WCAP-15364

Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: T. J. Laubham

EXECUTIVE SUMMARY

The purpose of this report is to document the generation of pressure-temperature limit curves for Braidwood Unit 1 for normal operation at 16, 22 and 32 EFPY using the latest methodologies (i.e. WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G., ASME Code Case N-588, ASME Code Case N-640 and WCAP-15315). Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values. The 1/4T and 3/4T values are summarized in Tables 4-16 through 4-21 and were calculated using the circumferential weld WF-562 (Heat 442011) (limiting material for circumferentially oriented flaws, Code Case N-588) and nozzle shell forging 5P-7016 (limiting material for axial flaws). The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all EFPY values evaluated. Hence, only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-6.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated 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 (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. 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 unirradiated RT_{NDT} (IRT_{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"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Braidwood Station Unit No. 1 reactor pressure vessel (Specifically from lower shell forging [49D867/49C813]-1-1). Also supplied was a weldment made with non-copper coated weld wire using the automatic sub-arc welding process (Weld wire heat # 442011 Linde 80 flux, lot number 8061, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company contracted Westinghouse to analyze surveillance capsule W from the Braidwood Unit 1 reactor vessel. As a part of this analysis and the current uprating program, Westinghouse is to generated new heatup and cooldown curves for 16, 22 and 32 EFPY. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10 CFR Part 50[3] (WCAP-15315[4], "Reactor Vessel Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants",
- ASME Code Case N-588[5] (where applicable), and
- Methodology of the 1996 version of ASME Section XI, Appendix G[6].
- The P-T Curves will be developed WITHOUT margins or instrumentation errors.

Circumferential Flaw methodology (ASME Code Case N-588) in combination with 1996 Appendix G to ASME Section XI and ASME Code Case N-640 (Kic methodology) will be used to develop the P-T Curves for the limiting circumferential weld material. In addition, the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (WCAP-15315) will be employed. Axial Flaw methodology, 1996 Appendix G to ASME Section XI and ASME Code Case N-640 (Kic methodology) along with the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (WCAP-15315) will be employed to develop P-T Curves for the limiting forging/base metal material. The final P-T Curves will be developed by generating a P-T Curve based on the most limiting circumferential weld ART value P-T Curve and/or the P-T Curve based on the most limiting axial flaw material ART value. Hence the final P-T Curves will be a composite curve based of the limiting data from the two curves (ie. Circ flaw or axial flaw case).

The purpose of this report is to present the calculations and the development of the Commonwealth Edison Company Braidwood Unit 1 heatup and cooldown curves for 16,22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Commonwealth Edison Company, the surveillance weld data from the Braidwood Unit 1 and Braidwood Unit 2 surveillance programs has been integrated. The Braidwood Unit 1 surveillance weld metal was fabricated from the same weld wire heat as the Braidwood Unit 2 surveillance weld metal (Heat No. 442011). This weld metal is in the upper to lower shell gird weld seam WF-562 of both Units. Per WCAP-15366[7], all the surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies 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_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640 of Appendix G of the ASME Code, Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \qquad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steels.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Code Case N-640 of Appendix G of the ASME Code as follows:

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

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress
K_{It} = stress intensity factor caused by the thermal gradients
K_{Ic} = function of temperature relative to the RT_{NDT} of the material
C = 2.0 for Level A and Level B service limits
C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

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

$$K_{Im} = M_m * (pR_i \div t) \qquad (3)$$

Where M_m for an inside surface is given by:

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

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

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

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

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

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

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

Where:

Ri = vessel inner radius,
t = vessel wall thickness, and
p = internal pressure,

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

$K_{Ib} = M_b$ * maximum bending stress, where M_b is two-thirds of M_m

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

$$K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5} \quad (4)$$

where:

CR = the cooldown rate in °F/hr.

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

$$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5} \quad (5)$$

where:

HU = the heatup rate in °F/hr.

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

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

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \quad (6)$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (7)$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (8)$$

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

Note, that equations 3 through 8 were added to the OPERLIM computer program, which is the Westinghouse computer program used to generate pressure-temperature limit curves. No other changes were made to the OPERLIM computer program with regard to the pressure-temperature curve calculation methodology. Hence, the pressure-temperature curve methodology described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) remains valid for the generation of the pressure-temperature curves documented in this report with the exceptions described above.

At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of a postulated flaw at the ¼T and ¾T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the

reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G 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 wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

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 ¼T 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 ΔT (temperature) developed during cooldown results in a higher value of K_{1C} at the ¼T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{1C} 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 ¼T location and, therefore, allowable pressures may unknowingly be violated 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 ¼T defect 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_{1C} for the ¼T crack during heatup is lower than the K_{1C} for the ¼T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{1C} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the ¼T flaw is considered. 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.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a ¼T flaw located at the ¼T location from the outside surface is assumed. 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 the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. 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 possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Code Case N-588 for Circumferential Weld Flaws:

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

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

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

G-2214.1 Membrane Tension...

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

$$K_{Im} = M_m \times (pR_i / t)$$

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

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

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

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circ. weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G contains the requirements for the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure (3106 psig), which is 621 psig for the Braidwood Unit 1 reactor vessel.

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

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

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

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \qquad (9)$$

Initial RT_{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[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may 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 adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (10)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depthx)} = f_{surface} * e^{(-0.24x)} \qquad (11)$$

where x inches (vessel inner radius and beltline thickness is 86.625 inches and 8.5 inches, respectively)[10] is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 10 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections and the results are presented in Section 6 of WCAP-15316[11]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Table 4-2 through 4-4, herein, contains the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluence values used to calculate the ART values for all beltline materials in the Braidwood Unit 1 reactor vessel for 16, 22 and 32 EFPY. Additionally, the calculated surveillance capsule fluence values are presented in Table 4-5.

Ratio Procedure and Temperature Adjustment:

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2 Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12th & 13th of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize ΔRT_{NDT}, measured to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\ measured} + 1.0*(T_{capsule} - T_{plant}) \qquad (12)$$

The irradiation temperatures of the Braidwood Unit 1 & 2 reactor vessels are as follows:

TABLE 4-1
Irradiation Temperatures of the Braidwood Unit 1 & 2 Reactor Vessels

Fuel Cycle	Braidwood 1	Capsule	Braidwood 2	Capsule
1	557°F	U	557°F	U
2	551°F	- -	551°F	- -
3	551°F	- -	551°F	- -
4	551°F	X	551°F	X
5	551-554°F	- -	551°F *	- -
6	554°F	- -	551°F *	- -
7	554°F *	W	550°F **	W
Average Temp.	553°F		552°F	

* These values are from ComEd and are documented in BRW-DIT-97-321[15].
** This value is from ComEd and is documented in BRW-DIT-2000-0010[16].

Capsules U and X were exposed to the same average operating temperature of 553°F. Capsule W was exposed to the same average operating temperature of 553°F for 5 of 7 cycles. For cycles 6 and 7, the Braidwood Unit 1 capsule W saw only a 1°F increase in average operating temperature. Thus, the average inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest. Hence, no temperature adjustments are required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-6 through 4-9. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 7, all available surveillance data for Braidwood Unit 1 is credible. This assessment also calculated the vessel weld (including temperature and chemistry adjustment) chemistry factor using Braidwood Unit 1 stand alone data and it was determined to be 33.5°F. Position 2.1 chemistry factors are calculated in Table 4-15.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Braidwood Unit 1, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}\ .$$

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Braidwood Unit 1, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I .

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 10.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 16 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2 ,E > 1.0 MeV)	1/4T (n/cm^2 ,E > 1.0 MeV)	3/4T (n/cm^2 ,E > 1.0 MeV)
Intermediate Shell Forging [49D383/49C344]-1-1	1.03×10^{19}	6.19×10^{18}	2.23×10^{18}
Lower Shell Forging [49D687/49C813]-1-1	1.03×10^{19}	6.19×10^{18}	2.23×10^{18}
Nozzle Shell Forging 5P-7016	3.07×10^{18}	1.84×10^{18}	6.65×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.00×10^{19}	6.00×10^{18}	2.17×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	3.07×10^{18}	1.84×10^{18}	6.65×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-3

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2 ,E > 1.0 MeV)	1/4T (n/cm^2 ,E > 1.0 MeV)	3/4T (n/cm^2 ,E > 1.0 MeV)
Intermediate Shell Forging [49D383/49C344]-1-1	1.41×10^{19}	8.47×10^{18}	3.05×10^{18}
Lower Shell Forging [49D687/49C813]-1-1	1.41×10^{19}	8.47×10^{18}	3.05×10^{18}
Nozzle Shell Forging 5P-7016	4.20×10^{18}	2.52×10^{18}	9.09×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.37×10^{19}	8.23×10^{18}	2.97×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	4.20×10^{18}	2.52×10^{18}	9.09×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-4

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2 ,E > 1.0 MeV)	1/4T (n/cm^2 ,E > 1.0 MeV)	3/4T (n/cm^2 ,E > 1.0 MeV)
Intermediate Shell Forging [49D383/49C344]-1-1	2.05×10^{19}	1.23×10^{19}	4.44×10^{18}
Lower Shell Forging [49D687/49C813]-1-1	2.05×10^{19}	1.23×10^{19}	4.44×10^{18}
Nozzle Shell Forging 5P-7016	6.08×10^{18}	3.65×10^{18}	1.32×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.99×10^{19}	1.19×10^{19}	4.31×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	6.08×10^{18}	3.65×10^{18}	1.32×10^{18}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-5
Calculated Integrated Neutron Exposure of the Braidwood Unit 1
Surveillance Capsules Tested to Date

Capsule	Fluence
U	3.87×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.24×10^{19} n/cm^2, (E > 1.0 meV)
W	2.09×10^{19} n/cm^2, (E > 1.0 MeV)

Contained in Table 4-6 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[12], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-6
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[a]
Intermediate Shell Forging [49D867/49C813]-1-1 (Tangential Orientation)	U	5.78°F
	X	38.23°F
	W	24.14°F
Intermediate Shell Forging [49D867/49C813]-1-1 (Axial Orientation)	U	0.0°F[b]
	X	28.75°F
	W	37.11°F
Surveillance Program Weld Metal (Heat # 442011)	U	17.06°F
	X	30.15°F
	W	49.68°F
Heat Affected Zone	U	56.31°F
	X	92.8°F
	W	84.86°F

Notes:

(a) From capsule W analysis results Calculated using the measured Charpy data plotted using CVGRAPH, Verision 4.1[11].

(b) Actual value for ΔRT_{NDT} is -16.07. This physically should not occur, therefore for conservatism (i.e. higher Chemistry Factor) a value of zero will be reported.

Table 4-7 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 1 base materials in the beltline region. Table 4-8 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 1 surveillance weld material, while Table 4-9 presents the overall best estimate average for that heat of weld. Table 4-10 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-10 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-12. Table 4-11 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-12.

TABLE 4-7
Calculation of the Best Estimate Cu and Ni Weight Percent for the Braidwood Unit 1 Forging Materials

Reference	Intermediate Shell Forging [49D383/49C344]-1-1 MK 24-2		Lower Shell Forging [49D867/49C813]-1-1 MK 24-3	
	Cu %	Ni %	Cu %	Ni %
13	0.05	0.73	---	---
11	---	---	0.03	0.73
11 (Charpy EL-6)	---	---	0.052	0.746
11 (Charpy ET-57)	---	---	0.046	0.736
Charpy ET-31[a]	---	---	0.045	0.687
Charpy EL-38[a]	---	---	0.049	0.725
Charpy ET-40[a]	---	---	0.047	0.721
Charpy EL-42[a]	---	---	0.053	0.809
Best Estimate Average[b]	0.05	0.73	0.05	0.74

Notes:
(a) Charpy Specimen From Capsule W of Braidwood Unit 1 (Ref 11).
(b) The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-8
Calculation of the Average Cu and Ni Weight Percent for the Braidwood Unit 1 Surveillance Weld Material Only (Heat # 442011)

Reference	Weight % Copper	Weight % Nickel
WCAP-14824, Rev.2[a]	0.032	0.671
Charpy EW-32[b]	0.032	0.674
Charpy EW-33[b]	0.032	0.660
Charpy EW-34[b]	0.030	0.661
Charpy EW-42[b]	0.033	0.690
Surveillance Weld Average	0.03	0.67

Notes:
(a) This is the average of 29 data points. Therefore, to average with the new Charpy data, multiply the average copper and nickel values from Ref. 17 by 29, add four new points and divide by 33.
(b) Charpy specimen from Capsule W of Braidwood Unit 1 (Ref. 11).

TABLE 4-9

Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Units 1 & 2 Weld Material (Using Braidwood 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
B&W Weld Qualification BAW-2261	WCAP-14824, Rev. 2	0.028	0.63
B&W Weld Qualification BAW-2261	WCAP-14824, Rev. 2	0.03	0.65
B&W Weld Qualification BAW-2261	WCAP-14824, Rev. 2	0.04	0.67
Braidwood Unit 1 Surv. Data Ave.(a)	Table 4-8	0.032	0.671
Braidwood Unit 2 Surv. Data Ave.(b)	WCAP-15369	0.034	0.710
BEST ESTIMATE AVERAGE(c)		0.03	0.67

Notes:

(a) The weld material in the Braidwood Unit 1 surveillance program was made of the same wire and flux as the reactor vessel inter. to lower shell girth seam weld (Weld seam WF-562, Wire Heat No. 442011, Flux Type Linde 80, Flux Lot No. 8061). The weld wire is type Linde MnMoNi (Low Cu-P).

(b) The Braidwood Unit 2 surveillance weld is representative of that used in the Braidwood Unit 1 reactor vessel core region girth seam (WF-562) heat number 442011, with a Linde 80 type flux, lot number 8061 (i.e. The Braidwood Unit 2 surveillance weld was fabricated with Linde 80 type flux, Lot number 0344).

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-10
Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}[a]
Closure Head Flange 2030-V-1	0.11	0.67	-20
Vessel Flange 122N357VA1	---	0.77	-10
Nozzle Shell Forging 5P7016	0.04	0.73	10
Intermediate Shell Forging [49D383/49C344]-1-1 [Table 4-7]	0.05	0.73	-30
Lower Shell Forging [49D867/49C813]-1-1 [Table 4-7]	0.05	0.74	-20
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011) [Table 4-9]	0.03	0.67	40
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	0.04	0.46	-25
Surveillance Weld Braidwood Unit 1 (Heat 442011) [Table 4-9]	0.03	0.67	- - -
Surveillance Weld Braidwood Unit 2 (Heat 442011) [Table 4-9]	0.03	0.71	- - -

Notes:
(a) The Initial RT_{NDT} values for the forgings and welds are based on measured data.

TABLE 4-11

Calculation of Chemistry Factors for Braidwood Unit 1 using Surveillance Capsule Data

Material	Capsule	Capsule f[a] (°F)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Lower Shell Forging [49D867/49C813]-1-1 (Tangential)	U	0.387	0.737	5.78	4.26	0.543
	X	1.24	1.060	38.23	40.52	1.124
	W	2.09	1.201	24.14	28.99	1.442
Lower Shell Forging [49D867/49C813]-1-1 (Axial)	U	0.387	0.737	0.0[e]	0.0	0.543
	X	1.24	1.060	28.75	30.48	1.124
	W	2.09	1.201	37.11	44.57	1.442
				SUM:	148.82	6.218
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (148.82°F) \div (6.218) = \mathbf{23.9°F}$					
Braidwood Unit 1 Surveillance Weld Metal	U	0.387	0.737	17.06[d]	12.57	0.543
	X	1.24	1.060	30.15[d]	31.96	1.124
	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2 Surveillance Weld Metal	U	0.40	0.746	0.0[d,f]	0.0	0.557
	X	1.23	1.058	26.3[d]	27.83	1.119
	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19°F) \div (6.273) = \mathbf{25.7°F}$					

Notes:

(a) f = Calculated fluence from the Braidwood Unit 1 capsule W dosimetry analysis results[11] and Calulated fluence from the Braidwood Unit 2 capsule W analysis[14], ($\times 10^{19}$ n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 1 taken from Ref. 11 and for Braidwood Unit 2 taken from Ref. 14.

(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted. The chemistry factors of the surveillance welds and the vessel weld is 41°F, hence, the ratio factor is 1.0. In addition, the average inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest. Hence, no temperature adjustments are required.

(e) Actual value of ΔRT_{NDT} is -16.07. This physically should not occur, therefore for conservatism (i.e. higher chemistry factor) a value of zero will be used.

(f) Measured value is –0.58. 0 is assumed for conservatism.

Contained in Table 4-12 is a summary of the calculated chemistry factors of the Braidwood Unit 1 beltline materials based on the Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1 methodology.

TABLE 4-12
Summary of the Braidwood Unit 1 Beltline Material Chemistry Factor Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Intermediate Shell Forging [49D383/49C344]-1-1	31.0°F	- - -
Lower Shell Forging [49D867/49C813]-1-1	31.0°F	23.9°F
Nozzle Shell Forging 5P7016	26.0°F	- - -
Intermediate Shell to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	41.0°F	25.7°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	54.0°F	- - -
Braidwood Unit 1 Surveillance Program Weld Metal	41.0°F	- - -
Braidwood Unit 1 & 2 Surveillance Program Weld Metal	41.0°F	- - -

Contained in Tables 4-13 through 4-15 is a summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Braidwood Unit 1 reactor vessel beltline materials for 16, 22 and 32 EFPY.

TABLE 4-13
Calculation of the ¼T and ¾T Fluence Factor Values used for the Generation of the 16 EPFY Heatup/Cooldown Curves

Material	¼T F (n/cm^2 ,E > 1.0 MeV)	¼T FF	¾T f (n/cm^2 ,E > 1.0 MeV)	¾T FF
Intermediate Shell Forging [49D383/49C344]-1-1	6.19×10^{18}	0.866	2.23×10^{18}	0.596
Lower Shell Forging [49D687/49C813]-1-1	6.19×10^{18}	0.866	2.23×10^{18}	0.596
Nozzle Shell Forging 5P-7016	1.84×10^{18}	0.550	6.65×10^{17}	0.340
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	6.00×10^{18}	0.857	2.17×10^{18}	0.589
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	1.84×10^{18}	0.550	6.65×10^{17}	0.340

TABLE 4-14
Calculation of the ¼T and ¾T Fluence Factor Values
used for the Generation of the 22 EFPY Heatup/Cooldown Curves

Material	¼T F (n/cm^2 ,E > 1.0 MeV)	¼T FF	¾T f (n/cm^2 ,E > 1.0 MeV)	¾T FF
Intermediate Shell Forging [49D383/49C344]-1-1	8.47×10^{18}	0.953	3.05×10^{18}	0.675
Lower Shell Forging [49D687/49C813]-1-1	8.47×10^{18}	0.953	3.05×10^{18}	0.675
Nozzle Shell Forging 5P-7016	2.52×10^{18}	0.626	9.09×10^{17}	0.398
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	8.23×10^{18}	0.945	2.97×10^{18}	0.668
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	2.52×10^{18}	0.626	9.09×10^{17}	0.398

TABLE 4-15
Calculation of the ¼T and ¾T Fluence Factor Values
used for the Generation of the 32 EFPY Heatup/Cooldown Curves

Material	¼T F (n/cm^2 ,E > 1.0 MeV)	¼T FF	¾T f (n/cm^2 ,E > 1.0 MeV)	¾T FF
Intermediate Shell Forging [49D383/49C344]-1-1	1.23×10^{19}	1.058	4.44×10^{18}	0.774
Lower Shell Forging [49D687/49C813]-1-1	1.23×10^{19}	1.058	4.44×10^{18}	0.774
Nozzle Shell Forging 5P-7016	3.65×10^{18}	0.722	1.32×10^{18}	0.475
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.19×10^{19}	1.049	4.31×10^{18}	0.766
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	3.65×10^{18}	0.722	1.32×10^{18}	0.475

Contained in Tables 4-16 and 4-21 are the calculations of the ART values used for the generation of the 16, 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-16

Calculation of the ART Values for the ¼T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 16[b] EFPY ($x 10^{19}$)	¼-t f ($x 10^{19}$)	¼-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.03	0.619	0.866	-30	26.8	0	13.4	26.8	24
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.03	0.619	0.866	-20	26.8	0	13.4	26.8	34
Lower Shell Forging → using S/C Data				23.9	1.03	0.619	0.866	-20	20.7	0	10.4	20.7	21
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.00	0.600	0.857	40	35.1	0	17.0	34.0	109
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.00	0.600	0.857	40	22.0	0	11.0	22.0	84
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.307	0.184	0.550	10	14.3	0	7.2	14.3	39
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.307	0.184	0.550	-25	29.7	0	14.9	29.7	34

NOTES:

(a) Initial RT_{NDT} values are measured values.

(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(d) ΔRT_{NDT} = CF * FF

(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).

(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible.

TABLE 4-17

Calculation of the ART Values for the ¾T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 16[b] EFPY (x 10^{19})	3/4-t f (x 10^{19})	¾-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.03	0.223	0.596	-30	18.5	0	9.3	18.5	7
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.03	0.223	0.596	-20	18.5	0	9.3	18.5	17
Lower Shell Forging → using S/C Data				23.9	1.03	0.223	0.596	-20	14.2	0	7.1	14.2	8
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.00	0.217	0.589	40	24.1	0	12.1	24.1	88
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.00	0.217	0.589	40	15.1	0	7.6	15.1	70
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.307	0.067	0.340	10	8.8	0	4.4	8.8	28
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.307	0.067	0.340	-25	18.4	0	9.2	18.4	12

NOTES:

(a) Initial RT_{NDT} values are measured values.

(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(d) ΔRT_{NDT} = CF * FF

(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).

(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible.

TABLE 4-18
Calculation of ART Values for the ¼T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 22[b] EFPY (x 10^{19})	¼-t f (x 10^{19})	¼-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.41	0.847	0.953	-30	29.5	0	14.8	29.5	29
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.41	0.847	0.953	-20	29.5	0	14.8	29.5	39
Lower Shell Forging → using S/C Data				23.9	1.41	0.847	0.953	-20	22.8	0	11.4	22.8	26
Inter. To Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.37	0.823	0.945	40	38.7	0	19.4	38.7	117
Inter. To Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.37	0.823	0.945	40	24.3	0	12.2	24.3	89
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.420	0.252	0.626	10	16.3	0	8.2	16.3	43
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.420	0.252	0.626	-25	33.8	0	16.9	33.8	43

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

TABLE 4-19
Calculation of ART Values for the ¾T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	F @ 22[b] EFPY ($x 10^{19}$)	3/4-t f ($x 10^{19}$)	¾-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	1.41	0.305	0.675	-30	20.9	0	10.5	20.9	12
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	1.41	0.305	0.675	-20	20.9	0	10.5	20.9	22
Lower Shell Forging → using S/C Data				23.9	1.41	0.305	0.675	-20	16.1	0	8.1	16.1	12
Inter. To Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.37	0.297	0.668	40	27.4	0	13.7	27.4	95
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.37	0.297	0.668	40	17.2	0	8.6	17.2	74
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.420	0.091	0.398	10	10.3	0	5.2	10.3	31
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.420	0.091	0.398	-25	21.5	0	10.8	21.5	18

NOTES:

(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) $\Delta RT_{NDT} = CF * FF$
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

TABLE 4-20
Calculation of ART Values for the ¼T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 32[b] EFPY (x 10^{19})	¼-t f (x 10^{19})	¼-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	2.05	1.23	1.058	-30	32.8	0	16.4	32.8	36
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	2.05	1.23	1.058	-20	32.8	0	16.4	32.8	46
Lower Shell Forging → using S/C Data				23.9	2.05	1.23	1.058	-20	25.3	0	12.7	25.3	31
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.99	1.19	1.049	40	43.0	0	21.5	43.0	126
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.99	1.19	1.049	40	27.0	0	13.5	27.0	94
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.608	0.365	0.722	10	18.8	0	9.4	18.8	48
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.608	0.365	0.722	-25	39.0	0	19.5	39.0	53

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

TABLE 4-21
Calculation of ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF	f @ 32[b] EFPY ($\times 10^{19}$)	3/4-t f ($\times 10^{19}$)	¾-t FF	I[a]	ΔRT_{NDT}[d]	σ_I	σ_Δ	M	ART[c]
Intermediate Shell Forging	49D383/49C344-1-1	0.05	0.73	31.0	2.05	0.444	0.774	-30	24.0	0	12.0	24.0	18
Lower shell Forging	49D867/49C813-1-1	0.05	0.74	31.0	2.05	0.444	0.774	-20	24.0	0	12.0	24.0	28
Lower Shell Forging → using S/C Data				23.9	2.05	0.444	0.774	-20	18.5	0	9.3	18.5	17
Inter. to Lower Shell Circ. Weld Metal[e]	WF-562	0.03	0.67	41.0	1.99	0.431	0.766	40	31.4	0	15.7	31.4	103
Inter. to Lower Shell Circ. Weld Metal[f] → using S/C Data				25.7	1.99	0.431	0.766	40	19.7	0	9.9	19.7	79
Nozzle Shell Forging	5P-7016	0.04	0.73	26.0	0.608	0.132	0.475	10	12.4	0	6.2	12.4	35
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.608	0.132	0.475	-25	25.7	0	12.9	25.7	26

NOTES:
(a) Initial RT_{NDT} values are measured values.
(b) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).
(c) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)
(d) ΔRT_{NDT} = CF * FF
(e) The CF for the Braidwood Unit 1 Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 Weld (WF-562, heat # 442011) and the Braidwood 2 Weld (WF-562, Heat # 442011).
(f) The Braidwood 1 and 2 surveillance programs contain the same heat of weld metal (heat # 442011). Hence, the surveillance program results have been integrated and they are credible

The intermediate to lower shell circumferential weld and the nozzle shell forging are the limiting beltline material for all heatup and cooldown curves to be generated. The ART value associated with this material will be used in all three sets of curves. The circumferential weld ART will be used when generating curves for Code Case N-588 (ie. Circ. Flaw). The ART associated with the limiting axial material must also be considered to determine if this case would be more conservative or overlap the circumferential flaw curves. Contained in Table 4-22 is a summary of the limiting ARTs to be used in the generation of the Braidwood Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-22

Summary of the Limiting ART Values used in Generation of the Braidwood Unit 1 Reactor Vessel Heatup and Cooldown Curves

EFPY	¼ T Limiting ART	¾ T Limiting ART
Limiting Circumferential Material (Weld Seam WF-562)		
16	84	70
22	89	74
32	94	79
Limiting Axial Material (Nozzle Shell Forging 5P-7016)		
16	39	28
22	43	31
32	48	35

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-6 present the 16, 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology, Code Case N-640 and Code Case N-588. The heatup and cooldown curves generated for the limiting circumferential material (Weld Seam WF-562) utilizing Code Case N-588 and the 1996 Appendix G methodology are less conservative than the heatup and cooldown curves generated for the axial flaw case. This is true throughout the entire temperature range, including the criticality curve. Hence, the heatup and cooldown curves presented in this reoprt were generated utilizing the limiting axial material ART (Nozzle Shell Forging 5P-7016), Code Case N-640 and the 1996 ASME Appendix G methodology.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-6. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1, 5-3 and 5-5 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640 and Appendix G to Section XI of the ASME Code[6] as follows:

$$1.5K_{Im} < K_{Ic} \qquad (13)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RTNDT)]}$,
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must 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 calculated as described in Section 3 of this report. The minimum temperature for the inservice hydrostatic leak test for the Braidwood Unit 1 reactor vessel at 16 EFPY is 100°F at 2485 psig using the 1996 App. G Methodology and Code Case N-640. The minimum temperature for the inservice hydrostatic leak test for the Braidwood Unit 1 reactor vessel at 22 EFPY is 103°F at 2485 psig using the 1996 App. G Methodology and Code Case N-640. The minimum temperature for the inservice hydrostatic leak test for the Braidwood Unit 1 reactor vessel at 32 EFPY is 108°F at 2485 psig using the 1996 App. G Methodology and Code Case N-640. The approximately vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-6 define all of the above limits for ensuring prevention of nonductile failure for the Braidwood Unit 1 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-6 are presented in Tables 5-1 through 5-3, respectively.

Additionally, Westinghouse Engineering has reviewed the minimum boltup temperature requirements for the Braidwood Unit 1 reactor pressure vessel. According to Paragraph G-2222 of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, the reactor vessel may be bolted up and pressurized to 20 percent of the initial hydrostatic test pressure at the initial RT_{NDT} of the material stressed by the boltup. Therefore, since the most limiting initial RT_{NDT} value is -10°F (vessel flange), the reactor vessel can be bolted up at this temperature.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 16 EFPY: ¼T, 39°F
¾T, 28°F

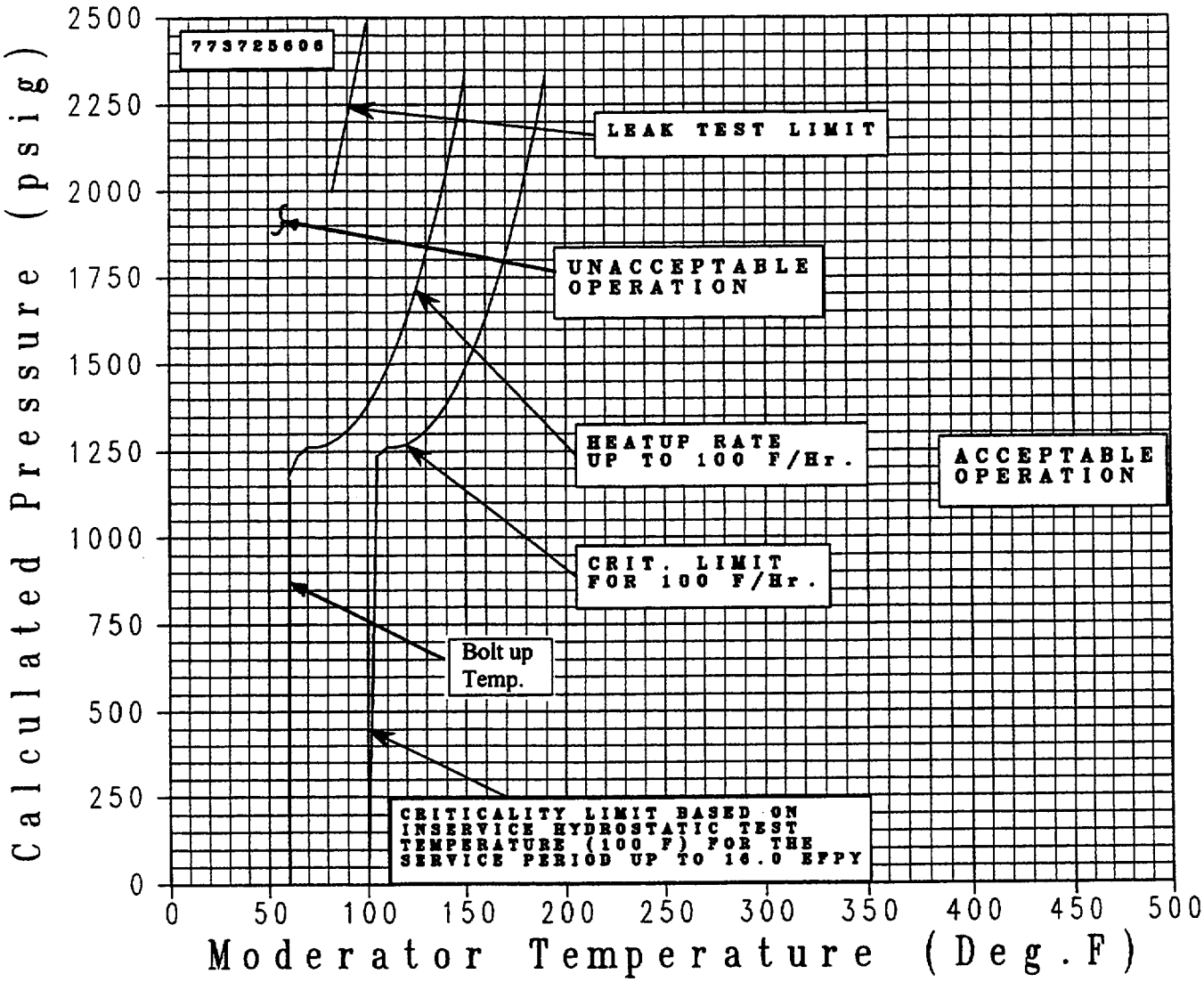

FIGURE 5-1 Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 16 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 16 EFPY: ¼T, 39°F
¾T, 28°F

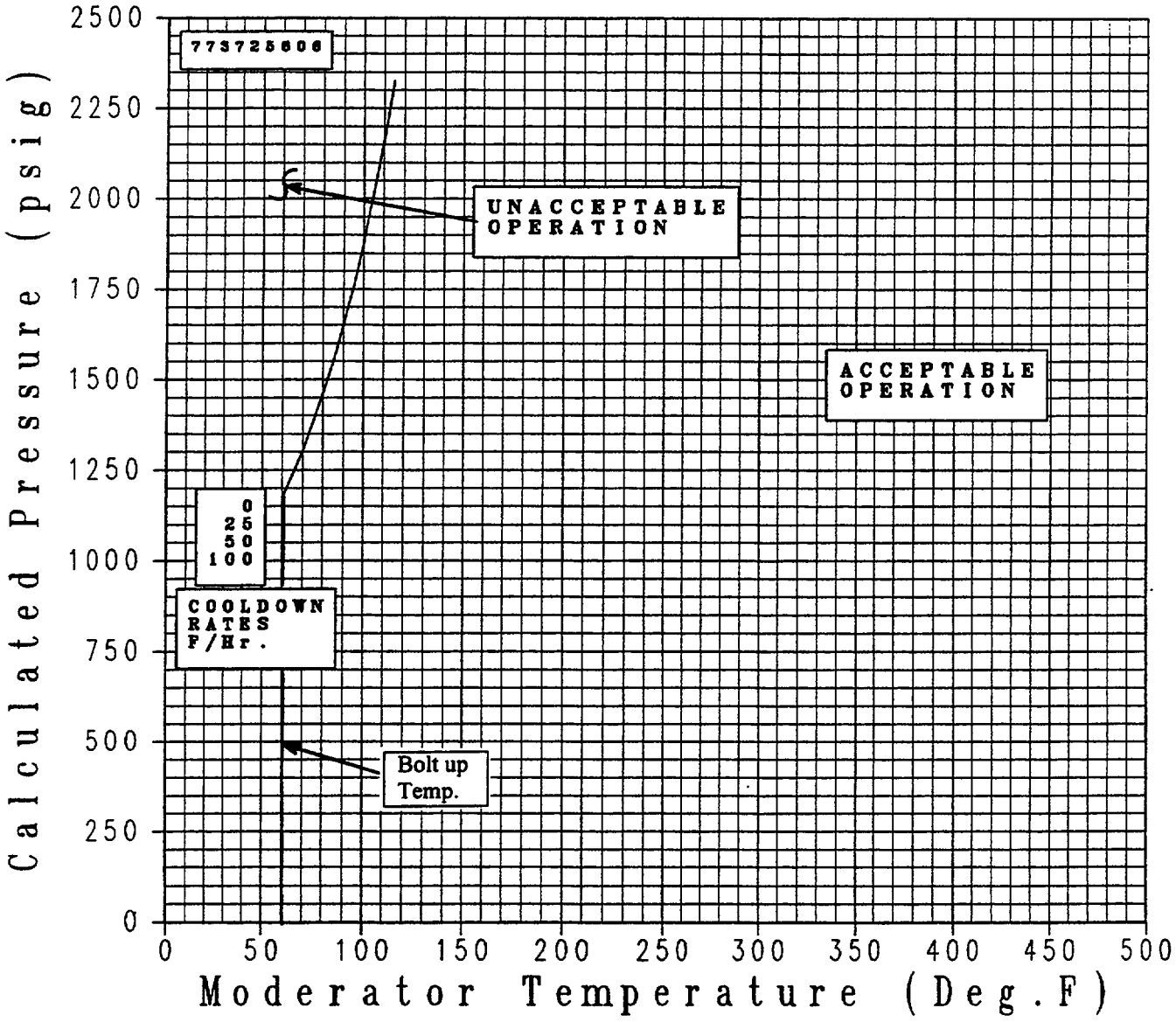

FIGURE 5-2 Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 16 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins for Instrumentation Errors)

TABLE 5-1

Braidwood Unit 1 Heatup and Cooldown Data Points for the 16 EFPY Curves
Using 1996 Appendix G and Code Case N-640 Methodology
(Without Margins of for Instrumentation Errors)

Run = 773725606		Cooldown Curves					
Steady State		25 Deg. °F/Hr		50 Deg. °F/Hr		100 Deg. °F/Hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	1177	60	1177	60	1177	60	1177
65	1237	65	1237	65	1237	65	1237
70	1304	70	1304	70	1304	70	1304
75	1377	75	1377	75	1377	75	1377
80	1459	80	1459	80	1459	80	1459
85	1549	85	1549	85	1549	85	1549
90	1648	90	1648	90	1648	90	1648
95	1758	95	1758	95	1758	95	1758
100	1880	100	1880	100	1880	100	1880
105	2014	105	2014	105	2014	105	2014
110	2162	110	2162	110	2162	110	2162
115	2326	115	2326	115	2326	115	2326

Run = 773725606		Heatup Curves					
		100 Deg. °F/Hr		Crit. Limit		Leak Test Limit	
		Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
		60	0	100	0	82	2000
		60	1177	105	1237	99	2485
		65	1237	110	1263		
		70	1263	115	1263		
		75	1263	120	1271		
		80	1271	125	1286		
		85	1286	130	1310		
		90	1310	135	1342		
		95	1342	140	1382		
		100	1382	145	1431		
		105	1431	150	1488		
		110	1488	155	1555		
		115	1555	160	1631		
		120	1631	165	1717		
		125	1717	170	1814		
		130	1814	175	1924		
		135	1924	180	2046		
		140	2046	185	2183		
		145	2183	190	2335		
		150	2335				

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 22 EFPY: ¼T, 43°F
¾T, 31°F

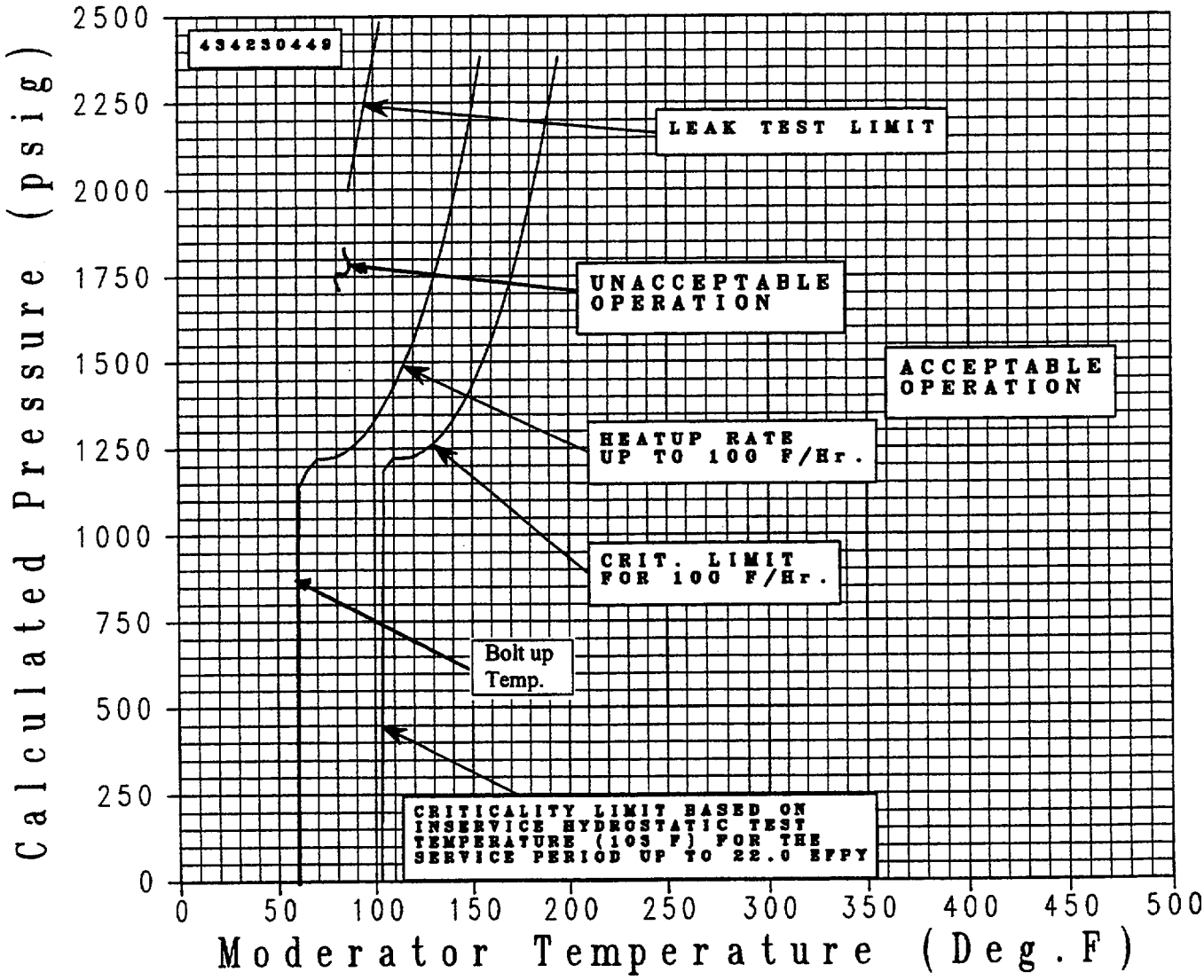

FIGURE 5-3 Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 22 EFPY: ¼T, 43°F
¾T, 31°F

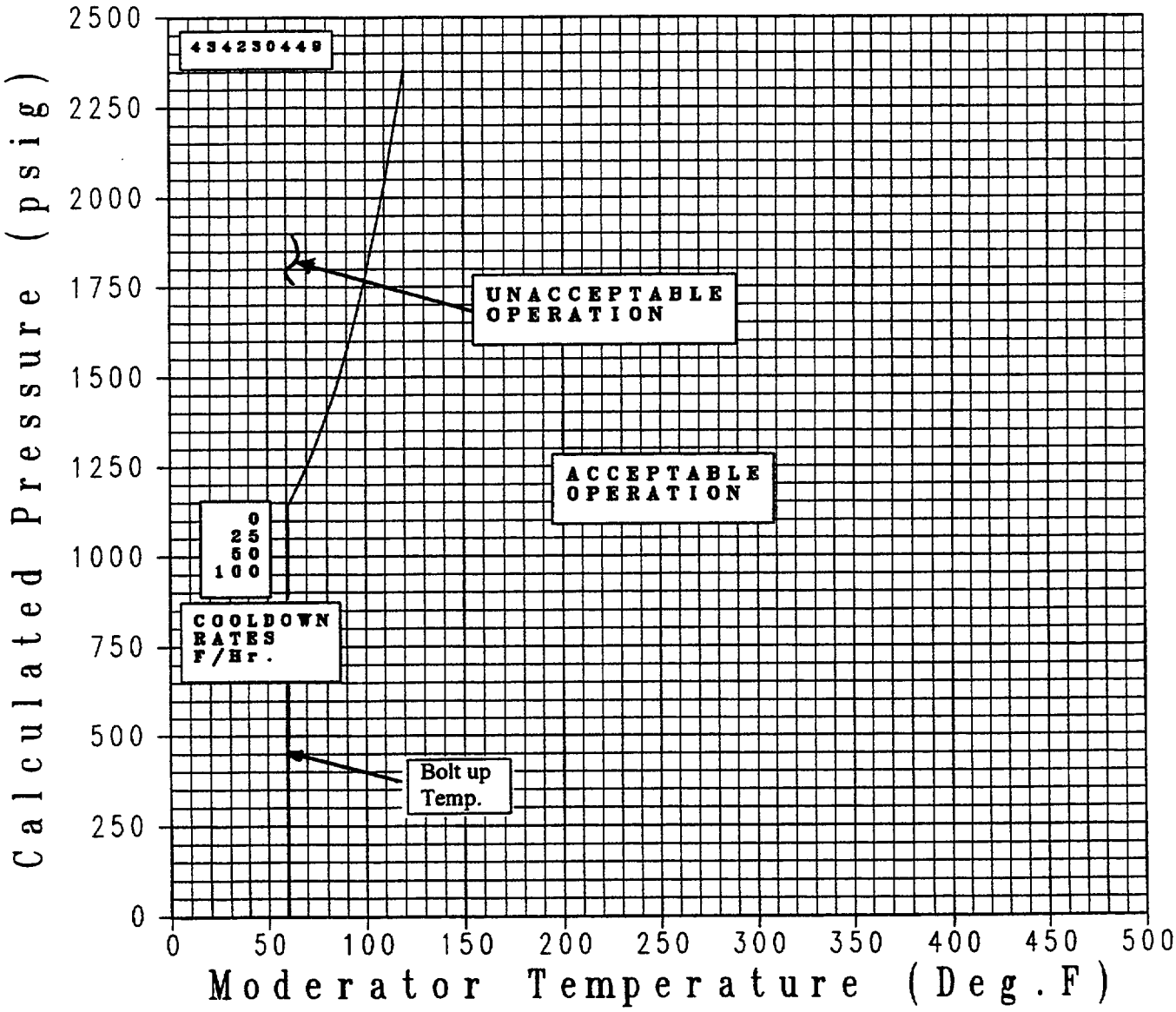

FIGURE 5-4 Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins of for Instrumentation Errors)

TABLE 5-2
Braidwood Unit 1 Heatup and Cooldown Curve Data Points for the 22 EFPY Curves
Using 1996 Appendix G and Code Case N-640 Methodology
(Without Margins of for Instrumentation Errors)

Run = 434230449		Cooldown Curves					
Steady State		25 Deg. °F/Hr		50 Deg. °F/Hr		100 Deg. °F/Hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	1133	60	1133	60	1133	60	1133
65	1188	65	1188	65	1188	65	1188
70	1250	70	1250	70	1250	70	1250
75	1318	75	1318	75	1318	75	1318
80	1393	80	1393	80	1393	80	1393
85	1476	85	1476	85	1476	85	1476
90	1568	90	1568	90	1568	90	1568
95	1669	95	1669	95	1669	95	1669
100	1781	100	1781	100	1781	100	1781
105	1905	105	1905	105	1905	105	1905
110	2042	110	2042	110	2042	110	2042
115	2194	115	2194	115	2194	115	2194
120	2361	120	2361	120	2361	120	2361

Run = 434230449	Heatup Curves					
	100 Deg. °F/Hr		Crit. Limit		Leak Test Limit	
	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
	60	0	103	0	86	2000
	60	1133	105	1188	103	2485
	65	1188	110	1221		
	70	1222	115	1221		
	75	1222	120	1229		
	80	1229	125	1243		
	85	1243	130	1264		
	90	1264	135	1294		
	95	1294	140	1331		
	100	1331	145	1376		
	105	1376	150	1430		
	110	1430	155	1492		
	115	1492	160	1563		
	120	1563	165	1644		
	125	1644	170	1736		
	130	1736	175	1838		
	135	1838	180	1954		
	140	1954	185	2082		
	145	2082	190	2225		
	150	2225	195	2384		
	155	2384				

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 32 EFPY: ¼T, 48°F
¾T, 35°F

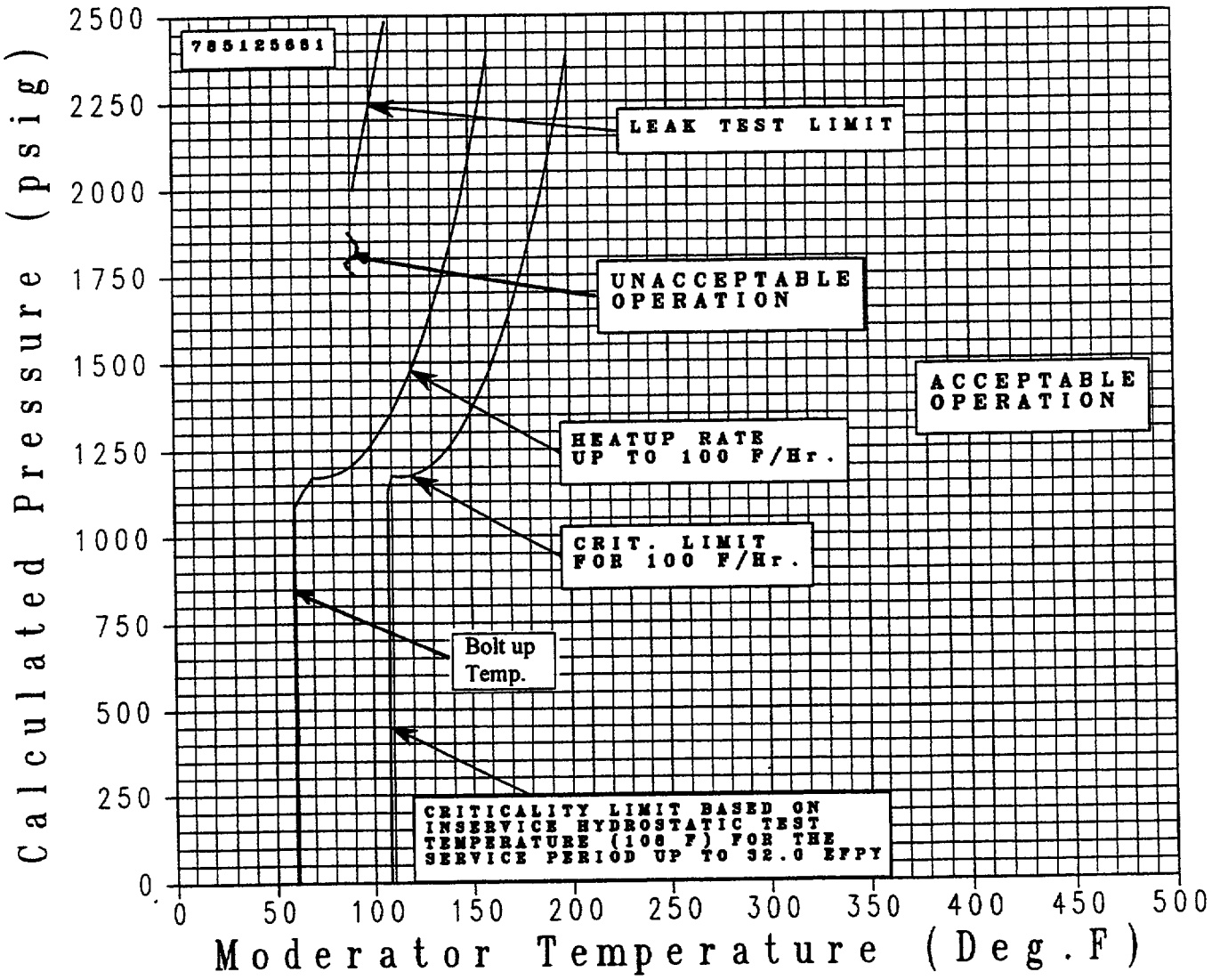

FIGURE 5-5 Braidwood Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: NOZZLE SHELL FORGING 5P-7016
LIMITING ART VALUES AT 32 EFPY: 1/4T, 48°F
3/4T, 35°F

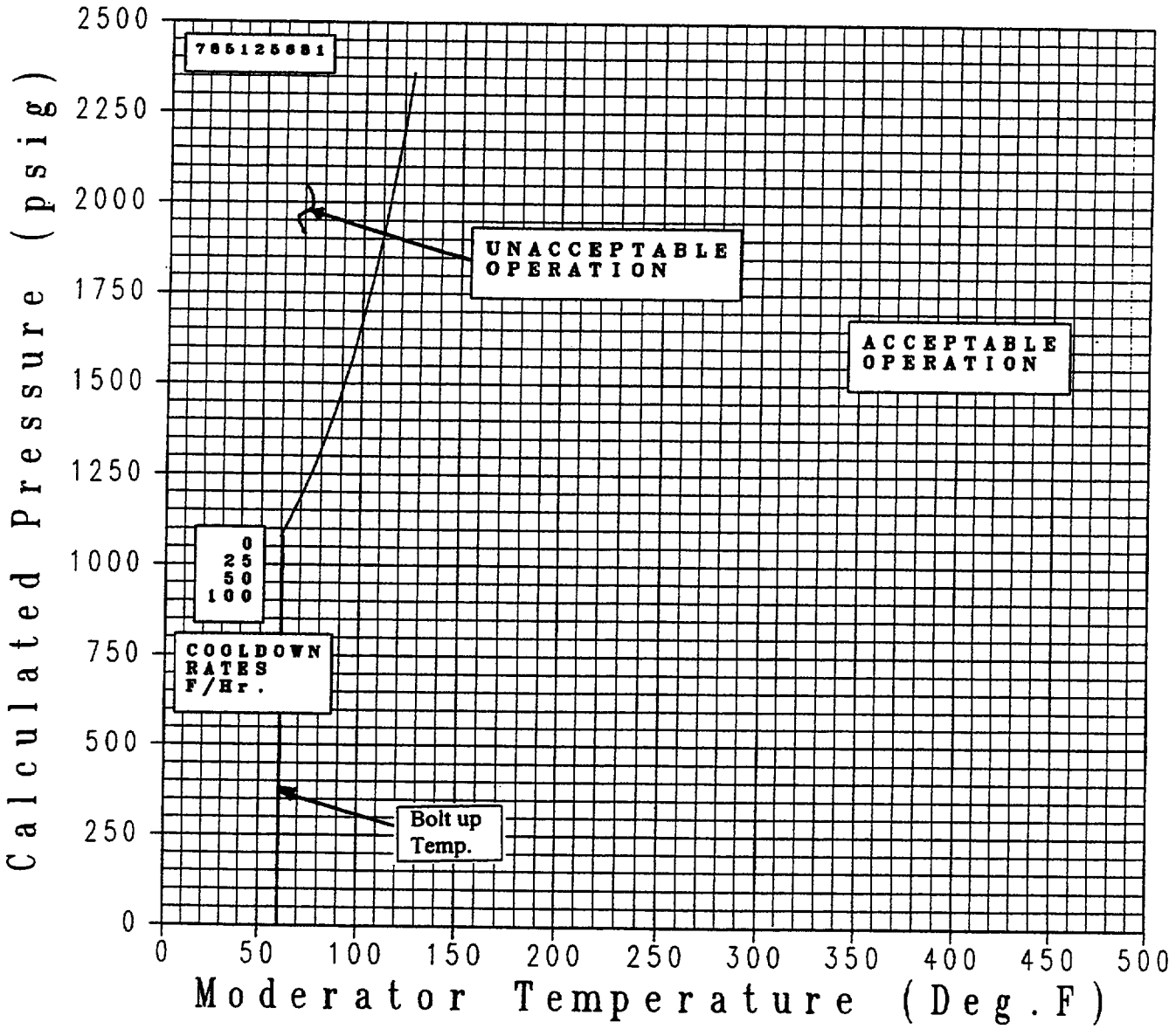

FIGURE 5-6 Braidwood Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G and Code Case N-640 Methodology (Without Margins of for Instrumentation Errors)

TABLE 5-5

Braidwood Unit 1 Heatup and Cooldown Curve Data Points for the 32 EFPY Curves Using 1996 Appendix G and Code Case N-640 Methodology

(Without Margins of for Instrumentation Errors)

Run = 785125681		Cooldown Curves					
Steady State		25 Deg. °F/Hr		50 Deg. °F/Hr		100 Deg. °F/Hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	1082	60	1078	60	1079		
65	1133						
70	1188						
75	1250						
80	1318						
85	1393						
90	1476						
95	1568						
100	1669						
105	1781						
110	1905						
115	2042						
120	2194						
125	2361						

Run = 785125681	Heatup Curves					
	100 Deg. °F/Hr		Crit. Limit		Leak Test Limit	
	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
	60	0	108	0	91	2000
	60	1082	108	1133	108	2485
	65	1133	110	1170		
	70	1172	115	1170		
	75	1172	120	1176		
	80	1176	125	1188		
	85	1188	130	1207		
	90	1207	135	1234		
	95	1234	140	1267		
	100	1267	145	1308		
	105	1308	150	1357		
	110	1357	155	1414		
	115	1414	160	1479		
	120	1479	165	1554		
	125	1554	170	1638		
	130	1638	175	1732		
	135	1732	180	1838		
	140	1838	185	1956		
	145	1956	190	2088		
	150	2088	195	2235		
	155	2235	200	2397		
	160	2397				

6 REFERENCES

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

2 Cases of ASME Boiler and Pressure Vessel Code, Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", Approved March 1999.

3 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", Federal Register, Volume 60, No. 243, dated December 19, 1995.

4 WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants", W. Bamford, et al., October 1999.

5 Cases of ASME Boiler and Pressure Vessel Code, Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels Section XI, Division 1", Approved December 12, 1997.

6 ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G, "Fracture Toughness Criteria for Protection Against Failure", December 1995.

7 WCAP-15366, "Commonwealth Edison Company Braidwood Unit 1 Surveillance Program Credibility Evaluation", E. Terek, March 2000.

8 WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J. D. Andrachek, et al., January 1996.

9 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15373

Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

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September 2000

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek

EXECUTIVE SUMMARY

The purpose of this report is to generate pressure-temperature limit curves for Braidwood Unit 2 for normal operation at 12, 16, 22 and 32 EFPY using the Following methodologies: WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-15315. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART). The 1/4T and 3/4T values are summarized in Tables 4-18 through 4-25 and were calculated using the circumferential weld WF-562, Heat 442011 (The limiting material for circumferentially oriented flaws, Code Case N-588) and nozzle shell forging 5P-7056 (The limiting material for axial flaws). The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all curves at each EFPY value evaluated. Hence, only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-8.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated 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 (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. 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 unirradiated RT_{NDT} (IRT_{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"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Braidwood Station Unit No. 2 reactor pressure vessel (Specifically from lower shell forging 50D102-1/50C97-1). Also supplied was a weldment made with weld wire heat # 442011 Linde 80 flux, lot number 0344, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company contracted Westinghouse to analyze surveillance capsule W from the Braidwood Unit 2 reactor vessel and perform an evaluation for a 5% Uprating. As a part of these analysis Westinghouse generated new heatup and cooldown curves for 12, 16, 22 and 32 EFPY. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- Regulatory Guide 1.99, Revision 2[1],
- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10 CFR Part 50[3] per WCAP-15315, "Reactor Vessel Head/Flange Requirements Evaluation for Operating PWR and BWR Plants"[4],
- ASME Code Case N-588[5] (where applicable),
- Methodology of the 1996 ASME B&P Vessel Code, Section XI, Appendix G,[6], and
- The PT Curves will be developed WITHOUT margins or instrumentation errors.

Based on the above methodologies, two sets of PT Curves will be generated. Set one will consist of the circumferential flaw methodology (ASME Code Case N-588) in combination with 1996 Appendix G to ASME Section XI and the K_{IC} methodology (ASME Code Case N-640) for the limiting circumferential weld material. Set two will consist of the 1996 Appendix G to ASME Section XI and the K_{IC} methodology (ASME Code Case N-640) for the limiting forging/base metal material. Both sets of curves will used the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (from WCAP-15315). The final PT curves to be presented herein will be the most limiting set of curves. If the situation arises where portions of each set of curves are limiting, then composite curves will be generated that are based on the most limiting data (i.e. Circ. Flaw or Axial Flaw Case).

The purpose of this report is to present the calculations and the development of the Commonwealth Edison Company Braidwood Unit 2 heatup and cooldown curves for 12, 16, 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Commonwealth Edison Company, the surveillance weld data from the Braidwood Unit 1 and Unit 2 surveillance programs has been integrated. Note that Braidwood Unit 1 surveillance weld is identical to the surveillance weld (Heat No. 442011) at Braidwood Unit 2. Per WCAP-15368[7], all the surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies 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_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[2, 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where, K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2 and SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im}	=	stress intensity factor caused by membrane (pressure) stress
K_{It}	=	stress intensity factor caused by the thermal gradients
K_{Ic}	=	function of temperature relative to the RT_{NDT} of the material
C	=	2.0 for Level A and Level B service limits
C	=	1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

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

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

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

$$M_m = 1.85 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.926\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 3.21 \text{ for } \sqrt{t} > 3.464$$

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

$$M_m = 1.77 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.893\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 3.09 \text{ for } \sqrt{t} > 3.464$$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

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

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

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

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

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

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

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

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

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G 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 wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

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 ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} 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 may unknowingly be violated 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 defect 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_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. 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.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. 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 the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. 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 possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

Code Case N-588: Circumferential Welds:

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

The earlier ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds.

In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

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

G-2214.1 Membrane Tension...

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

$$K_{Im} = M_m \times (pR_i / t)$$

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

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

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

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note again, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circ. weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[3] 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 when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which is 621 psig for Braidwood Unit 2 reactor vessel.

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

Improved knowledge of the fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{IC} in development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[2].

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants"[4], concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using K_{IC} toughness. Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region. It is therefore clear that no additional boltup requirements are necessary, and the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\,RT_{NDT} + \Delta RT_{NDT} + Margin \qquad (7)$$

Initial RT_{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[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may 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 adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (9)$$

where x inches (vessel beltline thickness is 8.5 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections and the results are presented in Section 6 of WCAP-15369[10] and WCAP-15316, Rev. 1[11]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Tables 4-2 through 4-5, herein, contain the calculated pak vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Braidwood Unit 2 reactor vessel. Additionally, the Braidwood Unit 2 calculated surveillance capsule fluence values are presented in Table 4-6.

Ratio Procedure and Temperature Adjustment:

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2, Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the measured ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted measured ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12th and 13th of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradition damage... To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT,\,measured}$ to T_{plant} is made as follows:
>
> $$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\,measured} + 1.0*(T_{capsule} - T_{plant}) \qquad (10)$$

Per Reference 12, page C-5, following are the average operating inlet temperatures of the Braidwood Unit 1 and 2 reactor vessels along with when the capsules were removed.

TABLE 4-1
Irradiation Temperature of the Braidwood Unit 1 and 2 Reactor Vessels

Fuel Cycle	Braidwood 1	Capsule	Braidwood 2	Capsule
1	557°F	U	557°F	U
2	551°F	- -	551°F	- -
3	551°F	- -	551°F	- -
4	551°F	X	551°F	X
5	551-554°F	- -	551°F*	- -
6	554°F	- -	551°F*	- -
7	554°F*	W	550°F**	W
Average Temp.	553°F		552°F	

* Information provided by ComEd in NDIT No. BRW-DIT-97-321[13].

** Information provided by ComEd in NDIT No. BRW-DIT-2000-0010[14].

Capsules U and X were exposed to the same average operating temperature of 553°F. Capsule W was exposed to the same average operating temperature of 553°F for 5 of the 7 cycles. For cycles 6 and 7, capsule W saw only a difference of 1°F in average operating temperature. Thus, the inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest and no temperature adjustments are required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-8 through 4-10. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 7, the surveillance weld data for Braidwood Unit 2 is credibile while the surveillance forging material is non-credible. In addition, Reference 7 also shows that the Table chemistry factor is non-conservative and the surveillance chemistry factor should be used with a full margin term. This assessment also calculated the vessel weld (including temperature and chemistry adjustment) chemistry factor using Braidwood Unit 2 stand alone data and Braidwood Unit 1 and 2 combined data. The chemistry factor was determined to be 25.7°F. Position 2.1 chemistry factors are calculated in Table 4-12.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Braidwood Unit 2, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}\ .$$

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Braidwood Unit 2, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I .

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 8.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 12 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	7.43×10^{18}	4.46×10^{18}	1.61×10^{18}
Lower Shell Forging 50D102-1/50C97-1	7.43×10^{18}	4.46×10^{18}	1.61×10^{18}
Nozzle Shell Forging 5P-7056	2.16×10^{18}	1.30×10^{18}	4.68×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	7.20×10^{18}	4.32×10^{18}	1.56×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	2.16×10^{18}	1.30×10^{18}	4.68×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-3

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 16 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	9.87×10^{18}	5.93×10^{18}	2.14×10^{18}
Lower Shell Forging 50D102-1/50C97-1	9.87×10^{18}	5.93×10^{18}	2.14×10^{18}
Nozzle Shell Forging 5P-7056	2.86×10^{18}	1.72×10^{18}	6.19×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	9.54×10^{18}	5.73×10^{18}	2.07×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	2.86×10^{18}	1.72×10^{18}	6.19×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-4

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	1.35×10^{19}	8.11×10^{18}	2.92×10^{18}
Lower Shell Forging 50D102-1/50C97-1	1.35×10^{19}	8.11×10^{18}	2.92×10^{18}
Nozzle Shell Forging 5P-7056	3.91×10^{18}	2.35×10^{18}	8.47×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.31×10^{19}	7.87×10^{18}	2.84×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	3.91×10^{18}	2.35×10^{18}	8.47×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-5

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 49D963-1/49C904-1	1.96×10^{19}	1.18×10^{19}	4.24×10^{18}
Lower Shell Forging 50D102-1/50C97-1	1.96×10^{19}	1.18×10^{19}	4.24×10^{18}
Nozzle Shell Forging 5P-7056	5.67×10^{18}	3.40×10^{18}	1.23×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.89×10^{19}	1.13×10^{19}	4.09×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	5.67×10^{18}	3.40×10^{18}	1.23×10^{18}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-6
Calculated Integrated Neutron Exposure of the Braidwood Unit 2
Surveillance Capsules Tested to Date

Capsule	Fluence
U	4.00×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.23×10^{19} n/cm^2, (E > 1.0 MeV)
W	2.25×10^{19} n/cm^2, (E > 1.0 MeV)

Contained in Table 4-7 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[15], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-7

Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[(a)]
Lower Shell Forging 50D102-1/50C97-1 (Tangential Orientation)	U	-9.73
	X	-9.42
	W	4.53
Lower Shell Forging 50D102-1/50C97-1 (Axial Orientation)	U	-0.13
	X	33.94
	W	33.2
Surveillance Program Weld Metal (Heat # 442011)	U	-0.58
	X	26.3
	W	23.9
Heat Affected Zone	U	-34.45
	X	-9.54
	W	4.03

Notes:

(a) From capsule W analysis results[10].

Table 4-8 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 2 base materials in the beltline region. Table 4-9 contains the calculation of the best estimate weight percent copper and nickel for the Braidwood Unit 2 surveillance weld material, while Table 4-10 presents the overall best estimate average for that heat of weld. Table 4-11 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-11 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-13. Table 4-12 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-13.

TABLE 4-8

Calculation of the Best Estimate Cu and Ni Weight Percent for the Braidwood Unit 2 Forging Materials

Reference	Intermediate Shell Forging 49D963-1/49C904-1		Lower Shell Forging 50D102-1/50C97-1	
	Cu %	Ni %	Cu %	Ni %
WCAP-11188[16]	0.03	0.71	---	---
WCAP-11188[16]	---	---	0.06	0.75
WCAP-11188[16]	---	---	0.057	0.77
Charpy FL-6[17]	---	---	0.049	0.745
Ref. 18			0.056	0.804
Charpy FL-33[10]	---	---	0.065	0.680
Charpy FL-42[10]	---	---	0.065	0.690
Charpy FT-43[10]	---	---	0.060	0.795
Charpy FT-45[10]	---	---	0.066	0.840
Best Estimate Average(a)	0.03	0.71	0.06	0.76

Note:

(a) The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-9

Calculation of the Average Cu and Ni Weight Percent for the Braidwood Unit 2 Surveillance Weld Material Only (Heat # 442011)

Reference	Weight % Copper	Weight % Nickel
WCAP-14824, Rev.2(a)	0.033	0.708
Charpy FW-38(b)	0.044	0.703
Charpy FW-34(b)	0.036	0.774
Charpy FW-41(b)	0.037	0.747
Charpy FW-43(b)	0.042	0.670
Surveillance Weld Average(c)	0.03	0.71

Note:

(a) This is the average of 32 data points.

(b) Charpy Specimens From Capsule W of Braidwood Unit 2 (Ref. 10).

(c) The Surveillance weld average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-10

Calculation of Best Estimate Cu and Ni Weight Percent Values for the Braidwood Units 1 & 2 Weld Material (Using Braidwood 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
B&W WQ: BAW-2261	Ref. 12	0.028	0.63
B&W WQ: BAW-2261	Ref. 12	0.03	0.65
B&W WQ: BAW-2261	Ref. 12	0.04	0.67
Braidwood Unit 2 Surv. Data Ave.[a]	Table 4-5	0.034	0.710
Braidwood Unit 1 Surv. Data Ave.[b]	Ref. 11	0.032	0.671
BEST ESTIMATE AVERAGE		0.03[c]	0.67[c]

NOTES:

(a) The weld material in the Braidwood Unit 2 surveillance program was made of the same wire and flux type as the reactor vessel inter. to lower shell girth seam weld (Weld seam WF-562, Wire Heat No. 442011, Flux Type Linde 80, Flux Lot No. 8061). The surveillance weld flux lot # is 0344.

(b) The Braidwood Unit 1 surveillance weld is identical to that used in the Braidwood Unit 2 reactor vessel core region girth seam (WF-562) heat number 442011, with a Linde 80 type flux, lot number 8061.

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-11

Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 3P6566/5P7547/4P6986	---	0.75	20
Vessel Flange 124P455	0.07	0.70	20
Nozzle Shell Forging 5P7056	0.04	0.90	30
Intermediate Shell Forging 49D963-1/49C904-1	0.03	0.71	-30
Lower Shell Forging 50D102-1 / 50C97-1	0.06	0.76	-30
Inter. to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	0.03	0.67	40
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	0.04	0.46	-25
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03, 0.03	0.67, 0.71	- - -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

TABLE 4-12

Calculation of Chemistry Factors for Braidwood Unit 2 using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging	U	0.400	0.746	0.0[e]	0	0.557
50D102-1/50C97-1	X	1.23	1.058	0.0[e]	0	1.119
(Tangential)	W	2.25	1.220	4.53	5.53	1.488
Lower Shell Forging	U	0.400	0.746	0.0[e]	0	0.557
50D102-1/50C97-1	X	1.23	1.058	33.94	35.91	1.119
(Axial)	W	2.25	1.220	33.2	40.50	1.488
				SUM:	81.94	6.328
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (81.94) \div (6.328) =$ **12.9°F**					
Braidwood Unit 1	U	0.387	0.737	17.06[d]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[d]	31.96	1.124
Weld Metal	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[d, e]	0	0.557
Surveillance	X	1.23	1.058	26.3[d]	27.83	1.119
Weld Metal	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) =$ **25.7°F**					

Notes:

(a) f = Calculated fluence from the Braidwood Unit 2 capsule W dosimetry analysis results [10],the Braidwood Unit 1 calculated fluences are from capsule W analysis[11], (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 2 taken from WCAP-15369[10]. Braidwood Unit 1 values are from WCAP-15316 Rev 1 (Ref. 11).

(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted (i.e. the ratio factor is 1.0).

(e) Actual values of ΔRT_{NDT} are -9.73 (Cap U Tang.), -9.42 (Cap. X Tang.), -0.13 (Cap. U Axial), -0.58 (Cap U Weld). This physically should not occur, therefore for conservatism (i.e. higher chemistry factor) a value of zero will be used.

TABLE 4-13

Summary of the Braidwood Unit 2 Reactor Vessel Beltline Material Chemistry Factors Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	**Chemistry Factor**	
	Position 1.1	**Position 2.1**
Intermediate Shell Forging 49D963-1/49C904-1	20.0°F	- - -
Lower Shell Forging 50D102-1/50C97-1	37.0°F	12.9°F
Nozzle Shell Forging 5P7056	26.0°F	- - -
Intermediate Shell to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	41.0°F	25.7°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	54.0°F	- - -
Braidwood Unit 1 & 2 Surveillance Program Weld Metal	41.0°F	- - -

Contained in Tables 4-14 through 4-17 is the summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Braidwood Unit 2 reactor vessel beltline materials for 12, 16, 22 and 32 EFPY.

TABLE 4-14

Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 12 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	4.46×10^{18}	0.775	1.61×10^{18}	0.519
Lower Shell Forging 50D102-1/50C97-1	4.46×10^{18}	0.775	1.61×10^{18}	0.519
Nozzle Shell Forging 5P-7056	1.30×10^{18}	0.471	4.68×10^{17}	0.282
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	4.32×10^{18}	0.767	1.56×10^{18}	0.512
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	1.30×10^{18}	0.471	4.68×10^{17}	0.282

TABLE 4-15

Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 16 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	5.93×10^{18}	0.854	2.14×10^{18}	0.586
Lower Shell Forging 50D102-1/50C97-1	5.93×10^{18}	0.854	2.14×10^{18}	0.586
Nozzle Shell Forging 5P-7056	1.72×10^{18}	0.534	6.19×10^{17}	0.328
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	5.73×10^{18}	0.844	2.07×10^{18}	0.578
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	1.72×10^{18}	0.534	6.19×10^{17}	0.328

TABLE 4-16

Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 22 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	8.11×10^{18}	0.941	2.92×10^{18}	0.663
Lower Shell Forging 50D102-1/50C97-1	8.11×10^{18}	0.941	2.92×10^{18}	0.663
Nozzle Shell Forging 5P-7056	2.35×10^{18}	0.609	8.47×10^{17}	0.384
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	7.87×10^{18}	0.933	2.84×10^{18}	0.656
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	2.35×10^{18}	0.609	8.47×10^{17}	0.384

TABLE 4-17

Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Material	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 49D963-1/49C904-1	1.18×10^{19}	1.046	4.24×10^{18}	0.762
Lower Shell Forging 50D102-1/50C97-1	1.18×10^{19}	1.046	4.24×10^{18}	0.762
Nozzle Shell Forging 5P-7056	3.40×10^{18}	0.703	1.23×10^{18}	0.460
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	1.13×10^{19}	1.034	4.09×10^{18}	0.752
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	3.40×10^{18}	0.703	1.23×10^{18}	0.460

Contained in Tables 4-18 through 4-25 are the calculations of the ART values used for the generation of the 12, 16, 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-18

Calculation of the ART Values for the 1/4T Location @ 12 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 12[a] EFPY	¼-t f ($\times 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.743	0.446	0.775	-30	15.5	0	7.75	15.5	1
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.743	0.446	0.775	-30	28.7	0	14.35	28.7	27
Lower shell Forging → using S/C Data				12.9	0.743	0.446	0.775	-30	10.0	0	17.0	34.0	14
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.720	0.432	0.767	40	31.4	0	15.7	31.4	103
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.720	0.432	0.767	40	19.7	0	9.85	19.7	79
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.216	0.130	0.471	30	12.2	0	6.1	12.2	54
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.216	0.130	0.471	-25	25.4	0	12.7	25.4	26

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-19

Calculation of the ART Values for the 3/4T Location @ 12 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 12[a] EFPY	¾-t f ($\times 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.743	0.161	0.519	-30	10.4	0	5.2	10.4	-9
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.743	0.161	0.519	-30	19.2	0	9.6	19.2	8
Lower shell Forging → using S/C Data				12.9	0.743	0.161	0.519	-30	6.7	0	17.0	34.0	11
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.720	0.156	0.512	40	21.0	0	10.5	21.0	82
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.720	0.156	0.512	40	13.2	0	6.6	13.2	**66**
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.216	0.0468	0.282	30	7.3	0	3.65	7.3	**45**
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.216	0.0468	0.282	-25	15.2	0	7.6	15.2	5

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-20

Calculation of the ART Values for the 1/4T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 16[a] EFPY	¼-t f ($\times 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.987	0.593	0.854	-30	17.1	0	8.55	17.1	4
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.987	0.593	0.854	-30	31.6	0	15.8	31.6	33
Lower shell Forging → using S/C Data				12.9	0.987	0.593	0.854	-30	11.0	0	17.0	34.0	15
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.954	0.573	0.844	40	34.6	0	17.0	34.0	109
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.954	0.573	0.844	40	21.7	0	10.85	21.7	**83**
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.286	0.172	0.534	30	13.9	0	6.69	13.9	**58**
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.286	0.172	0.534	-25	28.8	0	14.4	28.8	33

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-21

Calculation of the ART Values for the 3/4T Location @ 16 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 16[a] EFPY	¾-t f ($\times 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	0.987	0.214	0.586	-30	11.7	0	5.85	11.7	-7
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	0.987	0.214	0.586	-30	21.7	0	10.85	21.7	13
Lower shell Forging → using S/C Data				12.9	0.987	0.214	0.586	-30	7.6	0	17.0	34.0	12
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.954	0.207	0.578	40	23.7	0	11.85	23.7	87
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	0.954	0.207	0.578	40	14.9	0	7.45	14.9	**70**
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.286	0.0619	0.328	30	8.5	0	4.25	8.5	**47**
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.286	0.0619	0.328	-25	17.7	0	8.85	17.7	10

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-22

Calculation of the ART Values for the 1/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY	¼-t f ($\times 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.35	0.811	0.941	-30	18.8	0	9.4	18.8	8
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.35	0.811	0.941	-30	34.8	0	17.0	34.0	39
Lower shell Forging → using S/C Data				12.9	1.35	0.811	0.941	-30	12.1	0	17.0	34.0	16
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.31	0.787	0.933	40	38.3	0	19.15	38.3	117
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.31	0.787	0.933	40	24.0	0	12.0	24.0	**88**
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.391	0.235	0.609	30	15.8	0	7.9	15.8	**62**
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.391	0.235	0.609	-25	32.9	0	16.45	32.9	41

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-23

Calculation of the ART Values for the 3/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY	¾-t f ($\times 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.35	0.292	0.663	-30	13.3	0	6.65	13.3	-3
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.35	0.292	0.663	-30	24.5	0	12.25	24.5	19
Lower shell Forging → using S/C Data				12.9	1.35	0.292	0.663	-30	8.6	0	17.0	34.0	13
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.31	0.284	0.656	40	26.9	0	13.45	26.9	94
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.31	0.284	0.656	40	16.9	0	8.45	16.9	**74**
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.391	0.0847	0.384	30	10.0	0	5.0	10.0	**50**
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.391	0.0847	0.384	-25	20.7	0	10.35	20.7	16

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-24

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY	¼-t f ($\times 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.96	1.18	1.046	-30	20.9	0	10.45	20.9	12
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.96	1.18	1.046	-30	38.7	0	17.0	34.0	43
Lower shell Forging → using S/C Data				12.9	1.96	1.18	1.046	-30	13.5	0	17.0	34.0	18
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.89	1.13	1.034	40	42.4	0	21.2	42.4	125
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.89	1.13	1.034	40	26.6	0	13.3	26.6	93
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.567	0.340	0.703	30	18.3	0	9.15	18.3	67
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.567	0.340	0.703	-25	38.0	0	19.0	38.0	51

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

TABLE 4-25

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY	¾-t f ($\times 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	49D963-1/ 49C904-1	0.03	0.71	20.0	1.96	0.424	0.762	-30	15.2	0	7.6	15.2	0
Lower Shell Forging	50D102-1/ 50C97-1	0.06	0.76	37.0	1.96	0.424	0.762	-30	28.2	0	14.1	28.2	26
Lower shell Forging → using S/C Data				12.9	1.96	0.424	0.762	-30	9.8	0	17.0	34.0	14
Inter. to Lower Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	1.89	0.409	0.752	40	30.8	0	15.4	30.8	102
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				25.7	1.89	0.409	0.752	40	19.3	0	9.65	19.3	79
Nozzle Shell Forging	5P-7056	0.04	0.90	26.0	0.567	0.123	0.460	30	12.0	0	6.0	12.0	54
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-645	0.04	0.46	54.0	0.567	0.123	0.460	-25	24.8	0	12.4	24.8	25

NOTES:

(a) Fluence, f, is the calculated peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV).

(b) $ART = I + \Delta RT_{NDT} + M$ (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) $\Delta RT_{NDT} = CF * FF$

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Braidwood 1 and 2 Welds (WF-562, heat # 442011).

The girth weld WF-562 and the nozzle shell forging 5P-7056 are the limiting beltline materials for all heatup and cooldown curves to be generated. The ART value associated with these materials will be used in all four sets of curves. The girth weld ART will be used when generating curves for Code Case N-588 (ie. Circ. Flaw). The ART associated with the limiting axial material must also be considered to determine if this case would be more conservative or overlap the circ. flaw curves. Contained in Tables 4-26 through 4-29 is a summary of the limiting ARTs to be used in the generation of the Braidwood Unit 2 reactor vessel heatup and cooldown curves.

TABLE 4-26

Summary of Adjusted Reference Temperature (ART) values at the1/4T and 3/4T Locations for 12 EFPY

Material	**12 EFPY**	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging 49D963-1/49C904-1	1	-9
Lower Shell Forging 50D102-1/50C97-1	27	8
- Using Surveillance Data	14	11
Circumferential Weld WF-562	103	82
- Using Surveillance Data	**79[a]**	**66[a]**
Nozzle Shell Forging 5P-7056	**54[b]**	**45[b]**
Circumferential Weld WF-645	26	5

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-1 and 5-2. See note (b).

(b) These ART values, using the '96 App. G Methodology produced a more conservative curve, with no overlap, than the curves with the circ. flaw ART values using Code Case N-588 Methodology.

TABLE 4-27

Summary of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T Locations for 16 EFPY

Material	16 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 49D963-1/49C904-1	4	-7
Lower Shell Forging 50D102-1/50C97-1	33	13
- Using Surveillance Data	15	12
Circumferential Weld WF-562	109	87
- Using Surveillance Data	**83[a]**	**70[a]**
Nozzle Shell Forging 5P-7056	**58[b]**	**47[b]**
Circumferential Weld WF-645	33	10

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-3 and 5-4. See note (b).

(b) These ART values, using the '96 App. G Methodology produced a more conservative curve, with no overlap, than the curves with the circ. flaw ART values using Code Case N-588 Methodology.

TABLE 4-28

Summary of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T Locations for 22 EFPY

Material	22 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 49D963-1/49C904-1	8	-3
Lower Shell Forging 50D102-1/50C97-1	39	19
- Using Surveillance Data	16	13
Circumferential Weld WF-562	117	94
- Using Surveillance Data	**88**[(a)]	**74**[(a)]
Nozzle Shell Forging 5P-7056	**62**[(b)]	**50**[(b)]
Circumferential Weld WF-645	41	16

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-5 and 5-6. See note (b).

(b) These ART values, using the '96 App. G Methodology produced a more conservative curve, with no overlap, than the curves with the circ. flaw ART values using Code Case N-588 Methodology.

TABLE 4-29

Summary of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T Locations for 32 EFPY

Material	32 EFPY	
	1/4T ART	3/4T ART
Intermediate Shell Forging 49D963-1/49C904-1	12	0
Lower Shell Forging 50D102-1/50C97-1	43	26
- Using Surveillance Data	18	14
Circumferential Weld WF-562	125	102
- Using Surveillance Data	93[a]	79[a]
Nozzle Shell Forging 5P-7056	67[b]	54[b]
Circumferential Weld WF-645	51	25

NOTES:

(a) These ART values were used to generate the Braidwood Unit 2 heatup and cooldown curves in Figures 5-7 and 5-8. See note (b).

(b) These ART values, using the '96 App. G Methodology produced a more conservative curve, with no overlap, than the curves with the circ. flaw ART values using Code Case N-588 Methodology.

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-8 present the 12, 16, 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology[6] and Code Case N-588[5], respectively. The heatup and cooldown curves that are presented herein are actually curves generated using the 1996 App. G methodology with the *lower axial flaw ART value*. The reason is, these curves are more conservative than the curves generated using Code Case N-588 methodology with the *higher circ. flaw ART value*. This is true throughout the entire temperature range, including criticality.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-8. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1, 5-3, 5-5 and 5-7 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[2] (approved in February 1999) as follows:

$$1.5K_{Im} < K_{Ic} \qquad (11)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RT_{NDT})]}$,
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must 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 calculated as described in Section 3 of this report. The minimum temperatures for the inservice hydrostatic leak test for the Braidwood Unit 2 reactor vessel at 12, 16, 22 and 32 EFPY are 114°F, 118°F, 122°F and 127°F at 2485 psig 1996 App. G Methodology. The vertical line drawn from

these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-8 define all of the above limits for ensuring prevention of nonductile failure for the Braidwood Unit 2 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-8 are presented in Tables 5-1 through 5-8, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 12 EFPY: 1/4T, 79°F (N-588) & 54°F ('96 App. G)

3/4T, 66°F (N-588) & 45°F ('96 App. G)

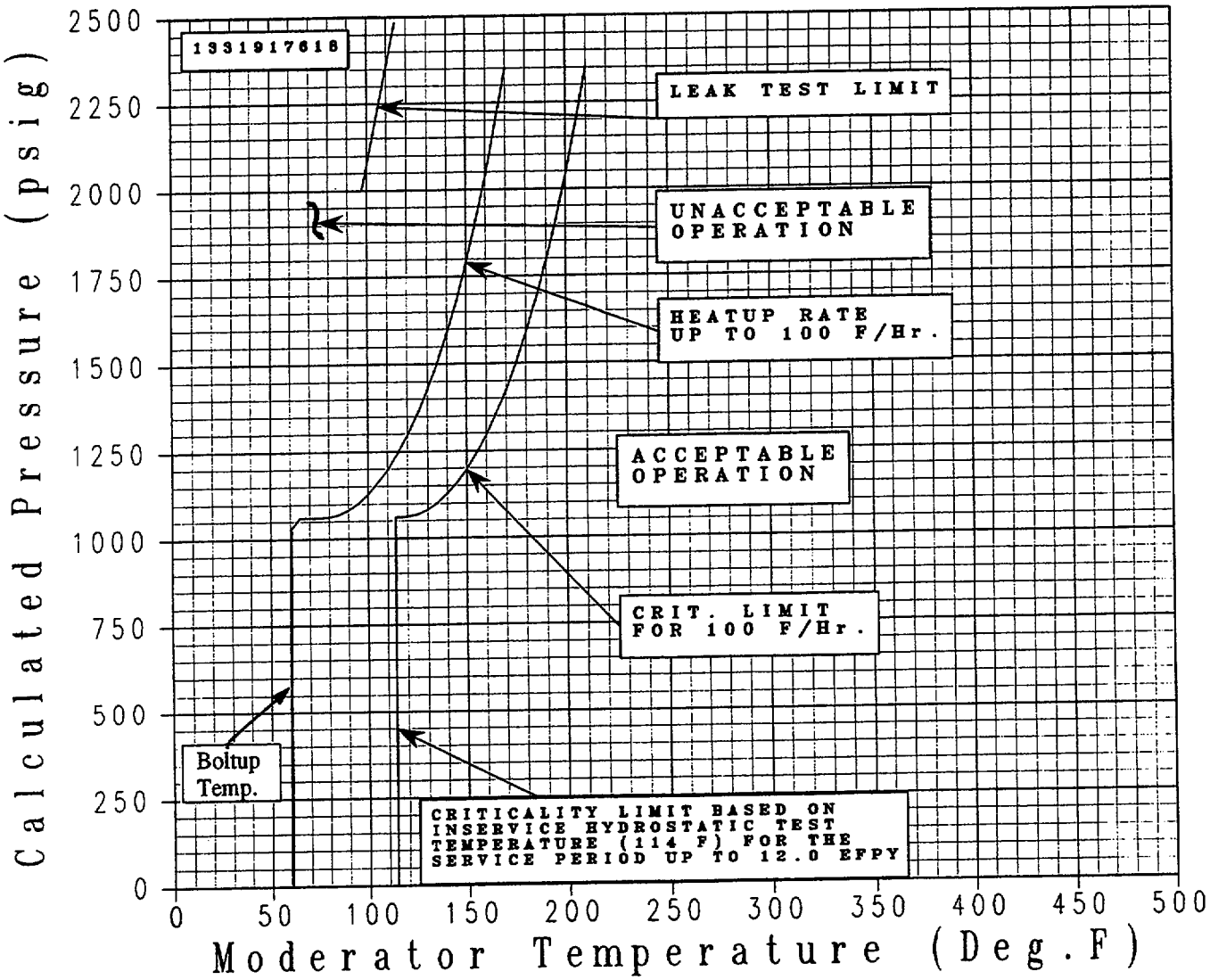

FIGURE 5-1 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 12 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 12 EFPY: 1/4T, 79°F (N-588) & 54°F ('96 App. G)

3/4T, 66°F (N-588) & 45°F ('96 App. G)

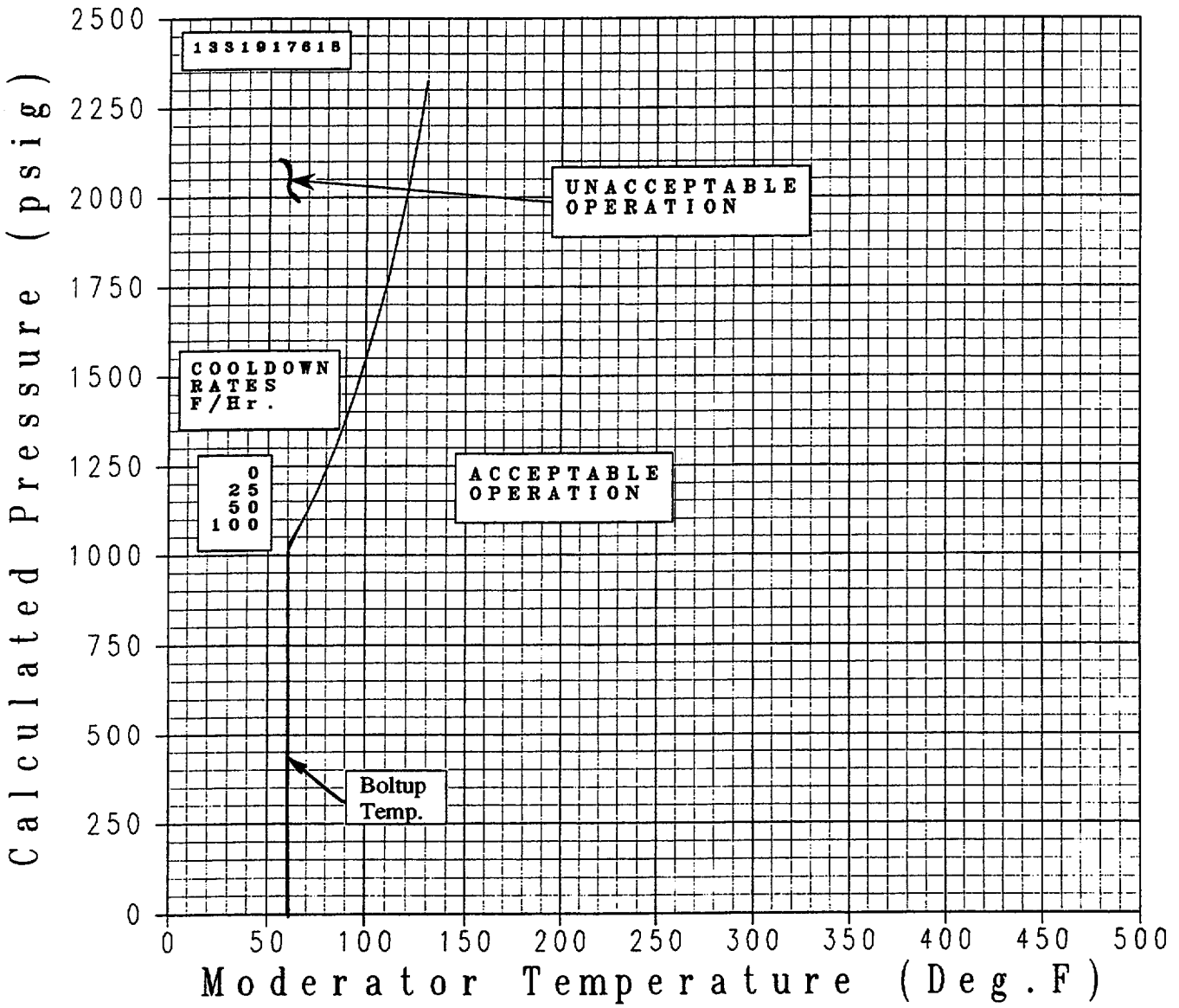

FIGURE 5-2 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 12 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

TABLE 5-1

Braidwood Unit 2 Heatup Data at 12 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	114	0	97	2000
60	1028	114	1028	114	2485
65	1061	114	1061		
70	1061	115	1061		
75	1061	120	1062		
80	1062	125	1069		
85	1069	130	1083		
90	1083	135	1103		
95	1103	140	1128		
100	1128	145	1161		
105	1161	150	1199		
110	1199	155	1244		
115	1244	160	1297		
120	1297	165	1357		
125	1357	170	1425		
130	1425	175	1501		
135	1501	180	1587		
140	1587	185	1683		
145	1683	190	1791		
150	1791	195	1910		
155	1910	200	2042		
160	2042	205	2189		
165	2189	210	2352		
170	2352				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-2

Braidwood Unit 2 Cooldown Data at 12 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	1028	60	1017	60	1011	60	1022
65	1073	65	1067	65	1067		
70	1122	70	1122				
75	1177						
80	1237						
85	1304						
90	1377						
95	1459						
100	1549						
105	1648						
110	1758						
115	1880						
120	2014						
125	2162						
130	2326						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. . Thus, only the '96 App. G Axial Flaw results are presented in this Report.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 16 EFPY: 1/4T, 83°F (N-588) & 58°F ('96 App. G)

3/4T, 70°F (N-588) & 47°F ('96 App. G)

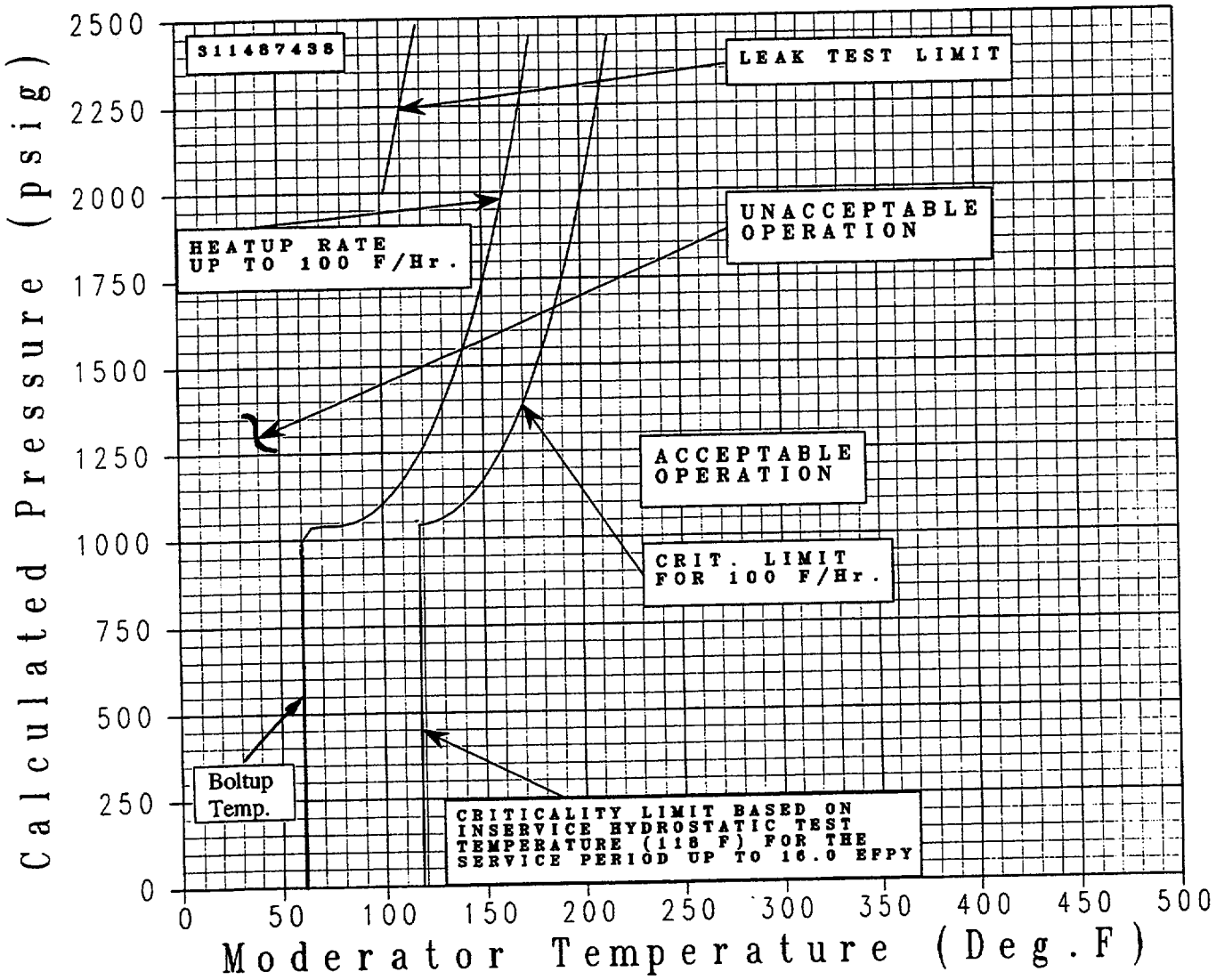

FIGURE 5-3 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 16 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 16 EFPY: 1/4T, 83°F (N-588) & 58°F ('96 App. G)
3/4T, 70°F (N-588) & 47°F ('96 App. G)

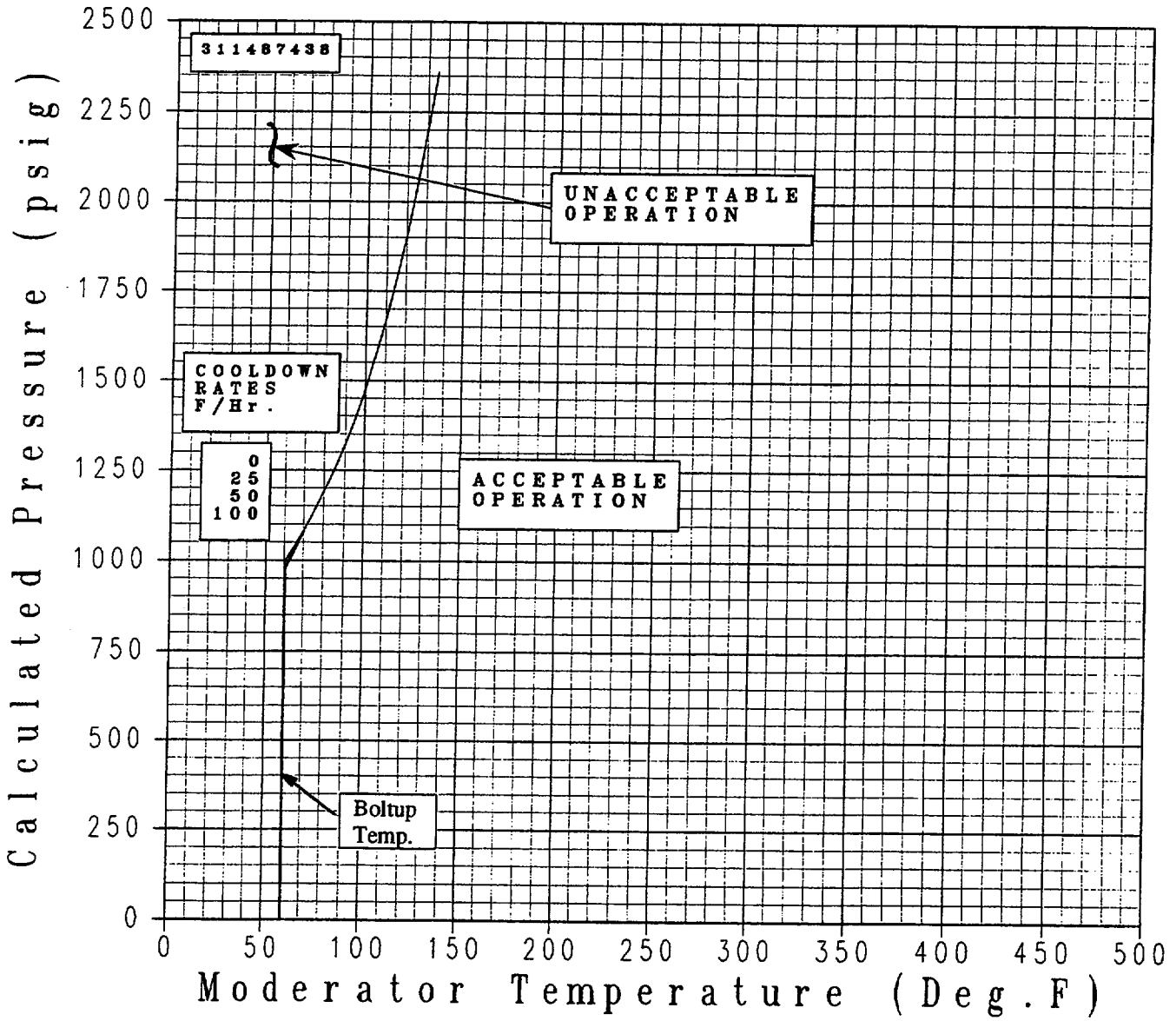

FIGURE 5-4 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 16 EFPY Code Case N-588 vs. 1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

TABLE 5-3
Braidwood Unit 2 Heatup Data at 16 EFPY Using Code Case N-588 vs. 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	118	0	101	2000
60	995	118	1037	118	2485
65	1037	118	1042		
70	1042	120	1042		
75	1042	125	1048		
80	1042	130	1061		
85	1048	135	1079		
90	1061	140	1104		
95	1079	145	1134		
100	1104	150	1171		
105	1134	155	1214		
110	1171	160	1264		
115	1214	165	1322		
120	1264	170	1387		
125	1322	175	1460		
130	1387	180	1543		
135	1460	185	1635		
140	1543	190	1738		
145	1635	195	1852		
150	1738	200	1980		
155	1852	205	2121		
160	1980	210	2277		
165	2121	215	2450		
170	2277				
175	2450				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. . Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-4

Braidwood Unit 2 Cooldown Data at 16 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	995	60	980	60	970	60	970
65	1037	65	1026	65	1022		
70	1082	70	1077				
75	1133						
80	1188						
85	1250						
90	1318						
95	1393						
100	1476						
105	1568						
110	1669						
115	1781						
120	1905						
125	2042						
130	2194						
135	2361						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. . Thus, only the '96 App. G Axial Flaw results are presented in this Report.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 22 EFPY 1/4T, 88°F (N-588) & 62°F ('96 App. G)

3/4T, 74°F (N-588) & 50°F ('96 App. G)

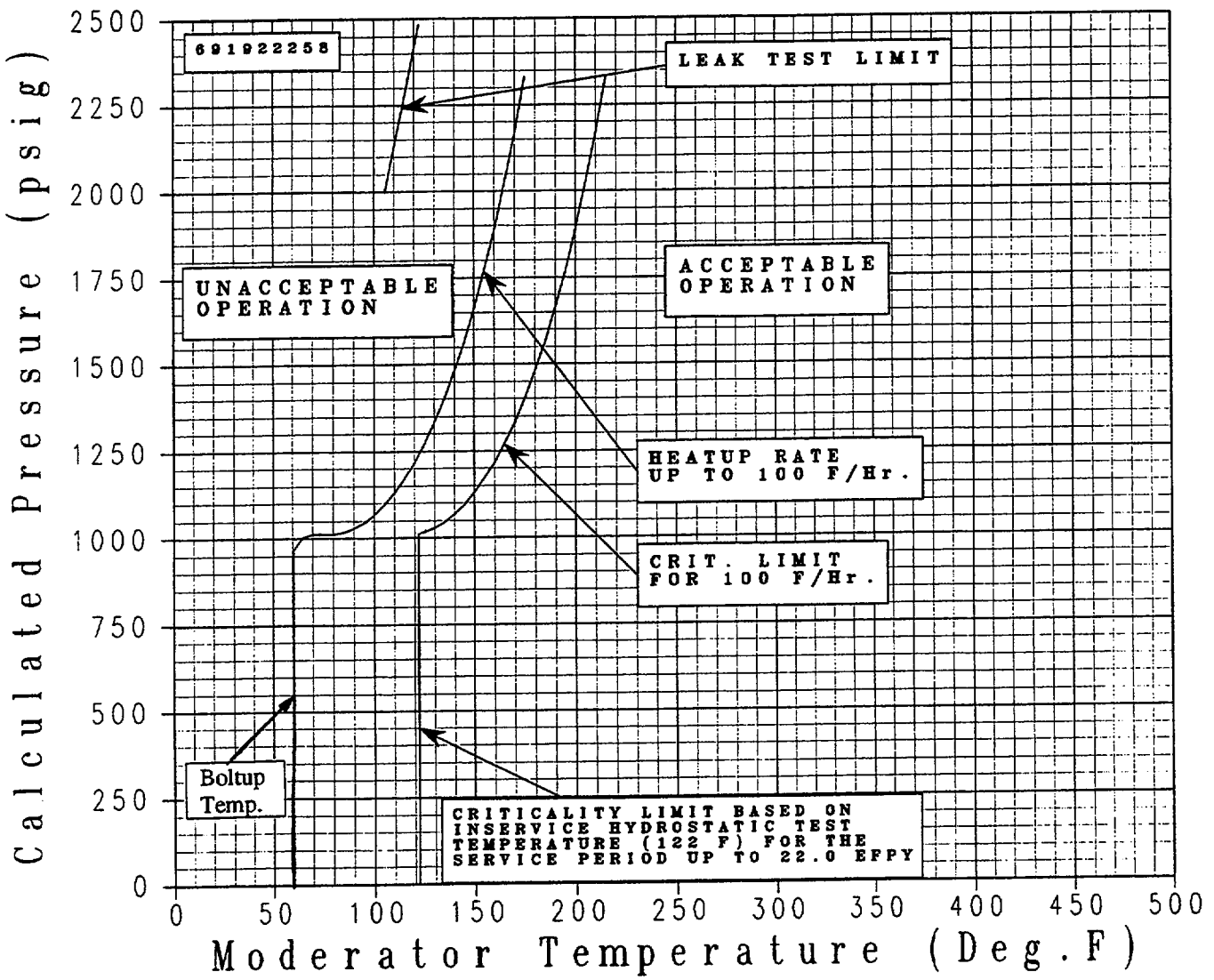

FIGURE 5-5 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 88°F (N-588) & 62°F ('96 App. G)
3/4T, 74°F (N-588) & 50°F ('96 App. G)

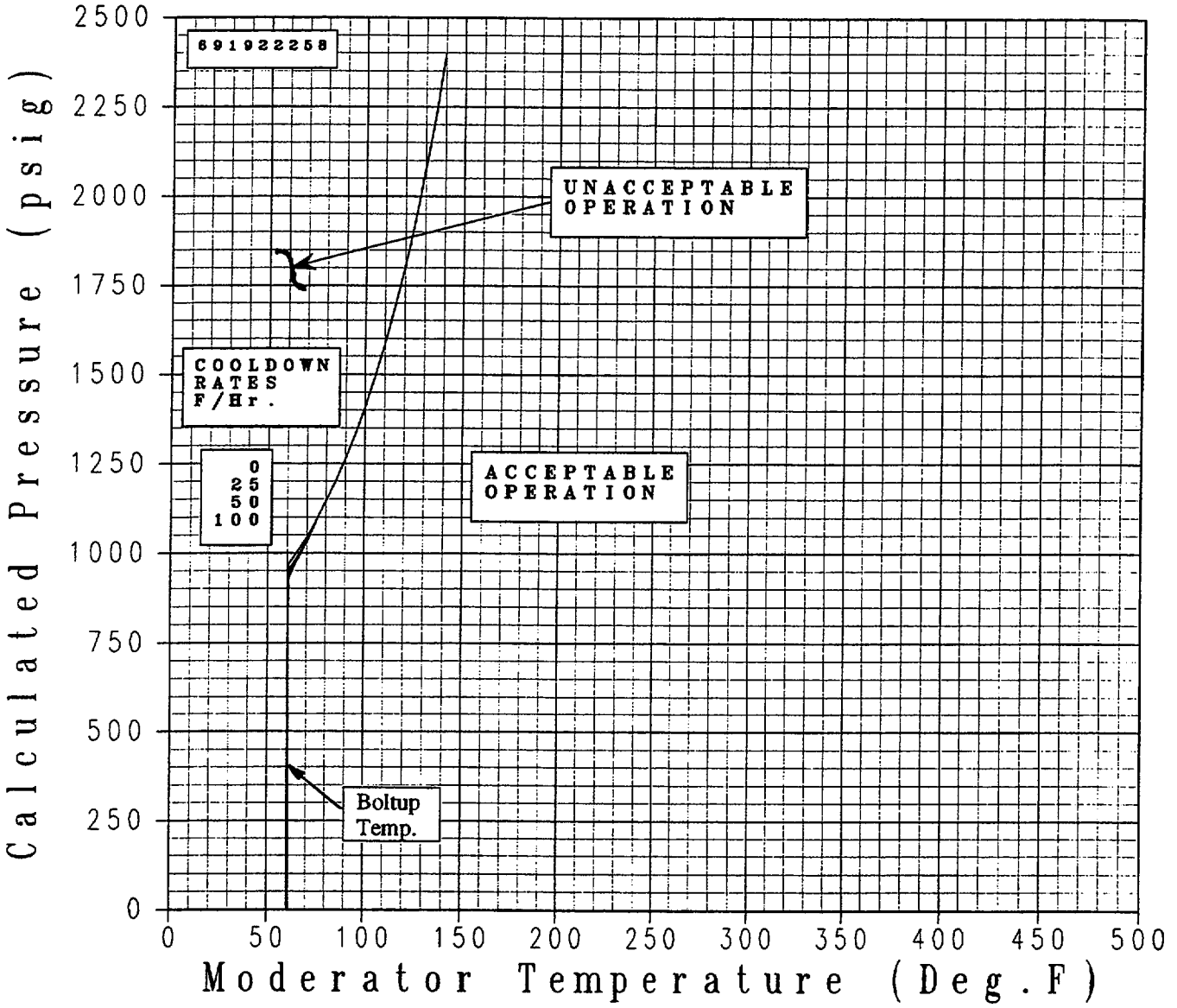

FIGURE 5-6 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

TABLE 5-5

Braidwood Unit 2 Heatup Data at 22 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	122	0	105	2000
60	965	122	1003	122	2485
65	1003	122	1013		
70	1013	125	1018		
75	1013	130	1030		
80	1013	135	1046		
85	1018	140	1069		
90	1030	145	1097		
95	1046	150	1131		
100	1069	155	1171		
105	1097	160	1218		
110	1131	165	1272		
115	1171	170	1333		
120	1218	175	1402		
125	1272	180	1479		
130	1333	185	1566		
135	1402	190	1663		
140	1479	195	1770		
145	1566	200	1890		
150	1663	205	2023		
155	1770	210	2170		
160	1890	215	2332		
165	2023				
170	2170				
175	2332				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. . Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-6

Braidwood Unit 2 Cooldown Data at 22 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	965	60	946	60	932	60	921
65	1003	65	989	65	980		
70	1045	70	1036	70	1033		
75	1092	75	1088				
80	1143						
85	1200						
90	1263						
95	1332						
100	1409						
105	1494						
110	1587						
115	1691						
120	1805						
125	1932						
130	2071						
135	2226						
140	2396						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. . Thus, only the '96 App. G Axial Flaw results are presented in this Report.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 32 EFPY 1/4T, 93°F (N-588) & 67°F ('96 App. G)

3/4T, 79°F (N-588) & 54°F ('96 App. G)

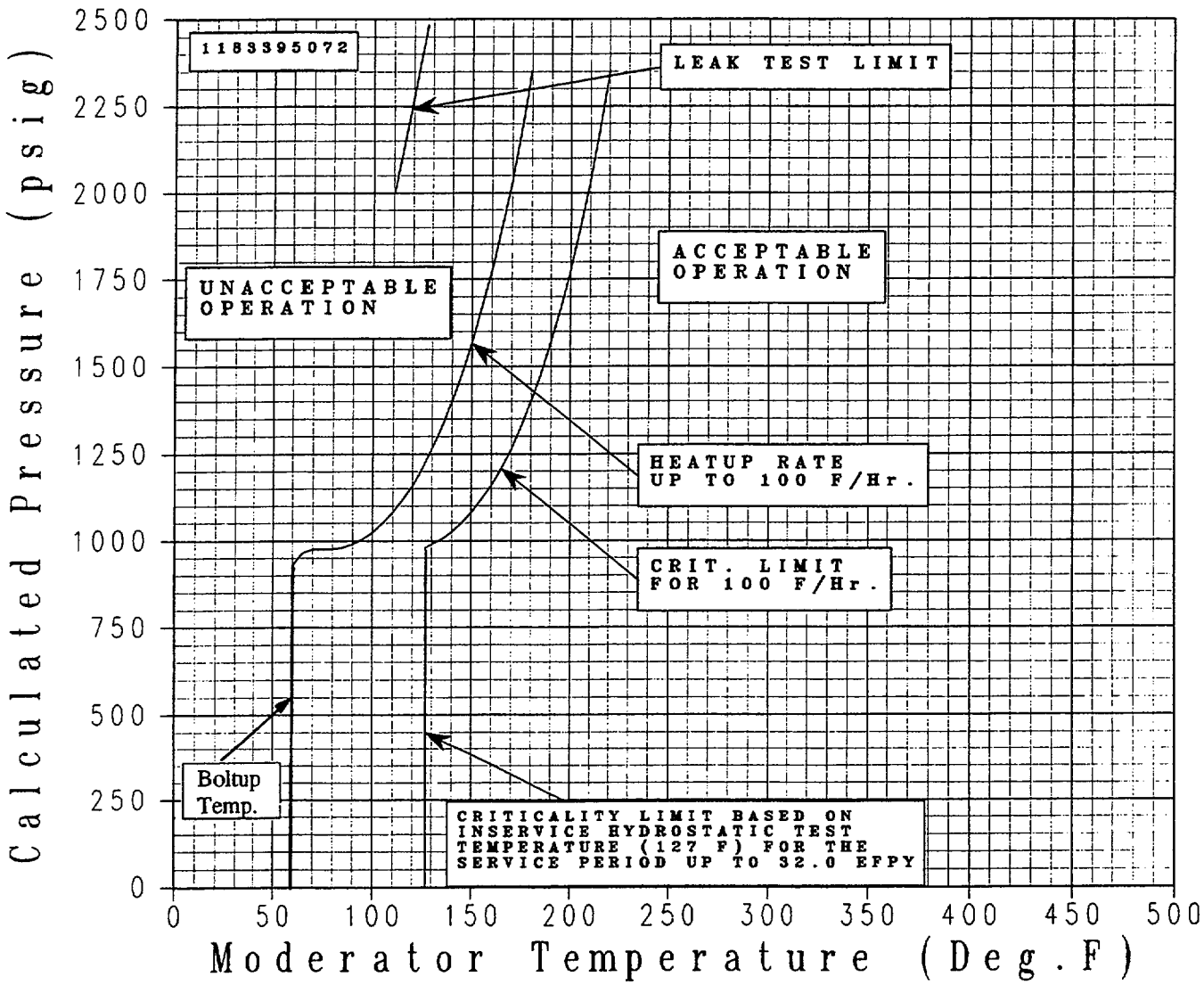

FIGURE 5-7 Braidwood Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-562 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 93°F (N-588) & 67°F ('96 App. G)
3/4T, 79°F (N-588) & 54°F ('96 App. G)

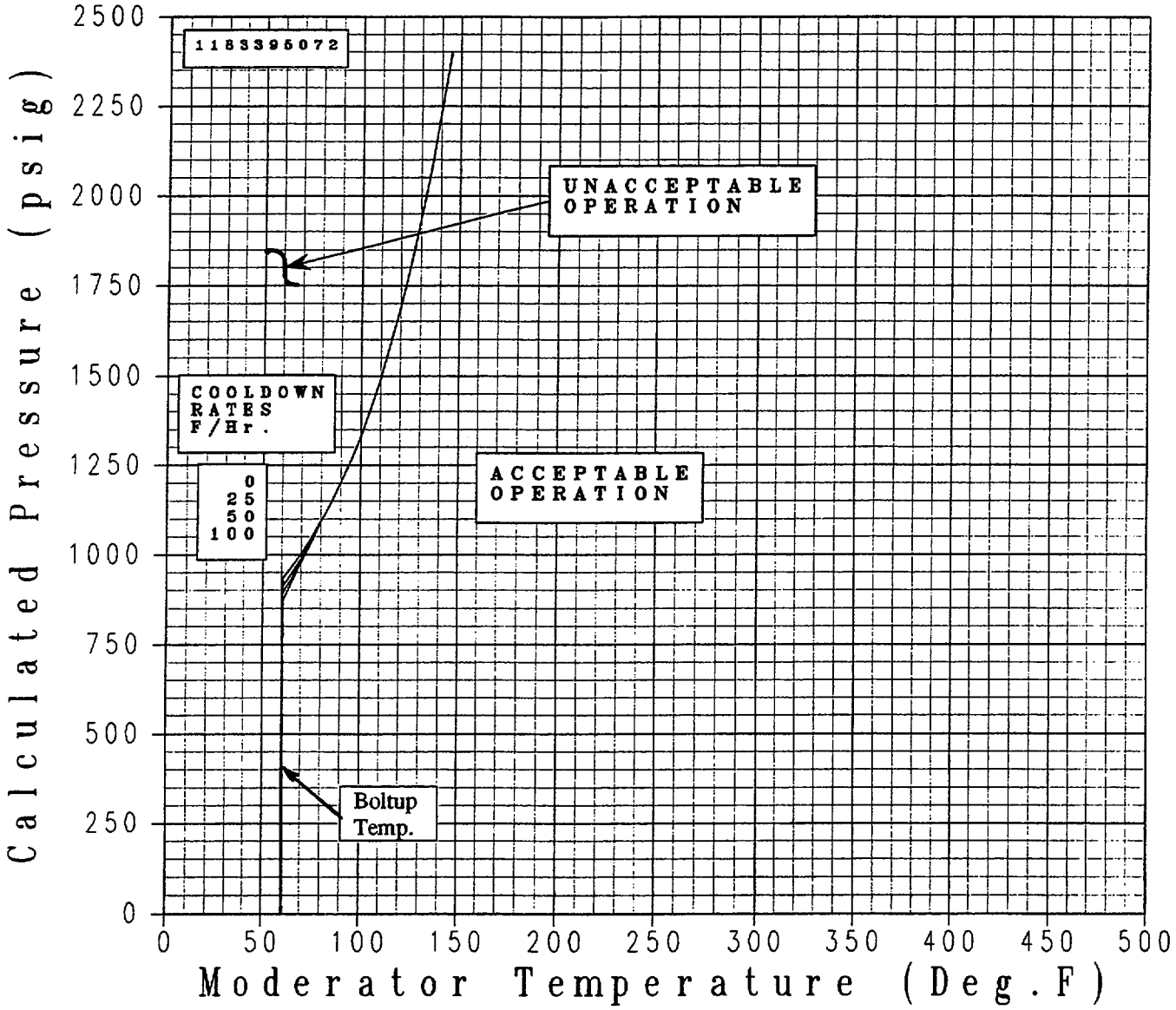

FIGURE 5-8 Braidwood Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Code Case N-588 vs.1996 Appendix G with Axial ART (Without Margins of for Instrumentation Errors)

TABLE 5-7

Braidwood Unit 2 Heatup Data at 32 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Heatup Curves		Configuration #: 361907406			
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	127	0	110	2000
60	931	127	965	127	2485
65	965	127	979		
70	977	127	977		
75	977	127	981		
80	977	130	990		
85	981	135	1005		
90	990	140	1025		
95	1005	145	1051		
100	1025	150	1081		
105	1051	155	1118		
110	1081	160	1161		
115	1118	165	1210		
120	1161	170	1266		
125	1210	175	1329		
130	1266	180	1400		
135	1329	185	1480		
140	1400	190	1569		
145	1480	195	1668		
150	1569	200	1778		
155	1668	205	1901		
160	1778	210	2036		
165	1901	215	2186		
170	2036	220	2353		
175	2186				
180	2353				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. . Thus, only the '96 App. G Axial Flaw results are presented in this Report.

TABLE 5-8

Braidwood Unit 2 Cooldown Data at 32 EFPY Using Code Case N-588 vs. 1996 App. G Methodology

(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	931	60	908	60	889	60	866
65	965	65	946	65	932	65	921
70	1003	70	989	70	980		
75	1045	75	1036	75	1033		
80	1092	80	1088				
85	1143						
90	1200						
95	1263						
100	1332						
105	1409						
110	1494						
115	1587						
120	1691						
125	1805						
130	1932						
135	2071						
140	2226						
145	2396						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall. Thus, only the '96 App. G Axial Flaw results are presented in this Report.

6 REFERENCES

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

2 ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

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

6 ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components", Appendix G, "Fracture Toughness Criteria for Protection Against Failure", 1996.

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12 WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration For Byron and Braidwood", T. J. Laubham, et al., November 1997. Ref. Errata letter CAE-97-233, CCE-97-316, "Transmittal of Updated Tables to WCAP-14824 Rev. 2"

13 Nuclear Design Information Transmittal NDIT No. BRW-DIT-97-321, Rev. 0, "Braidwood Station Historical Tcold Data for Units 1 & 2", Dated October 22, 1997.

14 Nuclear Design Information Transmittal NDIT No. BRW-DIT-2000-0010, Rev. 0, "Braidwood Station Historical Tcold Data for Units 1 & 2 Cycle 7", Dated January 20, 2000.

15 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

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WCAP-15365

Evaluation of Pressurized Thermal Shock for Braidwood Unit 1

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September 2000

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: T. J. Laubham

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Braidwood Unit 1 reactor vessel beltline materials based upon the results of the Surveillance Capsule W evaluation. The conclusion of this report is that all the beltline materials in the Braidwood Unit 1 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (32 EFPY) and life extension (48 EFPY). Specifically, the intermediate shell to lower shell circumferential weld, WF-562 was the most limiting material with 32 and 48 EFPY PTS values of 99°F and 101°F respectively.

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the Braidwood Unit 1 reactor vessel using the results of the surveillance Capsule W evaluation. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Braidwood Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6 of WCAP-15316, Revision 1[1]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The latest revision of the PTS Rule, 10 CFR Part 50.61[2], was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is also described in Regulatory Guide 1.99, Revision 2[3].

2. The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of license (EOL) fluence, RT_{PTS}.

3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and
  300°F for circumferential weld materials.

3 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence at the clad/base metal interface for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \quad (1)$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2

$$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \quad (2)$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

σ_Δ not to exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in RT_{NDT}, due to irradiation, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \quad (3)$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF, when using credible surveillance data, is determined using Equation 5.

The EOL Fluence (f) is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}.

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \tag{4}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF for surveillance materials is determined from Equation 5.

$$CF = \frac{\sum [A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum [f_i^{(0.56-0.20\log f_i)}]} \tag{5}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}. For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{PTS, measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{PTS} = \Delta RT_{PTS, measured} + 1.0*(T_{capsule} - T_{plant})$$

Note that the temperature adjust methodology has been reinforce by the NRC at the NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998.

4 VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Braidwood Unit 1 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as, "the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". Figure 1 identifies and indicates the location of all beltline region materials for the Braidwood Unit 1 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from WCAP-15364, Table 4-10[4]. The best estimate copper and nickel content is also documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for Braidwood Unit 1 reactor vessel beltline material properties are also shown in Table 1. As a note, per WCAP-15316, Revision 1, all vessel materials not listed in Table 1 experience a fluence less the 10^{17} n/cm^2 for both 32 and 48 EFPY

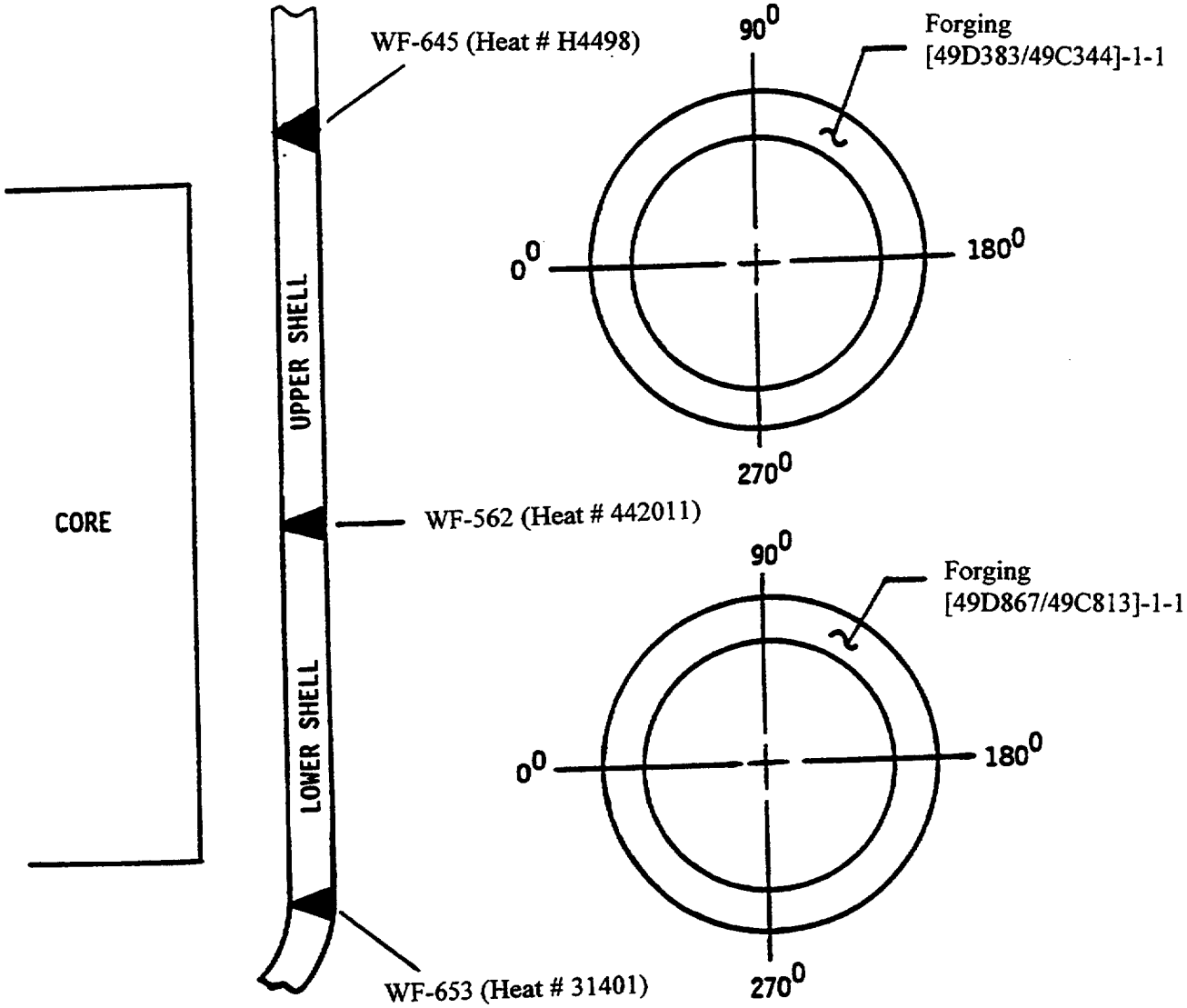

Figure 1: Identification and Location of Beltline Region Materials for the Braidwood Unit 1 Reactor Vessel

Table 1
Braidwood Unit 1 Reactor Vessel Beltline Unirradiated Material Properties

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT}[a]
Nozzle Shell Forging 5P7016	0.04	0.73	10
Intermediate Shell Forging [49D383/49C344]-1-1	0.05	0.73	-30
Lower Shell Forging [49D867/49C813]-1-1	0.05	0.74	-20
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	0.03	0.67	40
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	0.04	0.46	-25
Lower Shell Forging to Bottom Head Ring Circ. Weld Seam WF-653 (Heat 31401)	0.19	0.58	-40
Surveillance Weld Braidwood Unit 1 (Heat 442011)	0.03	0.67	- - -
Surveillance Weld Braidwood Unit 2 (Heat 442011)	0.03	0.71	- - -

Notes:
a) The Initial RT_{NDT} values for the forgings and welds are based on measured data.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the clad base metal interface of the Braidwood Unit 1 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the results of the Capsule W analysis. See Section 6.0 of the Capsule W analysis report, WCAP-15316, Revision 1[1].

TABLE 2
Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for Braidwood Unit 1 at 32 (EOL) and 48 (Life Extension) EFPY

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Intermediate Shell Forging [49D383/49C344]-1-1	45°	2.05×10^{19} n/cm^2	3.06×10^{19} n/cm^2
Lower Shell Forging [49D867/49C813]-1-1	45°	2.05×10^{19} n/cm^2	3.06×10^{19} n/cm^2
Intermediate to Lower Shell Forging Circ. Weld WF-562	45°	1.99×10^{19} n/cm^2	2.98×10^{19} n/cm^2
Nozzle Shell Forging [5P-7016]	45°	0.608×10^{19} n/cm^2	0.91×10^{19} n/cm^2
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645	45°	0.608×10^{19} n/cm^2	0.91×10^{19} n/cm^2

6 DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Braidwood Unit 1 reactor vessel for fluence values at the EOL (32 EFPY) and life extension (48 EFPY).

Per 10 CFR Part 50.61, Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. The Braidwood Unit 1 surveilance program data has evaluated and shown to be credible in WCAP-15366[5]. The related surveillance program results have been included in this PTS evaluation.

As presented in Table 3, chemistry factor values for Braidwood Unit 1 based on average copper and nickel weight percent values were calculated using Tables 1 and 2 from 10 CFR 50.61[2]. Additionally, chemistry factor values based on credible surveillance capsule data from Braidwood Units 1 and 2[4,5], are calculated in Table 4. Tables 5 and 6 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and life extension (48 EFPY).

TABLE 3

Interpolation of Chemistry Factors Using Tables 1 and 2 of 10 CFR Part 50.61

Material	CU wt. %	Ni wt. %	Chemistry Factor, °F
Intermediate Shell Forging [49D383/49C344]-1-1	0.05	0.73	31.0°F
Lower Shell Forging [49D867/49C813]-1-1	0.05	0.74	31.0°F
Nozzle Shell Forging 5P-7016	0.04	0.73	26.0°F
Intermediate to Lower Shell Circ. Weld WF-562 (Heat 442011)	0.03	0.67	41.0°F
Nozzle Shell to Intermediate Shell Circ. Weld WF-645 (Heat H4498)	0.04	0.46	54.0°F
Braidwood Unit 1 Surveillance Program Weld Metal	0.03	0.67	41.0°F
Braidwood Unit 2 Surveillance Program Weld Metal	0.03	0.71	41.0°F

TABLE 4

Calculation of Chemistry Factors using Surveillance Capsule Data Per Regulatory Guide 1.99, Revision 2, Position 2.1

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging [49D867/49C813]-1-1 (Tangential)	U	0.387	0.737	5.78	4.26	0.543
	X	1.24	1.060	38.23	40.52	1.124
	W	2.09	1.201	24.14	28.99	1.442
Lower Shell Forging [49D867/49C813]-1-1 (Axial)	U	0.387	0.737	0.0[e]	0.0	0.543
	X	1.24	1.060	28.75	30.48	1.124
	W	2.09	1.201	37.11	44.57	1.442
				SUM:	148.82	6.218
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (148.82) \div (6.218) = 23.9°F$					
Braidwood Unit 1 Surveillance Weld Metal	U	0.387	0.737	17.06[d]	12.57	0.543
	X	1.24	1.060	30.15[d]	31.96	1.124
	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2 Surveillance Weld Metal	U	0.40	0.746	0.0[d,f]	0.0	0.557
	X	1.23	1.058	26.3[d]	27.83	1.119
	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) = 25.7°F$					

Notes:

(a) f = Calculated fluence from the Braidwood Unit 1 capsule W dosimetry analysis results [1]. Calculated fluence from the Braidwood Unit 2 capsule W analysis[6], (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 1 documented in the Braidwood Unit 1 capsule W analysis[1]. The Braidwood Unit 2 ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 2 documented in the Braidwood Unit 2 capsule W analysis[6].

(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted (i.e. the ratio factor is 1.0) (See Reference 5).

(e) Actual value of ΔRT_{NDT} is -16.07. This physically should not occur, therefore for conservatism (i.e. higher chemistry factor) a value of zero will be used.

(f) Measured value is –0.58. 0 is assumed for conservatism.

TABLE 5
RT_{PTS} Calculation for Braidwood Unit 1 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence ($x10^{19}$ n/cm², E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging (Heat # 49D383/49C344-1-1)	2.05	1.20	31.0	37.2	34	-30	41
Lower Shell Forging (Heat # 49D867/49C813-1-1)	2.05	1.20	31.0	37.2	34	-20	51
Lower Shell Forging → Using S/C Data	2.05	1.20	23.9	28.7	17	-20	26
Inter. to Lower Shell Circ. Weld Metal (Seam # WF-562)	1.99	1.19	41.0	48.8	48.8	40	138
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	1.99	1.19	25.7	30.6	28	40	99
Nozzle Shell Forging (Heat # 5P-7016)	0.608	0.86	26.0	22.4	22.4	10	55
Nozzle Shell to Inter. Shell Circ. Weld Metal (Seam # WF-645)	0.608	0.86	46.0	39.6	39.6	-25	54

Notes:
(a) Initial RT_{NDT} values are measured values
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)
(c) $\Delta RT_{PTS} = CF * FF$

TABLE 6

RT_{PTS} Calculation for Braidwood Unit 1 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence ($x10^{19}$ n/cm², E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging (Heat # 49D383/49C344-1-1)	3.06	1.30	31.0	40.3	34	-30	44
Lower Shell Forging (Heat # 49D867/49C813-1-1)	3.06	1.30	31.0	40.3	34	-20	54
Lower Shell Forging → Using S/C Data	3.06	1.30	23.9	31.1	31.1	-20	42
Inter. To Lower Shell Circ. Weld Metal (Seam # WF-562)	2.98	1.29	41.0	52.9	52.9	40	146
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	2.98	1.29	25.7	33.2	28	40	101
Nozzle Shell Forging (Heat # 5P-7016)	0.909	0.97	26.0	25.2	25.2	10	60
Nozzle Shell to Inter. Shell Circ. Weld Metal (Seam # WF-645)	0.909	0.97	46.0	44.6	44.6	-25	64

Notes:

(a) Initial RT_{NDT} values are measured values

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) ΔRT_{PTS} = CF * FF

7 CONCLUSIONS

As shown in Tables 5 and 6, all of the beltline region materials in the Braidwood Unit 1 reactor vessel have EOL (32 EFPY) RT_{PTS} and Life Extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds. Specifically, the intermediate to lower shell circumferential Weld, WF-562, was the most limiting material with 32 and 48 EFPY PTS values of 99°F and 101°F respectively.

8 REFERENCES

1 WCAP-15316, Revision 1, Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program", E. Terek, et al., December 1999.

2 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

4 WCAP-15364, "Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operaton", E. Terek, March, 2000.

5 WCAP-15366, "Commonwealth Edison Company Braidwood Unit 1 Surveillance Program Credibility Evaluation", E. Terek, March, 2000.

6 WCAP-15369, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program", T. J. Laubham, et al., March 2000.

WCAP-15381

Evaluation of Pressurized Thermal Shock for Braidwood Unit 2

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September 2000

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Braidwood Unit 2 reactor vessel beltline materials based upon the results of the Surveillance Capsule W evaluation. The conclusion of this report is that all the beltline materials in the Braidwood Unit 2 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (32 EFPY) and life extension (48 EFPY). Specifically, the intermediate shell to lower shell circumferential weld, WF-562 was the most limiting material with 32 and 48 EFPY PTS values of 98°F and 101°F respectively.

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the Braidwood Unit 2 reactor vessel using the results of the surveillance Capsule W evaluation. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Braidwood Unit 2 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from Section 6 of WCAP-15369[5]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The latest revision of the PTS Rule[1], 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is also described in Regulatory Guide 1.99, Revision 2[2].

2. The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of license (EOL) fluence, RT_{PTS}.

3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and
  300°F for circumferential weld materials.

3 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \tag{1}$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2

$$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \tag{2}$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

σ_Δ not to exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{NDT}, due to irradiation, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \tag{3}$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

The EOL Fluence (f) is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}..

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \tag{4}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF for surveillance materials is determined from Equation 5.

$$CF = \frac{\sum [A_i * f_i^{(0.28-0.10 \log fi)}]}{\sum [f_i^{(0.56-0.20 \log fi)}]} \tag{5}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}. For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{PTS,\ measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{PTS} = \Delta RT_{PTS\ measured} + 1.0*(T_{capsule} - T_{plant})$$

Note that the temperature adjust methodology has been reinforce by the NRC at the NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998.

4 VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Braidwood Unit 2 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". Figure 1 identifies and indicates the location of all beltline region materials for the Braidwood Unit 2 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from WCAP-15369[7]. The best estimate copper and nickel content is also documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for Braidwood Unit 2 reactor vessel beltline material properties are also shown in Table 1.

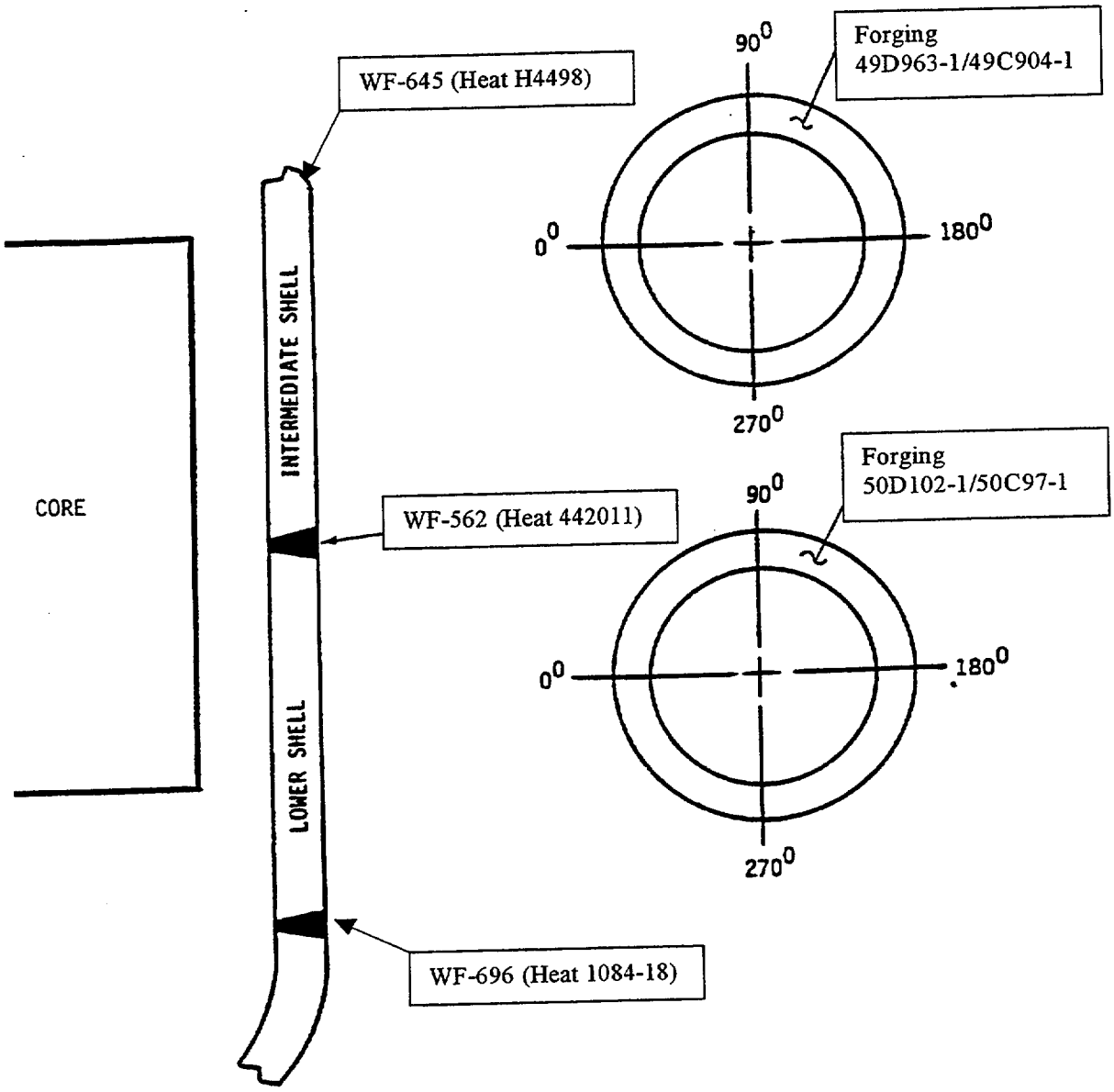

Figure 1: Identification and Location of Beltline Region Materials for the Braidwood Unit 2 Reactor Vessel

Table 1

Braidwood Unit 2 Reactor Vessel Beltline Unirradiated Material Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[(a)]
Closure Head Flange 3P6566/5P7547/4P6986	---	0.75	20
Vessel Flange 124P455	0.07	0.70	20
Nozzle Shell Forging 5P7056	0.04	0.90	30
Intermediate Shell Forging 49D963-1/49C904-1	0.03	0.71	-30
Lower Shell Forging 50D102-1 / 50C97-1	0.060	0.759	-30
Inter. to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	0.033	0.666	40
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	0.042	0.46	-25
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.032, 0.034	0.67, 0.71	- - -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the Braidwood Unit 2 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the results of the Capsule W analysis. See Section 6.0 of the Capsule W analysis report, WCAP-15369[5] .

TABLE 2

Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for Braidwood Unit 2 at 32 (EOL) and 48 (Life Extension) EFPY

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Nozzle Shell Forging 5P-7056	30°	5.67×10^{18} n/cm^2	8.49×10^{18} n/cm^2
Intermediate Shell Forging 49D963-1/49C904-1	30°	1.96×10^{19} n/cm^2	2.94×10^{19} n/cm^2
Lower Shell Forging 50D102-1/50C97-1	30°	1.96×10^{19} n/cm^2	2.94×10^{19} n/cm^2
Intermediate to Lower Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	30°	1.89×10^{19} n/cm^2	2.83×10^{19} n/cm^2
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-645 (Heat H4498)	30°	5.67×10^{18} n/cm^2	8.49×10^{18} n/cm^2

6 DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Braidwood Unit 2 reactor vessel for fluence values at the EOL (32 EFPY) and life extension (48 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For Braidwood Unit 2, the related surveillance program results have been included in this PTS evaluation. (See Reference 8 for the credibility evaluation of the Braidwood Unit 2 surveillance data.)

As presented in Table 3, chemistry factor values for Braidwood Unit 2 based on average copper and nickel weight percent were calculated using Tables 1 and 2 from 10 CFR 50.61[1]. Additionally, chemistry factor values based on credible surveillance capsule data from Braidwood Units 1 and 2 are calculated in Table 4. Tables 5 and 6 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and life extension (48 EFPY).

TABLE 3

Interpolation of Chemistry Factors Using Tables 1 and 2 of 10 CFR Part 50.61

Material	Ni, wt %	Chemistry Factor, °F
Intermediate Shell Forging 49D963-1/49C904-1 Given Cu wt% = 0.03	0.71	20.0°F
Lower Shell Forging 50D102-1/50C97-1 Given Cu wt% = 0.06	0.76	37.0°F
Nozzle Shell Forging 5P-7056 Given Cu wt% = 0.04	0.90	26.0°F
Intermediate to Lower Shell Circ. Weld WF-562 Given Cu wt% = 0.03	0.67	41.0°F
Nozzle Shell to Intermediate Shell Circ. Weld WF-645 Given Cu wt% = 0.04	0.46	54.0°F
Braidwood Unit 1 and 2 Surveillance Program Weld Metal Given Cu wt% = 0.03	0.67, 0.71	41.0°F

TABLE 4

Calculation of Chemistry Factors using Surveillance Capsule Data Per Regulatory Guide 1.99, Revision 2, Position 2.1

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging	U	0.400	0.746	0.0[e]	0	0.557
50D102-1/50C97-1	X	1.23	1.058	0.0[e]	0	1.119
(Tangential)	W	2.25	1.220	4.53	5.53	1.488
Lower Shell Forging	U	0.400	0.746	0.0[e]	0	0.557
50D102-1/50C97-1	X	1.23	1.058	33.94	35.91	1.119
(Axial)	W	2.25	1.220	33.2	40.50	1.488
				SUM:	81.94	6.328
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (81.94) \div (6.328) =$ **12.9°F**					
Braidwood Unit 1	U	0.387	0.737	17.06[d]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[d]	31.96	1.124
Weld Metal[f]	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[d, e]	0	0.557
Surveillance	X	1.23	1.058	26.3[d]	27.83	1.119
Weld Metal[f]	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) =$ **25.7°F**					

Notes:

(a) f = Calculated fluence from the Braidwood Unit 2 capsule W dosimetry analysis results[5], the Braidwood Unit 1 calculated fluences are from capsule W analysis[6], (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values for Braidwood Unit 2 taken from App. B. Braidwood Unit 1 values are from WCAP-15316 Rev 1 (Ref. 6).

(d) The surveillance weld metal ΔRT_{NDT} values have not been adjusted (i.e. the ratio factor is 1.0).

(e) Actual values of ΔRT_{NDT} are -9.73 (Cap U Tang.), -9.42 (Cap. X Tang.), -0.13 (Cap. U Axial), -0.58 (Cap U Weld). This physically should not occur, therefore for conservatism (i.e. higher chemistry factor) a value of zero will be used.

(f) The inlet operating temperature of both Braidwood Unit 1 and Unit 2 are essentially the same for the time of interest and no temperature adjustments are required.

TABLE 5
RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence, x 10^{19} (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging	1.96	1.18	20	23.6	23.6	-30	17
Lower Shell Forging	1.96	1.18	37	43.7	34	-30	48
Lower Shell Forging (Using S/C Data)	1.96	1.18	12.9	15.2	34	-30	19
Nozzle Shell Forging	0.567	0.841	26	21.9	21.9	30	74
Inter. to Lower Shell Circ. Weld Metal	1.89	1.17	41	48.0	48.0	40	136
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	1.89	1.17	25.7	30.1	28	40	**98**
Nozzle Shell to Inter. Shell Circ. Weld Metal	0.567	0.841	54	45.4	45.4	-25	66

Notes:

(a) Initial RT_{NDT} values are measured values (See Table 1)

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) ΔRT_{PTS} = CF * FF

TABLE 6

RT_{PTS} Calculation for Braidwood Unit 2 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence, x 10^{19} (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging	2.94	1.29	20	25.8	25.8	-30	22
Lower Shell Forging	2.94	1.29	37	47.7	34	-30	52
Lower Shell Forging (Using S/C Data)	2.94	1.29	12.9	16.6	34	-30	21
Nozzle Shell Forging	0.849	0.954	26	24.8	24.8	30	80
Inter. to Lower Shell Circ. Weld Metal	2.83	1.28	41	52.5	52.5	40	145
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	2.83	1.28	25.7	32.9	28	40	**101**
Nozzle Shell to Inter. Shell Circ. Weld Metal	0.849	0.954	54	51.5	51.5	-25	78

Notes:

(a) Initial RT_{NDT} values are measured values (See Table 1)

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F)

(c) ΔRT_{PTS} = CF * FF

7 CONCLUSIONS

As shown in Tables 5 and 6, all of the beltline region materials in the Braidwood Unit 2 reactor vessel have EOL (32 EFPY) RT_{PTS} and Life Extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds. Specifically, the intermediate to lower shell circumferential Weld, WF-562, was the most limiting material with 32 and 48 EFPY PTS values of 98°F and 101°F respectively.

8 REFERENCES

1 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

3 WCAP-11188, "Commonwealth Edison Company Braidwood Station Unit No. 2 Reactor Vessel Radiation Surveillance Program", L. R. Singer, December 1986.

4 WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance weld Metal Integration for Byron & Braidwood", T. J.Laubham, et. al., November, 1997.

5 WCAP-15369, "Analysis of Capsule W from the Commonwealth Edison Co. Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program", T. J. Laubham, et al., March 2000.

6 WCAP-15316, Revision 1, "Analysis of Capsule W from the Commonwealth Edison Co. Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program", E.Terek, et.al., December 1999.

7 CN-EMT-00-4, "Braidwood Unit 2 Surveillance Capsule W Calculations, Development of New Pressure-Temperature Limit Curves and PTS Calculations Using Uprated Fluences", T.J. Laubham, Dated February 2000.

8 WCAP-15368, "Commonwealth Edison Co. Braidwood Unit 2 Surveillance Program Credibility Evaluation", T. J. Laubham, March 2000.

ATTACHMENT G.9-1

The following documents are attached for information.

Reference G.9.1 NUREG-1002, Supplement No. 1, "Safety Evaluation Report related to the operation of Braidwood Station, Units 1 and 2," Section 3.9.3.3, "Design and Installation of Pressure Relief Devices"

Reference G.9.2 NUREG-0876, Supplement No. 5, "Safety Evaluation Report related to the operation of Byron Station, Units 1 and 2," Section 3.9.3.3, "Design and Installation of Pressure Relief Devices"

Reference G.9.3 Letter from L. N. Olshan (NRC) to H. E. Bliss (ComEd), dated August 18, 1988; Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2 (TAC Nos. 56200 and 63240) transmitting Technical Evaluation Report (TER) providing the results of the NRC's review on Byron Units 1 and 2 response to NUREG-0737, Item II.D.1

Reference G.9.4 Letter from S. P. Sands (NRC), to T. K. Kovach (ComEd), dated May 21, 1990; Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Braidwood Station, Units 1 and 2 (TAC Nos. 64019 and 64046) transmitting Technical Evaluation Report (TER) providing the results of the NRC's review on Braidwood Units 1 and 2 response to NUREG-0737, Item II.D.1

By letter dated September 26, 1984, the applicant committed to an inspection program based on the manufacturer's recommendations. The staff has reviewed and approved the manufacturer's generic turbine integrity methodology which provides procedures for estimating crack growth, missile generation probability and volumetric inspection intervals. Based on the manufacturer's recommendations, the applicant is required to volumetrically inspect all low-pressure turbine rotors every third refueling outage. In addition, an acceptable turbine valve inspection program has been incorporated into Section 4.3.4.2 of the Technical Specifications. Therefore, License Condition A(4) is no longer necessary.

The staff concludes that the turbine missile risks for the proposed plant design are in compliance with the requirements of GDC 4 and are acceptable. Thus, Outstanding Item B(1) is considered closed.

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Preoperational Vibration and Dynamic Effects of Testing on Piping

The staff reviewed the applicant's detailed program for pipe vibration documented in Preoperational Test No. BWPT-EM-12, Revision 1 (see Inspection Report No. 50-456/86-32 dated August 14, 1986). Specified piping was monitored by instrumentation and visually inspected by the applicant during preoperational hot functional testing conducted during March 1986. The following systems were inspected for vibrations:

- reactor coolant system
- component cooling system
- chemical and volume control system (safe shutdown portion)
- residual heat removal system
- safety injection system
- essential service water system
- containment spray system (except spray headers)
- chilled water (control room) system
- fuel pool cooling and cleanup system
- reactor coolant pressurizer system

On the basis of a review of the applicant's program, the staff has concluded that the preoperational hot functional tests have demonstrated that piping vibrations in the systems inspected are within acceptable limits and that the piping can expand thermally in a manner consistent with the intent of the design. Therefore, Confirmatory Issue A(3) is considered closed.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.3 Design and Installation of Pressure Relief Devices

As required by TMI Action Plan Item II.D.1, all PWR plant licensees and applicants are required to demonstrate that their pressurizer safety valves (SVs), power-operated relief valves (PORVs), PORV block valves, and all associated

discharge piping will function adequately under conditions predicted for design-basis transients and accidents. In response to this requirement, the Electric Power Research Institute (EPRI), on behalf of the PWR Owners Group, has completed a full-scale valve testing program and the Owners Group has submitted these test results to the NRC (letter from O. Kingsley (Alabama Power) to S. Chilk (NRC) dated July 27, 1982, transmitting WCAP-10105, a report performed for the Westinghouse Owners Group). Additionally, each PWR plant applicant for an OL was required to submit a report by fuel load time which would demonstrate the operability of these valves and the associated piping.

The applicant responded to this requirement with submittals dated July 1, 1982, and October 26, 1982, that contain information from the EPRI valve test program results which apply to Braidwood Units 1 and 2. A December 30, 1983, submittal also states that the safety and relief valve discharge piping and supports are being modified to ensure functionability.

The staff has not completed a detailed review of the applicant's submittals; however, on the basis of a preliminary review, the staff finds that the general approach of using the EPRI test results to demonstrate operability of the safety valves, PORVs, and PORV block valves is acceptable. The applicant's submittals note that Braidwood uses safety valves, PORVs, and PORV block valves similar to valves that performed satisfactorily for test sequences that bound conditions the valve could be exposed to.

In summary, on the basis of preliminary review, the staff has concluded that the applicant's general approach to responding to this item is acceptable and provides adequate assurance that reactor coolant system overpressure protection systems at Braidwood can adequately perform their intended functions. If the detailed review reveals that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping are needed to ensure that all intended design margins are present, the staff will require that the applicant make appropriate modifications. This is a Confirmatory Issue.

The staff has determined that an exemption is required from GDC 4 to Appendix A, which requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of discharging fluids. Based on the aforementioned low probability of a full area double ended pipe break and the reduced consequences of a pipe break at low power, the staff concludes that the exemption from GDC 4 up to 5% power will not endanger life or property or the common defense and security and is otherwise in the public interest.

3.8 Design of Seismic Category I Structures

3.8.3 Other Seismic Category I Structures

License Condition 2 in the original SER required that all the questions pertaining to the analysis, design and erection of masonry walls, including any modifications resulting from the staff's review, be resolved prior to the beginning of the power operation after the first refueling outage.

Since that time, however, additional information has been obtained from the applicant in letters dated December 5, 1983 and July 16, 1984 which indicates that the walls have been analyzed in compliance with the NRC regulatory requirements contained in SRP Section 3.8.4. Comparison of the maximum calculated stresses to the allowable stresses specified in the SRP indicates that the calculated stresses are below the allowables. Further, the applicant provided a summary of results of tests performed on walls similar to those at the Byron plant to estimate the factor of safety against failure. The test results indicate that the average factor of safety is 5.6 for loads under OBE load combinations and 3.55 under the SSE load combinations.

The applicant surveyed 458 walls and determined that 13 out of the surveyed walls had structural cracks. The cracked walls have been identified and reanalyzed by the applicant to demonstrate that these cracks have no effect on structural integrity of the walls.

In view of the above the staff concludes that the design of masonry walls at the Byron plant is conservative and complies with the staff's acceptance criteria. Therefore, the staff concludes that no additional actions are required regarding the masonry wall issue and considers it resolved.

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures

3.9.3.3 Design and Installation of Pressure Relief Devices

As required by NUREG-0737, Item II.D.1, all PWR plant licensees and applicants are required to demonstrate that their pressurizer safety valves (SV), power operated relief valves (PORVs), PORV block valves, and all associated discharge piping will function adequately under conditions predicted for design basis transients and accidents. In response to this requirement, the Electric Power Research Institute (EPRI), on behalf of the PWR Owners Group, has completed a full scale valve testing program and the Owners Group has submitted these test

results to the NRC. Additionally, each PWR plant applicant for an OL was required to submit a report by fuel load which would demonstrate the operability of these valves and the associated piping.

Commonwealth Edison responded to this requirement with submittals dated July 1, 1982 and October 26, 1982 that contain information from the EPRI valve test program results which apply to Byron 1 and 2. A December 30, 1983 submittal also states that the safety and relief valve discharge piping and supports are being modified to insure functionability.

The staff has not completed a detailed review of the applicant's submittals; however, based on a preliminary review the staff finds that the general approach of using the EPRI test results to demonstrate operability of the safety valves, PORVs and PORV block valves is acceptable. The applicant's submittals note that Byron utilizes safety valves, PORVs and PORV block valves similar to valves that performed satisfactorily for test sequences that bound conditions that the valve could be exposed to.

In summary, based on preliminary review, the staff has concluded that the applicant's general approach to responding to this item is acceptable and provides adequate assurance that the Byron Reactor Coolant System Overpressure Protection Systems can adequately perform their intended functions. If the completion of the detailed review reveals that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping are needed to assure that all intended design margins are present, the staff will require that the applicant make appropriate modifications.

3.9.3.4 Component Supports

The upper lateral support of the steam generators as described in FSAR Section 3.9.3.4.1.3 consists of two hydraulic snubbers on each of the four steam generators. The original snubbers were manufactured by Boeing and were in place during hot functional testing of Byron 1. At the request of the staff, the applicant had additional qualification testing conducted on snubbers that were identical to the snubbers installed at Byron 1. The testing was conducted by ITT Grinnell in June 1984. The test results were unacceptable and indicated deficiencies in the snubber design. Consequently, the applicant procured snubbers of a staff approved design manufactured by Paul Munroe. The applicant removed the original Boeing snubbers and will replace them with Paul Munroe snubbers. Since the Technical Specifications (3.7.8 and 3.4.5) do not require these snubbers to be operable prior to entering Mode 4, the staff has concluded that the above steam generator snubber replacement program is acceptable.

The staff has determined that an exemption is required from GDC 2 to Appendix A, which requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. The staff concludes that the exemption from the requirement of GDC 2 prior to entering Mode 4 will not endanger life or property or the common defense and security and is otherwise in the public interest. The staff reaches this conclusion because, prior to Mode 4, (1) the Technical Specifications do not require these snubbers, (2) the steam generators are not needed for decay heat removal, and (3) postulated reactor coolant system pipe break would not produce any offsite doses.

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

AUG 25 REC'D

August 18, 1988

Docket Nos.: 50-454 and 50-455

Mr. Henry E. Bliss
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: NUREG-0737, ITEM II.D.1, PERFORMANCE TESTING ON RELIEF AND SAFETY VALVES FOR BYRON STATION, UNITS 1 AND 2 (TAC NOS. 56200 AND 63240)

The enclosed Technical Evaluation Report (TER) was prepared with the assistance of EG&G, Idaho under contract with the NRC staff. The TER provides the results of the staff and EG&G's review of the licensee's submittals in response to TMI Action Plan Requirements, NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves. The staff endorses the findings contained in the TER. Based on these results, we conclude that you have provided an acceptable response.

This TER imposes one additional requirement. As discussed in Section 5, in order to demonstrate continued operability of the safety valves, you should develop and adopt plant procedures to inspect the valves after each lift involving loop seal or water discharge.

Sincerely,

Leonard N. Olshan

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Project III,
IV, V and Special Projects

Enclosure: As stated

EGG-NTA-8028

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
BYRON UNITS 1 and 2
DOCKET NO. 50-454, 50-455

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January 1988

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Prepared for the
U.S. Nuclear Regulatory Commission
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FIN No. A6492

ABSTRACT

Light water reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this review examined the response of the Licensee for the Byron Nuclear Power Station, Units 1 and 2, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee provided an acceptable response, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met.

FIN No. A6492--Evaluation of OR Licensing Actions-NUREG-0737, II.D.1

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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
BYRON NUCLEAR POWER PLANT, UNITS 1 AND 2
DOCKET NO. 50-454, 50-455

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant system. There were instances of valves opening below set pressure, valves opening above set pressure, and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island (TMI-2) sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend that programs be developed and executed which would reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR, are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (a) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage, (b) the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions are not

exceeded during normal operation or anticipated transient events, and (c) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure that the General Design Criteria are met, the NUREG-0578 position was issued as a requirement in a letter dated September 13, 1979 by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR), to ALL OPERATING NUCLEAR POWER PLANTS. This requirement has since been incorporated as Item II.D.1 of NUREG-0737, Clarification of TMI Action Plan Requirements (Reference 2), which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

1. Conduct testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

2. Determine valve expected operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2.

3. Choose the single failures such that the dynamic forces on the safety and relief valves are maximized.

4. Use the highest test pressures predicted by conventional safety analysis procedures.

5. Include in the relief and safety valve qualification program the qualification of the associated control circuitry.

6. Provide test data for Nuclear Regulatory Commission (NRC) staff review and evaluation, including criteria for success or failure of valves tested.

7. Submit a correlation or other evidence to substantiate that the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must be considered.

8. Qualify the plant specific safety and relief valve piping and supports by comparing to test data and/or performing appropriate analysis.

2. PWR OWNERS' GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer power operated relief valves, safety valves, block valves, and associated piping systems. The Commonwealth Edison Co. (CECo), owner of the Byron Nuclear Power Station, Units 1 and 2, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which were transmitted to the NRC by Reference 3. The applicability of these reports is discussed below.

EPRI developed a plan (Reference 4) for testing PWR safety, relief, and block valves under conditions which bound actual plant operating conditions. EPRI, through the valve manufacturers, identified the valves used in the overpressure protection systems of the participating utilities. Representative valves were selected for testing with a sufficient number of the variable characteristics that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). EPRI, through the Nuclear Steam Supply System (NSSS) vendors, evaluated the FSARs of the participating utilities and arrived at a test matrix which bounded the plant transients for which overpressure protection would be required (Reference 6).

EPRI contracted with Westinghouse Electric Corp. to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Byron, Units 1 and 2, were designed by Westinghouse this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were tested at the Combustion Engineering Company, Kressinger Development Laboratory, located in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valves tests are reported in Reference 9.

The primary objective of the EPRI/C-E Valve Test Program was to test each of the various types of primary system safety valves used in PWRs for the full range of fluid conditions under which they may be required to operate. The conditions selected for test (based on analysis) were limited to steam, subcooled water, and steam to water transition. Additional objectives were to (a) obtain valve capacity data, (b) assess hydraulic and structural effects of associated piping on valve operability, and (c) obtain piping response data that could ultimately be used for verifying analytical piping models.

Transmittal of the test results meets the requirement of Item 6 of Section 1.2 to provide test data to the NRC.

3. PLANT SPECIFIC SUBMITTAL

A preliminary plant specific evaluation of the adequacy of the overpressure protection system for Byron, Units 1 and 2, was submitted by CECo to the NRC on October 26, 1982 (Reference 11). This was followed by a submittal of additional information regarding piping and support adequacy on December 30, 1983 (Reference 12). A request for additional information was submitted to CECo by the NRC on March 26, 1987 to which CECo responded on December 2, 1987 (References 13 and 14).

The response of the overpressure protection system to Anticipated Transients Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review. Neither the Licensee nor the NRC have evaluated the performance of the system for these events.

4. REVIEW AND EVALUATION

4.1 Valves Tested

Byron, Units 1 and 2, are four-loop PWRs designed by the Westinghouse Electric Co. Each unit is equipped with three (3) safety valves, two (2) PORVs, and two (2) PORV block valves in its overpressure protection system. The safety valves are 6-in. Crosby Model HB-BP-86, 6M6, spring loaded valves with loop seal internals. The design set pressure is 2485 psig and the rated steam flow capacity is 420,000 lbm/h. The PORVs are 3-in. Copes-Vulcan Model D-100-160 globe valves with 316 SS stellited plugs and 17-4 PH cages. The PORV opening set pressure is 2335 psig and the rated steam flow capacity is 210,000 lbm/h. The PORV block valves are 3-in. Westinghouse Model 3000GM88 gate valves with Limitorque SB-00-15 motor operators. The inlet pipe to the safety valve includes a hot loop seal (282°F); the inlet to the PORV has a cold loop seal (170°F).

Safety valves, PORVs, and PORV block valves identical to those used at Byron, Units 1 and 2, were included in the EPRI tests. Since there is no difference between the valves tested and the valves installed at the plant, the test results for these valves are directly applicable to Byron, Units 1 and 2. Therefore, those parts of the criteria of Items 1 and 7 as identified in Section 1.2 of this report regarding applicability of the test valves are fulfilled.

4.2 Test Conditions

As stated above, Byron, Units 1 and 2, are four-loop PWRs designed by the Westinghouse Electric Corp. The valve inlet fluid conditions that bound the overpressure transients for Westinghouse designed PWR Plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection, and cold overpressurization events. The expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

4.2.1 FSAR Steam Transients

For the Byron PWRs, the limiting FSAR transients resulting in steam discharge through the safety valves alone and in steam discharge through both the safety and relief valves are the loss of load event (for maximum pressurizer pressure) and the locked rotor event (for the maximum pressurization rate).

In the case when the safety valves actuate alone, the maximum pressurizer pressure and maximum pressurization rate are predicted to be 2555 psia and 144 psi/s, respectively. The maximum developed backpressure in the outlet piping is 533 psia (Reference 14). Hot loop seals are used at the safety valve inlet. The loop seal temperature at the inlet to the valve is 282°F. The safety valves at Byron, Units 1 and 2, use manufacturer's recommended ring settings.

EPRI tests representative of the valve inlet fluid conditions for the limiting transient were selected for the plant specific evaluation. In selecting the EPRI tests, the safety valve ring settings were considered. For steam flow conditions, four loop seal discharge tests (Test No. 929, 1406, 1415, 1419) were applicable to Byron, Units 1 and 2. These tests were performed with valve ring settings representative of the typical ring settings used in Westinghouse PWRs including Byron. The ring settings used in these tests were (-71, -18) or (-77, -18). These represent the upper and lower ring positions measured from the level position referenced to the bottom of the disc ring. Since both the test ring settings and the in-plant ring settings were determined by the valve manufacturer, the Crosby Valve and Gage Co., using the same methods and the same standard of performance, these two sets of ring settings are considered comparable to each other. The loop seal temperature measured in the tests ranged from 90 to 360°F at the valve inlet. The maximum pressurizer (tank 1) pressures were in the range of 2675 to 2760 psia and the pressurization rate was 90 to 360 psi/s. The backpressures developed in the tests were 245 to 710 psia. The above data show that the inlet fluid conditions and backpressures of these tests envelop the corresponding fluid data predicted for the Byron safety valves.

When both the safety valves and PORVs are actuated, the maximum pressurizer pressure is predicted to be 2532 psia and the maximum pressurization rate is 130 psi/s. In the EPRI tests on the Copes-Vulcan PORV, the maximum steam pressure at valve opening was 2715 psia, which bounds the predicted pressure at Byron. A test simulating loop seal discharge was conducted at a pressure of 2725 psia with a water temperature of 134°F at the valve inlet. The backpressure developed at the outlet of the PORVs is not an important consideration because the air operated PORVs used at the Byron plant are not sensitive to backpressure (Reference 6). Therefore, the EPRI test inlet fluid conditions for the PORV with steam discharge are representative of the plant specific transient conditions.

4.2.2 FSAR Liquid Transients

The limiting FSAR transient resulting in liquid discharge through the PORVs and safety valves is the main feedline break accident (Reference 7). The submittal did not address the transient conditions that involve liquid discharge through safety valves and PORVs. The Licensee stated that its decision not to evaluate the Byron 1 and 2 safety valves and PORVs for liquid discharge was based on a probabilistic risk study presented in Appendix A of Reference 11. This study concluded that liquid discharge in the feedline break, extended high pressure injection, or low temperature overpressurization event was very unlikely to occur. Therefore, liquid discharge through the safety valves and PORVs was not considered.

However, the Westinghouse Valve Inlet Fluid Condition Report (Reference 7) stated that the main feedline pipe rupture event was classified as a Class IV licensing event. That is, one which was not expected to take place but was postulated because its consequences include the potential for the release of a significant amount of radioactive material. Also, NUREG-0737 specifically requires the safety valves and PORVs be qualified for inlet fluid conditions resulting from transients and accidents referenced in Regulatory Guide 1.70, Rev. 2. The feedwater line break is specifically defined in Regulatory Guide 1.70, Rev. 2. From a review of the feedwater line break analysis for Byron (see below), it is

clear that the feedwater line break is most likely to be the limiting transient for providing high pressure liquid to the safety valves, a fluid for which they were not originally designed. Therefore, in accordance with the NUREG requirements, the safety valves and PORVs should be qualified for inlet conditions typical of the feedline break event even though the probabilistic analysis showed the frequency of occurrence is extremely low.

The Byron feedline break analysis indicated that the safety valves and PORVs opened on saturated steam at about seven minutes into the transient and steam to saturated liquid transition would follow at thirteen minutes into the event (Appendix A, Reference 11). It is apparent that liquid discharge through the safety valves and PORVs cannot be ruled out. Therefore, valve operability will be reviewed using the feedline break data provided in Reference 7.

Reference 7 showed that, in a feedline break accident at Byron, the maximum pressure at the safety valve inlet during liquid discharge was calculated to be 2508 psia and the pressurization rate was 3.5 psi/s. Fluid temperatures at the valve inlet range from 615 to 635°F and the maximum liquid surge rate into the pressurizer is 569 gpm.

In a feedline break accident resulting in safety valve actuation, water discharge is always preceded by steam and steam to water transition flows. Among the EPRI tests performed on the 6M6 valve, Tests 931a and 931b were performed for loop seal/steam, steam to water transition, and water discharge conditions. The valve ring settings and inlet pipe configuration used in these tests were comparable to those of the in-plant safety valves. In Test No. 931a, the maximum inlet pressure was 2578 psia. The pressurization rate was 2.5 psi/s, the inlet fluid temperature was 117°F and the tank fluid temperature was 635°F. After the valve closed in Test 931a, the system was allowed to repressurize and the valve cycled on approximately 640°F water (Test 931b). Because the inlet temperature and pressure of the tests compare favorably with the predicted in-plant conditions, the results of these tests are applicable to the Byron safety valves.

The expected fluid conditions at the safety valve inlet were based on a Westinghouse analysis that assumed the PORVs were not operable during the feedline break transient. If the PORVs are operable, the same fluid conditions postulated for the safety valve inlet can also be expected at the PORV inlet (Reference 6). In the EPRI tests, high temperature water discharge and steam to water transition tests were performed with the Copes-Vulcan PORV. In the water discharge test, Test No. 76-CV-316-2W, the maximum pressure at the valve inlet was 2535 psia and the temperature was 647°F. In the transition test, Test No. 77-CV-316-7S/W, the maximum inlet pressure was 2532 psia and the water temperature was 657°F. The inlet fluid conditions for these tests compare well with the predicted maximum pressure and temperature of 2508 psia and 635°F for the Byron plant. Therefore, these tests are adequate to represent the in-plant PORV performance in the feedline break event.

4.2.3 Extended High Pressure Injection Event

The limiting extended high pressure injection event is the spurious actuation of the safety injection system at power (Reference 7). For a four-loop plant, both the safety valves and PORVs will be challenged. Both steam and water discharge are expected. In this event, however, the safety valves or PORVs open on steam and liquid discharge would not be observed until the pressurizer becomes water solid. According to Reference 7, this would not occur until at least 20 minutes into the event which allows ample time for operator action. Thus the potential for liquid discharge in extended HPI events can be disregarded.

4.2.4 Low Temperature Overpressurization Transient

The PORV is used for low temperature overpressure protection (LTOP) during the low temperature stages of reactor start-up and shutdown operations. The expected valve inlet conditions in the low temperature overpressure protection mode were given in Reference 14 as pressures ranging from 350 to 2450 psig and water temperatures ranging from 70 to 450°F. It is also possible for the PORV to actuate under high and low pressure steam conditions.

For steam discharge through the PORV, the high pressure steam tests discussed above cover the low pressure steam conditions predicted for LTOP. For water discharge conditions, there were two low pressure and low temperature water tests performed on the Copes-Vulcan PORV with stellited plug and 17-4 PH cage. The tests were conducted at an inlet pressure of 675 psia and water temperatures of 105 and 442°F, respectively. In addition there was a high pressure water test with a pressure of 2535 psia and a water temperature of 647°F. These conditions are representative of those at Byron. Therefore, the EPRI tests can be used to evaluate the performance of the Byron PORV during LTOP transients.

4.2.5 PORV Block Valve Fluid Conditions

The block valves at Byron are Westinghouse 3 in. gate valves, Model 3000GM88, with Limitorque SB-00-15 operators. The block valves are required to operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the PORVs. The 3-in. Westinghouse 3GM88 valve with Limitorque SB-00-15 operator was subjected to full pressure steam conditions (to 2485 psia) in the EPRI-Marshall tests. Later tests were performed by Westinghouse using subcooled water as a test fluid. Tests were conducted at differential pressures across the valve discharge ranging from 800 to 2600 psi and with flow rates ranging from 60 to 600 gpm. The Westinghouse tests showed that the torque required to operate the valve is almost entirely dependent on the differential pressure and is rather insensitive to momentum loading. Thus operability of the valve is nearly the same for steam and liquid discharge, and the results of the EPRI-Marshall tests can be used to assess operability of the valve.

4.2.6 Test Conditions Summary

The test sequences and analyses described above demonstrate that the test conditions bound the conditions for the plant valves. They also verify that Items 2 and 4 of Section 1.2 were met, in that conditions for the operational occurrences were determined and the highest predicted pressures were chosen for the test. The part of Item 7, which requires showing that the test conditions are equivalent to conditions prescribed in the FSAR, was also met.

4.3 Operability

4.3.1 Safety Valves

The EPRI tests representative of the steam discharge condition for the Byron safety valves are the loop seal tests on the Crosby 6M6 valve, Test No. 929, 1406, 1415, 1419. In all these tests (except Test No. 1415), the valve fluttered or chattered during loop seal discharge and stabilized when steam flow started. The valve opened within ±2% of the design set pressure and closed with 5.1 to 9.4% blowdown. Up to 111% of rated flow was achieved at 3% accumulation with valve lift positions at 92 to 94% of rated lift. These tests demonstrated that the valve performed its function in spite of the initial chatter during loop seal discharge.

In Test 1419, the valve chattered on closing and the test was terminated after the valve was manually opened to stop the chatter. This result does not indicate a valve closing problem for the Byron safety valve since an identical test (Test 1415) had already demonstrated that the valve performed satisfactorily and exhibited no sign of instability. The closing chatter in Test 1419 may possibly be a result of the repeated actuation of the valve in loop seal and water discharge tests. As shown in Table 4.3.1, the 6M6 test valve was subjected to seventeen steam, water, and transition tests. In the first four or five tests, the valve fluttered and chattered during loop seal discharge but stabilized and closed successfully. After Test 913, there were four instances in which the test was terminated due to chattering on closing. Galled guiding surfaces and damaged internal parts were found during each inspection and the damaged parts were refurbished or replaced before the next test started. The test results showed that the valve performed acceptably in the test following each repair, but that closing chatter recurred in a subsequent test. Test 1415 was performed immediately after valve maintenance and the valve performed stably. The next test (Test 1419) encountered chatter in closing though it was a repeat of Test 1415. These results suggest that inspection and maintenance are important to the continued operability of the valves. The Licensee should develop a formal procedure requiring that the safety valves be inspected

TABLE 4.3.1 EPRI TESTS ON CROSBY HB-BP-86 6M6 SAFETY VALVE

Seqn No.	Test No.	Ring Setting	Test Type		Stability	Leakage Pre (gpm)	Leakage Post (gpm)
1	903	1	Steam		Stable	0	0
				Inspection/Repair			
2	906a,b,c	1	L.S.		Stable	0	0
3	908	1	L.S.		f/c	0	0
4	910	1	L.S.		f/c	0	0
				Inspection/Repair			
5	913	2	L.S.		f/c	0	1.0
6	914a,b,c	2	L.S. Transition		Terminated	0	Large
				Inspection/Repair			
7	917	3	L.S.		f/c	0	0
8	920	3	L.S.		Terminated	0	0
				Inspection/Repair			
9	923	3	L.S.		f/c	0	0
10	926a,b,c,d	3	Transition		Stable	0.36	0.08
				Inspection/Repair			
11	929	4	L.S.		f/c	0	0
12	931a,b	4	L.S. Transition		c	0	0
13	932	4	Water		Terminated	0	--
				Inspection/Repair			
14	1406	4	L.S.		f/c	0	0.63
				Inspection/Repair			
15	1411	4	Steam		Stable	0.76	0.37
				Inspection/Repair			
16	1415	4	L.S.		Stable	0	0
17	1419	4	L.S.		Terminated	0	1.5
				Inspection/Repair			

c--chatter

f/c--flutter/chatter

L.S.--loop seal

Ring setting--four different sets of ring settings were tested. Actual ring positions not shown.

Terminated--Test terminated after valve was manually opened to stop chatter.

after each actuation and the procedure should be incorporated into the plant operating procedures or licensing documents such as the plant technical specifications.

Blowdown in these tests (5.1 to 9.4%) was in excess of the 5% value specified by the valve manufacturer and the ASME Code. Westinghouse performed an analysis, "Safety Valve Contingency Analysis in Support of the EPRI Safety/Relief Valve Testing Program--Volume 3: Westinghouse Systems," EPRI NP-2047-LD, October 1981, on the effects of increased blowdown and concluded that no adverse effects on plant safety occurred in that the reactor core remained covered. Therefore, the increased blowdown that occurred in the Crosby 6M6 steam tests is considered acceptable.

As discussed in Section 4.2.2, the limiting FSAR transient resulting in liquid discharge is the main feedline break accident. Tests 931a and 931b with typical plant ring settings of (-71, -18) simulate the expected Byron feedwater line break conditions. Test 931a was a loop seal/steam/water transition test. The 6M6 valve initially opened, fluttered or chattered in a partial lift position during loop seal discharge, then popped open, stabilized on steam, and closed with a 12.7% blowdown. Test 931b was a saturated water test. The 6M6 valve opened on 640°F water, chattered, and then stabilized. The valve closed with 4.8% blowdown. For these tests the valve opened within -1% and +3% of the set pressure. The maximum calculated surge rate at Byron, Units 1 and 2, during the feedline break transient is 569 gpm. The 6M6 valve tested by EPRI passed 2355 gpm at 2415 psia and 641°F which is much higher than the predicted flow rate for Byron. The above results demonstrate that the Byron safety valves would be adequate to perform the required water relief function.

Bending moments as high as 298,750 in-lb (Test 908) were induced on the discharge flange of the Crosby 6M6 test valve, which had no adverse effect on valve performance. Because this applied moment exceeds the maximum estimated bending moment of 177,460 in-lb for the Byron valves (Reference 14), the performance of the plant valves is also expected to be unaffected by bending moments imposed during discharge transients.

For the tests to be an adequate demonstration of safety valve stability, the test inlet piping pressure drop should exceed the plant pressure drop. The test inlet pressure drop for the Crosby 6M6 valve on the loop seal configuration was 263 psid on opening and 181 psid on closing. The values calculated for the Byron, Units 1 and 2, safety valves were 235 and 120 psid for opening and closing, respectively (see Reference 14). Therefore, the plant valves should be as stable as the test valves.

4.3.2 Power Operated Relief Valves

The EPRI tests on the Copes-Vulcan PORV with 316 SS stellited plug and 17-4 PH cage demonstrated that the valve opened and closed on demand in loop seal/steam, steam, water, and steam to water transition conditions. The opening and closing time were within the 2.0 second opening and closing time normally required for Westinghouse PWRs. The lowest steam flow rate observed in the tests was 232,000 lb/h which exceeded the rated flow of 210,000 lb/h for the Byron PORVs.

The predicted value of the maximum bending moment induced at the Byron PORV discharge flange was calculated to be 65,860 in-lb (Reference 14). This exceeds the maximum tested bending moment of 43,000 in-lb. Therefore, operability of the Byron PORVs with the maximum expected bending moment cannot be shown directly using the EPRI data. However, CECo stated in Reference 14 that Westinghouse qualified the Byron PORVs for these loads by analysis. The maximum loading of 65,860 in-lb results in stresses of 71% of the allowable value. Therefore, the Copes-Vulcan PORVs at Byron are expected to operate with the higher bending moments. This is reasonable because the EPRI test bending moments represent the maximum value tested, not the maximum permissible bending moment.

4.3.3 Electric Control Circuitry

NUREG-0737 Item II.D.1 states that the control circuitry associated with the PORVs shall also be qualified for design basis accidents and transients. The Nuclear Regulatory Commission staff agreed that meeting the

licensing requirements of 10 CFR 50.49 for this electrical equipment is satisfactory and that specific testing per the NUREG-0737 requirements is not necessary. CECo included the PORV controls that would be located in a harsh environment in the Byron 10 CFR 50.49 environmental qualification program (Reference 14) thereby ensuring that the control circuitry will function properly. CECo stated the NRC reviewed and approved the Byron environmental qualification program (NUREG-0876, Supplement 5).

4.3.4 PORV Block Valves

The PORV block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2, results from the high pressure steam tests can be used to evaluate valve operation over the full range of inlet conditions. The tests on the Westinghouse 3GM88 valve with Limitorque SB-00-15 actuator showed that the valve successfully opened and closed on command once the torque switch was set to its maximum. The plant block valves were also modified by adjusting the torque switches to optimal values for opening and closing thrust, adjusting the pinion gear ratios, and changing the wiring from torque controlled to limit controlled stroking. With these changes, the plant valves are expected to operate acceptably.

4.3.5 Operability Summary

The above discussion, demonstrating that the valves operated satisfactorily, verifies that the part of Item 1 of Section 1.2, which requires conducting tests to qualify the valves, and that part of Item 7, which requires the effect of discharge piping on operability be considered, were met provided the Licensee documents a formal procedure for the inspection of the safety valves as discussed in Section 4.3.1. Also, the licensing action for 10 CFR 50.49 satisfies Item 5 of Section 1.2.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports upstream and downstream of the safety valves and PORVs extending from the pressurizer nozzle to the pressurizer relief tank. The piping was designed for deadweight, internal pressure, thermal expansion, earthquake, and safety and relief valve

discharge conditions. The calculation of the time histories of hydraulic forces due to valve discharge, the method of structural analysis, and the load combinations and stress evaluation are discussed below.

4.4.1 Thermal Hydraulic Analysis

Pressurizer fluid conditions were selected for use in the thermal hydraulic analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI, and low temperature overpressurization events including the single failure that would maximize the forces on the valve.

The safety valve and PORV discharge transients were analyzed as two separate events. This approach is acceptable, since the safety valves and PORVs have different set points and will not lift simultaneously. Also the sequential discharge of the PORVs and safety valves (i.e., PORV discharge followed by safety valve discharge) would not generate higher piping loads than the separate PORV and safety valve discharge events. Therefore, the valve discharge conditions considered for the piping and support analyses were the steam discharge condition resulting from the simultaneous actuation of all safety valves and the steam discharge condition resulting from the simultaneous actuation of all PORVs. Because water seals are maintained upstream of the safety valves and PORVs, steam discharge conditions would generate the highest loads on the piping system when the water slug is expelled from the loop seal and forced down the discharge piping. Therefore, the selection of the steam discharge cases as the limiting conditions for the evaluation of the piping loads is considered adequate.

For this analysis, steam at a pressure of 2575 psia and enthalpy of 1130 Btu/lb was assumed to be discharged through the safety valves. A loop seal having an enthalpy distribution based upon a temperature profile consistent with EPRI hot loop seal Test 917 was assumed to be present upstream of the safety valves. That is, a temperature of approximately 300°F at the valve inlet and saturation temperature at the steam-water interface. The safety valves were assumed to open linearly in 0.040 s.

The initial conditions for steam discharge of the PORVs were: 2350 psia and enthalpy of 1162.4 Btu/lb. Cold water seals (170°F) were assumed upstream of the PORVs. The PORVs were assumed to open linearly in 1.0 s.

The valve opening times used for the safety valves and PORVs were greater than those measured in the EPRI tests. For the Crosby 6M6 valves used at Byron, measured pop times were on the order of 0.020 s. For the Copes-Vulcan PORVs, the measured main disk opening time was on the order of 0.40 s. However, with Byron using loop seals upstream of both the safety valves and PORVs, the opening times used in the analysis are considered adequate because the peak piping loads are due to the passage of the loop seals and valve characteristics such as opening time are less important in the calculation of these loads. This conclusion is supported by the analysis of EPRI Test 917 presented in Reference 15. When a valve opening time of 0.090 s was used, compared to a measured pop time of 0.015 s, the measured forces were still adequately calculated.

The thermal hydraulic analysis was performed using the Westinghouse computer code, ITCHVALVE. ITCHVALVE calculates the fluid parameters as a function of time. The unbalanced force or wave force in each piping segment is calculated from the fluid properties obtained from the ITCHVALVE analysis using another Westinghouse program, FORFUN. The forcing functions on the piping system resulting from the fluid transients were obtained from these calculations.

The adequacy of the ITCHVALVE/FORFUN programs for thermal hydraulic analyses was verified by comparing the analytical and test results for thermal hydraulic loadings in safety valve discharge piping for two EPRI tests (Test Nos. 908 and 917). The detailed comparisons of the ITCHVALVE predicted force time histories and the EPRI test results were presented in Reference 14 and these comparisons are considered satisfactory.

The thermal hydraulic and stress analyses of the Byron safety valve and PORV piping and supports were performed by the Westinghouse Electric Co. as a consultant to the Licensee. The typical Westinghouse analysis for such piping systems was fully reviewed in previous submittals for similar PWR plants such as Diablo Canyon Units 1 and 2 (Reference 16). The method of

analysis used by Westinghouse including the analysis assumptions, the structural modeling as well as the key parameters used in computer inputs such as the node spacing, calculation time interval, valve opening time, etc. was examined and found to be acceptable. Because the Byron piping analysis followed the same method and procedure used in previous Westinghouse analyses, the analysis method is considered acceptable. The safety valve and PORV flow rates used in the analysis were greater than 120% of the rated flow. The conservative factor contained in these flow rates is more than sufficient to account for the 10% derating of the safety valve required by the ASME Code and includes allowance for uncertainties or errors.

4.4.2 Stress Analysis

The structural responses of the piping system due to safety valve/PORV discharge transients were calculated using the modal superposition method. The fluid force time histories generated from the FORFUN program were used as forcing functions on the structure. The Westinghouse series of structural analysis programs, WESTDYN, FIXFM3, and WESTDYN2, were used to calculate the piping natural frequencies, mode shapes, nodal displacements, and the internal forces and support reactions. The FIXFM3 code calculated the displacements at the structural node points using the forcing functions generated by FORFUN and the modal data from WESTDYN. The structural displacements were then used by WESTDYN2 to compute the piping internal loads and support loads.

The WESTDYN series of structural programs mentioned above was previously reviewed and approved by the NRC (Reference 17). The adequacy of these programs for piping discharge analysis was further verified by comparing the solutions generated by these programs with the EPRI safety valve test results (Reference 18).

The important parameters in the structural analysis were reviewed and found acceptable. The time step size was 0.0005 s. Damping of 1% was used for the OBE and 2% for the SSE. Lumped mass spacing was determined to ensure all appropriate mode shapes were accurately represented. For the thermal hydraulic analysis, the cutoff frequency was greater than 333 Hz.

The piping upstream of the safety valves and PORVs was analyzed for the requirements of the ASME Code, Section III, 1977 Edition, Addenda through Summer 1979. The downstream piping was analyzed for the requirements of the ANSI B31.1 Code, 1973 Edition, Summer 1973 Addenda. The load combinations and stress limits used to evaluate the upstream and downstream piping are identical to those recommended by EPRI (Reference 19). The Licensee provided a comparison of the highest stresses in the piping against the applicable stress limits for the load combinations defined above. All stresses were within allowable stress limits (Reference 14).

The upstream pipe supports were designed in accordance with ASME Section III, Subsection NF, and the downstream supports were designed in accordance with ANSI B31.1 (Reference 14). The load combinations were consistent with the load combinations in the EPRI Submittal Guide (Reference 19), and all stresses were less than the code allowables.

In EPRI tests performed on the Crosby 6M6 safety valve, pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as the loop seal water was discharged. This phenomenon was not accounted for in the structural analysis of the system. The piping upstream of the safety valves in the EPRI tests was 8-in. Schedule 160 and 6-in. Schedule XX while at Byron, Units 1 and 2, the piping is 6-in. Schedule 160. The test piping did not sustain any discernible damage during the tests. Thus, the plant piping is also not expected to be damaged during similar oscillations, and an analysis for these pressure oscillations is not necessary for this plant.

4.4.3 Piping and Support Summary

The selection of a bounding case for the piping evaluation and the piping and support analysis demonstrate that the requirements of Item 3 and Item 8 of Section 1.2 outlined in this report were met.

5. EVALUATION SUMMARY

The Licensee for Byron, Units 1 and 2, provided an acceptable response to the requirements of NUREG-0737, and thereby reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met. The rationale for this conclusion is given below.

The Licensee participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant steam discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Licensee's justifications indicated direct applicability of the prototypical valve and valve performances to the in-plant valves and systems intended to be covered by the generic test program. The Licensee must document a formal procedure to inspect the safety valves each time they discharge the loop seal or water. The plant specific piping was shown by analysis to meet code requirements.

Thus, the requirements of Item II.D.1 of NUREG-0737 were met (Items 1-8 in Paragraph 1.2) and, thereby demonstrate by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15). Furthermore, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment was constructed in accordance with high quality standards (General Design Criterion No. 30).

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

May 21, 1990

MAY 25 1990

Docket Nos. STN 50-456
and STN 50-457

Mr. Thomas J. Kovach
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Dear Mr. Kovach:

SUBJECT: NUREG-0737, ITEM II.D.1, PERFORMANCE TESTING ON RELIEF AND SAFETY VALVES FOR BRAIDWOOD STATION, UNIT NOS. 1 AND 2 (TAC NOS. 64019 AND 64046)

The enclosed Technical Evaluation Report (TER) was prepared with the assistance of EG&G, Idaho, under contract with the NRC staff. The TER provides the results of the staff and EG&G's review of Commonwealth Edison Company's (CECo), submittals in response to TMI Action Plan Requirements, NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves. The staff endorses the findings contained in the TER. Based on these results, we conclude that you have provided an acceptable response.

This TER proposed one additional requirement and the staff agrees with the finding in the report. As discussed in Section 5, in order to demonstrate continued operability of the safety valves, CECo must develop and adopt plant procedures to inspect the valves after each lift involving loop seal or water discharge.

Sincerely,

Stephen P. Sands

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Enclosure:
As stated

cc w/enclosure:
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TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
BRAIDWOOD UNITS 1 and 2
DOCKET NO. 50-456, 50-457

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ABSTRACT

Light water reactors have experienced a number of occurrences of improper performance of safety and relief valves installed in the primary coolant system. As a result, the authors of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) and subsequently NUREG-0737 (Clarification of TMI Action Plan Requirements) recommended that programs be developed and completed which would reevaluate the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. This report documents the review of these programs by the Nuclear Regulatory Commission (NRC) and their consultant, EG&G Idaho, Inc. Specifically, this review examined the response of the Licensee for the Braidwood Nuclear Power Station, Units 1 and 2, to the requirements of NUREG-0578 and NUREG-0737 and finds that the Licensee provided an acceptable response, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met.

FIN No. D6005--Evaluation of CW Licensing Actions-NUREG-0737, II.D.1

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TECHNICAL EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)

BRAIDWOOD NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NO. 50-456, 50-457

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant system. There were instances of valves opening below set pressure, valves opening above set pressure, and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island (TMI-2) sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend that programs be developed and executed which would reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient, and accident conditions. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR, are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (a) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage, (b) the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions are

not exceeded during normal operation or anticipated transient events, and (c) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure that the General Design Criteria are met, the NUREG-0578 position was issued as a requirement in a letter dated September 13, 1979 by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR), to ALL OPERATING NUCLEAR POWER PLANTS. This requirement has since been incorporated as Item II.D.1 of NUREG-0737, Clarification of TMI Action Plan Requirements (Reference 2), which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

1. Conduct testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

2. Determine valve expected operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2.

3. Choose the single failures such that the dynamic forces on the safety and relief valves are maximized.

4. Use the highest test pressures predicted by conventional safety analysis procedures.

5. Include in the relief and safety valve qualification program the qualification of the associated control circuitry.

6. Provide test data for Nuclear Regulatory Commission (NRC) staff review and evaluation, including criteria for success or failure of valves tested.

7. Submit a correlation or other evidence to substantiate that the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must be considered.

8. Qualify the plant specific safety and relief valve piping and supports by comparing to test data and/or performing appropriate analysis.

2. PWR OWNERS' GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer power operated relief valves, safety valves, block valves, and associated piping systems. The Commonwealth Edison Co. (CECo), owner of the Braidwood Nuclear Power Station, Units 1 and 2, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which were transmitted to the NRC by Reference 3. The applicability of these reports is discussed below.

EPRI developed a plan (Reference 4) for testing PWR safety, relief, and block valves under conditions which bound actual plant operating conditions. EPRI, through the valve manufacturers, identified the valves used in the overpressure protection systems of the participating utilities. Representative valves were selected for testing with a sufficient number of the variable characteristics that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). EPRI, through the Nuclear Steam Supply System (NSSS) vendors, evaluated the FSARs of the participating utilities and arrived at a test matrix which bounded the plant transients for which overpressure protection would be required (Reference 6).

EPRI contracted with Westinghouse Electric Corp. to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Braidwood, Units 1 and 2, were designed by Westinghouse this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company Marshall Steam Station located in Terrell, North Carolina. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were tested at the Combustion Engineering Company, Kressinger Development Laboratory, located in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valves tests are reported in Reference 9.

The primary objective of the EPRI/C-E Valve Test Program was to test each of the various types of primary system safety valves used in PWRs for the full range of fluid conditions under which they may be required to operate. The conditions selected for test (based on analysis) were limited to steam, subcooled water, and steam to water transition. Additional objectives were to (a) obtain valve capacity data, (b) assess hydraulic and structural effects of associated piping on valve operability, and (c) obtain piping response data that could ultimately be used for verifying analytical piping models.

Transmittal of the test results meets the requirement of Item 6 of Section 1.2 to provide test data to the NRC.

3. PLANT SPECIFIC SUBMITTAL

A preliminary plant specific evaluation of the adequacy of the overpressure protection system for Braidwood, Units 1 and 2, was submitted by CECo to the NRC on October 26, 1982 (Reference 11). This was followed by a submittal of additional information regarding piping and support adequacy on December 30, 1983 (Reference 12). A request for additional information was submitted to CECo by the NRC on March 26, 1987 to which CECo responded on December 2, 1987 (References 13 and 14).

The response of the overpressure protection system to Anticipated Transients Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review. Neither the Licensee nor the NRC have evaluated the performance of the system for these events.

4. REVIEW AND EVALUATION

4.1 Valves Tested

Braidwood, Units 1 and 2, are four-loop PWRs designed by the Westinghouse Electric Co. Each unit is equipped with three (3) safety valves, two (2) PORVs, and two (2) PORV block valves in its overpressure protection system. The safety valves are 6-in. Crosby Model HB-BP-86, 6M6, spring loaded valves with loop seal internals. The design set pressure is 2485 psig and the rated steam flow capacity is 420,000 lbm/h. The PORVs are 3-in. Copes-Vulcan Model D-100-160 globe valves with 316 SS stellited plugs and 17-4 PH cages. The PORV opening set pressure is 2335 psig and the rated steam flow capacity is 210,000 lbm/h. The PORV block valves are 3-in. Westinghouse Model 3000GM88 gate valves with Limitorque SB-00-15 motor operators. The inlet pipe to the safety valve includes a hot loop seal (282°F); the inlet to the PORV has a cold loop seal (170°F).

Safety valves, PORVs, and PORV block valves identical to those used at Braidwood, Units 1 and 2, were included in the EPRI tests. Since there is no difference between the valves tested and the valves installed at the plant, the test results for these valves are directly applicable to Braidwood, Units 1 and 2. Therefore, those parts of the criteria of Items 1 and 7 as identified in Section 1.2 of this report regarding applicability of the test valves are fulfilled.

4.2 Test Conditions

As stated above, Braidwood, Units 1 and 2, are four-loop PWRs designed by the Westinghouse Electric Corp. The valve inlet fluid conditions that bound the overpressure transients for Westinghouse designed PWR Plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection, and cold overpressurization events. The expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

4.2.1 FSAR Steam Transients

For the Braidwood PWRs, the limiting FSAR transients resulting in steam discharge through the safety valves alone and in steam discharge through both the safety and relief valves are the loss of load event (for maximum pressurizer pressure) and the locked rotor event (for the maximum pressurization rate).

In the case when the safety valves actuate alone, the maximum pressurizer pressure and maximum pressurization rate are predicted to be 2555 psia and 144 psi/s, respectively. The maximum developed backpressure in the outlet piping is 533 psia (Reference 14). Hot loop seals are used at the safety valve inlet. The loop seal temperature at the inlet to the valve is 282°F. The safety valves at Braidwood, Units 1 and 2, use manufacturer's recommended ring settings.

EPRI tests representative of the valve inlet fluid conditions for the limiting transient were selected for the plant specific evaluation. In selecting the EPRI tests, the safety valve ring settings were considered. For steam flow conditions, four loop seal discharge tests (Test No. 929, 1406, 1415, 1419) were applicable to Braidwood, Units 1 and 2. These tests were performed with valve ring settings representative of the typical ring settings used in Westinghouse PWRs including Braidwood. The ring settings used in these tests were (-71, -18) or (-77, -18). These represent the upper and lower ring positions measured from the level position referenced to the bottom of the disc ring. Since both the test ring settings and the in-plant ring settings were determined by the valve manufacturer, the Crosby Valve and Gage Co., using the same methods and the same standard of performance, these two sets of ring settings are considered comparable to each other. The loop seal temperature measured in the tests ranged from 90 to 360°F at the valve inlet. The maximum pressurizer (tank 1) pressures were in the range of 2675 to 2760 psia and the pressurization rate was 90 to 360 psi/s. The backpressures developed in the tests were 245 to 710 psia. The above data show that the inlet fluid conditions and backpressures of these tests envelop the corresponding fluid data predicted for the Braidwood safety valves.

When both the safety valves and PORVs are actuated, the maximum pressurizer pressure is predicted to be 2532 psia and the maximum pressurization rate is 130 psi/s. In the EPRI tests on the Copes-Vulcan PORV, the maximum steam pressure at valve opening was 2715 psia, which bounds the predicted pressure at Braidwood. A test simulating loop seal discharge was conducted at a pressure of 2725 psia with a water temperature of 134°F at the valve inlet. The backpressure developed at the outlet of the PORVs is not an important consideration because the air operated PORVs used at the Braidwood plant are not sensitive to backpressure (Reference 6). Therefore, the EPRI test inlet fluid conditions for the PORV with steam discharge are representative of the plant specific transient conditions.

4.2.2 FSAR Liquid Transients

The limiting FSAR transient resulting in liquid discharge through the PORVs and safety valves is the main feedline break accident (Reference 7). The submittal did not address the transient conditions that involve liquid discharge through safety valves and PORVs. The Licensee stated that its decision not to evaluate the Braidwood 1 and 2 safety valves and PORVs for liquid discharge was based on a probabilistic risk study presented in Appendix A of Reference 11. This study concluded that liquid discharge in the feedline break, extended high pressure injection, or low temperature overpressurization event was very unlikely to occur. Therefore, liquid discharge through the safety valves and PORVs was not considered.

However, the Westinghouse Valve Inlet Fluid Condition Report (Reference 7) stated that the main feedline pipe rupture event was classified as a Class IV licensing event. That is, one which was not expected to take place but was postulated because its consequences include the potential for the release of a significant amount of radioactive material. Also, NUREG-0737 specifically requires the safety valves and PORVs be qualified for inlet fluid conditions resulting from transients and accidents referenced in Regulatory Guide 1.70, Rev. 2. The feedwater line break is specifically defined in Regulatory Guide 1.70, Rev. 2. From a review of the feedwater line break analysis for Braidwood (see below), it is

clear that the feedwater line break is most likely to be the limiting transient for providing high pressure liquid to the safety valves, a fluid for which they were not originally designed. Therefore, in accordance with the NUREG requirements, the safety valves and PORVs should be qualified for inlet conditions typical of the feedline break event even though the probabilistic analysis showed the frequency of occurrence is extremely low.

The Braidwood feedline break analysis indicated that the safety valves and PORVs opened on saturated steam at about seven minutes into the transient and steam to saturated liquid transition would follow at thirteen minutes into the event (Appendix A, Reference 11). It is apparent that liquid discharge through the safety valves and PORVs cannot be ruled out. Therefore, valve operability will be reviewed using the feedline break data provided in Reference 7.

Reference 7 showed that, in a feedline break accident at Braidwood, the maximum pressure at the safety valve inlet during liquid discharge was calculated to be 2508 psia and the pressurization rate was 3.5 psi/s. Fluid temperatures at the valve inlet range from 615 to 635°F and the maximum liquid surge rate into the pressurizer is 569 gpm.

In a feedline break accident resulting in safety valve actuation, water discharge is always preceded by steam and steam to water transition flows. Among the EPRI tests performed on the 6M6 valve, Tests 931a and 931b were performed for loop seal/steam, steam to water transition, and water discharge conditions. The valve ring settings and inlet pipe configuration used in these tests were comparable to those of the in-plant safety valves. In Test No. 931a, the maximum inlet pressure was 2578 psia. The pressurization rate was 2.5 psi/s, the inlet fluid temperature was 117°F and the tank fluid temperature was 635°F. After the valve closed in Test 931a, the system was allowed to repressurize and the valve cycled on approximately 640°F water (Test 931b). Because the inlet temperature and pressure of the tests compare favorably with the predicted in-plant conditions, the results of these tests are applicable to the Braidwood safety valves.

The expected fluid conditions at the safety valve inlet were based on a Westinghouse analysis that assumed the PORVs were not operable during the feedline break transient. If the PORVs are operable, the same fluid conditions postulated for the safety valve inlet can also be expected at the PORV inlet (Reference 6). In the EPRI tests, high temperature water discharge and steam to water transition tests were performed with the Copes-Vulcan PORV. In the water discharge test, Test No. 76-CV-316-2W, the maximum pressure at the valve inlet was 2535 psia and the temperature was 647°F. In the transition test, Test No. 77-CV-316-7S/W, the maximum inlet pressure was 2532 psia and the water temperature was 657°F. The inlet fluid conditions for these tests compare well with the predicted maximum pressure and temperature of 2508 psia and 635°F for the Braidwood plant. Therefore, these tests are adequate to represent the in-plant PORV performance in the feedline break event.

4.2.3 Extended High Pressure Injection Event

The limiting extended high pressure injection event is the spurious actuation of the safety injection system at power (Reference 7). For a four-loop plant, both the safety valves and PORVs will be challenged. Both steam and water discharge are expected. In this event, however, the safety valves or PORVs open on steam and liquid discharge would not be observed until the pressurizer becomes water solid. According to Reference 7, this would not occur until at least 20 minutes into the event which allows ample time for operator action. Thus the potential for liquid discharge in extended HPI events can be disregarded.

4.2.4 Low Temperature Overpressurization Transient

The PORV is used for low temperature overpressure protection (LTOP) during the low temperature stages of reactor start-up and shutdown operations. The expected valve inlet conditions in the low temperature overpressure protection mode were given in Reference 14 as pressures ranging from 350 to 2450 psig and water temperatures ranging from 70 to 450°F. It is also possible for the PORV to actuate under high and low pressure steam conditions.

For steam discharge through the PORV, the high pressure steam tests discussed above cover the low pressure steam conditions predicted for LTOP. For water discharge conditions, there were two low pressure and low temperature water tests performed on the Copes-Vulcan PORV with stellited plug and 17-4 PH cage. The tests were conducted at an inlet pressure of 675 psia and water temperatures of 105 and 442°F, respectively. In addition there was a high pressure water test with a pressure of 2535 psia and a water temperature of 647°F. These conditions are representative of those at Braidwood. Therefore, the EPRI tests can be used to evaluate the performance of the Braidwood PORV during LTOP transients.

4.2.5 PORV Block Valve Fluid Conditions

The block valves at Braidwood are Westinghouse 3 in. gate valves, Model 3000GM88, with Limitorque SB-00-15 operators. The block valves are required to operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the PORVs. The 3-in. Westinghouse 3GM88 valve with Limitorque SB-00-15 operator was subjected to full pressure steam conditions (to 2485 psia) in the EPRI-Marshall tests. Later tests were performed by Westinghouse using subcooled water as a test fluid. Tests were conducted at differential pressures across the valve discharge ranging from 800 to 2600 psi and with flow rates ranging from 60 to 600 gpm. The Westinghouse tests showed that the torque required to operate the valve is almost entirely dependent on the differential pressure and is rather insensitive to momentum loading. Thus operability of the valve is nearly the same for steam and liquid discharge, and the results of the EPRI-Marshall tests can be used to assess operability of the valve.

4.2.6 Test Conditions Summry

The test sequences and analyses described above demonstrate that the test conditions bound the conditions for the plant valves. They verify that Items 2 and 4 of Section 1.2 were met, in that conditions for the operational occurrences were determined and the highest predicted pressures were chosen for the test. The part of Item 7, which requires showing that the test conditions are equivalent to conditions prescribed in the FSAR, was also met.

4.3 Operability

4.3.1 Safety Valves

The EPRI tests representative of the steam discharge condition for the Braidwood safety valves are the loop seal tests on the Crosby 6M6 valve, Test No. 929, 1406, 1415, 1419. In all these tests (except Test No. 1415), the valve fluttered or chattered during loop seal discharge and stabilized when steam flow started. The valve opened within ±2% of the design set pressure and closed with 5.1 to 9.4% blowdown. Up to 111% of rated flow was achieved at 3% accumulation with valve lift positions at 92 to 94% of rated lift. These tests demonstrated that the valve performed its function in spite of the initial chatter during loop seal discharge.

In Test 1419, the valve chattered on closing and the test was terminated after the valve was manually opened to stop the chatter. This result does not indicate a valve closing problem for the Braidwood safety valve since an identical test (Test 1415) had already demonstrated that the valve performed satisfactorily and exhibited no sign of instability. The closing chatter in Test 1419 may possibly be a result of the repeated actuation of the valve in loop seal and water discharge tests. As shown in Table 4.3.1, the 6M6 test valve was subjected to seventeen steam, water, and transition tests. In the first four or five tests, the valve fluttered and chattered during loop seal discharge but stabilized and closed successfully. After Test 913, there were four instances in which the test was terminated due to chattering on closing. Galled guiding surfaces and damaged internal parts were found during each inspection and the damaged parts were refurbished or replaced before the next test started. The test results showed that the valve performed acceptably in the test following each repair, but that closing chatter recurred in a subsequent test. Test 1415 was performed immediately after valve maintenance and the valve performed stably. The next test (Test 1419) encountered chatter in closing though it was a repeat of Test 1415. These results suggest that inspection and maintenance are important to the continued operability of the valves. The Licensee should develop a formal procedure requiring that the safety

TABLE 4.3.1 EPRI TESTS ON CROSBY HB-BP-86 6M6 SAFETY VALVE

Seqn No.	Test No.	Ring Setting	Test Type		Stability	Leakage Pre (gpm)	Leakage Post (gpm)
1	903	1	Steam		Stable	0	0
				Inspection/Repair			
2	906a,b,c	1	L.S.		Stable	0	0
3	908	1	L.S.		f/c	0	0
4	910	1	L.S.		f/c	0	0
				Inspection/Repair			
5	913	2	L.S.		f/c	0	1.0
6	914a,b,c	2	L.S. Transition		Terminated	0	Large
				Inspection/Repair			
7	917	3	L.S.		f/c	0	0
8	920	3	L.S.		Terminated	0	0
				Inspection/Repair			
9	923	3	L.S.		f/c	0	0
10	926a,b,c,d	3	Transition		Stable	0.36	0.08
				Inspection/Repair			
11	929	4	L.S.		f/c	0	0
12	931a,b	4	L.S. Transition		c	0	0
13	932	4	Water		Terminated	0	--
				Inspection/Repair			
14	1406	4	L.S.		f/c	0	0.63
				Inspection/Repair			
15	1411	4	Steam		Stable	0.76	0.37
				Inspection/Repair			
16	1415	4	L.S.		Stable	0	0
17	1419	4	L.S.		Terminated	0	1.5
				Inspection/Repair			

c--chatter

f/c--flutter/chatter

L.S.--loop seal

Ring setting--four different sets of ring settings were tested. Actual ring positions not shown.

Terminated--Test terminated after valve was manually opened to stop chatter.

valves be inspected after each actuation and the procedure should be incorporated into the plant operating procedures or licensing documents such as the plant technical specifications.

Blowdown in these tests (5.1 to 9.4%) was in excess of the 5% value specified by the valve manufacturer and the ASME Code. Westinghouse performed an analysis, "Safety Valve Contingency Analysis in Support of the EPRI Safety/Relief Valve Testing Program--Volume 3: Westinghouse Systems," EPRI NP-2047-LD, October 1981, on the effects of increased blowdown and concluded that no adverse effects on plant safety occurred in that the reactor core remained covered. Therefore, the increased blowdown that occurred in the Crosby 6M6 steam tests is considered acceptable.

As discussed in Section 4.2.2, the limiting FSAR transient resulting in liquid discharge is the main feedline break accident. Tests 931a and 931b with typical plant ring settings of (-71, -18) simulate the expected Braidwood feedwater line break conditions. Test 931a was a loop seal/steam/water transition test. The 6M6 valve initially opened, fluttered or chattered in a partial lift position during loop seal discharge, then popped open, stabilized on steam, and closed with a 12.7% blowdown. Test 931b was a saturated water test. The 6M6 valve opened on 640°F water, chattered, and then stabilized. The valve closed with 4.8% blowdown. For these tests the valve opened within -1% and +3% of the set pressure. The maximum calculated surge rate at Braidwood, Units 1 and 2, during the feedline break transient is 569 gpm. The 6M6 valve tested by EPRI passed 2355 gpm at 2415 psia and 641°F which is much higher than the predicted flow rate for Braidwood. The above results demonstrate that the Braidwood safety valves would be adequate to perform the required water relief function.

Bending moments as high as 298,750 in-lb (Test 908) were induced on the discharge flange of the Crosby 6M6 test valve, which had no adverse effect on valve performance. Because this applied moment exceeds the maximum estimated bending moment of 177,460 in-lb for the Braidwood valves (Reference 14), the performance of the plant valves is also expected to be unaffected by bending moments imposed during discharge transients.

For the tests to be an adequate demonstration of safety valve stability, the test inlet piping pressure drop should exceed the plant pressure drop. The test inlet pressure drop for the Crosby 6M6 valve on the loop seal configuration was 263 psid on opening and 181 psid on closing. The values calculated for the Braidwood, Units 1 and 2, safety valves were 235 and 120 psid for opening and closing, respectively (see Reference 14). Therefore, the plant valves should be as stable as the test valves.

4.3.2 Power Operated Relief Valves

The EPRI tests on the Copes-Vulcan PORV with 316 SS stellited plug and 17-4 PH cage demonstrated that the valve opened and closed on demand in loop seal/steam, steam, water, and steam to water transition conditions. The opening and closing time were within the 2.0 second opening and closing time normally required for Westinghouse PWRs. The lowest steam flow rate observed in the tests was 232,000 lb/h which exceeded the rated flow of 210,000 lb/h for the Braidwood PORVs.

The predicted value of the maximum bending moment induced at the Braidwood PORV discharge flange was calculated to be 65,860 in-lb (Reference 14). This exceeds the maximum tested bending moment of 43,000 in-lb. Therefore, operability of the Braidwood PORVs with the maximum expected bending moment cannot be shown directly using the EPRI data. However, CECo stated in Reference 14 that Westinghouse qualified the Braidwood PORVs for these loads by analysis. The maximum loading of 65,860 in-lb results in stresses of 71% of the allowable value. Therefore, the Copes-Vulcan PORVs at Braidwood are expected to operate with the higher bending moments. This is reasonable because the EPRI test bending moments represent the maximum value tested, not the maximum permissible bending moment.

4.3.3 Electric Control Circuitry

NUREG-0737 Item II.D.1 states that the control circuitry associated with the PORVs shall also be qualified for design basis accidents and transients. The Nuclear Regulatory Commission staff agreed that meeting the

licensing requirements of 10 CFR 50.49 for this electrical equipment is satisfactory and that specific testing per the NUREG-0737 requirements is not necessary. CECo included the PORV controls that would be located in a harsh environment in the Braidwood 10 CFR 50.49 environmental qualification program (Reference 14) thereby ensuring that the control circuitry will function properly. CECo stated the NRC reviewed and approved the Braidwood environmental qualification program (NUREG-1002, Supplement 2).

4.3.4 PORV Block Valves

The PORV block valve must be capable of closing over a range of steam and water conditions. As described in Section 4.2, results from the high pressure steam tests can be used to evaluate valve operation over the full range of inlet conditions. The tests on the Westinghouse 3GM88 valve with Limitorque SB-00-15 actuator showed that the valve successfully opened and closed on command once the torque switch was set to its maximum. The plant block valves were also modified by adjusting the torque switches to optimal values for opening and closing thrust, adjusting the pinion gear ratios, and changing the wiring from torque controlled to limit controlled stroking. With these changes, the plant valves are expected to operate acceptably.

4.3.5 Operability Summary

The above discussion, demonstrating that the valves operated satisfactorily, verifies that the part of Item 1 of Section 1.2, which requires conducting tests to qualify the valves, and that part of Item 7, which requires the effect of discharge piping on operability be considered, were met provided the Licensee documents a formal procedure for the inspection of the safety valves as discussed in Section 4.3.1. Also, the licensing action for 10 CFR 50.49 satisfies Item 5 of Section 1.2.

4.4 Piping and Support Evaluation

This evaluation covers the piping and supports upstream and downstream of the safety valves and PORVs extending from the pressurizer nozzle to the pressurizer relief tank. The piping was designed for deadweight, internal pressure, thermal expansion, earthquake, and safety and relief valve

discharge conditions. The calculation of the time histories of hydraulic forces due to valve discharge, the method of structural analysis, and the load combinations and stress evaluation are discussed below.

4.4.1 Thermal Hydraulic Analysis

Pressurizer fluid conditions were selected for use in the thermal hydraulic analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI, and low temperature overpressurization events including the single failure that would maximize the forces on the valve.

The safety valve and PORV discharge transients were analyzed as two separate events. This approach is acceptable, since the safety valves and PORVs have different set points and will not lift simultaneously. Also the sequential discharge of the PORVs and safety valves (i.e., PORV discharge followed by safety valve discharge) would not generate higher piping loads than the separate PORV and safety valve discharge events. Therefore, the valve discharge conditions considered for the piping and support analyses were the steam discharge condition resulting from the simultaneous actuation of all safety valves and the steam discharge condition resulting from the simultaneous actuation of all PORVs. Because water seals are maintained upstream of the safety valves and PORVs, steam discharge conditions would generate the highest loads on the piping system when the water slug is expelled from the loop seal and forced down the discharge piping. Therefore, the selection of the steam discharge cases as the limiting conditions for the evaluation of the piping loads is considered adequate.

For this analysis, steam at a pressure of 2575 psia and enthalpy of 1130 Btu/lb was assumed to be discharged through the safety valves. A loop seal having an enthalpy distribution based upon a temperature profile consistent with EPRI hot loop seal Test 917 was assumed to be present upstream of the safety valves. That is, a temperature of approximately 300°F at the valve inlet and saturation temperature at the steam-water interface. The safety valves were assumed to open linearly in 0.040 s.

The initial conditions for steam discharge of the PORVs were: 2350 psia and enthalpy of 1162.4 Btu/lb. Cold water seals (170°F) were assumed upstream of the PORVs. The PORVs were assumed to open linearly in 1.0 s.

The valve opening times used for the safety valves and PORVs were greater than those measured in the EPRI tests. For the Crosby 6M6 valves used at Braidwood, measured pop times were on the order of 0.020 s. For the Copes-Vulcan PORVs, the measured main disk opening time was on the order of 0.40 s. However, with Braidwood using loop seals upstream of both the safety valves and PORVs, the opening times used in the analysis are considered adequate because the peak piping loads are due to the passage of the loop seals and valve characteristics such as opening time are less important in the calculation of these loads. This conclusion is supported by the analysis of EPRI Test 917 presented in Reference 15. When a valve opening time of 0.090 s was used, compared to a measured pop time of 0.015 s, the measured forces were still adequately calculated.

The thermal hydraulic analysis was performed using the Westinghouse computer code, ITCHVALVE. ITCHVALVE calculates the fluid parameters as a function of time. The unbalanced force or wave force in each piping segment is calculated from the fluid properties obtained from the ITCHVALVE analysis using another Westinghouse program, FORFUN. The forcing functions on the piping system resulting from the fluid transients were obtained from these calculations.

The adequacy of the ITCHVALVE/FORFUN programs for thermal hydraulic analyses was verified by comparing the analytical and test results for thermal hydraulic loadings in safety valve discharge piping for two EPRI tests (Test Nos. 908 and 917). The detailed comparisons of the ITCHVALVE predicted force time histories and the EPRI test results were presented in Reference 14 and these comparisons are considered satisfactory.

The thermal hydraulic and stress analyses of the Braidwood safety valve and PORV piping and supports were performed by the Westinghouse Electric Co. as a consultant to the Licensee. The typical Westinghouse analysis for such piping systems was fully reviewed in previous submittals for similar PWR plants such as Diablo Canyon Units 1 and 2 (Reference 16). The method of

analysis used by Westinghouse including the analysis assumptions, the structural modeling as well as the key parameters used in computer inputs such as the node spacing, calculation time interval, valve opening time, etc. was examined and found to be acceptable. Because the Braidwood piping analysis followed the same method and procedure used in previous Westinghouse analyses, the analysis method is considered acceptable. The safety valve and PORV flow rates used in the analysis were greater than 120% of the rated flow. The conservative factor contained in these flow rates is more than sufficient to account for the 10% derating of the safety valve required by the ASME Code and includes allowance for uncertainties or errors.

4.4.2 Stress Analysis

The structural responses of the piping system due to safety valve/PORV discharge transients were calculated using the modal superposition method. The fluid force time histories generated from the FORFUN program were used as forcing functions on the structure. The Westinghouse series of structural analysis programs, WESTDYN, FIXFM3, and WESTDYN2, were used to calculate the piping natural frequencies, mode shapes, nodal displacements, and the internal forces and support reactions. The FIXFM3 code calculated the displacements at the structural node points using the forcing functions generated by FORFUN and the modal data from WESTDYN. The structural displacements were then used by WESTDYN2 to compute the piping internal loads and support loads.

The WESTDYN series of structural programs mentioned above was previously reviewed and approved by the NRC (Reference 17). The adequacy of these programs for piping discharge analysis was further verified by comparing the solutions generated by these programs with the EPRI safety valve test results (Reference 18).

The important parameters in the structural analysis were reviewed and found acceptable. The time step size was 0.0005 s. Damping of 1% was used for the OBE and 2% for the SSE. Lumped mass spacing was determined to ensure all appropriate mode shapes were accurately represented. For the thermal hydraulic analysis, the cutoff frequency was greater than 333 Hz.

The piping upstream of the safety valves and PORVs was analyzed for the requirements of the ASME Code, Section III, 1977 Edition, Addenda through Summer 1979. The downstream piping was analyzed for the requirements of the ANSI B31.1 Code, 1973 Edition, Summer 1973 Addenda. The load combinations and stress limits used to evaluate the upstream and downstream piping are identical to those recommended by EPRI (Reference 19). The Licensee provided a comparison of the highest stresses in the piping against the applicable stress limits for the load combinations defined above. All stresses were within allowable stress limits (Reference 14).

The upstream pipe supports were designed in accordance with ASME Section III, Subsection NF, and the downstream supports were designed in accordance with ANSI B31.1 (Reference 14). The load combinations were consistent with the load combinations in the EPRI Submittal Guide (Reference 19), and all stresses were less than the code allowables.

In EPRI tests performed on the Crosby 6M6 safety valve, pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as the loop seal water was discharged. This phenomenon was not accounted for in the structural analysis of the system. The piping upstream of the safety valves in the EPRI tests was 8-in. Schedule 160 and 6-in. Schedule XX while at Braidwood, Units 1 and 2, the piping is 6-in. Schedule 160. The test piping did not sustain any discernible damage during the tests. Thus, the plant piping is also not expected to be damaged during similar oscillations, and an analysis for these pressure oscillations is not necessary for this plant.

4.4.3 Piping and Support Summary

The selection of a bounding case for the piping evaluation and the piping and support analysis demonstrate that the requirements of Item 3 and Item 8 of Section 1.2 outlined in this report were met.

5. EVALUATION SUMMARY

The Licensee for Braidwood, Units 1 and 2, provided an acceptable response to the requirements of NUREG-0737, and thereby reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 were met. The rationale for this conclusion is given below.

The Licensee participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant steam discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Licensee's justifications indicated direct applicability of the prototypical valve and valve performances to the in-plant valves and systems intended to be covered by the generic test program. The Licensee must document a formal procedure to inspect the safety valves each time they discharge the loop seal or water. The plant specific piping was shown by analysis to meet code requirements.

Thus, the requirements of Item II.D.1 of NUREG-0737 were met (Items 1-8 in Paragraph 1.2) and, thereby demonstrate by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) were designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15). Furthermore, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment was constructed in accordance with high quality standards (General Design Criterion No. 30).

6. REFERENCES

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