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FINAL REPORT

ON

REACTOR VESSEL

PRESSURE-TEMPERATURE LIMITS

FOR

CALVERT CLIFFS UNIT 1

FOR

12 EFFECTIVE FULL POWER YEARS

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1.0 INTRODUCTION

The following sections describe the basis for development of reactor vessel beltline pressure-temperature limitations for the Calvert Cliffs Unit 1 Nuclear Generating Station. These limits are calculated to meet the regulations of 10 CFR Part 50 Appendix A,(1) Design Criterion 14 and Design Criterion 31. These design criteria required that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

The pressure-temperature limits are developed using the requirements of 10 CFR 50 Appendix G(2). This appendix describes the requirements for developing the pressure-temperature limits and provides the general basis for these limitations. The margins of safety against fracture provided by the pressure-temperature limits using the requirements of 10 CFR Part 50 Appendix G are equivalent to those recommended in the ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure."(3) The general guidance provided in those procedures has been utilized to develop the Calvert Cliffs Unit 1 pressure-temperature limits with the requisite margins of safety for the heatup and cooldown conditions.

The Reactor Pressure Vessel beltline pressure-temperature limits are based upon the irradiation damage prediction methods of Regulatory Guide 1.99 Revision 02(4). This methodology has been used to calculate the limiting material Adjusted Reference Temperatures for Calvert Cliffs Unit 1 and have utilized fluence values for 12 Effective Full Power Years (EFPY).

This report provides reactor vessel beltline pressure-temperature limits in accordance with 10 CFR 50 Appendix G for 12 EFPY. The events analyzed are the isothermal, 10, 20, 30, 40, 50, 75 and 100°F/hr cooldown conditions and the 10, 20, 30, 40, 50, 60, 70 and 75°F/hr heatup conditions. These conditions were analyzed to provide a data base of reactor vessel P-T limits for use in establishing Low Temperature Overpressure Protection requirements.

Low Temperature Overpressure Protection (LTOP) enable temperatures are calculated based upon the guidance provided in USNRC Standard Review Plan (SRP) 5.2.2.[5] Using this guidance the temperatures at which the LTOP system must be aligned to the RCS under heatup and cooldown conditions are established for automatic protection of the Appendix G P-T limits.

2.0 ADJUSTED REFERENCE TEMPERATURE PROJECTIONS

In order to develop pressure-temperature limits over the design life of the reactor vessel, adjusted reference temperatures (ART) for the controlling beltline material need to be determined. The adjusted reference temperatures of reactor vessel beltline materials for Calvert Cliffs Unit 1 have been calculated at the 1/4t and 3/4t locations after 12 EFPY of operation. By comparing ART data for each material, the controlling material for Calvert Cliffs Unit 1 has been determined.

The adjusted reference temperatures (ART) have been calculated using the procedures in Regulatory Position 1.1 of Regulatory Guide 1.99 Revision 02[4]. The calculative procedure for the ART values for each material in the beltline is given by the following expression:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material. ΔRT_{NDT} is the mean value of the adjustment in the reference temperature caused by irradiation and is given by the following expression:

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

CF is the chemistry factor for the beltline materials which is a function of residual element content, i.e., weight percent copper and nickel. Regulatory Guide 1.99 Revision 02 provides values for the chemistry factors for welds and for base metal plates and forgings. The term f is the neutron fluence at any depth in the vessel. The neutron fluence at any depth is given by the following expression:

$$f = f_{surf}\ (e^{-0.24x})$$

The term f_{surf} is the calculated value of the neutron fluence ($10^{19} n/cm^2$, E> 1MeV) at the inner wetted surface of the vessel at the location of the postulated defect (1/4t or 3/4t), and x is the depth into the vessel wall from the inner wetted surface in inches.

Margin is the quantity that is added to obtain a conservative upper bound value of ART. The margin term is given by the following expression:

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

The terms σ_I and σ_Δ represent the standard deviation for initial RT_{NDT} and the standard deviation of the mean value for the reference temperature shift.

The following information provides the basis for the calculated ART values for Calvert Cliffs Unit 1:

1. Material data were obtained from Reference 6, including copper content, nickel content and initial reference temperature (RT_{NDT}). These data are summarized in Table 1 for Calvert Cliffs Unit 1.

2. Peak neutron fluence for the Calvert Cliffs Unit 1 beltline region was determined to be 1.93×10^{19} n/cm^2 (E>1 MeV) at 12 EFPY (Reference 7).

3. Shell course minimum reference thickness is 8.625 in. for both the lower and intermediate shells (Reference 8).

4. Calculations were based on the procedures in Regulatory Position 1.1 of NRC Regulatory Guide 1.99, Rev. 2 (Reference 4). Uncertainty in initial RT_{NDT} was taken as 0°F for measured values and 17°F for welds without measured values (Reference 9).

Adjusted reference temperatures for all beltline materials at the 1/4t and 3/4t locations after 12 EFPY were calculated using Regulatory Guide 1.99 Revision 02 and the results of the calculation are listed in Table 2 for Calvert Cliffs Unit 1. The controlling materials are shown in Table 2; the term "controlling" means having the highest ART for a given time and position within the vessel wall. The highest, or limiting, ARTs are then used to develop the pressure-temperature limits for the corresponding time period.

In the case of Calvert Cliffs Unit 1, the intermediate shell longitudinal welds are controlling at the 1/4t and 3/4t locations after 12 EFPY based on the predicted ART values of 222°F and 162.5°F respectively.

According to Position 1.1 of Regulatory Guide 1.99, Revision 2[4], the uncertainty in the value of initial RT_{NDT} is to be estimated from the precision of test method when a "measured" value of initial

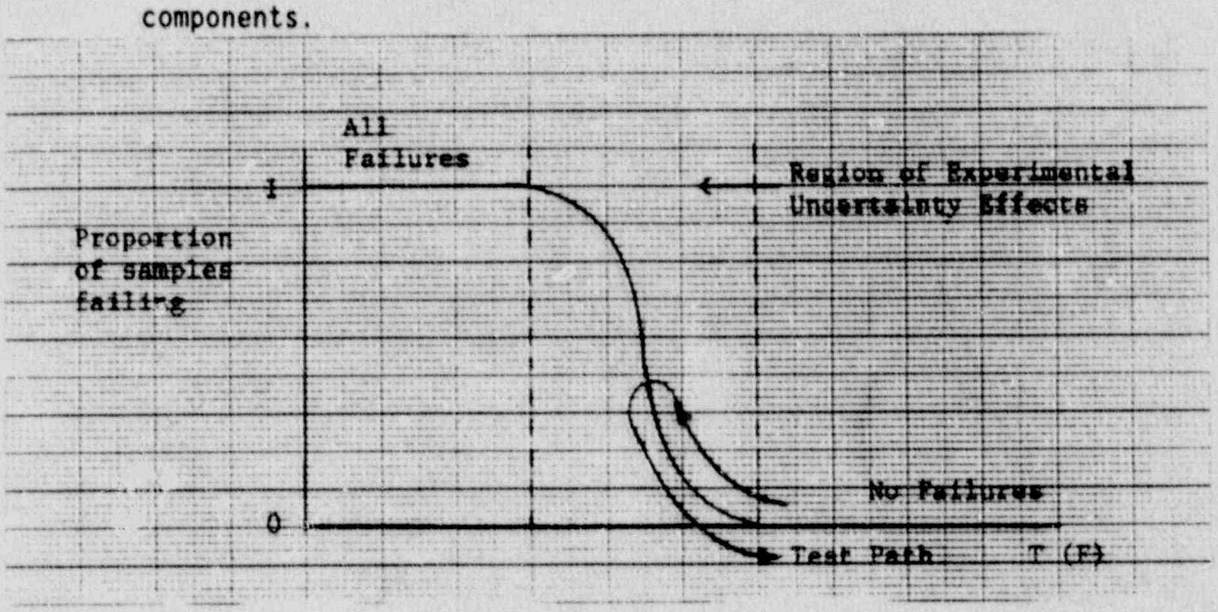
RT_{NDT} is available. RT_{NDT} is derived in accordance with NB2300 of the ASME Boiler and Pressure Vessel Code, Section III. It involves both a series of drop weight (ASTM E208) and Charpy impact (ASTM E23) tests on the material. The RT_{NDT} resulting from this two test method evaluation is conservatively biased. The elements of this conservatism include:

1) Choice for RT_{NDT} is the higher of NDT or T_{CV} -60°F. The drop-weight test is performed to obtain NDT and a full Charpy impact curve is developed to obtain T_{CV} for a given material. The combination of the two test methods gives protection against the possibility of errors in conducting either test and, with the full Charpy curve, demonstrates that the material is typical of reactor pressure vessel steel. Choice of the more conservative of the two (i.e., the higher of NDTT or T_{CV} - 60°F) assures that tests at temperatures above the reference temperature will yield increasing values of toughness, and verifies the temperature dependence of the fracture toughness implicit in the K_{IR} curve (ASME Code, Section III, Appendix G).

2) Selection of the most adverse Charpy results for T_{CV}. In accordance with NB2300, a temperature, T_{CV}, is established at which three Charpy specimens exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. The three specimens will typically exhibit a range of lateral expansion and absorbed energy consistent with the variables inherent in the test: Specimen temperature, testing equipment, operator, and test specimen (e.g., dimensional tolerance and material homogeneity). All of these variables are controlled using process and procedural controls, calibration and operator training, and they are conservatively bounded by using the lowest measurement of the three specimens. Furthermore, two related criteria are used, lateral expansion and absorbed energy, where consistency between the two measurements provides

further assurance that they are realistic and the material will exhibit the intended strength, ductility and toughness implicit in the K_{IR} curve.

- Inherent conservatism in the protocol used in performing the drop-weight test. The drop-weight test procedure was carefully designed to assure attainment of explicit values of deflection and stress concentration, eliminating a specific need to account for below nominal test conditions and thereby guaranteeing a conservative direction of these uncertainty components. In addition, the test protocol calls for decreasing temperature until the <u>first</u> failure is encountered, followed by increasing the test temperature 10°F above the point where the <u>last</u> failure is encountered. This in fact assures that one has biased the resulting estimate toward a low failure probability region of the temperature versus failure rate function diagrammed below. The effect of this protocol is to conservatively accommodate the integrated uncertainty components.



Given the three elements of conservatism described above, values of initial RT_{NDT} obtained in accordance with NB2300 will result in a conservative measure of the reference temperature. The conservative bias of the NB2300 methodology and the drop-weight test protocol essentially eliminate the uncertainty which might result from the precision of an individual drop-weight or Charpy impact test. Therefore, when measured values of RT_{NDT} are available, the estimate of uncertainty in initial RT_{NDT} is taken as zero.

3.0 PRESSURE - TEMPERATURE LIMITS

3.1 GENERAL APPROACH FOR CALCULATING PRESSURE-TEMPERATURE LIMITS

The analytical procedure for developing reactor vessel pressure-temperature limits utilizes the methods of Linear Elastic Fracture Mechanics (LEFM) found in the ASME Boiler and Pressure Vessel Code Section III, Appendix G (Reference 3) in accordance with the requirements of 10 CFR Part 50 Appendix G (Reference 2). For these analyses, the Mode I (opening mode) stress intensity factors are used for the solution basis. The general method utilizes Linear Elastic Fracture Mechanics procedures. Linear Elastic Fracture Mechanics relates the size of a flaw with the allowable loading which precludes crack initiation. This relation is based upon a mathematical stress analysis of the beltline material fracture toughness properties as prescribed in Appendix G to Section III of the ASME Code.

The reactor vessel beltline region is analyzed assuming a semi-elliptical surface flaw oriented in the axial direction with a depth of one quarter of the reactor vessel beltline thickness and an aspect ratio of one to six. This postulated flaw is analyzed at both the inside diameter location (referred to as the 1/4t location) and the outside diameter location (referred to as the 3/4t location) to assure the most limiting condition is achieved. The above flaw geometry and orientation is the maximum postulated defect size (reference flaw) described in Appendix G to Section III of the ASME Code.

At each of the postulated flaw locations the Mode I stress intensity factor, K_I, produced by each of the specified loadings is calculated and the summation of the K_I values is compared to a reference stress intensity, K_{IR}, which is the critical value of K_I for the material and temperature involved. The result of this method is a relation of pressure versus temperature for each reactor vessel operating period which precludes brittle fracture. K_{IR} is obtained from a reference fracture toughness curve for low alloy reactor pressure vessel steels as defined in Appendix G to Section III of the ASME Code. This governing curve is defined by the following expression:

$$K_{IR} = 26.78 + 1.223\, e^{[.0145(T-ART + 160)]}$$

where,

K_{IR} = reference stress intensity factor, Ksi $\sqrt{in}$

T = temperature at the postulated crack tip, °F

ART = adjusted reference nil ductility temperature at the postulated crack tip, °F

For any instant during the postulated heatup or cooldown, K_{IR} is calculated by the metal temperature at the tip of the flaw, and by the value of adjusted reference temperature at that flaw location. Also for any instant during the heatup or cooldown the temperature gradients across the reactor vessel wall are calculated (see Section 2.3) and the corresponding thermal stress intensity factor, K_{IT}, is determined. Through the use of superposition, the thermal stress intensity is subtracted from the available K_{IR} to determine the allowable pressure stress intensity factor and consequently the allowable pressure.

In accordance with the ASME Code Section III Appendix G requirements, the general equations for determining the allowable pressure for any assumed rate of temperature change during Service Level A and B operation are:

$$2K_{IM} + K_{IT} < K_{IR}$$

$$1.5K_{IM} + K_{IT} < K_{IR} \text{ (Inservice Hydrostatic Test)}$$

where,

K_{IM} = Allowable pressure stress intensity factor, Ksi$\sqrt{in}$

K_{IT} = Thermal stress intensity factor, Ksi$\sqrt{in}$

K_{IR} = Reference stress intensity, Ksi$\sqrt{in}$

The pressure-temperature limits provided in this report account for the temperature differential between the reactor vessel base metal and the reactor coolant bulk fluid temperature. Correction for elevation and RCS flow induced pressure differences between the reactor vessel beltline and pressurizer, are included in the development of the pressure-temperature limits. Consequently, the P-T limits are provided on coordinates of pressurizer pressure versus indicated RCS temperature.

The pressure correction factors are based upon the differential pressure due to the elevation difference between the reactor vessel beltline wall and the pressurizer. This term of the pressure correction factor is equal to 15.0 psi. The pressure correction factors are also based upon flow induced pressure drops across the reactor core through the hot leg pipe up to the surge line nozzle. This term of the pressure correction factor has two values which are dependent upon the Reactor Coolant Pump (RCP) combination utilized during operation. During cooldown at temperatures of T_c > 150°F, the flow induced pressure drop is based upon the RCS flow rates resulting from four operating RCPs and is equal to 52.0 psi. During cooldown at temperatures of $T_c \leq 150°F$, the flow induced pressure drop is zero based upon no operating RCPs. During heatup for all RCS temperatures, the flow induced pressure drop is assumed to be that associated with operation of four reactor coolant pumps. Instrument uncertainties have been included in the pressure-temperature limits. Consequently, two pressure correction factors are utilized in correcting the reactor vessel beltline region pressure to pressurizer pressure depending upon the cold leg temperature. The uncertainty associated with the pressure indication instrument loop is 48 psi and the uncertainty associated with the temperature indication instrument loop is 10°F. The following pressure correction factors have been utilized:

COOLDOWN		HEATUP	
T_c (°F)	PRESSURE CORRECTION FACTOR (PSI)	T_c (°F)	PRESSURE CORRECTION FACTOR (PSI)
> 150°F	100 psi	All RCS Temp.	100 psi
≤ 150°F	63 psi		

By explicitly accounting for the temperature differential between the flaw tip base metal temperature and the reactor coolant bulk fluid temperature, and the pressure differential between the beltline region of the reactor vessel and the pressurizer, the P-T limits are correctly represented on coordinates of pressurizer pressure and cold leg temperature.

3.2 THERMAL ANALYSIS METHODOLOGY

The Mode I thermal stress intensity factor is obtained through a detailed thermal analysis of the reactor vessel beltline wall using a computer code. In this code a one dimensional three noded isoparametric finite element is used for performing the radial conduction-convection heat transfer analysis. The vessel wall is divided into 10 elements and an accurate distribution of temperature as a function of radial location and transient time is calculated. The code utilizes convective boundary condition on the inside wall of the vessel and an insulative boundary on the outside wall of the vessel. Variation of material properties through the vessel wall are permitted allowing for the change in material thermal properties between the cladding and the base metal.

In general, the temperature distribution through the reactor vessel wall is governed by a partial differential equation,

$$pC \frac{\partial T}{\partial t} = K \quad \frac{\partial^2 T}{\partial r^2} \quad + \quad \frac{1}{r} \frac{\partial T}{\partial r}$$

subject to the following boundary conditions at the inside and outside wall surface locations:

$$\text{At } r = r_i \qquad -K \frac{\partial T}{\partial r} = h \; (T - T_c)$$

$$\text{At } r = r_o \qquad \frac{\partial T}{\partial r} = 0$$

where,

p	=	density, lb/ft^3
C	=	specific heat, btu/lb-°F
K	=	thermal conductivity, btu/hr-ft-°F
T	=	vessel wall temperature, °F
r	=	radius, ft
t	=	time, hr
h	=	convective heat transfer coefficient, btu/hr-ft^2-°F
T_c	=	RCS coolant temperature, °F
r_i, r_o	=	inside and outside radii of vessel wall, ft

The above is solved numerically using a finite element model to determine wall temperature as a function of radius, time, and thermal rate. Thermal stress intensity factors are determined by the calculated temperature difference through the beltline wall using thermal influence coefficients specifically generated for this purpose. The influence coefficients depend upon geometrical parameters associated with the maximum postulated defect, and the geometry of the reactor vessel beltline region (i.e., r_o/r_i, a/c, a/t), along with the assumed unit loading.

The thermal stress intensity factors are determined by the temperature difference and temperature profile through the beltline wall using thermal influence coefficients and superposition. ASME code Section III Appendix G recognizes the limitations of the method it provides for calculating K_{IT} because of the assumed temperature profile. Since a detailed heat transfer analysis results in varying temperature profiles (and consequently varying thermal stresses), an alternate method for calculating K_{IT} was employed as required by Article G-2214.3 of Reference 3. The alternate method employed used a polynomial fit of the temperature profile and superposition using influence coefficients to calculate K_{IT}. The influence coefficients were calculated using a 2-dimensional finite element model of the reactor vessel. The influence coefficients were corrected for 3 dimensional effects using ASTM Special Technical Publication 677 (Reference 10).

3.3 COOLDOWN LIMIT ANALYSIS

During cooldown, membrane and thermal bending stresses act together in tension at the reactor vessel inside wall. This results in the pressure stress intensity factor, K_{IM}, and the thermal stress intensity factor, K_{IT}, acting in unison creating a high stress intensity. At the reactor vessel outside wall the tensile pressure stress and the compressive thermal stress act in opposition resulting in a lower total stress than at the inside wall location.

Also neutron embrittlement, the shift in RT_{NDT} and the associated reduction in fracture toughness are less severe at the outside wall compared to the inside wall location. Consequently, the inside flaw location is more limiting and is analyzed for the cooldown event.

Utilizing the material metal temperature and adjusted reference temperature at the 1/4t location, the reference stress intensity is determined. From the method provided in Section 2.3, the through wall temperature gradient is calculated for the assumed cooldown rate to determine the thermal stress intensity factor. In general, the thermal stress intensity factors are found using the temperature difference through the wall as a function of transient time as described in Section 2.3. They are then subtracted from the available K_{IR} value to find the allowable pressure stress intensity factor and consequently the allowable pressure.

The cooldown pressure-temperature curves are thus generated by calculating the allowable pressure on the reference flaw at the 1/4t location based upon

$$K_{IM} = \frac{K_{IR} - K_{IT}}{2}$$

where,

K_{IM} = Allowable pressure stress intensity as a function of coolant temperature, $Ksi\sqrt{in}$

K_{IR} = Reference stress intensity as a function of coolant temperature, $Ksi\sqrt{in}$

K_{IT} = Thermal stress intensity as a function of coolant temperature, $Ksi\sqrt{in}$

To develop a composite pressure-temperature limit for the cooldown event, the isothermal pressure-temperature limit must be calculated. The isothermal pressure-temperature limit is then compared to the pressure-temperature limit associated with a cooling rate and the more restrictive allowable pressure-temperature limit is chosen resulting in a composite limit curve for the reactor vessel beltline.

Table 3 provides the results for the isothermal, 10, 20, 30, 40, 50, 75 and 100°F/hr cooldown pressure-temperature limits. These tables provide the allowable pressure versus reactor coolant temperature for the various cooldown conditions. The allowable pressure is in units of Ksi while the temperature is in units of °F. Figures 3, 4, 8 and 9 provide a graphical presentation of the cooldown pressure-temperature limits found in Table 3. It is permissible to linearly interpolate between the cooldown pressure-temperature limits.

3.4 HEATUP LIMIT ANALYSIS

During a heatup transient, the thermal bending stress is compressive at the reactor vessel inside wall and is tensile at the reactor vessel outside wall. Internal pressure creates a tensile stress at the inside wall as well as the outside wall locations. Consequently, the outside wall location has the larger total stress when compared to the inside wall. However, neutron embrittlement (the shift in material RT_{NDT} and the associated reduction in fracture toughness) is greater at the inside location than the outside. Therefore, both the inside and outside flaw locations must be analyzed to assure that the most limiting condition is achieved.

As described in the cooldown case, the reference stress intensity factor is calculated by the metal temperature at the tip of the flaw and by the adjusted reference temperature at the flaw location. For heatup the reference stress intensity is calculated for both the 1/4t and 3/4t locations. Using the finite element method described in Section 2.3, the temperature profile through the wall and the metal temperatures at the tip of the flaw are calculated for the transient history. This information is used to calculate the thermal stress intensity factor at the 1/4t and 3/4t locations using the calculated wall gradient and thermal influence coefficients. The allowable pressure stress intensity is then determined by superposition of the thermal stress intensity factor with the available reference stress intensity at the flaw tip. The allowable pressure is then derived from the calculated allowable pressure stress intensity factor.

It is interesting to note that a sign change occurs in the thermal stress through the reactor vessel beltline wall. Assuming a reference flaw at the 1/4t location the thermal stress tends to alleviate the pressure stress indicating the isothermal steady state condition would represent the limiting P-T limit. However, the isothermal condition may not always provide the limiting pressure-temperature limit for the 1/4t location during a heatup transient. This is due to the correction of the base metal temperature to the Reactor Coolant System (RCS) fluid temperature at the inside wall by accounting for clad and film temperature differentials.

For a given heatup rate (non-isothermal), the differential temperature through the clad and film increases as a function of thermal rate resulting in a higher RCS fluid temperature at the inside wall than the isothermal condition for the same flaw tip temperature and pressure. Therefore to ensure the accurate representation of the 1/4t pressure-temperature limit during heatup, both the isothermal and heatup rate dependent pressure-temperature limits are calculated to ensure the limiting condition was achieved. These limits account for clad and film differential temperatures and for the gradual buildup of wall differential temperatures with time, as do the cooldown limits.

At the 3/4t location the pressure stress and thermal stresses are tensile resulting in the maximum stress at that location. Pressure-temperature limits were calculated for the 3/4t location accounting for clad and film differential temperature and the buildup of wall temperature gradients with time using the method described in Section 2.3. The allowable pressure was derived based upon a flaw at the 3/4t location by superposition of the thermal stress intensity with the available reference stress intensity for the metal temperature and adjusted reference temperature at that position.

To develop composite pressure-temperature limits for the heatup transient, the isothermal, 1/4t heatup, and 3/4t heatup pressure-temperature limits are compared for a given thermal rate. Then the most restrictive pressure-temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for the heatup event.

Table 3 provides the results for the 10, 20, 30, 40, 50, 60, 70 and 75°F/hr heatup pressure-temperature limits. These tables provide the allowable pressure versus reactor coolant temperature for the various heatup conditions. The allowable pressure is in units of ksi while the temperature is in units of °F. Figures 1, 2, 6 and 7, provide a graphical presentation of the heatup pressure-temperature limits found in Table 3. It is permissible to linearly interpolate between the heatup pressure-temperature limits.

3.5 HYDROSTATIC TEST AND CORE CRITICAL LIMIT ANALYSIS

Both 10 CFR Part 50 Appendix G and the ASME Code Appendix G require the development of pressure-temperature limits which are applicable to inservice hydrostatic tests. For hydrostatic tests performed subsequent to loading fuel into the reactor vessel, the minimum test temperature is determined by evaluating K_I, the mode I stress intensity factors. The evaluation of K_I is performed in the same manner as that for normal operation heatup and cooldown conditions except the factor of safety applied to the pressure stress intensity factor is 1.5 versus 2.0. From this evaluation, a pressure-temperature limit which is applicable to inservice hydrostatic tests is established. The minimum temperature for the inservice hydrostatic test pressure can be determined by entering the curve at the test pressure (1.1 times normal operating pressure) and locating the corresponding temperature. The inservice hydrostatic test limit is provided for 12 EFPY in Table 4 and is shown in Figure 5.

Appendix G to 10 CFR Part 50, specifies pressure-temperature limits for core critical operation to provide additional margin during actual power operation. The pressure-temperature limit for core critical operation is based upon two criteria. These criteria are that the reactor vessel must be at a temperature equal to or greater than the minimum temperature required for the inservice hydrostatic test, and be at least 40°F higher than the minimum pressure-temperature curve for normal operation heatup or cooldown.

Note, that the core critical limits established above are solely based upon fracture mechanics considerations, and do not consider core reactivity safety analyses which can control the temperature at which the core can be brought critical.

3.6 LOWEST SERVICE TEMPERATURE, MINIMUM BOLTUP TEMPERATURE, AND MINIMUM PRESSURE LIMITS

In addition to the computation of the reactor vessel beltline P-T limits, additional limits have been provided for reference. These additional limits are the Lowest Service Temperature, Minimum Boltup Temperature, and Minimum Pressure Limits. These limits are described below.

The Lowest Service Temperature is the minimum allowable temperature at pressures above 20% of the pre-operational system hydrostatic test pressure (625 psia). This temperature is defined as equal to the most limiting RT_{NDT} for the balance of Reactor Coolant System (RCS) components plus 100°F, per Article NB 2332 of Section III of the ASME Boiler and Pressure Vessel Code.

The maximum RT_{NDT} for the balance of the RCS components is estimated as 50°F. Therefore, the Lowest Service Temperature is equal to 100°F + 50°F + 10°F = 160°F.

The minimum pressure limit is the break point between the minimum boltup temperature and the Lowest Service Temperature. Defined by the ASME Boiler and Pressure Vessel Code as 20% of the pre-operational hydrostatic test pressure, the minimum pressure is as follows when pressure correction factors for elevation and flow are taken into account:

Cooldown		Heatup	
562 psia	$T_c \leq 150°F$	525 psia	All RCS Temperatures
525 psia	$T_c > 150°F$		

The minimum boltup temperature is the minimum allowable temperature at pressures below the 20% of the pre-operational system hydrostatic test pressure. The minimum is defined as the initial RT_{NDT} for the material of the higher stressed region of the reactor vessel plus any effects for irradiation per Article G-2222 of Section III of the ASME Boiler and Pressure Vessel Code. The initial reference temperature of the reactor vessel and closure head flanges was determined using the certified material test reports and Branch Technical Position MTEB 5-2. The maximum initial RT_{NDT} associated with the stressed region of the closure head flange is -10°F. The minimum boltup temperature including temperature instrument uncertainty is -10 + 10°F = 0°F. However, for conservatism a minimum boltup temperature of 70°F is utilized.

4.0 LTOP ENABLE TEMPERATURES

Standard Review Plan 5.2.2, Overpressure Protection[5], has defined the temperature at which the Low Temperature Overpressure Protection (LTOP) system should be operable during startup and shutdown conditions. This temperature known as the LTOP enable temperature is defined 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 calculations. Below the LTOP enable temperature the LTOP system must be aligned to the RCS to prevent exceeding the applicable technical specification and Appendix G limits in the event of a transient.

Based upon this definition and upon the results of the Appendix G calculations, LTOP enable temperatures for cooldown and heatup conditions have been calculated for Calvert Cliffs Unit 1. In addition, LTOP enable temperatures have been calculated using specific heatup transients with changing thermal rates. The LTOP enable temperatures associated with the heatup transients which utilize changing thermal rates credit soak times for reducing thermal stress and meet the criteria for the LTOP enable temperatures as defined in Standard Review Plan 5.2.2.

The LTOP enable temperature for a cooldown is based upon the isothermal P-T limit. Consequently the LTOP enable temperature is equal to the 1/4t adjusted reference temperature + 90°F. Therefore for cooldown the LTOP enable temperatures equal 322°F for 12 EFPY when the temperature instrumentation uncertainty of 10°F is included.

The LTOP enable temperatures for heatup are provided in Table 5.

5.0 DATA

Reactor Vessel Data			Reference
Design Pressure	=	2500 psia	8
Design Temperature	=	650°F	8
Operating Pressure	=	2250 psia	8
Beltline Thickness	=	8.625 in	8
Inside Radius	=	86.9575 in	8
Cladding Thickness	=	.3125 in	8

Material-SA 533 Grade B Class 1			Reference
Thermal Conductivity	=	23.8 BTU/hr-ft-°F	11
Youngs Modulus	=	28×10^6 psi	11
Coefficient of Thermal Expansion	=	7.77×10^{-6} in/in/°F	11
Specific Heat	=	.12 BTU/lb-°F	
Density	=	.283 lb/in^3	

Stainless Steel Cladding

Thermal Conductivity = 10 BTU/hr-ft-°F

Adjusted Reference Temperature Values 12 EFPY		Reference
1/4t	222°F	
3/4t	162.5°F	
Fast Neutron Fluence = 1.93×10^{19} n/cm^2		7

Film coefficient on inside surface = 1000 BTU/hr-ft^2-°F

Pressure Correction Factors For Elevation and Flow (Reference 7)

	Cooldown		Heatup
RCS temperature $\leq$ 150°F	dp = 63 psia	All RCS	dp = 100 psia
RCS temperature > 150°F	dp = 100 psia	Temperatures	

Temperature Instrument Correction (Reference 7)

dt = 10°F

6.0 REFERENCES

1. Code of Federal Regulations, 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants", January 1988.

2. Code of Federal Regulations, 10 CFR Part 50, Appendix G "Fracture Toughness Requirements", January 1988.

3. ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure", 1986 Edition.

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

5. U. S. Nuclear Regulatory Commission Standard Review Plan (SRP) 5.2.2, Overpressure Protection, Revision 2, November 1988.

6. Reactor Vessel Weld Materials for Calvert Cliffs Unit 1 Supplemental Surveillance Program, C-E Report 02987-MCC-002 Revision 0, November 1989.

7. T. L. Cook to P. J. Hijeck "Transmittal of Data for PT Curve Calculation", BG&E letter NEV-90-256, April 4, 1990.

8. Instruction Manual, Reactor Vessel Assembly, Calvert Cliffs Station, Baltimore Gas & Electric Co., C-E Book No. 72167/73167 June 1972.

9. Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCA with Loss of Feedwater for the C-E NSSS, CEN-189, December 1981.

10. "Semi-Elliptical Cracks in a Cylinder Subjected to Stress Gradients", J. Heliot, R. C. Labbens, and Pellisser-Tanon, ASTM Special Technical Publication 677, August 1979.

11. ASME Boiler and Pressure Vessel Code Section III, Appendix I, "Design Stress Intensity Values, Allowable Stresses, Material Properties, and Design Fatigue Curves", 1986 Edition.

Table 1

CALVERT CLIFFS UNIT 1
REACTOR VESSEL BELTLINE MATERIALS

Component	Identification Number	Cu(%)	Ni(%)	RT_{NDT}
Intermediate Shell Long. Welds	2-203 A/C	.21	.87	-50°F
Lower Shell Long. Welds	3-203 A/C	.21	.69	-56°F[1]
Inter./Lower Shell Girth Weld	9-203	.23	.23	-80°F
Intermediate Shell Plate	D-7206-1	.11	.55	20°F
	D-7206-2	.12	.64	-30°F
	D-7206-3	.12	.64	10°F
Lower Shell Plate	D-7207-1	.13	.54	10°F
	D-7207-2	.11	.56	-10°F
	D-7207-3	.11	.53	-20°F

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(1) Generic value for submerged arc welds

Table 2

CALVERT CLIFFS UNIT 1 BELTLINE
ART CALCULATION

Identification Number	Chemistry Factor	RT_{NDT}, °F	i		RT_{NDT}, °F	ART, °F	RT_{NDT}, °F	ART, °F
2-203 A/C	208.2	-50	0	28	216	222	156.5	162.5
3-203 A/C	178.8	-56	17	28	186	195.5	134	143.5
9-203	120.4	-80	0	28	125	101	90.5	66.5
D-7206-1	73.5	20	0	17	76	130	55	109
D-7206-2	83.6	-30	0	17	87	91	63	67
D-7206-3	83.6	10	0	17	87	131	63	107
D-7207-1	89.2	10	0	17	93	137	67	111
D-7207-2	73.6	-10	0	17	76	100	55	79
D-7207-3	73.3	-20	0	17	76	90	55	69

TABLE 3

BG&E CALVERT CLIFFS UNIT 1
P-ALLOWABLE (KSI) VS. RCS TEMPERATURE (DEG. F)
FOR 12 EFPY, NORMAL OPERATION

RCS TEMP DEG F	HEATUP P-ALLOWABLE (KSI)									COOLDOWN P-ALLOWABLE (KSI)							
	ISO THERMAL	10 F/ HOUR	20 F/ HOUR	30 F/ HOUR	40 F/ HOUR	50 F/ HOUR	60 F/ HOUR	70 F/ HOUR	75 F/ HOUR	ISO THERMAL	10 F/ HOUR	20 F/ HOUR	30 F/ HOUR	40 F/ HOUR	50 F/ HOUR	75 F/ HOUR	100 F/ HOUR
60	0.4158	--	--	--	--	--	--	--	--	0.4528	0.4130	0.3731	0.3334	0.2938	0.2543	0.1562	0.0592
70	0.4188	--	--	--	--	--	--	--	--	0.4558	0.4161	0.3765	0.3371	0.2977	0.2585	0.1611	0.0651
80	0.4223	--	--	--	--	--	--	--	--	0.4593	0.4198	0.3805	0.3413	0.3022	0.2633	0.1668	0.0718
90	0.4262	0.4262	0.4262	0.4262	0.4262	0.4262	0.4262	0.4262	0.4262	0.4632	0.4241	0.3850	0.3461	0.3074	0.2689	0.1734	0.0797
100	0.4308	0.4308	0.4308	0.4308	0.4308	0.4308	0.4308	0.4308	0.4308	0.4678	0.4290	0.3903	0.3517	0.3134	0.2753	0.1810	0.0887
110	0.4361	0.4361	0.4361	0.4361	0.4265	0.4197	0.4155	0.4135	0.4130	0.4731	0.4346	0.3963	0.3582	0.3203	0.2827	0.1897	0.0990
120	0.4423	0.4423	0.4423	0.4379	0.4191	0.4068	0.3982	0.3926	0.3906	0.4793	0.4412	0.4033	0.3657	0.3284	0.2913	0.1999	0.1112
130	0.4494	0.4494	0.4494	0.4432	0.4183	0.4004	0.3873	0.3779	0.3739	0.4864	0.4488	0.4115	0.3744	0.3376	0.3012	0.2116	0.1252
140	0.4576	0.4576	0.4576	0.4532	0.4227	0.3998	0.3820	0.3684	0.3630	0.4946	0.4576	0.4208	0.3844	0.3483	0.3126	0.2252	0.1413
150	0.4670	0.4670	0.4670	0.4670	0.4318	0.4038	0.3815	0.3638	0.3566	0.5040	0.4677	0.4317	0.3960	0.3607	0.3259	0.2409	0.1599
160	0.4780	0.4780	0.4780	0.4780	0.4449	0.4122	0.3855	0.3638	0.3546	0.4780	0.4424	0.4072	0.3724	0.3381	0.3042	0.2219	0.1442
170	0.4907	0.4907	0.4907	0.4907	0.4621	0.4248	0.3937	0.3679	0.3569	0.4907	0.4560	0.4217	0.3879	0.3546	0.3219	0.2430	0.1693
180	0.5053	0.5053	0.5053	0.5053	0.4833	0.4414	0.4059	0.3759	0.3629	0.5053	0.4716	0.4384	0.4058	0.3738	0.3424	0.2672	0.1982
190	0.5223	0.5223	0.5223	0.5223	0.5087	0.4624	0.4224	0.3881	0.3733	0.5223	0.4897	0.4578	0.4265	0.3959	0.3661	0.2952	0.2314
200	0.5418	0.5418	0.5418	0.5418	0.5389	0.4878	0.4432	0.4046	0.3876	0.5418	0.5107	0.4802	0.4504	0.4215	0.3935	0.3277	0.2696
210	0.5645	0.5645	0.5645	0.5645	0.5645	0.5181	0.4686	0.4254	0.4064	0.5645	0.5349	0.5060	0.4781	0.4510	0.4250	0.3648	0.3135
220	0.5906	0.5906	0.5906	0.5906	0.5906	0.5540	0.4992	0.4509	0.4298	0.5906	0.5628	0.5360	0.5101	0.4852	0.4617	0.4084	0.3650
230	0.6209	0.6209	0.6209	0.6209	0.6209	0.5958	0.5355	0.4821	0.4581	0.6209	0.5952	0.5705	0.5470	0.5248	0.5039	0.4583	0.4243
240	0.6559	0.6559	0.6559	0.6559	0.6559	0.6449	0.5782	0.5190	0.4925	0.6559	0.6326	0.6105	0.5898	0.5705	0.5528	0.5160	0.4924
250	0.6963	0.6963	0.6963	0.6963	0.6963	0.6963	0.6281	0.5623	0.5329	0.6963	0.6758	0.6567	0.6392	0.6233	0.6093	0.5830	0.5707
260	0.7430	0.7430	0.7430	0.7430	0.7430	0.7430	0.6863	0.6133	0.5806	0.7430	0.7257	0.7101	0.6963	0.6843	0.6744	0.6596	0.6607
270	0.7970	0.7970	0.7970	0.7970	0.7970	0.7970	0.7540	0.6730	0.6366	0.7970	0.7835	0.7719	0.7623	0.7549	0.7501	0.7493	0.7661
280	0.8595	0.8595	0.8595	0.8595	0.8595	0.8595	0.8326	0.7423	0.7012	0.8595	0.8503	0.8433	0.8386	0.8365	0.8373	0.8521	0.8595
290	0.9317	0.9317	0.9317	0.9317	0.9317	0.9317	0.9237	0.8226	0.7776	0.9317	0.9275	0.9258	0.9269	0.9309	0.9317	0.9317	0.9317
300	1.0152	1.0152	1.0152	1.0152	1.0152	1.0152	1.0152	0.9165	0.8655	1.0152	1.0152	1.0152	1.0152	1.0152	1.0152	1.0152	1.0152
310	1.1117	1.1066	1.1046	1.1054	1.1087	1.1117	1.1117	1.0250	0.9676	1.1117	1.1117	1.1117	1.1117	1.1117	1.1117	1.1117	1.1117
320	1.2232	1.2110	1.2022	1.1966	1.1941	1.1946	1.1972	1.1502	1.0864	1.2232	1.2232	1.2232	1.2232	1.2232	1.2232	1.2232	1.2232
330	1.3522	1.3316	1.3150	1.3022	1.2928	1.2867	1.2836	1.2836	1.2227	1.3522	1.3522	1.3522	1.3522	1.3522	1.3522	1.3522	1.3522
340	1.5012	1.4710	1.4454	1.4241	1.4069	1.3938	1.3835	1.3774	1.3753	1.5012	1.5012	1.5012	1.5012	1.5012	1.5012	1.5012	1.5012
350	1.6736	1.6322	1.5962	1.5651	1.5388	1.5171	1.4990	1.4854	1.4798	1.6736	1.6736	1.6736	1.6736	1.6736	1.6736	1.6736	1.6736
360	1.8728	1.8185	1.7705	1.7282	1.6913	1.6598	1.6324	1.6099	1.6008	1.8728	1.8728	1.8728	1.8728	1.8728	1.8728	1.8728	1.8728
370	2.1031	2.0340	1.9719	1.9166	1.8675	1.8249	1.7866	1.7548	1.7409	2.1031	2.1031	2.1031	2.1031	2.1031	2.1031	2.1031	2.1031
380	2.3694	2.2830	2.2049	2.1345	2.0713	2.0149	1.9649	1.9218	1.9013	2.3694	2.3694	2.3694	2.3694	2.3694	2.3694	2.3694	2.3694
390	2.6772	2.5709	2.4741	2.3863	2.3069	2.2360	2.1710	2.1139	2.0891	2.6772	2.6772	2.6772	2.6772	2.6772	2.6772	2.6772	2.6772
400	2.9000	2.9000	2.7854	2.6775	2.5792	2.4905	2.4093	2.3371	2.3044	2.9000	2.9000	2.9000	2.9000	2.9000	2.9000	2.9000	2.9000
410	--	--	2.9000	2.9000	2.8940	2.7850	2.6847	2.5954	2.5537	--	--	--	--	--	--	--	--
420	--	--	--	--	2.9000	2.9000	2.9000	2.8927	2.8427	--	--	--	--	--	--	--	--
430	--	--	--	--	--	--	--	2.9000	2.9000	--	--	--	--	--	--	--	--

TABLE 4

BG&E CALVERT CLIFFS UNIT 1
P-ALLOWABLE (KSI) VS. RCS TEMP (DEG. F)
FOR 12 EFPY, HYDROSTATIC OPERATION

RCS TEMP DEG F	HYDROSTATIC (KSI)
60	0.5878
70	0.5918
80	0.5963
90	0.6016
100	0.6078
110	0.6148
120	0.6230
130	0.6325
140	0.6434
150	0.6561
160	0.6707
170	0.6876
180	0.7071
190	0.7297
200	0.7558
210	0.7860
220	0.8209
230	0.8612
240	0.9078
250	0.9617
260	1.0240
270	1.0961
280	1.1793
290	1.2756
300	1.3869
310	1.5155
320	1.6643
330	1.8362
340	2.0350
350	2.2648
360	2.5304
370	2.8375

Table 5

CALVERT CLIFFS 1
LTOP ENABLE TEMPERATURES
(INCLUDING TEMPERATURE INSTRUMENT UNCERTAINTY)
12 EFPY

Cooldown LTOP Enable Temperature equals 322°F for all rates.

Heatup Rate	Limiting Location	Enable Temp 1/4t	Limiting Location	Enable Temp 3/4t
10°F/hr	1/4t	327°F	1/4t	270°F
20°F/hr	1/4t	330°F	1/4t	278°F
30°F/hr	1/4t	336°F	1/4t	286°F
40°F/hr	1/4t	340°F	1/4t	294°F
50°F/hr	1/4t	345°F	1/4t	302°F
60°F/hr	1/4t	350°F	1/4t	310°F
70°F/hr	1/4t	354°F	1/4t	318°F
75°F/hr	1/4t	356°F	1/4t	321°F

Heatup Transients with Changing Rates

Case 1

Rate	Temperature Range	LTOP Enable Temp.
60°F/hr*	70°F- 305°F	
10°F/hr	305°F- 327°F	327°F
60°F/hr*	327°F- 550°F	

Case 2

Rate	Temperature Range	LTOP Enable Temp.
60°F/hr*	70°F- 260°F	
20°F/hr	260°F- 330°F	330°F
60°F/hr*	330°F- 550°F	

* This rate is chosen as the maximum heatup rate based upon the minimum allowable pressure associated with the 60°F/hr P-T limit (429.5 psia) being greater than the peak transient pressure for the LTOP event (422.7 psia provided by BG&E) as measured at the pressurizer.

FIGURE 1

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
HEATUP

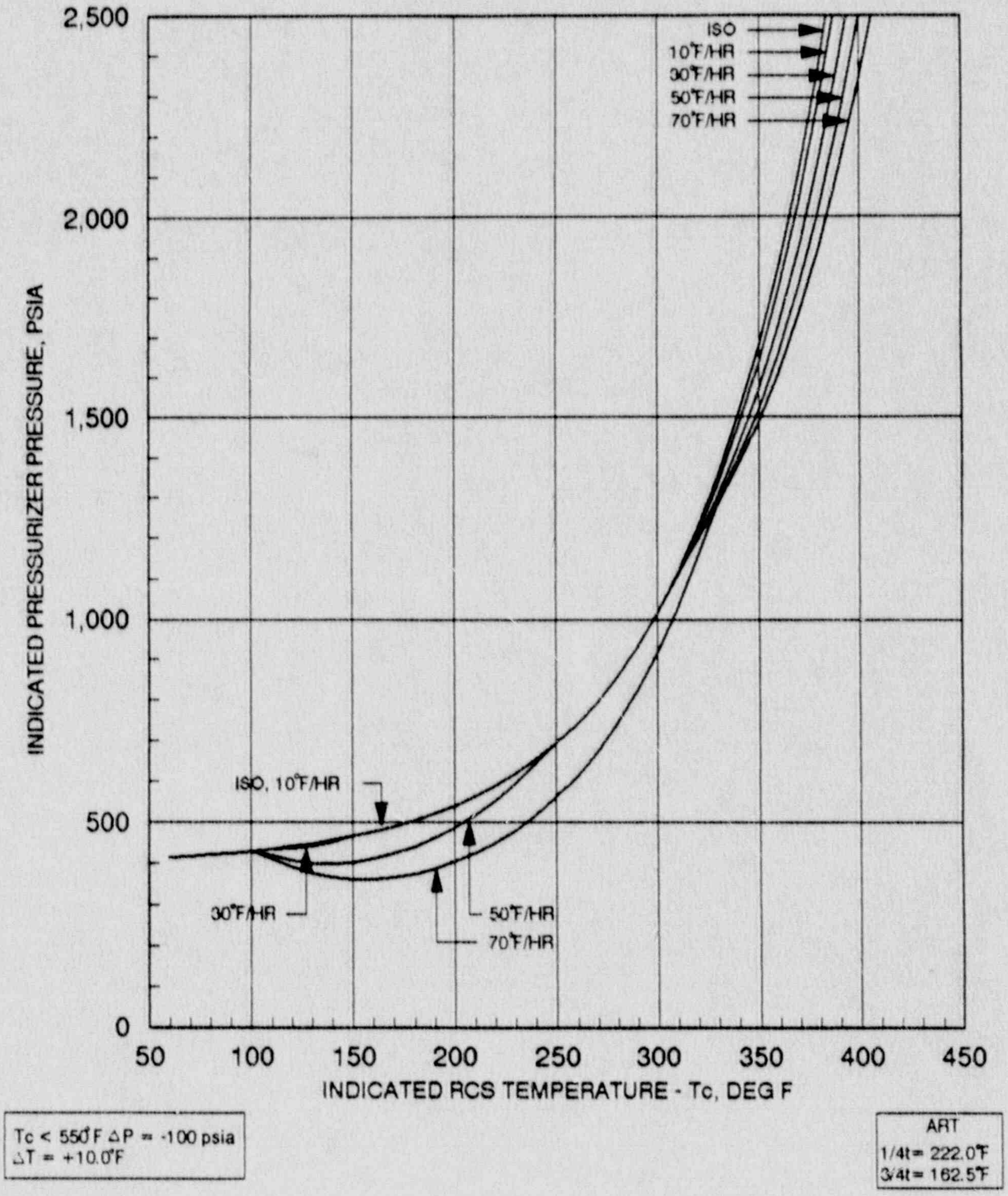
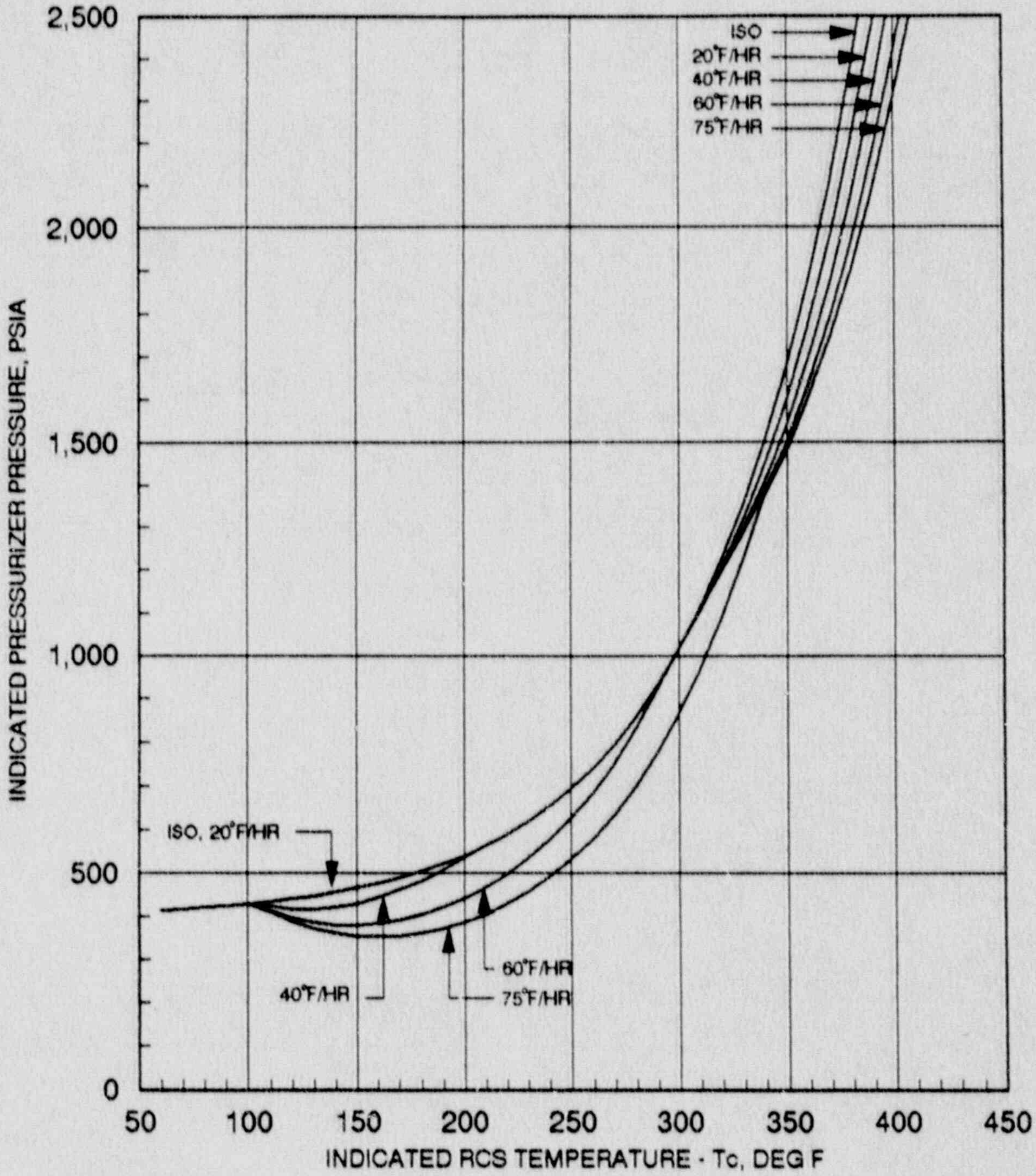


FIGURE 2

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
HEATUP



Tc < 550°F ΔP = -100 psia
ΔT = +10.0°F

ART
1/4t= 222.0°F
3/4t= 162.5°F

FIGURE 3

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
COOLDOWN

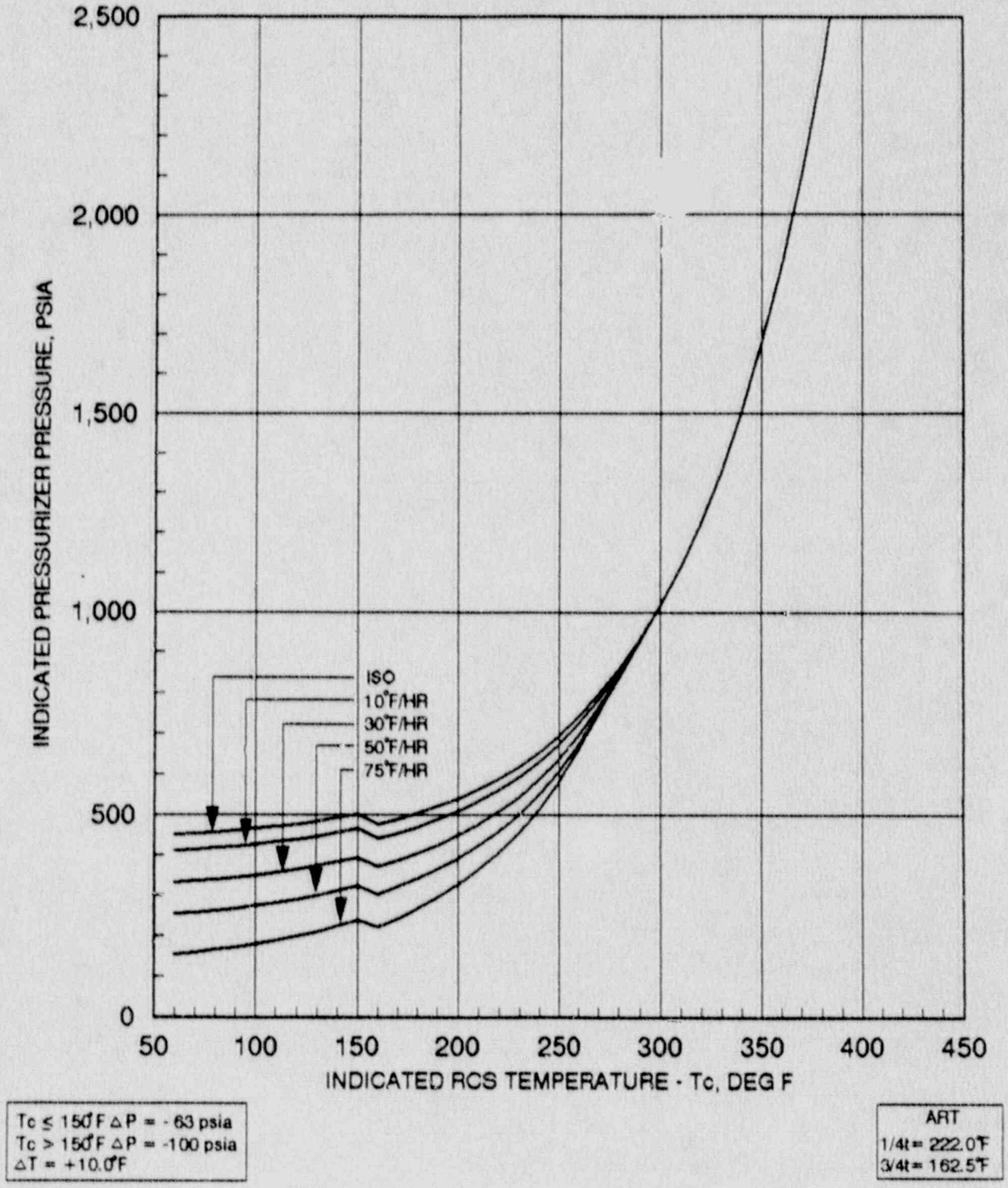
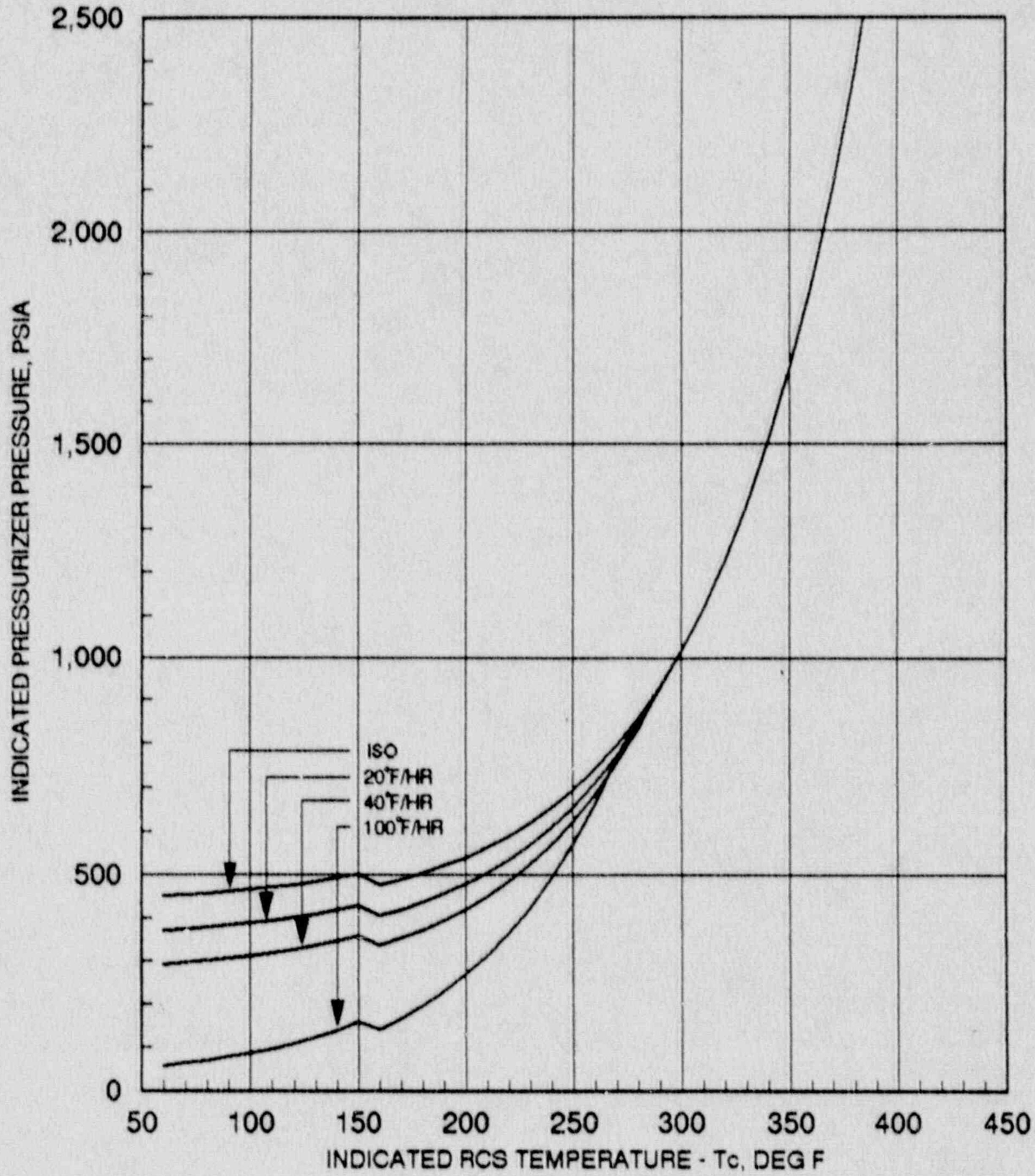


FIGURE 4

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
COOLDOWN



Tc ≤ 150°F ΔP = - 63 psia
Tc > 150°F ΔP = -100 psia
ΔT = +10.0°F

ART
1/4t= 222.0°F
3/4t= 162.5°F

FIGURE 5

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS
12 EFPY, HYDROSTATIC

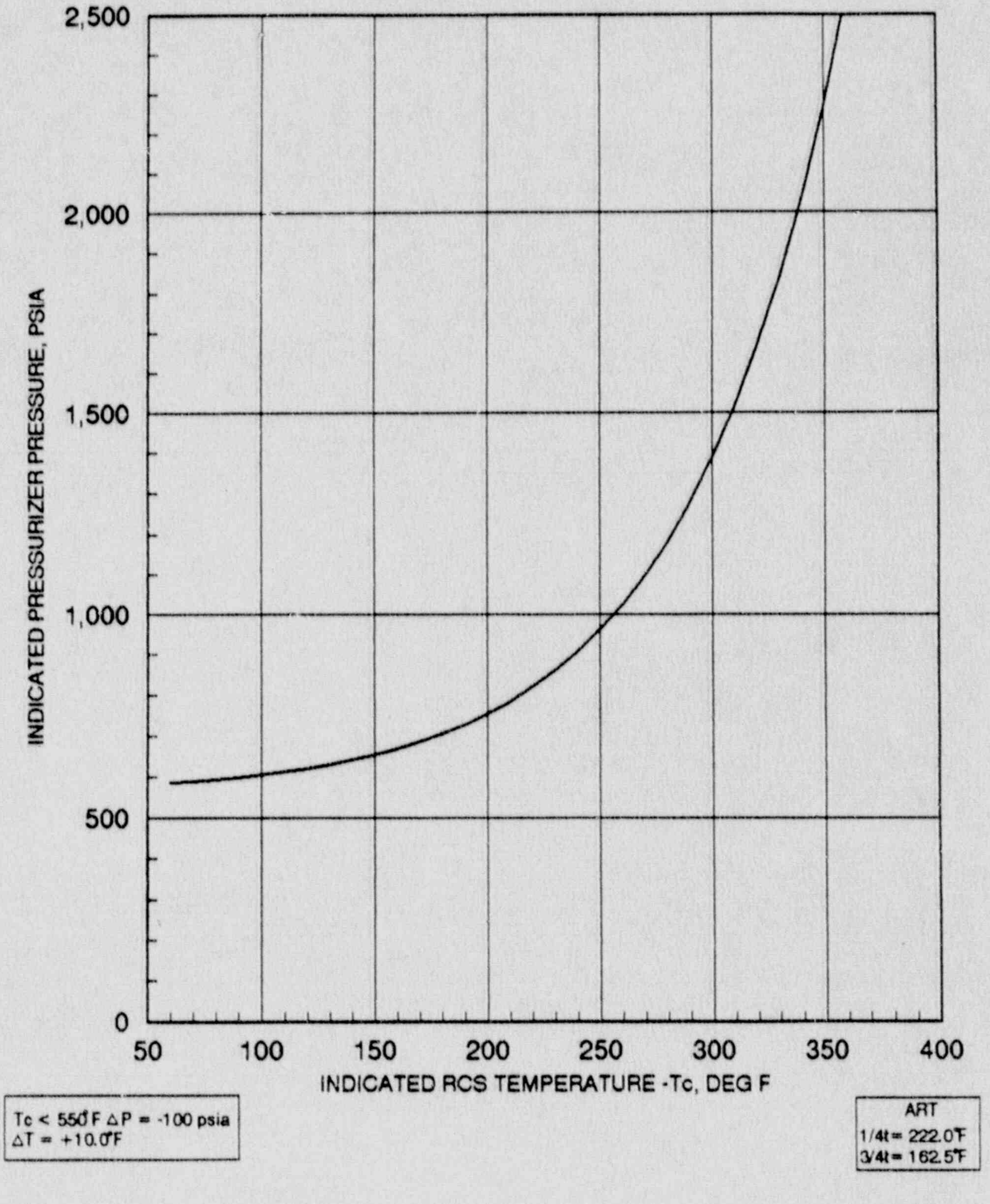


FIGURE 6

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
HEATUP

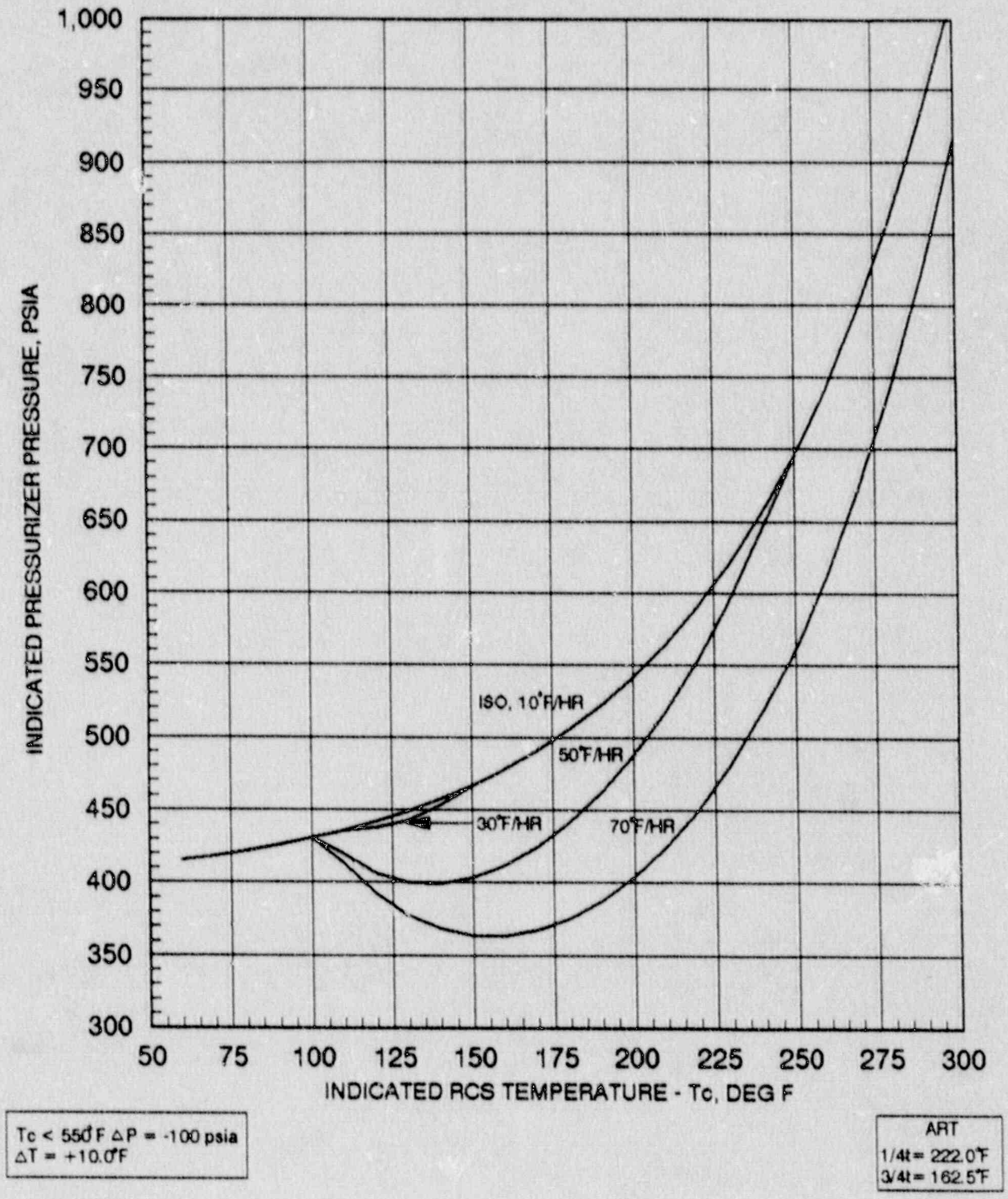
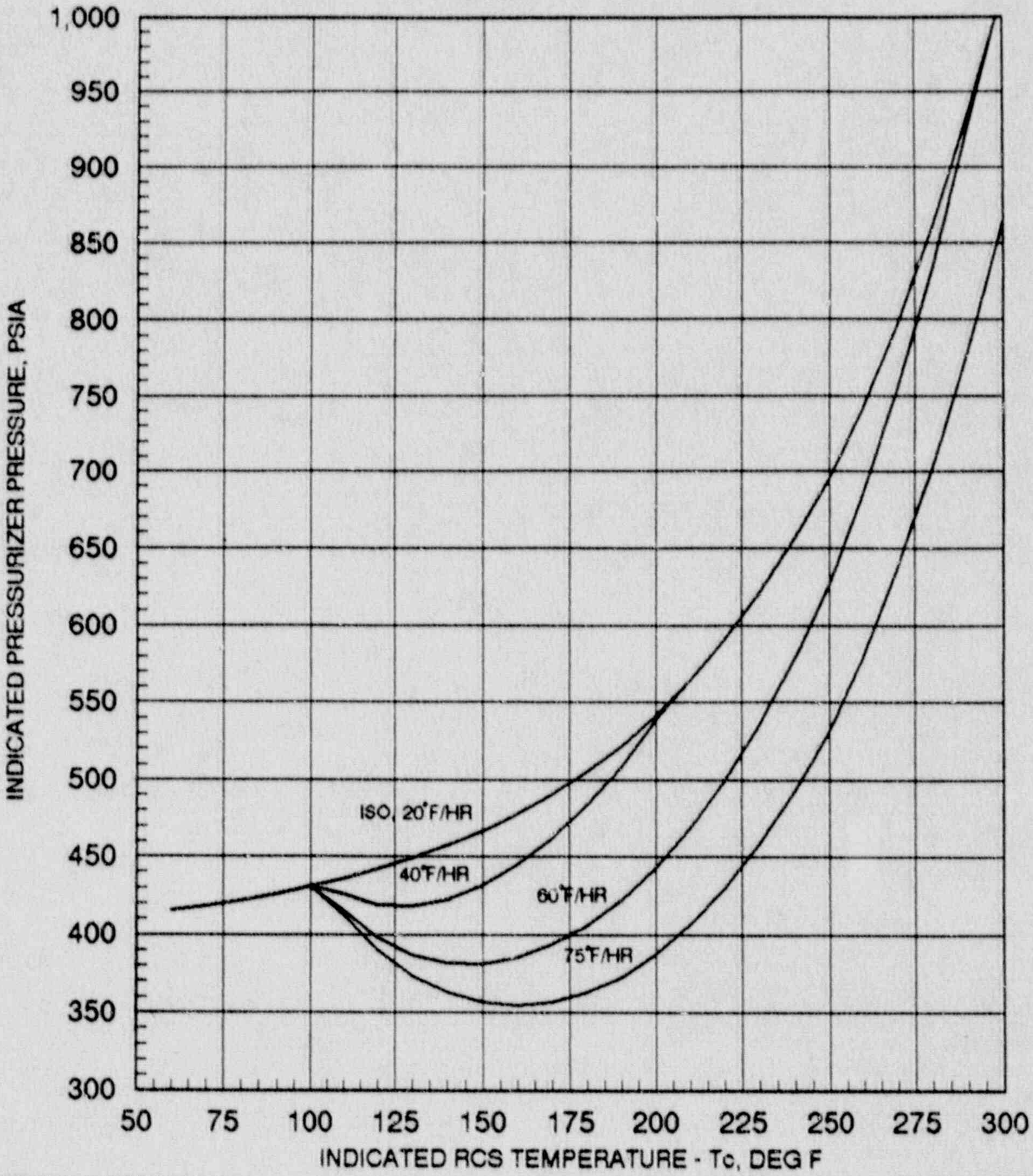


FIGURE 7

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
HEATUP

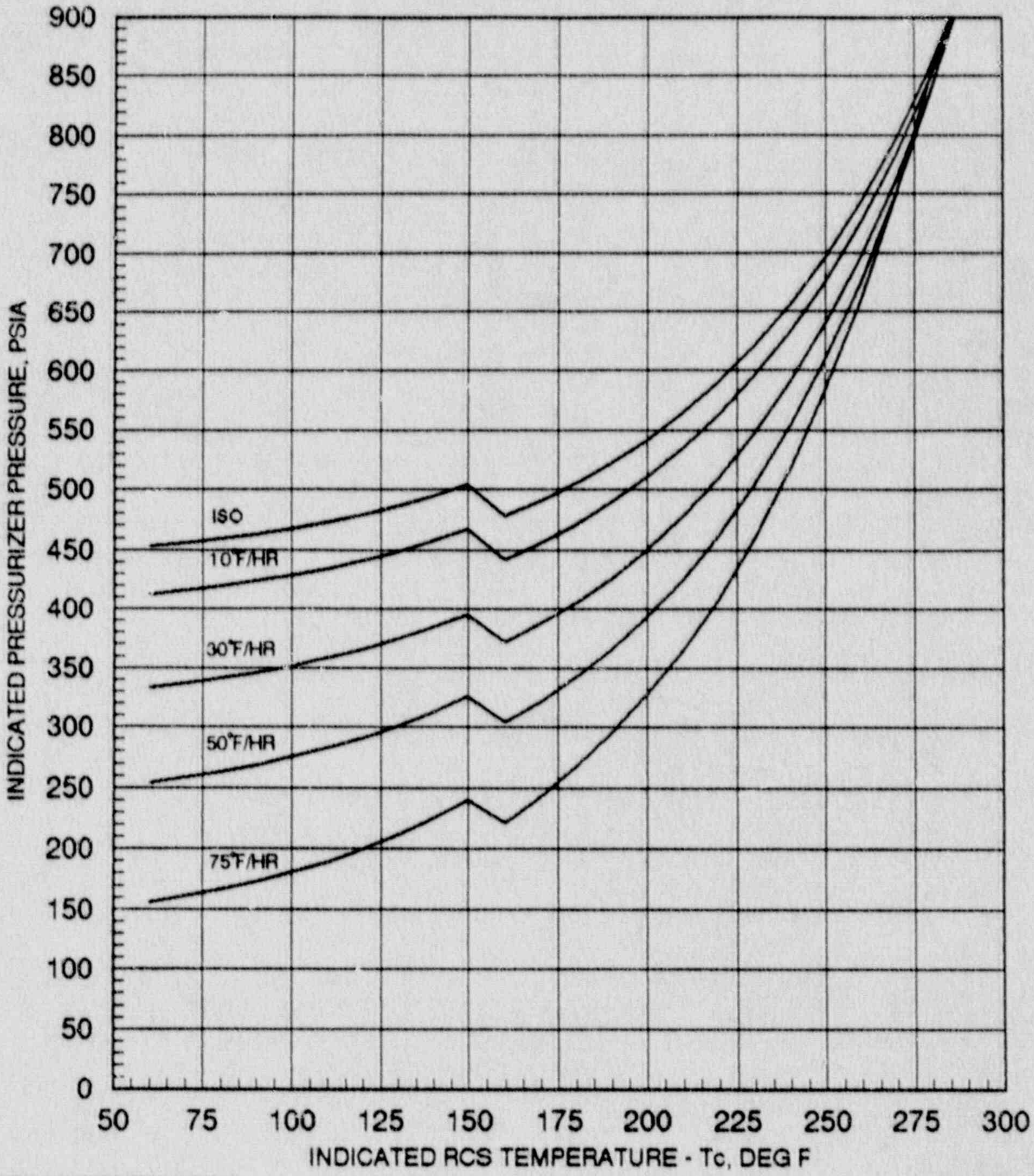


Tc < 550°F ΔP = -100 psia
ΔT = +10.0°F

ART
1/4t = 222.0°F
3/4t = 162.5°F

FIGURE 8

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
COOLDOWN

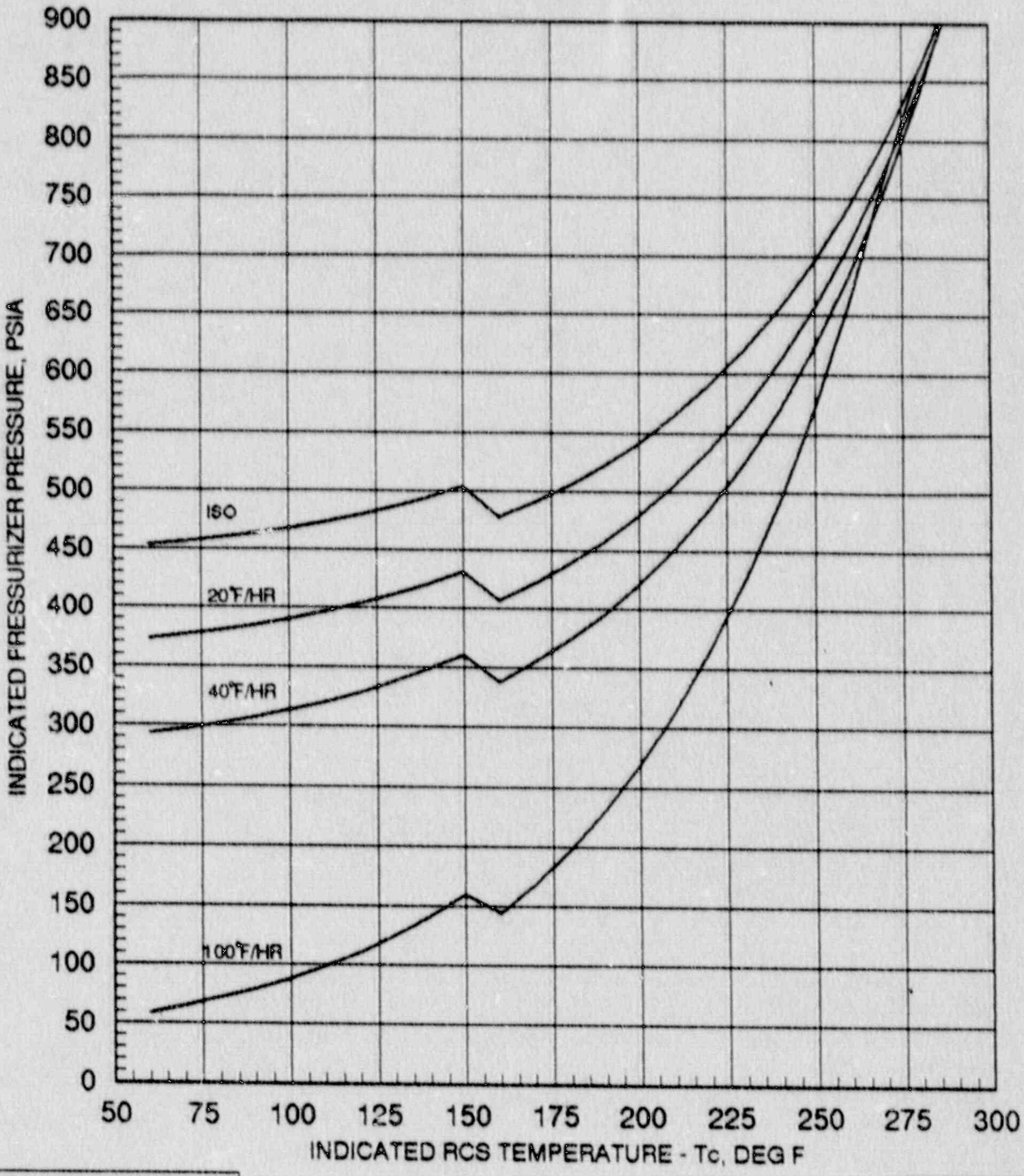


Tc ≤ 150°F ΔP = - 63 psia
Tc > 150°F ΔP = -100 psia
ΔT = +10.0°F

ART
1/4t= 222.0°F
3/4t= 162.5°F

FIGURE 9

BG&E CALVERT CLIFFS UNIT 1
BELTLINE P-T LIMITS, 12 EFPY
COOLDOWN



Tc ≤ 150°F ΔP = - 63 psia
Tc > 150°F ΔP = -100 psia
ΔT = +10.0°F

ART
1/4t= 222.0°F
3/4t= 162.5°F

ATTACHMENT 2

FIGURE 3.4-2a
CALVERT CLIFFS UNIT 1 HEATUP CURVE, 12 EFPY
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS

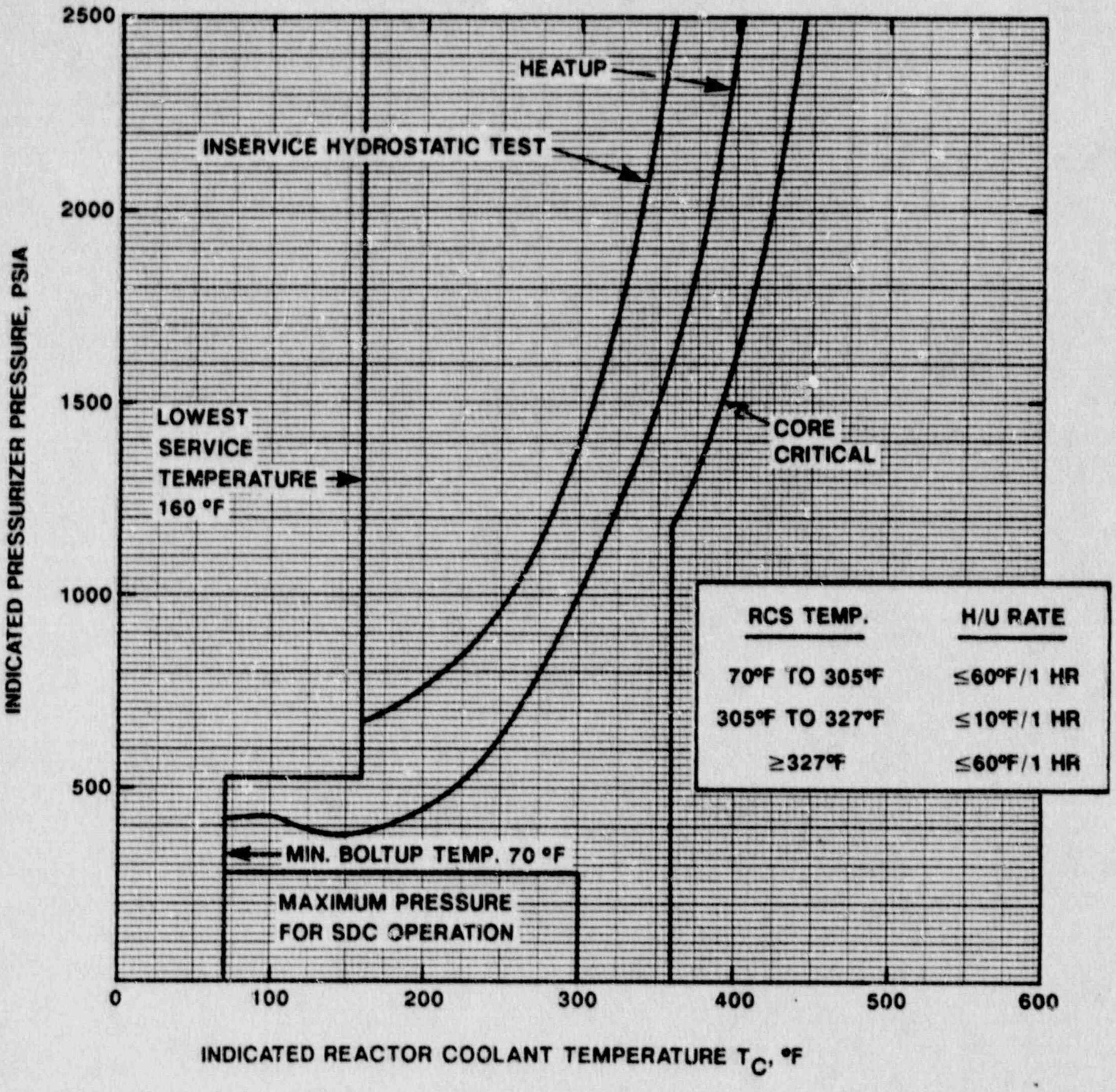


FIGURE 3.4-2b
CALVERT CLIFFS UNIT 1 COOLDOWN CURVE, 12 EFPY
REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITS