Condition is entered. -------------------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-----NOTES----- ~~1.~~ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ~~2. Figures 3.4.9-2 and 3.4.9-3 shall be adjusted as required by Figure 3.4.9-1 for the reactor vessel shell and fluid temperature limits.~~ ---------- Verify: a. RCS pressure and RCS temperature are within the applicable limits specified in ~~Figures 3.4.9-2 and 3.4.9-3; and~~ the PTLR; and b. RCS heatup and cooldown rates are within the limits specified in the PTLR. ~~≤ 100°F averaged over a 1 hour period.~~	30 minutes
SR 3.4.9.2	-----~~NOTE~~----- ~~Figure 3.4.9-4 shall be adjusted as required by Figure 3.4.9-1 for the reactor vessel shell fluid temperature limits.~~ ---------- Verify RCS pressure and RCS temperature are within the criticality limits specified in ~~Figure 3.4.9-4.~~ the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.3	-----NOTE----- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ----- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~≤ 50°F.~~ within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	-----NOTE----- Only required to be performed when tensioning the reactor vessel head bolting studs. ----- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F.~~ within the limits specified in the PTLR.	30 minutes
SR 3.4.9.5	-----NOTE----- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. ----- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F.~~ within the limits specified in the PTLR.	30 minutes
SR 3.4.9.6	-----NOTE----- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. ----- Verify reactor vessel flange and head flange temperatures are ~~≥ 70°F.~~ within the limits specified in the PTLR.	12 hours

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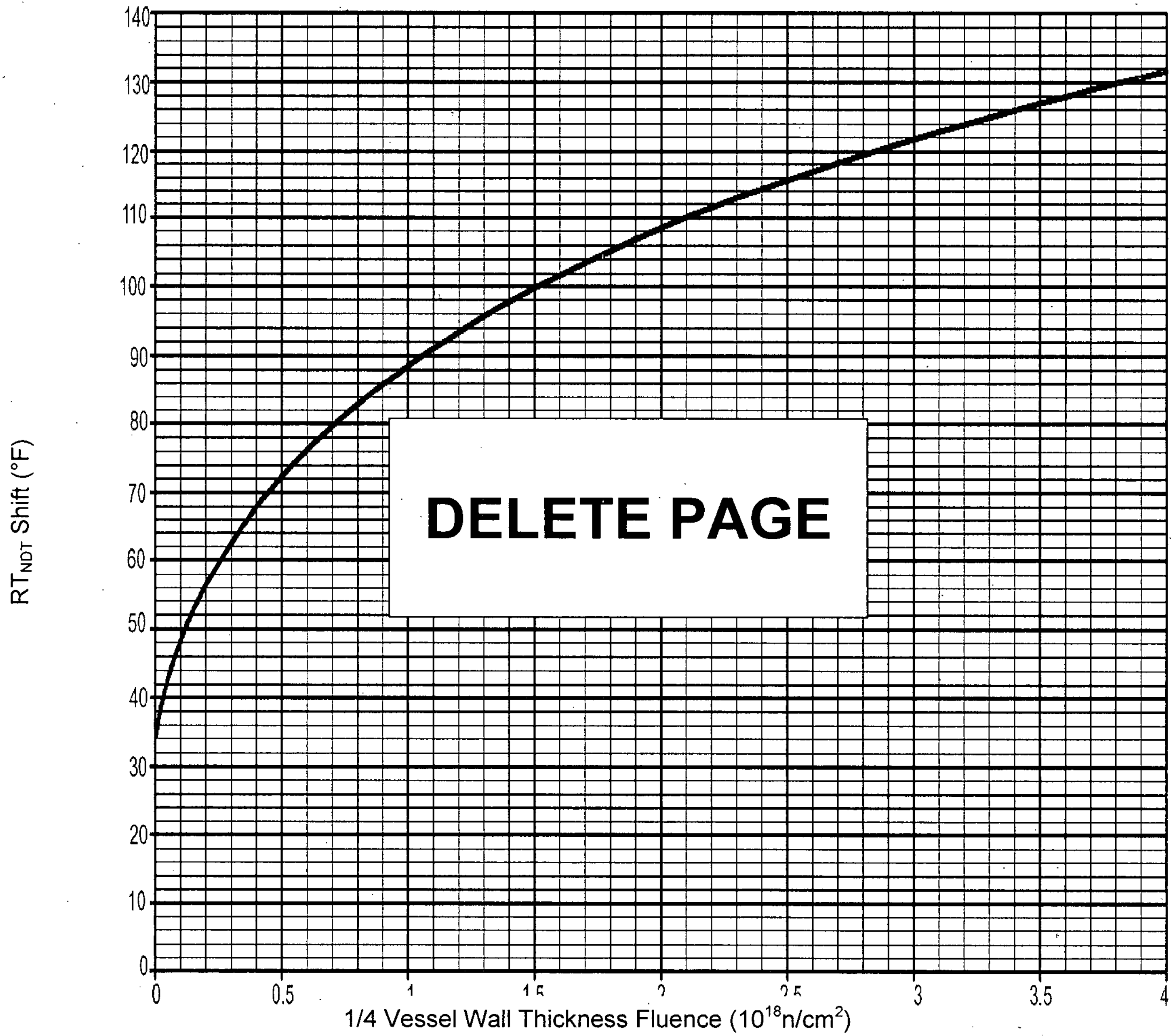

Figure 3.4.9-1
Core Beltline Operating Limits Curve Adjustment Versus Fluence

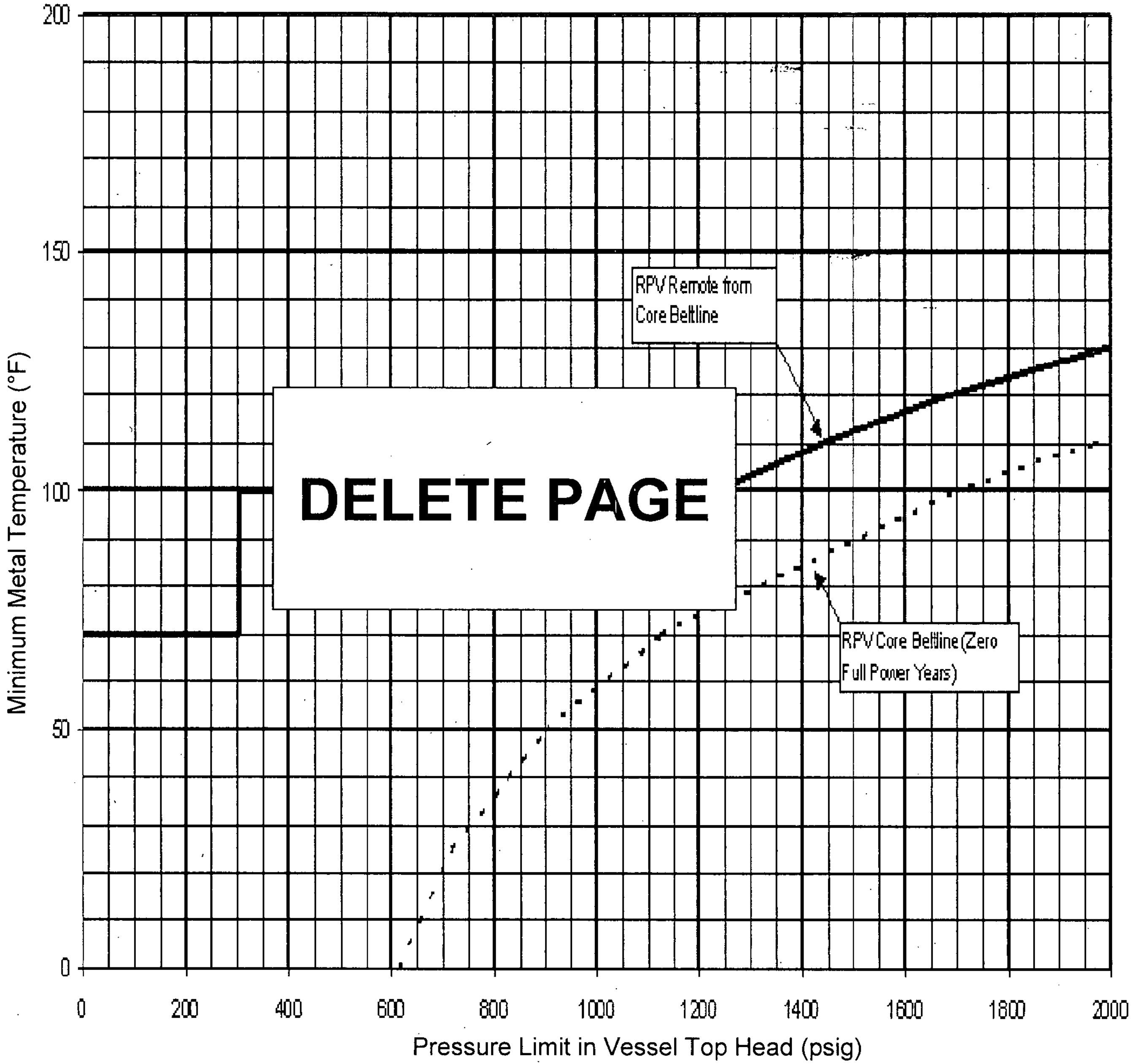

Figure 3.4.9-2
RCS Pressure Versus Temperature Limits
Inservice Leak and Hydrostatic Testing

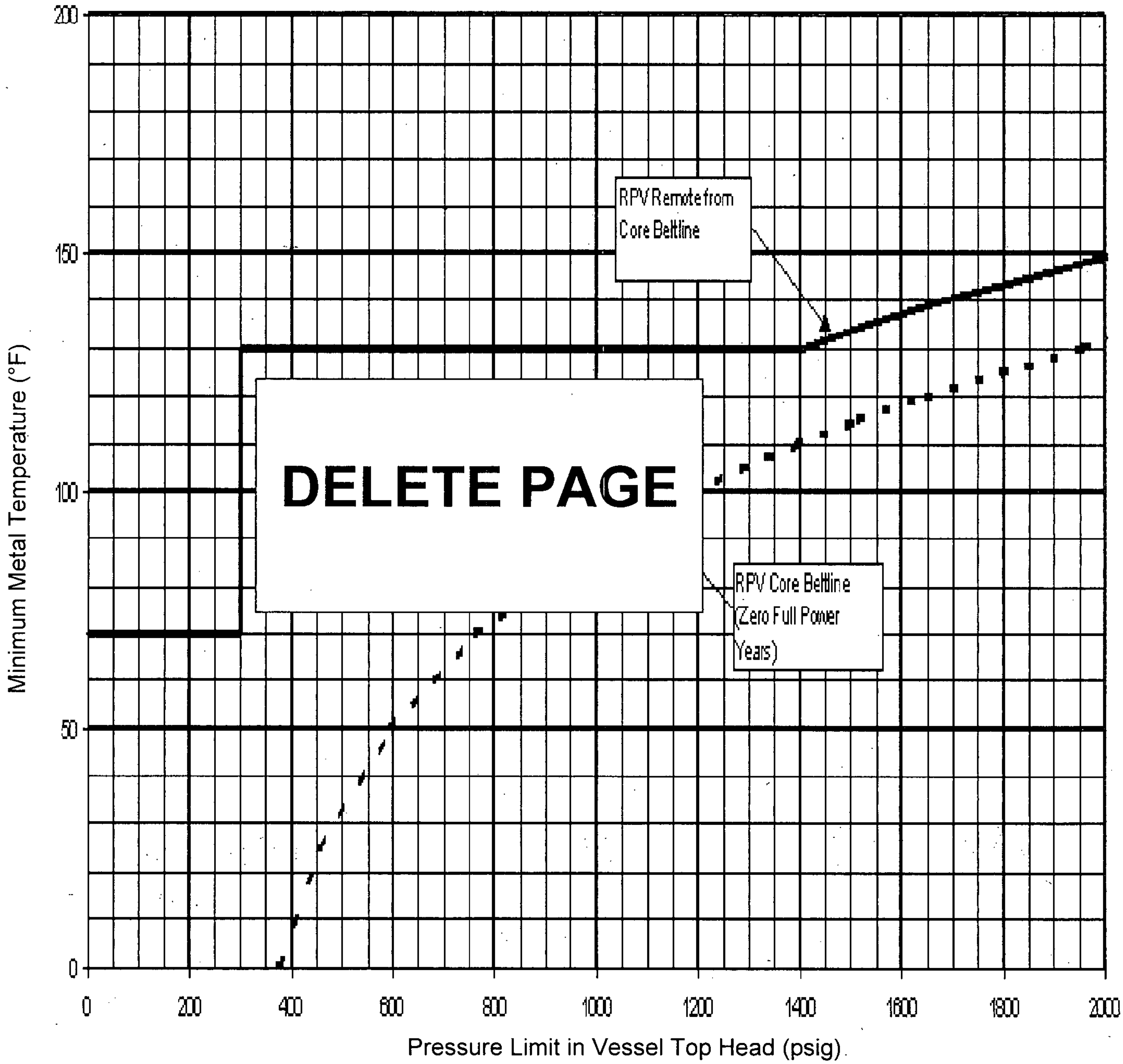

Figure 3.4.9-3
RCS Pressure Versus Temperature Limits
Non-Nuclear Heatup and Cooldown

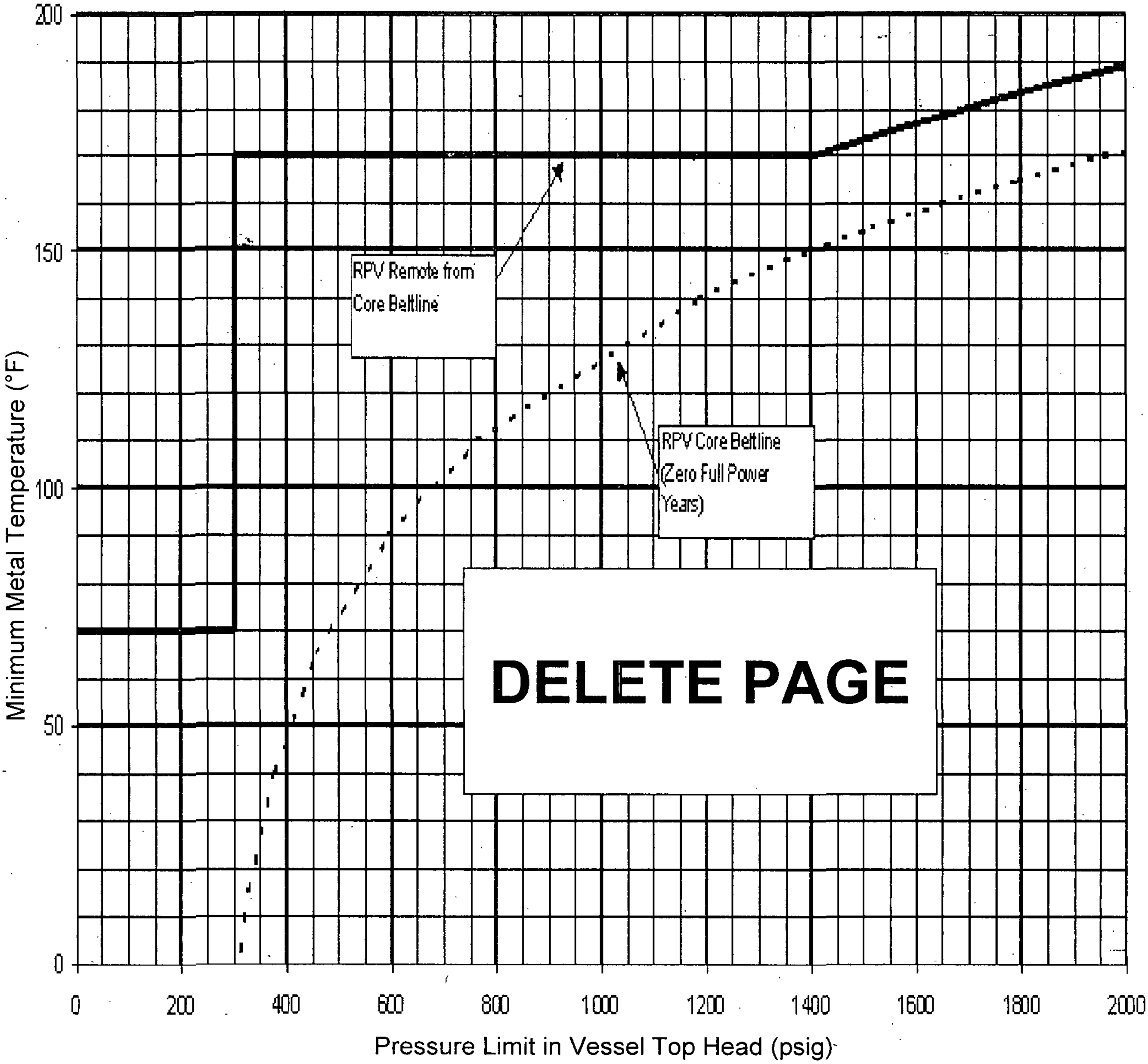

Figure 3.4.9-4
RCS Pressure Versus Temperature Limits
Critical Operation

5.6 Reporting Requirements

5.6.4 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

   2. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," Revision 0, dated April 2007.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

<u>ENCLOSURE 3</u>

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISE THE TECHNICAL SPECIFICATIONS TO INCLUDE A PRESSURE TEMPERATURE LIMITS REPORT

DRAFT TECHNICAL SPECIFICATION BASES CHANGES

(10 pages follow)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The ~~Specification~~ PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic testing, and criticality, and also limits for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and recommendations of Regulatory Guide 1.99, Revision 2 (Ref. 5).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

BASES

BACKGROUND (continued)

During inservice leak and hydrostatic testing, the reactor vessel shell temperatures (reactor vessel shell adjacent to shell flange, reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region) shall be at or above the temperatures specified in the PTLR. ~~shown on the two curves of Figure 3.4.9-2, where the dashed curve, "RPV Core Beltline (Full Power Years)," is increased by the core beltline temperature adjustment from Figure 3.4.9-1.~~ The reactor vessel bottom head temperature shall be at or above the temperatures specified in the PTLR. ~~shown on the solid curve of Figure 3.4.9-2, "RPV Remote from Core Beltline," with no adjustment from Figure 3.4.9-1.~~ During heatup by non-nuclear means and cooldown following nuclear shutdown the RCS temperatures (reactor vessel shell adjacent to shell flange, reactor vessel bottom drain, recirculation loops A and B, reactor vessel bottom head) shall be at or above the higher of the temperatures specified in the PTLR. ~~of Figure 3.4.9-3 where the dashed curve, "RPV Core Beltline (Zero Full Power Years)," is increased by the expected shift in RT_{NDT} from Figure 3.4.9-1.~~ The shift specified in the PTLR. ~~shown in Figure 3.4.9-1~~ includes both the delta RT_{NDT} and margin, as required by Regulatory Guide 1.99, Revision 2.

The P/T criticality limits include the Reference 1 requirement that they be at least 40°F above the non-critical heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leak and hydrostatic testing. During all operation with a critical reactor, the RCS temperatures (reactor vessel shell adjacent to shell flange, reactor vessel bottom head, and reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region) shall be at or above the higher of the temperatures specified in the PTLR. ~~of Figure 3.4.9-4 where the dashed curve, "RPV Core Beltline (Zero Full Power Years)," is increased by the expected shift in RT_{NDT} from Figure 3.4.9-1.~~

~~Reference 7~~ Reference X approved the P/T curves required by this Specification.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the limits specified in the PTLR, ~~Figures 3.4.9-2 and 3.4.9-3,~~ and during RCS heatup, cooldown, and inservice leak and hydrostatic testing, heatup and cooldown rates are within the limits specified in the PTLR; ~~≤ 100°F averaged over a 1 hour period;~~

b. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is within the limit specified in the PTLR ~~≤ 50°F~~ during recirculation pump startup;

c. RCS pressure and temperature are within the criticality limits specified in the PTLR ~~Figure 3.4.9-4~~; and

d. The reactor vessel flange and the head flange temperatures are within the limits specified in the PTLR ~~≥ 60°F~~ when tensioning the reactor vessel head bolting studs and when the reactor head is tensioned.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

BASES

LCO (continued)

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existence, size, and orientation of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline. Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

Verification that operation is within limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

The following locations must be monitored during RCS heatup and cooldown operations: a) reactor vessel shell adjacent to shell flange; b) reactor vessel bottom drain; c) recirculation loops A and B; and d) reactor vessel bottom head. The following locations must be monitored during inservice leak and hydrostatic testing: a) reactor vessel shell adjacent to shell flange; b) reactor vessel bottom head; and c) reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region.

Surveillance for heatup and cooldown may be discontinued when three consecutive measurements at each location are within 5°F. Surveillance for inservice leak and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified by ~~two~~ a Notes. The Note ~~1~~ requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leak and hydrostatic testing. ~~Note 2 requires Figures 3.4.9-2 and 3.4.9-3 to be adjusted as required by Figure 3.4.9-1 (core beltline temperature adjustment) for the reactor vessel shell and fluid temperatures. This ensures the proper RCS pressure and temperature limits are met based on reactor fluence.~~

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The following locations must be monitored to verify compliance with the P/T criticality curve limits: a) reactor vessel shell adjacent to shell flange; b) reactor vessel bottom drain; c) recirculation loops A and B; and d) reactor vessel bottom head.

~~This SR has been modified by a Note that requires Figure 3.4.9-4 to be adjusted as required by Figure 3.4.9-1 (core beltline temperature adjustment) for the reactor vessel shell and fluid temperatures. This ensures the proper RCS pressure and temperature limits are met based on reactor fluence.~~

SR 3.4.9.3

Differential temperature within the limit ensures that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances at the reactor nozzles and bottom head region. In addition, compliance with this limit ensures that the assumption of the analysis for the startup of an idle recirculation loop (Ref. 8) is satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limit will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.3 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 has been modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, the ΔT limit is not required. The Note also states the SR is only required to be met during a recirculation pump startup, since this is when the stresses occur.

SR 3.4.9.4, SR 3.4.9.5, and SR 3.4.9.6

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the ~~specified limits.~~ limits specified in the PTLR.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.4 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.5 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature ≤ 80°F in MODE 4. SR ~~3.4.9.5~~ 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature ≤ 100°F in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-66, 1961.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. ~~Letter from Darl Hood (NRC) to Jeffrey S. Forbes (NMC), "Monticello Nuclear Generating Plant - Issuance of Amendment 133, Revision to Pressure-Temperature Curves," dated February 24, 2003.~~ (Not used)
8. GE Service Information Letter No. 517, Supplement 1, "Analysis Basis for Idle Recirculation Loop Startup," dated August 26, 1998.

BASES

REFERENCES (continued)

9. SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," (latest approved version, see PTLR).

10. Amendment XX, (Approving PTLR for Monticello).