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Constellation Energy Group

Nine Mile Point
Nuclear Station

August 15, 2003
NMP2L 2096

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Nine Mile Point Unit 2
Docket No. 50-410

License Amendment Request Pursuant to 10 CFR 50.90: Revision of Reactor Pressure Vessel Pressure-Temperature Limits

Gentlemen:

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC, (NMPNS) hereby requests an amendment to Nine Mile Point Unit 2 (NMP2) Operating License NPF-69. The proposed changes to the Technical Specifications (TSs) contained herein would revise the Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves specified in Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits." Specifically, the proposed changes replace the current P-T limit curves contained in TS Figures 3.4.11-1 through 3.4.11-5 with new (recalculated) P-T curves. The TS Bases (B 3.4.11) will be revised to reflect the proposed changes to the TSs.

The new P-T limit curves are based, in part, on an alternative methodology and will be valid for 22 Effective Full Power Years (EFPY). The current P-T limit curves are valid for 12.8 EFPY. The estimated maximum EFPY at the end of the current operating cycle (Cycle 9), assuming a 100% capacity factor, is 12.16 EFPY. Therefore, the current P-T limit curves could expire as soon as the Fall of 2004 (mid-cycle during Cycle 10).

The alternative methodology uses the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1," in calculating the new RCS P-T limits. This alternative methodology has been endorsed by the ASME. Note that the use of this alternative methodology no longer requires an exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The NRC has recently approved generic (unconditional) application of ASME Code Case N-640 in ASME Section XI inservice inspection programs, effective August 7, 2003 [68 FR 40469, July 8, 2003], by including it in Revision 13 of Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1," and by its incorporation by reference into 10 CFR 50.55a(b), "Codes and standards."

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In addition, prior to its generic approval, the NRC has previously granted exemptions allowing use of the ASME N-640 Code Case and approved the associated TS changes for a number of other Boiling Water Reactor (BWR) plants, including: Pilgrim (ADAMS Accession Numbers ML010720448 and ML010790519), Brunswick Units 1 and 2 (ADAMS Accession Numbers ML012760157 and ML012780286), and Susquehanna Units 1 and 2 (ADAMS Accession Numbers ML013520568 and ML013520605).

The procedures and methodology that were previously used to calculate the RCS P-T limit curves were revised to recalculate the curves based, in part, on the ASME N-640 Code Case. During the recalculation of the P-T limit curves, an update to the neutron fluence calculations was also implemented. Although the new curves are for an increase of 9.2 EFPY, the limits have not changed significantly. This is because the old limits were based on preoperational fluence estimates which are approximately a factor of 2 larger than the actual fluence. The neutron fluence values for the Reactor Pressure Vessel (RPV) have been recalculated using methods that comply with Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," and these updated fluence values have been used in the P-T limit calculation. Therefore, the new P-T limit curves were developed using the ASME N-640 Code Case in conjunction with the updated neutron fluence values.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that the changes involve no significant hazards considerations.

NMPNS requests approval of this application and issuance of the TS amendment by February 15, 2004 with 60 days allowed for implementation. The amendment is needed for the Spring 2004 refueling outage (RFO9) to support plant heatup and cooldown and system leakage testing during the outage, as well as, provide valid P-T limit curves for Cycle 10. The proposed P-T limit curves will enhance overall plant safety by widening the P-T operating window, especially in the region of low temperature operations. Safety benefits that would be realized during system leakage and hydrostatic pressure testing at the lower test temperatures include a reduction in the challenges to operators in maintaining a high temperature in a limited operating band, personnel safety while conducting inspections in primary containment at elevated temperatures, and increased availability of plant systems, including the Residual Heat Removal System, due to a reduction of the heatup and test time.

This letter contains no new commitments as reflected in Section 5.3 of Attachment 1.

Pursuant to 10CFR50.91(b)(1), NMPNS has provided a copy of this license amendment request and the associated analyses regarding no significant hazards considerations to the appropriate state representative.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 15, 2003.

Sincerely,

Peter E. Katz
Vice President Nine Mile Point

PEK/CDM/bjh

Attachments:
1. Evaluation of Proposed Technical Specification Changes
2. Proposed Technical Specification Changes (Mark-up)
3. Changes to Technical Specification Bases Pages (Mark-up For Information Only)
4. Report No. MPM-502840

cc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR (2 copies)
Mr. John P. Spath, NYSERDA

ATTACHMENT 1

EVALUATION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

Subject: License Amendment Request Pursuant to 10 CFR 50.90: Revision of Reactor Pressure Vessel Pressure-Temperature Limits

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-69 for Nine Mile Point Unit 2 (NMP2).

The proposed changes would amend the Operating License to revise the Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves specified in Technical Specification (TS) Section 3.4.11, "RCS Pressure and Temperature (P/T) Limits." Specifically, the proposed changes replace the current P-T limit curves contained in TS Figures 3.4.11-1 through 3.4.11-5 with new (recalculated) P-T curves. The TS Bases (B 3.4.11) will be revised to reflect the proposed changes to the TSs.

The proposed changes to the TSs and associated changes to the TS Bases are indicated in the mark-up pages provided in Attachments 2 and 3, respectively. The TS Bases changes are provided for information only and will be controlled by the TSs Bases control program.

The new P-T limit curves are based, in part, on an alternative methodology and will be valid for 22 Effective Full Power Years (EFPY). The current P-T limit curves are valid for 12.8 EFPY. The estimated maximum EFPY at the end of the current operating cycle (Cycle 9), assuming a 100% capacity factor, is 12.16 EFPY. Therefore, the current P-T limit curves could expire as soon as the Fall of 2004 (mid-cycle during Cycle 10). In view of this, Nine Mile Point Nuclear Station, LLC, (NMPNS) is requesting approval of the proposed changes and issuance of the TS amendment by February 15, 2004 with 60 days allowed for implementation. In addition, the amendment will support plant heatup and cooldown and system leakage testing activities associated with the Spring 2004 refueling outage (RFO9).

The proposed P-T limit curves have been developed using the alternative methodology permitted by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640, "Alternate Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1." Code Case N-640 permits the use of an alternative fracture toughness curve (i.e., K_{Ic} in lieu of K_{Ia}) for the development of P-T limit curves. Note that the use of this alternative methodology no longer requires an exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The NRC has recently approved generic (unconditional) application of ASME Code Case N-640 in ASME Section XI inservice inspection programs, effective August 7, 2003 [68 FR 40469, July 8, 2003], by including it in Revision 13 of Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1," and by its incorporation by reference into 10 CFR 50.55a(b), "Codes and standards."

A report summarizing the inputs, methodology, and results of the calculations used in the development of the proposed (new) P-T limit curves is included in Attachment 4.

2.0 PROPOSED CHANGE

TS Figures 3.4.11-1 through 3.4.11-5 are being replaced with new figures to provide revised (recalculated) RCS P-T limit curves. The revised P-T limit curves will be valid for up to 22 EFPY and are applicable to system leakage and hydrostatic testing and RPV heatup and cooldown during reactor core non-criticality (non-nuclear) and criticality (nuclear) conditions.

The Bases for TS Section 3.4.11 are being revised to reflect the changes to the TSs.

3.0 BACKGROUND

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 31, "Fracture prevention of reactor coolant pressure boundary," the reactor coolant pressure boundary is required to be designed with sufficient margin to assure that, when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner. The GDC also requires consideration of the uncertainties in determining the effects of irradiation on material properties. These requirements are reiterated in 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation." The requirements of 10 CFR 50.60 are described in 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

Fracture toughness and reactor vessel material surveillance program requirements as specified in 10 CFR 50, Appendices G and H, must be considered in establishing RCS P-T limits. Appendix G specifies that the fracture toughness and testing requirements for reactor vessel material meet the requirements of the ASME B&PV Code and requires that the beltline material in the surveillance capsules be tested in accordance with the requirements of 10 CFR 50, Appendix H. Appendix G of 10 CFR 50 endorses ASME B&PV Code, Section XI, Appendix G, as providing a conservative method for developing reactor vessel P-T limits. In addition, Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," requires that the methods described in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," be used to predict the effects of neutron irradiation on vessel embrittlement by calculating the Adjusted Reference Temperature (ART) and the Charpy Upper Shelf Energy (USE). The ART is defined as the sum of the initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method.

Pursuant to 10 CFR 50, Appendix G, the materials used in the NMP2 Reactor Pressure Vessel (RPV) have been evaluated to determine their initial RT_{NDT} and these values were used to develop the initial and current RCS P-T limit curves.

The current P-T limit curves were approved by the NRC on January 11, 1991 and issued as Amendment No. 26 to the NMP2 TSs. The P-T limit curves and associated requirements were converted to the BWR Improved Standard TSs (NUREG-1434, Rev. 1) format (i.e., renumbered from Specification 3/4.4.6.1 to Specification 3.4.11) in accordance with Amendment No. 91, which was implemented in December 2000. Approval of the current P-T limit curves was based on the conformance of the limits to the requirements of 10 CFR 50, Appendix G, and Generic Letter 88-11. The current P-T limits satisfied Generic Letter 88-11 since the method used to calculate the ART conformed to Regulatory Guide 1.99, Revision 2. No surveillance capsules had been removed from the NMP2 reactor vessel at the time, so the ART was determined by Section C.1 of Regulatory Guide 1.99, Revision 2.

4.0 TECHNICAL ANALYSIS

In December 2000, NMPNS contracted with MPM Technologies, Inc. to recalculate the NMP2 P-T limit curves. The recalculated (proposed) P-T limit curves are based, in part, on fluence values calculated using methods that comply with Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." In addition, the proposed P-T limit curves include improvements that have been made to the calculational methodology contained in Section XI, Appendix G, of the ASME B&PV Code. The proposed new P-T limit curves have been calculated for 22 EFPY.

The methodology improvements were the application of ASME B&PV Code Case N-640, which permits fracture toughness curve K_{Ic}, as found in ASME B&PV Code, Section XI, Appendix A, to be used in lieu of curve K_{Ia} of Section XI, Appendix G, for the development of P-T limit curves. The proposed (new) P-T limit curves for NMP2 were, therefore, developed in accordance with 10 CFR 50, Appendix G, and the 1989 Edition of ASME B&PV Code, Section XI, Appendix G, as modified by the ASME N-640 Code Case. Use of the 1989 Edition of the ASME Code is acceptable based on 10 CFR 50, Appendix G, and 10 CFR 50.55a(b)(2). Application of the methodology improvements of ASME N-640 Code Case is further discussed in Section 4.1 below.

4.1 Application of ASME N-640 Code Case

The proposed P-T limits were developed based on the methodology specified in Section XI, Appendix G, of the ASME B&PV Code, as modified by ASME B&PV Nuclear Code Case N-640. ASME Code Case N-640 permits the use of alternate material fracture toughness when developing minimum vessel temperatures. Specifically, fracture toughness K_{Ic} values as defined in ASME B&PV Code, Section XI, Appendix A, Figure A-4200-1, were used in lieu of the K_{Ia} values defined in ASME B&PV Code, Section XI, Appendix G, Figure G-2210-1, for the development of the proposed (new) P-T limit curves.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T limit curves is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heatup and cooldown processes that an RPV normally undergoes. These slow heatup and cooldown limits are enforced by NMP2 TS 3.4.11. Surveillance Requirement 3.4.11.1 states that RCS heatup and cooldown rates are to be $\leq 100°$ F/HR in any one hour period.

Use of this approach is justified by the initial conservatism of the K_{Ia} curve when it was incorporated into the ASME B&PV Code in 1974. This initial conservatism was necessary due to the limited knowledge of RPV material fracture toughness at the time. Since that time, considerable knowledge has been gained regarding fracture toughness of RPV materials and their fracture response to applied loads. This increased knowledge has served to demonstrate that the fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect against potential RPV failure, and the K_{Ic} fracture toughness curve provides an adequate margin of safety for such a failure.

The acceptability of, and technical basis for, the use of ASME Code Case N-640 is described in "Technical Basis for Revised P-T Limit Curve Methodology," by W. H. Bamford (Westinghouse Electric), S. N. Malik (NRC), et. al. This methodology was presented at the 2000 ASME Pressure Vessels and Piping Conference. In general, the revised methodology removes excess conservatism in the current ASME, Section XI, Appendix G, approach. Performance of leak tests at artificially high temperatures could impact test personnel safety, challenge operators with maintaining a high temperature in a limited operating band, and decrease the availability of plant systems, including the Residual Heat Removal System, due to the longer RPV heatup and test time.

Notwithstanding that the use of the ASME N-640 Code Case changes the methodology used to calculate the proposed P-T limit curves, the modified methodology continues to satisfy the guidance contained in the 1989 Edition of ASME B&PV Code, Section XI, Appendix G. Therefore, it follows that the proposed P-T limit curves will also continue to satisfy the intent of the guidance contained in 10 CFR 50, Appendices G and H.

The NRC has found the application of the ASME N-640 Code Case acceptable. A number of nuclear facilities have previously requested the use of the N-640 Code Case and their applications have been approved by the NRC. [Reference: Pilgrim (ADAMS Accession Numbers ML010720448 and ML010790519), Brunswick Units 1 and 2 (ADAMS Accession Numbers ML012760157 and ML012780286), and Susquehanna Units 1 and 2 (ADAMS Accession Numbers ML013520568 and ML013520605)]. As previously discussed, the use of the ASME N-640 Code Case no longer requires an exemption from the requirements of 10 CFR 50.60. The NRC has recently approved generic (unconditional) application of the N-640 Code Case in ASME Section XI inservice inspection programs, effective August 7, 2003 [68 FR 40469, July 8, 2003], by including it in Revision 13 of Regulatory Guide 1.147 and by its incorporation by reference into 10 CFR 50.55a(b).

Based on the technical basis provided in "Technical Basis for Revised P-T Limit Curve Methodology," by W. H. Bamford (Westinghouse Electric), S. N. Malik (NRC), et. al., and continued compliance with 10 CFR 50, Appendices G and H, NMPNS has concluded that the proposed P-T limit curves maintain an adequate margin of safety for brittle fracture.

4.2 Updated Fluence Calculations

GDC 31 and 10 CFR 50, Appendix G, require the prediction of the effects of neutron irradiation on vessel embrittlement. In accordance with Generic Letter 88-11, the NRC requires the methods described in Regulatory Guide 1.99, Revision 2, to be used to predict these effects. The Regulatory Guide requires the ART and USE factors to be calculated to account for the effects of neutron embrittlement. If the ART and USE satisfy the limits contained in the Regulatory Guide and 10 CFR 50, Appendix G, then the vessel materials provide adequate margin against brittle fracture. One of the key components used in the calculations of both the ART and the USE is RPV neutron fluence.

Regulatory Guide 1.190 provides guidance for the calculation and measurement of RPV neutron fluence. The neutron fluence values calculated using the methodology described in Regulatory Guide 1.190 satisfy the requirements of 10 CFR 50, Appendix G, and Regulatory Guide 1.99, Revision 2. Accordingly, ART values calculated using these neutron fluence values would also satisfy 10 CFR 50, Appendix G, and Regulatory Guide 1.99, Revision 2, and thereby satisfy Generic Letter 88-11.

The NRC issued Generic Letter 88-11 to alert Licensees to the May 1988 issuance of Revision 2 to Regulatory Guide 1.99. The Generic Letter requested an analysis of the impact of Revision 2 to Regulatory Guide 1.99 on the TS P-T limit curves and requested that all actions necessary be completed within two plant outages of the effective date of issuance. The existing (original) P-T limit curves were revised in response to Generic Letter 88-11 since the ART for the limiting vessel beltline material (Plate C3147) using Revision 2 of the Regulatory Guide was higher than that previously calculated for 12.8 EFPY. The ART for the revised (current) P-T limit curves was determined using Section C.1 of Regulatory Guide 1.99, Revision 2, since no surveillance capsules had been removed from the NMP2 reactor vessel at the time. The net effect of the increase in the ART was a narrowing of the P-T window for heatup and cooldown operations.

The current P-T limit curves were approved by the NRC on January 11, 1991 and issued as Amendment No. 26 [Reference: TAC Nos. 71519 and 76399] to the NMP2 TSs. The P-T limit curves and associated requirements were converted to the BWR Improved Standard TSs (NUREG-1434, Rev. 1) format (i.e., renumbered from Specification 3/4.4.6.1 to Specification 3.4.11) in accordance with Amendment No. 91 [Reference: TAC No. MA3822], which was implemented in December 2000. The current P-T limits are valid for 12.8 EFPY and satisfy Generic Letter 88-11 since the method used to calculate the ART is consistent with Regulatory Guide 1.99, Revision 2. Accordingly, the current P-T limit curves and supporting RPV neutron fluence values satisfy the requirements of 10 CFR 50, Appendix G.

The RPV neutron fluence values used for the proposed (new) P-T limit curves were updated following analysis of the 3° vessel surveillance capsule that was withdrawn at the end of Fuel Cycle 7 (March 3, 2000). The updated fluence values and analyses results were reported to the NRC in Letter NMP2L 2015, dated March 8, 2001 (ADAMS Accession No. ML010750205). The MPM Technologies, Inc. methodology was used to determine the updated RPV neutron fluence values for the development of the proposed P-T limit curves. The MPM methodology fully satisfies the requirements of Regulatory Guide 1.190. The details regarding the neutron transport model, analysis procedures, and compliance with Regulatory Guide 1.190 are provided in Report Nos. MPM-1200676, "Nine Mile Point Unit 2 3-Degree Surveillance Capsule Report," MPM-301624, Revision 1, "Nine Mile Point Unit 2 Shroud Neutron Transport and Uncertainty Analysis," and MPM-301624A, Revision 1,"Nine Mile Point Unit 2 Shroud Neutron Transport and Uncertainty Analysis: Addendum," which were transmitted to the NRC for review in Letters NMP2L 2015, dated March 8, 2001, and NMP1L 1708, dated January 15, 2003 (ADAMS Accession No. ML030290056). Letter NMP1L 1708 also transmitted Report No. MPM-402781, "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations," for NRC review and approval pursuant to the benchmarking requirements of Regulatory Guide 1.190 for plant-specific (Nine Mile Point Unit 1 (NMP1) and NMP2) qualification of the MPM neutron transport calculational methodology.

Using the updated fluence values, revised values of the ART were calculated for 22 EFPY. According to Regulatory Guide 1.99, Revision 2, the vessel beltline plate or weld with the largest ART is limiting. Based on the revised ART values, the limiting beltline material at the wetted surface location is plate C3147. Since the ART is determined by summing the initial RT_{NDT}, ΔRT_{NDT}, and Margin, and because the ART for plate C3065-2 is close to that of plate C3147, it is also necessary to verify the limiting plate by calculation using the updated fluences for the 1/4T and 3/4T locations. Based on the results of these calculations, the limiting beltline material is Plate C3147 at the 1/4T location and Plate C3065-2 at the 3/4T location. Thus, as further discussed in Section 4.3 of Report No. MPM-50840 (Attachment 4), the P-T limit curves are lower bound limits conservatively based on two beltline materials. It is important to note that Plate C3147 is limiting at both the 1/4T and 3/4T positions at 22 EFPY if the σ_I (standard deviation) Margin term in the ART expression is taken as 0° F, which has been justified for plants based on measured initial RT_{NDT} values. Instead, σ_I has been conservatively maintained at 14.5° F (Reference: NMP2 Updated Safety Analysis Report, Section 5.3.2.1.3). The predicted ART for the limiting beltline material at 1/4T (Plate C3147) will remain less than 200° F for the operating life of the vessel as required by Regulatory Guide 1.99, Revision 2.

Using the updated fluence values, revised values of the USE were calculated. The revised USE values showed an unexpected slight increase due to irradiation, which is a phenomenon that has been observed in other plants and may be related to low fluence improvement of the matrix material. The limiting beltline material (Plate C3147) remained the same and the predicted shelf drop was less than 15%. Therefore, the 10 CFR 50, Appendix G, requirement to maintain a 50 ft-lb USE throughout the operating life of the vessel is satisfied for NMP2.

As discussed above, the RPV neutron fluence values used for the proposed (new) P-T limit curves were updated following withdrawal and analysis of the 3° vessel surveillance capsule. Although the new curves are for an increase of 9.2 EFPY, the limits have not changed significantly. This is because the old limits were based on the original General Electric preoperational fluence estimates which are approximately a factor of 2 larger than the actual fluence. The updated fluence values were calculated using the MPM Technologies, Inc. methodology, which satisfies the requirements of Regulatory Guide 1.190. The Regulatory Position 1.4 uncertainty analyses and comparisons with benchmark measurements and calculational benchmark problems (as provided in NUREG/CR-6115) have been completed and the Position 1.4 methodology qualification and uncertainty estimates have been satisfied. The benchmark analyses were submitted to the NRC for approval (Letter NMP1L 1708) as required to support plant-specific (applicable to NMP1 and NMP2) qualification of the MPM methodology. Supplemental information was provided in Letter NMP1L 1749, dated July 31, 2003, in response to an NRC Request for Additional Information (RAI), dated May 6, 2003, regarding the benchmark analyses. As requested, the RAI response provided a justification for use of the ORIGEN 2.1 code for fission source determinations and reported the results of additional analyses performed for the pool critical assembly (PCA) benchmark and available NMP1 surveillance capsule dosimetry. As reported in the RAI response, successful calculation of the PCA benchmark has been achieved and comparisons of the calculated and measured reaction rates for the NMP1 dosimetry sets indicate agreement well within the ±20% requirement for fluence calculational uncertainty. Moreover, the calculations do not exhibit any significant bias. Therefore, the supplemental information provided in the RAI response confirmed that the MPM methodology used for the calculation of the NMP1 and NMP2 fluence values fully satisfies the Regulatory Guide 1.190 requirement for fluence methodology qualification by measurement and calculational benchmarks.

Based on the acceptable results of the verifications and benchmarking comparisons of the updated RPV neutron fluence values and calculational methodology, NMPNS has concluded that the updated fluence values calculated for the proposed NMP2 P-T limit curves are consistent with the requirements of Regulatory Guide 1.190. Accordingly, the USE and ART values calculated using these updated fluence values satisfy 10 CFR 50, Appendix G, and Regulatory Guide 1.99, Revision 2, respectively, and thereby satisfy Generic Letter 88-11.

4.3 Conclusion

NRC regulations require that P-T limit curves provide an adequate margin of safety to the conditions at which brittle fracture may occur. These requirements are set forth in GDC 31 and 10 CFR 50, Appendices G and H. Generic Letter 88-11 and Regulatory Guides

1.99 and 1.190 provide guidance for compliance with the requirements of GDC 31 and Appendices G and H. The Appendices reference the requirements and guidance of Section XI, Appendix G, of the ASME B&PV Code for the development of P-T limit curves. The methodologies described in Regulatory Guides 1.99 and 1.190 and the ASME Code will provide P-T limit curves with the requisite margin against brittle fracture. The proposed P-T limit curves are consistent with these methodologies, as modified by application of ASME Code Case N-640.

ASME Code Case N-640 proposes an alternative to a requirement contained in Section XI, Appendix G, of the ASME B&PV Code. The alternate fracture toughness for RPV materials permitted by the Code Case is based on the additional knowledge gained since the inception of 10 CFR 50, Appendix G. The more appropriate assumptions and provisions allowed by the Code Case maintain a margin of safety that is consistent with the intent of 10 CFR 50, Appendices G and H.

The use of the ASME N-640 Code Case no longer requires an exemption from the requirements of 10 CFR 50.60. The NRC has recently approved generic (unconditional) application of the N-640 Code Case in ASME Section XI inservice inspection programs, effective August 7, 2003 [68 FR 40469, July 8, 2003], by including it in Revision 13 of Regulatory Guide 1.147 and by its incorporation by reference into 10 CFR 50.55a(b). In addition, prior to its generic approval, the NRC has granted exemptions allowing use of ASME Code Case N-640 and approved the associated TS changes for a number of other Boiling Water Reactor (BWR) plants, including: Pilgrim (ADAMS Accession Numbers ML010720448 and ML010790519), Brunswick Units 1 and 2 (ADAMS Accession Numbers ML012760157 and ML012780286), and Susquehanna Units 1 and 2 (ADAMS Accession Numbers ML013520568 and ML013520605).

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration Analysis

The proposed changes to the Technical Specifications (TSs) would replace the current Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves with revised curves that are based, in part, on the alternate methodology of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Case N-640. The TS Bases will be revised to reflect the proposed changes to the TSs.

Nine Mile Point Nuclear Station, LLC, (NMPNS) has evaluated whether or not 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

The proposed changes do not involve physical changes to the plant or alter the RCS pressure boundary (i.e., there are no changes in operating pressure, materials, or seismic loading). The proposed P-T limit curves and supporting changes provide continued assurance that the fracture toughness of the Reactor Pressure Vessel (RPV) is consistent with analysis assumptions and NRC regulations. The proposed P-T curves were developed in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, and ASME B&PV Code, Section XI, Appendix G, as modified by the alternate criteria and methods of ASME B&PV Code Case N-640. The more appropriate assumptions and provisions allowed by the Code Case maintain sufficient margins of safety to assure that, when stressed, the RPV boundary will behave in a non-brittle manner. Use of this methodology provides assurance that the probability of a rapidly propagating fracture will be minimized. The proposed P-T limit curves and supporting changes will prohibit operation in regions where it is possible for brittle fracture of reactor vessel materials to occur, thereby assuring that the integrity of the RCS pressure boundary is maintained. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed P-T limit curves and supporting changes do not affect the design or assumed accident performance of any structure, system, or component, or introduce any new modes of system operation or failure modes. Compliance with the proposed P-T curves and supporting requirements will provide sufficient protection against brittle fracture of reactor vessel materials to assure that the RCS pressure boundary performs as 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. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   NRC regulations require that P-T limit curves provide an adequate margin of safety to the conditions at which brittle fracture may occur. These requirements are set forth in 10 CFR 50, Appendix A, General Design Criterion (GDC) 31 and 10 CFR 50, Appendices G and H. Generic Letter 88-11 and Regulatory Guides 1.99 and 1.190 provide guidance for compliance with the requirements of GDC 31 and Appendices G and H. The Appendices reference the requirements and guidance of Section XI, Appendix G, of the ASME B&PV Code for the development of P-T limit curves. The methodologies described in Regulatory

Guides 1.99 and 1.190 and the ASME Code will provide P-T limit curves with the requisite margin against brittle fracture. The proposed P-T limit curves are consistent with these methodologies, as modified by the application of ASME Code Case N-640, which has recently been approved by the NRC for generic (unconditional) application in ASME Section XI inservice inspection programs. ASME Code Case N-640 proposes an alternative to a requirement contained in Section XI, Appendix G, of the ASME B&PV Code. The alternate fracture toughness for RPV materials permitted by the Code Case is based on the additional knowledge gained since the inception of 10 CFR 50, Appendix G. The more appropriate assumptions and provisions allowed by the Code Case maintain a margin of safety that is consistent with the intent of 10 CFR 50, Appendices G and H. The proposed P-T limit curves and supporting requirements provide assurance that the established P-T limits are not exceeded. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NMPNS concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The proposed P-T limit curves are consistent with the NRC approved alternate assessment criteria and methods of ASME B&PV Code Case N-640, and satisfy the requirements of GDC 31; 10 CFR 50.60; 10 CFR 50, Appendix G; and the 1989 Edition of ASME B&PV Code, Section XI, Appendix G, as modified by the Code Case. The proposed P-T limit curves also satisfy Generic Letter 88-11 by using methods consistent with Regulatory Guide 1.99, Revision 2, and Regulatory Guide 1.190.

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.

5.3 Commitments

The following table identifies those actions committed to by NMPNS in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENTS	Due Date/Event
None	None

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

The current versions of Technical Specification pages 3.4.11-6 through 3.4.11-10 have been marked-up by hand to reflect the proposed changes.

INSERT
FIGURE 3.4.11-1

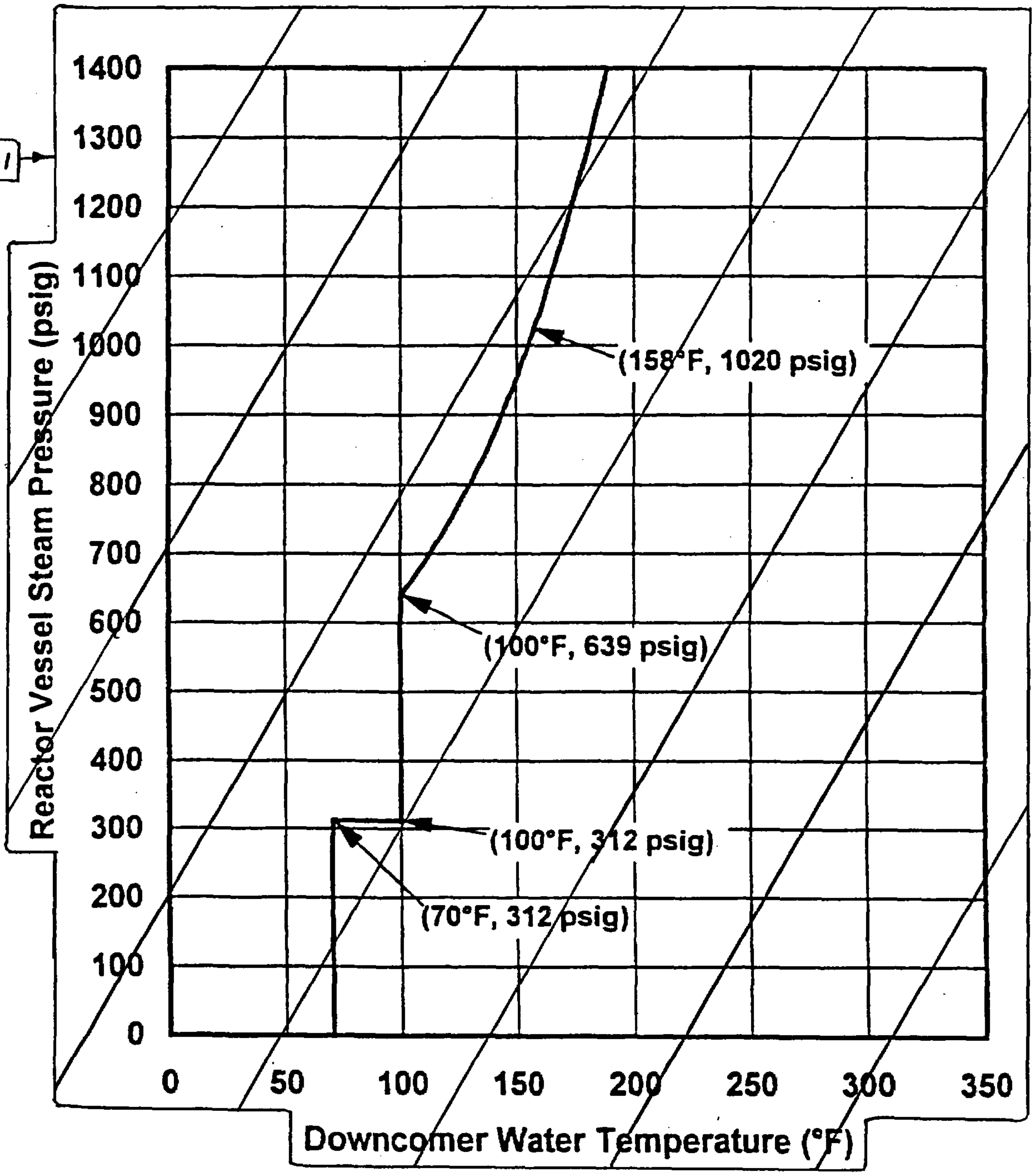
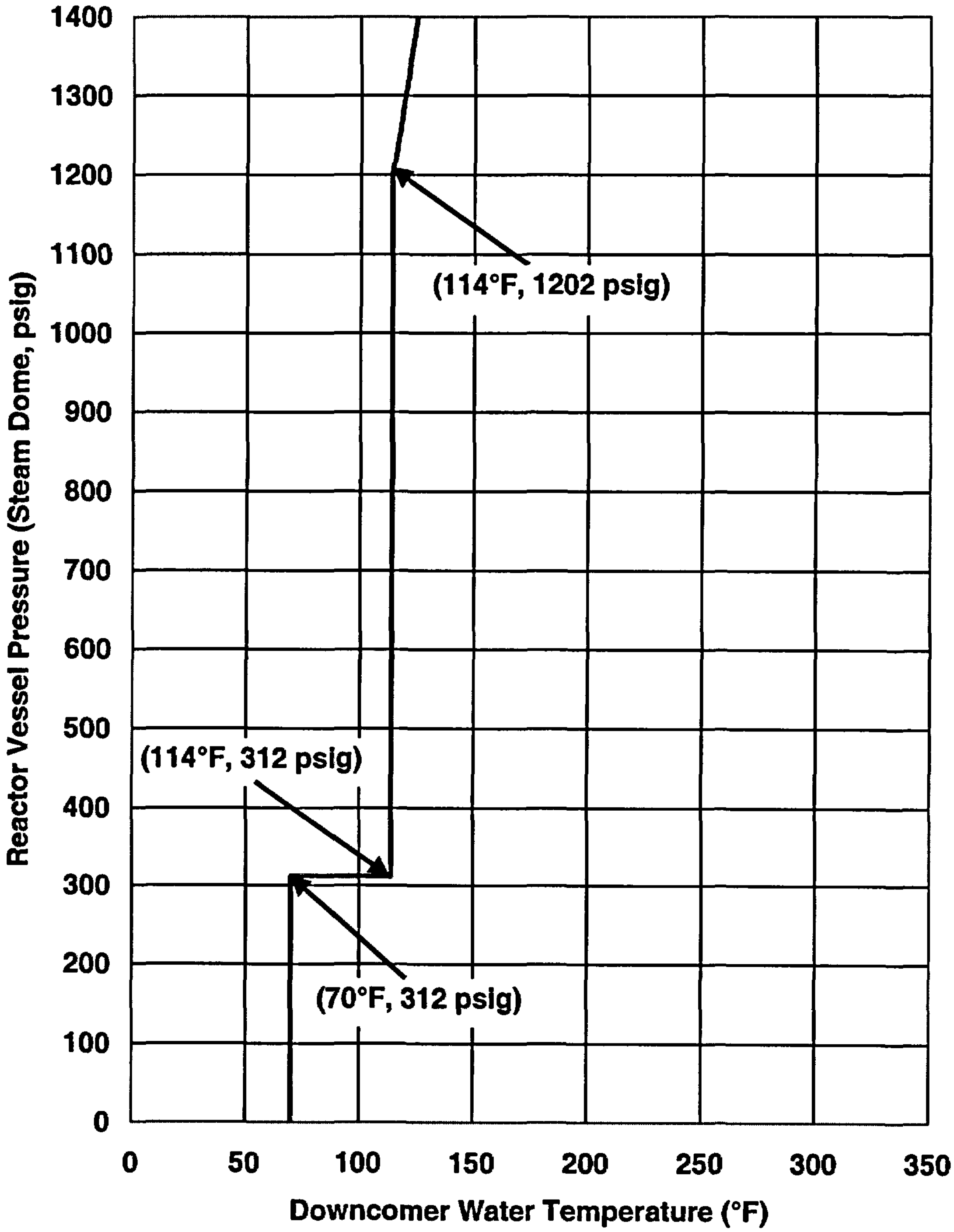


Figure 3.4.11-1 (Page 1 of 1)
Non-Nuclear System Leakage and Hydrostatic Testing Curve

INSERT FIGURE 3.4.11-1



INSERT FIGURE 3.4.11-2

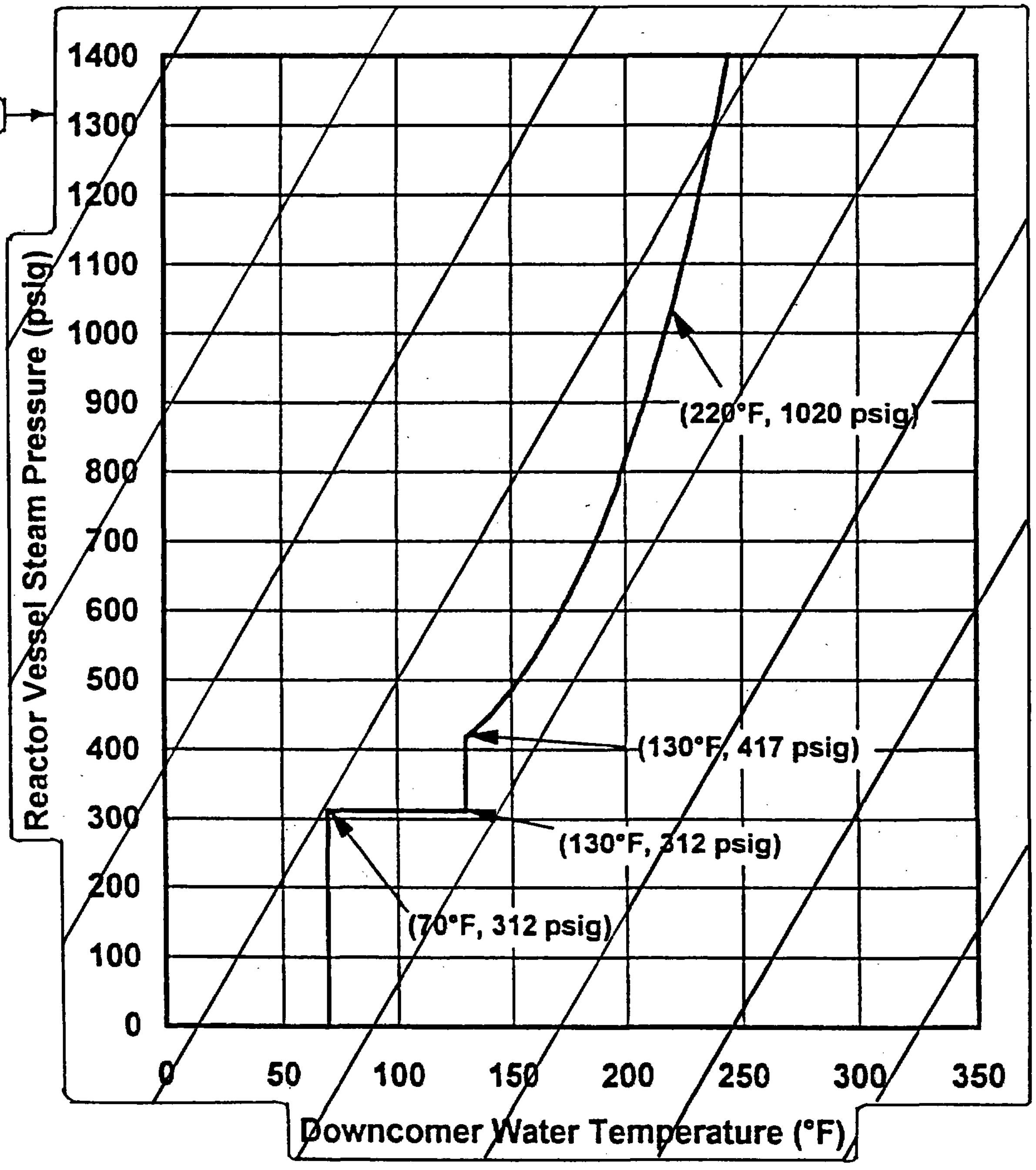
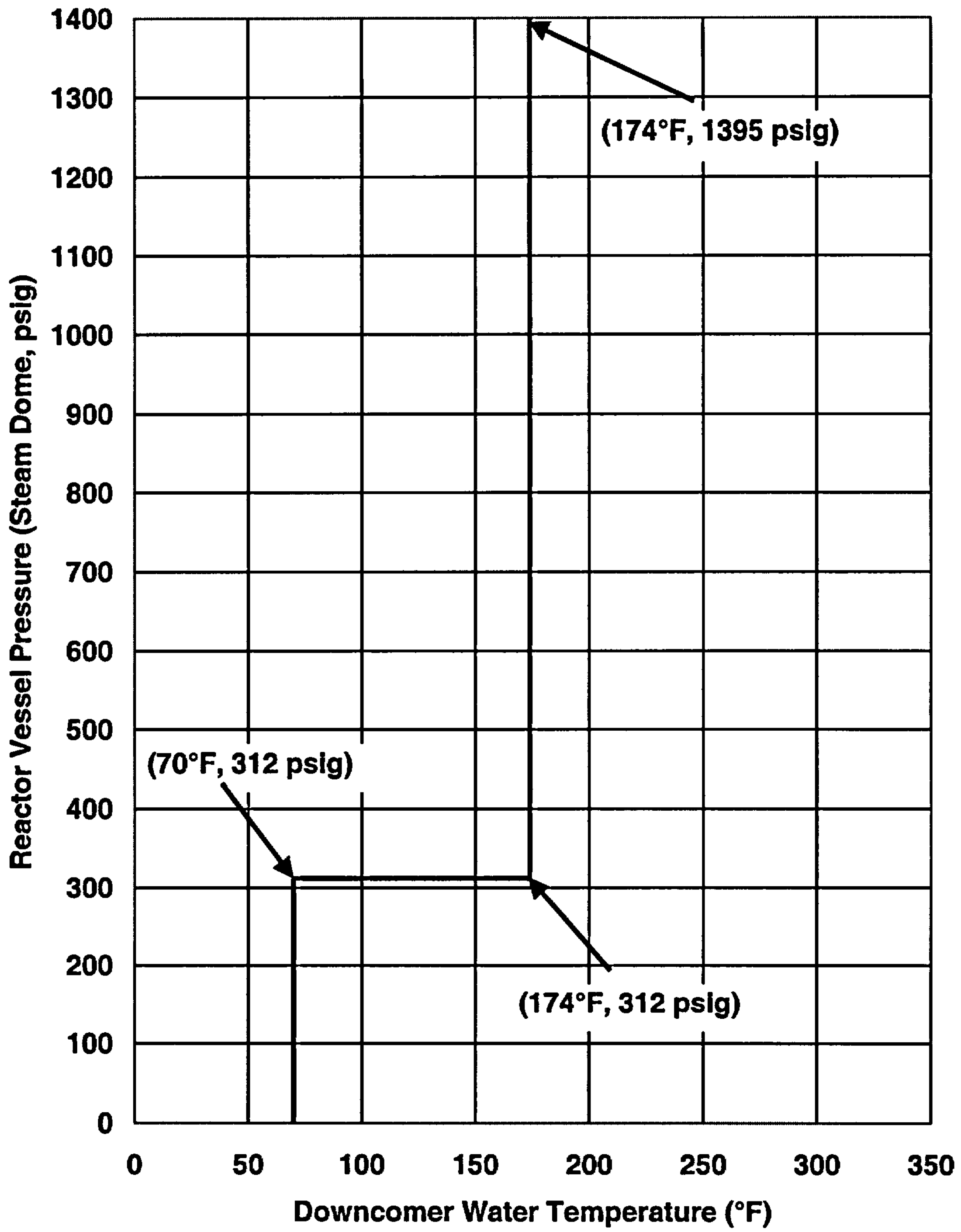


Figure 3.4.11-2 (Page 1 of 1)
Non-Nuclear Heatup Curve

INSERT FIGURE 3.4.11-2



INSERT FIGURE 3.4.11-3

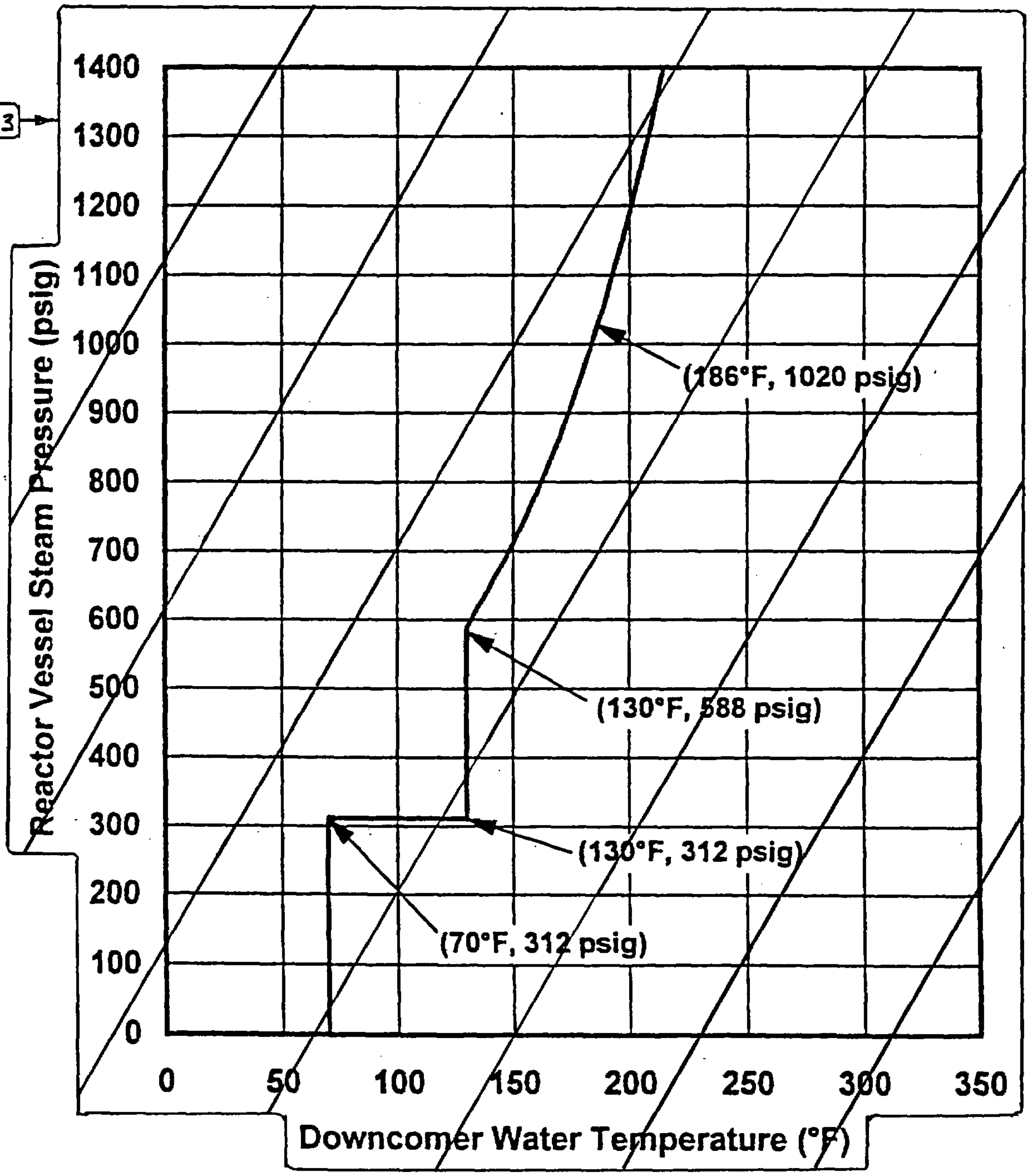
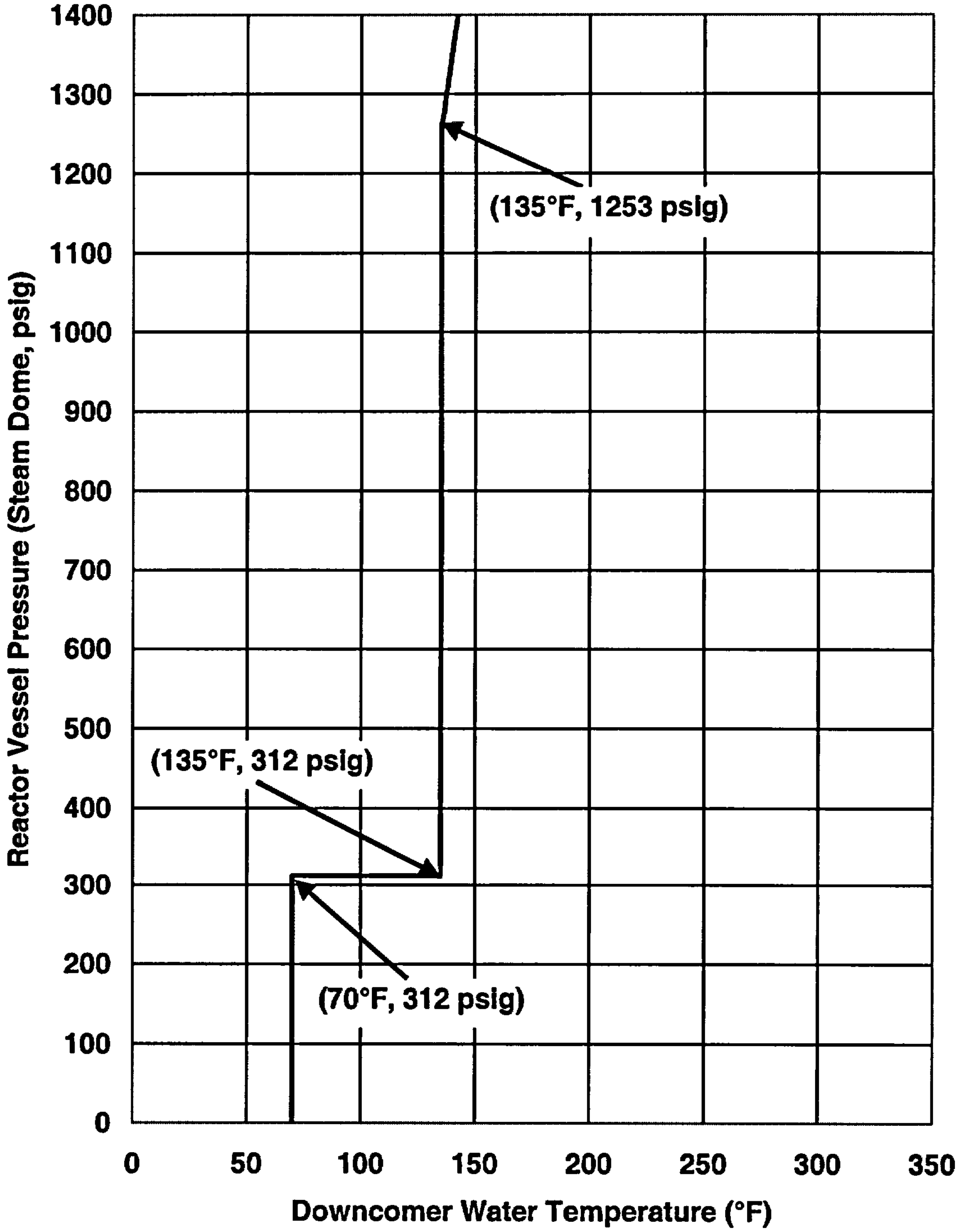


Figure 3.4.11-3 (Page 1 of 1)
Non-Nuclear Cooldown Curve

INSERT FIGURE 3.4.11-3



INSERT FIGURE 3.4.11-4

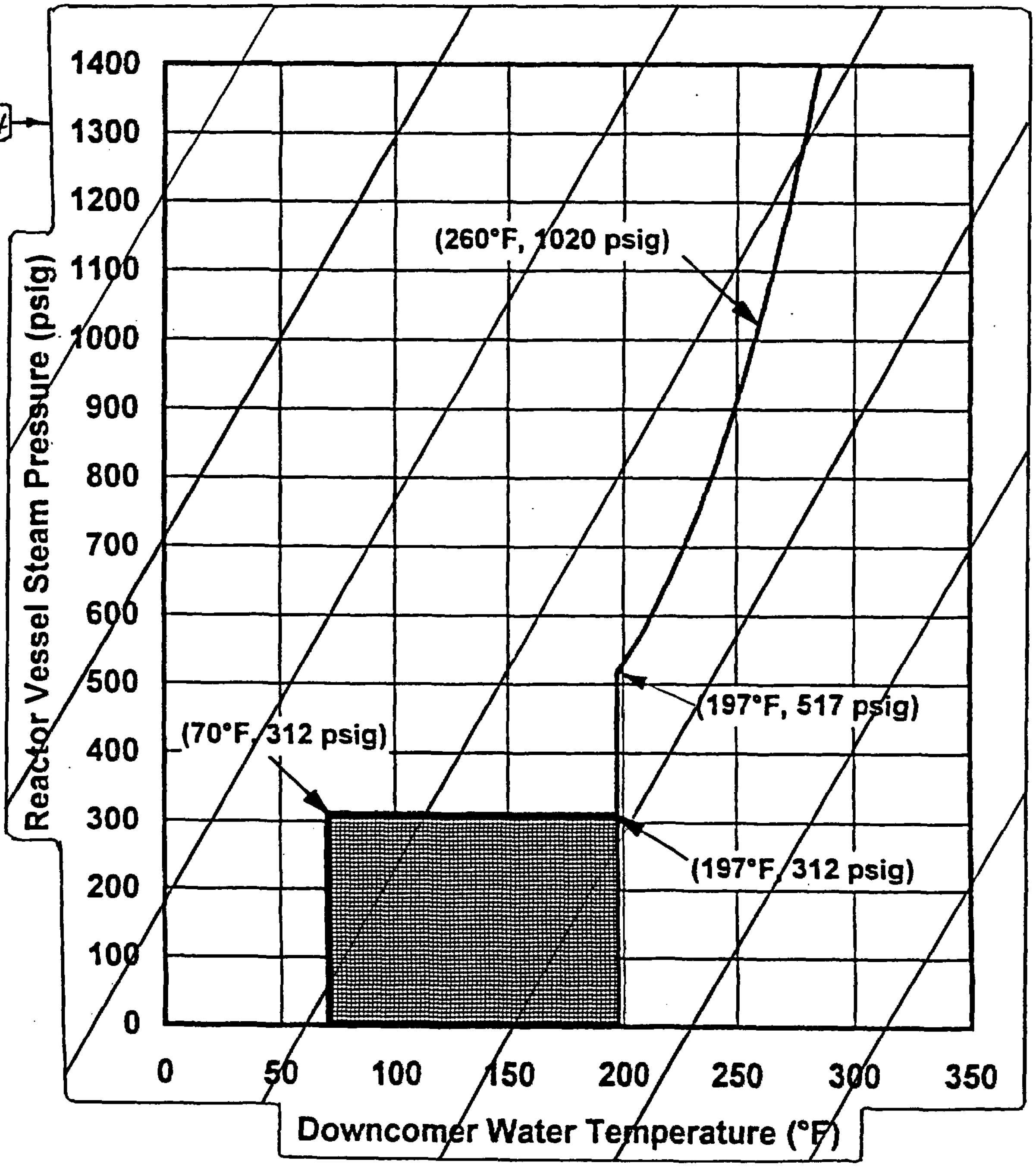
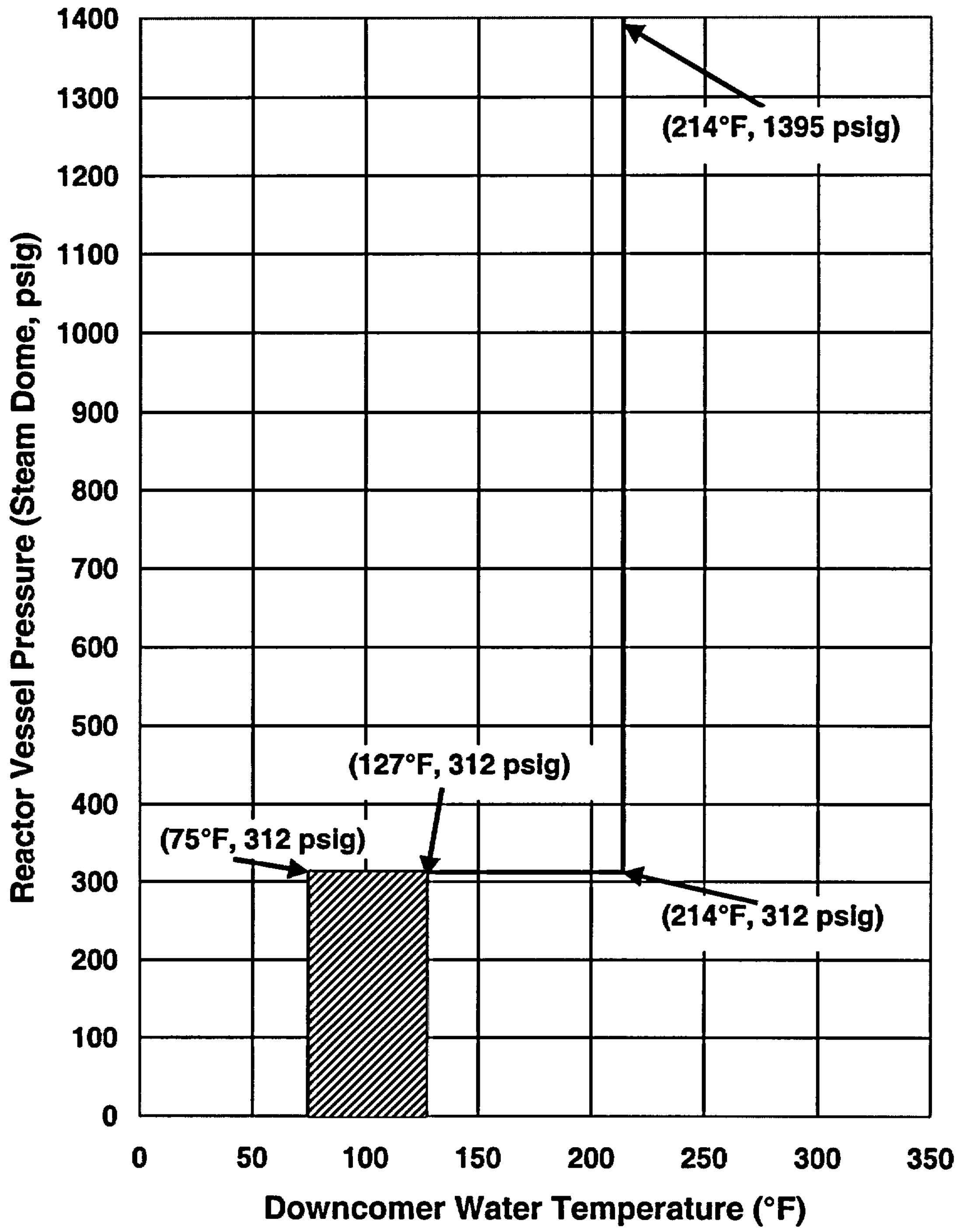


Figure 3.4.11-4 (Page 1 of 1)
Nuclear Heatup Curve

INSERT FIGURE 3.4.11-4



INSERT FIGURE 3.4.11-5

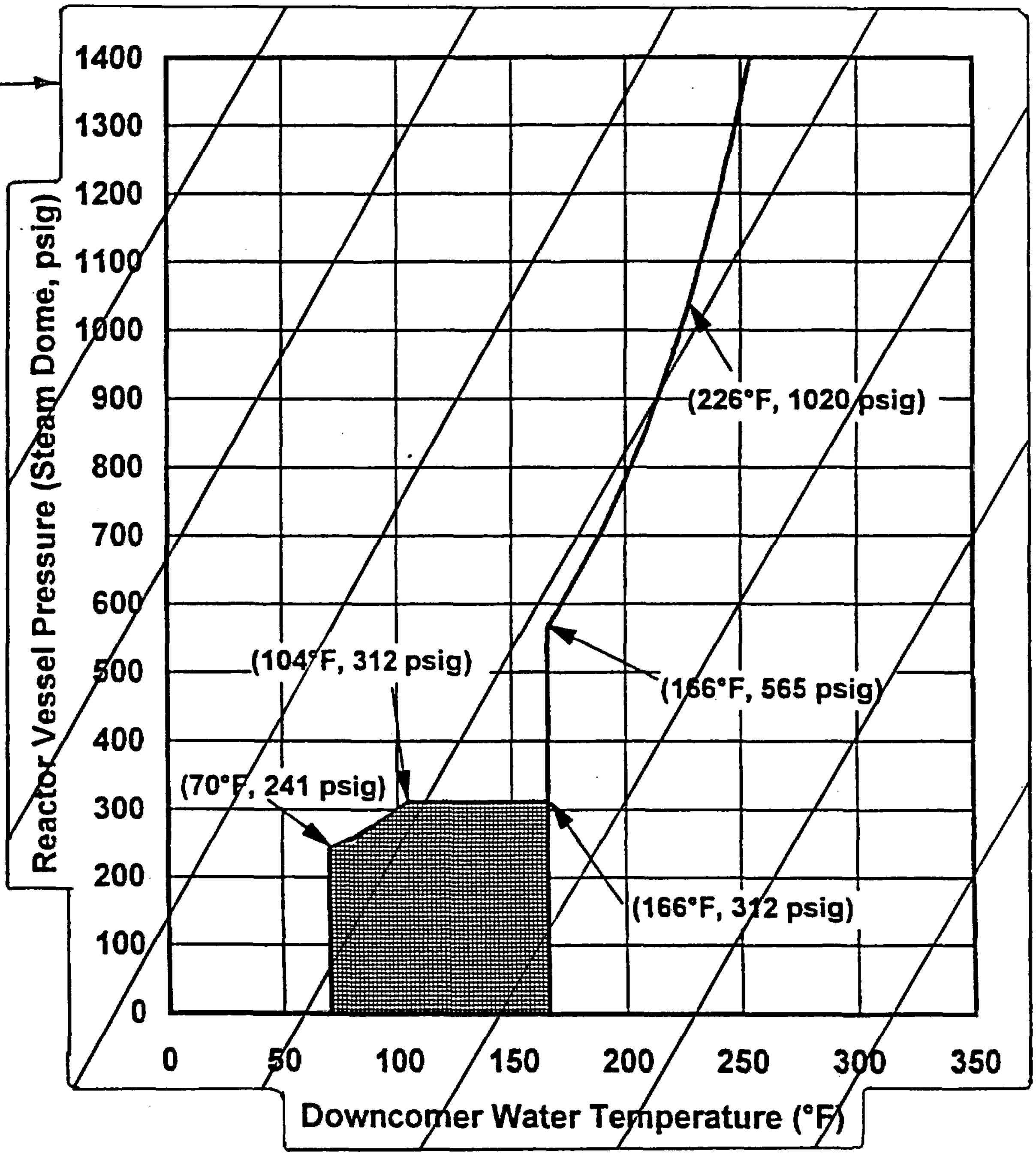
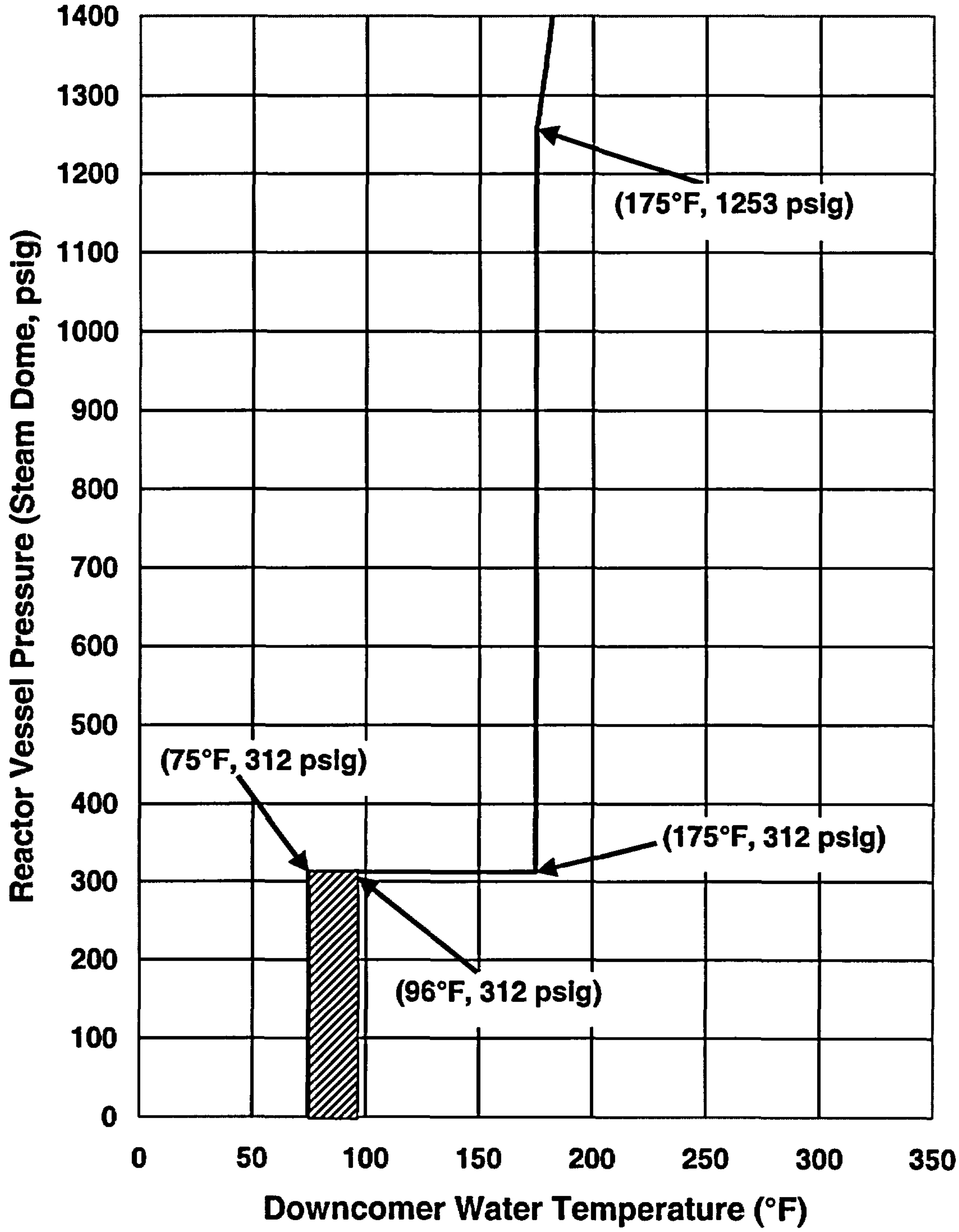


Figure 3.4.11-5 (Page 1 of 1)
Nuclear Cooldown Curve

INSERT FIGURE 3.4.11-5



ATTACHMENT 3

CHANGES TO TECHNICAL SPECIFICATION BASES PAGES

(FOR INFORMATION ONLY)

The current version of Technical Specification Bases pages B 3.4.11-1, B3.4.11-3, and B3.4.11-10 have been marked-up by hand to reflect the proposed changes. These Bases pages are provided for information only and do not require NRC issuance.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

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

The Specification contains P/T limit curves for heatup, cooldown, system leakage and hydrostatic testing, and criticality, and also limits the maximum rate of change of reactor coolant temperature. The P/T limit curves are applicable up to 12/8 effective full power years.

22

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

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

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

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. References 7 and 8 ~~approved~~ provide the basis for the curves and limits required by this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of Reference 9.

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3, 3.4.11-4, and 3.4.11-5, heatup and cooldown rates are ≤ 100°F in any 1 hour period during RCS heatup, cooldown, and system leakage and hydrostatic testing, and the RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any 1 hour period when the RCS temperature and pressure are not within the limits of Figure 3.4.11-2 or Figure 3.4.11-3, as applicable;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup, and during increases in THERMAL POWER or jet pump loop flow while in single loop operation at low THERMAL POWER or jet pump loop flow;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup, and during increases in THERMAL POWER or jet pump loop flow while in single loop operation at low THERMAL POWER or jet pump loop flow;

d. RCS pressure and temperature are within the criticality limits specified in Figures 3.4.11-4 and 3.4.11-5, prior to achieving criticality; and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

<u>SR 3.4.11.7, SR 3.4.11.8, and SR 3.4.11.9</u> (continued)

The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. ASTM E 185-82, July 1982.

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. ~~Letter from S.A. Varga (NRC) to C.V. Mangan (NMPC), "Issuance of Facility Operating License No. NPF-69 - Nine Mile Point Nuclear Station, Unit 2," dated July 2, 1987.~~ INSERT REF 7

8. ~~Letter from D.S. Brinkman to B.R. Sylvia (NMPC), "Issuance of Amendment 26 for Nine Mile Point Nuclear Station, Unit 2," dated January 11, 1991.~~ INSERT REF 8

9. 10 CFR 50.36(c)(2)(ii).

10. USAR, Section 15.4.4.

INSERT REF 7

Report No. MPM-502840, "Pressure-Temperature Operating Curves for Nine Mile Point Unit 2," July 2003.

INSERT REF 8

ASME Code Case N-640, "Alternate Reference Fracture Toughness for Development of P-T Limit Curves Section XI, Division 1."

ATTACHMENT 4

REPORT NO. MPM-502840

Pressure-Temperature Operating Curves for Nine Mile Point Unit 2

MPM
....SERVING CLIENT NEEDS
THROUGH ADVANCED TECHNOLOGY

July, 2003

Report No. MPM-502840

Final Report

entitled

Pressure-Temperature Operating Curves for Nine Mile Point Unit 2

Prepared for:

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July, 2003

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Preparer

7/31/03
Date

Robert B S H

Checker

7/31/03
Date

M P Manahan, Sr.

MPM Approval

7/31/03
Date

Nuclear Quality Assurance Certification

This document certifies that MPM has performed all work under Constellation Purchase Order Number 00-30028 in accordance with the requirements of the Purchase Order. All work has been performed under the MPM Nuclear Quality Assurance Program.

M P Manahan, Sr.

M. P. Manahan, Sr.
President

7/31/03
Date

Richard P. Erhard

R. Erhard
QA Manager

7/31/03
Date

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> "Pressure-Temperature Operating Curves for Nine Mile Point Unit 2", Report MPM-502840, MPM Technologies, Inc., 2161 Sandy Drive, State College, PA 16803-2283, July, 2003.

MPM Technologies, Inc.

M P Manahan, Sr.

M. P. Manahan, Sr.
President

Contents

Executive Summary

Reactor pressure vessel (RPV) materials undergo a transition in fracture behavior from brittle to ductile as the test temperature of the material is increased. Charpy V-notch tests are conducted in the nuclear industry to monitor changes in the fracture behavior during irradiation. Neutron irradiation to fluences above ~5 x 10^{16} n/cm^2 causes an upward shift in the Charpy curve as a function of temperature and an increase in the ductile-to-brittle transition temperature (DBTT). In order to ensure safe operation of a nuclear power plant during heatup, cooldown, and leakage/hydrotest conditions, it is necessary to conservatively calculate allowable stress loadings for the ferritic RPV materials. These allowable loadings can be conveniently presented as a plot of measured coolant pressure versus measured coolant temperature (P-T curves). Appendix G to 10CFR50 and Appendix G to Section XI (equivalent to Appendix G to Section III) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code present a procedure for obtaining the allowable loadings for ferritic pressure-retaining materials in Class 1 components. Neutron damage within the RPV during plant operation is accounted for in the allowable pressure loading by calculating an adjusted reference temperature (ART). Regulatory Guide 1.99, Revision 2 (RG1.99(2)) defines the ART as the sum of the initial unirradiated nil-ductility reference temperature (RT_{NDT}), plus the RT_{NDT} irradiation induced shift (ΔRT_{NDT}), plus a margin term. Within the nuclear industry, the ΔRT_{NDT} is determined from the Charpy transition curve shift indexed at 30 ft-lbs of absorbed energy. Regulatory Guide 1.99, Revision 2, presents a model and procedure for calculation of the ΔRT_{NDT} based on the material chemistry and fluence.

The P-T limits currently in use for Nine Mile Point Unit 2 (NMP-2) were calculated up to 12.8 effective full power years (EFPY). The new P-T limits reported here are valid for up to 22 EFPY. Although the new limits are for an increase of 9.2 EFPY, the P-T limits have not changed significantly from the 12.8 EFPY limits as a result of a fluence increase. This is because the old limits are based on GE's preoperational fluence estimates which are approximately a factor of 2 larger than the actual fluence. MPM has recently performed fluence calculations for NMP-2 using methods which are in compliance with Regulatory Guide 1.190, and these updated fluence results have been used in the P-T limit calculation.

The applied loadings for the postulated defects in the vessel wall must be bounded by the material toughness. The ASME K_{IR} curve has been used in the past for this purpose and is the lower bound to static, dynamic, and crack arrest fracture toughness data as a function of test temperature. ASME Code Case N-640 allows for replacement of the K_{IR}/K_{Ia} curve with the ASME lower bound to static fracture toughness data (K_{IC} curve). The P-T limit curves reported here have been calculated using the ASME Code Case N-640.

In summary, P-T operating curves for NMP-2 have been calculated for up to 22 EFPY of operation. These new P-T curves satisfy the requirements of 10CFR50, Appendix G and the ASME Code. Operation of NMP-2 in accordance with the revised P-T operating limits will preclude brittle fracture of the RPV materials. Safety margins for brittle fracture are in

accordance with those specified in 10CFR50, Appendix G and Appendix G to Section III/Section XI of the ASME Code. Therefore, the revised P-T limits do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not introduce the possibility of a new or different kind of accident, and do not significantly reduce existing margins of safety.

1.0 Introduction

To ensure safe operation of a nuclear power plant during heatup, cooldown, and leakage/ hydrostatic testing conditions, it is necessary to conservatively calculate allowable stress loadings for the ferritic reactor pressure vessel (RPV) materials. These allowable loadings are presented as a plot of measured coolant pressure versus measured coolant temperature (P-T curves). The P-T limits include the instrument error and conservatively account for the head of water from the measurement location in the steam dome to the bottom of the beltline region. The P-T curves currently in use at Nine Mile Point Unit 2 (NMP-2) were determined using the K_{IR} curve as the material bound and are valid for up to 12.8 EFPY . The current P-T limits were originally reported in Reference [1-1] and were updated to the Improved Technical Specification (ITS) format in 1998 [1-2].

The new P-T curves have been prepared to extend the valid operating period to 22 EFPY. Although the new limits will increase the exposure from 12.8 to 22 EFPY, P-T limits based on the ASME reference stress intensity factor (K_{IR} curve) would not change significantly because the old limits are based on GE's preoperational fluence estimates which are approximately a factor of 2 larger than the actual fluence. The recently calculated fluences for NMP-2 [1-3], determined using methods which are in compliance with Regulatory Guide 1.190, have been used in the new P-T limit calculations. In addition, the new leak test curve includes a thermal loading for heatup at a rate of up to 20 F/hour. Therefore, since the thermal loading is included in the leak test P-T limit, there is no need for a vessel thermal soak prior to pressurization to 1035 psig during the leak test.

The applied loadings for the postulated defects in the vessel wall must be bounded by the material toughness. The ASME K_{IR} curve has been used in the past for this purpose and is the lower bound to static, dynamic, and crack arrest fracture toughness data as a function of test temperature. Code Case N-640 allows for replacement of the K_{IR}/K_{Ia} curve with the ASME lower bound to static fracture toughness data (K_{IC} curve). The P-T limit curves reported here have been calculated using the ASME Code Case N-640. To maintain consistency with previous P-T models for NMP-2, the ASME K_{Ic} curve was implemented under Appendix G to Section III of the Code. The equivalence between Appendix G of Section III and Appendix G of Section XI will be discussed later in the report.

This report documents the methods used to obtain the revised P-T curves. Section 2.0 presents the methodology used for P-T curve calculation. Section 3.0 briefly summarizes the results of neutron transport calculations which were performed to more accurately determine the peak flux at the vessel inner diameter (ID) surface. Section 4.0 reviews the plant data used in the calculations and presents the resulting P-T curves. Section 5.0 briefly summarizes the work performed and recommends implementation of the new curves.

1.1 Chapter 1 Reference

[1-1] Manahan, M. P., "Pressure-Temperature Operating Curves for Nine Mile Point Unit 2", Report Submitted to NMPC, November, 1989.

[1-2] Letter Report from M. P. Manahan to T. Kurtz, "Improved Technical Specifications for Nine Mile Point Unit 2", June 30, 1998.

[1-3] MPM Report entitled, "Nine Mile Point Unit 2 3-Degree Surveillance Capsule Report", MPM-1200676, December, 2000.

2.0 Calculative Procedure

The regulations governing the calculation of P-T limits for the reactor coolant pressure boundary are found in the Code of Federal Regulations (CFR), the ASME Boiler and Pressure Vessel Code, and Regulatory Guides. As stated in Reference [2-1], the following are the regulations requiring P-T limits:

> "Paragraph 50.55a of 10 CFR Part 50, "Codes and Standards," requires that structures, systems, and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. In addition, General Design Criterion 1 of Appendix A of 10 CFR Part 50, "Quality Standards and Records," requires that the codes and standards used to assure quality products in keeping with the safety function be identified and evaluated to determine their adequacy.
>
> General Design Criterion 14 of Appendix A of 10 CFR Part 50, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. Likewise, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance and testing, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. Further, in order to assess the structural integrity of the reactor vessel, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires, in part, an appropriate materials surveillance program for the reactor vessel beltline region."

The fracture toughness requirements for the reactor pressure vessel (RPV) for testing and operational conditions are specified in Section IV of 10CFR50, Appendix G. The updated version of Appendix G [2-2] contains a table which summarizes the pressure and temperature requirements for the reactor pressure vessel. This appendix requires implementation of the acceptance and performance criteria of Appendix G [2-5, 2-6]of the ASME code. The basis for the technical requirements of the ASME code are discussed in Reference [2-3]. Appendix G to 10CFR50 requires that the effects of neutron irradiation on the RT_{NDT} of the beltline materials must be included in the P-T curve calculations. The guidance provided in the latest revision to Regulatory Guide 1.99 may be used for this purpose.

The calculations performed for NMP-2 fully satisfy the requirements of 10CFR50 Appendix G and Appendix G to Section XI/Section III of the ASME Code. In addition, the application of the Code Case N-640 under Section XI (now incorporated into Section XI) of the Code is consistent with the use of the Code Case under Section III. Further details are provided in the report sections which follow. The two key models used in the calculation, the ΔRT_{NDT} model and the P-T limit model, are briefly summarized below.

2.1 RT_{NDT} Shift Determination

Neutron damage within the RPV during plant operation is accounted for in the allowable pressure loading by calculating an adjusted reference temperature (ART). Regulatory Guide 1.99, Revision 2 [2-4] (RG1.99(2)) defines the ART as the sum of the initial unirradiated nil-ductility reference temperature (RT_{NDT}), plus the RT_{NDT} irradiation induced shift (ΔRT_{NDT}), plus a margin term. Within the nuclear industry, the ΔRT_{NDT} is determined from the Charpy transition curve shift indexed at 30 ft-lbs of absorbed energy. The ART for the vessel beltline region enters the P-T calculations directly via the ASME reference stress intensity factor relation (K_{IR}, K_{Ia} , K_{Ic}). Historically, the K_{IR} has been used to represent the lower bound to static and dynamic data and this lower bound in dominated by K_{Ia} . The Code Case N-640 allows for the replacement of the K_{IR}/K_{Ia} curve by the static curve (K_{Ic}).

It is necessary to provide reasonable and conservative estimates of the shift in nil-ductility reference temperature for the period of time over which the P-T curves will be used. The ΔRT_{NDT} for NMP-2 was calculated using the guidance given in Revision 2 to Regulatory Guide 1.99 [2-4]. The functional form for the RG1.99(2) model is as follows:

$$\Delta RT_{NDT} = (CF)\ f^{(0.28-0.1 \log f)}$$

where,

CF	=	chemistry factor: based on Cu and Ni content in the RG1.99(2) Tables, or on the fitted surveillance data under Regulatory Position 2.1 when two or more credible surveillance data points are available
f	=	fast fluence ($E > 1$ MeV) in units of 10^{19} n/cm^2

As discussed in Reference [1-3], the 3-degree surveillance capsule data are credible. However, since there is only one credible surveillance data point for NMP-2, RG1.99(2) Position 1.1 has been used to determine the limiting beltline material. The result of this analysis was to show that plate C3147 is limiting at the 1/4 T and plate C3065-2 is limiting at the 3/4 T. Therefore, the final P-T curves are lower bound limits based on these two beltline materials. Further details are provided in Section 4.0.

2.2 Pressure-Temperature (P-T) Curve Development

Appendix G [2-5, 2-6] of the ASME Boiler and Pressure Vessel Code presents a procedure for obtaining the allowable loadings for ferritic pressure-retaining materials in Class 1 components. This procedure is based on the principles of linear elastic fracture mechanics. The calculative method used to determine the NMP-2 P-T curves satisfies the requirements of the ASME procedure.

The model uses the following governing relation for calculation of heatup and cooldown curves for the reactor vessel:

$$K_{IC} > 2\,K_{IM} + K_{IT} \qquad (2\text{-}1)$$

The Code requires that a semi-elliptical, axially oriented 1/4 thickness (T) reference flaw be postulated at the inside (1/4 T) and the outside (3/4 T) surfaces of the vessel to calculate the applied stress intensity factors. As a result of this assumption, equation (2-1) can be re-written as follows:

$$K_{IC} > 2\,M_m\,\sigma + M_t\,\Delta T_{max} \qquad (2\text{-}2)$$

where,

σ	=	vessel hoop stress (ksi)
K_{IT}	=	stress intensity factor produced by a radial thermal gradient across the wall (ksi$\sqrt{in}$)
K_{IM}	=	stress intensity factor corresponding to membrane tension (ksi$\sqrt{in}$)
K_{IC}	=	stress intensity factor curve for static testing (ksi$\sqrt{in}$)
M_m	=	stress intensity index for membrane stress ($\sqrt{in}$)
M_t	=	stress intensity index for thermal stress (ksi$\sqrt{in}$ /F)
ΔT_{max}	=	temperature difference through the vessel wall during heatup and cooldown (F).

The vessel hoop stress is calculated using the finite wall thickness equation. This approach is conservative relative to the thin wall equation specified in Appendix G to Section XI. Therefore, the membrane stress is calculated in accordance with the requirements of Appendix G to Section III and Appendix G to Section XI.

The static stress intensity factor curve, K_{IC}, is calculated using the relationship given in Appendices A and G of Section XI of the Code and is implemented under Code Case N-640:

$$K_{IC} = 33.2 + 20.734 \exp(0.02\,(T-RT_{NDT})) \qquad (2\text{-}3)$$

where,

T = vessel metal temperature (F)

RT_{NDT} = nil-ductility reference temperature of the limiting RPV material.

(Note: Equation 2-3 is functionally equivalent to the Appendix A, Section XI equation, which was listed in earlier versions of the Code. The Appendix A equation is given by: $K_{IC} = 33.2 + 2.806 \exp(0.02\,(T-RT_{NDT} + 100))$

This analytical representation for K_{IC} is based on the lower bound of static critical stress intensity values measured as a function of temperature for specimens of SA533B and SA508 steel.

For leak/hydro test, the Code allows reduction of the membrane stress intensity safety factor from 2 to 1.5. The thermal stress intensity factor may be eliminated by ensuring the thermal gradient is negligible prior to pressurization for testing. The leak/hydro equation for the case where the core is not critical is therefore given by:

$$K_{IC} > 1.5\ K_{IM} \tag{2-4}$$

Use of equation 2-4 requires a thermal soak prior to pressurization to the leak/hydro test pressure. An alternative approach is to build the thermal loading into the leak/hydro test curve. In this case, the governing equation is:

$$K_{IC} > 1.5\ K_{IM} + K_{IT} \tag{2-5}$$

In equation 2-5, K_{IT} is defined by the maximum vessel thermal gradient for the heating rate specified. In the case of NMP-2, a non-critical heating rate of less than 20 F/hour has been specified. Therefore, the governing equations and safety factors specified in Appendix G to Section III are the same as those in Section XI.

2.2.1 Limiting Stress Intensity Factor Conditions

For the case of vessel heatup, two conditions are analyzed: the stresses at the 1/4 thickness (1/4 T) location; and the 3/4 thickness (3/4 T) location. The conditions analyzed meet the requirements of Appendix G to Section III and Appendix G to Section XI.

Heatup

At the 1/4 T position, the thermal stresses on heatup are compressive and the membrane stresses are tensile. Therefore, the most highly stressed condition is when the thermal stresses equal zero at an isothermal condition. Thus, the hypothetical case of an isothermal heatup, 0 F/hr, is considered and applied to the heatup curves for conservatism.

For a postulated outside surface flaw with a crack tip at the 3/4 T position, the thermal stresses and membrane stresses are tensile and therefore additive. As a result, the maximum thermal stresses for a particular heating rate are superimposed on the pressure stresses in order to develop conservative heatup curves. Therefore, at the 3/4 T position, a total of 6 cases are considered: 0 F/hr; 20 F/hr; 40 F/hr; 60 F/hr; 80 F/hr; and 100 F/hr. The most limiting of the 1/4 T and 3/4 T (i.e., inside or outside) conditions is used to form the heatup curve.

Cooldown

For the case of cooldown, the pressure calculations need only be performed for an inside surface flaw at the 1/4 T location since the membrane and thermal stresses are tensile and additive. The 3/4 T location (i.e., outside surface) will always be stressed to a lesser or equal value and thus need not be considered. Therefore, at the 1/4 T position, a total of six cases are considered for cooldown: 0 F/hr; 20 F/hr; 40 F/hr; 60 F/hr; 80 F/hr; and 100 F/hr.

Leak/Hydro Test

For leakage and hydrostatic testing, the ASME rules allow elimination of the thermal gradient term provided that the vessel will be thermally equilibrated at the time the test pressure is applied. Therefore, the limiting loading condition is the membrane stress at the 1/4 T (i.e., inside surface) position. In the current P-T analysis, the thermal stresses were included in the leak test curve so that the thermal soak would not have to be performed after the test temperature is reached. A maximum ramp of 20 F/hr was analyzed and the thermal stresses were conservatively added at the 3/4 T position.

2.2.2 Thermal Analysis

The temperature gradients in the pressure vessel wall at several heating and cooling rates are determined by performing transient thermal analyses using the Livermore multi-dimensional transient temperature distribution code TRUMP [2-7]. The transient conditions result in a temperature difference through the vessel wall and a temperature difference from the inner diameter (ID) surface to the 1/4 T and 3/4 T positions. Thermal conductivity data for American Society for Testing and Materials (ASTM) A533 Grade B low alloy steel is taken from Appendix I of Section III of the ASME Code [2-8], and specific heat data are obtained from the thermophysical properties of matter data series (TRPC) Data Series [2-9].

2.2.3 Stress Intensity Indices

The ASME stress intensity indices (M_m and M_t) are calculated using the procedures of Appendix G of the ASME Code [2-5, 2-6]. Use of the thermal stress intensity index, M_t, provided in Appendix G of the ASME Code is appropriate provided the temperature change starts at a steady state condition, has a rate of change of less than 100 F/hr, and the shape of the thermal gradient is approximately as given in Figure G-2214-3 of Appendix G to Section III (Figure G-2214-2 of Appendix G to Section XI). The former conditions were satisfied as described above and the latter condition was satisfied by comparison of the thermal transient output with the ASME thermal gradient profile [2-10]. The M_m used in the calculations was determined using Figure G-2214-1of Appendix G to Section III and is conservative as compared to the Section XI value. The M_m used is 2.3602 for plate C3147 and 2.3573 for plate C3065-2. Under Appendix G for Section XI, M_m for a 6.1875 inch wall is 2.3034 for the postulated axial defect. Therefore, the model satisfies the Section III and Section XI Appendix G requirements.

2.2.4 Allowable Coolant Temperature and Pressure

It is essential that the pressure and temperature variables plotted be consistent with the readings taken in the control room during plant operation. Based on discussions with plant personnel, the most useful plot is reactor vessel top dome pressure versus beltline downcomer water temperature. During plant operation, the temperature is measured in the recirculation pump suction line. This temperature is conservative as compared with the beltline downcomer water temperature since the actual reactor beltline temperature will be higher due to gamma heating effects and coolant flow frictional heating. Also, the instrument error must be included in a conservative manner. Therefore, we have:

$$T_C = T_{MEASURED} - T_{ERROR} \tag{2-6}$$

where,

T_C	=	downcomer coolant temperature
$T_{MEASURED}$	=	coolant temperature measured in the recirculation pump suction line
T_{ERROR}	=	temperature measurement instrument error

The pressure, since it is measured in the vessel steam region, must be corrected to the pressure at the bottom of the beltline region. This can be done by adding the head of water from the top of the vessel to the bottom of the downcomer. Thus the pressure drop is conservatively estimated so that the pressure can be measured in the steam region of the vessel. Also, the pressure measurement error must be included in a conservative manner:

$$P_{RV} = P_{MEASURED} + P_{HEAD} + P_{ERROR} \tag{2-7}$$

where,

P_{RV}	=	core pressure at the bottom of the beltline region
$P_{MEASURED}$	=	the core pressure measured in the top dome
P_{HEAD}	=	head of water from the top of the vessel to the bottom of the beltline region
P_{ERROR}	=	pressure measurement instrument error.

2.3 Chapter 2 References

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[2-3] Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials," August 1972.

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[2-5] ASME Boiler and Pressure Vessel Code, Section III, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Nonductile Failure,".

[2-6] ASME Boiler and Pressure Vessel Code, Section XI, Appendix G for Nuclear Power Plant Components, Division 1, Article G-2000, "Vessels,".

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Temperature Distributions in Multidimensional Systems," Lawrence Radiation Laboratory, Livermore, Report UCRL-14754, Revision 2 (1968).

[2-8] ASME Boiler and Pressure Vessel Code, Section III, Appendix I.

[2-9] Touloukian, Y. S., Power, R. W., Ho, C. Y., and Klemens, P. G, The TPRC Data Series, Thermalphysical Properties of Matter, Volume 1, Thermal Conductivity, Metallic Elements and Alloys, Purdue Research Foundation, June 1972.

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3.0 Neutron Fluence Calculation

3.1 Introduction

The neutron exposure of reactor structures is determined by a neutron transport calculation, or a combination of neutron transport calculations, to represent the distribution of neutron flux in three dimensions. The calculation determines the distribution of neutrons of all energies from their source from fission in the core region to their eventual absorption or leakage from the system. The calculation uses a model of the reactor geometry that includes the significant structures and geometrical details necessary to define the neutron environment at locations of interest.

A previous set of calculations was carried out for NMP-2 to determine the vessel and shroud exposure [1-3, 3-1]. The details regarding the transport model, analysis procedures, and results will only be briefly reviewed here. A full description of the transport work for NMP-2 is provided in References [1-3] and [3-1]. The results presented here are only for the key input needed for the P-T analysis. Data on the spatial flux distribution, the dosimetry analysis results, and the methodology benchmarking are presented in the referenced reports.

3.2 Neutron Transport Model

The transport calculations for NMP-2 were carried out in R-θ and R-Z geometry using the DORT two-dimensional discrete ordinates code [3-2] and the BUGLE-96 cross-section library [3-3]. The DORT code is an update of the DOT code which has been in use for this type of problem for many years. The BUGLE-96 library is a 47 energy group ENDF/B-VI based data set produced specifically for light water reactor applications (an update of the earlier SAILOR library). This library contains cross-sections collapsed using a BWR core spectrum which were used for the core region. Outside the core region, cross sections collapsed using PWR downcomer and PWR vessel spectra were used. The difference between BWR and PWR collapsing in these regions is not significant. In these analyses, anisotropic scattering was treated with a P3 expansion of the scattering cross-sections, and the angular discretization was modeled with an S8 order of angular quadrature. These procedures are in accordance with ASTM Standard E-482 [3-4].

R-θ Calculations

All structures outside the core were modeled with a cylindrical symmetry except for the inclusion of a surveillance capsule centered at 3° and jet pump structures located in the downcomer region. The jet pumps are only approximate models of two pumps with a central pipe (riser) in between. These structures were modeled as 2 slabs of stainless steel each centered at a radius of approximately 112.28 inches [3-5]. The slabs representing the pumps are at about 22 and 36.5 degrees, and the riser is at about 29.3 degrees. The slabs extend over approximately 3.85 degrees and have a thickness of 0.477 inches. The pipe slab extends over about 4.25 degrees and is 0.523 inches thick.

The R-θ model included 186 mesh points in the radial direction covering the range from the center of the core to ten inches into the biological shield. This large number of mesh points was used to accurately calculate the neutron flux transport from the core edge to the outside of the vessel. In the azimuthal direction, 48 mesh points were used to model a single octant of the reactor. Inspection of the fuel loading patterns indicated that only minor deviations from an octant symmetry were present and these were ignored. The 48 points provided good definition of the variation of the core edge with angle and defined the azimuthal flux variation.

The core region used a homogenized material distribution which includes the fuel, fuel cladding, and the water. The water region in the fuel contains both liquid water and steam. The fraction occupied by steam (known as the void fraction) varies by assembly and axial position within the fuel. Values of void fraction for each cycle at the middle of the cycle (moc), and at the additional times during the cycle for cycle 7, were supplied by Constellation for each assembly at 25 axial nodes [3-6, 3-7, 3-8]. Inspection of these values indicated that while some assemblies exhibit significant variation in the void fraction, some groups of neighboring assemblies had close to the same void fraction. To model the void fraction variation in the R-θ model, the outer rows of assemblies were divided into seven regions of approximately uniform water material density, and the average water density for the assemblies in each of these regions was calculated by multiplying the base water density (0.7365 g/cc) by 1.0 minus the void fraction. The inner assemblies were assigned to an eighth region and the core average void fraction was used for this region. Each one of these regions had a void fraction assigned as the average midplane void fraction value for the assemblies in the region. These average void fraction values were different for each case analyzed.

Water density in the bypass region was varied between 0.7585 g/cc at the inlet and 0.7394 g/cc at the outlet [3-5]. The value at midplane was taken to be an average of these values. The downcomer water density was calculated for a temperature of 534 F and a pressure of 1037 psia.

The DOTSOR code (available as part of the LEPRICON code package [3-9]), was used to convert the cycle power distributions from x,y to R,θ coordinates and to place the source in each mesh cell. The source per group was defined by an average fission spectrum calculated for a fission breakdown by isotope determined for the average burnup of the outer assemblies for each case. The main isotopes that contribute to the fission spectrum are U-235 and Pu-239, but contributions from U-238, Pu-240, and Pu-241 were also included. This is a good approximation to the fission spectrum because the outer assemblies were all burned assemblies with similar burnup, and the fission spectrum only slowly varies with burnup. Almost all of the neutrons that reach the capsule and vessel originate in the outer rows of fuel bundles.

The source calculations used the appropriate power distribution for all the fuel bundles in the first octant together with pin power distributions for the outer rows of bundles. The pin power distributions were used to model the spatial variation of the source within the bundles and took into account the gaps between bundles and water rods in the center. Equal pin power weighting was used for interior fuel bundles. he variation in relative pin power distributions within similar bundles between cycles was determined to be small [3-1], and so the cycle 7 9x9 moc pin power distributions were used in the calculations for all the cases.

The ORIGEN 2.1 code [3-10] was used to calculate the effects of burnup on the neutron source. The Casmo-Simulate data were not available at the time the original analyses were done. The calculations were carried out using an ORIGEN BWR cross section library appropriate for high burnup fuel. The initial fuel composition for each cycle was taken to be the average initial composition for the outer assemblies. The effects of the varying axial initial enrichment, burnup, and void fraction were ignored in this calculation and are assumed to have negligible impact because the effects of the change in parameters are minor. The ORIGEN code calculated the fission fraction by isotope and the average energy deposited in the reactor per fission (κ). The isotopic fission fractions were used to determine the fission spectrum and the average number of neutrons per fission (ν). The normalization of the neutron source in the DORT calculations is directly proportional to ν/κ which slowly varies with burnup.

The ORIGEN results were validated by comparison to NMP-2 calculated fuel compositions as a function of fuel burnup [3-11] and by comparison of fluence results for NMP-1 obtained using ORIGEN with those obtained using isotopic compositions from Casmo-Simulate. NMP-1 cycle 9 was calculated using fission fractions determined both by Casmo-Simulate and by ORIGEN. For this case, the fuel in the outer two rows of bundles had an initial average enrichment of 2.77% and an average burnup in the outer rows of about 22000 MWd/MTU. The ORIGEN calculation resulted in a higher fraction of fissions in Pu for this burnup and thus the normalization of the transport runs for the ORIGEN derived neutron source was higher by about 0.6%. Results for the two calculations were evaluated by plotting the neutron flux above 1 MeV at the vessel inner radius at the maximum axial elevation. It was observed that the curves are in close agreement, with the ORIGEN source producing a calculated fluence rate that is uniformly higher by 1.3%. Thus the effect of the difference in neutron spectrum is about 0.7%. Further details are provided in Reference [3-12].

R-Z Calculations

A second set of transport calculations were performed for each case in R-Z geometry. For this calculation, the core was divided into 3 radial regions. Two of these regions consisted of each of the outer two rows of assemblies averaged over the octant. The third region consisted of the inner part of the core. The neutron source in each of these regions was calculated using a radial source averaged over the octant (calculated by DOTSOR as for the R,θ case) together with an average axial power shape for each region. The axial power distribution was supplied for each assembly in 25 nodes, each representing 6 inches of core height. Neutron source outside the equivalent core radius was eliminated.

Each radial region was also divided into axial regions according to variation in void fraction. The void fraction was also given for each assembly in 25 axial nodes. Except for nodes near the bottom of the core which had zero void fraction, each node was modeled as a separate region for the calculation. This resulted in a total of 70 regions in the core, each with a distinct cross section set. In addition, the GE11 fuel bundles contain 8 part length fuel pins that end at 96 inches above the bottom of the active fuel (BAF). The volume of these pins was replaced with water at axial meshes above the 96 inch level. The bypass region was also modeled with a varying axial water density. The bypass region was divided into 12 subregions within the core height, each with a different water density.

For the R-Z model, the core radius was taken to be that which gave the equivalent core volume. Regions above and below the core were not modeled exactly but consisted of a one-foot high water reflector with vacuum boundaries at the top and bottom of the model. The model had 186 mesh points in the radial direction as in the R-θ model except with slightly different boundaries near the core edge. In the axial direction, the model had 68 mesh points with 38 in the core region.

Flux Synthesis

As indicated above, the calculations were carried out in 2 dimensions. In order to estimate the fluence rate in the 3 dimensional geometry, the following equation was used to evaluate the flux, ϕ, for each cycle case:

$$\phi\,(R,\theta,Z) = \phi\,(R,\theta) * \phi\,(R,Z) \,/\, \phi\,(R)\,.$$

In this equation, $\phi\,(R,\theta)$ is taken from the DORT R, θ calculation (normalized to the power at midplane in the model region), and $\phi\,(R,Z)$ is from the R,Z calculation normalized to the power in the entire core. A third calculation determined $\phi\,(R)$ using a one-dimensional cylindrical model normalized at core midplane. The model for the one-dimensional calculation used the same radial geometry as the R,Z calculation.

Power and Void Fraction Representation

The fluence estimates were based on neutron transport calculations performed using fuel power and void fraction distributions taken at the midpoint of cycles 1 through 6, and at five representative times during cycle 7. The detailed evaluation of the variation in flux level due to changes in fission distributions and void fraction distributions during cycle 7 was made to allow for accurate determination of dosimeter activities from the surveillance capsule that was withdrawn at the end of this cycle. It also provides an indication of the variation in flux level that occurs during a fuel cycle.

During reactor operation, the neutron flux level at any point in the shroud or vessel will vary due to changes in fuel composition, power distributions within the core, and water void fraction. These changes occur between fuel cycles due to changes in fuel loading and fuel design, and within a fuel cycle due to fuel burnup and resultant changes in power shape, control rod position, fission contributions by nuclide, and void fraction vs. axial height in each fuel bundle. In order to ensure that the fuel cycle data input to the model was representative, Constellation performed an analysis of the axial power shapes. For cycles 1 through 6, the core average axial power shape was plotted versus cycle exposure. An exposure-weighted cycle average power shape was calculated based on all of the individual power shapes. Power shapes close to the middle of cycle (moc) were compared with the cycle average shape to determine which shape was representative of the entire cycle. For cycle 7, Constellation once again examined the core average axial power shapes and the shapes were plotted throughout the cycle. Five cases were selected: beginning-of-cycle (boc); before middle-of-cycle (bmoc); middle-of-cycle (moc); after middle-of-cycle (amoc); and near the end-of-cycle (neoc). An axial shape which was most representative of each regime was chosen to represent that segment of cycle exposure.

The power shape analysis approach for selecting power shape inputs, while not highly sophisticated, does result in power shapes that are representative of the fuel cycle (or fuel cycle segment). Power shape throughout a typical cycle's worth of operation has similar characteristics from cycle to cycle. Power starts out being preferentially produced in the bottom of the core via rod pattern manipulation, causing a spectral shift and enhanced Pu production. The Pu produced in the early part of the cycle is beneficial for "squeezing" extra energy out of the core toward the EOC when control blades are not available for power shaping. During MOC, the axial segments of the core burned harder in the early cycle cause the power shape to flatten. As the cycle comes to a close, and rods are nearly fully withdrawn, the power shifts to the top of the core and the reactor is subsequently shut down for refueling as end-of-cycle (eoc) is achieved. These cycle characteristics are repeatable for all cycles which allows one to choose a moc shape as representative of the average over the entire cycle.

3.3 Vessel Fluence Results

The fluence to the reactor vessel was determined from the calculations for each cycle using the flux synthesis. The flux shape was found to vary somewhat from cycle to cycle due to the differences in fuel loading pattern and due to differences in axial power shape and void fraction. Inspection of the azimuthal variation of the fast flux indicated that the maximum value in the vessel occurs at approximately 26°. This is shown in Figure 3-1 which is a plot of the fluence (E > 1 MeV) at the end of cycle 7 at core midplane. The fluence is shown for the vessel IR which is the clad-base metal interface, at the1/4 T position, and at the 3/4 T position.

The peak fluence point varies axially, both during cycles and between cycles. Therefore, the maximum fluence point must be determined by integrating the flux at several axial heights to find the peak value. The maximum fluence point at the end of cycle 7 is at about 30 inches above midplane. This is shown in Figure 3-2 which plots the fluence (E > 1 MeV) at the end of cycle 7 versus axial distance from core midplane for the IR, 1/4 T, and 3/4 T positions. The fluence in this figure is at the maximum azimuth.

Values for the calculated maximum vessel fluence E > 1 Mev, fluence E > 0.1 MeV, and dpa are given in Table 3-1 for the inner radius of the vessel clad, the vessel base metal IR, the 1/4 T position and the 3/4 T position calculated at the end of cycle 7 (8.72 EFPY). Exposure values extrapolated to 22 EFPY are also given in Table 3-1. These have been extrapolated using cycle 7 average flux and dpa/s values since future cycles are projected to be similar to cycle 7. Since the maximum flux point for cycle 7 is slightly closer to axial midplane, the maximum vessel fluence at 22 EFPY was determined by integrating the flux at various axial points and taking the maximum value which was found to occur at 24 inches above midplane. The difference between the maximum value at 24 inches above midplane and 30 inches above midplane at 22 EFPY is only a small fraction of a percent (about 0.3% for fluence (E > 1 MeV) and this difference is not deemed to be significant. The values in Table 3-1 are the calculated maxima and thus the axial position of the fluence values in this table for 8.72 and 22 EFPY are not the same.

Radiation embrittlement effects are usually correlated with fluence E > 1 MeV. However, it is generally thought that dpa might be a better correlation parameter within the vessel wall and, if this is correct, the use of the fluence E > 1 MeV values within the vessel are non-conservative. Accordingly, a dpa attenuation factor is used for fluence determination through the vessel. This can be done using calculated dpa attenuation or using a formulation specified in RG 1.99(2). The fluence values using both these attenuation methods are given in Reference [1-3] for 8.72 and 22 EFPY. The RG1.99(2) and DPA attenuation methods both yield comparable results. The RG1.99(2) result is slightly higher at the 1/4 T position and slightly lower at the 3/4 T position. The RG1.99(2) attenuation was used in the P-T curve calculations.

3.4 Compliance with RG 1.190

The United States Nuclear Regulatory Commission (NRC) has issued RG 1.190 on Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence [3-13]. This guide covers recommended practices for neutron transport calculations and applies to other reactor components in addition to the primary emphasis on the pressure vessel. The MPM methodology fully satisfies all of the RG 1.190 requirements. Details concerning compliance of the MPM model with RG 1.190 are provided in Reference [3-1]. Reference [3-1] also provides results for the uncertainty assessment. Benchmarking of the MPM transport methodology for BWR analyses is reported in Reference [3-14].

3.5 Chapter 3 References

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[3-11] Letter, Shashi K. Dhar to M. P. Manahan, File Code ESB2-M99-043, July 29, 1999.

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[3-13] Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, U. S. Nuclear Regulatory Commission, March 2001.

[3-14] "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations," Report MPM-402781, MPM Technologies, Inc., 2161 Sandy Drive, State College, PA 16803-2283, January, 2003.

Table 3-1 NMP-2 Calculated Maximum Vessel Fluence and dpa at End of Cycle 7 (8.72 EFPY) and at 22 EFPY.

Position	Fluence (E > 1 MeV) n/cm^2	Fluence (E > 0.1 MeV) n/cm^2	dpa
End of Cycle 7 (8.72 EFPY)			
Clad IR	1.98E+17	3.60E+17	3.09E-04
Vessel IR	1.95E+17	3.67E+17	3.04E-04
Vessel 1/4 T	1.31E+17	3.24E+17	2.12E-04
Vessel 3/4 T	4.34E+16	1.72E+17	8.34E-05
After 22 EFPY[a]			
Clad IR	5.71E+17	1.03E+18	8.90E-04
Vessel IR	5.62E+17	1.06E+18	8.74E-04
Vessel 1/4 T	3.76E+17	9.29E+17	6.08E-04
Vessel 3/4 T	1.25E+17	4.86E+17	2.37E-04

a. Extrapolated using maximum values of flux (E > 1 MeV), flux (E > 0.1 MeV), and dpa/s averaged over cycle 7. At the vessel IR these values are 8.78E8 n/cm^2/s, 1.64E9 n/cm^2/s, and 1.36E-12 s^{-1}, respectively. Note that due to a slight shift in the axial position of the maximum flux point, the difference in maximum fluence values between 8.72 and 22 EFPY is not directly proportional to these maximum values but the differences are a small fraction of a percent.

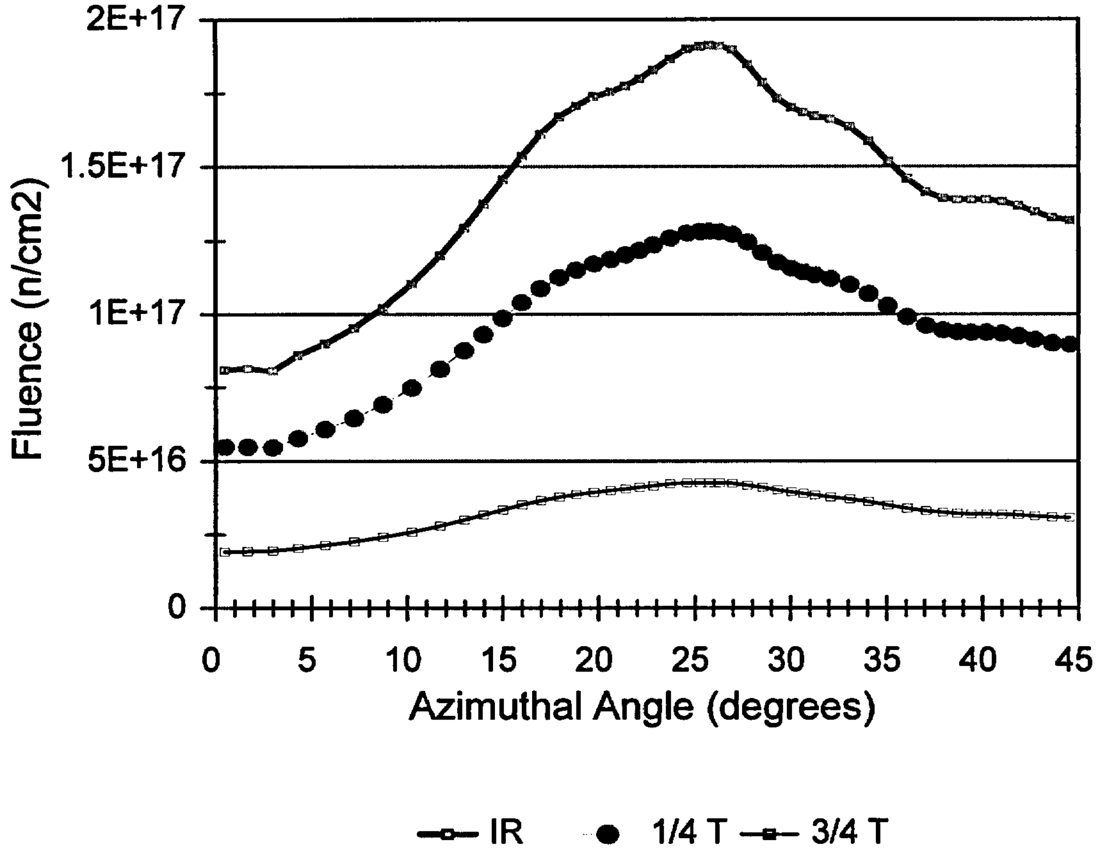


Figure 3-1 Reactor Vessel Fluence at the End of Cycle 7 at Core Midplane.

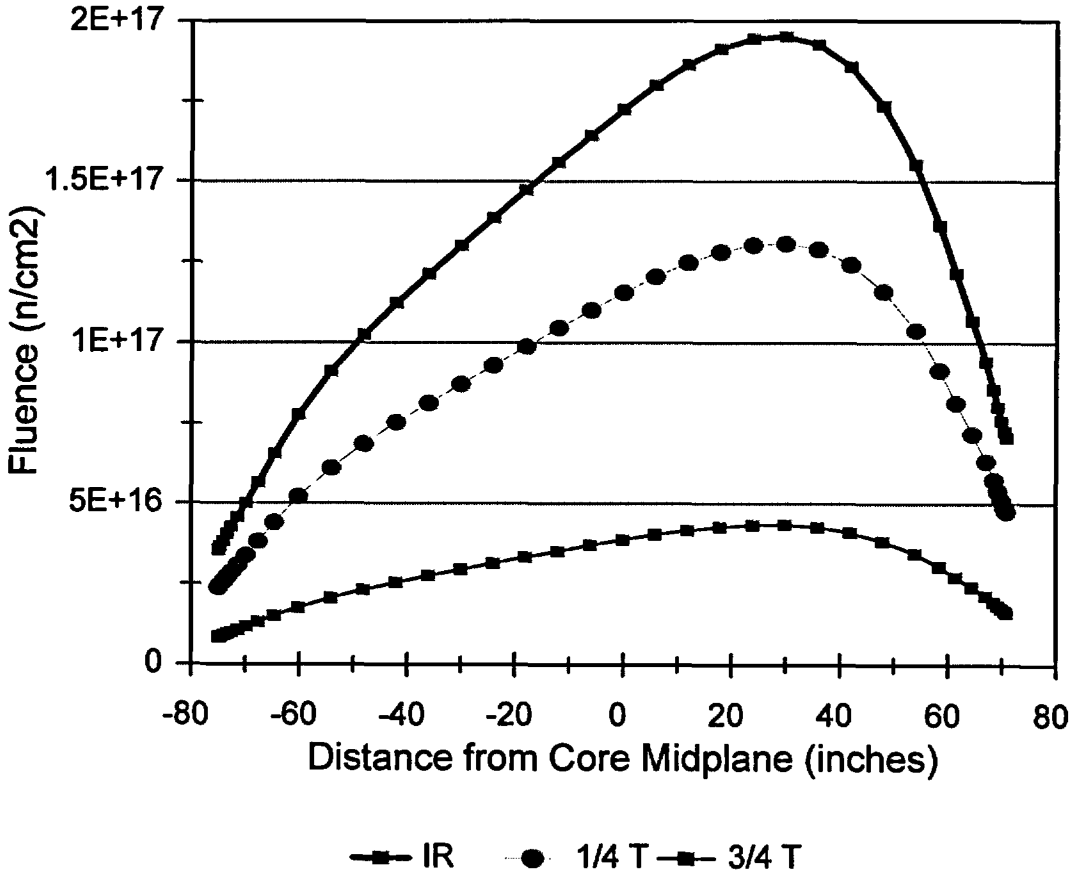


Figure 3-2 Reactor Vessel Fluence at the End of Cycle 7 at Azimuthal Maximum .

4.0 Pressure-Temperature Curve Analysis

P-T curves for up to 22 effective full power years (EFPY) were calculated for NMP-2. This section of the report contains a brief review of the data development necessary to calculate the P-T curves as well as the P-T curves themselves.

4.1 Peak Neutron Fluence Determination

The results of discrete ordinates transport calculations for NMP-2 were reported in Section 3.0. The peak vessel ID (wetted surface) fast (E > 1 MeV) fluence was determined by extrapolation of the Cycle 7 flux. Using the Reference RG1.99(2) attenuation, the 1/4 T fluence and the 3/4 T fluence were calculated to be:

$$\phi t_{1/4\,T} = [5.71x10^{17}\ n/cm^2]\ \exp\,[-0.24\ (1.7344)\,]$$
$$= 3.77x10^{17}\ n/cm^2$$

$$\phi t_{3/4\,T} = [5.71x10^{17}\ n/cm^2]\ \exp\,[-0.24\ (4.828)\,]$$
$$= 1.79x10^{17}\ n/cm^2$$

4.2 Surveillance Data Assessment

RG1.99(2) requires assessment of surveillance data to determine whether the data are credible. Credibility is judged by five criteria given in the regulatory guide. The surveillance capsule credibility analysis results are reported in [1-3] and the 3-degree capsule data are credible. Since only one capsule has been pulled to date, the ART was calculated using Position 1.1.

4.3 ART For Limiting RPV Material

P-T curve calculations are performed using mechanical property data for the RPV material which is limiting over the operating period for which the P-T curves are valid. Using the RG1.99(2) procedure, the beltline plate with the largest ART is limiting for the beltline region. The ART at 22 EFPY is given by:

$$ART = RT_{NDT} + \Delta RT_{NDT} + Margin$$

The mean ΔRT_{NDT} is calculated using:

$$\Delta RT_{NDT} = CF \times FF\ (F)$$

where,

CF = RG1.99(2) chemistry factor (F), either from the Tables or from the surveillance data fitting procedure (Regulatory Position 2.1 can be used when 2 or more credible

		surveillance data points are available)
FF	=	RG1.99(2) fluence factor = $f^{(0.28 - 0.1 \log f)}$
f	=	fast neutron fluence in units of 10^{19} n/cm^2
Margin	=	$2(\sigma_\Delta{}^2 + \sigma_I{}^2)^{0.5}$
σ_Δ	=	28 F for welds and 17 F for plates
σ_I	=	standard deviation for initial RT_{NDT}

The results of the ART calculation at the wetted surface are given in Table 4-1. The ART data presented in this report have been calculated using a σ_I of 14.5 F. The industry position has been that for plants with measured initial RT_{NDT} values, the σ_I term can be taken as zero. Although measured data are available for NMP-2, the σ_I of 14.5 F has been maintained for conservatism. Using the RG1.99(2) model, the beltline plate or weld with the largest ART is limiting. As shown in the Table 4-1, plate C3147 is the limiting beltline material at a fluence of 5.71 E+17 n/cm^2. FSAR Appendix 5A states that the shell course, head, and closure flange materials have RT_{NDT} values below 10 F. In addition, the nozzle forgings have RT_{NDT} values which do not exceed -20 F. Therefore, based on the data given in Table 4-1, and the FSAR Appendix 5A data, it is concluded that the limiting material is plate C3147 material at a fluence of 5.71 E+17 n/cm^2.

Since the ART is determined by summing the initial RT_{NDT}, ΔRT_{NDT}, and Margin, and because the ART for plate C3065-2 is close to that of plate C3147, it is necessary to verify the limiting plate by calculation using the 1/4 T and 3/4 T fluences. These results are shown in Tables 4-2 and 4-3. As shown in the Tables, plate C3147 is limiting at the 1/4 T and plate C3065-2 is limiting at the 3/4 T. Calculations have shown that, with a σ_I of 14.5 F, plate C3065-2 is limiting at the 3/4 T until about 30.8 EFPY. The final P-T curves reported here are lower bound limits conservatively based on these two beltline materials. It is important to note that if σ_I is taken as zero, then plate C3147 is limiting at both the 1/4 T and 3/4 T positions at 22 EFPY. Thus, the P-T curves have been calculated by conservatively defining two limiting plates.

4.4 Thermal Transient Analysis

The temperature gradients in the pressure vessel wall at several heating and cooling rates were determined by transient thermal analyses using the TRUMP [4-1] computer program which has been integrated into the PT Curve Version 2.0 computer program [4-2]. The pressure vessel wall was modeled as a cylinder having an internal radius of 126.6875 inches, a wall thickness of 6.1875 inches, and a clad thickness of 0.1875 inches. Reference [4-3] provides a discussion of the NMP-2 ferritic material thickness specification. The plate was ordered at 6.4375 inches to meet a minimum desired wall thickness of 6.1875 inches. Therefore, the 6.1875 inch thickness was used in the calculations since this thickness leads to more conservative stresses. It is important to note that no structural credit is taken for the cladding. However, the cladding is included in the thermal stress calculations and in the DPA attenuation calculation to ensure conservative results.

The pressure vessel wall was divided into 17 nodal elements. The inner nodal element was thermally coupled with a high thermal coefficient to a boundary node whose temperature change was controlled at the desired heating or cooling rates. The outer surface of the outer nodal element and the ends of all nodal elements were modeled as adiabatic surfaces, i.e., no heat flow through these surfaces. Six heating and cooling rates were evaluated. These included: 0, 20, 40, 60, 80, and 100F per hour. For each of the heating studies, the model was set at a uniform temperature of 30 F and heated up to 540 F. For each of the cooling studies, the model was initially set at a uniform temperature of 540 F and cooled at a constant rate to 70 F. The temperature difference through the RPV wall and the temperature difference between the ID surface and the 1/4 T and 3/4 T positions were calculated every 1 F of the thermal transient ramp for each heatup and cooldown rate.

4.5 Minimum Temperature for Critical Core Operation

Table 1 of 10CFR50 Appendix G summarizes the pressure and temperature requirements for the reactor pressure vessel. A copy of the Appendix G Table 1 is given in Table 4-4. The minimum temperature requirements for non-critical operations are based on the highest reference temperature of the closure flange bolt region that is highly stressed by the bolt preload with, or without, additional margin. For critical operations, the RPV temperature must exceed the larger of the minimum permissible temperature (T_{CRIT}) for the in-service system hydrostatic pressure test or the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload plus margin. Using the ASME reference stress intensity factor relation, the minimum metal temperatures for core critical operation are:

plate C3147	plate C3065-2
$T_{CRIT}\big|_{1/4\,T} = 91F$	$T_{CRIT}\big|_{1/4\,T} = 87F$
$T_{CRIT}\big|_{3/4\,T} = 81F$	$T_{CRIT}\big|_{3/4\,T} = 83F$

During cooldown, the 1/4 T position is limiting because the membrane and tensile stresses are additive and higher than those at the 3/4 T. The coolant temperature (T_c) which corresponds to T_{CRIT} can be calculated as follows:

$$T_c\ (\text{cooldown}) = T_{CRIT}\big|_{1/4\,T} - \Delta T_{1/4\,T} + T_{error}$$

plate C3147	plate C3065-2
T_c (cooldown) = 96 F	T_c (cooldown) = 92 F

During heatup, the 3/4 T limits at 100°F/hr. We have:

$$T_c\ (\text{heatup}) = T_{CRIT}\big|_{3/4\,T} + \Delta T_{3/4\,T} + T_{error}$$

plate C3147	plate C3065-2
T_c (heatup) = 125 F	T_c (heatup) = 127 F

4.6 Summary of Plant Parameters

There are several key plant parameters which are used in P-T curve calculations. These parameters are summarized in Table 4-5. The instrumentation used to measure reactor vessel pressure during plant heatup, cooldown, and leak/hydro testing, and the instrument uncertainties, are documented in [4-3]. For heatup/cooldown and leak testing, the pressure measurement uncertainty used in the P-T calculations is ±20.9 psig and the temperature measurement uncertainty is ±5.0 F. These uncertainties have been included in the P-T curves.

4.7 10CFR50 Appendix G Minimum Temperature Requirements

The 10CFR50 Appendix G minimum temperature requirements (Table 4-4) are stated (para. IV.A.2.c) to "pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature". Furthermore (para. IV.A), for the beltline region, "the effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis." Notice that the requirements are in terms of metal temperatures and not coolant temperatures. Most, if not all, operating nuclear power plants do not have thermocouples attached to the closure flange or vessel wall, much less at the deepest point on the crack front of the flaws assumed in the analysis. Therefore, in order to be able to apply these minimum metal temperature requirements at the plant, the metal temperature requirements must be expressed in terms of the measured coolant temperature and measured pressure. The process of converting the minimum required metal temperatures to minimum measured coolant temperatures must account for any temperature gradients in the vessel due to heating or cooling transients as well as any potential temperature and pressure sensor inaccuracies. In addition, any head of water between the pressure measurement location and the critical region of the vessel must be included in the conversion process. As mentioned previously, for NMP-2, recirculation water temperature is available to the operator in the control room during operation. The recirculation water temperature is a conservative representation of the downcomer water temperature due to gamma heating and flow path heating.

For the beltline region, converting the required minimum metal temperatures to required minimum measured coolant temperature is analytically straightforward because the flaw depth is precisely defined and the temperature differences across the vessel wall are calculated in the thermal transient analysis. However, converting the required closure flange bolt region metal temperature to measured coolant temperature is more difficult because of the more complicated geometry as compared with the beltline region of the vessel shell. Further, 10CFR50 Appendix G does not define a postulated flaw for the closure flange. In order to determine a coolant temperature which corresponds to the closure flange required minimum metal temperature, it is necessary to define a point in the flange or the adjacent shell where the minimum required temperature is to be satisfied. 10CFR50 Appendix G provides no guidance on selecting this critical material location. The Reference [1-1] coolant temperature requirements based on minimum required metal temperatures for the closure flange bolt region were computed based on the critical material being at the wetted surface (PTCurve Version 1.0 model).

In light of the Code Case N-640 approval which allows use of the ASME K_{IC} equation in place of the ASME K_{IR}/K_{Ia} equation, MPM reviewed the P-T curve margins and minimum temperature requirements as part of the PTCurve Version 2.0 preparation. A decision was made to add more conservatism to the 10CFR50 Appendix G Table limits in PTCurve 2.0. Under the new PTCurve Version 2.0 model, the closure flange region critical material location is assumed to be at the same distance from the wetted surface as the 3/4 T position considered in the analysis of the postulated OD beltline region crack. Thus, in PTCurve Version 2.0, the maximum $\Delta T(3/4T)$ for the considered heating ramps, plus the temperature sensor error, are added to the heatup and leak test minimum metal temperature requirements for the closure flange bolt region. During cooldown, the metal temperature is always at or above the coolant temperature, so the $\Delta T(1/4T)$ is still conservatively taken as zero in the PTCurve Version 2.0 model. The net effect of this model change is to increase the 10CFR50 Appendix G related minimum coolant temperature requirements for heatup and leak test.

4.8 P-T Curves

PT Curve, Version 2.0 [4-2] was used to calculate the NMP-2 P-T limits. It was necessary to run PTCurve twice to obtain the limits for plate C3147 which is limiting at the 1/4 T and for plate C3065-2 which is limiting at the 3/4 T. The final P-T limits are the lower bound pressures from the two cases. The results of the analyses are provided in Figures 4-1 through 4-5 and in Tables 4-6 through 4-10.

As described in Section 2.0, heating and cooling rates of up to 100 F per hour have been analyzed to obtain the most limiting conditions for heatup and cooldown operations. Lower bound curves from heating/cooling rates up to 100 F/hr, in conjunction with the specifications of 10CFR50 Appendix G, were used to obtain the P-T curve shown in Figures 4-1 through 4-4. In accordance with the guidance provided in 10CFR50 Appendix G, a total of four P-T curves have been prepared for heatup/cooldown operations which are valid through 22 EFPY including heatup and cooldown for non-critical operations and heatup and cooldown for critical operations. The minimum temperature for boltup is 70 F. If the plant is operated between 70 F and 75 F, only the non-critical heatup and cooldown curves apply. Instrument uncertainties have been included in the P-T curves. Core critical operation is permitted below 312 psig in the shaded region provided the water level is within the normal range for power operation.

The leak/hydro test curve shown in Figure 4-5 was calculated with a thermal loading which corresponds to a 20 F/hr ramp. Therefore, during leak/hydro testing, the non-critical heatup curve should be used to achieve the desired leak/hydro test temperature (114 F at 1035 psig), and then, the leak/hydro test curve may be used to pressurize the vessel to the desired test pressure without the need for a thermal soak. The use of this "no soak" leak/hydro test curve requires that:

- heating to the test temperature did not exceed 20 F/hr, and
- continued coolant heating during the leak/hydro test does not exceed 20 F/hr

After the test is completed, the heatup/cooldown curves must be used for reactor operation. These curves allow heating and cooling rates of up to 100 F/hr.

4.9 Chapter 4 References

[4-1] Edwards, A. L., "TRUMP: A Computer Program for Transient and Steady-State Temperature Distributions in Multidimensional Systems," Lawrence Radiation Laboratory, Livermore, Report UCRL-14754, Revision 2 (1968).

[4-2] PT Curve™ Version 2.0, MPM Technologies, Inc. Code Manual, Document No. MPM-502201, May, 2002

[4-3] Salvagno, A. M., "Review of NMP2 Input Data for P-T Curves", Letter ES01-011, January 16, 2001

[4-4] Manahan, M. P., "Leak/Hydro Test Curve for Nine Mile Point Unit 2", MPM Calculation No. MPM-298725, February 28, 1998

[4-5] RVID2: Source Document NMP2L 1595, November 20, 1995, Generic Letter 92-01, Supplement 1, "Reactor Vessel Structural Integrity", from R. B. Abbott to the NRC

Table 4-1 Analysis of NMP-2 Beltline Materials at 22 EFPY at the Wetted Surface to Identify Limiting ART

Material ID	Wetted Surface Fluence (n/cm^2)	RG1.99 Fluence Factor (FF)	Cu Content (wt %)	Ni Content (wt %)	RG 1.99 Chemistry Factor (CF) (F)	RG 1.99 Source of CF	Initial RT_{NDT} (TL) (F)	ΔRT_{NDT} (F)	Margin (F)	ART (F)
Plate C3065-1	5.71×10^{17}	0.314	0.06	0.63	37.0	Table	-10	11.6	31.2	32.9
Plate C3121-2	5.71×10^{17}	0.314	0.09	0.65	58.0	Table	0	18.2	34.3	52.5
Plate C3147-1	5.71×10^{17}	0.314	0.11	0.63	74.5	Table	0	23.4	37.3	60.7
Plate C3147-2 (1)	5.71×10^{17}	0.314	0.11	0.63	74.5	Table	0	23.4	37.3	60.7
Plate C3066-2	5.71×10^{17}	0.314	0.07	0.64	44.0	Table	-20	13.8	32.1	26.0
Plate C3065-2	5.71×10^{17}	0.314	0.06	0.63	37.0	Table	10	11.6	31.2	52.9
Weld 5P5657/0931 (1,2)	5.71×10^{17}	0.314	0.07	0.71	95.0	Table	-60	29.9	41.6	11.5
Weld 5P5657/0931 (1,3)	5.71×10^{17}	0.314	0.04	0.89	54.0	Table	-60	17.0	33.6	-9.4
Weld 5P6214B/0331(2)	5.71×10^{17}	0.314	0.02	0.82	27.0	Table	-50	8.5	30.2	-11.3
Weld 5P6214B/0331(3)	5.71×10^{17}	0.314	0.014	0.70	22.8	Table	-40	7.2	29.9	-3.0
Weld 4P7465/0751 (2)	5.71×10^{17}	0.314	0.02	0.82	27.0	Table	-60	8.5	30.2	-21.3
Weld 4P7465/0751 (3)	5.71×10^{17}	0.314	0.02	0.80	27.0	Table	-60	8.5	30.2	-21.3
Weld 4P7216/0751 (2)	5.71×10^{17}	0.314	0.045	0.800	61.0	Table	-50	19.2	34.8	3.9
Weld 4P7216/0751 (3)	5.71×10^{17}	0.314	0.035	0.820	47.5	Table	-80	14.9	32.6	-32.5

(1) These materials are also in the surveillance program.
(2) Single wire submerged arc process [4-5].
(3) Tandem wire submerged arc process [4-5].

Table 4-2 Analysis of NMP-2 Beltline Materials at 22 EFPY at the 1/4 T Position to Identify Limiting ART

Material ID	1/4 T Fluence (n/cm^2)	RG1.99 Fluence Factor (FF)	Cu Content (wt %)	Ni Content (wt %)	RG 1.99 Chemistry Factor (CF) (F)	RG 1.99 Source of CF	Initial RT_{NDT} (TL) (F)	ΔRT_{NDT} (F)	Margin (F)	ART (F)
Plate C3065-1	3.77×10^{17}	0.250	0.06	0.63	37.0	Table	-10	9.3	30.4	29.7
Plate C3121-2	3.77×10^{17}	0.250	0.09	0.65	58.0	Table	0	14.5	32.4	46.9
Plate C3147-1	3.77×10^{17}	0.250	0.11	0.63	74.5	Table	0	18.6	34.5	53.1
Plate C3147-2 (1)	3.77×10^{17}	0.250	0.11	0.63	74.5	Table	0	18.6	34.5	53.1
Plate C3066-2	3.77×10^{17}	0.250	0.07	0.64	44.0	Table	-20	11.0	31.0	22.0
Plate C3065-2	3.77×10^{17}	0.250	0.06	0.63	37.0	Table	10	9.3	30.4	49.7
Weld 5P5657/0931 (1,2)	3.77×10^{17}	0.250	0.07	0.71	95.0	Table	-60	23.8	37.5	1.3
Weld 5P5657/0931 (1,3)	3.77×10^{17}	0.250	0.04	0.89	54.0	Table	-60	13.5	32.0	-14.5
Weld 5P6214B/0331(2)	3.77×10^{17}	0.250	0.02	0.82	27.0	Table	-50	6.8	29.8	-13.5
Weld 5P6214B/0331(3)	3.77×10^{17}	0.250	0.014	0.7	22.8	Table	-40	5.7	29.6	-4.7
Weld 4P7465/0751 (2)	3.77×10^{17}	0.250	0.02	0.82	27.0	Table	-60	6.8	29.8	-23.5
Weld 4P7465/0751 (3)	3.77×10^{17}	0.250	0.02	0.8	27.0	Table	-60	6.8	29.8	-23.5
Weld 4P7216/0751 (2)	3.77×10^{17}	0.250	0.045	0.800	61.0	Table	-50	15.3	32.8	-2.0
Weld 4P7216/0751 (3)	3.77×10^{17}	0.250	0.035	0.820	47.5	Table	-80	11.9	31.3	-36.8

(1) These materials are also in the surveillance program.
(2) Single wire submerged arc process [4-5].
(3) Tandem wire submerged arc process [4-5].

Table 4-3 Analysis of NMP-2 Beltline Materials at 22 EFPY at the 3/4 T Position to Identify Limiting ART

Material ID	3/4 T Fluence (n/cm^2)	RG1.99 Fluence Factor (FF)	Cu Content (wt %)	Ni Content (wt %)	RG 1.99 Chemistry Factor (CF) (F)	RG 1.99 Source of CF	Initial RT_{NDT} (TL) (F)	ΔRT_{NDT} (F)	Margin (F)	ART (F)
Plate C3065-1	1.79×10^{17}	0.161	0.06	0.63	37.0	Table	-10	5.9	29.6	25.5
Plate C3121-2	1.79×10^{17}	0.161	0.09	0.65	58.0	Table	0	9.3	30.5	39.8
Plate C3147-1	1.79×10^{17}	0.161	0.11	0.63	74.5	Table	0	12.0	31.4	43.3
Plate C3147-2 (1)	1.79×10^{17}	0.161	0.11	0.63	74.5	Table	0	12.0	31.4	43.3
Plate C3066-2	1.79×10^{17}	0.161	0.07	0.64	44.0	Table	-20	7.1	29.8	16.9
Plate C3065-2	1.79×10^{17}	0.161	0.06	0.63	37.0	Table	10	5.9	29.6	45.5
Weld 5P5657/0931 (1,2)	1.79×10^{17}	0.161	0.07	0.71	95.0	Table	-60	15.3	32.8	-12.0
Weld 5P5657/0931 (1,3)	1.79×10^{17}	0.161	0.04	0.89	54.0	Table	-60	8.7	30.3	-21.1
Weld 5P6214B/0331(2)	1.79×10^{17}	0.161	0.02	0.82	27.0	Table	-50	4.3	29.3	-16.3
Weld 5P6214B/0331(3)	1.79×10^{17}	0.161	0.014	0.7	22.8	Table	-40	3.7	29.2	-7.1
Weld 4P7465/0751 (2)	1.79×10^{17}	0.161	0.02	0.82	27.0	Table	-60	4.3	29.3	-26.3
Weld 4P7465/0751 (3)	1.79×10^{17}	0.161	0.02	0.8	27.0	Table	-60	4.3	29.3	-26.3
Weld 4P7216/0751 (2)	1.79×10^{17}	0.161	0.045	0.800	61.0	Table	-50	9.8	30.6	-9.6
Weld 4P7216/0751 (3)	1.79×10^{17}	0.161	0.035	0.820	47.5	Table	-80	7.6	30.0	-42.4

(1) These materials are also in the surveillance program.
(2) Single wire submerged arc process [4-5].
(3) Tandem wire submerged arc process [4-5].

Table 4-4 10CFR50 Appendix G Pressure and Temperature Requirements for the Reactor Pressure Vessel

TABLE 1—PRESSURE AND TEMPERATURE REQUIREMENTS FOR THE REACTOR PRESSURE VESSEL

Operating condition	Vessel pressure [1]	Requirements for pressure-temperature limits	Minimum temperature requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	(2)
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	(2) +90 °F (6)
1.c No fuel in the vessel (Preservice Hydrotest Only).	ALL	(Not Applicable)	(3) +60 °F
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤20%	ASME Appendix G Limits	(2)
2.b Core not critical	>20%	ASME Appendix G Limits	(2) +120 °F (6)
2.c Core critical	≤20%	ASME Appendix G Limits + 40 °F	Larger of [(4)] or [(2) + 40 °F]
2.d Core critical	>20%	ASME Appendix G Limits + 40 °F	Larger of [(4)] or [(2) + 160 °F]
2.e Core critical for BWR (5)	≤20%	ASME Appendix G Limits + 40 °F	(2) + 60 °F

[1] Percent of the preservice system hydrostatic test pressure.
[2] *The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.*
[3] The highest reference temperature of the vessel.
[4] The minimum permissible temperature for the inservice system hydrostatic pressure test.
[5] For boiling water reactors (BWR) with water level within the normal range for power operation.
[6] Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

Table 4-5 Summary of Key Parameters Used in Calculation of NMP-2 Leak/Hydro Test P-T Curve

Parameter	Value	Reference
radius to vessel base metal/clad interface	126.6875 inches	VPF#3516-213-2 and VPF#3516-214-4
vessel base metal wall thickness	6.1875 inches	VPF #s 3516-193-5, 3516-213-2, 3516-214-4, USAR Table 1.3-1, and [4-3]
vessel clad thickness	0.1875 inches	VPF#3516-213-2, VPF#3516-214-4, and [4-3]
peak vessel wall wetted surface fast fluence (E>1MeV)	5.71×10^{17} n/cm^2	MPM Report MPM-1200676 [1-3]
initial RT_{NDT} for plate C3147 initial RT_{NDT} for plate C3065-2	0 F 10 F	USAR Table 5.3-2
RT_{NDT} of closure head flange region	10 F	USAR Section 5.3.1.5.1.
yield strength of plate C3147 yield strength of plate C3065-2	64.8 ksi 69.4 ksi	Lukens test certificate
average operating temperature of downcomer at full power	534 F	USAR Figure 5.1 and Constellation File Code ES 99-166 [4-3]
system operating pressure	1020 psig	USAR Section 10.1 and NEDC-31994P, Rev. 1, Table 1-2 [4-3]
preoperational system hydrostatic test pressure	1563 psig	USAR Section 5.3.2.1.2. and Vendor Manual, File Sequence N20766 [4-3]
in-service system leak test pressure	1035 psig	[4-3]
in-service system hydro test pressure	N/A	All ASME required testing is performed at nominal system operating pressure [Sa01]
temperature instrument error	5 F	[4-3]
pressure instrument error	20.9 psig	[4-3]
standard deviation for initial RT_{NDT}	14.5 F	[4-4]
head of water from the top dome to the bottom of the active fuel	27.33 psig	VPF #s 3516-193-5 and 3516-213-2 [4-3]

Table 4-6 Minimum Beltline Downcomer Water Temperature for Pressurization During Heat-Up (Core Not Critical) (Heating Rate ≤ 100 F/Hr) for Up o 22 Effective Full Power Years of Operation

REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)[a]
0	70
312	70
312	80
312	90
312	100
312	110
312	120
312	130
312	140
312	150
312	160
312	170
312	174
1395	174

Instrument Uncertainties Have Been Included in this Table

[a] Reactor Vessel Beltline Downcomer Water Temperature is Measured at Recirculation Loop Suction

Table 4-7 Minimum Beltline Downcomer Water Temperature For Pressurization During Cooldown (Core Not Critical) (Cooling Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation

REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)[a]
0	70
312	70
312	80
312	90
312	100
312	110
312	120
312	130
312	135
1253	135

Instrument Uncertainties Have Been Included in this Table

[a] Reactor Vessel Beltline Downcomer Water Temperature is Measured at Recirculation Loop Suction

Table 4-8 Minimum Beltline Downcomer Water Temperature For Pressurization During Heatup (Core Critical) (Heating Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation

REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)[a]
0	75
312	75
312	80
312	90
312	100
312	110
312	120
312	127[b]
312	130
312	140
312	150
312	160
312	170
312	180
312	190
312	200
312	210
312	214
1395	214

Instrument Uncertainties Have Been Included in this Table

[a] Reactor Vessel Beltline Downcomer Water Temperature is Measured at Recirculation Loop Suction

[b] Water Level Must be in Range for Power Operation if Core is Critical Below 127 F

Table 4-9 Minimum Beltline Downcomer Water Temperature For Pressurization During Cooldown (Core Critical) (Cooling Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation

REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)[a]
0	75
312	75
312	80
312	90
312	96[b]
312	100
312	110
312	120
312	130
312	140
312	150
312	160
312	170
312	175
1253	175

Instrument Uncertainties Have Been Included in this Table

[a] Reactor Vessel Beltline Downcomer Water Temperature is Measured at Recirculation Loop Suction

[b] Water Level Must be in Range for Power Operation if Core is Critical Below 96 F

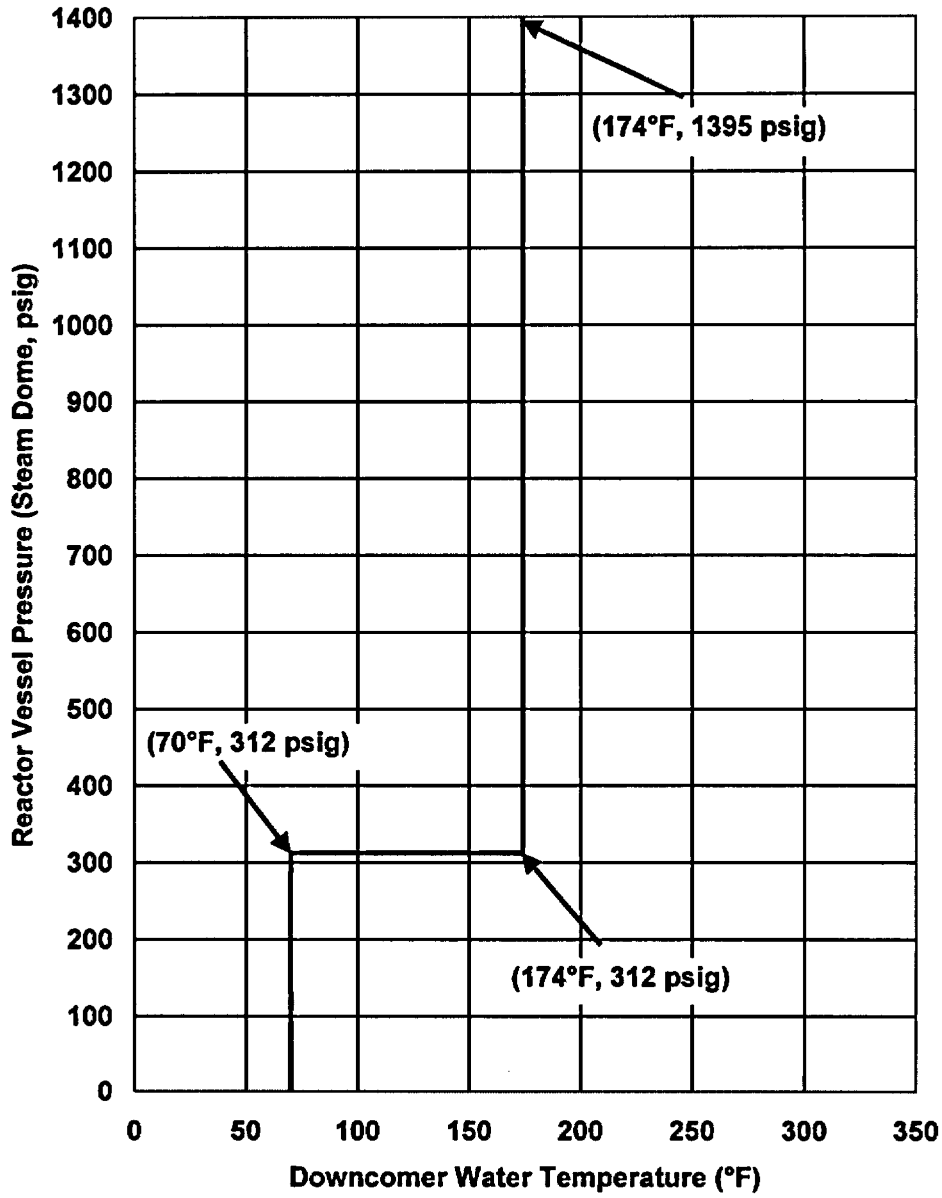
Table 4-10 Minimum Beltline Downcomer Water Temperature For Pressurization During In-Service Hydrostatic Testing and Leak Testing (Core Not Critical) for Up to 22 Effective Full Power Years of Operation

REACTOR PRESSURE (psig) IN TOP DOME	REACTOR VESSEL BELTLINE DOWNCOMER WATER TEMPERATURE (F)[a]
0	70
312	70
312	80
312	90
312	100
312	110
312	114
1202	114 [b]

Instrument Uncertainties Have Been Included in this Table

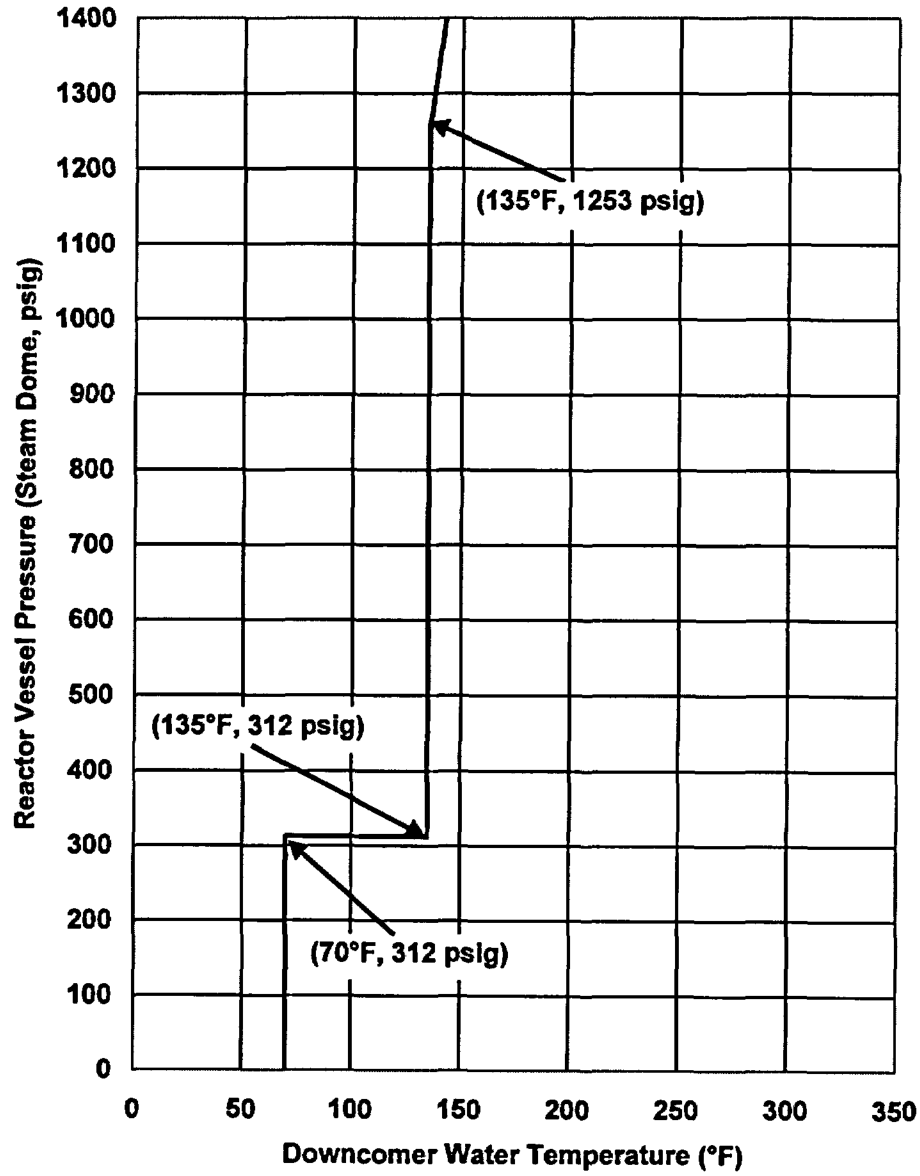
[a] Reactor Vessel Beltline Downcomer Water Temperature is Measured at Recirculation Loop Suction

[b] The Minimum Temperature for Leak Test at 1035 psig is 114 F



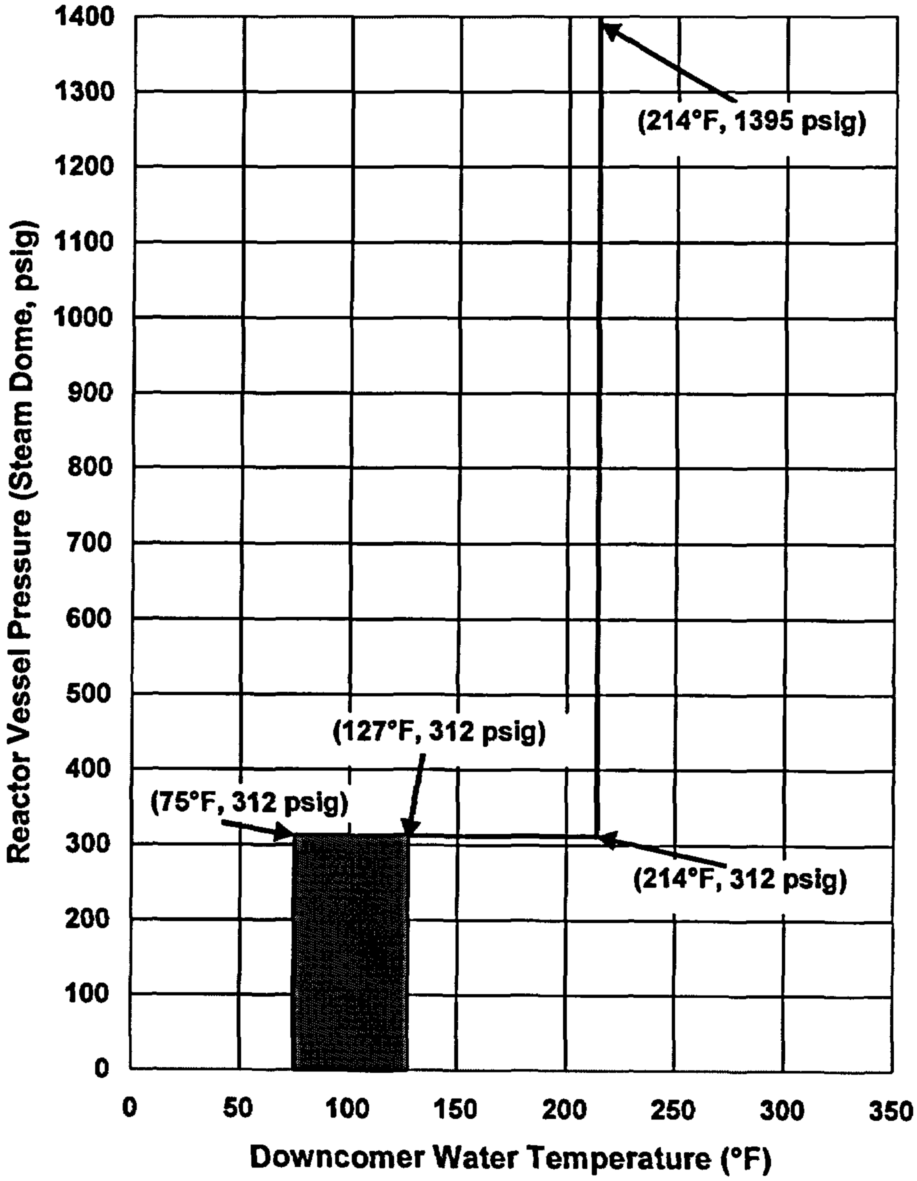
Instrument Uncertainties Have Been Included in this Figure

Figure 4-1 Minimum Beltline Downcomer Water Temperature for Pressurization During Heatup (Core Not Critical) (Heating Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation



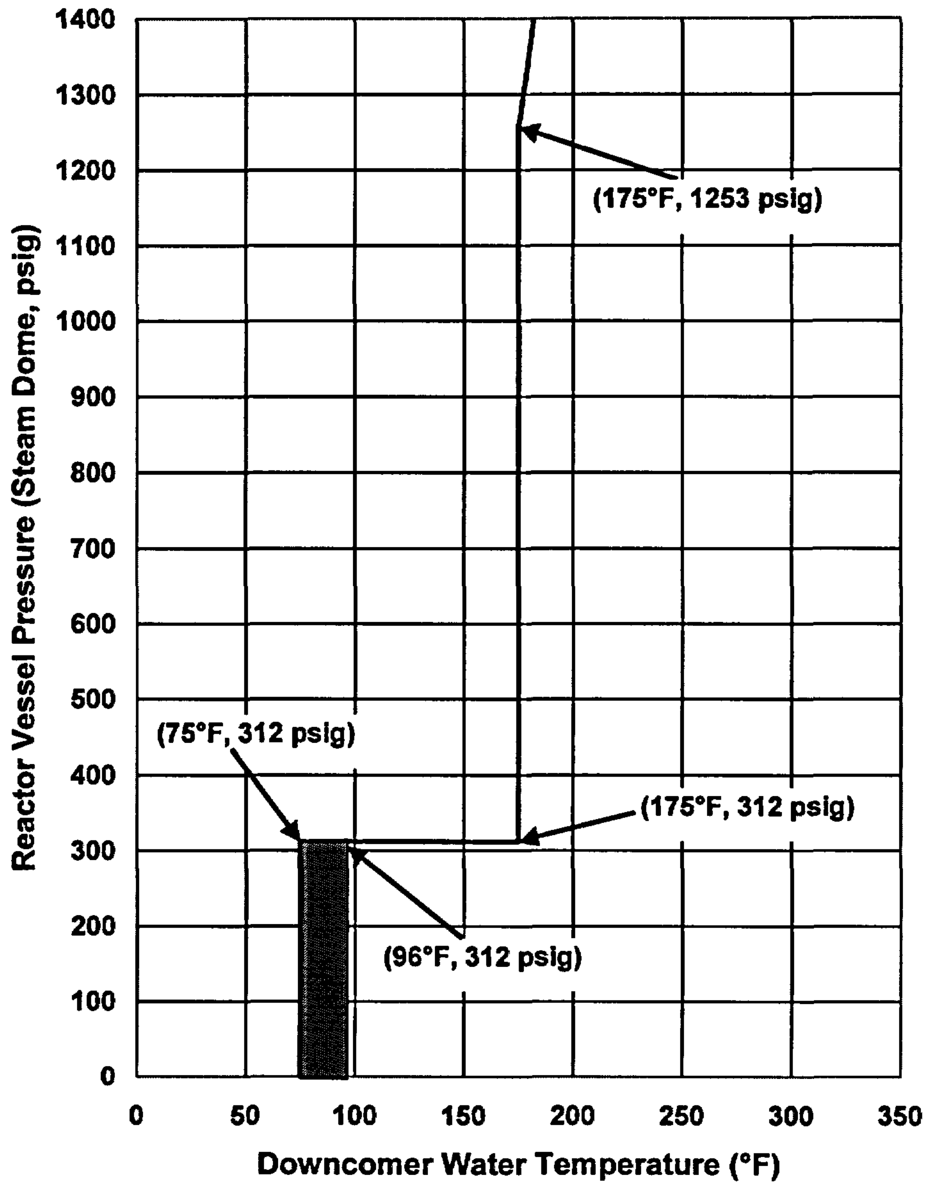
Instrument Uncertainties Have Been Included in this Figure

Figure 4-2 Minimum Beltline Downcomer Water Temperature for Pressurization During Cooldown (Core Not Critical) (Cooling Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation



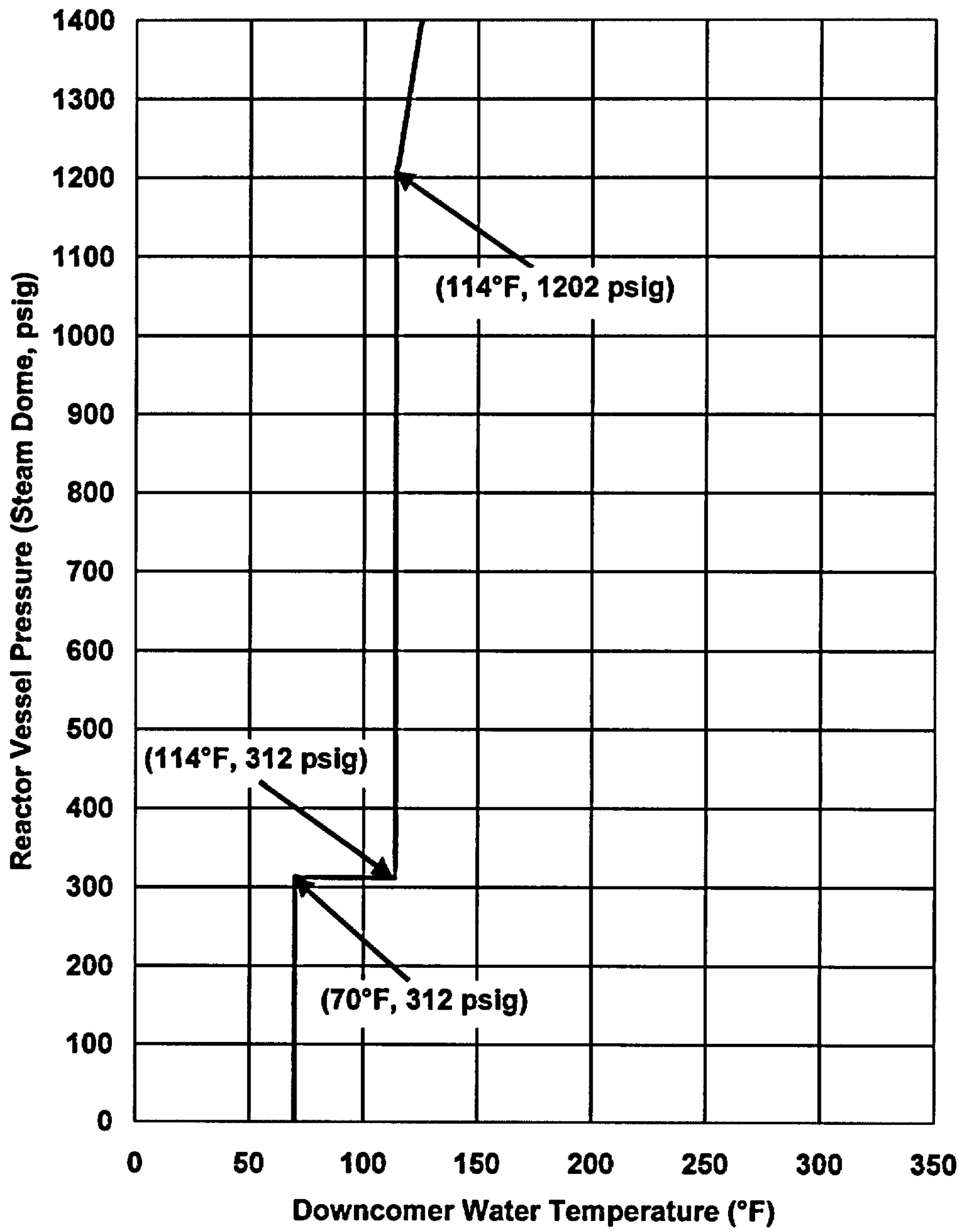
Instrument Uncertainties Have Been Included in this Figure

Figure 4-3 Minimum Beltline Downcomer Water Temperature for Pressurization During Heatup (Core Critical) (Heating Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation



Instrument Uncertainties Have Been Included in this Figure

Figure 4-4 Minimum Beltline Downcomer Water Temperature for Pressurization During Cooldown (Core Critical) (Cooling Rate ≤ 100 F/Hr) for Up to 22 Effective Full Power Years of Operation



Instrument Uncertainties Have Been Included in this Figure

Figure 4-5 Minimum Beltline Downcomer Water Temperature for Pressurization During In-Service Hydrostatic Testing and Leak Testing (Core Not Critical) for Up to 22 Effective Full Power Years of Operation

5.0 Summary

P-T operating curves for NMP-2 have been calculated for up to 22 EFPY of operation. MPM recommends implementation of the updated P-T limits in the near future to ensure NRC approval prior to exceeding the regulatory limit of 12.8 EFPY for the current curves. These new P-T curves satisfy the requirements of 10CFR50, Appendix G and the ASME Code. Operation of NMP-2 in accordance with the revised P-T operating limits will preclude brittle fracture of the RPV materials.

Safety margins for brittle fracture are in accordance with those specified in 10CFR50, Appendix G and Appendix G to Section XI/Section III of the ASME Code. Therefore, the revised P-T limits do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not introduce the possibility of a new or different kind of accident, and do not significantly reduce existing margins of safety.

The P-T limits determined for operation of NMP-2 will be implemented through plant procedures. It is important to note several key points in preparation of the plant procedure which implements the P-T limits.

- The minimum temperature for boltup is 70 F.
- If the plant is operated between 70 F and 75 F, only the non-critical heatup and cooldown curves apply.
- Instrument uncertainties have been included in the P-T curves.
- The P-T curves are valid up to 22 EFPY.
- Core critical operation is permitted below 312 psig in the shaded region provided the water level is within the normal range for power operation.
- The leak test curve includes a thermal loading based on a maximum heatup ramp rate of 20 F/hr. The leak test curve must only be used during pressurization for leak testing. Coolant heating must be less than 20 F/hr during the time that the leak test curve is used.
- During leak testing after a shutdown, use the non-critical heatup curve to reach the leak test temperature (114 F at 1035 psig) using a ramp rate of less than 20 F/hr. When the leak test temperature has been reached, stop the heating and switch to the leak test curve for pressurization to the test pressure. After completion of the leak test, decrease the pressure to within the non-critical heatup or cooldown curve limits, and continue operation using the appropriate heatup or cooldown curve (with heating and cooling rates up to 100 F/hr). The leak test curve must only be used during pressurization for leak testing.

6.0 Nomenclature

amoc	after middle-of-cycle
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BAF	bottom of active fuel
boc	beginning-of-cycle
bmoc	before middle-of-cycle
BWR	boiling water reactor
CFR	Code of Federal Regulations
CF	RG1.99(2) chemistry factor
DBTT	ductile-brittle transition temperature
eoc	end-of-cycle
EFPY	effective full power years
FF	RG1.99(2) fluence factor
F	degrees Fahrenheit
FSAR	Final Safety Analysis Report
ID	inner diameter
IR	inner radius
ITS	Improved Technical Specification
K_{IC}	ASME reference stress intensity factor curve for static testing
K_{Ia}	ASME reference stress intensity factor curve for crack arrest testing
K_{IR}	ASME reference stress intensity factor curve for static, dynamic, and crack arrest testing
M_m	ASME membrane stress intensity index
M_t	ASME thermal stress intensity index
moc	middle-of-cycle
neoc	near end-of-cycle
NMP-2	Nine Mile Point Unit 2
NMPC	Niagara Mohawk Power Corporation
NRC	Nuclear Regulatory Commission
P_o	operating pressure
P-T	pressure-temperature
PR-EDB	Power Reactor Embrittlement Data Base
PTCurve	MPM Technologies, Inc. allowable RPV pressure-temperature code package
RG1.99(2)	Regulatory Guide 1.99 (Revision 2)
RPV	reactor pressure vessel
ΔRT_{NDT}	neutron induced shift in RT_{NDT}
RT_{NDT}	nil-ductility reference temperature
σ_I	standard deviation for the initial RT_{NDT}
σ_Δ	standard deviation for ΔRT_{NDT}
σ_y	material yield strength
T	vessel wall thickness
ΔT_{30}	Charpy curve shift indexed at the 30 ft-lb
T_c	downcomer coolant temperature

T_{CRIT}	minimum temperature for plant operation using nuclear heat with the core critical
TL	transverse-longitudinal fracture specimen orientation
TRPC	thermophysical properties of matter data series
TRUMP	Livermore multi-dimensional transient temperature distribution code
WRC	Welding Research Council