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December 29, 2023

L-MT-23-047
10 CFR 50.90

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

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

License Amendment Request: Revision to the MNGP Pressure Temperature Limits Report to Change the Neutron Fluence Methodology and Incorporate New Surveillance Capsule Data

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requests a license amendment to replace the current neutron fluence methodology with a newer methodology, and to revise the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP). The TS change updates Specification 5.6.5, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to reflect the current version (Revision 1) of the Structural Integrity Associates (SIA) methodology report SIR-05-044-A, "Pressure Temperature Limits Report Methodology for Boiling Water Reactors."

The current MNGP PTLR hydrostatic pressure and leak test curve has minimal margin to the 212°F operating restriction for reactor pressure vessel (RPV) testing. To gain additional operational margin NSPM proposes to revise the PTLR. Two substantial changes are proposed for this PTLR revision: first, replacement of the current neutron fluence methodology with the TransWare Enterprises, Inc., Radiation Analysis Modeling Application as the licensing basis methodology, and second, incorporation of new MNGP plant-specific surveillance capsule data.

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Attachment 1 to the enclosure provides the existing TS page marked-up to show the proposed change. Attachment 2 to the enclosure provides the retyped TS page.

Enclosure 2 provides a copy of the revised MNGP PTLR report. Enclosure 3 provides a copy of the calculation for the adjusted reference temperatures and reference temperature shifts for the RPV components. Enclosure 4 provides a copy of the calculation for generating the MNGP PTLR curves.

In accordance with 10 CFR 50.91, “Notice for public comment; State consultation” paragraph (b), NSPM is notifying the State of Minnesota by providing a copy of this application, with this enclosure and attachments, to the State of Minnesota designated official.

NSPM requests issuance of this proposed license amendment within twelve months following completion of NRC acceptance review.

If there are any questions or if additional information is needed, please contact Mr. Richard Loeffler at (612) 342-8981 or Rick.A.Loeffler@xcelenergy.com.

Summary of Commitments

This letter makes no new commitments and no revisions to any existing commitments.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on December 29, 2023.

Sara L. Scott Digitally signed by Sara L. Scott
Date: 2023.12.29 13:50:53 -06'00' for

Shawn Hafen
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures / Attachments

cc: Administrator, Region III, US NRC
Project Manager, Monticello, US NRC
Resident Inspector, Monticello, US NRC
State of Minnesota

LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

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<u>Encl. 3.0</u>	“Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts” (NSPM Calculation No. 22-035) (SI No. 2100300.302, Revision 4)
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LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), is submitting a license amendment request (LAR). This LAR replaces the current neutron fluence methodology with a newer methodology, described further below, and revises the Technical Specifications (TS) for the Monticello Nuclear Generating Plant (MNGP) to reflect the most recent version of a Structural Integrity Associates (SIA), Inc., licensing topical report (LTR) for developing a Pressure Temperature Limits Report (PTLR). The specific proposed change to Specification 5.6.5, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” updates the methodology listed under item 5.6.5.b.1 from the 2007 (Revision 0) to the current, 2013 version (Revision 1), of the SIA LTR SIR-05-044-A, “Pressure Temperature Limits Report Methodology for Boiling Water Reactors” (Reference 1).

The current MNGP PTLR hydrostatic pressure and leak test curve has minimal margin to the 212°F operating restriction for reactor pressure vessel (RPV) testing. To gain additional operating margin for the remainder of the current renewed operating license period, NSPM proposes to revise the PTLR. Two substantial changes are included in this proposed PTLR revision. First, the current General Electric neutron fluence methodology is replaced by the TransWare Enterprises, Inc., (hereafter TransWare) Radiation Analysis Modeling Application (RAMA) as the licensing basis methodology to estimate RPV fluence. Second, results from evaluation of the MNGP 120-degree surveillance capsule removed during the spring 2021 MNGP Refueling Outage (RFO) are incorporated into the supporting analyses.

2.0 DETAILED DESCRIPTION

2.1 Background

In February 2013, through Amendment No. 172 (Reference 2) NSPM was approved to revise the MNGP TS in accordance with the guidance of NRC Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” (Reference 3) and Technical Specification Task Force (TSTF) – TSTF-419, “Revise PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS PTLR” (Reference 4) to revise and relocate the MNGP Pressure Temperature (P-T) limit curves to a PTLR.

The 2013 amendment revised the P-T limits based on the methodology documented in Revision 0 of the SIA LTR SIR-05-044-A (Reference 5). The fast neutron fluence calculations supporting that amendment were performed in accordance with the established General Electric calculational methodology (Reference 6). The PTLR (currently Revision 1) is applicable for the MNGP through the end of the license renewal period, i.e., up to 54 effective full power years (EFPY).

2.2 System Design and Operation

The Limiting Condition for Operation (LCO) of Specification 3.4.9, “RCS Pressure and Temperature (P/T) Limits,” establishes operating limits that provide margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). 10 CFR 50, Appendix G, “Fracture Toughness Requirements,” specifies material fracture toughness requirements for ferritic materials of the RCPB of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences (AOOs) and during system hydrostatic tests. These P-T limits are instituted through this specification which directs reference to the PTLR (and the curves and associated tables therein).

Each P-T limit curve defines an acceptable region for plant operation. The usual use of the curves is for operational guidance during heatup or cooldown operations and during AOOs – with the reactor being in a critical condition when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable operating region. The curves also provide guidance during certain other pressure testing conditions (i.e., inservice leak rate testing and / or hydrostatic testing).

Due to the effects of neutron irradiation embrittlement accumulated by the reactor, the P-T limit curves contained in plant TSs are updated periodically to ensure that the limit curves are always valid beyond the EFPYs that the plant has accumulated.

2.3 Current Technical Specification Requirements

Specification 3.4.9 provides the reactor coolant system pressure, temperature, heatup, and cooldown rates, and the recirculation pumps starting temperature requirements be maintained within limits through their specification within the PTLR and provides the Conditions, Required Actions, and Completion Times that must be met in order to maintain the required safety margins. The Conditions, Required Actions, and Completion Times remain unchanged by this proposed LAR. The proposed change revises Specification 5.6.5, “Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),” which lists under item 5.6.5.b the 2007 (Revision 0) version of the SIA LTR SIR-05-044-A as the NRC approved PTLR methodology basis.

2.4 Reason for the Proposed Changes

The current MNGP PTLR hydrostatic pressure and leak test curve has minimal margin to the 212°F operating restriction for RPV testing. The 120-degree surveillance capsule was removed during the spring 2021 MNGP RFO. The development of the new RAMA neutron fluence projections for the RPV combined with the surveillance capsule results support development of a revised PTLR that exhibits increased operational margin for the hydrostatic pressure and leak test curve, which could be used for the remainder of the current license renewal period.

2.5 Description of the Proposed Licensing Basis Changes

The licensing basis changes proposed in this LAR update the neutron fluence projections for the RPV based on new surveillance capsule data and new neutron fluence projections with the RAMA methodology. The TS administrative specification, i.e., Specification 5.6.5, governing development and approval of the PTLR, is being revised from reflecting the 2007 (Revision 0) to the 2013 (Revision 1) version of the PTLR methodology topical report, which is the current version of the SIR 05-044-A LTR. There are no changes to Specification 3.4.9, “RCS Pressure and Temperature (P/T) Limits.”

Attachment 1 to this enclosure provides the existing TS page marked-up to show the proposed change. Attachment 2 to this enclosure provides the retyped TS page.

3.0 TECHNICAL EVALUATION

3.1 General Discussion

10 CFR 50 Appendix G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including AOOs and during system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Due to the effects of neutron irradiation embrittlement accumulated by the reactor vessel, the P-T limit curves contained in plant TSs are updated periodically to ensure that the limit curves are always valid for beyond the EFPY that the plant has accumulated.

<u>P-T Curve Axes Inversion for the MNGP</u>

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 within the PTLR.

For the purposes of discussion within this LAR, however, NSPM will use the more common P-T curve nomenclature.

3.2 Development of the Revised P-T Limit Curves – Generic Letter (GL) 96-03 Considerations

The revised PTLR was developed in accordance with the template PTLR of the SIR-05-044-A LTR and meets the seven GL 96-03 criteria:

(1) Section 3.0, "Methodology," of the PTLR refers to the neutron fluence calculational methodology references and provides the values of neutron fluences used in the adjusted reference temperature (ART) calculation.

(2) Appendix A of the PTLR describes the MNGP reactor vessel materials surveillance program. The BWRVIP LTRs describe the administration of the material surveillance program including the surveillance capsule reports for the MNGP.

(3) Low Temperature Overpressure Protection System limits are not applicable to Boiling Water Reactors (BWRs).

(4) Section 3.0, "Methodology," of the PTLR describes the method for calculating the ART values using RG 1.99, Revision 2.

(5) Section 5.0, "Discussion," of the PTLR describes the application of fracture mechanics in the construction of P-T limits and provides information regarding the ANSYS finite element analyses for the feedwater nozzle (non-beltline) and recirculation inlet nozzle (beltline) performed to generate part of the P-T limits.

(6) Section 4.0, “Operating Limits,” of the PTLR discusses the minimum temperature requirements in 10 CFR 50, Appendix G which are applied to P-T limits for bolt-up temperature and hydrotest temperature. The SIR-05-044-A LTR provides detailed information regarding the minimum temperature requirements for bolt-up temperature and hydrotest temperature.

(7) Appendix A of the PTLR, which discusses the MNGP reactor vessel materials surveillance program, includes how multiple surveillance capsules are used in the ART calculation. The referenced reports and calculations describe how the data from multiple surveillance capsules are used and the determination of the chemistry factor from the surveillance data are used in the ART calculations.

3.3 Neutron Fluence Determination

10 CFR 50 Appendices G and H, “Fracture Toughness Requirements,” and “Reactor Vessel Material Surveillance Program Requirements,” respectively, present requirements that guide fluence determinations. Appendix G specifies fracture toughness requirements for the carbon

and low-alloy ferritic materials of the pressure-retaining components of the RCPB. Appendix H specifies requirements for a material surveillance program that serves to monitor changes in the fracture toughness properties of the ferritic materials in the reactor beltline region.

Implementing guidance addressing the two appendices is provided in two regulatory guides. US NRC Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Material,” (Reference 7) addresses the requirements of 10 CFR 50, Appendix G for determining the neutron fluence used in the evaluation of fracture toughness in light water nuclear reactor pressure vessel ferritic materials. The ART values for the limiting beltline materials were calculated (see Enclosure 3) in accordance with this regulatory guide.

RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” (Reference 8) addresses the requirements for determining the fast neutron fluence and uncertainty in the fluence predictions used in fracture toughness evaluations. The previous neutron fluence calculations supporting the P-T limits in the current PTLR (approved by Amendment 172 in 2013 (Reference 2)) were performed in accordance with the General Electric fluence methodology, NEDO-32983P-A, “Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations” (Reference 6). The present PTLR (currently Revision 1) is applicable for the MNGP through the end of the first license renewal period, i.e., up to 54 EFPY.

The RAMA fluence methodology was developed by TransWare under sponsorship for the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel Internals Program (BWRVIP), for the purpose of calculating fast neutron fluence in RPVs and reactor vessel internal components. As prescribed in RG 1.190, RAMA has been benchmarked and qualified against industry standard benchmarks for both BWR and pressurized water reactor (PWR) designs. In addition, RAMA has been compared with several plant-specific dosimetry measurements and reported fluence from several commercial operating reactors. The results of the benchmarks and comparisons to measurements show that RAMA accurately predicts specimen activities, RPV fluence, and vessel component fluence in all light water reactor types. This prior work was extended in the Seabrook Station license renewal analysis (Reference 9) further validating the use of RAMA for all light water reactor designs.

The RAMA methodology has been used for determining fast neutron fluence in both BWR and PWR pressure vessels with no discernable bias in the computed results. Utilization of the RAMA fluence methodology is subject to several conditions, including that a plant geometry-specific validation must be performed, as discussed below.

In May 2005, the NRC issued a safety evaluation (SE), enclosed within EPRI report BWRVIP-114NP-A, “BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual, Final Report,” (Reference 10), that evaluated the RAMA fluence methodology. Subsection 4.1, “BWR RPV Neutron Fluence,” of the SE states, “the staff concludes that the BWRVIP methodology, as described in these reports, provides an acceptable best-estimate plant-specific prediction of the fast ($E \geq 1.0$ MeV) neutron fluence for BWR RPVs.” The conclusion goes on to state:

> With respect to the calculation of BWR RPV neutron fluence, the staff concludes that based on the plant-specific benchmark data presently available, no calculational bias is required for the application of the methodology to plants of similar geometrical design to Susquehanna and Hope Creek, i.e., BWR-IV plants. However, in order to provide continued confidence in the proposed neutron fluence methodology for the BWR RPVs, the acceptance of this methodology is subject to the following conditions for plants which do not have geometries similar to the cited BWR-IV's:
>
> - To apply the RAMA methodology to plant groups which have geometries that are different than the cited BWR-IV's, at least one plant-specific capsule dosimetry analysis must be provided to quantify the potential presence of a bias and assure that the uncertainty is within the RG 1.190 limits.
>
>   and
>
> - Justification is necessary for a specific application based on geometrical similarity to an analyzed core, core shroud, and RPV geometry. That is, a licensee who wishes to apply the RAMA methodology for the calculation of RPV neutron fluence must reference, or provide, an analysis of at least one surveillance capsule from a RPV with a similar geometry.

On January 9, 2023, NSPM submitted an application for subsequent license renewal (SLR) for the MNGP (Reference 11). Neutron fluence projections were performed for the MNGP RPV and reactor vessel internals components and plant structures applying the RAMA fluence methodology for the projected twenty-year period of subsequent extended operation, i.e., through 72 EFPY – conservatively bounding the 54 EFPY fluence assumed through the end of the current Monticello Renewed Facility Operating License. On July 11, 2023, NSPM submitted a third supplement to the SLR application (Reference 12) containing non-proprietary and proprietary versions of a TransWare topical report discussing the MNGP fluence methodology and qualification of that model (Reference 13). This TransWare report supported development of the MNGP PTLR discussed herein.

With respect to the indented section above describing conditions that must be met for application of the RAMA fluence methodology, this report also addresses the above conditions. Specifically, since the MNGP is a BWR-III plant design:

- Plant-specific capsules dosimetry analysis were provided to quantify the potential presence of a bias and ensure that the uncertainty is within the RG 1.190 limits.

- The MNGP RPV was modeled providing an analyzed core, core shroud, and RPV geometry.

- Also, Subsection 4.1, first paragraph of the SE states, “This acceptance is limited to the axial region defined by the core active fuel height.” However, since the 2005 SE, conservatisms in later model applications have allowed extension to include the extended beltline region.

In the 2005 NRC SE the RAMA fluence methodology was indicated as approved for the Susquehanna and Hope Creek BWRs for the applications discussed therein. Subsequently, the use of RAMA has been approved for several other BWR licenses as discussed in the precedents section of this LAR.

3.4 Reactor Vessel Material Surveillance Program

10 CFR 50 Appendix H requires a material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region which result from exposure to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. Section III of Appendix H specifies that a material surveillance program is required for light water nuclear power reactors if the peak fast neutron fluence with energy greater than 1 MeV (E > 1 MeV) at the end of the design life of the vessel is expected to exceed 10^{17} n/cm^2. Section III also allows for an Integrated Surveillance Program (ISP) in which representative materials for the reactor are irradiated in one or more other reactors of sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

MNGP is a member of the BWRVIP ISP which is administered by the EPRI and the BWR Owners’ Group. The ISP combines the domestic BWR surveillance programs into a single integrated program. This program uses similar heats of materials in the surveillance programs of various BWRs nuclear plants to represent the limiting materials in other BWR RPVs.

The scope of the program is described in the BWRVIP ISP guidance, and the technical basis of the program is described in BWRVIP-78, “BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan” (Reference 14). The ISP capsule removal schedule is included in BWRVIP-86, Revision 1-A, “BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program Implementation Plan,” (Reference 15). On April 22, 2003, NSPM committed to implement the BWRVIP ISP in place of its original surveillance programs for the MNGP in Amendment No. 135 (Reference 16). MNGP is currently operating in and is licensed to use the BWRVIP ISP during the Renewed License period of extended operation. [(1)]

1. Adoption of BWRVIP-321-A, “Boiling Water Reactor Vessel and Internals Project, Plan for Extension of the BWR Integrated Surveillance (ISP) Through the Second License Renewal (SLR),” is projected for the proposed SLR period.

The most recently removed surveillance capsule, the 120-degree surveillance capsule, was irradiated from initial startup through 30 cycles of operation before it was removed from the RPV during the spring 2021 refueling outage in accordance with 10 CFR 50, Appendix H. This was the last of the three surveillance capsules installed in the MNGP reactor. Results for this capsule are available in BWRVIP-347, “BWR Vessel and Internals Project, Testing and Evaluation of the Monticello 120° ISP(E) Surveillance Capsule” (Reference 17) and the test results will be added to the next revision of BWRVIP-135, “BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations,” (Reference 18) and were used in the preparation of the PTLR revision discussed and provided herein.

3.5 Chemistry and Adjusted Reference Temperature

Chemistry in the context of P-T curve calculation is related to the copper and nickel contents in the RPV shell material. The copper and nickel metal contents are needed to calculate the ART. The reactor vessel beltline copper and nickel values were obtained from the evaluation of the MNGP reactor vessel plate, weld, and forging materials in the SIA calculation for evaluation of the ART and reference temperature shifts which included the results of the three surveillance capsules. The copper and nickel values were used with Table 1, “Chemistry Factor for Welds, °F,” of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1, “Adjusted Reference Temperature,” of the guide for the welds. The copper and nickel values were used with Table 2, “Chemistry Factor for Base Metal, °F,” of RG 1.99 to determine a CF per Paragraph 1.1 of the guide for the plates and forgings.

Enclosure 3 provides a copy of the calculation describing the method for calculating the ART using RG 1.99, Revision 2.

3.6 MNGP Feedwater and Recirculation Inlet Nozzles Finite Element Analyses

Plant-specific MNGP feedwater nozzle (non-beltline) and recirculation inlet nozzle (beltline) analyses were performed to determine through-wall pressure stress distributions and thermal stress distributions due to bounding thermal transients. The results are design inputs providing the quarter-T nozzle stress intensity factors for the feedwater and recirculation inlet nozzles in Enclosure 4.

3.7 Pressure-Temperature Limit Curves Generation for 72 EFPY

Enclosure 4 provides a copy of the calculation for development of the P-T limit curves for the beltline, bottom head, and non-beltline regions of the MNGP RPV for 72 EFPYs of operation in accordance with the guidance of Revision 1 of the SIA LTR SIR-05-044-A which satisfies the requirements of 10 CFR 50 Appendix G and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Nonmandatory Appendix G.

The curves were developed for the following plant conditions: Pressure Test (Curve A), Normal Operation – Core Not Critical (Curve B), and Normal Operation – Core Critical (Curve C). Separate curves are provided for each of the following three regions of the RPV as well as a composite curve for the entire RPV:

1. The beltline region (includes nozzles where 1/4T fluence > 1 x 10^{17} n/cm^2),
2. The bottom head region,
3. The non-beltline region, including the top head flange,
4. Composite curve (bounding curve for all regions)

For the beltline region, the P-T curves incorporate components with the neutron fluence greater than 1 x 10^{17} n/cm^2 (E >1 MeV). The instrument nozzles are not in the beltline region and are not included in the P-T curve evaluations. The Feedwater nozzles are assumed to be the bounding component for non-beltline components.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

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 ASME B&PV Code were used to generate the P-T limits. Also, Appendix G requires that applicable surveillance data from 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.

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 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.

- PDC 1.2.11 -- Class I Equipment and Structures

  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.

- AEC 70 GDC 33 -- Reactor Coolant Pressure Boundary Capability

  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 Prevention

  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 the 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.

4.2 Precedent

The proposed LAR is similar to the following NRC approved license amendments for BWRs where the RAMA neutron fluence calculational methodology was utilized.

1. The Peach Bottom Atomic Power Station, Units 2 and 3, received amendments in October 2019 where the RAMA neutron fluence methodology was approved for use by the NRC through the subsequent license renewal period (Reference 19).

2. Units 1 and 2 at the Limerick Generating Station received amendments in September 2021 where the RAMA neutron fluence methodology was approved for use by the NRC through the license renewal period in addition to the P-T curves being relocated to a PTLR (Reference 20).

3. The Hope Creek Generating Station received an amendment in December 2017 where the RAMA neutron fluence methodology was approved for use by the NRC through the license renewal period in addition to the P-T curves being relocated to a PTLR (Reference 21).

4. The Columbia Generating Station received an amendment in November 2022 replacing the existing P-T curves within the TS with a PTLR. The curves are valid based on analyses projected through the license renewal period (Reference 22)

5. Units 1 and 2 at the LaSalle County Station received amendments in November 2023 where the RAMA neutron fluence methodology was approved for use by the NRC through the license renewal period and the P-T curves were relocated to a PTLR (Reference 23)

Therefore, based on the considerations discussed above, NSPM has determined that the proposed change does not require any exemptions or relief from regulatory requirements other than the TS, and does not affect conformance with the intent of any GDC differently than described in the USAR.

4.3 No Significant Hazards Consideration Analysis

In accordance with 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), requests an amendment to facility Renewed Operating License DPR-22, Technical Specifications (TS) for Monticello Nuclear Generating Plant (MNGP).

It is proposed to revise the MNGP Technical Specifications (TS), specifically Specification 5.6.5, “Pressure Temperature Limits Report (PTLR),” to revise the PTLR curves to reflect new surveillance capsule results, apply a different fluence methodology, and extend the applicability of the curves through 72 effective full power years (EFPY).

These new curves have been developed applying the analytical methodology described in Revision 1 of the Structural Integrity Associates (SIA) Report, SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," which has received NRC approval. The curves were developed applying a different fluence methodology – TransWare implementation of the Radiation Analysis Modeling Application (RAMA) fluence methodology.

NSPM has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   The proposed license amendment requests adoption of the NRC approved RAMA neutron fluence calculational methodology together with an update of the TS to reflect the current NRC approved version of a SIA PTLR development methodology report for preparation of MNGP Pressure-Temperature (P-T) limit curves. The revised MNGP PTLR was developed based on these methodologies and templates provided within these reports.

   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 RAMA and the SIA methodology for calculating the P-T limit curves provides an equivalent level of assurance that RCPB integrity will be maintained, as required by 10 CFR 50 Appendix G.

   Additionally, 10 CFR 50, Appendix H, provides the NRC criteria for design and implementation of reactor pressure vessel (RPV) material surveillance programs for operating lightwater reactors. Implementing these NRC approved methodologies does not reduce the ability to protect the RCPB as specified in Appendix G, nor do these changes increase the probability of malfunction of plant equipment, or the failure of plant structures, systems, or components. Incorporation of the new RAMA fluence methodology for calculating P-T curves 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. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed license amendment for adoption of the NRC approved RAMA neutron fluence calculational methodology together with an update of the TS to reflect the current NRC approved version of a SIA PTLR development methodology does 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.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The proposed license amendment for adoption of the NRC approved RAMA neutron fluence calculational methodology together with an update of the TS to reflect the current NRC approved version of a SIA PTLR development methodology. Calculating the MNGP P-T limits using these NRC approved methodologies, 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 operability requirements for equipment assumed to operate for accident mitigation.

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

Based on the above, NSPM concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c); accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (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.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed changes do 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, "Standards for Protection Against Radiation," nor do they change an inspection or surveillance requirement. The proposed changes do 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 effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for a categorical exclusion set forth in 10 CFR 51.22, "Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," specifically paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed change.

6.0 REFERENCES

1. Boiling Water Reactor Owner's Group (BWROG) Licensing Topical Report (LTR) BWROG TP-11-022-A, Revision 1 (Structural Integrity Associates, Inc. Report SIR-05-044, Revision 1-A), “Pressure-Temperature Limits Report Methodology for Boiling Water Reactors,” dated August 2013 (Agency Documents and Management System (ADAMS) Accession No. ML13277A557)

2. NRC letter to NSPM, “Monticello Nuclear Generating Plant – Issuance of Amendment to Revise and Relocate Pressure Temperature Curves to a Pressure Temperature Limits Report (TAC No. ME7930),” dated February 27, 2013 (ADAMS Accession Number ML13025A155)

3. NRC Generic Letter 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” dated January 31, 1996 (ADAMS Accession Number ML031110004)

4. Technical Specification Task Force (TSTF) -419, “Revise PTLR Definition and References in ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR,” dated September 16, 2001 (ADAMS Accession Number ML012690234)

5. BWROG LTR BWROG TP-11-022-A, Revision 0 (Structural Integrity Associates, Inc. Report SIR-05-044, Revision 1-A), “Pressure Temperature Limits Report Methodology for Boiling Water Reactors,” dated August 2007

6. GE Energy, Nuclear Report NEDO-32983P-A, Revision 2, “Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,” dated January 2006 (ADAMS Accession Number ML072480121)

7. NRC Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” Revision 2, dated May 1988 (ADAMS Accession Number ML003740284)

8. NRC Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001 (ADAMS Accession Number ML010890301)

9. NRC Safety Evaluation to NextEra Energy Seabrook, LLC, “Safety Evaluation Report, With Open Items Related to the License Renewal of Seabrook Station,” Docket Number 50-443, dated June 2012

10. NRC letter to Bill Eaton, BWRVIP Chairman Entergy Operations, Inc., “Safety Evaluation of Proprietary EPRI Reports, ‘BWR Vessel and Internals Project, RAMA

Fluence Methodology Manual (BWRVIP-114),' 'RAMA Fluence Methodology Benchmark Manual Evaluation of Regulatory Guide 1.190 Benchmark Problems (BWRVIP-115),' 'RAMA Fluence Methodology Susquehanna Unit 2 Surveillance Capsule Fluence Evaluation for Cycles 1-5 (BWRVIP-117),' and 'RAMA Fluence Methodology Procedures Manual (BWRVIP-121),' and 'Hope Creek Flux Wire Dosimeter Activation Evaluation for Cycle 1 (TWE-PSE-001-R-001)' (TAC No. MB9765)," dated May 13, 2005

11. NSPM letter to NRC, "Monticello Nuclear Generating Plant Docket No. 50-263, Renewal License Numbers DPR-22 Application for Subsequent Renewal Operating License," (Letter L-MT-23-001) dated January 9, 2023 (ADAMS Accession No. ML23009A353)

12. NSPM letter to NRC, "Subsequent License Renewal Application Supplement 3," (Letter L-MT-23-030) dated July 11, 2023 (ADAMS Accession Number ML23193B026)

13. NSPM Calculation No: 23-008, Revision 1, TransWare Topical Report, "Monticello Nuclear Generating Plant Fluence Methodology Report" (TransWare Doc. No. MNT-FLU-001-R-001-LNP, Revision 0, April 2023) and Attachment 1, "Qualification of the Monticello Reactor Fluence Model – Cycles 1 to 30" (TransWare Doc. No. MNT-FLU-001-R-001-LNP, Attachment 1, Revision 1, June 2023) [This calculation and it's attachment were enclosed within Letter L-MT-23-030.]

14. BWRVIP-78NP, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," dated April 2000 (ADAMS Accession No. ML003704011)

15. BWRVIP-86, Revision 1-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," (ADAMS Accession Nos. ML023190487)

16. NRC letter to Nuclear Management Company (NMC), LLC, "Monticello Nuclear Generating Plant – Issuance of Amendment Re: Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program (TAC No. MB6460)," dated April 22, 2003 (ADAMS Accession No. ML030830591)

17. BWRVIP-347, "BWR Vessel and Internals Project, Testing and Evaluation of the Monticello 120° ISP(E) Surveillance Capsule," Final Report, October 2022

18. BWRVIP-135, Revision 4 (current), "BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations." (ADAMS Accession Number ML22332A455) (non-proprietary version)

19. NRC letter to Exelon Generation Company, LLC, "Safety Evaluation Report Related to the Subsequent License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3, Docket Nos. 50-277 and 50-278," dated February 2020

20. NRC letter to Exelon Nuclear, "Limerick Generating Station, Units 1 and 2 – Issuance of Amendment Nos. 253 and 215 Re: Technical Specification Changes Related to Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report (EPID L-2020-LLA-0221)," dated September 28, 2021, (ADAMS Accession Number ML21181A044)

21. Letter from the U.S. NRC to PSEG Nuclear LLC, "Hope Creek Generating Station – Issuance of Amendment to Revise and Relocate the Pressure-Temperature Limit Curves to a Pressure and Temperature Limits Report (CAC No. MF9502, EPID L-2017-LLA-0204)," dated December 14, 2017 (ADAMS Accession Number ML17324A840)

22. NRC letter to Energy Northwest, "Columbia Generating Station – Issuance of Amendment No. 268 to Revise Technical Specification 3.4.11 "RCS Pressure and Temperature (P/T) Limits" (EPID L-2021-LLA-0191)," dated November 23, 2022 (ADAMS Accession Number ML22263A445)

23. NRC letter to Constellation Energy Generation, LLC, "LaSalle County Station, Units 1 and 2 – Issuance of Amendment Nos. 260 and 245 to Renewed Facility Operating Licenses Re: Relocation of Pressure and Temperature Limit Curves to the Pressure Temperature Report" (EPID L-2022-LLA-0173)," dated November 8, 2023 (ADAMS Accession Number ML23286A260)

ENCLOSURE 1, ATTACHMENT 1

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

TECHNICAL SPECIFICATION PAGES (MARKUP)

(1 Page Follows)

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~~ 1, dated ~~April 2007.~~ August 2013.

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

ENCLOSURE 1, ATTACHMENT 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

TECHNICAL SPECIFICATION PAGES (RETYPED)

(1 Page Follows)

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   1. SIR-05-044-A, “Pressure-Temperature Limits Report Methodology for Boiling Water Reactors,” Revision 1, dated August 2013. |

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

ENCLOSURE 2

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

XCEL ENERGY CORPORATION
MONTICELLO NUCLEAR GENERATING PLANT (MNGP)
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)
UP TO 72 EFFECTIVE FULL-POWER YEARS (EFPY)

REVISION 2

(32 Pages Follow)

Xcel Energy Corporation

Monticello Nuclear Generating Plant (MNGP)

Pressure and Temperature Limits Report (PTLR)

up to 72 Effective Full-Power Years (EFPY)

Revision 2

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1.0 PURPOSE

The purpose of the Monticello Nuclear Generating Plant (MNGP) Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

1. Reactor Coolant System (RCS) Pressure versus Temperature limits during Heat-up, Cool-down and Hydrostatic/Class 1 Leak Testing.
2. RCS Heat-up and Cool-down rates.
3. Reactor Pressure Vessel (RPV) to RCS coolant ΔT (ΔTemperature) requirements during Recirculation Pump startup.
4. RPV bottom head coolant temperature to RPV coolant temperature ΔT requirements during Recirculation Pump startup.
5. RPV boltup temperature limits.

This report has been prepared in accordance with the requirements of the current and previous revisions of Licensing Topical Reports SIR-05-044 contained within BWROG-TP-11-022-A, Revision 1 [1].

2.0 APPLICABILITY

This report is applicable to the MNGP RPV for up to 72 Effective Full-Power Years (EFPY).

The following MNGP Technical Specifications (TS) are affected by the information contained in this report:

TS 3.4.9 RCS Pressure/Temperature (P/T) Limits

3.0 METHODOLOGY

The limits in this report were derived as follows:

1. The methodology used is in accordance with Reference [1], “Pressure – Temperature Limits Report Methodology for Boiling Water Reactors,” August 2013, incorporating the NRC Safety Evaluation in Reference [2].

2. The neutron fluence is calculated in accordance with NRC Regulatory Guide 1.190 (RG 1.190) [3] as documented in Reference [5].

3. The adjusted reference temperature (ART) values for the limiting beltline materials are calculated in accordance with NRC Regulatory Guide 1.99, Revision 2 (RG 1.99) [4], as documented in Reference [5, 6].

4. The pressure and temperature limits, which were calculated in accordance with Reference [1], are documented in Reference [6].

5. This revision of the pressure and temperature limits report is to incorporate the following changes:

    - Revision 2: to incorporate new irradiation fluence data [5, 10] that go out to 72 EFPY of the RPV and new chemistry factor from the 120 degree surveillance capsule [24].

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Safety Analysis Report (USAR), can be made pursuant to 10 CFR 50.59 [8], provided the above methodologies are utilized. After issuance, the revised PTLR is submitted to the NRC for awareness.

The requirement for 10 CFR 50 Appendix H submission of capsule report information is not directly associated with PTLR processing.

4.0 OPERATING LIMITS

The pressure-temperature (P-T) curves included in this report represent steam dome pressure versus minimum vessel metal temperature and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) core not critical operation, referred to as Curve B; and (c) core critical operation, referred to as Curve C.

Complete P-T curves were developed for 72 EFPY for MNGP, as documented in Reference [6], and are provided in Figure 1 through Figure 3 for MNGP. A tabulation of the curves is included in Table 1 through Table 3. The adjusted reference temperature (ART) tables for 72 EFPY for the MNGP vessel beltline materials are shown in Table 4 [5].

The resulting P-T curves are based on the geometry, design and materials information for the MNGP vessel. The following conditions apply to operation of the MNGP vessel:

- Heat-up/Cool-down rate limit during Hydrostatic Class 1 Leak Testing (Figure 1: Curve A): **≤ 25°F/hour**[1] [1].
- Normal Operating Heatup and Cooldown rate limit (Figure 2: Curve B – core non-critical, and Figure 3: Curve C – core critical): **100°F/hr**[2] [6].
- RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 145°F** [1].
- Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 50°F** [1].

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[1] Interpreted as the temperature change in any 1-hour period is less than or equal to 25°F.

[2] Interpreted as the temperature change in any 1-hour period is less than or equal to 100°F.

- RPV head flange, RPV flange and adjacent shell temperature limit during vessel bolt-up **≥ 60˚F** [6].

5.0 DISCUSSION

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust beltline P-T curves to account for irradiation effects. RG 1.99 [4] provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper (Cu) and nickel (Ni) values were obtained from the evaluation of the MNGP vessel plate, weld, and forging materials [5]; this evaluation included the results of three surveillance capsules. The Cu and Ni values were used with Table 1 of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds. The Cu and Ni values were used with Table 2 of RG 1.99 to determine a CF per Paragraph 1.1 of RG 1.99 for plates and forgings.

Per Reference [5] and in accordance with Appendix A of Reference [1], the MNGP representative weld and plate surveillance materials data were reviewed from the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) [11]. The representative plate material for MNGP (*C2220*) in the ISP is the same as the lower intermediate shell plate material in the vessel beltline region of MNGP. For the plate heat *C2220*, since the scatter in the fitted results is less than 1-sigma (17°F), the margin term (σ_Δ = 17°F) is cut in half for the material when calculating the ART. The representative heat of the weld material (*5P6756*) in the ISP is not the same as the limiting weld material in the vessel beltline region of MNGP. Therefore, the CFs from the tables in RG1.99 were used in the determination of the ART values of all MNGP materials except for plate heat *C2220*. Reference [5] used a chemistry factor (CF) of *180* from Reference [11]. However, the latest ISP data show that the CF value for plate heat *C2220* changes to 174 [24] which is used in the ART calculation in P-T curve evaluations [6].

The peak RPV ID fluence value of 5.94×10^{18} n/cm^2 at 72 EFPY used in the P-T curve evaluations was obtained from Reference [10]. The fluence value applies to the limiting beltline lower intermediate shell plates (Heat No. C2220-1 and C2220-2). The fluence value for the lower intermediate shell plates is based upon an attenuation of 0.738 for a postulated ¼ flaw. Consequently, the 1/4T fluence for 72 EFPY for the limiting lower intermediate shell plates is 4.38×10^{18} n/cm^2 [5]. Using the CF value of 174 from the latest ISP data [24], the limiting ART value for beltline plates and welds is 178.1°F for MNGP [6].

The RPV ID fluence value of 7.08×10^{17} n/cm^2 at 72 EFPY used in the P-T curve evaluation of the recirculation inlet nozzle was obtained from Reference [10]. The fluence value applies to the limiting recirculation inlet nozzle (Heat No. E21VW). The fluence value for the recirculation inlet nozzle is based upon an attenuation of 0.738 for a postulated ¼ flaw. As a result, the 1/4T fluence for the limiting recirculation inlet nozzle is 5.23×10^{17} n/cm^2 at 72 EFPY for MNGP. There are no additional forged or instrument nozzles in the extended beltline at 72 EFPY. The limiting ART value for the recirculation inlet nozzle is 116.6°F for MNGP at 72 EFPY [5].

The P-T curves for the core not critical and core critical operating conditions at a given EFPY apply for both the 1/4T and 3/4T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (inside surface flaw) and the 3/4T location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cool-down and is in the outer wall during heat-up. However, as a conservative simplification, the thermal gradient stresses at the 1/4T location are assumed to be tensile for both heat-up and cool-down. This results in the approach of applying the maximum tensile stresses at the 1/4T location. This approach is conservative because irradiation effects cause the allowable fracture toughness at the 1/4T to be less than that at 3/4T for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, and for a given pressure, the coolant saturation temperature is well above the P-T curve limiting temperature. Consequently, the material fracture toughness at a given pressure would exceed the allowable fracture toughness.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves specify a coolant heatup and cooldown temperature rate of ≤ 100°F/hr for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound RPV thermal transients. For the hydrostatic pressure and leak test curves (Curve A), a coolant heatup and cooldown temperature of ≤ 25°F/hr must be maintained. The P-T limits and corresponding limits of either Curve A or B may be applied, if necessary, while achieving or recovering from test conditions. So, although Curve A applies during pressure testing, the limits of Curve B may be conservatively used during pressure testing if the pressure test heatup/cooldown rate limits cannot be maintained.

The initial RT_{NDT}, chemistry (weight-percent copper and nickel), and ART at the 1/4T location for all RPV beltline materials significantly affected by fluence (i.e., fluence > 10^{17} n/cm^2 for E > 1 MeV) is shown in Table 4 for 72 EFPY [6]. Use of initial RT_{NDT} values in the determination of P-T curves for MNGP was approved by the NRC in Reference [9].

The only computer code used in the determination of the MNGP P-T curves was the ANSYS Mechanical, Release 18.1 [12] finite element computer program. ANSYS finite element analyses were used to develop the stress distributions through the feedwater nozzle (non-beltline) and recirculation inlet nozzle (beltline) as well as the vessel shell, and these stress distributions were used in the determination of the stress intensity factors for the feedwater and recirculation inlet nozzles [13, 14] and vessel shell. At the time that each of the analyses above was performed, the ANSYS program was controlled under the vendor’s 10 CFR 50 Appendix B [15] Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 83-11, Supplement 1 [16] was performed as a part of the computer program verification by comparing the solutions produced by the computer code to hand calculations for several problems.

The plant-specific MNGP feedwater nozzle analyses were performed to determine through-wall pressure stress distributions and thermal stress distributions due to bounding thermal transients. Detailed information regarding the analyses can be found in Reference [13]. The following inputs were used as input to the finite element analysis:

- A one-quarter symmetric, three-dimensional finite element model of the feedwater nozzle was constructed and is shown in Figure 4. Temperature dependent material properties, taken from the MNGP Code of Record [17], were used in the evaluation.
- Heat transfer coefficients were calculated at different flow rates. The analysis used the conservative forced convection coefficients and applied it to all wetted surfaces [13]. Therefore, the heat transfer coefficients used in the analysis bound the actual operating conditions in the feedwater nozzle at MNGP.
- With respect to operating conditions, stress distributions were developed for two bounding thermal transients. A thermal shock, which represents the maximum thermal shock for the feedwater nozzle during normal operating conditions, and a thermal ramp were analyzed [13]. The thermal stress distributions, corresponding to the limiting times presented in Reference [13], along a linear path through the nozzle corner is used as shown in Figure 5. The boundary integral equation/influence function (BIE/IF) methodology presented in Reference [1] is used to calculate the thermal stress intensity factor, K_{It}, due to the thermal stresses by fitting a third order polynomial equation to the path stress distribution for the thermal load case.
- With respect to pressure stress, a unit pressure of 1000 psig was applied to the internal surfaces of the 3-D model in Reference [13]. The pressure stress distribution was taken along a linear path through the nozzle corner as shown in Figure 5. The BIE/IF methodology presented in Reference [1] was used to calculate the applied pressure stress intensity factor, K_{Ip}, by fitting a third order polynomial equation to the path stress distribution for the pressure load case. The resulting K_{Ip} may be linearly scaled to determine the K_{Ip} for various RPV internal pressures.

The plant-specific MNGP recirculation inlet nozzle analysis was performed to determine through-wall pressure stress distributions and thermal stress distributions due to bounding thermal transients. Detailed information regarding the analysis can be found in Reference [14]. The following inputs were used as input to the finite element analysis:

- A one-quarter symmetric, three-dimensional finite element model of the recirculation inlet nozzle was constructed and is shown in Figure 6. Temperature dependent material properties, taken from the MNGP Code of Record [18], were used in the evaluation.
- Heat transfer coefficients were calculated with the most severe thermal shock for the nozzle blend radius in safety valve blow down (SVBD). The heat transfer coefficients were conservatively calculated based on the full temperature difference of the transient, rather than the RPV to coolant temperature difference [14]. Therefore, the heat transfer coefficients used in the analysis bound the actual operating conditions in the recirculation inlet nozzle at MNGP.
- With respect to operating conditions, the thermal transient that would produce the highest tensile stresses at the 1/4T location is the 100°F/hr SVBD transient [14]. Therefore, the stresses represent the bounding stresses in the recirculation inlet nozzle associated with 100°F/hr heatup/cooldown limits associated with the P-T curves for a nozzle in the beltline region.
- With respect to pressure stress, a unit pressure of 1010 psig was applied to the internal surfaces of the 3-D model in Reference [14]. The pressure stress distribution was taken along a linear path through the nozzle corner as shown in top of Figure 7. The BIE/IF methodology presented in Reference [1] was used to calculate the applied pressure stress intensity factor, K_{Ip}, by fitting a third order polynomial equation to the path stress distribution for the pressure load case. The resulting K_{Ip} may be linearly scaled to determine the K_{Ip} for various RPV internal pressures.

Table 5 summarizes the pressure stress intensity factor and maximum thermal stress intensity factor for both feedwater and recirculation inlet nozzle.

6.0 REFERENCES

1. BWROG-TP-11-022-A, Revision 1, Pressure Temperature Limits Report Methodology for Boiling Water Reactors, August 2013. (ADAMS Accession No. ML13277A557)
2. U.S. NRC Letter to BWROG dated May 16, 2013, “Final Safety Evaluation for Boiling Water Reactor Owners’ Group Topical Report BWROG-TP-11-022, Revision 1, November 2011, ‘Pressure-Temperature Limits Report Methodology for Boiling Water Reactors’” (TAC NO. ME7649, ADAMS Accession No. ML13277A557).
3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence”, March 2001.
4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”, May 1988.
5. SI Calculation No. 2100300.302P, Revision 2, “Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts.” **CONTAINS PROPRIETARY INFORMATION**.
6. SI Calculation No. 2200284.303, Revision 0, “Monticello Pressure-Temperature Limit Curves Generation for 72 EFPY.”
7. U.S. Code of Federal Regulations, Title 10, Part 50, Appendix G, “Fracture Toughness Requirements,” December 12, 2013.
8. U.S. Code of Federal Regulations, Title 10, Part 50, Section 59, “Changes, tests and experiments,” August 28, 2007.
9. Application for Renewed Operation License, Monticello Nuclear Generating Plant, March 2005.
10. “Monticello Nuclear Generating Station Reactor Pressure Vessel Fluence Evaluation – Subsequent License Renewal,” TransWare Enterprises, MNT-FLU-001-R-002, Revision 0. June 2022. SI File NO. 2100300.201.
11. *BWRVIP-135, Revision 4: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations*. EPRI, Palo Alto, CA: 2014. 3002003144. **EPRI PROPRIETARY INFORMATION**.
12. ANSYS Mechanical APDL (UP20170403) and Workbench (March 31, 2017), Release 18.1, SAS IP, Inc.
13. SI Calculation No. 2200284.302P, Revision 0, “Finite Element Stress Analysis of Monticello RPV Feedwater Nozzle.” **CONTAINS PROPRIETARY INFORMATION**.
14. SI Calculation No. 2200284.301P, Revision 0, “Finite Element Stress and Fracture Mechanics Analysis of Monticello RPV Recirculation Inlet Nozzle.” **CONTAINS PROPRIETARY INFORMATION**.

15. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, Appendix B, “Quality Assurance for Nuclear Power Plants and Fuel Reprocessing Plants”.

16. U.S. Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, “License Qualification for Performing Safety Analyses”, June 24, 1999.

17. ASME Boiler and Pressure Vessel Code Section III, <u>Rules for Construction of Nuclear Vessels</u>, 1977 Edition with Addenda through Summer1978.

18. ASME Boiler and Pressure Vessel Code, Section III including Appendices, 1980 Edition with Addenda through Winter 1980.

19. U.S. Code of Federal Regulations, Title 10, Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” January 31, 2008.

20. *BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan*. EPRI, Palo Alto, CA: 2012. 1025144. **EPRI PROPRIETARY INFORMATION.**

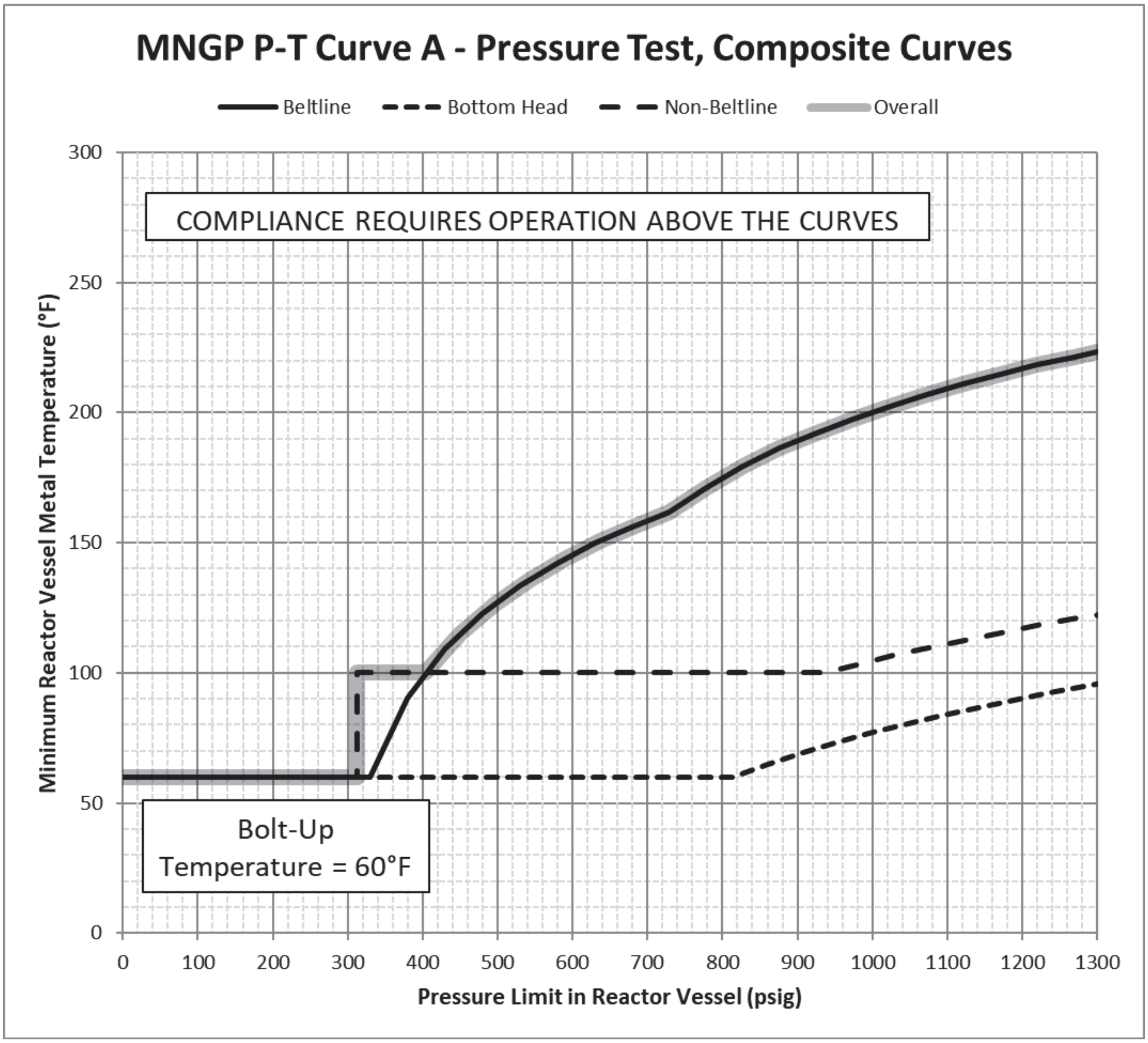
21. BWRVIP-250NP, *BWR Vessel and Internals Project, Testing and Evaluation of the Monticello Unit 1 120° Surveillance Capsule*. EPRI, Palo Alto, CA: 2011. 1022850.

22. SI File No. 2000115.401, Revision 0, “Updated Evaluation of the EPRI BWRVIP ISP Capsule Withdrawal Schedule,” May 3, 2021.

23. NRC (L. M. Padovan) letter to NMC (D. L. Wilson), "Monticello Nuclear Generating Plant - Issuance of Amendment re: Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program (TAC No. MB6460)", dated April 22, 2003.

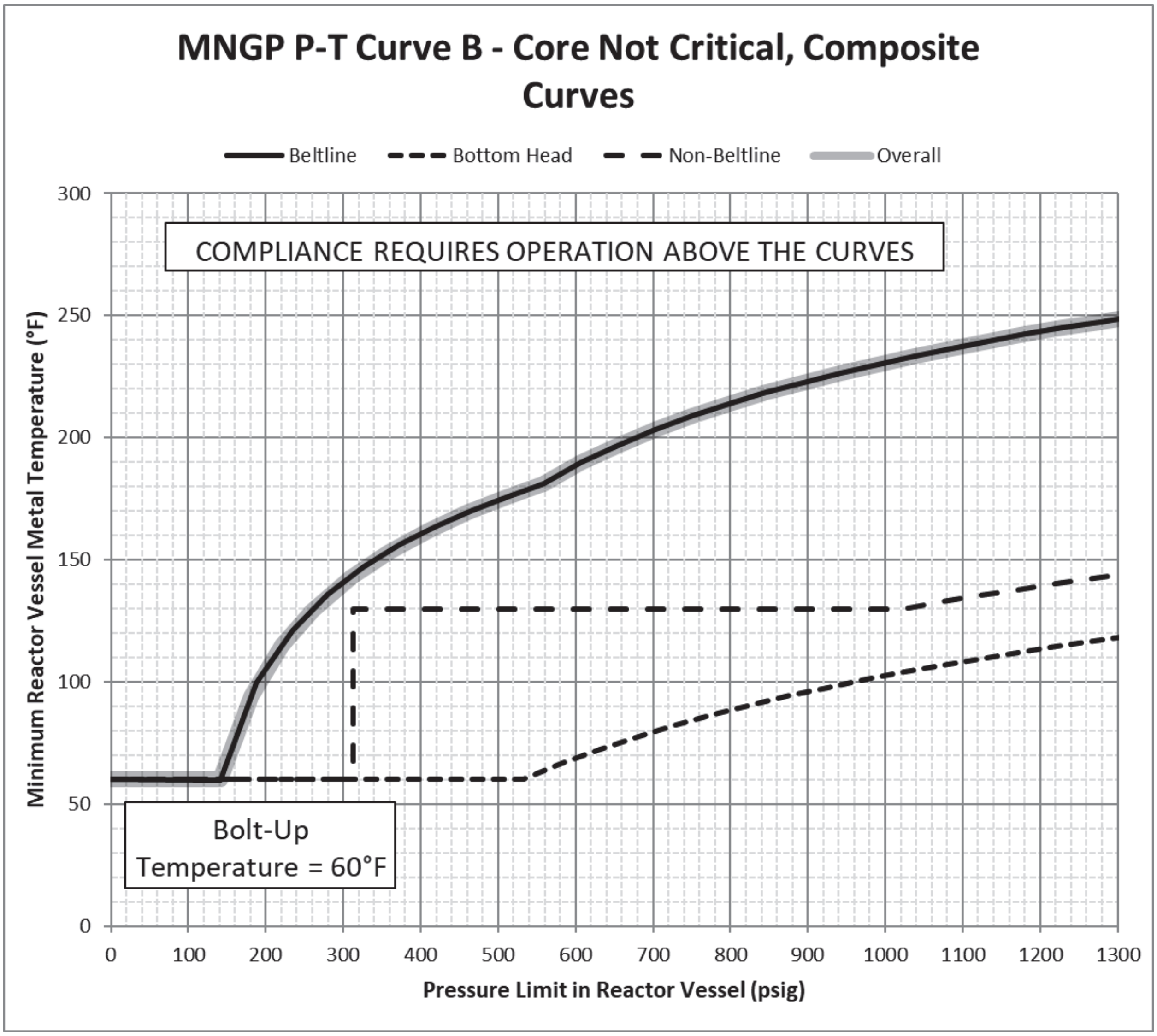
24. BWRVIP Letter 2022-053, September 14, 2022, from Bob Carter to Russell Lidberg, Subject: Notification of New BWRVIP Integrated Surveillance Program (ISP) Data Applicable to the Monticello Reactor Pressure Vessel (RPV).

Figure 1: MNGP P-T Curve A (Hydrostatic Pressure and Leak Tests) for 72 EFPY



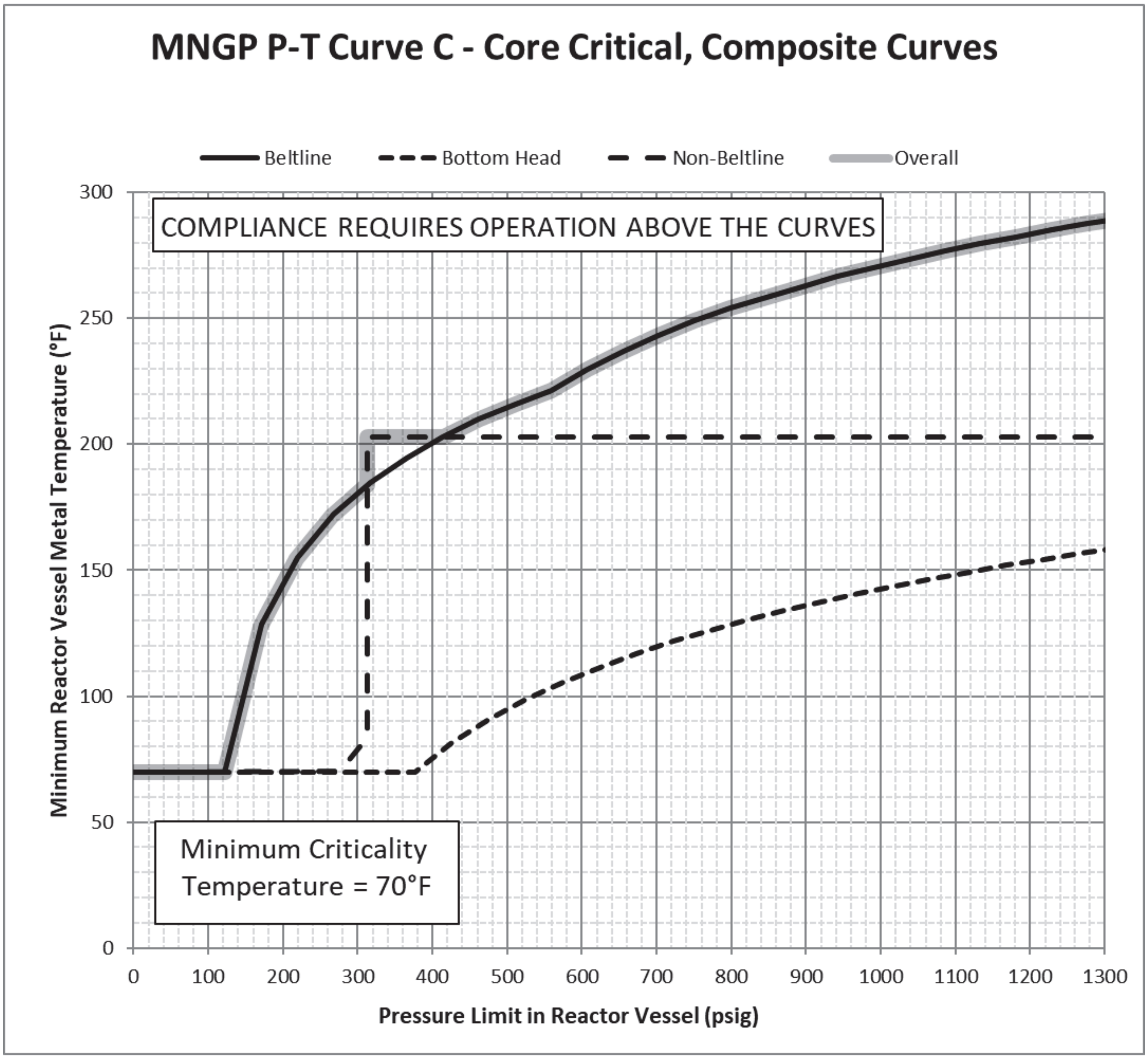
Note: The minimum reactor vessel metal temperature at 0 psig is applicable for RPV operation under a vacuum.

Figure 2: MNGP P-T Curve B (Normal Operation – Core Not Critical) for 72 EFPY



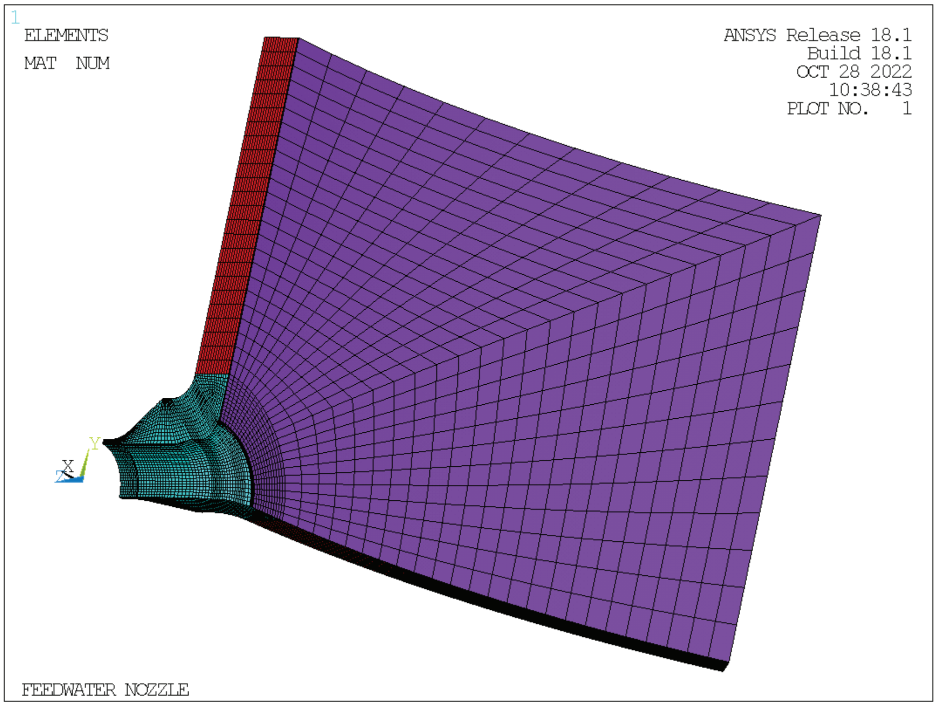
Note: The minimum reactor vessel metal temperature at 0 psig is applicable for RPV operation under a vacuum.

Figure 3: MNGP P-T Curve C (Normal Operation – Core Critical) for 72 EFPY



Note: The minimum reactor vessel metal temperature at 0 psig is applicable for RPV operation under a vacuum.

Figure 4: MNGP Feedwater Nozzle 3-D Finite Element Model [13]



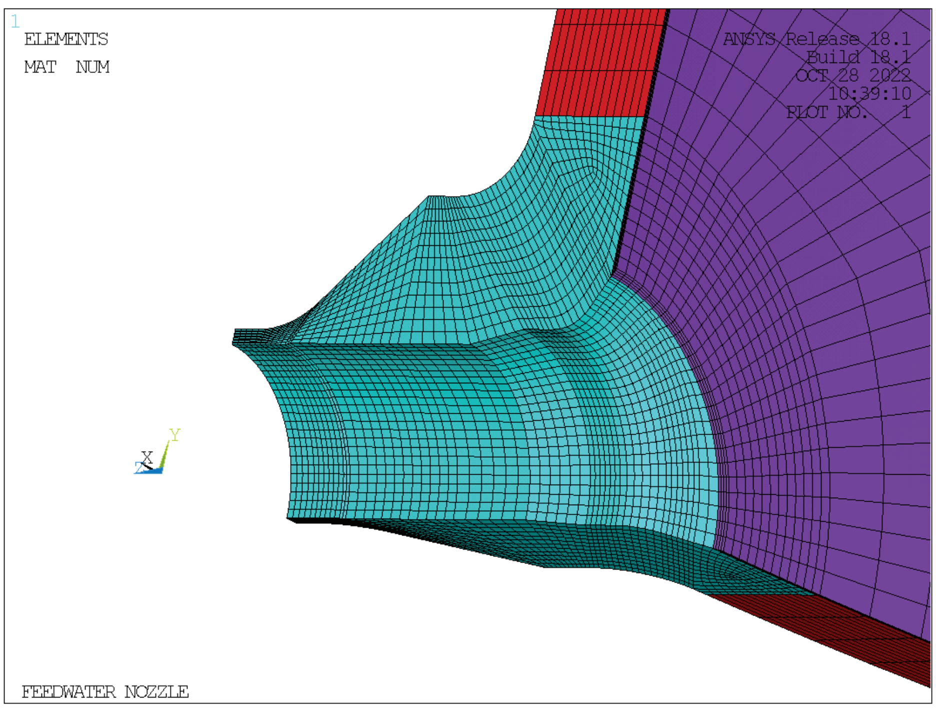


Figure 5: MNGP Feedwater Nozzle Stress Extraction Path [13]

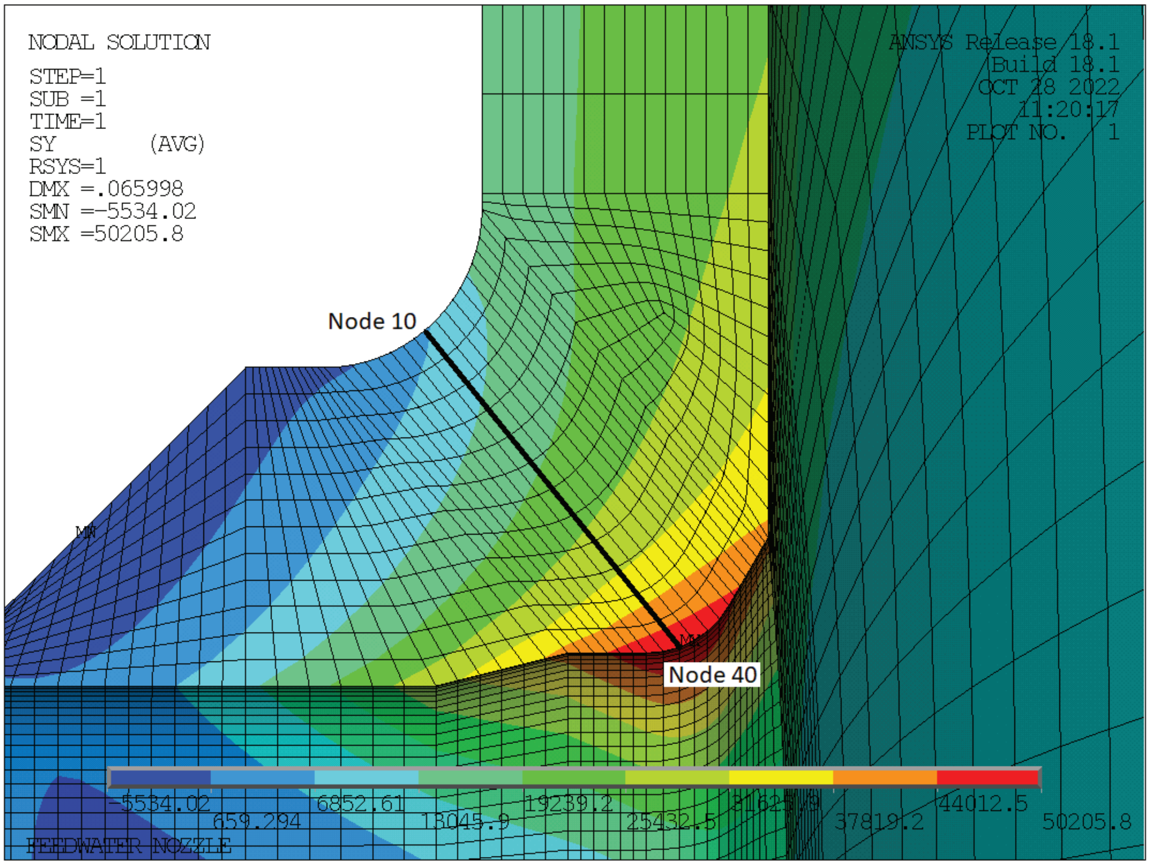
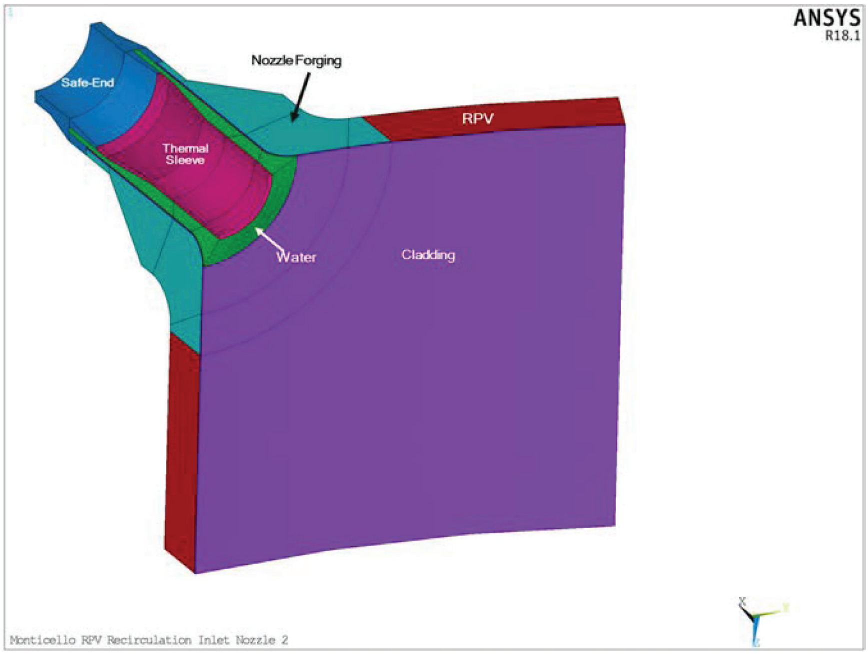


Figure 6: MNGP Recirculation Nozzle Finite Element Model [14]



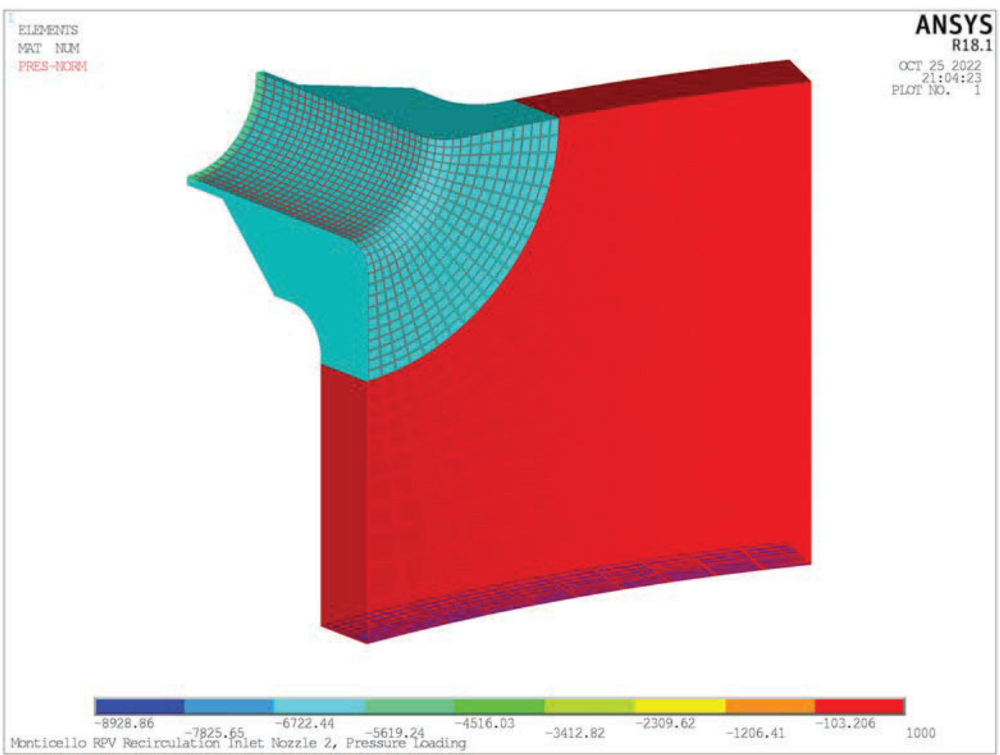


Figure 7: MNGP Recirculation Nozzle Stress Extraction Path [14]

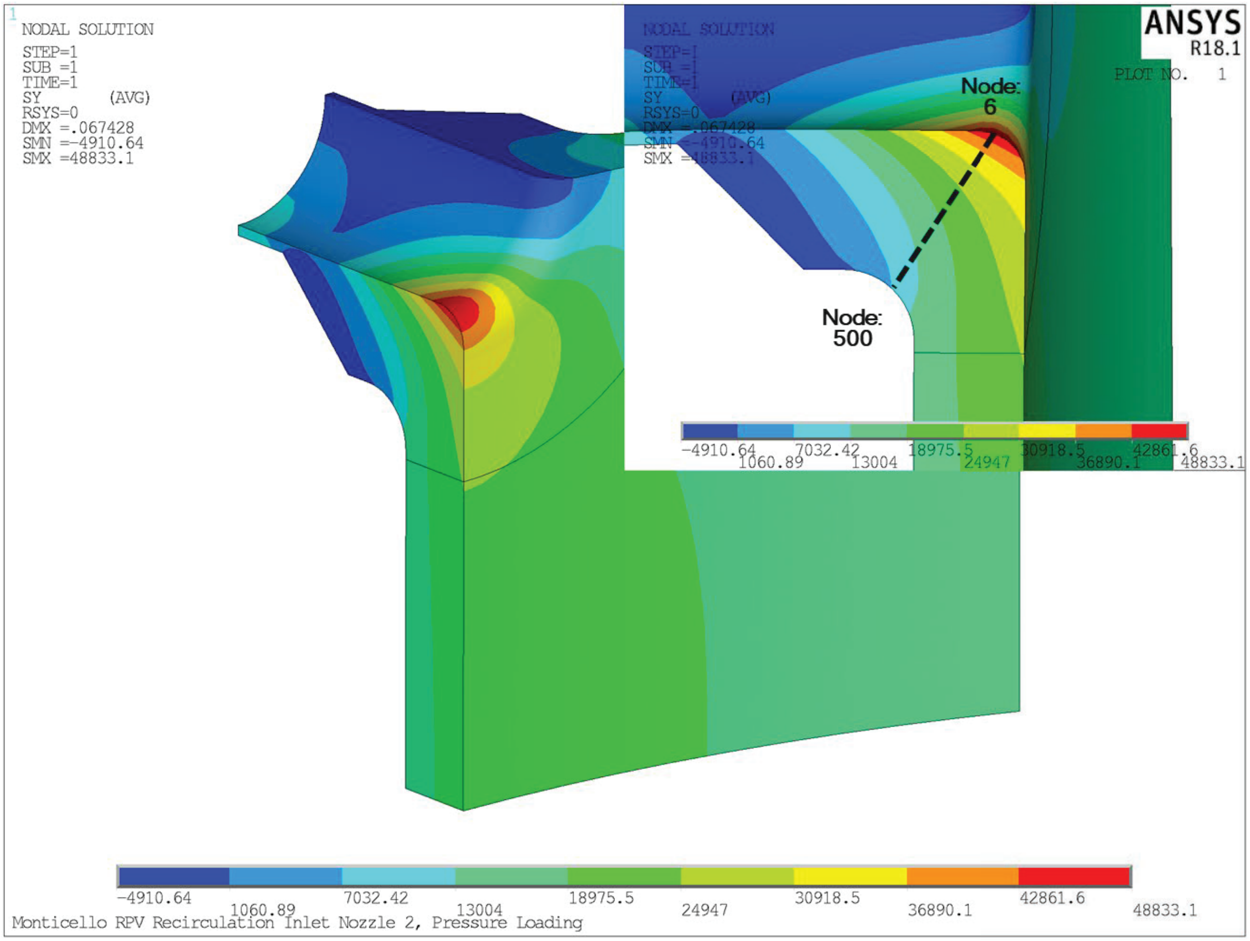


Table 1: MNGP Pressure Test (Curve A) P-T Curves for 72 EFPY

Beltline Region

Curve A - Pressure Test

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
60.0	0.0
60.0	329.7
90.3	379.7
109.1	429.6
122.7	479.5
133.4	529.5
142.2	579.4
149.6	629.3
156.1	679.2
161.9	729.2
171.4	778.2
179.3	827.3
186.2	876.3
192.2	925.4
197.6	974.4
202.4	1023.5
206.9	1072.5
210.9	1121.6
214.7	1170.6
218.2	1219.7
221.5	1268.7
224.5	1317.8
227.4	1366.8
230.2	1415.9
232.8	1464.9
235.2	1514.0
237.6	1563.0

Table 1: MNGP Pressure Test (Curve A) P-T Curves for 72 EFPY (continued)

Bottom Head Region

Curve A - Pressure Test

P-T Curve Temperature °F	P-T Curve Pressure psi
60.0	0.0
60.0	812.8
64.8	859.7
69.2	906.6
73.2	953.4
77.0	1000.3
80.5	1047.2
83.7	1094.1
86.8	1141.0
89.6	1187.9
92.3	1234.8
94.9	1281.7
97.4	1328.6
99.7	1375.4
102.0	1422.3
104.1	1469.2
106.1	1516.1
108.1	1563.0

Table 1: MNGP Pressure Test (Curve A) P-T Curves for 72 EFPY (continued)

Non-Beltline Region

Curve A - Pressure Test

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
60.0	0.0
60.0	312.6
100.0	312.6
100.0	936.3
103.6	984.5
106.9	1032.7
110.0	1080.9
113.0	1129.1
115.8	1177.3
118.4	1225.6
120.9	1273.8
123.3	1322.0
125.6	1370.2
127.7	1418.4
129.8	1466.6
131.8	1514.8
133.7	1563.0

Table 2: MNGP Core Not Critical (Curve B) P-T Curves for 72 EFPY

Beltline Region

Curve B - Core Not Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
60.0	0.0
59.6	141.2
99.6	187.6
121.1	234.1
135.8	280.5
147.1	326.9
156.1	373.4
163.8	419.8
170.4	466.2
176.1	512.6
181.3	559.1
189.7	606.9
196.9	654.7
203.2	702.5
208.8	750.3
213.9	798.1
218.4	845.9
222.6	893.7
226.5	941.5
230.1	989.3
233.4	1037.1
236.6	1084.9
239.5	1132.7
242.3	1180.5
245.0	1228.4
247.5	1276.2
249.8	1324.0
252.1	1371.8
254.3	1419.6
256.4	1467.4
258.4	1515.2
260.3	1563.0

Table 2: MNGP Core Not Critical (Curve B) P-T Curves for 72 EFPY (continued)

<u>Bottom Head Region</u>

Curve B - Core Not Critical

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
60.0	0.0
60.0	533.1
66.6	582.2
72.4	631.2
77.6	680.3
82.4	729.3
86.7	778.3
90.6	827.4
94.3	876.4
97.8	925.5
101.0	974.5
104.0	1023.6
106.8	1072.6
109.5	1121.6
112.1	1170.7
114.5	1219.7
116.8	1268.8
119.0	1317.8
121.1	1366.8
123.2	1415.9
125.1	1464.9
127.0	1514.0
128.8	1563.0

Table 2: MNGP Core Not Critical (Curve B) P-T Curves for 72 EFPY (continued)

Non-Beltline Region

Curve B - Core Not Critical

P-T Curve Temperature °F	P-T Curve Pressure psi
60.0	0.0
60.0	312.6
130.0	312.6
130.0	1022.7
132.7	1071.8
135.3	1120.9
137.8	1170.0
140.2	1219.1
142.4	1268.3
144.5	1317.4
146.6	1366.5
148.6	1415.6
150.5	1464.8
152.3	1513.9
154.1	1563.0

Table 3: MNGP Core Critical (Curve C) P-T Curves for 72 EFPY

Beltline Region

Curve C - Core Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
70.0	0.0
70.0	122.2
128.5	170.8
155.2	219.3
172.2	267.8
184.7	316.4
194.6	364.9
202.8	413.4
209.8	462.0
215.9	510.5
221.3	559.1
229.7	606.9
236.9	654.7
243.2	702.5
248.8	750.3
253.9	798.1
258.4	845.9
262.6	893.7
266.5	941.5
270.1	989.3
273.4	1037.1
276.6	1084.9
279.5	1132.7
282.3	1180.5
285.0	1228.4
287.5	1276.2
289.8	1324.0
292.1	1371.8
294.3	1419.6
296.4	1467.4
298.4	1515.2
300.3	1563.0

Table 3: MNGP Core Critical (Curve C) P-T Curves for 72 EFPY (continued)

Bottom Head Region

Curve C - Core Critical

P-T Curve Temperature *°F*	P-T Curve Pressure *psi*
70.0	0.0
70.0	376.2
81.5	425.6
90.9	475.1
98.7	524.5
105.5	574.0
111.5	623.4
116.9	672.9
121.7	722.3
126.1	771.8
130.2	821.2
133.9	870.7
137.4	920.1
140.6	969.6
143.7	1019.0
146.6	1068.5
149.3	1117.9
151.9	1167.4
154.3	1216.8
156.7	1266.3
158.9	1315.7
161.1	1365.2
163.1	1414.6
165.1	1464.1
167.0	1513.5
168.8	1563.0

Table 3: MNGP Core Critical (Curve C) P-T Curves for 72 EFPY (continued)

<u>Non-Beltline Region</u>

Curve C - Core Critical

P-T Curve Temperature *°F*	**P-T Curve Pressure** *psi*
70.0	0.0
70.0	276.4
83.7	312.6
203.0	312.6
203.0	1563.0

Table 4: MNGP 1/4T ART Table for 72 EFPY

Component No.	Heat	Lot	% Cu	% Ni	CF	Initial RT_{NDT} (°F)	72EFPY 1/4T Fluence (n/cm^2)	Fluence Factor f	ΔRT_{NDT} (°F)	σ_i (°F)	σ_Δ (°F)	72 EFPY 1/4T ART (°F)
Lower Shell Plates (Course 1)												
I-16	A0946-1	N/A	0.14	0.56	98	27	2.80E+18	0.653	64.1	0	17.0	125.1
I-17	C2193-1	N/A	0.17	0.5	119	0	2.80E+18	0.653	77.3	0	17.0	111.3
Lower-Intermediate Shell Plates (Course 2)												
I-14	C2220-1	N/A	*0.16*	*0.64*	174	27	4.38E+18	0.770	134.1	0	8.5	178.1
I-15	C2220-2	N/A	*0.16*	*0.64*	174	27	4.38E+18	0.770	134.1	0	8.5	178.1
Upper/Int Shell Plates (Course 3)												
I-12	C2089-1	N/A	0.35	0.5	200	0	2.38E+17	0.191	38.2	0	17.0	72.2
I-13	C2613-1	N/A	0.35	0.49	198	27	2.38E+17	0.191	37.9	0	17.0	98.9
Lower Shell (Course 1) Axial Welds												
VLAA-1 & VLAA-2	-	E8018N	0.1	0.99	135	-65.6	1.73E+18	0.535	72.2	12.7	28.0	68.1
Lower-Intermediate Shell (Course 2) Axial Welds:												
VLBA-1 & VLBA-2	-	E8018N	0.1	0.99	135	-65.6	1.55E+18	0.510	68.8	12.7	28.0	64.7
Upper/Int Shell (Course 3) Axial Welds:												
VLCB-1 & VLCB-2	-	E8018N	0.1	0.99	135	-65.6	1.56E+17	0.147	19.8	12.7	9.9	-13.5
Circumferential Welds												
VCBA-2	-	E8018N	0.1	0.99	135	-65.6	2.80E+18	0.653	88.0	0	28.0	78.4
VCBB-3	-	E8018N	0.1	0.99	135	-65.6	2.38E+17	0.191	25.8	0	12.9	-14.0
N2 Nozzle												
N2 Nozzle	E21VW	N/A	*0.18*	0.86	142	40	5.23E+17	0.300	42.6	0	17.0	116.6

Table 5: Nozzle Stress Intensity Factors

Nozzle	Applied Pressure, $K_{Ip\text{-}app}$	Thermal, K_{It}
Feedwater	70.59 for 1,000 psi pressure	10.37
Recirculation Inlet	75.20 for 1,010 psi pressure	25.28

K_I in units of ksi-in$^{0.5}$

APPENDIX A

MONTICELLO REACTOR VESSEL MATERIALS SURVEILLANCE PROGRAM

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements [19], the 300-degree surveillance capsule was removed and tested from the Monticello reactor vessel in 2007. The surveillance capsules contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The methods and results of testing are presented in References [20, 21]. The 120-degree capsule was withdrawn in the spring 2021 refueling outage [22, 24]. The latest testing results [24] will be added to the next revision of BWRVIP 135, Reference [11], and have been used in the preparation of this report. This was the final capsule installed in the MNGP reactor.

MNGP is licensed to use the BWRVIP ISP during the Renewed License period of extended operation. The BWRVIP ISP meets the requirements of 10 CFR 50, Appendix H, for Integrated Surveillance Programs, and has been approved by NRC. Xcel Energy committed to use the ISP in place of its original surveillance programs in the amendments issued by the NRC regarding the implementation of the Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program, dated April 22, 2003 [23].

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

EVALUATION OF ADJUSTED REFERENCE TEMPERATURES AND REFERENCE TEMPERATURE SHIFTS

(NSPM CALCULATION NO. 22-035)

(SI NO. 2100300.302, REVISION 4)

(21 Pages Follow)

Xcel Energy	Calculation Signature Sheet

Approval: 602000018581

Document Information	
NSPM Calculation (Doc) No: 22-035	Revision: 2
Title: EVALUATION OF ADJUSTED REFERENCE TEMPERATURES AND REFERENCE TEMPERATURES SHIFTS	
Facility: ☒ MT ☐ PI	Unit: ☒ 1 ☐ 2
Safety Class: ☒ SR ☐ Aug Q ☐ Non SR	
Type: Calc Sub-Type: OTH	

NOTE:	Print and sign name in signature blocks, as required.

Major Revisions		☐ N/A
EC Number: 601000003566	☒ Vendor Calc:	
Vendor Name or Code: Structural Integrity Associates	Vendor Doc No: 21000300.302	
Description of Revision: Correct OT fluence value for shell course 1 in Table 2 and correctons in Table 2-4		
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record		☒
Prepared by: (sign) / (print) Vendor	Date:	
Reviewed by: (sign) / (print) Matthw Sears via MOC ITEM 60000112096	Date:	
Type of Review: ☐ Design Verification ☐ Engr Review ☒ OAR ☐ EOC		
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test		
Approved by: (sign) / (print) Paul Young via MOC ITEM 60000112095	Date:	

Xcel Energy

Calculation Signature Sheet

Minor Revisions	☐ N/A
EC No:	☐ Vendor Calc:
Minor Rev. No:	
Description of Change:	
Pages Affected:	
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record	☐
Prepared by: (sign) / (print)	Date:
Reviewed by: (sign) / (print)	Date:
Type of Review: ☐ Design Verification ☐ Engr Review ☐ OAR ☐ EOC	
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test	
Approved by: (sign) / (print)	Date:

Summary of Verification (summary is required for Design Verification):

☒ No Comments
☐ See attached QF0528
☐

Superseded Calculations:

Facility	Calc Document Number	Title

Does the Calculation:

☐ YES ☒ No Affect piping or supports? (If YES, Attach MT Form 3544.) MONTI ONLY

☐ YES ☒ No Require Fire Protection Review? (Using QF2900, Fire Protection Program Impact Screen, determine if a Fire Protection Review is required.) If YES, document the engineering review in the EC. If NO, then attach completed QF2900 to the associated EC.

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CALCULATION 22-035 REV 2

Structural Integrity Associates, Inc.®

File No.: 2100300.302
Project No.: 2100300
Quality Program Type: ☒ Nuclear ☐ Commercial

CALCULATION PACKAGE

PROJECT NAME:
MNGP Subsequent License Renewal TLAA

CONTRACT NO.:
4000020831

CLIENT:
Xcel Energy, Inc.

PLANT:
Monticello Nuclear Generating Plant

CALCULATION TITLE:
Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts

Document Revision	Affected Pages	Revision Description	Project Manager Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1 - 14 A-1 - A-2	Initial Issue	Approved By: Daniel B. Denis, PE 07/19/2022	Prepared By: Jianxin Wang 07/19/2022 Checked By: Sam Ranganath 07/19/2022
1	10-14	Adjusted the 3/4T fluence value results from dpa to attenuation method and initial RT_{NDT} values for plates I-16 and I-17	Approved By: Daniel B. Denis, PE 07/27/2022	Prepared By: Jianxin Wang 07/27/2022 Checked By: Sam Ranganath 07/27/2022

2	6-8, 11-15 A-2	Calculate circumferential weld VCBB-3 separately. Incorporated client comments on text on Page 6, 7 and 8.	Approved By: Daniel B. Denis, PE 09/16/2022	Prepared By: Jianxin Wang 09/16/2022 Checked By: Sam Ranganath 09/16/2022
3	2, 3, 7, 10, 12, 13, 14	Remove proprietary marks/notes per References [17] [18]	Approved By: Daniel B. Denis, PE 03/14/2023	Prepared By: Jianxin Wang 03/14/2023 Checked By: Sam Ranganath 03/14/2023
4	7, 8, 12, 13, 14	Correct the 0T fluence value for shell course 1 in Table 2. Correct the σ_l term for VCBB-2 and VCBB-3 in Table 2-4.	Approved By: Stephen Parker 05/23/2023	Prepared By: Jianxin Wang 05/23/2023 Checked By: Daniel B. Denis, PE 05/23/2023

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1.0 INTRODUCTION

Radiation embrittlement of reactor pressure vessel (RPV) materials causes a decrease in fracture toughness. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.99, Revision 2 (RG1.99R2) [1] describes general procedures to evaluate the effects of neutron irradiation embrittlement on the alloy steel used in RPVs. In order to perform this evaluation, RG1.99R2 requires calculation of Adjusted Reference Temperature (ART) and Reference Temperature Shift (ΔRT_{NDT}) values. The ART values are then used to determine the local fracture toughness of the RPV wall and pressure-temperature limits, according to ASME Code, Section XI, Non-mandatory Appendix G [2] evaluations.

In 2011, SI performed a calculation of the ART and ΔRT_{NDT} values developed for all MNGP plates, welds and nozzles exposed to fluence levels greater than 1.0×10^{17} n/cm^2 [3]. Those calculations were based on the updated fluence calculations provided at that time, including the increase in neutron flux due to EPU. The ART and ΔRT_{NDT} values were calculated at 36, 40, and 54 effective full power years (EFPY). The reported values for 54 EFPY are intended to be applicable through the end of MNGP's current extended operating period (i.e., 60 years).

The purpose of this calculation is to develop 1/4T and 3/4T ART and ΔRT_{NDT} values for each MNGP RPV ferritic material exposed to end-of-life for fluence greater than 1.0×10^{17} n/cm^2 at the projected fluence levels for 80 years (72 EFPY) with updated fluence values [16].

2.0 METHODOLOGY

When surveillance data are limited or unavailable, RG1.99R2 [1] specifies that ART is calculated with the following equation:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \tag{1}$$

The "Initial RT_{NDT}" term refers to the reference temperature of nil ductility transition for the non-irradiated material.

The reference temperature shift, ΔRT_{NDT}, is defined in RG1.99R2 [1] as the shift in the reference temperature resulting from neutron irradiation. ΔRT_{NDT} is calculated from the product of the chemistry factor (CF) and fluence factor (FF) as follows:

$$\Delta RT_{NDT} = CF \cdot FF \tag{2}$$

The CF is a function of the weight percent copper (Cu) and weight percent nickel (Ni) of the weld and base metal (plate or forging) materials. Tables 1 and 2 of RG1.99R2 [1] provide the standard CF values used in this calculation.

The FF is based on the accumulated fast neutron exposure (E > 1 MeV) and is typically corrected by the thickness at the location of interest. The FF can be read directly from Figure 1 of RG1.99R2, or calculated using the following equation [1]:

$$FF = f^{0.28-0.10\log(f)} \tag{3}$$

Due to attenuation effects, the fluence decreases with distance into the RPV wall. Per RG1.99R2 [1], the calculated or measured fluence from the inside surface of the RPV is attenuated using the following formula:

$$f = f_{surf} \cdot e^{-0.24x} \tag{4}$$

Where: f = fast neutron fluence (10^{19} n/cm^2, E > 1 MeV)
f_{surf} = fast neutron fluence at the RPV inside surface (i.e., at base metal / cladding interface, same units as f)
x = depth into the RPV wall from the inside surface (inches)

For ASME Code, Section XI, non-mandatory Appendix G [2] evaluations, the "x" value is taken at one-quarter of the base metal thickness (1/4T) and three-quarter thickness of the base metal (3/4T). The fast neutron fluence can be attenuated through the stainless steel cladding on the inside surface of the RPV. By design, however, the cladding is treated purely as a lining, and not as a load-bearing member. Thus, for the purposes of this evaluation, the inside surface neutron fluence is considered to be at the base metal / cladding interface.

Margin (M), a conservative term defined in RG1.99R2 [1], accounts for uncertainty in the initial reference temperature and for variance in ΔRT_{NDT}. The margin is calculated using the following formula:

$$Margin = 2 \cdot \sqrt{\sigma_I^{\,2} + \sigma_\Delta^{\,2}} \tag{5}$$

Where: σ_I = the standard deviation for the initial RT_{NDT} (°F)
σ_Δ = the standard deviation for ΔRT_{NDT} (°F)

RG1.99R2 [1] states that the standard value of σ_Δ is 28°F for welds and 17°F for base metal (plates or forgings), and σ_Δ need not exceed 0.5 times the mean reference temperature shift (0.5 * ΔRT_{NDT}).

The σ_I term, which is related to the uncertainty in the precision of the Initial RT_{NDT}, is applied for values that are determined by measurement and also when generic or default values are used. For MNGP components where a σ_I value is not explicitly identified, σ_I is assumed to be equal to 0°F (Assumption 1 in Section 4.0) from heat-specific data.

When surveillance data exist (e.g., the ISP Representative Material or other Supplemental Surveillance Program (SSP) material) containing an identical match for the heat number of the vessel beltline material being evaluated, a separate procedure is used to evaluate the ART. This procedure first determines the credibility of the data and, using best estimate chemistry values, calculates a fitted CF. The fitted CF is then compared to the Table CF (defined above in Equation 2), and the greater of the two is used in subsequent ART calculations. If the surveillance data are credible, the margin (σ_Δ) may be cut in half, as specified in Section 2.1 of RG1.99R2 [1]. Detailed procedures to evaluate surveillance data in the manner described above can be found in Section 3 of Reference [4].

3.0 DESIGN INPUTS

The MNGP RPV is constructed of a series of plates, numbered 10 through 17 from top to bottom [7]. Two plates are joined at each elevation via circumferential and vertical welds. As shown in Figure 1, from Reference [16], at 72 EFPY the upper elevation of the RG1.99R2 fluence threshold ($1.0x10^{17}$ n/cm^2) is 190.48 inches above the bottom of active fuel (BAF) and the top of the beltline at 72 EFPY is at an elevation of 383.72 inches. Reference [7] specifies that the weld separating the lower intermediate shell plates (14 and 15) from the upper intermediate shell plates (12 and 13) is located at an elevation of 366.125 inches. Therefore, the upper intermediate plates must also be included in the ART evaluation.

The chemical composition of the MNGP RPV plates is obtained from several sources. The nickel content of the lower plates (A0946-1) and upper intermediate plates (C2089-1) is obtained from Reference [8]. The copper content of the lower plates is obtained from Table 4-1 of Reference [9]. Copper content is not available for the upper-intermediate plates; for conservatism, the bounding value of 0.35% copper specified in Section 1.1 of RG1.99R2 [1] is applied to these components (Assumption 2 in Section 4.0).

Attachment 1 of Reference [10] specifies updated copper and nickel values for the lower intermediate plates (C2220-1); these values supersede any prior information for these components. Reference [10] also specifies a fitted chemistry factor of 180.0°F based on the BWRVIP-135, Rev. 4 ISP data, which exceeds the default chemistry factor specified in the tables of Reference [1]. According to the discussion in the Attachment to Reference [10], the surveillance data used to determine the modified chemistry factor is credible. Therefore, the σ_{Δ} margin term is cut in half for the lower intermediate plates.

The 120° capsule information was unavailable at original authorship of this calculation. The fitted CF utilizing that information has been confirmed to be lower than the 180.0°F CF value, so the current analysis is conservative for Subsequent License Renewal. The value is currently EPRI proprietary and has not been directly referenced.

Initial RT_{NDT} values for the MNGP RPV plates are obtained from Table 5-1 of Reference [11]. In certain cases, multiple values are provided, based on different evaluation methods that are equally relevant. In such cases, it is assumed that selecting the minimum reported value is applicable for the ART calculations Assumption 3 in Section 4.0.

The vertical and circumferential welds that join the RPV plates must also be considered during the ART evaluation. Information on specific welds is not available; rather, Reference [12] provides parameters for a bounding beltline weld. Chemical composition information for the beltline weld is provided in Table 4-1 of Reference [12]. As described in Sections 3.1 and 3.2 of the same document, the Initial RT_{NDT} value for the bounding beltline weld is calculated from 45 tests performed on a sample specimen. The average calculated value is -65.6°F, with a standard deviation of 12.7°F. For the ART evaluation, these values are applied as the Initial RT_{NDT} and σ_I, respectively. These data have been publicly docketed (in submittals and in RVID2) and are considered non-proprietary.

According to the drawing in Reference [7], the centerline N2 recirculation inlet nozzles in the MNGP RPV are located at an elevation of 186 inches above the bottom of the reactor vessel. According to Reference [16], at 72 EFPY the lower elevation of the $1.0x10^{17}$ n/cm^2 fluence threshold corresponds to an RPV elevation of 190.48 inches. However, the elevation of the uppermost blend radius of the N2 nozzle is within the beltline, as shown in Appendix E of Reference [6]. Therefore, the N2 nozzles must also be included in the ART evaluation for 72 EFPY.

Similar to the upper intermediate shell plates, documentation of the copper content of the N2 nozzles is not available. Section 3.2 of Reference [13] provides a conservative estimate of copper content of 0.18% based on a statistical evaluation of beltline nozzles in other BWR plants (Assumption 4 in Section 4.0). Nickel content for each nozzle is identified in the RPV test reports in Reference [14]. The average of the reported values is 0.86%; this value, the best-estimate nickel content, is used to determine an N2 ART value. The Initial RT_{NDT} value is obtained from Table 5-2 of Reference [11], where a value of 40°F is common to all of the N2 nozzles.

Based on the boundary of the extended beltline [16] and examination of the RPV drawing [7], the N2 nozzles are the only forged nozzles in the extended beltline at 72 EFPY. There are no instrument nozzles in the extended beltline at 72 EFPY.

The maximum projected fluence levels for 72 EFPY at surface were taken from the latest report [16] but the vessel thickness for the Monticello RPV beltline region materials of 5.0625" [7] at base metal and cladding interface is used for fluence attenuation calculation. Although the fluence levels in the RPV at the 1/4T and 3/4T depths through the vessel thickness calculated per Equation 4 are less than the values in Reference [16], the methodology is consistent with the previous licensing document [11].

4.0 ASSUMPTIONS

The assumption made in order to define the evaluation approach and perform the analysis are summarized in the following list. The application of these assumptions is indicated throughout the document using a set of parentheses containing the appropriate assumption number; for example, Assumption #3 would be indicated as (Assumption 3 in Section 4.0).

1. According to RG1.99R2, the σ_I term is equal to the standard deviation of the Initial RT_{NDT} when that quantity is estimated from physical measurements [1]. However, for the MNGP evaluation, a number of components do not have a measured Initial RT_{NDT}; rather, a bounding value is estimated via alternative means. Values calculated by this method include substantial conservatism, rendering it unnecessary to create additional conservatism via the σ_I term. Consequently, for MNGP ART calculations, σ_I is set equal to zero unless the Initial RT_{NDT} for the component in question is estimated directly from measured data (e.g., in the case of the welds) or another source documents the specific σ_I term to utilize.
2. The copper content of the MNGP upper intermediate RPV shell plates is not documented. RG1.99R2 states that in cases where chemical composition is unknown, a conservative value of 0.35% copper may be used [1]. This approach is used herein to evaluate the ART values for the upper intermediate plates.
3. The Initial RT_{NDT} values listed in Tables 5-1 and 5-2 of Reference [11] are calculated by one of four different methods, as described in the footnotes accompanying the tables. In many cases, the values reported in Reference [11] have been conservatively increased from the estimated value. Additionally, multiple evaluation methods are often applicable for a particular RPV component. All of the methods are valid, so it is assumed that the minimum initial RT_{NDT} value reported for each component may be used for the ART evaluation. The values obtained by application of this assumption are consistent with those in MNGP's licensing basis documents [15].

4. Documentation of the copper content of the MNGP N2 nozzles is unavailable. However, this information is available for beltline nozzles at other BWR plants. Section 3.2 of Reference [13] offers an estimate of the copper content in nozzle forgings by means of statistical evaluation of available industry forging data. It is assumed that this approach is conservative and therefore applicable for the purposes of MNGP ART calculations.

5.0 CALCULATIONS

The methodology in Section 2.0 is used to evaluate the ART and ΔRT_{NDT} values for MNGP, based on the design inputs in Section 3.0 and consistent with the assumptions in Section 4.0. The design inputs, and resultant 0T, 1/4T and 3/4T ART values are given in Table 2, Table 3, and Table 4 for 72 EFPY.

6.0 CONCLUSIONS

This document contains ART and ΔRT_{NDT} values calculated in accordance with RG1.99R2 [1] for all MNGP plates, welds, and forgings exposed to fluence greater than $1.0x10^{17}$ n/cm^2. Design inputs are collected from a variety of sources, as discussed in Section 3.0. The calculated ART and ΔRT_{NDT} values at 0T, 1/4T and 3/4T are provided for 72 EFPY in Table 2, Table 3, and Table 4.

The bounding 0T ART value for the RPV plates and welds is 197.8°F and for the N2 nozzles is 123.9°F at 72 EFPY. The bounding 1/4T ART value for the RPV plates and welds is 182.7°F and for the N2 nozzles is 116.6°F at 72 EFPY. The bounding 3/4T ART value for the RPV plates and welds is 154.3°F and for the N2 nozzles is 100.5°F at 72 EFPY.

7.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988.
2. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” 2004 Edition.
3. SI Calculation 1000847.301, Rev. 0, “Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts,” Performed for Monticello P-T Curves Revision According to the PTLR Methodology, January 2011.
4. *BWRVIP-135, Revision 4: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations*. EPRI, Palo Alto, CA, 2021. 3002020996. **EPRI PROPRIETARY INFORMATION**.
5. Not used.
6. GE Hitachi Nuclear Energy Report No. NEDC-33307P, “Pressure-Temperature Curves for Nuclear Management Company LLC Monticello Nuclear Generating Plant,” Revision 0, February 2008, **GE PROPRIETARY INFORMATION**, SI File No. 1000847.203P.
7. Chicago Bridge and Iron Company Drawing No. 1, Revision 8, “General Plan, 17’2” I.D. x 63’-2” Ins Heads Nuclear Reactor,” NX-8290-13, SI File No. NSP-21Q-210.
8. Chicago Bridge and Iron Company Drawing No. R-7, Revision 0, “Skirt Knuckle, Heads & Shell & Misc Heat Number Summary for 17’-2” ID x 63’-2” INS. HDS. Nuclear Reactor,” NX-8290-133, SI File No. NSP-21Q-213.
9. GE Nuclear Energy Report No. SASR 88-99, “Implementation of Regulatory Guide 1.99, Revision 2 for the Monticello Nuclear Generating Plant,” Revision 1, January 1989, SI File No. NSP-21Q-202.
10. Letter from B. Carter (EPRI) to D. Potter (MNGP), “Evaluation of the Monticello 300° Surveillance Capsule Data,” BWR Vessel and Internals Project (BWRVIP), March 23, 2009, **EPRI PROPRIETARY INFORMATION**, SI File No. 1000207.202P.
11. Structural Integrity Associates, Inc. Report No. SIR-97-003, “Review of the Test Results of Two Surveillance Capsules, and Recommendations for the Materials Properties and Pressure-Temperature Curves to be used for the Monticello Reactor Pressure Vessel,” Revision 3, March 1999, SI File No. NSP-21Q-401.
12. GE Nuclear Energy Report No. SASR 87-61, “Revision of Pressure-Temperature Curves to Reflect Improved Beltline Weld Toughness Estimate for the Monticello Nuclear Generating Plant,” Revision 1, December 1987, SI File No. NSP-21Q-201.
13. *BWRVIP-173: BWR Vessel and Internals Project, Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials*. EPRI, Palo Alto, CA: 2007. 1014995. **EPRI PROPRIETARY INFORMATION**.
14. “Pressure Vessel Record, Exhibit D, Certified Test Reports,” Author, Date, and Revision Not Identified, SI File No. NSP-21Q-233.
15. Monticello Nuclear Generating Plant, “Application for Renewed Operating License,” Docket No. 50-263, March 2005.

16. “Monticello Nuclear Generating Station Reactor Pressure Vessel Fluence Evaluation – Subsequent License Renewal,” TransWare Enterprises, MNT-FLU-001-R-002, Revision 0. June 2022. SI File NO. 2100300.201.
17. Email from Nathan Palm (EPRI) to Max Smith (XCEL ENERGY), dated 03/09/2023, “RE: Epri Confirmation of Classifying,” SI File No 2100300.210.
18. Email from Max Smith (XCEL ENERGY) to Dan Denis (SI), dated 03/10/2023, “FW: BWRVIP 135 Rev 4 Proprietary Issue,” SI File No 2100300.210.

Table 1. Maximum E > 1.0 MeV Neutron Fluence for Monticello RPV Beltline Region at 72 EFPY

Beltline Component	Peak I.D. Fluence n/cm^2 [16]	1/4T Fluence n/cm^2	3/4T Fluence n/cm^2
Lower Shell (Course 1)	3.79E+18	2.80E+18	1.52E+18
Lower/Int Shell (Course 2)	5.94E+18	4.38E+18	2.39E+18
Upper/Int Shell (Course 3)	3.23E+17	2.38E+17	1.30E+17
Lower (Course 1) Axial Welds (VLAA-1 and VLAA-2)	2.35E+18	1.73E+18	9.45E+17
Lower- Int. (Course 2) Axial Welds (VLBA-1 and VLBA-2)	2.10E+18	1.55E+18	8.44E+17
Upper/Int Shell (Course 3) (VLCB-1 and VLCB-2)	2.12E+17	1.56E+17	8.52E+16
Horizontal Weld (VCBA-2)	3.79E+18	2.80E+18	1.52E+18
Horizontal Weld (VCBB-3)	3.23E+17	2.38E+17	1.30E+17
N2 Nozzles	7.08E+17	5.23E+17	2.85E+17

Notes:

1. Thickness of 5.0625” from base metal and cladding interface is used for fluence attenuation at 1/4T and 3/4T.
2. The fluence values at 1/4T and 3/4T are calculated using the attenuation method per Equation (4) to be consistent with the previous licensing document [11].

Table 2. Surface ART Values for Monticello RPV Components at 72 EFPY

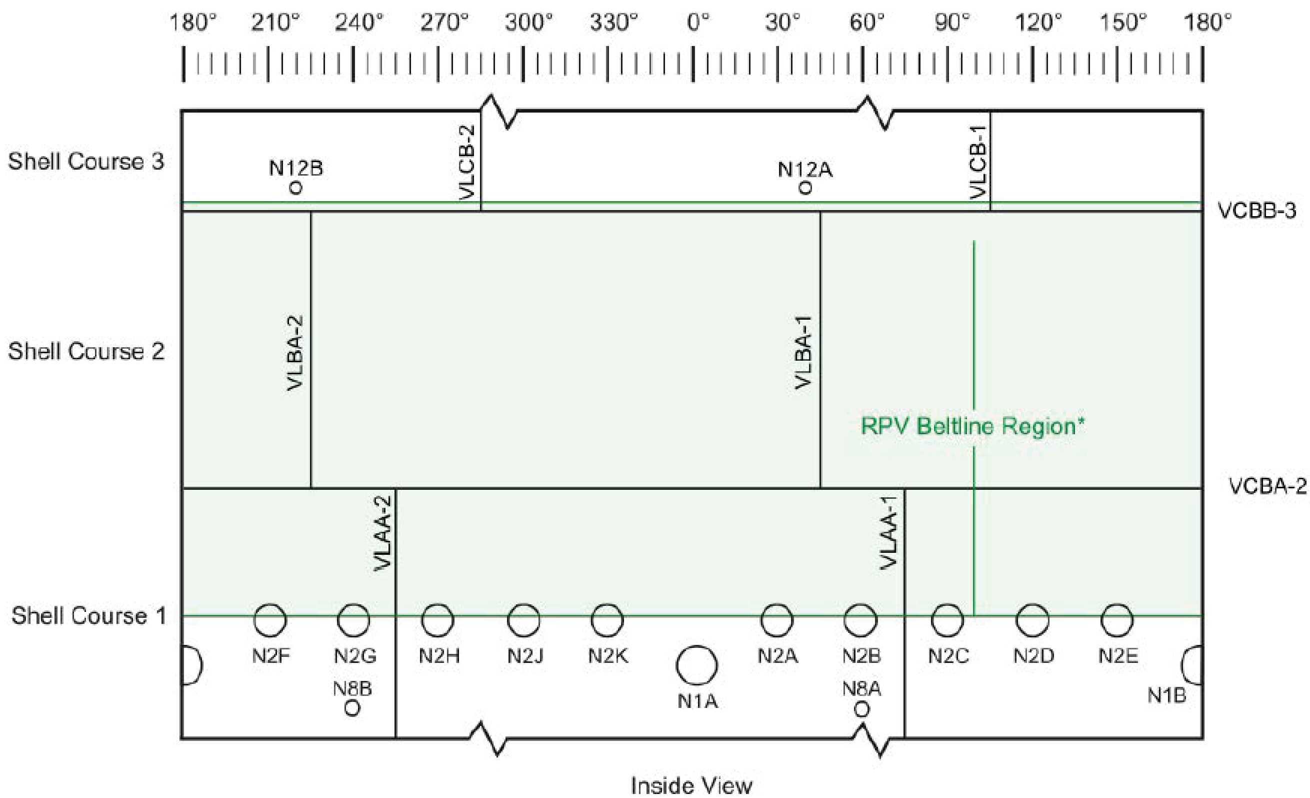
Component No.	Heat	Lot	% Cu	% Ni	CF	Initial RT_{NDT} (°F)	72EFPY 0T Fluence (n/cm^2)	Fluence Factor f	ΔRT_{NDT} (°F)	σ_i (°F)	σ_Δ (°F)	72 EFPY 0T ART (°F)
Lower Shell Plates (Course 1)												
I-16	A0946-1	N/A	0.14	0.56	98	27	3.79E+18	0.732	71.8	0	17.0	132.8
I-17	C2193-1	N/A	0.17	0.5	119	0	3.79E+18	0.732	86.7	0	17.0	120.7
Lower-Intermediate Shell Plates (Course 2)												
I-14	C2220-1	N/A	0.16	0.64	180	27	5.94E+18	0.854	153.8	0	8.5	197.8
I-15	C2220-2	N/A	0.16	0.64	180	27	5.94E+18	0.854	153.8	0	8.5	197.8
Upper/Int Shell Plates (Course 3)												
I-12	C2089-1	N/A	0.35	0.5	200	0	3.23E+17	0.229	45.7	0	17.0	79.7
I-13	C2613-1	N/A	0.35	0.49	198	27	3.23E+17	0.229	45.5	0	17.0	106.5
Lower Shell (Course 1) Axial Welds												
VLAA-1 & VLAA-2	-	E8018N	0.1	0.99	135	-65.6	2.35E+18	0.609	82.1	12.7	28.0	78.0
Lower-Intermediate Shell (Course 2) Axial Welds:												
VLBA-1 & VLBA-2	-	E8018N	0.1	0.99	135	-65.6	2.10E+18	0.581	78.4	12.7	28.0	74.3
Upper/Int Shell (Course 3) Axial Welds:												
VLCB-1 & VLCB-2	-	E8018N	0.1	0.99	135	-65.6	2.12E+17	0.178	24.1	12.7	12.0	-6.6
Circumferential Welds												
VCBA-2	-	E8018N	0.1	0.99	135	-65.6	3.79E+18	0.732	98.7	12.7	28.0	94.6
VCBB-3	-	E8018N	0.1	0.99	135	-65.6	3.23E+17	0.229	30.9	12.7	15.5	5.3
N2 Nozzle												
N2 Nozzle	E21VW	N/A	0.18	0.86	142	40	7.08E+17	0.351	49.9	0	17.0	123.9

Table 3. 1/4T ART Values for Monticello RPV Components at 72 EFPY

Component No.	Heat	Lot	% Cu	% Ni	CF	Initial RT_{NDT} (°F)	72EFPY 1/4T Fluence (n/cm^2)	Fluence Factor f	ΔRT_{NDT} (°F)	σ_i (°F)	σ_Δ (°F)	72 EFPY 1/4T ART (°F)
Lower Shell Plates (Course 1)												
I-16	A0946-1	N/A	0.14	0.56	98	27	2.80E+18	0.653	64.1	0	17.0	125.1
I-17	C2193-1	N/A	0.17	0.5	119	0	2.80E+18	0.653	77.3	0	17.0	111.3
Lower-Intermediate Shell Plates (Course 2)												
I-14	C2220-1	N/A	0.16	0.64	180	27	4.38E+18	0.770	138.7	0	8.5	182.7
I-15	C2220-2	N/A	0.16	0.64	180	27	4.38E+18	0.770	138.7	0	8.5	182.7
Upper/Int Shell Plates (Course 3)												
I-12	C2089-1	N/A	0.35	0.5	200	0	2.38E+17	0.191	38.2	0	17.0	72.2
I-13	C2613-1	N/A	0.35	0.49	198	27	2.38E+17	0.191	37.9	0	17.0	98.9
Lower Shell (Course 1) Axial Welds												
VLAA-1 & VLAA-2	-	E8018N	0.1	0.99	135	-65.6	1.73E+18	0.535	72.2	12.7	28.0	68.1
Lower-Intermediate Shell (Course 2) Axial Welds:												
VLBA-1 & VLBA-2	-	E8018N	0.1	0.99	135	-65.6	1.55E+18	0.510	68.8	12.7	28.0	64.7
Upper/Int Shell (Course 3) Axial Welds:												
VLCB-1 & VLCB-2	-	E8018N	0.1	0.99	135	-65.6	1.56E+17	0.147	19.8	12.7	9.9	-13.5
Circumferential Welds												
VCBA-2	-	E8018N	0.1	0.99	135	-65.6	2.80E+18	0.653	88.0	12.7	28.0	83.9
VCBB-3	-	E8018N	0.1	0.99	135	-65.6	2.38E+17	0.191	25.8	12.7	12.9	-3.6
N2 Nozzle												
N2 Nozzle	E21VW	N/A	0.18	0.86	142	40	5.23E+17	0.300	42.6	0	17.0	116.6

Table 4. 3/4T ART Values for Monticello RPV Components at 72 EFPY

Component No.	Heat	Lot	% Cu	% Ni	CF	Initial RT_{NDT} (°F)	72EFPY 3/4T Fluence (n/cm^2)	Fluence Factor f	ΔRT_{NDT} (°F)	σ_i (°F)	σ_Δ (°F)	72 EFPY 3/4T ART (°F)
Lower Shell Plates (Course 1)												
I-16	A0946-1	N/A	0.14	0.56	98	27	1.52E+18	0.506	49.7	0	17.0	110.7
I-17	C2193-1	N/A	0.17	0.5	119	0	1.52E+18	0.506	59.9	0	17.0	93.9
Lower-Intermediate Shell Plates (Course 2)												
I-14	C2220-1	N/A	0.16	0.64	180	27	2.39E+18	0.613	110.3	0	8.5	154.3
I-15	C2220-2	N/A	0.16	0.64	180	27	2.39E+18	0.613	110.3	0	8.5	154.3
Upper/Int Shell Plates (Course 3)												
I-12	C2089-1	N/A	0.35	0.5	200	0	1.30E+17	0.131	26.1	0	13.0	52.1
I-13	C2613-1	N/A	0.35	0.49	198	27	1.30E+17	0.131	25.9	0	13.0	78.8
Lower Shell (Course 1) Axial Welds												
VLAA-1 & VLAA-2	-	E8018N	0.1	0.99	135	-65.6	9.45E+17	0.406	54.7	12.7	27.4	49.5
Lower-Intermediate Shell (Course 2) Axial Welds:												
VLBA-1 & VLBA-2	-	E8018N	0.1	0.99	135	-65.6	8.44E+17	0.384	51.8	12.7	25.9	43.8
Upper/Int Shell (Course 3) Axial Welds:												
VLCB-1 & VLCB-2	-	E8018N	0.1	0.99	135	-65.6	8.52E+16	0.098	13.3	12.7	6.6	-23.7
Circumferential Welds												
VCBA-2	-	E8018N	0.1	0.99	135	-65.6	1.52E+18	0.506	68.2	12.7	28.0	64.1
VCBB-3	-	E8018N	0.1	0.99	135	-65.6	1.30E+17	0.131	17.6	12.7	8.8	-17.1
N2 Nozzle												
N2 Nozzle	E21VW	N/A	0.18	0.86	142	40	2.85E+17	0.213	30.2	0	15.1	100.5



Notes: This drawing is not to scale.
RPV beltline region is shown for 72 EFPY

Figure 1 Monticello RPV Beltline Region at 72 EFPY [16]

APPENDIX A

SUPPORTING FILES

Supporting Files

1. 2100300.302 Rev 4.xlsm Excel file contains the detailed ART calculations

ENCLOSURE 4

MONTICELLO NUCLEAR GENERATING PLANT

LICENSE AMENDMENT REQUEST

REVISION TO THE MNGP PRESSURE TEMPERATURE LIMITS REPORT TO CHANGE THE NEUTRON FLUENCE METHODOLOGY AND INCORPORATE NEW SURVEILLANCE CAPSULE DATA

MONTICELLO PRESSURE-TEMPERATURE LIMIT CURVES GENERATION FOR 72 EFPY

(NSPM CALCULATION NO. 23-012)

(SI NO. 2200284.303, REVISION 0)

(78 Pages Follow)

Xcel Energy	**Calculation Signature Sheet**

Approval: 602000018581

Document Information	
NSPM Calculation (Doc) No: 23-012	Revision: 0
Title: Monticello Pressure-Temperature Limit Curves Generation for 72 EFPY	
Facility: ☒ MT ☐ PI	Unit: ☒ 1 ☐ 2
Safety Class: ☒ SR ☐ Aug Q ☐ Non SR	
Type: **Calc** Sub-Type:	

NOTE:	**Print and sign name in signature blocks, as required.**

Major Revisions		☐ N/A
EC Number: 601000004404	☒ Vendor Calc: 2200284.303	
Vendor Name or Code: Structural Integrity Associates (SIA)	Vendor Doc No: 2200284.303	
Description of Revision: New calculation issuance		
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record		☒
Prepared by: (sign) Vendor / (print) SIA		Date: 12/18/23
Reviewed by: (sign) MOC 600001117674 / (print) Russell Lidberg		Date: See MOC 600001117674
Type of Review: ☐ Design Verification ☐ Engr Review ☒ OAR ☐ EOC		
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test		
Approved by: (sign) MOC 600001117673 / (print) Gus Hernandez		Date: MOC 600001117673

Xcel Energy

Calculation Signature Sheet

Minor Revisions	☒ N/A
EC No:	☐ Vendor Calc:
Minor Rev. No:	
Description of Change:	
Pages Affected:	
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record	☐
Prepared by: (sign) / (print)	Date:
Reviewed by: (sign) / (print)	Date:
Type of Review: ☐ Design Verification ☐ Engr Review ☐ OAR ☐ EOC	
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test	
Approved by: (sign) / (print)	Date:

Summary of Verification (summary is required for Design Verification):
☐ No Comments
☒ See attached QF0528
☐

Superseded Calculations:

Facility	Calc Document Number	Title
MNGP	11-005	Revised P-T Curves Calculation

Does the Calculation:

☐ YES ☒ No Affect piping or supports? (If YES, Attach MT Form 3544.) MONTI ONLY

☐ YES ☒ No Require Fire Protection Review? (Using QF2900, Fire Protection Program Impact Screen, determine if a Fire Protection Review is required.) If YES, document the engineering review in the EC. If NO, then attach completed QF2900 to the associated EC.

Design Revie Comment Form

Sheet 1 of 1

DOCUMENT NUM ER/ TITLE: File No: 2200284.303P

REVISION: 0 DATE: 12/2 /22

ITEM	REVIE ER S COMMENTS	PREPARER S RESOLUTION	REVIE ER S DISPOSITION
1 (RL)	Reference 1 lists SIR-05-044 revision 1. MNGP Tech-Specs section 5.6.5 lists SIR-05-044-A as the approved methodology for development of P-T curves. e either need a statement that the methodology used to develop this PTLR revision meets the requirements of both SIR-05-044 revisions or MNGP will need to do a LAR to update the Tech-Specs.	ill add a statement at the end of Section 1: The method used in this calculation meets the requirements of both the current revision and previous revision of SIR-05-044 [1].”	Acceptable
2 (RL)	Section 2 has a repeat sentence In some cases, a region may contain more than one component which is considered for development of the associated P-T curve.”	ill delete the repeated one.	Acceptable
3 (RL)	The 72 EFPY T fluence values in table 2 (pg 13) are different than the values listed in the Transware report MNT-FLU-001-R-002, table 3-2. Please e plain why.	The current 1/4T fluence values were calculated based on attenuation method per Regulatory Guide 1.99 Revision 2, which is consistent with previous TLAA. It was agreed between SI and Monticello when developing the ART values. The values are close to Transware s report.	Acceptable

Xcel Energy

Design Revie Comment Form

ITEM	REVIE ER S COMMENTS	PREPARER S RESOLUTION	REVIE ER S DISPOSITION
Reviewer: Russell Lidberg	Date: 1/3/23	Preparer: Vendor	Date: 1/3/23

Calculation 23-010/EC 601000004404: Monticello Pressure-Temperature Limit Curves Generation for 72 EFPY

Xcel Energy

External Design Document Suitability Review Checklist

External Design Document Being Reviewed:Monticello Pressure-Temperature Limit Curves Generation for 72 EFPY

Title: **Monticello Pressure-Temperature Limit Curves Generation for 72 EFPY**

Number: 2200284.303 Rev: 0 Date: 12/18/23

This design document was received from:

Organization Name: Structural Integrity Associates PO or DIA Reference: 4000024678

The purpose of the suitability review is to ensure that a calculation, analysis or other design document provided by an External Design Organization complies with the conditions of the purchase order and/or Design Interface Agreement (DIA) and is appropriate for its intended use. The suitability review does not serve as an independent verification. Independent verification of the design document supplied by the External Design Organization should be evident in the document, if required.

The reviewer should use the criteria below as a guide to assess the overall quality, completeness and usefulness of the design document. The reviewer is not required to check calculations in detail.

REVIEW

		Reviewed	N/A
1	Design Inputs used by the External Design Organization are appropriate.	☒	☐
2	Assumptions are described and reasonable.	☒	☐
3	Applicable codes, standards and regulations are identified and met.	☒	☐
4	Applicable construction and operating experience is considered.	☒	☐
5	Applicable structure(s), system(s), and component(s) are listed.	☒	☐
6	Formulae and equations are documented. Unusual symbols are defined.	☒	☐
7	Acceptance criteria are identified, adequate and satisfied.	☒	☐
8	Results are reasonable compared to inputs.	☒	☐
9	Source documents are referenced.	☒	☐
10	The document is appropriate for its intended use.	☒	☐
11	The document complies with the terms of the Purchase Order and/or DIA.	☒	☐
12	Inputs, assumptions, outputs, etc. which could affect plant operation are enforced by adequate procedural controls.	☒	☐
13	Plant impact has been identified and either implemented or controlled. If not identified in the document itself, identify the plant impacts and their associated tracking A/Rs and descriptions are listed in Table 1.	☒	☐
14	Design and Operational Margin have been considered and documented	☒	☐

Comments:

Completed by: Russell Lidberg 600001117674 Date: 12/18/23

Xcel Energy	External Design Document Suitability Review Checklist

TABLE 1

Initiate an AR to track open items and plant impacts (e.g., procedure revisions, validation of assumptions, database updates, etc.), if any.

Item No.	AR Tracking Number	PLANT IMPACT DESCRIPTION
1	none	
2		
3		
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15		

Xcel Energy

Design Information Transmittal DIT

Approval: ______________________________

From:	Russell Lidberg, Program Engineering, MNGP cel Energy				
To :	Dan Denis, P.E., Senior Consulaltant, Structural Integrity Associates				
Mod or Trac ing Number:	600000988024	Date:	12/16/22	DIT No:	PTLR-001
Mod Title:	PTLR Update to 72 EFPY				
Plant:	MNGP	Unit 1 ☒ Unit 2 ☐ Common ☐		Quality Classification	Safety Related

S CT:PT R pdate to FP

Check if applicable:

☐ **This DIT confirms information previously transmitted orally on _____ by _____ .**

☒ **This information is preliminary. See e planation belo .**

S RC F INF RM TI N (Source documents should be uniquely identified)

RVIP Letter 2022-053

D SCRIPTI N F INF RM TI N (rite the information being transmitted or list each document being transmitted)

Notification of New RVIP Integrated Surveillance Program (ISP) Data Applicable to the Monticello Reactor Pressure Vessel (RPV)

DISTRI TI N (Recipients should receive all attachments unless otherwise indicated. All attachments are uncontrolled unless otherwise indicated)

Dan Denis, P.E., Senior Consulaltant, Structural Integrity Associates

PPR D

Approver Name	Position	Signature	Date
Russell Lidberg	Program Engineer	Russell Lidberg	12/16/22

A copy of the DIT (along with any attachments not on file) should be sent to the Modification Folder.

Xcel Energy	Fire rote tion rogr m m t reen

Approval: 602000013088

Plant MT ☒ PI ☐

Activity Number: 601000004404 Activity Owner: Russell Lidberg
Type of Activity: ECR
rief Description of Activity: Revising the PTLR to incorporate updates to fluence pro ections and surveillance capsule data
References: PTLR, 23-010, 23-011, 23-012

1. Fire Protection Impact Considerations (Questions should be assessed for impact to Fire Protection Fundamentals, Calculations, Procedure, Programs, Systems and Components)		
		Does the Activity:
☐ Yes	☒ No	1. Add, remove, or create an opening (not filled by an approved penetration seal, door, or damper) in any Fire arrier wall, ceiling , or floor .
☐ Yes	☒ No	2. Add, remove, or modify any fire door, radiant heat shield, thermal shield, or fire damper with components other than replacement in ind
☐ Yes	☒ No	3. Add, remove, or modify the air flow CFM or the VAC discharge within three feet of a detector (e.g., air flow can affect the amount of air available for combustion and the timing of fire detection actuation)
☐ Yes	☒ No	4. Permanently add, remove, or modify any fire protective coating or wrap on any electrical raceway component (e.g., cable, conduit, or cable tray) (Ref the raceway drawings)
☐ Yes	☒ No	5. Add or remove any coatings (i.e., foam insulation, sound dampeners, and floor coatings)
☐ Yes	☒ No	6. Add, remove, or modify fireproofing or passive fire protection of any structural steel (Ref the system drawing)
☐ Yes	☒ No	. Add, relocate, or modify any wall, fence, door, building, trailer, or other structure that could affect the access or egress to safe shutdown components This includes impacts on fire brigade access, emergency lighting, and illumination levels.
☐ Yes	☒ No	8. Potentially affect personnel safety or SSCs within 50 feet of a large power transformer (10 MVA) due to fire or e plosion (including debris, pro ectiles, and blast effect) from a catastrophic failure of the transformer (SOER 10-1)
☐ Yes	☒ No	9. Add or remove a component in a FPP credited system
☐ Yes	☒ No	10. Modify the normal operating or failure position of a component in any FPP credited system (Ref the system P ID)
☐ Yes	☒ No	11. Alter normal or emergency system performance or operational characteristics associated with a FPP credited system (e.g., flow rates, temperatures, available volumes or capacities, pressures)(Ref the system P ID)
☐ Yes	☒ No	12. Modify any process monitoring (e.g., flow, temperature, level, pressure) instrumentation, including tubing or indication for any FPP credited system
☐ Yes	☒ No	13. Alter normal or available tan inventory water levels in a FPP credited system
☐ Yes	☒ No	14. Add, modify, or remove a component such that any unanaly ed flow bloc age, flow diversion, or inventory loss path are introduced in any FPP credited system
☐ Yes	☒ No	15. Alter any line si e or configuration within a FPP credited system
☐ Yes	☒ No	16. Affect a Fire Detection System component (e.g., power supply, cable type or routing, transmitter, control switch, control module)

Xcel Energy	Fire rote tion rogr m m t reen

1. Fire Protection Impact Considerations (Questions should be assessed for impact to Fire Protection Fundamentals, Calculations, Procedure, Programs, Systems and Components)		
		Does the Activity:
☐ Yes	☒ No	1 . Affect Fire Detection actuation logic, set points, interloc s, or software applications
☐ Yes	☒ No	18. Add, delete, or relocate detectors, or change detector type
☐ Yes	☒ No	19. Introduce potential obstructions (including possible overlapping obstructions) to the effective operation of fire suppression sprin lers, halon, or CO2 discharge no les Obstructions could include addition or relocation of cable trays, VAC ductwor , or panels, which could affect air flow, spray patterns, or bloc access to fire protection equipment.
☐ Yes	☒ No	20. Potentially impact the effectiveness of a gaseous suppression system (Any change to increase room volume or affect room integrity or nominal room operating temperature.)
☐ Yes	☒ No	21. Affect the location or type of manual fire suppression equipment (e.g., fire e tinguishers, hose stations, hydrants)
☐ Yes	☒ No	22. Affect access to manual or automatic fire suppression equipment or controls (e.g., alarms, pull stations, fire e tinguishers, hydrants, hose stations, isolation valves)
☐ Yes	☒ No	23. Impact suppression system piping or hangers
☐ Yes	☒ No	24. Affect the performance characteristics of any Fire Suppression system (E amples include water system flow rate or supply pressure, gaseous system pressure or volume, or system initiation time.)
☐ Yes	☒ No	25. Impact the magnitude of e pected fires by permanently adding or removing fi ed combustibles or flammable materials
☐ Yes	☒ No	26. Increase or decrease the quantity of oil in an area
☐ Yes	☒ No	2 . Impact any oil collection system, including the Reactor Coolant Pump Oil Collection System (if applicable)
☐ Yes	☒ No	28. Add or remove equipment containing oils (i.e., pumps, motors, oil filled transformers, air compressors)
☐ Yes	☒ No	29. Add any cable that is not IEEE-383-19 4 or an approved alternative.
☐ Yes	☒ No	30. Impact the performance or capacity of any fire propagation or water control features, including curbs, drains, or di es in a FPP credited system
☐ Yes	☒ No	31. Impact Fire Protection programmatic / procedural elements including: • Control of transient combustibles, including storage of combustibles, control of ha ardous materials, and combustible or flammable gases • Coatings program controls involving coating thic ness increases or combustible ratings • Controls for ignition sources, including the ot or program and temporary heating devices • FP Impairment logging, trac ing, and compensatory measures • Fire rigade staffing, structure, training/drills, equipment, communications, pre-fire plans, fire-related operating procedures, and off-site firefighting assistance • Fire protection surveillance procedures
☐ Yes	☒ No	32. Affect fire brigade training related to controlling the release of radioactivity
☐ Yes	☒ No	33. Impact the communication systems credited for use by the Fire rigade or by Operations during Post-Fire SSD activities
☐ Yes	☒ No	34. Install or reconfigure any ma or plant structures (e.g., walls, floors) that could impact the credited radio systems effectiveness

Xcel Energy	Fire rote tion rogr m m t reen

1. Fire Protection Impact Considerations (Questions should be assessed for impact to Fire Protection Fundamentals, Calculations, Procedure, Programs, Systems and Components)		
		Does the Activity:
☐ Yes	☒ No	35. Affect fire brigade capability by impacting access to suppression equipment or accessibility to any fire area Consider any change in building access, egress, paths of travel, and change in door status from normally unloc ed to normally loc ed. (Assume a large fire fighter wearing breathing apparatus and encumbered with equipment.)
☐ Yes	☒ No	36. Impact to the ISFSI Fire a ards Analysis including: • The receiving location, physical location or quantity of any combustible gases or flammable or combustible liquids stored in tan s or contained in plant equipment • Combustible loading within the ISFSI protected area fence • Equipment used to load or transport the cas onto the transit vehicle or to the ori ontal Storage Module (SM) • The path of travel for loaded cas s • Security procedures for allowing access into the Owner Controlled Area (OCA) for delivery truc s containing flammable or combustible liquids or gases
NEIL Impact (reference FP-E-NEIL-01)		
☐ Yes	☒ No	3 . Impact a NEIL insured Structure, fire protection system or component • Add a new structure • Change the occupancy classification of any part of a NEIL insured structure • Add a new NEIL required fire protection system • Add, modify, or remove an e isting NEIL required fire detection or fire suppression system • Create an addition to an e isting NEIL insured structure • Replace roof dec ing or covering • Does the change affect an interior finish such that it would not meet NEIL requirements • Reduce the fire rating of a NEIL required fire rated barrier • Add to, renovate, or alter the fire protection water supply or distribution systems, or use the fire protection water supply and distribution systems for other than emergency use • Add oil filled components over 50 gallons oil capacity, or increase the oil capacity of an e isting component greater than 50 gallons • Add to, renovate, or alter oil collection systems, fire barriers, or fire protection systems for oil filled components

☐ Yes	☒ No	Is there a potential impact to Classical FP program requirements as indicated by a Yes on any of questions 1-36 above (if yes, provide details below)
☐ Yes	☒ No	If a potential impact is identified, does the activity ma e a change to the Fire Protection Program (FP Program engineer document review below)

☐ Yes	☒ No	Is NEIL Impacted y this Change (as indicated by a Yes on question 3 above) (If so forward this form to the Fire Marshal to complete Section 8)

Xcel Energy	Fire rote tion rogr m m t reen

1. Fire Protection Impact Considerations (Questions should be assessed for impact to Fire Protection Fundamentals, Calculations, Procedure, Programs, Systems and Components)	
	Does the Activity:

2. Post-Fire Safe Shutdown Capability Considerations (Questions should be assessed for impact to achieving and maintaining safe shutdown in the event of a fire)		
		Does the Activity
☐ Yes	☒ No	1. Add or remove a component in a NSCA/Appendi R credited system
☐ Yes	☒ No	2. Modify the normal operating or failure position of a component in any NSCA/Appendi R credited system
☐ Yes	☒ No	3. Add, modify, or remove a component such that any unanaly ed flow bloc age, flow diversion, or inventory loss path are introduced in any NSCA/Appendi R credited system
☐ Yes	☒ No	4. Alter any line si e or configuration within a NSCA/Appendi R credited system
☐ Yes	☒ No	5. Add a branch line or modify an e isting branch line that may affect the mechanical boundary of any NSCA/Appendi R credited system
☐ Yes	☒ No	6. Alter normal or emergency system performance or operational characteristics associated with a NSCA/Appendi R credited system (e.g., flow rates, temperatures, available volumes or capacities, pressures)
☐ Yes	☒ No	. Modify any process monitoring (e.g., flow, temperature, level, pressure) instrumentation, including tubing or indication for any NSCA/Appendi R credited system
☐ Yes	☒ No	8. Alter normal or available tan inventory water levels in a NSCA/Appendi R credited system
☐ Yes	☒ No	9. Impact any NSCA/ Appendi R room heat up calculations or thermal hydraulic analyses
☐ Yes	☒ No	10. Modify or replace an MOV actuator in any NSCA/Appendi R credited system
☐ Yes	☒ No	11. Add, delete or modify the cable si e, cable design (including use of spare terminals or conductors), or cable routing for any NSCA/Appendi R credited system
☐ Yes	☒ No	12. Modify the control, power, indication or annunciation circuit for any NSCA/Appendi R credited component/system
☐ Yes	☐ No	13. Alter switchyard brea er alignments or interconnects
☐ Yes	☒ No	14. Alter the communication systems credited for use by the Fire rigade or by Operations during Post-Fire SSD activities
☐ Yes	☒ No	15. Install or reconfigure any ma or plant structures (e.g., walls, floors) that could impact the credited radio systems effectiveness
☐ Yes	☒ No	16. Permanently add, remove, or modify any fire protective coating or wrap on any electrical raceway component (e.g., cable, conduit, or cable tray)
☐ Yes	☒ No	1 . Modify or impact any performance characteristic of any fi ed eight hour battery emergency lighting unit required by Appendi R (MT only)

Xcel Energy	Fire rote tion rogr m m t reen

☐ Yes	☒ No	18. Introduce potential obstructions to emergency lighting units or their associated postfire shutdown access or egress paths
☐ Yes	☒ No	19. Is any plant equipment being modified such that electrical coordination is not being maintained
☐ Yes	☒ No	20. Is Engineering udgment being used to achieve electrical coordination (i.e., crediting cable length for time current characteristic curves with overlap)
☐ Yes	☒ No	Is there a potential impact to the NSCA/ Appendi R Safe Shutdown Capability as indicated by a Yes on any of questions 1-20 above (if yes, provide details below)
☐ Yes	☒ No	If a potential impact is identified, does the activity ma e a change to the NSCA/ Appendi R Safe Shutdown Capability (FP Program engineer document review below)

3.		***O*** *. Section 3-6 do not apply to Monticello.* Non-Power Operations Assessment Considerations (Questions should be assessed for impact to NPO fundamentals, Strategies, Procedures, Calculations, Analysis, Systems and Components.)
		Does the Activity:
☐ Yes	☐ No	1. Add, or remove a component in a NPO credited system
☐ Yes	☐ No	2. Modify the normal operating or failure position of a component in any NPO credited system
☐ Yes	☐ No	3. Add, modify, or remove a component such that any unanaly ed flow bloc age, flow diversion, or inventory loss path are introduced in any NPO credited system
☐ Yes	☐ No	4. Alter any line si e or configuration within a NPO credited system
☐ Yes	☐ No	5. Add a branch line or modify an e isting branch line that may affect the mechanical boundary of any NPO credited system
☐ Yes	☐ No	6. Alter normal or emergency system performance or operational characteristics associated with a NPO credited system (e.g., flow rates, temperatures, available volumes or capacities, pressures)
☐ Yes	☐ No	. Modify any process monitoring (e.g., flow, temperature, level, pressure) instrumentation, including tubing or indication for any NPO credited system
☐ Yes	☐ No	8. Alter normal or available tan inventory water levels in a NPO credited system
☐ Yes	☐ No	9. Modify or replace an MOV actuator in any NPO credited system
☐ Yes	☐ No	10. Add, delete or modify the cable si e, cable design (including use of spare terminals or conductors), or cable routing for any NPO credited system
☐ Yes	☐ No	11. Modify the control, power, indication or annunciation circuit for any NPO credited component/system
☐ Yes	☐ No	12. Alter switchyard brea er alignments or interconnects
☐ Yes	☐ No	13. Alter the communication systems credited for use by the Fire rigade or by Operations during Post-Fire SSD activities
☐ Yes	☐ No	14. Install or reconfigure any ma or plant structures (e.g., walls, floors) that could impact the credited radio systems effectiveness
☐ Yes	☐ No	15. Permanently add, remove, or modify any fire protective coating or wrap on any electrical raceway component (e.g., cable, conduit, or cable tray)
☐ Yes	☐ No	16. Is any plant equipment being modified such that electrical coordination is not being maintained
☐ Yes	☐ No	1 . Is Engineering udgment being used to achieve electrical coordination (i.e., crediting cable length for time current characteristic curves with overlap)

☐ Yes	☐ No	Is there a potential impact to the Non-Power Operations (NPO) analysis as indicated by a Yes on any of questions 1-1 above (if yes, provide details below)

Xcel Energy	Fire rote tion rogr m m t reen

☐ Yes	☐ No	If a potential impact is identified, does the activity ma e a change to the Non Power Operations (NPO) analysis (FP Program engineer document review below)

Xcel Energy	Fire rote tion rogr m m t reen

4. ***O*** *. Section 3-6 do not apply to Monticello.* Radioactive Release Considerations (Questions should be assessed for impact to Rad Release fundamentals, Fire Fighting Strategies, Training, Procedures, Calculations, Analysis, Engineering Controls, Systems and Components.)		
		Does the Activity:
☐ Yes	☐ No	1. Affect VAC flow rates and paths within Radiation Control Areas
☐ Yes	☐ No	2. Affect the ability to control or monitor the release of radioactive materials during fire suppression activities
☐ Yes	☐ No	3. Affect fire brigade training related to controlling the release of radioactivity
☐ Yes	☐ No	4. Permanently remove a penetration seal, add a new seal with an unapproved seal type or material, or replace a penetration seal with an unapproved seal type or material for an RCA
☐ Yes	☐ No	5. Add, remove, or create an opening (not filled by an approved penetration seal, door, or damper) in any fire barrier wall, ceiling, or floor in an RCA
☐ Yes	☐ No	6. Impact the performance or capacity of any fire propagation or water control features, including curbs, drains, or di es in a FPP credited system
☐ Yes	☐ No	. Add or remove a potential containment release path. Evaluation should include potential intersystem LOCA paths.

☐ Yes	☐ No	Is there a potential impact to Radioactive Release considerations as indicated by a Yes on any of questions 1- above (if yes, provide details below)
☐ Yes	☐ No	If a potential impact is identified, does the activity ma e a change to the Fire Protection Program credited Radioactive Release Considerations (FP Program engineer document review below)

5.		***O*** *. Section 3-6 do not apply to Monticello.* NFPA 805 Section Methodology Requirements or Previously Approved Alternatives (Questions should be assessed for impact to NSCA fundamentals, Fire Modeling, Procedures, Calculations, Analysis, and Evaluations.)
		Does the Activity:
☐ Yes	☐ No	1. Impact the methodology of NFPA 805 section 2.4
☐ Yes	☐ No	a. Fire Modeling (2.4.1)
☐ Yes	☐ No	b. Nuclear Safety Capability Assessment (2.4.2)
☐ Yes	☐ No	c. Fire Ris Evaluation (2.4.3)
☐ Yes	☐ No	d. Plant Change Evaluation (2.4.4)

☐ Yes	☐ No	Is there a potential impact to NFPA 805 Methodology requirements (if yes, provide details below)
☐ Yes	☐ No	If a potential impact is identified, are NFPA 805 Methodology requirements or Previously Approved Alternatives impacted by this change (FP Program engineer document review below)

For any YES answers above provide details how the activity impacts the NFPA 805 Methodology:

Xcel Energy	Fire rote tion rogr m m t reen

6. **O** . *Section 3-6 do not apply to Monticello.*
Fire PRA (Questions should be assessed for impact to Rad Release fundamentals, Fire Modeling, Procedures, Calculations, Analysis, and Evaluations.)

		Does the Activity:
☐ Yes	☐ No	1. Add or remove a component in a FPRA credited system
☐ Yes	☐ No	2. Modify the normal operating or failure position of a component in any FPRA credited system
☐ Yes	☐ No	3. Add, modify, or remove a component such that any unanaly ed flow bloc age, flow diversion, or inventory loss path are introduced in any FPRA credited system
☐ Yes	☐ No	4. Alter normal or emergency system performance or operational characteristics associated with a FPRA credited system (e.g., flow rates, temperatures, available volumes or capacities, pressures)
☐ Yes	☐ No	5. Modify any process monitoring (e.g., flow, temperature, level, pressure) instrumentation, including tubing or indication for any FPRA credited system
☐ Yes	☐ No	6. Modify or replace an MOV actuator in any FPRA credited system
☐ Yes	☐ No	. Add, delete or modify the cable si e, cable design (including use of spare terminals or conductors), or cable routing for any FPRA credited system
☐ Yes	☐ No	8. Add any cable that is not IEEE-383-19 4 or an approved alternative per NFPA 805 FAQ 06-0022 Rev 3
☐ Yes	☐ No	9. Modify the control, power, indication or annunciation circuit for any FPRA credited component/system
☐ Yes	☐ No	10. Alter switchyard brea er alignments or interconnects
☐ Yes	☐ No	11. Modify or impact any performance characteristic of any fi ed eight hour battery emergency lighting unit
☐ Yes	☐ No	12. Introduce potential obstructions to emergency lighting units or their associated post-fire shutdown access or egress paths
☐ Yes	☐ No	13. Add a new ignition source
☐ Yes	☐ No	14. Modify, relocate, remove, or change the equipment name of an e isting ignition source
☐ Yes	☐ No	15. Alter the physical dimensions or cabinet venting characteristics of an e isting ignition source
☐ Yes	☐ No	16. Add, modify, or remove ventilation or cable penetration openings in an electrical cabinet (e.g., electrical panel, unction bo , MCC, switchgear)
☐ Yes	☐ No	1 . Introduce permanent intervening combustibles or flammable materials
☐ Yes	☐ No	18. Increase or decrease the quantity of oil in an area
☐ Yes	☐ No	19. Alter any oil collection system, including the Reactor Coolant Pump Oil Collection System (if applicable)
☐ Yes	☐ No	20. Add or remove equipment containing oils (i.e., pumps, motors, oil filled transformers, air compressors)
☐ Yes	☐ No	21. Does the change add, delete, or relocate fire detectors, or change fire detector type
☐ Yes	☐ No	22. Does the change add, delete, change type of fi ed suppression system, or change effective one covered by the suppression system
☐ Yes	☐ No	23. Affect the performance characteristics of any Fire Suppression system (E amples include distance from suppression or system initiation time.)
☐ Yes	☐ No	24. Add, remove, or create an opening in any wall, ceiling, or floor
☐ Yes	☐ No	25. Permanently add, remove, or modify any fire protective coating or wrap on any electrical raceway component (e.g., cable, conduit, or cable tray)
☐ Yes	☐ No	26. Add, remove, or modify fireproofing or passive fire protection of any structural steel

Xcel Energy	Fire rote tion rogr m m t reen

6.	***O***	*. Section 3-6 do not apply to Monticello.* Fire PRA (Questions should be assessed for impact to Rad Release fundamentals, Fire Modeling, Procedures, Calculations, Analysis, and Evaluations.)
		Does the Activity:
☐ Yes	☐ No	2 . Add, remove, or modify any fire door, radiant heat shield, thermal shield, or fire damper with components other than replacement in ind
☐ Yes	☐ No	28. Add, relocate, or modify any wall fence, door, building, trailer, or other structure that could affect the access or egress to FPRA components
☐ Yes	☐ No	29. Result in an increase or decrease in the amount of open floor space in a room or area
☐ Yes	☐ No	30. Impact the performance or capacity of any fire propagation or water control features, including curbs, drains, or di es in a FPRA credited system
☐ Yes	☐ No	31. Alter any local or Control Room instrumentation or controls, including changes to layout or function, of a credited FPRA system
☐ Yes	☐ No	32. E ceed the ma fill capacity of any cable tray
☐ Yes	☐ No	33. Affect plant operating procedures such as altering normal or emergency systems operation or alignments (including offsite power) or Operations responses to abnormal (including fire and annunciators) or emergency conditions
☐ Yes	☐ No	34. Is any plant equipment being modified such that electrical coordination is not being maintained
☐ Yes	☐ No	35. Is engineering udgment being used to achieve electrical coordination (i.e., crediting cable length for time characteristic curves with overlap)
☐ Yes	☐ No	Is there a potential impact to the Fire PRA as indicated by a Yes on any of questions 1-35 above (if yes, provide details below)
☐ Yes	☐ No	If a potential impact is identified, does the activity ma e a change to the Fire Protection Program credited Fire PRA (FP Program engineer document review below)

Xcel Energy	Fire rote tion rogr m m t reen

. C D R		
		Does the Activity:
☐ Yes	☒ No	re irements re im te t e nge t en a. Forward the form to the Fire Marshal to complete Section 8. otenti im t s not i enti ie in e tions t en a. Activity Owner / RE shall sign and date below as preparer. b. No additional FP Program evaluation is required. c. hen used with an engineering change. this form should be retained with the engineering change pac age. d. 10 CFR 50.59 Applicability Determination should be mar ed No for affecting FPP. otenti im t s i enti ie in e tions t en ition revie is re ire a. Activity Owner / RE shall sign and date below as preparer. b. Contact the Fire Protection Program Engineer to review if the activity ma es a change to the Fire Protection Program t e F rogr m ngineer revie in e tions etermines t t t e tivit m es nge to t e F a. FP Program Engineer shall sign and date below as the reviewer b. Initiate a Fire Protection Change Review (QF2901) or Fire Protection Change Evaluation (QF2902), as applicable. c. 10 CFR 50.59 Applicability Determination should be mar ed Yes for affecting FPP. If required to be retained per FP-PE-FP-01, QF2900 Fire Protection Program Impact Screen S ALL be retained for the life of the plant with the change pac age.

R

Preparer(s):

Russell Lidberg / Russell Lidberg 600001117674 12/18/23

Print Name / Sign Date

Fire Protection Program Engineer (if required):

n/a

Print Name / Sign Date

Xcel Energy	**Fire Protection Program Impact Screen**

8. **N I Revie**
This section is to be completed by the Fire Marshal/ Fire Protection Coordinator when the NEIL property loss impact question at the end of Section 1 is marked 'yes'.
If a planned addition, renovation or alteration involves a structure, system or component that is or will be insured by NEIL (See FP-E-NEIL-01 for list of applicable SSCs) and the change is permanent (in place over 180 days) and any of the following questions are answered yes, a NEIL design review is required in accordance with FP-E-NEIL-01 and the NEIL Loss Control Standards. Does the change activity: • Add a new structure? • Change the occupancy classification of any part of a NEIL insured structure? • Add a new NEIL required fire protection system? • Add, modify, or remove an e isting NEIL required fire detection or fire suppression system? • Create an addition to an e isting NEIL insured structure? • Replace roof dec ing or covering? • Does the change affect an interior finish such that it would not meet NEIL requirements? • Reduce the fire rating of a NEIL required fire rated barrier? • Add to, renovate, or alter the fire protection water supply or distribution systems, or use the fire protection water supply and distribution systems for other than emergency use? • Add oil filled components over 50 gallons oil capacity, or increase the oil capacity of an e isting component greater than 50 gallons? • Add to, renovate, or alter oil collection systems, fire barriers, or fire protection systems for oil filled components? If any of the above questions were answered yes , provide details of compliance with NEIL requirements, any NEIL deviations and NEIL acceptance below.
Fire Marshal: n/a Print Name / Sign   Date **Fire Protection Program nginee if required :** Print Name / Sign   Date

File No.: 2200284.303
Project No.: 2200284
Quality Program Type: ☒ Nuclear ☐ Commercial

CALCULATION PACKAGE

PROJECT NAME:
Monticello P-T Limit Curves Generation for 72 EFPY (with SLR TLAA Synergy)

CONTRACT NO.:
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CLIENT:
Xcel Energy

PLANT:
Monticello Nuclear Generating Plant

CALCULATION TITLE:
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1.0 INTRODUCTION

Pressure-temperature (P-T) limit curves for the beltline, bottom head, and non-beltline regions of the Monticello Nuclear Generating Plant (MNGP) reactor pressure vessels (RPV) were developed for 54 effective full power years (EFPY) in Reference [13]. This calculation updates the P-T curves for 72 EFPY of operation. The P-T curves are prepared using the method documented in the Boiling Water Reactor Owner's Group (BWROG) Licensing Topical Reports (LTRs), "Pressure Temperature Limits Report Methodology for Boiling Water Reactors" [1] which satisfies the requirements of 10CFR50 Appendix G [3] and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Nonmandatory Appendix G [4]. The method used in this calculation meets the requirements of both the current revision and previous revision of SIR-05-044 [1].

2.0 METHODOLOGY

A full set of P-T curves, applicable to the following plant conditions, are prepared:

1. Pressure Test (Curve A),
2. Normal Operation – Core Not Critical (Curve B), and
3. Normal Operation – Core Critical (Curve C).

For each plant condition above, separate curves are provided for each of the following three regions of the RPV as well as a composite curve for the entire RPV:

1. The beltline region (includes nozzles where 1/4T fluence > 1 x 10^{17} n/cm^2),
2. The bottom head region,
3. The non-beltline region, including the top head flange,
4. Composite curve (bounding curve for all regions)

In some cases, a region may contain more than one component which is considered for development of the associated P-T curve. For the beltline region, the P-T curves incorporate components with fluence > 1 x 10^{17} n/cm^2 (E > 1 MeV). The instrument nozzles are not in the beltline region per Reference [19] and will not be included in the P-T curves evaluations. The Feedwater nozzle is assumed to be the bounding component for non-beltline, see Assumption 2 in Section 3.0. For MNGP, the curve for each vessel region identified above is composed from the bounding P-T limits determined from the following:

1. Beltline:
    a. Beltline shell
    b. Recirculation inlet nozzle, N2
2. Non-beltline
    a. Feedwater (FW) nozzle
    b. 10CFR50 Appendix G limits [3]
3. Bottom Head:
    a. Bottom head penetrations (in-core monitor housings, control rod drive housings)

Consequently, separate P-T curves are prepared for each component considered for each region, then a bounding curve is drawn from the individual P-T curves. Complete sets of P-T curves, as identified above, are provided for 72 EFPY of operation for the limiting Service Level A/B (Normal/Upset) thermal transient.

The methodology for calculating P-T curves, described below, is taken from Reference [1] unless specified otherwise.

The P-T curves are calculated by means of an iterative procedure, in which the following steps are performed:

Step 1: A fluid temperature, T, is assumed. The P-T curves are calculated considering a postulated flaw with a 6:1 aspect ratio that extends ¼ of the way through the vessel wall. The temperature at the postulated flaw tip is conservatively assumed equal to the coolant temperature.

Step 2: The static fracture toughness, K_{Ic}, is computed using the following equation from [4]:

$$K_{Ic} = 33.2 + 20.734 \cdot e^{0.02(T-ART)} \tag{1}$$

Where:	K_{Ic}	=	the lower bound static initiation critical fracture toughness (ksi√in).
	T	=	the metal temperature at the tip of the postulated ¼T flaw (°F).
	ART	=	the Adjusted Reference Temperature (ART) for the limiting material in the RPV region under consideration (°F).

Step 3: The allowable stress intensity factor due to pressure, K_{Ip}, is calculated as:

$$K_{Ip} = \frac{K_{Ic} - K_{It}}{SF} \tag{2}$$

Where:	K_{Ip}	=	the allowable stress intensity factor due to membrane (pressure) stress (ksi√in).
	K_{Ic}	=	the lower bound static fracture toughness calculated in Eq. (1) (ksi√in).
	K_{It}	=	the thermal stress intensity factor (ksi√in) from through wall thermal gradients.
	SF	=	the ASME Code recommended safety factor, based on the reactor condition. For hydrostatic and leak test conditions (i.e., P-T Curve A), SF = 1.5. For normal operation, both core non-critical and core critical (i.e., P-T Curves B and C), SF = 2.0.

When calculating values for Curve A, the thermal stress intensity factor is neglected (K_{It} = 0), since the hydrostatic leak test is performed at or near isothermal conditions.

For Curve B and Curve C calculations, K_{It} is computed in different ways based on the evaluated region. For the beltline, with the exception of nozzles, and bottom head regions, K_{It} is determined using the following equation [4] for a postulated inside surface connected flaw:

$$K_{It} = 0.953 \times 10^{-3} \cdot CR \cdot t^{2.5} \tag{3}$$

Where:	CR	=	the cooldown rate of the vessel (°F/hr).
	t	=	the RPV wall thickness (in).

For the FW nozzle/upper vessel region and the N2 recirculation nozzle, K_{It} is obtained from the stress distribution output of plant-specific finite element analyses (FEA). A polynomial curve-fit is determined for the through-wall stress distribution at the bounding time point. The linear elastic fracture mechanics (LEFM) solution for K_{It} is obtained from Reference [1]:

$$K_{It\_Poly} = \sqrt{\pi a}\left[0.706C_{0t} + \frac{2a}{\pi}\cdot 0.537C_{1t} + \frac{a^2}{2}\cdot 0.448C_{2t} + \frac{4a^3}{3\pi}\cdot 0.393C_{3t}\right] \tag{4}$$

Where:			
	a	=	¼T postulated flaw depth, a = ¼ t (in).
	t	=	thickness of the cross-section through the nozzle at the limiting path near the inner blend radius (in).
	C_{0t},C_{1t}, C_{2t},C_{3t}	=	thermal stress polynomial coefficients, obtained from a curve-fit of the extracted stresses from a transient FEA [11, 12].

The thermal stress polynomial coefficients are based on the assumed polynomial form of $\sigma(x) = C_0 + C_1 \cdot x + C_2 \cdot x^2 + C_3 \cdot x^3$. In this equation, "$x$" represents the radial distance in inches from the inside surface to any point on the crack face.

The transient FEA is performed assuming a fixed thermal shock between a high and a low temperature. In reality, the actual thermal shock varies for each evaluation step, as the maximum temperature is bounded by the pressure-temperature saturation curve. Thus, the value of K_{It} calculated in Equation 4 can be scaled to account for the maximum thermal shock, as shown in the following expression:

$$K_{It} = K_{It\_Poly}\cdot F_{scaling} = K_{It\_Poly}\cdot\left(\frac{T_{sat} - T_{low}}{T_{high} - T_{low}}\right) \tag{5}$$

Where:			
	K_{It}	=	the scaled thermal stress intensity factor, which is subsequently used in Equation 2 ($ksi\sqrt{inch}$)
	K_{It_Poly}	=	the thermal stress intensity factor computed from the polynomial expression defined in Equation 4 ($ksi\sqrt{inch}$)
	$F_{scaling}$	=	the scaling factor to apply to the polynomial stress intensity factor
	T_{sat}	=	the saturation temperature of the reactor (°F)
	T_{low}	=	the lower limit of the thermal shock applied to the FEA (°F)
	T_{high}	=	the upper limit of the thermal shock applied to the FEA (°F)

T_{sat} is determined from the pressure-temperature saturation curve. A power fit of this curve is developed in Appendix C, resulting in the following equation:

$$T_{sat} = 119.3\cdot 0.7987^{\frac{1}{P_{sat}}}\cdot P_{sat}^{\;0.2198} \tag{6}$$

In the above equation, P_{sat} is the saturation pressure corresponding to T_{sat}. For the purposes of this evaluation, P_{sat} is conservatively applied at the final P-T curve pressure (P_{P-T}), which is calculated below in Equation 12. This results in an iterative calculation process for each evaluation step, where a saturation pressure is assumed, a scaling factor is determined, the final pressure is computed, and the assumed saturation pressure is adjusted until the results achieve a suitable level of convergence.

Step 4: The allowable internal pressure of the RPV is calculated differently for each evaluation region. For the beltline region, with the exception of nozzles, the allowable pressure is determined as follows:

$$P_{allow} = \frac{K_{Ip} \cdot t}{M_m \cdot R_i} * 1000 \quad (7)$$

Where: P_{allow} = the allowable RPV internal pressure (psig).

K_{Ip} = the allowable stress intensity factor due to membrane (pressure) stress, as defined in Eq. (2) (ksi√in).

t = the RPV wall thickness (in).

M_m = the membrane correction factor for an inside surface axial flaw:

$M_m = 1.85$ for $\sqrt{t} < 2$

$M_m = 0.926 \sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$

$M_m = 3.21$ for $\sqrt{t} > 3.464$.

R_i = the inner radius of the RPV, per region (in).

For the bottom head region, the allowable pressure is calculated with the following equation:

$$P_{allow} = \frac{2 \cdot K_{Ip} \cdot t}{SCF \cdot M_m \cdot R_i} * 1000 \quad (8)$$

Where: SCF = conservative stress concentration factor to account for bottom head penetration discontinuities; SCF = 3.0 per Reference [1].

P_{allow}, K_{Ip}, t, M_m and R_i are defined in Eq. (7).

For the FW nozzle/ upper vessel region, and the N2 nozzle, the allowable pressure is determined from a ratio of the allowable and applied stress intensity factors. The applied factor can be determined from an FEA that determines the stresses due to the internal pressure on the nozzle and RPV. The methodology for this approach is as follows:

$$P_{allow} = \frac{K_{Ip} \cdot P_{ref}}{K_{Ip-app}} \quad (9)$$

Where: P_{ref} = RPV internal pressure at which the FEA stress coefficients (Eq. (10)) are determined (psi).

K_{Ip-app} = the applied pressure stress intensity factor (ksi√in).

P_{allow} and K_{Ip} are defined as in Eq. (7).

The applied pressure stress intensity factor for the FW nozzle and N2 nozzle is determined using a polynomial curve-fit approximation for the through-wall pressure stress distribution from a plant-specific FEA and the LEFM solution given in Eq. (8) [1]:

$$K_{Ip-app} = \sqrt{\pi a}\left[0.706C_{0p} + \frac{2a}{\pi}\cdot 0.537C_{1p} + \frac{a^2}{2}\cdot 0.448C_{2p} + \frac{4a^3}{3\pi}\cdot 0.393C_{3p}\right] \qquad (10)$$

Where: a = ¼ through-wall postulated flaw depth, a = ¼ t (in).

t = thickness of the cross-section through the limiting nozzle inner blend radius corner (in).

C_{0p},C_{1p}, C_{2p},C_{3p} = pressure stress polynomial coefficients, obtained from a curve-fit of the extracted stresses from an FEA.

Step 5: Steps 1 through 4 are repeated in order to generate a series of P-T points; the fluid temperature is incremented with each repetition. Calculations proceed in this iterative manner until 1,300 psig is reached. This value bounds the design pressure given in Section 4.

Step 6: Table 1 below summarizes the minimum temperature requirements contained in 10CFR50, Appendix G [3, Table 1], which are applicable to the material highly stressed by the main closure flange bolt preload (non-beltline curve). Additional minimum temperature requirements for bolt-up are included as shown in Table 1 below.

Note that the minimum bolt-up temperature of 60°F, is used here, consistent with the position given in Reference [1]. Further, some utilities specifically request that the minimum moderator temperature used in the plant shutdown margin evaluation be applied as a minimum bolt-up temperature requirement; it is also included in Table 1 but not required by MNGP. An additional 60°F margin is recommended in 10 CFR50, Appendix G [3, Table 1]. For P-T Curves A and B, this 60°F margin is only a recommendation, but for Curve C, the 60°F margin is required.

Table 1. Summary of Minimum Temperature Requirements for P-T Limit Curves.

Curve	Pressure Range	Minimum Metal Temperature	P-T Limits
A	$P \leq 20\% P_h$	Maximum of: • $RT_{NDT,max}$, • 60°F [1], • T_{SDM}	ASME Appendix G [4] requirements
	$P > 20\% P_h$	$RT_{NDT,max}$ + 90°F	ASME Appendix G [4] requirements
B	$P \leq 20\% P_h$	Maximum of: • $RT_{NDT,max}$, • 60°F [1], • T_{SDM}	ASME Appendix G [4] requirements
	$P > 20\% P_h$	$RT_{NDT,max}$ + 120°F	ASME Appendix G [4] requirements
C	$P \leq 20\% P_h$	Maximum of: • $RT_{NDT,max}$ + 60°F, • 60°F [1], • T_{SDM}	ASME Appendix G [4] requirements + 40°F
	$P > 20\% P_h$	Maximum of: • $RT_{NDT,max}$ + 160°F, • T_{ISHT}	ASME Appendix G [4] requirements + 40°F

Where: P_h is the pre-service hydrotest pressure, 1563 psig for MNGP [8].
$RT_{NDT,max}$ is the maximum RT_{NDT} of the vessel materials highly stressed by the bolt preload.
T_{SDM} is the temperature used in the shutdown margin evaluation.
T_{ISHT} is the minimum temperature at which the maximum in-service hydrotest pressure (1025 psig) [8] is allowed per Curve A.

Step 7: Uncertainty in the RPV pressure and metal temperature measurements is incorporated by adjusting the P-T curve pressure and temperature using the following equations:

$$T_{P-T} = T + U_T \tag{11}$$

$$P_{P-T} = P_{allow} - P_H - U_P \tag{12}$$

Where:
- T_{P-T} = The allowable coolant (metal) temperature (°F).
- U_T = The coolant temperature instrument uncertainty (°F).
- P_{P-T} = The allowable reactor pressure (psig).
- P_H = The pressure head to account for the water in the RPV (psig). Can be calculated from the following expression: $P_H = \rho \cdot \Delta h$.
- ρ = Water density at ambient temperature (lbm/in^3).
- Δh = Elevation of full height water level in RPV (in).
- U_P = The pressure instrument uncertainty (psig).

Steps 1 through 7, above, are implemented for all components and in all regions.

Nozzles in the beltline introduce stress concentration effects and have the potential to be more limiting than the generic beltline P-T curves. Nozzles or discontinuities outside the beltline are considered to be bounded by the upper vessel / feedwater nozzle or bottom head region P-T curves [1]. Beltline nozzles may be bounded by the upper vessel / feedwater nozzle curve if all of the following are met: the feedwater nozzle experiences more severe thermal transients, the feedwater nozzle RT_{NDT} is greater than or equal to the beltline nozzle ART, and the beltline and feedwater nozzle have similar transition geometry (blend radius).

The P-T Curves for hydrostatic leak test (Curve A) and normal operation – core not critical (Curve B) may be computed by following Steps 1 through 7. Values for Curve C, the core-critical operating curve, are generated from the requirements of 10CFR50 Appendix G [3] and the Curve A and Curve B limits. Table 1 of Reference [3] requires that core critical P-T limits be 40°F above any Curve A or Curve B limits at all pressures. 10CFR50 Appendix G [3] also stipulates that, above the 20% pressure transition point, the Curve C temperatures must be either the reference temperature (RT_{NDT}) of the closure flange region plus 160°F, or the temperature required for the hydrostatic pressure test, whichever is greater.

For P-T Curves A and B, the initial fluid temperature assumed in Step 1 is typically taken at the bolt-up temperature of the closure flange minus coolant temperature instrument uncertainty. According to Reference [3], the minimum bolt-up temperature is equal to the limiting material RT_{NDT} of the regions affected by bolt-up stresses. Consistent with Reference [1], the minimum bolt-up temperature shall not be lower than 60°F. Thus, the minimum bolt-up temperature shall be 60°F or the material RT_{NDT}, whichever is higher.

For P-T Curve C, when the reactor is critical, the initial fluid temperature is equal to the calculated minimum criticality temperature in this region. Table 1 of Reference [3] indicates that, for a BWR with normal operating water levels, the allowable temperature for initial criticality at the closure flange region is equal to the reference temperature (RT_{NDT}) at the flange region plus 60°F.

3.0 ASSUMPTIONS

The 10CFR50 Appendix G [3] and ASME Code [4] requirements and methods are considered to be supported in their respective technical basis documentation. Therefore, the assumptions inherent in the ASME B&PV Code methods utilized for this evaluation are not specifically identified and justified in this calculation. Only those assumptions specific to this calculation are identified and justified here. The following assumptions are used in preparation of the MNGP P-T curves:

1. The full-vessel height is used in the calculation of the static head contributed by the coolant in the RPV.

   This assumption is conservative in that the static head at the non-beltline regions is slightly lower than that of the bottom head curve; however, the difference in static head is small. Therefore, the added complexity in considering different static head values for each region of the vessel is not considered beneficial.

2. The FW nozzle is the bounding non-beltline component of the RPV.

   This assumption is made because:

   a. The geometric discontinuity caused by the nozzle penetration in the RPV shell causes a stress concentration which results in larger pressure induced stresses than would be calculated in the shell regions of the RPV.

b. The FW nozzle experiences more severe thermal transients than most of the other nozzles because of the feedwater injection temperature [5], which causes larger thermal stresses than are experienced in the shell region of the RPV.

c. Although some other nozzles can experience thermal transients, which would cause thermal stresses larger than those calculated for the shell regions of the RPV, and some nozzles are larger diameter than the FW nozzle, which could result in a slightly larger K_{Ip}, the combined stresses from the applied thermal and pressure loads are considered to bound all other non-beltline discontinuities [1, Section 2.5.3].

3. Application of a SCF = 3.0 to the membrane pressure stress in the bottom head bounds the effect of the bottom head penetrations on the stress field in this region of the vessel.

   Bottom head penetrations will create geometric discontinuities in the bottom head hemisphere resulting in high localized stresses. This effect must be considered in calculating the stress intensity factor from internal pressure. Rather than performing a plant-specific analysis, SI applies a conservative SCF for a circular hole in a flat plate subjected to a uniaxial load to the membrane stress in the shell caused by the internal pressure. The assumption of SCF = 3.0 is conservative because:

   a. It applies a peak SCF to the membrane stress which essentially intensifies the stress through the entire shell thickness and along the entire crack face of the postulated flaw rather than intensifying the stress local to the penetration and considering the stress attenuation away from the penetration,

   b. Review of SCFs for circular holes in plates subjected to an equi-bi-axial stress state as well as SCFs for arrays of circular holes in shells, shows that the SCF is likely closer to 2-2.5 rather than 3.0 [5].

   Consequently, the method utilized by SI is expedient, as intended, and conservatively bounds the expected effect of bottom head penetrations because a bounding SCF is used and applied as a membrane stress correction factor.

4.0 DESIGN INPUTS

The design inputs, also included in Appendix A, used to develop the MNGP P-T curves are identified below.

1. Limiting RT_{NDT} and ART:

Non-beltline RT_{NDT}:	40°F [6, Table 3] *(Bounding RT_{NDT} for non-beltline region, excluding bottom head.)*
Closure Flange RT_{NDT}:	10°F [6, Table 3] *(Bounding RT_{NDT} for materials highly stressed by bolt preload)*
Bottom Head RT_{NDT}:	26°F [6, Table 3]

Quarter T Recirculation Inlet (N2) Nozzle ART (72 EFPY): 116.6°F [10, p. 13]

Quarter T Beltline ART (72 EFPY): The limiting 1/4T beltline ART value was calculated to be 182.7°F [10, p. 13] for plates heat C2220 with a chemistry factor (CF) of 180 from Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) [15]. However, the latest ISP data shows that the CF value for plate heat C2220 changes to 174 [16].

The corresponding limiting 1/4T beltline ART value changed to 178.1°F as shown in the updated 1/4T ART calculations in Table 2 below.
(The limiting ART value of all beltline materials including plates and welds, used for P-T limit curve calculations for each EFPY)

Table 2. Updated MNGP 72 EFPY ¼T ART Calculation with Latest ISP CF Value.

Component No.	Heat	Lot	% Cu	% Ni	CF	Initial RT_{NDT} (°F)	72EFPY 1/4T Fluence (n/cm^2)	Fluence Factor f	ΔRT_{NDT} (°F)	σ_i (°F)	σ_Δ (°F)	72 EFPY 1/4T ART (°F)
Lower Shell Plates (Course 1)												
I-16	A0946-1	N/A	0.14	0.56	98	27	2.80E+18	0.653	64.1	0	17.0	125.1
I-17	C2193-1	N/A	0.17	0.5	119	0	2.80E+18	0.653	77.3	0	17.0	111.3
Lower-Intermediate Shell Plates (Course 2)												
I-14	C2220-1	N/A	0.16	0.64	**174** [16]	27	4.38E+18	0.770	**134.1**	0	8.5	**178.1**
I-15	C2220-2	N/A	0.16	0.64	**174** [16]	27	4.38E+18	0.770	**134.1**	0	8.5	**178.1**
Upper/Int Shell Plates (Course 3)												
I-12	C2089-1	N/A	0.35	0.5	200	0	2.38E+17	0.191	38.2	0	17.0	72.2
I-13	C2613-1	N/A	0.35	0.49	198	27	2.38E+17	0.191	37.9	0	17.0	98.9
Lower Shell (Course 1) Axial Welds												
VLAA-1 & VLAA-2	-	E8018N	0.1	0.99	135	-65.6	1.73E+18	0.535	72.2	12.7	28.0	68.1
Lower-Intermediate Shell (Course 2) Axial Welds:												
VLBA-1 & VLBA-2	-	E8018N	0.1	0.99	135	-65.6	1.55E+18	0.510	68.8	12.7	28.0	64.7
Upper/Int Shell (Course 3) Axial Welds:												
VLCB-1 & VLCB-2	-	E8018N	0.1	0.99	135	-65.6	1.56E+17	0.147	19.8	12.7	9.9	-13.5
Circumferential Welds												
VCBA-2	-	E8018N	0.1	0.99	135	-65.6	2.80E+18	0.653	88.0	0	28.0	78.4
VCBB-3	-	E8018N	0.1	0.99	135	-65.6	2.38E+17	0.191	25.8	0	12.9	-14.0
N2 Nozzle												
N2 Nozzle	E21VW	N/A	0.18	0.86	142	40	5.23E+17	0.300	42.6	0	17.0	116.6

Notes:

1. All from values are the same from Reference [10, Table 3] except the bold highlighted values. The CF values of shell plates I-14 and I-15 are based on Reference [16].

2. Minimum Bolt-Up Temperature:

Bolt-Up Temperature: 60°F [13, p. 8]

3. RPV Dimensions:

Full vessel height: 758 inches [7]
(Used to calculate maximum water head during pressure test and conservatively applied for normal operation as well.)

RPV inside radius: 103.1875 inches [7] *(to base metal)*
RPV shell thickness: 5.0625 inches [7] *(to base metal)*
Bottom head inside radius: 103.1875 inches [7] *(to base metal)*
Bottom head shell thickness: 5.9375 inches [7] *(to base metal)*

4. Heat-up / Cool-down Rate: 100°F/hr [9, p. A-7]

5. Quarter T Nozzle Stress Intensity Factors:

FW Nozzle [11, Table 7]:
1000 psi Internal Pressure: 70.59 ksi-in$^{0.5}$
Limiting thermal transient: 10.37 ksi-in$^{0.5}$

Recirculation Inlet (N2) Nozzle [12, Table 13]:
1010 psi Internal Pressure: 75.20 ksi-in$^{0.5}$
Limiting thermal transient: 25.28 ksi-in$^{0.5}$

6. Operating Pressure

Design Pressure: 1250 psig [8]
Normal Operating Pressure 1025 psig [8]

7. Hydro-test pressure:

Pre-Service: 1563 psig *(i.e. 1.25*Design pressure)*
In-Service: 1025 psig *(i.e. 1.0*Normal operating pressure)*

8. Applicable ASME XI Code Year [4]: 2004 Edition [13]

5.0 CALCULATIONS

The P-T curves in this calculation were developed using an Excel spreadsheet listed in Appendix D, which is independently verified for use on a project-specific basis in accordance with SI's Nuclear QA program [17, 18]. P-T limits are evaluated for 72 EFPY. P-T limits are calculated from 0 to 1300 psig. Supporting calculations for all P-T curves are included in Appendix B and represent the P-T curves for individual RPV components. The tabulated results in Table 3 through Table 14 present bounding composite P-T curves for the three RPV regions (beltline, non-beltline, and bottom head). As discussed in Section 5.1 for instance, the beltline curve A in Table 3 bounds three underlying component curves (beltline shell, Feedwater nozzles, and N2 Recirculation Inlet nozzles), shown in Table B-1 through Table B-3.

The bottom head methodology for calculating the allowable pressure shown in Eq. (8), using an SCF of 3.0 to account for bottom head penetration discontinuities, is applied for the thinner side plates of the MNGP bottom head, which bounds the thicker portion of the bottom head center plates with respect to the resulting P-T limits.

5.1 Pressure Test (Curve A)

The minimum bolt-up temperature of 60°F minus instrument uncertainty (0°F) is applied to all regions as the initial temperature in the iterative calculation process. The static fracture toughness (K_{Ic}) is calculated for all regions using Equation (1). The resulting value of K_{Ic}, along with a safety factor of 1.5 is used in Equation (2) to calculate the pressure stress intensity factor (K_{Ip}). The allowable RPV pressure is calculated for the beltline, bottom head and upper vessel regions using Equations (7, 8, and 9), as appropriate. For the feedwater nozzle / upper vessel region, the additional constraints specified in Step 6 of Section 2.0 are applied. Final P-T limits for temperature and pressure are obtained from Equations (11) and (12), respectively.

Since the thermal stress intensity factor is taken as zero for Curve A, the cool-down rate does not affect the results for Curve A.

Values for the beltline region curves for 72 EFPY are listed in Table 3. Data for the bottom head region curve for 72 EFPY is listed in Table 4. Data for the non-beltline (feedwater nozzle / upper vessel) region curve for 72 EFPY is listed in Table 5. The data for each region is plotted, and the resulting composite Curve A for 72 EFPY is provided in Figure 1 and tabulated in Table 6. Additional data and curves for each region are included in Appendix B.

5.2 Normal Operation – Core Not Critical (Curve B)

The minimum bolt-up temperature of 60°F for MNGP minus coolant temperature instrument uncertainty (0°F), is applied to all regions as the initial temperature in the iterative calculation process. The static fracture toughness (K_{Ic}) is calculated for all regions using Eq. (1). The thermal stress intensity factor (K_{It}) is calculated for the FW nozzle and N2 recirculation Inlet nozzle using Eq. (4).

The resulting values of K_{Ic} and K_{It}, along with a safety factor of 2.0, are used in Eq. (2) to calculate the pressure stress intensity factor (K_{Ip}). The allowable RPV pressure is calculated for the beltline, bottom head, and non-beltline regions using Eq. (7, 8, and 9), as appropriate. For the non-beltline (FW nozzle / upper vessel) region, the additional constraints specified in Step 6 of Section 2.0 are applied. Final P-T limits for temperature and pressure are obtained from Eq. (11 and 12), respectively.

The data resulting from each P-T curve calculation is tabulated. Values for the beltline region at 72 EFPY are listed in Table 7. Data for the bottom head region are listed in Table 8. Data for the non-beltline (feedwater nozzle / upper vessel) region are listed in Table 9. The data for each region is plotted, and the resulting data for composite Curve B for 72 EFPY is provided in Figure 2 and tabulated in Table 10. Additional data and curves for each region are included in Appendix B.

5.3 Normal Operation – Core Critical (Curve C)

The pressure and temperature values for Curve C are calculated in a similar manner as Curve B, with several exceptions. The initial evaluation temperature is calculated as the limiting non-beltline RT_{NDT} that is highly stressed by the bolt preload (in this case, that of the closure flange region: 10°F per Section 4.0) plus 60°F, resulting in a minimum criticality temperature of 70°F). When the pressure exceeds 20% of the pre-service system hydrostatic test pressure (20% of 1,563 psig = 313 psig), the P-T limits are specified as 40°F higher than the Curve B values. The minimum temperature above the 20% of the pre-service system hydrostatic test pressure is always greater than the reference temperature (RT_{NDT}) of the closure region plus 160°F or is taken as the minimum temperature required for the hydrostatic pressure test. The final Curve C values are taken as the absolute maximum between the regions of the beltline, the bottom head, and the non-beltline.

The data resulting from each P-T curve calculation is tabulated. Values for the beltline region at 72 EFPY are listed in Table 11. Data for the bottom head region are listed in Table 12. Data for the non-beltline (FW nozzle / upper vessel) region are listed in Table 13. The data for each region is plotted, and the resulting data for composite Curve C for 72 EFPY is provided in Figure 3 and tabulated in Table 14. Additional data and curves for each region are included in Appendix B.

5.4 Overall Composite Curves

Overall composite curves A, B, and C are plotted in Figure 4.

6.0 CONCLUSIONS

P-T curves are developed for MNGP using the methodology, assumptions, and design inputs defined in Sections 2.0, 3.0, and 4.0, respectively. P-T curves are developed for the beltline, bottom head, and non-beltline regions, considering limiting thermal transients at 72 EFPY, for the following plant conditions: Pressure Test (Curve A), Normal Operation – Core Not Critical (Curve B), and Normal Operation – Core Critical (Curve C). Tabulated pressure and temperature values are provided for all regions and EFPY in Table 3 through Table 14. The accompanying P-T curve plots are provided in Figure 1 through Figure 4.

7.0 REFERENCES

1. Licensing Topical Report (LTR) BWROG-TP-11-022-A (SIR-05-044), Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," August 2013, ADAMS Accession No. ML13277A557.
2. Not used.
3. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements," November 29, 2019.
4. ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 2004 Edition.
5. Pilkey, W. D., Pilkey, D. F., "Peterson's Stress Concentration Factors," 3rd Ed., C. 2008, John Wiley & Sons, Inc.
6. Structural Integrity Associates Calculation No. NSP-21Q-303, Revision 1, "Determination of the Initial RT_{NDT} and ART Values for the Monticello RPV Materials."
7. CB&I Drawing No. 1, Revision 8, "General Plan. 17'2" ID x 63' 2" Ins Heads Nuclear Reactor," NX-8290-13, SI File No. NSP-21Q-210.
8. GE Design Specification No. 23A1581, Revision 3, "Reactor Vessel - Recirculation Inlet Safe End," SI File No. 1000720.202.
9. GE Report No. SASR 88-99, Revision 1, "Implementation of Regulatory Guide 1.99, Revision 2 for the Monticello Nuclear Generating Plant," January 1989, SI File No. NSP-21Q-202
10. SI Calculation No. 2100300.302P, Revision 2, "Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts." **CONTAINS PROPRIETARY INFORMATION.**
11. SI Calculation No. 2200284.302P, Revision 0, "Feedwater Nozzle Fracture Mechanics Evaluation for Pressure-Temperature Limit Curve Development." **CONTAINS PROPRIETARY INFORMATION.**
12. SI Calculation No. 2200284.301P, Revision 0, "Finite Element Stress and Fracture Mechanics Analysis of Monticello RPV Recirculation Inlet Nozzle." **CONTAINS PROPRIETARY INFORMATION.**
13. SI Calculation No. 1000847.303, Revision 2, "Revised P-T Curves Calculation,"
14. GE Design Specification No. 22A6996, Revision 0, "Reactor Vessel System Cycling," SI File No. 1000847.201.
15. *BWRVIP-135, Revision 4: BWR Vessel and Internals Project, Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations*. EPRI, Palo Alto, CA: 2014. 3002003144. **EPRI PROPRIETARY INFORMATION**.
16. BWRVIP Letter 2022-053, September 14, 2022, from Bob Carter to Russell Lidberg, Subject: Notification of New BWRVIP Integrated Surveillance Program (ISP) Data Applicable to the Monticello Reactor Pressure Vessel (RPV).
17. SI Nuclear QA Procedure, "Design and Analysis Control," Revision 8.1, SI File No. QP03-01.
18. SI Nuclear QA Procedure, "Calculation Verification Activities," Revision 9, SI File No. QP03-07.

19. “Monticello Nuclear Generating Station Reactor Pressure Vessel Fluence Evaluation - Subsequent License Renewal,” TransWare Enterprises, MNT-FLU-001-R-002, Revision 0. June 2022. SI File No. 2100300.201

Table 3. MNGP Beltline Region, Curve A, for 72 EFPY

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	329.7
90.3	379.7
109.1	429.6
122.7	479.5
133.4	529.5
142.2	579.4
149.6	629.3
156.1	679.2
161.9	729.2
171.4	778.2
179.3	827.3
186.2	876.3
192.2	925.4
197.6	974.4
202.4	1023.5
206.9	1072.5
210.9	1121.6
214.7	1170.6
218.2	1219.7
221.5	1268.7
224.5	1317.8
227.4	1366.8
230.2	1415.9
232.8	1464.9
235.2	1514.0
237.6	1563.0

Table 4. MNGP Bottom Head Region, Curve A, for 72 EFPY

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	812.8
64.8	859.7
69.2	906.6
73.2	953.4
77.0	1000.3
80.5	1047.2
83.7	1094.1
86.8	1141.0
89.6	1187.9
92.3	1234.8
94.9	1281.7
97.4	1328.6
99.7	1375.4
102.0	1422.3
104.1	1469.2
106.1	1516.1
108.1	1563.0

Table 5. MNGP Non-Beltline Region, Curve A, for 72 EFPY

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	312.6
100.0	312.6
100.0	936.3
103.6	984.5
106.9	1032.7
110.0	1080.9
113.0	1129.1
115.8	1177.3
118.4	1225.6
120.9	1273.8
123.3	1322.0
125.6	1370.2
127.7	1418.4
129.8	1466.6
131.8	1514.8
133.7	1563.0

Table 6. MNGP Overall Composite Curve, Curve A, for 72 EFPY

Curve A - Pressure Test	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	312.6
100.0	312.6
100.0	403.1
115.0	449.7
126.5	496.3
135.9	542.8
143.7	589.4
150.6	636.0
156.5	682.6
161.9	729.2
171.4	778.2
179.3	827.3
186.2	876.3
192.2	925.4
197.6	974.4
202.4	1023.5
206.9	1072.5
210.9	1121.6
214.7	1170.6
218.2	1219.7
221.5	1268.7
224.5	1317.8
227.4	1366.8
230.2	1415.9
232.8	1464.9
235.2	1514.0
237.6	1563.0

Table 7. MNGP Beltline Region, Curve B, for 72 EFPY

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
59.6	141.2
99.6	187.6
121.1	234.1
135.8	280.5
147.1	326.9
156.1	373.4
163.8	419.8
170.4	466.2
176.1	512.6
181.3	559.1
189.7	606.9
196.9	654.7
203.2	702.5
208.8	750.3
213.9	798.1
218.4	845.9
222.6	893.7
226.5	941.5
230.1	989.3
233.4	1037.1
236.6	1084.9
239.5	1132.7
242.3	1180.5
245.0	1228.4
247.5	1276.2
249.8	1324.0
252.1	1371.8
254.3	1419.6
256.4	1467.4
258.4	1515.2
260.3	1563.0

Table 8. MNGP Bottom Head Region, Curve B, for 72 EFPY

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	533.1
66.6	582.2
72.4	631.2
77.6	680.3
82.4	729.3
86.7	778.3
90.6	827.4
94.3	876.4
97.8	925.5
101.0	974.5
104.0	1023.6
106.8	1072.6
109.5	1121.6
112.1	1170.7
114.5	1219.7
116.8	1268.8
119.0	1317.8
121.1	1366.8
123.2	1415.9
125.1	1464.9
127.0	1514.0
128.8	1563.0

Table 9. MNGP Non-Beltline Region, Curve B, for 72 EFPY

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	312.6
130.0	312.6
130.0	1022.7
132.7	1071.8
135.3	1120.9
137.8	1170.0
140.2	1219.1
142.4	1268.3
144.5	1317.4
146.6	1366.5
148.6	1415.6
150.5	1464.8
152.3	1513.9
154.1	1563.0

Table 10. MNGP Overall Composite Curve, Curve B, for 72 EFPY

Curve B - Core Not Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
60.0	0.0
60.0	141.2
94.1	181.0
115.1	220.8
129.5	260.6
143.0	310.3
153.5	360.1
162.1	409.8
169.3	459.6
175.6	509.3
181.3	559.1
189.7	606.9
196.9	654.7
203.2	702.5
208.8	750.3
213.9	798.1
218.4	845.9
222.6	893.7
226.5	941.5
230.1	989.3
233.4	1037.1
236.6	1084.9
239.5	1132.7
242.3	1180.5
245.0	1228.4
247.5	1276.2
249.8	1324.0
252.1	1371.8
254.3	1419.6
256.4	1467.4
258.4	1515.2
260.3	1563.0

Table 11. MNGP Beltline Region, Curve C, for 72 EFPY

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	122.2
128.5	170.8
155.2	219.3
172.2	267.8
184.7	316.4
194.6	364.9
202.8	413.4
209.8	462.0
215.9	510.5
221.3	559.1
229.7	606.9
236.9	654.7
243.2	702.5
248.8	750.3
253.9	798.1
258.4	845.9
262.6	893.7
266.5	941.5
270.1	989.3
273.4	1037.1
276.6	1084.9
279.5	1132.7
282.3	1180.5
285.0	1228.4
287.5	1276.2
289.8	1324.0
292.1	1371.8
294.3	1419.6
296.4	1467.4
298.4	1515.2
300.3	1563.0

Table 12. MNGP Bottom Head Region, Curve C, for 72 EFPY

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	376.2
81.5	425.6
90.9	475.1
98.7	524.5
105.5	574.0
111.5	623.4
116.9	672.9
121.7	722.3
126.1	771.8
130.2	821.2
133.9	870.7
137.4	920.1
140.6	969.6
143.7	1019.0
146.6	1068.5
149.3	1117.9
151.9	1167.4
154.3	1216.8
156.7	1266.3
158.9	1315.7
161.1	1365.2
163.1	1414.6
165.1	1464.1
167.0	1513.5
168.8	1563.0

Table 13. MNGP Non-Beltline Region, Curve C, for 72 EFPY

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	276.4
83.7	312.6
203.0	312.6
203.0	1563.0

Table 14. MNGP Overall Composite Curve, Curve C, for 72 EFPY

Curve C - Core Critical	
P-T Curve Temperature	**P-T Curve Pressure**
°F	*psi*
70.0	0.0
70.0	122.2
127.8	169.8
154.4	217.4
171.4	265.0
183.9	312.6
203.0	312.6
203.0	414.7
209.9	462.8
215.9	510.9
221.3	559.1
229.7	606.9
236.9	654.7
243.2	702.5
248.8	750.3
253.9	798.1
258.4	845.9
262.6	893.7
266.5	941.5
270.1	989.3
273.4	1037.1
276.6	1084.9
279.5	1132.7
282.3	1180.5
285.0	1228.4
287.5	1276.2
289.8	1324.0
292.1	1371.8
294.3	1419.6
296.4	1467.4
298.4	1515.2
300.3	1563.0

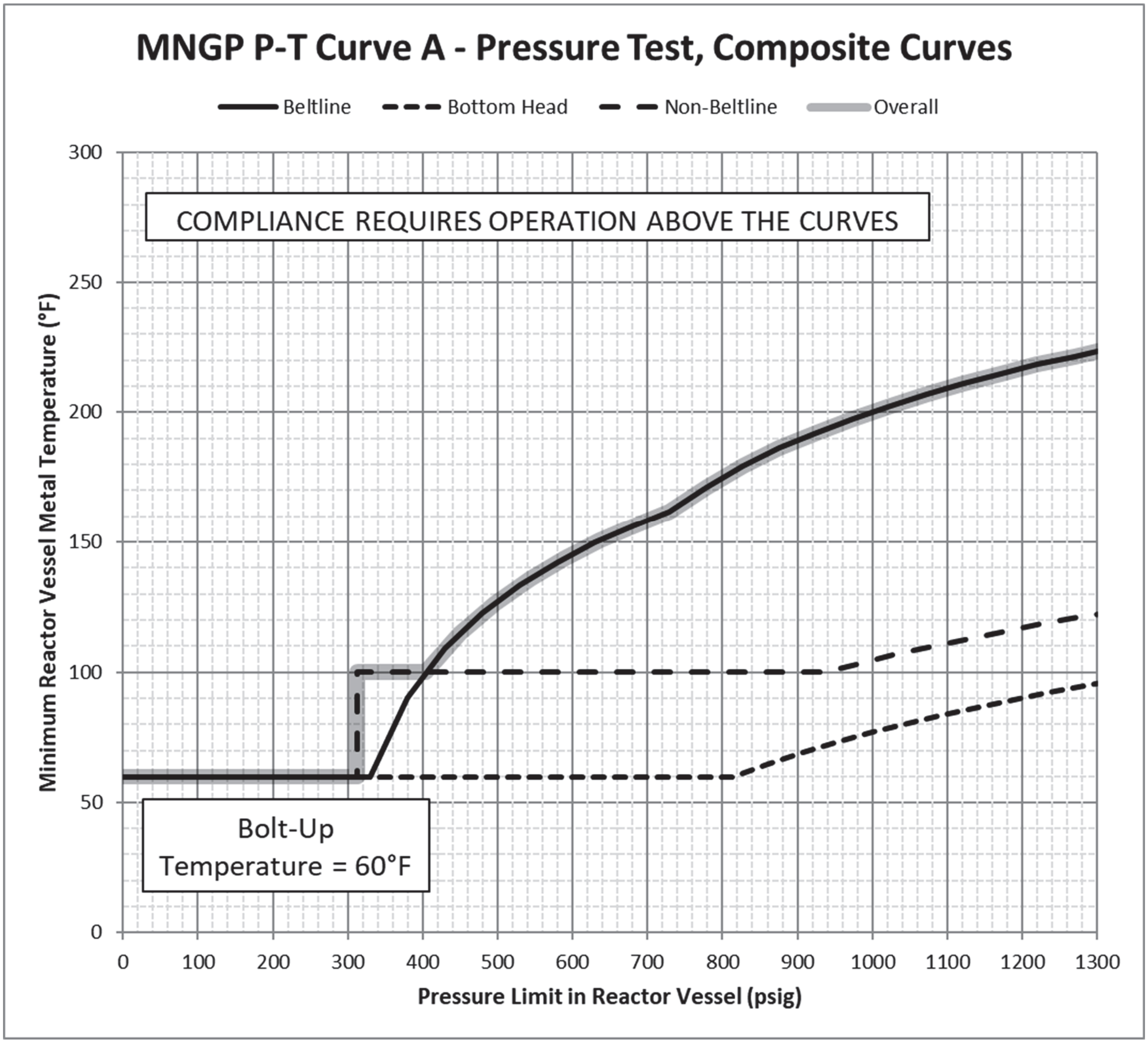


Figure 1. MNGP P-T Curve A (Hydrostatic Pressure and Leak Test), 72 EFPY

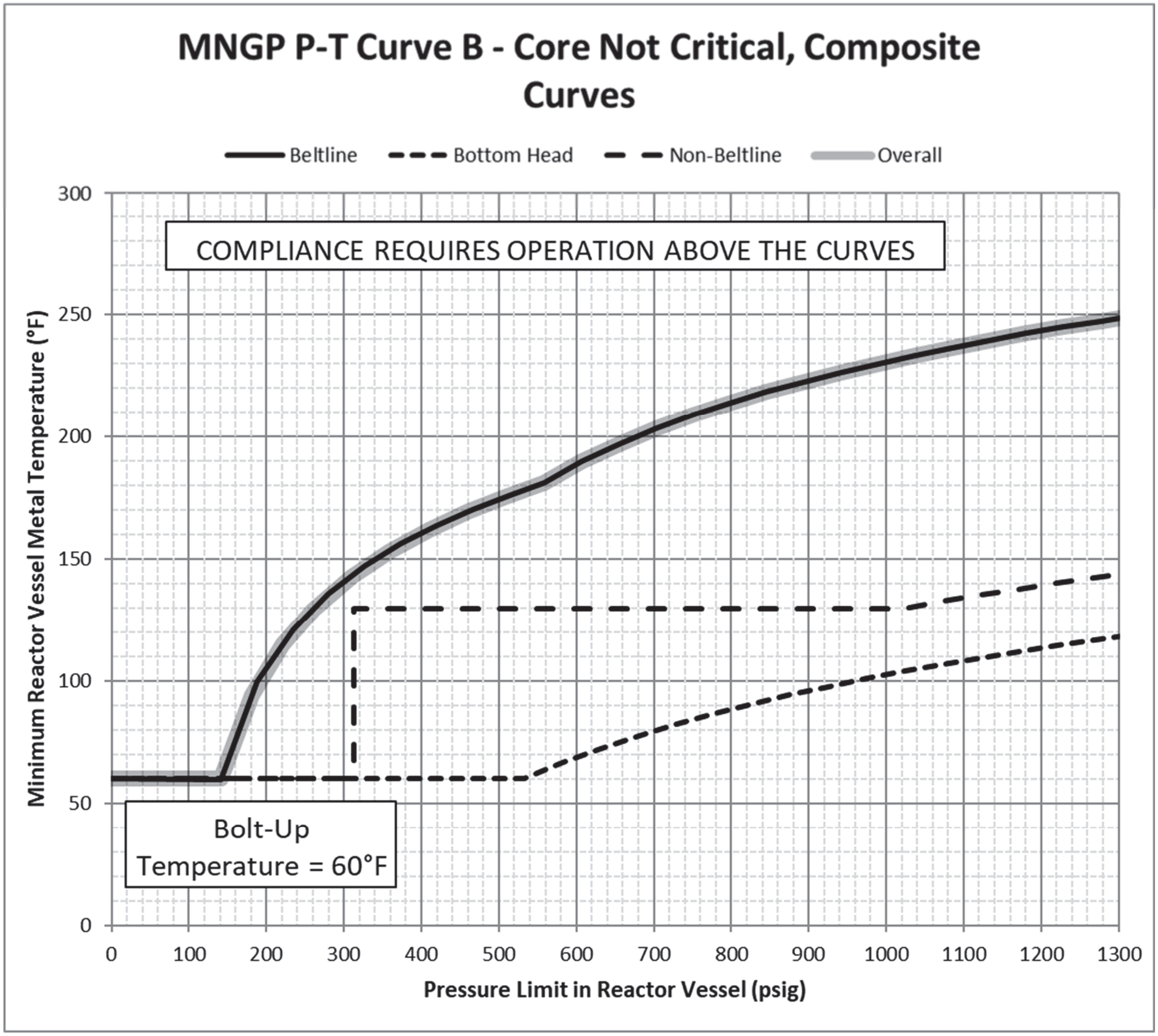


Figure 2. MNGP P-T Curve B (Normal Operation – Core Not Critical), 72 EFPY

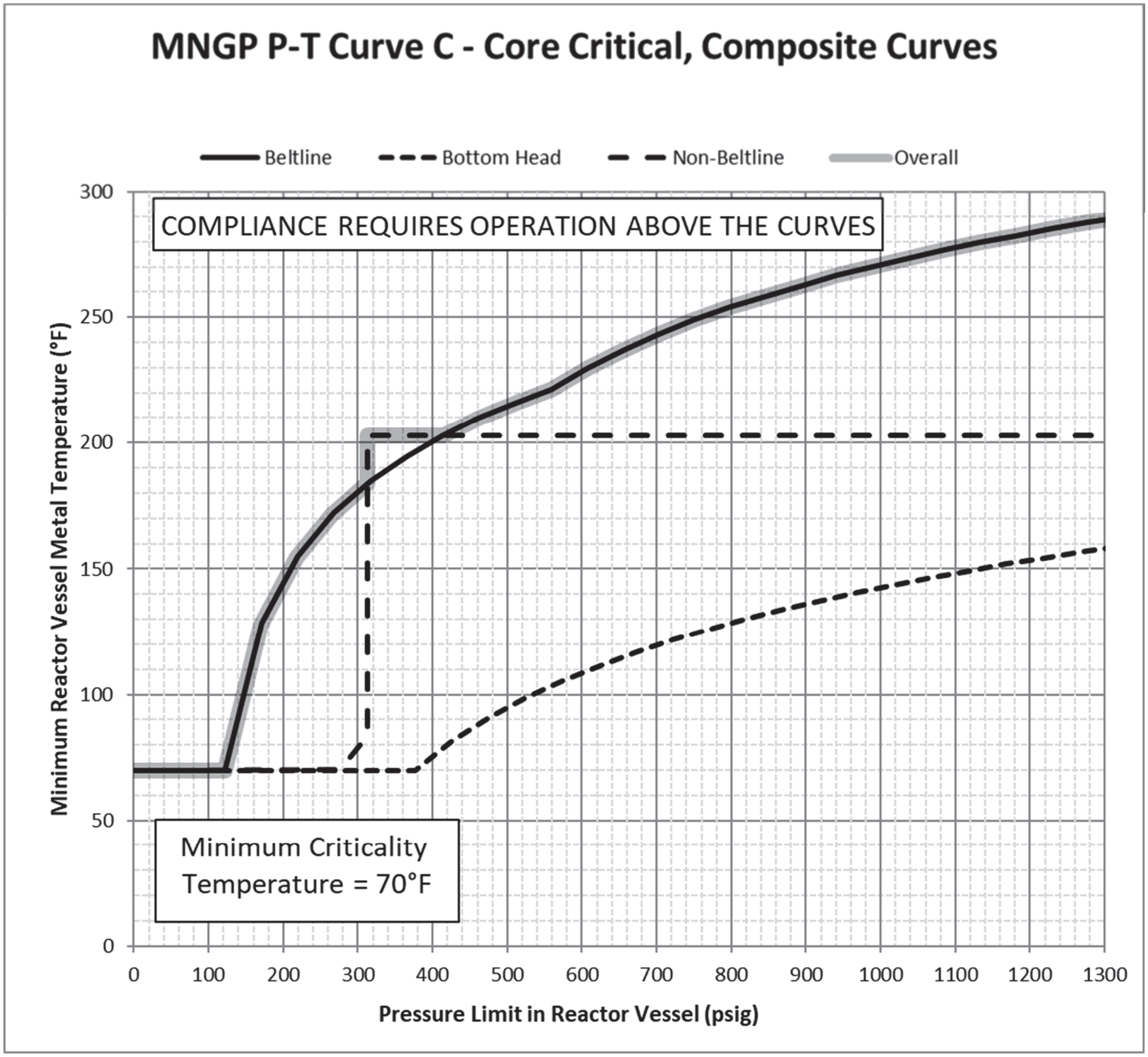


Figure 3. MNGP P-T Curve C (Normal Operation – Core Critical), 72 EFPY

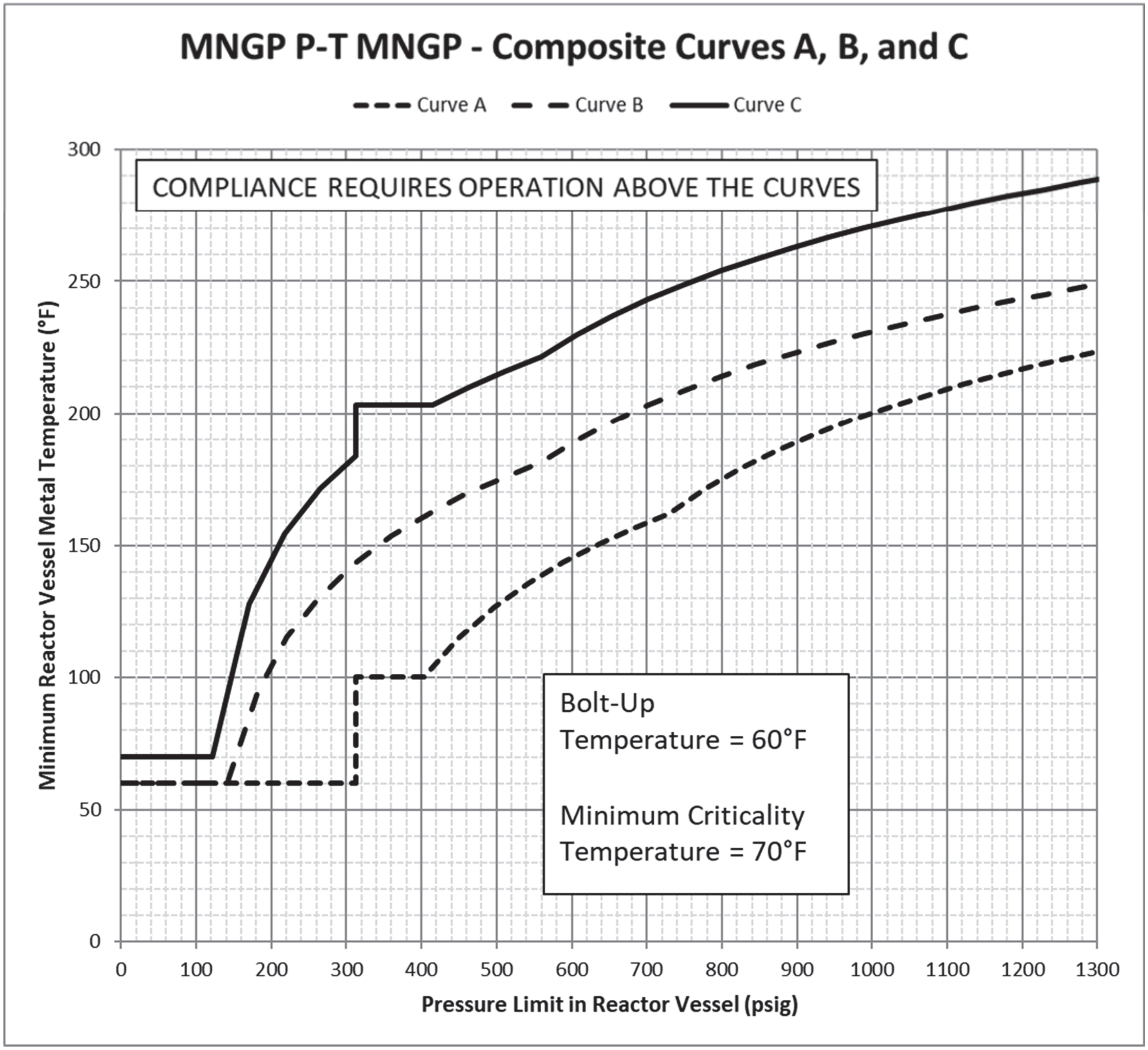


Figure 4. MNGP Overall Composite Curves A, B, and C, 72 EFPY

APPENDIX A

P-T CURVE INPUT LISTING

Table A-1: MNGP Nozzle Stress Intensity Factors

Nozzle	Applied Pressure, $K_{Ip\text{-}app}$	Thermal, K_{It}	Reference
Feedwater	70.59 for 1,000 psi pressure	10.37	[11]
Recirculation Inlet (N2)	75.20 for 1010 psi pressure	25.28	[12]

K_I in units of ksi-in$^{0.5}$

Table A-2: MNGP Unit 1 P-T Curve Input Listing

General Parameters	Values
Unit System for Tables and Plots	English
Temperature Instrument Uncertainty Adjustment (°F)	0
Pressure Instrument Uncertainty Adjustment (psig)	0
Water Density (lbm/ft3)	62.4
Full-Vessel Water Height (in)	758
Safety Factor for Curve A	1.5
Safety Factor for Curves B and C	2
Bolt-up Temperature (°F)	60
ART of Closure Flange Region (°F)	10
Default Temperature Increment for Tables (°F)	10
Default Pressure Increment for Composite Tables (psig)	50
Starting Pressure for Curves (psig)	0
Atmospheric Pressure Adjustment (psi)	14.7
Preservice hydrotest pressure (psi)	1563
In-service hydrotest pressure (psi)	1025
Minimum in-service hydrotest temperature (°F)	203

Beltline Parameters	Values
Adjusted Reference Temperature (°F)	178.1
Vessel Radius (in)	103.1875
Vessel Thickness (in)	5.0625
Heat-up / Cool-down Rate (°F/hr)	100

Additional Beltline Nozzle Parameters (N2 Recirculation Inlet)	Values
Adjusted Reference Temperature (°F)	116.6
Applied Pressure Stress Intensity Factor (ksi*in^0.5)	75.2
Applied Thermal Stress Intensity Factor (ksi*in^0.5)	25.28
Scale KIT based on Saturation Temperature?	Yes
Minimum Transient Temperature (°F)	100
Maximum Transient Temperature (°F)	549
Reference Pressure for Thermal Transient (psig)	1010

Bottom Head Parameters	**Values**
Adjusted Reference Temperature (°F)	26
Vessel Radius (in)	103.1875
Vessel Thickness (in)	5.9375
Heat-up / Cool-down Rate (°F/hr)	100
Stress Concentration Factor	3

Upper Vessel (Feedwater Nozzle) Parameters	**Values**
Adjusted Reference Temperature (°F)	40
Applied Pressure Stress Intensity Factor (ksi*in^0.5)	70.59
Applied Thermal Stress Intensity Factor (ksi*in^0.5)	10.37
Minimum Thermal Stress Intensity Factor (ksi*in^0.5)	
Scale K_{IT} based on Saturation Temperature?	Yes
Minimum Transient Temperature (°F)	100
Maximum Transient Temperature (°F)	548
Reference Pressure for Thermal Transient (psig)	1000

APPENDIX B

SUPPORTING CALCULATIONS

Table B-1: MNGP Beltline Region, Curve A Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
60.0	35.2	23.4	60.0	0.0
60.0	35.2	23.4	60.0	524.5
70.0	35.6	23.7	70.0	531.3
80.0	36.1	24.1	80.0	539.6
90.0	36.8	24.5	90.0	549.7
100.0	37.5	25.0	100.0	562.1
110.0	38.5	25.7	110.0	577.2
120.0	39.7	26.5	120.0	595.6
130.0	41.1	27.4	130.0	618.2
140.0	42.9	28.6	140.0	645.7
150.0	45.0	30.0	150.0	679.4
160.0	47.6	31.8	160.0	720.4
170.0	50.8	33.9	170.0	770.6
180.0	54.7	36.5	180.0	831.9
190.0	59.5	39.7	190.0	906.8
200.0	65.3	43.6	200.0	998.2
210.0	72.4	48.3	210.0	1109.9
220.0	81.1	54.1	220.0	1246.3
230.0	91.7	61.2	230.0	1412.9
240.0	104.7	69.8	240.0	1616.3

Table B-2: MNGP Recirculation Inlet Nozzle, Beltline Region, Curve A Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
60.0	39.9	26.6	60.0	0.0
60.0	39.9	26.6	60.0	329.7
70.0	41.4	27.6	70.0	343.0
80.0	43.2	28.8	80.0	359.2
90.0	45.4	30.3	90.0	379.0
100.0	48.1	32.1	100.0	403.1
110.0	51.4	34.2	110.0	432.6
120.0	55.4	36.9	120.0	468.6
130.0	60.3	40.2	130.0	512.6
140.0	66.3	44.2	140.0	566.3
150.0	73.6	49.1	150.0	632.0
160.0	82.6	55.1	160.0	712.1
170.0	93.5	62.4	170.0	810.1
180.0	106.9	71.3	180.0	929.6
190.0	123.2	82.1	190.0	1075.7
200.0	143.1	95.4	200.0	1254.1
210.0	167.5	111.6	210.0	1472.0
220.0	197.2	131.5	220.0	1738.2

Table B-3: MNGP Bottom Head Region, Curve A Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	*ksi*in$^{1/2}$*	*ksi*in$^{1/2}$*	*°F*	*psig*
60.0	74.1	49.4	60.0	0.0
60.0	74.1	49.4	60.0	812.8
70.0	83.2	55.5	70.0	915.5
80.0	94.3	62.8	80.0	1040.9
90.0	107.8	71.8	90.0	1194.1
100.0	124.3	82.9	100.0	1381.2
110.0	144.4	96.3	110.0	1609.8

Table B-4: MNGP FW Nozzle / Non-Beltline, Curve A Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	*ksi*in$^{1/2}$*	*ksi*in$^{1/2}$*	*°F*	*psig*
60.0	64.1	42.8	60.0	0.0
60.0	64.1	42.8	60.0	578.3
70.0	71.0	47.3	70.0	643.0
80.0	79.3	52.9	80.0	722.0
90.0	89.6	59.7	90.0	818.5
100.0	102.0	68.0	100.0	936.3
110.0	117.3	78.2	110.0	1080.2
120.0	135.9	90.6	120.0	1256.1
130.0	158.6	105.8	130.0	1470.8
140.0	186.4	124.3	140.0	1733.1

Table B-5: MNGP Beltline Region, Curve B Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
60.0	35.2	14.8	60.0	0.0
60.0	35.2	14.8	60.0	321.8
70.0	35.6	15.0	70.0	326.9
80.0	36.1	15.3	80.0	333.1
90.0	36.8	15.6	90.0	340.7
100.0	37.5	16.0	100.0	350.0
110.0	38.5	16.5	110.0	361.3
120.0	39.7	17.1	120.0	375.2
130.0	41.1	17.8	130.0	392.1
140.0	42.9	18.7	140.0	412.8
150.0	45.0	19.8	150.0	438.0
160.0	47.6	21.1	160.0	468.8
170.0	50.8	22.7	170.0	506.4
180.0	54.7	24.6	180.0	552.4
190.0	59.5	27.0	190.0	608.5
200.0	65.3	29.9	200.0	677.1
210.0	72.4	33.5	210.0	760.9
220.0	81.1	37.8	220.0	863.1
230.0	91.7	43.1	230.0	988.1
240.0	104.7	49.6	240.0	1140.7
250.0	120.5	57.5	250.0	1327.1
260.0	139.9	67.2	260.0	1554.8
270.0	163.5	79.0	270.0	1832.8

Table B-6: MNGP Recirculation Inlet Nozzle, Beltline Region, Curve B Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
60.0	39.9	22.8	60.0	0.0
60.0	39.9	12.6	60.0	141.2
70.0	41.4	13.2	70.0	149.8
80.0	43.2	14.0	80.0	160.2
90.0	45.4	14.9	90.0	172.9
100.0	48.1	16.1	100.0	188.5
110.0	51.4	17.5	110.0	207.7
120.0	55.4	19.3	120.0	231.3
130.0	60.3	21.4	130.0	260.5
140.0	66.3	24.1	140.0	296.5
150.0	73.6	27.4	150.0	341.0
160.0	82.6	31.5	160.0	395.9
170.0	93.5	36.5	170.0	463.5
180.0	106.9	42.8	180.0	546.9
190.0	123.2	50.4	190.0	649.6
200.0	143.1	59.8	200.0	776.1
210.0	167.5	71.4	210.0	931.6
220.0	197.2	85.6	220.0	1122.8
230.0	233.5	103.1	230.0	1357.7
240.0	277.8	124.6	240.0	1646.1

Table B-7: MNGP Bottom Head Region, Curve B Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	$ksi*in^{1/2}$	$ksi*in^{1/2}$	°F	psig
60.0	74.1	33.0	60.0	0.0
60.0	74.1	33.0	60.0	533.1
70.0	83.2	37.5	70.0	610.2
80.0	94.3	43.0	80.0	704.2
90.0	107.8	49.8	90.0	819.2
100.0	124.3	58.0	100.0	959.5
110.0	144.4	68.1	110.0	1130.9
120.0	169.1	80.4	120.0	1340.3
130.0	199.2	95.5	130.0	1596.0

Table B-8: MNGP FW Nozzle / Non-Beltline, Curve B Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	*ksi*in$^{1/2}$*	*ksi*in$^{1/2}$*	*°F*	*psig*
60.0	64.1	32.1	60.0	0.0
60.0	64.1	28.1	60.0	370.9
70.0	71.0	31.4	70.0	417.6
80.0	79.3	35.4	80.0	474.8
90.0	89.6	40.4	90.0	544.8
100.0	102.0	46.5	100.0	630.7
110.0	117.3	53.9	110.0	735.9
120.0	135.9	63.0	120.0	864.8
130.0	158.6	74.1	130.0	1022.7
140.0	186.4	87.8	140.0	1215.9
150.0	220.3	104.5	150.0	1452.5
160.0	261.8	124.9	160.0	1742.0

Table B-9: MNGP Beltline Region, Curve C Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
30.0	34.3	14.4	70.0	0.0
30.0	34.3	14.4	70.0	311.4
40.0	34.5	14.5	80.0	314.2
50.0	34.8	14.7	90.0	317.6
60.0	35.2	14.8	100.0	321.8
70.0	35.6	15.0	110.0	326.9
80.0	36.1	15.3	120.0	333.1
90.0	36.8	15.6	130.0	340.7
100.0	37.5	16.0	140.0	350.0
110.0	38.5	16.5	150.0	361.3
120.0	39.7	17.1	160.0	375.2
130.0	41.1	17.8	170.0	392.1
140.0	42.9	18.7	180.0	412.8
150.0	45.0	19.8	190.0	438.0
160.0	47.6	21.1	200.0	468.8
170.0	50.8	22.7	210.0	506.4
180.0	54.7	24.6	220.0	552.4
190.0	59.5	27.0	230.0	608.5
200.0	65.3	29.9	240.0	677.1
210.0	72.4	33.5	250.0	760.9
220.0	81.1	37.8	260.0	863.1
230.0	91.7	43.1	270.0	988.1
240.0	104.7	49.6	280.0	1140.7
250.0	120.5	57.5	290.0	1327.1
260.0	139.9	67.2	300.0	1554.8
270.0	163.5	79.0	310.0	1832.8

Table B-10: MNGP Recirculation Inlet Nozzle, Beltline Region, Curve C Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
30.0	36.9	21.2	70.0	0.0
30.0	36.9	11.1	70.0	122.2
40.0	37.7	11.6	80.0	127.8
50.0	38.7	12.0	90.0	134.0
60.0	39.9	12.6	100.0	141.2
70.0	41.4	13.2	110.0	149.8
80.0	43.2	14.0	120.0	160.2
90.0	45.4	14.9	130.0	172.9
100.0	48.1	16.1	140.0	188.5
110.0	51.4	17.5	150.0	207.7
120.0	55.4	19.3	160.0	231.3
130.0	60.3	21.4	170.0	260.5
140.0	66.3	24.1	180.0	296.5
150.0	73.6	27.4	190.0	341.0
160.0	82.6	31.5	200.0	395.9
170.0	93.5	36.5	210.0	463.5
180.0	106.9	42.8	220.0	546.9
190.0	123.2	50.4	230.0	649.6
200.0	143.1	59.8	240.0	776.1
210.0	167.5	71.4	250.0	931.6
220.0	197.2	85.6	260.0	1122.8
230.0	233.5	103.1	270.0	1357.7
240.0	277.8	124.6	280.0	1646.1

Table B-11: MNGP Bottom Head Region, Curve C Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	ksi*in$^{1/2}$	ksi*in$^{1/2}$	°F	psig
30.0	55.7	23.7	70.0	0.0
30.0	55.7	23.7	70.0	376.2
40.0	60.6	26.2	80.0	418.5
50.0	66.7	29.3	90.0	470.1
60.0	74.1	33.0	100.0	533.1
70.0	83.2	37.5	110.0	610.2
80.0	94.3	43.0	120.0	704.2
90.0	107.8	49.8	130.0	819.2
100.0	124.3	58.0	140.0	959.5
110.0	144.4	68.1	150.0	1130.9
120.0	169.1	80.4	160.0	1340.3
130.0	199.2	95.5	170.0	1596.0

Table B-12: MNGP FW Nozzle / Non-Beltline, Curve C Calculations, 72 EFPY

Gage Fluid Temperature	K_{Ic}	K_{Ip}	P-T Curve Temperature	P-T Curve Pressure
°F	*ksi*in$^{1/2}$*	*ksi*in$^{1/2}$*	*°F*	*psig*
30.0	50.2	25.1	70.0	0.0
30.0	50.2	21.4	70.0	276.4
40.0	53.9	23.2	80.0	301.8
50.0	58.5	25.4	90.0	332.8
60.0	64.1	28.1	100.0	370.9
70.0	71.0	31.4	110.0	417.6
80.0	79.3	35.4	120.0	474.8
90.0	89.6	40.4	130.0	544.8
100.0	102.0	46.5	140.0	630.7
110.0	117.3	53.9	150.0	735.9
120.0	135.9	63.0	160.0	864.8
130.0	158.6	74.1	170.0	1022.7
140.0	186.4	87.8	180.0	1215.9
150.0	220.3	104.5	190.0	1452.5
160.0	261.8	124.9	200.0	1742.0

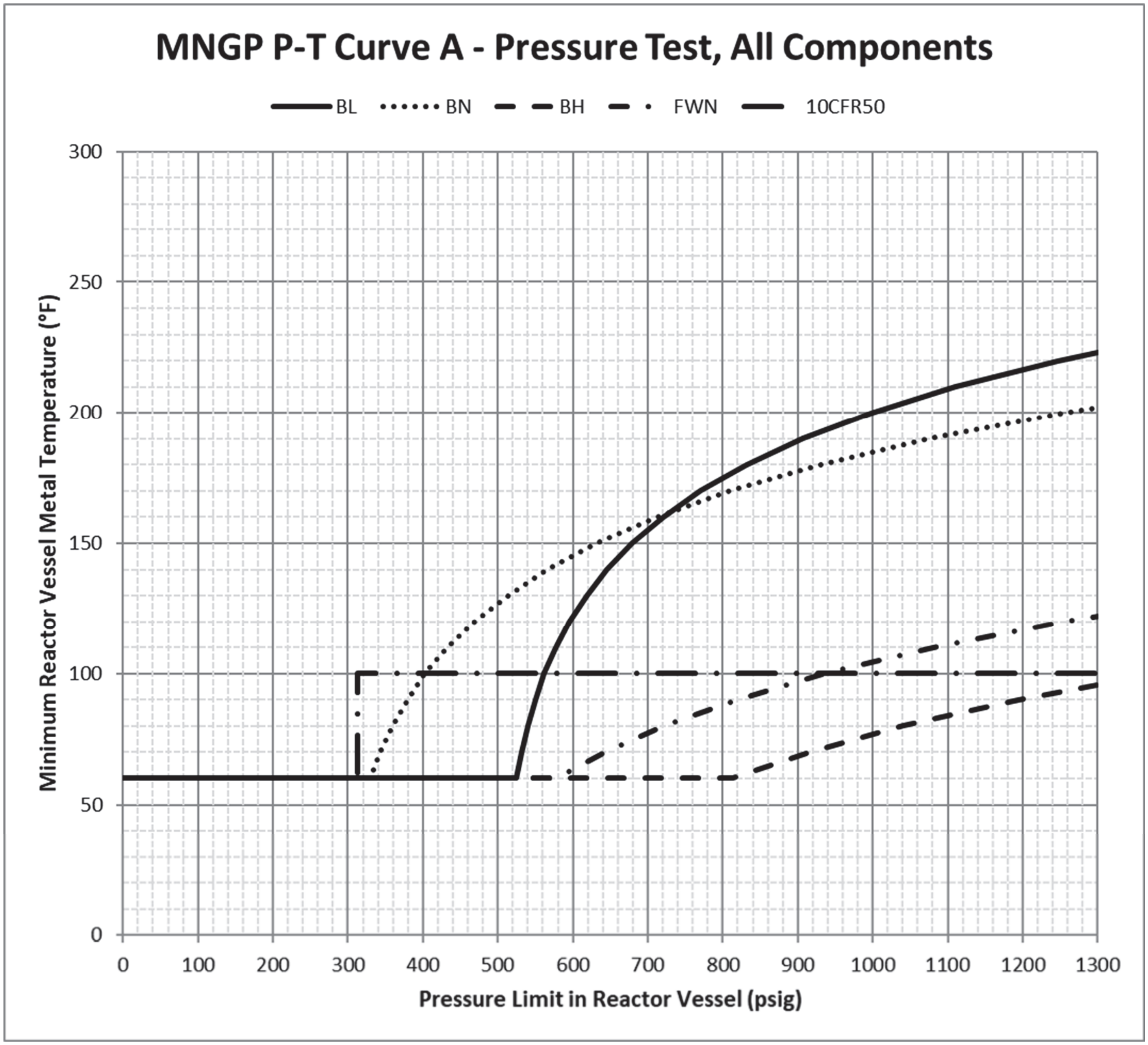


Figure B-1: MNGP P-T Curve A (Hydrostatic Pressure and Leak Test), 72 EFPY

Note: BL is Beltline, BH is Bottom Head, FWN is Feedwater Nozzle, BN is Recirculation Inlet Nozzle

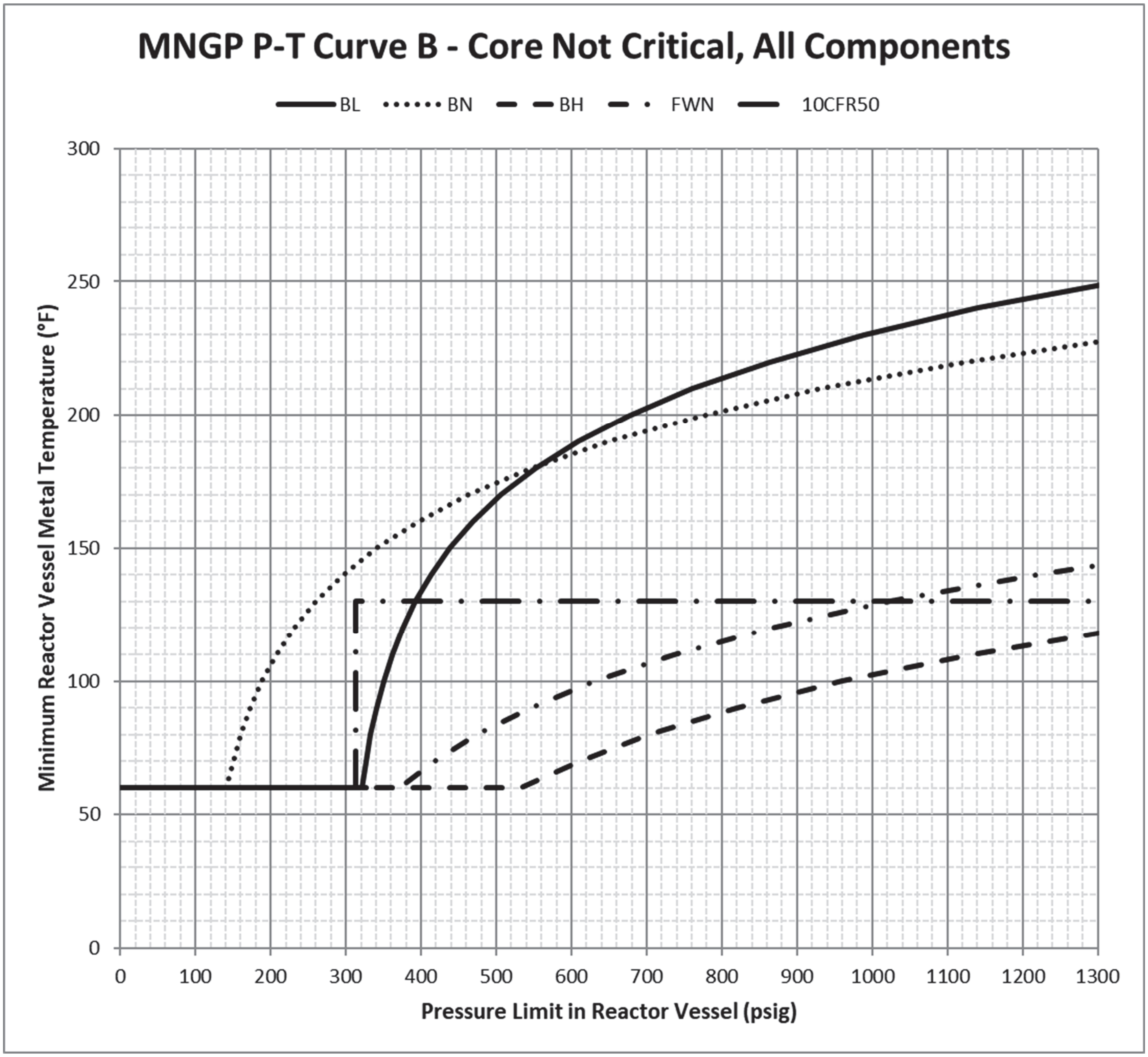


Figure B-2: MNGP P-T Curve B (Normal Operation - Core Not Critical), 72 EFPY

Note: BL is Beltline, BH is Bottom Head, FWN is Feedwater Nozzle, BN is Recirculation Inlet Nozzle

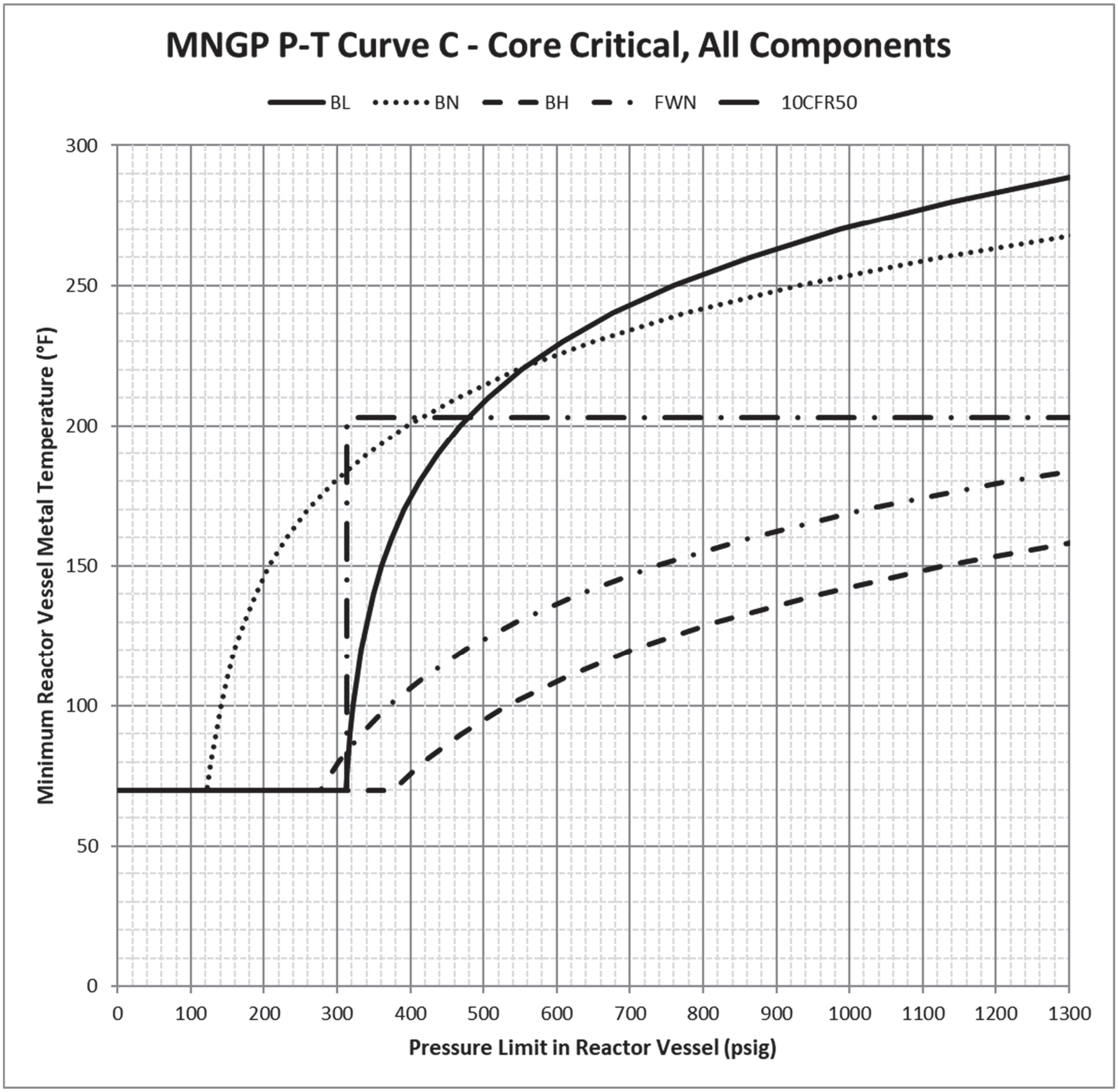


Figure B-3: MNGP P-T Curve C (Normal Operation – Core Critical), 72 EFPY
Note: BL is Beltline, BH is Bottom Head, FWN is Feedwater Nozzle, BN(N2) is Recirculation Inlet Nozzle

APPENDIX C

DEVELOPMENT OF SATURATION STEAM CURVE FITS

Curve Fit for Saturated Steam

Reference: Steam Table data obtained from "Steam Tables, Properties of Saturated and Superheated Steam," CE Power Systems, 7th Printing.

Curve Fit: $T_{sat} = 119.3*(0.7987)^{(1/Psat)} * P_{sat}^{0.2198}$

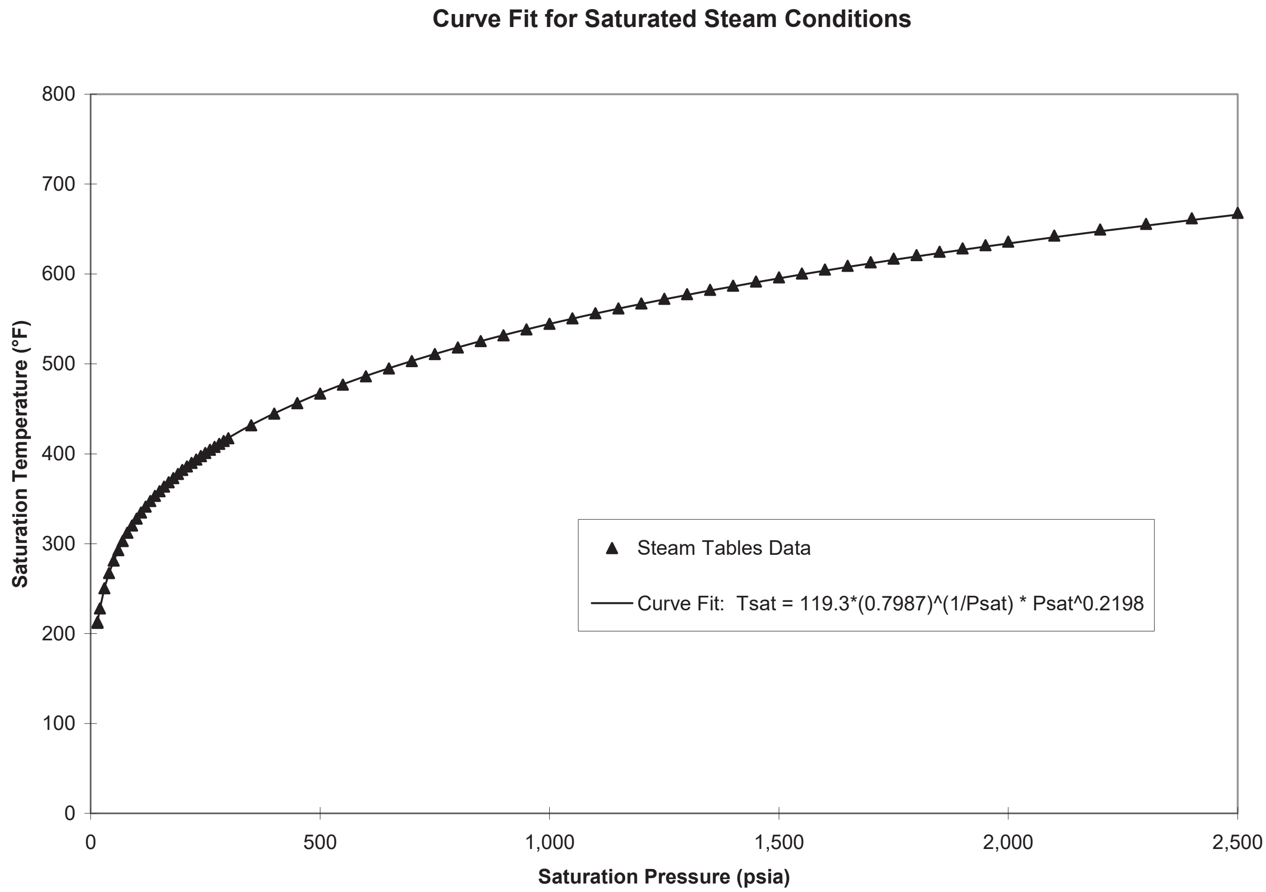
Pressure P_{sat} (psia)	Temperature T_{sat} (°F)	Curve Fit T_{sat} (°F)	Difference (°F)	Error (%)
14.696	212.00	212.10	0.10	0.05%
15	213.03	213.13	0.10	0.05%
20	227.96	227.89	-0.07	-0.03%
30	250.34	250.07	-0.27	-0.11%
40	267.25	266.89	-0.36	-0.13%
50	281.02	280.62	-0.40	-0.14%
60	292.71	292.32	-0.39	-0.13%
70	302.93	302.55	-0.38	-0.13%
80	312.04	311.69	-0.35	-0.11%
90	320.28	319.96	-0.32	-0.10%
100	327.82	327.54	-0.28	-0.09%
110	334.79	334.54	-0.25	-0.07%
120	341.27	341.06	-0.21	-0.06%
130	347.33	347.16	-0.17	-0.05%
140	353.04	352.91	-0.13	-0.04%
150	358.43	358.34	-0.09	-0.03%
160	363.55	363.49	-0.06	-0.02%
170	368.42	368.40	-0.02	-0.01%
180	373.08	373.08	0.00	0.00%
190	377.53	377.57	0.04	0.01%
200	381.80	381.87	0.07	0.02%
210	385.91	386.01	0.10	0.03%
220	389.88	390.00	0.12	0.03%
230	393.70	393.84	0.14	0.04%
240	397.39	397.56	0.17	0.04%
250	400.97	401.16	0.19	0.05%
260	404.44	404.65	0.21	0.05%
270	407.80	408.03	0.23	0.06%
280	411.07	411.32	0.25	0.06%
290	414.25	414.51	0.26	0.06%
300	417.35	417.62	0.27	0.07%
350	431.73	432.06	0.33	0.08%
400	444.60	444.97	0.37	0.08%
450	456.28	456.67	0.39	0.08%
500	467.01	467.39	0.38	0.08%
550	476.94	477.30	0.36	0.08%
600	486.20	486.54	0.34	0.07%
650	494.89	495.19	0.30	0.06%
700	503.08	503.33	0.25	0.05%
750	510.84	511.03	0.19	0.04%
800	518.21	518.34	0.13	0.03%
850	525.24	525.31	0.07	0.01%
900	531.95	531.95	0.00	0.00%
950	538.39	538.32	-0.07	-0.01%
1000	544.58	544.43	-0.15	-0.03%

Curve Fit for Saturated Steam

Reference: Steam Table data obtained from "Steam Tables, Properties of Saturated and Superheated Steam," CE Power Systems, 7th Printing.

Curve Fit: $T_{sat} = 119.3*(0.7987)^{(1/Psat)} * P_{sat}^{0.2198}$

Pressure P_{sat} (psia)	Temperature T_{sat} (°F)	Curve Fit T_{sat} (°F)	Difference (°F)	Error (%)
1050	550.53	550.31	-0.22	-0.04%
1100	556.28	555.97	-0.31	-0.06%
1150	561.82	561.43	-0.39	-0.07%
1200	567.19	566.71	-0.48	-0.08%
1250	572.38	571.83	-0.55	-0.10%
1300	577.42	576.78	-0.64	-0.11%
1350	582.32	581.59	-0.73	-0.13%
1400	587.07	586.26	-0.81	-0.14%
1450	591.70	590.80	-0.90	-0.15%
1500	596.20	595.22	-0.98	-0.16%
1550	600.59	599.53	-1.06	-0.18%
1600	604.87	603.73	-1.14	-0.19%
1650	609.05	607.83	-1.22	-0.20%
1700	613.13	611.84	-1.29	-0.21%
1750	617.12	615.75	-1.37	-0.22%
1800	621.02	619.58	-1.44	-0.23%
1850	624.83	623.32	-1.51	-0.24%
1900	628.56	626.99	-1.57	-0.25%
1950	632.22	630.58	-1.64	-0.26%
2000	635.80	634.10	-1.70	-0.27%
2100	642.76	640.94	-1.82	-0.28%
2200	649.45	647.53	-1.92	-0.30%
2300	655.89	653.89	-2.00	-0.30%
2400	662.11	660.04	-2.07	-0.31%
2500	668.11	665.99	-2.12	-0.32%
		Maximum =	0.39	0.08%
		Minimum =	-2.12	-0.32%
		Average =	-0.41	-0.07%
		Std. Deviation	0.71	0.12%


Curve Fit for Saturated Steam Conditions
800
700
600
500
400
300
200
100
0
Saturation Temperature (°F)
0
500
1,000
1,500
2,000
2,500
Saturation Pressure (psia)
Steam Tables Data
Curve Fit: Tsat = 119.3*(0.7987)^(1/Psat) * Psat^0.2198

APPENDIX D

SUPPORTING FILES

Supporting Files	Comment
1. 2200284.303P R0.xlsx (File name remains, although information is not proprietary)	Excel file contains the detailed P-T curve calculations for MNGP