AmerGen SM
AmerGen Energy Company, LLC
Three Mile Island Unit 1
Route 441 South, P.O. Box 480
Middletown, PA 17057
Telephone: 717-944-7621
An Exelon/British Energy Company

10 CFR 50.12
10 CFR 50.60
10 CFR 50.61
10 CFR 50.90

March 29, 2001
5928-01-20035

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

Dear Sir or Madam:

SUBJECT: THREE MILE ISLAND, UNIT 1 (TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
LICENSE AMENDMENT REQUEST NO. 308 –
PRESSURE TEMPERATURE OPERATING CURVES

In accordance with 10 CFR 50.4(b)(1), enclosed is License Amendment Request No. 308.

The purpose of this License Amendment Request is to revise the pressure-temperature (P-T) limits of TMI Unit 1 Technical Specification Section 3.1.2. The proposed amendment will revise the reactor coolant system heatup, cooldown, and inservice leak hydrostatic test limitations for the Reactor Coolant System (RCS) to a maximum of 29 Effective Full Power Years (EFPY) in accordance with 10 CFR 50, Appendix G. Further, the proposed amendment revises TMI Unit 1 Technical Specification Sections 3.1.12 and 4.5.2 for Low Temperature Overpressure Protection (LTOP) requirements to reflect the revised P-T limits of the reactor vessel. These changes rely on NRC approved methodology for determining allowable pressure-temperature limits.

In accordance with 10 CFR 50.61, the fracture toughness requirements for protection against pressurized thermal shock events have been evaluated for reactor operation up to 29 EFPY.

During relatively low temperature RCS operations, the present Technical Specifications require no more than one Reactor Coolant Pump (RCP) be operated per loop. RCP operation at low pressure with either one pump in one RCS loop, or with one RCP in each RCS loop, results in gradual RCP impeller wear from cavitation.

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The proposed Technical Specification changes will permit TMI Unit 1 to be operated during LTOP conditions with two RCPs in operation in a single loop. This change, which has been implemented at all other B&W plants, substantially improves NPSH margin for the RCPs, thereby eliminating impeller cavitation wear.

The NRC has approved similar changes for Oconee Nuclear Station, Units 1, 2 and 3 in Amendment Nos. 307, dated October 1, 1999.

The proposed amendment also includes two exemption requests pursuant to 10 CFR 50.12 from certain requirements of 10 CFR 50.60(a), and one exemption request pursuant to 10 CFR 50.12 from certain requirements of 10 CFR 50.61(a), as described below.

- Code Case N-588 provides procedures for determining reactor vessel pressure and temperature limits derived from postulating a circumferentially oriented reference flaw in a circumferential weld. This code case was approved for use by ASME Section XI on December 12, 1997. The NRC has previously approved the use of Code Case N-588 for Oconee in NRC letter dated July 29, 1999.

- Code Case N-640 provides an alternate method for determining the fracture toughness of reactor vessel materials for use in determining pressure-temperature limits. The code case was approved for use by ASME Section XI on February 26, 1999. The NRC has previously approved the use of Code Case N-640 for Oconee in NRC letter dated July 29, 1999.

- Pressurized Thermal Shock (PTS) evaluation for TMI Unit 1 reactor vessel weld metal WF-70 utilizes the Master Curve approach using the data found in BAW-2202 for determining the initial reference temperature value as an alternative to Paragraph NB-2331 of the ASME Code required by 10 CFR 50.61(a)(5). The NRC has previously approved use of the Master Curve approach for the Zion Nuclear Power Station Units 1 and 2 in NRC letter dated February 22, 1994.

Using the standards in the 10 CFR 50.92, AmerGen Energy Company, LLC (AmerGen) has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this License Amendment Request is provided to the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection, as well as the chief executives of the township and county in which the facility is located.

Approval of this license amendment is requested by August 1, 2001, to support implementation prior to exceeding the existing 17.7 EFPY limits and to enable implementation prior to shutdown for the Cycle 14 refueling outage (September 7, 2001) due to the benefits of reduced cavitation wear on the reactor coolant pump impellers at low temperature operation and facilitation of transition to decay heat removal provided by the proposed changes.

If any additional information is needed, please contact David J. Distel at (610) 765-5517.

Sincerely yours,

Mark E. Warner
Vice President, TMI Unit 1

MEW/djd

Enclosures: 1) TMI Unit 1 License Amendment Request No. 308
Safety Evaluation and No Significant Hazards Consideration
2) Framatome ANP Document I.D. 32-5011638-01, Pressure Temperature
Limits for 29 EFPY for TMI Unit 1
3) Affected TMI Unit 1 Technical Specification Pages

cc: H. J. Miller, USNRC, Regional Administrator, Region I
T. G. Colburn, USNRC, Senior Project Manager, TMI Unit 1
J. D. Orr, USNRC, Senior Resident Inspector, TMI Unit 1
D. Allard, Director, Bureau of Radiation Protection – PA Department
Of Environmental Resources
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
File No. 01021

AMERGEN ENERGY COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 308

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF DAUPHIN)

This License Amendment Request is submitted in support of Licensee's request to change the Technical Specifications for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed marked up pages for the TMI Unit 1 Technical Specifications are also included. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

AmerGen Energy Company, LLC

BY: ______________________
Vice President, TMI Unit 1

Sworn and Subscribed to before me

this 29th day of March 2001.

____
Notary Public

Notarial Seal
Suzanne C. Miklosik, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Nov. 22, 2003
Member, Pennsylvania Association of Notaries

ENCLOSURE 1

TMI Unit 1 License Amendment Request No. 308

Safety Evaluation and No Significant Hazards Consideration

1.0 License Amendment Request No. 308

AmerGen Energy Company, LLC (AmerGen) requests that the following changed replacement pages be inserted into the existing Technical Specification:

Revised Technical Specification Pages: ii, vii, 3-3, 3-4, 3-5, 3-5a (Figure 3.1-1), 3-5b (Figure 3.1-2), 3-18d, 3-18e, 3-18f, and 4-41.

Marked up pages showing the requested changes are provided in Enclosure 3.

2.0 Reason for Change

The proposed amendment revises the pressure-temperature (P-T) limits of Technical Specification Section 3.1.2 for TMI Unit 1. The proposed amendment will revise the heatup, cooldown, and inservice leak hydrostatic test limitations, and the respective heatup and cooldown rates, for the Reactor Coolant System (RCS). The service period for the new P-T limits will be to a maximum of 29 Effective Full Power Years (EFPY) for TMI Unit 1. Technical Specification 3.1.2 Bases is revised to describe the new limits.

Technical Specification Sections 3.1.2.4 and 3.1.2.5 are revised to remove reference to Sections V.B and V.C of Appendix G, as these sections no longer exist in the regulations. This is an administrative change.

The proposed amendment also revises Technical Specification Figures 3.1-1 and 3.1-2 to permit TMI Unit 1 to be operated during low temperature conditions with two Reactor Coolant Pumps (RCP) in operation in a single loop.

The proposed amendment also revises Technical Specification Section 3.1.12 Low Temperature Overpressure Protection (LTOP) setpoints. The RCS Power Operated Relief Valve (PORV) low setpoint is being revised to 552 psig as a result of the P-T limit changes, which is the error adjusted maximum setpoint. The enable temperature for the PORV setpoint (T.S. 3.1.12.2) and the LTOP setpoint (T.S. 3.1.12.1) is revised to 329°F to be consistent with the new P-T bases. The temperature limit for High Pressure Injection (HPI) Pump breaker rack-in and HPI testing (T.S. 3.1.12.3 and 4.5.2.1) is revised to 329°F to be consistent with the new P-T bases. The LTOP setpoint for pressurizer level is revised to 100 inches to be consistent with the new P-T bases. Technical Specification 3.1.12 Bases is revised to describe the new limits and setpoints.

Technical Specification Sections 3.1.12.1, 3.1.12.2, and 3.1.12.3 are also revised to reorganize and clearly delineate the LTOP system protection parameters and applicable conditions. However, the only significant revisions from the existing specifications involve the actual parameter values as described above. Technical Specification Sections 3.1.12.2.a and 3.1.12.2.b are administratively revised to delete reference to nominal setpoint pressure values which do not affect the specified maximum and minimum setpoint values.

Table of Contents Page ii is editorially revised to reflect the title change to Sections 3.1.12 and to correct a previously issued typographical error in the listed titles of Section 3.4.1 and 3.4.2.

Background

Currently, TMI Unit 1 P-T limits have been evaluated for up to 17.7 EFPY. This amendment request provides AmerGen's evaluation of these limits in order to: 1) permit operation of two RCPs in one RCS loop, and 2) extend the evaluation period of the new P-T Limits to 29 EFPY. These changes rely on recently approved methodology for determining allowable P-T limits which is described below in more detail.

During relatively low temperature RCS operations, the present Technical Specifications require no more than one RCP be operated per loop. RCP operation at low pressure with either one pump in one RCS loop, or with one RCP in each RCS loop, results in gradual RCP impeller wear from cavitation.

The proposed Technical Specification changes will permit TMI Unit 1 to be operated during low temperature conditions with two RCPs in operation in a single loop. With both RCPs operating in a loop, flow through each pump is significantly reduced thereby reducing the required NPSH and eliminating cavitation induced impeller wear. Additionally, the change to allow two RCPs per loop facilitates transition to decay heat removal in the cooldown mode.

AmerGen requested Framatome ANP to perform reactor vessel integrity assessments and generate new P-T limit curves for TMI Unit 1 using NRC approved methodologies. These curves have been developed and envelope operation up to 29 EFPY for TMI Unit 1. These assessments, P-T limit curves, and the LTOP analysis and limits proposed in this amendment application satisfy the requirements of 10 CFR 50.60(a) and 10 CFR 50.61 with three exceptions. These exceptions have been reviewed and approved by NRC in other plant licensing applications and involve ASME Boiler and Pressure Vessel Code Cases which have not been included in the latest revisions to NRC Regulatory Guides 1.147, 1.85, and 1.84; and use of the Master Curve approach for Pressurized Thermal Shock (PTS) evaluation of weld WF-70.

These exceptions are as follows:

A. ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels" (Reference 1), provides reactor coolant temperature, pressure and system heatup and cooldown rate limits with margins derived from postulating a circumferentially oriented reference flaw in a circumferential weld. Code Case N-588 is utilized for TMI Unit 1 as described in the following Section 3.3.

B. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1" (Reference 2), provides reactor coolant temperature, pressure and system heatup and cooldown rate limits with margins consistent with the fastest rate of temperature change allowed (K_{Ic} fracture toughness curve in lieu of K_{Ia}). Code Case N-640 is utilized for TMI Unit 1 as described in the following Section 3.3.

C. PTS evaluation for TMI Unit 1 reactor vessel weld metal WF-70 utilizes the Master Curve approach using the data found in BAW-2202 for determining the initial reference temperature value as an alternative to Paragraph NB-2331 of the ASME Code required by 10 CFR 50.61(a)(5). This approach is utilized for TMI Unit 1 as described in the following Section 3.6.

AmerGen requests exemptions to use ASME Code Cases N-588 and N-640 pursuant to 10 CFR 50.60(b), and to utilize the Master Curve approach pursuant to 10 CFR 50.61(a)(5), under 10 CFR 50.12. These exemption requests are provided in Sections 3.7, 3.8, and 3.9 below.

3.0 Safety Evaluation Justifying Change

3.1 Fluence Program

The B&W Owner's Group (BWOG) has conducted the Reactor Vessel Integrity Program for more than ten years with the objective of assuring the continued licensability of the participants' reactor vessels. Reactor Vessel Surveillance Program requirements are defined in 10 CFR 50, Appendix H and BAW-1543A, Revision 2 "Integrated Reactor Vessel Material Surveillance Program," (Reference 3) which addresses the method of compliance with Appendix H. In addition to 10 CFR 50, Appendix H, the NRC's Safety Evaluation Report for BAW-1543A, Revision 2 specifies the need for an integrated surveillance program and requires that each reactor in the integrated program have the capability to monitor reactor vessel neutron fluence. The BWOG cavity dosimetry program is described in Topical Report BAW-1875 (Reference 4), "The B&WOG Cavity Dosimetry Program," which was approved by NRC in June 1986. In Cycle 7, TMI Unit 1 began monitoring vessel fluence using dosimetry installed in the TMI Unit 1 reactor cavity. Cavity dosimetry measurements are used in a well defined process to determine the best estimate fluence in the reactor vessel. While the fluence is determined analytically, the measurements play a crucial role in establishing that the calculations are within the acceptance criteria and in determining the uncertainty in the fluence calculations.

Fluence analyses as part of the reactor vessel surveillance program have three fundamental objectives:

- determine the maximum neutron fluence at the reactor vessel as a function of reactor operation
- predict the reactor vessel neutron fluence in the future, and
- determine the dosimeter activation within the surveillance capsule.

The calculational methodology for predicting the fluence using the cavity dosimetry was validated in the benchmark phase of the cavity dosimetry program. The benchmark consisted of both surveillance capsule and cavity dosimetry comparisons of calculations to dosimetry measurements. The results of these benchmarks are documented in FTI Topical Report BAW-2241P-A (Reference 5). This report was approved by the NRC in a Safety Evaluation dated February 18, 1999. The results demonstrate that the TMI Unit 1 reactor vessel fluence data used to develop the P-T limits are sufficient for safety and licensing evaluations of reactor vessel embrittlement.

BAW-2241P-A, Revision 1, was used to calculate the neutron fluence exposure to the pressure vessel of the TMI Unit 1 nuclear reactor. The fast neutron fluence ($E > 1$ MeV) at the pressure vessel (upper and lower forgings, as well as specific welds) was calculated in accordance with the requirements of the U.S. NRC Draft Regulatory Guide DG-1053, as described in detail in the FTI fluence topical report, BAW-2241P-A.

The energy-dependent flux at the capsule was used to determine the calculated activity of each dosimeter. Neutron transport calculations in two dimensional geometry were used to obtain energy dependent flux distributions throughout the core. Reactor conditions were representative of an average over the Cycle 5 - 6, Cycle 7, Cycle 8, Cycle 8 - 9, and Cycle 10 - 11 irradiation time periods. Geometric detail was selected to explicitly represent the surveillance capsule assembly and the reactor vessel. The calculated activities were adjusted to account for known biases (photo-fission, non-saturation, and short half-life), and compared directly to the measured activities. The measurements were used to show that the calculational results are reasonable and to show that TMI Unit 1 results are consistent with the FTI benchmark database of uncertainties.

Fluence values used for this P-T limit submittal are based on calculations provided in References 6 and 7. These fluence values are provided on enclosed Tables 1 and 2 (Reference 7).

3.2 Determination of Adjusted RT_{NDT} (ART)

The projected 29 EFPY ART values at the ¼ Thickness (¼ T) and ¾ Thickness (¾ T) locations for the beltline regions of the TMI Unit 1 reactor vessel were calculated by Framatome ANP (Reference 7). These calculations were in accordance with Regulatory Guide 1.99, Revision 2. The Regulatory Guide 1.99 credibility criteria are applied by Framatome ANP to determine the appropriate margin term. The calculations determined

the ART for the various reactor vessel (RV) materials using Regulatory Guide 1.99, Revision 2, Regulatory Positions 1.1 and 2.1. The selected controlling values are those RV locations with the highest ART for ¼ T and ¾ T whether determined using Regulatory Position 1.1 or 2.1 methodology. The fluence values for all reactor vessel materials at the limiting location for each material are summarized in Tables 1 and 2. The controlling values were determined to be the upper shell to lower shell circumferential weld (100%) (WF-25).

3.3 Determination of Pressure-Temperature Limits

Introduction

The proposed P-T limits for TMI Unit 1 were developed using FTI computer code PTPC 3.3, as modified by ASME Code Case N-588 for circumferential flaws in welds and by Code Case N-640 for use of the K_{Ic} fracture toughness curve. The methods and criteria employed to establish operating pressure and temperature limits are described in NRC approved Topical Report BAW-10046A (Reference 8). The method of analysis consists of determining the P-T limits for the beltline region, the nozzle region and the closure head region of the reactor vessel for normal heatup, normal cooldown, and inservice leak hydrostatic test.

Technical Justification for Use of Code Case N-640

For TMI Unit 1, ASME Code Case N-640 is utilized in the development of the proposed pressure temperature limits. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The effect of granting this exemption is to change the fracture toughness curve used for development of the P-T curves from K_{Ia} to K_{Ic}. The other margins involved with the ASME XI, Appendix G process of determining P-T limit curves remain unchanged. The unchanged margins are: 1) a flaw which is ¼ vessel thickness in depth and 3/2 the vessel thickness in length, 2) safety factor of two on pressure stress for heatup and cooldown and a safety factor of 1.5 for testing, and 3) upper bound adjusted reference temperature (RT_{NDT}).

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curves is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heatup and cooldown process of a reactor coolant system, with the fastest rate allowed being 100°F per hour. The rate of change of pressure and temperature is virtually constant at low temperatures; therefore, the reactor vessel thermal stress is essentially nil for this transient. During development of Code Case N-640 and the accompanying Appendix G code change, the ASME Section XI, Working Group on Operating Plant Criteria (WGOPC), performed assessments of the margins inherent to K_{Ia} using realistic heatup and cooldown curves. These assessments

led to the conclusion that utilization of the K_{Ia} curve was excessively conservative and the K_{Ic} curve provided adequate margin protection from brittle fracture.

Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. The initial conservatism was necessary due to limited experience and knowledge of the fracture toughness of reactor pressure vessel materials over time and usage. The conservatism also provided margin thought to be necessary to cover uncertainties and a number of postulated but unquantified effects.

Since 1974, additional knowledge has been gained from examination and testing of reactor pressure vessels that had been subject to the effects of neutron embrittlement in both an operating and test environment. The K_{Ia} curve was based on 125 data points. The K_{Ic} curve is based on more than 1500 data points. The additional data has significantly reduced the uncertainties associated with embrittlement effects and reduced other uncertainties. The added data ensures the K_{Ic} curve adequately statistically bounds the data. The new information indicates the lower bound on fracture toughness provided by the K_{Ia} curve is extremely conservative and is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure.

P-T limit curves based on the K_{Ic} methodology will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations. There are two primary safety benefits in opening the lower temperature operating window. The first safety benefit is a reduction in the likelihood of a challenge to RCS power operated relief valve during low temperature operations. The second safety benefit is opening the low temperature pressure window sufficiently to enable normal operation of two RCPs in a single RCS loop. With two RCPs operating in a loop, the required NPSH for the operating pumps is reduced and impeller cavitation wear is eliminated thereby reducing maintenance and radiation exposure. Additionally, the change to allow two RCPs per loop facilitates transition to decay heat removal in the cooldown mode.

<u>Technical Justification for Use of Code Case N-588</u>

For TMI Unit 1 ASME Code Case N-588 is also utilized in the development of the proposed P-T limits. Code Case N-588 provides benefits in terms of calculating P-T limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The ¼T surface reference flaw is a postulated flaw which accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. When considering a reference flaw with respect to a weld, the reference flaw would represent any prior existing defect that may have been introduced during fabrication. Thus, the intended application of a reference flaw is to account for prior existing defects that could physically exist within the

geometry of the weldment. The current ASME Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating P-T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. The possibility an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that axial defects be postulated in plates/forging and axial welds. The fabrication of reactor pressure vessels for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These procedural controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Industry experience with the repair of weld indications found during pre-service inspection, and data taken from destructive examination of actual vessel welds, confirms any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead orientation. Therefore, any potential defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

Due to progress made in NDE techniques over the last thirty years, it is very unlikely to have large, undetected defects present in the beltline region of reactor vessels. It is further unlikely to have axial cracks originating from a circumferential weld perpendicular to the weld seam orientation in reactor vessels. Both experience and engineering studies indicate the primary degradation mechanism affecting the beltline region of the reactor vessel is neutron embrittlement. No other service induced degradation mechanism exists at a pressurized water reactor to cause a prior existing defect located in the beltline region of the reactor vessel to grow while inservice. Based on these considerations, and the fact the P-T limit for reactor operation is the limiting pressure for any of the materials in the vessel, it is not necessary to include additional conservatism in the assumed flaw orientation for circumferential welds. ASME Section XI Code Case N-588 and the accompanying Appendix G Code change corrected this inconsistency in assumed flaw orientation for circumferential welds in vessels when calculating operating P-T limits.

Summary of Assumptions, Analysis, and Results

The analysis has conservatively provided limits that meet the requirements of Appendix G utilizing Code Case N-588 and Code Case N-640 for TMI Unit 1. The proposed Technical Specification P-T limit curves are presented as indicated pressure vs. temperature since the limits include margins for instrument error. The analyses for TMI Unit 1 are summarized in Enclosure 2.

Calculations for the proposed P-T limits utilized the following inputs:

- The operational reactor coolant pump constraints are those listed in the proposed Technical Specifications for various RCS temperature bands and are described in detail in Section 3.4 below and Enclosure 2.

- The following linear heatup ramps which bound the limits of the Technical Specification:

  60 - 570 °F: 50 °F/hr, or 15 °F/18-minute steps

- The cooldown transient was analyzed as a step transient which bounds the Technical Specification and is defined as follows:

570 °F - 255 °F:	100 °F/hr or 15 °F/9-minute steps
255 °F - 70 °F:	30 °F/hr or 15 °F/30-minute steps
At 240 °F:	Decay heat removal system initiated which is modeled as a step change from 240 °F to 200 °F and held at 200 °F for one minute. Following this hold period, a step temperature increase to 235 °F is made. It is assumed that two RCPs in one loop are operating during this transient. This analysis bounds the situation where only one RCP is operating on initiation of decay heat removal.

For a given transient, the maximum allowable pressure as a function of fluid temperature was obtained through a point-by-point comparison of the results at the ¼T and ¾T locations of the limiting beltline weld, the nozzle region and the closure head region. The maximum allowable pressure was taken to be the lowest of the calculated allowable pressures for a given time point. The resulting loci of the points determine the P-T limit curves.

3.4 Justification for Changed Operating Reactor Coolant Pump Combinations

The limits on allowable operating RCP combinations controls the pressure differential between the reactor vessel wall and pressure measurement point and are used as inputs for calculating the heatup, cooldown, and the inservice leak hydrostatic test limit curves. For example, with one reactor coolant pump operating in a loop, the pressure differential between the low range pressure transmitter tap and the actual pressure at the vessel beltline is approximately 20 psi. With two RCPs in the same loop, this pressure

differential is approximately 50 psi. The differential pressure created by the operation of two RCPs is conservatively accounted for in the development of the P-T limit curves so the upper pressure limit will not be exceeded.

As a result of the proposed P-T limits described in the foregoing Section 3.3, the pressure limits at low temperatures have been increased. The increase in the pressure limit provides sufficient pressure margin to accommodate operation of two RCPs in a loop or one RCP in each loop at low temperatures as proposed in Technical Specification Figures 3.1-1 and 3.1-2. Location adjustment for the allowable operating RCP combinations is summarized in Enclosure 2.

3.5 <u>LTOP Limits Justification</u>

Background

The low temperature P-T limits provide restrictions for the protection from nonductile failure of the RCS under transient conditions. The LTOP System protects the reactor vessel from excessive pressures at low temperature conditions. LTOP events occur as a result of equipment malfunction or operator error that results in mass or energy addition to the RCS. The B&W plant design is less likely to exceed Appendix G limits because of restrictions that preclude water-solid operation of the pressurizer. The design basis events for the LTOP System are as follows:

A. Erroneous actuation of the High Pressure Injection System
B. Erroneous opening of the core flood tank discharge valve
C. Erroneous addition of nitrogen to the pressurizer
D. Makeup control valve (makeup to the RCS) failing full open
E. All pressurizer heaters erroneously energized
F. Temporary loss of the Decay Heat Removal System's capability to remove decay heat from the RCS
G. Thermal expansion of the RCS after starting an RCP due to stored thermal energy in the steam generator

Consequences of LTOP Events

Each of the postulated LTOP events were analyzed to determine the rate of RCS pressure increase and/or the total amount of pressure increase that the system would experience. A stand alone thermal hydraulic model of the pressurizer was used for these predictions. Capabilities to model RCS inventory increases (e.g., makeup, HPI), inventory decreases (e.g., letdown), RCS expansion, and pressurizer heaters were included. A range of initial pressures and pressurizer levels were applied so that the pressurization rates could be applied to different initial P-T operating conditions. A brief summary of each transient response is provided below.

Erroneous actuation of the High Pressure Injection (HPI) system – this event would be the most limiting LTOP transient. However, HPI actuation results in a very rapid pressurization of the RCS and precludes achieving the necessary 10 minutes for operator action. Thus, this event is prevented below the LTOP enable temperature through plant procedures.

Erroneous actuation of the core flood tank discharge valve – this event is precluded by closing and locking out the breakers of the motor operated block valves before the RCS pressure decreases below the CFT pressure (600 psig). This will occur prior to cooling below the ART .

Erroneous addition of nitrogen to the pressurizer – this event can not overpressurize the RCS because of plant equipment that regulates the nitrogen pressure to 150 psig (i.e., pressure regulator and relief valves).

Makeup control valve (makeup to the RCS) fails full open – this event results in a pressurization rate of 20 to 30 psi/minute and is the most limiting of the remaining LTOP events.

All pressurizer heaters erroneously energized – this event is a slow transient (9 to 12 psi/minute) and is bounded by the failed makeup control valve event.

Temporary loss of the Decay Heat Removal System's (DHRS) capability to remove decay heat from the RCS – this event is a slow transient (7 psi/minute) and is bounded by the failed makeup control valve event.

Thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator – this event results in a finite increase in pressure that is less than the margin between the Appendix G and LTOP limits. Because of the presence of a pressurizer bubble, this event is much less severe than at other PWRs.

In summary, the most limiting, credible event is the failed open makeup control valve. Because of system design differences, the plant response is sensitive to the makeup pump head-capacity curve and system resistance. The above system descriptions including actions and administrative controls are unchanged from the existing TMI Unit 1 UFSAR description as previously approved by NRC (reference NRC SER dated July 28, 1980).

Methods to Protect the Reactor Vessel Pressure Limit

The TMI Unit 1 Low Temperature Overpressure mitigation system is both redundant and functionally diverse. The plant by virtue of a steam or gas blanket in the pressurizer space and the relatively small size and heat capacity of the Once Through Steam Generators, is not susceptible to heat addition transients. The plant is not operated in a water solid condition for normal heatup and cooldown evolutions which slows the dynamic pressure response during credible mass addition events. To provide protection from overpressurization, the plant is equipped with a dual setpoint pilot operated relief valve

that is set below the determined LTOP limit. In addition, plant operation is limited (i.e., combination of operating pressure and pressurizer level limitations) so that in the event of the most limiting LTOP event (a fully failed open makeup control valve) and a failure of the pilot operated relief valve to operate, the plant dynamic pressure response would provide at least 10 minutes during which time operator action is assumed to terminate the pressurization event before reaching the LTOP limit.

Two means of setting operating limits have been used for the failed open makeup control valve. This first approach assumes that the plant is operating at the maximum allowable pressure (as defined by bounds of the Appendix G heatup and cooldown limits and the PORV setpoint) at the time at which the failed open makeup control valve event occurs. Then, using plant specific makeup flow vs. RC pressure curves, the maximum allowable initial pressurizer level that will cause the tenth minute pressure to equal the LTOP pressure is determined. Thus, if the pressurizer level is maintained below this value for temperatures less than the enable temperature and if the RC pressure is less than the Appendix G heatup/cooldown pressure, the LTOP limit will not be exceeded during ten minutes of the failed open makeup control valve event.

The second approach is similar except that the maximum allowable pressurizer level is set and the maximum allowable pressure vs. temperature curve is determined. If this curve results in higher allowable operating pressures than the Appendix G heatup/cooldown curve, the LTOP limit is protected by the Appendix G curves (for this pressurizer level). If this curve results in lower allowable operating pressures, then this developed curve is implemented as the limiting operator curve.

3.5.1 LTOP Pressure-Temperature Limit

The LTOP allowable pressure versus temperature for the reactor vessel is limited to 100% of the steady-state Appendix G NDT limits (ASME Code Case 640). Thus, the flaw size, critical depth, allowable crack growth, and the calculational methodology are identical to those used in the Appendix G calculations. The use of steady state temperatures, rather than the transient resulting from technical specification heatup and cooldown limits is based on the likelihood that LTOP events occur during steady-state operations. This steady state approach has been approved by the NRC for B&W and other operating PWR plants.

The LTOP transient is also dependent on the initial pressurizer level (≤100 inches for TMI Unit 1), the makeup flow rate, and any non-condensable gasses that might be in the pressurizer. LTOP is a concern only below the enable temperature. The initial pressurizer level of ≤ 100 inches was chosen as the analytical input value for development of the revised LTOP analysis and resulting limits since this value represents the level currently utilized by TMI Unit 1 for normal heatup and cooldown conditions.

ASME Code 1995 Edition through 1996 Addendum defines the LTOP enable temperature as the RT_{NDT} temperature of the limiting material plus 50°F. The limiting Adjusted Reference Temperature is 250.5°F (251°F) for weld WF-25. During a cooldown, the coolant temperature is always less than (or equal to) the ¼ T temperature,

thus, it is conservative to use the coolant temperature as the LTOP setpoint. Therefore, the enable temperature will be the ART +50. Assuming a 12°F temperature uncertainty, the cooldown enable temperature is 313°F. However during a heatup, the ¼ T temperature is always less than the coolant temperature. Therefore, the setpoint must be the water temperature plus the difference in these two locations. As described in Enclosure 2, the maximum ¼ T temperature will be 16°F less than the coolant temperature (during a maximum allowable heatup rate). Therefore, the enable temperature will be the ART + 50 + 16. Assuming a 12°F temperature uncertainty, the heatup enable temperature is 329°F. The proposed Technical Specifications include the enable temperature of 329°F since this value satisfies the requirements for both heatup and cooldown scenarios.

The PORV setpoint for LTOP protection must be set at the minimum pressure to ensure the reactor vessel LTOP limits are not violated. Per Enclosure 2, the maximum allowed pressure at 100°F with 2/0 RCPs operating is 641 psig. Using the 69 psi correction to the pressurizer (low range tap) and using the 20 psi set point uncertainty, the PORV must be set at 641-20-69 = 552 psig.

Administrative controls are implemented during LTOP conditions to ensure the LTOP event initiators are not credible or that 10 minutes are available for operator action. The proposed administrative controls are listed in the proposed TMI Unit 1 Technical Specification Sections 3.1.12.1, 3.1.12.2, 3.1.12.3, and 4.5.2.1.

3.6 Pressurized Thermal Shock (PTS)

In accordance with 10 CFR 50.61, TMI Unit 1 has reevaluated the pressurized thermal shock reference temperature (RT_{PTS}) as part of generating the new Appendix G limits. The RT_{PTS} values applicable to the projected end-of-license period (29 EFPY) for the TMI Unit 1 reactor vessel beltline materials are listed in Table 3. These values are calculated in accordance with 10 CFR 50.61. The controlling beltline material for the TMI Unit 1 reactor vessel is the lower shell longitudinal weld, SA-1526, with a RT_{PTS} value of 256.4°F (Reference 12). The screening criterion for this weld metal is 270°F.

The extrapolated 29 EFPY fluences for the TMI Unit 1 reactor vessel beltline materials are listed in Table 3. These fluence values are at the clad-base metal interface on the inside surface of the reactor vessel where the material in question receives the greatest fluence. Since there is no inside surface fluence for the lower shell longitudinal weld (outside diameter 63%), SA-1494, the RT_{PTS} value for this weld metal is not applicable and is not evaluated as part of this calculation.

For the TMI Unit 1 reactor vessel beltline materials, measured initial reference temperature (IRT_{NDT}) values are not available with the exception of the weld metal WF-70. Therefore, generic mean values for each class of material are used for these materials based on available data as allowed in 10 CFR 50.61. The generic IRT_{NDT} values for the

TMI Unit 1 reactor vessel materials are based on statistical evaluations performed on available data. For the weld metal WF-70, the IRT_{NDT} is based on the alternative Master Curve approach using the data found in BAW-2202 (Reference 9). This approach has been previously approved by the NRC for the Zion Nuclear Power Station, Units 1 and 2. An exemption request pursuant to 10 CFR 50.61(a)(5) is provided in Section 3.9.

Table 3 lists the IRT_{NDT} values for the TMI Unit 1 reactor vessel beltline materials. The ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = (\text{Chemistry Factor}) \times (\text{Fluence Factor})$$

The Chemistry Factor is a function of the material's copper and nickel content. The Chemistry Factor is determined from Table 1 (for weld metals) and Table 2 (for base metals) in 10 CFR 50.61. Linear interpolation is permitted. The copper and nickel contents for the TMI Unit 1 reactor vessel beltline region materials are reported in BAW-2313, Revision 2 (Reference 10). The weld metal values are the best-estimate values based on the mean of the source means of each weld wire heat.

Table 3 lists the Chemistry Factors and the fluence factors for the TMI Unit 1 reactor vessel beltline region materials at 29 EFPY.

In accordance with 10 CFR 50.61, the Fluence Factor is determined as follows:

Fluence Factor = $f^{(0.28 - 0.10 \log f)}$, where f is the best-estimate neutron fluence at the clad-base-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence.

The calculated 29 EFPY ΔRT_{NDT} values for the TMI Unit 1 reactor vessel beltline materials are listed in Table 3.

The margin calculation is done to obtain conservative, upper bound values for the RT_{PTS} calculation. It is calculated as the square root of the sum of the squares of the standard deviations for the initial RT_{NDT} (σ_1) and the ΔRT_{PTS} (σ_Δ). If a measured value of initial RT_{NDT} for the material in question is available, σ_1 is to be estimated from the precision of the test method. If not, and generic mean values for that class of material are used, σ_1 is the standard deviation obtained from the set of data used to establish the mean. The standard deviation σ_Δ for ΔRT_{PTS}, is 28°F for welds and 17°F for base metal. The margin term is then calculated according to the equation below and the results are listed in Table 3.

$$M = \boldsymbol{Margin} = 2\sqrt{\sigma_1^2 + \sigma_\Delta^2}$$

However, in accordance with 10 CFR 50.61, σ_Δ need not exceed 0.50 times the mean value of ΔRT_{PTS}.

When two or more credible surveillance data sets are available, these data may be used to determine the RT_{PTS} of the reactor vessel beltline materials as follows:

First, if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the reactor vessel weld, the measured values of ΔRT_{NDT} should be adjusted by multiplying the values by the ratio of the chemistry factor for the reactor vessel weld to that for the surveillance weld. Second, using the ΔRT_{NDT} and its corresponding fluence, the chemistry factor may be calculated by multiplying each adjusted ΔRT_{NDT} by the corresponding fluence factor, summing the products, and dividing by the sum of the squares of the fluence factors:

$$CF = \frac{\Sigma \Delta RT_{NDT} * ff}{\Sigma ff^2}$$

The TMI Unit 1 plant specific reactor vessel surveillance program (RVSP) provides data for predicting the reference temperature shift for the base metal plate heat number C2789-2. In addition, the Master Integrated Reactor Vessel Surveillance Program (MIRVP) described in BAW-1543, Revision 4, provides surveillance data for weld metals WF-70 (wire heat 72105), WF-25 (wire heat 299L44), and SA-1526 (wire heat 299L44) for predicting the reference temperature shift. In these cases, the chemistry factor determined from the generic Tables in 10 CFR 50.61 using the weld wire heat "best-estimate" copper and nickel contents is considered conservative and is used in the RT_{PTS} calculation for the TMI Unit 1 reactor vessel beltline welds SA-1526, WF-25, and WF-70.

3.7 Justification for ASME Code Case N-588 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1," in lieu of the 10 CFR 50, Appendix G.

10 CFR 50.12 Requirements: The requested exemption to allow use of ASME Code Case N-588 to determine stress intensity factors for postulated defects in maximum postulated defects for circumferential welds meets the criteria of 10 CFR 50.12 as addressed below. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law: No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety: 10 CFR 50, Appendix G, requires, in part, that Article G-2120 of ASME XI, Appendix G, be used to determine the maximum postulated defects in reactor pressure vessels (RPV) when determining pressure-temperature limits for the vessel. These limits are determined for normal operation and pressure test conditions. Article G-2120 specifies, in part, that the postulated defect be in the surface of the vessel material and normal (perpendicular in the plane of the material) to the direction of maximum stress. ASME XI, Appendix G, also provides methodology to determine the stress intensity factors for a maximum postulated defect normal to the maximum stress. The purpose of this article is, in part, to prevent nonductile fractures by providing procedures to identify the most limiting postulated fractures to be considered in the development of pressure-temperature limits.

   Due to progress made in NDE techniques over the last thirty years, it is very unlikely to have large, undetected defects present in the beltline region of reactor vessels. If is further unlikely to have axial cracks originating from a circumferential weld perpendicular to the weld seam orientation in reactor vessels. Both experience and engineering studies indicate that the primary degradation mechanism affecting the beltline region of the reactor vessel is neutron embrittlement. No other service induced degradation mechanism exists at a pressurized water reactor to cause a prior existing defect located in the beltline region of the reactor vessel to grow while inservice. Based on these considerations, and the fact that the pressure temperature (P-T) limit for reactor operation is the limiting pressure for any of the materials in the vessel, it is not necessary to include additional conservatism in the assumed flaw orientation for circumferential welds. ASME Section XI, Code Case N-588, and the accompanying Appendix G Code change corrected this inconsistency in assumed flaw orientation for circumferential welds in vessels when calculating operating P-T limits.

   Code Case N-588 provides benefits in terms of calculating P-T limits by revising the Section XI, Appendix G reference flaw orientation for circumferential welds in reactor vessels. The reference flaw is a postulated flaw that accounts for the possibility of a prior existing defect that may have gone undetected during the fabrication process. When considering a reference flaw with respect to a weld, the reference flaw would represent any prior existing defect that my have been introduced during fabrication. Thus, the intended application of a reference flaw

is to account for prior existing defects that could physically exist within the geometry of the weldment. The current ASME Section XI, Appendix G approach mandates the consideration of an axial reference flaw in circumferential welds for purposes of calculating P-T limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. The possibility that an axial flaw may extend from a circumferential weld into a plate/forging or axial weld is already adequately covered by the requirement that axial defects be postulated in plates/forging and axial welds. The fabrication of reactor pressure vessels for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These procedural controls were also designed to minimize defects that could be introduced into the weld during the fabrication process.

Industry experience with the repair of weld indications found during pre-service inspection, and data taken from destructive examination of actual vessel welds, confirms that any remaining defects are small, laminar in nature, and do not cross transverse to the weld bead orientation. Therefore, any potential defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

ASME Code Case N-588 addresses this issue by allowing consideration of maximum postulated defects oriented circumferentially with circumferential welds. Code Case N-588 also provides appropriate procedures to determine limiting circumferential weld defects and associated stress intensify factors for use in developing reactor pressure vessel P-T limits per ASME XI, Appendix G procedures. The procedures allowed by Code Case N-588 are conservative and provide a margin of safety in the development of reactor pressure vessel pressure temperature operating and pressure test limits which will prevent nonductile fractures.

The proposed P-T limits include restrictions on allowable operating conditions and equipment operability requirements to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in Technical Specification 3.1.2. Therefore, this exemption does not present an undue risk to the public health and safety.

3. <u>The requested exemption will not endanger the common defense security</u>: The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60: Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulation if special circumstances are present. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) – demonstrates that the underlying purpose of the regulation will continue to be achieved;

   (a)(2)(iii) – would result in undue hardship or other costs that are significant if the regulation is enforced and;

   (a)(2)(v) – will provide only temporary relief from the applicable regulation and licensee has made good faith efforts to comply with the regulations.

   10 CFR 50.12(a)(2)(ii): The underlying purpose of 10 CFR 50, Appendix G and ASME XI, Appendix G, is to satisfy the requirement that: 1) The reactor coolant pressure boundary be operated in a manner having sufficient margin to ensure that when stressed the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and (2) P-T operating and test curves provide margin in consideration of uncertainties in determining the effects of irradiation on material properties.

   Application of Code Case N-588 to determine P-T operating and test limit curves per ASME XI, Appendix G, provides appropriate procedures to determine limiting maximum postulated defects and considering those defects in the P-T limits. This application of the code case maintains the margin of safety originally contemplated for plates/forgings and axial welds.

   Therefore, use of Code Case N-588, as described above, satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

   10 CFR 50.12(a)(2)(iii): The Reactor Coolant System pressure-temperature operating window is defined by the P-T operating and test curves developed in accordance with the ASME XI, Appendix G procedure. Continued operation with these P-T curves without the relief provided by ASME Code Case N-588 would unnecessarily restrict the pressure-temperature operating window for TMI Unit 1. This restriction requires that under certain low temperature conditions that only one reactor coolant pump in a reactor coolant loop be operated. The effect of this restriction is undesirable degradation of reactor coolant pump impellers due to cavitation with either one pump or one pump in each loop in operation. Further, the proposed LTOP guidelines will reduce the potential for an undesired challenge to the reactor coolant system power operated relief valve.

This constitutes an unnecessary burden that can be alleviated by the application Code Case N-588 in the development the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME Code Case N-588 does not reduce the margin of safety originally contemplated by either the NRC or ASME.

10 CFR 50.12(a)(2)(v): The exemption provides only temporary relief from the applicable regulation and TMI Unit 1 has made a good faith effort to comply with the regulation. We request that the exemption be granted until such time than the NRC generically approves ASME Code Case N-588 for use by the nuclear industry. However, to retain sufficient pressure-temperature operating margin to the end of the proposed TMI Unit 1 Technical Specification pressure temperature limits, an exemption to use Code Case N-588 is required.

Code Case N-588, Conclusion for Exemption Acceptability: Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-588 allows postulation of a circumferential defect in circumferential welds to be considered in lieu of requiring the defect to be oriented across the weld from one plate or forging to the adjoining plate or forging. This circumstance was not considered at the time ASME XI, Appendix G was developed and imposes restrictions on P-T operating limits beyond those originally contemplated.

This proposed alternative is acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-588 for TMI Unit 1 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in Technical Specification 3.1.2. Therefore, this exemption does not present an undue risk to the public health and safety.

3.8 Justification for ASME Code Case N-640 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-640, "Alternative Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I," in lieu of 10 CFR 50, Appendix G.

10 CFR 50.12 Requirements: The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME XI, Appendix G to determine the pressure-temperature limits meets the criteria of 10 CFR 50.12 as discussed below. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law: No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety: The revised pressure-temperature (P-T) limits being proposed for TMI Unit 1 rely in part, on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME XI, Appendix G process of determining P-T curves remain unchanged.

   Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heat up and cooldown process of a reactor vessel.

   Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel materials over time and usage. Since 1974, additional knowledge has been gained about the affect of usage on reactor pressure vessel materials. The additional knowledge demonstrates the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure.

   P-T curves based on the K_{Ic} curves will enhance overall plant safety by opening the pressure-temperature operating window with the greatest benefit in the region of low temperature operations. The two primary safety benefits in opening the low temperature operating window is a reduction in the challenges to RCS power operated relief valves and elimination of RCP impeller cavitation wear.

3. The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by this exemption request.

4. <u>Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60</u>: Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) – demonstrates the underlying purpose of the regulation will continue to be achieved;

   (a)(2)(iii) – would result in undue hardship or other costs that are significant if the regulation is enforced and;

   (a)(2)(v) – will provide only temporary relief from the applicable regulation and the licensee has made good faith efforts to comply with the regulations.

   <u>10 CFR 50.12(a)(2)(ii)</u>: ASME XI, Appendix G, provides procedures for determining allowable loading on the reactor pressure vessel and is approved for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P-T operating and test curves satisfied the underlying requirement for: 1) The reactor coolant pressure boundary be operated in a manner having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and 2) P-T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

   The ASME XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

   <u>10 CFR 50.12(a)(2)(iii)</u>: The Reactor Coolant System pressure-temperature operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME XI, Appendix G procedure. Continued operation of TMI Unit 1 with these P-T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the pressure-temperature operating window. This restriction requires, under certain low temperature conditions, only one reactor coolant pump in a reactor coolant loop to be operated. The effect of this restriction is undesirable degradation of reactor coolant pump impellers due to cavitation sustained with either one pump or one pump in each loop in operation.

This constitutes an unnecessary burden that can be alleviated by the application of Code Case N-640 in the development of the proposed P-T curves. Implementation of the proposed P-T curves as allowed by ASME Code Case N-640 does not significantly reduce the margin of safety.

10 CFR 50.12(a)(2)(v): The exemption provides only temporary relief from the applicable regulation and TMI Unit 1 has made a good faith effort to comply with the regulation. We request the exemption be granted until such time that the NRC generically approves ASME Code Case N-640 for use by the nuclear industry. However, to retain sufficient pressure-temperature operating margin to the end of the proposed TMI Unit 1 Technical Specification pressure-temperature limits, we require an exemption to use Code Case N-640.

Code Case N-640, Conclusion for Exemption Acceptability: Compliance with the specified requirements of 10 CFR 50.60 would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the fracture toughness lower bound used by ASME XI, Appendix G, in the determination of reactor coolant pressure-temperature limits. This proposed alternative is acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-640 for TMI Unit 1 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in Technical Specification 3.1.2. Therefore, this exemption does not present an undue risk to the public health and safety.

3.9 Justification for Exemption from 10 CFR 50.61(a)(5) Determination of Reference Temperature

The following information provides the basis for the exemption request to 10 CFR 50.61(a)(5) as it relates to the determination of the reference temperature for material used in reactor vessels. The stated section requires that the reference temperature be determined in accordance with Paragraph NB-2331 of the ASME Code.

10 CFR 50.12 Requirements: The requested exemption to allow use of the Master Curve approach using the data found in BAW-2202 to determine the initial reference temperature for nil ductility transition (IRT_{NDT}) for weld WF-70 meets the criteria of 10 CFR 50.12 as discussed below. 10 CFR 50.12 states the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law: No law exists which precludes the activities covered by this exemption request. 10 CFR 50.61(a)(5) allows the use of other methods when approved by The Director, Office of Nuclear Reactor Regulation.

2. The requested exemption does not present an undue risk to the public health and safety: The proposed methodology utilizes drop-weight test data and fracture mechanics to demonstrate the non-brittle nature of the WF-70 weld material in the temperature range of interest. This approach is supported by actual material fracture toughness testing data found in BAW-2202, "Fracture Toughness Characterization of WF-70 Weld Metal," BWNT, September 1993. The material behavior demonstrated by the WF-70 test results is bounded by the ASME Code reference toughness curve.

   Additionally, flux reduction measures have been implemented to further limit the amount of radiation induced embrittlement experienced by the TMI Unit 1 reactor vessel.

3. The requested exemption will not endanger the common defense and security: The common defense and security are not in any way compromised by this exemption request. The proposed change does not alter the physical plant in any manner.

At least one of the special circumstances are present per 10 CFR 50.12(a)(2):

10 CFR 50.12(a)(2)(ii) indicates that an exemption would be warranted if application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The PTS rule, 10 CFR 50.61, was adopted to ensure that operating reactors were adequately protected from pressurized thermal shock. This is accomplished through limitations on the amount of radiation-induced embrittlement to be tolerated. In the case of TMI Unit 1, weld material WF-70 does not meet the PTS screening criteria using the methodology provided in the ASME Code. However, using the alternate methodology proposed by AmerGen which utilizes fracture mechanics, the limiting weld material will meet the PTS screening criteria. Additionally, the material behavior of WF-70 more closely resembles the behavior predicted by the fracture mechanics approach. Thus, there is confidence that the TMI Unit 1 reactor vessel will not be susceptible to PTS for the duration of the projected end-of-license period of 29 EFPY. This achieves the underlying purpose of the rule without applying the provision of the rule which requires application of the methodology provided by the ASME Code.

Absent the requested exemption, TMI Unit 1 will either cease operation prematurely or be required to expend significant resources to demonstrate the non-brittle nature of the TMI Unit 1 reactor vessel. This exemption will allow the utilization of a methodology which predicts non-brittle behavior for the duration of the operating license.

3.10 Affected UFSAR Sections

The TMI Unit 1 UFSAR will be revised with the implementation of this Technical Specification amendment request to reflect the changes to the P-T limits and the LTOP analysis described in this amendment application. This UFSAR revision will be made in accordance with 10 CFR 50.71(e).

3.11 References

1. ASME Code Case N-588, Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division I, December 12, 1997.

2. ASME Code Case N-640, Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I, February 26, 1999.

3. BAW-1543A, Revision 2, Integrated Reactor Vessel Material Surveillance Program, A. L. Lowe, Jr., et.al., B&W Nuclear Division, May 1985.

4. BAW-1875, The B&WOG Cavity Dosimetry Program, S. Q. King, August 1985.

5. BAW-2241P-A, Revision 1, Fluence and Uncertainty Methodologies, J. R. Worsham, III, et. al., B&WOG Materials Committee April 1999.

6. Framatome ANP Calculation No. 86-5010023-00, TMI Cycle 5-11 Final Report, December 15, 2000.

7. Framatome ANP Calculation No. 32-5011059-00, TMI-1 Reactor Vessel Adjusted RTNDT Values for 23 & 29 EFPY, January 17, 2001.

8. BAW-10046A, Revision 2, Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G, H. W. Behnke, et. al., BWNT, June 1986.

9. BAW-2202, Fracture Toughness Characterization of WF-70 Weld Metal, BWNT, September 1993.

10. BAW-2313, Revision 2, B&WOG Reactor Vessel Working Group Reactor Vessel Materials and Surveillance Data Information, Volume 1 and 2, FTI, December 1999.

11. BAW-1543, Revision 4, Master Integrated Reactor Vessel Surveillance Program, BWNT, February 1993.

12. Framatome ANP Calculation No. 32-5011130-00, Pressurized Thermal Shock Reference Temperatures for TMI-1, January 26, 2001.

TABLE 1

Adjusted Reference Temperature Evaluation for the TMI Unit 1 Reactor Vessel Materials at the ¼-Thickness Location

Applicable Through 29 EFPY

Reactor Vessel Beltline Region Material	Material Ident.	Heat Number	Type	Cu Wt%	Ni wt%	29 EFPY Fluence, n/cm^2		Chemistry Factor	Fluence Factor	ΔRT_{NDT}, °F	Initial RT_{NDT}, °F	Margin, °F	T/4 ART_{NDT}
						Wetted Surface	T/4 Location						
Regulatory Guide 1.99, Revision 2, Position 1.1													
Lower Nozzle Belt Forging (LNB)	ARY 59	123S454	SA-508 Cl. 2	0.08	0.72	9.5609E+18	5.5925E+18	51.0	0.837	42.7	+3	70.7	116.4
Upper Shell Plate (US)	C2789-1	C2789-1	SA-302 Gr. B Modified	0.09	0.57	10.3758E+18	6.0692E+18	58.0	0.860	49.9	+1	63.6	114.5
Upper Shell Plate (US)	C2789-2	C2789-2	SA-302 Gr. B Modified	0.09	0.57	10.3758E+18	6.0692E+18	58.0	0.860	49.9	+1	63.6	114.5
Lower Shell Plate (LS)	C3307-1	C3307-1	SA-302 Gr. B Modified	0.12	0.55	10.3813E+18	6.0724E+18	82.0	0.860	70.5	+1	63.6	135.1
Lower Shell Plate (LS)	C3251-1	C3251-1	SA-302 Gr. B Modified	0.11	0.50	10.3813E+18	6.0724E+18	73.0	0.860	62.8	+1	63.6	127.4
LNB to US Circ. Weld (100%)	WF-70	72105	Linde 80 Flux	0.32	0.58	9.5609E+18	5.5925E+18	199.3	0.837	166.8	-26	56.0	196.8
US Longit. Weld (Both 100%)	WF-8	8T1762	Linde 80 Flux	0.19	0.57	7.3168E+18	4.2799E+18	152.4	0.764	116.4	-5	68.5	179.9
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80 Flux	0.34	0.68	10.0859E+18	5.8996E+18	220.6	0.852	188.0	-7	69.5	[250.5]
LS Longit. Weld (100%)	SA-1526	299L44	Linde 80 Flux	0.34	0.68	6.5700E+18	3.8430E+18	220.6	0.735	162.1	-7	69.5	224.6
LS Longit. Weld (ID 37%)	SA-1526	299L44	Linde 80 Flux	0.34	0.68	6.5700E+18	3.8430E+18	220.6	0.735	162.1	-7	69.5	224.6
LS Longit Weld (OD 63%)	SA-1494	8T1554	Linde 80 Flux	0.16	0.57	N/A	N/A	143.9	N/A	N/A	-5	N/A	N/A

[ ] - Controlling values of the adjusted reference temperatures.

TABLE 2

Adjusted Reference Temperature Evaluation for the TMI Unit 1 Reactor Vessel Materials at the ¾-Thickness Location

Applicable Through 29 EFPY

Reactor Vessel Beltline Region Material	Material Ident.	Heat Number	Type	Cu wt%	Ni wt%	29 EFPY Fluence, n/cm^2		Chemistry Factor	Fluence Factor	ΔRT_{NDT}, °F	Initial RT_{NDT}, °F	Margin, °F	3/4T ART_{NDT}
						Wetted Surface	3/4T Location						
Regulatory Guide 1.99, Revision 2, Position 1.1													
Lower Nozzle Belt Forging (LNB)	ARY 59	123S454	SA-508 Cl. 2	0.08	0.72	9.5609E+18	2.0318E+18	51.0	0.573	29.2	+3	68.5	100.7
Upper Shell Plate (US)	C2789-1	C2789-1	SA-302 Gr. B Modified	0.09	0.57	10.3758E+18	2.2050E+18	58.0	0.593	34.4	+1	63.6	99.0
Upper Shell Plate (US)	C2789-2	C2789-2	SA-302 Gr. B Modified	0.09	0.57	10.3758E+18	2.2050E+18	58.0	0.593	34.4	+1	63.6	99.0
Lower Shell Plate (LS)	C3307-1	C3307-1	SA-302 Gr. B Modified	0.12	0.55	10.3813E+18	2.2062E+18	82.0	0.593	48.6	+1	63.6	113.2
Lower Shell Plate (LS)	C3251-1	C3251-1	SA-302 Gr. B Modified	0.11	0.50	10.3813E+18	2.2062E+18	73.0	0.593	43.3	+1	63.6	107.9
LNB to US Circ. Weld (100%)	WF-70	72105	Linde 80 Flux	0.32	0.58	9.5609E+18	2.0318E+18	199.3	0.573	114.2	-26	56.0	144.2
US Longit. Weld (Both 100%)	WF-8	8T1762	Linde 80 Flux	0.19	0.57	7.3168E+18	1.5549E+18	152.4	0.511	77.9	-5	68.5	141.4
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80 Flux	0.34	0.68	10.0859E+18	2.1434E+18	220.6	0.586	129.3	-7	69.5	[191.8]
LS Longit. Weld (100%)	SA-1526	299L44	Linde 80 Flux	0.34	0.68	6.5700E+18	1.3962E+18	220.6	0.487	107.4	-7	69.5	169.9
LS Longit. Weld (ID 37%)	SA-1526	299L44	Linde 80 Flux	0.34	0.68	6.5700E+18	N/A	220.6	N/A	N/A	-7	N/A	N/A
LS Longit Weld (OD 63%)	SA-1494	8T1554	Linde 80 Flux	0.16	0.57	N/A	1.3962E+18	143.9	0.487	70.1	-5	68.5	133.6

[ ] - Controlling values of the adjusted reference temperatures.

TABLE 3

TMI Unit 1 Reactor Vessel Beltline Material Pressurized Thermal Shock Reference Temperature Summary Applicable to the End-of-License Period (29 EFPY)

Reactor Vessel Beltline Region Material	Material Ident.	Heat Number	Type	Cu wt%	Ni wt%	29 EFPY Fluence at Clad-Base Metal Interface, n/cm^2	Chemistry Factor	Fluence Factor	ΔRT_{PTS}, °F	Initial RT_{NDT}, °F	Margin, °F	RT_{PTS}, °F	Screening Criteria
10 CFR 50.61 (Tables)													
Lower Nozzle Belt Forging (LNB)	ARY 59	123S454	SA-508 Cl. 2	0.08	0.72	9.2217E+18	51.0	0.977	49.8	+3	70.7	123.5	270
Upper Shell Plate (US)	C2789-1	C2789-1	SA-302 Gr. B Modified	0.09	0.57	10.0161E+18	58.0	1.000	58.0	+1	63.6	122.6	270
Upper Shell Plate (US)	C2789-2	C2789-2	SA-302 Gr. B Modified	0.09	0.57	10.0161E+18	58.0	1.000	58.0	+1	63.6	122.6	270
Lower Shell Plate (LS)	C3307-1	C3307-1	SA-302 Gr. B Modified	0.12	0.55	10.0148E+18	82.0	1.000	82.0	+1	63.6	146.6	270
Lower Shell Plate (LS)	C3251-1	C3251-1	SA-302 Gr. B Modified	0.11	0.50	10.0148E+18	73.0	1.000	73.0	+1	63.6	137.6	270
LNB to US Circ. Weld (100%)	WF-70	72105	Linde 80 Flux	0.32	0.58	9.2217E+18	199.3	0.977	194.7	-26	56.0	224.7	300
US Longit. Weld (Both 100%)	WF-8	8T1762	Linde 80 Flux	0.19	0.57	7.2299E+18	152.4	0.909	138.5	-5	68.5	202.0	270
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80 Flux	0.34	0.68	9.7392E+18	220.6	0.993	219.1	-7	69.5	281.6	300
LS Longit. Weld (100%)	SA-1526	299L44	Linde 80 Flux	0.34	0.68	6.4879E+18	220.6	0.879	193.9	-7	69.5	256.4	270
LS Longit. Weld (ID 37%)	SA-1526	299L44	Linde 80 Flux	0.34	0.68	6.4879E+18	220.6	0.879	193.9	-7	69.5	256.4	270
LS Longit Weld (OD 63%)	SA-1494	8T1554	Linde 80 Flux	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

4.0 Environmental Consideration

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that my be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

AmerGen has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described in Item 5.0 of this evaluation.

2. The proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. The proposed amendment ensures operation within applicable safety limits and margins of safety. The changes do no modify the reactor coolant pressure boundary nor make any physical changes to the facility design, material, or construction standards.

3. The proposed license amendment will not result in a significant increase in individual or cumulative occupational radiation exposure. The consequences of any design basis accident are not affected by this change. The proposed changes do not affect the integrity of the reactor coolant pressure boundary or any fission product barrier. Occupational exposures are not affected by the proposed changes.

5.0 No Significant Hazards Consideration

AmerGen has determined that this License Amendment Request poses no significant hazards considerations as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

These proposed Technical Specification changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N-588 and N-640. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides safety limits and margins of safety which ensure that failure of a reactor vessel will not occur.

The proposed changes do not impact the capability of the reactor coolant pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore, do not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards.

The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed Pressure-Temperature (P-T) limits, Low Temperature Overpressure (LTOP) limits and setpoints, and allowable operating reactor coolant pump combinations are not considered to be an initiator or contributor to any accident analysis addressed in the TMI Unit 1 UFSAR.

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR 100. In addition, the proposed changes do not affect any fission product barrier. The revised PORV LTOP setpoint is established to protect reactor coolant pressure boundary. The changes do not degrade or prevent the response of the PORV or safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the mitigation of the radiological consequences of an accident previously evaluated.

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

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed license amendment revises the TMI Unit 1 reactor vessel P-T limits. LTOP limits and setpoints, and allowable operating reactor coolant pump combinations. Compliance with 10 CFR 50 Appendix G, includes utilization of ASME XI, Appendix G, as modified by Code Cases N-588 and N-640 to meet the underlying intent of the regulations.

Operation of TMI Unit 1 in accordance with these proposed Technical Specification changes will not create any failure modes not bounded by previously evaluated accidents.

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

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed Technical Specification (TS) changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N-588 and N-640. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides safety limits and margins of safety which ensure that failure of a reactor vessel will not occur.

No plant safety limits, set points, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

6.0 Implementation

AmerGen requests that the amendment authorizing this change become effective upon issuance and implemented within 30 days.

ENCLOSURE 2

Framatome ANP Document I.D. 32-5011638-01

29 EFPY Pressure Temperature Limits TMI Unit 1

FRAMATOME TECHNOLOGIES

CALCULATIONAL SUMMARY SHEET (CSS)

DOCUMENT IDENTIFIER 32-5011638-01

TITLE TMI-1 29 EFPY PT Limits

PREPARED BY:		REVIEWED BY:	
NAME	J. A. Weimer	NAME	BL Boman
SIGNATURE.	James A Weimer	SIGNATURE.	BL Boman
TITLE	Sr Princ. Engrg DATE 3/02/01	TITLE	Supervisory Engrg DATE 03/02/01
COST CENTER	41016 REF. PAGE(S) 2	TM STATEMENT: REVIEWER INDEPENDENCE	Hudlin

PURPOSE AND SUMMARY OF RESULTS:

PURPOSE

This document provides revised input for OP-1102-1 and OP 1102-11, the operator pressure – temperature curves used for maintaining the structural integrity of the reactor vessel, RC pumps, surge line, and decay heat removal system during heatups and cooldowns up through EOL conditions (29 EFPY). This analysis also provides the bases for the final technical specification NDT heatup and cooldown requirements and establishes the PORV set point when operating under Low Temperature Over pressure (LTOP) constraints. The limits provided in this analysis are for heatups and cooldowns with 2/0 RC pump operation (compared to previously used 1 pump operation) at low RC pressures. The primary purpose of Revision 01 was to provided Technical Specification curves (Attachment 2). Revision 1 also provided some other minor corrections discussed on page 3. **Revision 1 completely replaces Revision 0.**

SUMMARY OF RESULTS

Figures 1 through 4 show the heatup and cooldown limits for normal RC pump operation (2/0, 2/1, and 2/2). Figures 5 and 6 show the cooldown limits for a situation where RC pump(s) have failed (cooldown with 0/2 or any one-pump operation).

The "hard" limits (NDT, LTOP, and subcooling margin) are established conservatively such that they are the same for all heatups and cooldowns and are the same when using either pressure instrument (wide or low range). This analysis assumed that if 2/0 is not available, the plant will not heatup, therefore 2/0 is the only heatup option. The pressure instrument uncertainty on these limits was 4 psi for the low range pressure and 25 psi for the wide range pressure. The low and wide range temperature uncertainties were 7F and 12F, respectively. The wide range uncertainties were applied to the entire range for the NDT and LTOP limits. The subcooling margin assumes no instrument uncertainty (just the required 25F temperature reduction) per Reference 2. The surge line limit assumes no instrument uncertainty also per Reference 2. The rod drop limits are identical to previous limits used at TMI-1 (per Reference 2). The soft limits (NPSH, seal staging, and DHRS operation) assume the low range uncertainties below 450 psig and the wide range above 450 psig per Reference 2. Since DHRS and seal staging limits are essentially only used in the low range, they will have only low range uncertainties. The low range pressure device (in the pressurizer) must be used for these limits below 450 psig and the wide range (hot leg tap) above 450 psig. A transition zone is provided for the change over between 450 psig and 500 psig (the upper limit of the low range pressure transducer). Since the PORV set point (with its 22 psi opening uncertainty) is 552 psig (or 485 psig if original set points are used), it will limit the LTOP/NDT curves above this pressure while the plant is below the enable temperature. The various limits of operation are listed with notes for each figure. In general, heatups require 2/0 RC pump operation (with the second pump started within ~5 minutes of the first). Due to NPSH concerns relative to the DHRS limit, the 3rd RC pump should not be started until after DHRS is terminated and the RCS temperature is greater than 200F. Normal cooldowns require 2/0 RC pump operation (or 0/2 in special circumstances) below 450 psig prior to bringing the DHRS into service. Above 450 psig and 400F, any RC pump combination is permissible. Due to the differences in the two pressure indications (with the pressurizer tap indicating a higher pressure than the hot leg tap by ~15 psi), there will be an additional ~15 psi conservatism in the NDT/LTOP limits when using the pressurizer indication for P-T limits. This is because the location correction was applied as if the hot leg tap was being used in the low range (below 450 psig).

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE / VERSION / REV	CODE / VERSION / REV
None Used	

THIS DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK

YES ( ) NO (x)

Results (continued)
The PORV low pressure set point is corrected to the top of the pressurizer (the pressure that opens the PORV). The location correction to both the low and wide range pressure taps for each limit is based on the most conservative correction associated with the pump combinations described on each curve.

Each limit uses the more conservative tube plugging criteria for the limit (maximum allowable plugging (20% per generator) or cycle 10 specific plugging, whichever provides the worst case correction factor). Per Reference 2, these curves are only for monitoring pressure on the "A" hot leg (or pressurizer) since the "B" hot leg tap will never be used for cooldowns or heatups (as requested by TMI-1 personnel). Note that the PORV set point for LTOP was calculated to be 552 psig but may conservatively remain at 485 psig until the equipment can be modified for this pressure at a later date. This indicated pressurizer level must be equal to or less than 100" while operating below the LTOP enable temperature.

The minimum RCS indicated temperature for RC pump startup during a heatup is 100F and the maximum indicated temperature for RC pump operation during a cooldown is 200F.

Attachment 2 shows the heatup and cooldown composite limit and some appropriate notes for the NDT limits. While the NDT limits can be monitored form either the "A" or "B" hot leg, the procedural PT window must use the "A" hot leg (per request from TMI-1). Attachment 2 also shows the ISLH limits with similar notes.

REFERENCES

1. FRA ANP Doc. 32-5011029-00 "TMI-1 Uncorrected P-T Limits at 29 EFPY 02/01.
2. FRA ANP Doc. 51-5010400-01 "TMI-1 Analytical Input Summary" 03/02/01.
3. FRA ANP Doc. 51-1212232-01 "Key Elevations for all Plants" 3/94
4. TMI-1 Plant Heatup to 525F procedure 1102-1 Rev 154.
5. FRA ANP Doc. 32-5011491-00 " FSPLIT Hydraulic Analysis For TMI-1" 02/01.
6. FRA ANP Doc. 32-5009876-00 "TMI-1 LTOP Failed Open MU Control Valve" 02/01.
7. FRA ANP Doc. 32-5011059-00 "TMI1 Reactor Vessel ART Values for 23 & 29 EFPY", 01/01.
8. CRANE Tech. Paper 410 Flow of Fluids through Valves, Fittings, & Pipe".
9. FRA ANP Doc. 32-5010021-00 "TMI-1 Cy. 5-11 Fluence Calculation" 02/01.
10. FRA ANP Doc. 32-1176254-00 "177 FA Plant Hydraulics Model", 1/93.
11. TMI-1 Doc. C-1101-220-5360-030 Rev 0 Sheet No 11.
12. BAW-2127 "Pressurizer Surge Line Thermal Stratification" 12/90.
13. TMI-1 Cooldown procedure 1102-11 Rev 123.
14. FRA ANP Doc. 32-5001328-00 "Asymmetric SG Tube Plugging" 9/98.
15. FRA ANP Doc. 32-1232660-00 GCALC Calc of Mass Flux Rates" 5/94.
16. FRA ANP Doc. 32-5011261-00 "Seal Injection analysis for TMI-1" 2/01.

References 4, 11, and 13 can be obtained through the TMI-1 document control system WRG Date 3/5/01

RECORD OF REVISION

Revision	Description of Change
00	Original Release
01	Attachment 2 was added to facilitate TMI-1 NRC Submittal. Word on page 8 under "Enable Temperature 2nd sentence changed from "greater" to "less". Seven pressures on Table 3 (above 2400 psig) were incorrectly transposed from the spreadsheet & were reduced by ~15 psi. Deleted ambiguous statement on page 8 Reference 2 (AIS) was revised

DISCUSSION

During heatups and cooldowns, the RCS is effectively at a constant temperature at any given time, however, the RC pressure can vary significantly around the loop. Therefore, the various limits that are location dependent must be corrected to the location of pressure measurement. Due to the fact that the location correction is different to the low (pressurizer) and wide (hot leg) range pressure transducers, the PT limits will have to be established for each location, depending on which instrument is required for P-T monitoring. The pressure indication in the pressurizer should not change as RC pumps are started or stopped and the "A" hot leg tap indication should change by approximately 5 psi during 2/0 pump operation (see unrecoverable loss of path 12 during 2/0 operation in Reference 5). However, the pressures at the limiting locations (e.g., pump suction, downcomer beltline, etc.) can change significantly as pumps are started or stopped. As the RCS pressure increases from the low to high range, there will be a step change associated with the pressurizer to hot leg tap transition location adjustment. There will also be a step change associated with the instrument uncertainty associated with each instrument when transitioning from low to high range. Per reference 2, NDT limits will use the wide range uncertainties (25 psi and 12F) even in the low range. Therefore, PT windows will be developed for the following situations.

1. A nominal heatup and cooldown PT window will be developed for the low range pressure transducer in the pressurizer for pressures below 450 psig[1]. All limits requiring instrument uncertainty except NDT and LTOP will be based on the low range tap's 4 psi uncertainty. The NDT will use a larger uncertainty that will provide for conservative results and a smooth transition to the high range instruments. These limits will also have an additional conservatism (~15 psi[2]) based on the fact that they will be corrected to the hot leg tap while the low range pressurizer tap will actually be monitored.
2. The second nominal heatup and cooldown PT window will be developed for pressures greater than ~450 psig. All applicable limits will be based on the wide range transducers in the "A" hot leg and will be applicable to all possible RC pump operating conditions.
3. The third set of PT windows will include the non-standard low range cooldown conditions based on a loss of one or more RC pumps.

The following details the development of the PT limits for TMI-1

CALCULATIONS

PRESSURE LOCATION CORRECTIONS

All the PT window limits apply to different portions or components in the RCS. The limits are calculated at the required locations. However, the plant uses only two locations of indicated pressure, the pressurizer low range tap and the hot leg wide range tap. Therefore, all limits need to be corrected to one or both of these locations. This was accomplished by developing a hydraulics model of the RCS and analyzing it for various temperatures and pump combinations. There are two different hydraulic conditions analyzed for, the maximum allowable SG tube plugging and a minimum SG tube plugging. The minimum plugging will be the SG plugging established in cycle 10. The plant will likely never experience less than cycle 10 plugging (8.43% in SG A and 2.65% in SG B) and therefore it will be considered the minimum percent plugging prior to 29 EFPY.

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1 The low range is acceptable to 500 psig but the PT limits will assume the transition to high range is ~450 psig to provide operations with some margin to transfer signals.

2 As shown below in Table 6, the pressure inside the pressurizer ranges from ~14 to 18 psi higher than the hot leg tap pressure for 2/0 pump operation depending on the temperature (primarily due to elevation differences).

The NDT and LTOP limits will use the Cycle 10 hydraulic results because it will produce the highest DP from the beltline to the pressure instruments (e.g., the highest flow rate). The DHRS limits will use Cycle 10 plugging for the same reason. The NPSH and seal staging will use 20% plugged because this will provide the highest DP between the pressure tap and the pump suction locations. The surge line limits, rod drop limits, and subcooling margins do not apply because they are identical to previous requirements (Reference 2).

TMI-1 is permitted to have 15% tube plugging in one SG and 25% in the other. Reference 14 shows that when there is a 10% difference in SG plugging, the pump inlet pressures can vary by as much as 3 psi (see 0%-10% case in Reference 14). However, the pump flow also decreases by ~1.5% which in turn reduces the required NPSH. As will be shown below, a ~3% flow reduction through the pump at 200F (2/0 operation) increases the available NPSH by ~8 psi and therefore, 1.5% would have ~4 psi more NPSH available. Also, since NPSH is a soft limit and a few psi higher or lower will not have a significant effect on the pumps, the 20%/20% case will be used for the 15%/25% plugging case for NPSH calculations. Also, as discussed below, the seal staging limits are believed to have sufficient margin built into them based on the fact the pressure drop across the seal increases when pumps are started; therefore a few psi will not affect these results as they will be well above the limit. Another conservatism applied to this calculation is that the location corrections calculated below for a specific temperature will be rounded up and applied to all temperatures up to the next higher temperature calculated (no interpolation). This will provide conservative DP for most temperatures. Therefore, any small affect on the flow and pressure distribution due to 15% - 25% plugging vs 20% - 20% plugging does not need to be included in the NPSH or seal staging limits.

The base FSPLIT analyses in Reference 5 used the 20% tube plugging. One case at 200F was analyzed for each pump combination with the cycle 10 plugging model (200BE in Table 1 below). The changes (or deltas) in the various pressure differences in the loop was compared to the 20% plugging case and this difference was applied to all temperatures. This will be conservative for all temperatures higher than 200F and slightly non-conservative for the 100F cases. However, since more tube plugging has occurred since cycle 10 and since the larger DPs occur above 200F (during 3 and 4 pump operation), and since the NDT/LTOP analyses has an additional ~15 psi conservatisms inherent to the calculations at 100F (see discussion below), the final results will be still be conservative for all temperatures.

When no RC pumps are operating, the pressure differences are due solely to elevation differences. The DHRS is approximately 8 ft (of water) below the low range pressure tap, and 38.2 ft below the wide range tap (Reference 3)[3]. The beltline is an additional 10.5 ft below this point (49.7 ft below the wide range tap. The pump suction is approximately 39.2 ft below the hot leg tap and ~9 ft below the low range pressure indication (at 100" in the pressurizer). The no-pump pressure differences are shown below using 62.5 lb/cuft density. The calculated beltline and drop line corrections are increased (in the table) so that the 200F 20% tube plug case is the same as the 200F BE case. The same DP correction is then used for all the temperatures.

3 The relative elevation of the wide range tap is 60.46 ft. The relative elevation of the low range pressure (per Reference 2) is ~3 ft above the surge line or 30.25 ft. The pump suction is defined as approximately at the horizontal cold leg center-line or 21.25 ft. The center of the hot leg at the drop line is ~1 ft above the horizontal hot leg relative elevation or 22.25 ft

Table 1 Location Corrections For TMI-1

Temperature	Low range (pressurizer) DP to Operating Pump Suction	Wide Range (hot leg) DP to Pump Operating Suction	Wide Range DP to Beltline	Low range DP to Drop Line
1/0 Pump Combination				
20% Tube plugging except BE cases which are ~6-8% tube plugging			+1 psi DP*** for BE case conversion	+0.7 psi DP*** for BE case conversion
100	-59.8	-45.2	37.3	15.8
200	-56.3	-42.2	36.5	15.3
250	-54.6	-40.8	35.8	15.4
300	-52.8	-39.4	35.0	15.2
400	-49.0	-36.5	33.0	14.4
500	-44.4	-33.1	30.2	13.4
540	-42.0	-31.4	28.9	13.0
200 BE	-52.6	-38.4	36.5	15.3
0/1 Pump Combination				
			+0.2 psi DP	0 psi DP *
100	-70.7	-57.8	23.9	-7.0
200	-67.4	-54.7	23.2	-7.3
250	-65.6	-53.1	22.7	-7.3
300	-63.6	-51.5	22.1	-7.1
400	-59.2	-47.9	20.8	-6.6
500	-53.7	-43.6	18.9	-5.9
540	-51.0	-41.5	18.1	-5.5
200BE	-64.4	-51.7	23.2	-8.0
1/2 Pump Combination				
	1 pump loop		+2.2 psi DP	0 psi DP
100	-105.2	-90.9	64.5	-25.3
200	-98.5	-84.8	64.1	-25.8
250	-95.4	-82.0	61.8	-25.6
300	-92.3	-79.2	60.3	-25.2
400	-85.5	-73.5	56.6	-23.7
500	-77.5	-66.6	52.1	-21.7
540	-73.6	-63.4	49.5	-20.4
200 BE	-91.5	-29.7	64.1	-27.9
2/1 Pump Combination				
	1 pump loop		+4.5 psi DP	+2.1 psi DP
100	-75.1	-56.9	102.0	36.0
200	-68.0	-50.3	100.9	36.3
250	-65.2	-47.8	99.5	36.0
300	-62.5	-45.7	97.3	35.5
400	-57.3	-41.7	91.9	33.9
500	-51.6	-37.4	84.8	31.3
540	-49.0	-35.6	80.8	30.1
200 BE	-58.9	-40.9	100.9	36.3
0/2 Pump Combination				
			+0.4 psi DP	0 psi DP
100	-128.8	-115.1	32.4	-41.7
200	-121.3	-108.7	30.9	-42.0
250	-117.7	-105.3	30.2	-41.6
300	-113.9	-102.0	29.4	-40.7
400	-105.8	-94.7	27.6	-38.4
500	-96.0	-86.1	25.3	-35.0
540	-91.3	-82.9	23.6	-33.3
200 BE	-114.7	-102.1	30.9	-45.0
2/0 Pump Combination				
			+3.9 psi DP	+3.1 psi DP
100	-81.6	-63.6	87.4(To Low Rng ~69**psid)	53.8
200	-74.5	-56.3	86.8	53.4
250	-71.3	-53.6	85.4	52.9
300	-68.4	-51.2	83.7	52.0

400	-62.8	-46.7	79.1	49.4
500	-56.6	-42.1	72.8	45.7
540	-53.7	-40.0	70.0	43.8
200 BE	-64.8	-46.2	86.8	53.4
2/2 Pump Combination				
			+4.6 psi DP	+0.1 psi DP
300	-55.7	-39.5	111.3	4.6
400	-51.2	-36.1	105.4	4.4
500	-46.0	-32.4	96.9	4.3
540	-43.7	-30.7	92.6	4.3
300 BE	-47.9	-31.5	111.3	4.6
0/0 Pump Combination				
100	4	17	22	3

* For 0/1 and 0/2 the drop line nozzle pressure is higher than the indicated pressure (see negative numbers in Table) and therefore it is conservative to use the 20% tube plugging numbers.
** This 69 psid shows the conservatism that will be included if the low range pressure is monitored.
***The difference between the "200" case and the "200BE" case was applied to the other temperatures.

LTOP

Background

Attachment 1 includes a discussion of the LTOP limit and related topics. If the LTOP 10 minute transient limit (discussed in Attachment 1) is more conservative than the NDT heatup or cooldown limits (e.g. lower pressure allowed), the LTOP will inherently protect against heatup and cooldown NDT limits. Note that the LTOP transient is also dependent on the initial pressurizer level (less than 102 inches for TMI-1), the MU flow rate, and any non-condensable gasses that might be in the pressurizer. LTOP is a concern only below the enable temperature. Attachment 1 discussion assumes that potential HPI flow and CFT flow are locked out below the enable temperature.

Enable Temperature

Per Reference 7, the Adjusted Reference Temperature (ART) is 250.5F (rounded to 251F). Code Case N514 requires that the LTOP limits be in effect when the ¼ thickness (¼ T) temperature is less than the ART + 50F. During a cooldown, the coolant temperature is always less than (or equal to) the ¼ T temperature and therefore it is conservative to use the coolant temperature as the LTOP enable set point. However, during a heatup, the ¼ T temperature is always less than the coolant temperature. Therefore, the set point must be the water temperature plus the difference in these two locations. Per Reference 1, the maximum ¼ T temperature will be 16F lower than the coolant temperature (during a maximum allowable heatup rate). Therefore, the enable temperature will be the ART + 50 + 16. Both cases will assume a 12 F temperature uncertainty (as discussed above) resulting in a cooldown enable temperature of 313F and a heatup temperature of 329F.

LTOP Limit

The LTOP limit establishes two heatup/cooldown criteria. The minimum LTOP limit (at 60F) is the pressure used to establish the PORV set-point when the RCS is below the enable temperature. Also, the failed-open MU control valve transient (discussed in detail in Attachment 1) dictates that the maximum pressure during heatups and cooldowns must be low enough to prevent exceeding the LTOP limit for 10 minutes of full makeup flow (at which time operator action is assumed to terminate the pressurization). When the LTOP curve becomes PORV limited, the margin to the actual LTOP limit increases significantly.

There are five uncorrected steady state LTOP location limits (beltline, axial weld, closure head, outlet nozzle and inlet nozzle) that apply between 60F to the enable temperature (with closure head up to 190F) per Reference 1. The axial flaw is more limiting than any other core region location[4]. The closure head is more limiting than the axial flaw above 65F (where the closure head is a constant 625 psig). However, since the closure head correction to the hot leg pressure tap is ~35 psid less than the beltline correction (during 2/0 operation) it will not be actually limiting until it is ~35 psi lower than the axial weld or ~660 psig. The closure head location is ~8ft above the hot leg centerline or~4 psi lower than the center line pressure (node 4 in Reference 5). Prior to pump stars, this correction is essentially ~18 ft elevation or 8 psid. In case 200F 2/0 pump operation, node 4 is at 223.3 psia while the hot leg tap is 174.6 psia. This is a DP of (223.3-4) – 174.6 or ~45 psid (head closure higher than the hot leg tap pressure) per case T2P200.out in Ref. 5. This makes the correction factor 45 psid compared to ~87 psid for the beltline correction factor or 42 psi difference (conservatively use 35 psi difference). Therefore, during 2/0 operation, when the axial weld limit is greater than 660 psig (625+35), the closure head will dominate, remaining at constant 660 psig until the RCS temperature exceeds 180F (where the head closure limit increases from 625 psig to 1250 psig). Prior to any RCPs operating, when the axial weld is greater than 633 psig (625 + 8), the closure head will dominate. The location correction to the hot leg wide range pressure tap are based on 0/0 pumps operating up to 100F (where the first two pumps are started) and then based on 2/0[5] pumps up to 200F and finally 2/1 pumps up to the enable temperature (see Table 2 below). This PT limit is corrected to the RCS "A" hot leg pressure tap (instead of the low range tap in the pressurizer). Since the surge line limit prevents RC pump operation below ~100°F, this was be chosen as the minimum temperature for RC pump operation (see Reference 2). The minimum 3rd pump start was specified at 200F (per Reference 4) but waiting till 250F to start the 3rd pump may provide a larger PT window (see Figure 1). The 4th pump start (for DP correction) will be conservatively set at the enable temperature of 329F for location correction (4th pump start is presently set at 400F per Reference 4). The final operating procedures will have a vertical limit at this enable temperature, which will also include the step pressure change due to increased DP for 4-pump operation (even though the 4th pump cannot be started till 400F). Reference 5 shows all the FSPLIT analyses that were used to establish the location correction for the RCS pressure from the "A" hot leg tap to the RV beltline.

Reference 6 defines the pressure difference needed to accommodate the 10-minute failed open makeup control valve transient. The makeup flow rate as a function of RCS pressure and initial pressurizer level used in Reference 6 (100 inches indicated) was based on Reference 2. Reference 6 also assumed no non-condensable gasses in the pressurizer, also based on Reference 2. The flow rates used in this analysis were:

RCS Pressure @ Hot Leg tap psig	Maximum Makeup Flow (gpm)
101.8	300
193.2	295
282.9	290
371.1	285
457.7	280
626.1	270

4 Per Reference 9 the actual location of the limiting weld is 30 cm below the beltline or ~1 ft. Therefore, the beltline pressure drop correction will be used for this weld.

5 The 0/2 correction factor results in a higher allowable LTOP limit curve. However, since the PORV set point will be established based on the more conservative (lower) 2/0 pump combination, the 10 minute transient will also be established from the 2/0 pump combination for all startup and cooldown pump configurations.

An equation was developed to describe the initial allowable pressure as a function of the final pressure for the 10-minute transient.

$$\text{Init Press}^{6} = 147058.8\{-1.2588+[1.2588^2 - 13.6E\text{-}6(-14.404 - (P_{final} + 14.7))]^{0.5}\} - 14.7$$

Where P_{final} = the pressure at the end of the 10-minutes of failed open makeup control valve flow (based on Reference 6 inputs) and Init Press is the initial pressure (psig). Setting Pfinal to the maximum allowable LTOP pressure, Reference 1, the LTOP limit can be determined as "Initial Pressure".

Example – for a maximum allowable pressure of 625 psig (Pfinal=625), the 10-minute transient equation would predict 504.2 psig maximum allowable LTOP pressure (Initial Pressure=504) to withstand the 10-minute transient. Figure 1 of Reference 6 shows that an initial pressure of 504 psig (519 psia) will increase in pressure by 121 psi resulting in 625 psig.

Figure 7 shows the maximum allowable initial pressure to permit 10 minutes of full makeup and not exceed the LTOP limit. The bases of this figure is from Table 2 below. Below the enable temperature, the RCS must always be below this pressure when MU is not isolated and the RCS is not opened to atmosphere (see PORV Flow section below).

Column 3 is the limiting pressure at the temperature of column 1 from Reference 1. Columns 4 through 8 are a list of DP correction factors for different pump combination with the one used in bold print. Column 9 is the solution to the above equation with column 3 being the Pfinal input. Column 10 subtracts the appropriate (bold) correction factors of columns 4 through 8 from columns 9 (for 2/0, 2/1, 2/2 startup). Column 11 is the same as column 10 except it uses the 0/2, 1/2, 2/2 heatup. Finally, column 12 is the same as column 10 except it is "smoothed" to show limits that do not increase as temperatures decrease.

PORV Set-point

The PORV set-point for LTOP protection must be set at the minimum pressure to ensure the reactor vessel LTOP limits are not violated. The PORV must be set to protect for the minimum allowable pressure. This will be either at 60F (with pumps off), or 100F with 2/0 pumps operating whichever is the more limiting. At 60F, the LTOP limit is 623 psig (Reference 1). Using the 23 psi correction to the hot leg tap and using the 20 psi set point uncertainty (Reference 2), the PORV must be set at 623-20-23= 580 psig at 60F. At 100F, the LTOP limit is 641 psig (Reference 1). Using the 69 psi correction to the pressurizer (low range tap) and using the 20 psi set point uncertainty, the PORV must be set at 641-20-69 = 552 psig. Note that the pressure in the pressurizer is ~18 psi higher than the hot leg tap at 100F 2/0 pump operation per Table 1 above (at 100 inch pressurizer level). Therefore, the location correction to the pressurizer is 87 psi (Table 1) minus 18 or 69 psid. The head of steam to the top of the pressurizer is negligible. The PORV senses the pressurizer pressure and not the hot leg pressure.

6 Reference 6 presented the final pressure as a function of initial pressure. Using the quadratic equation, initial pressure was determined was determined as a function of final pressure as shown above.

Table 2 LTOP Limits

Temp, F	Temp F	Limiting Max Allowable Pressure	RC Pump Combination DP correction (Ref 7) Beltline to Hot Leg Tap					Initial Pressure Limit @ Beltline	Startup With 2/0, 2/1, 2/2	Startup With 0/2, 1/2, 2/2	Min 2/0 Envelop Including PORV setpoint psig
		psig	0/0	2/0	0/2	2/1	1/2	psig			
Ref 1	+12F Uncrt	Ref 1	Conservative application of Location corrections from above					Ref 6	Location and uncrt (-25 psi) corrected		
60	72	623	**23**	Not legal	Not legal	Not legal	Not legal	502.6	454.6	454.6	403.9
65	77	625	**23**	Not legal	Not legal	Not legal	Not legal	504.2	456.2	456.2	403.9
70	82	626	**23**	Not legal	Not legal	Not legal	Not legal	505.0	457.0	457.0	403.9
75	87	628	**23**	Not legal	Not legal	Not legal	Not legal	506.6	458.6	458.6	403.9
80	92	630	**23**	Not legal	Not legal	Not legal	Not legal	508.2	460.2	460.2	403.9
85	97	632	**23**	Not legal	Not legal	Not legal	Not legal	509.7	461.7	461.7	403.9
90	102	633**	**23**	Not legal	Not legal	Not legal	Not legal	510.5	462.5	462.5	403.9
95	107	633 **	**23**	Not legal	Not legal	Not legal	Not legal	510.5	462.5	462.5	403.9
100	112	633 **	**23**	Not legal	Not legal	Not legal	Not legal	510.5	462.5	462.5	403.9
100.1	112.1	641	23	**88**	33	Not legal	Not legal	516.9	**403.9**	458.9	403.9
105	117	644	23	**88**	33	Not legal	Not legal	519.3	406.3	461.3	406.3
110	122	647	23	**88**	33	Not legal	Not legal	521.6	408.6	463.6	408.6
115	127	652	23	**88**	33	Not legal	Not legal	525.6	412.6	467.6	412.6
120	132	656	23	**88**	33	Not legal	Not legal	528.8	415.8	470.8	415.8
125	137	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
130	142	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
135	147	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
140	152	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
145	157	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
150	162	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
155	167	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
160	172	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
165	177	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
170	182	660 **	23	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
175	187	660 **	22	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
180	192	660 **	22	**88**	33	Not legal	Not legal	531.9	418.9	473.9	418.9
180.1	192.1	765	22	**88**	33	Not legal	Not legal	615.1	502.1	557.1	502.1
185	197	781	22	**88**	33	Not legal	Not legal	627.7	514.7	569.7	514.7
190	202	799	22	**88**	33	Not legal	Not legal	642.0	529.0	584.0	529.0
195	207	819	22	**88**	33	Not legal	Not legal	657.8	544.8	599.8	544.8
200	212	842	22	**88**	33	Not legal	Not legal	676.0	547.9	618.0	547.9
200.1	212.1	842	22	87	33	**100**	65	676.0	551.0	586.0	552.0
210	222	893	22	87	33	**100**	65	716.4	591.4	626.4	552.0
215	227	923	22	87	33	**100**	65	740.1	615.1	650.1	552.0
220	232	956	22	87	33	**100**	65	766.2	641.2	676.2	552.0
225	237	992	22	87	33	**100**	65	794.7	669.7	704.7	552.0
230	242	1033	22	87	33	**100**	65	827.1	702.1	737.1	552.0
235	247	1077	22	87	33	**100**	65	861.9	736.9	771.9	552.0
240	252	1127	22	87	33	**100**	65	901.5	776.5	811.5	552.0
245	257	1181	22	87	33	**100**	65	944.1	819.1	854.1	552.0
250	262	1241	22	87	33	**100**	65	991.5	866.5	901.5	552.0
250	262.01	1241	22	86	31	**100**	65	991.5	866.5	901.5	552.0

255	267	1308	22	86	31	100	65	1044.5	919.5	954.5	552.0
260	272	1381	22	86	31	100	65	1102.1	977.1	1012.1	552.0
265	277	1462	22	86	31	100	65	1166.1	1041.1	1076.1	552.0
270	282	1552	22	86	31	100	65	1237.1	1112.1	1147.1	552.0
275	287	1651	22	86	31	100	65	1315.2	1190.2	1225.2	552.0
280	292	1761	22	86	31	100	65	1402.0	1277.0	1312.0	552.0
285	297	1882	22	86	31	100	65	1497.3	1372.3	1407.3	552.0
290	302	2016	22	86	31	100	65	1602.9	1477.9	1512.9	552.0
295	307	2164	22	86	31	100	65	1719.4	1594.4	1629.4	552.0
300 *	312	2327	22	86	31	100	65	1847.6	1722.6	1757.6	552.0
305	317	2508	22	86	31	100	65	1989.9	1864.9	1899.9	552.0
310	322	2708	22	86	31	100	65	2147.1	2022.1	2057.1	552.0
315	327	2929	22	86	31	100	65	2320.5	2195.5	2230.5	552.0
317	329	3026	22	86	31	100	65	2396.8	2271.8	2306.8	552.0
317	329.01	3026	22	86	31	100	65	2396.6	2271.6	2306.6	2271.6
320	332	3172	22	86	31	100	65	2511.1	2386.1	2421.1	2386.1

*Note that this table includes the heatup enable temperature. As shown on Figure 7, the enable temperature (where the PORV must be at the lower setpoint) could be as low as 313F for cooldowns.

** Closure Head Limited

LTOP Pressurizer Level Vs Pressure (Nitrogen)

When the RCS is filled and vented and the pressurizer has a blanket of nitrogen (vs. steam), the 10 minute transient limit previously defined will not apply. The N_2 will not condense and therefore, will pressurize faster during the MU event. However, TMI-1 will have the MU pumps locked out during this condition (Reference 2) and therefore pressurization with a nitrogen blanket is not a concern.

MU Tank Volume

If the MU tank volume depletes (or the transient is terminated due to MU pump NPSH limits), in less than 10 minutes, the pressurization transient will stop. At the present time, the MU tank volume will allow for more than the 10 minute flow time and cannot be used to minimize the maximum LTOP pressure. However, it is mentioned here to potentially be used in the future if other initial conditions ever change.

PORV Flow

The PORV flow rate must be large enough to prevent pressure excursions created by the water entering the pressurizer (in order to assure that the LTOP limit is maintained). Reference 2 shows that the PORV has a flow area of 0.94 in^2. Per Reference 8 page A22, the maximum pressure drop at 485[7] psig through a contraction/expansion (valve) with a conservatively assumed loss factor of 2 is .59 x 500 psia or 295 psid. This results in a down stream pressure of 205 psia at the choked flow point. Using the sonic velocity equation from Reference 8, $v=(k \times g \times 144 \times P'/rho)^{0.5}$ k= isentropic exponent = 1.27 (page A9 Reference 8), rho = density = 1.078 lb/cuft , P'= 205 psia. $V= (1.27 \times 32.2 \times 144 \times 205/1.078)^{0.5}$ =1058 ft/sec. The maximum volumetric flow (for an area of .94 in^2), is 1058 x .94/144 x 60 = 414 cuft per minute. Reference 15 shows that using the critical flow tables (Homogenous Equilibrium Method), the maximum flow is 217 cuft/min. This uses a beta ratio (diameter ratio) of zero, where for all Re numbers above 200 Cd ~0.6 (Reference 8), 500 psia stagnation pressure, and saturated enthalpy of 1204.6 BTU/lb, the flow rate is 600 lb/sec-ft^2. This is

7 This is less than the allowable PORV set point and therefore will be conservative for this calculation. This pressure was used because TMI-1 may elect to maintain the present PORV set point of ~485 psig.

600 x .0065 ft^2 x 60 sec = 235 lb/min. At 1.08 lb/cuft this is 217 cuft/min. Even with this conservative calculation, the maximum flow rate is much greater than the maximum makeup flow at 500 psig is 275 gpm or 36 cuft per minute. Therefore, the pressurizer pressure will not exceed the set point pressure due to approaching sonic velocities during an LTOP event. The minimum flow area need to pass 36 cuft/min is 36/217 x 0.94 in^2 = 0.16 in^2. The PORV should never experience a solid water flow rate under these circumstances because the pressurizer cannot fill (from 100" initial level) during the 10-minute MU transient. Furthermore, the MU tank will deplete long before the pressurizer fills.

Water Flow

The LTOP scenario is not required for protection when an opening in the RCS can pass at least as much water as the MU can supply to the RCS. Using a Beta (diameter) Ratio approaching zero, flow coeff C= discharge Coeff C_d = ~0.6

From Reference 8 for Beta~0
Flow (gpm)=236 d^2 C x $(DP/rho)^{0.5}$ Where d=diameter in inches
For 485 psid, 62.1 = rho
275 gpm = d^2 x 395.5
Solving for d = .834 in resulting in an area of 0.55 in^2
The same calculation using the critical flow tables (P=500 psia, h=29.5 btu/lb @275 gpm results in an area of 0.54 in^2

Final LTOP Limit

The final (enveloping) LTOP limit is shown on Figure 7. It will be 403.9 psig up to 100F (112F with temperature uncertainty). It will then increase to ~419 psig to be closure head limit at ~137F and maintain this limit until 180F (192 with uncertainty). It will then increase to the PORV set point of 552 psig at ~200F and remain at this pressure until the RCS temperature exceeds 329F. Figure 7 shows the recommended PT points for plotting.

Cooldown & Heatup NDT

The heatup and cooldown NDT limits were calculated in Reference 1 for two types of transients, a ramp and a step change. The maximum allowable rates of temperature changes cases are shown in Reference 2 and analyzed in Reference 1. Some initial (preliminary) parameterizations were performed on DHRS initiation and RC pump shut-off to insure conservative results. Reference 1 analyzed the enveloping worst case of each. The conservatisms used in these analyses are discussed in Reference 2. In general, these analyses used step changes in water temperatures every time a system was started or stopped (e.g., cold water in cold legs just after the first RC pump start, cold water in DHRS just after system initiation, etc.). Similar to the LTOP analyses, the limits of each weld location were compared to find the most limiting. These values were then corrected to reflect the pressure at the hot leg tap. This approach includes an inherent ~15 psi conservatism when monitoring the low range pressurizer tap. The 12F and 25 psi instrument uncertainties were then added to the curves. Finally, an enveloping "smoothed" curve was generated for both heatup and cooldown.

Figures 8 shows the limiting uncorrected results of Reference 1 for both the ramp and step cooldowns for the axial weld and the closure head limits. Per Reference 1, these are the limiting cooldown NDT requirements in the TMI vessel. Per discussion above, the closure head correction factor is ~35 psi less than the beltline correction factor and therefore, the Reference 1 closure head was increased 35 psi to put both curves on the same correction basis. Figures 8 also shows these results with the location corrections and the instrument uncertainty included. The location correction

conservatively assumed four RC pump operating down to 329F and then 2/1 (three pumps) down to the allowable pump shut-off limit of 200F (analyzed at 190F to incorporate some conservatism). Above 310F, the pressures listing in Table 3 are based on the outlet nozzle; however, the belt line correction factor was conservatively used (either of these sets of data is above the pressurizer safety valve set-point). The 2/0 correction was not used so as to provide for potential pump switching at low pressures where 3 pump is the maximum allowable number below 400F. The final enveloping CD NTD limit is calculated in Table 3 below. Note that the change from the 3-pump to 4-pump correction factor was included in the vertical step that occurs at the enable temperature (even though the 4th pump cannot be operating below 400F). The step at ~235F is due to the DHRS initiation (see Reference 1 for additional discussion). Per Reference 2, the wide range (worst case) uncertainties is applied to the entire range. Similar to the LTOP case, the location correction is to the hot leg tap, which is ~15 psi conservative when monitoring the pressurizer (low range) tap below 450 psig. Figure 8 also shows a set of recommended P-T points to describe the final cooldown limit. Table 3 shows the detailed results of Figure 8.

Figure 9 shows the same results for the heatup NDT limits. It includes the limiting uncorrected results from Reference 1, the location corrections and the instrument uncertainty. The location correction assumes that the 2/0 RC pumps start at 100F, the 3rd pump starts at 200F and the 4th pump starts at 329F, the final heatup enable temperature (conservatively below the allowable 400F 4th pump startup temperature). Figure 9 also shows a set of recommended P-T points to describe the final heatup limit. Table 4 shows the detailed results of Figure 9. Similar to the cooldown limits, above 335F, the pressures listing in Table 4 are based on the outlet nozzle; however, the beltline correction factor was conservatively used (either of these sets of data is above the pressurizer safety valve set-point).

Table 3 Cooldown NDT Limits

Temp, F	Temp F	Min Press	RC Pump Combination DP correction (Ref 5) Beltline to Hot Leg Tap						Location corrected	Cooldown 2/1, 2/2, 2/0	Min 2/0 Envelop setpoint psig
		psig	0/0	2/0	0/2	2/1	1/2	2/2			
Ref 1	+12F Uncrt	Ref 1	Conservative application of Location corrections from above							+25 psi Uncrt	
70	82	535	23	Not legal	Not legal	Not legal	Not legal	Not legal	512	487	487
75	87	537	23	Not legal	Not legal	Not legal	Not legal	Not legal	514	489	489
80	92	539	23	Not legal	Not legal	Not legal	Not legal	Not legal	516	491	491
85	97	542	23	Not legal	Not legal	Not legal	Not legal	Not legal	519	494	494
90	102	545	23	Not legal	Not legal	Not legal	Not legal	Not legal	522	497	497
95	107	548	23	Not legal	Not legal	Not legal	Not legal	Not legal	525	500	500
100	112	552	23	Not legal	Not legal	Not legal	Not legal	Not legal	529	504	504
105	117	556	23	Not legal	Not legal	Not legal	Not legal	Not legal	533	508	508
110	122	560	23	Not legal	Not legal	Not legal	Not legal	Not legal	537	512	512
115	127	565	23	Not legal	Not legal	Not legal	Not legal	Not legal	542	517	517
120	132	571	23	Not legal	Not legal	Not legal	Not legal	Not legal	548	523	523
125	137	577	23	Not legal	Not legal	Not legal	Not legal	Not legal	554	529	529
130	142	584	23	Not legal	Not legal	Not legal	Not legal	Not legal	561	536	536
135	147	592	23	Not legal	Not legal	Not legal	Not legal	Not legal	569	544	544
140	152	601	23	Not legal	Not legal	Not legal	Not legal	Not legal	578	553	553
145	157	611	23	Not legal	Not legal	Not legal	Not legal	Not legal	588	563	563
150	162	622	23	Not legal	Not legal	Not legal	Not legal	Not legal	599	574	574
155	167	633 **	23	Not legal	Not legal	Not legal	Not legal	Not legal	610	585	585

160	172	633 **	23	Not legal	Not legal	Not legal	Not legal	Not legal	610	585	585
165	177	633 **	23	Not legal	Not legal	Not legal	Not legal	Not legal	610	585	585
170	182	633 **	23	Not legal	Not legal	Not legal	Not legal	Not legal	610	585	585
175	187	633 **	23	Not legal	Not legal	Not legal	Not legal	Not legal	610	585	585
180	192	633 **	23	Not legal	Not legal	Not legal	Not legal	Not legal	610	585	585
180.1	192.1	706	23	Not legal	Not legal	Not legal	Not legal	Not legal	683	658	620*
190	202	733	23	Not legal	Not legal	Not legal	Not legal	Not legal	710	685	638*
195	207	756	23	Not legal	Not legal	Not legal	Not legal	Not legal	733	708	656*
200	212	781	23	Not legal	Not legal	Not legal	Not legal	Not legal	758	733	656
200.1	212.1	781	22	87	33	100	65	Not legal	681	656	656
205	217	808	22	87	33	100	65	Not legal	708	683	683
210	222	838	22	87	33	100	65	Not legal	738	713	713
215	227	842	22	87	33	100	65	Not legal	742	717	717
220	232	842	22	87	33	100	65	Not legal	742	717	717
225	237	842	22	87	33	100	65	Not legal	742	717	717
230	242	842	22	87	33	100	65	Not legal	742	717	717
235	247	842	22	87	33	100	65	Not legal	742	717	717
240	252	842	22	87	33	100	65	Not legal	742	717	717
240.1	252.1	1110	22	87	33	100	65	Not legal	1010	985	985
245	257	1181	22	87	33	100	65	Not legal	1081	1056	1056
250	262	1241	22	87	33	100	65	Not legal	1141	1116	1116
255	267	1308	22	87	33	100	65	Not legal	1208	1183	1183
260	272	1381	22	87	33	100	65	Not legal	1281	1256	1256
265	277	1462	22	87	33	100	65	Not legal	1362	1337	1337
270	282	1552	22	87	33	100	65	Not legal	1452	1427	1427
275	287	1651	22	87	31	100	65	Not legal	1551	1526	1526
280	292	1761	22	87	31	100	65	Not legal	1661	1636	1636
285	297	1882	22	87	31	100	65	Not legal	1782	1757	1757
290	302	2016	22	87	31	100	65	Not legal	1916	1891	1891
295	307	2164	22	87	31	100	65	Not legal	2064	2039	2039
300	312	2327	22	86	31	100	65	Not legal	2227	2202	2202
305	317	2508	22	86	31	100	65	Not legal	2408	2383	2383
310	322	2544	22	86	31	100	65	Not legal	2444	2419	2419
315	327	2548	22	86	31	100	65	Not legal	2448	2423	2423
317	329	2549	22	86	31	100	65	Not legal	2449	2424	2424
317	329	2549	22	86	31	100	65	112	2437	2412	2412
320	332	2550	22	86	31	100	65	112	2438	2413	2413
325	337	2553	22	86	31	100	65	112	2441	2416	2416
330	342	2556	22	86	31	97	61	112	2444	2419	2419
335	347	2559	22	86	31	97	61	112	2447	2422	2422
340	352	2562	22	86	31	97	61	112	2450	2425	2425
345	357	2565	22	86	31	97	61	112	2453	2428	2428
350	362	2569	22	86	31	97	61	112	2457	2432	2432
355	367	2573	22	86	31	97	61	112	2461	2436	2436
360	372	2577	22	86	31	97	61	112	2465	2440	2440
365	377	2581	22	86	31	97	61	112	2469	2444	2444
370	382	2584	22	86	31	97	61	112	2472	2447	2447
375	387	2589	22	86	31	97	61	112	2477	2452	2452
380	392	2594	22	86	31	97	61	112	2482	2457	2457
385	397	2600	22	86	31	97	61	112	2488	2463	2463
390	402	2606	22	86	31	97	61	112	2494	2469	2469
395	407	2612	22	86	31	97	61	112	2500	2475	2475

400	412	2618	22	86	31	97	61	**112**	2506	2481	2481
405	417	2625	22	80	31	97	61	**112**	2513	2488	2488
410	422	2632	22	80	31	97	61	**112**	2520	2495	2495
415	427	2640	22	80	31	97	61	**112**	2528	2503	2503
420	432	2647	22	80	31	97	61	**112**	2535	2510	2510
425	437	2656	22	80	31	97	61	**112**	2544	2519	2519
430	442	2664	22	80	31	97	61	**112**	2552	2527	2527
435	447	2673	22	80	31	97	61	**112**	2561	2536	2536
440	452	2682	22	80	28	92	57	**106**	2576	2551	2551
445	457	2692	22	80	28	92	57	**106**	2586	2561	2561
450	462	2702	22	80	28	92	57	**106**	2596	2571	2571
455	467	2713	22	80	28	92	57	**106**	2607	2582	2582
460	472	2724	22	80	28	92	57	**106**	2618	2593	2593
465	477	2736	22	80	28	92	57	**106**	2630	2605	2605
470	482	2748	22	80	28	92	57	**106**	2642	2617	2617
475	487	2761	22	80	28	92	57	**106**	2655	2630	2630
480	492	2774	22	80	28	92	57	**106**	2668	2643	2643
485	497	2788	22	80	28	92	57	**106**	2682	2657	2657
490	502	2803	22	80	28	92	57	**106**	2697	2672	2672
495	507	2818	22	80	28	92	57	**106**	2712	2687	2687
500	512	2834	22	80	28	92	57	**106**	2728	2703	2703
505	517	2850	22	80	28	92	57	**106**	2744	2719	2719
510	522	2865	22	80	28	92	57	**106**	2759	2734	2734
515	527	2883	22	80	28	92	57	**106**	2777	2752	2752
520	532	2901	22	80	28	92	57	**106**	2795	2770	2770
525	537	2921	22	80	28	92	57	**106**	2815	2790	2790
530	542	2940	22	80	28	92	57	**106**	2834	2809	2809
535	547	2961	22	80	28	92	57	**106**	2855	2830	2830
540	552	2982	22	75	26	92	52	**106**	2876	2851	2851
545	557	3003	22	75	26	85	52	**96**	2907	2882	2882
550	562	3024	22	75	26	85	52	**96**	2928	2903	2903
555	567	3046	22	75	26	85	52	**96**	2950	2925	2925
560	572	3066	22	75	26	85	52	**96**	2970	2945	2945
565	577	3083	22	75	26	85	52	**96**	2987	2962	2962
570	582	3093	22	75	26	85	52	**96**	2997	2972	2972

*The pressures are reduced to provide a "smoothed" curve.

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Table 4 Heatup NDT Limits

Temp, F	Temp F	Min Press	RC Pump Combination DP correction (Ref 5) Beltline to Hot Leg Tap						Location corrected	Startup With 2/0, 2/1, 2/2	Min 2/0 Envelop setpoint psig
		psig	0/0	2/0	0/2	2/1	1/2	2/2			
Ref 1	+12F Uncrt	Ref 1	Conservative application of Location corrections from above								
60	72	623	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	600	575	479
65	77	619	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	596	571	479

70	82	615	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	592	567	479
75	87	611	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	588	563	479
80	92	607	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	584	559	479
85	97	603	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	580	555	479
90	102	599	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	576	551	479
95	107	595	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	572	547	479
100	112	592	**23**	Not legal	Not legal	Not legal	Not legal	Not legal	569	544	479
100.1	112.1	592	23	**88**	33	Not legal	Not legal	Not legal	504	479	479
105	117	592	23	**88**	33	Not legal	Not legal	Not legal	504	479	479
110	122	594	23	**88**	33	Not legal	Not legal	Not legal	506	481	481
115	127	597	23	**88**	33	Not legal	Not legal	Not legal	509	484	484
120	132	602	23	**88**	33	Not legal	Not legal	Not legal	514	489	489
125	137	608	23	**88**	33	Not legal	Not legal	Not legal	520	495	495
130	142	616	23	**88**	33	Not legal	Not legal	Not legal	528	503	503
135	147	624	23	**88**	33	Not legal	Not legal	Not legal	536	511	511
140	152	634	23	**88**	33	Not legal	Not legal	Not legal	546	521	521
145	157	646	23	**88**	33	Not legal	Not legal	Not legal	558	533	533
150	162	659	23	**88**	33	Not legal	Not legal	Not legal	571	546	546
155	167	660 **	23	**88**	33	Not legal	Not legal	Not legal	572	547	547
160	172	660 **	23	**88**	33	Not legal	Not legal	Not legal	572	547	547
165	177	660 **	23	**88**	33	Not legal	Not legal	Not legal	572	547	547
170	182	660 **	22	**88**	33	Not legal	Not legal	Not legal	572	547	547
175	187	660 **	22	**88**	33	Not legal	Not legal	Not legal	572	547	547
180	192	660 **	22	**88**	33	Not legal	Not legal	Not legal	572	547	547
180.1	192.1	726	22	**88**	33	Not legal	Not legal	Not legal	638	613	613
185	197	738	22	**88**	33	Not legal	Not legal	Not legal	650	625	625
190	202	751	22	**88**	33	Not legal	Not legal	Not legal	663	638	638
195	207	766	22	**88**	33	Not legal	Not legal	Not legal	678	653	653
200	212	783	22	**88**	33	Not legal	Not legal	Not legal	695	670	658*
200.1	212.1	783	22	87	33	**100**	65	Not legal	683	658	658
210	222	821	22	87	33	**100**	65	Not legal	721	696	696
215	227	843	22	87	33	**100**	65	Not legal	743	718	718
220	232	867	22	87	33	**100**	65	Not legal	767	742	742
225	237	894	22	87	33	**100**	65	Not legal	794	769	769
230	242	924	22	87	33	**100**	65	Not legal	824	799	799
235	247	957	22	87	33	**100**	65	Not legal	857	832	832
240	252	994	22	87	33	**100**	65	Not legal	894	869	869
245	257	1034	22	87	33	**100**	65	Not legal	934	909	909
250	262	1079	22	86	31	**100**	65	Not legal	979	954	954
255	267	1128	22	86	31	**100**	65	Not legal	1028	1003	1003
260	272	1182	22	86	31	**100**	65	Not legal	1082	1057	1057
265	277	1242	22	86	31	**100**	65	Not legal	1142	1117	1117
270	282	1308	22	86	31	**100**	65	Not legal	1208	1183	1183
275	287	1381	22	86	31	**100**	65	Not legal	1281	1256	1256
280	292	1461	22	86	31	**100**	65	Not legal	1361	1336	1336
285	297	1550	22	86	31	**100**	65	Not legal	1450	1425	1425

290	302	1648	22	86	31	**100**	65	Not legal	1548	1523	1523
295	307	1756	22	86	31	**100**	65	Not legal	1656	1631	1631
300	312	1876	22	86	31	**97**	61	Not legal	1779	1754	1754
305	317	2008	22	86	31	**97**	61	Not legal	1911	1886	1886
310	322	2154	22	86	31	**97**	61	Not legal	2057	2032	2032
315	327	2315	22	86	31	**97**	61	Not legal	2218	2193	2193
317	329	2386	22	86	31	**97**	61	Not legal	2289	2264	2249*
317	329	2386	22	86	31	97	61	**112**	2274	2249	2249
320	332	2493	22	86	31	97	61	**112**	2381	2356	2356
325	337	2690	22	86	31	97	61	**112**	2578	2553	2553
330	342	2907	22	86	31	97	61	**112**	2795	2770	2770
335	347	3093	22	86	31	97	61	**112**	2981	2956	2956
340	352	3093	22	86	31	97	61	**112**	2981	2956	2956
345	357	3093	22	86	31	97	61	**112**	2981	2956	2956
350	362	3093	22	86	31	97	61	**112**	2981	2956	2956
355	367	3093	22	86	31	97	61	**112**	2981	2956	2956
360	372	3093	22	86	31	97	61	**112**	2981	2956	2956
365	377	3093	22	86	31	97	61	**112**	2981	2956	2956
370	382	3093	22	86	31	97	61	**112**	2981	2956	2956
375	387	3093	22	86	31	97	61	**112**	2981	2956	2956
380	392	3093	22	86	31	97	61	**112**	2981	2956	2956
385	397	3093	22	86	31	97	61	**112**	2981	2956	2956
390	402	3093	22	86	31	97	61	**112**	2981	2956	2956
395	407	3093	22	86	31	97	61	**112**	2981	2956	2956
400	412	3093	22	80	28	92	57	**106**	2987	2962	2962
405	417	3093	22	80	28	92	57	**106**	2987	2962	2962
410	422	3093	22	80	28	92	57	**106**	2987	2962	2962
415	427	3093	22	80	28	92	57	**106**	2987	2962	2962
420	432	3093	22	80	28	92	57	**106**	2987	2962	2962
425	437	3093	22	80	28	92	57	**106**	2987	2962	2962
430	442	3093	22	80	28	92	57	**106**	2987	2962	2962
435	447	3093	22	80	28	92	57	**106**	2987	2962	2962
440	452	3093	22	80	28	92	57	**106**	2987	2962	2962
445	457	3093	22	80	28	92	57	**106**	2987	2962	2962
450	462	3093	22	80	28	92	57	**106**	2987	2962	2962
455	467	3093	22	80	28	92	57	**106**	2987	2962	2962
460	472	3093	22	80	28	92	57	**106**	2987	2962	2962
465	477	3093	22	80	28	92	57	**106**	2987	2962	2962
470	482	3093	22	80	28	92	57	**106**	2987	2962	2962
475	487	3093	22	80	28	92	57	**106**	2987	2962	2962
480	492	3093	22	80	28	92	57	**106**	2987	2962	2962
485	497	3093	22	80	28	92	57	**106**	2987	2962	2962
490	502	3093	22	80	28	92	57	**106**	2987	2962	2962
495	507	3093	22	80	28	92	57	**106**	2987	2962	2962
500	512	3093	22	75	26	92	52	**106**	2987	2962	2962
505	517	3093	22	75	26	85	52	**96**	2997	2972	2972

510	522	3093	22	75	26	85	52	**96**	2997	2972	2972
515	527	3093	22	75	26	85	52	**96**	2997	2972	2972
520	532	3093	22	75	26	85	52	**96**	2997	2972	2972
525	537	3093	22	75	26	85	52	**96**	2997	2972	2972
530	542	3093	22	75	26	85	52	**96**	2997	2972	2972
535	547	3093	22	75	26	85	52	**96**	2997	2972	2972
540	552	3093	22	75	26	85	52	**96**	2997	2972	2972
545	557	3093	22	75	26	85	52	**96**	2997	2972	2972
550	562	3093	22	75	26	85	52	**96**	2997	2972	2972
555	567	3093	22	75	26	85	52	**96**	2997	2972	2972
560	572	3093	22	75	26	85	52	**96**	2997	2972	2972

*The pressures are reduced to provide a "smoothed" curve.
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Combined LTOP, HU, CD, Head Closure Limits

Figure 10 shows the enveloping CD, HU, LTOP limits on the same plot. The LTOP requirements are more limiting than the entire HU or CD limits. The only difference in the procedural NDT/LTOP curves will be the enable temperature (discussed above). The cooldown enable temperature is 313F and the heatup enable temperature is 329F. The final procedures will show the limiting heatup/cooldown LTOP limit as shown on Figure 10. Attachment 2 shows a composite of the heatup and cooldown limits along with some pertinent notes.

In Service Leak Hydro (ISLH)

The ISLH limit is not currently in the TMI-1 heatup and cooldown procedures but is included in the calculation to be consistent with previous requirements to include ISLH calculation (even though they are apparently not used by any procedures). The Reference 1 uncorrected limit and the RC pump location correction are shown on Figure 11. Unlike the HU and CD limits, this curve is the overall limiting PT for all the HU and CD ISLH results from Reference 1. It also assumed the worst case pump start corrections (e.g., 2/0 pumps were operating at 100F which is conservative for the cooldown where they are off at 200F). This curve is basically limited by the head closure up to 150F and ramp CD (from belt line) up to ~250F and then ramp HU (from belt line) to 320F, and ramp cooldown (outlet nozzle) for the remainder of the curve. Also, since the head closure was most limiting prior to RC pump starts, it was given an effective zero location correction up to 100F (RC pump start) and a location correction of 88 psi- 35 psi = 53 psi (based on the 625 psi limit at the head closure) up to 150F. This pressure established the absolute minimum ISLH pressure. Table 5 lists the enveloping ISLH curve. Attachment 2 shows the final ISLH curve from Figure 11 along with some pertinent notes.

Table 5 ISLH Limits

T*emp, F	Temp F	Min Press	RC Pump Combination DP correction (Ref 5) Beltline to Hot Leg Tap						Location corrected	Instrument Uncertainty
		psig	0/0	2/0	0/2	2/1	1/2	2/2		
Ref 1	+12F Uncrt	Ref 1	Conservative application of Location corrections from above						Ref 5	-25 psi
60	72	625	**0**	Not legal	Not legal	Not legal	Not legal	Not legal	625	(600)547
100	112	625	**0**	Not legal	Not legal	Not legal	Not legal	Not legal	625	(600)547
100.01	112	625	23	**53**	33	Not legal	Not legal	Not legal	572	547
150	162	625	23	**53**	33	Not legal	Not legal	Not legal	572	547
150	162	829	23	**88**	33	Not legal	Not legal	Not legal	741	716
155	167	845	23	**88**	33	Not legal	Not legal	Not legal	757	732
160	172	864	23	**88**	33	Not legal	Not legal	Not legal	776	751
165	177	885	23	**88**	33	Not legal	Not legal	Not legal	797	772
170	182	909	23	**88**	33	Not legal	Not legal	Not legal	821	796
175	187	923	23	**88**	33	Not legal	Not legal	Not legal	835	810
190	202	977	23	**88**	33	Not legal	Not legal	Not legal	889	864
195	207	1008	23	**88**	33	Not legal	Not legal	Not legal	920	895
200	212	1041	23	**88**	33	Not legal	Not legal	Not legal	953	928
200	212	1041	22	87	33	**100**	65	Not legal	941	916
205	217	1068	22	87	33	**100**	65	Not legal	968	943
210	222	1094	22	87	33	**100**	65	Not legal	994	969
215	227	1122	22	87	33	**100**	65	Not legal	1022	997
220	232	1122	22	87	33	**100**	65	Not legal	1022	997
225	237	1122	22	87	33	**100**	65	Not legal	1022	997
230	242	1122	22	87	33	**100**	65	Not legal	1022	997
235	247	1122	22	87	33	**100**	65	Not legal	1022	997
240	252	1122	22	87	33	**100**	65	Not legal	1022	997
240	252	1325	22	87	33	**100**	65	Not legal	1225	1200
245	257	1379	22	87	33	**100**	65	Not legal	1279	1254
250	262	1438	22	87	33	**100**	65	Not legal	1338	1313
255	267	1504	22	87	33	**100**	65	Not legal	1404	1379
260	272	1576	22	87	33	**100**	65	Not legal	1476	1451
265	277	1656	22	87	33	**100**	65	Not legal	1556	1531
270	282	1744	22	87	33	**100**	65	Not legal	1644	1619
275	287	1841	22	87	33	**100**	65	Not legal	1741	1716
280	292	1948	22	87	33	**100**	65	Not legal	1848	1823
285	297	2067	22	87	33	**100**	65	Not legal	1967	1942
290	302	2197	22	87	33	**100**	65	Not legal	2097	2072
295	307	2342	22	87	33	**100**	65	Not legal	2242	2217
300	312	2501	22	87	33	**100**	65	Not legal	2401	2376
305	317	2677	22	87	33	**100**	65	Not legal	2577	2552
310	322	2872	22	87	33	**100**	65	Not legal	2772	2747
315	327	3087	22	87	33	**100**	65	Not legal	2987	2962
320	332	3324	22	87	33	**100**	65	Not legal	3224	3199
325	337	3404	22	87	33	**100**	65	Not legal	3304	3271*
330	342	3408	22	86	31	**100**	65	Not legal	3308	3271*
330	342	3408	22	86	31	97	61	**112**	3296	3271
335	347	3412	22	86	31	97	61	**112**	3300	3275
340	352	3416	22	86	31	97	61	**112**	3304	3279

345	357	3421	22	86	31	97	61	**112**	3309	3284
350	362	3425	22	86	31	97	61	**112**	3313	3288
355	367	3430	22	86	31	97	61	**112**	3318	3293
360	372	3436	22	86	31	97	61	**112**	3324	3299
365	377	3442	22	86	31	97	61	**112**	3330	3305
370	382	3445	22	86	31	97	61	**112**	3333	3308
375	387	3452	22	86	31	97	61	**112**	3340	3315
380	392	3459	22	86	31	97	61	**112**	3347	3322
385	397	3466	22	86	31	97	61	**112**	3354	3329
390	402	3474	22	86	31	97	61	**112**	3362	3337
395	407	3482	22	86	31	97	61	**112**	3370	3345
400	412	3491	22	86	31	97	61	**112**	3379	3354
405	417	3500	22	86	31	97	61	**112**	3388	3363
410	422	3509	22	86	31	97	61	**112**	3397	3372
415	427	3519	22	86	31	97	61	**112**	3407	3382
420	432	3530	22	86	31	97	61	**112**	3418	3393
425	437	3541	22	86	31	97	61	**112**	3429	3404
430	442	3552	22	86	31	97	61	**112**	3440	3415
435	447	3563	22	86	31	97	61	**112**	3451	3426
440	452	3576	22	86	31	97	61	**112**	3464	3439
445	457	3589	22	86	31	97	61	**112**	3477	3452
450	462	3603	22	86	31	97	61	**112**	3491	3466
455	467	3617	22	86	31	97	61	**112**	3505	3480
460	472	3632	22	86	31	97	61	**112**	3520	3495
465	477	3648	22	86	31	97	61	**112**	3536	3511
470	482	3664	22	86	31	97	61	**112**	3552	3527
475	487	3681	22	86	31	97	61	**112**	3569	3544
480	492	3699	22	86	31	97	61	**112**	3587	3562
485	497	3718	22	86	31	97	61	**112**	3606	3581
490	502	3737	22	86	31	97	61	**112**	3625	3600
495	507	3757	22	86	31	97	61	**112**	3645	3620
500	512	3778	22	86	31	97	61	**112**	3666	3641
505	517	3800	22	86	31	97	61	**112**	3688	3663
510	522	3820	22	86	31	97	61	**112**	3708	3683
515	527	3844	22	80	28	92	57	**106**	3738	3713
520	532	3869	22	80	28	92	57	**106**	3763	3738
525	537	3894	22	80	28	92	57	**106**	3788	3763
530	542	3921	22	80	28	92	57	**106**	3815	3790
535	547	3948	22	80	28	92	57	**106**	3842	3817
540	552	3976	22	80	28	92	57	**106**	3870	3845
545	557	4004	22	80	28	92	57	**106**	3898	3873
550	562	4033	22	80	28	92	57	**106**	3927	3902
555	567	4061	22	80	28	92	57	**106**	3955	3930
560	572	4088	22	80	28	92	57	**106**	3982	3957
565	577	4111	22	80	28	92	57	**106**	4005	3980
570	582	4123	22	80	28	92	57	**106**	4017	3992

*The pressures are reduced to provide a "smoothed" curve.

NPSH

This section will develop the minimum allowable NPSH for heatups and cooldowns. The methodology for this development is as follows.

1. Determine the flow rate (gpm) for each pump combination at various temperatures from 100F to 540F. This will be done using FSPLIT (Reference 5) with the RC pump head-capacity curve from Reference 2. FSPLIT will use the 20% tube plugged model from Reference 5 since this will minimize the pump suction pressure due to the additional pressure drop in the SG. The reverse loss factor in the RC pump is 12.4 VH based on Reference 10.
2. Using the required NPSH data for Westinghouse pumps from Reference 2, determine the required NPSH for each of these flow rates. Note that Figure 6 of Reference 2 is based on 70F water and since the vapor pressure at this temperature is less than 1 psia, this figure is effectively the actual required NPSH (in feet and in psia).
3. Vary the pressurizer pressure in FSPLIT until the NPSHr is achieved at the pump suction.
4. Monitor the various important pressures around the loop for each of these cases.
5. Perform a hand calculation check to assure that the proper NPSHa is being calculated. NPSHa = NPSHr - 1 Cold Leg Velocity Head + Psat.
6. Plot the final NPSH requirement as the pressure at the pressurizer tap (low range) and hot leg tap (wide range) vs RCS temperature.

The NPSH for this analysis will be based on a normal heatup with 2/0 and a 2/0 cooldown (or in a pump failure condition with 0/2). These pump combinations will provide less risk to cavitation than the 1 pump operation previously used at TMI-1. This is primarily due to the fact that the NPSHr in the 2/0 pump flow region (~108,000 gpm per pump) is likely based on test data and not extrapolated, as is the case for 130000+ gpm of 1/0 pumps. Furthermore, the extrapolation to the 1 pump flows is on an exponential increase and a small change in the slope in this region results in a large change in required NPSH, thereby allowing for a much larger uncertainty. A small change in slope in the 2/0 NPSHr region results in a small change in the NPSHr.

Pump Flow rates

Reference 2 shows the required NPSH (in feet) of the Westinghouse pumps as a function of pump flow in gpm. The first step is to determine the pump flow rates of the various pump combination at different temperatures. This is achieved by using the FSPLIT RCS model developed in Reference 5.

Sample calculation

Analyze a 1/0 pump combination at 540F coolant temperature with FSPLIT to determine the pump flow rate. Per Table 1 in Reference 5 it is ~135000 gpm. From Figure 6 in Reference 2, this is an NPSHr of approximately 775 ft.

At 540F, (46.8 lb/ft^3), this is 775 x 46.8/144 = 251.9 psi NPSH required.
NPSHr = Ppump –Pvapor press +Pvelocity head
At 540F, Pvapor press= 962.8 psia
At 135330 gpm (results from FSPLIT) in a 28" ID pipe (4.276 ft^2 area, Reference 3), the velocity is 135330/7.48/60/4.276= 70.5 ft/sec. This results in a VH of 70.5^2 x 46.8/64.3/144 = 25.1 psi

Therefore, the pump suction pressure P(pump suction) must be at least 251.9+962.8-25.1=1189.6 psia. This pressure is then corrected to the pressure tap location. These calculations for all pump combinations and temperatures are shown in Table 6 below.

The FSPLIT analysis for this case (1/0 @ 540F) predicted a 1221 psia at the hot leg tap and 1232.9 psia at the surge line. The pressurizer tap indication is corrected to the 348 ft 10.75 inch location in the pressurizer (Reference 2). Since the required pressurizer level is 100 inches, it is ~3 feet above the surge line at 323 ft (see Reference 2). The tap indicates pressure at the 348 ft 10.75 inch elevation or 25 ft 10.75 inches above the water level in the pressurizer. The pressurizer water space will be effectively saturated liquid and the steam space saturated steam. The correction factor from the surge line to the tap will be -3 ft x Rho(liquid)/144 – 25.9 ft x Rho (steam)/144. This results in 1231.6 psia or 1216.9 psig at the pressurizer tap [1232.9-(3 x 44.6/144) - (25.9 x 2.15 / 144) = 1232.9-1.3=1231.6]

Other NPSHr for different flow rates are tabulated below.

2/2 pumps	~95000 gpm	220 ft
2/1 pumps(1 pmp loop)	~128000 gpm	660 ft
2/0 pumps	~105000 gpm	300 ft
1/0	~135000 gpm	775 ft

The 0/1 pump combination analysis used the same case except the pressure tap and pressurizer were assumed on the B loop and FSPLIT nodes 11 and 13 were used in place of nodes 10 and 12. All these analyses used the 20% tube plugging model to minimize the pressure at the pump suction. When all the fuel assemblies in TMI-1 use the fine mesh debris grids, the pressure drop in the core will increase, decreasing RCS flow rate slightly. This decrease will in turn decrease the pressure drop between the pressure taps and the pump suction. These two effects will decrease the required NPSH and decrease the DP to tap correction, making the present prediction conservative.

This same calculation was performed for 2/0 pump combination (and consequently 0/2), 2/1 (and 1/2) and 2/2 pump combinations for a range of temperatures from 100F to 540F (the RCS cannot be at a constant temperature above 540F). These results are shown in Table 6 below. One calculation was performed for the 1/1 pump combination to show that the required NPSH is only marginally less than 1/0 and therefore, 1/0 was used for both 1/0 and 1/1 pump combinations (see Reference 5). Table 6 also shows a 200F case using the tube plugging results from Cycle 10 at TMI-1 (shown as a best estimate case). This case was analyzed to show the higher flow (and consequently higher pressure drops in the core and hot legs). These cases were used to modify the calculated pressure changes for the DHRS and NDT limits since lower tube plugging is conservative for these cases.

The NPSHr was plotted on Figures 12 and 13. Figure 12 shows the results of all the pump combinations for the low range indication (with 4 psi pressure and 7F temperature uncertainties) and Figure 13 shows it for the hot leg pressure tap (wide range) with 25 psi and 12F uncertainty. The only case from Figure 13 used in the procedures is the 0/1 pump combination (extrapolated to 570F).

Finally, since the percent degraded head associated with the Westinghouse NPSHr curves is not known (it is assumed to be 1%) it is recommended that operation near the NPSH curve be minimized. When practical, maintain at least 50 psi margin to the NPSH limit (i.e., do not operate for long periods of time right on or very close to the NPSH limit).

Table 6 NPSH Calculations

Velocity Head			Press	Sat	Water	Press @	Press		Press	Flow	Water	Flow	NPSHr	FSPLIT		Saturation
C leg	H leg a	H leg b	Pzr Tap	Density	Temp	Surge Line	Ptap-Ppzr	HL tap	Pmp suct	Pump	Density	Pump	ft to psia	NPSHa	NPSHa	Pressure
psi	psi	psi	psia	lb/cuft	Deg F	psia	psid	Psia	Psia	lb/sec	lb/cuft	gpm	psia	psia	ft	psia
Pump Combination 1/0 Reqd NPSH= 775 ft																
32.6	6.7	0.06	362.0	52.3	100	363.1	-14.6	347.4	302.2	18525	62.06	133976	334.0	333.9	774.8	0.95
31.8	6.7	0.06	358.9	52.3	200	360	-14.1	344.8	302.6	18059	60.14	134776	323.7	323.1	773.6	11.53
31.4	6.6	0.06	368.8	52.3	250	369.9	-13.8	355.0	314.2	17695	58.84	134977	316.7	315.8	772.9	29.82
30.7	6.5	0.06	397.6	51.9	300	398.7	-13.4	384.2	344.8	17262	57.33	135142	308.5	308.5	774.9	67.01
28.8	6.2	0.06	557.3	50.1	400	558.4	-12.5	544.8	508.3	16195	53.70	135359	289.0	289.9	777.4	247.26
26.3	5.6	0.05	961.9	46.7	500	963.1	-11.3	950.6	917.5	14799	49.00	135556	263.7	262.9	772.6	680.86
25.1	5.4	0.05	1231.6	44.6	540	1232.9	-10.6	1221.0	1189.6	14111	46.80	135330	251.9	251.8	774.8	962.79
32.6	7.1	0.07	359.0	52.3	200BE	360.1	-14.2	344.8	306.4	18235	60.14	136089	323.7	327.5	784.2	11.53
Pump Combination 0/1 Reqd NPSH=775 ft																
32.6	0.06	6.7	372.9	52.3	100	374	-12.9	360.0	302.2	18525	62.06	133976	334.0	333.8	774.5	0.95
31.9	0.06	6.7	370.0	52.3	200	371.1	-12.7	357.3	302.6	18059	60.14	134776	323.7	323.1	773.6	11.53
31.4	0.06	6.6	379.8	52.3	250	380.9	-12.5	367.3	314.2	17695	58.84	134977	316.7	315.8	772.9	29.82
30.7	0.06	6.5	408.4	51.9	300	409.5	-12.1	396.3	344.8	17262	57.33	135142	308.5	308.5	774.9	67.01
28.8	0.06	6.2	567.5	50.1	400	568.6	-11.3	556.2	508.3	16195	53.70	135359	289.0	289.9	777.4	247.26
26.3	0.05	5.6	971.2	46.7	500	972.4	-10.1	961.1	917.5	14799	49.00	135556	263.7	262.9	772.6	680.86
25.1	0.05	5.4	1240.5	44.6	540	1241.8	-9.5	1231.0	1189.5	14111	46.80	135330	251.9	251.8	774.8	962.79
32.6	0.07	7.1	370.8	52.3	200BE	371.9	-12.7	358.1	306.4	18235	60.14	136089	323.7	327.5	784.2	11.53
Pump Combination 1/2 Reqd NPSH= 660 ft																
29.4	4.9	25.2	306.1	52.9	100	307.2	-14.3	291.8	255.7	17569	62.03	127124	284.3	283.9	659.1	0.95
28.7	4.9	25.0	306.4	52.9	200	307.5	-13.7	292.7	258.9	17110	60.14	127693	275.6	276.1	661.1	11.53
28.2	4.8	24.9	316.0	52.9	250	317.1	-13.4	302.6	270	16761	58.84	127852	269.7	268.8	657.8	29.82
27.5	4.7	24.5	347.4	52.5	300	348.5	-13.1	334.3	302.8	16347	57.33	127979	262.8	263.3	661.4	67.01
25.8	4.4	23.1	509.4	50.6	400	510.6	-12.0	497.4	468.2	15330	53.70	128130	246.1	246.7	661.5	247.26
23.6	4.1	21.2	918.7	47	500	919.9	-10.9	907.8	881.3	14003	49.00	128265	224.6	224.1	658.6	680.86
22.5	3.9	20.2	1190.2	44.9	540	1191.5	-10.2	1180.0	1154.8	13348	46.80	128012	214.5	214.5	660.0	962.79
29.2	5.1	26.7	308.5	53.5	200BE	309.6	-13.8	294.7	265	17237	60.14	128641	275.6	282.6	676.7	11.53
Pump Combination 2/1 Reqd NPSH= 660 ft																
29.4	25.0	4.9	276.0	53.5	100	277.1	-18.2	257.8	255.7	17569	62.03	127124	284.3	283.9	659.1	0.95
28.7	25.0	4.9	275.9	53.5	200	277	-17.7	258.2	258.9	17110	60.14	127693	275.6	276.1	661.1	11.53
28.2	24.9	4.8	285.8	53.5	250	286.9	-17.4	268.4	270	16761	58.84	127852	269.7	268.8	657.8	29.82
27.5	24.5	4.7	317.6	52.9	300	318.7	-16.8	300.8	302.8	16347	57.33	127979	262.8	263.3	661.4	67.01

25.8	23.1	4.4	481.2	50.6	400	482.4	-15.6	465.6	468.2	15330	53.70	128130	246.1	246.7	661.5	247.26
23.6	21.2	4.1	892.8	47	500	894	-14.2	878.6	881.3	14003	49.00	128265	224.6	224.1	658.6	680.86
22.5	20.2	3.9	1165.6	44.9	540	1166.9	-13.4	1152.2	1154.8	13348	46.80	128012	214.5	214.5	660.0	962.79
29.2	26.7	5.1	275.9	53.5	200BE	277	-18.0	257.9	265	17237	60.14	128641	275.6	282.6	676.7	11.53
Pump Combination 0/2 Reqd NPSH= 300 ft																
18.9	0.5	27.5	240.2	54.8	100	241.3	-13.7	226.5	111.4	14041	62.03	101596	129.2	128.6	298.5	0.95
18.9	0.5	27.7	239.7	54.8	200	240.8	-12.6	227.1	118.4	13883	60.14	103610	125.3	125.8	301.2	11.53
18.7	0.5	27.4	251.1	54.5	250	252.2	-12.4	238.7	133.4	13654	58.84	104152	122.6	122.2	299.1	29.82
18.4	0.5	26.9	282.0	54.1	300	283.2	-11.9	270.1	168.1	13353	57.33	104539	119.4	119.4	299.9	67.01
17.3	0.5	25.4	448.2	51.8	400	449.4	-11.1	437.1	342.4	12563	53.70	105003	111.9	112.4	301.4	247.26
15.9	0.4	23.3	863.1	47.8	500	864.4	-9.9	853.2	767.1	11492	49.00	105264	102.1	102.2	300.3	680.86
15.2	0.4	22.2	1136.1	45.7	540	1137.4	-8.4	1127.7	1044.8	10957	46.80	105082	97.5	97.2	299.1	962.79
	0.5	29.4	242.8	54.8	200BE	243.9	-12.6	230.2	128.1	14317	60.14	106849	125.3	136	325.6	11.53
Pump Combination 2/0 Reqd NPSH= 300 ft																
18.9	27.5	0.5	193.0	54.8	100	194.1	-18.0	175.0	111.4	14041	62.03	101596	129.2	128.6	298.5	0.95
18.9	27.7	0.5	192.9	54.8	200	194	-18.2	174.7	118.4	13883	60.14	103610	125.3	125.8	301.2	11.53
18.7	27.4	0.5	204.7	54.5	250	205.8	-17.7	187.0	133.4	13654	58.84	104152	122.6	122.2	299.1	29.82
18.4	26.9	0.5	236.5	54.1	300	237.7	-17.2	219.3	168.1	13353	57.33	104539	119.4	119.4	299.9	67.01
17.3	25.4	0.5	405.2	51.8	400	406.4	-16.1	389.1	342.4	12563	53.70	105003	111.9	112.4	301.4	247.26
15.9	23.3	0.4	823.7	47.8	500	825	-14.5	809.2	767.1	11492	49.00	105264	102.1	102.2	300.3	680.86
15.2	22.2	0.4	1098.5	45.7	540	1099.8	-13.7	1084.8	1044.8	10957	46.80	105082	97.5	97.2	299.1	962.79
20.1	29.4	0.5	192.9	54.8	200BE	194	-18.6	174.3	128.1	14317	60.14	106849	125.3	136	325.6	11.53
Pump Combination 2/2 Reqd NPSH= 220 ft																
14.9	21.8	21.8	195.5	54.1	300	196.7	-16.2	179.3	139.8	12026	57.33	94150	87.6	87.6	220.0	67.01
14.1	20.6	20.6	366.2	51.8	400	367.4	-15.1	351.1	315	11325	53.70	94655	82.0	81.9	219.6	247.26
12.9	18.9	18.9	788.7	47.8	500	790	-13.6	775.1	742.7	10359	49.00	94886	74.9	74.8	219.8	680.86
12.3	18.0	18.0	1064.6	45.7	540	1065.9	-13.0	1051.6	1020.9	9874	46.80	94695	71.5	70.4	216.6	962.79
15.7	22.91	23.45	195.8	54.1	300BE	197	-16.4	179.4	147.9	12332	57.33	96546	87.6	96.6	242.6	67.01

DHRS Pressures

Per Reference 2, the maximum allowable pressure in the DHRS is at the decay heat pump discharge and is a function of temperature. The limiting values for this analysis are 505 psig up to 250F, sloping to 470 psig at 300F. Since the limit did not exceed 300F on the previous PT curves, it will stop at 300F in the revised PT limits. Per Reference 2, the total head added by the pump is 350 ft at 3000 gpm and the elevation change from the DH pump to the center of the hot leg (at the drop line) is 50 ft. The unrecoverable losses in the DHRS line is based on TMI calculation C-1101-220-5360-030 Rev 0 Sheet No 11 (Reference 11). At 3000 gpm it is;

$$\text{Loss(psid)} = (32.79 \times (3000/6000)^2 + 8.35 \times (3000/3000)^2) \times \text{Rho}/144 =$$
$$(32.79 \times (.5)^2 + 8.35) \times \text{Rho}/144 = 0.1149 \times \text{Rho}$$

Finally, the pressure in the hot leg (at the drop line) is corrected to the low range pressurizer pressure tap. The hot leg pressure is based on the DP calculated in FSPLIT between node 9 (the drop line) and node 12 (the tap). The pressurizer pressure will be calculated as discussed above for NPSH. Also, the DHRS limit will always use the low range pressure transmitter since the DHRS system will always be operated below 450 psig.

A sample calculation for 2/0 operation at 200F follows.
Plimit = 505 psig. The pump head at 3000 gpm is 350 ft x 60.14 lb/ft^3 /144 = 146.17 psid making the allowable pump suction pressure 358.83 psig. The unrecoverable loss in the line to the RCS hot leg is 6.9 psid solving the above equation. The elevation change is 50 ft or 20.9 psid, resulting in a pressure of 358.83 –20.88 +6.91 = 344.86 at the hot leg center line above the decay heat drop line. The velocity head of 3000 gpm in the 12 inch pipe (using 0.6013 ft^2) is [(3000 gal/min)/(7.4805 gal/ft^3) / .6013 ft^2 /60 sec/min]^2 x 60.14lb/ft^3 / 64.4 ft/sec^2 / 144 in^2 / ft^2 = 0.8 psi. This will be neglected. The correction to the pressurizer tap is –53.4 psid (see Table 1 of FSPLIT results above) resulting in 291.46 psig. Table 7 below shows this calculation for each point analyzed. Note from Table 1 that the DP correction from the drop line to the pressurizer is 47.6 psid for the 20% tube plugging (conservative for NPSH), and 3 psi higher when cycle 10 plugging results are used (less than 20% plugged). The cycle 10 FSPLIT benchmark case from Reference 5 was used to increase the DPs from Table 1 for the DHRS limits. The final modification to the calculated pressures will be the 4 psi uncertainty or 287.46 psig. When the RCS temperature is 200F+7F uncertainty equal 207F and the limiting pressurizer tap indication for DHRS operation is 290.7 psig. At this condition (2/0 pump operation), the DHRS pump discharge pressure will be 505 psig.

The DHRS limit is a non-symmetric limit during cooldowns. 2/0 and 0/2 will have different limits since the drop line is only on the B loop. During a 2/0 cooldown, the pressure at the drop line is ~50 psi higher than the pressure at the surge line, however, during 0/2 pump operation the drop line pressure is ~45 psi lower than the surge line due to the difference in flow directions. Therefore, the allowable indicated pressure at the pressurizer tap will be ~95 to 100 psi higher in 0/2 than 2/0. Since 0/2 is only permitted during cooldowns, a separate 0/2 limit will be provided for these conditions. DHRS limits for the different pump combinations are shown on Figure 14.

Table 7 DHRS Limits for TMI-1						
1/0 Pump Combination						
Temperature (F)		DHRS Plimit	DHRS Pump	Photleg	Low range	Ppzr
Nominal	With Uncrt	psig	suction	Dropline	Press psig	-uncrt(4 psi)
100	107	505	354.2	339.7	324.0	320.0
200	207	505	358.8	344.9	329.6	325.6
250	257	505	362.0	348.3	332.9	328.9
300	307	470	330.7	317.3	302.1	298.1
200BE		505	358.8	344.9	329.6	322.1
0/1 Pump Combination						
100	107	505	354.2	339.7	346.8	342.8
200	207	505	358.8	344.9	352.2	348.2
250	257	505	362.0	348.3	355.6	351.6
300	307	470	330.7	317.3	324.4	320.4
200BE		505	358.8	344.9	352.9	345.4
1/2 Pump Combination						
100	107	505	354.2	339.8	365.1	361.1
200	207	505	358.8	344.9	370.6	366.6
250	257	505	362.0	348.3	373.9	369.9
300	307	470	330.7	317.3	342.5	338.5
200BE		505	358.8	344.9	372.7	368.7
2/1 Pump Combination						
100	107	505	354.2	339.8	303.8	299.8
200	207	505	358.8	344.9	308.5	304.5
250	257	505	362.0	348.3	312.3	308.3
300	307	470	330.7	317.3	281.8	277.8
200BE		505	358.8	344.9	308.5	304.5
0/2 Pump Combination						
100	107	505	354.2	339.8	381.5	377.5
200	207	505	358.8	344.9	386.8	382.8
250	257	505	362.0	348.3	389.9	385.9
300	307	470	330.7	317.3	358.1	354.1
200BE		505	358.8	344.9	389.8	385.8
2/0 Pump Combination						
100	107	505	354.2	339.8	286.0	282.0
200	207	505	358.8	344.9	291.4	287.4
250	257	505	362.0	348.3	295.4	291.4
300	307	470	330.7	317.3	265.4	261.4
200BE		505	358.8	344.9	291.4	283.9
0/0 Pump Combination						
100	107	505	354.2	339.8	336.2	332.2
These limits assume 3000 gpm flow rate. A lower flow rate will require a lower limit.						

SUBCOOLING LIMIT

Subcooling margin can be lost if the pressure is too low or temperature too high. Per Reference 2, this limit is based on different temperature indicators than the other limits on the PT curve. Therefore, per Reference 2, the 25F required subcooling margin is essentially the uncertainty and this limit will not utilize any instrument uncertainties. Per conversations with TMI personnel, this curve is essentially used as a general guide-line and other instrumentation on the control panel is

actually used for subcooling margin. This curve is shown on Figures 1 through 6. It is a plot of Psat Vs. Tsat-25F.

Temperature (F)	Psat of Temp +25F (psig)	Temperature (F)	Psat of Temp +25F (psig)
75	-13.8	350	169.6
100	-12.8	375	232.6
125	-11.0	400	311.2
150	-8.0	425	407.8
175	-3.2	450	525.2
200	4.2	475	666.2
225	15.1	500	833.6
250	30.7	525	1030.7
275	52.3	550	1261.1
300	81.5	575	1528.5
325	119.9	600	1837.6

SURGE LINE

The surge line limit will be identical to the other B&W plants (per Reference 2) and shown in Reference 12. No instrument uncertainty is needed for the surge line PT as discussed in Reference 2. Also, there is no explicit location correction for the surge line limit when monitoring the "A" loop pressure since the pressure indication is effectively at (or very near) the surge line. Note that per Reference 12, the development surge line limit assumed some violations of this limit would occur during the lifetime of the plant. This curve is shown on Figures 1 through 6. The values used to generate this curve are listed below. Note that the Reference 4 (original TMI-1 curve) surge line curve is slightly different than that suggested by Reference 12.

Temperature (F)	Wide and Low Range Pressure (psig)
66	150
100	232
129	325
179	325
200	413
225	526
275	526
310	728
346.5	1000
389	1400
422	1800
450.5	2200
450.5	2500

SEAL STAGING

Reference 16 calculated the RCS low range pressures that would provide ~200 psid across the No. 1 seal. (The minimum required pressure across the seal is 200 psid per the Westinghouse pump

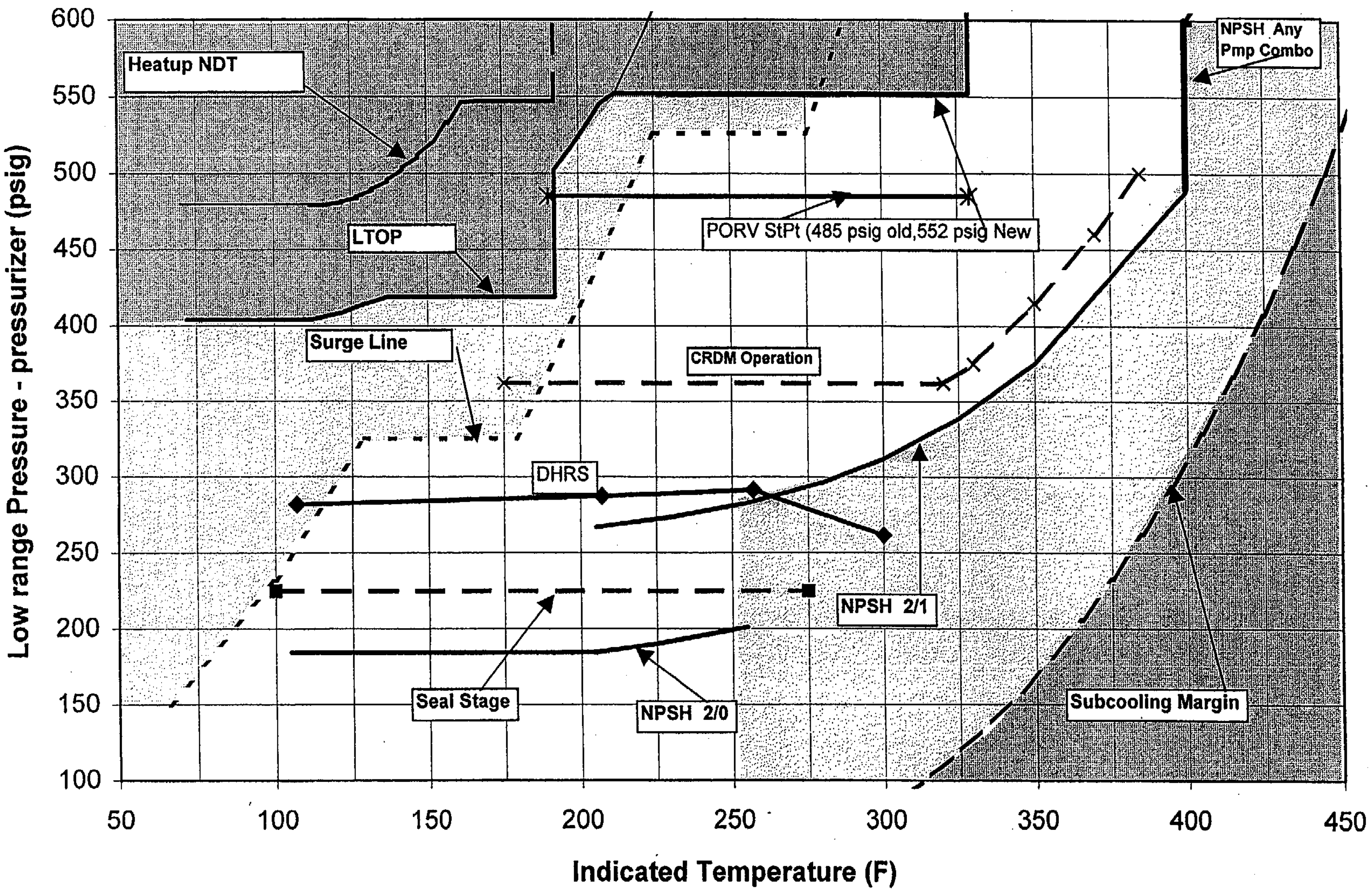
manual). The key element of this prediction is that the seal DP actually increases when the RC pumps start and if 200 psid exists prior to pump start, the DP will exceed 200 psid after pump start. The limit was determined to be 225 psig (low range) when the seal return pressure is established at 40 or greater psig (MU-39-PI1) just prior to pump start. This curve is shown on Figures 1,3 ,5, and 6.

ROD DROP LIMITS

References 4 and 13 show the dissolved gas limit (for rod trip associated problems). Per Reference 2, this limit will be applied exactly as it has in the past. Note that CRDM venting procedures could possibly preclude the need for this limit. Since the CRDM pressure is the same during 2/0 or 0/2 pump operation, this limit will apply to both. This curve is shown on Figures 1, 3, and 5. The values for this limit were interpreted from Reference 4 and are listed below.

Temperature (F)	Low Range Prressure (psig)
175	362
320	362
330	375
350	415
370	460
385	500

Figure 1
2/0, 2/1, Low Range Heatup For TMI-1 29 EFPY



Note For Figure 1
NDT and LTOP limits include 25 psi and 12F instrument uncertainty.

Surge line and Subcooling margin have no instrument uncertainty associated with them.

DHRS and NPSH have 4 psi and 7F uncertainty.

The seal staging limit assumes that MU-39-PI1 is 40 psig or less when the 1^{st} RC pump is started. For every psi it is greater than 40 psig, the seal staging pressure (225 psig) should be increased by a psi.

Operate with 2/0 pumps between 100F and 200F-250F. Do not start 3^{rd} pump until the DHRS has been secured. Do not start 1st pump until the RCS is ≥ 100F.

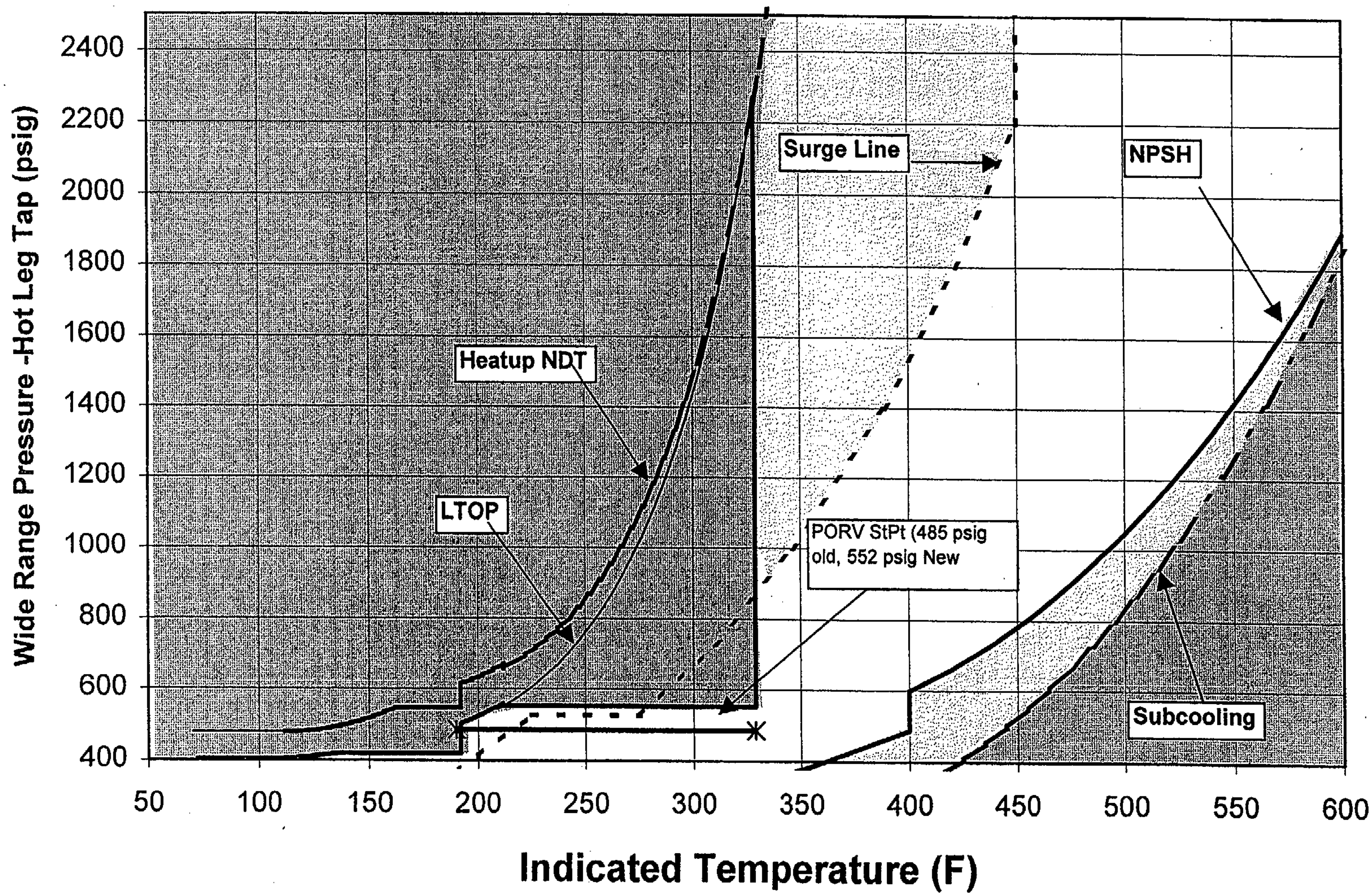
Start the 2^{nd} pump of 2/0 combination within 5 minutes of the 1^{st} pump to minimize cavitation of 1/0 pump operation at these low pressures.

The DHRS limit uses a flow rate of 3000 gpm. A lower flow rate will require a lower limit.

Assure that the automatic pressurizer level control is 100" (or less) and assure that this 100" limit is maintained until the RCS temperature is ≥ 329F.

This figure assumes a maximum HU ramp rate of 50F/hr or 15F step increases followed by an 18 minute soak.

Figure 2
2/0, 2/1, 2/2 Wide Range Heatup For TMI-1 29 EFPY



Note For Figure 2
NDT , LTOP, and NPSH limits include 25 psi and 12F instrument uncertainty.

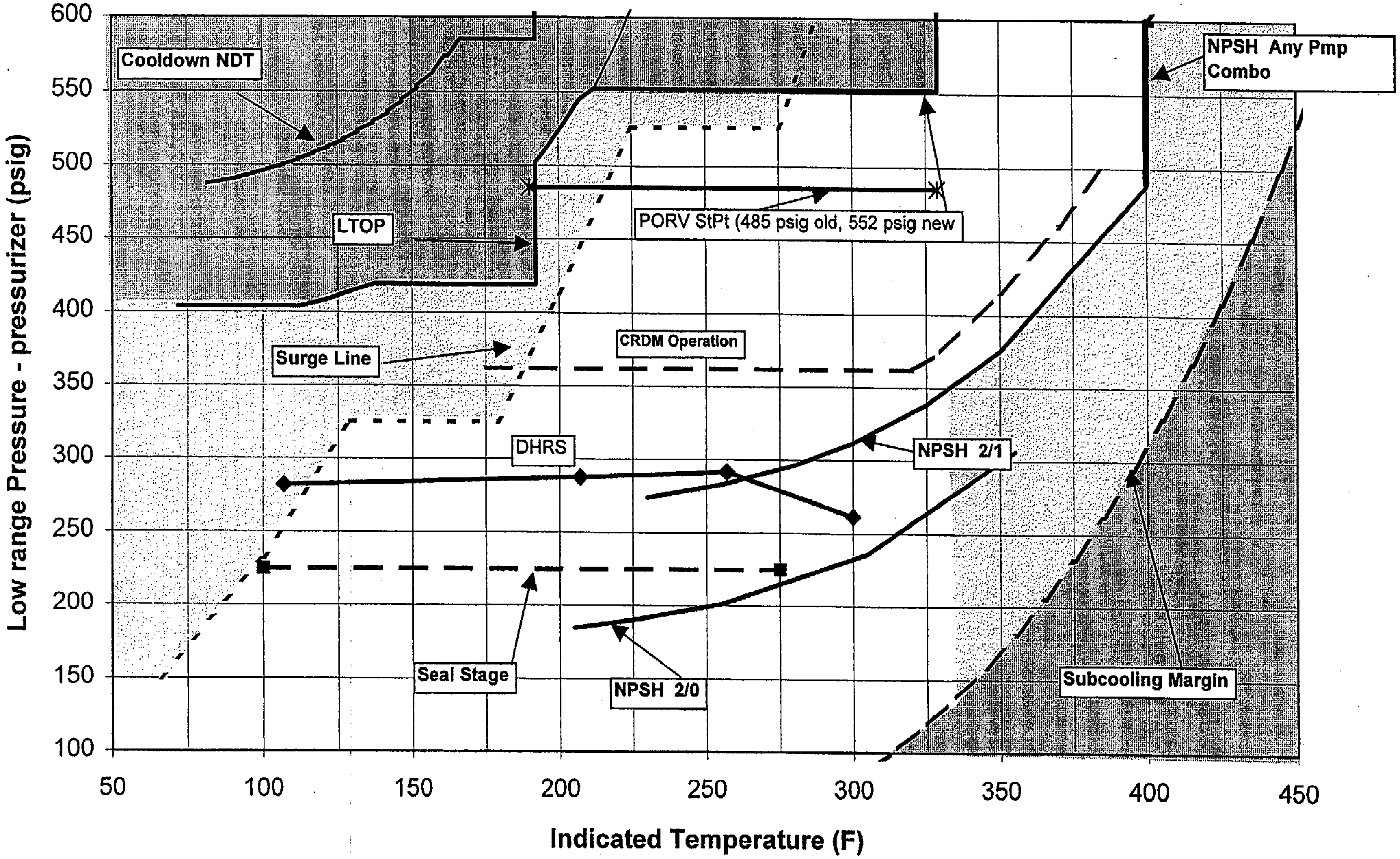
Surge line and Subcooling margin have no instrument uncertainty associated with them.

Any RC pump combination is permissible above 400F.

Assure that the automatic pressurizer level control is set to 100" (or less) and assure that this 100" limit is maintained till the RCS temperature is ≥ 329F.

This Figure assumes a maximum HU ramp rate of 50F/hr or 15F step increases followed by an 18 minute soak.

Figure 3
2/0, 2/1, Low Range Cooldown For TMI-1 29 EFPY



Note For Figure 3
NDT , LTOP, and NPSH limits include 25 psi and 12F instrument uncertainty.

Surge line and Subcooling margin have no instrument uncertainty associated with them.

Any RC pump combination is permissible above 400F.

DHRS and NPSH have 4 psi and 7F uncertainty.

Do not operate the DHRS until if more than 2 RC pumps are operating.

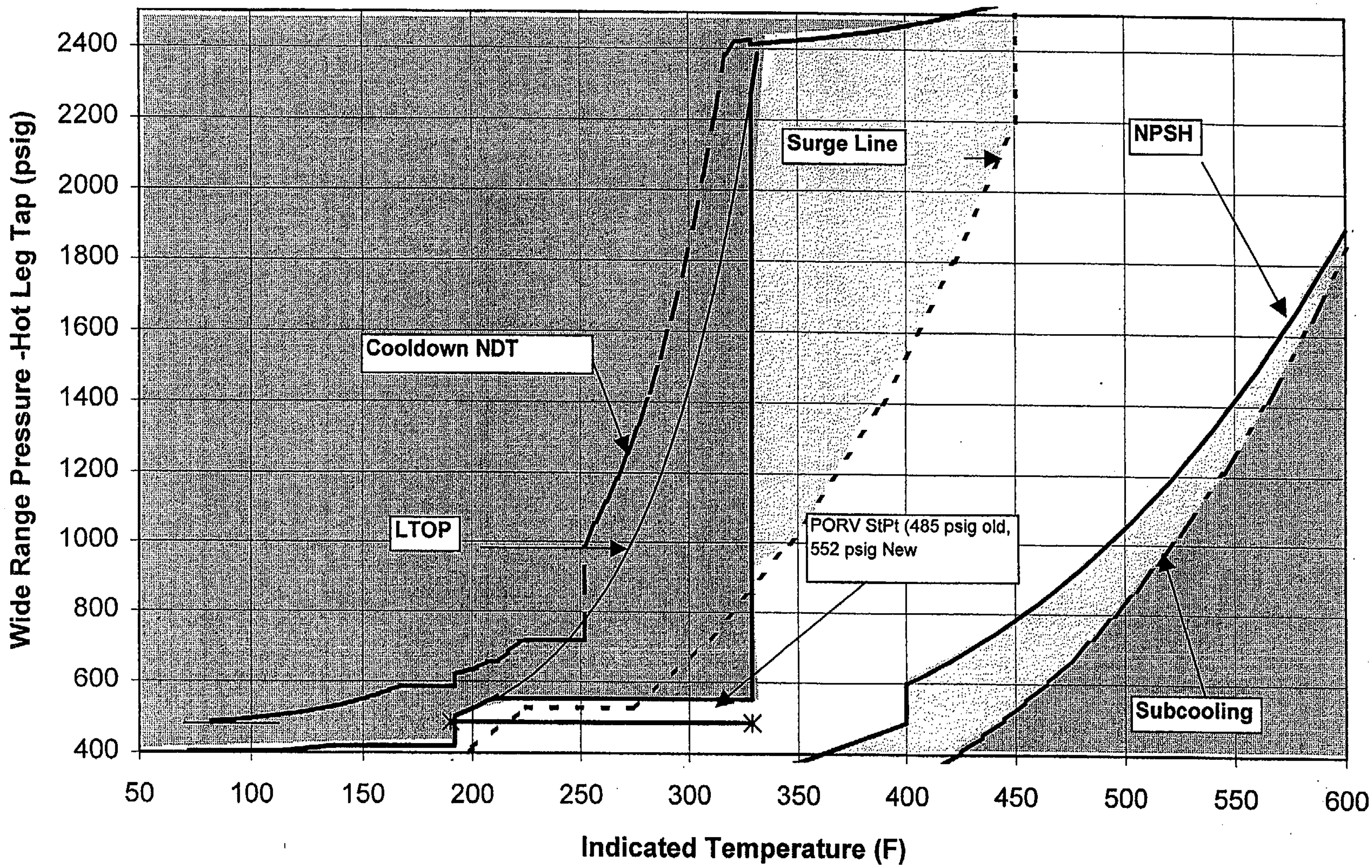
The DHRS limit uses a flow rate of 3000 gpm. A lower flow rate will require a lower limit.

Terminate the 2nd pump of the 2/0 combination immediately after stopping the 1st pump to minimize cavitation of 1/0 pump operation at these low pressures. All pumps must be secured when the RCS temperature is less than 200F.

Assure that the pressurizer automatic level control is 100" (or less) before cooling to 329F.

This Figure assumes a maximum CD ramp rate of 100F/hr above 255F and 30F/hr below 255F or step changes of 15F with a 9 minute soak above 255F and 15F step changes with a 30 minute soak below 255F.

Figure 4
Wide Range Cooldown For TMI-1 29 EFPY



Note For Figure 4
NDT , LTOP, and NPSH limits include 25 psi and 12F instrument uncertainty.

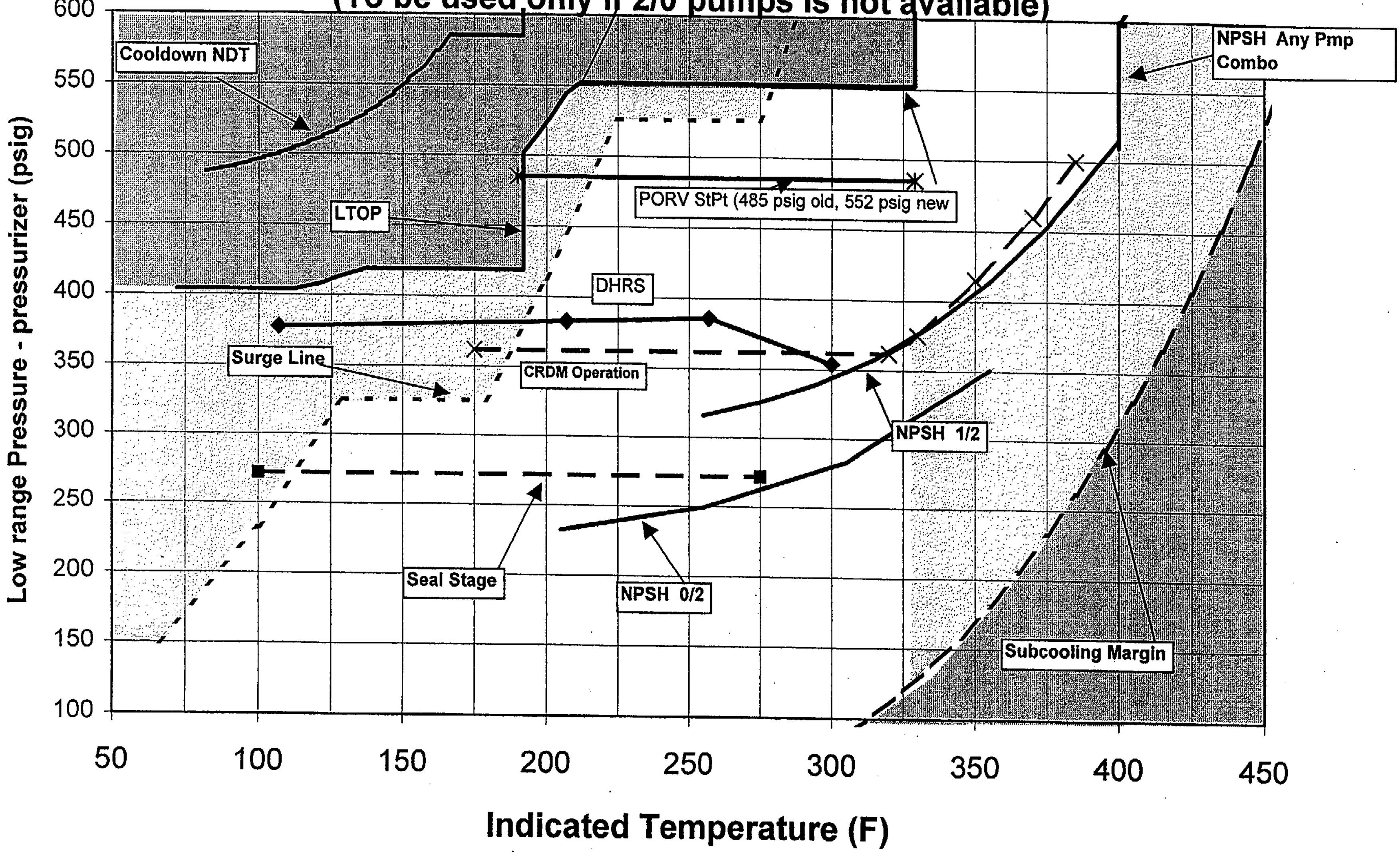
Surge line and Subcooling margin have no instrument uncertainty associated with them.

Any RC pump combination is permissible above 400F. Below 400F and 500 psig, see Figure 3 or 5.

Assure that the pressurizer automatic level control is 100" (or less) before cooling to 329F.

This Figure assumes a maximum CD ramp rate of 100F/hr above 255F and 30F/hr below 255F or step changes of 15F with a 9 minute soak above 255F and 15F step changes with a 30 minute soak below 255F.

Figure 5
1/2, 0/2, Low Range Cooldown For TMI-1 29 EFPY
(To be used only if 2/0 pumps is not available)



Note For Figure 5
NDT , LTOP, and NPSH limits include 25 psi and 12F instrument uncertainty.

Surge line and Subcooling margin have no instrument uncertainty associated with them.

Any RC pump combination is permissible above 400F.

DHRS and NPSH have 4 psi and 7F uncertainty.

Do not operate the DHRS until if more than 2 RC pumps are operating..

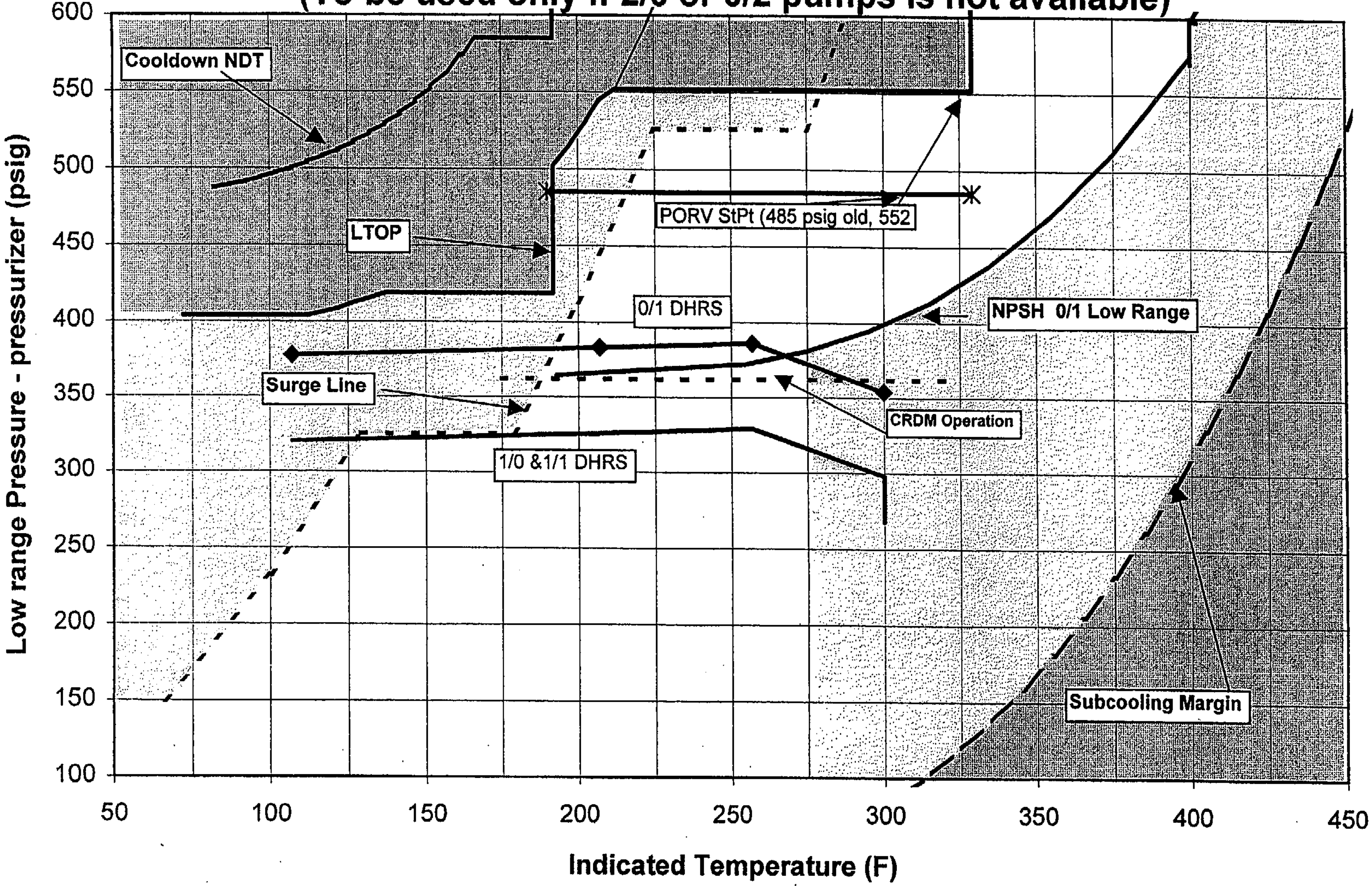
The DHRS limit uses a flow rate of 3000 gpm. A lower flow rate will require a lower limit.

Terminate the 2nd pump of the 0/2 combination immediately after stopping the 1st pump to minimize cavitation of 1/0 pump operation at these low pressures. All pumps must be secured when the RCS temperature is less than 200F.

Assure that the pressurizer automatic level control is 100" (or less) when the RCS temperature is < 329F.

This Figure assumes a maximum CD ramp rate of 100F/hr above 255F and 30F/hr below 255F or step changes of 15F with a 9 minute soak above 255F and 15F step changes with a 30 minute soak below 255F.

Figure 6
0/1, 1/0, 1/1 Low Range Cooldown For TMI-1 29 EFPY
(To be used only if 2/0 or 0/2 pumps is not available)



Note For Figure 6
NDT , LTOP, and NPSH limits include 25 psi and 12F instrument uncertainty.

Surge line and Subcooling margin have no instrument uncertainty associated with them.

DHRS and NPSH have 4 psi and 7F uncertainty.

Assure that the pressurizer automatic level control is 100" (or less) when the RCS temperature is < 329F.

If cooldown on this curve is necessary (assuming multiple RC pump failures) do the following:

- Maintain RCS pressure above 400 psig while performing the necessary preliminary steps to initiate DHRS.
- Depressurize below the allowable DHRS limit, initiate the DHRS and terminate RC pump(s) as quickly as possible (since the RC pump will experience cavitation during the depressurization).

This Figure assumes a maximum CD ramp rate of 100F/hr above 255F and 30F/hr below 255F or step changes of 15F with a 9 minute soak above 255F and 15F step changes with a 30 minute soak below 255F.

Figure 7
LTOP Pressure Limit

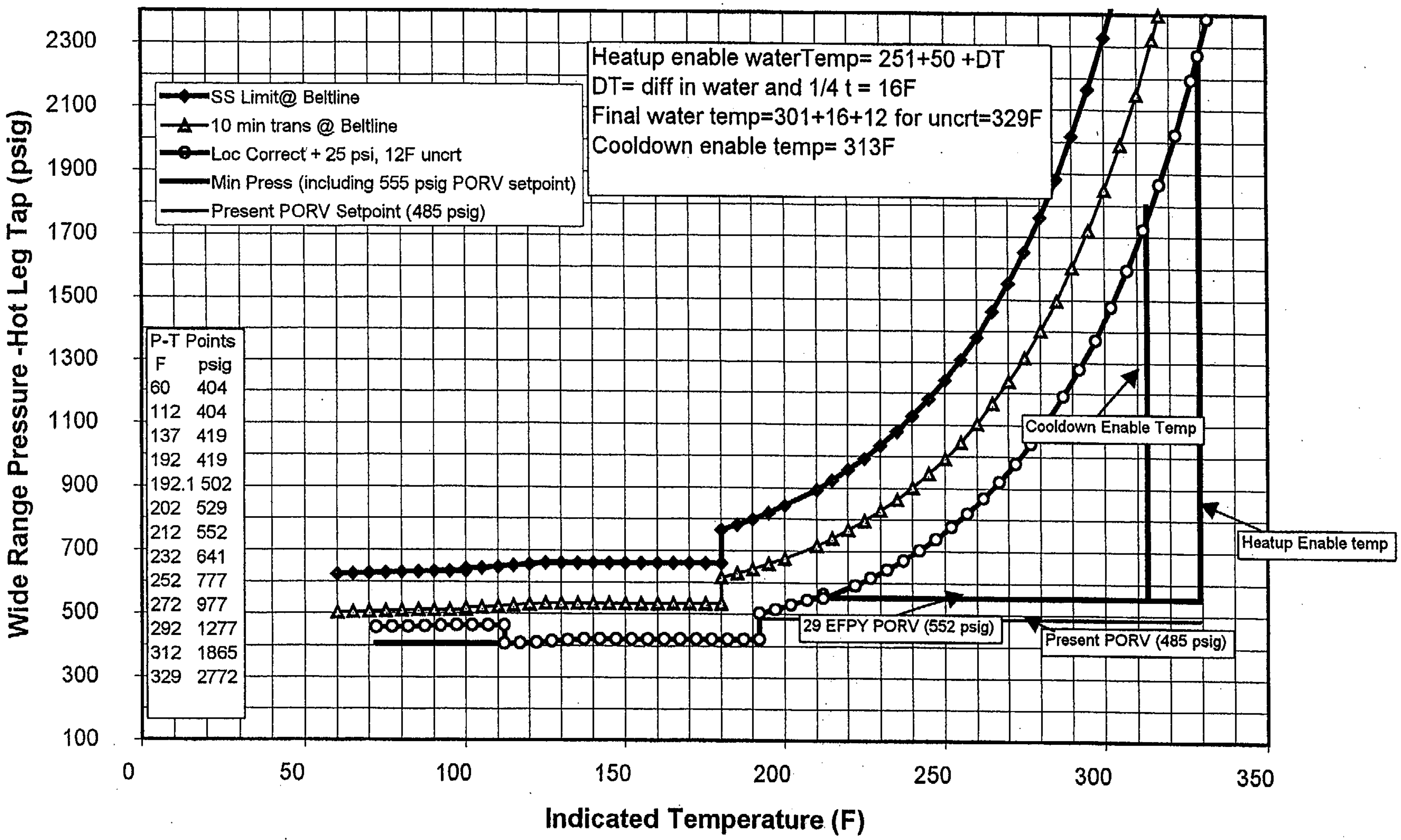


Figure 8
TMI-1 NDT Cooldown Limits

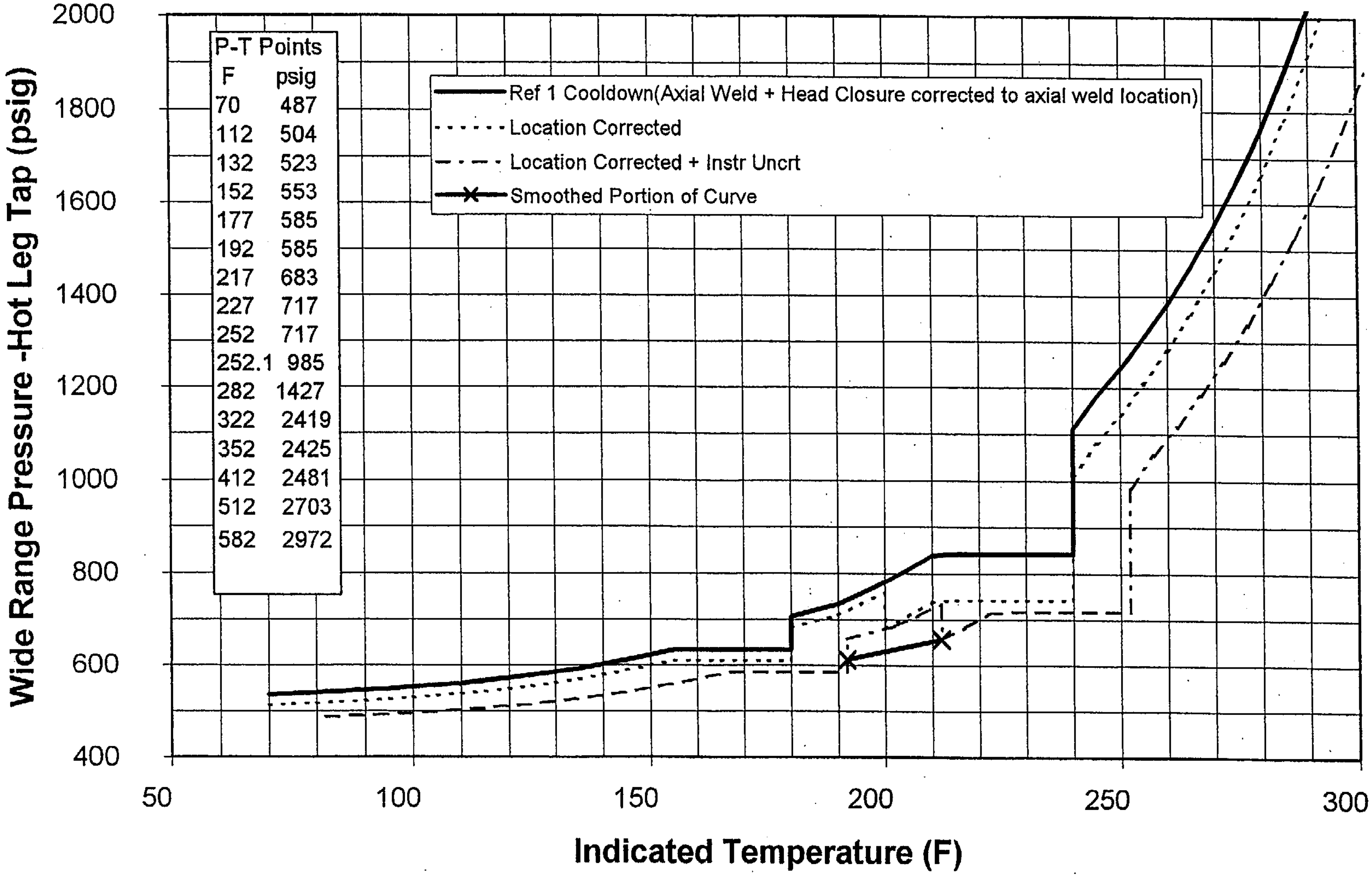


Figure 9
TMI-1 NDT Heatup Limits

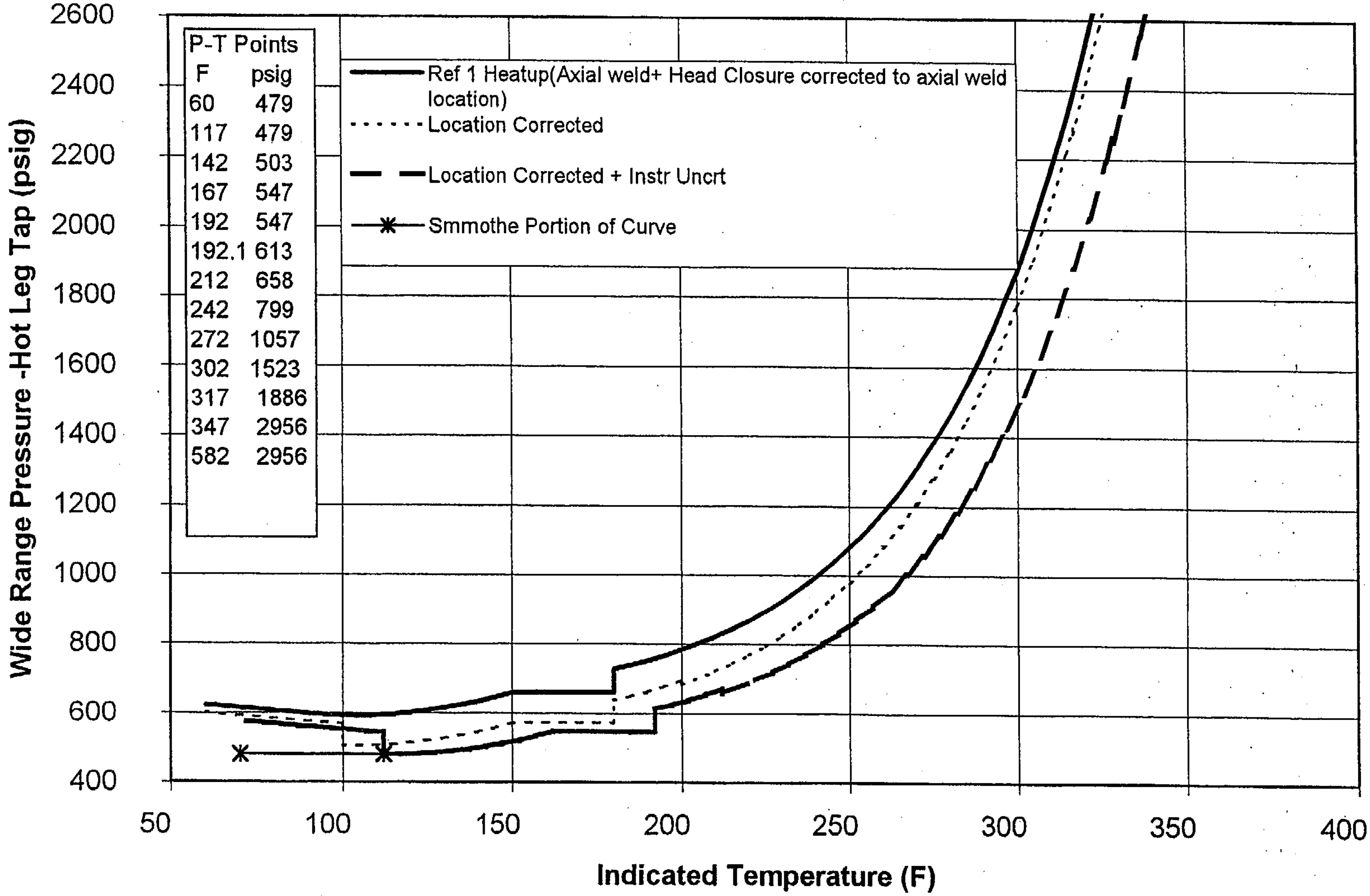


Figure 10
Combined HU, CD, and LTOP/PORV Limits

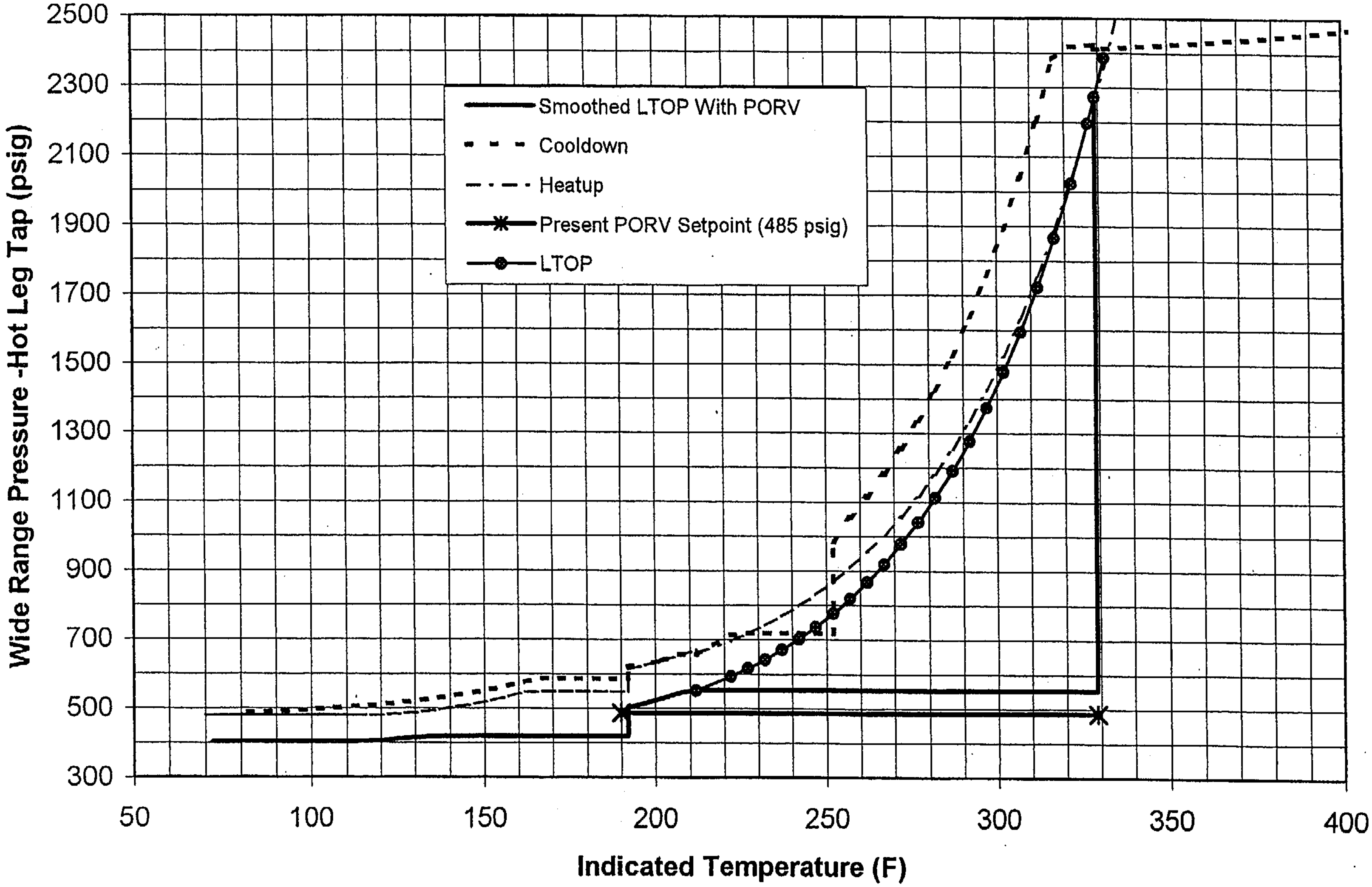


Figure 11
TMI-1 NDT ISLH Limits

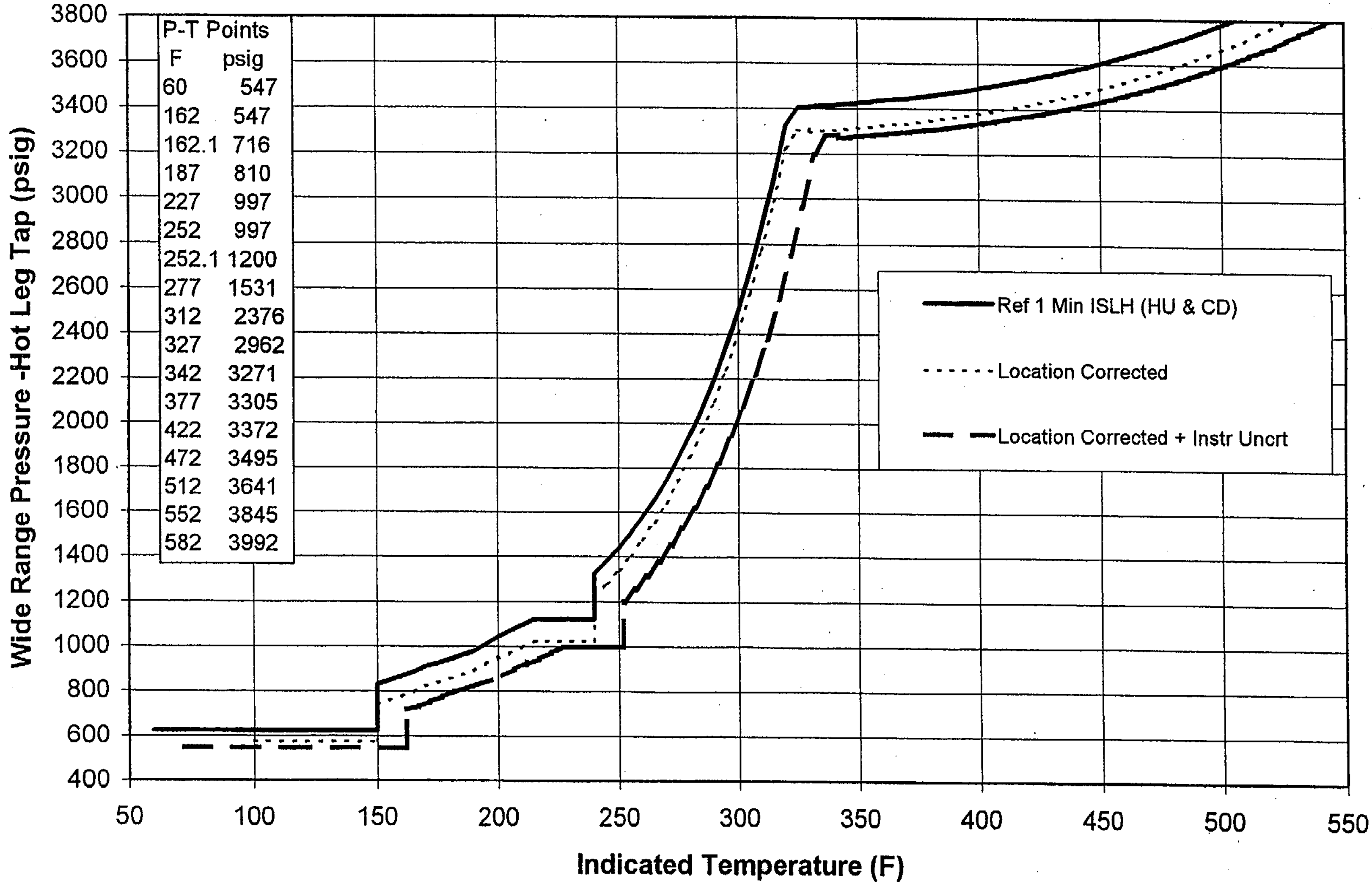


FIGURE 12
NPSH Limits (with Low range Instr Uncertainty) for all pump Combinations

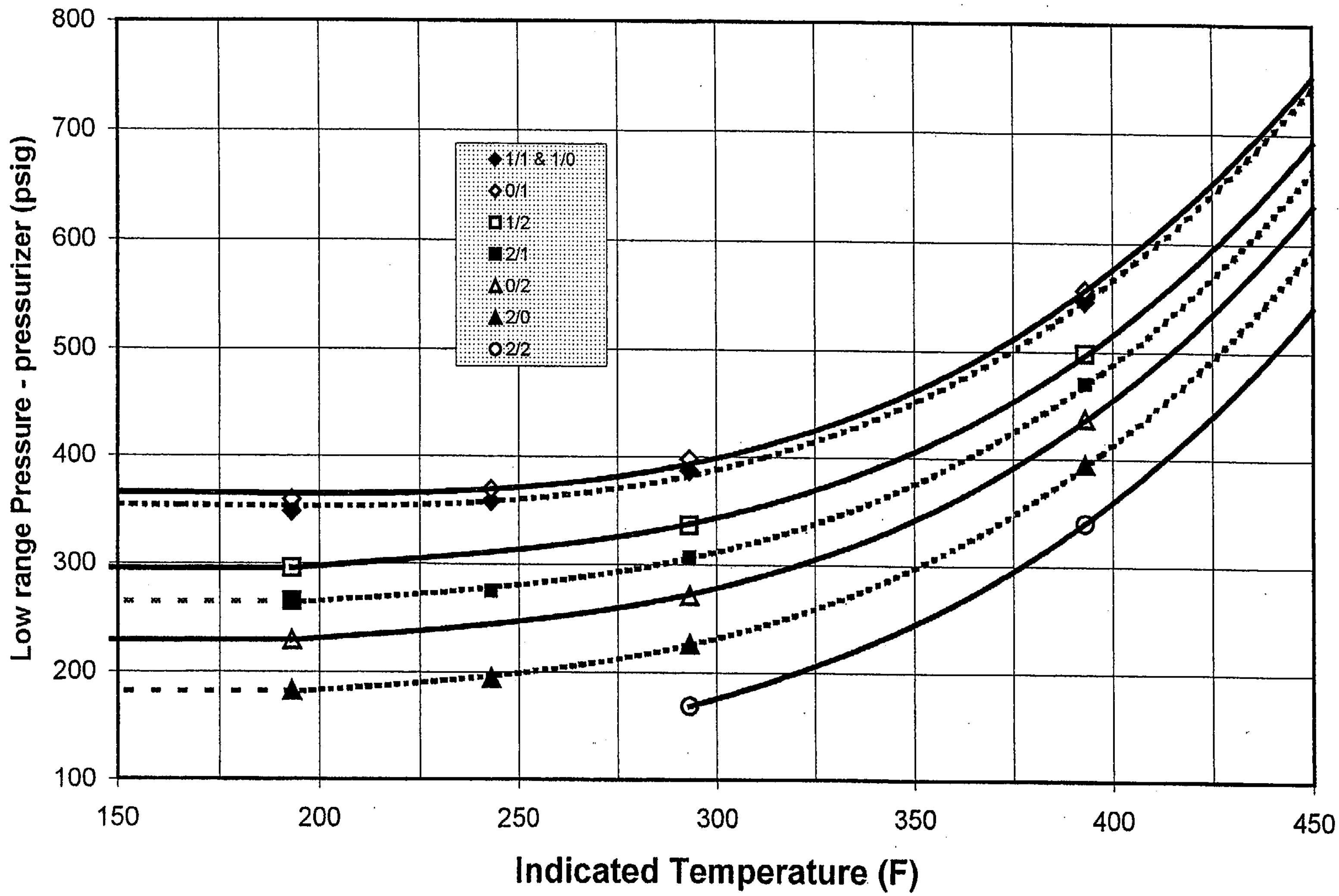


FIGURE 13
NPSH Limits (with Wide Range Instr Uncertainty) for all pump Combinations

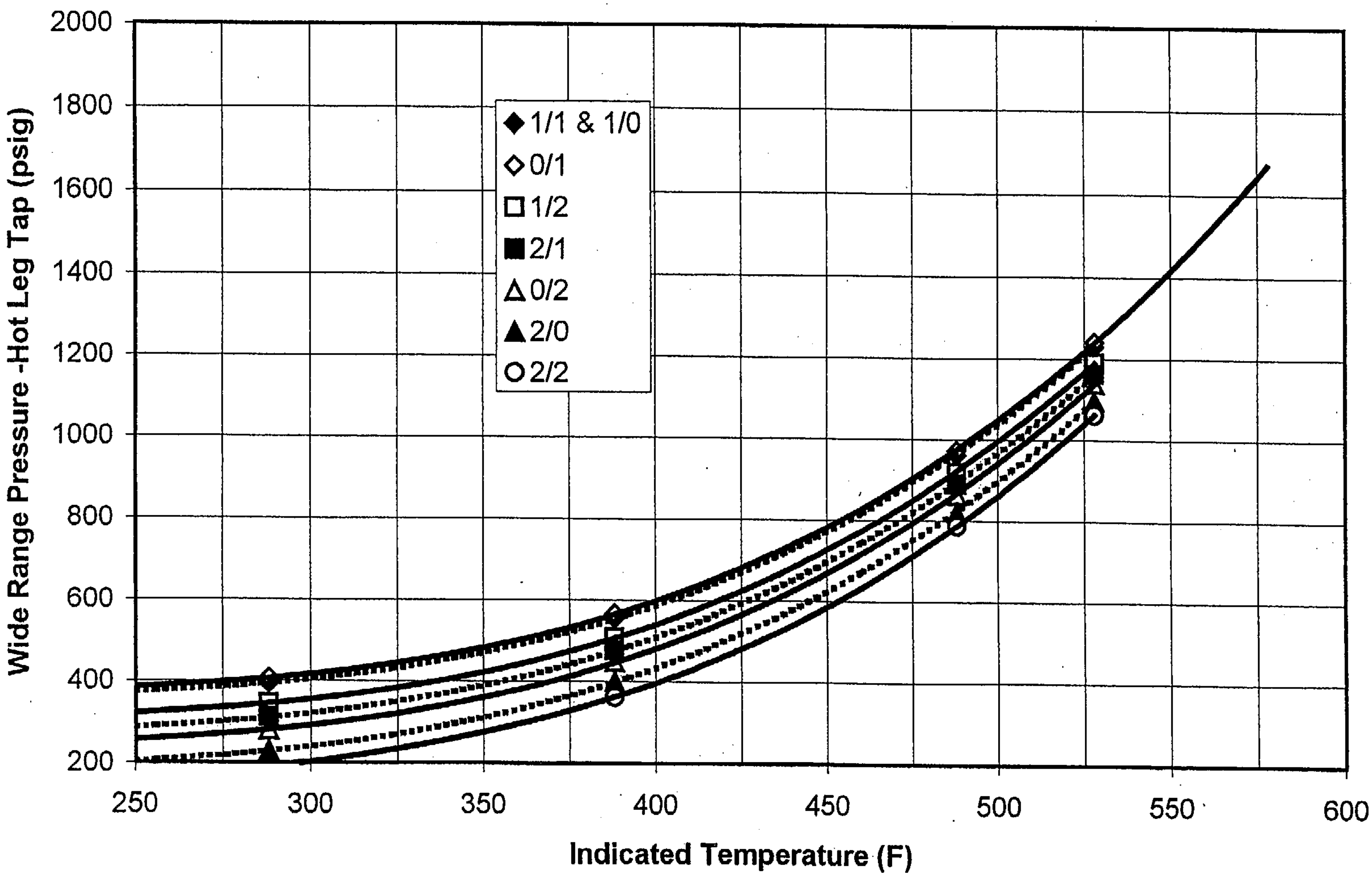
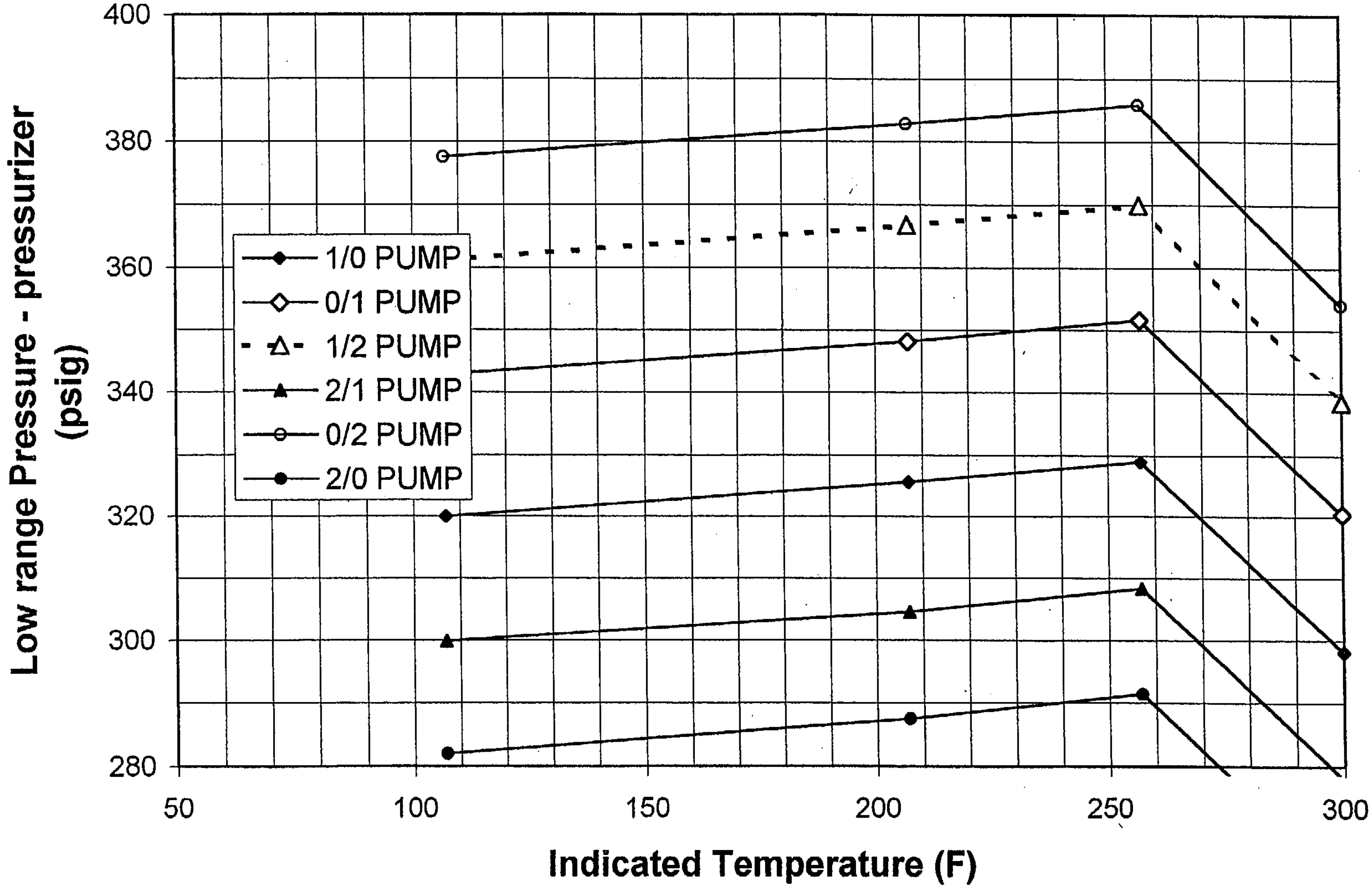


Figure 14
DHRS OPERATION LIMITS (with Instr Uncrt)



ATTACHMENT 1 LTOP Discussion

The methodology for protecting against LTOP events at B&W designed plants is described in this section. As such, the following elements are presented and discussed.

- o Definition of reactor vessel pressure limit
- o Definition of enable temperature
- o Definition of LTOP transients
- o Consequences of LTOP events
- o Definition of methods to protect the reactor vessel pressure limit

Definition of the Reactor Vessel Pressure Limit

The LTOP allowable pressure versus temperature for the reactor vessel is limited to 100% of the steady-state Appendix G NDT limits (ASME Code Case 640). Thus, the flaw size, critical depth, allowable crack growth, and the calculational methodology are identical to those used in the Appendix G calculations. The use of steady state temperatures, rather than the transient resulting from technical specification heatup and cooldown limits is based on the likelihood that LTOP events occur during steady-state operations. It is not unreasonable to conclude that the likelihood of an LTOP event occurring without notice by plant operations during heatups or cooldowns is small. This steady state approach has been approved by the NRC for B&W and other operating PWR plants.

Definition of Enable Temperature

ASME Code 1995 Edition through 1996 Addendum defines the LTOP enable temperature as the RTndt temperature of the limiting material plus 50°F.

Definition of LTOP Transients

LTOP events occur as the result of equipment malfunction or operator error that results in mass or energy addition to the reactor coolant system. In the B&W plant operating history, only once has the technical specification Appendix G limits been violated due to an LTOP event. Because of restrictions that preclude water-solid operation of the pressurizer (i.e., a steam or nitrogen bubble is maintained with the reactor vessel head on), this plant design is less likely to exceed Appendix G limits.

In the B&W design, mass can be injected into the system through: (1) the four HPI nozzles (one or two of which also serve as normal makeup); (2) the core flood nozzles through which the core flood tank system, decay heat removal system, and low pressure injection system (LPI) can provide added inventory; and (3) the pressurizer spray nozzle via HPI, LPI, or the nitrogen addition system. Energy can be added to the RCS via: (1) failure of the decay heat removal system; (2) actuation of the pressurizer heaters; and (3) reactor coolant pumps. As a result, the following transients were postulated and evaluated for their potential to increase reactor vessel pressure:

- Erroneous actuation of the High Pressure Injection (HPI) system
- Erroneous opening of the core flood tank discharge valve
- Erroneous addition of nitrogen to the pressurizer
- Makeup control valve (makeup to the RCS) fails full open
- All pressurizer heaters erroneously energized
- Temporary loss of the Decay Heat Removal System's (DHRS) capability to remove decay heat from the RCS
- Thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator

Consequences of LTOP Events

Each of the postulated LTOP events were analyzed to determine the rate of RCS pressure increase and/or the total amount of pressure increase that the system would experience. A stand alone thermal hydraulic model of the pressurizer was used for these predictions. Capabilities to model RCS inventory increases (e.g., makeup, HPI), inventory decreases (e.g., letdown), RCS expansion, and pressurizer heaters were included. A range of initial pressures and pressurizer levels were applied so that the pressurization rates could be applied to different initial P-T operating conditions. A brief summary of each transient response is provided below.

Erroneous actuation of the High Pressure Injection (HPI) system - this event would be the most limiting LTOP transient. However, HPI actuation results in a very rapid pressurization of the RCS and precludes achieving the necessary 10 minutes for operator action. Thus, this event is prevented below the LTOP enable temperature through plant procedures.

Erroneous opening of the core flood tank discharge valve - this event is precluded by closing and locking out the breakers of the motor operated block valves before the RCS pressure decreases below the CFT pressure (600 psig). This will occur prior to cooling below the ART

Erroneous addition of nitrogen to the pressurizer - this event can not overpressurize the RCS because of plant equipment that regulates the nitrogen pressure to 150 psig (i.e., pressure regulator and relief valves).

Makeup control valve (makeup to the RCS) fails full open - this event results in a pressurization rate of 20 to 30 psi/minute and is the most limiting of the remaining LTOP events.

All pressurizer heaters erroneously energized - this event is a slow transient (9 to 12 psi/minute) and is bounded by the failed makeup control valve event.

Temporary loss of the Decay Heat Removal System's (DHRS) capability to remove decay heat from the RCS - this event is a slow transient (7 psi/minute) and is bounded by the failed makeup control valve event.

Thermal expansion of the RCS after starting an RC pump due to stored thermal energy in the steam generator - this event results in a finite increase in pressure that is less than the margin between the Appendix G and LTOP limits. Because of the presence of a pressurizer bubble, this event is much less severe than at other PWRs.

In summary, the most limiting, credible event is the failed open makeup control valve. Because of system design differences, the plant response is sensitive to the makeup pump head-capacity curve and system resistance. This requires each plant to evaluate a plant specific response.

<u>Definition of Methods to Protect the Reactor Vessel Pressure Limit</u>

In general, each plant is equipped with either : (1) a dual setpoint pilot operated relief valve that is set below the LTOP limit, or (2) an additional relief valve (e.g., decay heat removal system relief valve) that is also set below the LTOP limit. In the event of relief valve failure, plant operation is limited (i.e., combination of operating pressure, pressurizer level) such that, in the event of the most limiting LTOP event, failed open makeup control valve, either: (1) ten minutes are available between the time the pressure-temperature operating limits are exceeded and the LTOP limits are violated, thus, providing adequate time for the operator to terminate the event, or (2) the available makeup tank volume would be exceeded and thus terminate the event before the LTOP limit is violated.

Two means of setting operating limits have been used for the failed open makeup control valve. The first approach assumes that the plant is operating at the maximum allowable pressure (as defined by bounds of the Appendix G heatup and cooldown limits and the PORV setpoint) at the time at which the failed open makeup control valve event occurs. Then, using plant specific makeup flow vs. RC pressure curves, the maximum allowable initial pressurizer level that will cause the tenth minute pressure to equal the LTOP pressure is determined. Thus, if the pressurizer level is maintained below this value for temperatures less than the enable temperature and if the RC pressure is less than the Appendix G

heatup/cooldown pressure, the LTOP limit will not be exceeded during ten minutes of the failed open makeup control valve event.

The second approach is similar except that the maximum allowable pressurizer level is set and the maximum allowable pressure vs. temperature curve is determined. If this curve results in higher pressures than the Appendix G heatup/cooldown curve, the LTOP limit is protected by the Appendix G curves (for this pressurizer level). If this curve results in lower pressures, than this curve is implemented as the limiting operator curve.

In performing either approach, the integrated makeup flow is determined. This allows the makeup inventory that will be exhausted during ten minutes of operation to be determined which can then be used as a means of LTOP protection.

In addition, RCS vent size calculations that will prevent pressurization during the failed open makeup control valve are calculated to provide backup LTOP protection. For example, one plant can prevent RCS pressurization if a $0.75 in^2$ vent is available. Thus, if the PORV is declared inoperable, a vent (e.g., steam generator hand-hole) can be opened to protect against LTOP events.

ATTACHMENT 2 HU & CD & ISLH COMBINED (SUBMITTAL) CURVES

The following two curves were constructed to assist in the TMI-1 NRC submittal. Figure 3.1-1 is a composite of the limiting points for both the heatup and cooldown NDT limits. Figure 3.1-2 is the same curve as Figure 11 with some notes added. The following table is the bases of the HU/CD curve and Table 5 above is the bases for the ISLH curve. Note that the actual PT analyses started RC pumps at 90F (with the operational limit set to 100F) and the cooldown RC pump limit operation was analyzed at 190F and the final curve "smoothing" used adds the conservatism that allows the operational limit set to 190F. The heatup pump start limit will insure a conservative analysis (using the 7F low range temperature limit) compared to actual pumps start. Also note that the allowable pump combination listed below are not necessarily allowed by procedures for heatups and cooldowns (so as to protect other equipment). The ART values listed on the figures are from References 1 and 7.

Limiting Composite Heatup & Cooldown PT points

Temp	Press	Temp	Press	Temp	Press	Temp	Press
F	psig	F	psig	F	psig	F	psig
82	479	207	653	329	2249	452	2551
87	479	212	656	332	2356	457	2561
92	479	217	677	337	2416	462	2571
97	479	222	696	342	2419	467	2582
102	479	227	717	347	2422	472	2593
107	479	232	717	352	2425	477	2605
112	479	237	717	357	2428	482	2617
117	479	242	717	362	2432	487	2630
122	481	247	717	367	2436	492	2643
127	484	252	717	372	2440	497	2657
132	489	252.1	869	377	2444	502	2672
137	495	257	909	382	2447	507	2687
142	503	262	954	387	2452	512	2703
147	511	267	1003	392	2457	517	2719
152	521	272	1057	397	2463	522	2734
157	533	277	1117	402	2469	527	2752
162	546	282	1183	407	2475	532	2770
167	547	287	1256	412	2481	537	2790
172	547	292	1336	417	2488	542	2809
177	547	297	1425	422	2495	547	2830
182	547	302	1523	427	2503	552	2851
187	547	307	1631	432	2510	557	2882
192	547	312	1754	437	2519	562	2903
192.1	585	317	1886	442	2527	567	2925
202	638	322	2032	447	2536	572	2945
		327	2193			582	2972

Figure 3.1-1
Reactor Coolant System Heatup/Cooldown Limitations
[Applicable through 29 EFPY]

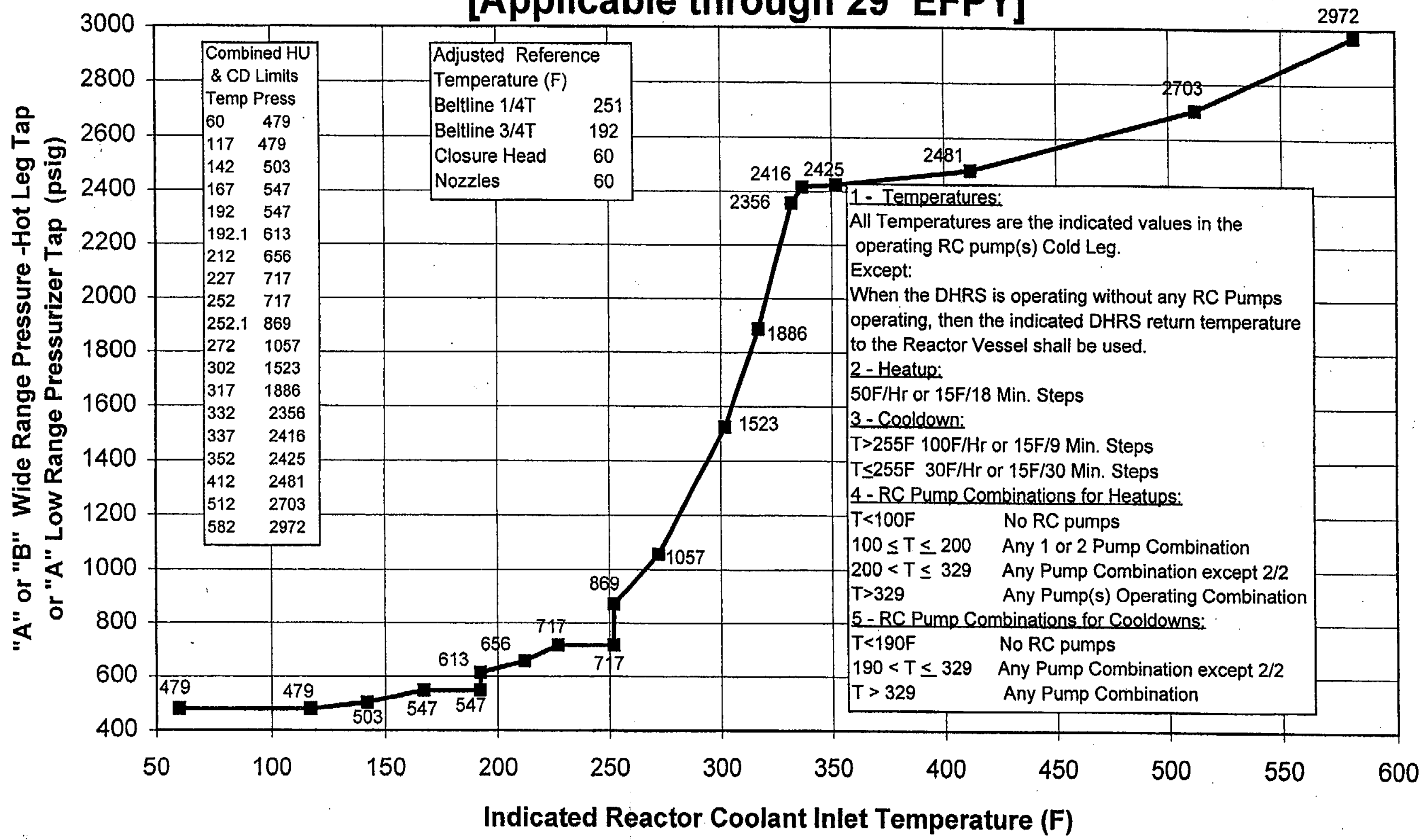
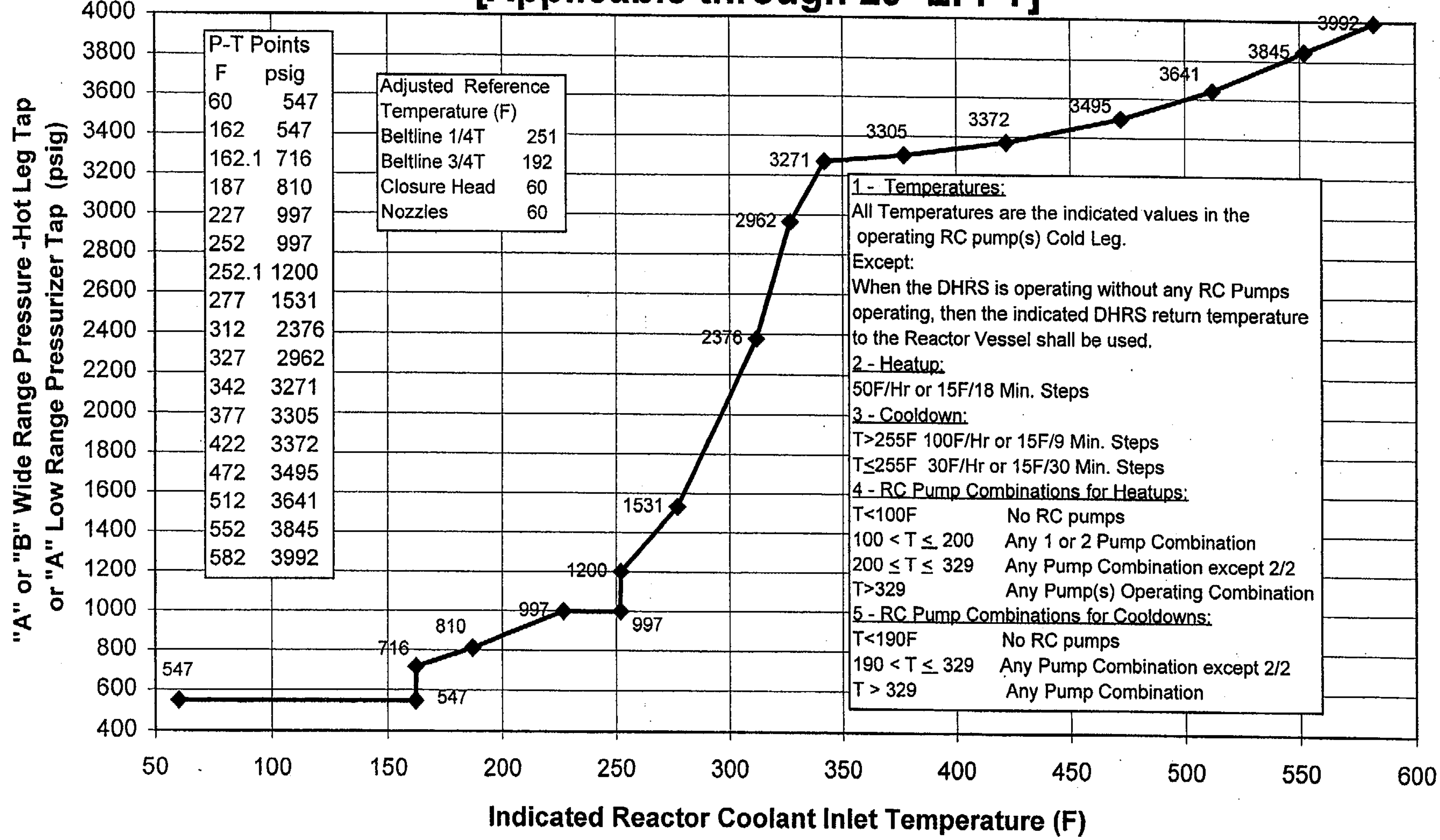


Figure 3.1-2
Reactor Coolant Inservice Leak Hydrostatic Test
[Applicable through 29 EFPY]



ENCLOSURE 3

Affected TMI Unit 1 Technical Specification Pages

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LIST OF FIGURES

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objectives

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

To assure that reactor vessel integrity by maintaining the stress intensity as a result of operational plant heatup and cooldown conditions and inservice leak and hydro test conditions below values which may result in non-ductile failure.

Specification

3.1.2.1 For operations until ~~17.7~~ 29 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding ~~17.7~~ 29 effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G. ~~Section V.B.~~ The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

3.1.2.5 The updated proposed technical specifications referred to in 3.1.2.4 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G. ~~Section V.C.~~

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional temperature step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 240°F indicated RCS temperature.

The unirradiated reference nil ductility temperature (RT_{NDT}) for the surveillance region materials were determined in accordance with 10 CFR 50, Appendixes G and H. For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated as described in Reference No. ~~6~~ 5.

The predicted RT_{NDT} was calculated using the respective predicted neutron fluence at ~~17.7~~ 29 effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2, Section C.1.1 for the plate metals and for the limiting weld metals (SA-1526 & WF-25).

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 4. Extrapolation of reactor vessel fluxes ~~(average of cycles 8 and 9),~~ and corresponding fluence accumulations, based on predicted fuel cycle design conditions through ~~17.7~~ 29 effective full power years of operation are described in Reference~~s 5 and~~ 6.

Amendment No. ~~29, 124, 157, 176~~ 208

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Based on the predicted RT_{NDT} after ~~17.7~~ 29 effective full power years of operation, the pressure/temperature limits of Figure 3.1-1 and 3.1-2 have been established by FTI calculation, Reference No. 7, in accordance with the requirements of 10 CFR 50, Appendix G. Also, see Reference 4. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2 and ASME Code Section XI, Appendix G, as modified by ASME Code Case N-640 and N-588. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves. The pressure limit lines on Figure 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. A 12°F error in measured temperature.

c. System pressure is measured in ~~either~~ RCS "A" loop, hot leg.

d. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) BAW-1901, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program

(4) BAW-1901, Supplement 1, Analysis of Capsule TMI-1C, GPU Nuclear, Three Mile Island Nuclear Station - Unit 1, Reactor Vessel Materials Surveillance Program, Supplement 1 Pressure - Temperature Limits.

(5) FTI Calculation No. 32-5011059-00, "TMI-1 Reactor Vessel Adjusted RT_{NDT} Values for 23 and 29 EFPY." ~~BAW-2108, Rev. 1, B&WOG Materials Committee Report "Fluence Tracking System"~~

(6) FTI Calculation No. 86-5010023-00, "TMI Cycle 5-11 Final Report." ~~GPU Nuclear calculation No. C-1101-221-E520-013 Rev. 0, "TMI-1 Reactor Vessel Welds Fluence, RT_{PTS} and RT_{NDT} per R.G. 1.99 R-2, Pos. No. 1"~~

(7) FTI Calculation No. ~~32-5001065-01, "TMI-1 P/T Limits," March 1998.~~ 32-5011638-01, "TMI-1 29 EFPY P/T Limits."

Figure 3.1-1
Reactor Coolant System Heatup/Cooldown Limitations
[Applicable through 29 EFPY]

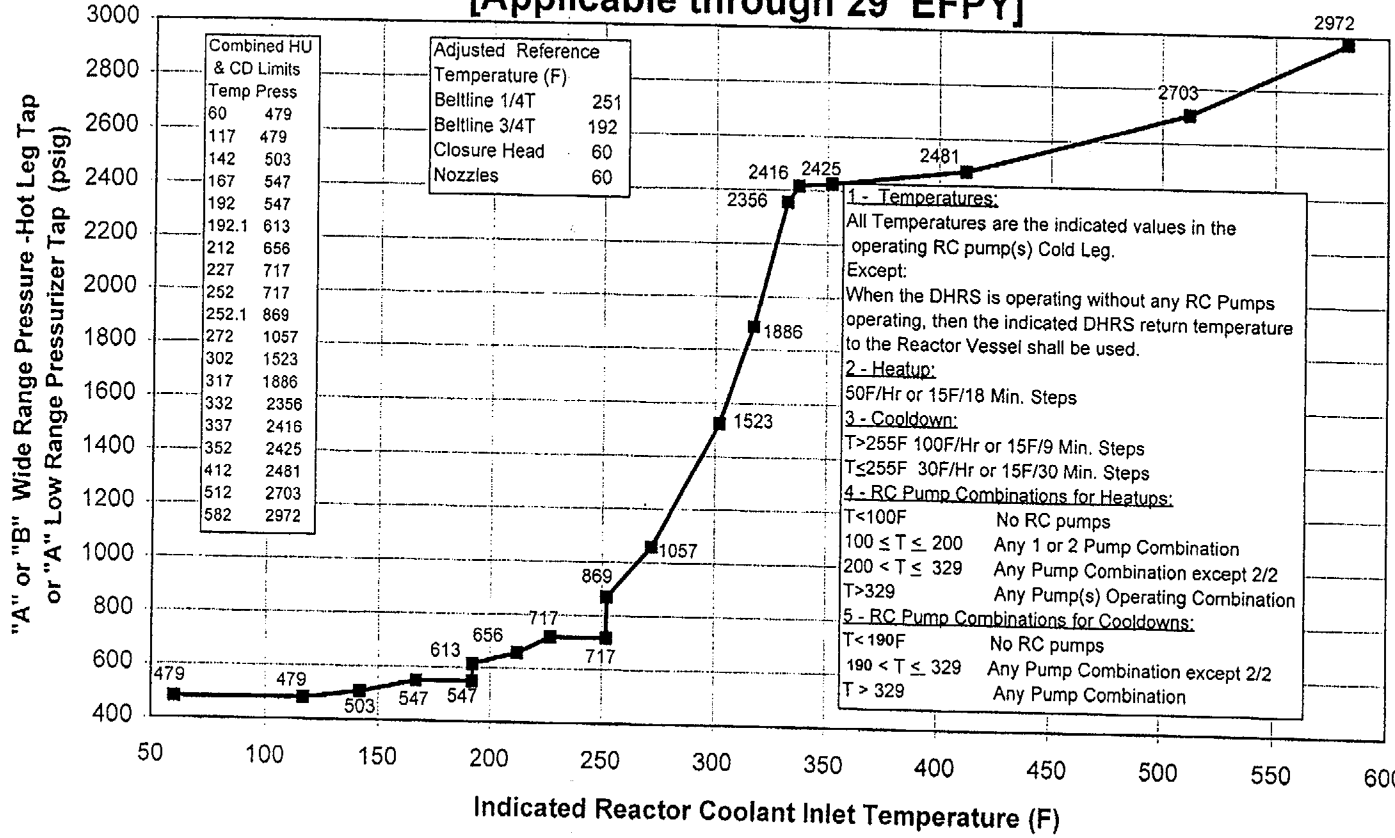
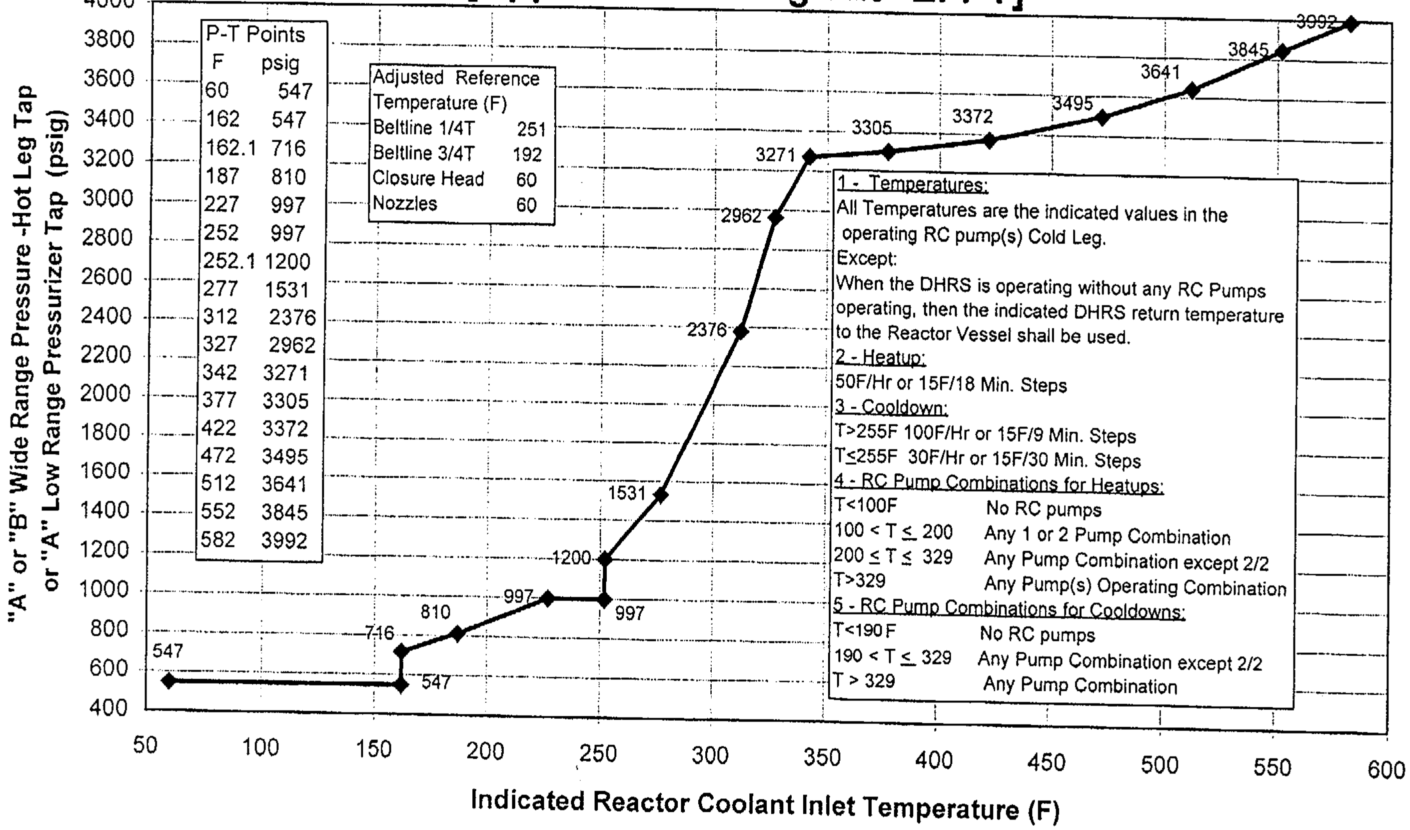


Figure 3.1-2
Reactor Coolant Inservice Leak Hydrostatic Test
[Applicable through 29 EFPY]



Amendment 25, 134, 167, 176, 208

3.1.12 Pressurizer Power Operated Relief Valve (PORV), ~~and~~ Block Valve, and Low Temperature Overpressure Protection (LTOP)

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.~~1~~3 When the indicated RCS temperature is below ~~332~~ 329°F the PORV shall not be taken out of service, nor shall it be isolated from the system unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out.
b. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed.
c. Head of the Reactor Vessel is removed.

3.1.12.2 The PORV settings shall be as follows:

b.~~1~~. Unless the Low Temperature Overpressure Protection Setpoint is in effect, the ~~Above 275°F ± 12°F - Minimum 2425 psig (Nominal 2450 psig).~~ PORV lift setpoint will be a minimum of 2425 psig.

With the PORV setpoint below the minimum value, within 8 hours either:

1. restore the setpoint above the minimum value, or
2. close the associated block valve, or
3. close the PORV, and remove power from PORV
4. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

a.~~2~~. Low Temperature Overpressure Protection Setpoint ~~Below 275°F ± 12°F - Maximum 510 psig (Nominal 485 psig).~~

1. When indicated RCS temperature is ≤ 329°F, the LTOP system shall be operable as defined in Specification 3.1.12.1 and
2. The PORV will have a maximum lift setpoint of 552 psig.

With~~in~~ the PORV setpoint above the maximum value, within 8 hours either:

1. restore the setpoint below the maximum value, or
2. verify pressurizer level is ≤ 100 inches indicated and satisfy the requirements of Technical Specification 3.1.12.~~1~~3 allowing the PORV to be taken out of service.

Amendment No. 56, 78, 149, 167, 186

3.1.12.1 LTOP Protection

~~3.1.12.3~~ If the reactor vessel head is installed and ~~T_{avg} is ≤332°F~~ indicated RCS temperature is ≤ 329 °F:

a. High Pressure Injection Pump breakers shall ~~not~~ be racked ~~in unless:~~ out or

~~a.~~ MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed, and

b. Pressurizer level maintained ~~is~~ ≤ ~~220~~ 100 inches. If pressurizer level is > ~~220~~ 100 inches, restore level to ≤ ~~220~~ 100 inches within 1 hour.

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

a. With the PORV Block Valve inoperable, within 1 hour either:

1. restore the PORV Block Valve to OPERABLE status or

2. close the PORV (verify closed) and remove power from the PORV

3. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the PORV block valve inoperable, restore the inoperable valve to OPERABLE status prior to startup from the next COLD SHUTDOWN unless the COLD SHUTDOWN occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a COLD SHUTDOWN occurs within this 90 day period, restore the inoperable valve to OPERABLE status prior to startup for the next fuel cycle.

Bases

If the PORV is removed from service while the RCS is below ~~332~~ 329°F, sufficient measures are incorporated to prevent severe overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2. Above ~~287~~ 329°F ~~(275°F + 12°F)~~, the PORV setpoint has been chosen to limit the potential for inadvertent discharge or cycling of the PORV. Other action such as removing the power to the PORV has the same effect as raising the setpoint which also satisfies this requirement. There is no upper limit on this setpoint as the Pressurizer Safety Valves (T.S. 3.1.1.3) provide the required overpressure relief.

Below ~~263~~ 329°F ~~(275°F - 12°F)~~, the PORV setpoint is reduced to provide the required low temperature overpressure relief when high pressure sources and flowpaths are in service. There is no lower limit on the pressure actuation specified as lower setpoints also provide this same protection.

In both cases, the ~~275°F ± 12°F~~ setting is specified to reflect the nominal value which allows for normal variations in the temperature setpoint while maintaining the tolerances assumed in the bases for T.S. 3.1.2. Either pressure actuation setpoint is acceptable within the temperature range between ~~263~~ 313°F and ~~287~~ 329°F. 329

With RCS temperatures less than ~~332~~°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than ~~220~~ 100 inches to allow time for action to prevent severe overpressurization in the event of any single failure.

The PORV block valve is required to be OPERABLE during the HOT STANDBY, STARTUP, and POWER OPERATION in order to provide isolation of the PORV discharge line to positively control potential RCS depressurization.

For protection from severe overpressurization during HPI testing, refer to Section 4.5.2.1.c.

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification

4.5.2.1 High Pressure Injection

a. During each refueling interval and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

b. The test will be considered satisfactory if the valves **(MU-V-14A/B & 16A/B/C/D)** have completed their travel and the make-up pumps are running as evidenced by **system flow**. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A/B/C/D shall be conducted only under either of the following conditions:

Indicated RCS temperature 329

1) ~~Tavg~~ shall be greater than ~~332~~°F.

2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. During each refueling period and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps listed in 4.5.1.1b have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.