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**Non-Proprietary**

# U.S. EPR Pressure-Temperature Limits Methodology for RCS Heatup and Cooldown

Technical Report

April 2009

AREVA NP Inc.

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## **ABSTRACT**

This technical report provides the methodology for developing pressure-temperature (P-T) limits for protecting the integrity of the reactor coolant pressure boundary (RCPB) of the U.S. EPR. The methodology includes the applicable requirements of the ASME Boiler and Pressure Vessel (BPV) Code, Section XI, Appendix G (2004 Edition) and the requirements of 10 CFR 50, Appendix G. 10 CFR 50, Appendix G requires that P-T limits be established and that the P-T limits established are in compliance with the ASME BPV Code.

The components of the RCPB are designed to withstand the effects of cyclic loading due to system pressure and temperature changes during normal heatup and cooldown of the reactor coolant system and during anticipated operational occurrences. The P-T limits provide a margin of safety to preclude non-ductile failure of the RCPB during these operational transients.

This technical report also contains a generic pressure-temperature limits report (PTLR) for the U.S. EPR based on bounding material properties.

### Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	1.0	Added paragraph to note that this report also contains the generic PTLR for the U.S. EPR design.
2	3.1	Added description of neutron fluence method per GL96-03.
3	3.2	Expanded description of ART calculation method per GL 96-03.
4	3.3	Described the material surveillance program per GL96-03.
5	3.0	Moved specific material properties to Section 6.0 (generic PTLR).
6	4.0	Moved LTOP enable temperature discussion to Section 5.4.
7	5.1 – 5.3	New section - Added LTOP setpoint methodology per GL96-03 throughout 5.0 sections.
8	5.4	Added justification for LTOP enable temperature basis.
9	6.1 – 6.5	New section - Added Generic PTLR for the U.S. EPR design satisfying GL96-03 content requirements throughout 6.0 sections.
10	ALL	Editorial changes throughout

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## Nomenclature

<b>Acronym</b>	<b>Definition</b>
AOO	Anticipated Operational Occurrence
ART	Adjusted Reference Temperature
CF	Chemistry factor
CFR	Code of Federal Regulations
EFPY	Effective Full Power Year
FF	Fluence factor
$IRT_{NDT}$	Initial Reference Temperature for Nil-Ductility Transition
ISI	Inservice Inspection
ISLH	Inservice Leak and Hydrostatic
$K_{IR}$	Crack Arrest Critical Stress Intensity Factor
$K_{Ia}$	Crack Arrest Critical Stress Intensity Factor
$K_{Ic}$	Critical or Reference Stress Intensity Factor
LCO	Limiting Condition for Operation (Technical Specifications)
LTOP	Low-Temperature Overpressure Protection
NDE	Non-Destructive Examination
PSRV	Pressurizer Safety Relief Valve
P-T	Pressure-Temperature
PTLR	Pressure-Temperature Limits Report
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
$RT_{NDT}$	Reference Temperature for Nil-Ductility Transition
WRCB	Welding Research Council Bulletin



## 1.0 INTRODUCTION

This document presents the methodology for developing pressure-temperature (P-T) limits for the reactor coolant pressure boundary (RCPB) of the U.S. EPR. This technical report is based on the methodologies of the ASME BPV Code, Section XI, Appendix G (Reference 1) and 10 CFR Part 50, Appendix G, Fracture Toughness Requirements (Reference 2).

The reactor pressure vessel (RPV) beltline materials used in the U.S. EPR satisfy Charpy upper-shelf energy requirements throughout the 60-year life of the vessel; therefore, this report does not address low upper-shelf toughness issues.

The components of the RCPB are designed to withstand the effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (i.e., heatup) and shutdown (i.e., cooldown) operations, and operational transients. Because of inherent conservatism in the methodology, the P-T limits provide a margin of safety to preclude non-ductile failure of the RCPB during changes in pressure and temperature. Due to neutron irradiation over the plant lifetime, the RPV is the component most subject to non-ductile failure. The P-T limit methodology described herein applies to the limiting components of the RPV.

This technical report also contains a generic pressure-temperature limits report (PTLR) for the U.S. EPR based on bounding material properties.

## **2.0 FRACTURE TOUGHNESS REQUIREMENTS**

### **2.1 Background**

The NRC has established fracture toughness requirements for ferritic materials in pressure-retaining components of the RCPB to provide adequate margins of safety over its service lifetime. 10 CFR 50.60, Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation (Reference 3) requires compliance with the fracture toughness requirements set forth in 10 CFR Part 50, Appendix G which additionally requires compliance with Reference 1. These fracture toughness requirements, including P-T limits, apply during inservice leak and hydrostatic (ISLH) testing and any condition of normal operation, including anticipated operational occurrences (AOO). Compliance with these requirements protects the structural integrity of the RCPB, specifically the RPV. The regulations in 10 CFR 50.60 and the associated 10 CFR Part 50, Appendix G provide the general basis for these limits.

10 CFR Part 50, Appendix G requires that P-T limits be at least as conservative as the limits obtained by following Reference 1. The U.S. EPR complies with the 10 CFR Part 50, Appendix G requirement to develop P-T limits in accordance with the specifications of Reference 1.

### **2.2 *P-T Limits and Minimum Temperature Requirements***

P-T limits are prescribed to avoid encountering pressure, temperature and temperature rate of change conditions that might cause undetected flaws to propagate and cause non-ductile failure of the RCPB. The P-T limits and minimum temperature requirements are derived from 10 CFR Part 50, Appendix G and Reference 1.

**2.2.1 10 CFR Part 50, Appendix G**

10 CFR Part 50, Appendix G Section IV Paragraphs A.2.a, b, c, and d specify the following P-T and minimum temperature requirements: (Note: A typographical error in Section IV Paragraphs A.2.a, b and c is corrected from “Table 3” to “Table 1.”)

- a. “Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in Table 1, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. In Table 1, the vessel pressure is defined as a percentage of the preservice system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.
- b. The pressure-temperature limits identified as "ASME Appendix G limits" in Table 1 require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.
- c. The minimum temperature requirements given in Table 1 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in Table 1, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in Table 1.
- d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.”

The minimum temperature requirements for the material reference temperature for nil-ductility transition ( $RT_{NDT}$ ) in Table 1, “Pressure and Temperature Requirements for the Reactor Pressure Vessel,” of 10 CFR Part 50, Appendix G (included as Table 2–1 in this report), are partially based on ASME BPV Code Section III, NB-2300 (Reference 4) and are incorporated in the development of the P-T limit curves.

### **2.2.2 ASME Code**

In 1987, Reference 1 adopted the requirements of ASME BPV Code Section III, Appendix G (Reference 5), except for the name of the fracture toughness curve,  $K_{IR}$ , which was labeled as  $K_{Ia}$ . This was done for consistency with the corresponding curve in Section XI of the ASME Code which is labeled as  $K_{Ia}$ . This was also done to place the nuclear power plant operational requirements in Reference 1 rather than in Reference 5. Section III addresses design and construction of nuclear power plants and has no provision for the effects of neutron irradiation on RPV beltline materials. If neutron fluence levels are greater than anticipated during the service lifetime of RCPB components, the P-T limits are re-evaluated to account for the potential degradation in material toughness.

The methodology of Reference 1 postulates the pre-existence of certain surface flaws in the limiting components of the RPV. The Reference 1 methods are described in Section 4.0 of this Technical Report.

### **2.2.3 Limiting Components**

The components of the RCPB are designed to withstand the effects of cyclic loads due to system pressure and temperature changes. The P-T limits establish operating limits that provide a margin to preclude non-ductile failure of the RPV and piping of the RCPB. For P-T limits, the closure head, outlet nozzle and beltline region are the three components analyzed.

The RPV is a vertically mounted cylindrical vessel consisting of forged shells, a transition ring, nozzles, and heads. There are no longitudinal seam welds. The welds in the RPV beltline region are shown on Figure 2-1.

Due to neutron radiation induced embrittlement, the beltline region of the reactor vessel is the most critical component. The closure head is important in the development of P-T limits due to preoperational bolt-up stress and the minimum temperature requirements in Table 2-1. RPV nozzles are analyzed with a postulated nozzle corner crack as

described in Welding Research Council Bulletin (WRCB) 175 (Reference 6). Other RCPB components are considered bounded by these three analyzed components.

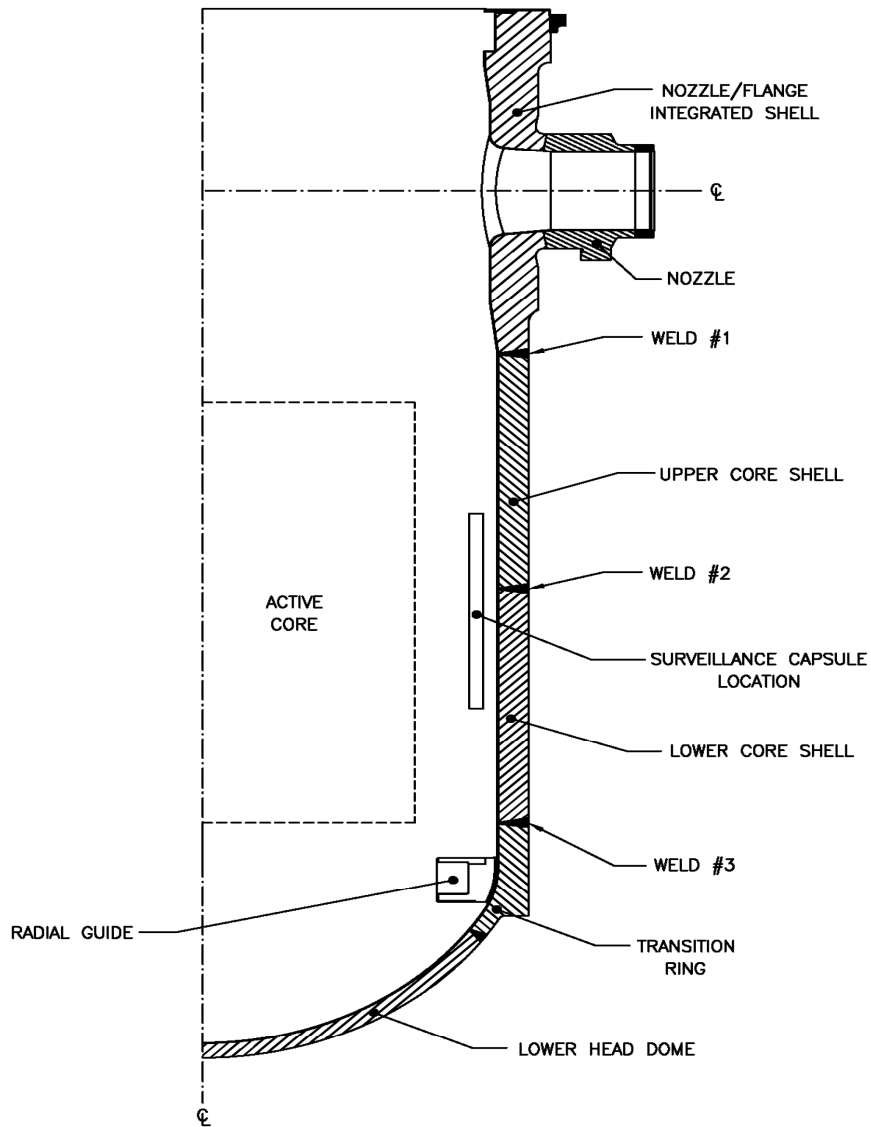
**Table 2–1 Pressure and Temperature Requirements for the Reactor Pressure Vessel**

Operating Condition	Vessel Pressure <sup>(1)</sup>	Requirements for Pressure-Temperature Limits	Minimum Temperature Requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤ 20%	ASME Appendix G Limits	(2)
1.b Fuel in the vessel	> 20%	ASME Appendix G Limits	(2) + 90 °F (6)
1.c No fuel in the vessel, (Preservice Hydrotest Only)	ALL	(Not Applicable)	(3) + 60 °F
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences (AOOs):			
2.a Core not critical	≤ 20%	ASME Appendix G Limits	(2)
2.b Core not critical	> 20%	ASME Appendix G Limits	(2) + 120 °F (6)
2.c Core critical	≤ 20%	ASME Appendix G Limits + 40 °F	Larger of [(4)] or [(2) + 40 °F]
2.d Core critical	> 20%	ASME Appendix G Limits + 40 °F	Larger of [(4)] or [(2)+160 °F]
2.e Core critical for BWR <sup>(5)</sup>	≤ 20%	ASME Appendix G Limits + 40 °F	(2) + 60 °F

Notes:

1. Percent of the preservice system hydrostatic test pressure.
2. The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
3. The highest reference temperature of the vessel.
4. The minimum permissible temperature for the inservice system hydrostatic pressure test.
5. For boiling water reactors (BWR) with water level within the normal range for power operation.
6. Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

**Figure 2-1 RPV Beltline Welds**



### 3.0 MATERIAL PROPERTIES

#### 3.1 *Neutron Fluence Methodology*

The methodology for determining the projected RPV neutron fluences used in the adjusted reference temperature calculation is defined in BAW-2241P-A, Fluence and Uncertainty Methodologies (Reference 12). The methodology in Reference 12 conforms to the guidance of Regulatory Guide 1.190 (Reference 13).

The U.S. EPR heavy reflector significantly reduces the neutron flux ( $E > 1$  MeV) on the RPV. The reduction in RPV fluence due to the heavy reflector has been determined by applying the methodology in Reference 12 and replacing the heavy reflector with downcomer water in the discrete ordinates transport code (DORT) model. A comparative evaluation was performed using Monte Carlo N-Particle Transport Code (MCNP) and demonstrates a comparable reduction in neutron flux. The comparative evaluation confirms the fluence projections.

Measured data from the material surveillance program described in Section 3.3 will supplement the calculated fluence predictions. The surveillance capsule withdrawal schedule and capsule lead factors allow indication of neutron irradiation in advance of vessel irradiation damage.

#### 3.2 *Calculation of Adjusted Reference Temperature*

The adjusted reference temperature (ART) for each material in the RPV beltline is calculated in accordance with RG 1.99, Revision 2 (Reference 7) according to the following expression:

$$\text{ART} = \text{Initial } RT_{\text{NDT}} + \Delta RT_{\text{NDT}} + \text{Margin} \quad (3-1)$$

The limiting ART values are used to derive the required minimum temperatures and the fracture toughness  $K_{Ic}$  curve in Reference 1.

**3.2.1 Initial  $RT_{NDT}$** 

The initial  $RT_{NDT}$  is the reference temperature for the RPV beltline material in the unirradiated condition, evaluated in accordance with ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331 (Reference 4). 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 test results to establish a mean and standard deviation for the class.

**3.2.2 Reference Temperature Adjustment ( $\Delta RT_{NDT}$ )**

$\Delta RT_{NDT}$  is the mean value of the adjustment in reference temperature caused by irradiation and is calculated using the following expression:

$$\Delta RT_{NDT} = (CF) * (FF) \quad (3-2)$$

Where CF is the chemistry factor and FF is the fluence factor.

The chemistry factor (CF) is determined from the copper and nickel content for each RPV beltline region material using Table 1 (for weld metals) and Table 2 (for base metals) of Reference 7. Linear interpolation is permitted. When determining the CF, the “weight percent copper” and “weight-percent nickel” are best estimate values for the material.

The fluence factor (FF) is determined as follows in accordance with Reference 7:

$$FF = f^{(0.28 - 0.10 \log f)} \quad (3-3)$$

Where  $f$  = neutron fluence ( $10^{19}$  n/cm<sup>2</sup>,  $E > 1$  MeV). The neutron fluence at any depth in the RPV wall is determined in accordance with Reference 7 as follows:

$$f = f_{\text{surf}} (e^{-0.24x}) \quad (3-4)$$



where  $f_{\text{surf}}$  ( $10^{19}$  n/cm<sup>2</sup>,  $E > 1$  MeV) is the calculated value of the neutron fluence at the inner wetted surface of the vessel, and  $x$  (inches) is the depth into the vessel wall measured from the vessel inside (wetted) surface.

To verify that the ART for each vessel beltline material is a bounding value for the RPV, plant-specific information shall be considered when available. This information includes, but is not limited to, the reactor vessel operating temperature and surveillance program results. The results from the plant-specific surveillance program must be integrated into the ART estimate if the plant-specific surveillance data have been deemed credible as judged by the following criteria:

- The materials in the surveillance capsules must be representative of those that are controlling with regard to radiation embrittlement.
- Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30 ft-lb temperature unambiguously.
- Where there are two or more sets of surveillance data from one reactor, the scatter of  $\Delta RT_{\text{NDT}}$  values must be less than 28°F for welds and 17°F for base metals. Even if the range in the capsule fluences is large (i.e., two or more orders of magnitude), the scatter may not exceed twice those values.
- The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within  $\pm 25^\circ\text{F}$ .
- The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the database for the material.

The surveillance data that are deemed credible must be used to determine a material-specific value of CF for use in Equation 3-2. The material-specific value of CF is determined from the following equation:

$$CF = \frac{\sum_{i=1}^n [A_i \times FF_i]}{\sum_{i=1}^n FF_i^2} \quad (3-5)$$

Where  $n$  is the number of surveillance data points,  $A_i$  is the measured value of  $\Delta RT_{\text{NDT}}$  adjusted to account for known differences in chemical composition and irradiation temperature between the capsules and the vessel, and  $FF_i$  is the fluence factor for each surveillance data point.

### 3.2.3 Margin

The margin is the quantity that is added to obtain conservative, upper-bound values of the ART for use in calculations required by Appendix G to 10 CFR Part 50. The margin is determined by the following expression:

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_{\Delta}^2} \quad (3-6)$$

Here,  $\sigma_I$  is the standard deviation for the initial  $RT_{\text{NDT}}$  and  $\sigma_{\Delta}$  is the standard deviation for  $\Delta RT_{\text{NDT}}$ . If a measured value of the initial  $RT_{\text{NDT}}$  for the material in question is available,  $\sigma_I$  is to be estimated from the precision of the test method. If generic mean values are used,  $\sigma_I$  is the standard deviation obtained from the set of data used to establish the mean.

The standard deviation for  $\Delta RT_{\text{NDT}}$ ,  $\sigma_{\Delta}$ , is 28°F for welds and 17°F for base metals, except that  $\sigma_{\Delta}$  need not exceed 0.5 times  $\Delta RT_{\text{NDT}}$ . For cases in which the results from a credible plant-specific surveillance program are used, the value of  $\sigma_{\Delta}$  to be used is 14°F for welds and 8.5°F for base metals; the value of  $\sigma_{\Delta}$  need not exceed 0.5 times  $\Delta RT_{\text{NDT}}$ .

### 3.3 U.S. EPR RPV Material Surveillance Program

The U.S. EPR RPV material surveillance program monitors changes in the mechanical properties of the ferritic steel in the RPV beltline region due to the thermal and

irradiation environment. The material surveillance program complies with Appendix H to 10 CFR Part 50 and ASTM E 185-82 (Reference 14).

The material surveillance program uses four specimen capsules containing representative RPV material samples, neutron dosimeters, and temperature monitors. The number of capsules meets the minimum requirements of both ASTM E 185-02 (Reference 15) and ASTM E 185-82. All four capsules contain the same type and number of mechanical test specimens, neutron dosimeters, and temperature monitors. The capsules are located one on either side of the 7° and 187° locations, measured from the main axis of the vessel, and are attached to the outside of the core barrel in the downcomer region at the mid-elevation of the reactor core.

The RPV materials selected for the material surveillance program are those that are adjacent to the active core. Using the maximum initial  $RT_{NDT}$  values, maximum nickel and copper contents allowed in the RPV and a 60 effective full power year (EFPY) fluence, the limiting RPV beltline material for the U.S. EPR is predicted to be the upper core shell to lower core shell weld. This prediction is made in accordance with 10 CFR 50.61. Based on the predictions of the most susceptible materials and on the requirements of ASTM E185-82 and ASTM E185-02, the following materials are included in the material surveillance program:

- Upper to lower core shell weld (Weld #2).
- Lower core shell to transition ring weld (Weld #3), if different from Weld #2.
- Upper core shell forging.
- Lower core shell forging.
- Heat affected zone (HAZ) from a core shell forging and RPV Weld #2.

For each of the beltline materials selected, Charpy V-notch, tension, and compact fracture (CT) specimens are included, except for the HAZ for which only Charpy V-notch specimens are required.

In addition to the four capsules that are assembled for irradiation, surplus material sufficient to fabricate four additional capsules is archived. The total material quantity complies with the minimum requirements of both ASTM E185-82 and ASTM E185-02.

## **4.0 ANALYTICAL METHODS FOR DETERMINING P-T LIMITS**

### **4.1 *Pressure Boundary Components***

P-T limits are established for the limiting RCPB components as described in Section 2.2.3.

### **4.2 *Maximum Postulated Defects***

#### **4.2.1 *Beltline Region of Reactor Vessel***

The methods of Reference 1 postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-quarter of the RPV beltline thickness ( $1/4t$ ), and a length equal to 1.5 times the RPV beltline thickness, as shown in Figure 4-1. In Reference 1, there is no requirement as to which surface (inside or outside) on which to postulate the defect. In application, the defect is postulated on both internal and external surfaces of reactor vessels ( $1/4t$  and  $3/4t$  locations, respectively), even though there is no water environment and no known causes for crack initiation or propagation on the external surface of reactor vessels.

For circumferential seam welds in the RPV,  $1/4t$  defect is postulated in the circumferential direction as shown in Figure 4-2.

#### **4.2.2 *Closure Head***

The reactor closure head is subjected to high stresses due to bolt preload-induced bending and internal pressure. Because the bolt stresses induce tensile bending stresses on the external surface, an external surface defect is postulated. According to Article G-2120 of Reference 1, a flaw size less than  $1/4t$  may be postulated on an individual case basis. [

] Considering the ability to detect a flaw that is a fraction of this size during periodic inspection of this region, postulation of this size defect is

conservative. Because the [ ] location from the outside is insensitive to the inside downcomer reactor coolant temperature, the closure head limit is controlled by heatup rates. This limit bounds the cooldown limit of the closure head.

#### 4.2.3 *Inlet and Outlet Nozzle*

As allowed by Article G-2120 of Reference 1, a flaw with a depth of one inch is postulated in the corners of inlet and outlet nozzles. This assumption is based on past inservice inspection (ISI) results of nozzle corner radius regions in pressurized water reactors in the U.S., which have not shown any reportable indications. In addition, the reliability of defect detection and sizing capability demonstrated for nozzle corner regions has significantly improved since the performance demonstration initiative (PDI) program in 1994 (Reference 8). NRC regulations in 10 CFR 50.55a Codes and Standards (Reference 9) impose requirements for ISI.

Figure 5 of Reference 8 illustrates the probability of detecting and rejecting a flaw, 0.25-inch into the base metal, is equal to or greater than 90 percent and the probability of detecting and rejecting a flaw 0.5-inch or greater into the base material is 100 percent. Also, because the nozzle region does not have much exposure to neutron radiation compared with the beltline region, the limit curves for inlet and outlet nozzles do not change with increasing operating time.

#### 4.3 *Fracture Toughness of Reactor Vessel Steels*

Reference 1 requires the critical stress intensity factor  $K_{Ic}$ , defined by equation 4-1, be used in P-T limit calculations. (Terms and units are further defined in Reference 1.)

$$K_{Ic} = 33.2 + 20.734 \exp[0.02(T - RT_{NDT})] \quad (4-1)$$

Where:

$K_{Ic}$  = critical stress intensity factor

T = temperature

$RT_{NDT}$  = reference temperature for nil-ductility transition

#### 4.4 *Transient Temperature and Thermal Stress Calculation*

The following analytical processes are used in the determination of the allowable pressures for the generation of P-T limits.

The temperature profile through the reactor vessel wall is determined by solving the one-dimensional axisymmetric heat conduction equation:

$$\rho C_p \frac{\partial T}{\partial t} = k \left( \frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r} \right) \quad (4-2)$$

The equation is subjected to boundary conditions at the inside and the outside walls of the reactor vessel:

At the inside wall where  $r = R_i$ ,

$$-k \frac{\partial T}{\partial r} = h(T_w - T_b)$$

At the outside wall where  $r = R_o$ ,

$$\frac{\partial T}{\partial r} = 0$$

Where:

- $\rho$  = density
- $C_p$  = specific heat
- $k$  = thermal conductivity
- $T$  = vessel wall temperature
- $r$  = radius
- $t$  = time
- $h$  = convective heat transfer coefficient

$T_w$	= wall temperature
$T_b$	= bulk coolant temperature
$R_i$	= inside radius of vessel wall
$R_o$	= outside radius of vessel wall

Equation 4-2 is solved numerically using a finite difference or finite element method to determine the wall temperature as a function of radius, time and thermal transient rate. The temperature profile through the reactor vessel wall at a particular time allows determination of the corresponding thermal stresses from the theory of elasticity or a finite element model in the radial, or through-thickness, direction. This numerical procedure is particularly important for multi-rate heatup and cooldown cases.

#### **4.5 Stress Intensity Factors, $K_I$**

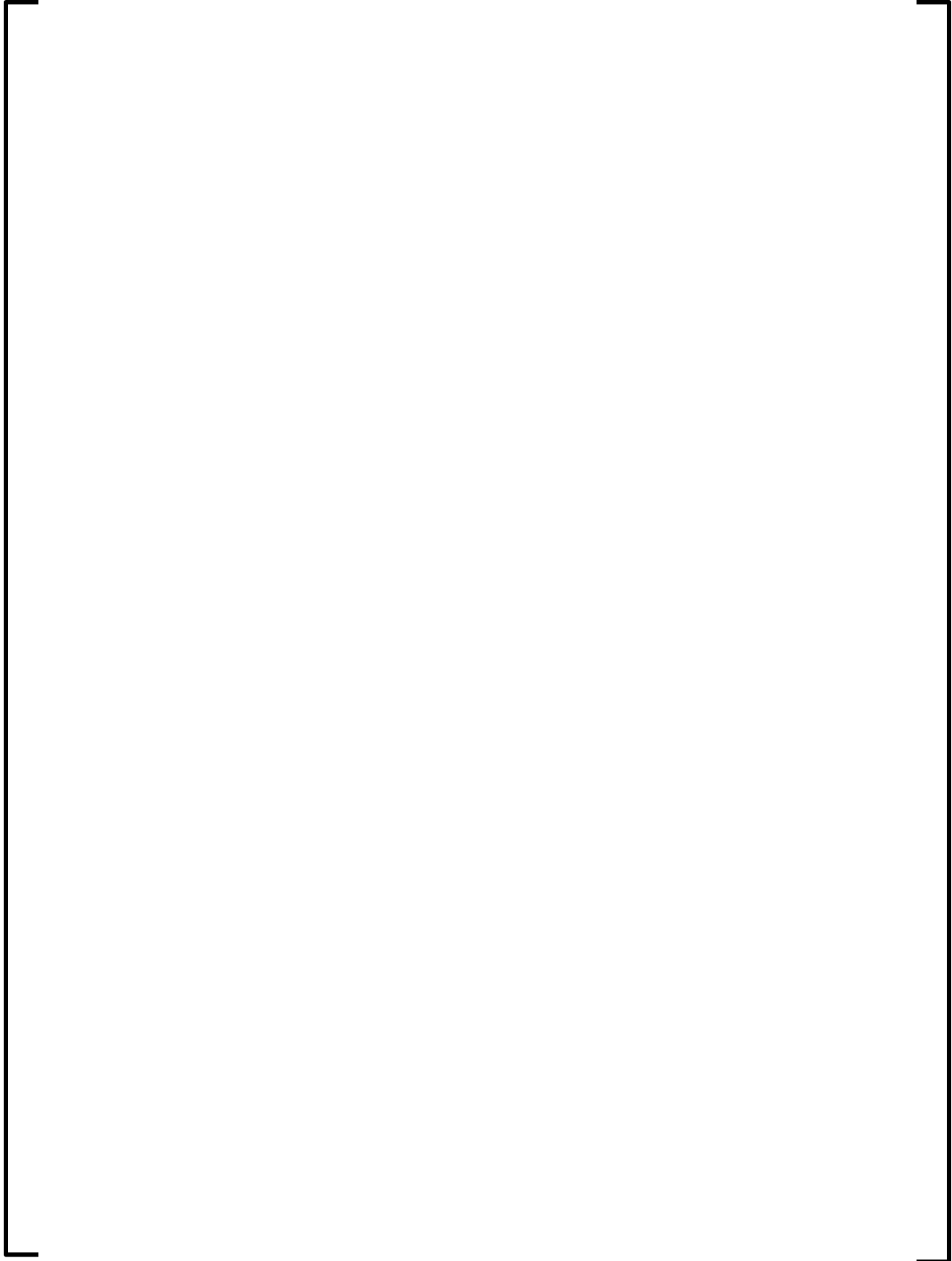
The use of any particular stress intensity factor solution presented in this document or in Reference 1 is allowed, because the resulting  $K_I$  values are appropriate. The choice of  $K_I$  equation usually depends on the form of stress profiles used in the solution. The applicable  $K_I$  equations include those provided in Reference 1 and are not repeated in this report. These terms and units are further defined in the references and are expressed in U.S. customary units.

##### **4.5.1 Longitudinal Semielliptical Surface Flaws**

The stress intensity factor equation for a longitudinal semielliptical surface flaw [ ] is considered well suited for the development of P-T limits because the solution is for a cylinder, not for a plate. The solution is based on the finite element analysis of cylinders with radius to thickness ratios of [ ]

]







Only the stress intensity factors defined in this document or in Reference 1 are used for the U.S. EPR.

#### **4.5.2 Circumferential Semielliptical Surface Flaws**

The  $K_I$  solution for a circumferential flaw shown is from [

]



Only the stress intensity factors defined in this document or in Reference 1 are used for the U.S. EPR.

#### **4.5.3 Nozzle Corner Semielliptical Surface Flaw**

The determination of the stress intensity factor for a nozzle corner crack is based on the method contained in Reference 6, which gives the following equation:

$$K_I = \sigma \sqrt{\pi a} F(a/r_n) \quad (4-11)$$

Where:  $F = 2.5 - 6.108(a/r_n) + 12(a/r_n)^2 - 9.1664(a/r_n)^3$

$\sigma$  = hoop stress

$a$  = crack depth

$r_n$  = apparent radius of nozzle, in which is given by the equation,

$$r_n = r_i + 0.29r_c$$

$r_i$  = actual inner radius of nozzle

$r_c$  = nozzle radius

The membrane stress is the hoop stress due to pressure and is determined using Lamé's solution for thick wall cylinders subjected to internal pressure. The maximum hoop stress is developed at the inside surface of the wall and is given by:

$$\sigma = p \frac{R_o^2 + R_i^2}{R_o^2 - R_i^2} \quad (4-12)$$

The maximum hoop stress at the inside surface is conservatively assumed as a uniform membrane stress across the entire wall thickness.

#### 4.6 *Determination of Appendix G Limits*

The governing equation for determining the pressure-temperature operating limit curves is Equation 1 from Article G-2215 of Reference 1:

$$(S.F.) K_{Im} + K_{It} \leq K_{Ic} \quad (4-13)$$

Where: S.F. = safety factor

= 2 for normal and upset conditions

= 1.5 for ISLH testing

$K_{Im}$  = stress intensity factor due to internal pressure

$K_{It}$  = stress intensity factor due to thermal gradient

$K_{Ic}$  = reference stress intensity factor defined in Equation 4-1

For a thermal transient analysis, a temperature profile is calculated for a given point in time during a heatup or cooldown transient.  $K_{Ic}$  is determined from the crack-tip temperature,  $T$ , and the material  $RT_{NDT}$  at a given location.  $RT_{NDT}$  is determined in accordance with Section 3.2 based on the neutron fluence and chemical composition of the material. The temperature profile (thermal gradient) determines the thermal stresses at various points throughout the reactor vessel wall. If the stress intensity factor is defined as for a pressure of one psi,  $\bar{K}_{Ip}$ , then the allowable pressure is determined using the following equation:

$$P_{allow} = \frac{K_{Ic} - K_{Ii}(T_f)}{(S.F.) \cdot \bar{K}_{Ip}} \quad (4-14)$$

Where:

$T_f$  = fluid temperature

$\bar{K}_{Ip}$  = unit pressure stress intensity factor (due to 1 psig)

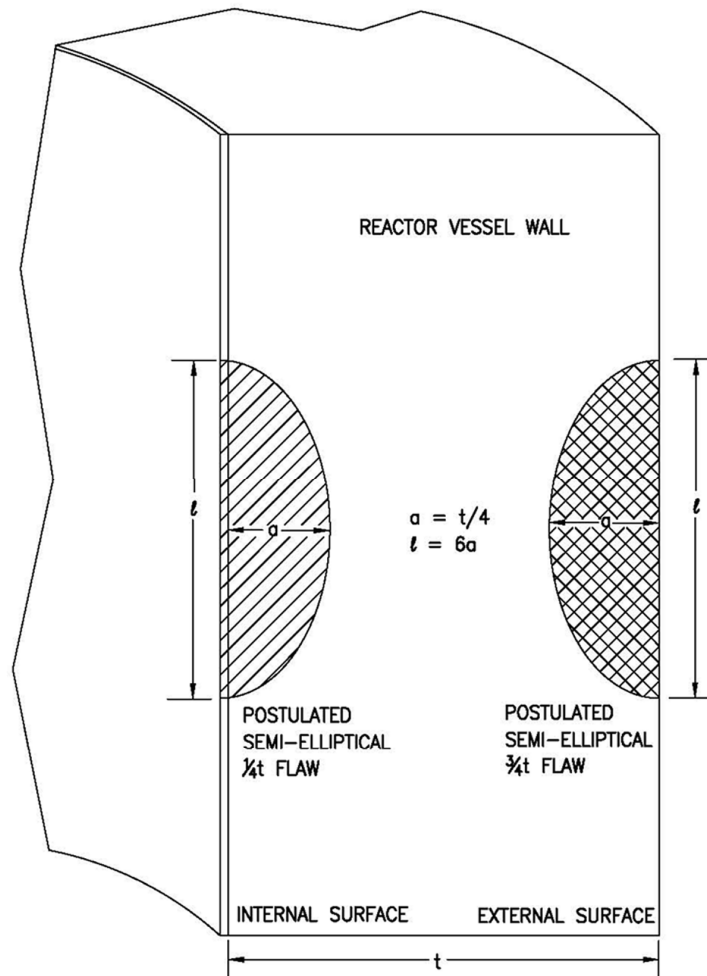
A plot showing the allowable pressure as a function of bulk coolant temperature is a P-T limit curve. The same procedure applies to the ISLH testing except the safety factor is 1.5, instead of 2.0.

#### **4.7 Minimum Bolt Preload Temperature**

In accordance with the requirements of 10 CFR Part 50, Appendix G, Table 1, item 2(a), the minimum bolt preload temperature is equal to the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload. For the U.S. EPR, the minimum bolt preload temperature  $\geq 50^\circ\text{F}$ . Therefore, the minimum bolt preload temperature should be  $50^\circ\text{F}$  or the highest reference temperature of the closure flange region, whichever is greater.

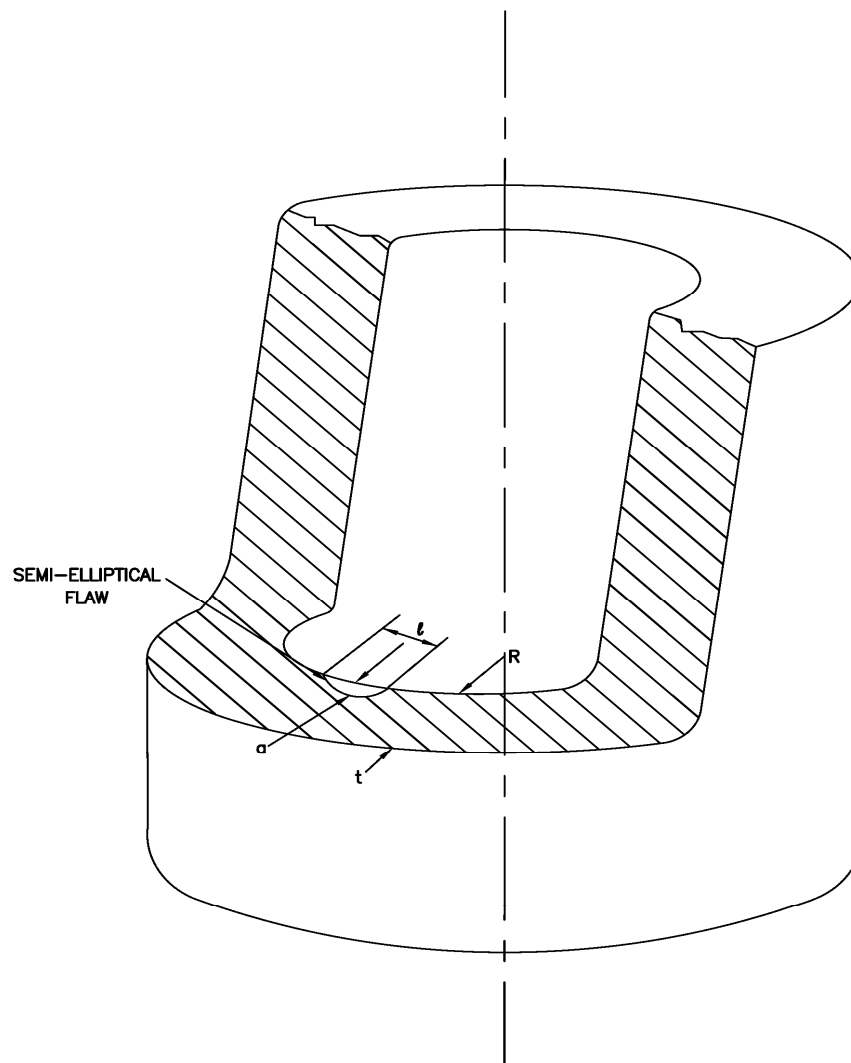
**4.8 Criticality Limit Temperature and Criticality Limit Curve Determination**

The criticality limit temperature is obtained by determining the minimum permissible temperature from the controlling P-T limits for ISLH heatup or cooldown at a pressure of 2500 psig (approximately 10 percent above the normal full-power operating pressure). The ISLH analysis conservatively considers the most limiting heatup and cooldown transients. The minimum permissible ISLH test temperature at 2500 psig is compared to  $RT_{NDT} + 160^{\circ}\text{F}$ . The larger temperature between these two becomes the criticality limit temperature. Item 2.d in Table 2-1 specifies that the criticality limit curve is the criticality limit temperature or the Appendix G limit curve shifted by  $40^{\circ}\text{F}$ , whichever is greater.

**Figure 4-1 Postulated Longitudinal Flaw in Reactor Vessel**

## NOTES:

1. POSTULATED FLAWS ON THE VERTICAL PLANE OF THE REACTOR VESSEL BELTLINE WALL
2. DRAWING NOT TO SCALE

**Figure 4-2 Circumferential Weld in Reactor Vessel Wall**

- NOTES:
1. POSTULATED FLAWS ON THE HORIZONTAL PLANE OF THE REACTOR VESSEL BELTLINE WALL
  2. DRAWING NOT TO SCALE
  3.  $a = t/4$ ;  $l = 6a$

## **5.0 LOW TEMPERATURE OVERPRESSURE PROTECTION**

### **5.1 *Low Temperature Overpressure Protection Features***

The U.S. EPR low temperature overpressure protection (LTOP) features are designed to provide the capability, during reactor operation at low temperature conditions, to automatically prevent the RCS pressure from exceeding the applicable limits established by 10 CFR Part 50, Appendix G. Two pressurizer safety relief valves (PSRV) open automatically in response to RCS pressure to provide the required relief capability. The reactor operator manually validates a permissive to enable LTOP based on the LTOP enable temperature requirement addressed in Section 5.4.

### **5.2 *Analytical Methodology***

The LTOP analyses consider all potential overpressure events to establish the limiting events. Potential events may be excluded from the LTOP analyses if the controls to prevent the events are in the plant Technical Specifications.

LTOP events are categorized into mass input and heat input events. Two mass input events – start of four MHSI pumps, and both charging pumps running with the control valve failed open – and one heat input event – startup of an RCP with the secondary side temperature 50°F hotter than the primary side temperature – were identified as the limiting events for analysis.

The LTOP transient analysis is performed using RELAP5/MOD2-B&W (Reference 16). The analyses assume the most limiting single active failure, in addition to the initiating event, and assume the most limiting allowable operating conditions and system configurations at the time of the postulated cause of the overpressure event. Assumed initial conditions and system configurations include:

- Water solid RCS.
- Isolated volume control system letdown.



- Maximum allowable initial RCS pressure.
- Range of initial RCS temperatures from the allowable minimum to the LTOP enable temperature.
- Maximum flow input (mass input events).
- Maximum heat input (heat input events).

Specified pressure instrument uncertainties are applied to the PSRV setpoints in the penalizing direction for overpressure (brittle fracture protection) and underpressure (reactor coolant pump operation). Control system delays and valve stroke times are modeled to accurately represent pressure overshoot and undershoot. Dynamic and static pressure differences between the pressure sensors and the RPV are applied when evaluating the peak pressures against the limits derived from 10 CFR Part 50, Appendix G.

### **5.3      *Setpoint Determination***

The LTOP analyses determine the maximum and minimum pressures for each analyzed transient for a given set of PSRV setpoints. The maximum pressure is higher than the PSRV setpoint due to control system delays and the time required for the valves to fully open. When the PSRVs open, the relief capacity is sufficient to reduce the system pressure. The system pressure then undershoots the setpoint until the PSRV reset pressure is reached and the valves are closed, accounting for control system delays and valve closure time. The PSRV opening and closing delays result in a repeating pressure oscillation that continues until the event is terminated.

The maximum pressure for each transient, with instrument uncertainty applied in the positive direction, is compared to the pressure-temperature limits derived from the fracture mechanics evaluation. The peak RPV pressure shall not exceed 100 percent of the applicable 10 CFR 50, Appendix G limit. The minimum pressure for each transient, with instrument uncertainty applied in the negative direction, is compared against limits

required for reactor coolant pump operation. Any set of PSRV setpoints that maintain RCS pressure within the upper and lower limits is acceptable.

#### **5.4 *LTOP Enable Temperature***

The LTOP enable temperature is defined in Article G-2215 of ASME Code Section XI, Appendix G as 200°F or the reactor coolant temperature corresponding to the reactor vessel metal temperature equal to  $RT_{NDT} + 50^{\circ}\text{F}$ , whichever is greater. In the latter case, the enable temperature accounts for the difference in temperature between the 1/4t crack-tip metal temperature and the reactor coolant temperature because the metal temperature is lower than the coolant temperature during heatup.

NRC Branch Technical Position 5-2 (Reference 17) defines the enable temperature as the water temperature corresponding to a metal temperature of at least  $RT_{NDT} + 90^{\circ}\text{F}$  at the beltline location (1/4t or 3/4t) that is controlling in the 10 CFR Part 50, Appendix G limit calculations. Since this Branch Technical Position definition was established, ASME Code Cases (Code Case N-514 in 1992, Code Case N-640 in 1999, and Code Case N-641 in 2000) have been developed to establish an acceptable approach to develop LTOP limits that provides the necessary safety margins while also providing for an adequate normal operating window. The Code Cases included a lower LTOP enable temperature requirement, which was subsequently incorporated into ASME Code Section XI as stated above. The U.S. EPR follows ASME Code, Section XI, Appendix G because it represents significant advances in fracture mechanics and in the analysis of reactor vessel integrity, while providing greater operational flexibility.

## **6.0 U.S. EPR GENERIC PRESSURE TEMPERATURE LIMITS REPORT**

### **6.1 *RCS Pressure Temperature Limits Report***

This PTLR has been prepared following the analytical methods described in Sections 2.0 through 5.0 of this report to meet the requirements of U.S. EPR Generic Technical Specifications Section 5.6.4. In particular, this PTLR specifies limits to satisfy Limiting Condition for Operation (LCO) 3.4.3, RCS Pressure and Temperature (P/T) Limits, and LCO 3.4.11, Low Temperature Overpressure Protection (LTOP). The content of this PTLR conforms to the requirements stated in NRC Generic Letter 96-03 (Reference 18).

This is a generic PTLR for the U.S. EPR design based on bounding material properties provided in design specifications and heatup and cooldown transients used in the design process. Actual material properties and operational transients will be addressed in the plant-specific PTLR.

### **6.2 *Normal Heatup and Cooldown Limits (LCO 3.4.3)***

A first step in establishing P-T limits is the collection of geometrical data and material properties including  $RT_{NDT}$  of the limiting materials. Table 6-1 and Table 6-2 provide the material properties used in the analysis.

For heatup and cooldown limits, an RCS coolant temperature-time history (hereinafter referred to as “temperature-time history”) is created. The temperature-time histories are based on plant system operations and are typical heatup and cooldown temperature-time histories, respectively. With this input, allowable pressures are calculated for each selected time point of the temperature-time history. Actual heatup or cooldown rates can be anywhere between steady-state condition and the specified limits represented by the temperature-time histories. The analyses include four different cases: normal heatup, normal cooldown, ISLH heatup, and ISLH cooldown.

The RCS temperature rate of change limits are:

- A maximum RCS heatup rate of 72°F per hour.
- A maximum RCS cooldown rate of 90°F per hour.

The 1/4t flaws are postulated at both inside and outside surfaces (1/4t and 3/4t flaws, respectively) of the reactor vessel beltline region, remote from discontinuities. A second region, namely the nozzle corner in the RPV beltline region, is also assessed for allowable pressure at each selected time point. For the nozzle, the flaw is postulated to be located on an inside surface.

Because flaws are postulated on the inside and outside surfaces in the beltline region, three allowable pressure values are determined. These are the transient 1/4t and 3/4t flaw values and steady-state 1/4t flaw value for both circumferential and axial flaws in the beltline region. Accordingly, a total of six pressures are computed for the beltline region. Four additional pressure values are determined for the two nozzle corner crack locations, representing transient and steady-state conditions of the inlet and outlet nozzles.

Finally, the allowable pressure is determined for a third region, namely the closure head region. The closure head limits are developed from a separate analysis, considering the postulated outside surface flaw, as described in Section 4.2.2. The normal heatup and cooldown stresses in the closure head region are obtained from a detailed finite element stress analysis of the reactor vessel closure. The general form of the stress intensity factor equation reported in Section 4.5.1 is used. The allowable pressure and temperature limits for the reactor vessel closure head are determined considering the requirements of Reference 1 and Reference 2.

Among the six pressure values of the beltline region, the lowest one is determined and considered to be the maximum allowable pressure value at each selected time point. In this manner the results at various selected temperature-time history time points are determined to produce a single lower bound P-T limit curve for normal heatup or normal cooldown. The pressure values, along with the associated RCS coolant temperatures

at the selected time points of the temperature-time history form the P-T limit curves. The lowest allowable pressure at each time point yields a single lower bound P-T limit curve for normal heatup or normal cooldown. The P-T curves for normal plant heatup and cooldown are presented in Figure 6-1 and Figure 6-2, respectively.

The P-T limit curve for the closure head region is calculated separately. The allowable pressure from plant startup is maintained as a constant value of 635 psig (20 percent of preservice hydrostatic test pressure) from the bolt preload temperature condition until the coolant temperature reaches the temperature where a calculated crack-tip metal temperature exceeds the minimum temperature requirement of Reference 1 and Table 2-1. The minimum required temperatures are subsequently determined at 1285 psig and at 2325 psig (full power, steady-state condition). The resulting closure head limit curves are included in Figure 6-1 and Figure 6-2 for heatup and cooldown, respectively.

For plant heatup, the closure head limit curve lower bounds both the beltline region and the nozzle corner limit curves. As noted, the closure head limit does not change throughout the lifetime of the plant. In the case of normal cooldown from steady-state conditions, as shown in Figure 6-2, the beltline P-T limit is controlling until it intersects with the closure head limit curve. At 635 psig, which corresponds to 20 percent of the preservice hydrostatic test pressure, the allowable temperature corresponds to the minimum temperature requirement of  $RT_{NDT} + 120^{\circ}\text{F}$  per item 2.b of Table 2-1. The P-T limit curve for the (inlet and outlet) nozzle corner region is not a controlling P-T limit region at any time during normal plant heatup or cooldown.

The P-T limits thus calculated are “uncorrected P-T limits,” meaning that measurement uncertainty due to instrument error or sensor location adjustment is not included. The sensor location adjustment is necessary due to the difference in sensor readings (pressure and temperature) at the measurement location compared to the corresponding pressures and temperatures at the controlling P-T limit region. Sensor location adjustment includes the effect of pump operation. These corrections are made to the uncorrected P-T limits and the resultant corrected P-T limits are presented in

Figure 6-1 and Figure 6-2. Any applicable pressure or temperature instrument error corrections have not been included in the curves.

### **6.2.1 Summary of P-T Limits**

As shown in Figure 6-1 and Figure 6-2, in the low temperature range, the closure head limits are controlling. The beltline limits are higher than the closure head limits. For normal heatup, the closure head limit is always controlling. For normal cooldown, the closure head limit is controlling in the low temperature region up to 116°F.

For ISLH testing, the closure head limits are controlling for ISLH heatup and up to 116°F for ISLH cooldown. At temperatures above 116°F, the beltline circumferential seam weld limit is the controlling P-T limit.

### **6.2.2 Criticality Limit**

Following the steps defined in Section 4.8, the criticality limit temperature corresponding to a pressure of 2500 psig is determined to be 220°F. This temperature is compared to the value of  $RT_{NDT} + 160^{\circ}\text{F}$  (i.e., 156°F). The criticality limit temperature is the higher of these two values (i.e., 220°F). Therefore, the criticality limit curve reflects a minimum temperature of 220°F and the normal heatup limit curve shifted by 40°F for temperatures higher than this criticality limit temperature as shown in Figure 6-1.

### **6.3 Reactor Vessel Material Surveillance Program**

The recommended RPV material surveillance program capsule withdrawal schedule is outlined in Table 6-3. Testing of each surveillance capsule will be performed in accordance with 10 CFR 50, Appendix H. The material data will be evaluated using the guidance of RG 1.99, which is described in Section 3.2. The P-T limits will be recalculated or the applicable EFPY will be adjusted, as necessary, to confirm that the 1/4t and 3/4t ART values of the RPV-based P-T limits are not exceeded.

## **6.4 Low Temperature Overpressure Protection**

### **6.4.1 PSRV Lift Setpoints for LTOP (LCO 3.4.11)**

The maximum PSRV lift setpoints for LTOP are determined in accordance with the methodology described in Section 5.0 and are presented in Table 6-4.

### **6.4.2 LTOP Arming Temperature**

The minimum LTOP enable temperature is defined as 200°F or the reactor coolant temperature corresponding to the reactor vessel metal temperature equal to  $RT_{NDT} + 50^\circ\text{F}$ , whichever is greater. According to Table 6-2, the highest  $RT_{NDT}$  is projected to be 126.5°F at 60 EFPY. Therefore,  $RT_{NDT} + 50^\circ\text{F}$  is 176.5°F. This latter value must be added to the difference between the 1/4t crack-tip metal temperature and the reactor coolant temperature, which is 29.0°F based on the analyzed heatup transient. Therefore, the LTOP enable temperature is 205.5°F plus any adjustments for measurement uncertainty.

The LTOP arming temperature setpoint, as used in the U.S. EPR Generic Technical Specifications, defines the applicability of LCO 3.4.10 and LCO 3.4.11. The LTOP arming temperature is specified in Table 6-4 and is above the minimum required by the fracture mechanics evaluation.

## **6.5 Supplemental Data Tables**

Table 6-1 identifies the material composition of RPV beltline components based on the maximum limits for vessel manufacture and are therefore conservative. The chemistry factors for the materials are extracted from RG 1.99 Revision 2, Table 1 and Table 2.

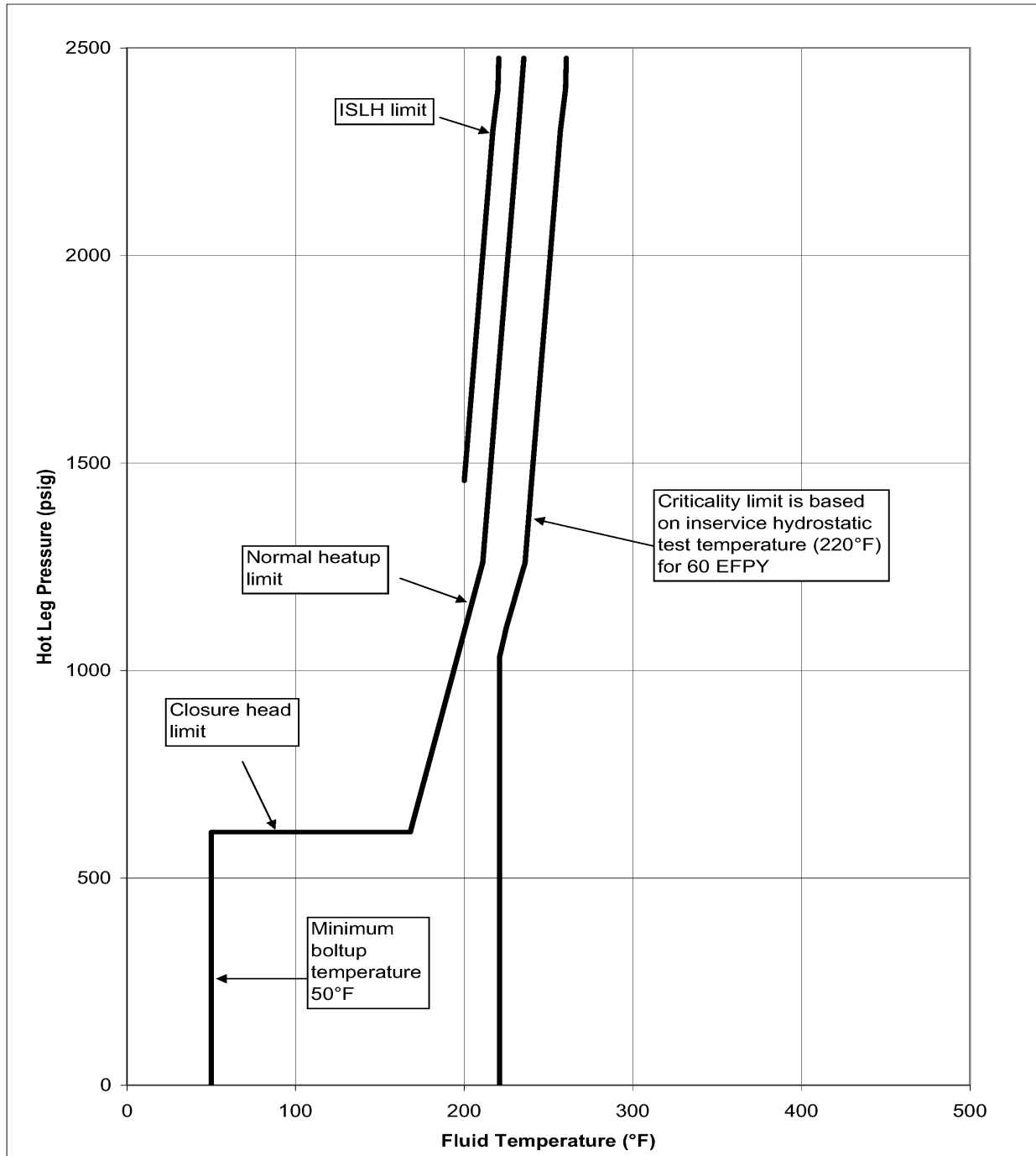
Table 6-1 contains the maximum projected neutron fluence values for the RPV beltline materials at 60 EFPY. Calculated fluence values are tabulated at the inner wetted surface of the RPV and at the 1/4T and 3/4T positions, where T is the vessel thickness measured from the inside (wetted) surface.

Table 6-2 identifies the initial  $RT_{NDT}$  for each vessel material and the ART at 60 EFPY for the 1/4T and 3/4T locations. The initial  $RT_{NDT}$  is the maximum limit for vessel manufacture and is therefore conservative.

Table 6-5 contains the pressurized thermal shock (PTS) reference temperatures,  $RT_{PTS}$ , for each beltline material at the projected end of life (EOL) of 60 EFPY calculated in accordance with 10 CFR 50.61. The copper and nickel content for each material is specified in Table 6-1 along with the projected fluence, conservatively based on a 24-month cycle core design. Table 6-5 also contains the PTS screening criteria from 10 CFR 50.61. The  $RT_{PTS}$  values are not projected to exceed the PTS screening criteria using the EOL fluence.



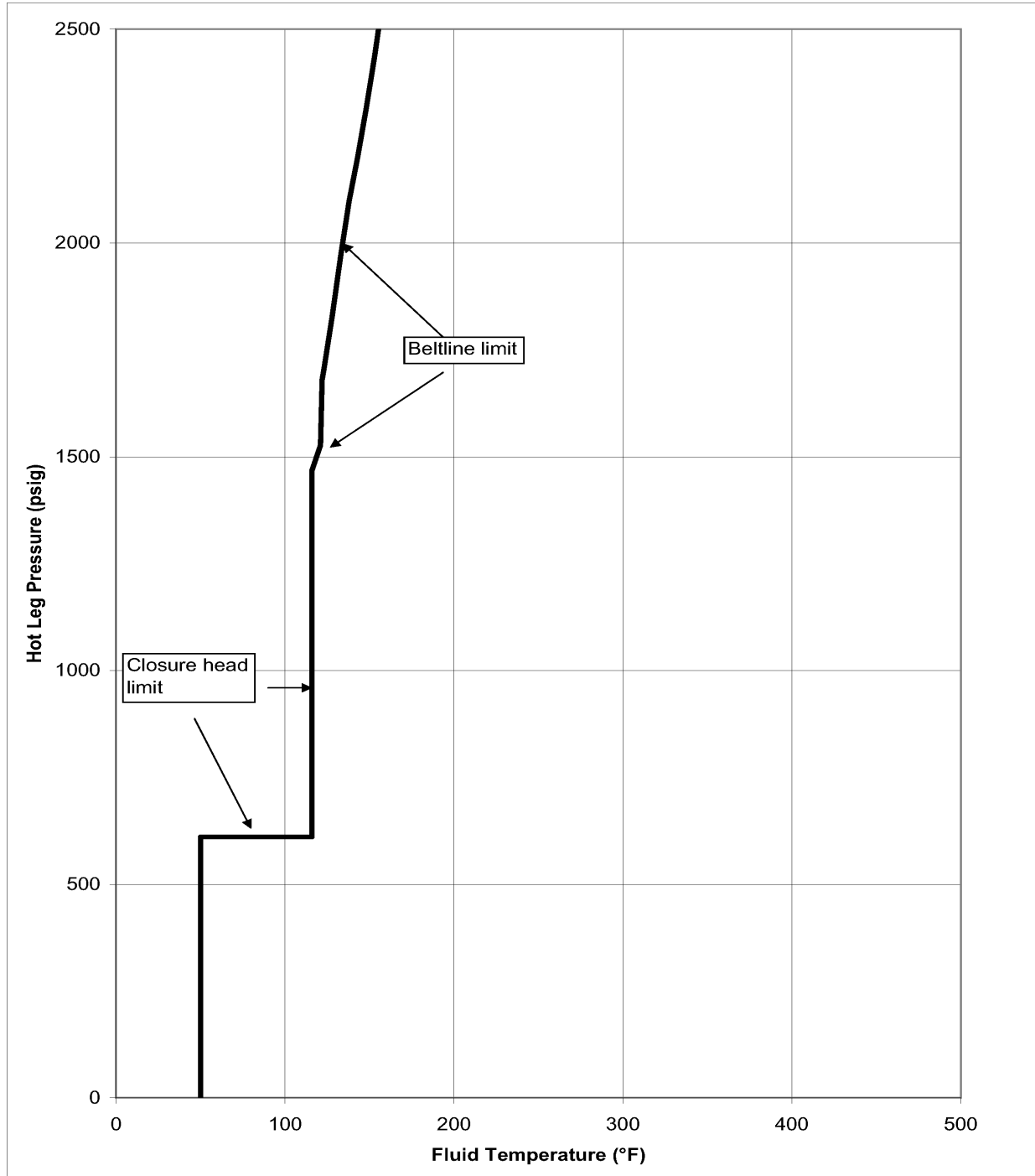
**Figure 6-1 U.S. EPR RCS P-T Limits – Normal Heatup with ISLH and Criticality Limit Curves Applicable to 60 EFPY**



Note:

1. P-T limit curves do not include margin for instrument uncertainty.

**Figure 6-2 U.S. EPR RCS P-T Limits – Normal Cooldown Applicable to 60 EPFY**



Note:

1. P-T limit curves do not include margin for instrument uncertainty.

**Table 6–1 Chemical Composition and Projected Fluence for the U.S. EPR Reactor Vessel Materials through 60 EFPY**

Material Description		Chemical Composition		Chemistry Factor <sup>(2)</sup>	60 EFPY Fluence, n/cm <sup>2</sup> <sup>(1)</sup>		
Reactor Vessel	Type	Cu wt %	Ni wt %		Inner Wetted Surface	1/4t Location <sup>(3)</sup>	3/4t Location <sup>(3)</sup>
Nozzle Shell <sup>(4)</sup>	SA-508 Grade 3 Class 1	0.08	0.80	51.0	2.103E+17	1.111E+17	3.410E+16
Upper Core Shell	SA-508 Grade 3 Class 1	0.06	0.80	37.0	1.375E+19	7.262E+18	2.230E+18
Lower Core Shell	SA-508 Grade 3 Class 1	0.06	0.80	37.0	1.375E+19	7.262E+18	2.230E+18
Transition Ring <sup>(5)</sup>	SA-508 Grade 3 Class 1	0.08	0.80	51.0	4.406E+18	2.327E+18	7.145E+17
Weld #1	Low Alloy Steel Weld	0.06	1.20	82.0	2.103E+17	1.111E+17	3.410E+16
Weld #2	Low Alloy Steel Weld	0.06	1.20	82.0	1.368E+19	7.225E+18	2.218E+18
Weld #3	Low Alloy Steel Weld	0.06	1.20	82.0	4.406E+18	2.327E+18	7.145E+17

## Notes:

wt % = weight percentage

t = wall thickness

1. Projected neutron fluence ( $E > 1$  MeV) conservatively based on a 24-month cycle core design.
2. Chemistry factors extracted from Table 1 and Table 2 of RG1.99 Revision 2.
3. Calculated using Equation 3 of RG 1.99 Revision 2 and a vessel thickness of 9.84 inches with a minimum cladding thickness of 0.20 inches.
4. The fluence value for the nozzle shell is conservatively assigned the fluence value for Weld #1.
5. The fluence value for the transition ring is conservatively assigned the fluence value for Weld #3.

**Table 6–2 Adjusted Reference Temperature for the U.S. EPR  
Reactor Vessel Materials through 60 EFPY**

Material Description		Initial RT <sub>NDT</sub> (°F)	ART, °F at 60 EFPY <sup>(1)</sup>	
Reactor Vessel	Type		1/4t Location	3/4t Location
Nozzle Shell	SA-508, Grade 3, Class 1	-4	8.0	1.2
Upper Core Shell	SA-508, Grade 3, Class 1	-4	63.4	40.2
Lower Core Shell	SA-508, Grade 3, Class 1	-4	63.4	40.2
Transition Ring	SA-508, Grade 3, Class 1	-4	57.8	32.0
Weld #1	Low Alloy Steel Weld	-4	15.4	4.2
Weld #2	Low Alloy Steel Weld	-4	126.5	93.4
Weld #3	Low Alloy Steel Weld	-4	95.4	53.8
Closure Head	SA-508, Grade 3, Class 1	-4	-4 <sup>(2)</sup>	N/A
Closure Head	Closure Head Weld	-4	-4 <sup>(2)</sup>	N/A

## Notes:

1. The margin term in the RG 1.99 Revision 2 expression for adjusted reference temperature is calculated according to RG 1.99 Revision 2, Equation 4. The standard deviation for initial RT<sub>NDT</sub> is 0°F because the initial RT<sub>NDT</sub> is specified as a maximum limit for vessel manufacture. The standard deviation for  $\Delta$ RT<sub>NDT</sub> is the lesser of 28°F for welds and 17°F for base metals and 0.50 times the mean value of  $\Delta$ RT<sub>NDT</sub> calculated from the chemistry factors and fluences in Table 6-1.
2. The projected fluence to the RPV head is insufficient to cause any measurable shift in RT<sub>NDT</sub>.

**Table 6–3 U.S. EPR Material Surveillance Program Recommended Specimen Withdrawal Schedule**

Capsule	EFPY	Target Capsule Fluence (n/cm <sup>2</sup> )	ASTM E185-82 Requirement <sup>(1)</sup>
1	6	$2.1 \times 10^{18}$	6 EFPY or $5 \times 10^{18}$ n/cm <sup>2</sup> , whichever comes first
2	15	$5.2 \times 10^{18}$	15 EFPY or EOL fluence at the vessel inside surface, whichever comes first
-	20	$7.3 \times 10^{18}$	Not required
-	40	$1.3 \times 10^{19}$	Not required
3	60	$2.1 \times 10^{19}$	EOL, but between one and two times EOL fluence at the vessel inside surface
4	Supplemental	To be determined	Not required

Note:

1. ASTM E185-82 requirements are based on a predicted  $\Delta$ RTNDT at the vessel inside surface  $\leq 100^\circ\text{F}$ . The predicted transition temperature shift is  $89^\circ\text{F}$  for circumferential weld #2 per RG1.99, Rev. 2 at the RPV inside surface at 60 EFPY using the limiting compositions of Cu and Ni contained in the weld and the highest initial RTNDT.

**Table 6–4 LTOP Settings**

Parameter	Values
Maximum PSRV Lift Setpoints	525 psig <sup>(1)</sup>
	541 psig <sup>(1)</sup>
LTOP Arming Temperature	248°F

Note:

1. Hot leg pressure indication. Setpoints do not include instrument uncertainty, which is accounted for in the LTOP analyses.

**Table 6–5 U.S. EPR Pressurized Thermal Shock Reference Temperature at EOL (60 EFPY)**

Reactor Vessel Location	60 EFPY Fluence, n/cm2 (Inner Wetted Surface) <sup>(1)</sup>	Chemistry Factor <sup>(2)</sup>	Fluence Factor <sup>(3)</sup>	$\Delta RT_{NDT}$ (°F)	Margin (°F) <sup>(4)</sup>	$RT_{PTS}$ (°F)	Screening Criterion (°F)
Nozzle Shell	2.103E+17	51.0	0.177	9.0	9.0	14.0	270
Upper Core Shell	1.375E+19	37.0	1.088	40.3	34.0	70.3	270
Lower Core Shell	1.375E+19	37.0	1.008	40.3	34.0	70.3	270
Transition Ring	4.406E+18	51.0	0.772	39.4	34.0	69.4	270
Weld #1	2.103E+17	82.0	0.177	14.4	14.4	24.8	300
Weld #2	1.368E+19	82.0	1.087	89.1	56.0	141.1	300
Weld #3	4.406E+18	82.0	0.772	63.3	56.0	115.3	300

## Notes:

1. The fluences at the inside wetted surface of the vessel are conservative compared to the fluences at the clad-base metal interface as stated in 10 CFR 50.61.
2. Chemistry factors are based on the copper and nickel contents specified in Table 6-1 and Table 1 and Table 2 of 10 CFR 50.61.
3. Fluence factor is the fluence term in Equation 3 of 10 CFR 50.61. The  $RT_{PTS}$  calculation conservatively uses the fluence at the inside wetted surface of the vessel.
4. The margin term is calculated according to 10 CFR 50.61, Equation 2. The standard deviation for initial or unirradiated  $RT_{NDT}$  is 0°F because the initial  $RT_{NDT}$  is specified as a maximum for vessel manufacture and will be measured. The standard deviation for  $\Delta RT_{NDT}$  is the lesser of 28°F for welds and 17°F for base metals and 0.50 times  $\Delta RT_{NDT}$ .

## **7.0 SUMMARY AND CONCLUSION**

This technical report presents the methodology for the development of P-T limits used to demonstrate compliance with the fracture toughness and requirements of 10 CFR Part 50, Appendix G and Reference 1. The P-T limits generated by this methodology provide an adequate margin of safety for protecting the integrity of the RCPB of the U.S. EPR during conditions of normal operation, including AOOs and ISLH tests, to which the RCPB may be subjected over its service lifetime.

## 8.0 REFERENCES

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