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WCAP-14040, Rev. 3
Project Number 694

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

Attention: Chief, Information Management Branch,
Division of Inspection and Support Programs

Subject: Westinghouse Owners Group
Transmittal of WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," (MUHP-3073)

Reference: 1) Westinghouse Owners Group Letter, R. Bryan to Document Control Desk, "Transmittal of WCAP-15315, Rev. 1, 'Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants'," OG-02-019, May 23, 2002.

This letter transmits five copies of the WCAP-14040, Rev. 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC review and approval. WCAP-14040-A, Rev. 2, was approved by the NRC on October 16, 1995, and contains a methodology for developing Reactor Coolant System (RCS) Pressure-Temperature (P-T) limit curves and Cold Overpressure Mitigation System (COMS) setpoints and enable temperature that can be referenced by licensees in the Administrative Controls Section of the Technical Specifications when relocating P-T limit curves, COMS setpoints and COMS enable temperature to a Pressure and Temperature Limits Report (PTLR).

Several ASME Nuclear Code Cases (N-588, N-640, and N-641) associated with the development of P-T limit curves and the COMS enable temperature have been approved by the ASME subsequent to the approval of WCAP-14040-NP-A, Rev. 2 in October 1995. Exemption requests have been approved by the NRC to allow the use of these ASME Nuclear Code Cases in the development of P-T limit curves.

WCAP-14040, Rev. 3 has been revised to incorporate these approved ASME Nuclear Code Cases into the methodology used to develop the P-T limit curves and COMS enable temperature that is contained in WCAP-NP-A, Rev. 2.

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WCAP-14040, Rev. 3 also contains an option to develop the P-T limit curves without the flange requirement, currently required by 10CFR50 Appendix G. The option to develop P-T limit curves without the flange requirement would require NRC approval of an exemption request, or rulemaking to eliminate the requirement. A Petition for Rulemaking to eliminate the flange requirement of 10CFR50 Appendix G from the P-T limit curves was submitted by Westinghouse Electric Co. in November 1999.

The technical justification for eliminating the flange requirement is contained in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," Rev. 0. WCAP-15315, Rev. 0 was submitted to the NRC with the Petition for Rulemaking to eliminate the flange requirement of 10CFR50 Appendix G by Westinghouse Electric Co., in November 1999. WCAP-15315, Rev. 1 contains the additional information for eliminating the flange requirement as requested by the NRC during a meeting between Westinghouse and the NRC on August 28, 2001. WCAP-15315, Rev. 1 is also being submitted for NRC review as justification for eliminating the flange requirement of 10CFR50 Appendix G (Reference 1).

The WOG is submitting WCAP-14040, Rev. 3 under the NRC licensing topical report program for review and acceptance for referencing in licensing actions. The objective is that once approved, each WOG member can reference a single methodology in the Administrative Controls Section of the Technical Specifications when relocating or revising P-T limit curves and COMS setpoints and enable temperature in a PTLR.

The WOG requests that the NRC complete the review of WCAP-14040, Rev. 3, by September 30, 2002. Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, “Processing Request for Reviews of Topical Reports,” the WOG requests that the NRC provide an estimate of the review hours, and target dates for any Request(s) for Additional Information and for completion of the Safety Evaluation for WCAP-14040, Rev. 3.

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Very truly yours,

Robert H Bryan

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enclosures

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WCAP - 14040
Revision 3

Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves

Westinghouse Electric Company LLC

WCAP-14040

Revision 3

Methodology Used to Develop Cold Overpressure Mitigating System Setpoints And RCS Heatup and Cooldown Limit Curves

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1.0 INTRODUCTION

1.1 BACKGROUND

The concept of a Pressure and Temperature Limits Report (PTLR) was introduced into the Technical Specifications during the development of NUREG 1431[1], Standard Technical Specifications for Westinghouse PWRs and is consistent with the philosophy of NRC Generic Letter 88-16[2]. The PTLR is similar to the Core Operating Limits Report (COLR), which is currently licensed for several plants and also contained in NUREG 1431. The COLR contains core related limit values which may change from cycle to cycle as they are related to a cycle specific core design. In the same way, a PTLR contains reactor vessel material related limits which may change every fluence cycle as they are related to reactor vessel material and strength. Implementation of the PTLR will allow licensees to relocate their RCS heatup and cooldown curves and COMS setpoints currently contained in the Technical Specifications to the PTLR. Additionally, the Vessel Fluence and Materials tables contained in the Technical Specifications or Bases can be relocated to licensee controlled documents. This process will allow changes to these tables, figures and values to be made without making a License Amendment Request (LAR). These figures are typically revised due to changes in the nil ductility reference temperature (RT_{NDT}), regulations and surveillance capsule withdrawal.

1.2 PURPOSE OF TOPICAL REPORT

In order to implement the PTLR, the analytical methods used to develop the pressure and temperature limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the Technical Specifications. The purpose of this report is to provide the current Westinghouse methodology for developing the RCS heatup and cooldown curves and COMS setpoints. When approved by the NRC, this methodology may be referenced by licensees to implement the PTLR.

This topical report does not provide all of the methodologies which can be used to develop RCS heatup and cooldown curves and COMS setpoints, but rather methodologies that can be referenced by licensees when approved by the NRC to license the PTLR concept.

1.3 CONTENT OF TOPICAL REPORT

This report contains the methodology used to develop the RCS heatup and cooldown curves in Section 2.0 and the methodology used to develop the COMS setpoints in Section 3.0. The methodology used to develop the COMS enable temperature is also discussed in Section 3.0.

2.0 PRESSURE-TEMPERATURE LIMIT CURVES

2.1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the beltline region of the reactor vessel. The most limiting RT_{NDT} of the material in the core (beltline) region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}). The unirradiated RT_{NDT} is defined as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (both normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron irradiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 (Radiation Embrittlement of Reactor Vessel Materials)[3]. Regulatory Guide 1.99, Revision 2, is used for the calculation of adjusted reference temperature (ART) values (irradiated RT_{NDT} with margins for uncertainties) at 1/4t and 3/4t locations. "t" is the thickness of the vessel at the beltline region measured from the clad/base metal interface (Note, thickness of cladding is neglected as specified in the ASME Code, Section III, paragraph NB-3122.3). Using the adjusted reference temperature values, pressure-temperature limit curves are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by Appendix G, Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code[5]. The procedure for establishing the pressure-temperature limits is entirely deterministic. The conservatisms included in the limits are (but not limited to):

- An assumed flaw in the wall of the reactor vessel has a depth equal to 1/4 of the thickness of the vessel wall and a length equal to 1-1/2 times the vessel wall thickness,

- A factor of 2 is applied to the membrane stress intensity factor (K_{IM}),

- 2-sigma margins are applied in determining the adjusted reference temperature (ART), and

- The limiting toughness is based upon a reference value [K_{Ia}, which is a lower bound of the dynamic crack initiation or arrest toughnesses, or K_{Ic}, which is a lower bound of static feature toughness].

This section describes the methodology used by Westinghouse to develop the allowable pressure-temperature relationships for normal plant heatup and cooldown rates that are included in the Pressure-Temperature Limits Report (PTLR). First, the methodology describing how the neutron fluence is calculated for the reactor vessel beltline materials is provided. Next, sections describing fracture toughness properties, adjusted reference temperature calculation, criteria for allowable pressure-temperature relationships, and pressure-temperature curve generation are provided.

2.2 NEUTRON FLUENCE METHODOLOGY

The methodology used to provide best estimate neutron exposure evaluations for the reactor pressure vessel is based on the underlying philosophy that, in order to minimize the uncertainties associated with vessel exposure projections, plant specific neutron transport calculations must be supported by benchmarking of the analytical approach, comparison with industry wide power reactor data bases of surveillance capsule and reactor cavity dosimetry, and, ultimately, by validation with plant specific surveillance capsule and reactor cavity dosimetry databases. That is, as a progression is made from the use of a purely analytical approach tied to experimental benchmarks to an approach that makes use of industry and plant specific power reactor measurements to remove potential biases in the analytical method, knowledge regarding the neutron environment applicable to a specific reactor vessel is increased and the uncertainty associated with vessel exposure projections is minimized.

2.2.1 Plant Specific Transport Calculations

Fast neutron exposure calculations for the reactor geometry are carried out using both forward and adjoint discrete ordinates transport techniques. A single forward calculation provides the relative energy distribution of neutrons for use as input to neutron dosimetry evaluations as well as for use in relating measurement results to the actual exposure at key locations in the pressure vessel wall. A series of adjoint calculations, on the other hand, establish the means to compute absolute exposure rate values using fuel cycle specific core power distributions; thus, providing a direct comparison with all dosimetry results obtained over the operating history of the reactor.

In combination, the absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra distributions from the forward calculation provide the means to:

1. Evaluate neutron dosimetry from surveillance capsule and reactor cavity locations.

2. Enable a direct comparison of analytical prediction with measurement.

3. Determine plant specific bias factors to be used in the evaluation of the best estimate exposure of the reactor pressure vessel.

4. Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves.

The forward transport calculation for the reactor is carried out in r,θ geometry using the DORT two-dimensional discrete ordinates code[11] and the BUGLE-93 cross-section library[12]. The BUGLE-93 library is a 47 neutron group, ENDFB-VI based, data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering is treated with a P_3 expansion of the scattering cross-sections and the angular discretization is modeled with an S_8 order of angular quadrature. The reference forward calculation is normalized to a core midplane power density characteristic of operation at the stretch rating for the reactor.

The spatial core power distribution utilized in the reference forward calculation is derived from statistical studies of long-term operation of Westinghouse 2-, 3-, and 4-loop plants. Inherent in the development of this reference core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2σ uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power is used.

Due to the use of this bounding spatial power distribution, the results from the reference forward calculation establish conservative exposure projections for reactors of each design operating at the stretch rating. Since it is unlikely that actual reactor operation would result in the implementation of a power distribution at the nominal $+2\sigma$ level for a large number of fuel cycles and, further, because of the widespread implementation of low leakage fuel management strategies, the fuel cycle specific calculations for specific reactors generally result in exposure rates well below these conservative predictions.

All adjoint analyses are also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the BUGLE-93 library. Adjoint source locations are chosen at several key azimuths on the pressure vessel inner radius. In addition, adjoint calculations were carried out for sources positioned at the geometric center of all surveillance capsules and, where applicable, at dosimetry locations in the reactor cavity. Again, these calculations are run in r,θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, $\phi(E > 1.0\ \text{MeV})$.

The importance functions generated from these individual adjoint analyses provide the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yield absolute predictions of neutron exposure at the locations of interest for each of the operating fuel cycles; and, establish the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles.

Having the importance functions and appropriate core source distributions, the response of interest can be calculated as:

$$\varphi(R_o, \theta_o) = \int_r \int_\theta \int_E \ell\,(r, \theta, E)\ r\ dr\ d\theta\ dE$$

where:

$\phi(R_o,\theta_o)$ = Neutron flux ($E > 1.0$ MeV) at radius R_o and azimuthal angle θ_o.

$\ell\,(r,\theta,E)$ = Adjoint importance function at radius r, azimuthal angle θ, and neutron source energy E.

$S(r,\theta, E)$ = Neutron source strength at core location r,θ and energy E.

It is important to note that the cycle specific neutron source distributions, $S(r,\theta,E)$, utilized with the adjoint importance functions, $\ell\,(r,\theta,E)$, permit the use not only of fuel cycle specific spatial variations of fission rates within the reactor core; but, also allow for the inclusion of the effects of the differing neutron yield per fission and the variation in fission spectrum introduced by the build-in of plutonium isotopes as the burnup of individual fuel assemblies increases.

2.2.2 Determination of Best Estimate Pressure Vessel Exposure

The best estimate exposure of the reactor pressure vessel is developed using a combination of absolute plant specific transport calculations based on the methodology discussed in Section 2.2.1 and plant specific measurement data from surveillance capsule and reactor cavity irradiations. In particular, the best estimate vessel exposure is obtained from the following relationship:

$$\Phi_{Best\ Est} = K\ \Phi_{Calc.}$$

Where:

$\Phi_{Best\ Est}$ = The best estimate fast neutron exposure at the location of interest.

K = The plant specific measurement/calculation (M/C) bias factor derived from all available surveillance capsule and reactor cavity dosimetry data.

$\Phi_{calc.}$ = The absolute calculated fast neutron exposure at the location of interest.

The approach defined in the above equation is based on the premise that the measurement data represent the most accurate plant specific information available at the locations of the dosimetry; and, further that the use of the measurement data on a plant specific basis essentially removes biases present in the analytical approach and mitigates the uncertainties that would result from the use of analysis alone. That is, at the measurement points the uncertainty in the best estimate exposure is dominated by the uncertainties in the measurement process. At locations within the pressure vessel wall, additional uncertainty is incurred due to the analytically determined relative ratios among the various measurement points and locations within the pressure vessel wall.

The implementation of this approach acts to remove plant specific biases associated with the definition of the core source, actual vs. assumed reactor dimensions, and operational variations in water density within the reactor. As a result, the overall uncertainty in the best estimate exposure projections within the vessel wall depend on the individual uncertainties in the measurement process, the uncertainty in the dosimetry location, and in the uncertainty in the calculated ratio of the neutron exposure at the point of interest to that at the measurement location.

The uncertainties in the measured flux are derived directly from the results of least squares evaluations of dosimetry data. The positioning uncertainties are taken from parametric studies of sensor position performed as part of an analytical sensitivity evaluation of the reactor design. The uncertainties in the exposure ratios relating dosimetry results to positions within the vessel wall are based on analytical sensitivity studies of the vessel thickness tolerance for cavity measurement data and on downcomer water density variations and vessel inner radius tolerance for the surveillance capsule measurements.

In general, pressure-temperature limits are generated for a particular EFPY (effective full power years) of plant operation. In some cases the fluence at the EFPY of interest is obtained directly from the dosimetry analysis. However, if the fluence is not available from the dosimetry analysis, the peak vessel inner radius fluence at the EFPY of interest is calculated as follows:

$$f = F\,C\,E \qquad (2.2\text{-}2)$$

Where:

f = the peak vessel inner radius fluence at the EFPY of interest (n/cm^2 (E > 1.0 MeV))

F = Best estimate peak flux at the pressure vessel inner radius (n/cm^2 - sec (E > 1.0 MeV))

C = seconds per year = 3.16×10^7 sec/yr

E = EFPY of interest

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[(4)], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[(8)]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[(9)] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1. Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{Cv}$ - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness requirements in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) must be provided for an exemption from the regulations to be granted by the NRC.

2.4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The adjusted reference temperature (ART) for each material in the beltline region is calculated in accordance with Regulatory Guide 1.99, Revision 2[3]. The most limiting ART values (i.e., highest value at 1/4t and 3/4t locations) are used in determining the pressure-temperature limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{2.4-1}$$

IRT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8] and calculated per Section 2.3. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for that class of material can be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the shift in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF\ f^{(0.28 - 0.10 \log f)} \tag{2.4-2}$$

CF (°F) is the chemistry factor and is a function of copper and nickel content. CF is given in Table 1 of Reference 3 for weld metal and in Table 2 in Reference 3 for base metal (Position 1.1 of Regulatory Guide 1.99, Revision 2). In Tables 1 and 2 of Reference 3 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in Regulatory Guide 1.99, Revision 2, Paragraph B.4) become available they may be used to calculate the chemistry factor per Position 2.1 of Regulatory Guide 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n} [A_i\ f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n} [f_i^{(0.28-0.10\log f_i)}]^2} \tag{2.4-3}$$

Where "n" is the number of surveillance data points, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point.

If Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a higher value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a lower value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4t, the following formula is used to attenuate the fast neutron fluence (E> 1 MeV) at the specified depth.

$$f = f_{surface}\, e^{(-0.24x)} \qquad (2.4\text{-}4)$$

where $f_{surface}$ 10^{19} n/cm^2, E > 1 MeV) is the value, calculated per Section 2.2, of the neutron fluence at the base metal surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (2.4-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by equation (2.4-2) plus two standard deviations ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by Appendix G to 10 CFR Part 50[4]. Margin is calculated by the following equation:

$$\text{Margin} = 2\,[(\sigma_I^2 + \sigma_\Delta^2)]^{0.5} \qquad (2.4\text{-}5)$$

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 ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per Regulatory Guide 1.99, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

2.5 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME Code requirements[5] for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates specify that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, the fracture toughness for the metal temperature at that time. Two values of fracture toughness may be used, K_{Ia} or K_{Ic}.

K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G, to Section XI of the ASME Code[5]. (Note that in Appendix G, to Section III of the ASME Code, the reference fracture toughness is denoted as K_{IR}, whereas in Appendix G of Section XI, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.) The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \exp\,[0.0145\,(T - RT_{NDT} + 160)] \qquad (2.5\text{-}1)$$

where,

K_{Ia} = lower bound of dynamic and crack arrest toughness as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RT_{NDT} is the adjusted reference temperature (ART) of Section 2.4.

(Note: In the calculation of K_{Ia}, a slightly lower (0.8%) and more conservative value is obtained using a constant of 1.223, instead of 1.233, which would give a higher allowable limit. However, a value of 1.223 is consistent with Welding Research Council Bulletin 175, and NRC Standard Review Plan 5.3.2.)

K_{Ic} is also obtained from Section XI of the ASME Code, for example in Appendix A, and is a lower bound of static fracture toughness. Since heatup and cooldown is a slow process, static properties are appropriate. The K_{Ic} curve is given by the following expression:

$$K_{Ic} = 33.20 + 20.734 \exp [0.0200 (T - RT_{NDT})] \tag{2.5-2}$$

The use of the K_{Ic} curve (Section XI, Appendix A) as a basis for developing P-T limit curves is currently contained in ASME Code Case N640. Use of the K_{Ic} fracture toughness will yield less limiting P-T curves, which is clearly a benefit.

However, the use of Code Case 640 presently includes a restriction on the setpoints for the Cold Overpressure Mitigation System (COMS). This maximum pressure for the COMS system is 100% of the pressure allowed by the P-T limit curves. This essentially disallows the use of Code Case N514 in these circumstances, meaning that the COMS system must protect to the actual P-T limit curve, rather than 110 percent, as allowed by Code Case N514.

The use of Code Case N640 has not yet been endorsed by the NRC, and therefore use of this Code Case will require an exemption under 10CFR50.60 paragraph (b), pertaining to proposed alternatives to the requirements of Appendices G and H.

The governing equation for generating pressure-temperature limit curves is defined in Appendix G of the ASME Code[5] as follows:

$$C\,K_{IM} + K_{IT} < \text{Reference Fracture Toughness} \tag{2.5-3}$$

where,

K_{IM} = stress intensity factor caused by membrane (pressure) stress,

K_{IT} = stress intensity factor caused by the thermal gradients through the vessel wall,

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

C = 1.5 for hydrostatic and leak test conditions when the reactor core is not critical

Reference Fracture Toughness = K_{Ia} or K_{Ic}, as discussed above

(Note: K_{IT} is set to zero for hydrostatic and leak test calculations since these tests are performed at isothermal conditions).

At specific times during the heatup or cooldown transient, the reference fracture toughness is determined by the metal temperature at the tip of the postulated flaw (the postulated flaw has a depth of one-fourth of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, paragraph G-2120), the appropriate value for RT_{NDT} at the same location, and the reference fracture toughness equation (2.5-1 or 2.5-2). The thermal stresses resulting from the temperature gradients through the vessel wall and the corresponding (thermal) stress intensity factor, K_{IT}, for the reference flaw are calculated as described in Section 2.6. From Equation (2.5-3), the limiting pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated as described in Section 2.6.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference 1/4t (t = reactor vessel wall thickness) flaw of Appendix G, Section XI to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the vessel wall because the thermal gradients that increase with increasing cooldown rates produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Allowable pressure-temperature curves are generated for steady-state (zero rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4t vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the temperature difference across the wall developed during cooldown results in a higher value of reference fracture toughness at the 1/4t location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in reference fracture toughness exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4t location and, therefore, allowable pressures could be lower if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4t flaw at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the reference fracture toughness for the inside 1/4t flaw during heatup is lower than the reference fracture toughness for the same flaw during steady-state conditions at the same coolant temperature. However, conditions may exist so that the effects of compressive thermal stresses and lower reference fracture toughness do not offset each other and the pressure-temperature curve based on finite heatup rates could become limiting. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature, the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained for the inside 1/4t flaw.

The third portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case of a 1/4t outside surface flaw. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and coolant temperature during the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate is analyzed on an individual basis.

Following the generation of the three pressure-temperature curves, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state data and finite heatup rate data for both inside and outside surface flaws. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is not possible to predict which condition is most limiting because of local differences in irradiation (RT_{NDT}), metal temperature and thermal stresses. With the composite curve, the pressure limit is at all times based on analysis of the most critical situation.

Finally, the 1983 Amendment to 10CFR50[4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. In addition, when the core is critical, the pressure-temperature limits for core operation (except for low power physics tests) require that the reactor vessel be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown. These limits are incorporated into the pressure-temperature limit curves wherever applicable.

A petition for rulemaking to eliminate the flange requirement contained in 10CFR50 Appendix G was submitted to the NRC by Westinghouse in November 1999. Until 10CFR50 Appendix G is revised to eliminate the flange requirement, it must be included in the P-T limits, unless an exemption request is submitted and approved by the NRC.

Figure 2.2 shows an example of a heatup curve using a heatup rate of 60°F/Hr applicable for the first 16 EFPY. Figure 2.3 shows an example of cooldown curves using rates of 0°, 20°, 40°, 60°, and 100°F/Hr applicable for the first 16 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 2.2 and 2.3. Note that the step in these curves are due to the previously described flange requirements [4].

2.6 PRESSURE-TEMPERATURE CURVE GENERATION METHODOLOGY

2.6.1 Thermal and Stress Analyses

The time-dependent temperature solution utilized in both the heatup and cooldown analysis is based on the one-dimensional transient heat conduction equation

$$\rho C \frac{\partial T}{\partial t} = K \left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r} \right] \tag{2.6.1-1}$$

with the following boundary conditions applied at the inner and outer radii of the reactor vessel,

$$\text{at } r = r_i, \quad -K \frac{\partial T}{\partial r} = h(T - T_c) \tag{2.6.1-2}$$

$$\text{at } r = r_o, \quad \frac{\partial T}{\partial r} = 0 \tag{2.6.1-3}$$

where,

r_i = reactor vessel inner radius

r_o = reactor vessel outer radius

ρ = material density

C = material specific heat

K = material thermal conductivity

T = local temperature

r = radial location

t = time

h = heat transfer coefficient between the coolant and the vessel wall

T_c = coolant temperature

These equations are solved numerically to generate the position and time-dependent temperature distributions, $T(r,t)$, for all heatup and cooldown rates of interest.

With the results of the heat transfer analysis as input, position and time-dependent distributions of hoop thermal stress are calculated using the formula for the thermal stress in a hollow cylinder given by Timoshenko[14].

$$\sigma_\theta (r,t) = \frac{E\alpha}{1-\nu} \frac{1}{r^2} \left[\frac{r^2 + r_i^2}{r_o^2 - r_i^2} \int_{r_i}^{r_o} T(r,t) r \, dr + \int_{r_i}^{r} T(r,t) r \, dr - T(r,t) r^2 \right] \tag{2.6.1-4}$$

where,

$\sigma_\theta(r,t)$ = hoop stress at location and time t

E = modulus of elasticity

α = coefficient of linear expansion

V = Poisson's ratio

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α are evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r)r\,dr}{r_o^2 - r_i^2} \tag{2.6.1-5}$$

E and α are calculated as a function of this equivalent temperature and the Eα product in equation (2.6.1-4) is treated as a constant in the computation of hoop thermal stress.

The linear bending (σ_b) and constant membrane (σ_m) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in Appendix A, to Section XI of the ASME Code(15). These stress components are used for determining the thermal stress intensity factors, K_{IT}, as described in the following subsection.

2.6.2 Steady-State Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at the 1/4t location where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius.

The allowable pressure $P(T_c)$ is a function of coolant temperature, and the pressure temperature curve is calculated for the steady state case at the assumed 1/4t inside surface flaw. First, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t}}{2.0} \tag{2.6.2-1}$$

where,

$K_I * (T\text{-}RT_{NDT})$ = allowable reference stress intensity factor as a function of $T\text{-}RT_{NDT}$ at 1/4t. K_I* = K_{Ia} for the original Appendix G approach, or K_I* = K_{Ic} for the new approach using Code Case N640.

Next, the maximum allowable pressure stress is determined using an iterative process and either of the following two options:

Option 1: Welding Research Bulletin 175 Method

$$Q = \varphi^2 - 0.212\left(\frac{\sigma_P}{\sigma_y}\right)^2 \qquad (2.6.2\text{-}2)$$

$$\sigma_P = \frac{K_{IM(max)}}{1.1\,M_K\sqrt{\dfrac{\pi a}{Q}}} \qquad (2.6.2\text{-}3)$$

$$K_{IP} = 1.1 M_K \sigma_P \sqrt{\frac{\pi a}{Q}} \qquad (2.6.2\text{-}4)$$

where,

Q = flaw shape factor modified for plastic zone size[16],

ϕ = is the elliptical integral of the 2nd kind (ϕ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw)[16],

0.212 = plastic zone size correction factor[16],

σ_P = pressure stress,

σ_y = yield stress,

1.1 = correction factor for surface breaking flaws,

M_K = correction factor for constant membrane stress [16], M_K as function of relativeflaw depth (a/t) is shown in Figure 2.4,

a = crack depth of 1/4t,

K_{IP} = pressure stress intensity factor.

Option 2: Section XI, Appendix G, Revised Method

$$\sigma_P = \frac{K_{IM(max)}}{M_m}$$

$$K_{IP} = M_m\,\sigma_p$$

where,

M_m = 1.84 for $\sqrt{t} < 2$

= 0.918 $\sqrt{t}$ for $2 < \sqrt{t} < 3.464$

= 3.18 for $\sqrt{t} > 3.464$

The maximum allowable pressure stress is determined by incrementing σ_p from an initial value of 0.0 psi until a pressure stress is found that computes a K_{IP} value within 1.0001 of the $K_{IM(max)}$ value. After the maximum allowable σ_p is found, the maximum allowable internal pressure is determined by

$$P(T_c) = \sigma_P \left[\frac{r_o^2 - r_i^2}{r_o^2 + r_i^2}\right] \quad (2.6.2\text{-}5)$$

where,

$P(T_c)$ = calculated allowable pressure as a function of coolant temperature.

2.6.3 Finite Cooldown Rate Analyses

Option 1: WRC Bulletin 175 Method. For each cooldown rate the pressure-temperature curve is calculated at the inside 1/4t location. First, the thermal stress intensity factor is calculated for a coolant temperature at a given time using the following equation from the Welding Research Council[16]:

$$K_{IT} = [\sigma_m 1.1 M_K + \sigma_b M_B]\sqrt{\frac{\pi a}{Q}} \quad (2.6.3\text{-}1)$$

where,

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_K = correction factor for membrane stress[16] (see Figure 2.4),

M_B = correction factor for bending stress[16], M_B as a function of relative flaw depth (a/t) is shown in Figure 2.5.

The flaw shape factor Q in equation (2.6.2-6) is calculated from[16]

$$Q = \phi^2 - 0.212\left(\frac{\sigma_m + \sigma_b}{\sigma_y}\right)^2 \qquad (2.6.3\text{-}2)$$

Option 2: Section XI, Appendix G Revised Method. Alternatively the thermal stress intensity factor may be calculated using a new stress intensity factor expression which first appeared in the 1996 Addendum of Section XI.

For membrane tension, the K_I corresponding to membrane tension for the postulated defect is:

$$K_{Im} = M_m (pR_i \div t) \qquad (2.6.3\text{-}3)$$

Where M_m for an inside surface is given by:

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

$M_m = 0.926 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$, and

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

Similarly, M_m for an outside surface flaw is given by:

$M_m = 1.77$ for $\sqrt{t} < 2$,

$M_m = 0.893 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$, and

$M_m = 3.09$ for $\sqrt{t} > 3.464$.

Where:

R_i = vessel inner radius,

t = vessel wall thickness, and

p = internal pressure

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where $M_b = 0.667\ M_m$

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{IT} = 0.953 \times 10^{-3}\ CR\ t^{2.5} \qquad (2.6.3\text{-}4)$$

where:

CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{IT} = 0.753 \times 10^{-3}\ HU\ t^{2.5} \qquad (2.6.3\text{-}5)$$

where:

HU = the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from ASME Section XI, Appendix G, Figure G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Section XI, Appendix G, Figure G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in G-2214.3 (a)(1) and (2) of Appendix G to ASME Section XI.

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a 1/4-thickness inside surface defect using the relationship:

$$K_{IT} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (2.6.3\text{-}6)$$

or similarly, K_{IT} during heatup for a 1/4-thickness outside surface defect using the relationship:

$$K_{IT} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \qquad (2.6.3\text{-}7)$$

where the coefficients C_0, C_1, C_2, and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.6.3\text{-}8)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Once K_{IT} is computed, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{IT}(T_c)_{1/4t}}{2.0} \qquad (2.6.3\text{-}9)$$

From $K_{IM(max)}$, the maximum allowable pressure is determined using the iterative process described above and equations (2.6.2-2) through (2.6.2-5).

The steady-state pressure-temperature curve of Section 2.6.2 is compared to the cooldown curves for the 1/4t inside surface flaw at each cooldown rate. At any time, the allowable pressure is the lesser of the two values, and the resulting curve is called the composite cooldown limit curve.

Finally, the 10 CFR Part 50[4] rule for closure flange regions is incorporated into the cooldown composite curve as described in Section 2.5.

2.6.4 Finite Heatup Rate Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia} or K_{Ic}) is determined in Equation (2.5-1) or (2.5-2) at both the 1/4t and 3/4t locations where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius. At the 3/4t location, a 1/4t flaw is assumed to originate on the outside of the vessel.

For each heatup rate a pressure-temperature curve is calculated at the 1/4t and 3/4t locations. First, the thermal stress intensity factor is calculated at the 1/4t and 3/4t locations for a coolant temperature at a given time using Option 1 or 2 from Section 2.6.3.

Once K_{IT} is computed, the maximum allowable membrane (pressure) stress intensity factors at the 1/4t and 3/4t locations are determined using the following equations:

$$\text{At } 1/4t, \quad K_{IM(max)1/4t} = \frac{K_I * (T - RT_{NDT})_{1/4t} - K_{IT}(T_c)_{1/4t}}{2.0} \qquad (2.6.4\text{-}1)$$

$$\text{At } 3/4t, \quad K_{IM(max)3/4t} = \frac{K_I * (T - RT_{NDT})_{3/4t} - K_{IT}(T_c)_{3/4t}}{2.0} \qquad (2.6.4\text{-}2)$$

From $K_{IM(max)1/4t}$ and $K_{IM(max)3/4t}$, the maximum allowable pressure at both the 1/4t and 3/4t locations is determined using the iterative process described in Section 2.6.2 and equations (2.6.2-2) through (2.6.2-5).

As was done with the cooldown case, the steady state pressure-temperature curve of Section 2.6.2 is compared with the 1/4t and 3/4t location heatup curves for each heatup rate, with the lowest of the three being used to generate the composite heatup limit curve. The composite curve is then adjusted for the 10 CFR Part 50[4] rule for closure flange requirements, as discussed in Section 2.5.

2.6.5 Hydrostatic and Leak Test Curve Analyses

The minimum inservice hydrostatic leak test curve is determined by calculating the minimum allowable temperature at two pressure values (pressure values of 2000 psig and 2485 psig, approximately 110% of operating pressure, are generally used). The curve is generated by drawing a line between the two pressure-temperature data points. The governing equation for generating the hydrostatic leak test pressure-temperature limit curve is defined in Appendix G, Section X1, of the ASME Code[5] as follows:

$$1.5\, K_{IM} < K_{Ia} \qquad (2.6.5\text{-}1)$$

where, K_{IM} is the stress intensity factor caused by the membrane (pressure) stress and K_{Ia} is the reference stress intensity factor as defined in equation (2.5-1). Note that the thermal stress intensity factor is neglected (i.e., $K_{IT} = 0$) since the hydrostatic leak test is performed at isothermal conditions.

The pressure stress is determined by,

$$\sigma_P = \left[\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2}\right] P \qquad (2.6.5\text{-}2)$$

where,

P = the input pressure (generally 2000 and 2485 psig)

Next, the pressure stress intensity factor is calculated for a 1/4t flaw by option 1:

$$K_{IM} = \left[1.1 M_K \sqrt{\frac{\pi a}{Q}}\right] \sigma_P \qquad (2.6.5\text{-}3)$$

or by option 2:

$$K_{IM} = M_m\, \sigma_p$$

where,

M_m = 1.84 for $\sqrt{t} < 2$

= 0.918 $\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

= 3.18 for $\sqrt{t} > 3.464$

The K_{IM} result is multiplied by the 1.5 factor of equation (2.5-2) and divided by 1000,

$$K_{HYD} = \frac{1.5 K_{IM}}{1000} \qquad (2.6.5\text{-}4)$$

Finally, the minimum allowable temperature is determined by setting K_{HYD} to K_{Ia} in equation (2.5-1) and solving for temperature T:

$$T = \frac{\ell n\left[\frac{(K_{HYD} - 26.78)}{1.223}\right]}{0.0145} + RT_{NDT} - 160.0 \qquad (2.6.5\text{-}5)$$

The 1983 Amendment to 10CFR50[3] has a rule which addresses the test temperature for hydrostatic pressure tests. This rule states that, when there is no fuel in the reactor vessel during hydrostatic pressure tests or leak tests, the minimum allowable test temperature must be 60°F above the adjusted reference temperature of the beltline region material that is controlling. If fuel is present in the reactor vessel during hydrostatic pressure tests or leak tests, the requirements of this section and Section 2.5 must be met.

2.7 CODE CASE N-588 FOR CIRCUMFERENTIAL WELD FLAWS

In 1997, ASME Section XI, Appendix G was revised to add methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The original ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds. In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Section XI, Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of G-2120 is

$$K_{IM} = M_m \, x \, (PR_i/t)$$

Where, M_m for an inside surface flaw is given by:

M_m = 0.89 for $\sqrt{t} < 2$,

M_m = 0.443 $\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 1.53 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 0.89 for $\sqrt{t} < 2$,

M_m = 0.443 $\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,

M_m = 1.53 for $\sqrt{t} > 3.464$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circumferential weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

2.8 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3106 psig), which is 621 psig for a typical Westinghouse reactor vessel design.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{ia} fracture toughness, in the mid 1970s.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{ic} in the development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1."

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using the K_{ic} toughness. Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region. It is therefore clear that no additional boltup requirements are necessary, and therefore the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves, once the requirements of 10CFR50 Appendix G are changed.

2.9 MINIMUM BOLTUP TEMPERATURE

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The

Westinghouse position is that the minimum boltup temperature be no lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT} whichever is higher.

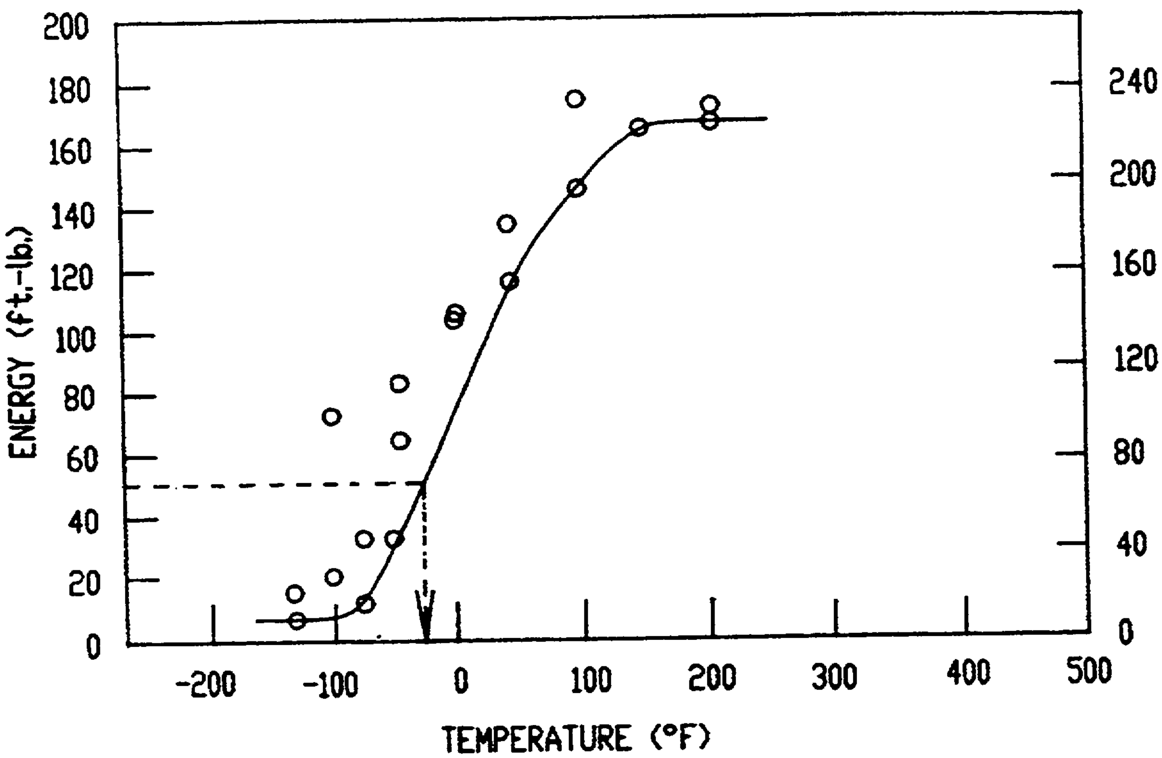


Figure 2.1 **Example of a Charpy Impact Energy Curve Used to Determine IRT_{NDT} (Note: 35 mils lateral expansion is required at indicated temperature)**

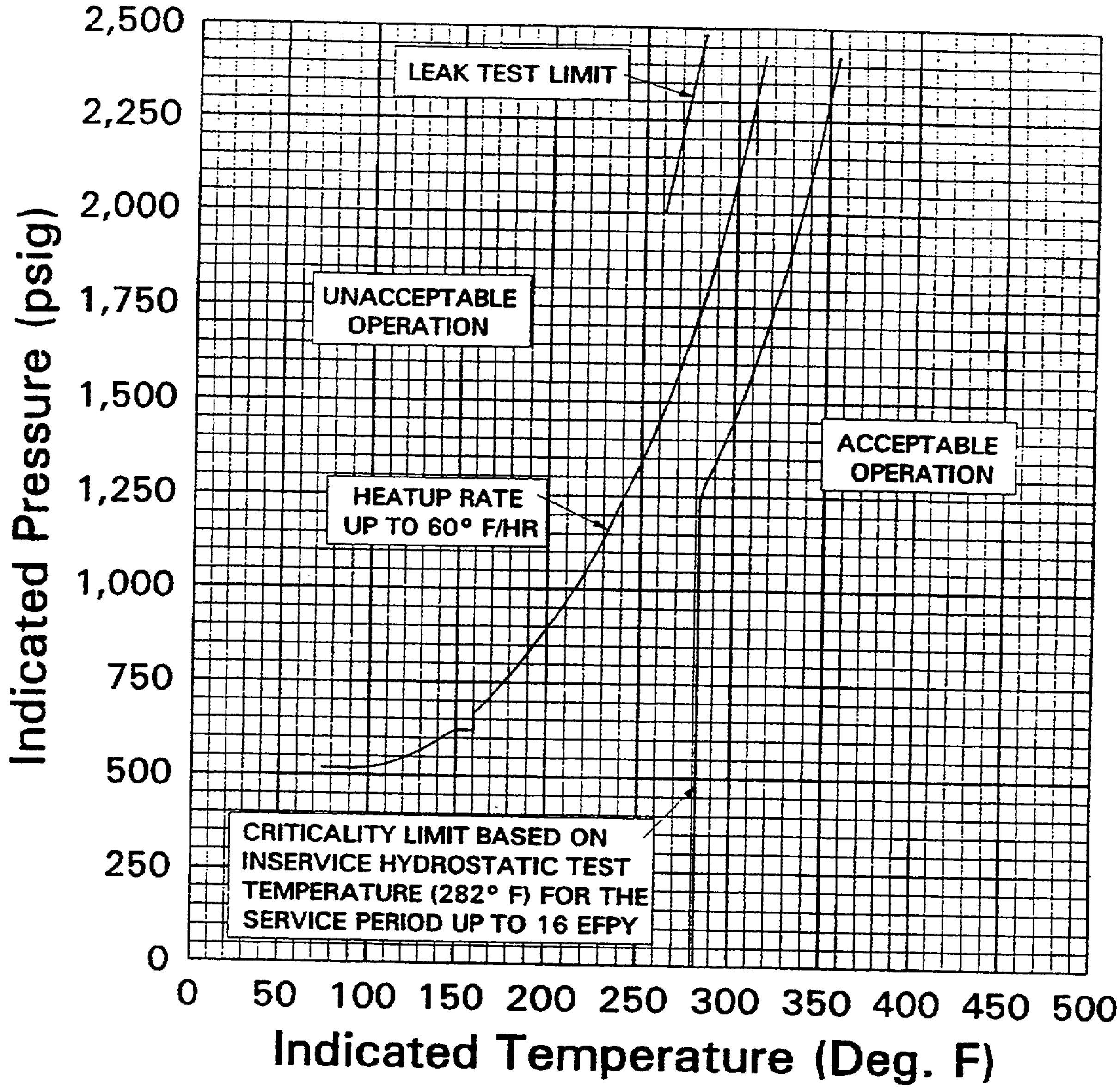


Figure 2.2 Heatup Pressure-Temperature Limit Curve For Heatup Rates up to 60°F/Hr

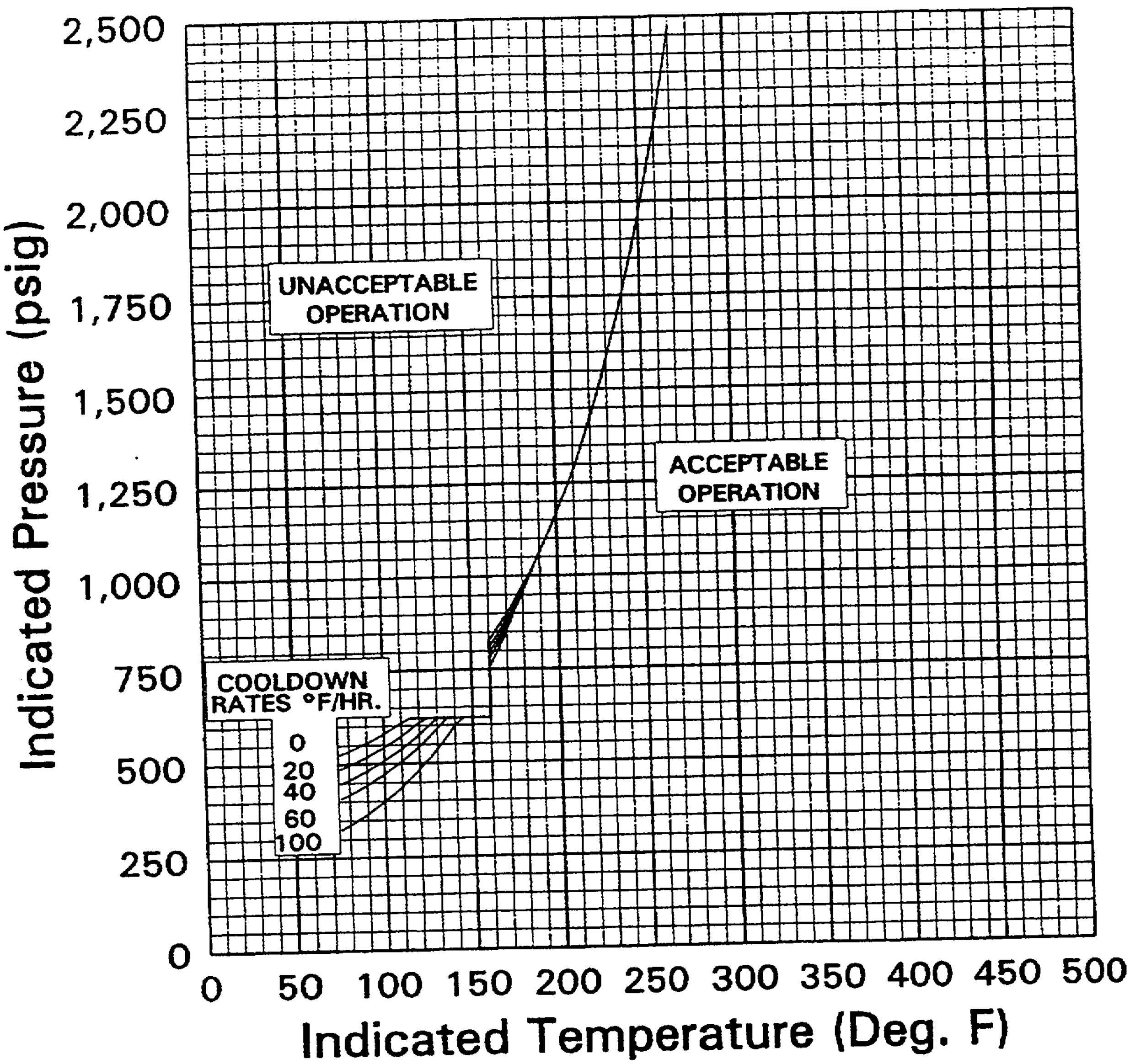


Figure 2.3 Cooldown Pressure-Temperature Limit Curves or Cooldown Rates up to 100°F/Hr

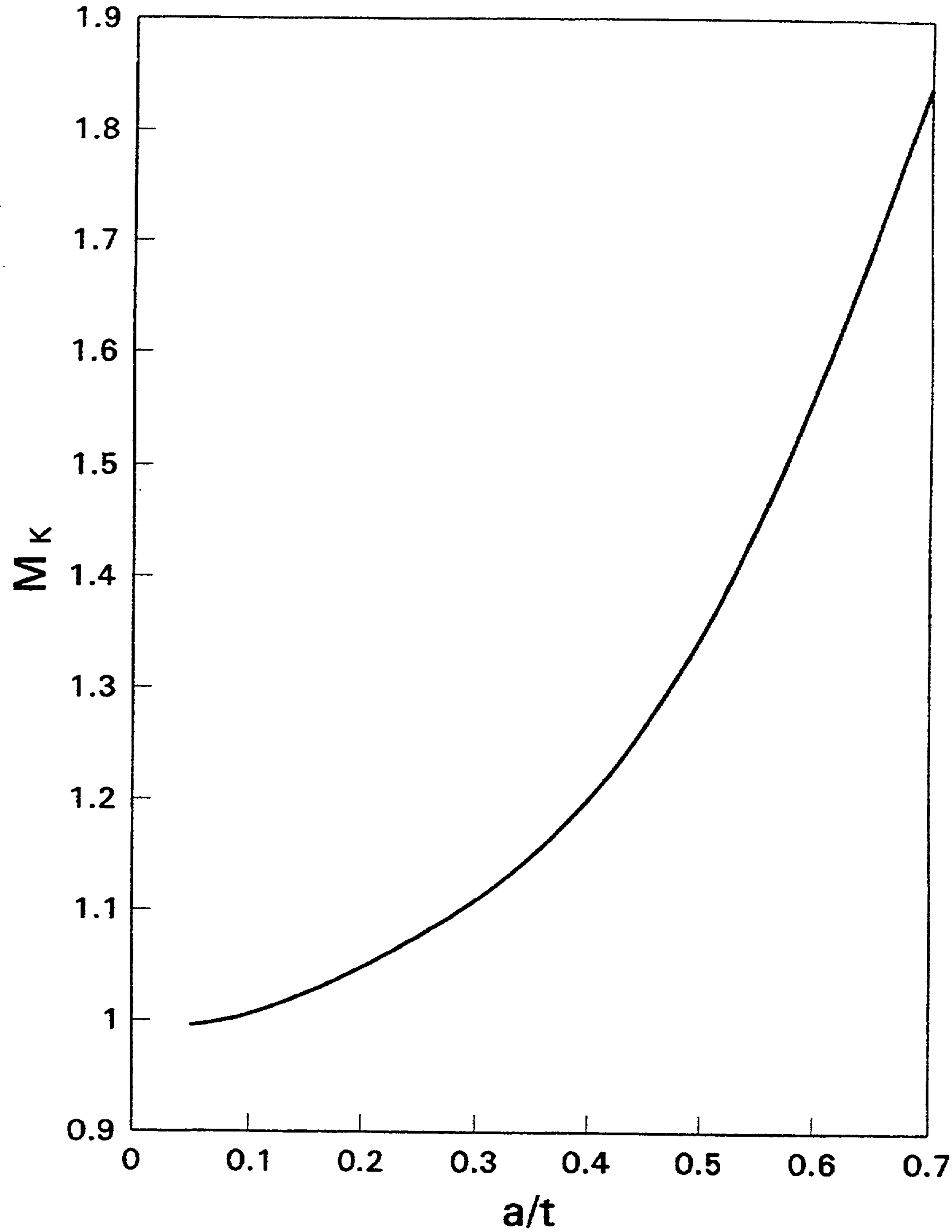


Figure 2.4 **Membrane Stress Correction Factor (M_K) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6 (Welding Research Bulletin 175 Method)**

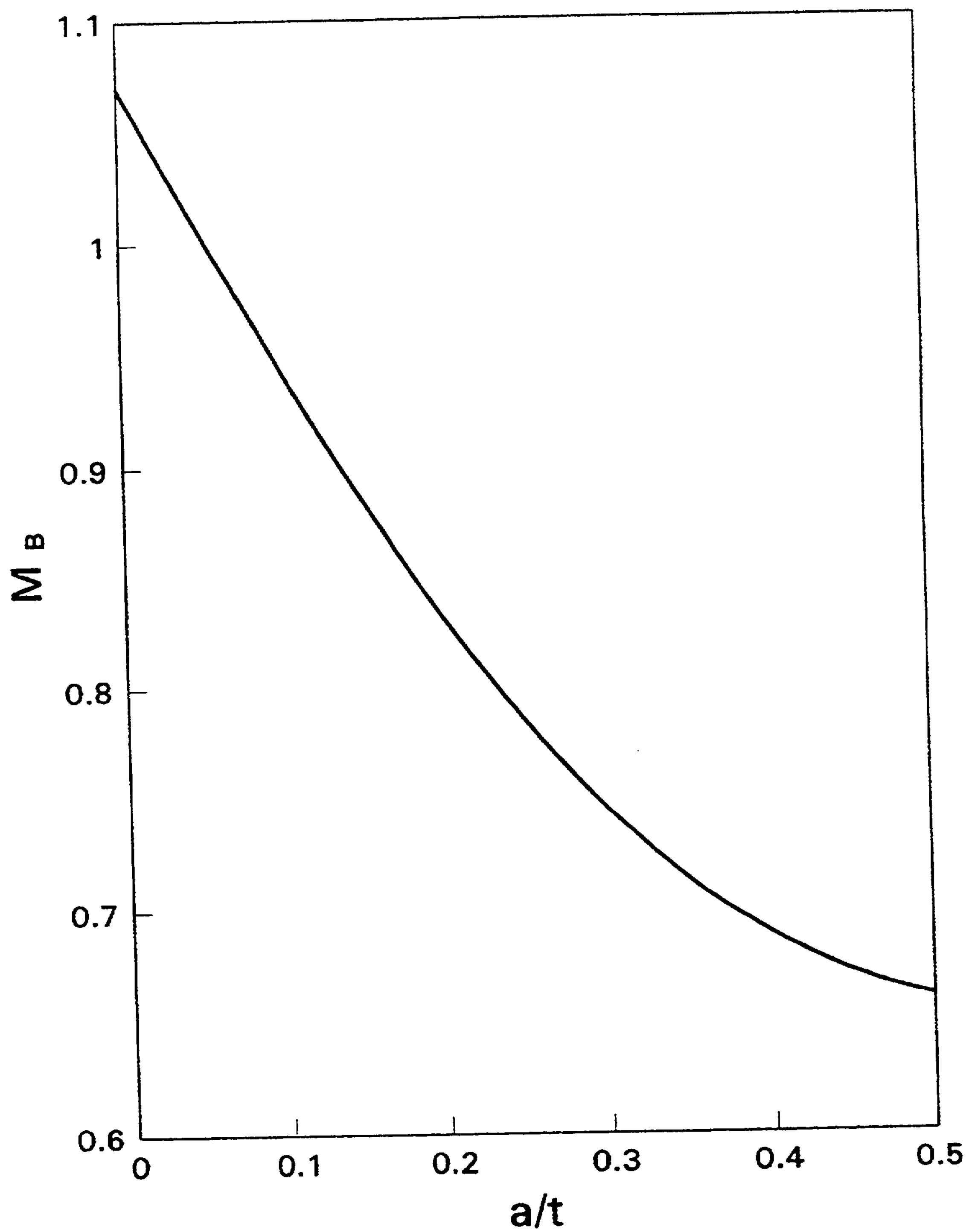


Figure 2.5 Bending Stress Correction Factor (M_B) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6 (Welding Research Bulletin 175 Method)

3.0 COLD OVERPRESSURE MITIGATING SYSTEM (COMS)

3.1 INTRODUCTION

The purpose of the COMS is to supplement the normal plant operational administrative controls and the water relief valves in the Residual Heat Removal System (RHRS) when they are unavailable to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. This has been achieved by conservatively choosing COMS setpoints which prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G[4] requirements. The COMS is designed to provide the capability, during relatively low temperature operation (typically less than 350°F), to automatically prevent the RCS pressure from exceeding the applicable limits. Once the system is enabled, no operator action is involved for the COMS to perform its intended pressure mitigation function. Thus, no operator action is modelled in the analyses supporting the setpoint selection, although operator action may be initiated to ultimately terminate the cause of the overpressure event.

The PORVs located near the top of the pressurizer, together with additional actuation logic from the wide-range pressure channels, are utilized to mitigate potential RCS overpressure transients defined below if the RHRS water relief valves are inadvertently isolated from the RCS. The COMS provides the supplemental relief capacity for specific transients which would not be mitigated by the RHRS relief valves. In addition, a limit on the PORV piping is accommodated due to the potential for water hammer effects to be developed in the piping associated with these valves as a result of the cyclic opening and closing characteristics during mitigation of an overpressure transient. Thus, a pressure limit more restrictive than the 10CFR50, Appendix G[4] allowable is imposed above a certain temperature so that the loads on the piping from a COMS event would not affect the piping integrity.

Two specific transients have been defined, with the RCS in a water-solid condition, as the design basis for COMS. Each of these scenarios assumes as an initial condition that the RHRS is isolated from the RCS, and thus the relief capability of the RHRS relief valves is not available. The first transient consists of a heat injection scenario in which a reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature and the RHRS has been inadvertently isolated. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into a water-solid RCS caused by the simultaneous isolation of the RHRS isolation of letdown and failure of the normal charging flow controls to the full flow condition. Various combinations of charging and safety injection flows may also be evaluated on a plant-specific basis; however, the mass injection transient used as a design basis should encompass the limiting pump(s) operability configuration permitted per the plant-specific Technical Specifications during the Modes when COMS is required to be in operation. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

3.2 COMS SETPOINT DETERMINATION

Westinghouse has developed the following methodology which is employed to determine PORV setpoints for mitigation of the COMS design basis cold overpressurization transients. This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel integrity.

3.2.1 Parameters Considered

The selection of proper COMS setpoints for actuating the PORVs requires the consideration of numerous system parameters including:

a. Volume of reactor coolant involved in transient

b. RCS pressure signal transmission delay

c. Volumetric capacity of the relief valves versus opening position

d. Stroke time of the relief valves (open & close)

e. Initial temperature and pressure of the RCS

f. Mass input rate into RCS

g. Temperature of injected fluid

h. Heat transfer characteristics of the steam generators

i. Initial temperature asymmetry between RCS and steam generator secondary water

j. Mass of steam generator secondary water

k. RCP startup dynamics

l. 10CFR50, Appendix G pressure/temperature characteristics of the reactor vessel

m. Pressurizer PORV piping/structural analysis limitations

n. Dynamic and static pressure difference between reactor vessel midplane and location of wide range pressure transmitter

These parameters are input to a specialized version of the LOFTRAN computer code which calculates the maximum and minimum system pressures.

3.2.2 Pressure Limits Selection

The function of the COMS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing COMS setpoints which prevent exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50[4]. The COMS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. In addition, the COMS also provides for an operational consideration to maintain the integrity of the PORV piping. A typical characteristic 10CFR50

Appendix G curve is shown by Figure 3.1 where the allowable system pressure increases with increasing temperature. This type of curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties. Superimposed on this curve is the PORV piping limit which is conservatively used, for setpoint development, as the maximum allowable pressure above the temperature at which it intersects with the 10CFR50 Appendix G curve.

When a relief valve is actuated to mitigate an increasing pressure transient, the release of a volume of coolant through the valve will cause the pressure increase to be slowed and reversed as described by Figure 3.2. The system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signalled to close. Note that the pressure continues to decrease below the reset pressure as the valve recloses. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the reactor coolant pump #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement and the PORV piping limitations) and the nominal RCP #1 seal performance criteria create a pressure range from which the setpoints for both PORVs may be selected as shown on Figures 3.3 and 3.4.

Where there is insufficient range between the upper and lower pressure limits to select PORV setpoints to provide protection against violation of both limits, setpoint selection to provide protection against the upper pressure limit violation shall take precedence.

3.2.3 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signalled to open at a specific pressure setpoint. However, as shown on Figure 3.2, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed. The maximum and minimum pressures reached (P_{MAX} and P_{MIN}) in the transient are a function of the selected setpoint (P_s) as shown on Figure 3.3. The shaded area represents an optimum range from which to select the setpoint based on the particular mass input case. Several mass input cases may be run at various input flow rates to bound the allowable setpoint range.

3.2.4 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e., the mass injection transient is not sensitive to temperature). The shaded area on Figure 3.4 describes the acceptable band for a heat input transient from which to select the setpoint for a particular initial reactor coolant temperature.

3.2.5 Final Setpoint Selection

By superimposing the results of multiple mass input and heat input cases evaluated, (from a series of figures such as 3.3 and 3.4) a range of allowable PORV setpoints to satisfy both conditions can be determined. Each of the two PORVs may have a different pressure setpoint versus temperature specification such that only one valve will open at a time and mitigate the transient (i.e., staggered setpoints). The second valve operates only if the first fails to open on command. This design supports a single failure assumption as well as minimizing the potential for both PORVs to open simultaneously, a condition which may create excessive pressure undershoot and challenge the RCP #1 seal performance criteria. However, each of the sets of staggered setpoints must result in the system pressure staying below the P_{MAX} pressure limit shown on Figures 3.3 and 3.4 when either valve is utilized to mitigate the transient.

The function generator used to program the pressure versus setpoint curves for each valve has a limited number of programmable break points (typically 9). These are strategically defined in the final selection process, with consideration given to the slope of any line segment, which is limited to approximately 24 psi/°F.

The selection of the setpoints for the PORVs considers the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50[(4)] or the peak RCS pressure based upon piping/structural analysis loads. The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. The upper pressure limits are already based on conservative assumptions (such as a safety factor of 2 on pressure stress, use of a lower bound K_{IR} curve and an assumed 1/4T flaw depth with a length equal to 1 1/2 times the vessel wall thickness) as discussed in section 2 of this report. However, uncertainties associated with instrumentation utilized by COMS will be determined using a process described by ISA Standard S67.04-1994. These uncertainties will be accounted for in the selection of COMS PORV setpoints.

While the RHR relief valves also provide overpressure protection for certain transients, these transients are not the same as the design basis transients for COMS. The RHR relief valve design basis precedes the development of the COMS design basis, and therefore the RHR relief valves may not provide protection against the COMS design basis events. The design basis described herein should be considered as applicable only when the pressurizer PORVs are used for COMS.

3.3 APPLICATION OF ASME CODE CASE N-514

ASME Code Case N-514[(17)] allows low temperature overpressure protection systems (LTOPS, as the code case refers to COMS) to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G, paragraph G-2215, of Section XI of the ASME Code[(5)]. (Note, that the setpoint selection methodology as discussed in Section 3.2.5 specifically utilizes the steady-state curve.) The application of ASME Code Case N-514 increases the operating margin in the region of the pressure-temperature limit curves where the COMS system is enabled. Code Case N-514 requires LTOPS to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. RT_{NDT} is the highest adjusted reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2.

3.4 ENABLE TEMPERATURE FOR COMS

The enable temperature is the temperature below which the COMS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is very conservative, and is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3. This definition is also supported by Westinghouse and can be used by requesting an exemption to the regulations or when ASME Code Case N-514 is formally approved by the NRC.

A significant improvement in the enable temperature can be obtained by application of code case N641. This code case incorporates the benefits of code cases N588, and N640. The resulting enable temperatures for the Westinghouse designs obtained using code case N641 are listed below.

The use of Code Case N641 has not yet been approved by the NRC, and therefore the use of this Code Case will require approval of an exemption request, as discussed in under 10CFR50.60 paragraph (b), pertaining to proposed alternatives to the requirements of Appendices G and H.

Vessel Type	Axial Flaw	Circumferential Flaw
2 – loop	RT_{NDT} + 23F	Any temperature
3 – loop	RT_{NDT} + 30F	RT_{NDT} – 174F
4 – loop	RT_{NDT} + 34F	RT_{NDT} – 110F

The RCS cold leg temperature limitation for starting an RCP is the same value as the COMS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the COMS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

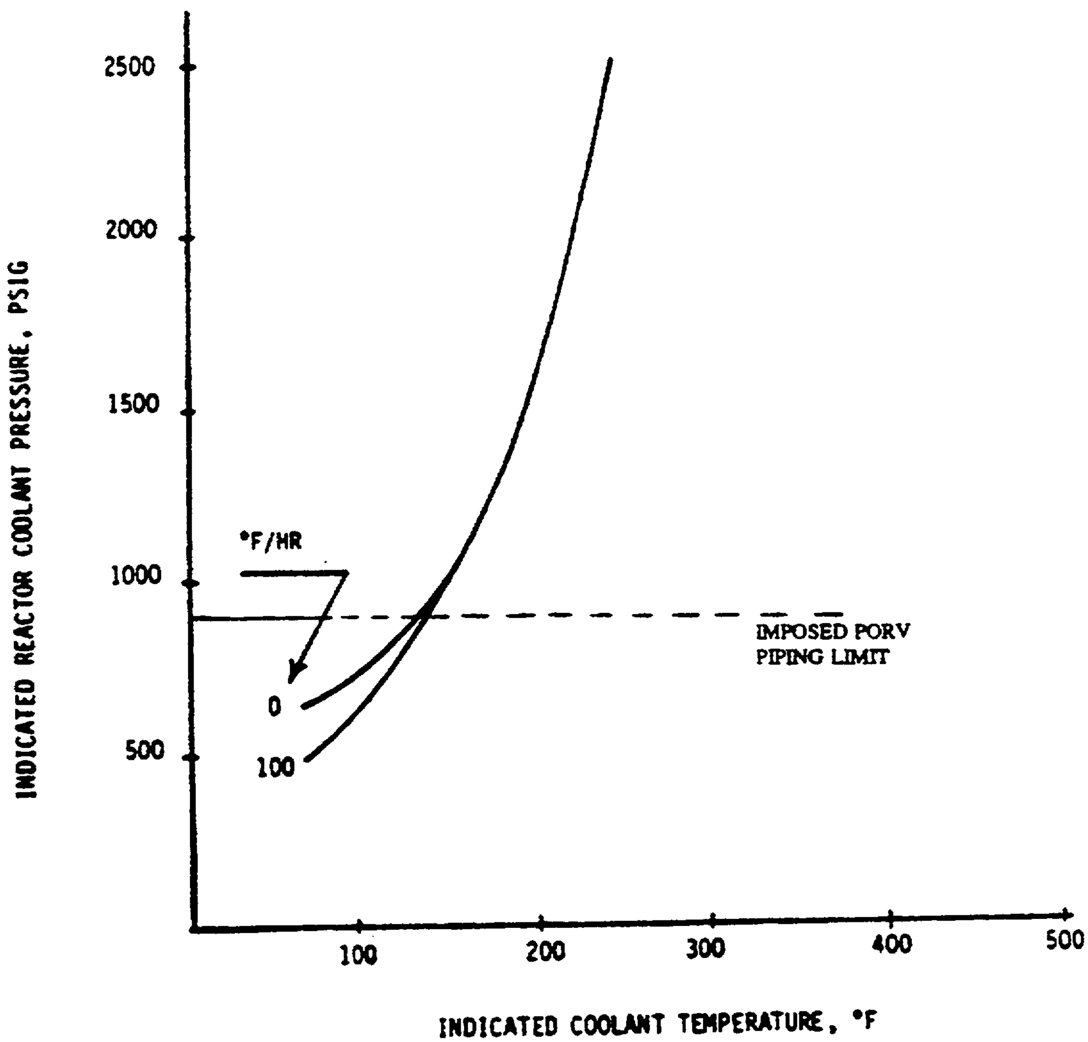


Figure 3.1 Typical Appendix G P/T Characteristics

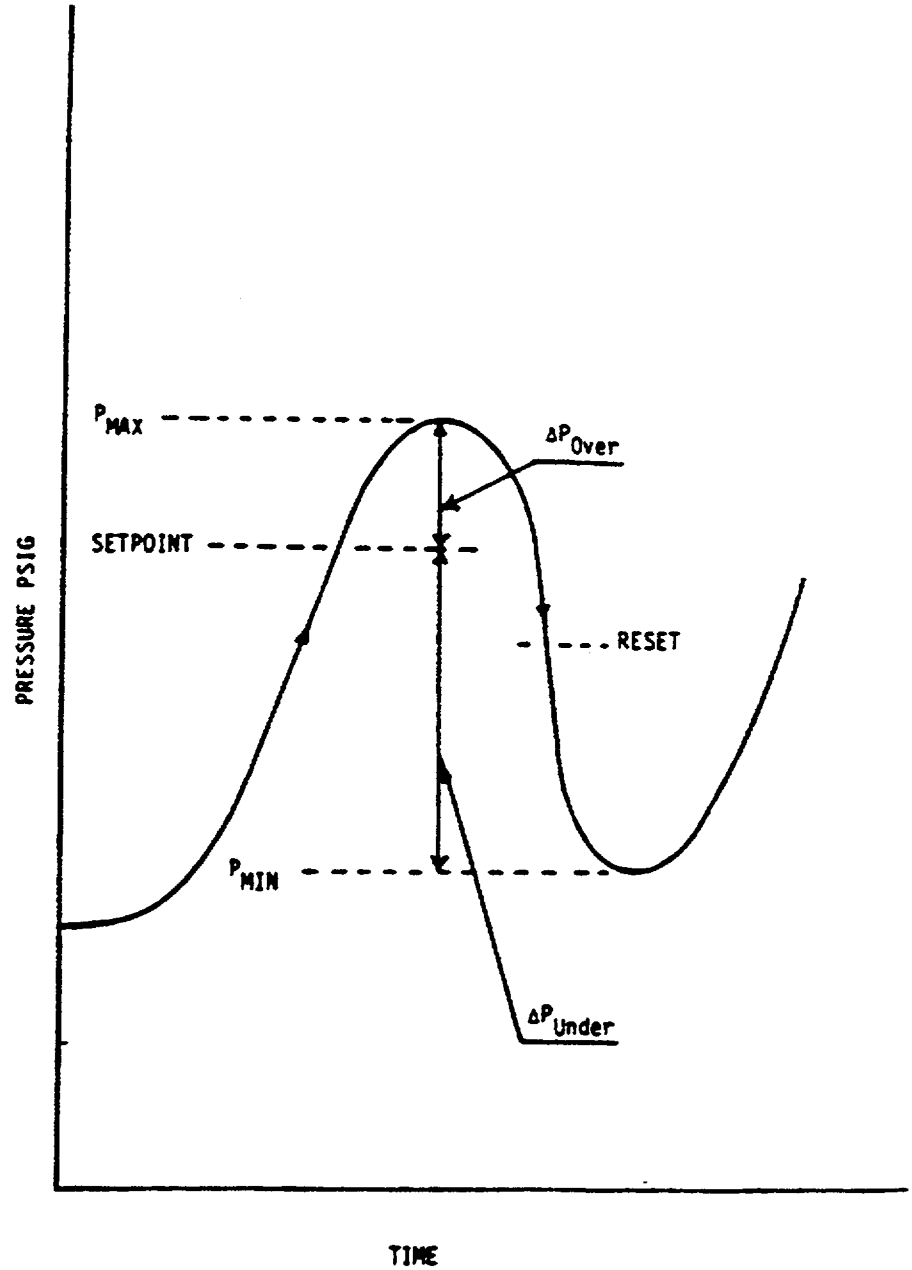
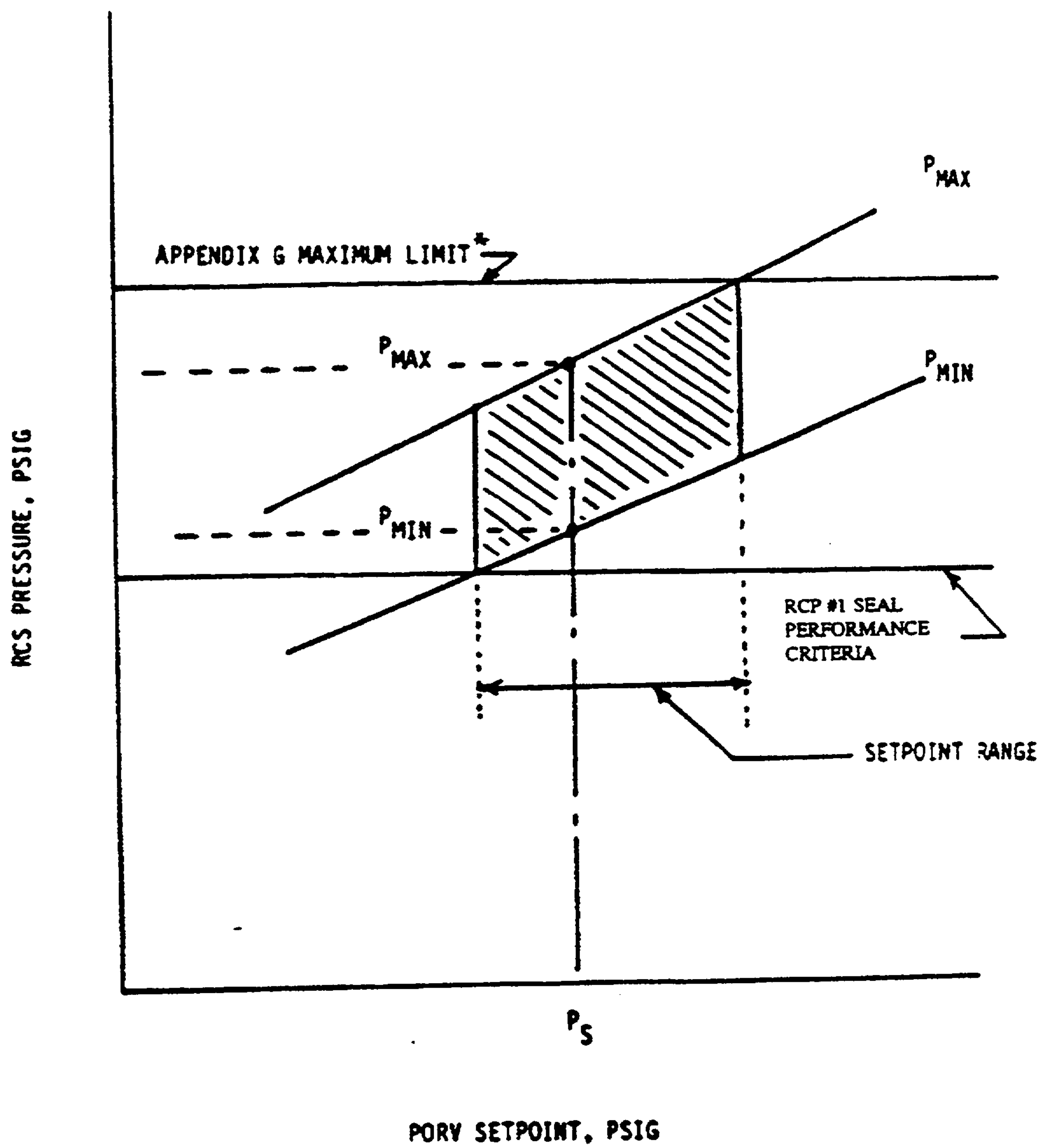
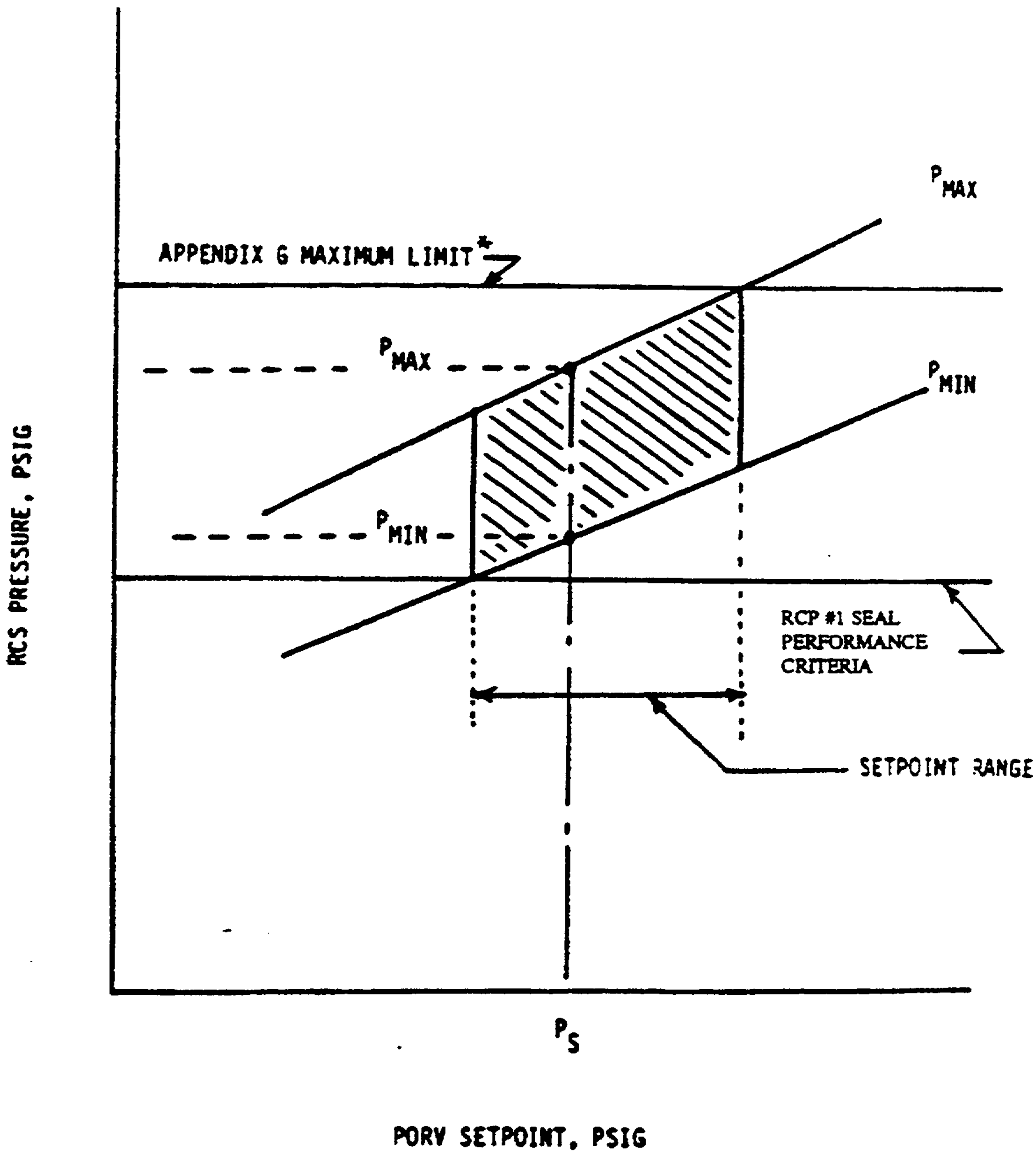


Figure 3.2 Typical Pressure Transient (1 Relief Valve Cycle)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

Figure 3.3 Setpoint Determination (Mass Input)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

Figure 3.4 Setpoint Determination (Heat Input)

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Plants," Revision 2.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," Generic Letter 88-16, October, 1988.

3. U.S. Nuclear Regulatory Commission, Radiation Embrittlement of Reactor Vessel Materials, Regulatory Guide 1.99, Revision 2, May, 1988.

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix G, Fracture Toughness Requirements.

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G, Fracture Toughness Criteria For Protection Against Failure.

6. R. G. Soltesz, R. K. Disney, J. Jedruch, and S. L. Ziegler, Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation. Vol. 5-Two-Dimensional Discrete Ordinates Transport Technique, WANL-PR(LL)-034, Vol. 5, August 1970.

7. ORNL RSIC Data Library Collection DLC-76 SAILOR Coupled Self-Shielded, 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

11. RSIC Computer Code Collection CCC-543, "TORT-DORT Two- and Three-Dimensional Discrete Ordinates Transport, Version 2.8.14," December, 1994.

12. RSIC Data Library Collection DLC- 175, "BUGLE-93, Production and Testing of the VITAMIN B-6 Fine Group and the BUGLE-93 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data," April 1994.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors," Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix A, Analysis of Flaws, Article A3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section X1, Division 1, "Low Temperature Overpressure Protection," Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

19. ASME Boiler and Pressure Vessel Code Case N640, Section XI, Division 1, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," February 26, 1996.

20. ASME Boiler and Pressure Vessel Code Case N588, Section XI, Division 1, "Alternative to Reference Flow Orientation of Appendix G for Circumferential Welds in Reactor Vessels," December 12, 1997.

21. ASME Boiler and Pressure Vessel Code Case N641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," January 17, 2000.

APPENDIX A

RELEVANT ASME NUCLEAR CODE CASES

Table A-1 Status of ASME Nuclear Code Cases Associated with the P-T Limit Curve/COMS Methodology

Code Case	Title	Approved by ASME	Codified in 10CFR50	Exemption Request Granted
514	Low Temperature Overpressure Protection	2/12/92	No	Yes
588	Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessel	12/12/97	No	Yes
640	Alternative Reference Fracture Toughness for Development of P-T Limit Curves	2/26/99	No	Yes
641	Alternative Pressure Temperature Relationship and Low Temperature Overpressure Protection System Requirement	1/17/00	No	Unknown

Approval Date: February 12, 1992

See Numeric Index for expiration and any reaffirmation dates.

Case N-514
Low Temperature Overpressure Protection
Section XI, Division 1

Inquiry: Section XI, Division 1, IWB-3730, requires that during reactor operation, load and temperature conditions be maintained to provide protection against failure due to the presence of postulated flaws in the ferritic portions of the reactor coolant pressure boundary. For those plants having low temperature overpressure protection (LTOP) systems, what load and temperature conditions under IWB-3730 may be used to provide protection against failure during reactor start-up and shutdown operation due to low temperature overpressure events that have been classified as Service Level A or B events?

Reply: It is the opinion of the Committee that for those plants having LTOP systems the following load and temperature conditions may be used to provide protection against failure during reactor start-up and shutdown operation due to low temperature overpressure events that have been classified as Service Level A or B events. LTOP systems shall be effective at coolant temperatures less than 200°F or at coolant temperatures[1] corresponding to a reactor vessel metal temperature[2] less than RT_{NDT} + 50°F, whichever is greater. LTOP systems shall limit the maximum pressure in the vessel to 110% of the pressure determined to satisfy Appendix G, para. G-2215 of Section XI, Division 1.

[1] The coolant temperature is the reactor coolant inlet temperature.

[2] The vessel metal temperature is the temperature at a distance one-fourth of the vessel section thickness from the inside surface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Rev. 2.

Approval Date: December 12, 1997
See Numeric Index for expiration and any reaffirmation dates.

Case N-588
Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels
Section XI, Division 1

Inquiry: Paragraph G-2120 specifies that postulated reference defects should be sharp, surface defects oriented normal to the direction of maximum stress. What alternative rules may be used for postulating a reference defect in a circumferential welds?

Reply: It is the opinion of the Committee that, as an alternative to the procedure for assuming axially oriented reference defects in all welds and base metal per G-2120, a circumferential orientation may be used specifically for circumferential welds.

CONTENTS

-1000 INTRODUCTION

-1100 Scope

This Case presents an alternative procedure for calculating applied stress intensity factors during normal operation and pressure test conditions. The procedure is based on the principles of linear elastic fracture mechanics. At each location being investigated a maximum postulated defect is assumed, and the *mode I stress intensity factor* K_I is produced by each of the specified pressure and thermal loadings. Different procedures are recommended for axial and circumferential weld orientations.

-2100 GENERAL REQUIREMENTS

-2120 Maximum Postulated Defects

The postulated defect used in this recommended procedure are sharp, surface defects oriented axially for plates, forgings and axial welds, and oriented circumferentially for circumferential welds. For section thicknesses of 4 in. to 12 in. The postulated defects have a depth of one-fourth of the section thickness and a length of $1\frac{1}{2}$ times the section thickness. Defects are postulated at both the inside and outside surfaces. For sections greater than 12 in. thick, the postulated defect for the 12 in. section is used. For sections less than 4 in. thick, the 1 in. deep defect is conservatively postulated. Smaller defect sizes[1] may be used on an individual case basis if a smaller size of maximum postulated defect can be ensured. Due to the safety factors recommended here, the prevention of nonductile fracture is ensured for some of the most important situations even if the defects were to be about twice as large in linear dimensions as this postulated maximum defect.

-2200 Level A and Level B Service Limits

-2210 Shells and Heads Remote from Discontinuities

-2211 Recommendations

The assumptions of this Subarticle are recommended for shell and head regions during Level A and B Service Limits.

-2212 Material Fracture Toughness

-2212.1 Reference Critical Stress Intensity Factor for Material. The K_{Ia} values of Fig. G-2210-1 are recommended.

-2212.2 Irradiation Effects. Subarticle A-4400 of Appendix A is recommended to define the change in reference critical stress intensity factor due to irradiation.

-2213 Maximum Postulated Defects

The recommended maximum postulated defects are described in -2120.

-2214 Calculated Stress Intensity Factors

-2214.1 Membrane Tension. The K_I corresponding to membrane tension for the postulated axial defect of -2120 is $K_{Im} = M_m \times (pR_i/t)$, where M_m for an inside axial surface flaw is given by

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

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

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

Similarly, M_m for an outside axial surface flaw is given by

$M_m = 1.77$ for $\sqrt{t} < 2$

$M_m = 0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

$M_m = 3.09$ for $\sqrt{t} > 3.464$

where

p = internal pressure (ksi)

R_i = vessel inner radius (in.)

t = vessel wall thickness (in.)

The K_I corresponding to membrane tension for the postulated circumferential defect of -2120 is $K_{Im} = M_m \times (pR_i/t)$, where M_m for an inside circumferential surface defect is given by

$M_m = 0.89$ for $\sqrt{t} < 2$

$M_m = 0.443\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

$M_m = 1.53$ for $\sqrt{t} > 3.464$

Similarly, M_m for an outside circumferential surface defect is given by

$M_m = 0.89$ for $\sqrt{t} < 2$

$M_m = 0.443\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

$M_m = 1.53$ for $\sqrt{t} > 3.464$

-2214.2 Bending Stress. The K_I corresponding to bending stress for postulated axial or circumferential defects of -2120 is $K_{Ib} = M_b \times$ maximum bending stress, where M_b is two-thirds of M_m.

-2214.3 Radial Thermal Gradient. The maximum K_I produced by a radial thermal gradient for a postulated axial or circumferential inside surface defect of

[1] WRCB 175 (Welding Research Council Bulletin 175) "PVRC Recommendations on Toughness Requirements for Ferritic Materials" provides procedures in para. 5(c)(2) for considering maximum postulated defects smaller than those described.

-2120 is $K_{It} = 0.953 \times 10^{-3} \times \text{CR} \times t^{2.5}$, where *CR* is the cooldown rate in F/hr., or, for a postulated axial or circumferential outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times \text{HU} \times t^{2.5}$, where *HU* is the heatup rate in F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions in -2214.3(a)(1) and (2).

(1) An assumed shape of the temperature gradient is approximately as shown in Fig. G-2214-2.

(2) The temperature change starts from a steady state condition and has a rate, associated with startup and shutdown, less than about 100°F/hr. The results would be overly conservative if applied to rapid temperature changes.

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution at any specified time during cooldown for a ¼-thickness axial or circumferential surface defect.

For an inside surface defect during cooldown

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)\sqrt{\pi a}$$

For an outside surface defect during heatup

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + (0.401C_3)\sqrt{\pi a}$$

The coefficients C_0, C_1, C_2, and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3$$

where x is a dummy variable that represents the radial distance, in., from the appropriate (i.e., inside or outside) surface and a is the maximum crack depth, in.

(c) For the startup condition, the allowable pressure vs. temperature relationship is the minimum pressure at any temperature, determined from (1) the calculated steady state results for the ¼-thickness inside surface defect, (2) the calculated steady state results for the ¼-thickness outside surface defect, and (3) the calculated results for the maximum allowable heatup rate using a ¼-thickness outside surface defect.

-2215 Allowable Pressure

The equations given in this Subarticle provide the basis for determination of the allowable pressure at any temperature at the depth of the postulated defect during Service Conditions for which Level A and Level B Service Limits are specified. In addition to the conservatism of these assumptions, it is recommended that a factor of 2 be applied to the calculated K_I values produced by primary stresses. In shell and head regions remote from discontinuities, the only significant loadings are: (1) general primary membrane stress due to pressure; and (2) thermal stress due to thermal gradient through the thickness during startup and shutdown. Therefore, the requirement to be satisfied and from which the allowable pressure for any assumed rate of temperature change can be determined is:

$$2K_{Im} + K_{It} < K_{Ia} \qquad (1)$$

throughout the life of the component at each temperature with K_{Im} from -2214.1, K_{It} from -2214.3, and K_{Ia} from Fig. G-2210-1.

The allowable pressure at any temperature shall be determined by the following procedure:

(a) For the startup condition, consider postulated defects in accordance with -2120, perform calculations for thermal stress intensity factors due to the specified range of heatup rates from -2214.3, calculate the K_{Ia} toughness for all vessel beltline materials from -2212 using temperatures and RT_{NDT} values for the corresponding locations of interest, and calculate the pressure as a function of coolant inlet temperature for each material and location. The allowable pressure vs. temperature relationship is the minimum pressure at any temperature determined from (1) the calculated steady-state ($K_{It} = 0$) results for the ¼ thickness inside surface postulated defects using the equation:

$$p = \frac{K_{Ia}}{2M_m} \cdot \left({}^{t}/R_i\right)$$

and (2) the calculated results from all vessel beltline materials for the heatup stress intensity factors using the corresponding ¼ thickness outside surface postulated defects and the equation

$$p = \frac{K_{Ia} - K_{It}}{2M_m} \cdot \left({}^{t}/R_i \right)$$

(b) For the cooldown condition, consider postulated defects in accordance with -2120, perform calculations for thermal stress intensity factors due to the specified range of cooldown rates from -2214.3, calculate the K_{Ia} toughness for all vessel beltline materials from -2212 using temperatures and RT_{NDT} values for the corresponding locations of interest, and calculate the pressure as a function of coolant inlet temperature for each material and location using the equation:

$$p = \frac{K_{Ia} - K_{It}}{2M_m} \cdot \left({}^{t}/R_i \right)$$

The allowable pressure vs temperature relationship is the minimum pressure at any temperature, determined from all vessel beltline materials for the cooldown stress intensity factors using the corresponding ¼ thickness inside surface postulated defects.

Approval Date: February 26, 1999

See Numeric Index for expiration and any reaffirmation dates.

Case N-640
Alternative Reference Fracture Toughness for Development of P-T Limit Curves
Section XI, Division 1

Inquiry: May the reference fracture toughness curve K_{Ic}, as found in Appendix A of Section XI, be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves?

Reply: It is the opinion of the Committee that the reference fracture toughness K_{Ic} of Fig. A-4200-1 of Appendix A may be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves. When this Case is employed, LTOP Systems shall limit the maximum pressure in the vessel to 100% of the pressure allowed by the P-T Limit Curves.

Approval Date: January 17, 2000

See Numeric Index for expiration and any reaffirmation dates.

Case N-641
Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements
Section XI, Division 1

Inquiry: What alternatives to Appendix G-2215 may be used for determination of pressure-temperature relationships and low temperature overpressure protection system effective temperatures and allowable pressures?

Reply: It is the opinion of the Committee that, as an alternative to Appendix G-2215, the following may be used.

-1000 INTRODUCTION

-1100 Scope

This Case presents alternative procedures for calculating pressure-temperature relationships and low temperature overpressure protection (LTOP) system effective temperatures and allowable pressures. These procedures take into account alternative fracture toughness properties, circumferential and axial reference flaws, and plant-specific LTOP effective temperature calculations.

-2215 Allowable Pressure

-2215.1 Pressure-Temperature Relationship. The equations below provide the basis for determination of the allowable pressure at any temperature at the depth of the postulated defect during Service Conditions for which Level A and Level B Service Limits are specified. In addition to the conservatism of these assumptions, it is recommended that a factor of 2 be applied to the calculated K_I values produced by primary stresses. In shell and head regions remote from discontinuities, the only significant loadings are: (1) general primary membrane stress due to pressure; and (2) thermal stress due to thermal gradient through the thickness during startup and shutdown. Therefore, the requirement to be satisfied and from which the allowable pressure for any assumed rate of temperature change can be determined is:

$$2K_{Im} + K_{It} < K_{Ic} \qquad (1)$$

throughout the life of the component at each temperature with K_{Im} from G-2214.1, K_{It} from G-2214.3, and K_{Ic} from Fig. G-2210-1.

The allowable pressure at any temperature shall be determined as follows.

(a) For the startup condition,

(1) consider postulated defects in accordance with G-2120;

(2) perform calculations for thermal stress intensity factors due to the specified range of heat-up rates from G-2214.3;

(3) calculate the K_{Ic} toughness for all vessel beltline materials from G-2212 using temperatures and RT_{NDT} values for the corresponding locations of interest; and

(4) calculate the pressure as a function of coolant inlet temperature for each material and location. The allowable pressure-temperature relationship is the minimum pressure at any temperature determined from

(a) the calculated steady-state ($K_{It} = 0$) results for the $\frac{1}{4}$ thickness inside surface postulated defects using the equation:

$$P = \frac{K_{Ic}}{2M_m}\left({}^{t}/_{R_i}\right)$$

(b) the calculated results from all vessel beltline materials for the heatup stress intensity factors using the corresponding $\frac{1}{4}$ thickness outside-surface postulated defects and the equation:

$$P = \frac{K_{Ic} - K_{It}}{2M_m}\left({}^{t}/_{R_i}\right)$$

(b) For the cooldown condition;

(1) consider postulated defects in accordance with G-2120;

(2) perform calculations for thermal stress intensity factors due to the specified range of cooldown rates from G-2214.3;

(3) calculate the K_{Ic} toughness for all vessel beltline materials from G-2212 using temperatures and RT_{NDT} values for the corresponding location of interest; and

(4) calculate the pressure as a function of coolant inlet temperature for each material and location using the equation:

$$P = \frac{K_{Ic} - K_{It}}{2M_m}\left(t/_{R_i}\right)$$

The allowable pressure-temperature relationship is the minimum pressure at any temperature, determined from all vessel beltline materials for the cooldown stress intensity factors using the corresponding ¼ thickness inside-surface postulated defects.

-2215.2 Low Temperature Overpressure Protection System. Plants having LTOP systems may use the following temperature and pressure conditions to provide protection against failure during reactor startup and shutdown operation due to low temperature overpressure events that have been classified Service Level A or B.

(a) LTOP System Effective Temperature. The LTOP system effective temperature T_e is the temperature at or above which the safety relief valves provide adequate protection against nonductile failure. LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (2) below. Alternatively, LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (3) below.

(1) a coolant temperature[1] of 200°F;

(2) a coolant temperature[1] corresponding to a reactor vessel metal temperature[2], for all vessel beltline materials, where T_e is defined for inside axial surface flaws as RT_{NDT} + 40°F, and T_e is defined for inside circumferential surface flaws as RT_{NDT} − 85°F;

(3) a coolant temperature[1] corresponding to a reactor vessel metal temperature[2], for all vessel beltline materials, where T_e is calculated on a plant specific basis for the axial and circumferential reference flaws using the following equation:

$$T_e = RT_{NDT} + 50 \ln [((F \cdot M_m (pR_i / t)) - 33.2) / 20.734]$$

where

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

(b) LTOP System Allowable Pressure. LTOP systems shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy Eq. (1) if K_{Ic} is used for determination of allowable pressure, or 110% of the pressure determined to satisfy Eq. (1) if K_{Ia} is used (as an alternative to K_{Ic}) for determination of allowable pressure.

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[1] The coolant temperature is the reactor coolant inlet temperature.

[2] The vessel metal temperature is the temperature at a distance one-fourth of the vessel section thickness from the clad-base-metal interface in the vessel beltline region. RT_{NDT} is the highest adjusted reference temperature, for weld or base metal in the beltline region, at a distance one-fourth of the vessel section thickness from the clad-base-metal interface as determined in accordance with Regulatory Guide 1.99, Rev. 2.