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A WISCONSIN ENERGY COMPANY

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10 CFR 50.90

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U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Ladies and Gentlemen:

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFCATIONS CHANGE REQUEST 219
ADOPTION OF PRESSURE AND TEMPERATURE LIMITS REPORT
AND REVISED P-T AND LTOP LIMITS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, Wisconsin Electric Power Company, licensee, hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively. The purpose of the proposed amendments is to implement a Pressure and Temperature Limits Report (PTLR) concurrent with implementation of Improved Standard Technical Specifications at the Point Beach Nuclear Plant (PBNP). An application to convert the PBNP custom Technical Specifications to Standard Technical Specifications based on NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, was submitted on November 15, 1999. That submittal was based on the incorporation of a PTLR and appropriate references in the proposed Standard Technical Specifications.

NRC Generic Letter 96-03 provides guidance for licensees allowing relocation of the Reactor Coolant System pressure temperature limit curves and Low Temperature Overpressure Protection (LTOP) system limits from the Technical Specifications (TS) to a PTLR. Relocation of these limits is allowed, provided that the limits are determined in accordance with NRC-approved methodologies that are referenced in the TS and any revisions to the PTLR are provided to the NRC.

Descriptions and bases for the proposed changes, a safety evaluation, and no significant hazards determination are provided. The proposed PTLR for the Point Beach units is also provided. The Pressure Temperature limits and Low Temperature Overpressure Protection system limits contained within this submittal are based on the analyses provided in support of this amendment request.

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We have determined that the proposed amendments do not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, we conclude that the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

We request that these amendments be reviewed and approved such that the PTLR may be implemented with the improved TS at PBNP.

Sincerely,

Mark A. Reddemann
Vice President
Point Beach Nuclear Plant

TGM/tja

Attachments

Subscribed and sworn before me on
this 10th day of March, 2000.

Christine K. Pozorski
Notary Public, State of Wisconsin

My commission expires 8/25/2002.

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
PSCW

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TECHNICAL SPECIFICATIONS CHANGE REQUEST 219
ADOPTION OF PRESSURE TEMPERATURE LIMITS REPORT
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POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
DESCRIPTIONS AND BASES FOR PROPOSED CHANGES

Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," describes the process for relocation of these limits from the Technical Specifications (TS) to a Pressure Temperature Limits Report (PTLR). The relocation of the pressure temperature limits curves and low temperature overpressure protection system limits is allowed provided these limits are determined using NRC approved methodologies, the applicable methodologies are referenced in the TS, and revisions or supplements to the PTLR are provided to the NRC.

Wisconsin Electric has submitted an application to convert the PBNP Technical Specifications, to the improved Standard Technical Specifications based on NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1. That submittal, dated November 15, 1999, is based on the incorporation of a PTLR and associated references in the proposed Technical Specifications.

The proposed changes to allow adoption of the PTLR and provide for its incorporation into the proposed converted Technical Specifications for Point Beach Nuclear Plant, Units 1 and 2 are described below.

1. Description of proposed change:

   Add the definition of a Pressure and Temperature Limits Report (PTLR) to the Technical Specifications. This definition corresponds to that proposed in our November 15, 1999, submittal for conversion to the Standard Technical Specifications and the definition is consistent with NUREG-1431. The Specification 5.6.5 reference is based on the proposed administrative section that will describe the PTLR requirements. The PTLR is defined as follows:

   Pressure and Temperature Limits Report: The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and enable temperature associated with the Low Temperature Overpressure Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

Basis for proposed change:

The addition of the PTLR definition is in accordance with the guidance provided in GL 96-03. The proposed definition is consistent with NUREG -1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1.

The proposed addition of the PTLR definition is an administrative change because the new definition does not, in itself, impose new requirements on the operation of the Point Beach Nuclear Plant.

2. Description of proposed change:

Add an administrative reporting requirement to the Administrative Controls section of the TS, which defines the approved methodologies to be used for determination of the Point Beach reactor vessel pressure and temperature limits. The proposed reporting requirement, which will be included in the proposed TS conversion, is as follows (the LCO references correspond to applicable proposed TS conversion and may change pending review and approval of amendments to implement the TS conversion):

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

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

(1) LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
(2) LCO 3.4.6, "RCS Loops-MODE 4"
(3) LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled"
(4) LCO 3.4.10,"Pressurizer Safety Valves"
(5) LCO 3.4.12,"Low Temperature Overpressure Protection (LTOP)"

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

WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996.

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

Basis for proposed change:

The reporting requirement proposed for addition to the TS is in accordance with the guidance in GL 96-03 and NUREG-1431. The addition of this requirement ensures that the RCS pressure and temperature limits and LTOP system limits are determined in accordance with NRC approved methodologies. Thus, reactor operation will continue to meet all regulatory and design basis requirements.

The addition of these methodologies to the Technical Specifications therefore, does not impose any new requirements on the operation of PBNP and is administrative.

3. Description of proposed change:

Relocate the Reactor Coolant System temperature and pressure limits shown on TS Figures 15.3.1-1 and 15.3.1-2 and the heatup and cooldown limits contained in TS 15.3.1.B.1.a, b, and c to the PTLR. Additionally, the Reactor Coolant System temperature and pressure limits shown on Figures 15.3.1-1 and 15.3.1-2 are expiring in 2001 and the new curves are provided. These curves are based on new analyses contained in the attached information.

Basis for proposed change:

As described in GL 96-03, relocation of these limits allows the licensee to maintain these limits efficiently and at a lower cost, provided the parameters for constructing the curves and setpoints are derived using methodology approved by the NRC. The current pressure and temperature limits for Point Beach expire in 2001 and PBNP is in the process of converting to Standard Technical Specifications. Therefore, this is a convenient time to adopt the PTLR. New analyses have been completed to determine the new limits for PBNP. These new analyses account for all factors described in the approved methodology document: WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996. The new analyses are provided, for NRC review, in a separate attachment to this submittal.

4. Description of proposed change:

Delete TS 15.3.1.A.4.a, which requires at least one pressurizer safety valve to be operable whenever the reactor head is on the vessel, and change the TS 15.3.1.A.4.b applicability requirement for two pressurizer safety valves to be operable from "whenever the reactor is critical," to "whenever the reactor coolant system temperature is ≥ the LTOP enable temperature specified in the PTLR."

Basis for proposed change:

These pressurizer safety valve requirements in the current Technical Specifications predate the Technical Specifications requirements for LTOP. The Pressurizer Safety valves were considered the primary protection for the RCS from overpressurization prior to the inclusion of LTOP requirements in the Techical Specifications. After LTOP was added to the PBNP TS and implemented, the Pressurizer Safety valves were no longer required for protection of the RCS below the LTOP enable temperature. Therefore, the Pressurizer Safety valve LCO could have been changed to only require operability of the Pressurizer Safety valves when the RCS conditions equal or exceed the LTOP enable temperature. The proposed change is consistent with the requirements in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants."

5. Description of proposed change:

Change the applicability of TS 15.3.15, "Low Temperature Overpressure Protection System," from "when the reactor coolant system temperature is < 355°F," to "when the reactor coolant system temperature is < the LTOP enable temperature specified in the PTLR."

Basis for proposed change:

This change incorporates the proper reference for the applicability of the LTOP LCO, based on the relocation of the LTOP enable temperature to the PTLR

6. Description of proposed change:

Relocate the temperature limit for reactor coolant pump starting in TS 15.3.15.B.2 to the PTLR.

Basis for proposed change:

The limitations placed on starting the reactor coolant pump are based on limiting the pressure transient that can occur when a reactor coolant pump is started and there is no pressure absorbing volume in the pressurizer or there is a large temperature differential between the RCS and the Steam Generator. The TS RCS temperature limit for starting the RCP is based on the enable temperature for the Low Temperature Overpressure Protection system. The LTOP enable temperature is being relocated to the PTLR, therefore, the TS temperature limit for starting the RCP must reference the PTLR for this parameter.

7. Description of proposed change:

Relocate the LTOP enable temperature in TS 15.3.15.A.1 to the PTLR and reduce the temperature from 355°F to 270°F.

Basis for proposed change:

As described in GL 96-03, relocation of these limits allows the licensee to maintain these limits efficiently and at a lower cost, provided the parameters for constructing the setpoints are derived using methodology approved by the NRC. The current pressure and temperature limits for Point Beach expire in 2001 and PBNP is in the process of converting to Standard Technical Specifications. Therefore, this is a convenient time to adopt the PTLR.

The new analyses result in a lower limit for the LTOP enable temperature. These new analyses account for all factors described in the approved methodology document: WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996. As previously stated, the new analyses are provided, for NRC review, as a separate attachment to this license amendment request.

8. Description of proposed change:

Relocate the LTOP pressurizer power operated relief valve setpoint in TS 15.3.15.A.1.a to the PTLR and increase the limit from 440 psig to 500 psig.

Bases for proposed change:

As described in GL 96-03, relocation of these limits allows the licensee to maintain these limits efficiently and at a lower cost, provided the parameters for constructing the setpoints are derived using methodology approved by the NRC. The current pressure and temperature limits for Point Beach expire in 2001 and PBNP is in the process of converting to Standard Technical Specifications. Therefore, this is a convenient time to adopt the PTLR.

The new analyses result in a higher limit for the LTOP pressurizer power operated relief valve setpoint. These new analyses account for all factors described in the approved methodology document: WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996. As previously stated, the new analyses are provided, for NRC review, as a separate attachment to this license amendment request.

9. Description of proposed change:

Add a requirement to TS 15.3.15.B.1 for isolation of each safety injection (SI) accumulator with pressure greater than or equal to the maximum allowed RCS pressure for the existing RCS cold leg temperature as determined by the pressure temperature limit curves in the PTLR. This change is proposed to make the LCO requirements for PBNP consistent with the LCO requirements contained in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants."

Basis for proposed change:

This proposed change is more restrictive because it establishes system alignment requirements that do not exist in the current TS. This system alignment limitation for the accumulators eliminates the need to consider accumulator injection for LTOP. This requirement was not established for PBNP previously, because the SI accumulators are normally isolated prior to reduction of RCS below the SI accumulator pressure, and the isolation valves are not re-opened until RCS pressure exceeds SI accumulator pressure. This normal re-alignment is performed to prevent inadvertent injection via the SI accumulators. Therefore, this LCO requirement was not previously considered to be necessary. This proposed additional TS requirement does not add any restrictions that could affect safety of the reactor or any other safety function at PBNP.

10. Description of proposed change:

Relocate the upper temperature limit in TS 15.3.15.A.2.a to the PTLR and decrease the limit from 355°F to 270°F.

Basis for proposed change:

As described in GL 96-03, relocation of these limits allows the licensee to maintain these limits efficiently and at a lower cost, provided the parameters for constructing the setpoints are derived using methodology approved by the NRC. The current pressure and temperature limits for Point Beach expire in 2001 and PBNP is in the process of converting to Standard Technical Specifications. Therefore this is a convenient time to adopt the PTLR.

The new analyses result in a lower limit for the LTOP enable temperature. These new analyses account for all factors described in the approved methodology document: WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996. As previously stated, the new analyses are provided, for NRC review, as a separate attachment to this license amendment request.

11. Description of proposed change:

Change the applicable TS bases to properly reflect the above changes.

Basis for proposed change:

The current Technical Specifications bases are being totally revised and rewritten for the conversion to standard TS. This change is accomplished via the November 15, 1999 submittal to convert the PBNP TS to standard TS.

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SAFETY EVALUATION

NRC Generic Letter 96-03 provides guidance for licensees, allowing relocation of the Reactor Coolant System (RCS) pressure temperature (P-T) limits curves and the Low Temperature Overpressure Protection (LTOP) limits from the Technical Specifications (TS). This relocation is allowed provided that the P-T limits curves and the LTOP limits are included in a Pressure Temperature Limits Report (PTLR) and are determined with NRC-approved methodologies referenced in the administrative controls reporting requirements section of the TS. The limits removed from the TS and relocated to the PTLR can then be changed using the referenced methodologies without prior NRC review and approval. Appropriate safety limits will continue to be maintained in the TS. Changes to the PTLR parameters will be reported to the NRC as directed by the proposed TS administrative controls reporting requirement.

This amendment request further proposes new pressure-temperature limits and low temperature overpressure protection limits, based on new analyses performed due to anticipated expiry of the current pressure-temperature limits within about 1 year. The new analyses are provided as an attachment to this letter.

This amendment request is proposed to allow implementation of a PTLR with implementation of the standardized Technical Specifications conversion at the Point Beach Nuclear Plant (PBNP). An amendment request was submitted for the proposed conversion of the PBNP custom TS to the standard TS on November 15, 1999. That submittal is based on implementation of the PTLR.

This amendment request adds the definition of the PTLR, and the reporting requirement as recommended by NUREG-1431, meeting the guidance in Generic Letter 96-03. The P-T limits and LTOP limits provided in this submittal that are proposed to be relocated are and will continue to be determined by analyses performed in accordance with NRC approved methodologies. The applicable approved methodologies are listed in the proposed TS administrative controls reporting requirement.

Therefore, the Specifications continue to meet the criteria of 10 CFR 50.36, Criterion 2, in that appropriate safety limits are maintained in the Technical Specifications. The reporting requirement ensures that the limits on the safety analyses continue to be met and that all changes to the design are performed in accordance with approved methodologies. The proposed Technical Specifications changes for relocation of the P-T limits and LTOP limits are administrative. The proposed changes for the new P-T and LTOP limits are based on the results of the analyses provided with this amendment request and performed in accordance with NRC approved methodology.

Operation of PBNP in accordance with these proposed changes continues to ensure safe operation of the plant and will not be inimical to the health and safety of the public.

GL 96-03 provides guidance for the relocation of the P-T and LTOP limits to the PTLR. The following information provides the explanation and disposition of the provisions for the methodology described in GL 96-03 and in the administrative controls section of standard Technical Specifications:

Provision 1: "The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued)."

The methodology for calculation of the Point Beach Units 1 and 2 neutron fluence is as described in WCAP-14040-NP-A, and as implemented in WCAP-12794, Rev. 4 for Unit 1 and WCAP-12795, Rev. 3 for Unit 2. The neutron fluence calculations for Point Beach are carried out using forward and adjoint formulations in r,θ geometry of the two-dimensional Discrete Ordinates Transport (DOT) code. The anisotropic scattering is treated with a P_3 expansion of the scattering cross section and the angular discretization is modeled with an S_8 order of angular quadrature. The core power distribution and neutron source distribution are estimated conservatively, accounting for spectral changes due to plutonium accumulation. The methodology uses the BUGLE-96 cross section library which is based on the data set of the Evaluated Nuclear Data File/B-VI (ENDF/B-VI). The DOT code was re-benchmarked to the ENDF/B-VI cross sections using the Poolside Critical Assembly (PCA) simulator experiment at the Oak Ridge National Laboratory (ORNL), surveillance capsule and cavity dosimetry measurements. The results of analytic sensitivity studies demonstrate that the methodology is capable of providing best estimate fluence evaluations within ± 20 percent (1σ).

This methodology has evolved over many years and has been validated using NRC-sponsored experimental measurements (i.e., the PCA experiment at ORNL as well as the reactor surveillance capsule database). The latest improvement was the introduction of the ENDF/B-VI-based inelastic scattering cross sections for iron. In WCAP-14040-NP-A, the DOT code was rebenchmarked with the new cross sections.

Provision 2: "The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves."

The reactor vessel material surveillance program is designed to monitor radiation effects on reactor vessel materials under actual operating conditions. The radiation effects are determined from changes in fracture toughness of the material; this is obtained by pre- and post-irradiation testing of vessel material specimens removed from surveillance capsules. Appendix H to 10 CFR Part 50 requires that the surveillance program satisfy ASTM Standard E-185 which specifies material selection, material testing, specimen sizes, and specimen quantities.

Point Beach Units 1 and 2 utilize the methodology of WCAP-14040-NP-A, which references Appendix H to 10 CFR Part 50. WCAP-14040-NP-A provides guidance on the derivation of a material property referred to as the initial reference temperature, IRT_{NDT}. WCAP-14040-NP-A utilizes the test method and derivation as defined in paragraph NB-2331 of Section III of the ASME Code and NRC Branch Technical Position MTEB 5-2 to determine IRT_{NDT}. The value is derived from the results of a series of Charpy V-notch impact tests and drop-weight tests. Initially, the nil-ductility transition temperature, T_{NDT}, is determined by drop-weight tests. Next, at a temperature not greater than T_{NDT} + 60°F, each specimen of the Charpy V-notch test shall exhibit at least 35 mils of lateral expansion and at least 50 ft-lb of absorbed energy. If the two requirements are met, T_{NDT} is the initial reference temperature (IRT_{NDT}). If the two criteria are not met, additional Charpy V-notch tests (in groups of three specimens) are performed to determine the temperature, T_{Cv}, at which the criteria are met. In this case T_{Cv} – 60°F is the IRT_{NDT}. If the Charpy V-notch test has not been performed at T_{NDT} + 60°F, or, if the test at this temperature does not exhibit the two requirements, the IRT_{NDT} can be obtained by a full Charpy impact curve developed from the minimum data points of all the Charpy tests performed.

The Point Beach Unit 1 and 2 pressure vessel surveillance programs are in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

Provision 3: "Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR."

The Point Beach LTOP systems are designed to provide the capability, during reactor operation at low-temperature conditions, to automatically prevent the reactor coolant system (RCS) pressure from exceeding the applicable limits established by Appendix G to 10 CFR Part 50. LTOP is manually enabled by reactor operators on the basis of its predetermined enable temperature during reactor startup and shutdown. WCAP-14040-NP-A specifies a methodology of developing a LTOP system which uses the pressurizer power-operated relief valves (PORVs) with variable setpoints. However, for Point Beach Units 1 and 2, a single setpoint is established based on the most limiting reactor vessel beltline materials at the minimum allowable pressure for reactor coolant system pressurization established pursuant to Appendix G to 10 CFR Part 50. After LTOP is enabled, it will automatically function to mitigate overpressure.

The design basis of the Point Beach LTOP systems considers both mass-addition and heat-addition transients. The limiting mass-addition transient is defined as a mass injection scenario when the RCS is water solid. The transient is postulated as isolation of the letdown system coupled with the full flow from one safety injection pump, combined with the full charging pump flow from all (three) charging pumps, at a time when all (two) reactor coolant pumps are operating. Operation of more than one safety injection pump is administratively controlled to disable the second safety injection pump when LTOP is enabled. This combination of operating pumps establishes the most limiting mass-addition transient for the Point Beach units.

The design basis heat input transient assumes the starting of the first reactor coolant pump during water solid conditions with a temperature difference between the reactor coolant system and the steam generator secondary-side of 50°F. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient, and pressure is relieved by a single power operated relief valve.

The major function of the LTOP system is to protect the structural integrity of the reactor vessel from excessive pressure and temperature loadings. In order to achieve this purpose, pressure-temperature limits established in accordance with the requirements of Appendix G to 10 CFR Part 50 are considered as the upper limits for the RCS during postulated transient conditions. However, since the overpressure events most likely occur during isothermal conditions in the RCS, the steady-state Appendix G limits are used for design of LTOP. Also, LTOP provides for an operational consideration to maintain the integrity of the PORV discharge piping. An upper pressure limit of 800 psia is selected for this purpose. This maximum pressure is selected on the basis of a generic study by Westinghouse using a type of PORV which would cause maximum back pressure in the piping during an overpressure transient.

The methodology for developing the PORV setpoints provides adequate protection for reactor vessel integrity and maintains proper operational margins. In calculating the PORV setpoints, plant parameters and transient conditions listed in Section 3.2.1 of WCAP-14040-NP-A are considered. This list contains initial RCS and steam generator parameters, PORV size and lifting characteristics, mass and heat input rate to RCS, pressure limits to be protected and other parameters and conditions. These data were included in a specialized version of the LOFTRAN computer code which is used to calculate the maximum RCS pressures due to overshoot of the RCS pressure under various overpressure transient conditions. Each of the two PORVs may have a different pressure setpoint. The staggered setpoints for the two PORVs would prevent excessive pressure undershoot that would challenge the RCP No. 1 seal performance criteria. However, each PORV will protect reactor vessel integrity, assuming a single failure of the other PORV.

Uncertainties in the pressure and temperature instrumentation utilized by the LTOP system are explicitly accounted for in determination of the LTOP setpoint. A process described by ISA Standard 67.04-1994 is used in the determination of these instrumentation uncertainties.

Provision 4: "The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2."

The methodology for determining the Point Beach adjusted reference temperature, ART, is as described in WCAP-14040-NP-A which conforms to NRC Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The ART is calculated by adding the initial nil-ductility transition temperature of the unirradiated material (IRT_{NDT}), the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin to account for uncertainties in the prediction method as follows:

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

The derivation of IRT_{NDT} was discussed above under Provision 2. The calculation of ΔRT_{NDT} due to irradiation conforms to RG 1.99, Revision 2, as follows:

$$\Delta RT_{NDT} = (CF)\ f^{(0.28 - 0.10 \log f)}$$

where, CF (in units of ^{o}F) is the chemistry factor, and f (in units of 10^{19} n/cm^2, E > 1.0 MeV) is the value of fast neutron fluence at a specific depth. The chemistry factor is obtained from RG 1.99, Revision 2 based on copper and nickel content of the vessel material. Another method of calculating the chemistry factor is to use credible surveillance data. The fast neutron fluence is calculated for any depth into the vessel wall by the following equation:

$$f = f_{surface} \exp(-0.24x)$$

where $f_{surface}$ (in units of 10^{19} n / cm^2, E > 1.0 MeV) is the neutron fluence at the inner wetted surface or clad-base metal interface of the vessel at the location of the postulated flaw, and x (in units of inches) is the depth into the vessel wall measured from the location of known neutron fluence.

The margin is included in the ART calculations to account for uncertainties in the values of IRT_{NDT}, copper and nickel contents, fluence and the calculational procedures. The margin is calculated by the following equation:

$$\text{Margin} = 2\,[\sigma_i^2 + \sigma_\Delta^2]^{1/2}$$

where, σ_i is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_i is estimated from the precision of the test method. For generic mean values, σ_i is the standard deviation from the set of data used to establish the mean. σ_Δ is 28°F for welds and 17°F for base metal per RG 1.99, Revision 2. σ_Δ is reduced by half, when credible surveillance data are used. σ_Δ need not exceed half the mean value of ΔRT_{NDT} for all cases.

Provision 5: "The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits."

The P-T limits for Point Beach Units 1 and 2 are calculated using linear elastic fracture mechanics in accordance with Appendix G to Section XI of the ASME Code. The method is based on restricting the stress intensity factor of the postulated defect to be less than the reference stress intensity factor of the limiting reactor vessel materials. In accordance with ASME Code Case ISI 99-29, the critical initiation stress intensity factor, K_{Ic}, is used in this calculation in lieu of the critical arrest stress intensity factor, K_{Ia}. The K_{Ic} is determined by the metal temperature and RT_{NDT} at the tip of the postulated flaw. The K_{Ic} curve in the ASME Code is given by the following equation:

$$K_{Ic} \quad = 33.2 + 20.734 \exp\,[0.02\,(T - RT_{NDT})]$$

Where, T is the metal temperature and RT_{NDT} is the ART value of the limiting vessel material at the 1/4t and 3/4t locations of the vessel wall.

The postulated reference flaw is assumed to have a depth of one-fourth of the beltline thickness and a length of 1.5 times the beltline thickness. In accordance with ASME Code Case ISI 99-29, the reference flaw in this calculation is postulated to be axially oriented for plates, forgings, and axial welds, and oriented circumferentially for circumferential welds.

The stress intensity factor caused by the postulated crack is limited to the reference stress intensity factor of the material as follows:

$$C \times K_{IM} + K_{IT} < K_{Ic}$$

Where, K_{IM} is the stress intensity factor caused by pressure (membrane) stress, K_{IT} is the stress intensity factor caused by the thermal stress, and C is a safety factor that is 2 for heatup and cooldown and 1.5 for hydrostatic and leak test conditions when the reactor core is not critical.

Equations used in the determination of K_{IT} and K_{IM} for Point Beach are as discussed in Chapter 7, *PWR Analysis Theoretical Report*, of the EPRI P-T Calculator for Windows® User's Manual, Version 3.0. The method of Raju and Newman is used in the determination of the stress intensity factors for Point Beach. This methodology is consistent with Appendix G to Section XI of the ASME Code and Standard Review Plan Section 5.3.2.

Provision 6: "Description of LTOP enabling temperature limit development methodology."

The "enable" temperature is the RCS temperature below which the LTOP system is required to function. This temperature was determined for Point Beach utilizing ASME Code Section XI Code Case ISI 99-29, "Plant Specific Enable Temperature." This Code Case specifies the enable temperature as the water temperature corresponding to a metal temperature of at least $RT_{NDT} + \tau$ at the quarter thickness beltline location. Where, τ may be determined on a plant specific basis, based on reactor vessel dimensions and design pressure, and the membrane stress correction factor, M_m, identified in Appendix G to ASME Section XI. Above this temperature, brittle fracture of the reactor vessel is not expected.

Provision 7: "The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves."

Appendix G of 10 CFR Part 50 imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F above the minimum temperature of the heatup and cooldown curves and must not be lower than the minimum temperature for the inservice pressure test. The minimum temperature requirement for pressurization of the RCS is calculated in accordance with the requirements of Appendix G to 10 CFR Part 50 and is identified on the Point Beach Pressure-Temperature Limits for Heatup diagram. The identified minimum bolt-up temperature includes consideration of possible instrument uncertainties for the RCS wide range temperature instrument loop.

Provision 8: "Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted *increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2σ) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{NDT} + 2\sigma$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology."*

The Point Beach Unit 1 and 2 pressure vessel surveillance programs are in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." However, surveillance specimens for the current limiting materials for the Point Beach reactor vessels were not included in the surveillance capsules when these plant specific surveillance programs were developed. Therefore, the results of the examinations of these specimens do not meet the credibility criteria of Regulatory Guide 1.99, Rev. 2 for Point Beach.

Surveillance data for the current limiting materials for the Point Beach reactor vessels are included in the plant specific surveillance programs for R.E. Ginna and Turkey Point Units 3 and 4. In addition, the B&W Owners Group's Master Integrated Reactor Vessel Surveillance Program (MIRVP) includes surveillance specimens for the limiting Point Beach materials. Surveillance data from these plant specific and integrated programs will be monitored by Point Beach, and if it is warranted, a supplement to the PTLR will be provided to the NRC to demonstrate how the results affect the approved methodology.

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATIONS CHANGE REQUEST 219
ADOPTION OF PRESSURE TEMPERATURE LIMITS REPORT
AND REVISED P-T AND LTOP LIMITS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
NO SIGNIFICANT HAZARDS DETERMINATION

In accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, Wisconsin Electric Power Company, licensee, requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively. The purpose of the proposed amendments is to implement a Pressure Temperature Limits Report (PTLR) concurrent with implementation of standardized Technical Specifications at the Point Beach Nuclear Plant (PBNP).

In accordance with the requirements of 10 CFR 50.91, Wisconsin Electric has evaluated operation of the Point Beach Nuclear Plant in accordance with the proposed changes against the standards of 10 CFR 50.92. Operation of the Point Beach Nuclear Plant in accordance with the proposed changes results in no significant hazards consideration. Our evaluation and basis for this conclusion follows.

1. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.**

   The proposed changes relocate the pressure-temperature limits and low temperature overpressure protection limits from the Technical Specifications to a Pressure Temperature Limits Report (PTLR). The proposed changes also provide revised pressure-temperature limits and revised low temperature overpressure protection limits. Appropriate design and safety limits are retained in the Specifications, thereby meeting the requirements of 10 CFR 50.36. Specific, approved methodologies used to determine and evaluate the parameter requirements are added to the Specifications and a reporting requirement is added to ensure the NRC is apprised of all changes. Operation of the PBNP will continue to meet all design and safety analysis requirements because approved methodologies are required to be used to evaluate and change parameters, and appropriate safety and design limits maintained in the Technical Changes. Therefore, neither the probability nor consequences of an accident previously evaluated can be increased.

2. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.**

   Operation of PBNP, in accordance with the proposed changes, will continue to meet all design and safety limits. Appropriate design and safety limits continue to be controlled within the Technical Specifications as they are presently. These changes will not result in a change to the design and safety limits under which PBNP operation has been determined to be acceptable. These changes cannot result in a new or different kind of accident from any accident previously evaluated.

3. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendment does not result in a significant reduction in a margin of safety.**

   Appropriate safety limits continue to be controlled by the Specifications. Changes to the relocated pressure-temperature and low temperature overpressure protection limits will be accomplished using NRC approved methodologies, thereby ensuring operation will continue within the bounds of the existing safety analyses including all applicable margins of safety. Therefore, operation in accordance with the proposed changes cannot result in a reduction in a margin of safety.

In summary, operation of Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated, does not create a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, operation in accordance with the proposed amendments involves no significant hazards consideration.

ATTACHMENT 4

EDITED TECHNICAL SPECIFICATIONS PAGES

o. Dose Equivalent I-131

Dose Equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988.

p. $\overline{E}$ - Average Disintegration Energy

$\overline{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ADD PROPOSED DEFINITION OF PTLR

D. Failure of Containment High-Range Radiation Monitor

A minimum of two in-containment radiation-level monitors with a maximum range of 10^8 rad/hr (10^7/hr for photons only) should be operable at all times except for cold shutdown and refueling outages. This is specified in Table 15.3.5-5, item 7. If the minimum number of operable channels are not restored to operable condition within seven days after failure, a special report shall be submitted to the NRC within thirty days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

E. Failure of Main Steam Line Radiation Monitors

If a main steam line radiation monitor (SA-11) fails and cannot be restored to operability in seven days, prepare a special report within thirty days of the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channel to operable status.

ADD PROPOSED
PTLR REPORTING
REQUIREMENT

B. Pressure/Temperature Limits

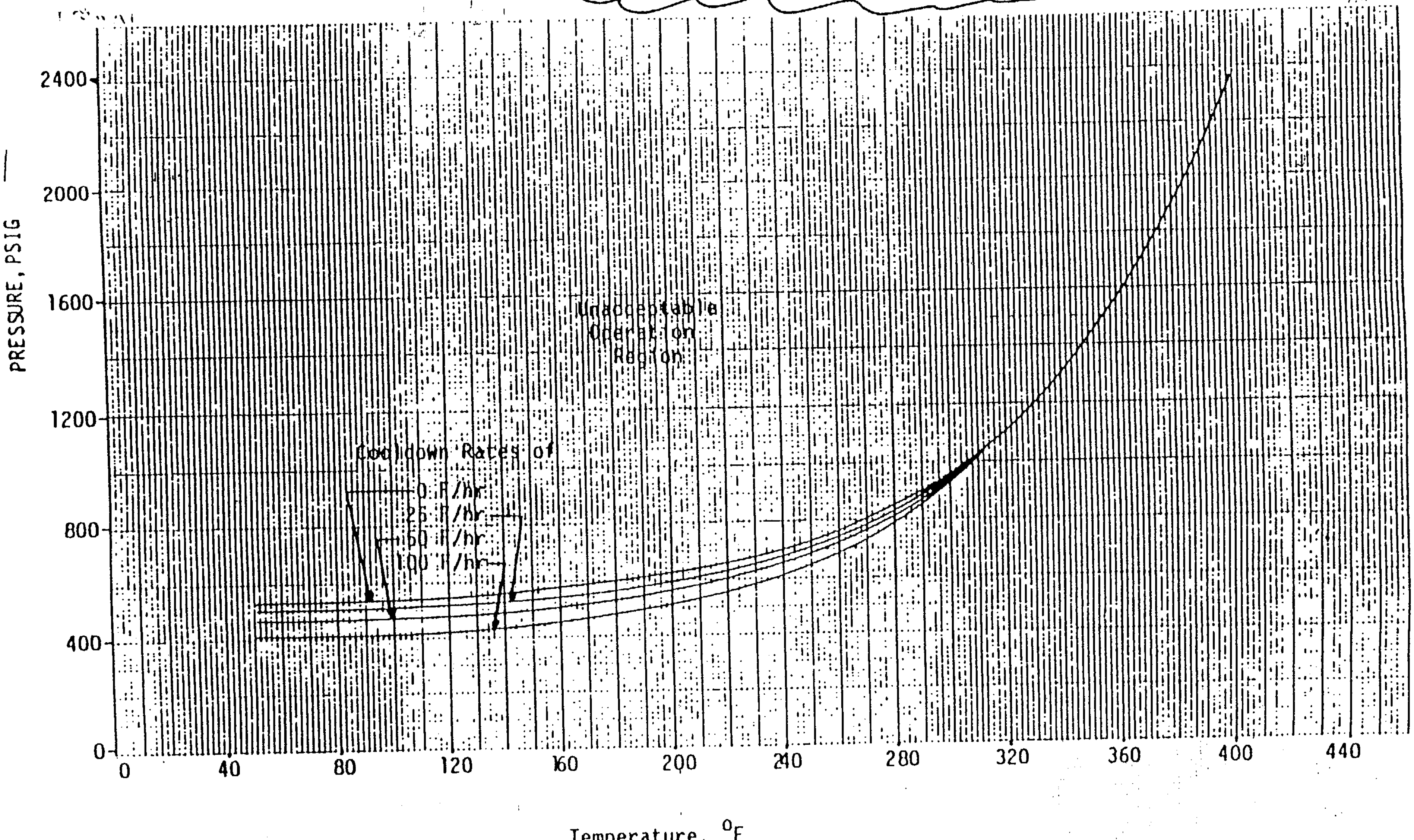
Specification:

1. The Reactor Coolant System temperature and pressure shall be limited in accordance with the limit lines shown in Figure 15.3.1-1 and 15.3.1-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:
   a. A maximum heatup of 100°F in any one hour,
   b. A maximum cooldown of 100°F in any one hour, and
   c. An average temperature change of ≤10°F per hour during inservice leak and hydrostatic testing operations.
2. The secondary side of the steam generator will not be pressurized above 200 psig if the temperature of the steam generator vessel shell is below 70°F.
3. The pressurizer temperature shall be limited to:
   a. A maximum heatup of 100°F in any one hour and a maximum cooldown of 200°F in any one hour, and
   b. A maximum spray water temperature differential between the pressurizer and spray fluid of not greater than 320°F.
4. The reactor vessel irradiation surveillance specimens are removed and examined, according to NRC approved schedules, to determine changes in material properties. The results of these examinations shall be considered in the evaluation of the prediction method to be used to update Figures 15.3.1-1 and 15.3.1-2. Revised figures shall be provided to the Commission at least sixty (60) days before the calculated exposure of the applicable reactor vessel exceeds the exposure for which the figures apply.

RELOCATE TO PTLR

Figure 15.3.1-2/PBNP Units 1 & 2
Cooldown Limitations Applicable to
23.6 Effective Full Power Years
(Approximately January 2001)

RELOCATE FIGURE
TO PTLR AND
CHANGE TO NEW LIMITS

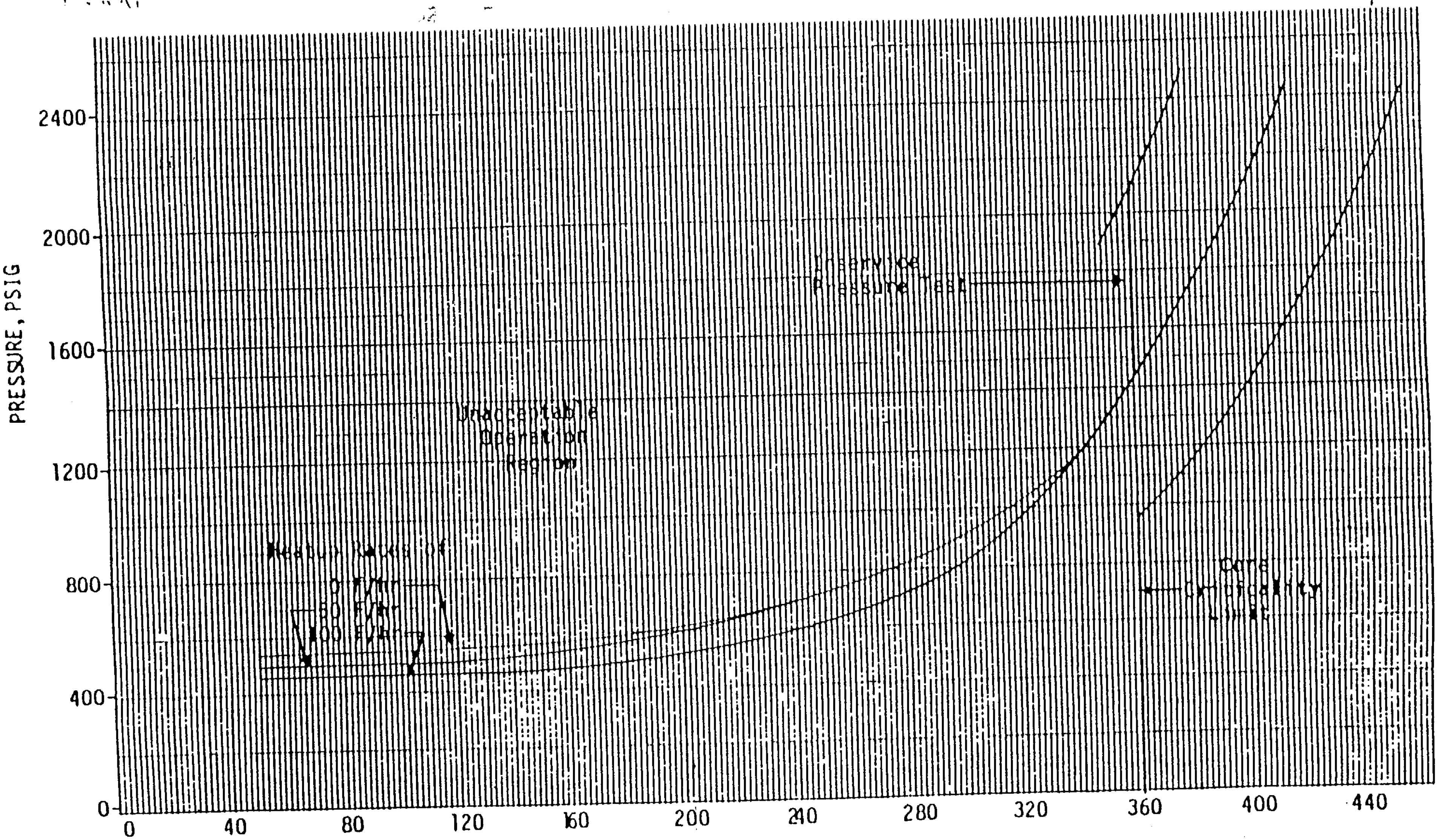


Unit 1 - Amendment No. 168
Unit 2 - Amendment No. 172

March 20, 1996

Figure 15.3.1-1/PBNP Units 1 & 2
Heatup Limitations Applicable to
23.6 Effective Full Power Years
(Approximately January 2001)

RELOCATE FIGURE
TO PTLR AND
CHANGE TO NEW LIMITS



Temperature, °F

(c) Residual Heat Removal Loop (A)*

(d) Residual Heat Removal Loop (B)*

(2) If the conditions of specification (1) above cannot be met, corrective action to return a second decay heat removal method to operable status as soon as possible shall be initiated immediately.

(3) If no decay heat removal method is in operation, except as permitted by (4) below, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(4) At least one of the above decay heat removal methods shall be in operation.

(a) All reactor coolant pumps and residual heat removal pumps may be deenergized for up to 1 hour in any 8 hour period provided:

(1) No operations are permitted that would cause dilution of reactor coolant system boron concentration, and

(2) Core outlet temperature is maintained at least 10°F below saturation temperature.

b. Reactor Coolant Temperature Less Than 140°F

(1) Both residual heat removal loops shall be operable except as permitted in items (3) or (4) below.

(2) If no residual heat removal loop is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(3) One residual heat removal loop may be out of service when the reactor vessel head is removed and the refueling cavity flooded.

(4) One of the two residual heat removal loops may be temporarily out of service to meet surveillance requirements.

4. Pressurizer Safety Valves

~~a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.~~

~~b.~~ Both pressurizer safety valves shall be operable whenever the reactor ~~is critical.~~ coolant temperature is ≥ LTOP enable temperature in the PTLR.

*Mechanical design provisions of the residual heat removal system afford the necessary flexibility to allow an operable residual heat removal loop to consist of the RHR pump from one loop coupled with the RHR heat exchanger from the other loop. Electrical design provisions of the residual heat removal system afford the necessary flexibility to allow the normal or emergency power source to be inoperable or tied together when the reactor coolant temperature is less than 200°F.

15.3.15 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

Applicability

Applies to operability of the low temperature overpressure protection (LTOP) system when the reactor coolant system temperature is < ~~355°F.~~ LTOP enable temperature in the PTLR.

Objective

To specify functional requirements and limiting conditions for operation on the use of the pressurizer power operated relief valves when used as part of the LTOP system and to specify further limiting conditions for operation when the reactor coolant system is operated at low temperatures.

Specification

A. System Operability

1. Except as specified in 15.3.15.A.2 below, the LTOP system shall be operable whenever the reactor coolant system is not open to the atmosphere and the temperature is < ~~355°F~~. Operability requirements are:

   a. Both pressurizer power operated relief valves operable at a setpoint ~~of ≤440 psig.~~ within the limits of the PTLR.

   b. Both power operated relief valve block valves are open.

2. The requirements of 15.3.15.A.1 may be modified as specified below:

   a. With one PORV inoperable while reactor coolant system temperature is >200°F but < ~~355°F~~, either restore the inoperable PORV to operable status within 7 days, or depressurize and vent reactor coolant system within the next 8 hours.

   b. With one PORV inoperable while reactor coolant system temperature is ≤200°F, either restore the inoperable PORV to operable status within 24 hours, or depressurize and vent the reactor coolant system within a total of 32 hours.

LTOP enable temperature in the PTLR

c. With both power operated relief valves inoperable while the reactor coolant system temperature is < ~~355°F~~, the reactor coolant system must be depressurized and vented within 8 hours.

3. If the reactor coolant system is vented per Specification 15.3.15.A.2.a, b, or c, the pathway must be verified at least once every 31 days when it is provided by a non-isolable pathway or by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the pathway every 12 hours.

B. Additional Limitations

1. When LTOP is required to be enabled by Specification 15.3.15.A.1: a. No ~~no~~ more than one high pressure safety injection pump shall be operable. The second high pressure safety injection pump shall be rendered inoperable whenever LTOP is required to be enabled by verifying that the motor circuit breakers have been removed from their electrical power supply circuits or by verifying that the discharge valves from the high pressure safety injection pumps to the reactor coolant system are shut and that power is removed from their operators.

b. Each accumulator whose pressure is greater than or equal to the maximum allowed RCS pressure for the existing RCS cold leg temperature as determined by the P-T limits curves, in the PTLR, shall be isolated.

2. A reactor coolant pump shall not be started when the reactor coolant system temperature is < ~~355°F~~ unless:

   a. There is a pressure absorbing volume in the pressurizer or in the steam generator tubes or

   b. The secondary water temperature of each steam generator is less than 50°F above the temperature of the reactor coolant system.

Basis

The Low Temperature Overpressure Protection System consists of a redundant means of relieving pressure during periods of water solid operation and when the reactor coolant system temperature is < ~~355°F~~. This method of water

ATTACHMENT 5

CALCULATION 2001-0001-00

RCS PRESSURE-TEMPERATURE LIMITS AND LTOP SETPOINTS
APPLICABLE THROUGH 32.2 EFPY – UNIT 1 AND 34.0 EFPY – UNIT 2

Nuclear Power Business Unit

EDMS CALCULATION INDEX UPDATE FORM

DATA PROVIDED BY ORIGINATOR		
EDMS Field	**Permitted Entry**	**Field Value**
Document ID	20 char max	2000-0001-00
Revision Number	3 digits max	0
Unit	0, 1, or 2	1 & 2
Approval Date	MMDDYYYY	3/6/2000
Title	255 char max	RCS Pressure-Temperature Limits and LTOP Setpoints Applicable Through 32.2 EFPY – Unit 1 and 34.0 EFPY – Unit 2
Originator	40 char max	J.R. Pfefferle
Status	"Active," "Superseded," or "Void"	Active
Discipline	E, IC, M, Civ, or Chem	M
Vendor	N/A for W.E. calcs	N/A
System(s)	CHAMPS identifier(s)	CHAMPS identifier(s): RC

ASSOCIATE DOCUMENTS (all fields must be completed; see back of form for directions)

Document ID	Revision	Action
WCAP-12794, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 1," Rev. 4, February 2000.	4	A
WCAP-12795, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 2," Rev. 3, August 1995.	3	A
NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.	2	A
ASME Boiler and Pressure Vessel Code, Sections III & XI, 1986 Edition.	-	A
ASME Code Case ISI 99-29, "Plant Specific Enable Temperature," 1999.	-	A
Point Beach Nuclear Plant Unit Nos. 1 and 2 Final Safety Analysis Report.	-	A
WEPCO Letter NPL 99-0569, "Monthly Operating Reports," October 7, 1999.	-	A
EPRI TR-107450, "P-T Calculator for Windows, Version 3.0," Revision 0, December, 1998.	0	A
Facility Operating License DPR-24 For Point Beach Nuclear Plant Unit No. 1, Wisconsin Electric Power Company, Docket No. 50-266, Page 5, Amendment No. 174, July 9, 1997.	-	A
Facility Operating License DPR-27 For Point Beach Nuclear Plant Unit No. 2, Wisconsin Electric Power Company, Docket No. 50-301, Page 5, Amendment No.178, July 9,1997.	-	A
ASTM E 29-93a, "Standard Practice for Using Significant Digits in Test Data to Determine Conformance with Specifications."	-	A
Instruction Manual 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.	-	A
Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.	-	A

ROUTE TO PBNP NUCLEAR INFORMATION MANAGEMENT

BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998.	-	A
WEPCO Calculation Addendum No. 98-0156-00-A, Revision 0, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," 9/22/1999.	0	A
CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997.	2	A
B&W Nuclear Technologies Calculation 32-1229659-00, "PB-1 & 2 Uncorrected P/T Limits at 32 & 34 EFPY," March 25, 1994.	-	A
WCAP-8738, "Heatup and Cooldown Limit Curves For Point Beach Nuclear Plant Unit No. 2," January, 1977.	-	A
ASME Steam Tables, Fifth Edition.	-	A
Westinghouse Calculation # SE/FSE-C-WEP-0159, "Point Beach Units 1 and 2 LTOP Analysis," November 27, 1996.	-	A
Vectra letter to Wisconsin Electric, "Low Temperature Overpressure Protection (LTOP) Preliminary Instrument Loop Uncertainty," March 5, 1996.	-	A
Vectra letter to Wisconsin Electric, "Wide Range RCS Hot and Cold Leg Temperature Instrument Uncertainty Calculation," May 29, 1996.	-	A
10 CFR 50, Appendix G, "Fracture Toughness Requirements," January 1, 1996 edition.	-	A
Westinghouse Report, "Pressure Mitigating Systems Transient Analysis Results," July 1977.	-	A
Westinghouse Report, "Supplement to the July 1977 Report, Pressure Mitigating Systems Transient Analysis Results," September, 1977.	-	A
WEPCO Calculation No. PB-89-036, Revision 0, "2RC-430/431C Pressure Assisted Operation Analysis," 10/10/1989.	0	A

ROUTE TO PBNP NUCLEAR INFORMATION MANAGEMENT

INSTRUCTIONS FOR COMPLETING PBF-1620 "ASSOCIATED DOCUMENTS"

1. All references cited by the calculation must be listed. Additional sheets may be attached, if necessary, and a reproduction of the reference section of the calculation may be used provided it contains the information required by PBF-1620.

2. Include sufficient identifying information for each document to permit NIM personnel to locate or create a record for the reference document (Document ID and Revision).

3. Action: Code as "A" (Add), or "D" (Delete). Revision 0 of a calculation or addendum will have only additions. However, a revision may require the deletion of obsolete references as well as additions of new or revised references. If multiple revisions of the same reference are applicable they shall be listed as separate line items and coded appropriately.

   "N/A" may only be used for one of two situations:

   a) When revising a calculation or addendum and the reference remains the same as cited in the previous revision, or

   b) The reference cited is not used by the calculation for data or information specific to PBNP. Typical examples are Codes, Industry Standards, textbooks, general use "handbooks," generic research papers, etc. In any case, if a future change in the reference could cause the results of the calculation to be affected, it shall not be coded "N/A."

ROUTE TO PBNP NUCLEAR INFORMATION MANAGEMENT

Nuclear Power Business Unit
CALCULATION DOCUMENT FORM

Calculation/Addendum Number:	**Title of Calculation/Addendum:**
2000-0001-00	RCS Pressure-Temperature Limits and LTOP Setpoints Applicable Through 32.2 EFPY – Unit 1 and 34.0 EFPY – Unit 2

☒ **Original Calculation/Addendum**
☐ **Revised Calculation/Addendum Revision #** ____

☒ **Supersedes Calculation/Addendum**
N-94-058, Rev. 2

☒ **QA Scope**
☐ **Non-QA Scope**

Associated Documents: See References p. 6 + 7.

Superseded By Calculation/Addendum # ____

This Calculation has been reviewed in accordance with NP 7.2.4. The review was accomplished by one or a combination of the following (as checked):

____ **A review of a representative sample of repetitive calculations.**

✓ **A detailed review of the original calculation.**

✓ **A review of the calculation against a similar calculation previously performed.**

✓ **A review by an alternate, simplified, or approximate method of calculation.**

Page Inventory:

Page 1 - 4 Form PBF-1608
Page 5 – 30 Calculations
Page 31 – 80 Attachments

Attachments:
Pages 31 through 80

Prepared By: James Pfefferle (Print Name)	Signature: [signature]	**Date:** 3-4-2000
Reviewed By: Thomas Spry (Print Name)	Signature: Thomas D. Spry	**Date:** 3-4-2000
Approved By: R. K. Hanneman (Print Name)	Signature: R.K. Hanneman	**Date:** 3/6/2000

Calculation Checklist (Optional for Non-QA Scope)

Item No.		Attribute Description	N/A	Author	Reviewer	
1.	**Purpose**					
	a.	Is the purpose clearly stated indicating issue to be resolved or information to be determined?		☒	☑ yes	☐ no
2.	**Methodology and Acceptance Criteria**					
	a.	Has the method/approach been described?		☒	☑ yes	☐ no
	b.	Have appropriate acceptance criteria and their sources been identified?		☒	☑ yes	☐ no
3.	**Assumptions**					
	a.	Are the assumptions provided with sufficient rationale to permit verification?		☒	☑ yes	☐ no
	b.	Have assumptions associated with pending plant or procedure changes that require verification been identified?	N/A	☐	☐ yes	☐ no
	c.	Have the requirements to revise governing calculations or verify pending assumptions been documented in a modification or an EWR?	N/A	☐	☐ yes	☐ no
5.	**References**					
	a.	Have all the appropriate references, including revisions and/or dates, been identified?		☒	☑ yes	☐ no
	b.	Are all references readily available in the PBNP Records System, as public documents, or attached?		☒	☑ yes	☐ no
4.	**Inputs**					
	a.	Have the applicable inputs and sources been identified?		☒	☑ yes	☐ no
	b.	Is the source for each input identified and listed in the References and/or Assumptions?		☒	☑ yes	☐ no
6.	**Calculation**					
	a.	Have formulae and inputs been provided consistent with the source document, including engineering units?		☒	☑ yes	☐ no
7.	**Computer-Aided Design Calculations (NP 7.2.4 Attachment A)**					
	a.	Has the computer program been validated per the requirements of Attachment A?		☒	☑ yes	☐ no
	b.	Have the program version and revision been identified on the computer run and in the calculation?		☒	☑ yes	☐ no

Item No.		Attribute Description	N/A	Author	Reviewer	
	c.	Is the input to the computer program adequately documented?		☒	☑ yes	☐ no
	d.	If spreadsheet or other simple computer aided tools are used in the calculation, have the formulae been documented in the calculation?	N/A	☐	☐ yes	☐ no
	e.	Have the attributes been documented in the calculation for any input or output data files supporting the calculation, including file name, date stamp, time stamp (hour and minute only), and file size?		☒	☑ yes	☐ no
8.	**Summary of Results and Conclusions**					
	a.	Do the summary of results and conclusions clearly state the calculation results and respond to the purpose?		☒	☑ yes	☐ no
	b.	Do the conclusions address the acceptability/ unacceptability of the results?		☒	☑ yes	☐ no
	c.	Has a CR been initiated to identify any unsatisfactory conditions?	N/A	☐	☐ yes	☐ no
	d.	Have all engineering judgments been provided with sufficient rationale?		☒	☑ yes	☐ no
9.	**Administrative**					
	a.	Have calculation format and content as noted in NP 7.2.4 been followed?		☒	☑ yes	☐ no
	b.	Have all required attachments been included in the document and numbered appropriately?		☒	☑ yes	☐ no
	c.	Has the calculation been prepared neatly and legibly with sufficient contrast to allow satisfactory record copies to be produced?		☒	☑ yes	☐ no
	d.	Are the calculation number, preparer's initials, preparation date, and page number provided on each page?		☒	☑ yes	☐ no
	e.	Have revisions been clearly identified by revision bars or other appropriate means (for revised calculations only)?	N/A	☐	☐ yes	☐ no
	f.	If the calculation is a revision, has a listing of successor documents, including impact and any required actions initiated, been incorporated as an attachment?	N/A	☐	☐ yes	☐ no

Item No.		Attribute Description	N/A	Author	Reviewer
	g.	If the calculation supersedes a previous calculation, is this noted on the cover sheet?		☒	☑ yes ☐ no
	h.	Has the calculation been appropriately identified as QA or Non-QA scope?		☒	☑ yes ☐ no
	i.	Has the review method been clearly identified on the cover page?			☑ yes ☐ no
	j.	Is all information required by PBF-1620 entered on the form?		☒	☑ yes ☐ no
	k.	If calculation creates a potential DBD open item, has a form PBF-1611 been prepared and in accordance with NP 7.7.3?		☒ No DBDOI created	☑ yes ☐ no

COMMENTS AND RESOLUTION

Reviewer Comments:	Resolution:
Comments resolved with preparer. Thomas D Spry 3/4/2000	

Purpose:

This calculation determines the pressure-temperature (P-T) limits applicable through the end of the Operating Licenses for Point Beach Units 1 and 2 when implementing ASME Code Section XI, Code Case ISI 99-29. This calculation also calculates an acceptable setpoint for the Low Temperature Overpressure Protection System applicable to both Point Beach units when implementing ASME Code Case ISI 99-29. All evaluations utilize the latest available best-estimate chemistry values and all applicable surveillance data for all vessel beltline materials.

This calculation is bounding for the two units, in that it determines the maximum pressure allowed for each unit through the end of the Operating License and utilizes the limiting allowable pressure of the two units to determine an acceptable LTOP setpoint. This calculation will consider the worst case combination of operating reactor coolant, charging, and residual heat removal pumps in conjunction with the design basis mass and heat input transients. The design basis mass input transient assumes that one safety injection pump is locked out of service during LTOP conditions.

Table of Contents:

References:

1. WCAP-12794, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 1," Rev. 4, February 2000.
2. WCAP-12795, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 2," Rev. 3, August 1995.
3. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
4. EPRI TR-107450, "P-T Calculator for Windows, Version 3.0," Revision 0, December,1998.
5. ASME Boiler and Pressure Vessel Code, Sections III & XI, 1986 Edition.
6. ASME Code Case ISI 99-29, "Plant Specific Enable Temperature," 1999.
7. Point Beach Nuclear Plant Unit Nos. 1 and 2 Final Safety Analysis Report.
8. WEPCO Letter NPL 99-0569, "Monthly Operating Reports," October 7, 1999.
9. Facility Operating License DPR-24 For Point Beach Nuclear Plant Unit No. 1, Wisconsin Electric Power Company, Docket No. 50-266, Page 5, Amendment No. 174, July 9, 1997.
10. Facility Operating License DPR-27 For Point Beach Nuclear Plant Unit No. 2, Wisconsin Electric Power Company, Docket No. 50-301, Page 5, Amendment No.178, July 9,1997.
11. ASTM E 29-93a, "Standard Practice for Using Significant Digits in Test Data to Determine Conformance with Specifications," 1993.
12. Instruction Manual 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
13. Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
14. BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998.
15. WEPCO Calculation Addendum No. 98-0156-00-A, Revision 0, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," 9/22/1999.
16. CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997.
17. B&W Nuclear Technologies Calculation 32-1229659-00, "PB-1 & 2 Uncorrected P/T Limits at 32 & 34 EFPY," March 25, 1994.
18. WCAP-8738, "Heatup and Cooldown Limit Curves For Point Beach Nuclear Plant Unit No. 2," January, 1977.
19. ASME Steam Tables, Fifth Edition.
20. Westinghouse Calculation # SE/FSE-C-WEP-0159, "Point Beach Units 1 and 2 LTOP Analysis," November 27, 1996.
21. Vectra letter to Wisconsin Electric, "Low Temperature Overpressure Protection (LTOP) Preliminary Instrument Loop Uncertainty," March 5, 1996.
22. Vectra letter to Wisconsin Electric, "Wide Range RCS Hot and Cold Leg Temperature Instrument Uncertainty Calculation," May 29, 1996.
23. 10 CFR 50, Appendix G, "Fracture Toughness Requirements," January 1, 1996 edition.
24. Westinghouse Report, "Pressure Mitigating Systems Transient Analysis Results," July 1977.

25. Westinghouse Report, "Supplement to the July 1977 Report, Pressure Mitigating Systems Transient Analysis Results," September, 1977.
26. WEPCO Calculation No. PB-89-036, Revision 0, "2RC-430/431C Pressure Assisted Operation Analysis," 10/10/1989.

Methods & Acceptance Criteria:

The methodology of this calculation follows the steps listed below:

Pressure-Temperature Limits

I. Determine projected EFPY for each unit by extrapolation of fluence data to end of Operating License.
II. Using References 1 and 2, determine the projected fluence for the limiting reactor vessel materials at the reactor vessel inside surface (clad-base metal interface) at the end of the plant Operating Licenses. See Tables 1 and 2 for a description of the methodology used.
III. Determine the corresponding reactor vessel fluence and fluence factors at the one-fourth thickness (1/4T) and three-fourths thickness (3/4T) location from the clad-base metal interface. See Tables 1 and 2 for a description of the methodology used.
IV. Determine the chemistry factor, initial properties, and margin term for the PBNP reactor vessel beltline materials using the latest available best-estimate chemistry values and all applicable surveillance data for all vessel beltline materials.
V. Determine the projected adjusted reference temperature (ART) at the 1/4T and 3/4T location for the reactor vessel beltline materials at EOL. See Tables 3 through 6 for a description of the methodology used.
VI. One heatup curve and one cooldown curve will be calculated to be applicable to both units at PBNP. This is done by determining the limiting weld material in both reactor vessels and using this material's properties (Cu and Ni content and fluence) as the basis for the RT_{NDT} calculations. The calculated RT_{NDT} adjusts the heatup and cooldown curves.
VII. Reg. Guide 1.99, Rev. 2 (Ref. 3) will be used to calculate adjusted reference temperatures (ARTs). The neutron attenuation function in Reg. Guide 1.99, Rev. 2 will be used to calculate ARTs at the 1/4T and 3/4T thickness locations in the vessel wall.
VIII. Instrument uncertainties will not be applied in the calculation of the heatup and cooldown curves. 10 CFR 50, Appendix G and ASME Code Section VIII, Appendix G do not require additional margins (instrument errors) be added to the curves.
IX. Input data will be entered into the P-T Calculator for Windows, Version 3.0 program (Ref. 4) to generate pressure-temperature curves.

Additional Methods and Acceptance Criteria applicable to the determination of LTOP setpoints are provided below.

Assumptions:

1. A capacity factor of 95% is assumed for future plant operation. This is based on higher capacity factors expected as a result of a transition from 12 to 18 month operating cycles and will be verified against future plant performance.
2. Changes in core design will not increase vessel inner-surface fluence beyond the values provided for a given EFPY at specific vessel altitude and azimuth locations in WCAPs

12794, Rev. 4 and 12795, Rev. 3. This will be monitored by evaluating ex-vessel dosimetry from the most limiting reactor vessel.

3. Reactor vessel thickness is based on the distance from the clad-base metal interface to the outside diameter.
4. The reactor vessel is assumed to be in an isothermal condition for evaluation of LTOP setpoints.
5. A surface defect with a depth of one-quarter of the vessel thickness and a length of one and one-half times the vessel thickness is postulated to exist on the reactor vessel inside surface (Ref. 5, Article G-2120).
6. No limits will be placed on the allowed number of reactor coolant, RHR, or charging pumps operating in the analysis of mass-input transient.
7. It is assumed that only one safety injection pump is capable of operation (i.e. one SI pump is locked out and incapable of injecting to the RCS) during LTOP conditions.
8. ASME Section XI Code Case ISI 99-29 (Ref. 6) will be utilized in the generation of pressure-temperature curves and LTOP setpoints.

Calculations:

I. EFPY Projection to End of Operating Licenses

A) Current EFPY

Inputs:

Maximum rated reactor thermal output: 1518.5 MWTh (Ref. 7)

The total thermal output for each unit as of October 1, 1999, is (Ref. 8):

Unit 1: Total thermal output = 289,065,806 MW hours - thermal
Unit 2: Total thermal output = 283,109,290 MW hours - thermal

Converting to EFPY:

Unit 1:

$$289{,}065{,}806 \text{ MWTh} \times \frac{\text{eff. full power}}{1518.5 \text{ MWTh}} \times \frac{1 \text{ day}}{24 \text{ hrs.}} \times \frac{1 \text{ year}}{365.25 \text{ days}} = 21.7 \text{ EFPY}$$

Unit 2:

$$283{,}109{,}290 \text{ MWTh} \times \frac{\text{eff. full power}}{1518.5 \text{ MWTh}} \times \frac{1 \text{ day}}{24 \text{ hrs.}} \times \frac{1 \text{ year}}{365.25 \text{ days}} = 21.3 \text{ EFPY}$$

B) EFPY from 10/1/1999 until End of License

Unit 1:

Calendar years from 10/1/1999 to 10/5/2010 (Unit 1 License expiration date, Ref. 9) is 11.01 years; a 95% capacity factor is assumed (Assumption 1):

11.01 years x 0.95 = 10.5 EFPY

Unit 2:
Calendar years from 10/1/1999 to 3/8/2013 (Unit 2 License expiration date, Ref. 10) is 13.39 years; a 95% capacity factor is assumed (Assumption 1):

13.39 years x 0.95 = 12.7 EFPY

C) **Total EFPY through End of License:**

Unit 1: 21.7 EFPY + 10.5 EFPY = 32.2 EFPY

Unit 2: 21.3 EFPY + 12.7 EFPY = 34.0 EFPY

II. Fluence and Adjusted Reference Temperature (ART) Evaluations (see Tables 1 through 6)

Inputs and calculational methods for fluence and adjusted reference temperature evaluations are specified in Tables 1 through 6.

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Table 1. Point Beach Unit 1 RPV Beltline 32.2 EFPY Fluence Values

Based on WCAP-12794, "Reactor Cavity Neutron Measurement Program For Wisconsin Electric Power Company Point Beach Unit 1," Rev. 4, February 2000. Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 32 EFPY, due to changes in core design at certain points in the operating history of the unit. As intermediate input to further calculations, these values are not rounded in accordance with ASTM E29 (Ref. 11).

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad *(D)*

Component Description	Heat or Heat/Lot	32 EFPY Inside Surface Fluence (E19 n/cm²)	32.2 EFPY Inside Surface Fluence (E19 n/cm²) *(A)*	32.2 EFPY 1/4T Fluence (E19 n/cm²) *(B)*	32.2 EFPY 1/4T Fluence Factor *(C)*	32.2 EFPY 3/4T Fluence (E19 n/cm²) *(B)*	32.2 EFPY 3/4T Fluence Factor *(C)*
Nozzle Belt Forging	122P237	0.547	0.550	0.3724	0.7269	0.1707	0.5322
Intermediate Shell Plate	A9811-1	2.64	2.65	1.794	1.160	0.8225	0.9452
Lower Shell Plate	C1423-1	2.24	2.25	1.523	1.116	0.6983	0.8993
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	0.547	0.550	0.3724	0.7269	0.1707	0.5322
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	1.74	1.75	1.185	1.047	N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	1.74	1.75	N/A	N/A	0.5431	0.8293
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	2.24	2.25	1.523	1.116	0.6983	0.8993
Lower Shell Long Seam (100%)	61782 (SA-847)	1.54	1.55	1.049	1.013	0.4811	0.7960

Footnotes:

(A) Interpolation of neutron exposure (in units of E19 n/cm², E>1 MeV) to a particular value of effective full power years (EFPY) is performed based on WCAP-12794, Revision 4. For example, for the nozzle belt forging, heat no. 122P237,

$$\text{fluence} = 0.547 + \left(\frac{0.796 - 0.547}{48\text{ EFPY} - 32\text{ EFPY}}\right) \times (32.2\text{ EFPY} - 32.0\text{ EFPY}) = 0.550\text{ E19 n/cm}^2$$

(B) From an inside surface fluence value (not including cladding), fluence is attenuated to a desired thickness using equation (3) of Regulatory Guide 1.99, Revision 2: $f = f_{surf} \times e^{-0.24x}$, where f_{surf} is expressed in units of E19 n/cm², E > 1MeV, and x is the desired depth in inches into the vessel wall. For example, for the nozzle belt forging, heat no. 122P237, at 32.2 EFPY, at a depth of ¼ of the 6.5" vessel wall (1.625"), $f = 0.550 \times e^{-0.24(1.625)} = 0.3724$ E19 n/cm².

(C) The dimensionless fluence factor is calculated using the fluence factor formula from equation (2) of Regulatory Guide 1.99, Revision 2: $ff = f^{(0.28 - 0.10 \log f)}$, where f is the fluence in units of E19 n/cm². For example, the 32.2 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237, $ff = 0.3724^{(0.28 - 0.10 \log 0.3724)} = 0.7269$.

(D) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969 (Ref. 12).

Table 2. Point Beach Unit 2 RPV Beltline 34.0 EFPY Fluence Values

Based on WCAP-12795, "Reactor Cavity Neutron Measurement Program For Wisconsin Electric Power Company Point Beach Unit 2," Rev. 3, August 1995. Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 32 EFPY, due to changes in core design at certain points in the operating history of the unit. As intermediate input to further calculations, these values are not rounded in accordance with ASTM E29 (Ref. 11).

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad *(D)*

Component Description	Heat or Heat/Lot	32 EFPY Inside Surface Fluence (E19 n/cm²)	34.0 EFPY Inside Surface Fluence (E19 n/cm²) *(A)*	34.0 EFPY 1/4T Fluence (E19 n/cm²) *(B)*	34.0 EFPY 1/4T Fluence Factor *(C)*	34.0 EFPY 3/4T Fluence (E19 n/cm²) *(B)*	34.0 EFPY 3/4T Fluence Factor *(C)*
Nozzle Belt Forging	123V352	0.548	0.5775	0.3910	0.7399	0.1792	0.5435
Intermediate Shell Forging	123V500	3.01	3.174	2.149	1.208	0.9851	0.9958
Lower Shell Forging	122W195	2.52	2.654	1.797	1.161	0.8237	0.9456
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	0.548	0.5775	0.3910	0.7399	0.1792	0.5435
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	2.49	2.606	1.764	1.156	0.8088	0.9405

Footnotes:

(A) Interpolation of neutron exposure (in units of E19 n/cm², E>1 MeV) to a particular value of effective full power years (EFPY) is performed based on WCAP-12795, Revision 3. For example, for the nozzle belt forging, heat no. 123V352,

$$\text{fluence} = 0.548 + \frac{(0.784 - 0.548)}{(48\text{ EFPY} - 32\text{ EFPY})} \times (34\text{ EFPY} - 32\text{ EFPY}) = 0.5775\text{E19 n/cm}^2$$

(B) From an inside surface fluence value (not including cladding), fluence is attenuated to a desired thickness using equation (3) of Regulatory Guide 1.99, Revision 2: $f = f_{surf} \times e^{-0.24x}$, where f_{surf} is expressed in units of E19 n/cm², E > 1MeV, and x is the desired depth in inches into the vessel wall. For example, for the nozzle belt forging, heat no. 123V352, at 34.0 EFPY, at a depth of ¼ of the 6.5" vessel wall (1.625"), $f = 0.5775 \times e^{-0.24(1.625)} = 0.3910$ E19 n/cm².

(C) The dimensionless fluence factor is calculated using the fluence factor formula from equation (2) of Regulatory Guide 1.99, Revision 2: $ff = f^{(0.28 - 0.10 \log f)}$, where f is the fluence in units of E19 n/cm². For example, the 34.0 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352, $ff = 0.3910^{(0.28 - 0.10 \log 0.3910)} = 0.7399$.

(D) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970 (Ref. 13).

Table 3. Point Beach Unit 1 RPV 1/4T Beltline Material Adjusted Reference Temperatures at 32.2 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 14), including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	1/4T 32.2 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	0.7269	55.97	0	17	34	140
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	1.160	102.08	26.9	17	63.64	167
"	"	"			79.3	Surv. Data *(B)*	"	91.99	"	8.5	56.42	149
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	1.116	61.71	26.9	17	63.64	126
"	"	"			35.8	Surv. Data *(B)*	"	39.95	"	8.5	56.42	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.7269	110.78	19.7	28	68.47	174
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	1.047	144.70	19.7	28	68.47	208
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	N/A	N/A	19.7	N/A	N/A	N/A
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table *(C)*	1.116	187.04	0	28	56	253 *(G)*
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	1.013	159.45	19.7	28	68.47	223
"	"	"			163.3	Surv. Data *(D)*	"	165.42	"	14	48.34	209 *(G)*

Footnotes:

(A) See Table 1.

(B) Credible Surveillance Data; see BAW-2325 for evaluation..

(C) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measuredΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).

(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," (Ref. 15) utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.

(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 122P237, ART = 50 +(77 x 0.7269) + 34 = 140°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.

(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.

(G) By inspection, these are the limiting material properties.

(H) Table 4. Point Beach Unit 2 RPV 1/4T Beltline Material Adjusted Reference Temperatures at 34.0 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 14), including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	1/4T 34.0 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	0.7399	56.23	0	17	34	130
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table *(B)*	1.208	70.06	0	17	34	144 *(G)*
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	1.161	35.99	0	17	34	110
"	"	"			42.8	Surv. Data *(C)*	"	49.69	"	8.5	17	107
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table *(H)*	0.7399	125.78	17	28	65.51	135
Intermed. To Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table *(D)*	1.156	208.08	19.7	28	68.47	272 *(G)*

Footnotes:

(A) See Table 2.
(B) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (34°F).
(C) Credible surveillance data; see BAW-2325 for evaluation.
(D) Non-credible surveillance data; Table CF value based on best-estimate chemistry is higher than best fit calculated using surveillance data, and therefore conservative.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 +(76 x 0.7399) + 34 = 130°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
(G) By inspection, these are the limiting material properties.
(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997 (Ref. 16).

Table 5. Point Beach Unit 1 RPV 3/4T Beltline Material Adjusted Reference Temperatures at 32.2 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998, including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	3/4T 32.2 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	0.5322	40.98	0	17	34	125
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	0.9452	83.18	26.9	17	63.64	148
"	"	"			79.3	Surv. Data *(B)*	"	74.95	"	8.5	56.42	132
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	0.8993	49.73	26.9	17	63.64	114
"	"	"			35.8	Surv. Data *(B)*	"	32.19	"	8.5	56.42	90
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.5322	81.11	19.7	28	68.47	145
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	N/A	N/A	19.7	N/A	N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	0.8293	130.70	19.7	28	68.47	194 *(G)*
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table *(C)*	0.8993	150.72	0	28	56	217 *(G)*
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	0.7960	125.29	19.7	28	68.47	189
"	"	"			163.3	Surv. Data *(D)*	"	129.99	"	14	48.34	173

Footnotes:

(A) See Table 1.
(B) Credible Surveillance Data; see BAW-2325 for evaluation.
(C) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).
(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 122P237, ART = 50 +(77 x 0.5322) + 34 = 125°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
(G) By inspection, these are the limiting material properties.

Table 6. Point Beach Unit 2 RPV 3/4T Beltline Material Adjusted Reference Temperatures at 34.0 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998, including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	3/4T 34.0 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	0.5435	41.31	0	17	34	115
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table *(B)*	0.9958	57.76	0	17	34	132 *(G)*
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	0.9456	29.31	0	17	34	103
"	"	"			42.8	Surv. Data *(C)*	"	40.47	"	8.5	17	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table *(H)*	0.5435	92.40	17	28	65.51	102
Intermed. To Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table *(D)*	0.9405	169.29	19.7	28	68.47	233 *(G)*

Footnotes:

(A) See Table 2.
(B) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (34°F).
(C) Credible surveillance data; see BAW-2325 for evaluation.
(D) Non-credible surveillance data; Table CF value based on best-estimate chemistry is higher than best fit calculated using surveillance data, and therefore conservative.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 +(76 x 0.5435)+ 34 = 115F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
(G) By inspection, these are the limiting material properties.
(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997.

Calculations (cont'd):

III. Pressure-Temperature Curves

Attachment 1, "P-T Calculator for Windows: Summary Report XYZ Reactor 60 F/hr Heatup up to 32 EFPY," documents the results of the test case run to verify proper execution of the P-T Calculator software program. Inputs and outputs used in the generation of the Point Beach pressure-temperature curves are provided in Attachment 2, "P-T Calculator for Windows: Summary Report Point Beach Heatup Limits – Unit 1 Axial Flaw," Attachment 3, "P-T Calculator for Windows: Summary Report Point Beach Heatup Limits – Unit 2 Circumferential Flaw," Attachment 4, "P-T Calculator for Windows: Summary Report Point Beach Cooldown Limits – Unit 1 Axial Flaw," and Attachment 5, "P-T Calculator for Windows: Summary Report Point Beach Cooldown Limits – Unit 2 Circumferential Flaw."

The data files used are as follows:

Input Case	File Name	Size	Modified
Test Case	Xyz.pwr	59KB	10/31/1998, 4:14 PM
Heatup Analysis	Pb_100hu.pwr	69KB	3/3/2000, 12:32 PM
Cooldown Analysis	Pb_100cd.pwr	69KB	3/3/2000, 12:36 PM

Most inputs to P-T Calculator for Windows (Ref. 4) were previously defined in this calculation. However, references for additional inputs are provided below:

Effective film heat transfer coefficient at clad/base metal interface:

h_{bm} = 690 Btu/hr-ft^2-°F = 4.792 Btu/hr-in^2-°F (Ref. 17)

Beltline Material Type:

Unit 1: SA-302, Grade B (Ref. 12)

Unit 2: SA-508, Class 2 (Ref. 18)

In addition, Figures 1 and 2 are provided as figures suitable for inclusion in the Pressure Temperature Limits Report based on the output of P-T Calculator for Windows. These figures include additional annotations and guidance to provide the most useful figures for the plant reactor operators.

IV. LTOP Setpoints

Methods & Acceptance Criteria for LTOP Setpoint Determination:

The methodology of this calculation follows the steps listed below:

I. From Tables 3 and 4 determine the limiting projected adjusted reference temperatures at the 1/4T location for the limiting reactor vessel materials at end of plant Operating License periods.

II. Determine the reference stress intensity factor corresponding to the metal temperature of interest and adjusted reference temperature of the limiting materials.
III. Determine the permissible stress intensity caused by membrane stress for an isothermal event at the temperature of interest.
IV. Determine the maximum allowable pressure corresponding to the reactor vessel permissible membrane stress intensity.
V. Correct for pressure instrument uncertainty and location bias in relation to the reactor vessel beltline.
VI. Correct for pressure overshoot due to PORV accumulation during the design basis mass input or heat input transient for the conditions of interest to determine the acceptable LTOP pressure setpoint. See page 20 for additional discussion of the methodology for analysis of the mass input transient and page 22 for additional discussion of the methodology for analysis of the heat input transient.
VII. Compare maximum attained pressure corresponding to given LTOP pressure setpoint with maximum allowable pressure corresponding to reactor pressure vessel permissible membrane stress intensity. If maximum attained is lower than maximum permissible, setpoint is acceptable.
VIII. Determine LTOP enable temperature using method of Code Case ISI 99-29 (Ref. 6).

Inputs:

Specific Volume of Compressed Water at 60°F, 600 psig: 1 lbm/0.0160 ft^3 (Ref. 19)

Pressure Instrument Location Bias:

Unit 1
- 41.3 psig w/one RCP in operation (Ref. 20)
- 70.3 psig w/two RCPs in operation

Unit 2
- 44.6 psig w/one RCP in operation (Ref. 20)
- 74.4 psig w/two RCPs in operation

The instrument location bias values associated with the elevation difference of the wide range pressure transmitter and the mid-plane of the reactor vessel are plant specific values and include the dynamic head effects of two RHR pumps operating (Ref 3).

Instrument Uncertainty:	Pressure	± 13 psi	(Ref. 21)
	Temperature	± 17.8 °F	(Ref. 22)

Reactor Vessel Thickness:	6.5 inches, without clad (Ref. 12 & 13)
One-Quarter Thickness:	1.625 inches
Reactor Vessel I.D.:	132.312 inches
Design Pressure:	2500 psig

By inspection of Tables 3 and 4 (above), it is determined that for Point Beach the limiting postulated axial 1/4T flaw is in Unit 1 weld heat 61782 with an adjusted reference temperature of 209°F, and the limiting postulated circumferential 1/4T flaw is in Unit 2 weld heat 72442 with an adjusted reference temperature of 272°F. The Point Beach LTOP setpoints can be conservative based on the limiting maximum allowable pressure of these two materials.

V. Determination of Setpoint for Mass Input Transient

A. Setpoint for Operation with Two RCPs and Three Charging Pumps

1. Calculation of Reference Critical Stress Intensity Factor (K_{IR}):

ASME Section XI Code Case ISI 99-29 allows the use of the lower bound of static fracture toughness (K_{Ic}) for developing the heatup and cooldown limits used in determination of the LTOP setpoint.

$$K_{Ic} = 33.2 + 20.734 \exp [0.02 (T_{min} - ART_{NDT})] \quad \text{(Ref. 6, Art. G-2110)}$$

By inspection of calculation inputs, the limiting material of either unit in the closure flange region that is highly stressed by bolt preload has a reference temperature of 60°F. In accordance with the requirements of 10 CFR 50, App. G (Ref. 23), the material temperature in this region must be greater than this reference temperature in order to pressurize the reactor vessel to ≤ 20% of its preservice hydrostatic test pressure with the reactor core not critical. Therefore, 60°F is the minimum temperature at which the RCS can be pressurized.

Substituting, the minimum allowable temperature (T_{min}) = 60 °F

Axial Flaw (Heat 61782)

$$K_{Ic} = 33.2 + 20.734 \exp [0.02 (60 - 209)] = 34.25 \text{ ksi-in}^{1/2}$$

Circumferential Flaw (Heat 72442)

$$K_{Ic} = 33.2 + 20.734 \exp [0.02 (60 - 272)] = 33.49 \text{ ksi-in}^{1/2}$$

To account for temperature instrument uncertainty during plant operation a correction is made to determine the minimum allowable indicated temperature.

Minimum allowable indicated temperature ($T_{min\ ind}$) = 60 °F + 17.8 °F =77.8 °F

2. Calculation of Maximum Allowable Pressure:

ASME Code Section XI Code Case ISI 99-29 permits the LTOP setpoint to be established such that the maximum pressure in the reactor vessel is limited to the pressure determined to satisfy ASME Section XI, Appendix G, Article G-2215.

Maximum Allowable Membrane Tension (K_{Im}):

$$2K_{Im} + K_{It} < K_{Ic}; \text{ where } K_{It} = 0 \text{ for isothermal conditions (Ref. 6, Article G-2215)}$$

Axial Flaw (Heat 61782)

$K_{Im} = K_{IC}/2 = 34.25/2 = 17.12$ ksi-in$^{1/2}$

Circumferential Flaw (Heat 72442)

$K_{Im} = K_{IC}/2 = 33.49/2 = 16.74$ ksi-in$^{1/2}$

Maximum Allowable Pressure:

$K_{Im} = M_m * P*R_i/t$ (Ref. 5, Article G-2214.1)

where: $P*R_i/t = P*D_i/(2*t)$

$M_m = 0.926\ t^{1/2}$ for inside axial surface flaws with: $2 \leq t^{1/2} \leq 3.464$

$M_m = 0.443\ t^{1/2}$ for inside circumferential surface flaws with: $2 \leq t^{1/2} \leq 3.464$

P = ASME XI App. G pressure limit, psig

D_i = inside diameter, inch

t = vessel thickness, inch

D = 132.312 inch, t = 6.5 inch (Ref. 12 & 13)

Axial Flaw (Heat 61782)

$$P = \frac{K_{Im} * (2*t)}{M_m * D} = \frac{17.12 \text{ ksi-in}^{1/2} * 2 * 6.5 \text{ inch}}{0.926 * (6.5 \text{ inch})^{1/2} * 132.312 \text{ inch}} = 712.5 \text{ psig}$$

Circumferential Flaw (Heat 72442)

$$P = \frac{K_{Im} * (2*t)}{M_m * D} = \frac{16.74 \text{ ksi-in}^{1/2} * 2 * 6.5 \text{ inch}}{0.443 * (6.5 \text{ inch})^{1/2} * 132.312 \text{ inch}} = 1456.3 \text{ psig}$$

By inspection, the postulated axial flaw is more limiting and will be used from this point forward in the evaluation of LTOP setpoints.

Maximum Allowable Indicated Pressure:

$P_{max\text{-}ind}$ = P - Location Bias - Instrument Uncertainty

= 712.5 psig – 70.3 psig - 13 psig = 629.2 psig

3. Determine Acceptable LTOP Setpoint:

The LTOP setpoint is determined for operation to a minimum reactor pressure vessel metal temperature of 60°F with two operating RHR pumps, up to two operating reactor coolant pumps, and up to three operating charging pumps (Assumptions 6 and 7). A setpoint of 500 psig is evaluated, which is the setpoint of the RHR high capacity relief valve, RH-861C (Ref. 7). Details of the evaluation of the setpoint are provided below for the mass input transient.

Methods

The mass input transient setpoint determination follows the methods described in Section 4 of Reference 24. A design basis mass input transient of one high pressure safety injection pump and three charging pumps discharging to the reactor coolant system while the system is solid with pressure relieved by one power operated relief valve is evaluated. The criteria for demonstrating that a 500 psig setpoint is acceptable is to determine the setpoint overshoot (ΔP) and add it to the setpoint. If this sum is less than the maximum allowable indicated pressure of 629.2 psig, the setpoint is considered to be acceptable.

The equation to use in the determination of setpoint overshoot for the mass input transient is as follows:

$$\Delta P\ (V, S, Z, X) = \Delta P_{REF}\ (X) * F_V * F_S * F_Z * \text{Exp. Ratio} \qquad \text{(Ref. 24)}$$

where: ΔP (V, S, Z, X) = setpoint overshoot, psig
- V = total RCS & RHR volume, ft^3
- S = relief valve setpoint, psig
- Z = relief valve opening time, sec.
- X = mass input rate, lb/sec
- $\Delta P_{REF}(x)$ = reference overshoot at mass input rate X, psi (Table 4.2.1, Ref. 24)
- F_V = RCS volume factor (Table 4.2.2, Ref. 24)
- F_S = relief valve setpoint factor (Table 4.2.4, Ref. 24)
- F_Z = relief valve opening time factor (Table 4.2.3, Ref. 24)
- Exp. Ratio = ratio of maximum overshoot with metal expansion considered versus without its consideration

The method described in Reference 24 was developed from a reference set of parameters which are as follows:

- X = mass input rate from the reference safety injection pump
- V = 6000 cubic foot primary system volume
- S = relief valve setpoint at 600 psig
- Z = reference 3 second opening valve

From the reference parameters and results of the various transient analyses, the factors F_Y, F_S, and F_Z were developed as described in Section 4.3 of Ref. 24. The report states that the development of these factors is conservative and plant specific analyses would result in peak values less than the peak values calculated using the algorithm outlined in the report.

Calculation

The Point Beach plant specific parameters are the same for both units and have the following values:

X = mass input rate for Point Beach is identical to the reference SI pump used in the analyses (Curve C of Figure 2.3.2 of Ref. 24).
V = 7200 cubic feet for total RCS and RHR volume (Ref. 7)
S = 500 psig for relief valve setpoint
Z = 2 seconds for relief valve open time (Ref. 26)

For an allowable pressure of greater than or equal to 630 psig the safety injection pump flow will be no more than 820 gpm (Curve C of Figure 2.3.2 of Ref. 24). To this the flow from three charging pumps at full flow (181.5 gpm) is added (Ref. 7, Table 9.3-3).

$X = 1001.5 \text{ gpm} * (1 \text{ minute}/60 \text{ seconds}) * (1 \text{ lbm}/0.0160 \text{ ft}^3) * (1 \text{ ft}^3/7.4805 \text{ gallon})$
$= 139.5 \text{ lb/sec}$

From Figure 4.2.1 of Ref. 24 the ΔP_{REF} for a 139.5 lb/sec flowrate is:

$\Delta P_{REF} = 191$ psig

From Figure 4.2.2 of Ref. 24 the F_V for a 7200 cubic foot RCS volume is:

$F_V = 0.91$ at 7200 ft^3

From Figure 4.2.4 of Ref. 24 the F_S for a 500 psig setpoint is:

$F_S = 1.14$

From Figure 4.2.3 of Ref. 24 the F_Z for a 2 second valve is:

$F_Z = 0.733$ at 2 seconds

The effect of metal expansion is evaluated using the method of Section 5.2 of Ref. 24. The effect on overshoot is related to the ratio of the value in peak pressure when metal expansion is assumed in the analysis to the value without metal expansion. Using the maximum values from Figure 5.2, the ratio is:

$$\text{Exp. Ratio} = \frac{\text{Maximum overshoot with metal expansion}}{\text{Maximum overshoot without metal expansion}} = \frac{115}{155} = 0.74$$

The resulting overshoot is:

$\Delta P = 191 * 0.91 * 0.733 * 1.14 * 0.74 = 107.5$ psi

Adding this to the setpoint results in:

$P_{MAX} = 500 + 109.5 = 607.5$ psig

This value is less than the maximum indicated allowable pressure of 629.2 psig for the mass input transient and is acceptable. Therefore, the setpoint of 500 psig is acceptable for operation with three charging and two reactor coolant pumps at reactor vessel metal temperatures greater than 60°F (RCS indicated water temperature of 78°F). This setpoint considers the effect of two operating RHR pumps (Ref. 20).

VI. Determination of Setpoint for Heat Input Transient

Methods

The design basis heat input transient assumes the starting of the first reactor coolant pump during water solid conditions with a temperature difference between the reactor coolant system and the steam generator of 50°F. Pressure is relieved by a single power operated relief valve. The information provided in the Supplement to the July 1977 Report (Ref. 25, "the supplement") is used to determine the setpoint overshoot for the heat input transient.

The following parameters are applicable to Point Beach (Ref. 7):

Steam generator heat transfer area	= 44,000 ft^2 - Unit 1*
	= 47,500 ft^2 - Unit 2*
RCS volume	= 6,259 ft^3
RCS/SG ΔT	= 50 °F
Initial RCS pressure	= 300 psig
Relief valve setpoint	= 500 psig
Relief valve opening time	= 2 seconds

* Although the Unit 2 replacement steam generators have a larger heat transfer area than the Unit 1 steam generators, because of material differences, the heat transfer capabilities of each steam generator design is equivalent. As a conservatism in this analysis, it is assumed that the heat transfer capability for each unit is proportional to the heat transfer area of the larger Point Beach Unit 2 steam generators.

A bounding assessment based on the overshoot with a 6000 ft^3 RCS, 500 psig setpoint, and 3 second relief valve opening time for the Point Beach LTOP setpoint will be made, after making a correction for steam generator heat transfer area. This assessment is bounding because:

1. A smaller system volume results in a larger overshoot pressure; and
2. A longer relief valve opening time results in a larger pressure accumulation.

Therefore, the actual pressure overshoot will be smaller than that estimated in this bounding assessment.

As a conservatism, the evaluation of the heat input transient includes a correction for the limiting Unit 2 steady state pressure bias due to two RCPs and two RHR pumps operating. This is done by reducing the maximum allowable pressure at the pressure instrument by the pressure bias due to two RCPs operating. This pressure bias correction is conservative because the maximum location pressure bias is not achieved until two reactor coolant pumps reach steady state flow conditions, whereas the limiting energy input transient occurs following the start of the first RCP. The allowable pressure for a postulated axial flaw in the Unit 1 limiting material is used to conservatively bound both units.

Calculations of the bounding cases of pressure overshoot for initial RCS temperatures of 100°F, 140°F, 180°F, and 250°F with bias correction for two reactor coolant pumps operating are provided below. These four temperatures represent all the temperatures analyzed in the supplement (Ref. 25).

A. Pressure Overshoot for Heat Input Transient at 100°F

1. Calculation of Reference Critical Stress Intensity Factor (K_{IR}):

Minimum temperature (T_{min}) = 100°F - 17.8°F = 82.2°F

K_{Ic} = 33.2 + 20.734 exp [0.02 (T_{min} - ART_{NDT})] (Ref. 5, Art. G-2110)
= 33.2 + 20.734 exp [0.02 (82.2 - 209)] = 34.84 ksi-in$^{1/2}$

2. Calculation of Maximum Allowable Pressure:

Maximum Allowable Membrane Tension (K_{Im}):

$2K_{Im} < K_{IC}$

$K_{Im} = K_{IC}/2 = 34.84/2 = 17.42$ ksi-in$^{1/2}$

Maximum Allowable Pressure:

$K_{Im} = M_m * P*R_i/t$ (Ref. 5, Article G-2214.1)

where: $P*R_i/t = P*D_i/(2*t)$
$M_m = 0.926\ t^{1/2}$ for inside axial surface flaws with: $2 \leq t^{1/2} \leq 3.464$
P = ASME XI App. G pressure limit, psig
D_i = inside diameter, inch
t = vessel thickness, inch

D = 132.312 inch, t = 6.5 inch (Ref. 12 & 13)

Axial Flaw (Heat 61782)

$$P = \frac{K_{Im} * (2*t)}{M_m * D} = \frac{17.42 \text{ ksi-in}^{1/2} * 2 * 6.5 \text{ inch}}{0.926 * (6.5 \text{ inch})^{1/2} * 132.312 \text{ inch}} = 725.0 \text{ psig}$$

Maximum Allowable Indicated Pressure:

$P_{max\text{-}ind} = P_{max}$ - Location Bias - Instrument Uncertainty

= 725.0 psig – 70.3 psig - 13 psig = 641.7 psig

3. Calculation of Overshoot Pressure

From Figure 16 of the supplement (Ref. 25), the Reference UA for an RCS volume of 6000 at 100°F is read as 0.083. This reference value is normalized to Point Beach by applying the ratio of steam generator heat transfer areas:

$$\text{Normalized UA @ 6000 ft}^3 = 0.083 * 47{,}500/58{,}000 = 0.068$$

Entering Figure 16 with UA = 0.068 we find:

$$\Delta P_{6K} = P_{MAX} - P_{SETPOINT} = 24 \text{ psi.}$$

The maximum pressure that can be reached with this bounding overshoot value is:

$$P_{MAX} = P_{SETPOINT} + \Delta P_{6K} = 500 \text{ psig} + 24 \text{ psi} = 524 \text{ psig.}$$

This value is less than the maximum indicated allowable pressure of 641.7 psig at an indicated RCS cold leg temperature of 100°F. Therefore, a setpoint of 500 psig is acceptable for the heat input transient at 100°F.

B. Pressure Overshoot for Heat Input Transient at 140°F

1. Calculation of Reference Critical Stress Intensity Factor (K_{IR}):

$$\text{Minimum temperature } (T_{min}) = 140°F - 17.8°F = 122.2°F$$

$$K_{Ic} = 33.2 + 20.734 \exp[0.02\,(122.2 - 209)] = 36.85 \text{ ksi-in}^{1/2}$$

2. Calculation of Maximum Allowable Pressure:

Maximum Allowable Membrane Tension (K_{Im}):

$$K_{Im} = K_{IC}/2 = 36.85/2 = 18.43 \text{ ksi-in}^{1/2}$$

Maximum Allowable Pressure:

Axial Flaw (Heat 61782)

$$P = \frac{K_{Im} * (2*t)}{M_m * D} = \frac{18.43 \text{ ksi-in}^{1/2} * 2 * 6.5 \text{ inch}}{0.926 * (6.5 \text{ inch})^{1/2} * 132.312 \text{ inch}} = 767.0 \text{ psig}$$

Maximum Allowable Indicated Pressure:

$$P_{max\text{-}ind} = P_{max} - \text{Location Bias} - \text{Instrument Uncertainty}$$

$$= 767.0 \text{ psig} - 70.3 \text{ psig} - 13 \text{ psig} = 683.7 \text{ psig}$$

3. Calculation of Overshoot Pressure

From Figure 16 of the supplement, the Reference UA for an RCS volume of 6000 at 140°F is read as 0.097. This reference value is normalized to Point Beach by applying the ratio of steam generator heat transfer areas:

$$\text{Normalized UA @ 6000 ft}^3 = 0.097 * 47{,}500/58{,}000 = 0.079$$

Entering Figure 16 with UA = 0.079 we find:

$$\Delta P_{6K} = P_{MAX} - P_{SETPOINT} = 48 \text{ psi.}$$

The maximum pressure that can be reached with this bounding overshoot value is:

$$P_{MAX} = P_{SETPOINT} + \Delta P_{6K} = 500 \text{ psig} + 48 \text{ psi} = 548 \text{ psig.}$$

This value is less than the maximum indicated allowable pressure of 683.7 psig at an indicated RCS cold leg temperature of 140°F. Therefore, a setpoint of 500 psig is acceptable for the heat input transient at 140°F.

C. Pressure Overshoot for Heat Input Transient at 180°F

1. Calculation of Reference Critical Stress Intensity Factor (K_{IR}):

$$\text{Minimum temperature } (T_{min}) = 180°F - 17.8°F = 162.2°F$$

$$K_{Ic} = 33.2 + 20.734 \exp[0.02\,(162.2 - 209)] = 41.33 \text{ ksi-in}^{1/2}$$

2. Calculation of Maximum Allowable Pressure:

Maximum Allowable Membrane Tension (K_{Im}):

$$K_{Im} = K_{IC}/2 = 41.33/2 = 20.66 \text{ ksi-in}^{1/2}$$

Maximum Allowable Pressure:

Axial Flaw (Heat 61782)

$$P = \frac{K_{Im} * (2*t)}{M_m * D} = \frac{20.66 \text{ ksi-in}^{1/2} * 2 * 6.5 \text{ inch}}{0.926 * (6.5 \text{ inch})^{1/2} * 132.312 \text{ inch}} = 859.8 \text{ psig}$$

Maximum Allowable Indicated Pressure:

$$P_{max\text{-}ind} = P_{max} - \text{Location Bias} - \text{Instrument Uncertainty}$$

$$= 859.8 \text{ psig} - 70.3 \text{ psig} - 13 \text{ psig} = 776.5 \text{ psig}$$

3. Calculation of Overshoot Pressure

From Figure 16 of the supplement, the Reference UA for an RCS volume of 6000 at 180°F is read as 0.114. This reference value is normalized to Point Beach by applying the ratio of steam generator heat transfer areas:

Normalized UA @ 6000 ft^3 = 0.114 * 47,500/58,000 = 0.093

Entering Figure 16 with UA = 0.093 we find:

$\Delta P_{6K} = P_{MAX} - P_{SETPOINT} = 78$ psi.

The maximum pressure that can be reached with this bounding overshoot value is:

$P_{MAX} = P_{SETPOINT} + \Delta P_{6K} = 500$ psig + 78 psi = 578 psig.

This value is less than the maximum indicated allowable pressure of 776.5 psig at an indicated RCS cold leg temperature of 180°F. Therefore, a setpoint of 500 psig is acceptable for the heat input transient at 180°F.

D. Pressure Overshoot for Heat Input Transient at 250°F

1. Calculation of Reference Critical Stress Intensity Factor (K_{IR}):

Minimum temperature (T_{min}) = 250°F - 17.8°F = 232.2°F

K_{Ic} = 33.2 + 20.734 exp [0.02 (232.2 - 209)] = 66.18 ksi-in$^{1/2}$

2. Calculation of Maximum Allowable Pressure:

Maximum Allowable Membrane Tension (K_{Im}):

$K_{Im} = K_{IC}/2 = 66.18/2 = 33.09$ ksi-in$^{1/2}$

Maximum Allowable Pressure:

Axial Flaw (Heat 61782)

$$P = \frac{K_{Im} * (2*t)}{M_m * D} = \frac{33.09 \text{ ksi-in}^{1/2} * 2 * 6.5 \text{ inch}}{0.926 * (6.5 \text{ inch})^{1/2} * 132.312 \text{ inch}} = 1377.1 \text{ psig}$$

Maximum Allowable Indicated Pressure:

$P_{max\text{-}ind} = P_{max}$ - Location Bias - Instrument Uncertainty

= 1377.1 psig – 70.3 psig - 13 psig = 1293.8 psig

3. Calculation of Overshoot Pressure

From Figure 16 of the supplement, the Reference UA for an RCS volume of 6000 at 250°F is read as 0.138. This reference value is normalized to Point Beach by applying the ratio of steam generator heat transfer areas:

Normalized UA @ 6000 ft^3 = 0.138 * 47,500/58,000 = 0.113

Entering Figure 16 with UA = 0.113 we find:

$\Delta P_{6K} = P_{MAX} - P_{SETPOINT}$ = 128 psi.

The maximum pressure that can be reached with this bounding overshoot value is:

$P_{MAX} = P_{SETPOINT} + \Delta P_{6K}$ = 500 psig + 128 psi = 628 psig.

This value is less than the maximum indicated allowable pressure of 1293.8 psig at an indicated RCS cold leg temperature of 250°F. Therefore, a setpoint of 500 psig is acceptable for the heat input transient at 250°F.

VII. Determination of LTOP Enable Temperature

The LTOP enable temperature will be determined based on the limiting axial and circumferential RT_{NDT} for the two units (i.e., a RT_{NDT} of 209°F for Unit 1 axial weld heat 61782 and of 272°F for Unit 2 circumferential weld heat 72442). In accordance with ASME Code Case ISI 99-29 (Ref. 6), the LTOP enable temperature may be determined as the greater of a coolant temperature of 200°F or a coolant temperature corresponding to a reactor vessel metal temperature of at least $RT_{NDT} + \tau$ at the quarter thickness beltline location.

Where:

τ = + 40°F for generic inside axial surface flaw, and
τ = – 85°F for generic inside circumferential surface flaw

Or may be computed on a plant specific basis by:

$\tau = 50 \ln [((F \bullet M_m (pR_i/t)) - 33.2)/20.734]$

F =1.1, accumulation factor for safety relief valves
M_m = the value of M_m determined in accordance with ASME XI, G-2214.1
p = vessel design pressure (ksi)
R_i = vessel inner radius (in.)
t = vessel wall thickness (in.)

The 1/4T metal temperature lag is determined in Attachment 5 from the output of P-T Calculator for Windows. The temperature of the reactor coolant is compared to the temperature at the 1/4T position for a 100°F/hr heatup rate. From Attachment 5, the maximum 1/4T temperature lag is 20.1°F at a 100°F/hr heatup rate. The temperature instrument uncertainty is 17.8°F (Ref. 22).

By inspection of Tables 3 and 4 (above), it is determined that for Point Beach the limiting postulated axial 1/4T flaw is in Unit 1 weld heat 61782 with an adjusted reference temperature of 209°F, and the limiting postulated circumferential 1/4T flaw is in Unit 2 weld heat 72442 with an adjusted reference temperature of 272°F.

For the Point Beach Unit 1 axial flaw, τ is determined on a plant specific basis as:

$$\begin{aligned}\tau &= 50 \ln [((1.1 \bullet 0.926 \bullet (6.5)^{1/2} (2.5 \text{ ksia} \bullet 66.16 / 6.5)) - 33.2) / 20.734] \\ &= 23.1°F\end{aligned}$$

$$\begin{aligned}\text{Enable Temperature} &= 209°F + 23.1°F + 20.1°F + 17.8°F \\ &= 270.0°F\end{aligned}$$

For the Point Beach Unit 2 circumferential flaw, the enable temperature is determined using the generic definition ($\tau = -85°F$) as:

$$\begin{aligned}\text{Enable Temperature} &= 272°F - 85°F + 20.1°F + 17.8°F \\ &= 224.9°F\end{aligned}$$

Therefore, the LTOP enable temperature is defined for Point Beach Units 1 and 2 as 270°F, which is the greater of: 200°F; 270°F; and 224.9°F.

Results and Conclusions:

This calculation provides heatup and cooldown pressure-temperature limits (Figures 1 and 2) applicable through the expiration of the Point Beach Units 1 and 2 Operating Licenses, and demonstrates that a Technical Specification LTOP setpoint of 500 psig with LTOP enable temperature of 270°F will provide acceptable protection of the reactor vessel from mass injection and heat-injection overpressure events at low temperatures through the expiration of the Point Beach Units 1 and 2 Operating Licenses.

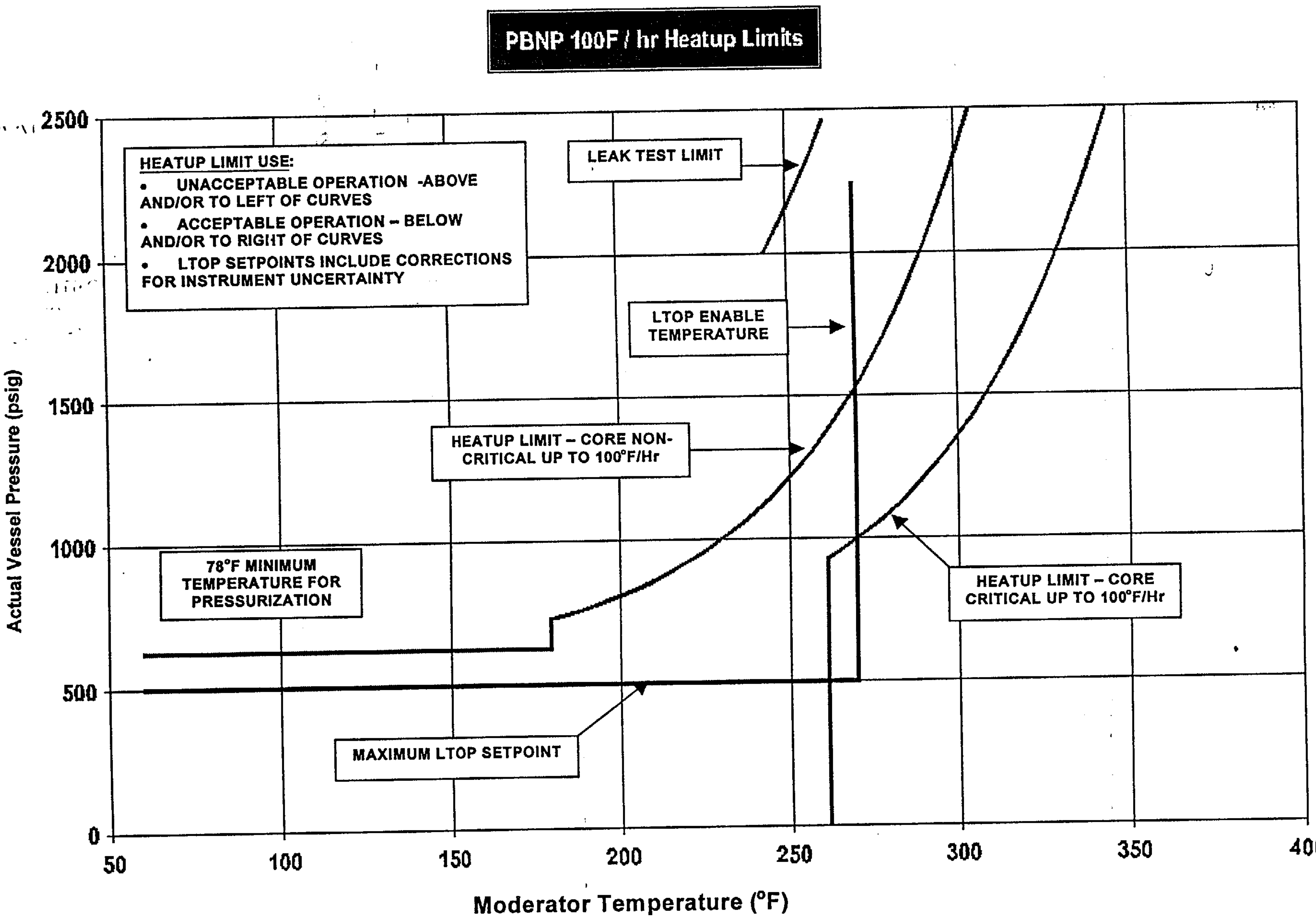

Figure 1: Point Beach Nuclear Plant Heatup Limits, Applicable Through 32.2 EFPY for Unit 1 and 34.0 EFPY for Unit 2

JRP

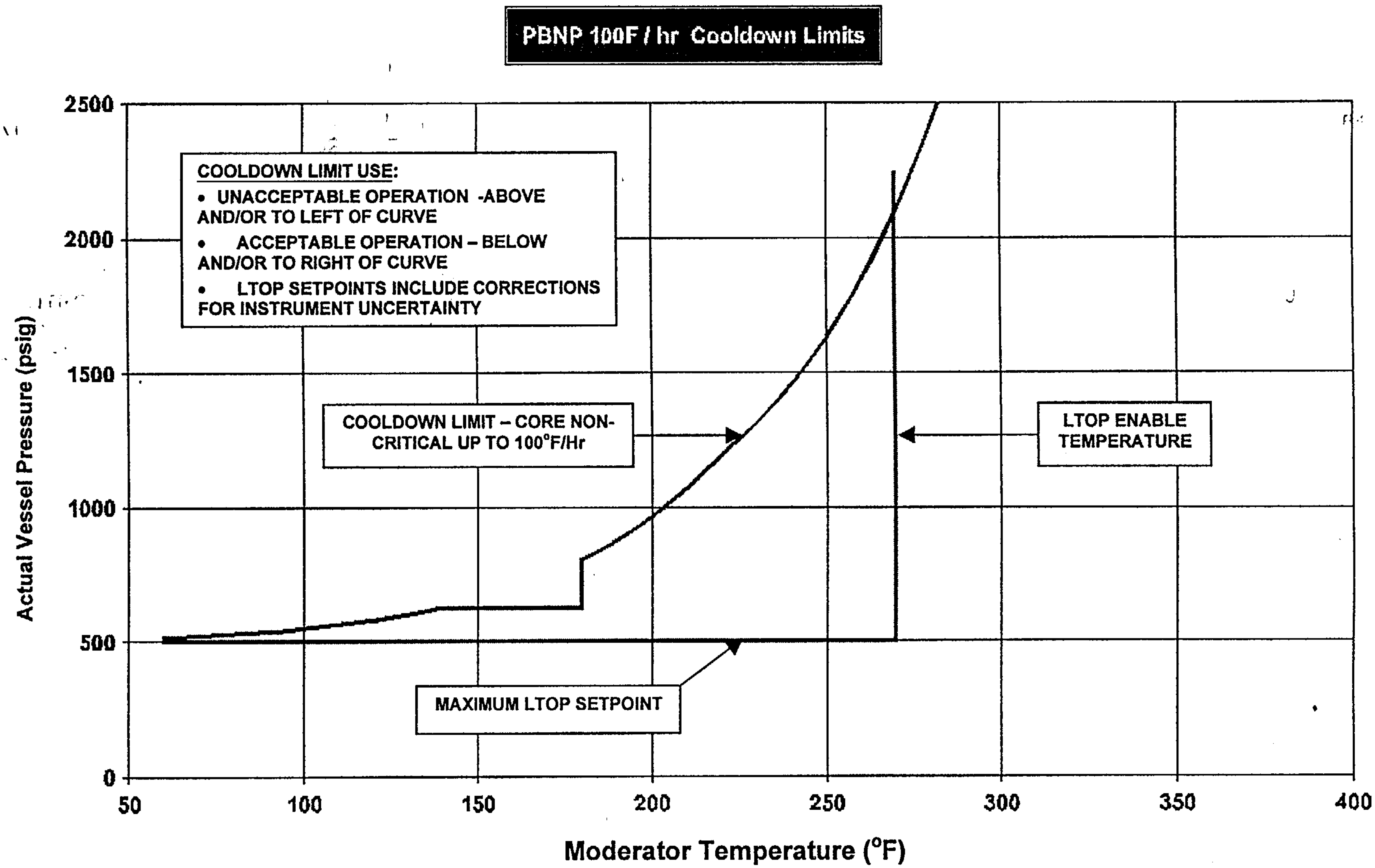

Figure 2: Point Beach Nuclear Plant Cooldown Limits, Applicable Through 32.2 EFPY for Unit 1 and 34.0 EFPY for Unit 2

Calc 2000-0001-00

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The Test Plan for P-T Calculator for Windows was performed using the XYZ Reactor 60 F/hr Heatup up to 32 EFPY dataset (file: xyz.pwr). This file was opened with P-T Calculator running in the PWR analysis module. Input data was verified as consistent with the screen print contained in Section 3 of the P-T Calculator User's Manual, the calculate function was performed using this data set, and a Summary Report of the results was printed. The input and output data on the Summary Report were compared to the PWR Example Problem results contained in Appendix A of the P-T Calculator User's Manual and verified to be consistent. Every 10th line of data of the Temperature Distribution, Fracture Toughness, and Thermal Stress Intensity Factor results were checked to confirm consistent calculation output. In addition, the 60 F/hr Heatup Curve for the XYZ Reactor (PWR) was judged to be generally consistent with the curve included in the User's Manual. Based on this review the Test Case results are deemed to be acceptable.

Verified by: James R. Riffle

Date: 3/4/2000

Calc 2000-0001-00

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P-T Calculator for Windows: Summary Report

XYZ Reactor 60 F/hr Heatup up to 32 EFPY

User Input Data:

```
The inner radius of vessel base metal:              78.5   in.
The outer radius of vessel base metal:             85.25   in.

The operating pressure:                             2235   psig
The pre-service test pressure:                      3110   psig
The in-service test pressure:                     2458.5   psig

The h value:                                      6.9444   Btu/in.^2-hr.-F

The pressure instrument error:                        60   psig
The temperature instrument error:                     10   F

The limiting material: A9153-1
The initial RT_NDT at closure flange:                100   F
The RT_NDT at 1/4 thickness up to 32 EFPY:           164   F
The RT_NDT at 3/4 thickness up to 32 EFPY:           149   F

The Analysis method:                     Raju-Newman method
The Flaw Orientation:                            Axial Flaw
The Fracture Criterion:                                K_IR

The 10 F/hr Loading is used to Generate ISLH P-T Curve

Normal Operation Safety Factor:                        2
ISLH Safety Factor:                                  1.5
Flange Region Thermal Lag:                             0 F

LTOP Set Point and Enable Temperature are Determined Per Appendix G
Set Point Scale Factor:                              110 %
LTOP Temperature Adjustment:                           0 F
LTOP Pressure Adjustment:                              0 psig
```

The coolant temperature variation

Time (hr.)	Temperature (DEG. F)
0	60
5	360

Run date 3/3/2000

J.R. [signature]

Calc 2000-0001-00
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3/4/2000

User Input Material Data:

The temperature independent material properties

The thermal conductivity:	1.95	Btu/in.-hr.-F
The thermal diffusivity:	60.48	in.^2/hr.
The Poisson's ratio:	0.3	

The temperature dependent material properties

Temperature (DEG. F)	Coeff. of Thermal Expansion (x 10^-6 in./in./F)
70	7.02
100	7.13
150	7.29
200	7.45
250	7.6
300	7.74
350	7.88
400	8.01
450	8.13
500	8.25
550	8.36
600	8.46
650	8.55
700	8.63
750	8.71
800	8.78

Temperature (DEG. F)	Young's Moduli (Msi)
70	29.2
200	28.5
300	28
400	27.4
500	27
600	26.4
700	25.3
800	23.9

Calc. 2000-0001-00
JL 3/4/2000 p 34-F8

Results: P-T Curve, Core Noncritical

Temperature (DEG.F)	Pressure (psig)
70	456.732
76	461.113
82	464.841
88	453.079
94	445.807
100	442.066
106	441.134
112	442.489
118	445.758
124	450.679
130	457.096
136	464.892
142	473.992
148	484.375
154	496.052
160	509.06
166	523.458
172	539.328
178	556.767
184	575.892
190	596.835
196	619.747
196.54	622
202	622
208	622
214	622
220	622
220	734.691
226	770.324
232	809.225
238	851.685
244	898.024
250	948.591
256	1003.77
262	1063.98
268	1129.68
274	1201.36
280	1279.58
286	1364.93
292	1458.05
298	1559.64
304	1670.48
310	1791.41
316	1923.33
322	2067.26
328	2224.28
334	2381.1
340	2549.85
346	2733.93
352	2934.74
358	3153.78
364	3392.73
370	3653.39

Calc. 2000 - 0001-00

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Results: P-T Curve, Core Critical

Temperature (DEG. F)	Pressure (psig)
304.407	0
304.407	1090.33
308	1129.68
314	1201.36
320	1279.58
326	1364.93
332	1458.05
338	1559.64
344	1670.48
350	1791.41
356	1923.33
362	2067.26
368	2224.28
374	2381.1
380	2549.85
386	2733.93
392	2934.74
398	3153.78
404	3392.73
410	3653.39

Calc. 2000-0001-00

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Results: P-T Curve for Inservice Hydrostatic Test

Temperature (DEG. F)	Pressure (psig)
284.967	2000
286	2020.53
292	2150.53
298	2292.35
304	2447.06
304.407	2458.5

Calc. 2000-0001-00

JEP 3/4/2000 p.37 of 80

Results: Temperature Distribution

Time (hr.)	T @ 1/8 T (DEG. F)	T @ 1/4 T (DEG. F)	T @ 3/8 T (DEG. F)	T @ 1/2 T (DEG. F)
0	60	60	60	60
0.1	63.4789	62.1974	61.3705	60.8016
0.2	68.1979	66.0245	64.4043	63.1661
0.3	73.2674	70.4387	68.225	66.4746
0.4	78.5741	75.2565	72.5994	70.4655
0.5	84.0562	80.3734	77.385	74.9646
0.6	89.6693	85.7137	82.4777	79.8433
0.7	95.3803	91.2208	87.7999	85.0055
0.8	101.164	96.8526	93.2935	90.3797
0.9	107.003	102.577	98.9152	95.9121
1	112.882	108.372	104.633	101.563
1.1	118.792	114.218	110.421	107.302
1.2	124.725	120.104	116.264	113.107
1.3	130.675	126.018	122.146	118.961
1.4	136.637	131.954	128.058	124.852
1.5	142.609	137.906	133.992	130.771
1.6	148.588	143.87	139.943	136.71
1.7	154.573	149.844	145.906	142.665
1.8	160.561	155.824	151.879	148.631
1.9	166.552	161.809	157.858	154.606
2	172.546	167.798	163.843	160.587
2.1	178.541	173.789	169.831	166.573
2.2	184.537	179.783	175.823	172.562
2.3	190.534	185.779	181.817	178.554
2.4	196.532	191.775	187.812	184.548
2.5	202.531	197.772	193.808	190.544
2.6	208.53	203.771	199.806	196.541
2.7	214.529	209.769	205.804	202.538
2.8	220.528	215.768	211.802	208.536
2.9	226.528	221.767	217.801	214.535
3	232.527	227.767	223.8	220.534
3.1	238.527	233.766	229.799	226.533
3.2	244.527	239.766	235.799	232.533
3.3	250.527	245.766	241.799	238.532
3.4	256.527	251.765	247.798	244.532
3.5	262.527	257.765	253.798	250.532
3.6	268.527	263.765	259.798	256.531
3.7	274.527	269.765	265.798	262.531
3.8	280.527	275.765	271.798	268.531
3.9	286.526	281.765	277.798	274.531
4	292.526	287.765	283.798	280.531
4.1	298.526	293.765	289.798	286.531
4.2	304.526	299.765	295.798	292.531
4.3	310.526	305.765	301.798	298.531
4.4	316.526	311.765	307.798	304.531
4.5	322.526	317.765	313.798	310.531
4.6	328.526	323.765	319.798	316.531
4.7	334.526	329.765	325.798	322.531
4.8	340.526	335.765	331.798	328.531
4.9	346.526	341.765	337.798	334.531
5	352.526	347.765	343.798	340.531

Calc. 2000-0001-00
JEL 3/4/2000
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Time (hr.)	T @ 5/8 T (DEG. F)	T @ 3/4 T (DEG. F)	T @ 7/8 T (DEG. F)	T @ 1 T (DEG. F)
0	60	60	60	60
0.1	60.4484	60.2676	60.1626	60.1186
0.2	62.2842	61.7237	61.3912	61.2716
0.3	65.1781	64.3131	63.7948	63.6129
0.4	68.8579	67.764	67.1059	66.877
0.5	73.1245	71.8595	71.0968	70.8329
0.6	77.8295	76.4366	75.5959	75.3057
0.7	82.862	81.3736	80.4745	80.1648
0.8	88.1392	86.5795	85.6368	85.3124
0.9	93.5992	91.9862	91.0109	90.6756
1	99.1958	97.5429	96.5433	96.1998
1.1	104.894	103.212	102.194	101.844
1.2	110.669	108.964	107.933	107.579
1.3	116.501	114.779	113.738	113.38
1.4	122.375	120.641	119.592	119.232
1.5	128.281	126.538	125.483	125.121
1.6	134.211	132.461	131.402	131.039
1.7	140.159	138.404	137.341	136.977
1.8	146.12	144.361	143.296	142.931
1.9	152.09	150.328	149.262	148.896
2	158.069	156.304	155.237	154.871
2.1	164.052	162.286	161.218	160.851
2.2	170.04	168.273	167.204	166.837
2.3	176.031	174.263	173.193	172.826
2.4	182.024	180.256	179.185	178.818
2.5	188.019	186.25	185.179	184.812
2.6	194.015	192.246	191.175	190.808
2.7	200.012	198.243	197.172	196.804
2.8	206.01	204.24	203.169	202.802
2.9	212.009	210.239	209.167	208.8
3	218.007	216.237	215.166	214.799
3.1	224.007	222.236	221.165	220.797
3.2	230.006	228.236	227.164	226.797
3.3	236.005	234.235	233.164	232.796
3.4	242.005	240.235	239.163	238.796
3.5	248.005	246.234	245.163	244.795
3.6	254.005	252.234	251.163	250.795
3.7	260.004	258.234	257.163	256.795
3.8	266.004	264.234	263.162	262.795
3.9	272.004	270.234	269.162	268.795
4	278.004	276.234	275.162	274.795
4.1	284.004	282.234	281.162	280.795
4.2	290.004	288.234	287.162	286.795
4.3	296.004	294.234	293.162	292.794
4.4	302.004	300.234	299.162	298.794
4.5	308.004	306.234	305.162	304.794
4.6	314.004	312.234	311.162	310.794
4.7	320.004	318.233	317.162	316.794
4.8	326.004	324.233	323.162	322.794
4.9	332.004	330.233	329.162	328.794
5	338.004	336.233	335.162	334.794

Calc. 2000-0001-00

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Results: Fracture Toughness and Thermal Stress Intensity Factors

Time (hr.)	K_IR @ 1/4 T (psi-sqrt in.)	K_IR @ 3/4 T (psi-sqrt in.)	K_TH @ 1/4 T (psi-sqrt in.)	K_TH @ 3/4 T (psi-sqrt in.)
0	29534.7	30204	0	0
0.1	29623.9	30217.3	-1278.8	843.125
0.2	29786.2	30290.6	-2538.48	1839.22
0.3	29984.9	30425	-3496.86	2611.18
0.4	30216.7	30612	-4217.44	3192.42
0.5	30481.4	30846.4	-4759.57	3629.68
0.6	30779.4	31125.5	-5168.29	3959.24
0.7	31111.9	31448	-5477.27	4208.27
0.8	31480.5	31814	-5711.64	4397.06
0.9	31887.3	32224.5	-5890.2	4540.79
1	32335	32681.3	-6025.68	4649.8
1.1	32826.5	33186.9	-6128.86	4732.76
1.2	33365.1	33744.3	-6208.56	4796.76
1.3	33954.7	34356.9	-6270.68	4846.58
1.4	34599.6	35029.1	-6319.63	4885.77
1.5	35304.5	35765.5	-6358.73	4917
1.6	36074.5	36571.3	-6390.45	4942.27
1.7	36915.5	37452.4	-6416.63	4963.08
1.8	37833.5	38415.2	-6438.65	4980.53
1.9	38835.7	39466.9	-6457.56	4995.47
2	39929.4	40615.3	-6474.13	5008.51
2.1	41122.9	41868.9	-6488.93	5020.14
2.2	42425.2	43237.3	-6502.41	5030.7
2.3	43846.1	44730.6	-6514.88	5040.45
2.4	45396.4	46360.1	-6526.6	5049.58
2.5	47087.9	48138.1	-6537.73	5058.25
2.6	48933.1	50078.1	-6548.43	5066.57
2.7	50946.3	52194.7	-6558.4	5074.31
2.8	53142.5	54503.8	-6567.97	5081.74
2.9	55538.4	57023.1	-6577.34	5089
3	58152.2	59771.4	-6586.54	5096.14
3.1	61003.6	62769.7	-6595.62	5103.17
3.2	64114.2	66040.7	-6604.59	5110.12
3.3	67507.7	69609	-6613.48	5117
3.4	71209.5	73501.7	-6622.3	5123.83
3.5	75247.9	77748.4	-6630.64	5130.29
3.6	79653.4	82381	-6638.29	5136.21
3.7	84459.4	87434.8	-6645.89	5142.09
3.8	89702.2	92948	-6653.45	5147.94
3.9	95421.6	98962.4	-6660.97	5153.76
4	101661	105523	-6668.45	5159.55
4.1	108467	112681	-6675.9	5165.31
4.2	115892	120489	-6683.31	5171.05
4.3	123992	129007	-6690.6	5176.69
4.4	132829	138299	-6696.51	5181.26
4.5	142468	148435	-6702.38	5185.8
4.6	152984	159494	-6708.21	5190.31
4.7	164455	171557	-6713.99	5194.79
4.8	176969	184716	-6719.75	5199.24
4.9	190621	199072	-6725.46	5203.66
5	205514	214733	-6731.14	5208.05

Calc. 2000-0001-00

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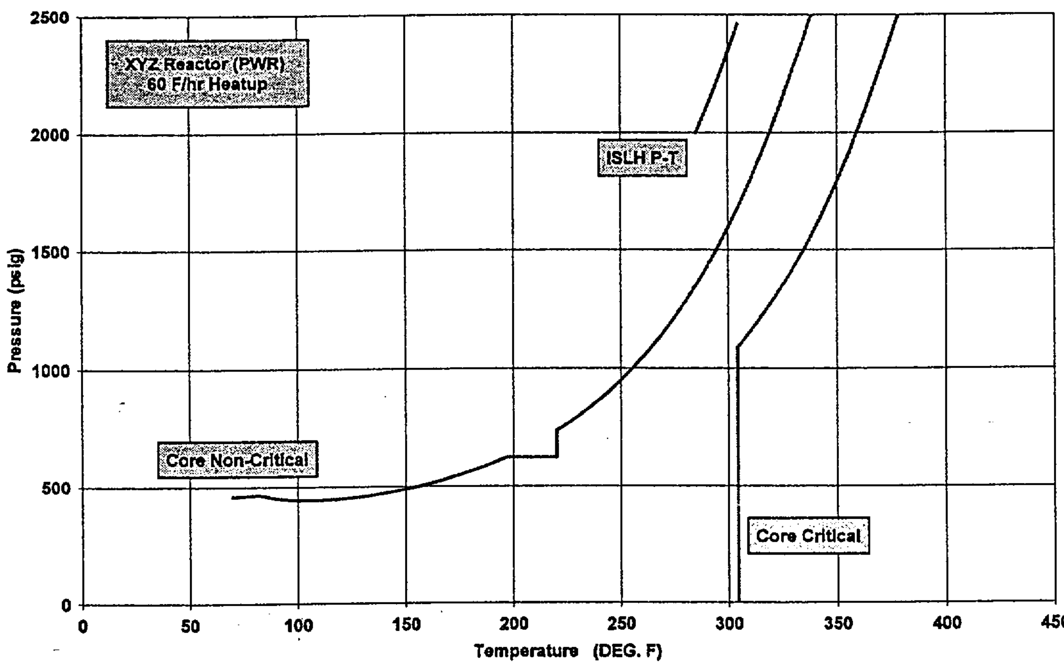

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JEL 3/4/2000

Calc 2000-0001-00
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P-T Calculator for Windows: Summary Report

Point Beach Heatup Limits - Unit 1 Axial Flaw

User Input Data:

The inner radius of vessel base metal:	66.156	in.
The outer radius of vessel base metal:	72.656	in.
The operating pressure:	2235	psig
The pre-service test pressure:	3110	psig
The in-service test pressure:	2458.5	psig
The h value:	5	Btu/in.^2-hr.-F
The pressure instrument error:	0	psig
The temperature instrument error:	0	F
The limiting material: 61782 (SA-847)		
The initial RT_NDT at closure flange:	60	F
The RT_NDT at 1/4 thickness up to 32 EFPY:	209	F
The RT_NDT at 3/4 thickness up to 32 EFPY:	194	F
The Analysis method:	Raju-Newman method	
The Flaw Orientation:	Axial Flaw	
The Fracture Criterion:	K_IC	
The 10 F/hr Loading is used to Generate ISLH P-T Curve		
Normal Operation Safety Factor:	2	
ISLH Safety Factor:	1.5	
Flange Region Thermal Lag:	0	F
LTOP Set Point and Enable Temperature are Input by the User		
Enable Temperature:	0	F
Set Point:	0	psig
LTOP Temperature Adjustment:	0	F
LTOP Pressure Adjustment:	0	psig

The coolant temperature variation

Time (hr.)	Temperature (DEG. F)
0	60
3.4	400

Run Date 3/3/2000
JFP

Calc 2000-0001-00
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User Input Material Data:

The temperature independent material properties

The thermal conductivity:	2.03333	Btu/in.-hr.-F
The thermal diffusivity:	62.928	in.^2/hr.
The Poisson's ratio:	0.3	

The temperature dependent material properties

Temperature (DEG. F)	Coeff. of Thermal Expansion (x 10^-6 in./in./F)
70	7.02
100	7.13
150	7.29
200	7.45
250	7.6
300	7.74
350	7.88
400	8.01
450	8.13
500	8.25
550	8.36
600	8.46
650	8.55
700	8.63
750	8.71
800	8.78

Temperature (DEG. F)	Young's Moduli (Msi)
70	29.2
200	28.5
300	28
400	27.4
500	27
600	26.4
700	25.3
800	23.9

Calc. 2000-0001-00

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Results: P-T Curve, Core Noncritical

Temperature (DEG.F)	Pressure (psig)
60	622
66.8	622
73.6	622
80.4	622
87.2	622
94	622
100.8	622
107.6	622
114.4	622
121.2	622
128	622
134.8	622
141.6	622
148.4	622
155.2	622
162	622
168.8	622
175.6	622
180	622
180	727.206
182.4	734.318
189.2	757.798
196	785.032
202.8	816.513
209.6	852.822
216.4	894.626
223.2	942.698
230	997.935
236.8	1061.36
243.6	1134.15
250.4	1217.65
257.2	1313.41
264	1423.21
270.8	1549.08
277.6	1693.36
284.4	1858.76
291.2	2048.32
298	2265.54
304.8	2514.47
311.6	2799.71
318.4	3126.55
325.2	3501.05
332	3930.17
338.8	4421.87
345.6	4985.23
352.4	5630.7
359.2	6370.24
366	7217.54
372.8	8188.31
379.6	9300.55
386.4	10574.9
393.2	12034.9
400	13707.6

Calc 2000-0001-00

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Results: P-T Curve, Core Critical

Temperature (DEG. F)	Pressure (psig)
261.182	0
261.182	928.436
263.2	942.698
270	997.935
276.8	1061.36
283.6	1134.15
290.4	1217.65
297.2	1313.41
304	1423.21
310.8	1549.08
317.6	1693.36
324.4	1858.76
331.2	2048.32
338	2265.54
344.8	2514.47
351.6	2799.71
358.4	3126.55
365.2	3501.05
372	3930.17
378.8	4421.87
385.6	4985.23
392.4	5630.7
399.2	6370.24
406	7217.54
412.8	8188.31
419.6	9300.55
426.4	10574.9
433.2	12034.9
440	13707.6

Calc 2000-0001-00

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Results: P-T Curve for Inservice Hydrostatic Test

Temperature (DEG. F)	Pressure (psig)
243.543	2000
243.6	2001.15
250.4	2158.08
257.2	2337.87
261.182	2458.5

Calc. 2000-0001-00

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Results: Temperature Distribution

Time (hr.)	T @ 1/8 T (DEG. F)	T @ 1/4 T (DEG. F)	T @ 3/8 T (DEG. F)	T @ 1/2 T (DEG. F)
0	60	60	60	60
0.068	63.3727	61.9627	61.1241	60.5845
0.136	68.3122	65.7821	63.9768	62.6505
0.204	73.6901	70.281	67.6879	65.689
0.272	79.3515	75.2355	72.0047	69.4571
0.34	85.2326	80.5456	76.7991	73.8073
0.408	91.2899	86.1414	81.978	78.627
0.476	97.4896	91.9679	87.4675	83.826
0.544	103.804	97.9809	93.208	89.3318
0.612	110.212	104.145	99.1515	95.0855
0.68	116.695	110.43	105.259	101.04
0.748	123.238	116.814	111.499	107.156
0.816	129.831	123.278	117.847	113.403
0.884	136.463	129.807	124.281	119.756
0.952	143.128	136.387	130.785	126.195
1.02	149.818	143.009	137.346	132.703
1.088	156.53	149.666	143.953	139.266
1.156	163.258	156.35	150.597	145.876
1.224	170	163.056	157.27	152.521
1.292	176.753	169.78	163.968	159.197
1.36	183.516	176.518	170.685	165.896
1.428	190.285	183.269	177.419	172.614
1.496	197.06	190.029	184.165	179.348
1.564	203.84	196.796	190.921	186.095
1.632	210.624	203.57	197.686	192.852
1.7	217.411	210.349	204.457	199.617
1.768	224.201	217.132	211.234	206.389
1.836	230.992	223.918	218.016	213.166
1.904	237.785	230.707	224.801	219.948
1.972	244.58	237.498	231.588	226.733
2.04	251.375	244.291	238.379	233.521
2.108	258.171	251.085	245.171	240.311
2.176	264.968	257.88	251.964	247.103
2.244	271.766	264.676	258.759	253.897
2.312	278.564	271.473	265.555	260.692
2.38	285.363	278.27	272.351	267.488
2.448	292.161	285.068	279.149	274.284
2.516	298.96	291.867	285.946	281.081
2.584	305.759	298.665	292.745	287.879
2.652	312.559	305.464	299.543	294.677
2.72	319.358	312.263	306.342	301.476
2.788	326.158	319.063	313.141	308.275
2.856	332.957	325.862	319.94	315.074
2.924	339.757	332.662	326.74	321.873
2.992	346.557	339.461	333.539	328.673
3.06	353.357	346.261	340.339	335.472
3.128	360.157	353.061	347.138	342.272
3.196	366.957	359.861	353.938	349.071
3.264	373.756	366.66	360.738	355.871
3.332	380.556	373.46	367.538	362.671
3.4	387.356	380.26	374.338	369.471

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Time (hr.)	T @ 5/8 T (DEG. F)	T @ 3/4 T (DEG. F)	T @ 7/8 T (DEG. F)	T @ 1 T (DEG. F)
0	60	60	60	60
0.068	60.2823	60.1538	60.0778	60.0405
0.136	61.7503	61.2077	60.8905	60.7743
0.204	64.2499	63.3152	62.7604	62.5649
0.272	67.5749	66.3152	65.5624	65.3005
0.34	71.5658	70.0423	69.1286	68.8128
0.408	76.0948	74.3579	73.3141	72.9547
0.476	81.0589	79.1495	78.0004	77.6057
0.544	86.3747	84.3258	83.0917	82.6684
0.612	91.9748	89.8131	88.5102	88.0639
0.68	97.8048	95.5519	94.1935	93.7285
0.748	103.821	101.494	100.091	99.6105
0.816	109.987	107.601	106.161	105.668
0.884	116.274	113.84	112.371	111.869
0.952	122.66	120.187	118.694	118.184
1.02	129.125	126.62	125.108	124.591
1.088	135.654	133.124	131.596	131.075
1.156	142.235	139.684	138.144	137.618
1.224	148.858	146.291	144.74	144.211
1.292	155.515	152.934	151.376	150.844
1.36	162.199	159.608	158.043	157.508
1.428	168.906	166.305	164.735	164.199
1.496	175.63	173.023	171.448	170.91
1.564	182.369	179.756	178.178	177.639
1.632	189.12	186.502	184.921	184.381
1.7	195.88	193.258	191.675	191.134
1.768	202.648	200.023	198.437	197.896
1.836	209.421	206.794	205.207	204.666
1.904	216.2	213.571	211.983	211.441
1.972	222.983	220.352	218.763	218.221
2.04	229.769	227.137	225.548	225.005
2.108	236.558	233.925	232.335	231.792
2.176	243.349	240.715	239.124	238.581
2.244	250.142	247.507	245.916	245.373
2.312	256.936	254.301	252.709	252.166
2.38	263.731	261.096	259.504	258.96
2.448	270.528	267.892	266.299	265.756
2.516	277.324	274.688	273.096	272.552
2.584	284.122	281.485	279.893	279.349
2.652	290.92	288.283	286.69	286.147
2.72	297.718	295.081	293.489	292.945
2.788	304.517	301.88	300.287	299.743
2.856	311.316	308.679	307.086	306.542
2.924	318.115	315.478	313.885	313.341
2.992	324.914	322.277	320.684	320.14
3.06	331.714	329.076	327.483	326.94
3.128	338.513	335.876	334.283	333.739
3.196	345.313	342.675	341.082	340.539
3.264	352.113	349.475	347.882	347.338
3.332	358.912	356.275	354.682	354.138
3.4	365.712	363.075	361.482	360.938

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Results: Fracture Toughness and Thermal Stress Intensity Factors

Time (hr.)	K_IC @ 1/4 T (psi-sqrt in.)	K_IC @ 3/4 T (psi-sqrt in.)	K_TH @ 1/4 T (psi-sqrt in.)	K_TH @ 3/4 T (psi-sqrt in.)
0	34253.1	34621.6	0	0
0.068	34295.3	34626	-1271.7	792.178
0.136	34382.2	34656.3	-2749.58	1925.49
0.204	34493.6	34719	-3990.41	2915.06
0.272	34628.3	34813	-5003.54	3728.03
0.34	34788.3	34937.8	-5827.92	4390.1
0.408	34976.4	35094.5	-6499.12	4929.12
0.476	35196	35285	-7046.4	5368.52
0.544	35451	35512.4	-7493.44	5727.31
0.612	35746.4	35780.6	-7859.36	6020.88
0.68	36087.5	36094.5	-8157.8	6260.26
0.748	36480.7	36459.8	-8401.46	6455.64
0.816	36933.5	36883.2	-8601.86	6616.24
0.884	37454.2	37372.7	-8767.23	6748.68
0.952	38052.6	37937.5	-8904.21	6858.29
1.02	38739.8	38588	-9018.2	6949.42
1.088	39528.7	39336.5	-9113.57	7025.58
1.156	40433.8	40196.9	-9193.83	7089.59
1.224	41472.1	41185.2	-9261.86	7143.77
1.292	42662.8	42319.9	-9319.95	7189.97
1.36	44028.1	43622.1	-9370	7229.69
1.428	45593.3	45116	-9413.51	7264.17
1.496	47387.4	46829.4	-9451.71	7294.37
1.564	49443.7	48794.1	-9485.61	7321.12
1.632	51800.3	51046.6	-9516	7345.05
1.7	54501	53628.6	-9543.25	7366.47
1.768	57595.9	56588.2	-9567.61	7385.58
1.836	61142.2	59980.1	-9590.08	7403.18
1.904	65205.7	63867.3	-9611.01	7419.54
1.972	69861.7	68321.9	-9630.68	7434.89
2.04	75196.5	73426.4	-9649.32	7449.41
2.108	81308.9	79275.4	-9667.11	7463.25
2.176	88312.2	85977.3	-9684.2	7476.54
2.244	96336.2	93656.4	-9699.05	7488.07
2.312	105530	102455	-9713.36	7499.18
2.38	116062	112536	-9727.27	7509.97
2.448	128130	124086	-9740.86	7520.49
2.516	141956	137319	-9754.16	7530.79
2.584	157797	152481	-9767.22	7540.9
2.652	175945	169852	-9780.08	7550.85
2.72	196737	189754	-9791.32	7559.54
2.788	220559	212556	-9801.46	7567.38
2.856	247852	238680	-9811.46	7575.11
2.924	279120	268610	-9821.32	7582.73
2.992	314944	302901	-9831.06	7590.26
3.06	355987	342188	-9840.69	7597.7
3.128	403010	387198	-9850.22	7605.06
3.196	456883	438766	-9859.21	7612.01
3.264	518604	497847	-9866.85	7617.91
3.332	589317	565534	-9874.41	7623.75
3.4	670332	643083	-9881.89	7629.53

Calc. 2000-0001-00

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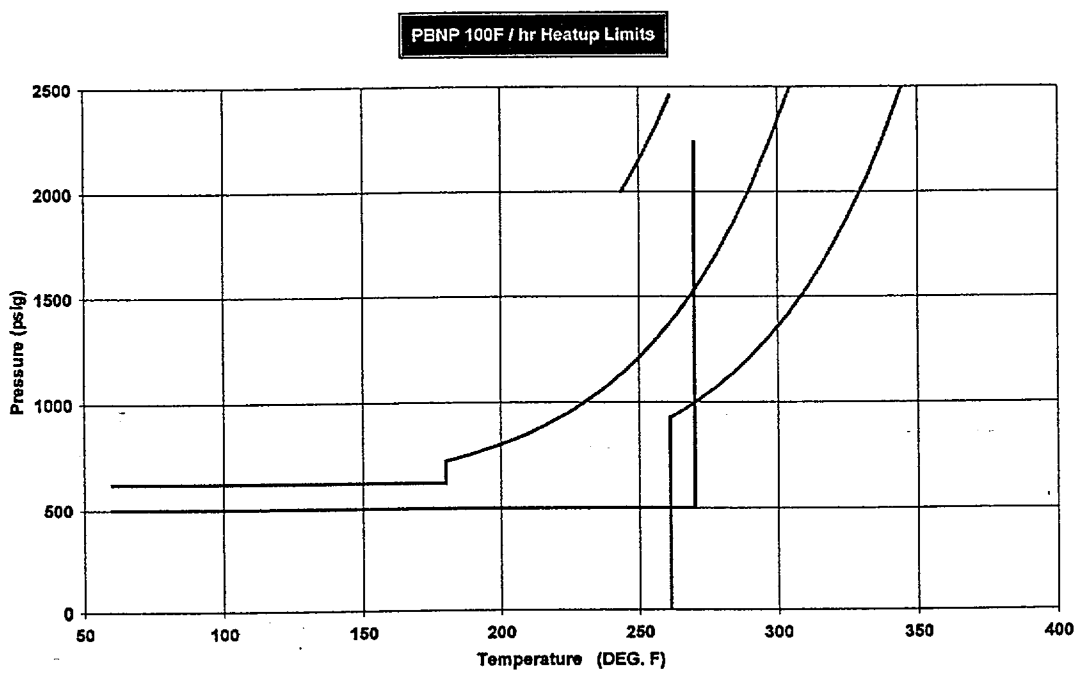

P-T Calculator for Windows: Summary Report

Point Beach Heatup Limits - Unit 2 Circumferential Flaw

User Input Data:

The inner radius of vessel base metal:	66.156	in.
The outer radius of vessel base metal:	72.656	in.
The operating pressure:	2235	psig
The pre-service test pressure:	3110	psig
The in-service test pressure:	2458.5	psig
The h value:	5	Btu/in.^2-hr.-F
The pressure instrument error:	0	psig
The temperature instrument error:	0	F
The limiting material: 72442 (SA-1484)		
The initial RT_NDT at closure flange:	60	F
The RT_NDT at 1/4 thickness up to 34 EFPY:	272	F
The RT_NDT at 3/4 thickness up to 34 EFPY:	233	F
The Analysis method:	Raju-Newman method	
The Flaw Orientation:	Circumferential Flaw	
The Fracture Criterion:	K_IC	
The 10 F/hr Loading is used to Generate ISLH P-T Curve		
Normal Operation Safety Factor:	2	
ISLH Safety Factor:	1.5	
Flange Region Thermal Lag:	0	F
LTOP Set Point and Enable Temperature are Input by the User		
Enable Temperature:	270	F
Set Point:	500	psig
LTOP Temperature Adjustment:	0	F
LTOP Pressure Adjustment:	0	psig

The coolant temperature variation

Time (hr.)	Temperature (DEG. F)
0	60
3.4	400

Run date 3/3/2000
JRG

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User Input Material Data:

The temperature independent material properties

The thermal conductivity:	2	Btu/in.-hr.-F
The thermal diffusivity:	61.488	in.^2/hr.
The Poisson's ratio:	0.3	

The temperature dependent material properties

Temperature (DEG. F)	Coeff. of Thermal Expansion (x 10^-6 in./in./F)
70	6.41
100	6.53
150	6.73
200	6.93
250	7.12
300	7.3
350	7.49
400	7.66
450	7.84
500	8.03
550	8.21
600	8.35
650	8.51
700	8.64
750	8.78
800	8.9

Temperature (DEG. F)	Young's Moduli (Msi)
70	27.8
200	27.1
300	26.7
400	26.1
500	25.7
600	25.2
700	24.6
800	23

Calc 2000-0001 -00

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Results: P-T Curve, Core Noncritical

Temperature (DEG.F)	Pressure (psig)
60	622
66.8	622
73.6	622
80.4	622
87.2	622
94	622
100.8	622
107.6	622
114.4	622
121.2	622
128	622
134.8	622
141.6	622
148.4	622
155.2	622
162	622
168.8	622
175.6	622
180	622
180	1327.66
182.4	1333.67
189.2	1353.92
196	1377.72
202.8	1405.53
209.6	1437.84
216.4	1475.25
223.2	1518.45
230	1568.25
236.8	1625.54
243.6	1691.42
250.4	1767.14
257.2	1854.09
264	1953.9
270.8	2068.44
277.6	2199.84
284.4	2350.59
291.2	2523.46
298	2721.65
304.8	2948.85
311.6	3209.28
318.4	3409.23
325.2	3635.17
332	3893.8
338.8	4189.91
345.6	4529.03
352.4	4917.44
359.2	5362.3
366	5871.85
372.8	6455.51
379.6	7124.06
386.4	7889.76
393.2	8766.92
400	9771.77

Calc. 2000-0004-00

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Results: P-T Curve, Core Critical

Temperature (DEG. F)	Pressure (psig)
224.886	0
224.886	1341.07
229.2	1353.92
236	1377.72
242.8	1405.53
249.6	1437.84
256.4	1475.25
263.2	1518.45
270	1568.25
276.8	1625.54
283.6	1691.42
290.4	1767.14
297.2	1854.09
304	1953.9
310.8	2068.44
317.6	2199.84
324.4	2350.59
331.2	2523.46
338	2721.65
344.8	2948.85
351.6	3209.28
358.4	3409.23
365.2	3635.17
372	3893.8
378.8	4189.91
385.6	4529.03
392.4	4917.44
399.2	5362.3
406	5871.85
412.8	6455.51
419.6	7124.06
426.4	7889.76
433.2	8766.92
440	9771.77

Calc 2000-0001-00

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Results: P-T Curve for Inservice Hydrostatic Test

Temperature (DEG. F)	Pressure (psig)
150	2000
150	2102.5
155.2	2113.6
162	2130.22
168.8	2149.23
175.6	2171
182.4	2195.91
189.2	2224.44
196	2257.09
202.8	2294.49
209.6	2337.32
216.4	2386.37
223.2	2442.55
224.886	2458.5

Calc 2000-0001-00

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Results: Temperature Distribution

Time (hr.)	T @ 1/8 T (DEG. F)	T @ 1/4 T (DEG. F)	T @ 3/8 T (DEG. F)	T @ 1/2 T (DEG. F)
0	60	60	60	60
0.068	63.3563	61.9389	61.1029	60.5675
0.136	68.2848	65.7357	63.9252	62.5992
0.204	73.6513	70.2116	67.6032	65.5965
0.272	79.2995	75.1405	71.8833	69.3185
0.34	85.1659	80.4234	76.6393	73.6204
0.408	91.2081	85.9918	81.7797	78.3921
0.476	97.3927	91.7918	87.2322	83.545
0.544	103.693	97.7797	92.9379	89.0075
0.612	110.087	103.92	98.8493	94.7214
0.68	116.558	110.185	104.928	100.639
0.748	123.09	116.55	111.142	106.723
0.816	129.673	122.997	117.466	112.942
0.884	136.296	129.51	123.879	119.269
0.952	142.953	136.077	130.366	125.686
1.02	149.637	142.688	136.911	132.174
1.088	156.342	149.334	143.504	138.722
1.156	163.065	156.01	150.136	145.316
1.224	169.803	162.709	156.8	151.949
1.292	176.553	169.426	163.489	158.614
1.36	183.311	176.16	170.199	165.304
1.428	190.078	182.905	176.926	172.015
1.496	196.851	189.661	183.666	178.742
1.564	203.629	196.426	190.418	185.483
1.632	210.411	203.197	197.179	192.236
1.7	217.197	209.973	203.947	198.997
1.768	223.985	216.754	210.722	205.765
1.836	230.775	223.538	217.501	212.54
1.904	237.568	230.326	224.284	219.319
1.972	244.361	237.115	231.07	226.102
2.04	251.156	243.907	237.859	232.888
2.108	257.952	250.7	244.649	239.677
2.176	264.749	257.495	251.442	246.468
2.244	271.546	264.29	258.236	253.261
2.312	278.344	271.087	265.031	260.055
2.38	285.142	277.884	271.827	266.85
2.448	291.94	284.681	278.624	273.646
2.516	298.739	291.479	285.421	280.443
2.584	305.538	298.278	292.219	287.24
2.652	312.337	305.077	299.018	294.038
2.72	319.137	311.876	305.816	300.836
2.788	325.936	318.675	312.615	307.635
2.856	332.736	325.474	319.414	314.434
2.924	339.535	332.273	326.213	321.233
2.992	346.335	339.073	333.013	328.032
3.06	353.135	345.873	339.812	334.832
3.128	359.935	352.672	346.612	341.631
3.196	366.735	359.472	353.411	348.431
3.264	373.534	366.272	360.211	355.23
3.332	380.334	373.072	367.011	362.03
3.4	387.134	379.872	373.811	368.83

Calc 2100-0001-00

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Time (hr.)	T @ 5/8 T (DEG. F)	T @ 3/4 T (DEG. F)	T @ 7/8 T (DEG. F)	T @ 1 T (DEG. F)
0	60	60	60	60
0.068	60.2701	60.1466	60.073	60.036
0.136	61.7029	61.166	60.8523	60.737
0.204	64.1553	63.2222	62.6686	62.4731
0.272	67.4265	66.1629	65.408	65.1451
0.34	71.3612	69.828	68.9087	68.5906
0.408	75.8345	74.0821	73.0292	72.6663
0.476	80.7451	78.8148	77.6533	77.254
0.544	86.0108	83.936	82.6864	82.2575
0.612	91.5649	89.3728	88.0515	87.5987
0.68	97.3532	95.0657	93.6864	93.2141
0.748	103.332	100.967	99.5402	99.0521
0.816	109.464	107.037	105.572	105.071
0.884	115.723	113.244	111.748	111.237
0.952	122.083	119.563	118.041	117.521
1.02	128.526	125.972	124.43	123.903
1.088	135.035	132.454	130.896	130.363
1.156	141.6	138.997	137.425	136.888
1.224	148.209	145.587	144.004	143.464
1.292	154.853	152.217	150.625	150.082
1.36	161.527	158.879	157.28	156.734
1.428	168.225	165.567	163.962	163.414
1.496	174.941	172.276	170.666	170.117
1.564	181.674	179.002	177.389	176.838
1.632	188.419	185.742	184.125	183.573
1.7	195.175	192.493	190.874	190.321
1.768	201.938	199.254	197.632	197.079
1.836	208.709	206.022	204.399	203.845
1.904	215.485	212.796	211.171	210.617
1.972	222.266	219.574	217.949	217.394
2.04	229.05	226.357	224.731	224.175
2.108	235.837	233.143	231.516	230.961
2.176	242.627	239.932	238.304	237.748
2.244	249.418	246.723	245.094	244.539
2.312	256.212	253.515	251.886	251.331
2.38	263.006	260.309	258.68	258.124
2.448	269.802	267.104	265.475	264.919
2.516	276.598	273.9	272.271	271.714
2.584	283.395	280.697	279.067	278.511
2.652	290.193	287.494	285.864	285.308
2.72	296.991	294.292	292.662	292.106
2.788	303.789	301.09	299.46	298.904
2.856	310.588	307.889	306.259	305.702
2.924	317.387	314.688	313.058	312.501
2.992	324.186	321.487	319.857	319.3
3.06	330.985	328.286	326.656	326.099
3.128	337.785	335.086	333.455	332.899
3.196	344.584	341.885	340.255	339.698
3.264	351.384	348.685	347.054	346.498
3.332	358.183	355.484	353.854	353.297
3.4	364.983	362.284	360.654	360.097

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Results: Fracture Toughness and Thermal Stress Intensity Factors

Time (hr.)	K_IC @ 1/4 T (psi-sqrt in.)	K_IC @ 3/4 T (psi-sqrt in.)	K_TH @ 1/4 T (psi-sqrt in.)	K_TH @ 3/4 T (psi-sqrt in.)
0	33498.7	33851.7	0	0
0.068	33510.5	33853.6	-1125.92	674.089
0.136	33535	33867	-2442.3	1648.47
0.204	33566.4	33895	-3554.59	2506.41
0.272	33604.4	33937.1	-4468.15	3215.79
0.34	33649.4	33993.2	-5215.91	3796.99
0.408	33702.4	34063.7	-5828.5	4273.08
0.476	33764.2	34149.4	-6331.23	4663.66
0.544	33836	34251.8	-6744.68	4984.76
0.612	33919.1	34372.6	-7085.57	5249.36
0.68	34015	34514	-7367.42	5468.01
0.748	34125.7	34678.6	-7601.24	5649.28
0.816	34253.1	34869.5	-7795.97	5800.11
0.884	34399.6	35090.1	-7958.87	5926.17
0.952	34568	35344.8	-8095.85	6032.06
1.02	34761.3	35638.1	-8211.71	6121.5
1.088	34983.3	35975.6	-8310.36	6197.55
1.156	35238	36363.6	-8394.98	6262.69
1.224	35530.2	36809.3	-8468.16	6318.92
1.292	35865.3	37321.1	-8532.01	6367.88
1.36	36249.5	37908.5	-8588.24	6410.93
1.428	36689.9	38582.3	-8638.25	6449.13
1.496	37194.8	39355.2	-8683.19	6483.38
1.564	37773.5	40241.5	-8723.96	6514.4
1.632	38436.8	41257.6	-8761.34	6542.78
1.7	39196.8	42422.5	-8796.26	6569.24
1.768	40067.8	43757.7	-8829.91	6594.68
1.836	41065.9	45288	-8861.71	6618.68
1.904	42209.6	47041.7	-8891.99	6641.5
1.972	43520	49051.5	-8921.03	6663.35
2.04	45021.5	51354.5	-8949.03	6684.4
2.108	46741.8	53993.5	-8976.19	6704.79
2.176	48712.9	57017.3	-9002.64	6724.64
2.244	50971.2	60482.1	-9026.99	6742.9
2.312	53558.7	64451.9	-9050.65	6760.63
2.38	56523.2	69000.4	-9073.9	6778.05
2.448	59919.7	74211.8	-9096.8	6795.2
2.516	63811.1	80182.7	-9119.42	6812.12
2.584	68269.5	87023.7	-9141.78	6828.86
2.652	73377.5	94861.6	-9163.94	6845.43
2.72	79229.7	103842	-9184.34	6860.69
2.788	85934.5	114130	-9203.14	6874.74
2.856	93616.1	125917	-9221.76	6888.66
2.924	102417	139422	-9240.21	6902.46
2.992	112500	154895	-9258.51	6916.14
3.06	124052	172621	-9276.67	6929.7
3.128	137287	192930	-9294.69	6943.17
3.196	152449	216198	-9312.01	6956.11
3.264	169821	242856	-9326.4	6966.87
3.332	189724	273398	-9340.69	6977.54
3.4	212526	308389	-9354.87	6988.14

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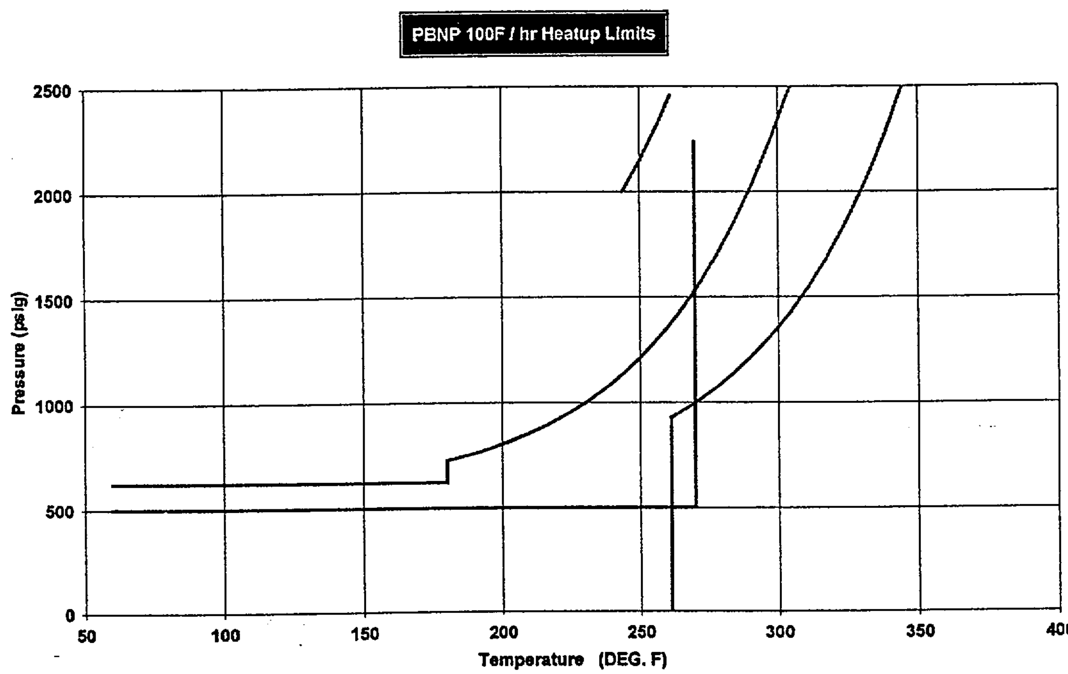

Calc 2000-0001-00

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P-T Calculator for Windows: Summary Report

Point Beach Cooldown Limits - Unit 1 Axial Flaw

User Input Data:

```
The inner radius of vessel base metal:              66.156  in.
The outer radius of vessel base metal:              72.656  in.

The operating pressure:                               2235  psig
The pre-service test pressure:                        3110  psig
The in-service test pressure:                       2458.5  psig

The h value:                                        24.375  Btu/in.^2-hr.-F

The pressure instrument error:                           0  psig
The temperature instrument error:                        0  F

The limiting material: SA-812 / 775
The initial RT_NDT at closure flange:                   60  F
The RT_NDT at 1/4 thickness up to 32 EFPY:             209  F
The RT_NDT at 3/4 thickness up to 32 EFPY:             194  F

The Analysis method:                    Raju-Newman method
The Flaw Orientation:                          Axial Flaw
The Fracture Criterion:                              K_IC

The 10 F/hr Loading is used to Generate ISLH P-T Curve

Normal Operation Safety Factor:                          2
ISLH Safety Factor:                                    1.5
Flange Region Thermal Lag:                             0 F

LTOP Set Point and Enable Temperature are Input by the User
Enable Temperature:                                  270 F
Set Point:                                           500 psig
LTOP Temperature Adjustment:                           0 F
LTOP Pressure Adjustment:                              0 psig
```

The coolant temperature variation

Time (hr.)	Temperature (DEG. F)
0	400
3.4	60

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JR

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User Input Material Data:

The temperature independent material properties

The thermal conductivity:	2.03333	Btu/in.-hr.-F
The thermal diffusivity:	62.928	in.^2/hr.
The Poisson's ratio:	0.3	

The temperature dependent material properties

Temperature (DEG. F)	Coeff. of Thermal Expansion (x 10^-6 in./in./F)
70	7.02
100	7.13
150	7.29
200	7.45
250	7.6
300	7.74
350	7.88
400	8.01
450	8.13
500	8.25
550	8.36
600	8.46
650	8.55
700	8.63
750	8.71
800	8.78

Temperature (DEG. F)	Young's Moduli (Msi)
70	29.2
200	28.5
300	28
400	27.4
500	27
600	26.4
700	25.3
800	23.9

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Results: P-T Curve, Core Noncritical

Temperature (DEG.F)	Pressure (psig)
60	514.334
66.8	518.214
73.6	522.725
80.4	528.014
87.2	534.155
94	541.241
100.8	549.41
107.6	558.819
114.4	569.649
121.2	582.107
128	596.43
134.8	612.889
138.077	622
141.6	622
148.4	622
155.2	622
162	622
168.8	622
175.6	622
180	622
180	805.43
182.4	820.674
189.2	870.199
196	926.98
202.8	992.072
209.6	1066.68
216.4	1152.19
223.2	1232.25
230	1313.65
236.8	1406.91
243.6	1513.76
250.4	1636.17
257.2	1776.41
264	1937.09
270.8	2121.17
277.6	2332.07
284.4	2573.69
291.2	2850.51
298	3167.67
304.8	3531.02
311.6	3947.31
318.4	4424.24
325.2	4970.66
332	5596.68
338.8	6313.9
345.6	7135.6
352.4	8077.01
359.2	9155.57
366	10391.3
372.8	11807
379.6	13428.9
386.4	15287.1
393.2	17416.1
400	19855.2

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Results: Temperature Distribution

Time (hr.)	T @ 1/8 T (DEG. F)	T @ 1/4 T (DEG. F)	T @ 3/8 T (DEG. F)	T @ 1/2 T (DEG. F)
0	400	400	400	400
0.068	395.95	397.63	398.62	399.265
0.136	390.542	393.394	395.414	396.911
0.204	384.789	388.539	391.371	393.569
0.272	378.812	383.268	386.742	389.497
0.34	372.662	377.676	381.659	384.854
0.408	366.376	371.832	376.217	379.761
0.476	359.983	365.788	370.491	374.311
0.544	353.504	359.586	364.541	368.579
0.612	346.959	353.259	358.413	362.623
0.68	340.36	346.833	352.145	356.492
0.748	333.72	340.329	345.765	350.22
0.816	327.046	333.764	339.298	343.839
0.884	320.345	327.149	332.761	337.369
0.952	313.624	320.495	326.17	330.831
1.02	306.887	313.811	319.534	324.238
1.088	300.136	307.103	312.865	317.602
1.156	293.375	300.376	306.168	310.932
1.224	286.606	293.633	299.449	304.234
1.292	279.83	286.878	292.714	297.515
1.36	273.049	280.114	285.965	290.779
1.428	266.265	273.343	279.205	284.03
1.496	259.477	266.565	272.437	277.27
1.564	252.686	259.783	265.663	270.502
1.632	245.894	252.997	258.882	263.727
1.7	239.1	246.208	252.098	256.947
1.768	232.305	239.417	245.311	250.163
1.836	225.508	232.624	238.521	243.375
1.904	218.711	225.829	231.729	236.585
1.972	211.914	219.034	224.935	229.793
2.04	205.116	212.237	218.14	222.999
2.108	198.317	205.44	211.343	216.204
2.176	191.518	198.642	204.547	209.408
2.244	184.719	191.844	197.749	202.611
2.312	177.92	185.045	190.951	195.813
2.38	171.12	178.246	184.152	189.015
2.448	164.321	171.447	177.354	182.216
2.516	157.521	164.648	170.555	175.418
2.584	150.722	157.848	163.755	168.619
2.652	143.922	151.049	156.956	161.819
2.72	137.122	144.249	150.156	155.02
2.788	130.322	137.449	143.357	148.22
2.856	123.522	130.65	136.557	141.421
2.924	116.722	123.85	129.757	134.621
2.992	109.922	117.05	122.957	127.821
3.06	103.122	110.25	116.158	121.021
3.128	96.3225	103.45	109.358	114.222
3.196	89.5225	96.6501	102.558	107.422
3.264	82.7226	89.8502	95.7579	100.622
3.332	75.9226	83.0502	88.9579	93.8218
3.4	69.1226	76.2502	82.158	87.0219

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Time (hr.)	T @ 5/8 T (DEG. F)	T @ 3/4 T (DEG. F)	T @ 7/8 T (DEG. F)	T @ 1 T (DEG. F)
0	400	400	400	400
0.068	399.643	399.804	399.897	399.945
0.136	397.947	398.569	398.933	399.07
0.204	395.17	396.205	396.821	397.042
0.272	391.55	392.918	393.738	394.027
0.34	387.266	388.898	389.879	390.222
0.408	382.456	384.297	385.405	385.792
0.476	377.23	379.236	380.446	380.866
0.544	371.675	373.812	375.102	375.549
0.612	365.86	368.101	369.453	369.922
0.68	359.84	362.162	363.565	364.05
0.748	353.656	356.043	357.485	357.984
0.816	347.344	349.782	351.256	351.765
0.884	340.93	343.408	344.907	345.424
0.952	334.435	336.945	338.463	338.987
1.02	327.876	330.412	331.946	332.475
1.088	321.267	323.823	325.369	325.902
1.156	314.618	317.19	318.746	319.282
1.224	307.938	310.522	312.085	312.624
1.292	301.233	303.827	305.396	305.937
1.36	294.507	297.109	298.683	299.226
1.428	287.767	290.375	291.952	292.497
1.496	281.013	283.626	285.207	285.752
1.564	274.25	276.867	278.451	278.996
1.632	267.48	270.1	271.685	272.231
1.7	260.703	263.325	264.912	265.459
1.768	253.921	256.546	258.133	258.681
1.836	247.136	249.762	251.35	251.898
1.904	240.347	242.974	244.564	245.112
1.972	233.556	236.184	237.774	238.322
2.04	226.763	229.392	230.983	231.531
2.108	219.969	222.599	224.189	224.738
2.176	213.174	215.803	217.395	217.943
2.244	206.377	209.007	210.599	211.147
2.312	199.58	202.211	203.802	204.351
2.38	192.782	195.413	197.005	197.553
2.448	185.984	188.615	190.207	190.755
2.516	179.185	181.816	183.408	183.957
2.584	172.386	175.018	176.61	177.158
2.652	165.587	168.219	169.811	170.359
2.72	158.788	161.419	163.011	163.56
2.788	151.988	154.62	156.212	156.761
2.856	145.189	147.82	149.413	149.961
2.924	138.389	141.021	142.613	143.162
2.992	131.59	134.221	135.813	136.362
3.06	124.79	127.421	129.013	129.562
3.128	117.99	120.622	122.214	122.763
3.196	111.19	113.822	115.414	115.963
3.264	104.39	107.022	108.614	109.163
3.332	97.5902	100.222	101.814	102.363
3.4	90.7903	93.422	95.0141	95.563

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Results: Fracture Toughness and Thermal Stress Intensity Factors

Time (hr.)	K_IC @ 1/4 T (psi-sqrt in.)	K_IC @ 3/4 T (psi-sqrt in.)	K_TH @ 1/4 T (psi-sqrt in.)	K_TH @ 3/4 T (psi-sqrt in.)
0	978758	1.30957e+006	0	0
0.068	934991	1.30456e+006	1625.31	-1020.87
0.136	861737	1.27355e+006	3322.47	-2338.51
0.204	785060	1.21627e+006	4693.04	-3439.26
0.272	709831	1.141e+006	5778.89	-4315.48
0.34	638239	1.05541e+006	6636.51	-5008.02
0.408	571496	965547	7313.2	-5554.57
0.476	510206	875802	7846.72	-5985.55
0.544	454561	789181	8266.98	-6325.09
0.612	404479	707576	8597.65	-6592.31
0.68	359703	632050	8857.47	-6802.33
0.748	319878	563072	9060.72	-6966.71
0.816	284600	500706	9218.65	-7094.53
0.884	253447	444753	9341.46	-7194.01
0.952	226006	394852	9436.48	-7271.05
1.02	201880	350554	9509.53	-7330.36
1.088	180701	311372	9565.2	-7375.64
1.156	162132	276812	9607.11	-7409.82
1.224	145867	246398	9638.13	-7435.21
1.292	131630	219677	9658.77	-7452.28
1.36	119175	196235	9672.02	-7463.4
1.428	108286	175690	9679.85	-7470.16
1.496	98767.5	157700	9683.4	-7473.46
1.564	90450.4	141958	9683.55	-7474.02
1.632	83184.7	128190	9680.99	-7472.4
1.7	76838.5	116154	9676.29	-7469.05
1.768	71296.2	105635	9669.25	-7463.83
1.836	66456.6	96444.4	9659.74	-7456.66
1.904	62231	88416.4	9649.15	-7448.62
1.972	58541.7	81404.9	9637.67	-7439.87
2.04	55320.9	75282	9625.49	-7430.55
2.108	52509.1	69935.6	9612.74	-7420.77
2.176	50054.5	65267.6	9599.52	-7410.63
2.244	47911.8	61192.1	9585.92	-7400.17
2.312	46041.5	57634.2	9571.14	-7388.79
2.38	44408.8	54528.3	9556.08	-7377.19
2.448	42983.7	51817	9540.8	-7365.41
2.516	41739.7	49450.2	9525.32	-7353.48
2.584	40653.9	47384.2	9509.67	-7341.41
2.652	39706.2	45580.8	9493.88	-7329.23
2.72	38878.9	44006.7	9477.95	-7316.94
2.788	38156.8	42632.6	9461.9	-7304.55
2.856	37526.5	41433.3	9445.74	-7292.08
2.924	36976.4	40386.4	9429.48	-7279.53
2.992	36496.2	39472.6	9413.12	-7266.91
3.06	36077.1	38675.1	9396.67	-7254.21
3.128	35711.2	37978.9	9380.13	-7241.45
3.196	35391.9	37371.2	9363.51	-7228.62
3.264	35113.2	36840.8	9345.56	-7214.76
3.332	34869.9	36377.9	9324.62	-7198.59
3.4	34657.6	35973.8	9303.58	-7182.35

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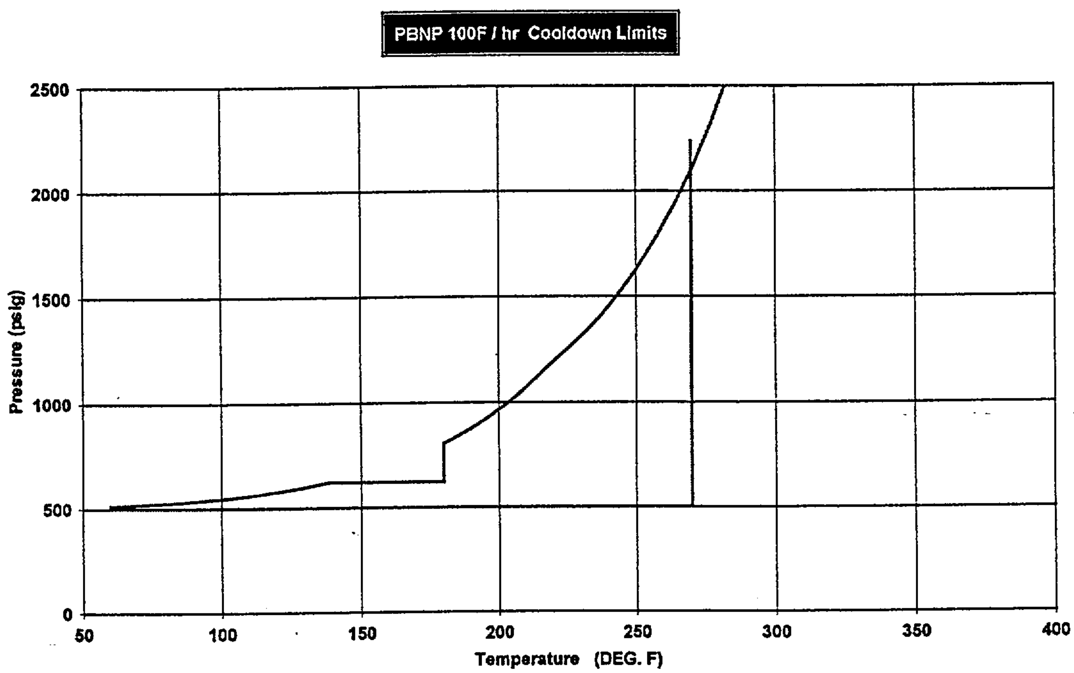

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P-T Calculator for Windows: Summary Report

Point Beach Cooldown Limits - Unit 2 Circumferential Flaw

User Input Data:

The inner radius of vessel base metal:	66.156	in.
The outer radius of vessel base metal:	72.656	in.
The operating pressure:	2235	psig
The pre-service test pressure:	3110	psig
The in-service test pressure:	2458.5	psig
The h value:	24.375	Btu/in.^2-hr.-F
The pressure instrument error:	0	psig
The temperature instrument error:	0	F
The limiting material: 72442 (SA-1484)		
The initial RT_NDT at closure flange:	60	F
The RT_NDT at 1/4 thickness up to 34 EFPY:	272	F
The RT_NDT at 3/4 thickness up to 34 EFPY:	233	F
The Analysis method:	Raju-Newman method	
The Flaw Orientation:	Circumferential Flaw	
The Fracture Criterion:	K_IC	
The 10 F/hr Loading is used to Generate ISLH P-T Curve		
Normal Operation Safety Factor:	2	
ISLH Safety Factor:	1.5	
Flange Region Thermal Lag:	0	F
LTOP Set Point and Enable Temperature are Input by the User		
Enable Temperature:	0	F
Set Point:	0	psig
LTOP Temperature Adjustment:	0	F
LTOP Pressure Adjustment:	0	psig

The coolant temperature variation

Time (hr.)	Temperature (DEG. F)
0	400
3.4	60

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User Input Material Data:

The temperature independent material properties

The thermal conductivity:	2	Btu/in.-hr.-F
The thermal diffusivity:	61.488	in.^2/hr.
The Poisson's ratio:	0.3	

The temperature dependent material properties

Temperature (DEG. F)	Coeff. of Thermal Expansion (x 10^-6 in./in./F)
70	6.41
100	6.53
150	6.73
200	6.93
250	7.12
300	7.3
350	7.49
400	7.66
450	7.84
500	8.03
550	8.21
600	8.35
650	8.51
700	8.64
750	8.78
800	8.9

Temperature (DEG. F)	Young's Moduli (Msi)
70	27.8
200	27.1
300	26.7
400	26.1
500	25.7
600	25.2
700	24.6
800	23

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Results: P-T Curve, Core Noncritical

Temperature (DEG.F)	Pressure (psig)
60	622
66.8	622
73.6	622
80.4	622
87.2	622
94	622
100.8	622
107.6	622
114.4	622
121.2	622
128	622
134.8	622
141.6	622
148.4	622
155.2	622
162	622
168.8	622
175.6	622
180	622
180	1274.72
182.4	1283.88
189.2	1313.78
196	1348.19
202.8	1387.78
209.6	1433.29
216.4	1485.59
223.2	1545.66
230	1614.69
236.8	1693.92
243.6	1784.8
250.4	1889.02
257.2	2008.49
264	2145.38
270.8	2302.15
277.6	2481.64
284.4	2632.29
291.2	2802.76
298	2998.07
304.8	3221.82
311.6	3478.17
318.4	3771.87
325.2	4108.36
332	4493.86
338.8	4935.53
345.6	5441.54
352.4	6021.27
359.2	6685.45
366	7446.39
372.8	8318.18
379.6	9316.98
386.4	10461.3
393.2	11772.3
400	13274.3

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Results: Temperature Distribution

Time (hr.)	T @ 1/8 T (DEG. F)	T @ 1/4 T (DEG. F)	T @ 3/8 T (DEG. F)	T @ 1/2 T (DEG. F)
0	400	400	400	400
0.068	395.972	397.66	398.647	399.286
0.136	390.577	393.451	395.476	396.972
0.204	384.838	388.621	391.47	393.675
0.272	378.875	383.377	386.881	389.654
0.34	372.74	377.814	381.838	385.062
0.408	366.468	371.997	376.435	380.019
0.476	360.089	365.979	370.746	374.615
0.544	353.623	359.801	364.83	368.926
0.612	347.089	353.495	358.733	363.009
0.68	340.5	347.089	352.492	356.912
0.748	333.868	340.601	346.136	350.67
0.816	327.202	334.05	339.689	344.314
0.884	320.508	327.448	333.17	337.868
0.952	313.793	320.805	326.594	331.349
1.02	307.06	314.13	319.972	324.772
1.088	300.313	307.43	313.313	318.15
1.156	293.556	300.709	306.625	311.491
1.224	286.79	293.971	299.915	304.803
1.292	280.017	287.222	293.186	298.093
1.36	273.238	280.461	286.442	291.364
1.428	266.455	273.693	279.687	284.62
1.496	259.668	266.918	272.923	277.865
1.564	252.879	260.138	266.152	271.101
1.632	246.088	253.354	259.374	264.329
1.7	239.295	246.567	252.592	257.552
1.768	232.5	239.777	245.806	250.77
1.836	225.704	232.985	239.018	243.984
1.904	218.908	226.191	232.227	237.196
1.972	212.11	219.396	225.434	230.405
2.04	205.313	212.6	218.64	223.612
2.108	198.514	205.803	211.844	216.818
2.176	191.716	199.006	205.048	210.022
2.244	184.917	192.208	198.251	203.226
2.312	178.118	185.41	191.453	196.429
2.38	171.318	178.611	184.655	189.631
2.448	164.519	171.812	177.856	182.833
2.516	157.719	165.013	171.057	176.034
2.584	150.92	158.213	164.258	169.235
2.652	144.12	151.414	157.459	162.436
2.72	137.32	144.614	150.66	155.637
2.788	130.52	137.815	143.86	148.838
2.856	123.72	131.015	137.061	142.038
2.924	116.921	124.215	130.261	135.238
2.992	110.121	117.415	123.461	128.439
3.06	103.321	110.615	116.661	121.639
3.128	96.5208	103.815	109.861	114.839
3.196	89.7208	97.0155	103.061	108.039
3.264	82.9209	90.2156	96.2616	101.239
3.332	76.1209	83.4156	89.4616	94.4394
3.4	69.3209	76.6157	82.6617	87.6395

Calc 2000-0001-00
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Time (hr.)	T @ 5/8 T (DEG. F)	T @ 3/4 T (DEG. F)	T @ 7/8 T (DEG. F)	T @ 1 T (DEG. F)
0	400	400	400	400
0.068	399.659	399.813	399.904	399.951
0.136	398.004	398.618	398.979	399.115
0.204	395.279	396.312	396.927	397.148
0.272	391.718	393.09	393.912	394.203
0.34	387.493	389.136	390.123	390.47
0.408	382.742	384.599	385.718	386.108
0.476	377.57	379.599	380.823	381.248
0.544	372.066	374.231	375.537	375.99
0.612	366.296	368.569	369.941	370.417
0.68	360.315	362.674	364.099	364.592
0.748	354.166	356.594	358.061	358.568
0.816	347.884	350.366	351.867	352.385
0.884	341.496	344.022	345.548	346.076
0.952	335.024	337.584	339.131	339.666
1.02	328.485	331.072	332.636	333.176
1.088	321.892	324.501	326.078	326.623
1.156	315.257	317.883	319.471	320.019
1.224	308.588	311.227	312.824	313.375
1.292	301.892	304.543	306.146	306.699
1.36	295.175	297.834	299.443	299.998
1.428	288.44	291.107	292.72	293.276
1.496	281.693	284.365	285.981	286.539
1.564	274.935	277.611	279.23	279.788
1.632	268.168	270.848	272.469	273.028
1.7	261.394	264.077	265.7	266.259
1.768	254.615	257.3	258.924	259.484
1.836	247.832	250.518	252.144	252.704
1.904	241.045	243.733	245.359	245.92
1.972	234.256	236.945	238.571	239.132
2.04	227.464	230.154	231.781	232.342
2.108	220.671	223.361	224.989	225.55
2.176	213.876	216.567	218.195	218.756
2.244	207.08	209.772	211.4	211.961
2.312	200.283	202.975	204.604	205.165
2.38	193.486	196.178	197.807	198.369
2.448	186.688	189.381	191.01	191.571
2.516	179.89	182.583	184.212	184.773
2.584	173.091	175.784	177.413	177.975
2.652	166.292	168.985	170.614	171.176
2.72	159.493	162.186	163.815	164.377
2.788	152.694	155.387	157.016	157.578
2.856	145.894	148.588	150.217	150.779
2.924	139.095	141.788	143.417	143.979
2.992	132.295	134.988	136.618	137.179
3.06	125.495	128.189	129.818	130.38
3.128	118.696	121.389	123.018	123.58
3.196	111.896	114.589	116.218	116.78
3.264	105.096	107.789	109.419	109.98
3.332	98.296	100.989	102.619	103.18
3.4	91.4961	94.1895	95.8189	96.3806

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Results: Fracture Toughness and Thermal Stress Intensity Factors

Time (hr.)	K_IC @ 1/4 T (psi-sqrt in.)	K_IC @ 3/4 T (psi-sqrt in.)	K_TH @ 1/4 T (psi-sqrt in.)	K_TH @ 3/4 T (psi-sqrt in.)
0	301411	618295	0	0
0.068	289151	616112	1508	-910.386
0.136	268482	602350	3091.87	-2097.67
0.204	246819	576696	4377.52	-3096.68
0.272	225551	542778	5400.23	-3895.55
0.34	205296	504031	6210.67	-4529.2
0.408	186396	463191	6851.89	-5030.7
0.476	169023	422271	7358.48	-5427.02
0.544	153236	382660	7758.01	-5739.68
0.612	139014	345247	8072.44	-5985.85
0.68	126288	310544	8319.21	-6179.16
0.748	114961	278788	8511.35	-6329.82
0.816	104920	250028	8658.62	-6445.5
0.884	96048.6	224188	8771.78	-6534.53
0.952	88229.8	201114	8857.83	-6602.38
1.02	81352.6	180609	8922.35	-6653.41
1.088	75313.4	162455	8969.74	-6691.06
1.156	70016.5	146431	9003.51	-6718.07
1.224	65375.5	132319	9026.45	-6736.63
1.292	61312.3	119915	9038.79	-6746.98
1.36	57757.1	109027	9043.38	-6751.3
1.428	54648.1	99481.8	9042.52	-6751.37
1.496	51930.2	91120.6	9037.32	-6748.05
1.564	49555.1	83802.2	9028.66	-6742.03
1.632	47479.9	77400.1	9017.25	-6733.87
1.7	45667.3	71802.2	9003.64	-6723.98
1.768	44084.2	66909.3	8987.79	-6712.36
1.836	42701.7	62633.7	8969.29	-6698.72
1.904	41494.5	58898.5	8949.66	-6684.2
1.972	40440.6	55636	8929.12	-6668.97
2.04	39520.4	52786.7	8907.84	-6653.16
2.108	38717	50298.7	8885.96	-6636.89
2.176	38015.8	48126.2	8863.58	-6620.23
2.244	37403.6	46229.4	8840.81	-6603.27
2.312	36869.2	44573.5	8818.93	-6586.96
2.38	36402.7	43127.8	8796.84	-6570.49
2.448	35995.5	41865.8	8774.5	-6553.82
2.516	35640.1	40764.2	8751.92	-6536.98
2.584	35329.8	39802.6	8729.15	-6519.98
2.652	35059	38963.1	8706.2	-6502.85
2.72	34822.6	38230.4	8683.08	-6485.6
2.788	34616.3	37590.8	8659.83	-6468.23
2.856	34436.2	37032.5	8636.43	-6450.76
2.924	34279	36545.2	8612.91	-6433.2
2.992	34141.8	36119.9	8589.27	-6415.55
3.06	34022.1	35748.6	8565.52	-6397.81
3.128	33917.5	35424.5	8541.65	-6379.98
3.196	33826.3	35141.7	8517.68	-6362.08
3.264	33746.7	34894.8	8493.61	-6344.1
3.332	33677.2	34679.3	8469.43	-6326.04
3.4	33616.5	34491.2	8445.15	-6307.91

Calc 2000-0001-00

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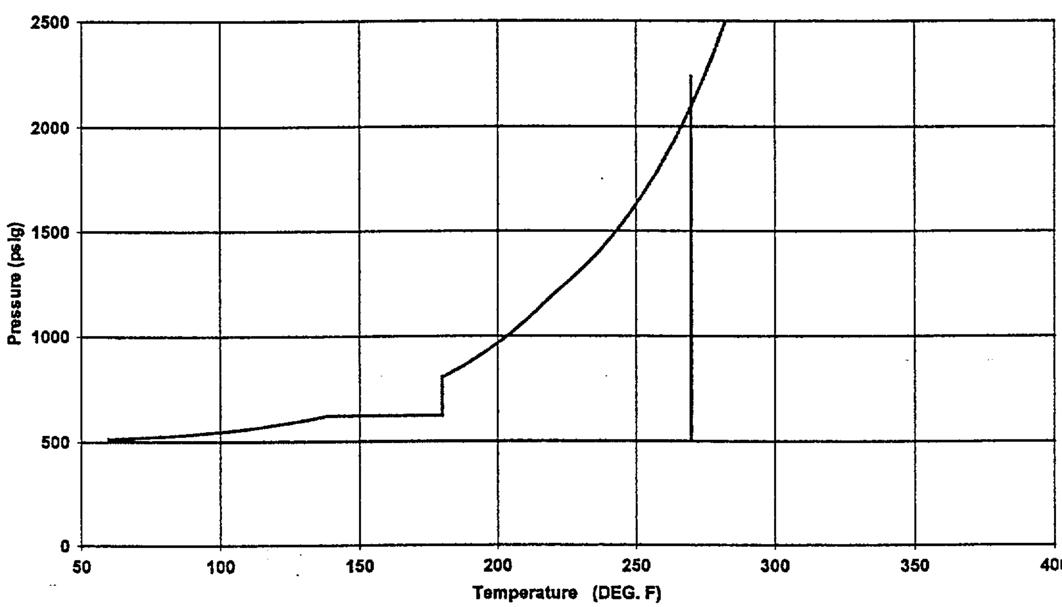
PBNP 100F / hr Cooldown Limits



Calc. 2000-0001-00

3/4/2000

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Temperature Lag During Heatup at 1/4t Location
Data From P-T Calculator

	Unit 1 Vessel			Unit 2 Vessel		
Time	IS Temp	1/4t Temp	Delta Temp	IS Temp	1/4t Temp	Delta Temp
0.00	60.00	60.00	0.00	60.00	60.00	0.00
0.07	66.80	61.96	4.84	66.80	61.94	4.86
0.14	73.60	65.78	7.82	73.60	65.74	7.86
0.20	80.40	70.28	10.12	80.40	70.21	10.19
0.27	87.20	75.24	11.96	87.20	75.14	12.06
0.34	94.00	80.55	13.45	94.00	80.42	13.58
0.41	100.80	86.14	14.66	100.80	85.99	14.81
0.48	107.60	91.97	15.63	107.60	91.79	15.81
0.54	114.40	97.98	16.42	114.40	97.78	16.62
0.61	121.20	104.14	17.06	121.20	103.92	17.28
0.68	128.00	110.43	17.57	128.00	110.18	17.82
0.75	134.80	116.81	17.99	134.80	116.55	18.25
0.82	141.60	123.28	18.32	141.60	123.00	18.60
0.88	148.40	129.81	18.59	148.40	129.51	18.89
0.95	155.20	136.39	18.81	155.20	136.08	19.12
1.02	162.00	143.01	18.99	162.00	142.69	19.31
1.09	168.80	149.67	19.13	168.80	149.33	19.47
1.16	175.60	156.35	19.25	175.60	156.01	19.59
1.22	182.40	163.06	19.34	182.40	162.71	19.69
1.29	189.20	169.78	19.42	189.20	169.43	19.77
1.36	196.00	176.52	19.48	196.00	176.16	19.84
1.43	202.80	183.27	19.53	202.80	182.91	19.89
1.50	209.60	190.03	19.57	209.60	189.66	19.94
1.56	216.40	196.80	19.60	216.40	196.43	19.97
1.63	223.20	203.57	19.63	223.20	203.20	20.00
1.70	230.00	210.35	19.65	230.00	209.97	20.03
1.77	236.80	217.13	19.67	236.80	216.75	20.05
1.84	243.60	223.92	19.68	243.60	223.54	20.06
1.90	250.40	230.71	19.69	250.40	230.33	20.07
1.97	257.20	237.50	19.70	257.20	237.12	20.08
2.04	264.00	244.29	19.71	264.00	243.91	20.09
2.11	270.80	251.08	19.72	270.80	250.70	20.10
2.18	277.60	257.88	19.72	277.60	257.49	20.11
2.24	284.40	264.68	19.72	284.40	264.29	20.11
2.31	291.20	271.47	19.73	291.20	271.09	20.11
2.38	298.00	278.27	19.73	298.00	277.88	20.12
2.45	304.80	285.07	19.73	304.80	284.68	20.12
2.52	311.60	291.87	19.73	311.60	291.48	20.12
2.58	318.40	298.67	19.73	318.40	298.28	20.12
2.65	325.20	305.46	19.74	325.20	305.08	20.12
2.72	332.00	312.26	19.74	332.00	311.88	20.12
2.79	338.80	319.06	19.74	338.80	318.67	20.13
2.86	345.60	325.86	19.74	345.60	325.47	20.13
2.92	352.40	332.66	19.74	352.40	332.27	20.13
2.99	359.20	339.46	19.74	359.20	339.07	20.13
3.06	366.00	346.26	19.74	366.00	345.87	20.13
3.13	372.80	353.06	19.74	372.80	352.67	20.13
3.20	379.60	359.86	19.74	379.60	359.47	20.13
3.26	386.40	366.66	19.74	386.40	366.27	20.13
3.33	393.20	373.46	19.74	393.20	373.07	20.13
3.40	400.00	380.26	19.74	400.00	379.87	20.13

Calc. 2000-0001-00

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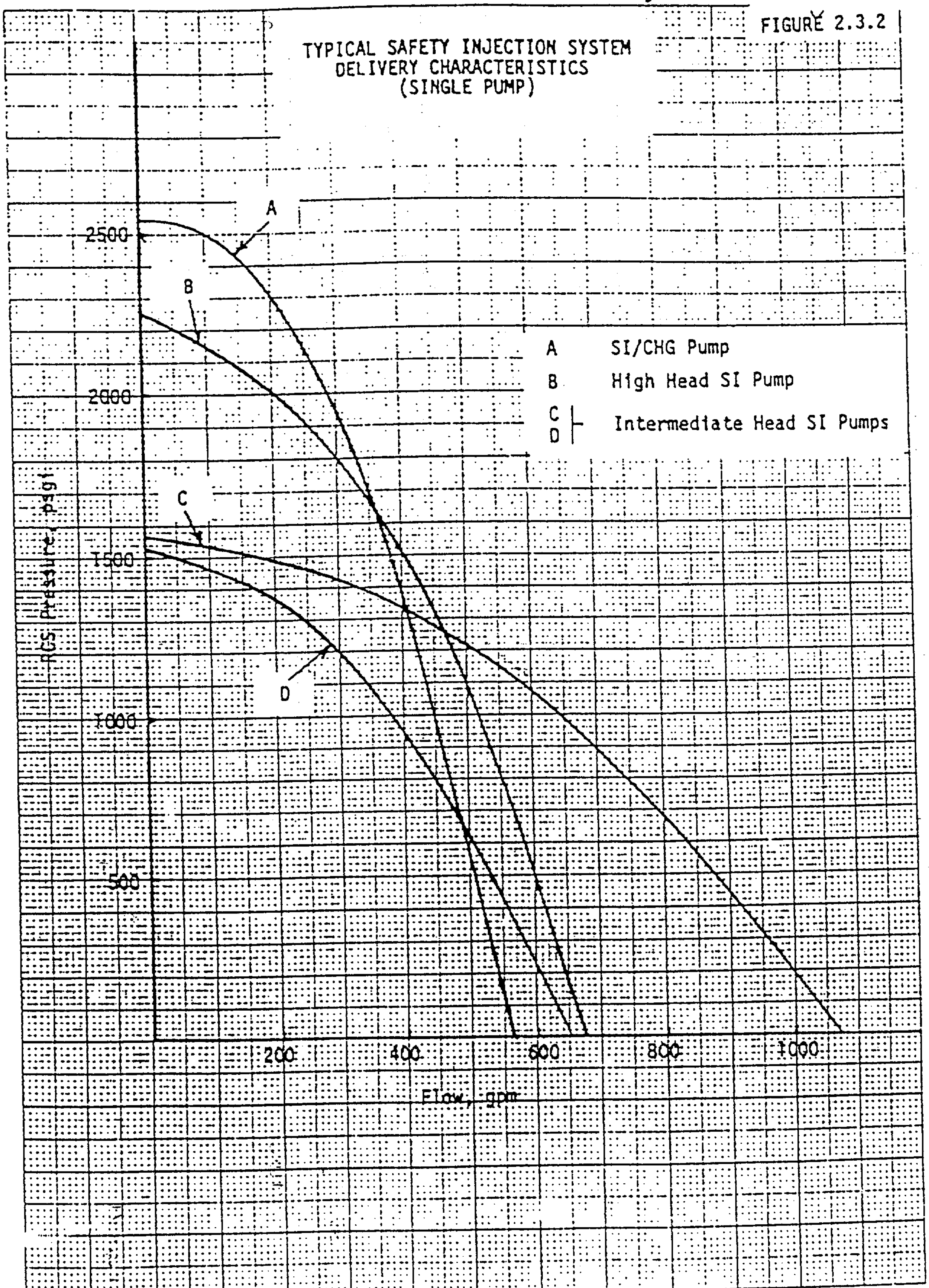

Calc. 2000-0001-00

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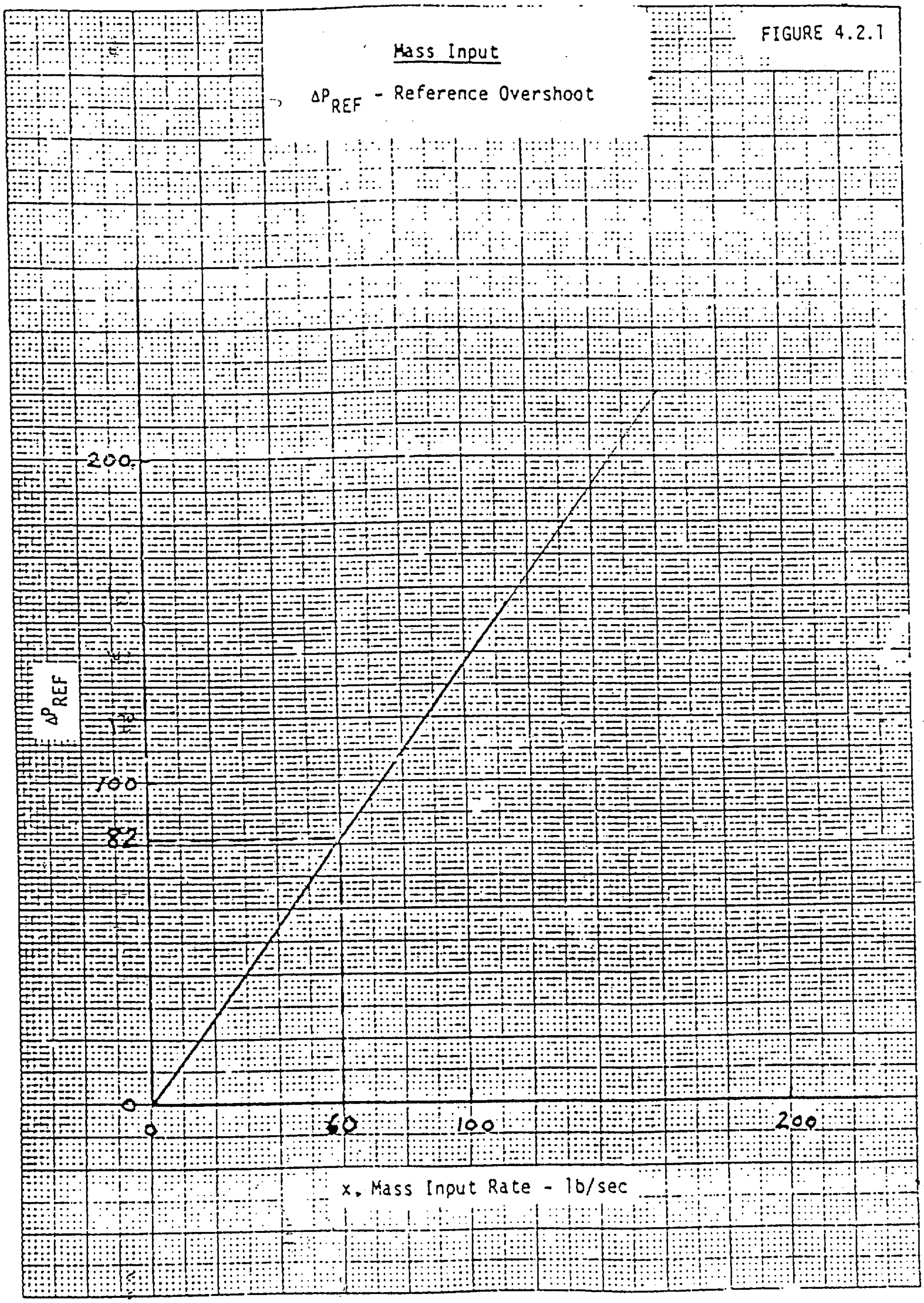
FIGURE 4.2.1



Calc. 2000-0001-00

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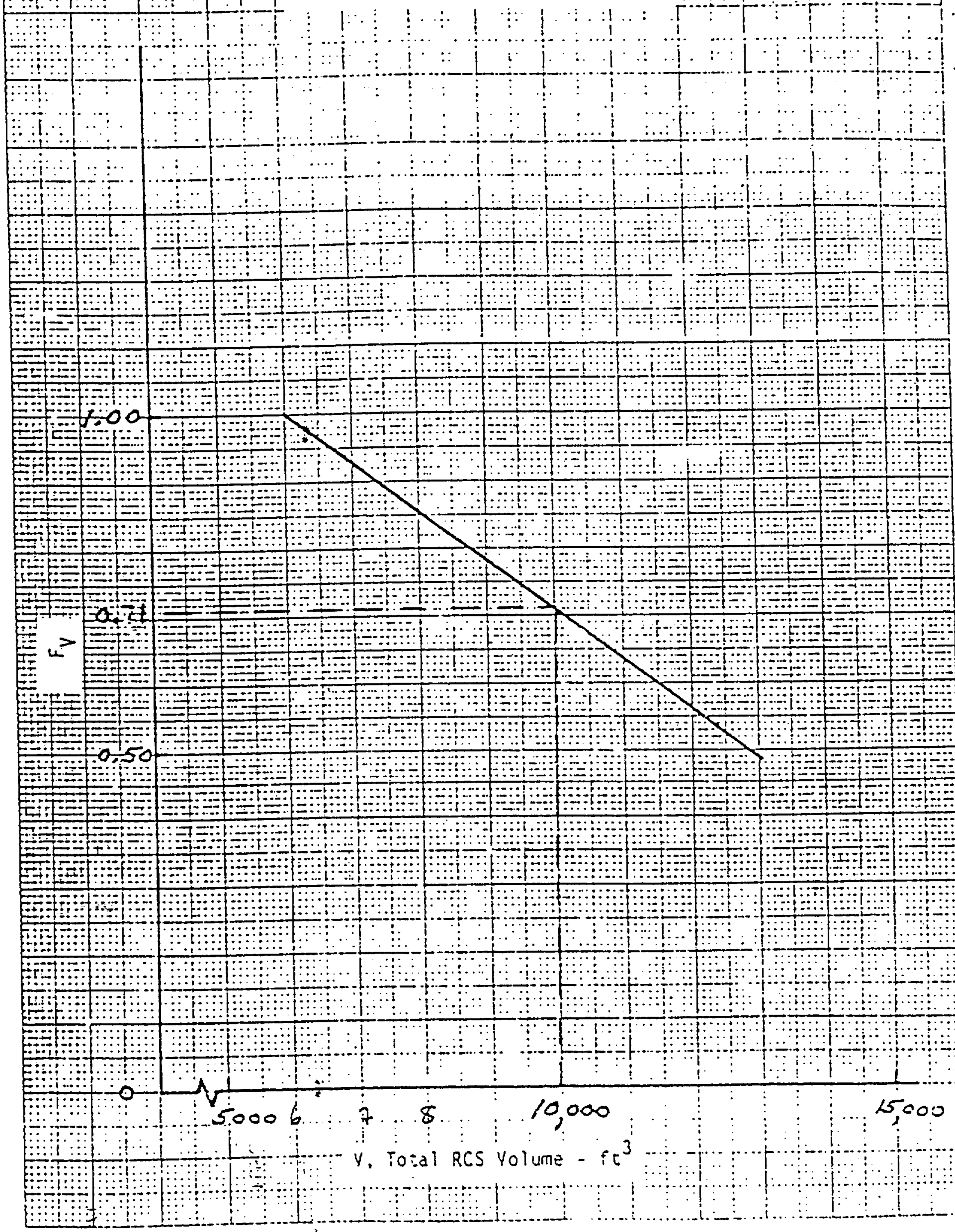
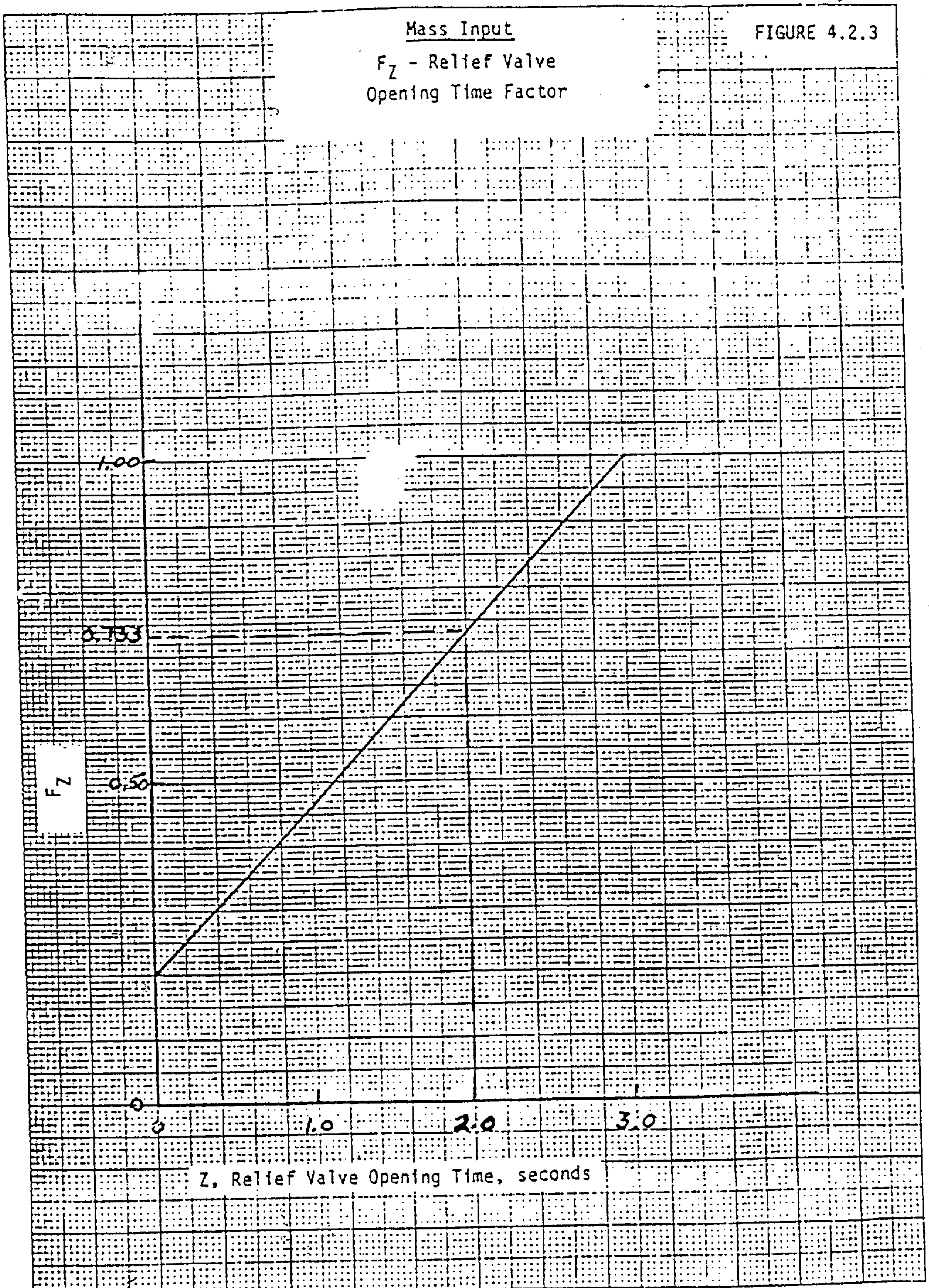
FIGURE 4.2.2

Mass Input

F_V - RCS Volume Factor




Mass Input
F_Z - Relief Valve
Opening Time Factor
FIGURE 4.2.3
1.00
0.733
0.50
F_Z
0
0
1.0
2.0
3.0
Z, Relief Valve Opening Time, seconds

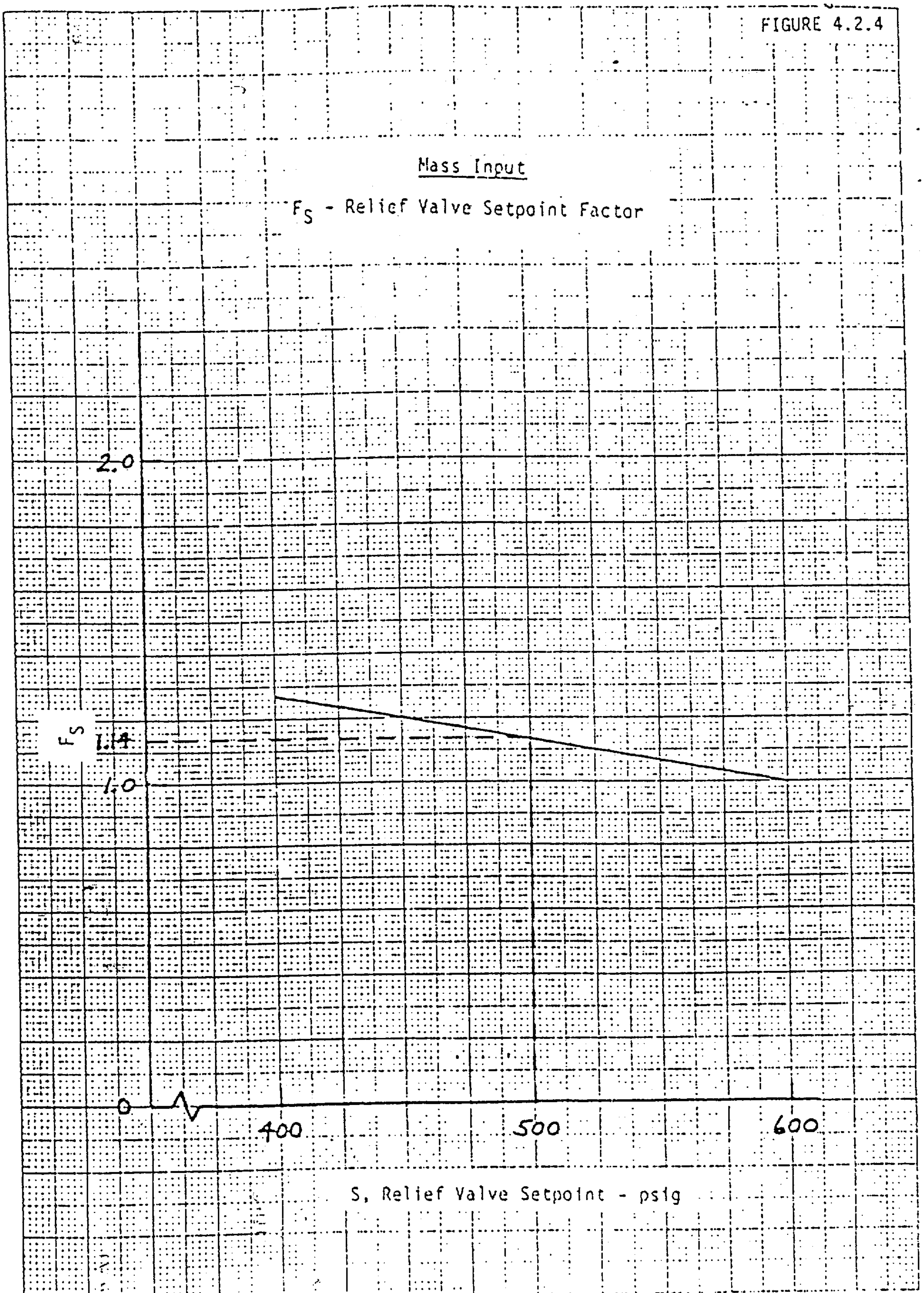


FIGURE 4.2.4
Mass Input
F_S - Relief Valve Setpoint Factor
F_S
2.0
1.4
1.0
0
400
500
600
S, Relief Valve Setpoint - psig

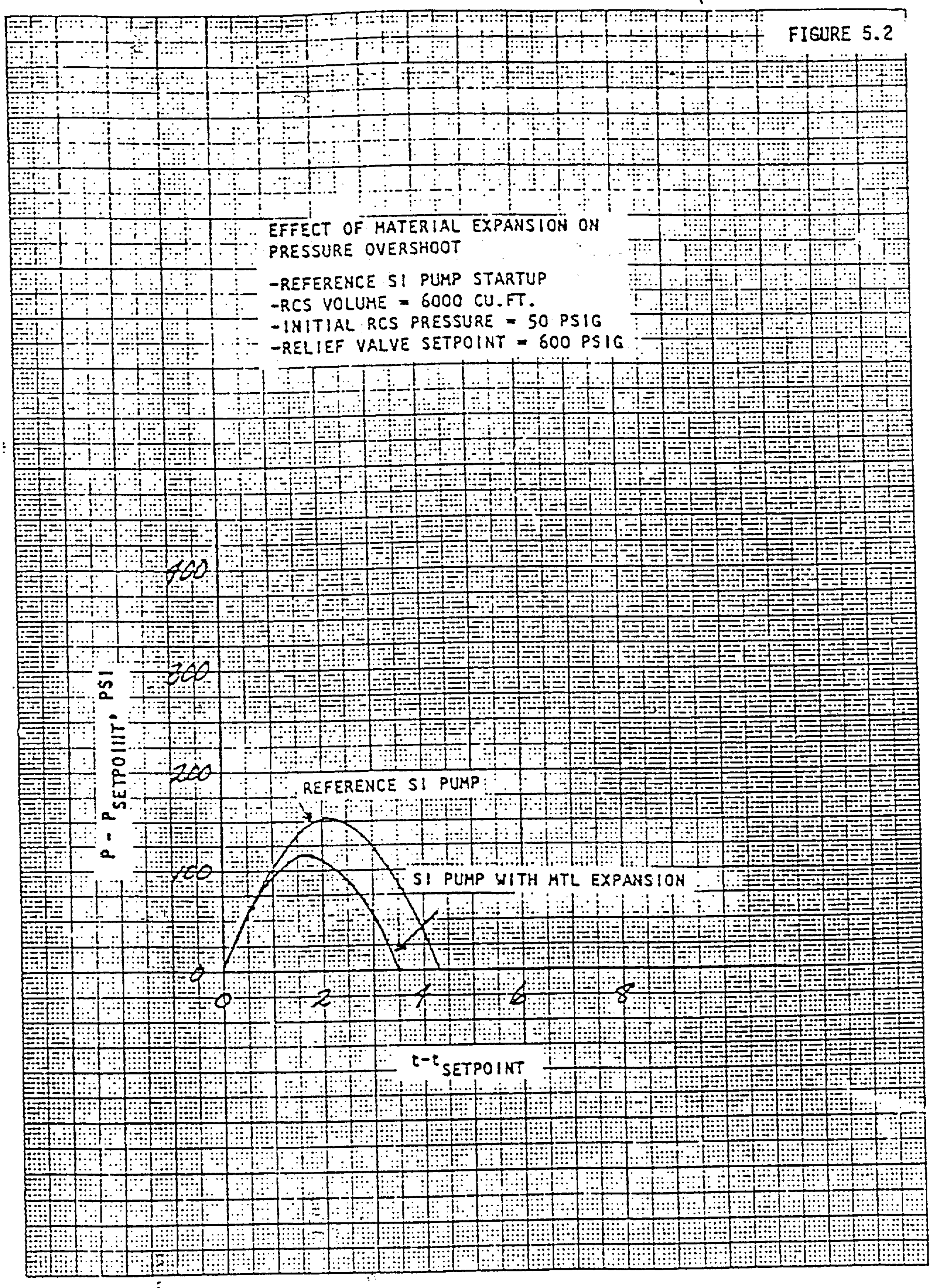
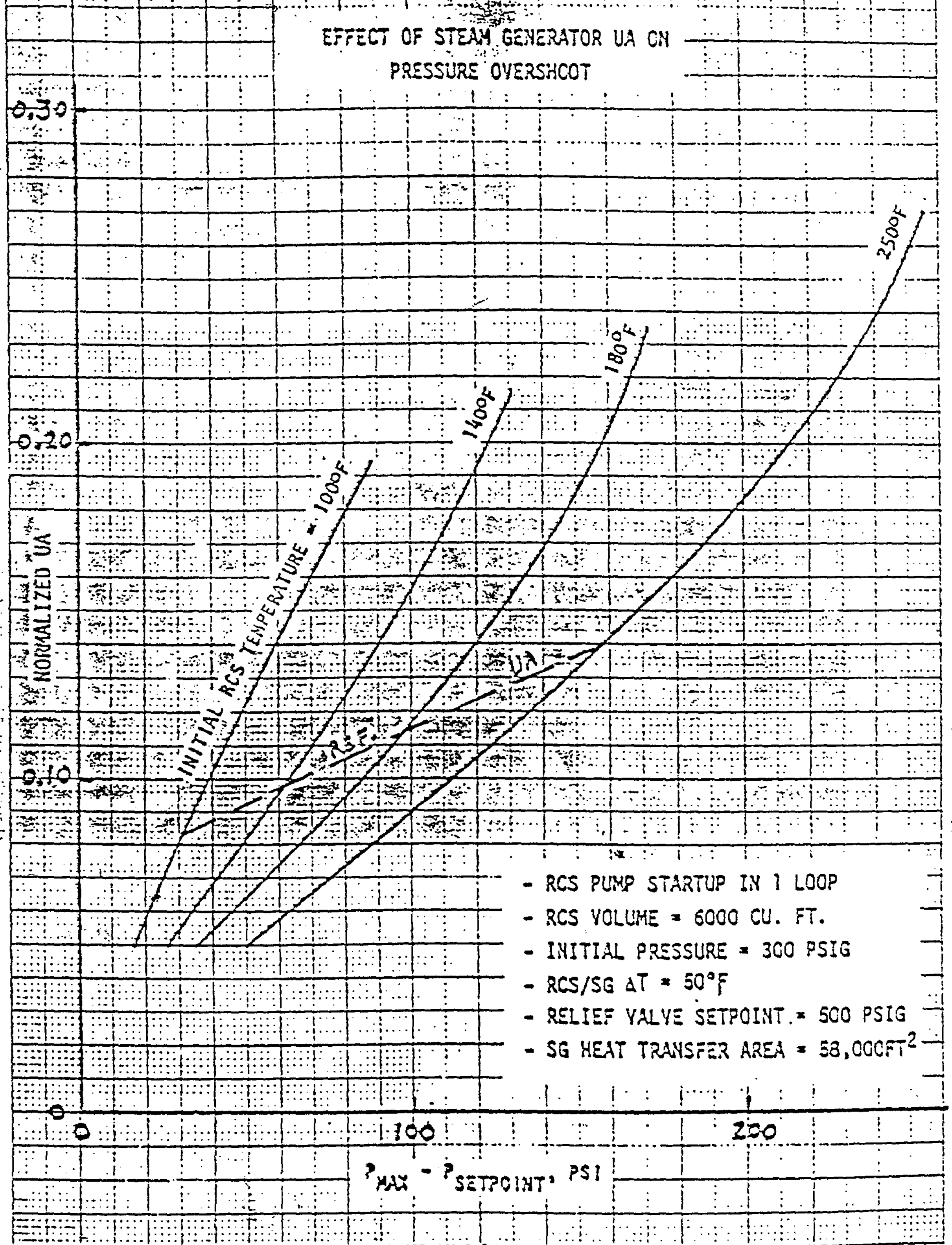

EFFECT OF MATERIAL EXPANSION ON PRESSURE OVERSHOOT
-REFERENCE SI PUMP STARTUP
-RCS VOLUME = 6000 CU.FT.
-INITIAL RCS PRESSURE = 50 PSIG
-RELIEF VALVE SETPOINT = 600 PSIG
400
300
200
100
0
REFERENCE SI PUMP
SI PUMP WITH MTL EXPANSION
$p - P_{SETPOINT}$, PSI
0
2
4
6
8
$t-t_{SETPOINT}$
FIGURE 5.2

Figure 16


EFFECT OF STEAM GENERATOR UA ON
PRESSURE OVERSHOOT
0.30
0.20
0.10
0
NORMALIZED UA
INITIAL RCS TEMPERATURE = 100°F
140°F
180°F
250°F
UA = R.S.F.
- RCS PUMP STARTUP IN 1 LOOP
- RCS VOLUME = 6000 CU. FT.
- INITIAL PRESSURE = 300 PSIG
- RCS/SG ΔT = 50°F
- RELIEF VALVE SETPOINT = 500 PSIG
- SG HEAT TRANSFER AREA = 58,000FT²
0
100
200
P MAX - P SETPOINT, PSI

ATTACHMENT 6

PROPOSED
PRESSURE AND TEMPERATURE LIMITS REPORT

Point Beach Nuclear Plant
Units 1 and 2

Pressure Temperature Limits Report
Revision 0 (Effective through: 32.2 EFPY – Unit 1; and 34.0 EFPY – Unit 2)

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This RCS Pressure and Temperature Limits Report (PTLR) for Point Beach Nuclear Plant Units 1 and 2 has been prepared in accordance with the requirements of Technical Specification 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

- 3.4.3 Pressure/Temperature (P-T) Limits
- 3.4.6 Low Temperature Overpressure Protection (LTOP) System

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. All changes to these limits must be developed using the NRC approved methodologies specified in Technical Specification 5.6.6. These limits have been determined such that all applicable limits of the safety analysis are met. All items that appear in capitalized type are defined in Technical Specification 1.1, "Definitions."

2.1 RCS Pressure and Temperature Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup rate of 100°F per hour.

b. A maximum cooldown rate of 100°F per hour.

2.1.2 The RCS P-T limits for heatup and cooldown are specified by Figures 1 and 2, respectively.

2.1.3 The minimum temperature for pressurization, using the methodology is 60°F, which when corrected for possible instrument uncertainties is a minimum indicated RCS temperature of 78°F (as read on the RCS cold leg meter) or 70°F using the hand-held, digital pyrometer.

2.2 Low Temperature Overpressure Protection System Enable Temperature (LCO 3.4.6)

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 270°F.

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.6)

2.3.1 Pressurizer Power Operated Relief Valve Lift Setting Limits

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is $\leq$ 500 psig (includes instrument uncertainty).

3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedules for Units 1 and 2 are provided in Tables 1 and 2, respectively.

The pressure vessel surveillance program is in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

Surveillance specimens for the limiting materials for the Point Beach reactor vessels are not included in the plant specific surveillance program. Therefore, the results of the examinations of these specimens do not meet the credibility criteria of USNRC Regulatory Guide 1.99, Rev. 2 for Point Beach Nuclear Plant, Units 1 and 2.

4.0 SUPPLEMENTAL DATA INFORMATION AND DATA TABLES

4.1 The RT_{PTS} values for the Point Beach Nuclear Plant limiting beltline materials is 274°F for Unit 1 and 288°F for Unit 2 at 32 EFPY.

4.2 Tables

Table Number	**Table Description**
Table 1	Point Beach Nuclear Plant, Unit 1 Reactor Vessel Surveillance Capsule Removal Schedule
Table 2	Point Beach Nuclear Plant, Unit 2 Reactor Vessel Surveillance Capsule Removal Schedule
Table 3	Point Beach Unit 1 RPV Beltline 32.2 EFPY Fluence Values
Table 4	Point Beach Unit 2 RPV Beltline 34.0 EFPY Fluence Values
Table 5	Point Beach Unit 1 RPV 1/4t Beltline Material Adjusted Reference Temperatures at 32.2 EFPY
Table 6	Point Beach Unit 2 RPV 1/4t Beltline Material Adjusted Reference Temperatures at 34.0 EFPY
Table 7	Point Beach Unit 1 RPV 3/4t Beltline Material Adjusted Reference Temperatures at 32.2 EFPY
Table 8	Point Beach Unit 2 RPV 3/4t Beltline Material Adjusted Reference Temperatures at 34.0 EFPY

5.0 REFERENCES

5.1 WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996.

5.2 WCAP-12794, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 1," Rev. 4, February 2000.

5.3 WCAP-12795, "Reactor Cavity Neutron Measurement Program for Point Beach Unit 2," Rev. 3, August 1995.

5.4 EPRI TR-107450, "P-T Calculator for Windows, Version 3.0," Revision 0, December,1998.

5.5 Westinghouse Report, "Pressure Mitigating Systems Transient Analysis Results," July 1977.

5.6 Westinghouse Report, "Supplement to the July 1977 Report, Pressure Mitigating Systems Transient Analysis Results," September, 1977.

5.7 Wisconsin Electric Calculation 2000-0001, Revision 0, RCS P-T Limits and LTOP Setpoints Applicable through 32.2 EFPY –Unit 1 and 34.0 EFPY – Unit 2.

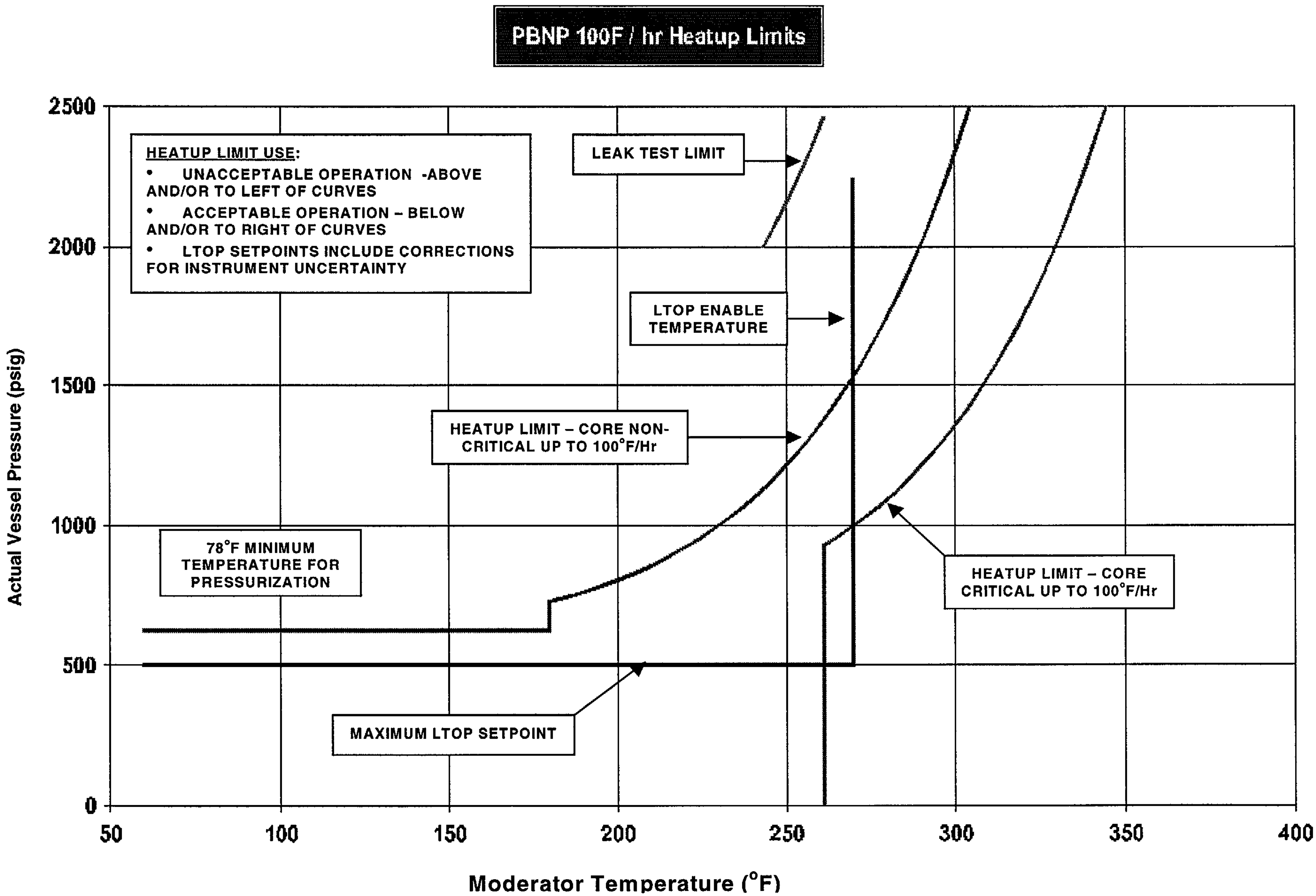


Figure 1. RCS Pressure-Temperature Limits for Heatup

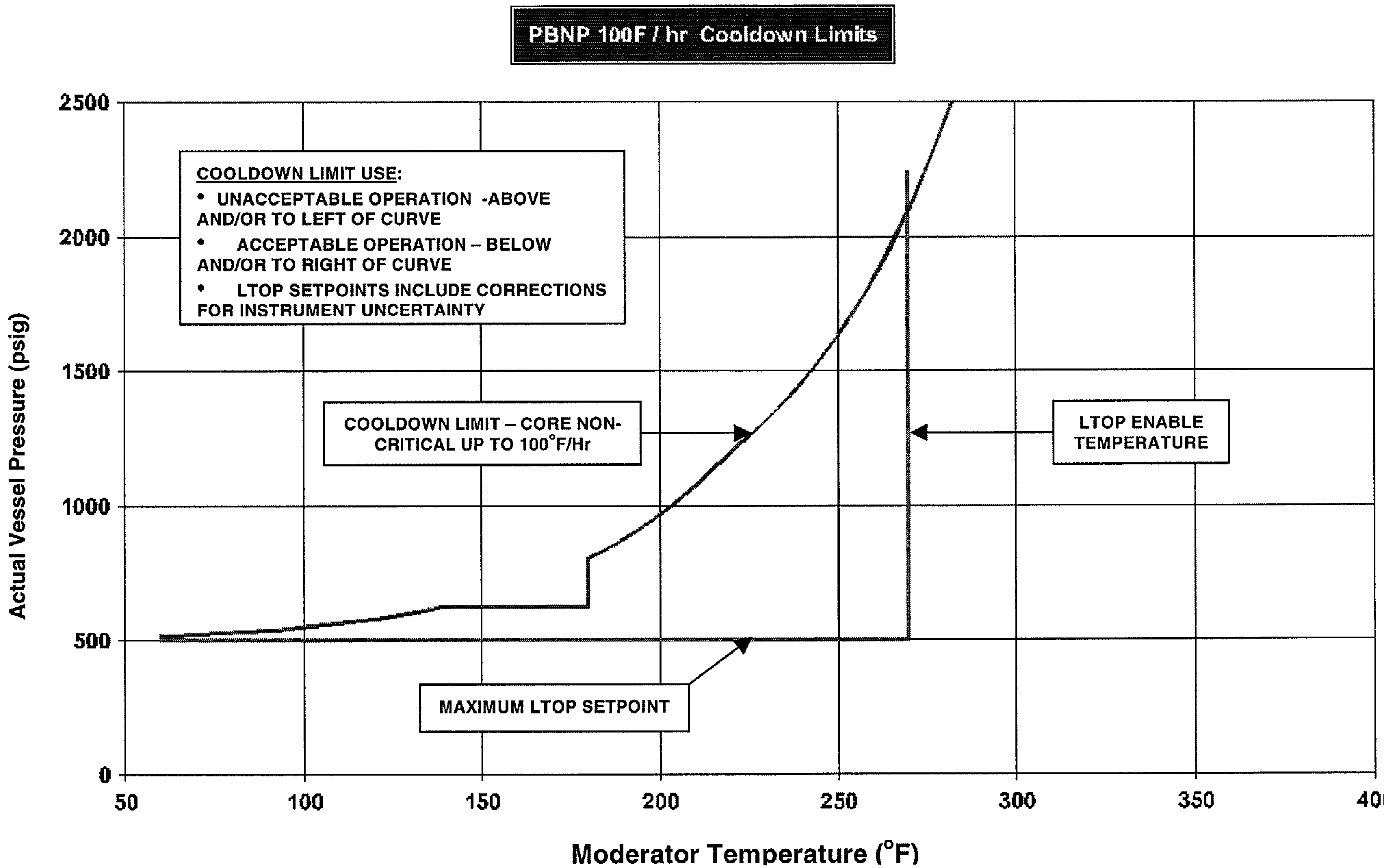


Figure 2. RCS Pressure-Temperature Limits for Cooldown

Table 1 Point Beach Nuclear Plant, Unit 1
Reactor Vessel Surveillance Capsule Removal Schedule

Capsule Identification Letter	Approximate Removal Date*
V	September 1972 (actual)
S	December 1975 (actual)
R	October 1977 (actual)
T	March 1984 (actual)
P	April 1994 (actual)
N	Standby

* The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

Table 2 Point Beach Nuclear Plant, Unit 2
Reactor Vessel Surveillance Capsule Removal Schedule

Capsule Identification Letter	Approximate Removal Date*
V	November 1974 (actual)
T	March 1977 (actual)
R	April 1979 (actual)
S	October 1990 (actual)
P	June 1997 (actual)
N	Standby

* The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

Table 3. Point Beach Unit 1 RPV Beltline 32.2 EFPY Fluence Values

Based on WCAP-12794, "Reactor Cavity Neutron Measurement Program For Wisconsin Electric Power Company Point Beach Unit 1," Rev. 4, February 2000. Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 32 EFPY, due to changes in core design at certain points in the operating history of the unit. As intermediate input to further calculations, these values are not rounded in accordance with ASTM E29 (Ref. 11).

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad *(D)*

Component Description	Heat or Heat/Lot	32 EFPY Inside Surface Fluence (E19 n/cm^2)	32.2 EFPY Inside Surface Fluence (E19 n/cm^2) *(A)*	32.2 EFPY 1/4T Fluence (E19 n/cm^2) *(B)*	32.2 EFPY 1/4T Fluence Factor *(C)*	32.2 EFPY 3/4T Fluence (E19 n/cm^2) *(B)*	32.2 EFPY 3/4T Fluence Factor *(C)*
Nozzle Belt Forging	122P237	0.547	0.550	0.3724	0.7269	0.1707	0.5322
Intermediate Shell Plate	A9811-1	2.64	2.65	1.794	1.160	0.8225	0.9452
Lower Shell Plate	C1423-1	2.24	2.25	1.523	1.116	0.6983	0.8993
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	0.547	0.550	0.3724	0.7269	0.1707	0.5322
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	1.74	1.75	1.185	1.047	N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	1.74	1.75	N/A	N/A	0.5431	0.8293
Intermed. to Lower Shell Circ. Weld (100%)	71249 (SA-1101)	2.24	2.25	1.523	1.116	0.6983	0.8993
Lower Shell Long Seam (100%)	61782 (SA-847)	1.54	1.55	1.049	1.013	0.4811	0.7960

Footnotes:

(A) Interpolation of neutron exposure (in units of E19 n/cm^2, E>1 MeV) to a particular value of effective full power years (EFPY) is performed based on WCAP-12794, Revision 4. For example, for the nozzle belt forging, heat no. 122P237,

$$\text{fluence} = 0.547 + \left(\frac{0.796 - 0.547}{48\text{ EFPY} - 32\text{ EFPY}}\right) \times (32.2\text{ EFPY} - 32.0\text{ EFPY}) = 0.550\text{ E19 n/cm}^2$$

(B) From an inside surface fluence value (not including cladding), fluence is attenuated to a desired thickness using equation (3) of Regulatory Guide 1.99, Revision 2: $f = f_{surf} \times e^{-0.24x}$, where f_{surf} is expressed in units of E19 n/cm^2, E > 1MeV, and x is the desired depth in inches into the vessel wall. For example, for the nozzle belt forging, heat no. 122P237, at 32.2 EFPY, at a depth of ¼ of the 6.5" vessel wall (1.625"), $f = 0.550 \times e^{-0.24(1.625)} = 0.3724$ E19 n/cm^2.

(C) The dimensionless fluence factor is calculated using the fluence factor formula from equation (2) of Regulatory Guide 1.99, Revision 2: $ff = f^{(0.28 - 0.10 \log f)}$, where f is the fluence in units of E19 n/cm^2. For example, the 32.2 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 122P237, $ff = 0.3724^{(0.28 - 0.10 \log 0.3724)} = 0.7269$.

(D) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969 (Ref. 12).

Table 4. Point Beach Unit 2 RPV Beltline 34.0 EFPY Fluence Values

Based on WCAP-12795, "Reactor Cavity Neutron Measurement Program For Wisconsin Electric Power Company Point Beach Unit 2," Rev. 3, August 1995. Note that the estimated fluence at a specific point in time is not linearly interpolated between zero and the estimated fluence at 32 EFPY, due to changes in core design at certain points in the operating history of the unit. As intermediate input to further calculations, these values are not rounded in accordance with ASTM E29 (Ref. 11).

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad *(D)*

Component Description	Heat or Heat/Lot	32 EFPY Inside Surface Fluence (E19 n/cm^2)	34.0 EFPY Inside Surface Fluence (E19 n/cm^2) *(A)*	34.0 EFPY 1/4T Fluence (E19 n/cm^2) *(B)*	34.0 EFPY 1/4T Fluence Factor *(C)*	34.0 EFPY 3/4T Fluence (E19 n/cm^2) *(B)*	34.0 EFPY 3/4T Fluence Factor *(C)*
Nozzle Belt Forging	123V352	0.548	0.5775	0.3910	0.7399	0.1792	0.5435
Intermediate Shell Forging	123V500	3.01	3.174	2.149	1.208	0.9851	0.9958
Lower Shell Forging	122W195	2.52	2.654	1.797	1.161	0.8237	0.9456
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	0.548	0.5775	0.3910	0.7399	0.1792	0.5435
Intermed. to Lower Shell Circ. Weld (100%)	72442 (SA-1484)	2.49	2.606	1.764	1.156	0.8088	0.9405

Footnotes:

(A) Interpolation of neutron exposure (in units of E19 n/cm^2, E>1 MeV) to a particular value of effective full power years (EFPY) is performed based on WCAP-12795, Revision 3. For example, for the nozzle belt forging, heat no. 123V352,

$$\text{fluence} = 0.548 + \left(\frac{0.784 - 0.548}{48\ \text{EFPY} - 32\ \text{EFPY}}\right) \times (34\ \text{EFPY} - 32\ \text{EFPY}) = 0.5775\text{E19 n/cm}^2$$

(B) From an inside surface fluence value (not including cladding), fluence is attenuated to a desired thickness using equation (3) of Regulatory Guide 1.99, Revision 2: $f = f_{surf} \times e^{-0.24x}$, where f_{surf} is expressed in units of E19 n/cm^2, E > 1MeV, and x is the desired depth in inches into the vessel wall. For example, for the nozzle belt forging, heat no. 123V352, at 34.0 EFPY, at a depth of ¼ of the 6.5" vessel wall (1.625"), $f = 0.5775 \times e^{-0.24(1.625)} = 0.3910$ E19 n/cm^2.

(C) The dimensionless fluence factor is calculated using the fluence factor formula from equation (2) of Regulatory Guide 1.99, Revision 2: $ff = f^{(0.28 - 0.10 \log f)}$, where f is the fluence in units of E19 n/cm^2. For example, the 34.0 EFPY 1/4T fluence factor for nozzle belt forging, heat no. 123V352, $ff = 0.3910^{(0.28 - 0.10 \log 0.3910)} = 0.7399$.

(D) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970 (Ref. 13).

Table 5. Point Beach Unit 1 RPV 1/4T Beltline Material Adjusted Reference Temperatures at 32.2 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 14), including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	1/4T 32.2 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	0.7269	55.97	0	17	34	140
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	1.160	102.08	26.9	17	63.64	167
"	"	"			79.3	Surv. Data *(B)*	"	91.99	"	8.5	56.42	149
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	1.116	61.71	26.9	17	63.64	126
"	"	"			35.8	Surv. Data *(B)*	"	39.95	"	8.5	56.42	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.7269	110.78	19.7	28	68.47	174
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	1.047	144.70	19.7	28	68.47	208
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	N/A	N/A	19.7	N/A	N/A	N/A
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table *(C)*	1.116	187.04	0	28	56	
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	1.013	159.45	19.7	28	68.47	223
"	"	"			163.3	Surv. Data *(D)*	"	165.42	"	14	48.34	209 *(G)*

Footnotes:

(A) See Table 1.

(B) Credible Surveillance Data; see BAW-2325 for evaluation..

(C) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).

(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," (Ref. 15) utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.

(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 122P237, ART = 50 +(77 x 0.7269) + 34 = 140°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.

(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.

(G) By inspection, these are the limiting material properties.

Table 6. Point Beach Unit 2 RPV 1/4T Beltline Material Adjusted Reference Temperatures at 34.0 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998 (Ref. 14), including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	1/4T 34.0 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	0.7399	56.23	0	17	34	130
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table *(B)*	1.208	70.06	0	17	34	144 *(G)*
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	1.161	35.99	0	17	34	110
"	"	"			42.8	Surv. Data *(C)*	"	49.69	"	8.5	17	107
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table *(H)*	0.7399	125.78	17	28	65.51	135
Intermed. To Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table *(D)*	1.156	208.08	19.7	28	68.47	272 *(G)*

Footnotes:

(A) See Table 2.
(B) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (34°F).
(C) Credible surveillance data; see BAW-2325 for evaluation.
(D) Non-credible surveillance data; Table CF value based on best-estimate chemistry is higher than best fit calculated using surveillance data, and therefore conservative.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 +(76 x 0.7399) + 34 = 130°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
(G) By inspection, these are the limiting material properties.
(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997 (Ref. 16).

Table 7. Point Beach Unit 1 RPV 3/4T Beltline Material Adjusted Reference Temperatures at 32.2 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998, including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	3/4T 32.2 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	122P237	+50	0.11	0.82	77	Table	0.5322	40.98	0	17	34	125
Intermediate Shell Plate	A9811-1	+1	0.20	0.06	88	Table	0.9452	83.18	26.9	17	63.64	148
"	"	"			79.3	Surv. Data *(B)*	"	74.95	"	8.5	56.42	132
Lower Shell Plate	C1423-1	+1	0.12	0.07	55.3	Table	0.8993	49.73	26.9	17	63.64	114
"	"	"			35.8	Surv. Data *(B)*	"	32.19	"	8.5	56.42	90
Nozzle Belt to Intermed. Shell Circ Weld (100%)	8T1762 (SA-1426)	-5	0.19	0.57	152.4	Table	0.5322	81.11	19.7	28	68.47	145
Intermediate Shell Long Seam (ID 27%)	1P0815 (SA-812)	-5	0.17	0.52	138.2	Table	N/A	N/A	19.7	N/A	N/A	N/A
Intermediate Shell Long Seam (OD 73%)	1P0661 (SA-775)	-5	0.17	0.64	157.6	Table	0.8293	130.70	19.7	28	68.47	194 *(G)*
Intermed. To Lower Shell Circ. Weld (100%)	71249 (SA-1101)	+10	0.23	0.59	167.6	Table *(C)*	0.8993	150.72	0	28	56	217 *(G)*
Lower Shell Long Seam (100%)	61782 (SA-847)	-5	0.23	0.52	157.4	Table	0.7960	125.29	19.7	28	68.47	189
"	"	"			163.3	Surv. Data *(D)*	"	129.99	"	14	48.34	173

Footnotes:

(A) See Table 1.
(B) Credible Surveillance Data; see BAW-2325 for evaluation.
(C) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between ratio-adjusted measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (56°F).
(D) Credible Surveillance Data; see WE Calculation Addendum 98-0156-00-A, "Evaluation of New Surveillance Data on Chemistry Factor for Weld Wire Heat 61782, Point Beach Unit 1," utilizing latest time-weighted temperature data for Point Beach Unit 1, which supersedes BAW-2325.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 122P237, ART = 50 +(77 x 0.5322) + 34 = 125°F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, 132-Inch I.D. Reactor Pressure Vessel, Babcock & Wilcox, September 1969.
(G) By inspection, these are the limiting material properties.

Table 8. Point Beach Unit 2 RPV 3/4T Beltline Material Adjusted Reference Temperatures at 34.0 EFPY

Unless otherwise noted, all ART input data obtained from BAW-2325, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," May 1998, including the most recent best-estimate chemistry values for welds, applying current B&WOG mean-of-the-sources approach. All beltline materials are included for comparison.

Vessel Manufacturer:	Babcock & Wilcox and Combustion Engineering
Plate and Weld Thickness (without cladding):	6.5", without clad *(F)*

Component Description	Heat or Heat/Lot	Initial RT_{NDT} (°F)	%Cu	%Ni	CF	CF Method	3/4T 34.0 EFPY Fluence Factor *(A)*	ΔRT_{NDT} (°F)	σ_I	σ_Δ	Margin (°F)	ART (°F) *(E)*
Nozzle Belt Forging	123V352	+40	0.11	0.73	76	Table	0.5435	41.31	0	17	34	115
Intermediate Shell Forging	123V500	+40	0.09	0.70	58	Table *(B)*	0.9958	57.76	0	17	34	132 *(G)*
Lower Shell Forging	122W195	+40	0.05	0.72	31	Table	0.9456	29.31	0	17	34	103
"	"	"			42.8	Surv. Data *(C)*	"	40.47	"	8.5	17	97
Nozzle Belt to Intermed. Shell Circ Weld (100%)	21935	-56	0.18	0.70	170	Table *(H)*	0.5435	92.40	17	28	65.51	102
Intermed. To Lower Shell Circ. Weld (100%)	72442 (SA-1484)	-5	0.26	0.60	180	Table *(D)*	0.9405	169.29	19.7	28	68.47	233*(G)*

Footnotes:

(A) See Table 2.
(B) Non-credible surveillance data; see BAW-2325 for evaluation. Table CF conservative because difference between measured ΔRT_{NDT} and predicted ΔRT_{NDT} based on Table CF is less than 2σ (34°F).
(C) Credible surveillance data; see BAW-2325 for evaluation.
(D) Non-credible surveillance data; Table CF value based on best-estimate chemistry is higher than best fit calculated using surveillance data, and therefore conservative.
(E) Adjusted reference temperature (ART) calculated per Regulatory Guide 1.99, Rev. 2. ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin, where ΔRT_{NDT} = Chemistry Factor x Fluence Factor, and Margin = $2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$, with σ_I defined as the standard deviation of the Initial RT_{NDT}, and σ_Δ defined as the standard deviation of ΔRT_{NDT}. For example, for nozzle belt forging, heat no. 123V352, ART = 40 +(76 x 0.5435)+ 34 = 115F. Calculated ART values are rounded to the nearest °F in accordance with the rounding-off method of ASTM Practice E29.
(F) Instruction Manual, Reactor Vessel, Point Beach Nuclear Plant No. 2, Combustion Engineering, CE Book #4869, October 1970.
(G) By inspection, these are the limiting material properties.
(H) Table CF value based on best-estimate chemistry data from CEOG Report "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," CE NPSD-1039, Revision 2, Final Report, June 1997.