WOG

Westinghouse Owners Group

OG-96-006

WCAP-14040-NP-A
Project Number 694

January 15, 1996

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

Attention: Chief, Planning, Program and Management Support Branch

Subject: Westinghouse Owners Group
Transmittal of Report: WCAP-14040-NP-A, Revision 2 [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"

Reference: 1) C.I. Grimes to R.A. Newton, Acceptance for Referencing of Topical Report WCAP-14040, Revision 1, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, (TAC # M91749), dated October 16, 1995.

This letter transmits twenty-three (23) copies of the approved topical report WCAP-14040-NP-A, Revision 2 [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", dated January 1996.

Also enclosed is:

1. One (1) copy of the Copyright Notice.

This report provides the final NRC approved Westinghouse Owners Group (WOG) technical documentation and methodology for developing the Reactor Coolant System (RCS) heatup and cooldown curves and Cold Overpressure Mitigating Systems (COMS) setpoints which our members may reference in the administrative controls section of technical specifications for license amendment applications and is provided in accordance with the procedures established in NUREG-0390, "Topical Reports Review Status".

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Correspondence with respect to the copyrighted aspects of this WCAP should be addressed to Mr. N.J. Liparulo, Manager Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, PA 15230-0355.

Very truly yours,

Lee Bush, Chairman
Licensing Subcommittee
Westinghouse Owners Group

LB/ygs

enclosures

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WESTINGHOUSE CLASS 3 (Non-Proprietary)

WCAP-14040-NP-A

METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES

WOG Program
MUHP-3024
Revision 2

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January 1996

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Technology Division
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 16, 1995

Mr. Roger A. Newton, Chairman
Westinghouse Owners Group
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230-0355

Dear Mr. Newton:

SUBJECT: ACCEPTANCE FOR REFERENCING OF TOPICAL REPORT WCAP-14040, REVISION 1, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES" (TAC # M91749)

REFERENCES:

1. L. BUSH TO DOCUMENT CONTROL DESK, TRANSMITTAL OF RESPONSE TO CONCERNS IDENTIFIED DURING REVIEW OF WCAP-14040, REV. 1, ENTITLED "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES," OG-95-54, JUNE 16, 1995.

2. L. BUSH TO DOCUMENT CONTROL DESK, ATTENTION C.I. GRIMES, TRANSMITTAL OF RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING WCAP-14040, REV. 1, ENTITLED "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES," OG-95-061, JULY 18, 1995.

3. L. BUSH TO DOCUMENT CONTROL DESK, ATTENTION C.I. GRIMES, TRANSMITTAL OF RESPONSE TO REQUEST FROM JULY 25, 1995 TELECON FOR ADDITIONAL INFORMATION (RAI) REGARDING WCAP-14040, REV 1, ENTITLED "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES", OG-95-067, AUGUST 15, 1995.

4. L. BUSH TO USNRC DOCUMENT CONTROL DESK ATTENTION C.I. GRIMES, TRANSMITTAL OF PAGE REVISION TO WCAP-14040, REV. 1, ENTITLED "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES," TO ADDRESS INSTRUMENTATION UNCERTAINTIES ASSOCIATED WITH COMS PORV SETPOINTS, OG-95-078, SEPTEMBER 26, 1995.

We have completed our review of the subject topical report and additional information provided in References 1-4 submitted by the Westinghouse Owners Group. We find the report to be acceptable for referencing in the administrative controls section of technical specifications for license amendment applications to the extent specified and under the limitations delineated in the report and the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report when the report appears as a reference in license amendment applications, except to ensure that the material in WCAP-14040 is applicable to the specific plant involved as indicated in the conclusion section of the safety evaluation. Licensees should confirm the applicability to their plants of the methodology in WCAP-14040.

Our acceptance applies only to the matters described in the report. It does not include approval of: 1) the plant-specific surveillance program; 2) the use of plant-specific surveillance data in calculating adjusted reference temperatures; and 3) if applicable, the use of plant-specific residual heat removal suction relief valves as a part of COMS because these are not included in WCAP-14040.

In accordance with procedures established in NUREG-0390, "Topical Reports Review Status," we request that the Westinghouse Owners Group publish this report within three months of receipt of this letter. The final version shall incorporate this letter and the enclosed safety evaluation report between the title page and the abstract. The final version shall include an -A (designating "accepted") after the report identification symbol.

Should our criteria or regulations change so that our conclusions as to the acceptability of the report are invalidated, licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation. If you have questions, please contact either Maggalean W. Weston at 301-415-3151 or John C. Tsao at 301-415-2702.

Sincerely,

CI Grimes

Christopher I. Grimes, Chief
Technical Specifications Branch
Office of Nuclear Reactor Regulation

Enclosure:
As stated

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SAFETY EVALUATION OF TOPICAL REPORT

METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES WCAP-14040, REVISION 1, WESTINGHOUSE ELECTRIC CORPORATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

1.0 INTRODUCTION

By letter dated December 20, 1994, the Westinghouse Owners Group (WOG) submitted for staff review a Westinghouse Electric Corporation topical report, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1 (Reference 1). Also, WOG responded to the staff's request for additional information by letters dated June 16, July 18, August 15, and September 26, 1995 (References 2, 3, 4, 5).

In the late 1980s, the NRC, with input from the nuclear industry, initiated a program to streamline plant technical specifications (TS), thereby improving the overall safety of plant operation. The improvement program produced the "Standard Technical Specifications (STS) for Westinghouse Plants," NUREG-1431. Revision 1 was issued on April 7, 1995. Under the STS format, the pressure-temperature (P-T) limit curves and the setpoints for the cold overpressure mitigating system (COMS) may be removed from TS and placed in a separate document, the "Pressure-Temperature Limits Report" (PTLR). Although relocated, the regulatory requirements for P-T and COMS limits are maintained in Limiting Condition for Operation (LCO) 3.X.X in the STS. In addition, the PTLR will be administratively controlled in Section 5.0 of the STS. Presently, the P-T and COMS limits in TS are revised periodically through the license amendment process, and require staff review and approval. With the PTLR concept, licensees may revise the P-T and COMS limits based on the 10 CFR 50.59 review process without the staff's prior approval.

The staff evaluated WCAP-14040, Revision 1 based on the set of provisions (see Table 1) in a draft generic letter published in the *Federal Register* for public comment on June 2, 1995. The provisions are categorized into the following seven topics relevant to the P-T and COMS limits: (1) neutron fluence calculation, (2) reactor vessel material surveillance program, (3) low-temperature overpressure protection system, (4) adjusted reference temperature, (5) fracture mechanics calculation, (6) minimum temperature requirement, and (7) use of surveillance data.

2.0 EVALUATION

Provision 1: Neutron Fluence Calculation

The staff based its review on a Westinghouse-proposed expanded writeup regarding the neutron methodology in the additional information (Reference 3). The neutron fluence calculations are carried out using forward and adjoint formulations in r,θ geometry of the two-dimensional Discrete Ordinates Transport (DOT) code. The anisotropic scattering is treated with a P_3 expansion of the scattering cross section and the angular discretization is modeled with an S_8 order of angular quadrature. The core power distribution and the neutron source distribution were estimated conservatively, accounting for spectral changes due to plutonium accumulation. The revised version of the report uses the BUGLE-93 cross section library which is based on the data set of the Evaluated Nuclear Data File, Version B-VI (ENDF/B-VI). The DOT code was rebenchmarked to the ENDF/B-VI cross sections using the Poolside Critical Assembly (PCA) simulator experiment at the Oak Ridge National Laboratory (ORNL), surveillance capsule and cavity dosimetry measurements. It is stated that results of analytic sensitivity studies showed that the methodology is capable of providing best estimate fluence evaluations within ± 20 percent (1σ).

The methodology summarized above evolved over many years and has been validated using NRC-sponsored experimental measurements (i.e., the PCA experiment at ORNL as well as the reactor surveillance capsule data base). The latest improvement was the introduction of the ENDF/B-VI-based inelastic scattering cross sections for iron. In WCAP-14040, the DOT code was rebenchmarked with the new cross sections. Methodologies incorporating the same elements as in WCAP-14040 have been used in staff-sponsored work, and formed the basis for similar methodology approvals. The staff finds that WCAP-14040 incorporates state-of-the-art fast neutron radiation transport; therefore, the staff finds it acceptable.

Provision 2: Reactor Vessel Material Surveillance Program

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

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

<u>Provision 3: Low Temperature Overpressure Protection System</u>

The methodology for calculating setpoints for the low temperature overpressure protection system is referred to as the cold overpressure mitigating system (COMS) in WCAP-14040. COMS is designed to provide the capability, during reactor operation at low-temperature conditions, to automatically prevent the reactor coolant system (RCS) pressure from exceeding the applicable limits established by Appendix G to 10 CFR Part 50. COMS is manually enabled by reactor operators on the basis of its predetermined enable temperature during reactor startup and shutdown. WCAP-14040 specifies a methodology of developing a COMS which uses the pressurizer power-operated relief valves (PORVs) with variable setpoints. After COMS is enabled, it will automatically function to mitigate overpressure. However, the WCAP-14040 method does not use the relief valve(s) at the suction line of the residual heat removal (RHR) system in the design basis of COMS.

The design basis of COMS considers both mass-addition and heat-addition transients. WCAP-14040 defines the mass-addition transient as a mass injection scenario when the RCS is water solid. The transient is postulated as the simultaneous isolation of the RHR and letdown systems coupled with a full charging pump flow because of the flow control failure. However, WCAP-14040 assumes only one charging pump running during the transient. The staff considers that the WCAP method is not conservative because it did not consider the worst-scenario case in which all charging pumps and safety injection pumps are running and injecting water into the RCS. The one-charging pump assumption is applicable only for plants whose TS restricts only one charging pump operable during the COMS operation. For those plants that do not have this TS restriction, an inadvertent actuation of safety injection may cause all operable charging and safety injection pumps to deliver flow to a water-solid RCS. In response to the staff's request for modification of its COMS design basis, WOG, in its letter dated August 15, 1995, stated that Section 3.1 of WCAP-14040, Revision 1 will be modified to state the following: "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 Specification during the Modes when COMS is required to be in operation." The staff considers that these additional design criteria would ensure that the most limiting mass-addition transients will be analyzed in the design of COMS.

For the heat-addition transient, WCAP-14040 assumes that the most limiting case is the startup of a reactor coolant pump (RCP) in a single loop with the RCS temperature being as much as 50 °F lower than the steam generator secondary-side temperature and the inadvertent isolation of the RHR system. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. This assumption is conservative provided that the plant TS restricts startup of an RCP when the steam generator secondary-side temperature is more than 50 °F higher than the RCS temperature.

The major function of COMS is to protect the structural integrity of the reactor vessel from excessive pressure and temperature loadings. In order to achieve this purpose, the P-T limits established for the RCS per the requirement of Appendix G to 10 CFR Part 50 are considered as the upper limits for the RCS during postulated transient conditions. However, since the overpressure events most likely occur during isothermal conditions in the RCS, the steady-state Appendix G limits are used for the design of COMS. Also, COMS provides for an operational consideration to maintain the integrity of the PORV piping. An upper pressure limit of 800 psia is selected for this purpose. This maximum pressure is selected on the basis of a generic study by Westinghouse using a type of PORV which would cause maximum back pressure in the piping during an overpressure transient. The lower limit of the RCS pressure during a transient is based on an operational consideration for maintaining a normal pressure differential across the RCP No. 1 seals for proper RCP operation. When 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.

The methodology for developing the PORV setpoints intends to provide adequate protection for reactor vessel integrity and maintain proper operational margins. In calculating the PORV setpoints, plant parameters and transient conditions listed in Section 3.2.1 of WCAP-14040 are considered. This list contains initial RCS and steam generator parameters, PORV size and lifting characteristics, mass and heat input rate to the RCS, pressure limits to be protected and other parameters and conditions. These data were included in a specialized version of the LOFTRAN computer code which calculates the maximum and minimum RCS pressures due to overshoot and undershoot of RCS pressure under various overpressure transient conditions. The function generator used to program the PORV setpoints curve has a number of programmable break points (typically nine points). The break points were selected so that the P-T limits are fully protected with the PORV setpoints curve established by connecting these break points. Each of the two PORVs may have a different pressure setpoints curve. The staggered setpoints for two PORVs would prevent excessive pressure undershoot that would challenge the RCP No. 1 seal performance criteria. However, each PORV with its setpoints will protect the P-T limits, assuming a single failure of the other PORV.

Section 3.2 of WCAP-14040, Revision 1 originally stated that since the P/T limits are conservatively determined, the uncertainties in the pressure and temperature instrumentation utilized by the COMS are not explicitly account for in the selection of the PORV setpoints. The staff finds this approach unacceptable. In response to the staff's request, WOG, in its letter dated September 26, 1995, committed to revise Section 3.2 of WCAP-14040 to state that the instrument uncertainties will be accounted for in the selection of PORV setpoints using a process described by Instrument of America (ISA) Standard 67.04-1994. The staff considers that the WOG proposed change of WCAP-14040 regarding its treatment of ins:rumentation uncertainties acceptable.

The "enable" temperature is the RCS temperature below which COMS is required to function. This temperature is specified in Branch Technical Position RSB 5-2, attached to Standard Review Plan (SRP) Section 5.2.2. It is the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90 °F at the 1/4t or 3/4t reactor vessel beltline location (t = beltline thickness). Above this temperature, brittle fracture of the reactor vessel is not expected. The enable temperature calculation in WCAP-14040 is consistent with the staff position stated in Branch Technical Position RSB 5-2 and, therefore, acceptable.

WCAP-14040 discussed ASME Code Case N-514, which provides an alternative setpoint method to Appendix G to Section III of the ASME Code and SRP Section 5.2.2. The code case allows (1) the maximum pressure of the PORV setpoints to 110 percent of the P-T limits and 2) the enable temperature of 200 °F or the RCS temperature corresponding to a reactor vessel metal temperature of at least RT_{NDT} + 50 °F, whichever is greater. Code Case N-514, which is incorporated into Appendix G to Section XI of the ASME Code (1994 Addenda), has not been endorsed by the NRC in its regulations. In the interim, NRC must approve an exemption to the regulations before this code case can be applied in the design of COMS.

Provision 4: Adjusted Reference Temperature (ART)

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

$$ART = IRT_{NDT} + \Delta RT_{NDT} + margin \qquad (1)$$

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

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

where, CF is the chemistry factor, and f is the fast neutron fluence at a specific depth. The chemistry factor is obtained from RG 1.99, Revision 2 based on copper and nickel content of the vessel material. Another method of calculating the chemistry factor is to use credible surveillance data. The fast neutron fluence is calculated for any depth into the vessel wall by the following equation:

$$f = f_{surface} \times \exp(-0.24 \times d) \quad (3)$$

where, $f_{surface}$ is the neutron fluence at the base metal surface of the vessel at the position of the postulated defect and d (in inches) is the depth into the vessel wall measured from the interface of vessel cladding and base metal.

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

$$\text{Margin} = 2\sqrt{(\sigma_i^2 + \sigma_\Delta^2)} \quad (4)$$

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

Provision 5: Fracture Mechanics Calculation

Westinghouse used linear elastic fracture mechanics in Appendix G to Section XI of the ASME Code in calculating P-T limits. The method is based on restricting the stress intensity factor of the postulated defect to be less than the reference stress intensity factor of the reactor vessel material, K_{Ia}. The K_{Ia} is determined by the metal temperature and RT_{NDT} at the tip of the postulated flaw. The flaw is assumed to have a depth of one-fourth of the beltline thickness and a length of 1.5 times the beltline thickness. The K_{Ia} curve in the ASME Code is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 \times \exp[0.0145(T-RT_{NDT} + 160)] \quad (5)$$

where, T is the metal temperature and RT_{NDT} is the ART value of the limiting vessel material at the 1/4t and 3/4t locations of the vessel wall. In Appendix G to Section III of the ASME Code (1995 Edition), the reference stress intensity factor is denoted as K_{IR}, whereas in Appendix G to Section XI, the reference stress intensity factor is denoted as K_{Ia}. However, K_{IR} and K_{Ia} curves are identical and their equations have the same functional forms and coefficients.

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

$$C \times K_{IM} + K_{IT} < K_{Ia} \quad (6)$$

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

The K_{IT} is determined using the following one-dimensional heat conduction equation and boundary conditions:

$$\rho C\ dT/dt = K\ [d^2T/dr^2 + 1/r\ dT/dr] \quad (7)$$

$$\text{at } r = r_i,\ -K\ dT/dr = h(T-T_c) \quad (8)$$

$$\text{at } r = r_o,\quad dT/dr = 0 \quad (9)$$

where, r_i and r_o are the reactor vessel inner and outer radius, respectively; ρ is the material density; C is the material specific heat; K is the material thermal conductivity; T is the local temperature; r is the radial location; t is the time; h is the heat transfer coefficient between the coolant and the vessel wall; and T_c is the coolant temperature. Solving Equations 7, 8, and 9 gives a vessel-wall location and time-dependent temperature distribution for specified heatup and cooldown rates. This temperature distribution is placed in the following equation for thermal stress in a hollow cylinder:

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

where, E is the modulus of elasticity, α is the coefficient of linear expansion, and v is Poisson's ratio. Solving this equation yields a position and time-dependent distribution of hoop thermal stress, $\sigma_\theta(r,t)$. The linear bending (σ_b) and constant membrane (σ_m) stress of the hoop thermal stress are approximated by the linearization technique in Appendix A of Section XI of the ASME Code. After determining the bending and membrane stresses, the K_{IT} is calculated by the following equation:

$$K_{IT} = [1.1\ \sigma_m\ M_k + \sigma_b\ M_b]\ \sqrt{(\pi a/Q)} \quad (10)$$

where, M_k and M_b are correction factors for membrane and bending stresses, respectively, and Q is the flaw shape factor. Westinghouse's use of Equation 10 is consistent with the Welding Research Council Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." The K_{IT} solution will be used in combination with K_{Ia} (Equation 5) to solve for $K_{IM(max)}$ using Equation 6:

$$K_{IM(max)} = (K_{Ia} - K_{IT})\ /\ 2.0 \quad (11)$$

The maximum allowable pressure stress for a given temperature is determined using an iterative process and the following three equations:

$$Q = \phi^2 - 0.212\ (\sigma_p/\sigma_y)^2 \quad (12)$$

$$\sigma_p = K_{IM(max)}/[1.1\ M_k\ \sqrt{(\pi a/Q)}] \quad (13)$$

$$K_{IP} = 1.1\ M_k\ \sigma_p\ \sqrt{(\pi a/Q)} \quad (14)$$

where ϕ is the elliptical integral of the second kind, σ_p is the pressure stress, σ_y is the yield stress, M_k is the correction factor for constant membrane stress, a is the crack depth at the 1/4t location, and K_{IP} is the stress intensity factor caused by vessel internal pressure.

The solution to the iterative process is the allowable pressure stress which can be used to calculate the allowable pressure by the following equation:

$$P(T_c) = \sigma_p \,[(r_o^2 - r_i^2)/(r_o^2 + r_i^2)] \qquad (15)$$

The steady-state, cooldown, and heatup P-T curves are determined using this process. For steady state, K_{IT} is zero and K_{Ia} is determined at the 1/4t location (the most restrictive location). For cooldown, K_{IT} and K_{Ia} are determined at the 1/4t location. The P-T curve at 1/4t is compared with the steady-state curve. The allowable pressure for cooldown is determined by the lesser of the two values, and the resulting curve is the composite cooldown limit curve. For heatup, K_{IT} and K_{Ia} are determined at the 1/4t and 3/4t locations. The P-T curves at 1/4t, 3/4t, and steady-state are compared. The lowest of the three for each heatup rate is used to generate the composite heatup limit curve. The composite cooldown limit curve and composite heatup limit curve provide the allowable operating range for operation. The staff finds the WCAP-14040 methodology consistent with Appendix G to Section III of the ASME Code and SRP Section 5.3.2.

Provision 6. Minimum Temperature Requirement

Appendix G of 10 CFR Part 50 imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. When the core is critical (other than for the purpose of low-level physics test), the temperature of the reactor vessel must not be lower than 40 °F above the minimum temperature of heatup and cooldown curves and must not be lower than the minimum temperature for the inservice hydrostatic pressure test. WCAP-14040 imposes these requirements to the P-T limit curves.

Provision 7. Use of Surveillance Data

WCAP-14040 stated that when two or more credible surveillance capsules have been removed, the measured increase in 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 plus two standard deviation specified in RG 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. WCAP-14040 is consistent with the staff prescribed provisions in Table 1.

3.0 CONCLUSION

The staff concludes that WCAP-14040, Revision 1, is acceptable because

(1) WCAP-14040 incorporates state-of-the-art fast neutron radiation transport.

(2) The COMS methodology satisfies SRP Section 5.2.2 and Branch Technical Position RSB 5-2.

(3) The fracture mechanics calculations conforms to Appendix G of 10 CFR Part 50 and SRP Section 5.3.2.

(4) WCAP-14040 conforms to RG 1.99, Revision 2, to calculate adjusted reference temperature.

(5) The methodology of calculating the minimum temperature in the P-T limit curves conforms to Appendix G to 10 CFR Part 50.

(6) WCAP-14040 satisfies the provisions described in a draft generic letter published in the *Federal Register* for public comment on June 2, 1995.

However, the staff recommends that applicants and licensees who use WCAP-14040, Revision 1 as the methodology for developing their plant-specific P-T limits and COMS setpoints, should address in their PTLR submittal the following topics that were not discussed in WCAP-14040:

(1) Plant-specific surveillance program.

(2) The use of plant-specific surveillance data in calculating adjusted reference temperatures.

(3) If applicable, the use of relief valves of the RHR system as a part of COMS.

4.0 REFERENCES

1. Letter from R.A. Newton of Westinghouse Owners Group to USNRC Document Control Desk (Attention J.R. Strosnider), Subject: Transmittal of WCAP-14040, Revision 1, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," December 20, 1994.

2. Letter from L. Bush of Westinghouse Owners Group to USNRC Document Control Desk (Attention Chief, Planning, Program and Management Support Branch), Subject: Transmittal of Response to Concerns Identified During Review of WCAP-14040, Revision 1, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," June 16, 1995.

3. Letter from L. Bush of Westinghouse Owners Group to USNRC Document Control Desk (Attention C.I. Grimes), Subject: Transmittal of Response to Request for Additional Information (RAI) Regarding WCAP-14040, Rev. 1, Entitled "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," OG-95-061, July 18, 1995.

4. Letter from L. Bush of Westinghouse Owners Group to USNRC Document Control Desk (Attention C.I. Grimes), Subject: Transmittal of Response

to Request From July 25, 1995 Telecon for Additional Information (RAI) Regarding WCAP-14040, Rev. 1, Entitled "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," OG-95-067, August 15, 1995.

5. Letter from L. Bush of Westinghouse Owners Group to USNRC Document Control Desk (Attention C.I. Grimes), Subject: Transmittal of Page Revision to WCAP-14040, Rev. I, Entitled "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," to Addrcss Instrumentation Uncertainties Associated with COMS PORV Setpoints, OG-95-078, September 26, 1995.

TABLE 1

REQUIREMENTS FOR METHODOLOGY AND PTLR

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR
1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).	Describe transport calculation methods, including computer codes and formulas used to calculate neutron fluence. Provide references.	Provide the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The withdrawal schedule for reactor vessel material surveillance specimens shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.	Briefly describe the surveillance program. Licensee transmittal letter should identify by title and number the report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. Topical/generic report contains placeholder only. Reference Appendix H to 10 CFR Part 50.	Provide the surveillance capsule withdrawal schedule, or reference by title and number those documents that contain the schedule. Reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data.
3. Low temperature overpressure protection (LTOP) system lift setting limits developed using NRC-approved methodologies may be included in the PTLR.	Describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics. Reference SRP Section 5.2.2; Code Case N-514; ASME Code, Appendix G, Section XI as applied in accordance with 10 CFR 50.55.	Provide setpoint curves or setpoint values.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.	Describe the method for calculating the ART using Regulatory Guide 1.99, Revision 2.	Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (t = vessel beltline thickness). PWRs - identify RT_{PTS} value in accordance with 10 CFR 50.61

REQUIREMENTS FOR METHODOLOGY AND PTLR (Continued)

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800, SRP Section 5.3.2, "Pressure-Temperature Limits."	Describe the application of fracture mechanics in constructing P-T curves based on ASME Code, Appendix G, Section XI, and SRP Section 5.3.2.	Provide the P-T curves for heatup, cooldown, criticality, and hydrostatic and inservice leak tests.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.	Describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T curves.	Identify minimum temperatures on the P-T curves,such as minimum boltup temperature and hydrotest temperature.
7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}, where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Describe procedure if measured value exceeds predicted value. <u>WHEN OTHER PLANT DATA ARE USED</u> 1. Identify the source(s) of data. 2.a Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability. **OR** 2.b Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation. Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.

LEGAL NOTICE

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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. Currently, there is no Westinghouse topical report that contains an NRC approved methodology for developing the RCS heatup and cooldown curves and COMS setpoints that can be referenced to implement the PTLR. 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 a methodology 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}),

- The limiting toughness is based upon a reference value (K_{Ia}), which is a lower bound of the dynamic crack initiation or arrest toughnesses, and

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

This section describes the methodology used by Westinghouse Electric Corporation 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:

$$\phi(R_0,\theta_0) = \int_r \int_\theta \int_E I(r,\theta,E)\ S(r,\theta,E)\ r\ dr\ d\theta\ dE$$

Where: $\phi(R_0,\theta_0)$ = Neutron flux (E > 1.0 MeV) at radius R_0 and azimuthal angle θ_0.

$I(r,\theta,E)$ = 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, $I(r,\theta,E)$, 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:

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

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 fracture toughness 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 10CFR

50.12(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:

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

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:

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

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:

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

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:

$$\text{Margin} = 2\sqrt{(\sigma_I^2 + \sigma_\Delta^2)} \qquad (2.4\text{-}5)$$

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, K_{Ia}, for the metal temperature at that time. 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} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RTNDT 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.)

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

$$C * K_{IM} + K_{IT} < K_{Ia} \qquad (2.5\text{-}2)$$

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

(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, K_{Ia} 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). 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 (2.5-2), 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 K_{Ia} at the

1/4t location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ia} 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 K_{Ia} for the inside 1/4t flaw during heatup is lower than the K_{Ia} 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 K_{Ia}'s 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 (RTNDT), 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.

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

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

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

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

$$\text{at } r = r_o, \qquad \frac{\partial T}{\partial r} = 0 \qquad (2.6.1\text{-}3)$$

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].

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

where,

$\sigma_\theta(r,t)$	=	hoop stress at location and time t
E	=	modulus of elasticity
α	=	coefficient of linear expansion
ν	=	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:

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

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:

$$K_{IM(max)} = \frac{K_{Ia}(T - RT_{NDT})_{1/4t}}{2.0} \qquad (2.6.2\text{-}1)$$

where,

$K_{Ia}(T\text{-}RT_{NDT})$ = allowable reference stress intensity factor as a function of $T\text{-}RT_{NDT}$ at 1/4t.

Next, the maximum allowable pressure stress is determined using an iterative process and the following three equations:

$$Q = \phi^2 - 0.212 \left[\frac{\sigma_p}{\sigma_y}\right]^2 \qquad (2.6.2\text{-}2)$$

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

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

where,

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

ϕ = is the elliptical integral of the 2nd kind ($\phi = 1.11376$ 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 relative flaw depth (a/t) is shown in Figure 2.4,

a = crack depth of 1/4t,

K_{IP} = pressure stress intensity factor.

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

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

where,

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

2.6.3 Finite Cooldown Rate Analyses

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]:

$$K_{IT} = \left[\sigma_m 1.1\, M_K + \sigma_b M_B\right] \sqrt{\frac{\pi a}{Q}} \tag{2.6.3-1}$$

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]:

$$Q = \phi^2 - 0.212 \left[\frac{\sigma_m + \sigma_b}{\sigma_y} \right]^2 \tag{2.6.3-2}$$

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:

$$K_{IM(max)} = \frac{K_{Ia}(T-RT_{NDT})_{1/4t} - K_{IT}(T_c)_{1/4t}}{2.0} \tag{2.6.3-3}$$

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}) is determined in Equation (2.5-1) 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 equation (2.6.3-1) 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:

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

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

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.

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 XI, 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. KIT=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,

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

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:

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

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, depending on whether the core is critical during the test.

2.7 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.

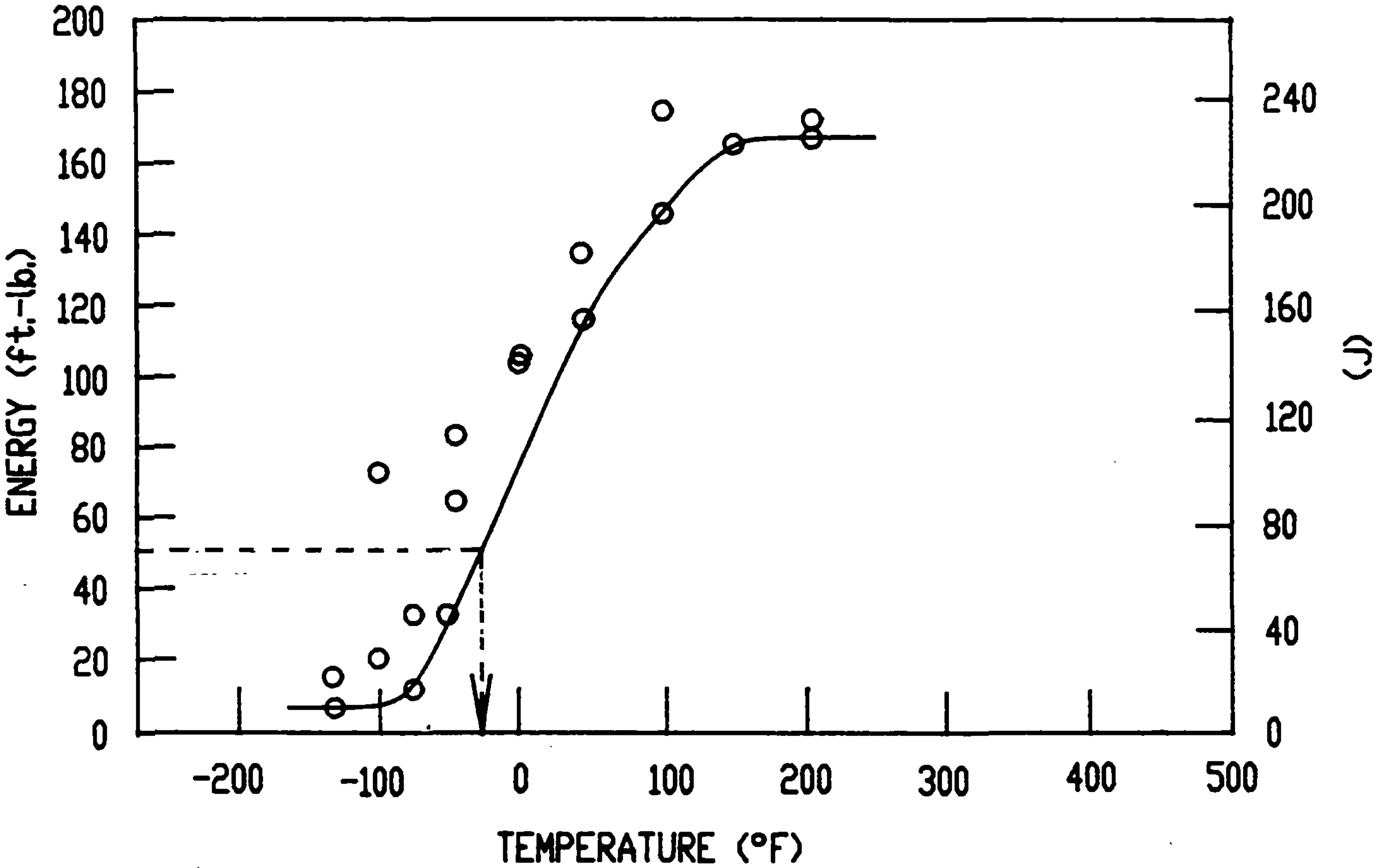


Figure 2.1: Example of a Charpy Impact Energy Curve Used to Determine IRT_{NDT} (Note: 35 mils lateral expansion is required at indicated temperature)

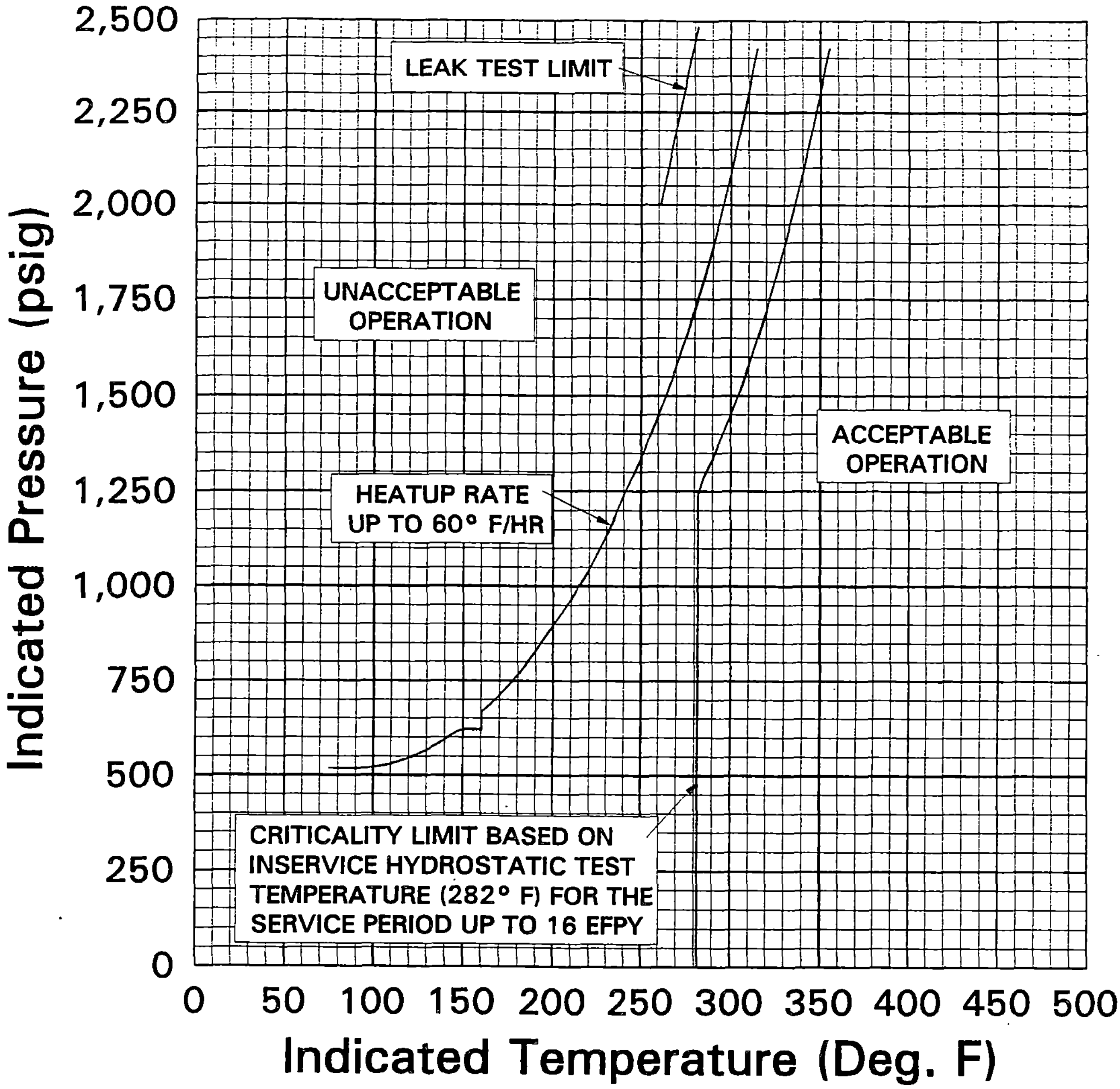


Figure 2.2: Heatup Pressure-Temperature Limit Curve For Heatup Rates up to 60 °F/Hr

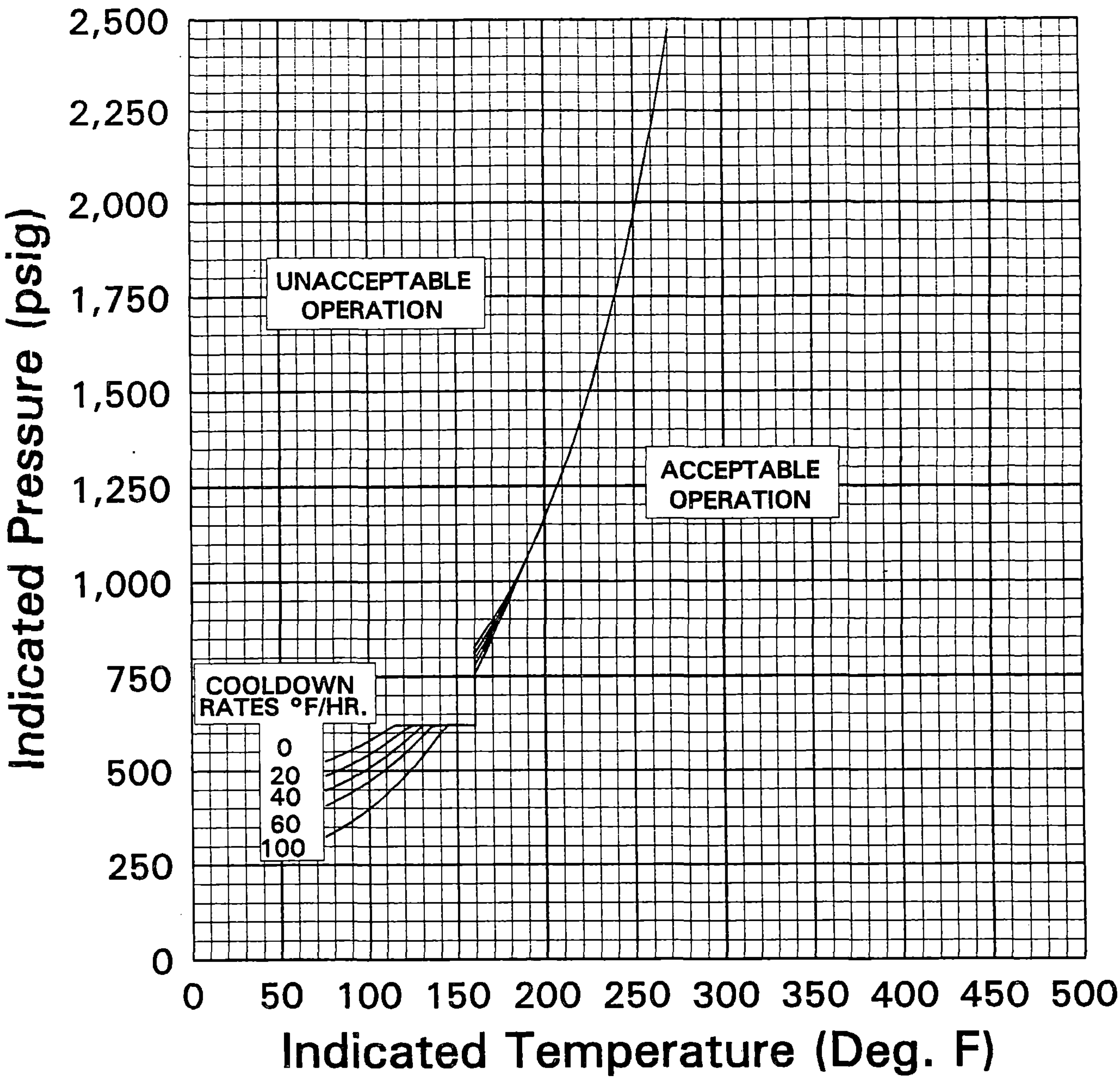


Figure 2.3: Cooldown Pressure-Temperature Limit Curves For Cooldown Rates up to 100°F/Hr

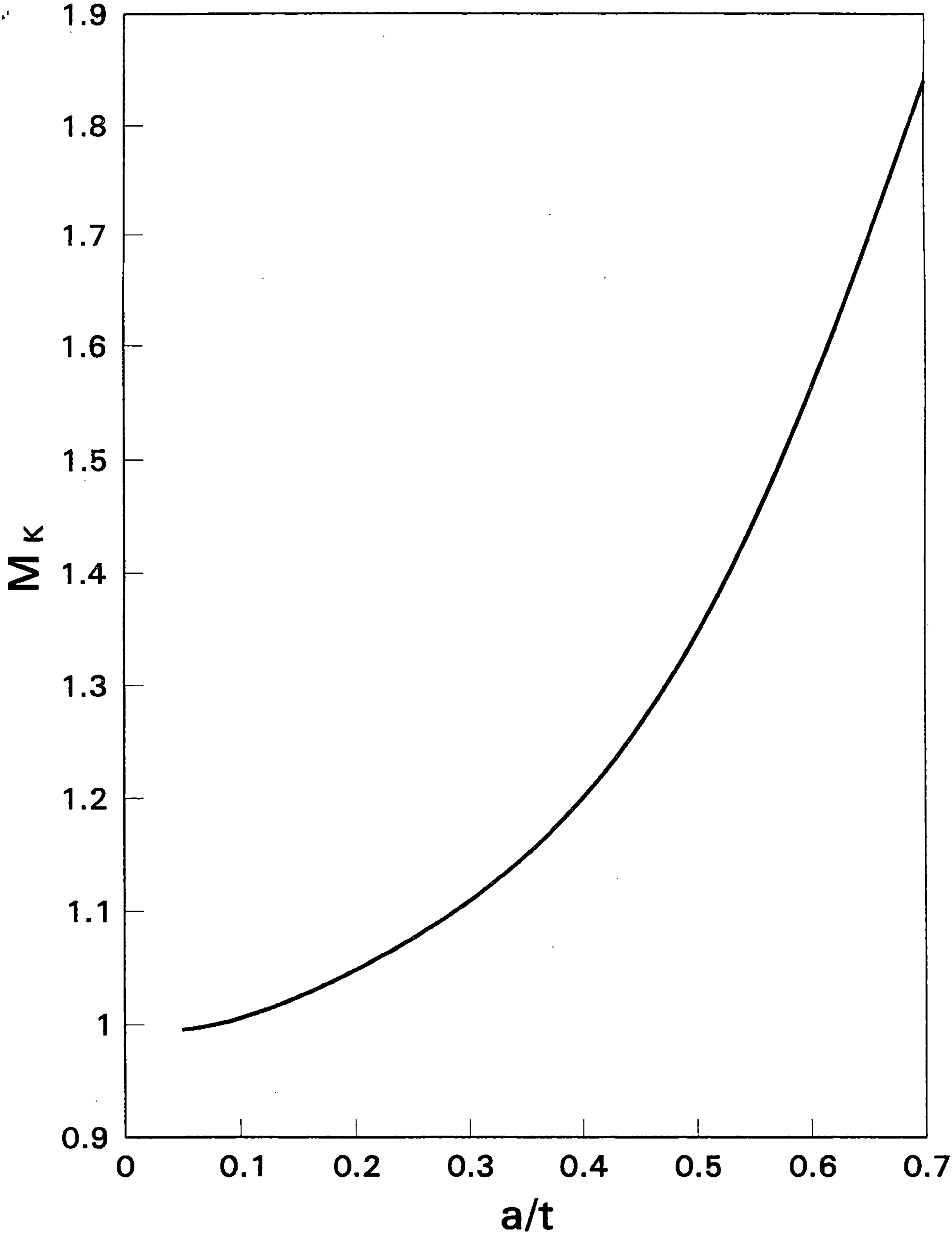


Figure 2.4: Membrane Stress Correction Factor (M_K) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6

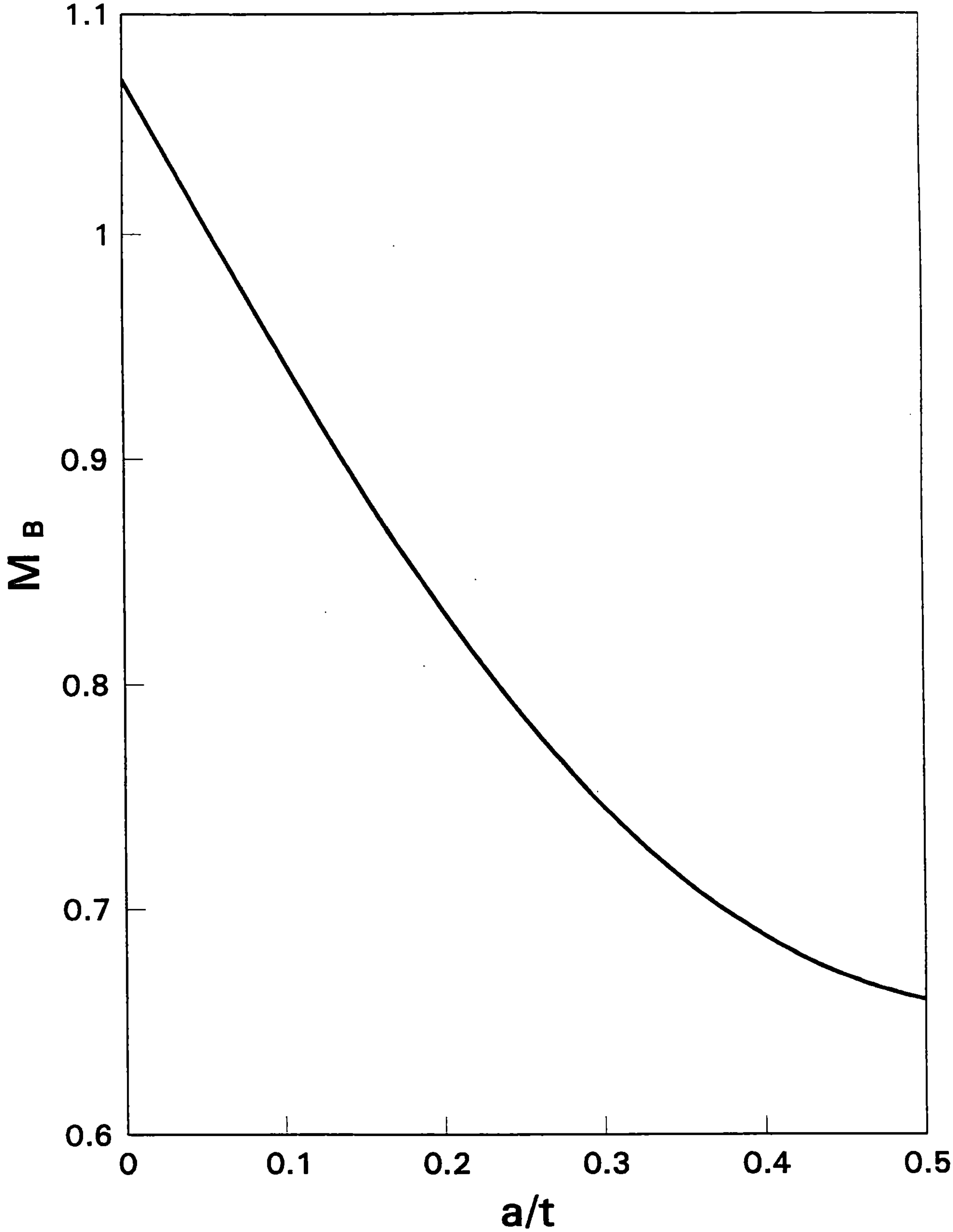


Figure 2.5: Bending Stress Correction Factor (M_B) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6.

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 desig ned 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 ¼T flaw depth with a length equal to 1 ½ 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. Although expected soon, use of Code Case N-514 has not yet been formally approved by the NRC. In the interim, an exemption to the regulations must be granted by the NRC before Code Case N-514 can be used in the determination of the COMS setpoints and enable temperature.

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 also supported by Westinghouse 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.

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

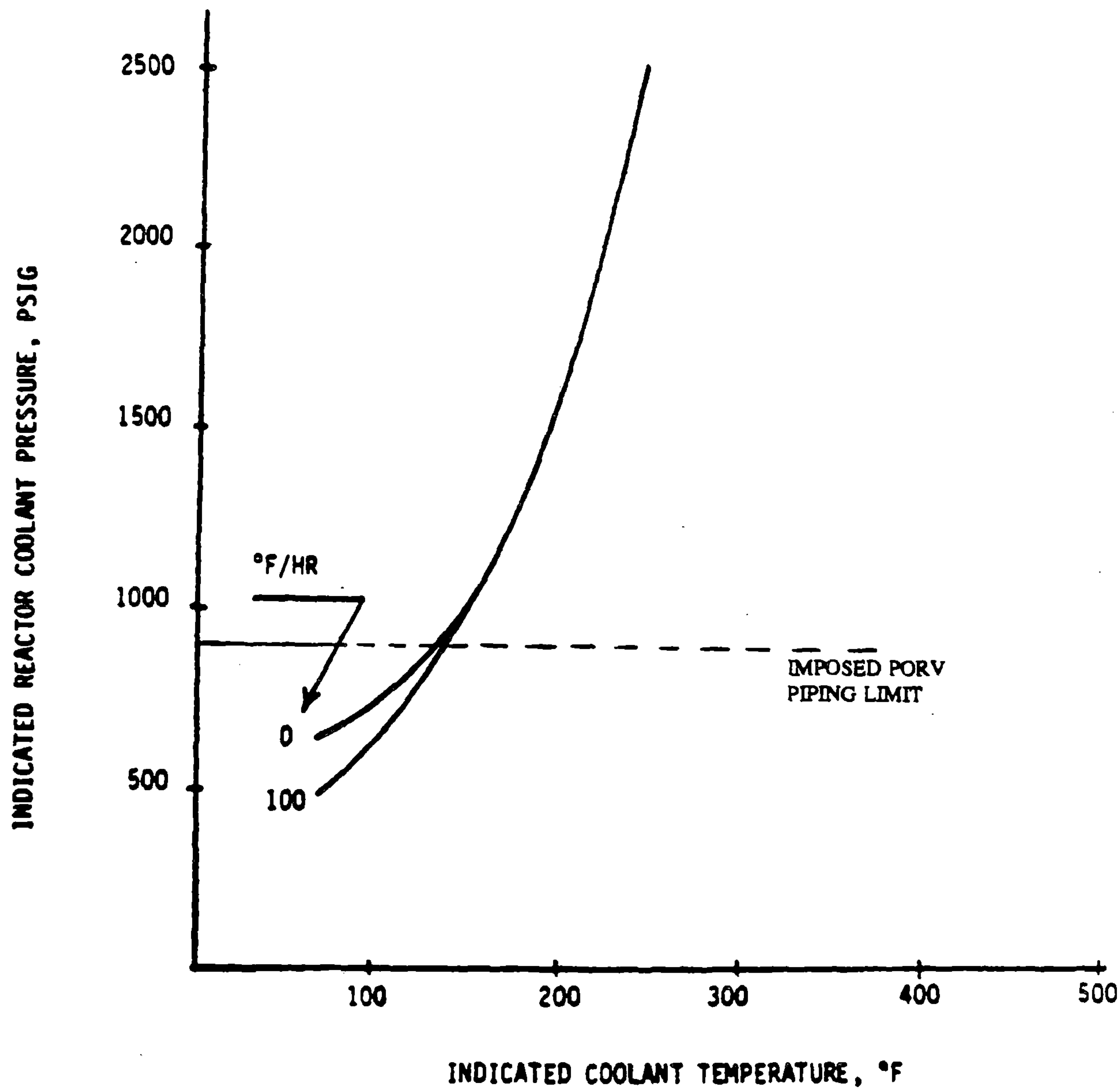


FIGURE 3.2

TYPICAL PRESSURE TRANSIENT

(1 RELIEF VALVE CYCLE)

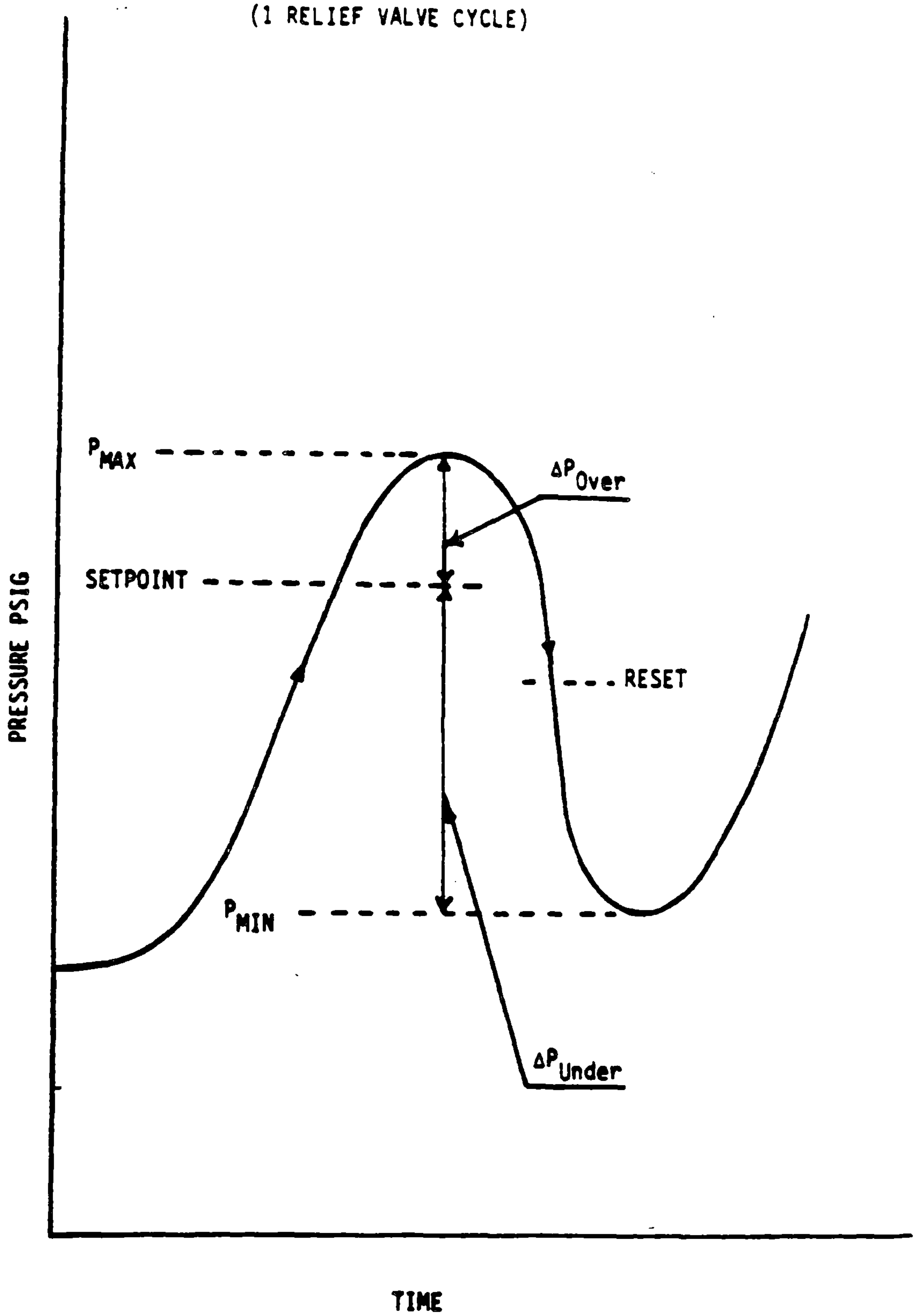
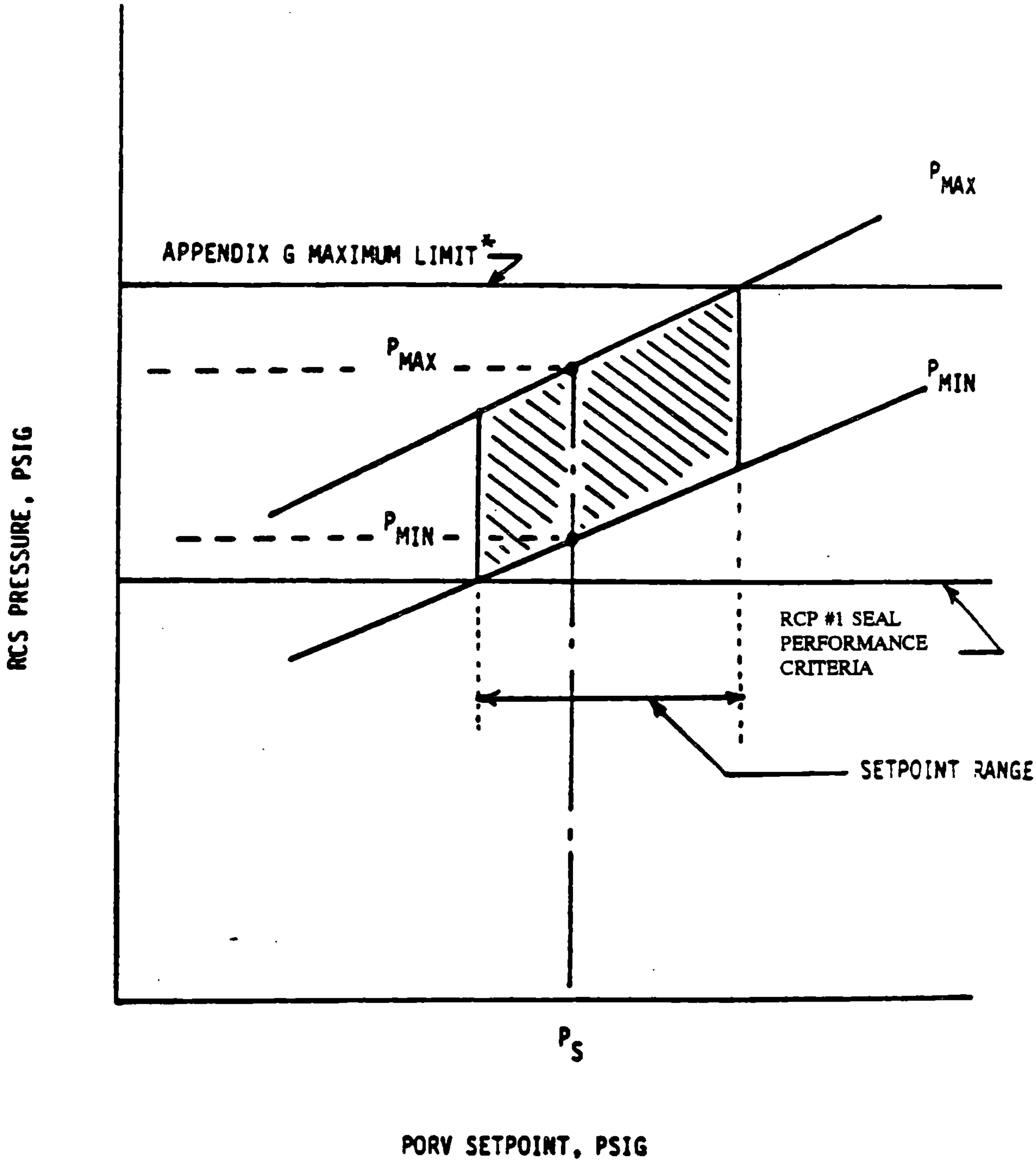


FIGURE 3.3

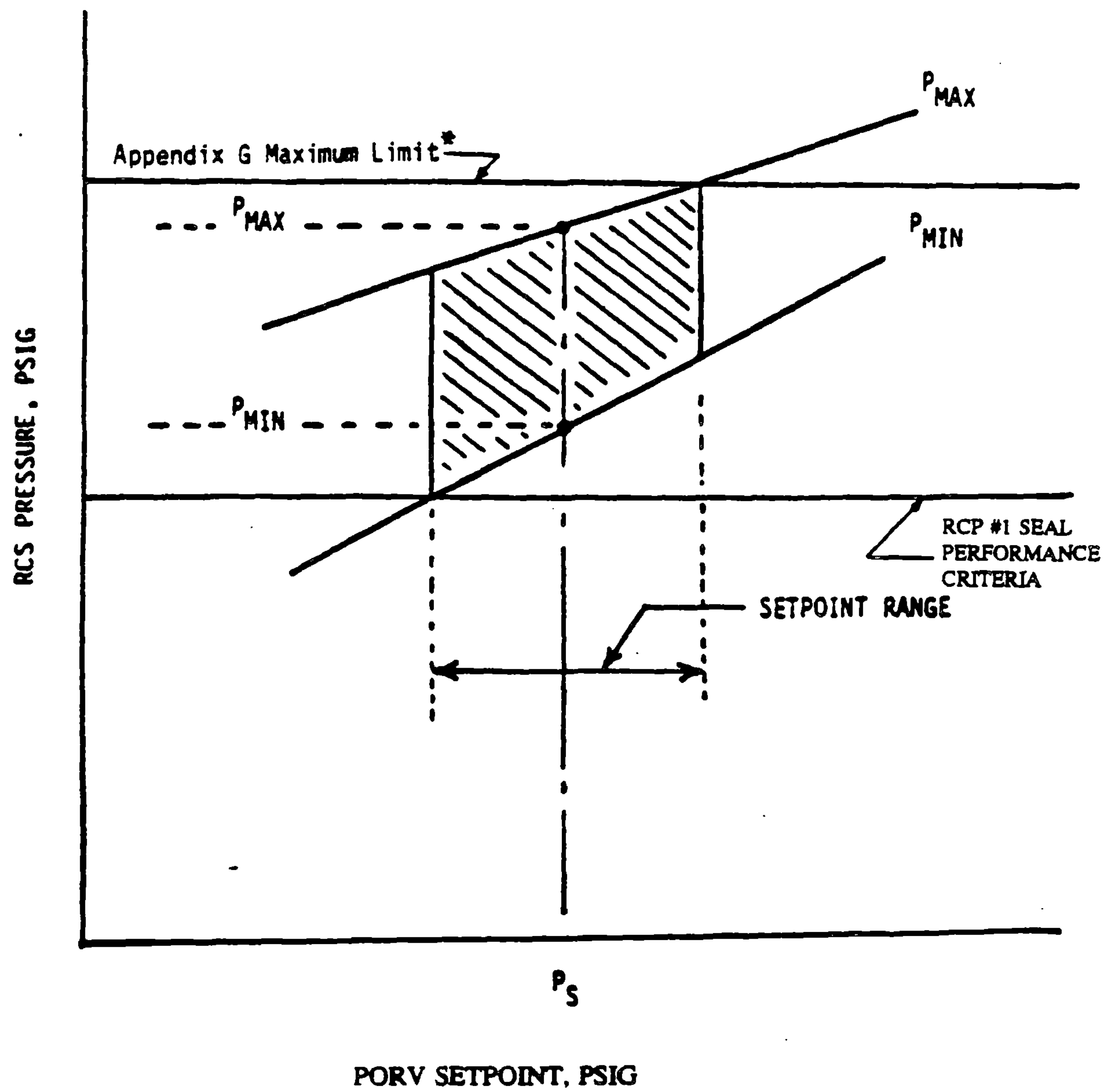
SETPOINT DETERMINATION (MASS INPUT)



* 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)



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

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors", Revision 0, September, 1992.

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", January, 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 A-3000, 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 XI, 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.

APPENDIX A

SAMPLE PRESSURE AND TEMPERATURE LIMITS REPORT

WATTS BAR UNIT 1

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

REVISION 4

RCS PRESSURE AND TEMPERATURE LIMITS REPORT FOR WATTS BAR UNIT 1

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This PTLR for Watts Bar Unit 1 has been prepared in accordance with the requirements of Technical Specification 5.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications affected by this report are listed below:

LCO 3.4.3. RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

2.0 RCS Pressure and Temperature Limits

The limits for LCO 3.4.3 are presented in the subsection which follows. These limits have been developed (Ref. 1, 4) using the NRC-approved methodologies specified in Specification 5.9.6.

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are (Ref. 1):

a. A maximum heatup Rate 100°F per hour.
b. A maximum cooldown Rate 100°F per hour.
c. A maximum temperature change of ≤ 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2.1-1 and 2.1-2 (Ref. 1).

NOTE: The heat-up and cool-down curves are based on beltline conditions and do not compensate for pressure differences between the pressure transmitter and reactor midplane/beltline or for instrument inaccuracies. Refer to Table 2.1-3 for pressure differences (Ref. 2). Site Engineering Setpoint and Scaling documents SSD-1-P-68, -63, -64, -66, and -70 provide the adjusted curves for temperature and pressure limits which are compensated for pressure differential and instrument inaccuracy to be used for heatup and cooldown. R

3.0 Cold Overpressure Mitigation System (LCO 3.4.12)

The lift setpoints for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsection which follows. These lift setpoints have been developed using the NRC-approved methodologies specified in Specification 5.9.6.

3.1 Pressurizer PORV Lift Setting Limits

The pressurizer PORV lift setpoints are specified by Figure 3.1-1 through 3.1-4 and Table 3.1-1 (Ref. 2). The limits for the COMS setpoints are contained in the 1.5 EFPY curves for heatup (Figure 3.1-5 and Table 3.1-2) and Cooldown (Figure 3.1-6 and Table 3.1-3) (Ref. 6) which are based on beltline conditions and are not compensated for pressure differences between the pressure transmitter and the reactor midplane/beltline or for instrument inaccuracies. Refer to Table 2.1-3 for pressure differences (Ref. 2).

NOTE: These setpoints include allowance for pressure difference between the pressure transmitter and reactor midplane. Site Engineering Setpoint and Scaling documents for instrument loop numbers 1-T-68-1B and 1-T-68-43B contain the adjusted curves compensated for pressure differential and instrument inaccuracy which provides the PORV lift limits for the COMS.

4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 4.0-1. The results of these examinations shall be used to update Figures 2.1-1, 2.1-2, and 3.1-1 through 3.1-4.

The pressure vessel steel surveillance program (Ref. 3) is in compliance with Appendix H to 10 CFR 50, entitled "Reactor Vessel Material Surveillance Program Requirements". The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure", to section III of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82. The removal schedule is provided in Table 4.0-1.

5.0 Supplemental Data Tables

Table 5.1 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2, predictions. This table was intentionally left blank since no capsules were removed to date.

Table 5.2 shows calculations of the surveillance material chemistry factors using surveillance capsule data. This table was intentionally left blank since no capsules were removed to date.

Table 5.3 provides the required Watts Bar Unit 1 reactor vessel toughness data. The bolt-up temperature is also included in this table.

Table 5.4 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves.

Table 5.5 provides a summary of the adjusted reference temperature (ART) values of the Watts Bar Unit 1 reactor vessel beltline materials at the 1/4-T and 3/4-T locations for 7 EFPY.

Table 5.6 shows example calculations of the adjusted reference temperature (ART) values at 7 EFPY for the limiting Watts Bar Unit 1 reactor vessel material (Intermediate Shell Forging 05).

Table 5.7 provides a summary of the fluence values used in the PTS evaluation.

Table 5.8 provides RT_{PTS} values for Watts Bar Unit 1 for 32 EFPY.

Table 5.9 provides RT_{PTS} values for Watts Bar Unit 1 for 48 EFPY.

REFERENCES

1. WCAP-13829 Revision 1, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 1", February 1995.

2. Westinghouse Letter to TVA, WAT-D-9448, "Revised COMS PORV Setpoints," August 27, 1993.

3. WCAP-9298, Revision 1, "Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program", April 1993.

4. Westinghouse Letter to TVA, WAT-D-9526, "COMS".

5. WCAP-14040, Revision 1, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", December 1994.

6. WCAP-14176, "Watts Bar Unit 1 Heatup and Cooldown Limit Curves For Normal Operation", September 1994.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
INITIAL RT_{NDT} 47 °F
LIMITING ART AT 7 EFPY: 1/4-T, 181.1 °F
3/4-T, 147.7 °F

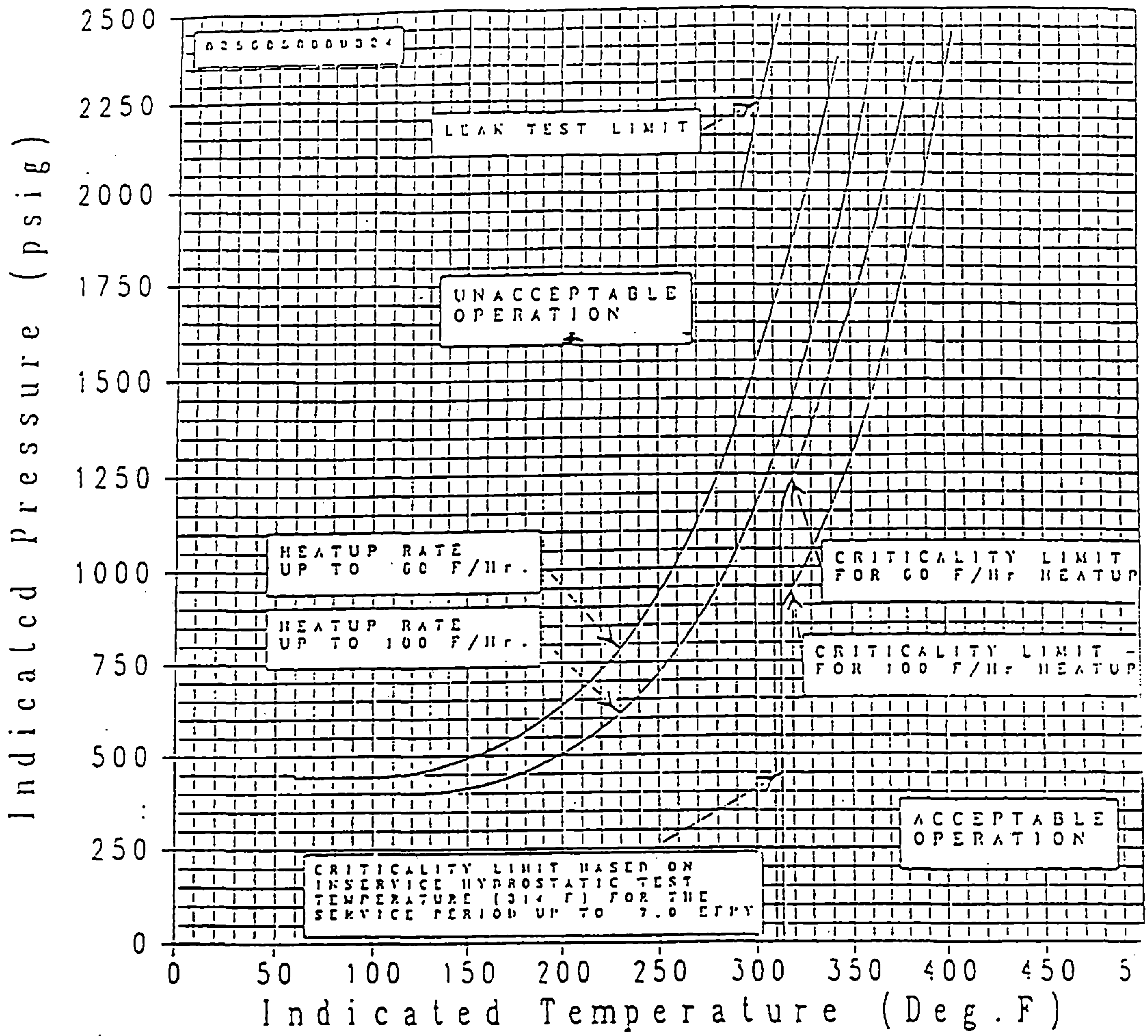


Figure 2.1-1

Watts Bar Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates of 60 and 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)

(Plotted Data (Ref. 1) provided on Table 2.1-1)

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data points plotted on Figure 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMITS (60 °F/HR)	CRITICALITY LIMITS (100 °F/HR)
60	443.05	401.48			
65	443.05	401.48			
70	443.05	401.48			
75	443.05	401.48			
80	443.05	401.48			
85	443.05	401.48			
90	443.05	401.48			
95	443.05	401.48			
100	443.05	401.48			
105	444.11	401.48			
110	446.29	401.48			
115	449.34	401.48			
120	453.33	401.48			
125	458.10	401.48			
130	463.72	402.34			
135	470.07	404.05			
140	477.24	406.56			
145	485.17	409.93			
150	493.84	414.14			
155	503.41	419.16			
160	513.86	425.01			
165	525.20	431.68			
170	537.41	439.16			
175	550.72	447.59			
180	565.12	456.94			
185	580.63	467.24			
190	597.27	478.55			
195	615.30	490.90			
200	634.60	504.29			

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data points plotted on Figure 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMITS (60 °F/HR)	CRITICALITY LIMITS (100 °F/HR)
205	655.51	518.93			
210	678.00	534.82			
215	702.06	551.92			
220	727.92	570.53			
225	755.91	590.51			
230	785.79	612.21			
235	818.05	635.46			
240	852.61	660.68			
245	889.70	687.66			
250	929.52	716.88			
255	972.27	748.15			
260	1018.19	781.76			
265	1067.49	818.06			
270	1120.42	856.92			
275	1177.15	898.66			
280	1237.98	943.47			
285	1303.13	991.58			
290	1373.09	1043.20			
293			2000		
295	1447.99	1098.55			
300	1527.97	1157.76			
305	1612.31	1221.51			
310	1686.16	1289.68			
314			2485	0 to 1120.42	0 to 856.92
315	1765.15	1362.69		1177.15	898.66
320	1849.89	1440.81		1237.98	943.47
325	1940.35	1524.60		1303.13	991.58
330	2037.07	1614.19		1373.09	1043.20

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data points plotted on Figure 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMITS (60 °F/HR)	CRITICALITY LIMITS (100 °F/HR)
335	2140.54	1709.92		1447.99	1098.55
340	2250.90	1812.16		1527.97	1157.76
345	2368.62	1921.54		1612.31	1221.51
350		2038.25		1686.16	1289.68
355		2162.68		1765.15	1362.69
360		2295.42		1849.89	1440.81
365		2436.49		1940.35	1524.60
370				2037.07	1614.19
375				2140.54	1709.92
380				2250.90	1812.16
385				2368.62	1921.54
390					2038.25
395					2162.68
400					2295.42
405					2436.49

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART AT 7 EFPY: 1/4-T, 181.1 °F
3/4-T, 147.7 °F

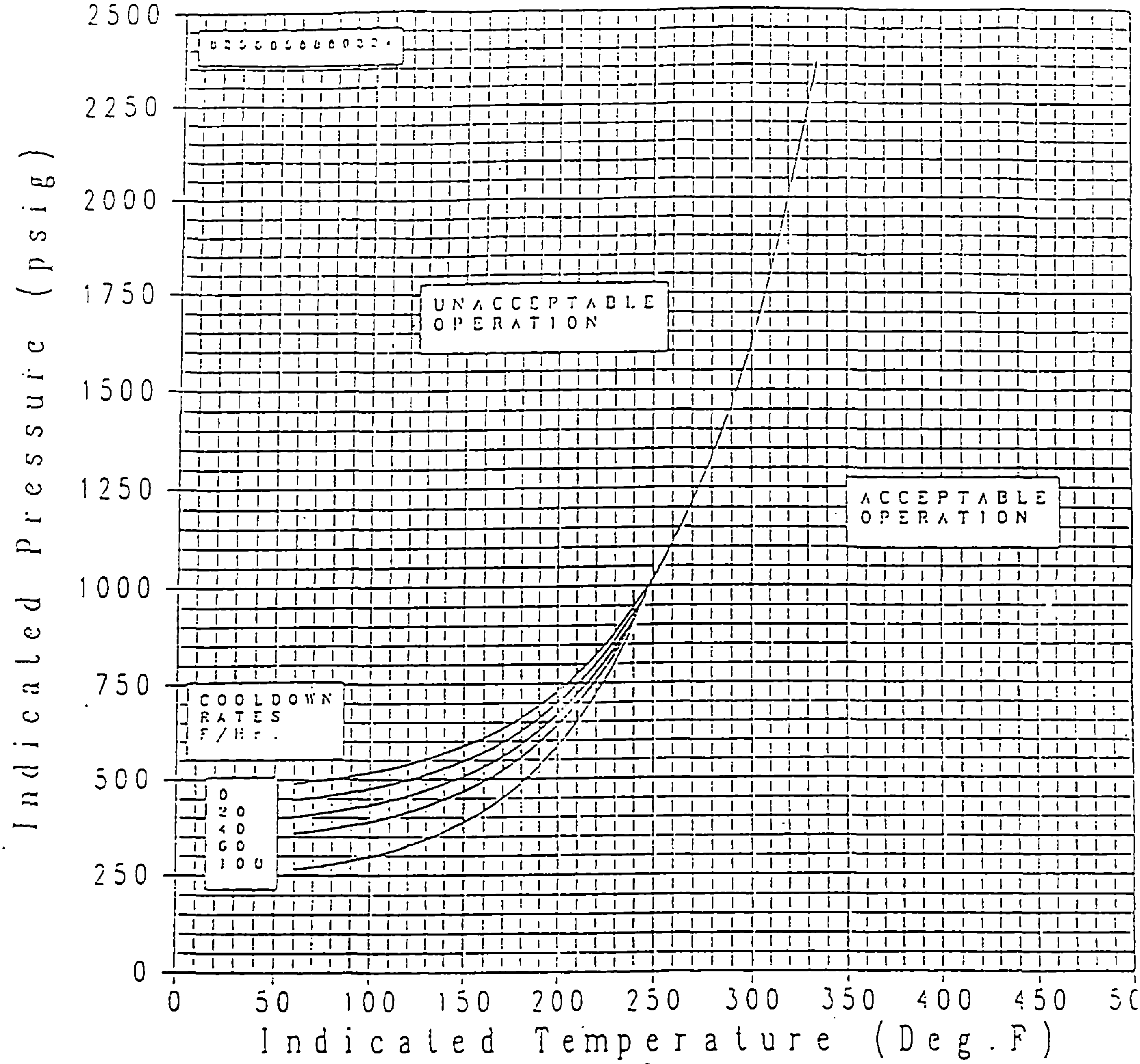


Figure 2.1-2

Watts Bar Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)

(Plotted Data (Ref. 1) provided on Table 2.1-2)

Table 2.1-2
Watts Bar Unit 1 Cooldown Limits
(Data plotted on Figure 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
60	266.59	359.39	404.02	447.48	490.13
65	269.45	362.11	406.70	450.24	492.77
70	272.56	365.11	409.67	453.18	495.72
75	276.04	368.41	412.91	456.38	498.88
80	279.82	371.97	416.40	459.81	502.28
85	283.98	375.89	420.21	463.54	505.94
90	288.53	380.12	424.32	467.55	509.87
95	293.54	384.74	428.79	471.89	514.09
100	298.99	389.73	433.61	476.55	518.64
105	304.97	395.18	438.77	481.60	523.52
110	311.45	401.06	444.40	487.03	528.77
115	318.55	407.40	450.52	492.81	534.42
120	326.20	414.30	457.10	499.12	540.39
125	334.59	421.80	464.24	505.93	546.91
130	343.66	429.89	471.92	513.26	553.93
135	353.56	438.60	480.25	521.17	561.47
140	364.21	448.06	489.20	529.67	569.58
145	375.84	458.32	498.80	538.75	578.31
150	388.41	469.38	509.23	548.62	587.56
155	402.06	481.35	520.50	559.26	597.63
160	416.74	494.17	532.63	570.71	608.47
165	432.73	508.12	545.63	582.92	620.12
170	449.91	523.16	559.73	596.19	632.50
175	468.61	539.30	574.96	610.49	645.97
180	488.79	556.81	591.21	625.85	660.44
185	510.54	575.72	608.89	642.29	675.98
190	534.09	595.96	627.89	660.09	692.57
195	559.46	617.94	648.27	679.12	710.56
200	586.80	641.47	670.32	699.74	729.70
205	616.49	667.02	693.97	721.92	750.49
210	648.35	694.35	719.52	745.65	772.78
215	682.74	723.96	746.96	771.30	796.64
220	719.95	755.76	776.41	798.79	822.23

Table 2.1-2
Watts Bar Unit 1 Cooldown Limits
(Data plotted on Figure 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
225	760.01	789.97	808.37	828.31	849.99
230	803.10	826.78	842.54	860.26	879.55
235	849.59	866.62	879.31	894.47	911.51
240	899.62	909.35	918.85	931.19	945.77
245	953.53	955.36	961.60	970.69	982.53
250	1011.36	1004.81	1007.39	1013.15	1022.00
255		1057.86	1056.69	1058.81	1064.40
260			1109.45	1107.90	1109.98
265					1158.95
270					1211.57
275					1268.05
280					1328.61
285					1393.44
290					1463.21
295					1537.95
300					1618.11
305					1703.92
310					1795.88
315					1894.39
320					1999.72
325					2112.46
330					2232.88
335					2361.41

Table 2.1-3

Pressure Differentials

Number of Pumps	Delta P (psi)
0	5.2
1	31.0
2	38.0
3	52.0
4	74.0

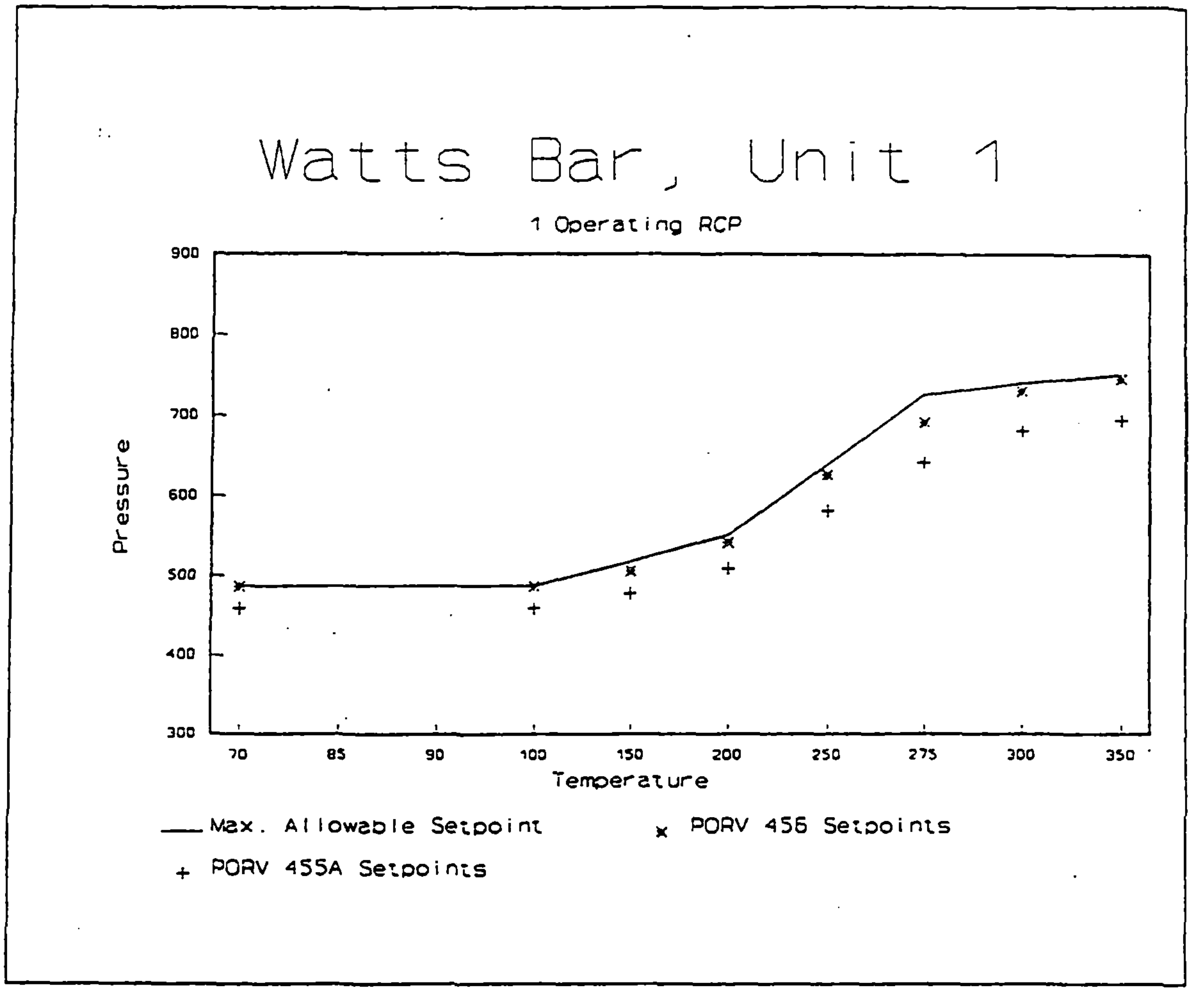


Figure 3.1-1
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

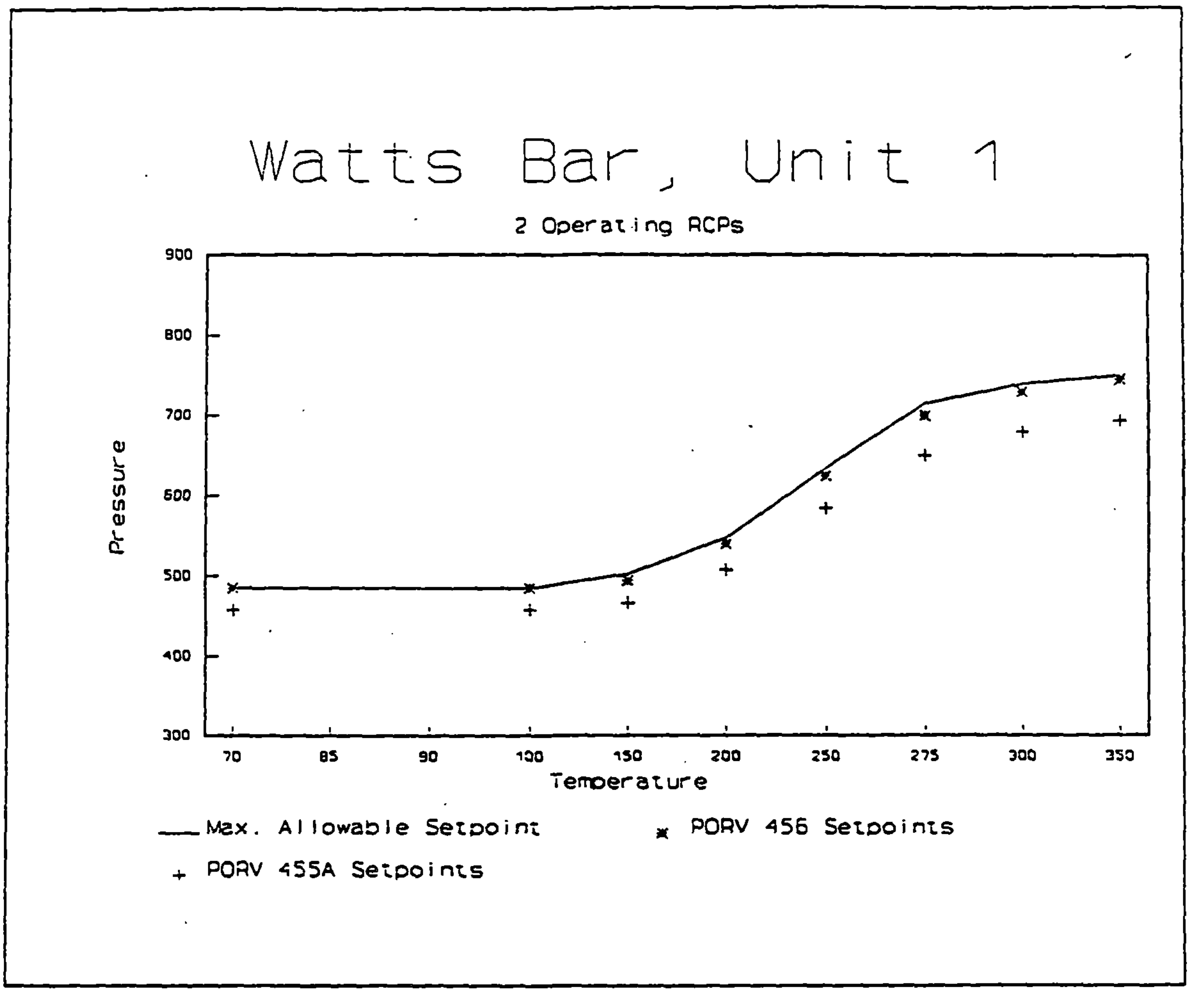


Figure 3.1-2
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

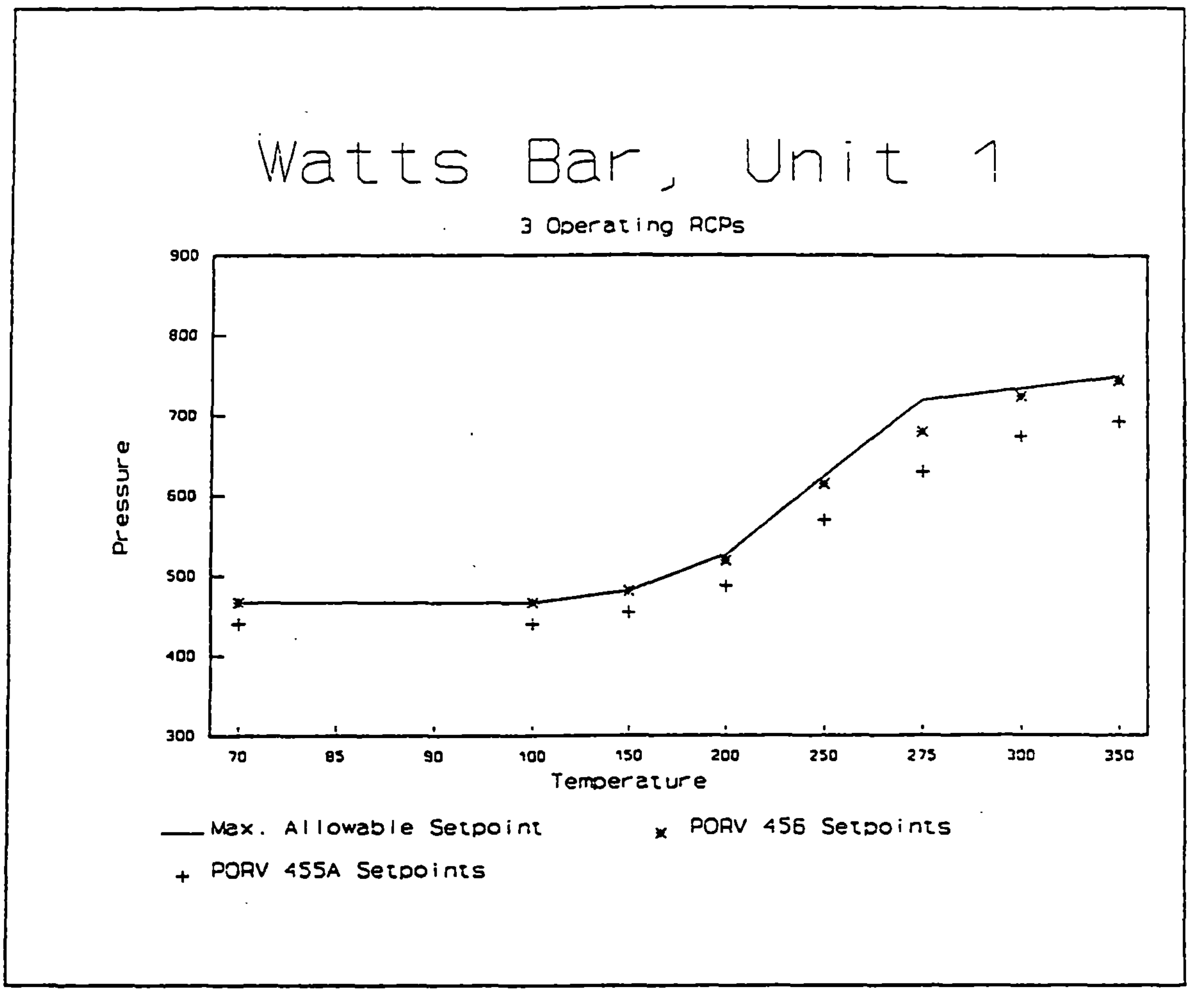


Figure 3.1-3
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

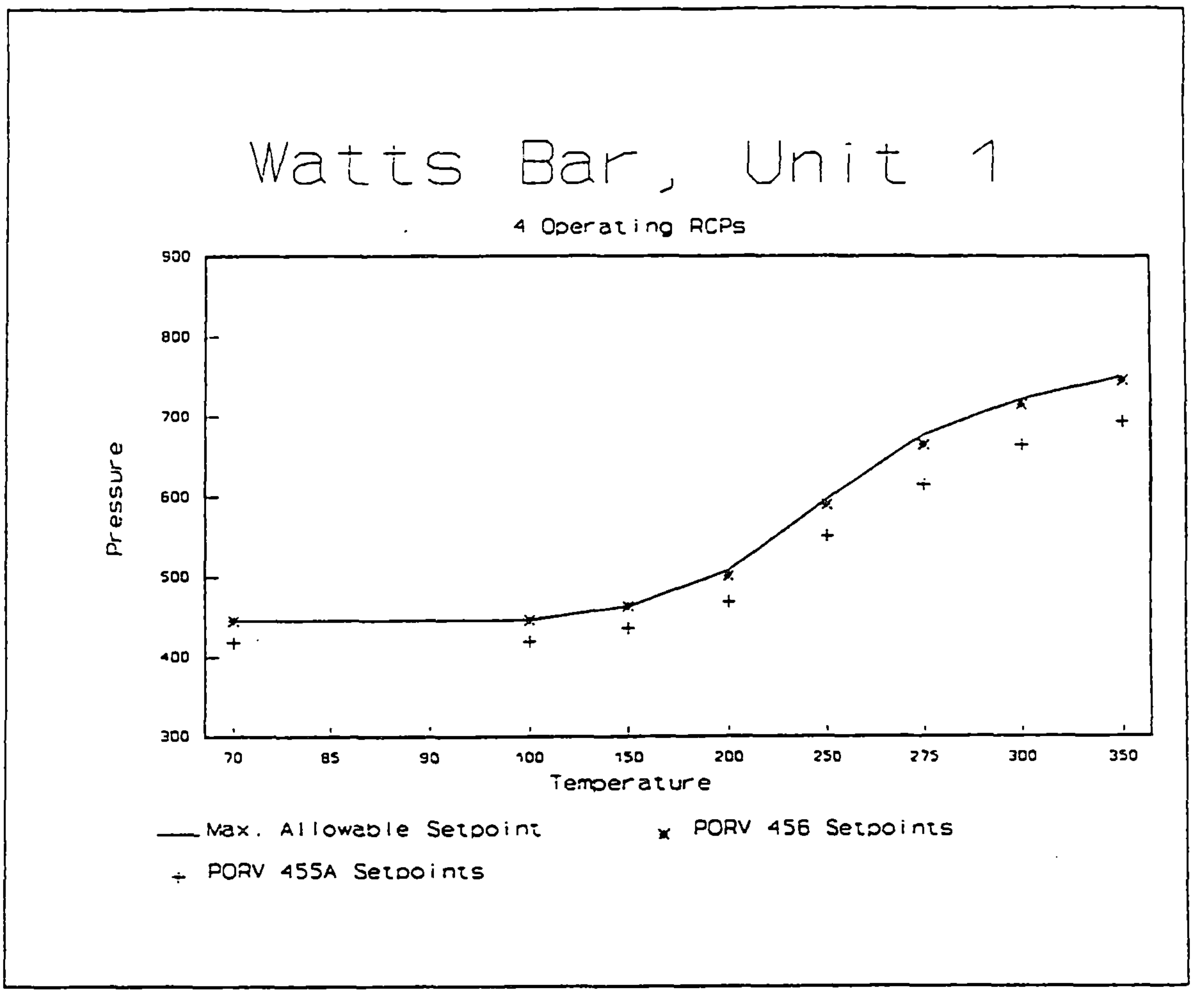


Figure 3.1-4
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

Table 3.1-1
Watts Bar Unit 1 PORV Setpoints vs Temperature
(Data (Ref.3) plotted on Figures 3.1-1 through 3.1-4)

TEMP (°F)	SETPOINTS (PSIG)							
	1 RCS PUMP OPERATING		2 RCS PUMPS OPERATING		3 RCS PUMPS OPERATING		4 RCS PUMPS OPERATING	
	PORV-334	PORV-340A	PORV-334	PORV-340A	PORV-334	PORV-340A	PORV-334	PORV-340A
70	486	459	485	458*	467	440*	445*	418*
100	486	459	485	458*	467	440*	445*	418*
150	505	477	495	467	482	455*	462	435*
200	540	508	540	508	520	488	500	468
250	625	580	625	585	615	570	590	550
275	690	640	700	650	680	630	665	615
300	730	680	730	680	725	675	715	665
350	745	690	745	690	745	690	745	690
450	2350	2350	2350	2350	2350	2350	2350	2350

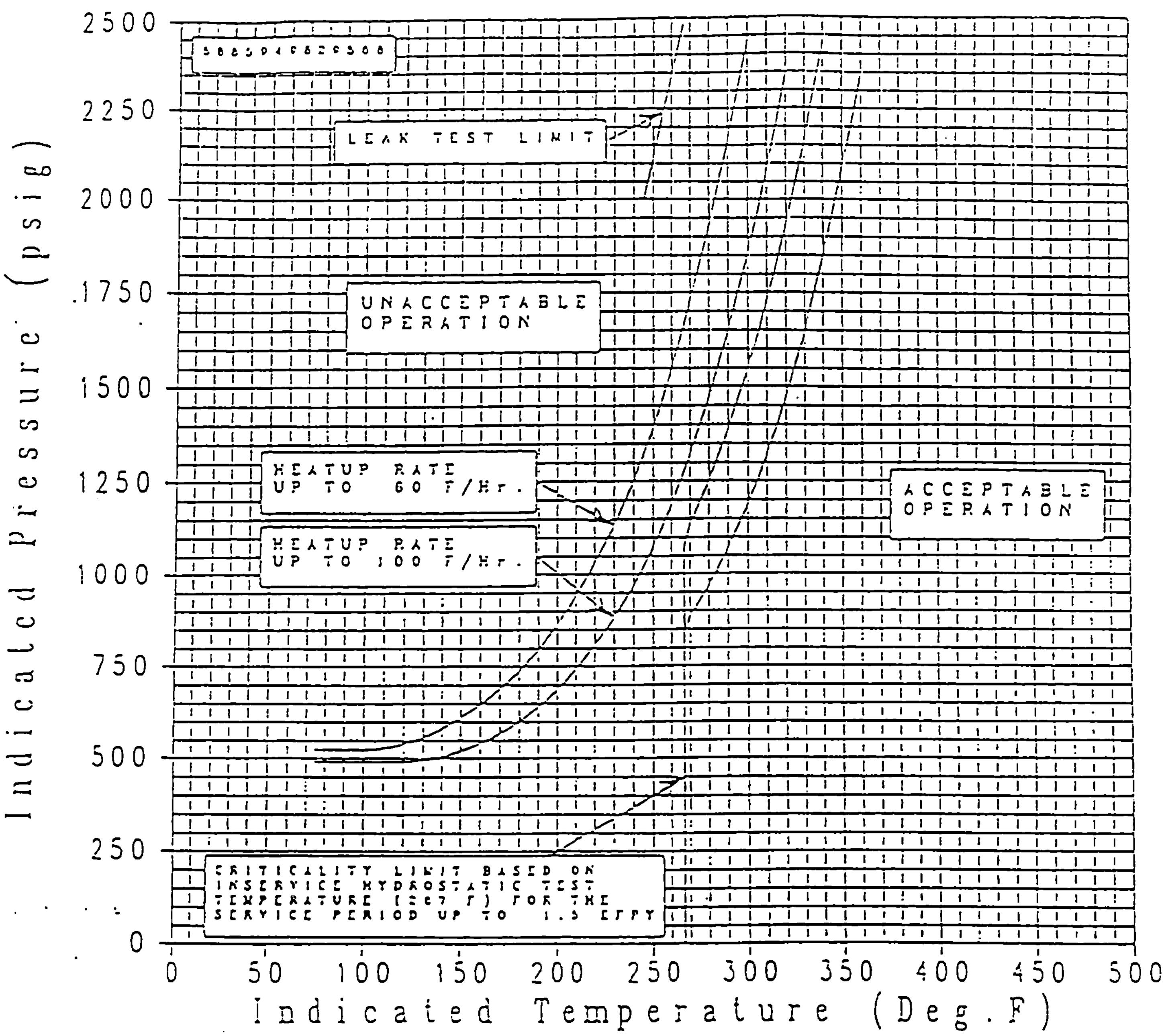
* Setpoint violates pump seal limit. The pump seal limit includes a 63 psig adjustment for pressure channel uncertainty.

Figure 3.1-5

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05

LIMITING ART AT 1.5 EFPY: 1/4-t, 133.2°F

3/4-t, 107.7°F



Watts Bar Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 100°F/hr) Applicable for the First 1.5 EFPY (Without Margins for Instrumentation Errors)

Table 3.1-2

Watts Bar Unit 1 Heatup Data at 1.5 EFPY Without Instrumentation Error Margins

(includes: Vessel Flange Req. (80°F, 621 psig))

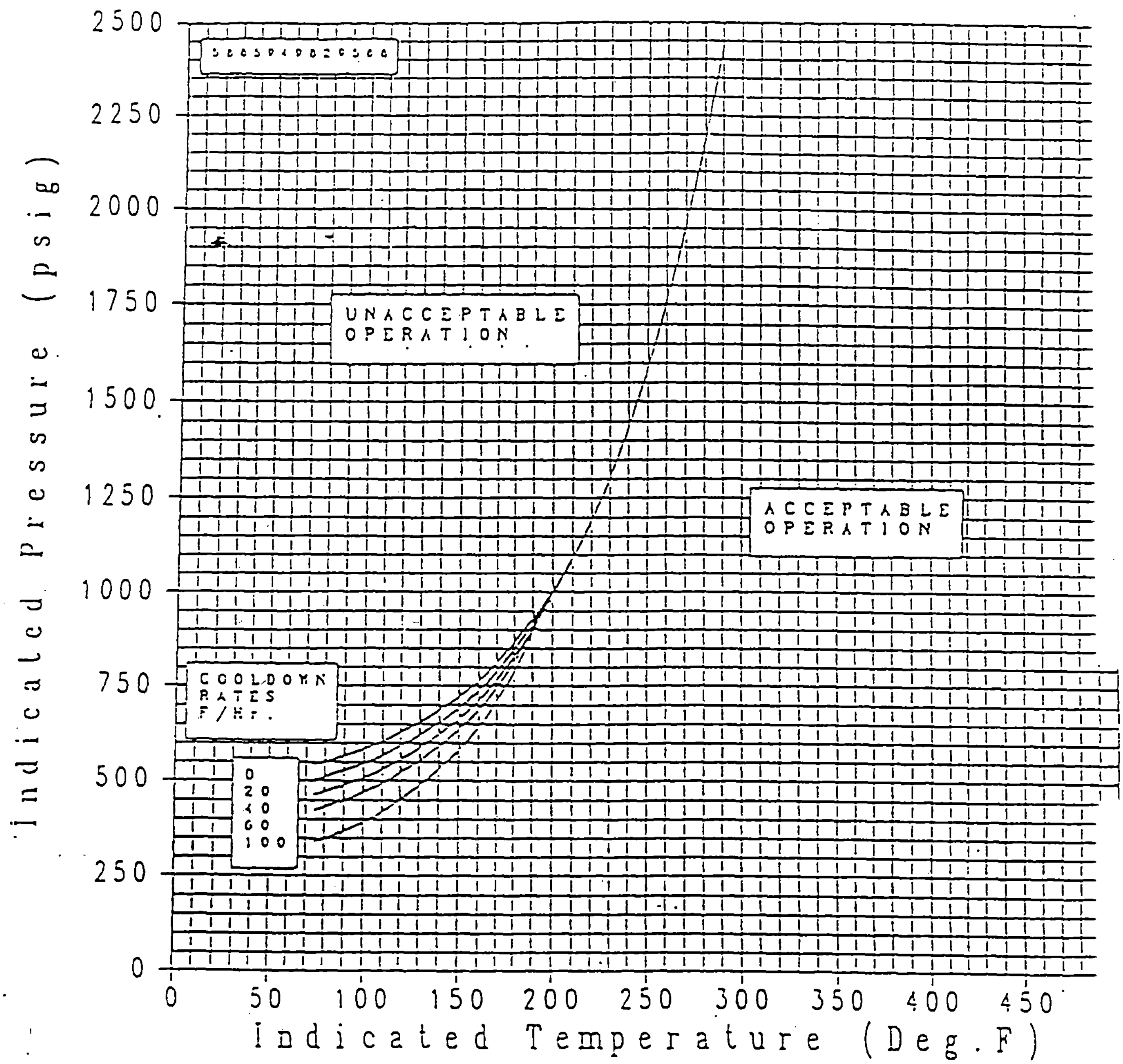
60 HU T (Deg.F)	60 HU P (psig)	Criticality Limit T (Deg.F)	Criticality Limit P (psig)	100 HU T (Deg.F)	100 HU P (psig)	Criticality Limit T (Deg.F)	Criticality Limit P (psig)	Hydrostatic Leak Test T (Deg.F)	Hydrostatic Leak Test P (psig)
75	524.60	267	0.00	75	492.13	267	0.00	245	2000
80	524.60	267	549.27	80	492.13	267	550.93	267	2485
85	524.60	267	538.00	85	492.13	267	536.42		
90	524.60	267	530.48	90	492.13	267	523.64		
95	524.60	267	526.23	95	492.13	267	513.32		
100	524.60	267	524.60	100	492.13	267	505.11		
105	525.49	267	525.49	105	492.13	267	499.14		
110	528.36	267	528.36	110	492.13	267	495.00		
115	533.22	267	533.22	115	492.13	267	492.76		
120	539.61	267	539.61	120	492.13	267	492.13		
125	547.81	267	547.81	125	493.17	267	493.17		
130	557.48	267	557.48	130	495.68	267	495.68		
135	568.69	267	568.69	135	499.72	267	499.72		
140	581.32	267	581.32	140	505.16	267	505.16		
145	595.36	267	595.36	145	512.05	267	512.05		
150	610.96	267	610.96	150	520.33	267	520.33		
155	628.13	267	628.13	155	530.05	267	530.05		
160	646.72	267	646.72	160	541.08	267	541.08		
165	667.12	267	667.12	165	553.71	267	553.71		
170	689.07	267	689.07	170	567.82	267	567.82		
175	713.03	267	713.03	175	583.39	267	583.39		
180	738.71	267	738.71	180	600.67	267	600.67		
185	766.62	267	766.62	185	619.67	267	619.67		
190	796.55	267	796.55	190	640.28	267	640.28		
195	828.76	267	828.76	195	662.92	267	662.92		
200	863.61	267	863.61	200	687.34	267	687.34		
205	902.97	267	900.97	205	714.03	267	714.03		
210	941.11	267	941.11	210	742.72	267	742.72		
215	984.24	267	984.24	215	773.88	267	773.88		
220	1030.58	267	1030.58	220	807.45	267	807.45		
225	1080.37	267	1080.37	225	843.62	267	843.62		
230	1133.75	270	1133.75	230	882.57	270	882.57		
235	1191.02	275	1191.02	235	924.51	275	924.51		
240	1252.38	280	1252.38	240	969.65	280	969.65		
245	1318.42	285	1318.42	245	1018.20	285	1018.20		
250	1388.87	290	1388.87	250	1070.40	290	1070.40		
255	1464.75	295	1464.75	255	1126.45	295	1126.45		
260	1545.88	300	1545.88	260	1186.56	300	1186.56		
265	1632.69	305	1632.69	265	1250.93	305	1250.93		
270	1725.62	310	1725.62	270	1320.19	310	1320.19		
275	1824.95	315	1824.95	275	1394.24	315	1394.24		
280	1931.19	320	1931.19	280	1473.72	320	1473.72		
285	2044.62	325	2044.62	285	1558.77	325	1558.77		
290	2165.98	330	2165.98	290	1649.72	330	1649.72		
295	2295.23	335	2295.23	295	1747.03	335	1747.03		
300	2432.91	340	2432.91	300	1851.11	340	1851.11		
				305	1962.23	345	1962.23		
				310	2080.77	350	2080.77		
				315	2207.32	355	2207.32		
				320	2342.13	360	2342.13		

Figure 3.1-6

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05

LIMITING ART AT 1.5 EFPY: 1/4-t, 133.2°F

3/4-t, 107.7°F



Watts Bar Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 1.5 EFPY (Without Margins for Instrumentation Errors)

Table 3.1-3

Watts Bar Unit 1 Cooldown Data at 1.5 EFPY Without Instrumentation Error Margins

(includes: Vessel Flange Req. (80°F, 621 psig))

Steady State		20 CD		40 CD		60 CD		100 CD	
T (Deg.F)	P (psig)	T (Deg.F)	P (psig)	T (Deg.F)	P (psig)	T (Deg.F)	P (psig)	T (Deg.F)	P (psig)
75	544.13	75	504.75	75	464.68	75	423.89	75	339.99
80	550.93	80	511.84	80	472.10	80	431.67	80	348.66
85	558.25	85	519.50	85	480.12	85	440.05	85	358.12
90	566.12	90	527.72	90	488.77	90	449.15	90	368.29
95	574.58	95	536.50	95	498.03	95	459.03	95	379.41
100	583.55	100	546.05	100	508.09	100	469.67	100	391.43
105	593.33	105	556.35	105	518.97	105	481.19	105	404.42
110	603.85	110	567.42	110	530.68	110	493.51	110	418.51
115	615.15	115	579.37	115	543.22	115	506.95	115	433.80
120	627.29	120	592.08	120	556.83	120	521.41	120	450.23
125	640.22	125	605.92	125	571.52	125	536.95	125	468.11
130	654.26	130	620.80	130	587.21	130	553.80	130	487.41
135	669.36	135	636.69	135	604.27	135	572.00	135	508.21
140	685.43	140	653.92	140	622.62	140	591.48	140	530.74
145	702.88	145	672.47	145	642.27	145	612.64	145	555.01
150	721.61	150	692.28	150	663.57	150	635.28	150	581.28
155	741.62	155	713.78	155	686.37	155	659.87	155	609.58
160	763.28	160	736.70	160	711.08	160	686.18	160	640.05
165	786.39	165	761.57	165	737.52	165	714.73	165	673.12
170	811.40	170	788.13	170	766.12	170	745.28	170	708.62
175	838.14	175	816.87	175	796.81	175	778.21	175	746.92
180	866.99	180	847.65	180	829.76	180	813.84	180	788.17
185	897.91	185	880.71	185	865.42	185	852.06	185	832.68
190	931.09	190	916.24	190	903.65	190	893.20	190	880.61
195	966.71	195	954.70	195	944.80	195	937.48	195	932.31
200	1035.22	200	995.83	200	989.00	200	985.13	200	987.97
205	1046.42	205	1040.05	205	1036.59	205	1036.40		
210	1090.65	210	1087.57	210	1087.70				
215	1138.18								
220	1189.21								
225	1243.97								
230	1302.68								
235	1365.87								
240	1433.29								
245	1506.09								
250	1583.90								
255	1667.11								
260	1756.73								
265	1852.40								
270	1954.86								
275	2064.38								
280	2181.38								
285	2306.69								
290	2440.17								

Table 4.0-1
Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Time[a][b][d]	Estimated Capsule Fluence (n/cm²)[c]
U	56.0°	3.6	1st Refueling Outage	3.60×10^{18}
W	124.0°	3.6	5.4	1.90×10^{19}
X	236.0°	3.6	8.9	3.19×10^{19}
Z	304.0°	3.6	17.8	6.38×10^{19}
V	58.5°	3.6	Stand-By	----
Y	238.5°	3.6	Stand-By	----

(a) Effective Full Power Years (EFPY) from plant startup.

(b) Removal times are based on not-to-exceed criteria of E185-82, Section 7.6.2. Capsules should be removed on the last cycle prior to reaching the indicated time.

(c) Based on design basis fluence of 3.18×10^{19} n/cm² (E > 1MeV).

(d) Withdraw two capsules before the vessel exceeds 5.4 EFPY. The results of the capsule analysis will be reviewed and should an amended removal schedule be required, two standby capsules are available for additional monitoring.[1] If the results of capsule testing predict an end of life use of < 50 ft-lb, TVA will perform the necessary analysis required by Appendix G, IV.A.1 to ensure adequate safety margins.[2]

TABLE 5.1

Comparison of the Watts Bar Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decrease with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence ($\times 10^{19}$ n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted[a] (°F)	Measured (°F)	Predicted[a] (%)	Measured (%)
Intermediate Shell Forging 05 (tangential)						
Intermediate Shell Forging 05 (axial)						
Weld Metal						
HAZ Metal						

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using average weight percent values of Cu an Ni.

NOTE: No capsules have been removed from the Watts Bar Unit 1 reactor vessel at this time.

TABLE 5.2						
Watts Bar Unit 1 Calculation of Chemistry Factors Using Surveillance Capsule Data						
Material	Capsule	Fluence (n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 05 (Tangential)						
Intermediate Shell Forging 05 (Axial)						
				Sum:		
	Chemistry Factor =					
Weld Metal						
				Sum:		
	Chemistry Factor =					

NOTE: No capsules have been removed from the Watts Bar Unit 1 reactor vessel at this time.

TABLE 5.3			
Watts Bar Unit 1 Reactor Vessel Toughness Table (Unirradiated)			
Material Description	Cu (%)[a]	Ni (%)[a]	Initial RT_{NDT} (°F)[b]
Closure Head Flange	0.13	0.75	-42
Vessel Flange	--	0.92	-40[c]
Intermediate Shell Forging 05	0.17	0.80	47
Lower Shell Forging 04	0.08	0.83	5
Circumferential Weld	0.05	0.70	-43

NOTES:
a) Average values of copper and nickel weight percent.
b) Initial RT_{NDT} values are measured values.
c) Used in the consideration of flange requirements for heatup/cooldown curves. Per methodology given in WCAP-14040, the minimum boltup temperature is 60°F.

TABLE 5.4					
Watts Bar Unit 1 Reactor Vessel Surface Fluence Values at 7 EFPY (n/cm^2, E > 1.0 MeV)					
Azimuthal	0°	15°	25°	35°	45°
Surface	4.13×10^{18}	6.15×10^{18}	6.96×10^{18}	5.67×10^{18}	6.50×10^{18}

TABLE 5.5		
Summary of ARTs for the Watts Bar Unit 1 Reactor Vessel Beltline Materials at the 1/4-T and 3/4-T Locations for 7 EFPY		
Component	7 EFPY[a]	
	1/4-T (°F)	3/4-T (°F)
Intermediate Shell Forging 05	181.11[b]	147.70[b]
Lower Shell Forging 04	77.68	56.54
Circumferential Weld	60.14	25.72

NOTES:
(a) Calculated using the peak vessel fluence of 6.96 x 10^{18} n/cm^2 (E > 1.0 MeV).
(b) Used to generate the heatup/cooldown curves.

TABLE 5.6		
Calculation of Adjusted Reference Temperatures at 7 EFPY for the Limiting Watts Bar Unit 1 Reactor Vessel Material (Intermediate Shell Forging 05)		
Parameter	Values	
Operating Time	7 EFPY	
Material	Inter. Shell Forging 05	Inter. Shell Forging 05
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F	132	132
Fluence (f), $\div 10^{19}$ n/cm^2 (E > 1.0 MeV) [a]	0.4188	0.1517
Fluence Factor (FF)[b]	0.758	0.505
ΔRT_{NDT} = CF x FF, °F	100.1	66.7
Initial RT_{NDT} (I), °F	47	47
Margin (M), °F [c]	34	34
ART = I + (CFxFF) + M, °F per Regulatory Guide 1.99, Revision 2	181.1	147.7

<u>NOTES:</u>

(a) Fluence, f, is based upon $f_{surf} = 6.96 \times 10^{18}$ n/cm^2. The Watts Bar Unit 1 reactor vessel wall thickness is 8.465 inches at the beltline region.

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

(c) Margin is calculated as $M = 2(\sigma_i^2 + \sigma_\Delta^2)^{0.5}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is 0°F since the initial RT_{NDT} value is a measured value. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for the forging, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}. R

TABLE 5.7					
Watts Bar Unit 1 Reactor Vessel Surface Fluence Values at 32 and 48 EFPY (n/cm^2, E > 1.0 MeV)					
EFPY	0°	15°	25°	35°	45°
32	1.89×10^{19}	2.81×10^{19}	3.18×10^{19}	2.59×10^{19}	2.97×10^{19}
48	2.84×10^{19}	4.22×10^{19}	4.77×10^{19}	3.89×10^{19}	4.46×10^{19}

TABLE 5.8							
RT_{PTS} Values for Watts Bar Unit 1 for 32 EFPY							
Material	CF (°F)	Surface Fluence (n/cm², E > 1.0 MeV)	FF	ΔRT_{NDT} (CF x FF) (°F)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Forging 05	132	3.18×10^{19}	1.30	171.6	47	34	253
Lower Shell Forging 04	51	3.18×10^{19}	1.30	66.3	5	34	106
Circ. Weld	68	3.18×10^{19}	1.30	88.4	-43	56	102

TABLE 5.9							
RT_{PTS} Values for Watts Bar Unit 1 for 48 EFPY							
Material	CF (°F)	Surface Fluence (n/cm², E > 1.0 MeV)	FF	ΔRT_{NDT} (CF x FF) (°F)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Forging 05	132	4.77×10^{19}	1.39	185.5	47	34	265
Lower Shell Forging 04	51	4.77×10^{19}	1.39	70.9	5	34	110
Circ. Weld	68	4.77×10^{19}	1.39	94.5	-43	56	108

SOURCE NOTES

1. NCO820285003
2. NCO820285004

APPENDIX B

NRC REQUEST FOR ADDITIONAL INFORMATION "RAI"

AND

CORRESPONDENCE WITH NRC

Westinghouse Owners Group

OG-95-078

WCAP-14040 - NP

September 26, 1995

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

Attention: Chief, Planning, Program and Management Support Branch

Attention: Mr. C. I. Grimes, Chief
Technical Specifications Branch

Subject: Westinghouse Owners Group
Transmittal of Page Revision to WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" to Address Instrumentation Uncertainties Associated with COMS PORV Setpoints

Reference: 1) R.A. Newton to Document Control Desk, Attention J.R. Strosnider, Transmittal of Report: WCAP-14040, Rev. 1, [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", OG-94-102, dated December 20, 1994.

As discussed in the September 21, 1995 telecon between the WOG, Westinghouse and Mr. Chu-Yu Liang of the NRC, WCAP-14040, Rev. 1, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", previously submitted by Reference 1), will be revised as indicated on the attached marked up page to address instrumentation uncertainties associated with COMS PORV setpoints.

Should you have any questions concerning this information, please call me at (708) 746-2084 x2890.

Very truly yours,

Lee Bush, Chairman
Licensing Subcommittee
Westinghouse Owners Group

LB/ygs
attachment

cc: WOG Steering Committee
WOG Primary Representatives
WOG Licensing Subcommittee Representatives
WOG Materials Subcommittee Representatives
Peter S. Tam, USNRC
Jack R. Strosnider, USNRC
John C. Tsao, USNRC
Harvey I. Abelson, USNRC
R.F. Saunders, Virginia Power
T.C. McMeekin, Duke Power
N.J. Liparulo, W
K.J. Voytell, W

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. ~~Since both~~ A the upper ~~and lower~~ pressure limits ~~values~~ are ~~conservatively determined, the uncertainties in the pressure and temperature instrumentation utilized by the COMS are not explicitly accounted for in the selection of the COMS PORV setpoints. Accounting for the effects of instrumentation uncertainty would impose additional unnecessary restrictions on the setpoint development, which is~~ already based on conservative assumptions ~~pressure limits~~ (such as a safety factor of 2 on pressure stress, use of a lower bound K_{IR} curve and an assumed ¼T flaw depth with a length equal to 1½ times the vessel wall thickness) as discussed in section 2 of this report. ~~, without a commensurate increase in the level of protection afforded to reactor vessel integrity.~~

Insert

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. Although expected soon, use of Code Case N-514 has not yet been formally approved by the NRC. In the interim, an exemption to the regulations must be

Insert

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.

OG-95-067 WCAP-14040 - NP

August 15, 1995

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

Attention: Chief, Planning, Program and Management Support Branch

Attention: Mr. C. I. Grimes, Chief
Technical Specifications Branch

Subject: Westinghouse Owners Group
Transmittal of Response to Request From July 25, 1995 Telecon for Additional Information (RAI) Regarding WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"

Reference: 1) C.I. Grimes to R.A. Newton, Request for Additional Information (RAI) Regarding WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", dated July 11, 1995.

2) L.Bush to Document Control Desk, Attention Chief, Planning, Program and Management Support Branch and C.I. Grimes, Chief Technical Specifications Branch "Transmittal of Response to Request for Additional Information (RAI) Regarding WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", OG-95-061, dated July 18, 1995.

As discussed in the July 25, 1995 telecon between Westinghouse, TVA, and Mr. Chu-Yu Liang of the NRC, WCAP-14040, Rev. 1, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", December, 1994 will be revised as discussed below in response to Questions 2, 5, and 6 in Enclosure 1 of Reference 1: (Note: Previous responses provided by Reference 2)

The next to the last sentence in the first paragraph on Page 3-2 will be revised as follows:

"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 end of second paragraph on Page 3-4 will be revised as follows:

"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."

The following paragraph will be added to Page 3-6 before the Section 3.3 discussion:

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.

Should you have any questions concerning this information, please call me at (708) 746-2084 x2890.

Very truly yours,

Lee Bush, Chairman
Licensing Subcommittee
Westinghouse Owners Group

LB/ygs

cc: WOG Steering Committee
WOG Primary Representatives
WOG Licensing Subcommittee Representatives
Peter S. Tam, USNRC
Jack R. Strosnider, USNRC
John C. Tsao, USNRC
Harvey I. Abelson, USNRC
R.F. Saunders, Virginia Power
T.C. McMeekin, Duke Power
N.J. Liparulo, W
K.J. Voytell, W

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 11, 1995

Mr. Roger A. Newton, Chairman
Westinghouse Owners Group
Westinghouse Electric Corporation
M/S 5-16 E,
P.O. Box 355
Pittsburgh, PA 15230-0355

Dear Mr. Newton:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING WCAP-14040, "METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES."

By letter dated December 20, 1994, Westinghouse submitted WCAP 14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup And Cooldown Limit Curves" for review and approval. The staff has reviewed the report and requests additional information, as delineated in Enclosure 1. On May 5, 1995, Ginna also submitted WCAP 14040 as a part of a license amendment request. Enclosure 2 requests additional information regarding the Ginna submittal.

In an effort to utilize limited NRC resources more effectively and efficiently, we are conducting this review generically. Therefore, we will submit all questions for licensees utilizing the WCAP to you.

Enclosure 3 includes a copy of an RAI submitted to Watts Bar on June 27, 1995 and a copy of Issues For Discussion in a Conference Call dated June 28, 1995. These questions should also be addressed for the topical report review.

Please respond, in writing, to this request by the time of our meeting tentatively scheduled for July 20, 1995. This schedule is necessary to meet the proposed fuel loading date of September 24, 1995 for Watts Bar.

Sincerely,

C.I. Grimes

C.I. Grimes, Chief
Technical Specifications Branch
Division of Project Support
Office of Nuclear Reactor Regulation

'AUG 0

Enclosure 1

REQUEST FOR ADDITIONAL INFORMATION
CONCERNING WCAP 14040, "METHODOLOGY USED TO DEVELOP
COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS
AND RCS HEATUP AND COOLDOWN LIMIT CURVES"

1. First sentence in Section 3.1 implies that the RHR relief valves are sufficient for cold overpressure mitigation in general. Is any generic study available to support this assertion?

2. Section 3.1 should be modified to clearly state that the most limiting mass addition transient should be considered for design of COMS. An inadvertent actuation of safety injection should be considered (assume all operable SI pumps deliver water into RCS) if it is more limiting than the case of a failure of the normal charging flow controls with letdown isolated.

3. In Section 3.2.1, should the steam generator pressure and RHR system pressure limits be considered as a part of parameters in the design of COMS?

4. Provide basis of assuming 800 psia as the PORV piping limit in COMS design.

5. Section 3.2.2 should be modified to indicate that in the case where the available range is insufficient to concurrently accommodate the upper and lower pressure limits, the upper pressure limits are given preference.

6. Section 3.2.5 Should be modified to indicate that the design of COMS should also consider the proper overpressure protection of RHR system.

7. Provide clarification to the second paragraph of Section 3.2.5 regarding the selection process for setpoints.

8. Section 3.2.5 indicates that in the selection of the PORV setpoints, the upper limits are specified by the minimum of the steady-state cooldown curve. Please discuss why the heatup curve is not considered in this process.

9. Section 3.2.5 indicates that the uncertainties in the pressure and temperature instrumentation utilized by the COMS are not accounted for in the selection of the COMS PORV setpoints. This is not acceptable to the staff. Please modify the methodology to consider the potential uncertainties in the instrumentation utilized by the COMS.

Enclosure 2

REQUEST FOR ADDITIONAL INFORMATION
CONCERNING RG&E SPECIFIC METHODOLOGY FOR DETERMINING LTOPS SETPOINTS

1. Section 3.1 should be modified to clearly state that the most limiting mass addition transient should be considered for design of COMS. An inadvertent actuation of safety injection should be considered (assume all operable SI pumps deliver water into RCS) if it is more limiting than the case of a failure of the normal charging flow controls with letdown isolated.

2. Provide basis of assuming 800 psia as the PORV piping limit in COMS design.

3. Section 3.2.5 indicates that in the selection of the PORV setpoints, the upper limits are specified by the minimum of the steady-state cooldown curve. Please discuss why the heatup curve is not considered in this process.

4. Section 3.2.5 indicates that the uncertainties in the pressure and temperature instrumentation utilized by the COMS are not accounted for in the selection of the COMS PORV setpoints. This is not acceptable to the staff. Please modify the methodology to consider the potential uncertainties in the instrumentation utilized by the COMS.

5. In Section 3.2.1, items n and q describe the same parameter. Please correct this error.

6. In Section 3.2.1, it is indicated that the computer code BWNT RELAP5/MOD2-B&W is used by RG&E for LTOP design. Please provide discussion on the applicability of this computer code for LTOP design at Ginna. Also, discuss the staff review status for this computer code.

ENCLOSURE 3

June 27, 1997

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: WATTS BAR UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING PRESSURE-TEMPERATURE LIMIT REPORT (PTLR) AND ACCOMPANYING WESTINGOUSE OWNERS GROUP WCAP-14040 (TAC NO. M89048 AND M92336)

Dear Mr. Kingsley:

By letters dated March 10, 1994, December 23, 1994, and March 29, 1995, Tennessee Valley Authority submitted (1) Watts Bar Unit 1 Reactor Coolant System Pressure and Temperature Limits Report, Revision 3; (2) Westinghouse WCAP-13829, "Heatup and Cooldown Limit Curves for Watts Bar Unit 1," Revision 1; and (3) Westinghouse WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1. The staff has reviewed this information and has developed several questions, as delineated in the enclosure.

Please respond within 20 days of receipt of this letter. This stringent schedule is needed to meet the proposed fuel loading date of September 24, 1995. The requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,
Original signed by
Peter S. Tam, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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REQUEST FOR ADDITIONAL INFORMATION
WATTS BAR UNIT 1
PRESSURE-TEMPERATURE LIMIT REPORT AND METHODOLOGY
(TAC M89048 AND M92336)

The staff requests additional information for Westinghouse Owners Group, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040, Revision 1:

1. On page 2-6, the bottom paragraph, WCAP-14040 states that "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 must be provided for an exemption from regulations to be granted by the NRC." Commonwealth Edison Company's Zion Units 1 and 2 exemption for their WF-70 weld was cited as an example.

   The staff notes that in addition to the technical justification, the utility must show that special circumstances are present as described in 10 CFR 50.12. In the case of Zion Units 1 and 2, the staff acceptance of the alternative procedure was contingent on the analysis of a significant amount of fracture toughness data for the WF-70 weld metal. Acceptance of such a procedure in a case where little or no fracture toughness data were available would be difficult in the absence of an officially sanctioned consensus standard. In addition to this technical justification, the utility met the special circumstances criteria in 10 CFR 50.12(a)(2).

   The staff requests TVA modify the paragraph (on page 2-6) to emphasize that special circumstances must be present to justify an exemption to the rule. Also if Zion Units 1 and 2 is used as an example, TVA should clarify the staff's justification for approval of the exemption.

2. On page 2-8, below Equation 2.4-4, it should state where $f_{surface}$ (10^{19} n/cm^2, E>1 MeV) .... The staff requests that the revised text be inserted.

3. On pages 2-16 and 2-18, the flaw shape factor, Q, is calculated two separate ways during heatup and cooldown (once for the pressure term and once for the thermal term). Explain the reason for the use of two flaw shape factors and justify why a generic Q value for the combined effect was not used.

4. In Equation 2.5-1 in the report,

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

   The above number is shown as 1.223, whereas in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1989 edition, the redlined number is shown as 1.233. The differences are also highlighted in WCAP-13829; where on page 3 the value used is 1.223 and on page 7 the value used is 1.233.

ENCLOSURE

The staff notes that in Welding Research Council, Bulletin 175 and in the NRC Standard Review Plan 5.3.2, the number is shown as 1.223. However, the WCAPs 14040 and 13829 reference the ASME code. Explain the discrepancy in your reports and justify the use of 1.223.

5. Demonstrate the method for determining the thermal stress intensity factor, K_{IT}, for heatup and cooldown. Specifically, provide a sample calculation of the process starting with Equation 2.6.1-1 on page 2-14. The sample calculation should be in sufficient detail to provide the staff a reproducible method for verifying the heatup and cooldown curves. The calculation should include the time-dependent temperature distribution determination (page 2-14), the use of the Timoshenko equation (pages 2-14 and 2-15), the determination of the thermal stresses (membrane and bending) employing the linearization technique from Appendix A of the ASME code (page 2-17), and the final value of K_{IT} using Equation 2.6.3-1 (page 2-17).

6. In Equation 2.6.3-3 (page 2-18) or Equations 2.6.4-1 and 2.6.4-2 (page 2-19) for cooldown and heatup respectively, the K_{Ia} input is calculated by using Equation 2.5-1 (page 2-10). The staff assumes the temperature input to Equation 2.5-1 is derived using the time-dependent temperature distribution solution described on page 2-14. If this is a valid assumption, demonstrate the use of Equation 2.6.3-3 or Equations 2.6.4-1 and 2.6.4-2 for cooldown and heatup respectively using the temperature distribution solution. If this is not a valid assumption, provide an explanation for the temperature value used in Equation 2.5-1 and subsequently the K_{Ia} used in Equation 2.6.3-3 or Equations 2.6.4-1 and 2.6.4-2.

7. The LTOP/COMS setpoints do not appear to include instrument error. Explain the absence of this factor and justify the exclusion of a margin for instrument error. (Westinghouse provided a response by letter dated June 16, 1995, which the staff is in the process of reviewing.)

The staff requests additional information for Westinghouse Owners Group, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 1," WCAP-13829, Revision 1.

1. The pressure-temperature limits curves in Figures 1 and 2 do not include margins for pressure differences between the wide-range pressure transmitter and the limiting reactor vessel beltline region (stated on page 1). Justify the exclusion of the pressure differences for the heatup and cooldown curves.

2. On page 2, the staff requests the last sentence be revised to read "The post-irradiation fracture toughness properties of the reactor vessel beltline material will be obtained directly from the Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program."

3. On page 6, the staff requests the first paragraph reference WCAP-14040 directly to clarify the methods in obtaining the pressure-temperature limits.

The staff requests additional information for Watts Bar Unit 1 RCS Pressure Temperature Limits Report, Revision 3.

1. On page A-27 in note (c), the margin equation is quoted from Regulatory Guide 1.99, Revision 2, incorrectly. The equation should be multiplied by a factor of two. The values in the table reflect the correct equation. The staff requests TVA to correct the equation.

- In sample calculations, use values from Watts Bar Unit 1, P-T limits for 7 EFPY.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 28, 1995

MEMORANDUM FOR: Docket File

FROM: Peter S. Tam, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Peter S. Tam

SUBJECT: WATTS BAR UNIT 1 - ISSUES FOR DISCUSSION IN A CONFERENCE CALL (TAC M89048 and M92336)

As a result of its review of Westinghouse Topical Report WCAP-14040, Revision 1, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints And RCS Heatup and Cooldown Limit Curves", the staff raised a number of issues (attached). These issues are faxed to Watts Bar site licensing personnel to prepare them for a conference call. The staff's comments do not currently constitute a formal position nor request for information.

Docket Numbers 50-390
50-391

Issues for Discussion

WCAP-14040, Revision 1

1. It is stated that: "..neutron source distributions utilized in these analyses include...the effects of varying neutron yield per fission and fission spectrum introduced by the build-up of Plutonium." The next paragraph states: "The source distribution is taken to be an average distribution with all fissions attributed to U-235." Please account for the apparent conflict in these statements.

2. The 47 group SAILOR library is based on ENDF/B-IV data. It is stated that: "...on the average for Westinghouse plants, surveillance capsule dosimetry indicates that the fluence calculations are biased low" and "...for conservatism the fluence values...are adjusted by the average bias indicated by the dosimetry measurements for that plant":

   2.1 Is the low bias a result of using ENDF/B-IV data in thermal shield plants?

   2.2 How was the DOT benchmarked to account for ENDF/B-IV and a thermal shield?

   2.3 Is it not more reasonable to bias the dosimetry by the plant specific bias?

   2.4 There is no mention of the associated plant specific uncertainty analysis to be performed to qualify the dosimetry. Please comment.

Westinghouse Owners Group

Domestic Utilities
American Electric Power
Carolina Power & Light
Commonwealth Edison
Consolidated Edison
Duquesne Light
Duke Power
Georgia Power
Florida Power & Light
Houston Lighting & Power
New York Power Authority
Northeast Utilities
Northern States Power
Pacific Gas & Electric
Public Service Electric & Gas
Rochester Gas & Electric
South Carolina Electric & Gas
Southern Nuclear
Tennessee Valley Authority
TU Electric
Union Electric
Virginia Power
Wisconsin Electric Power
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International Utilities
Electrabel
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Korea Electric Power
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Nuklearna Elektrana
Spanish Utilities
Taiwan Power
Vattenfall

OG-95-061 WCAP-14040 - NP

July 18, 1995

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

Attention: Chief, Planning, Program and Management Support Branch

Attention: Mr. C. I. Grimes, Chief
Technical Specifications Branch

Subject: Westinghouse Owners Group
Transmittal of Response to Request for Additional Information (RAI) Regarding WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"

Reference: 1) C.I. Grimes to R.A. Newton, Request for Additional Information (RAI) Regarding WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", dated July 11, 1995.

2) R.A. Newton to Document Control Desk, Attention J.R. Strosnider, Transmittal of Report: WCAP-14040, Rev. 1, [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", OG-94-102, dated December 20, 1994.

3) L. Bush to Document Control Desk, Attention Chief, Planning, Program and Management Support Branch, Transmittal of Response to Concerns Identified During Review of WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", OG-95-54, dated June 16, 1995.

This letter transmits twenty three (23) copies of the Westinghouse Owners Group (WOG) Response, attached, to the NRC Request for Additional Information (RAI) Regarding WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Reference 1). WCAP-14040, Rev. 1, [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", dated December 1994, was submitted by Reference 2) and the previous response addressing potential uncertainties in the instrumentation utilized by the COMS, was submitted by Reference 3).

The attachment provides WOG responses to the following generic questions, noting that responses to the plant specific questions will be provided by the individual licensees:

- o Nine questions - Enclosure 1 to Reference 1)
- o Seven questions - Enclosure 3 to Reference 1)
- o Two questions - Issues for Discussion, Reference 1)

The WOG acknowledges that clarifications and/or modifications to the Approved Version of the WCAP will be made after final resolution is reached on all issues.

This information is provided to support your preparation for our forthcoming meeting on July 20, 1995. Should you have any questions concerning this information, please call me at (708) 746-2084 x2890.

Very truly yours,

Lee Bush, Chairman
Licensing Subcommittee
Westinghouse Owners Group

LB/ygs

attachment

cc: WOG Steering Committee (1L,1A)
WOG Primary Representatives (1L,1A)
WOG Licensing Subcommittee Representatives (1L,1A)
Peter S. Tam, USNRC (1L,1A)
Jack R. Strosnider, USNRC (1L,1A)
John C. Tsao, USNRC (1L,1A)
Harvey I. Abelson, USNRC (1L,1A)
R.F. Saunders, Virginia Power (1L)
T.C. McMeekin, Duke Power (1L)
N.J. Liparulo, W (1L)
K.J. Voytell, W (1L)

WOG Responses to Enclosure 1 of NRC Letter
dated July 11, 1995, Grimes (NRC) to Newton (WOG)

REQUEST FOR ADDITIONAL INFORMATION
CONCERNING WCAP-14040, "METHODOLOGY USED TO DEVELOP
COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS
AND RCS HEATUP AND COOLDOWN LIMIT CURVES"

NRC Question 1

First sentence in Section 3.1 implies that the RHR relief valves are sufficient for cold overpressure mitigation in general. Is any generic study available to support this assertion?

RESPONSE:

The intent of the first sentence is not to imply that RHRS alone is sufficient to mitigate every cold overpressure event. In effect, COMS provides protection of the ASME Code limits during heat injection and mass injection transients when the RHR cannot provide sufficient relief. In the methodology of WCAP-14040, RHR is conservatively assumed to be completely isolated.

NRC Question 2

Section 3.1 should be modified to clearly state that the most limiting mass addition transient should be considered for design of COMS. An inadvertent actuation of safety injection should be considered (assume all operable SI pumps deliver water into RCS) if it is more limiting than the case of a failure of the normal charging flow controls with letdown isolated.

RESPONSE:

It is agreed that the most limiting case should be considered. However, the number and/or combination of operable pumps which would be assumed in the mass injection transient is plant specific, based on Technical Specification requirements.

NRC Question 3

In Section 3.2.1, should the steam generator pressure and RHR system pressure limits be considered as a part of parameters in the design of COMS?

RESPONSE:

RHRS is isolated for the transients described in WCAP-14040. Pressure/temperature limits based on the Appendix G methodology, as well as the PORV piping limit of 800 psig, are more limiting than steam generator pressure/temperature limits.

NRC Question 4

Provide basis of assuming 800 psia as the PORV piping limits in COMS design.

RESPONSE:

The PORV piping limit in COMS design is based on an analysis of water hammer effects on relief valve piping for certain classes of rapidly opening relief valves (i.e., Garrett valves) during water solid conditions. As other PORVs have relatively slow opening and closing times, water hammer effects are greatly reduced or effectively eliminated when compared to the Garrett type valves. The practice of taking conservative results of the Garrett analysis and applying them to all COMS setpoint evaluations is assumed.

NRC Question 5

Section 3.2.2 should be modified to indicate that in the case where the available range is insufficient to concurrently accommodate the upper and lower pressure limits, the upper pressure limits are given preference.

RESPONSE:

Where the available range is insufficient to protect both the lower pump seal limit and upper pressure/temperature limits based on the Appendix G methodology, setpoints are selected such that the upper limits are protected.

NRC Question 6

Section 3.2.5 should be modified to indicate that the design of COMS should also consider the proper overpressure protection of RHR system.

RESPONSE:

RHRS is assumed to be isolated during the transients described in WCAP 14040.

NRC Question 7

Provide clarification to the second paragraph of Section 3.2.5 regarding the selection process for setpoints.

RESPONSE:

As stated in the second paragraph of Section 3.2.5, breakpoints are selected as inputs into the function generator, which typically accommodates nine values. The function generator then interprets linearly between those breakpoints in order to define a contiguous setpoint curve at each pressure and temperature over the desired range. The 9 breakpoints are selected such that the pressure/temperature limits will not be intersected during this linear interpolation and are therefore protected.

NRC Question 8

Section 3.2.5 indicates that in the selection of the PORV setpoints, the upper limits are specified by the minimum of the steady-state cooldown curve. Please discuss why the heatup curve is not considered in this process.

RESPONSE:

For steady state conditions, there is only one pressure/temperature limit curve (no difference between the heatup and cooldown curves). For convenience, it is shown with other "cooldown" curves.

NRC Question 9

Section 3.2.5 indicates that the uncertainties in the pressure and temperature instrumentation utilized by the COMS are not accounted for in the selection of the COMS PORV setpoints. This is not acceptable to the staff. Please modify the methodology to consider the potential uncertainties in the instrumentation utilized by the COMS.

RESPONSE:

Refer to Westinghouse Owners Group letter OG-95-54, dated June 16, 1995, to the NRC.

WOG Responses to Enclosure 3 of NRC Letter
dated July 11, 1995, Grimes (NRC) to Newton (WOG)

REQUEST FOR ADDITIONAL INFORMATION
WATTS BAR UNIT 1
PRESSURE-TEMPERATURE LIMIT REPORT AND METHODOLOGY
(TAC M89048 AND M92336)

NRC Question 1

On page 2-6, the bottom paragraph, WCAP-14040 states that "Plants that do not follow the fracture toughness requirements in Branch Technical Position MTEB 5-2 to determine IRT_{NDI} can use alternative procedures. However, sufficient technical justification must be provided for an exemption from regulations to be granted by the NRC." Commonwealth Edison Company's Zion Units 1 and 2 exemption for their WF-70 weld was cited as an example.

The staff notes that in addition to the technical justification, the utility must show that special circumstances are present as described in 10CFR50.12. In the case of Zion Units 1 and 2, the staff acceptance of the alternative procedure was contingent on the analysis of a significant amount of fracture toughness data for the WF-70 weld metal. Acceptance of such a procedure in a case where little or not fracture toughness data were available would be difficult in the absence of an officially sanctioned consensus standard. In addition to this technical justification, the utility met the special circumstances criteria in 10CFR50.12(a)(2).

The staff requests TVA modify the paragraph (on page 2-6) to emphasize that special circumstances must be present to justify an exemption to the rule. Also if Zion Units 1 and 2 is used as an example, TVA should clarify the staff's justification for approval of the exemption.

RESPONSE:

As requested, the text on page 2-6 will be modified to emphasize that special circumstances, per criteria of 10CFR 50.12(a)(2), must be present to justify an exemption to the rules in Branch Technical Position MTEB 5-2. The example application for Zion Units 1 and 2 will be deleted from the report text.

NRC Question 2

On page 2-8, below Equation 2.4-4, it should state where $f_{surface}$ (10^{19} n/cm^2, E>1 MeV) . . . The staff requests that the revised text be inserted.

RESPONSE:

On page 2-8, below Equation 2.4-4, "10^{19}" will be inserted before "n/cm^2, E>1 Mev" used to define $f_{surface}$, as requested.

NRC Question 3

On pages 2-16 and 2-18, the flaw shape factor, Q, is calculated two separate ways during heatup and cooldown (once for the pressure term and once for the thermal term). Explain the reason for the use of two flaw shape factors and justify why a generic Q value for the combined effect was not used.

RESPONSE:

On both pages 2-16 (Equation 2.6.2-2) and 2-18 (Equation 2.6.3-2), the general form of the equation for flaw shape factor, Q, is the same:

$$Q = \phi^2 - 0.212\ (\sigma / \sigma_y)^2$$

For the pressure stresses on page 2-16, the stress, σ, has a constant value of σ_p through the thin walled pressure vessel. For the thermal stresses on page 2-18, the stress, σ, has a constant (membrane) component of σ_m and linear (bending) component of σ_b through the pressure vessel wall. Both of these equations for Q are used for both heatup and cooldown. There are no thermal stresses for constant temperatures at steady-state conditions (Section 2.6.2).

NRC Question 4

In equation 2.5-1 in the report,

$$Kia = 26.78 + 1.233 * \exp [0.0145 (T\text{-}RT_{NDT} + 160)] \qquad (2.5\text{-}1)$$

The above number is shown as 1.223, whereas in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, 1989 edition, the redlined number is shown as 1.233. The differences are also highlighted in WCAP-13829; where on page 3 the value used is 1.223 and on page 7 the value used is 1.233.

The staff notes that in Welding Research Council, Bulletin 175 and in the NRC Standard Review Plant 5.3.2, the number is shown as 1.223. However, the WCAPs 14040 and 13829 reference the ASME code. Explain the discrepancy in your reports and justify the use of 1.223.

RESPONSE:

In the calculation of K_{Ia}, a slightly lower (0.8%) and more conservative value of 1.223 is used instead of 1.233, which would give a higher allowable limit. As noted by the staff, the 1.223 value is consistent with Welding Research Council, Bulletin 175 and NRC Standard Review Plan 5.3.2. The text of WCAPs 14040 and 13829 will be modified to use the 1.223 value consistently and to add appropriate footnotes indicating its conservatism relative to the value in the ASME Code, Appendix G.

NRC Question 5

Demonstrate the method for determining the thermal stress intensity factor, K_{IT}, for heatup and cooldown. Specifically, provide a sample calculation of the process starting with Equation 2.6.1-1 on page 2-14. The sample calculation should be in sufficient detail to provide the staff a reproducible method for verifying the heatup and cooldown curves. The calculation should include the time-dependent temperature distribution determination (page 2-14), the use of the Timoshenko equation (pages 2-14 and 2-15), the determination of the thermal stresses (membrane and bending) employing the linearization technique from Appendix A of the ASME code (page 2-17), and the final value of K_{IT} using Equation 2.6.3-1 (page 2-17).

RESPONSE:

The heatup and cooldown limits are calculated by Westinghouse Computer Code OPERLIM in accordance with NRC approved methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April, 1975. The initial version of the OPERLIM program was documented and verified in WCAP-9186, "Documentation and Verification of the OPERLIM Computer Code," August, 1977. All subsequent revisions have also been documented and verified per Westinghouse configuration control and quality assurance procedures. The verification of WCAP-9186 includes comparison calculations of temperature and thermal stress distributions through the vessel wall, as well as calculation of the thermal stress intensity factor, K_{IT}.

NRC Question 6

In Equation 2.6.3-3 (page 2-18) or Equations 2.6.4-1 and 2.6.4-2 (page 2-19) for cooldown and heatup respectively, the K_{Ia} input is calculated by using Equation 2.5-1 (page 2-10). The staff assumes the temperature input to Equation 2.5-1 is derived using the time-dependent temperature distribution solution described on page 2-14. If this is a valid assumption, demonstrate the use of Equation 2.6.3-3 or Equations 2.6.4-1 and 2.6.4-2 for cooldown and heatup respectively using the temperature distribution solution. If this is not a valid assumption, provide an explanation for the temperature value used in Equation 2.5-1 and subsequently the K_{Ia} used in Equation 2.6.3-3 or Equations 2.6.4-1 and 2.6.4-2.

RESPONSE:

The staff assumption that the temperature input to calculate K_{Ia} in Equation 2.5-1 is derived from the time-dependent temperature distribution solution, as described on page 2-14, is correct. The calculation of the maximum membrane stress intensity factor, $K_{IM(max)}$, per Equations 2.6.3-3, 2.6.4.-1 and 2.6.4-2, is also performed by the OPERLIM Computer Code as documented and verified per Westinghouse configuration control and quality assurance procedures.

NRC Question 7

The LTOP/COMS setpoints do not appear to include instrument error. Explain the absence of this factor and justify the exclusion of a margin for instrument error. (Westinghouse provided a response by letter dated June 16, 1995, which the staff is in the process of reviewing.)

RESPONSE:

Refer to Westinghouse Owners Group letter OG-95-54, dated June 16, 1995 to the NRC.

The spatial core power distribution utilized in the reference forward calculation is derived from statistical studies of long-term operation of Westinghouse 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. A U-235 fission spectrum is employed in the definition of this reference source.

Due to the use of this bounding spatial power distribution, the results from the reference forward calculation establish conservative exposure projections for individual reactors. 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 this reactor result in exposure rates well below these conservative predictions.

These conservative fluence projections are used as a design basis for new reactors until a measurement data base is initiated by withdrawal of a surveillance capsule at the conclusion of the first operating fuel cycle.

Cycle Specific Adjoint Calculations

All adjoint analyses are also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation. Adjoint source locations are chosen at each of the azimuthal locations containing cavity dosimetry as well as 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 each internal surveillance capsule. 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 operating fuel cycle; 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:

$$\phi(R_0,\theta_0) = \int_r \int_\theta \int_E I(r,\theta,E)\ S(r,\theta,E)\ r\ dr\ d\theta\ dE$$

where: $\phi(R_0,\theta_0)$ = Neutron flux (E > 1.0 MeV) at radius R_0 and azimuthal angle θ_0.

$I(r,\theta,E)$ = 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, $I(r,\theta,E)$, permitted the use not only of fuel cycle specific spatial variations of fission rates within the reactor core; but, also allowed 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 increased. Therefore, in implementing the adjoint approach, the effects of plutonium fissioning are accounted for based on individual fuel assembly burnups on a fuel cycle by fuel cycle basis. Averaging of the source over multiple operating cycles is not required.

NRC Question 2

The 47 group SAILOR library is based on ENDF/B-IV data. It is stated that: "... on the average for Westinghouse Plants surveillance capsule dosimetry indicates that the fluence calculations are biased low" and "...for conservatism the fluence values ... are adjusted by the average bias indicated by the dosimetry measurements for that plant":

2.1 Is the low bias a result of using ENDF/B-IV data in thermal shield plants?

RESPONSE:

Yes, comparisons with neutron dosimetry from both internal surveillance capsules and reactor cavity dosimetry indicate that the use of ENDF/B-IV cross-sections in the transport analysis leads to an underprediction of approximately 15%.

As noted in WCAP-14040, Rev. 1 fluence evaluations completed prior to January 1995 made use of the ENDF/B-IV cross-section data sets. Subsequent to the release of the BUGLE-93 ENDF/B-VI based cross-section library in the fourth quarter of 1994, the Westinghouse Fluence Evaluation Methodology was updated and re-benchmarked to incorporate this more accurate data set.

2.2 How was the DOT benchmarked to account for ENDF/B-IV and a thermal shield?

RESPONSE:
The overall exposure evaluation methodology discussed in WCAP-14040, Rev. 1 is based on the underlying philosophy that, in order to minimize the uncertainties in vessel exposure projections, plant specific neutron transport calculations must be supported by;

1) Benchmarking of the analytical approach.
2) Comparison with power reactor surveillance capsule and reactor cavity industry wide data bases.
3) Comparison with plant specific measurements.

That is, as the 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.

The qualification of the Westinghouse transport methodology consisted of the following three parts:

1 - Comparisons with benchmark measurements from the PCA simulator at ORNL.

2 - Comparisons with a series of power reactor measurements that include data both from internal surveillance capsule dosimetry and reactor cavity dosimetry.

3 - An analytic sensitivity study investigating the dominant sources of uncertainty in the transport model.

The results of these studies demonstrate that the overall methodology is capable of providing best estimate fluence evaluations within $\pm 20\%$ 1σ.

In the application of this methodology on a plant specific basis, the qualification results are combined with all available plant specific measurement data to define the biases and uncertainties required to provide projections of the best estimate neutron exposure of the particular pressure vessel.

This benchmarking procedure was also applied to the ENDF/B-VI data set.

2.3 Is it not more reasonable to bias the dosimetry by the plant specific bias?

RESPONSE:
No, the measurement data base acts to remove bias that are introduced by methods approximations as well as by uncertainties in reactor dimensions, fluctuations in water temperatures, and variations in the neutron source in the reactor core. The application of the bias to the pure calculated results is as follows:

Using the Westinghouse neutron exposure methodology, the best estimate exposure of the reactor pressure vessel is developed using a combination of absolute plant specific transport calculations and all available plant specific measurement data. At a minimum this measurement data base will include dosimetry results from the materials surveillance capsules withdrawn over the course of the reactor lifetime as an integral part of the Reactor Vessel Surveillance Program, but may also include additional dosimetry from measurements obtained in the reactor cavity.

Combining the measurement data base with the plant specific calculations, 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.

Bias factors of approximately 1.15 are consistent with the use of the ENDF/B-IV based libraries in the plant specific transport calculations. Experience to date has shown that bias factors of approximately 0.92 are consistent with the use of the newer ENDF/B-VI data libraries in the plant specific transport calculations.

2.4 There is no mention of the associated plant specific uncertainty analysis to be performed to quality the dosimetry. Please comment.

RESPONSE:
The use of the bias factor derived from the measurement data base 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 uncertainty in the derived neutron flux for an individual measurement is obtained directly from the results of a least squares evaluation of dosimetry data. The least squares approach combines individual uncertainty in the calculated neutron energy spectrum, the uncertainties in dosimetry cross-sections, and the uncertainties in measured foil specific activities to produce a net uncertainty in the derived neutron flux at the measurement point. The associated uncertainty in the plant specific bias factor, K, derived from the M/C data base, in turn, depends on the total number of available measurements as well as on the uncertainty of each measurement. Because of the dependence on the size of the overall data base, plants that have incorporated supplemental reactor cavity dosimetry will, in general, have a lower uncertainty than plants with measurements from internal surveillance capsules alone.

The positioning uncertainties for dosimetry are taken from parametric studies of sensor position performed as part of the analytical sensitivity studies included in the qualification of the methodology. The uncertainties in the exposure ratios relating dosimetry results to positions within the vessel wall are again based on the analytical sensitivity studies of the vessel thickness tolerance for cavity data and on downcomer water density variations and vessel inner radius tolerance for surveillance capsule measurements. Thus, this portion of the overall uncertainty is controlled entirely by dimensional tolerances associated with the reactor design and by the operational characteristics of the reactor.

The net uncertainty in the bias factor, K, is combined with the uncertainty from the analytical sensitivity study to define the overall fluence uncertainty at the pressure vessel wall. Typically, the derived uncertainties in the bias factor, K, and the additional uncertainty from the analytical sensitivity studies combine to yield net uncertainties in the range of 10-15% for neutron fluence (E > 1.0 MeV).

WOG

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OG-95-54 WCAP-14040 - NP

June 16, 1995

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

Attention: Chief, Planning, Program and Management Support Branch

Subject: Westinghouse Owners Group
Transmittal of Response to Concerns Identified During Review of WCAP-14040, Rev. 1, Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"

Reference: 1) R.A. Newton to Document Control Desk, Attention J.R. Strosnider, Transmittal of Report: WCAP-14040, Rev. 1, [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", OG-94-102, dated December 20, 1994.

This letter transmits twelve (12) copies of the attachment "Cold Overpressure Mitigating System (COMS) Setpoints and Reactor Coolant System (RCS) Heatup and Cooldown Curves: Introduction of random Instrument Uncertainties Adds Unnecessary Conservatisms to WCAP-14040 Methodology" which addresses concerns raised by the NRC Staff in their review of WCAP-14040, Revision 1 [Non-Proprietary] Entitled "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", dated December 1994, previously submitted by Reference 1).

Also attached is:

1. One (1) copy of the Copyright Notice.

The attachment "Cold Overpressure Mitigating System (COMS) Setpoints and Reactor Coolant System (RCS) Heatup and Cooldown Curves: Introduction of random Instrument Uncertainties Adds Unnecessary Conservatisms to WCAP-14040 Methodology" provides the Westinghouse Owners Group (WOG) technical response to the NRC Staff concern raised at the May 8, 1995 telecon with TVA (lead plant Watts Bar) and Westinghouse. The NRC Staff comment was in regard to the pressure and temperature instrumentation uncertainties for the Cold Overpressure Mitigating System (COMS) setpoints as addressed in the fourth sentence on page 3-6, of WCAP-14040, Rev. 1.

Correspondence with respect to the copyrighted aspects of this WCAP should be addressed to Mr. N.J. Liparulo, Manager Nuclear Safety Regulatory and Licensing Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, PA 15230-0355.

Very truly yours,

Lee Bush, Chairman
Licensing Subcommittee
Westinghouse Owners Group

RAN/dac

attachments

cc: WOG Steering Committee (1L,1A)
Westinghouse Owners Group Primary Representatives (1L,1A)
Licensing Subcommittee Representatives (1L,1A)
Peter S. Tam, USNRC (1L,1A)
Jack R. Strosnider, USNRC (1L,1A)
John C. Tsao, USNRC (1L,1A)
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Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Curves: Introduction of Random Instrument Uncertainties Adds Unnecessary Conservatism to WCAP-14040 Methodology

The Cold Overpressure Mitigation System (COMS) supplements plant operating and administrative controls to protect the reactor vessel from exposure to conditions which may cause a crack to propagate in a brittle manner. WCAP-14040 (Reference 1) describes the methodogy endorsed by Westinghouse and the Westinghouse Owners Group for developing COMS setpoints and RCS Heatup and Cooldown Limit curves. This paper supports the Westinghouse and Westinghouse Owners Group position, as stated in Section 3.2.5 of WCAP-14040, in favor of excluding random instrument uncertainties from this COMS methodolgy. Specifically,

- The methodology of WCAP-14040 incorporates adequate conservatisms that considerably override the need to incorporate additional margins for random temperature and pressure uncertainties,

- Incorporating additional margins for random temperature and pressure uncertainties unnecessarily reduces operating flexibility, particularly at lower temperatures, between the Appendix G limit and the minimum pressure necessary for proper operation of reactor coolant pump seals, without a commensurate increase in protection of the reactor vessel integrity, and

- By reducing operating flexibility, the likelihood of COMS actuation is increased.

Heatup and Cooldown Curves

As referenced by 10 CFR Part 50 (Reference 4), requirements for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates are specified in the ASME Code, Appendix G of Section XI (Reference 3). It should be emphasized that development of heatup and cooldown curves based on Appendix G methodology incorporates significant conservatisms that provide more than adequate protection to the reactor vessel. Furthermore, neither the Appendix G methodology nor current federal regulations require the incorporation of random instrument uncertainties.

Conservatisms in the heatup and cooldown limits include:

a. An assumed defect in the reactor vessel wall has a depth equal to 1/4 of the thickness of the vessel wall (1/4T) and a length equal to 1-1/2 times the thickness of the vessel wall.

b. A factor of safety of 2 is applied to the membrane stress intensity factor (K_I, for pressure).

c. The limiting toughness is based upon a reference value (K_{IR}) which is a lower bound on the dynamic crack initiation or arrest toughnesses, and

d. A 2-sigma margin term is applied in determining the adjusted reference temperature (ART) that is used to calculate the limiting toughness.

Figure 5 from Reference 2 (attached) illustrates the relative impact of the conservatisms outlined in (a) through (c), as well as their cumulative impact when applied using the Appendix G methodology. [1] Pressure/temperature limits are shown for the assumed 1/4T critical flaw size and relaxed critical flaw sizes ranging from .5 to 1 inch. When compared to lesser flaw depths in this example, the required 1/4T flaw assumption reduces allowable pressure/temperature limits by approximately 400-1200 psi at lower temperatures, where brittle fracture is of most concern. Similarly, when a factor of safety of 2 is applied to the pressure/temperature limits, the allowable pressure limit is actually based on the stress intensity resulting from a pressure twice the magnitude of the limit (e.g., an allowable pressure of 400 psig is based on the stress intensity resulting from a pressure of 800 psig).

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1 The pressure/temperature limits were based on a three loop, high copper content reactor vessel having an end of life fluence.

Article G-2120 of the ASME code indicates that "due to the safety factors recommended here [in this Appendix], prevention of nonductile fracture is ensured for some of the most important situations even if the defects were about twice as large in linear dimensions as the postulated maximum defect. Figure G-2210-1 (attached) is a curve showing the relationship that can be conservatively expected between the critical, or reference, stress intensity factor K_{IA} and a temperature which is related to the reference nil ductility temperature (RT_{NDT}). This curve is based on the lower bound of static, dynamic and crack arrest critical K_I values measured as a function of temperature. No available data points for static, dynamic or arrest tests fall below this curve. An analytical approximation of this curve is:

$$K_{IA} = 26.78 + 1.233\ e^{[0.0145\ (T\ -\ RTNDT\ +\ 160)]}$$

An increase in the RT_{NDT} of the material makes the value of K_{IA} more conservative. The calculational methods for adjusted RT_{NDT} require that a 2-sigma margin be added to its value, thus introducing additional conservatism.

Acceptance criteria in Appendix E of the ASME code, which governs the evaluation of actual overpressure events on the integrity of the reactor vessel beltline region, also demonstrate the conservatism of the Appendix G methodology. Refering to Table E-2 of Appendix E, the minimum initiation crack size which can be assumed is less than 1 inch, compared to the 1/4T flaw assumption (approximately 2.1 inches) in the Appendix G methodology. Appendix E permits the evaluation of actual overpressure effects based on the crack initiation toughness outlined in Appendix A, Figure A-4200-1 (attached). As demonstrated above, Appendix G methodology relies on the reference stress intensity factor of Figure G-2210-1, (identical to the lower bound crack arrest values of Figure A-4200-1) which is considerably more limiting.

<u>Setpoint Methology</u>

As described in WCAP-14040, COMS setpoints are chosen to prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G requirements during two limiting design basis events, a heat injection transient and a mass injection transient. The heat injection transient is defined by the inadvertent startup of a reactor coolant pump when the RCS temperature is as much as 50F lower than the steam generator secondary side temperature. The mass injection transient is defined by the introduction of charging and/or safety injection flow, with letdown isolated. The following additional conservatisms are relevant to the analysis:

a. The RCS is water solid, with the RHRS isolated (RHRS valves not available to mitigate the transient).

b. The RCS is at cold conditions, enclosed by a rigid, nonyielding boundary. The pressure increase during an overpressure event is the direct result of the inability of the coolant to expand into a larger volume. In reality, the pressure boundary is somewhat elastic and would expand at elevated temperature and increasing pressure.

c. Failure of flow controls during the mass injection event. Charging and/or Safety Injection flow immediately steps up to and remains at full capacity.

d. A maximum allowable setpoint curve is developed from the most limiting pressure increase from the mass injection and heat injection case so that neither the Appendix G limit nor the PORV piping limit is exceeded.

e. Failure of one pressurizer PORV is assumed, such that the relief capacity of one PORV is adequate to maintain system pressure below the maximum pressure limit.

f. PORV flow capacity is linear with valve travel.

g. High secondary to primary heat transfer coefficient assumed.

h. The mass injection case is analyzed solely at cold condtiions, with the pressurizer water solid and at the conditions as the RCS. Injection flow is at a temperature equal to the RCS. The results from this analysis are applied in the setpoint development over the entire temperature range.

As such, the mass injection and heat injection events are well defined such that plant response is conservatively modelled. In addition, setpoint development as described in WCAP 14040 accounts for other known physical effects such as (1) heat transport effects which, during the heat injection event, account for a 50F temperature difference between the temperature sensors and the reactor vessel, and (2) the pressure difference between the wide range pressure transmitter and the reactor vessel beltline region.

In addition to the ultimate goal of protecting Appendix G limits, setpoints are selected such that resulting undershoots in pressure do not challenge the reactor coolant pump seal limits. Accounting for additional instrument uncertainties imposes additional operating contraints. At lower temperatures particularly, only a narrow window is available for operation between the Appendix G limit and the minimum pressure required in order to assure proper operation of the pump seals. Reducing the maximum allowable setpoint further in order to incorporate instrument uncertainty reduces this available margin further, and in extreme cases, can decrease the maximum allowable PORV setpoint below the pump seal limit. The magnitude of these uncertainties is insignificant when compared to conservatisms in the heatup and cooldown curves and they add only a neglible amount of margin to protecting the reactor vessel. However, the impact on plant operations could be significant. Since instrument uncertainties reduce operating flexibility, the likelihood of COMS actuation is increased. Thus, the likelihood of upset conditions in the plant is increased without a commensurate increase in protecting the reactor vessel from brittle fracture.

Conclusions

A number of issues have been discussed which underscore the conservatism of the WCAP-14040 methodology for developing heatup and cooldown curves and determining COMS setpoints which protect the reactor vessel from brittle fracture. Based on the above discussion, Westinghouse continues to support the exclusion of random instrument uncertainties in the COMS methodology and requests NRC acceptance of WCAP-14040 without exception.

References:

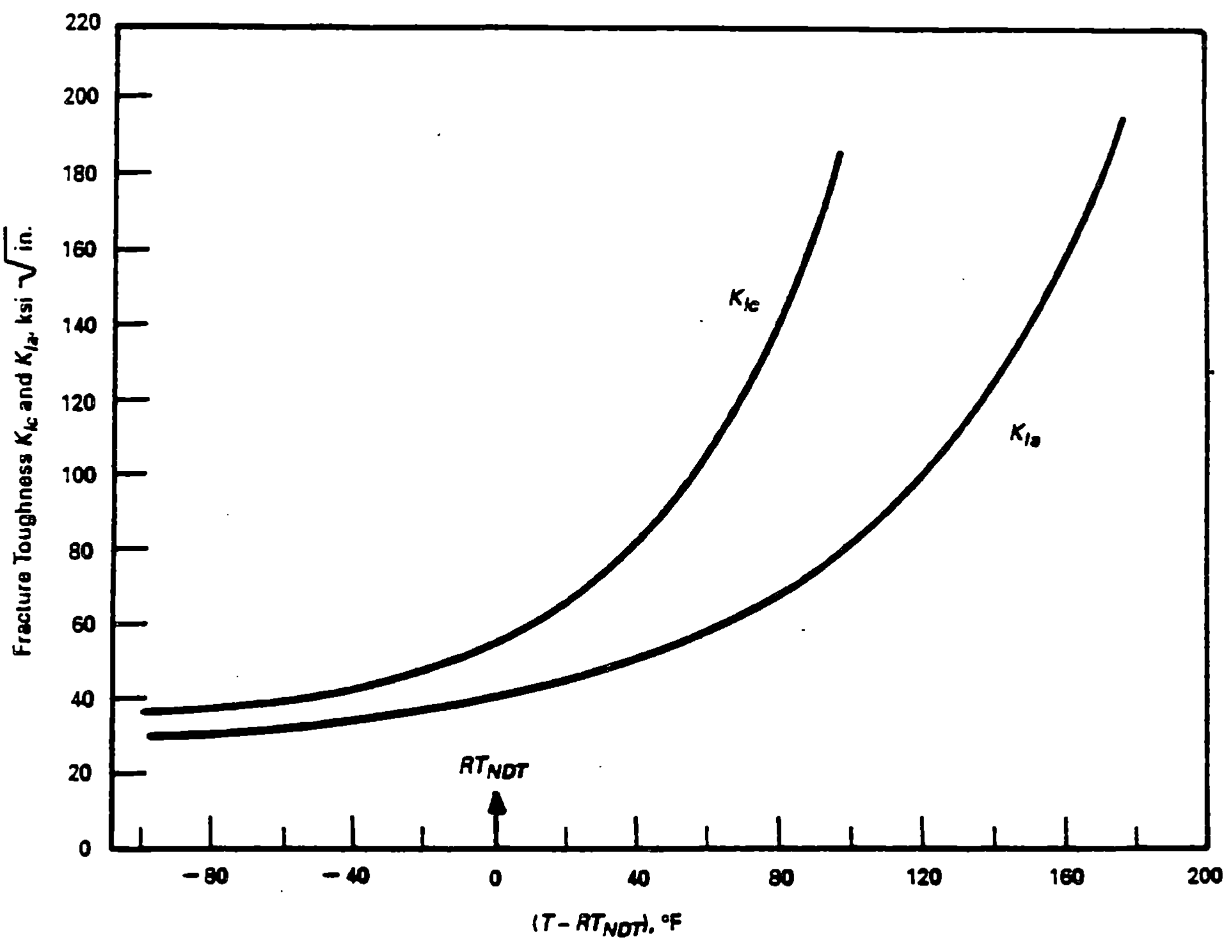
1. WCAP-14040, Rev. 1 "Methodology used to Develp Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Westinghouse Electric Corporation, December 1994.

2. "Influence of Material Property Variation on the Assessment of Structural Integrity of Nuclear Components", J. N. Chirigos and T. A. Meyer, Journal of Testinghouse and Evaluation, Vol. 6, No. 5, September 1978.

3. ASME Boiler and Pressure Vessel Code, Section III, Division I, Appendix G, "Protection Against Nonductile Failure"; Section XI, Division I, Appendix G, "Fracture Toughness Criteria for Protection Against Failure"; Section XI, Division I, Appendix E, "Evaluation of Unanticipated Operating Events".

4. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements", January 1, 1994.

Attachments



A92 FIG. A-4200-1 LOWER BOUND K_{Ia} AND K_{Ic} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS

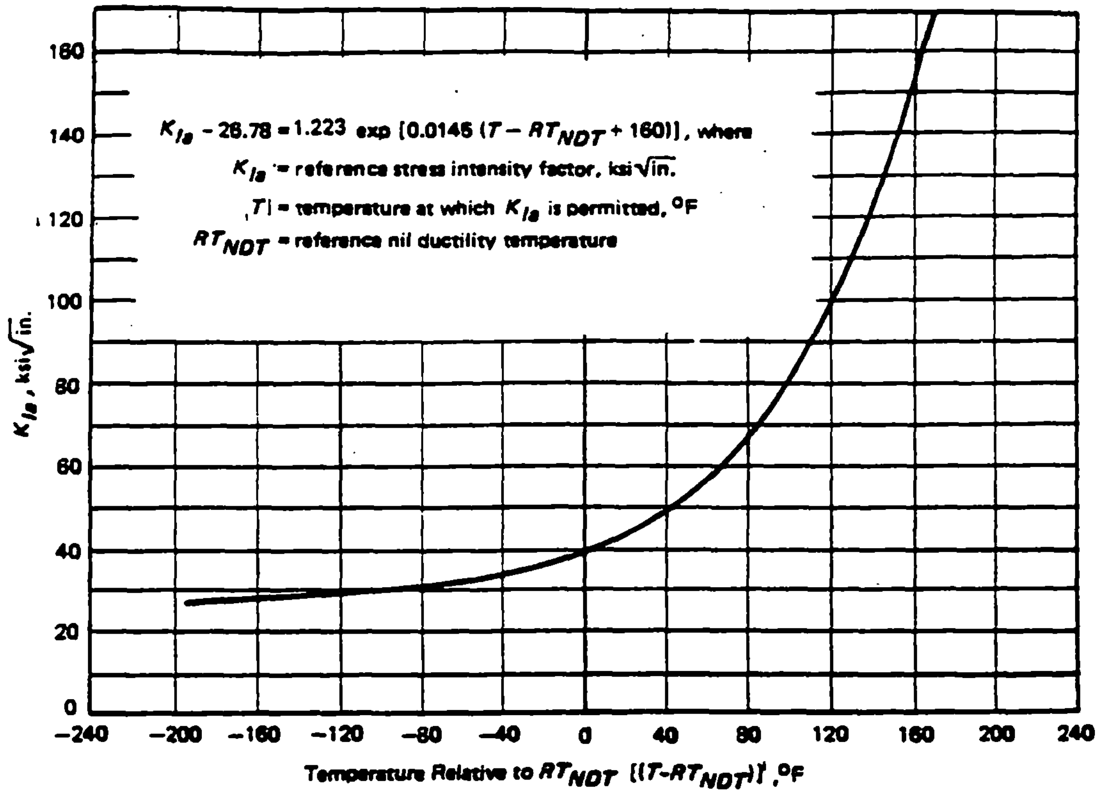


FIG. G-2210-1

G-2212 Material Fracture Toughness

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

G-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.

G-2213 Maximum Postulated Defect

The recommended maximum postulated defect is that described in G-2120.

G-2214 Calculated Stress Intensity Factors

G-2214.1 Membrane Tension. The K_I corresponding to membrane tension for the postulated defect of G-2120 is $K_{Im} = M_m \times$ membrane stress, where M_m is as shown in Fig. G-2214-1.

G-2214.2 Bending Stress. The K_I corresponding to bending stress for the postulated defect of G-2120 is $K_{Ib} = M_b \times$ maximum bending stress, where M_b is two-thirds of the M_m shown in Fig. G-2214-1.

G-2214.3 Radial Thermal Gradient. The K_I produced by a radial thermal gradient across a wall thickness for the postulated defect of G-2120 is $K_{It} = M_t \times$ temperature difference through the wall, °F, where M_t is as shown in Fig. G-2214-2.

(a) The M_t values in Fig. G-2214-2 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(1) An assumed shape of the temperature gradient is approximately as shown in Fig. G-2214-3.

(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) If the conditions of G-2214.3(a) are not met, other means must be used for calculating the K_I produced by thermal stress. For example, the moment produced by the radial thermal gradient may be calculated

Normal Condition Plant Operation

The standards and criteria applicable during normal plant operation are Appendixes G of both ASME-III and 10 CFR 50. The principal material property of interest is the material fracture toughness, which is given in the code as the K_{IR} curve in ASME-III, Appendix G, and as equations for initiation toughness K_{Ic} and arrest toughness K_{Ia} in Appendix A of ASME-XI. The K_{Ia} of ASME-XI corresponds to K_{IR} in ASME-III, Appendix G.

Figure 1 gives a plot of K_{Ic} and K_{Ia} (or K_{IR}) as obtained from the ASME-III, Appendix G. Temperature is plotted relative to the reference transition temperature, RT_{NDT}. The code gives no guidance as to the value of toughness on the upper shelf, which is assumed here to be 220 MPa·m$^{1/2}$ (200 ksi·in.$^{1/2}$). It is felt that this value is supported by existing data; however, the actual value taken is not too important for the task at hand, which will include an assessment of varying shelf toughness.

Fluence Dependence

An important aspect of identifying the toughness curves to be applied to a given situation is to account for the effect of irradiation. This can be accomplished by employing so-called trend curves that give the change in reference temperature with fluence at given copper and phosphorus levels. Regulatory Guide 1.99 presents such curves, as does Ref *1*.

As can be seen by examining these documents, there is a difference in the fluence dependence assumed on property changes. This difference is illustrated in Fig. 2. This figure presents ΔRT_{NDT} (°F) as a function of increasing fluence. Two fluence trends are shown in Fig. 2: one curve is ΔRT_{NDT} plotted as a function of the square root (½ power) of the fluence and the second curve is ΔRT_{NDT} plotted as a function of the cube root (⅓ power) of the fluence. The impact of different ΔRT_{NDT} fluence curves on reactor vessel structural integrity can now be assessed.

ASME-III, Appendix G Assessment

The Appendix G assessment involves determining the stress intensity factor owing to the design loading conditions acting on an assumed reference flaw usually taken as one fourth of the wall thickness (¼T) in depth. Appropriate safety factors are applied to the calculated stress intensity factors and it is required that the stress intensity factors (with the safety factors) be less than the value defined by the K_{IR} curve, that is, the reference fracture toughness.

The specific Appendix G evaluation of heat-up and cool-down in nuclear power plants will be discussed separately in a later section of this paper.

By considering the various design loading conditions such as steady state operation and step load change in power, the stress intensity factor is calculated for each loading condition, the appropriate safety factors are applied, and the resulting stress intensity factor is compared with the K_{IR} curve at the temperature where the particular loading condition occurs. An example of the results of this type of analysis is shown in Fig. 3 for an analysis of the beltline region of a four-loop reactor vessel at end-of-life fluence conditions.

The points plotted in Fig. 3 represent the stress intensity factors calculated for individual loading conditions. The position of the reference toughness curve (K_{IR} curve) has been established by two different methods: Method A, ⅓ power ΔRT_{NDT}-fluence dependence and Method B, ½ power ΔRT_{NDT}-fluence dependence. Even though there is a significant difference in the K_{IR} curve, for these loading conditions, in this particular example ample margin still remains with the curved portion of the K_{IR} curves.

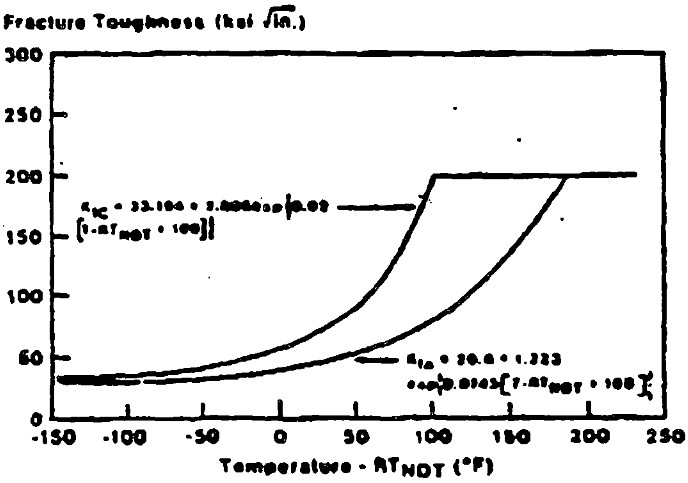


FIG. 1—*Reference fracture toughness curves [1 ksi·in.$^{1/2}$ = 1.1 MPa·m$^{1/2}$; °C = (°F − 32)/1.8].*

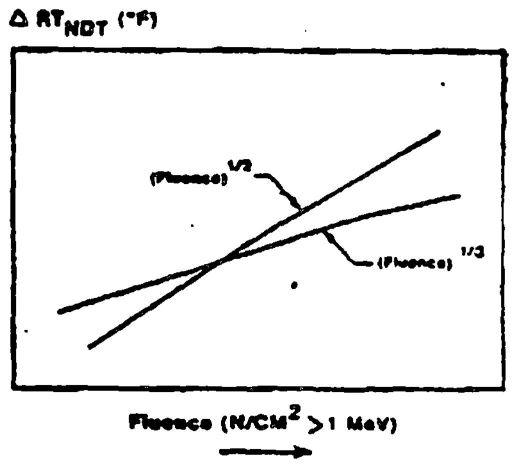


FIG. 2—*Change in reference temperature with fluence.*

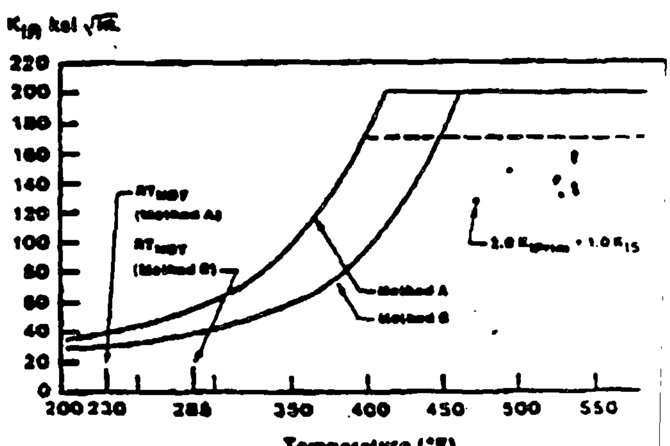


FIG. 3—*Temperature curve versus K_{IR} for four-loop reactor at the beltline [1 ksi·in.$^{1/2}$ = 1.1 MPa·m$^{1/2}$; °C = (°F − 32)/1.8].*

There is, however, a less satisfactory situation relative to the proximity of the individual K_I values to the assumed toughness at the shelf. There is ample margin relative to the assumed value of 220 MPa·m$^{1/2}$ (200 ksi·in.$^{1/2}$) and less, of course, at the value of 182 to 187 MPa·m$^{1/2}$ (165 to 170 ksi·in.$^{1/2}$), which represents the termination of the K_{IR} curve in Appendix G of ASME-III. It should be emphasized that there is considerable conservatism in the calculated stress intensity factors, arising mainly from the factor of 2 on pressure stress intensity and the assumption of a ¼T flaw, as required by ASME-III, Appendix G.

The upper shelf values portrayed in Fig. 3 are considered adequate, in particular for plants built to controlled copper limits. Results from on-going tests will help establish upper shelf levels for older plants with relatively high copper content.

Heat-Up and Cool-Down

Typical heat-up and cool-down limit curves are constructed by the methods of ASME-III, Appendix G, for specified heat-up and cool-down rates, that is, 33°C/h (60°F/h) cool-down rate.

A time period of reactor operation is chosen for which heat-up and cool-down limit curves are to be determined and the RT_{NDT} is then identified. The stress intensity factor K_I for thermal stress is calculated at a specified coolant temperature relative to the RT_{NDT}. The K_{IR} is determined from ASME-III, Appendix G and then the allowable reactor system pressure is calculated which satisfies the relation:

$$K_{IR} > 2K_I \text{ pressure} + K_I \text{ thermal}$$

This procedure is repeated for temperatures covering the range of reactor coolant temperatures of interest and the results are presented as curves of maximum system pressure versus system temperature, as shown in Fig. 4. The curves in Fig. 4 are representative curves for a three-loop, high-copper vessel having an end-of-life fluence.

Two end-of-life limit curves are shown in Fig. 4, one determined by the ⅓ power fluence dependence, the other by the ½ power fluence dependence of shift in RT_{NDT}. The case chosen was for a reactor design that operates at low average coolant temperature T_{avg}.

The criticality limit is determined by adding a 22°C (40°F) margin to the maximum pressure limit to prevent brittle fr[acture] based on ASME-III, Appendix G. Note that in this case re[actor] operation is marginal based on Method A (⅓ power depend[ence]) and impossible based on Method B (½ power dependence). It can be seen that the fluence dependence of ΔRT_{NDT} is a [sig]nificant parameter that must be established. It is essential the proper fluence dependence, ½ power versus ⅓ power [the] dependence, be established since it can determine whether re[actor] operation is possible at end-of-life conditions, with the assump[tion] of course, that the margins presently required will be applicab[le in] the future.

It appears that there is a more critical need to establish the e[ffect] of fluence on the shift in transition temperature rather than [the] effect of fluence on upper shelf toughness.

An indication of the conservatism in the ASME-III, Appendi[x G] limits can be seen in Fig. 5. This figure presents the maxim[um] pressure limit at end-of-life conditions for a three-loop ve[ssel] having high (0.30%) copper content.

The lowest curve is that obtained by using the ASME-[III] Appendix G, methods. The subsequent curves result by:

(1) relaxing the factor of 2 on K_I (pressure),
(2) using the K_{Ic} instead of the K_{IR} curve.

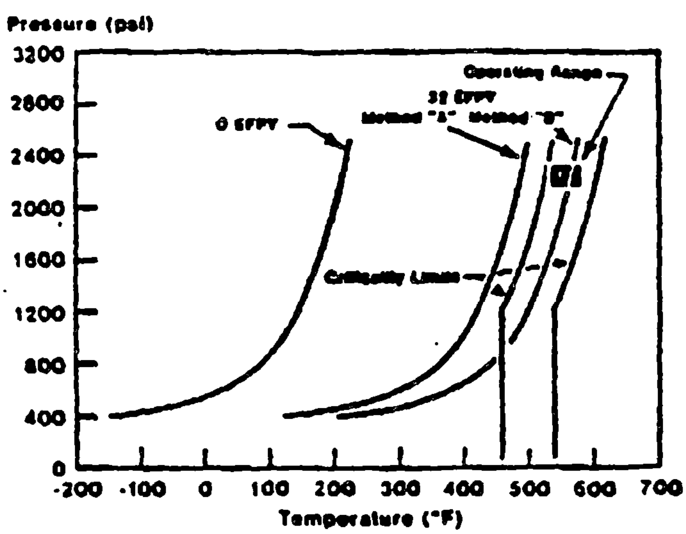


FIG. 4—*Operating limits according to ASME-III, Appendix G (1 psi = 6.9 kPa; °C = (°F − 32)/1.8; EFPY = effective full power years.*

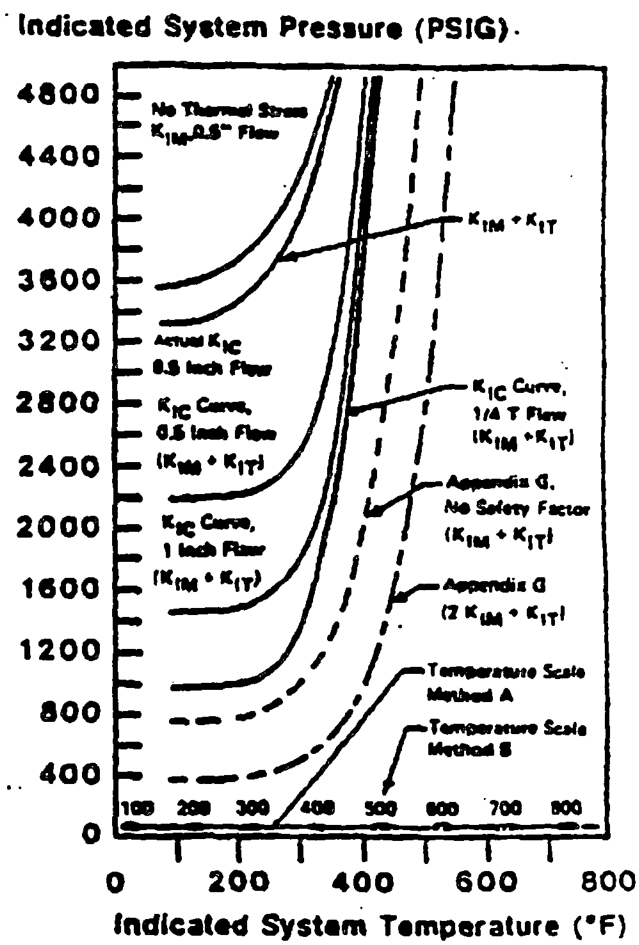


FIG. 5—*Reactor vessel cool-down limit curves. ΔT/Δt, maximum 33°C/h (60°F/h) (1 psig = 6.9 kPa; 1 in. = 25.4 mm; °C = (°F − 32)/1.8)*

(3) relaxing the assumed critical flaw size from ¼T to 25.4 and 12.7 mm (1 and ½ in.), and, finally,

(4) basing the curve on an assumed "actual toughness," which can be expected to fall above the code K_{Ic} curve, a value which may result from the testing of specimens contained in reactor vessel surveillance capsules.

Figure 5 illustrates explicitly the conservatism that exists in the current methods for defining maximum pressure limits. The lower temperature scale results when the toughness curve is shifted by the ½ power fluence dependence as opposed to the upper scale, which is based on the ⅓ power fluence dependence.

Figure 5 demonstrates the need to establish the correct method of extrapolating the effect of fluence on ΔRT_{NDT}, and it also illustrates the need to assure that shifting the reference toughness curve by use of Charpy data is indeed properly conservative. This can be accomplished by conducting irradiation tests that include both Charpy and fracture mechanics specimens that yield a direct measure of fracture toughness and comparing the shifts in toughness curves. Support for the contention that the method of accounting for the effects of irradiation on toughness using Charpy data to shift the K_{IR} curve is conservative can be seen in Fig. 6.

In Fig. 6 the dynamic fracture toughness obtained from 12.7-mm (½-in.) thick wedge-opening-loaded samples irradiated in two different surveillance capsules are plotted. These data, although limited, indicate that the toughness is significantly above the K_{IR} curve, which was positioned by using a ½ power fluence dependence on Charpy data. The dashed curve was drawn with the shape of the K_{Ic} curve and positioned by the point at $T - RT_{NDT} = 38$°C (100°F). The dashed curve is that used to define "actual" toughness in Fig. 5. This plot also suggests that the shelf toughness for irradiated (~7 × 10[18] n/cm²) material is greater than 193 MPa·m[½] (175 ksi·in.[½]).

Assessment of Vessel Integrity Under Accident Conditions

Accident conditions are those severe conditions which, although not expected to occur, are considered in the reactor vessel design because they can produce high thermal stresses in the re vessel. To evaluate the reactor vessel under thermal shock c tions [1] a description of the accident transient, usually sented in terms of the coolant pressure and temperature function of time into the transient, is first developed. From information the temperature distribution, or the temperature file, through the reactor vessel wall at fixed times into the tran is calculated. At a given time into the transient, the temper profile is used to determine the thermal stress distribution thr the wall. The thermal stress and the pressure stress (if any then used to determine the stress intensity factors for a postul flaw of varying depth through the wall. The calculation is usi performed at a given time into reactor life, typically at end-of which provides the maximum fluence at the inside diamete the vessel. The distribution of fluence through the vessel wal conjunction with the temperature profile at a given time into transient, then permits the determination of the fracture to ness, K_{Ic} and K_{Ia}, as a function of position in the vessel wall given point in vessel life and at the particular time into transient.

The results of such a calculation can be represented in a pl stress intensity factor as a function of a/t, where a is the c depth or position in the wall of thickness t. Such a plot is sh in Fig. 7.

In Fig. 7 the variation of K_I, K_{Ic}, and K_{Ia} through the v wall is presented for 600 s into the transient. Where $K_I \geq K_{Ic}$ ture initiation will occur, and if initiation occurs, arrest will o when $K_I < K_{Ia}$.

It can be seen in Fig. 7 that the K_I curve has just missed lower part of the K_{Ic} curve and that initiation will not occur i some crack depth greater than 0.6 a/t. Obviously, a slight cha in the position of the K_{Ic} (or K_I) curve could have lead to initia at flaw sizes less than 0.1 a/t.

If similar plots for longer times into the transient are c sidered, as presented in Fig. 8, it can be seen that the initia size could be very small or very large depending on how well K

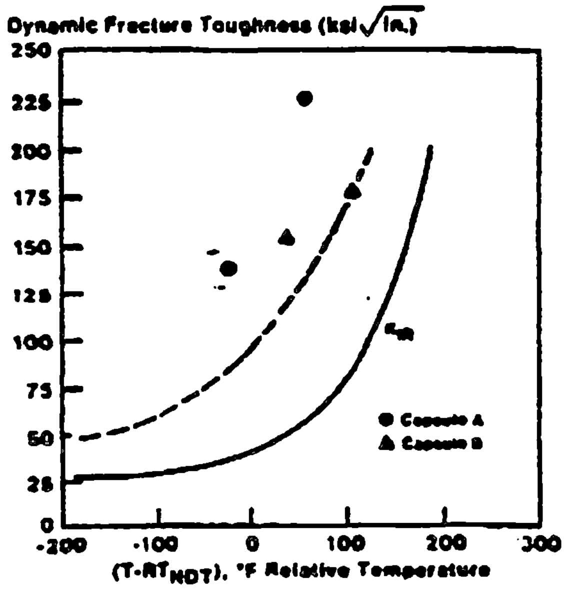


FIG. 6—*Surveillance data [1 ksi·in.[½] = 1.1 MPa·m[½]; °C = (°F − 32)/1.8].*

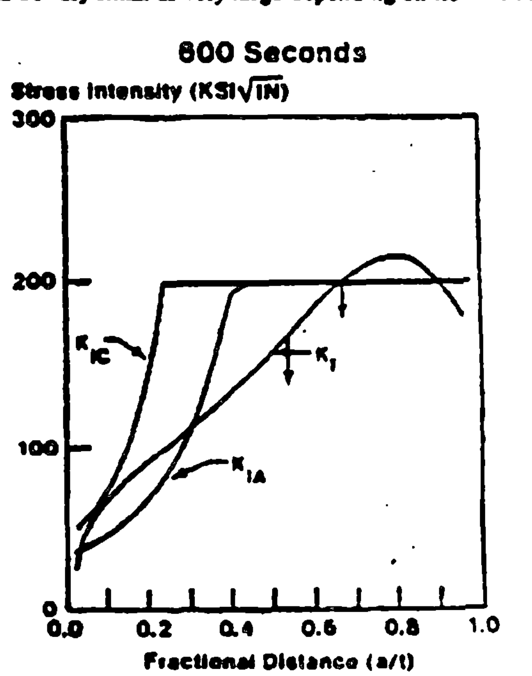


FIG. 7—*Thermal shock evaluation for 600 s into the transient (1 ksi·in.[½] = 1.1 MPa·m[½]).*

known. Actually, in this series of curves, the K_I value never reaches the K_{Ic} value until the upper shelf, where the critical size (minimum crack initiation depth) is very large in terms of the wall thickness. As illustrated here the results are very sensitive to the location of the toughness curve. Obviously, the method of shifting the unirradiated toughness curves can have a very significant impact on the calculated critical flaw size.

The relationship of the actual vessel toughness to the assumed design K_{Ic} curve can also be very significant in determining the actual critical flaw size determined for the vessel. A small change in toughness can change the intersection point from a very small to a very large value of a/t (or vice versa).

Effect of the Upper Shelf Toughness Value

The sensitivity of the calculated critical flaw size when $K_I > K_{Ic}$ in the upper shelf region is shown in Fig. 9. Here, as before, the standard shelf value has been taken as 220 MPa·m$^{1/2}$ (200 ksi·in.$^{1/2}$), which, for this case, yields a_c (minimum or critical flaw size) of ~0.68 a/t—very large. If the shelf were taken as 275 MPa·m$^{1/2}$ (250 ksi·in.$^{1/2}$) there would be no a_c value, which for all practical purposes is not much different than 0.68 a/t. If the shelf were 182 MPa·m$^{1/2}$ (165 ksi·in.$^{1/2}$), there is, of course, a reduction in the a_c value to 0.54 a/t, again, very large.

Thus in the case of an accident evaluation the exact value of the upper shelf is not very significant. The opposite is true for the lower portion of the K_{Ic} curve.

Figure 9 also illustrates that if initiation had occurred at about 0.08 a/t, arrest could be expected at a/t ~0.30, that is, where K_I falls below the K_{Ia} curve. Thus, the validity of the arrest concept and the location of the arrest curve can be extremely important, in particular when the initiation flaw size turns out to be small.

Effect of Thermal Properties

We have, to this point, focused attention on the uncertaintie the critical flaw size calculation that result from uncertaintie: the material property values: initiation toughness, arrest tou ness, and upper shelf toughness. We have not considered certainties in the K_I curve that may result from uncertainties material thermal properties. Also, uncertainties in material ther properties can result in changes in the location of the toughn curves since the thermal properties can change the temperat profiles.

In Fig. 10 a temperature profile that results from an assun reduction of 25% in thermal conductivity, because of irradiati is compared with the base case, that is, no reduction in therr conductivity.

It is known that neutron irradiation will reduce electrical c ductivity in metals and it is reasonable to assume a reduction thermal conductivity. To the best of the authors' knowledge thermal properties of irradiated reactor vessel steel have not b measured. The 25% reduction is arbitrary and was chosen illustrate an effect.

Very little change occurs in the temperature distribution n the vessel inside diameter, but towards the vessel outside diame there is about a 28°C (50°F) difference in temperature.

In arriving at this plot, a 25% reduction in conductivity of b clad and base metal was assumed. Other calculations show t the effect of the clad is negligible and the results portrayed due almost completely to the assumed reduction in base m conductivity.

The thermal stress profile resulting from this temperature p file is shown in Fig. 11. In this figure it can be seen that a sign cant increase in thermal stress has resulted from the 25% red tion in conductivity, increased tensile stress in the inner regi and increased compressive stress in the outer region. A fract

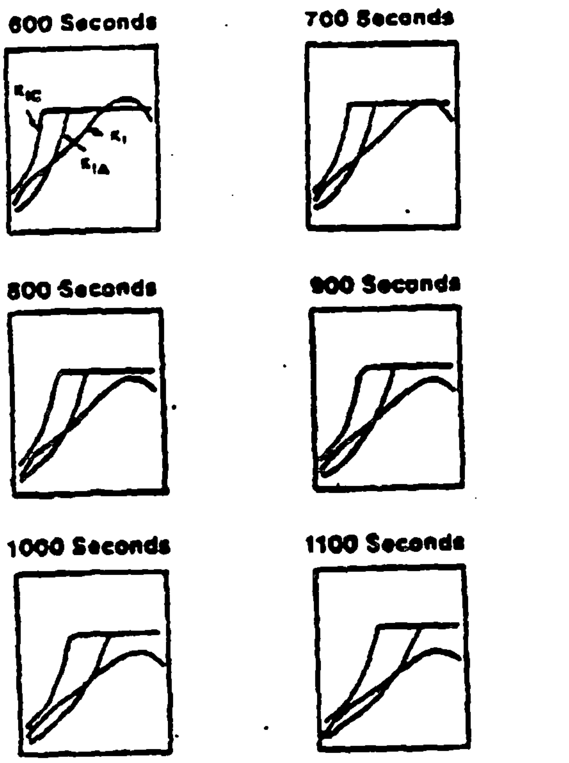


FIG. 8—*Thermal shock evaluation for longer times into the transient.*

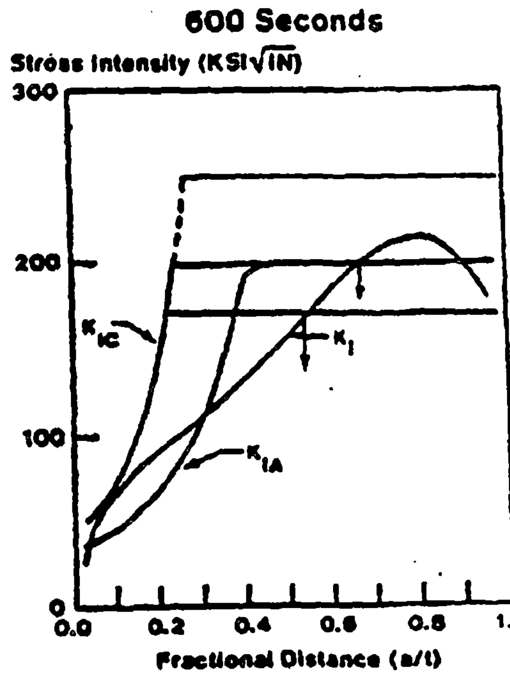


FIG. 9—*Thermal shock evaluation: effect of upper shelf (1 ksi·in.$^{1/2}$ = 1.1 MPa·m$^{1/2}$).*

mechanics analysis would be required to evaluate the impact of changes in thermal conductivity on the reactor vessel integrity.

Fatigue Crack Growth

Data obtained to date for reactor vessel steels indicate a large dependence of crack growth rate on loading frequency and environment—water versus air [2]. At low frequencies the appear to differ by a factor of about 20 owing to environ as shown in Fig. 12. This figure is schematic and is int only to illustrate the difference caused by the environmental and the large conservatism contained in fatigue calculations no consideration is given to the presence of the cladding, prevents access of the water to the base metal.

The presence of an environmental effect suggests the poss of an in-pile effect since the radiation environment can b down the water and lead to species which may control th hanced crack growth rates observed when tests are conduct water. This is an area that should be evaluated.

There is a dearth of data on crack growth rates pertine actual reactor vessel evaluations, that is, irradiated ma tested in a water environment. Limited data exist for irrad material tested in air [3,4] and no data exist for testing in a environment. There does not appear to be an effect of irradi on crack growth rates when the testing is performed in a environment.

Since the present analytical methods, which do not conside presence of the clad, are conservative for reactor vessel fa crack growth evaluations, the analysis results could be impr by obtaining and using more pertinent fatigue crack growth obtained in vacuum or air environments.

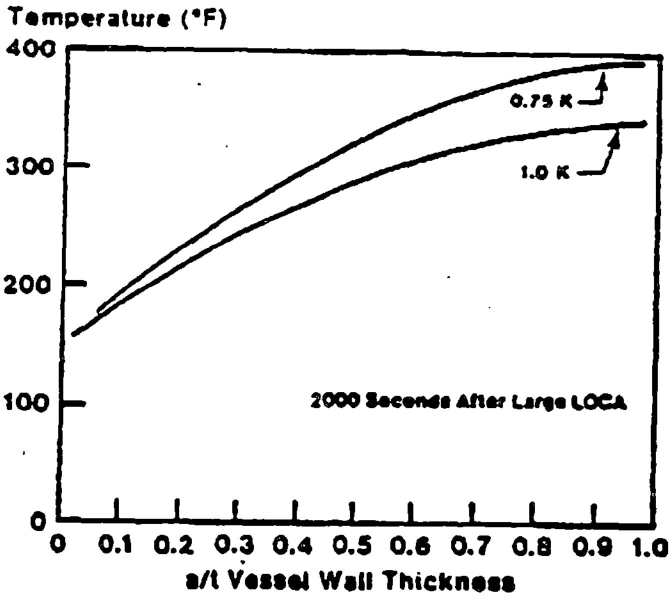


FIG. 10—*Temperature profiles: effect of thermal conductivity* [°C = (°F − 32)/1.8; LOCA = *loss-of-coolant accident*].

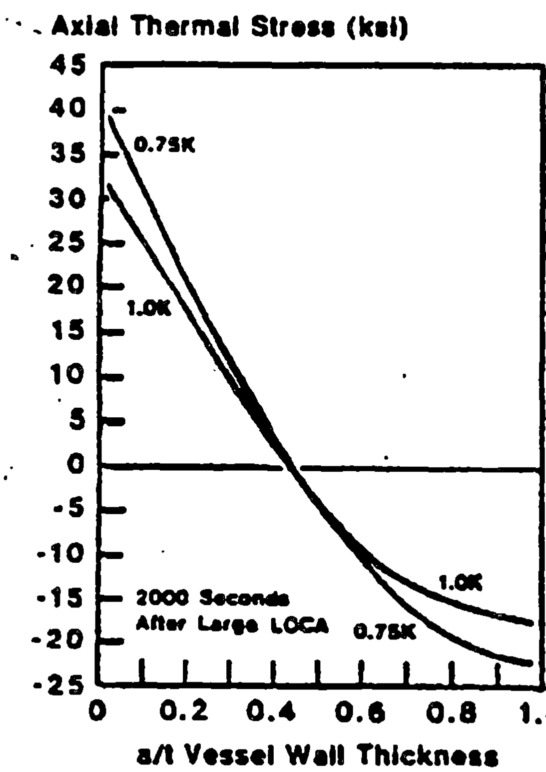


FIG. 11—*Thermal stress profiles: effect of thermal conductivity* (*1 ksi = 6.9 MPa*; LOCA = *loss-of-coolant accident*).

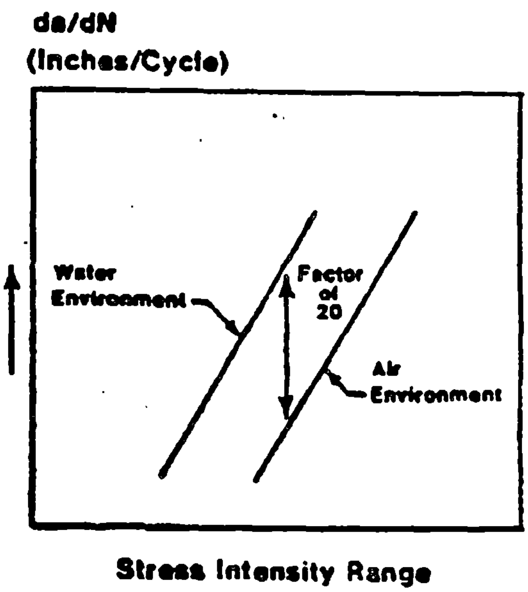


FIG. 12—*Schematic diagram of crack growth rate versus intensity range* (*1 in. = 25.4 mm*).

Summary

Several areas have been reviewed to identify conservatism i determination of material property variations and the sensitiv the structural integrity analysis results to the material pro variations. The areas covered were fluence dependence o transition temperature, fracture toughness and the drop in u shelf, the arrest toughness, the thermal properties, and the growth rate. Major interest is in defining the fluence depen of irradiation embrittlement, in particular for those evalu that require extrapolation into the future with resulting higher fluence. Accurate methods for predicting changes in transition temperature and defining fracture toughness for irradiated material are needed. Indications are that knowledge of toughness in the low toughness region of the fracture toughness curve is more important than toughness at the upper shelf.

When the assessment of a vessel indicates small flaw sizes for initiation and consequent reliance on arrest to demonstrate adequacy, then the arrest toughness for irradiated material must be accurately known.

Finally, the thermal properties of irradiated reactor vessel material should be evaluated since an irradiation effect can affect evaluation of vessel integrity under accident conditions.

As long as cladding is not considered in crack growth evaluation, then the behavior of irradiated material in a water environment must be established, and it should be determined whether an in-pile effect on crack growth rate in a water environment exists. Also, if consideration is to be given for the clad, more crack growth data in vacuum or air should be obtained to eliminate the present conservatism.

Acknowledgments

The authors wish to express their appreciation to J. M. Kran and O. Meenwis for their contributions to the technical ba for this paper.

References

[1] Bamford, W. H. and Buchalet, C. B., "Method for Fracture Mechar Analysis of Nuclear Reactor Vessels Under Severe Thermal Tr siems," WCAP-8510, Westinghouse Commercial Atomic Po Division, Westinghouse Electric Co., Pittsburgh, Pa., July 1976.

[2] Bamford, W. H., Moon, D. M., and Scott, K. V., "Evaluation Critical Factors in Crack Growth Rate Studies of LWR Steel presented at the Fourth Water Reactor Safety Information Meeti Gaithersburg, Md., Oct. 1976.

[3] Shahinian, P., Watson, H. E., and Hawthorne, J. R., "Fati Crack Growth Resistance of Several Neutron Irradiated Press Vessel Steels and Welds," *Transactions of the American Society Mechanical Engineers*, Vol. 96, Series H, 1974, pp. 242–245

[4] James, L. A. and Williams, J. A., "The Effect of Temperature Neutron Irradiation upon the Fatigue Crack Propagation Behavio ASTM A533B, Class 1 Steel," HEDL-TME-72-132, Hanf Engineering Development Laboratory, Atomic Energy Commissi Washington, D.C., Sept. 1972.

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OG-94-102

December 20, 1994

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