# US-APWR REACTOR VESSEL
# Pressure and Temperature Limits Report

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# Revision History

| Revision | Page | Description |
|---|---|---|
| 0 | All | Original Issue |
| 1 | Abstract<br>1-1 to 8-1<br><br>v | Fully revised to incorporate P-T limit curve values and contents required by NRC Generic Letter 96-03.<br><br>Updated List of Acronyms. |

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## Abstract

This report contains the methodology used to derive the pressure and temperature limits and the generic limits applicable to the US-APWR standard design and is submitted as part of the US-APWR Design Certificate.

The evaluations for the generic limits are based on the bounding properties of the reactor vessel beltline region materials and projected fluence.

This report is prepared following NRC Generic Letter 96-03.

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# List of Acronyms

The following list defines the acronyms used in this document.

| | |
|---|---|
| ART | Adjusted Reference Temperature |
| BTP | Branch Technical Position |
| BWR | Boiling Water Reactor |
| CS/RHR | Containment Spray/Residual Heat Removal |
| EFPY | Effective Full Power Years |
| EOL | End-of-Life |
| LCO | Limiting Condition of Operation |
| LTOP | Low Temperature Overpressure Protection |
| LWR | Light Water Reactor |
| P-T | Pressure-Temperature |
| PTLR | Pressure and Temperature Limits Report |
| PTS | Pressurized Thermal Shock |
| $RT_{PTS}$ | Pressurized Thermal Shock Reference Temperature |
| RCS | Reactor Coolant System |
| RHR | Residual Heat Removal |
| RHRS | Residual Heat Removal System |
| $RT_{NDT}$ | Reference Temperature |
| RWSP | Refueling Water Storage Pit |

## 1.0 INTRODUCTION

This report presents the Reactor Coolant System (RCS) Pressure-Temperature (P-T) limits for the US-APWR standard design in accordance with the requirements of Technical Specification 5.6.4. A description of the Low Temperature Overpressure Protection (LTOP) System setpoint is also provided in this report. In addition, the requirements of the reactor vessel material surveillance program are discussed.

The following two Technical Specification Limiting Conditions of Operation (LCO) are addressed in this report:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System

This report covers the US-APWR operation until the plant end-of-life (EOL) of 60 Effective Full Power Years (EFPY).

## 2.0 BACKGROUND

P-T limit curves are established in order to protect the reactor vessel by minimizing the possibility of fast fracture.

The following regulatory requirements concerning P-T limits on the reactor vessel pressure boundary are considered.

- 10 CFR 50.60 (Reference 1): compliance with the requirements of 10 CFR 50 Appendices G and H.
- 10 CFR 50 Appendix G (Reference 2): material testing and fracture toughness requirements.

The methods outlined in Appendix G of ASME Code Section XI (Reference 3), including defect sizes and safety factors, are applied in the analyses for protection against non-ductile failure. ASME Code Section XI Appendix G is applied rather than Appendix G of ASME Code Section III (Reference 4), as it is referenced by 10 CFR 50 Appendix G.

Representative P-T limit curves for the US-APWR up to 60EFPY, conservative based on the design life of 60 years, are established and the corresponding setpoint of LTOP system is determined.

# 3.0 NEUTRON FLUENCE

## 3.1 Calculation Methodology

Regulatory Guide 1.190 (Reference 5) provides the guidance for the application and qualification of a methodology for determining the neutron fluence experienced by materials in the beltline region of Light Water Reactor (LWR) reactor vessel. The calculation methodology for the US-APWR reactor vessel neutron fluence is in accordance with Regulatory Guide 1.190 and is described in Reference 6.

## 3.2 Calculated Value

The neutron fluences at 60EFPY used to evaluate base metal and weld within the beltline region are shown in Table 3-1, from Reference 6.

Table 3-1 Fast Neutron Fluence at 60EFPY

| **Material** | **Neutron Fluence** [E>1MeV, n/cm$^2$] |
|---|---|
| Base Metal | $9.8 \times 10^{18}$ (1) |
| Weld | $8.5 \times 10^{18}$ (2) |

Notes:
1. Maximum value at ID of beltline region.
2. Maximum value at ID of circumferential weld line between lower shell and transition ring.

# 4.0 MATERIAL PROPERTIES

## 4.1 Material Specification

For the US-APWR reactor vessel, the lower shell, the transition ring (SA-508 Gr.3 Cl.1) and the weld line between the lower shell and transition ring are fully or partially within the beltline region. These are considered to be the bounding material in terms of degradation due to irradiation.

In order to reduce effects of irradiation embrittlement on beltline region ferritic base and weld material during plant operation, copper, nickel phosphorus and sulfur content are limited as shown in Table 4-1.

In addition, considering irradiation embrittlement, the maximum reference temperature ($RT_{NDT}$) for the reactor vessel beltline ferritic materials is 0°F for forgings and -20°F for weld materials in the unirradiated condition, while that for the other ferritic materials, including the closure flange region, is 10°F.

## 4.2 Adjusted Reference Temperature

In accordance with Regulatory Guide 1.99, (Reference 7), the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

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

The ART value of each material within the beltline region is calculated and the limiting ART is used to establish P-T limit curves.

### 4.2.1 Initial $RT_{NDT}$

Initial $RT_{NDT}$ is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of ASME Code Section III (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 sufficient test results to establish a mean and standard deviation for the class.

### 4.2.2 Adjustment of Reference Temperature ($\Delta RT_{NDT}$)

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

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

CF(°F) is the chemistry factor, a function of copper and nickel content. CF is given in Table 1 of Reference 7 for welds and in Table 2 of Reference 7 for base metal. Linear interpolation is permitted. In Table 1 and 2 of Reference 7 “weight-percent copper” and "weight-percent nickel" are the best-estimate values for the material, or the upper limiting values given in the material specifications.

The value $f^{(0.28-0.10\log f)}$ is called as the fluence factor (FF).

The neutron fluence at any depth in the vessel wall, f ($10^{19}$n/cm$^2$, E>1MeV), is determined as follows:

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

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

When two or more credible surveillance data sets (as defined in the Discussion of Reference 7) become available from the reactor in question, they may be used to determine the adjusted reference temperature and the Charpy upper-shelf energy of the beltline materials as below:

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

Where n is the number of surveillance data points, $A_i$ is the measured value of $\Delta RT_{NDT}$ and $f_i$ is the fluence for each surveillance data point.

### 4.2.3 Margin

Margin is the quantity, °F, that is to be added to obtain conservative, upper-bound values of adjusted reference temperature for the calculations required by 10 CFR 50 Appendix G.

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

Here, $\sigma_I$ is the standard deviation for the initial $RT_{NDT}$. If a measured value of initial $RT_{NDT}$ for the material in question is available, $\sigma_I$ is to be estimated from the precision of the test method. If not, and generic mean values for that class of material are used, $\sigma_I$ is the standard deviation obtained from the set of data used to establish the mean.

The standard deviation for $\Delta RT_{NDT}$, $\sigma_\Delta$, is 28°F for welds and 17°F for base metal. When surveillance data sets are used to calculate CF, $\sigma_\Delta$ is 14°F for welds and 8.5°F for base metal. In each case the value of $\sigma_\Delta$ need not exceed one-half of $\Delta RT_{NDT}$.

### 4.3 Reactor Vessel Material Surveillance Program

The surveillance program for the reactor vessel consists of six capsules. The capsules include test specimens for the reactor vessel weld metal, base metal and heat affected zone (HAZ) metal.

Test specimens for the base metal are oriented in both the longitudinal and transverse direction compared to the principal forging direction of the forging material. Weld test plates for surveillance program specimens have their principal working direction parallel to the weld line so that specimens for the HAZ area are normal to the principal working direction.

Taking into consideration the minimum number of test specimens required by ASTM E-185 (Reference 8), as referenced by 10 CFR 50 Appendix H (Reference 9), a minimum of 9 tensile test specimens, 48 CVN test specimens and 6 CT fracture test specimens are contained in each capsule. The test specimens are enclosed in stainless steel sheaths to protect against corrosion.

Dosimeters and thermal monitors are also included in the capsules. The dosimeters are used to evaluate the neutron exposure experienced by the test specimens and reactor vessel shell. The thermal monitors consist of low melting point alloys that are used to monitor the maximum temperature of the test specimens. In accordance with Regulatory Guide 1.190 (Reference 5), the following dosimeters and thermal monitors are included in the capsules.

- Dosimeters
  (1) Iron
  (2) Copper
  (3) Nickel
  (4) Titanium
  (5) Niobium
  (6) Cobalt-aluminum (0.15% cobalt)
  (7) Cobalt-aluminum (cadmium shielded)
  (8) Uranium-238 (cadmium shielded)
  (9) Neptunium-237 (cadmium shielded)
- Thermal Monitors
  (1) 97.5% lead, 2.5% silver, (579°F melting point)
  (2) 97.5% lead, 1.75% silver, 0.75% tin (590°F melting point)

The guide baskets for the surveillance capsules are located so that the lead factors (ratio of the neutron flux at the location of the capsule to that at the reactor vessel inner surface at the peak fluence location) of the capsules are between 2 and 3.

The above range for the lead factors is specified to monitor the embrittlement properties of the reactor vessel materials in the future, and takes into consideration the recommendations of Reference 8. The calculated lead factors of the capsules for the US-APWR reactor vessel are 2.7 for two (2) capsules (Type A), 2.4 for two (2) capsules (Type B) and 2.1 for two (2) capsules (Type C).

**Table 4-1 Chemical Composition Requirements for Reactor Vessel Materials**

| Element | Beltline Region Forging (wt %) | Beltline Region As-Welded Weld Material (wt %) |
|---|---|---|
| **Copper** | 0.05 max. | 0.08 max. |
| **Nickel** | 1.00 max. | 0.95 max. |
| **Phosphorus** | 0.005 max. | 0.012 max. |
| **Vanadium** | 0.05 max. | 0.05 max. |
| **Sulfur** | 0.005 max. | 0.01 max. |

## 5.0 FRACTURE TOUGHNESS REQUIREMENTS

10 CFR 50 Appendix G (Reference 2) defines P-T limits and minimum temperature requirements for the reactor vessel by the operating condition, the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. Both P-T limits and minimum temperature requirements are summarized in Table 1 of 10 CFR 50 Appendix G (included as Table 5-1 in this report).

### 5.1 P-T Limits

Following 10 CFR 50 Appendix G, P-T limits for the reactor vessel are established to be at least as conservative as the following limits in accordance with ASME Code Section XI Appendix G:

P-T Limits for Heatup and Cooldown Operations

For heatup and cooldown operations, the material's $K_{Ic}$ must be greater than the sum of 2 times $K_{Im}$ and $K_{It}$, shown by the following relationship:

$2 \times K_{Im} + K_{It} < K_{Ic}$

where,

$K_{Im}$ = stress intensity factor caused by pressure ($\mathrm{ksi}\sqrt{\mathrm{in}}$)

$K_{It}$ = stress intensity factor caused by thermal gradients through the reactor vessel wall ($\mathrm{ksi}\sqrt{\mathrm{in}}$)

$K_{Ic}$ = the lower bound of static plane-strain fracture toughness ($\mathrm{ksi}\sqrt{\mathrm{in}}$)

Note that the constant value 2 is the safety factor for $K_{Im}$.

P-T Limits for Inservice Leak and Hydrostatic Tests

For inservice leak and hydrostatic tests, the $K_{Ic}$ of the material must be greater than 1.5 times $K_{Im}$, expressed as follows:

$1.5\ K_{Im} < K_{Ic}$

Note that the constant value 1.5 is the safety factor for $K_{Im}$, and $K_{It}$ is equal to 0.

P-T Limits for Core Critical Operation

When the reactor core is critical, the P-T limit required for the heatup and cooldown operations must be shifted by minimum 40°F.

### 5.1.1 Determination of $K_{Ic}$

$K_{Ic}$ is provided in ASME Code Section XI Appendix G as a relationship of a metal temperature and $RT_{NDT}$. The analytical approximation is given by the following equation.

$$K_{Ic} = 33.2 + 20.734 \exp[0.02(T - RT_{NDT})] \qquad (ksi\sqrt{in})$$

where,

T = metal temperature (°F)

$RT_{NDT}$ = reference temperature (°F) or ART (°F) for beltline materials at a specific time in the plant life, whose value takes into account irradiation embrittlement as specified in Section 4.2.

### 5.1.2 Determination of $K_{It}$

$K_{It}$ is determined in accordance with ASME Code Section XI Appendix G as shown below. Note that $K_{It}$ is equal to 0, for hydrostatic and leak tests which are performed under isothermal conditions.

An inside surface defect is assumed at 1/4 the wall thickness from the inner wall surface (1/4-T), and the following relationship is applied:

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

An outside surface defect is also assumed at 1/4 the wall thickness from the outer wall surface (3/4-T), and the following relationship is applied:

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

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

$$\sigma(x) = C_0 + C_1\left(\frac{x}{a}\right) + C_2\left(\frac{x}{a}\right)^2 + C_3\left(\frac{x}{a}\right)^3 \qquad (psi)$$

where,

x = variable representing the radial distance from the inside or outside surface to any point on the crack front of the defect (in)

a = maximum crack depth (in)

The radial position and time-dependent thermal hoop stress is calculated by the following equation for thermal stresses in hollow cylinders given by Timoshenko (Reference 10).

$$\sigma_\theta(r,t) = \frac{E\alpha}{1-\upsilon}\frac{1}{r^2}\left(\frac{r^2+R_i^2}{R_o^2-R_i^2}\int_{R_i}^{R_o} T(r,t)rdr + \int_{R_i}^{r} T(r,t)rdr - T(r,t)r^2\right) \qquad \text{(psi)}$$

where,

E = modulus of elasticity at the average temperature ($T_{ave}$)

$\alpha$ = coefficient of linear expansion at a time t

$\upsilon$ = Poisson's ratio

$R_i$ = inner radius (in)

$R_o$ = outer radius (in)

r = radial position (in)

The average temperature, $T_{ave}$ for E above is determined by the following equation.

$$T_{ave} = \frac{\int_0^{t_{th}} T(r)dr}{\int_0^{t_{th}} dr}$$

where,

$t_{th}$ = shell wall thickness (in)

The time-dependent metal temperature is determined by evaluating the following one-dimensional heat conduction equation.

$$\rho C\frac{\partial T}{\partial t} = \lambda\frac{1}{r}\frac{\partial}{\partial r}\quad r\frac{\partial T}{\partial r}$$

where,

$\rho$ = material density (lb/in$^3$)

C = material specific heat (Btu/(lb-°R))

$\lambda$ = material thermal conductivity (Btu/(ft-h-°F))

T = local temperature (°F)

t = time (hr)

### 5.1.3 Determination of $K_{Im}$

$K_{Im}$ is determined by the following equation in accordance with ASME Code Section XI Appendix G.

$$K_{Im} = M_m \times \left( \frac{PR_i}{t_{th}} \right) \qquad \text{(psi)}$$

For inside (1/4-T) axial surface flaws, $M_m$ is determined as follows:

$M_m$ = 1.85 for $\sqrt{t_{th}} < 2$

$M_m$ = $0.926\sqrt{t_{th}}$ for $2 \le \sqrt{t_{th}} \le 3.464$

$M_m$ = 3.21 for $\sqrt{t_{th}} > 3.464$

For outside (3/4-T) axial surface flaws, $M_m$ is determined as follows:

$M_m$ = 1.77 for $\sqrt{t_{th}} < 2$

$M_m$ = $0.893\sqrt{t_{th}}$ for $2 \le \sqrt{t_{th}} \le 3.464$

$M_m$ = 3.09 for $\sqrt{t_{th}} > 3.464$

where,

P = internal pressure (ksi)

### 5.1.4 Determination of P-T Limits

Based on the above relationships in this Section, relationships can be established for the P-T limit curves. The relationship for heatup and cooldown operations is as follows.

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

For heatup operation, the inner surface temperatures are higher than the outer surface temperatures, therefore stresses are in compression on the inner surfaces and in tension on the outer surfaces. However, since degredation due to irradiation is higher for the inner surfaces, the above equation is evaluated for both inner (1/4-T) and outer (3/4-T) surfaces, and the limiting pressures determine the limits of operation. For cooldown, only the inner (1/4-T) surfaces are evaluated as tension stresses occur on the inner surfaces, and effects of irradiation are also more conservative for the inner surfaces.

For hydrostatic and leak tests which are performed under isothermal conditions, $K_{It}$ is equal to 0 and the safety factor for $K_{Im}$ is 1.5. The relationship is as follows.

$$P = \frac{K_{Ic}}{1.5\, M_m \left( \dfrac{R_i}{R_o - R_i} \right)}$$

## 5.2 Minimum Temperature Requirements

Per 10 CFR 50 Appendix G, the following minimum temperature requirements are taken into consideration for the operating conditions, in addition to the P-T limits determined in Section 5.1.

### 5.2.1 Heatup and Cooldown Operations

Minimum Boltup Temperature

Minimum boltup temperature is the highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload, as stated in item 2.a of 10 CFR 50 Appendix G, Table 1 (included as Table 5-1 in this report). In the US-APWR standard design, the boltup temperature is specified to be equal to or higher than 70°F. Therefore, the minimum boltup temperature should be the larger of 70°F or the highest reference temperature of the material in the closure flange region.

Required Temperature for Closure Flange

Per item 2.b of 10 CFR 50 Appendix G, Table 1, when the reactor vessel pressure exceeds 20% of the preservice system hydrostatic test pressure, the material temperature is required to be equal to or higher than the highest reference temperature in the closure flange region, plus 120°F.

### 5.2.2 Core Critical Operation (Criticality Limit Temperature)

The criticality limit temperature shall be equal to or higher than the minimum permissible temperature for the inservice leak and hydrostatic test. For the US-APWR, the permissible temperature is determined from the P-T limit for the inservice leak and hydrostatic test at a pressure of 2500 psig (approximately 110% of the normal operating pressure) and compared to the highest reference temperature in the closure flange region, plus 40°F (for a vessel pressure ≤ 20% of the preservice system hydrostatic test pressure) or plus 160°F (for a vessel pressure > 20%). In each case of the vessel pressure, the criticality limit temperature is determined at the larger temperature or higher, in accordance with items 2.c and 2.d of 10 CFR 50 Appendix G, Table 1.

### 5.2.3 Preservice Hydrostatic Tests

For preservice hydrostatic tests prior to loading fuel in the reactor vessel, the minimum test temperature shall be greater than the highest reference temperature of the material in the closure flange region that is highly stressed by the stud bolt preload, plus 60°F, in accordance with item 1.c of 10 CFR 50 Appendix G, Table 1.

**Table 5-1 Pressure and Temperature Requirements for the Reactor Vessel**

| Operating condition | Vessel pressure [1] | 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: | | | |
| 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 ([5]) | ≤ 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.

# 6.0 LOW TEMPERATURE OVERPRESSURE PROTECTION

## 6.1 LTOP Design

LTOP is designed to meet the requirements of Branch Technical Position (BTP) 5-2 of NUREG-0800 (Reference 11), which requires that the LTOP system be designed and installed to prevent exceeding of the applicable technical specifications and Appendix G limits for the RCS while operating at low temperature.

The Residual Heat Removal System (RHRS) is provided with self-actuated water relief valves to prevent overpressure in this relatively low design pressure system caused either within the system itself or from transients transmitted from the RCS. The Containment Spray/Residual Heat Removal (CS/RHR) pump suction relief valves mitigate pressure transients originating in the RCS to maximum pressure values determined by the relief valve set pressure. The CS/RHR pump suction relief valves discharge to the Refueling Water Storage Pit (RWSP) in the containment.

## 6.2 Analyses of Overpressure Transients

Overpressure transient in the RCS at low temperature can be caused by either of two types of events; i.e., mass input or heat input. For mass input case, inadvertent actuation of two high head injection pumps is assumed as an initiating event and for heat input case, inadvertent start of one reactor coolant pump is assumed as a initiating event.

The MARVEL-M plant transient analysis code is used to calculate transient responses of primary coolant pressure for cold overpressure event. Additional details regarding the MARVEL-M code are provided in Reference 12.

BTP 5-2 requires that the system should be able to perform its function assuming any single active component failure. The LTOP system for the US-APWR consists of the CS/RHR pump suction relief valves, which are spring-loaded relief valves. As spring-loaded relief valves are passive components, the LTOP system is a passive system. Therefore, any single active failure does not affect the LTOP system.

<u>Mass Input Case</u>

Figure 6-1 is the plot of primary coolant pressure variation versus time for mass input case, obtained with the following conditions. The largest variation is 120 psi.

- Initial primary coolant temperature: 100°F
- Initial primary coolant pressure: 400 psig
- Solid RCS condition

- Two Residual Heat Removal (RHR) relief valves operable
- Inadvertent start of two high head injection pumps

<u>Heat Input Case</u>

Figure 6-2 is the plot of primary coolant pressure variation versus time for heat input case, obtained with the following conditions. The largest variation is 95 psi.

- Initial primary coolant temperature: 100°F
- Initial secondary coolant temperature: 150°F
- Initial primary coolant pressure: 400 psig
- Solid RCS condition
- Two RHR relief valves operable
- Inadvertent start of one reactor coolant pump

## 6.3 Methodology for Setpoint Determination

The LTOP setpoint is determined to cover with RCS P-T limits, the RCS pressure after the LTOP system actuated, even taking into account the pressure variation evaluated in Section 6.2.

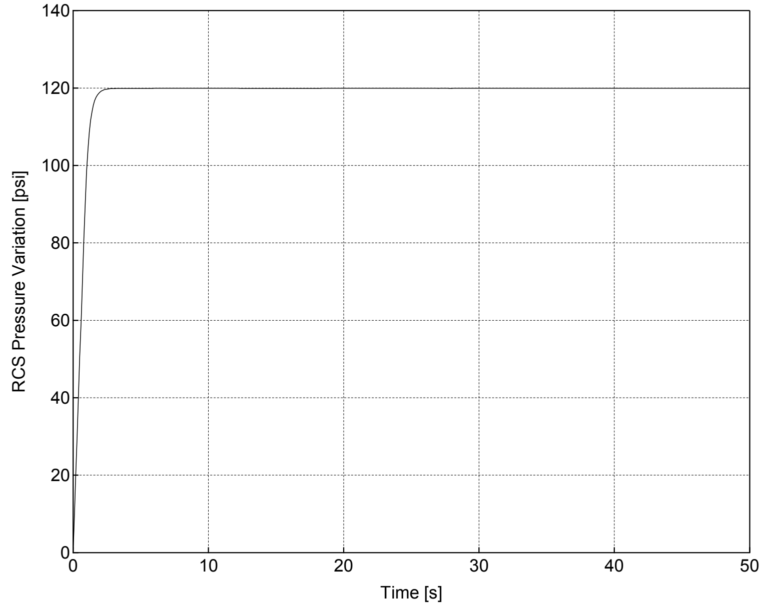


**Figure 6-1 Primary Coolant Pressure Variation (Mass Input Case)**

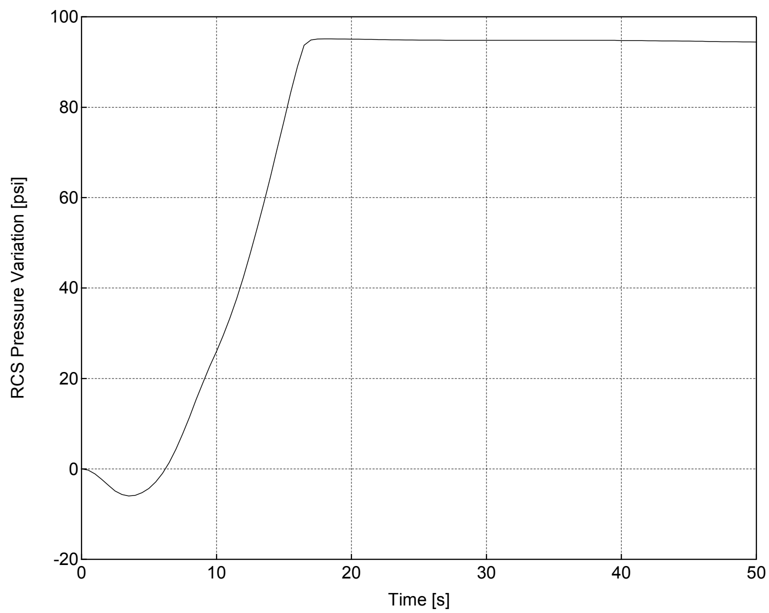


**Figure 6-2 Primary Coolant Pressure Variation (Heat Input Case)**

## 7.0 OPERATING LIMITS

As the generic Pressure and Temperature Limits Report (PTLR) for US-APWR, the items required by Generic Letter 96-03 have been prepared according to Section 3.0 to 6.0 to meet the requirements of LCO 3.4.3 “RCS Pressure and Temperature (P/T) Limits”, and LCO 3.4.12 “Low Temperature Overpressure Protection (LTOP)”.

This PTLR has been developed based on the evaluations of the beltline region materials exposed to the neutron fluence shown in Table 3-1 for 60EFPY. The material properties applied are the bounding values taken from the material specification as described in Section 4.1, as the actual values are not available.

In plant-specific PTLRs, items required by Generic Letter 96-03 will be discussed based on the actual material properties.

## 7.1 RCS Temperature Rate-of-Change Limits (LCO 3.4.3)

### 7.1.1 Maximum Heatup Rate

The RCS heatup rate limit is 50°F in any 1-hour period.

### 7.1.2 Maximum Cooldown Rate

The RCS cooldown rate limit is 100°F in any 1-hour period.

### 7.1.3 Maximum Temperature Change During Inservice Leak and Hydrostatic Testing

During inservice leak and hydrostatic testing operations above the heatup and cooldown limit curves, constant RCS temperature is assumed.

## 7.2 Calculation of Adjusted Reference Temperature

The bounding material specification for the copper and nickel weight percent values for the US-APWR reactor vessel beltline region materials are used to calculate the chemistry factors in accordance with Section 4.2. The limiting materials and the EOL ARTs at the 1/4-t and 3/4-t locations are presented in Table 7-1.

In accordance with Reference 13, the pressurized thermal shock reference temperature ($RT_{PTS}$) values are calculated in the same manner as the ART values. Therefore, $RT_{PTS}$ values of the beltline material are equal to the ART values and also shown in Table 7-1. The $RT_{PTS}$ values at EOL are not expected to exceed the pressurized thermal shock (PTS) screening criteria of Reference 13.

### 7.3 P-T Limit Curves for Heatup, Cooldown, Inservice Leak and Hydrostatic Testing, and Criticality (LCO 3.4.3)

#### 7.3.1 P-T Limit Curves for Heatup, Inservice Leak and Hydrostatic Testing, and Criticality

The P-T limit curves for heatup, inservice leak and hydrostatic testing, and criticality, applicable to 60EFPY, are shown in Figure 7-1.

#### 7.3.2 P-T Limit Curve for Cooldown

The P-T limit curve for cooldown, applicable up to 60EFPY, is shown in Figure 7-2.

#### 7.3.3 Minimum Boltup Temperature

The highest $RT_{NDT}$ of the material in the closure flange region is 10°F, which is the specification limit for ferritic materials out of the beltline region. According to Subsection 5.2.1, therefore, the minimum boltup temperature is 70°F as shown in Figure 7-1.

#### 7.3.4 Required Temperature for Closure Flange

The preservice system hydrostatic test is performed at a pressure of 3107 psig and the highest $RT_{NDT}$ in the closure flange region is 10°F. As described in Subsection 5.2.1, the temperature of the closure flange region is required to be equal to or higher than 130°F ($RT_{NDT}$+120°F), when the reactor vessel pressure exceeds 621 psig, which corresponds to 20% of the preservice system hydrostatic test pressure. While the required temperature is 130°F for cooldown operation (see Figure 7-2), 195°F is required for heatup operation (see Figure 7-1) taking into consideration lower reactor vessel material temperature compared to that of the RCS coolant.

#### 7.3.5 Criticality Limit Temperature

From Figure 7-1, the minimum permissible temperature for inservice leak and hydrostatic tests at 2500 psig is 195°F. Following Section 5.2.2, since the highest $RT_{NDT}$ in the closure flange region plus 160°F (170°F) is lower than the minimum permissible temperature of inservice leak and hydrostatic tests, the criticality limit temperature is required to be equal to or higher than 195°F, regardless of the reactor vessel pressure. Taking into account temperature differences between reactor vessel material and the RCS coolant, the critilality limit temperature is set at 280°F as shown in Figure 7-1.

#### 7.3.6 Preservice Hydrostatic Test Temperature

For preservice hydrostatic tests prior to loading fuel in the reactor vessel, the minimum test temperature is 70°F, which the highest $RT_{NDT}$ in the closure flange region plus 60°F, following Subsection 5.2.3.

### 7.4 Reactor Vessel Material Surveillance Program

A generic capsule withdrawal schedule for the US-APWR reactor vessel surveillance program is provided in Table 7-2. The schedule follows the requirements of ASTM E-185 as required by 10 CFR 50 Appendix H. ASTM E-185 recommends the capsule withdrawal schedule for a design life of 32 EFPY. The minimum number of capsule withdrawal sequences is three (3), as the maximum predicted $\Delta RT_{NDT}$ at the reactor vessel inner surface after 32EFPY is below 100°F. The data for the fluence at the reactor vessel inner surface for 60EFPY can be obtained from the 3rd capsule sequence at approximately 29EFPY.

The test results will be used to verify the validity of $\Delta RT_{NDT}$ that is the basis for the P-T limit curves. As the surveillance capsule results become available, the projected fluence and the $RT_{NDT}$ calculations will be adjusted, if necessary. The development of new P-T curves may become necessary from these adjustments.

Furthermore, once two or more credible surveillance data sets become available after capsule withdrawals from the reactor vessel, this data will be used to calculate chemistry factor values per Position 2.1 of the Reference 7 as described in Subsection 4.2.2.

### 7.5 LTOP System Setpoint (LCO 3.4.12)

Figure 7-1 and 7-2 show representative P-T limit curves for heatup and cooldown, respectively. In the low temperature range (below 130°F), the limit is flat and minimized. In this range, the upper pressure limit is 621 psig.

From the results of analyses described in Section 6.2, the largest pressure variation is 120 psi (mass input case). As the initial pressure condition is 400 psig, the maximum pressure is 520 psig. This pressure is lower than the upper limit, 621 psig. Therefore, the LTOP system ensures the RCS pressure is within the operating limit of the P-T limit curve.

In the P-T limit curve for cooldown, at lower temperature (below 95°F) the upper limit is lower than the flat line. As this pressure limit is too close to the RHR operating pressure (400 psig), it is difficult to prevent overpressure by a system. Therefore, when the RCS temperature is in this lower temperature range, sufficient open area is to be provided to ensure that this pressure limit is not exceeded (e.g, reactor vessel closure head should be removed).

From the above discussion on the validity of the LTOP system, the setpoint is 470 psig as shown in Table 7-3. The setpoint is to be updated when revised P-T limits conflict with the LTOP system limits.

## Table 7-1 Calculation of $RT_{NDT}$ / $RT_{PTS}$ at EOL (60EFPY)

| Location | | Initial $RT_{NDT}$ (°F) | f(1) | FF(2) | CF(3) (°F) | $\Delta RT_{NDT}$(4) (°F) | $\sigma_I$(5) (°F) | $\sigma_\Delta$(6) (°F) | Margin(7) (°F) | ART (°F) |
|---|---|---|---|---|---|---|---|---|---|---|
| Beltline Region Forgings | ID | 0 | 0.98 | 0.99 | 31 | 30.8 | 17 | 15.4 | 45.9 | 76.7 |
| | 1/4-T | | 0.52 | 0.82 | | 25.4 | 17 | 12.7 | 42.4 | 67.8 |
| | 3/4-T | | 0.15 | 0.50 | | 15.6 | 17 | 7.8 | 37.4 | 53.0 |
| Beltline Region Weld | ID | -20 | 0.85 | 0.95 | 108 | 103.1 | 17 | 28 | 65.5 | 148.6 |
| | 1/4-T | | 0.45 | 0.78 | | 84.3 | 17 | 28 | 65.5 | 129.8 |
| | 3/4-T | | 0.13 | 0.47 | | 50.9 | 17 | 25.5 | 61.2 | 92.1 |

Notes:
1. Fluence f ($10^{19}$n/cm$^2$, E>1MeV) at a depth of x (in inches) based on the fluence $f_{surf}$ ($10^{19}$n/cm$^2$, E>1MeV) at the ID is calculated by;
   $f = f_{surf}\ (e^{-0.24x})$, where fsurf is the fluence of ID from Table 3-1.
2. FF (Fluence Factor) = $f^{(0.28 - 0.10 \cdot \log f)}$.
3. Values from Table 1 and Table 2 of Regulatory Guide 1.99 (Reference 7) for Cu = 0.05 wt% and Ni = 1.0 wt% for the forgings, and Cu = 0.08 wt% and Ni = 0.95 wt% for the weld material.
4. $\Delta RT_{NDT}$ = CF × FF.
5. Standard deviation for Initial $RT_{NDT}$. 17°F selected from Table-P Footnote (5) of Reference 14.
6. Standard deviation for $\Delta RT_{NDT}$. $\sigma_\Delta$ = smaller of 17°F or 0.5 ×$\Delta RT_{NDT}$ for the forgings, and smaller of 28°F or 0.5 × ΔRTNDT for the weld material.
7. Margin determined by $\sqrt{\sigma_I^2 + \sigma_\Delta^2}$

**Table 7-2 US-APWR Reactor Vessel Material Surveillance Program (Withdrawal Schedule)**

| Sequence | Capsule Type | Withdrawal Schedule | Note |
|---|---|---|---|
| 1st | A | Approx. 3 EFPY | At the time when the predicted $\Delta RT_{NDT}$ of the capsule material is approximately 50°F. |
| 2nd | A | Approx. 12 EFPY | At the time when the accumulated neutron fluence of the capsule corresponds to the approximate 32EFPY fluence at the reactor vessel inner wall location. |
| 3rd | C | Approx. 29 EFPY | At the time when the accumulated neutron fluence of the capsule corresponds to the peak fluence 60EFPY (not less than once or greater than twice the 32EFPY) at the reactor vessel inner wall location. |
| 4th | B or C | Standby | Supplemental |
| 5th | B or C | Standby | Supplemental |
| 6th | B or C | Standby | Supplemental |

**Table 7-3 CS/RHR Pump Suction Relief Valve Design Data**

| | |
|---|---|
| **Number** | 4 |
| **Design pressure (psig)** | 900 |
| **Design temperature (°F)** | 400 |
| **Minimum required capacity per valve (gpm)** | 1,320 |
| **Set pressure (psig)** | 470 |
| **Fluid** | Reactor Coolant |
| **Inlet and Outlet size (in.)** | 6 |
| **Environmental condition**<br>**Ambient temperature (°F)**<br>**Relative humidity (%)** | <br>Up to 120<br>Up to 100 |

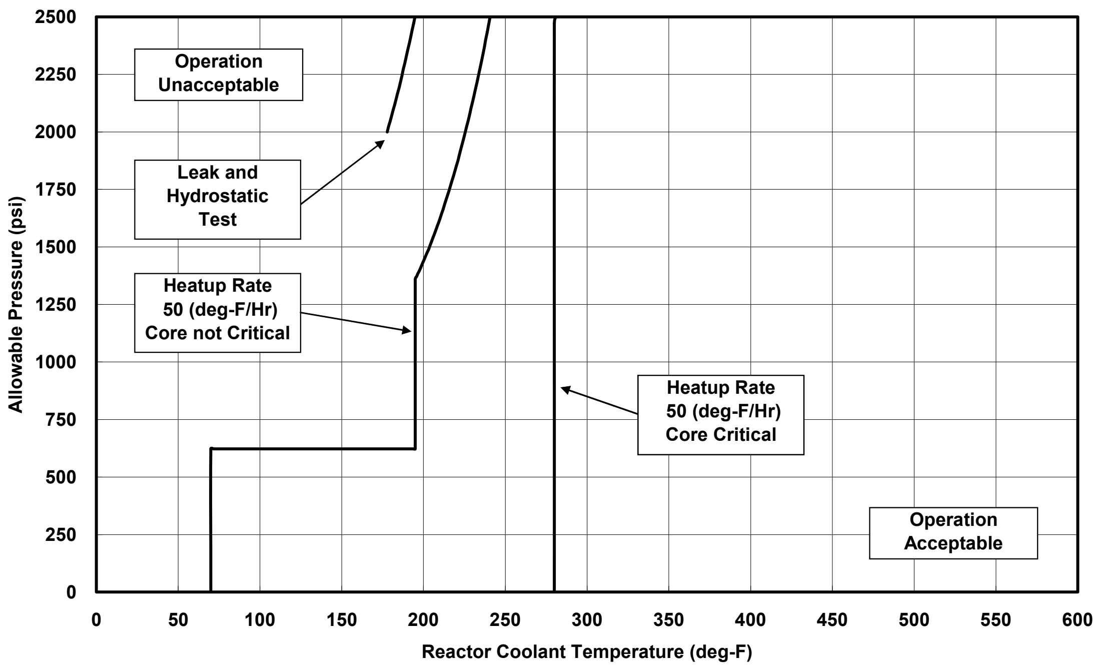


**Figure 7-1 Generic P-T Limit Curves for Heatup, Inservice Leak and Hydrostatic Testing, and Criticality up to 60EFPY**

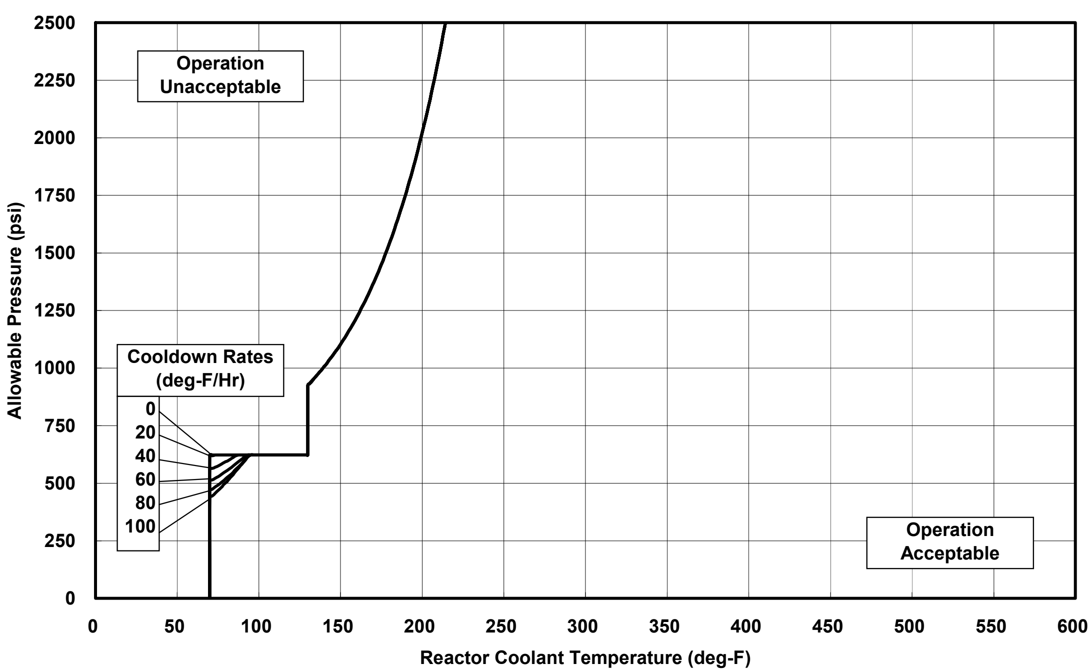


**Figure 7-2 Generic P-T Limit Curve for Cooldown up to 60EFPY**

## 8.0 REFERENCES

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