Licensing Technical Report

Pressure and Temperature Limits Methodology

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Abstract

This report describes the methodology used to develop the pressure-temperature (P-T) limits and the low temperature overpressure protection (LTOP) setpoint for the NuScale Power, LLC, NuScale Power Module (NPM). Plant operation within these limits protects the reactor coolant pressure boundary (RCPB) from non-ductile fracture.

This report bases its requirements and methodology for developing P-T limits on the requirements and the methodologies in Title 10 of the Code of Federal Regulations (CFR) Part 50 (10 CFR 50), Appendix G, and the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code (BPVC) Section XI, Appendix G; the P-T limits in the reactor pressure vessel (RPV) account for vessel embrittlement due to neutron fluence in accordance with Regulatory Guide (RG) 1.99. Representative P-T limits for the NPM are in tables and figures displaying maximum allowable reactor coolant system (RCS) pressure as a function of RCS temperature.

The NPM reactor vessel uses an LTOP system to provide protection against non-ductile failure due to LTOP events during reactor start-up and shutdown operation. The basis of the LTOP methodology in this report is ASME BPVC Section XI, Appendix G. The LTOP setpoints account for the effects of neutron embrittlement.

The basis for representative limits in this report is the projected 57 effective full-power years (EFPY) neutron fluence over the 60-year design life of the module. The P-T limits and LTOP setpoints applicable to operating modules are module-specific based on material properties of as-built reactor vessels. Plant licensees provide these limits, based on the methods provided in this report.

10 CFR 50.61 requires pressurized thermal shock (PTS) screening for the RPV beltline region of pressurized water reactors (PWRs). A PTS event is an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV.

Executive Summary

There are a number of Nuclear Regulatory Commission (NRC) regulations related to reactor coolant pressure boundary (RCPB) integrity, including General Design Criterion (GDC) 31; GDC 32; Title 10 of the Code of Federal Regulations (CFR) Part 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H. Collectively, these regulations require a licensee to

- ensure that the RCPB has sufficient margin to prevent non-ductile failure during all phases of operation, including postulated accident conditions, accounting for material changes due to neutron fluence and temperature history over the life of the RCPB.
- develop reactor vessel pressure-temperature (P-T) limits for the reactor pressure vessel (RPV), which are limitations on reactor operating pressure as a function of reactor coolant temperature for various operating conditions.
- develop and maintain an appropriate surveillance program to monitor reduction in material toughness in ferritic materials over the life of the reactor vessel.

This report presents the methodologies used to demonstrate that the regulatory requirements identified above are met or are not applicable to the NuScale Power Module (NPM) reactor vessel. Historically, P-T limits were in the plant's technical specifications. The NRC guidance in Generic Letter (GL) 96-03 provides a means of relocating the P-T limits to a pressure-temperature limits report (PTLR), which facilitates modifications to P-T limits as needed over the life of the plant. Moving the P-T limits from the technical specifications to the PTLR requires the licensee to develop methods and programs to address each of the following aspects:

- neutron fluence calculation method
- adjusted reference temperature (ART) calculation method to account for the effects of neutron embrittlement
- minimum temperature requirements for the reactor vessel during various operational and testing modes
- reactor vessel surveillance program (RVSP) for ferritic steel
- the low temperature overpressure protection (LTOP) setpoint calculation method

This report addresses each of these topics as applicable to the NPM design. A licensee may use the methods found in this report to develop a PTLR rather than maintaining P-T limits in the plant's technical specifications. This report also includes the pressurized thermal shock (PTS) screening results.

1.0 Introduction

1.1 Purpose

This report describes the methodology used to develop the NuScale Power Module (NPM) heatup and cooldown curves (pressure-temperature (P-T) curves) and low temperature overpressure protection (LTOP) setpoints. Operation within these limits protects the reactor vessel from brittle fracture. This report also provides an embrittlement analysis in accordance with Regulatory Guide (RG) 1.99 (Reference 6.1.1) and outlines whether the design requires a reactor vessel surveillance program (RVSP). This report includes the pressurized thermal shock (PTS) screening results.

1.2 Scope

This report provides a methodology for development of P-T limits for the NPM reactor coolant pressure boundary (RCPB) including

- heatup and cooldown curves and P-T limits for normal operation.
- the P-T limits for in-service leak and hydrostatic tests.
- the LTOP setpoints.

In addition, this report provides values for each of these items based on assumed material properties at an exposure of 57 effective full-power years (EFPY) fluence, which represents the end-of-design-life neutron exposure based on a 60-year design life of the module with an assumed 95 percent capacity factor. This report does not provide P-T limits for use in an as-built NPM; the P-T limits must be created on a module-specific basis with consideration of the material properties of the as-built reactor pressure vessel (RPV). Licensees may reference the methods contained in this report to develop their module-specific pressure-temperature limits report (PTLR), or they may choose to develop an alternative methodology.

This report includes the PTS screening results.

In accordance with Generic Letter (GL) 96-03 (Reference 6.1.2), this report addresses the following five methodology aspects:

- neutron fluence calculation method
- the adjusted reference temperature (ART) calculation method to account for the effects of neutron embrittlement, in accordance with Reference 6.1.1
- minimum temperature requirements for the reactor vessel during various operational and testing modes based on Appendix G of Reference 6.1.3
- the RVSP for ferritic steel
- the LTOP setpoint calculation method

Table 1-1 Abbreviations

Term	Definition
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CNV	containment vessel
EFPY	effective full-power years
GDC	General Design Criterion
ISLH	inservice leak and hydrostatic testing
LTOP	low temperature overpressure protection
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
P-T	pressure and temperature
PTLR	pressure and temperature limits report
PTS	pressurized thermal shock
RCPB	reactor coolant pressure boundary
RG	Regulatory Guide
RPV	reactor pressure vessel
RT_{NDT}	nil-ductility reference temperature
RVSP	reactor vessel surveillance program
RVV	reactor vent valve
SIF	stress intensity factor

2.0 Background

This report outlines the P-T limits methodology and the LTOP setpoints methodology that can be used by a licensee to create a module-specific PTLR for an NPM. In addition, this report outlines the neutron fluence calculation method, ART calculation method, minimum P-T curves, RVSP recommendations, LTOP setpoint calculation method, and PTS screening results.

2.1 Regulatory Requirements and Recommendations

2.1.1 General Design Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

General Design Criterion (GDC) 31 requires that the RCPB have sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner, and there is minimal probability of rapidly propagating fracture.

Changes in material properties must account for service temperatures and other conditions of the pressure boundary material under operating, maintenance, testing, and postulated accident conditions, as well as the uncertainties in determining

- material properties.
- the effects of irradiation on material properties.
- residual, steady state, and transient stresses.
- size of flaws.

2.1.2 General Design Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

General Design Criterion 32 requires that the RCPB be designed to permit periodic inspection and testing of important areas and an appropriate material surveillance program for the RPV.

2.1.3 10 CFR 50.60 - Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation

Regulation 10 CFR 50.60 (Reference 6.1.3) requires that light water reactors meet the fracture toughness and material surveillance program requirements set forth in Appendix G and Appendix H of Reference 6.1.3. Proposed alternatives to the requirements described in Appendix G and Appendix H of Reference 6.1.3 or portions thereof are allowed when the NRC grants an exemption under 10 CFR 50.12. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

2.1.4 10 CFR 50, Appendix G - Fracture Toughness Requirements

Appendix G of Reference 6.1.3 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation. Conditions of normal operation include anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

2.1.5 10 CFR 50, Appendix H - Reactor Vessel Material Surveillance Program Requirements

Appendix H of Reference 6.1.3 establishes the necessary material surveillance program to satisfy GDC 32 for light water reactors. Appendix H of Reference 6.1.3 requires that licensees establish and maintain a material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. The materials in the reactor vessel beltline region undergo exposure to neutron irradiation and to the thermal environment. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Upper RPV ferritic materials, which are outside the beltline, do not exceed the Appendix H of Reference 6.1.3 threshold for requiring an RVSP.

2.1.6 Generic Letter 96-03 - Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits

Reference 6.1.2 provides information that describes the methodology that licensees may use to create PTLRs.

2.1.7 Regulatory Guide 1.99 - Radiation Embrittlement of Reactor Vessel Materials

Reference 6.1.1 provides general procedures that calculate the effects of neutron embrittlement of low-alloy steels used in light water reactor vessels.

2.1.8 10 CFR 50.61 - Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

Regulation 10 CFR 50.61 requires PTS screening for the RPV beltline region of pressurized water reactors (PWRs). A PTS event is an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV. The NPM design supports an exemption to 10 CFR 50.61

due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

3.0 Analysis

3.1 Materials

In accordance with Appendix G of Reference 6.1.3, the calculations in this report apply to the pressure-retaining components of the RCPB. Because Appendix G of Reference 6.1.3 only contains data and methods applicable to ferritic materials, and because the NPM lower RPV is not made of ferritic materials, this report also evaluates ferritic materials in the upper RPV (i.e., the region above the upper flange).

Table 3-1 lists the materials in the RPV. Table 3-2 lists the materials in the containment vessel (CNV). Figure 3-1, Figure 3-2, and Figure 3-3 show the material distribution model for the RPV and the CNV.

Table 3-1 Reactor Pressure Vessel Material Distribution

Component	Material
Lower Seismic Cap	SA-693, Type 630, Condition H1100
Lower Head	SA-965, Grade FXM-19
Lower Flange Core Region Shell	SA-965, Grade FXM-19
Upper Flange Shell	SA-508, Grade 3 Class 2
Upper Feed Plenum Shell	SA-508, Grade 3 Class 2
Upper Steam Generator Shell	SA-508, Grade 3 Class 2
Upper Support Ledge Shell	SA-508, Grade 3 Class 2
Upper Support Ledge Shell Cladding	Alloy 690
RPV - CNV Support Ledge	SB-168, Alloy 690
Upper Steam Plenum Shell	SA-508, Grade 3 Class 2
Upper Pressurizer Shell	SA-508, Grade 3 Class 2
Upper Head	SA-508, Grade 3 Class 2
Interior and exterior cladding, except for the RPV upper support ledge exterior cladding	308L/309L

Table 3-2 Containment Vessel Material Distribution

Component	Material
Lower Seismic Support Pads	SA-479, Type 304
Lower Support Skirt	SA-182, Grade F304
Lower Head	SA-965, Grade FXM-19
Lower Core Region Shell	SA-965, Grade FXM-19
Lower Transition Shell	SA-965, Grade FXM-19
Buttering and Weld between the Lower Shell and Lower Transition Shell	Alloy 52/152
Lower Shell	SA-336, Grade F6NM
Lower Flange	SA-336, Grade F6NM
Upper Flange	SA-336, Grade F6NM
Upper Support Ledge Shell	SA-336, Grade F6NM
Upper Steam Generator Access Shell	SA-336, Grade F6NM
Upper Intermediate Shell	SA-336, Grade F6NM
Upper Manway Access Shell	SA-336, Grade F6NM
Upper Seismic Support Shell	SA-336, Grade F6NM
Upper Head	SA-336, Grade F6NM
Control Rod Drive Mechanism Top Head Cover	SA-182, Grade F6NM

Figure 3-1 Two Dimensional Model Material Distribution

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}}2(a),(c),ECI

Figure 3-2 Two Dimensional Model Containment Vessel Material Distribution

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}}2(a),(c),ECI

Figure 3-3 Two Dimensional Model Reactor Pressure Vessel Material Distribution

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}}2(a),(c),ECI

3.1.1 Neutron Fluence and Ferritic Materials

Per Appendix G of Reference 6.1.3, the calculations in this report apply to the pressure-retaining components of the RCPB. The lower RPV (i.e., the region below the upper flange) undergoes exposure to higher neutron fluence than other portions of the RCPB; however, the NPM lower RPV is made of austenitic stainless steel rather than ferritic materials (Table 3-1). Despite the higher neutron fluence in the lower RPV region, the use of austenitic stainless steel ensures safety of the RCPB because austenitic stainless steel has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

Appendix G of Reference 6.1.3, provides the following definition of the RPV beltline.

Beltline or Beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

The NPM design does not require testing Charpy upper shelf energy per Appendix G of Reference 6.1.3, for the following reason.

- Appendix G of Reference 6.1.3 applies to ferritic materials, while the portion of the NPM in the beltline region is austenitic stainless steel. The ASME BPVC Section III, NB-2311, does not require impact testing for austenitic stainless steel because these materials do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. Because impact testing is not required for austenitic stainless steel, the nil-ductility reference temperature (RT_{NDT}) cannot be calculated. The NRC endorsed ASME BPVC Section III in 10 CFR 50.55a.

Appendix H of Reference 6.1.3 specifically applies to ferritic steel because the requirements for an RVSP were developed for ferritic materials and there is no guidance for an RVSP for austenitic stainless steel. Furthermore, austenitic stainless steel has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Because the lower RPV is austenitic stainless steel, the ferritic portion of the RPV that experiences the highest fluence is evaluated against the Appendix H of Reference 6.1.3 criteria requiring an RVSP. The upper RPV lower flange has a design life peak fluence less than 1E+17 n/cm^2, E > 1 MeV, Section III.A of Appendix H of Reference 6.1.3 does not require an RVSP.

3.2 Adjusted Reference Temperature

There is no ART for the NPM because there is no need to adjust the RT_{NDT} for fluence because the peak neutron fluence at the top of the lower flange of the RPV is

{{ }}$^{2(a),(c),ECI}$, which is less than the 1E+17 n/cm^2, E > 1 MeV regulatory limit.

Appendix H of Reference 6.1.3 requires beltline material surveillance if the portions of the RPV experience a maximum fluence greater than 1.0E+17 n/cm^2, E > 1 MeV; however, the portion of the RPV experiencing the highest neutron fluence is the lower RPV, which is made of austenitic stainless steel. The ASME BPVC Section III, NB-2311, does not require impact testing for austenitic stainless steels because they do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. Without impact testing, RT_{NDT} cannot be calculated for austenitic stainless steel, and thus ART is not applicable. The NRC endorsed ASME BPVC Section III in 10 CFR 50.55a. Since the upper RPV is the only part of the RPV made of ferritic materials, an evaluation of the upper RPV experiencing the highest design life peak fluence indicates that the upper RPV neutron fluence would have to increase by a factor of {{ }}$^{2(a),(c),ECI}$ to experience a fluence greater than 1.0E+17n/cm^2, E > 1 MeV; therefore, there is no need to adjust the reference temperatures.

3.3 Scope of Pressure-Temperature Limits Analysis

In order to develop a P-T limits methodology for the NPM, this report calculates minimum P-T limits for the NPM upper RPV design based on the requirements of Appendix G of Reference 6.1.3 and based on the methodologies in ASME BPVC Section XI, Appendix G (Reference 6.1.5). Finite element models simulate thermal transient stress and analyze fracture mechanics.

3.3.1 Thermal Transients

Thermal transients, in the context of this evaluation, include two heat transfer mechanisms: convection and radiation.

Convection is considered on the following surfaces:

- internal surfaces of the RPV (free and forced convection)
- the annulus between the RPV and CNV when flooded during the heatup and cooldown transients (free convection)
- the outside of the CNV for locations submerged in the pool (free convection)

Radiation is considered in the following regions:

- between the RPV outer surface and the CNV inner surface
- between the lower and upper RPV in the gap in the RPV flange
- between the lower and upper CNV in the gap in the CNV flange
- between the upper CNV and the control rod drive mechanism access cover at the closure surface

Convection driven by condensation in the upper pressurizer is also a driving heat transfer mechanism that occurs during these transients when the pressurizer wall temperature dips below saturation temperature. This occurrence can increase the convective film coefficients.

The four thermal transients considered in this evaluation include

- heatup.
- power ascent.
- power descent.
- cooldown.

This report creates P-T limit curves for the following transient conditions:

- heatup, including power ascent. The heatup transient begins with the annulus between the RPV and CNV flooded with water.
- cooldown, starting with power descent. The cooldown transient includes the annulus between the RPV and the CNV flooded with water.
- inservice leak and hydrostatic testing (ISLH). The ISLH considers both steady state and heatup/cooldown transient conditions.

3.3.1.1 Heatup Transient

Figure 3-4 shows the heatup transient.

Figure 3-4 Transient Temperature for Heatup

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}}2(a),(c),ECI

3.3.1.2 Power Ascent Transient

Figure 3-5 shows the power ascent transient.

Figure 3-5 Power Ascent Transient Definition - Temperatures and Convection Coefficients

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}}2(a),(c),ECI

3.3.1.3 Power Descent Transient

Figure 3-6 shows the power descent transient.

Figure 3-6 Power Descent Transient Definition - Temperatures and Convection Coefficients

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}}2(a),(c),ECI

3.3.1.4 Cooldown Transient

Figure 3-7 shows the cooldown transient.

Figure 3-7 Cooldown Transient Definition - Temperatures and Convection Coefficients

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}}2(a),(c),ECI

3.3.2 Fracture Mechanics

This report analyzes axial and circumferential flaw locations at the most limiting thermal and pressure stress locations. Fracture mechanics analyses consider postulated flaws as follows:

- axial flaws: one-fourth thickness from the inner surface and one-fourth thickness from the outer surface.
- circumferential flaws: one-fourth thickness from the inner surface and one-fourth thickness from the outer surface.

3.3.3 Pressure and Temperature Limit Methodology

3.3.3.1 Pressure Boundary Components

In accordance with Appendix G of Reference 6.1.3, the calculations in this report bound the pressure-retaining components of the RCPB. The lower RPV is austenitic stainless steel (Table 3-1). Section 3.1 discusses the material distribution for the RPV and CNV. This evaluation considers the upper RPV because it contains ferritic materials.

3.3.3.2 Maximum Postulated Cracks

The methods of Appendix G, Article G-2214.1, of Reference 6.1.5 postulate the existence of a sharp surface crack in the RPV that is normal to the direction of the maximum stress. As specified in paragraph G-2120 of Reference 6.1.5, the crack depth is one-fourth of the RPV wall thickness, and the crack length is 1.5 times the wall thickness. This report considers both inside and outside surface cracks in axial and circumferential directions individually. For crack evaluations, a single crack is present in the RPV.

Figure 3-8 and Figure 3-9 show representations of circumferential and axial cracks, respectively.

Figure 3-8 Representation of Postulated Semi-Elliptical Circumferential Cracks in Reactor Pressure Vessel Wall

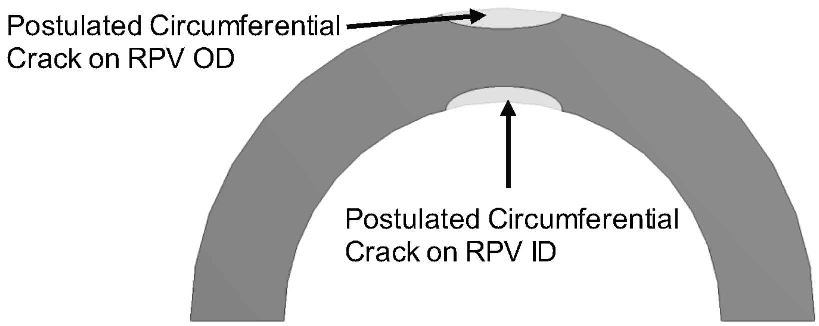
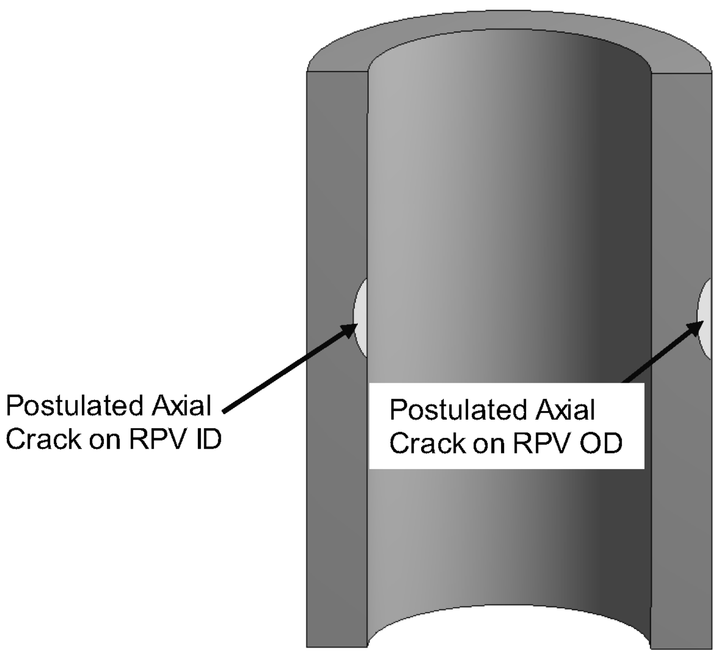


Figure 3-9 Representation of Postulated Semi-Elliptical Axial Cracks in Reactor Pressure Vessel Wall



3.3.3.3 Fracture Toughness

Appendix G, Article G-2110, of Reference 6.1.5 requires use of the critical stress intensity factor (SIF), K_{IC}, defined by Equation 3-1, in P-T limits calculations.

$$K_{IC} = 33.2 + 20.734\exp[0.02(T - RT_{NDT})] \quad \text{Eq. 3-1}$$

Where:

K_{IC} = Critical SIF measuring fracture toughness (ksi in $^{0.5}$).

T = Temperature at crack tip (degrees F).

RT_{NDT} = Reference temperature for nil-ductility transition (degrees F).

The conservative limit on upper shelf fracture toughness, K_{IC} from Equation 3-1, has an upper bound value of $200\ \mathrm{ksi} \cdot \mathrm{in}^{0.5}$, which is slightly lower than the upper cutoff of lower bound K_{IC} in Appendix G, Article G-2212 of Reference 6.1.5. The crack-tip temperatures needed for these fracture toughness calculations are from transient thermal analysis.

3.3.3.4 Fracture Mechanics Analysis

3.3.3.4.1 Calculation of Stress Intensity Factors due to Internal Pressure

Appendix G, Article G-2214.1, of Reference 6.1.5 provides a method to calculate K_{Im} corresponding to membrane tension for postulated axial and circumferential cracks. This method applies to locations away from geometric discontinuity where calculation of hoop stress and axial stress occurs directly through an influence coefficient M_m (M_{m_axial} for axial cracks and M_{m_circ} for circumferential cracks).

For postulated axial cracks:

$$K_{Im\_axial} = M_{m\_axial}(pR_i / t) \quad \text{Eq. 3-2}$$

Where:

p = internal pressure (ksi).

R_i = vessel inner radius (inches).

t = vessel wall thickness (inches).

On the inside surface:

$$M_{m\_axial} = \begin{cases} 1.85 & \text{for } t < 4 \text{ in} \\ 0.926\sqrt{t} & \text{for } 4 \text{ in} \le t \le 12 \text{ in} \\ 3.21 & \text{for } t > 12 \text{ in} \end{cases} \qquad \text{Eq. 3-3}$$

On the outside surface:

$$M_{m\_axial} = \begin{cases} 1.77 & \text{for } t < 4 \text{ in} \\ 0.893\sqrt{t} & \text{for } 4 \text{ in} \le t \le 12 \text{ in} \\ 3.09 & \text{for } t > 12 \text{ in} \end{cases} \qquad \text{Eq. 3-4}$$

And for postulated circumferential cracks on the inside or outside surface:

$$K_{Im\_circ} = M_{m\_circ}(pR_i/t) \qquad \text{Eq. 3-5}$$

$$M_{m\_circ} = \begin{cases} 0.89 & \text{for } t < 4 \text{ in} \\ 0.443\sqrt{t} & \text{for } 4 \text{ in} \le t \le 12 \text{ in} \\ 1.53 & \text{for } t > 12 \text{ in} \end{cases} \qquad \text{Eq. 3-6}$$

Equation 3-2 through Equation 3-6 are not valid for cracks postulated at locations with a geometric discontinuity. A finite element analysis crack model calculates the SIFs due to pressure for all locations. A unit pressure (1 psig) is applied to the RPV inner surface. The SIFs for the crack tip node at the deepest point are calculated for five contours. The maximum value from contours two through five for the deepest point is the maximum SIF (K_{Im}) for this evaluation. The first contour is not used because it is not accurate due to numerical inaccuracies in the stresses and strains at the crack tip.

3.3.3.4.2 Calculation of Stress Intensity Factors due to Thermal Stress

The hoop and axial thermal stresses are curve-fit to third order polynomial functions, which calculate thermal stress intensity factors K_{IT}. The format of the polynomial function is:

$$\sigma = c_0 + c_1\left(\frac{x}{a}\right) + c_2\left(\frac{x}{a}\right)^2 + c_3\left(\frac{x}{a}\right)^3 \qquad \text{Eq. 3-7}$$

Where c_0, c_1, c_2, and c_3 are coefficients.

σ =hoop stress or axial stress used to calculate SIF for postulated axial or circumferential crack (psi).

a =crack depth (inches).

x =distance from the appropriate (i.e., inside or outside) surface with $x = a$ at the deepest crack tip (inches).

Appendix G, Article G-2214.3(b) of Reference 6.1.5 provides generic equations to calculate K_{IT} for radial thermal gradient for any thermal stress distribution. For postulated axial and circumferential cracks away from geometry discontinuity, the following equations calculate SIFs.

For an inside surface crack during a cooldown transient:

$$K_{IT} = (1.0359c_0 + 0.6322c_1 + 0.4753c_2 + 0.3855c_3)\sqrt{\pi a} \qquad \text{Eq. 3-8}$$

For an outside surface crack during a heat up transient:

$$K_{IT} = (1.043c_0 + 0.630c_1 + 0.481c_2 + 0.401c_3)\sqrt{\pi a} \qquad \text{Eq. 3-9}$$

Where *a* is the crack depth (inches), and c_0, c_1, c_2 and c_3 are coefficients of the third order polynomial equation for hoop or axial thermal stresses.

Equation 3-8 and Equation 3-9 are not accurate for cracks postulated at locations with a geometric discontinuity. A finite element analysis crack model calculates the SIFs due to transient thermal stresses by the superposition principle. To do so, a unit pressure (1psig) is applied to the crack top face and crack bottom face in four separate steps.

1. Constant unit pressure, set $c_0 = 1$, $c_1 = 0$, $c_2 = 0$ and $c_3 = 0$ in Equation 3-7. The calculated SIF is $K_{It\_c_0}$.
2. Linear pressure along the crack depth direction, set $c_0 = 0$, $c_1 = 1$, $c_2 = 0$ and $c_3 = 0$ in Equation 3-7. The calculated SIF is $K_{It\_c_1}$.
3. Quadratic pressure along the crack depth direction, set $c_0 = 0$, $c_1 = 0$, $c_2 = 1$ and $c_3 = 0$ in Equation 3-7. The calculated SIF is $K_{It\_c_2}$.
4. Cubic pressure along the crack depth direction, set $c_0 = 0$, $c_1 = 0$, $c_2 = 0$ and $c_3 = 1$ in Equation 3-7. The calculated SIF is $K_{It\_c_3}$.

The SIFs for the crack tip node at the deepest point are calculated for five contours. The maximum value from the integrals of contour paths two through five is the maximum SIF. The proposed crack-specific equation to calculate SIFs for any axial/circumferential inside/outside surface cracks is:

$$K_{IT} = c_0 K_{It\_c_0} + c_1 K_{It\_c_1} + c_2 K_{It\_c_2} + c_3 K_{It\_c_3} \qquad \text{Eq. 3-10}$$

Where c_0, c_1, c_2, and c_3 are the actual coefficients of the 3rd order polynomial equation. If K_{IT} is negative, the allowable pressure calculation uses a zero value.

3.3.3.5 American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Appendix G, Limits

This section documents the Appendix G of Reference 6.1.5 methodology for calculating the RPV allowable pressure for preservice hydrostatic test, normal heatup and cooldown transients, and ISLH conditions. This report documents development of a representative set of P-T calculations.

The ASME BPVC allowable pressure is part of the Appendix G of Reference 6.1.3 requirements. Except for the preservice hydrostatic test, the requirement of Appendix G of Reference 6.1.3 is that the test temperature must be greater than 50 degrees F.

The fundamental equation that is used to calculate P-T limits with a required safety margin is given by:

$$K_{I\,applied} = K_{IC} \qquad \text{Eq. 3-11}$$

Where K_{IC} is the lower bound crack initiation fracture toughness factor for the material as represented in Equation 3-1, and $K_{I\,applied}$ is the stress intensity factor due to pressure and thermal gradient loads at the tip of the one-fourth T postulated cracks.

$$K_{I\,applied} = SF \cdot M_m \cdot (pR_i/t) + K_{IT} \qquad \text{Eq. 3-12}$$

Where SF is the required structural factor applied to the pressure loading, and dependent on which P-T limits curve is being evaluated, M_m is the influence coefficient from Section 3.3.3.4.1, and K_{IT} is calculated using the Section 3.3.3.4.2 methodology.

The allowable pressure associated with a specified temperature along a P-T limits curve is:

$$P = \frac{(K_{IC} - K_{IT})t}{SF \cdot M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{SF \cdot K_{Im}} \qquad \text{Eq. 3-13}$$

The appropriate K_{IT} and SF values used for various conditions are:

- For preservice hydrostatic tests, a steady-state condition ($K_{IT} = 0$) is applied, and the required structural factor $SF = 1$.

$$P = \frac{K_{IC}t}{M_m \cdot R_i} = \frac{K_{IC}}{K_{Im}} \qquad \text{Eq. 3-14}$$

Performance of the allowable pressure calculation occurs for the crack with highest M_m that bounds other cracks. The basis for the preservice limiting pressure is NUREG-0800, Section 5.3.2 (Reference 6.1.4).

For the heat up and cooldown transients, the thermal SIF K_{IT} calculation occurs at selected time points, and the required structural factor $SF = 2$.

$$P = \frac{(K_{IC} - K_{IT})t}{2M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{2K_{Im}} \qquad \text{Eq. 3-15}$$

For ISLH, the SIF K_{IT} from heat up and cooldown transients conservatively apply to the most limiting crack, and the required structural factor $SF = 1.5$.

$$P = \frac{(K_{IC} - K_{IT})t}{1.5M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{1.5K_{Im}} \qquad \text{Eq. 3-16}$$

3.3.3.6 10 CFR 50, Appendix G, Pressure and Temperature Limits

Appendix G of Reference 6.1.3 requires that the P-T limits are at least as conservative as limits obtained by following the Appendix G of Reference 6.1.5, methods presented in Section 3.3.3.5. Additionally, Table 1 of Appendix G of Reference 6.1.3 requires further limitations (Table 3-3).

Table 3-3 Pressure and Temperature Requirements for the Reactor Pressure Vessel

Operating Condition	Vessel Pressure(1)	Requirements for Pressure-Temperature Limits	Minimum Temperature Requirements
Hydrostatic Pressure and Leak Tests (core is not critical)			
Fuel in the Vessel	≤ 20%	ASME BPVC § XI App. G Limits	(2)
Fuel in the Vessel	> 20%	ASME BPVC § XI App. G Limits	(2) + 90 degrees F(5)
No Fuel in the Vessel (preservice hydrostatic test)	all	Not Applicable	(3) + 60 degrees F
Normal Operation (including heatup and cooldown), Including Anticipated Operational Occurrences			
Core Not Critical	≤ 20%	ASME BPVC § XI App. G Limits	(2)
Core Not Critical	> 20%	ASME BPVC § XI App. G Limits	(2) + 120 degrees F(5)
Core Critical	≤ 20%	ASME BPVC § XI App. G Limits + 40 degrees F	maximum of (4) or ((2) + 40 degrees F)
Core Critical	> 20%	ASME BPVC § XI App. G Limits + 40 degrees F	maximum of (4) or ((2) + 160 degrees 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 in-service system hydrostatic pressure test.
5. 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.

3.4 Reactor Vessel Surveillance Program Consideration

Appendix H of Reference 6.1.3 states:

> *The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment.*

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel does not exceed 1E+17 n/cm^2, E > 1 MeV.

Because Appendix H of Reference 6.1.3 applies to ferritic materials with peak neutron fluence at the end of the design life above 1E+17 n/cm^2, E > 1 MeV, the NPM reactor pressure vessel has no RVSP requirement in accordance with Appendix H of Reference 6.1.3 because the lower RPV is made of austenitic stainless steel and the maximum design life peak fluence of the ferritic portion of the RPV is below 1E+17 n/cm^2, E > 1 MeV. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic materials in the RPV beltline region.

3.5 Low Temperature Overpressure Protection

The NPM reactor vessel uses LTOP systems for protection against failure during reactor start-up and shutdown operation due to LTOP events classified as service level A or B events. Per Appendix G, paragraph G-2215, of Reference 6.1.5, LTOP systems must be effective at coolant temperatures less than 200 degrees F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 degrees F, whichever is greater. {{

}}2(a),(c),ECI

3.6 Pressurized Thermal Shock

Regulation 10 CFR 50.61 requires the PTS screening for the RPV beltline region of PWRs. A PTS event means an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with, or followed by, significant pressure within the RPV. The 10 CFR 50.61 definition of beltline is:

> *(The) RPV beltline means the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

Regulation 10 CFR 50.61 (Reference 6.1.3) does not define significant radiation damage. However, Appendix H of Reference 6.1.3 requires the monitoring of ferritic RPV beltline materials with peak neutron fluence at the end of the design life exceeding 1E+17 n/cm^2, E > 1 MeV.

The 10 CFR 50.61 (Reference 6.1.3) PTS screening methodology is based on calculating the reference temperature for PTS (RT_{PTS}). The RT_{PTS} means RT_{NDT} evaluated for the end of design life peak fluence for each of the vessel beltline materials using the 10 CFR 50.61 (Reference 6.1.3) procedures per the following 10 CFR 50.61 equation:

$$RT_{PTS} = RT_{NDT(u)} + \Delta RT_{NDT} + Margin \qquad \text{Eq. 3-17}$$

The $RT_{NDT(U)}$ is the reference temperature RT_{NDT} before service (unirradiated condition) established by impact testing per NB-2311 of Reference 6.1.6.

The 10 CFR 50.61(b)(2) (Reference 6.1.3) acceptance criteria for passing the PTS screening are: RT_{PTS} not to exceed 270 degrees F for plates, forgings, and axial welds, and RT_{PTS} not to exceed 300 degrees F for circumferential welds.

4.0 Results

4.1 Adjusted Reference Temperature

There is no ART because there is no need to adjust RT_{NDT} for fluence because the peak fluence at the top of the lower flange of the RPV is {{ }}2(a),(c),ECI, which is less than 1E+17n/cm^2, E > 1 MeV.

Appendix H of Reference 6.1.3 requires surveillance of ferritic materials that experience a maximum fluence greater than 1.0E+17 n/cm^2, E > 1 MeV. The upper RPV fluence would have to increase by a factor of {{ }}2(a),(c),ECI to experience a fluence greater than 1.0E+17n/cm^2, E > 1 MeV; therefore, there is no need to adjust the reference temperatures.

4.2 Pressure Temperature Limits

The Appendix G of Reference 6.1.3 P-T limits are based on the requirements presented in Table 3-3. {{

}}2(a),(c),ECI Table 4-1 presents the application of Appendix G, Table 1 of Reference 6.1.3 to the NPM reactor pressure vessel. Figure 4-1 through Figure 4-7 present the uncorrected P-T limits curves.

Table 4-1 Pressure-Temperature Limits for NuScale Power Module Reactor Pressure Vessel per 10 CFR 50, Appendix G

Operating Condition	Vessel Pressure(1)	Requirements for Pressure-Temperature Limits	Minimum Temperature Requirements
Hydrostatic Pressure and Leak Tests (core is not critical)			
Fuel in the Vessel	535.3 psig	ASME BPVC § XI App. G Limits	0 degrees F
Fuel in the Vessel	535.5 psig	ASME BPVC § XI App. G Limits	90 degrees F
No Fuel in the Vessel (preservice hydrostatic test)	535.5 psig	Not Applicable	60 degrees F
Normal Operation (including heatup and cooldown), Including Anticipated Operational Occurrences			
Core Not Critical	535.5 psig	ASME BPVC § XI App. G Limits	0 degrees F
Core Not Critical	535.5 psig	ASME BPVC § XI App. G Limits	120 degrees F
Core Critical	535.5 psig	ASME BPVC § XI App. G Limits + 40degrees F	90 degrees F
Core Critical	535.5 psig	ASME BPVC § XI App. G Limits + 40degrees F	160 degrees F

Notes:
1. Percent of the preservice system hydrostatic test pressure.

Table 4-2 Summary of Pressure-Temperature Limits - Normal

Normal Combined Heatup and Power Ascent Transient (Core Not Critical)		Composite Normal (Core Critical with RPV Pressure = 20 Percent Pressure = 535.3 psig)		Composite Normal (Core Critical with RPV Pressure > 20 Percent Pressure = 535.3 psig)		Normal Combined Power Descent and Cooldown	
		(Minimum core critical temperature determined from the steady state and transient ISLH curves)					
Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig
65	535	Reactor is not permitted to be critical below 90°F if ISLH testing is performed at steady-state or transient conditions.		Reactor is not permitted to be critical below 160°F if ISLH testing is performed at steady-state or transient conditions.		600	3260
120	535					220	3260
120	2230					210	2400
150	2230	90	0	160	0	150	1875
200	2285	90	535	160	1875	120	1875
300	2475	160	535	190	1875	120	535
600	2475	160	1875	240	2285	65	535
		190	1875	340	2475		
		240	2285	640	2475		
		340	2475				
		640	2475				

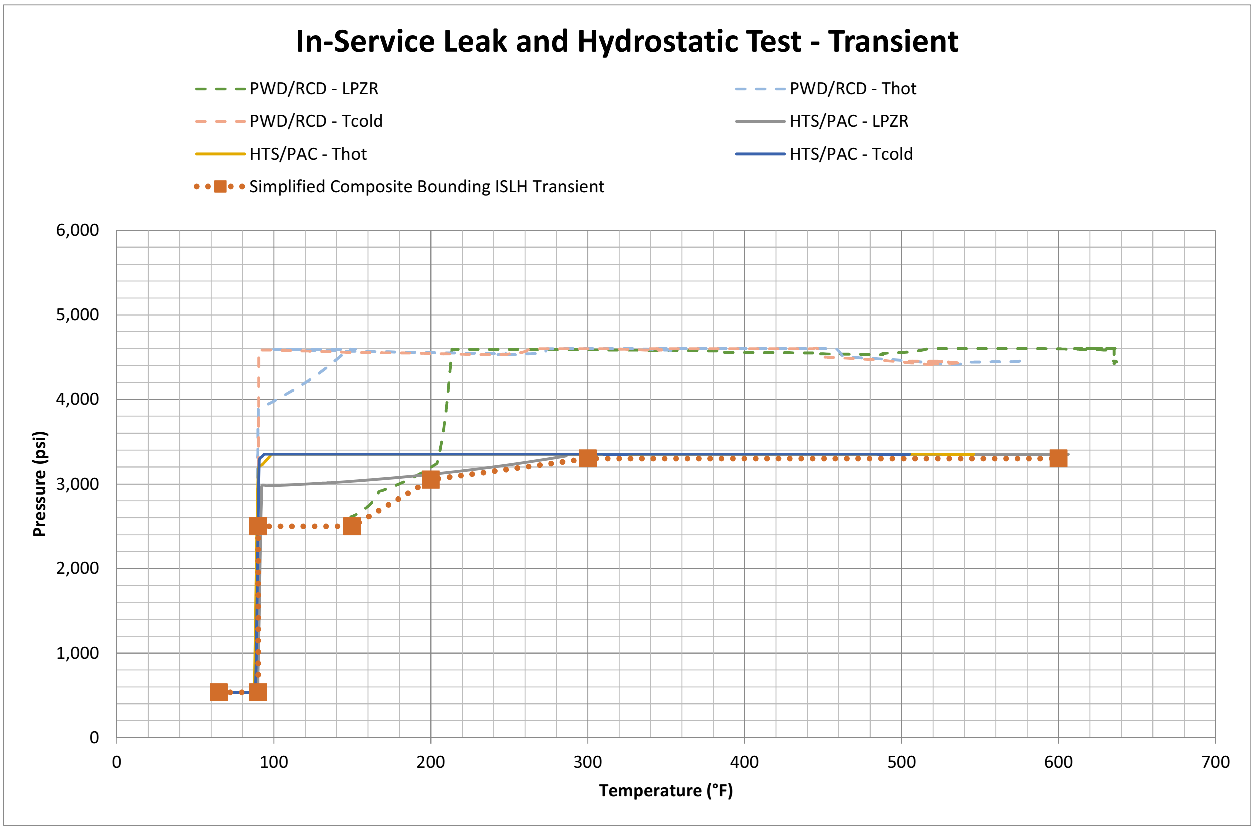
Note: Linear interpolation can be used to calculate the allowable pressures for the temperatures not listed in the table.

Table 4-3 Summary of Pressure-Temperature Limits - Inservice Leak and Hydrostatic Testing

ISLH for Combined Heatup and Power Ascent Transient		ISLH for Combined Power Descent and Cooldown Transient		Transient ISLH (Bounding)		Steady-State ISLH	
Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig
65	535	600	4350	65	535	65	535
90	535	220	4350	90	535	90	535
90	2980	210	3200	90	2500	90	3660
150	2980	150	2500	150	2500	95	3960
200	3050	90	2500	200	3050	100	4300
300	3300	90	535	300	3300	105	4610
600	3300	65	535	600	3300	600	4610

Note: Linear interpolation can be used to calculate the allowable pressures for the temperatures not listed in the table.

Figure 4-1 Pressure-Temperature Limits for Transient Inservice Leak and Hydrostatic Testing (Composite of Transients)



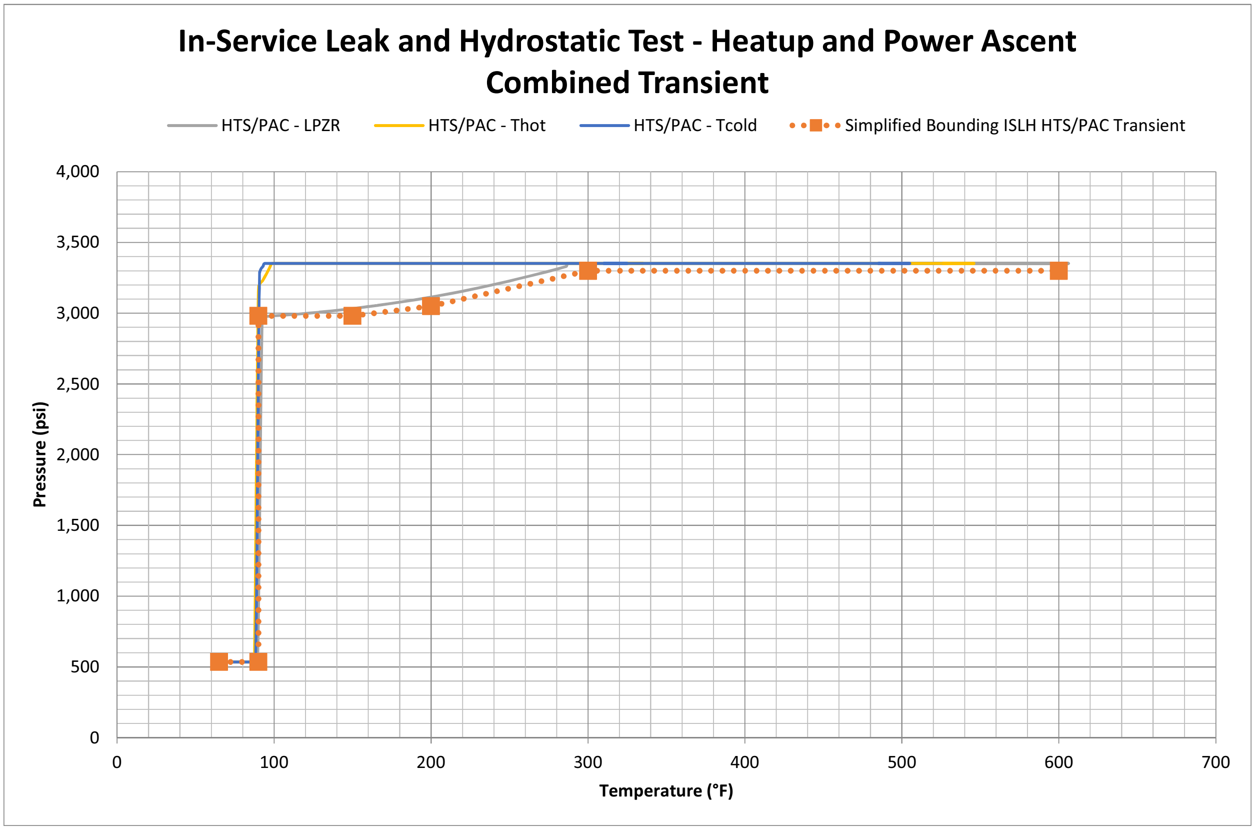
Notes:
1. This image is intended to be viewed in color.
2. The following are abbreviations used in the figure above:
   a. PWD: power descent
   b. RCD: cooldown
   c. LPZR: lower pressurizer region
   d. Thot: reactor coolant system hot temperature
   e. Tcold: reactor coolant system cold temperature
   f. HTS: heatup transient
   g. PAC: power ascent

Figure 4-2 Pressure-Temperature Limits for Steady-State Inservice Leak and Hydrostatic Testing

{{

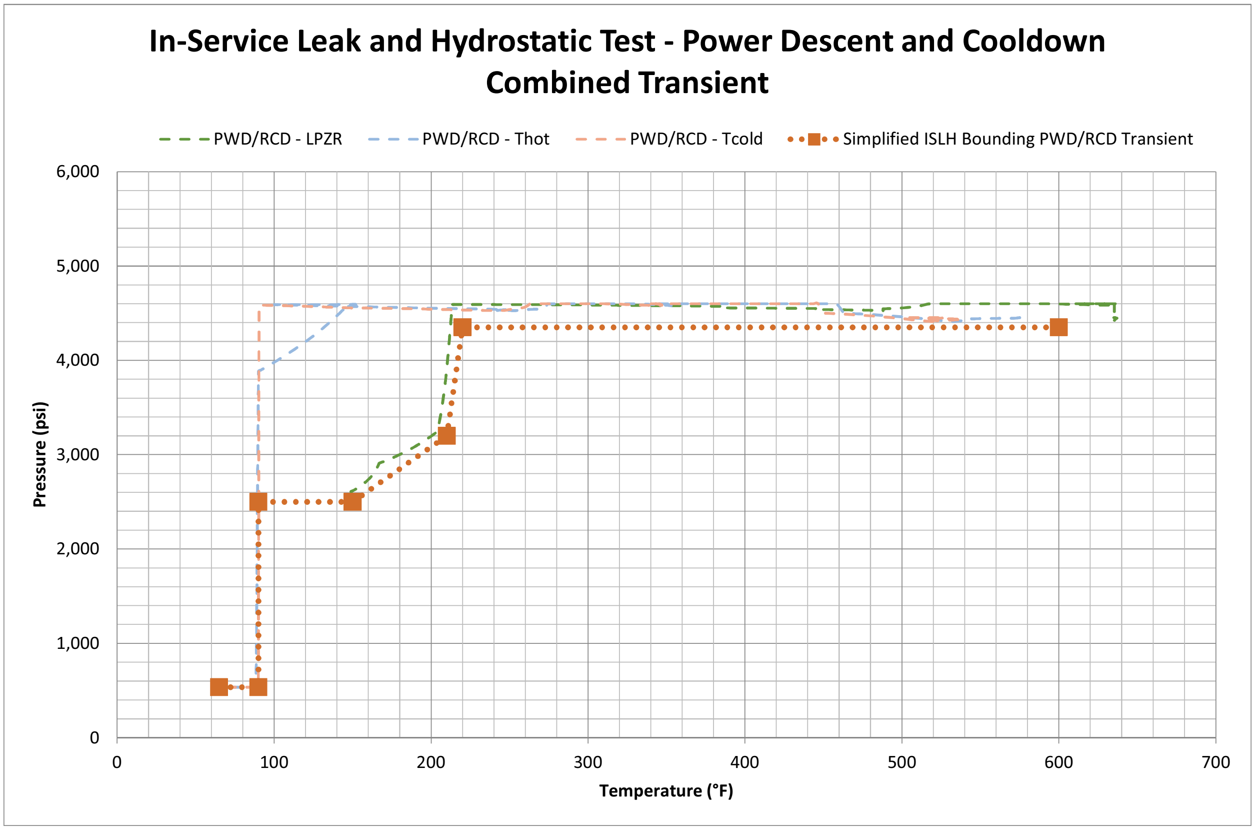
}}2(a),(c),ECI

Figure 4-3 Pressure-Temperature Limits for Bounding Heatup and Power Ascent Transient Inservice Leak and Hydrostatic Testing



Notes:
1. This image is intended to be viewed in color.
2. The following are abbreviations used in the figure above:
    a. HTS: heatup
    b. PAC: power ascent
    c. LPZR: lower pressurizer region
    d. Thot: reactor coolant system hot temperature
    e. Tcold: reactor coolant system cold temperature

Figure 4-4 Pressure-Temperature Limits for Bounding Power Descent and Cooldown Transient Inservice Leak and Hydrostatic Testing



Notes:
1. This image is intended to be viewed in color.
2. The following are abbreviations used in the figure above:
    a. PWD: power descent
    b. RCD: cooldown
    c. LPZR: lower pressurizer region
    d. Thot: reactor coolant system hot temperature
    e. Tcold: reactor coolant system cold temperature

Figure 4-5 Pressure-Temperature Limits for Bounding Normal Heatup and Power Ascent Transient

{{

}}2(a),(c),ECI

Figure 4-6 Pressure-Temperature Limits for Bounding Normal Power Descent and Cooldown Transient

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}}2(a),(c),ECI

Figure 4-7 Pressure-Temperature Limits for Core Critical Heatup/Power Ascent and Power Descent/Cooldown Transients

{{

}}2(a),(c),ECI

4.3 Low Temperature Overpressure Protection Setpoint Limits

The LTOP setpoint limits the maximum pressure in the reactor vessel to less than the pressure limit curves in Figure 4-8. It uses the minimum pressure from the heatup and cooldown curves. Overpressure protection occurs by opening the two reactor vent valves (RVVs) located on the head of the reactor vessel when exceeding the LTOP pressure setpoint. For a given cold temperature, a pressurizer pressure above the LTOP setpoint causes the module protection system to send a RVV open signal. Above {{ }}[2(a),(c),ECI], the reactor safety valves provide overpressure protection. The LTOP logic and components can continue to perform their function in the event of a single active failure.

The reactor safety valves do not lift when LTOP is enabled. This calculation accounts for pressure and temperature measurement uncertainties, the static pressure difference between the pressure measurement and the bottom of the RPV, the maximum delay in the RVV opening, and the delay in sensor response and module protection system processing time.

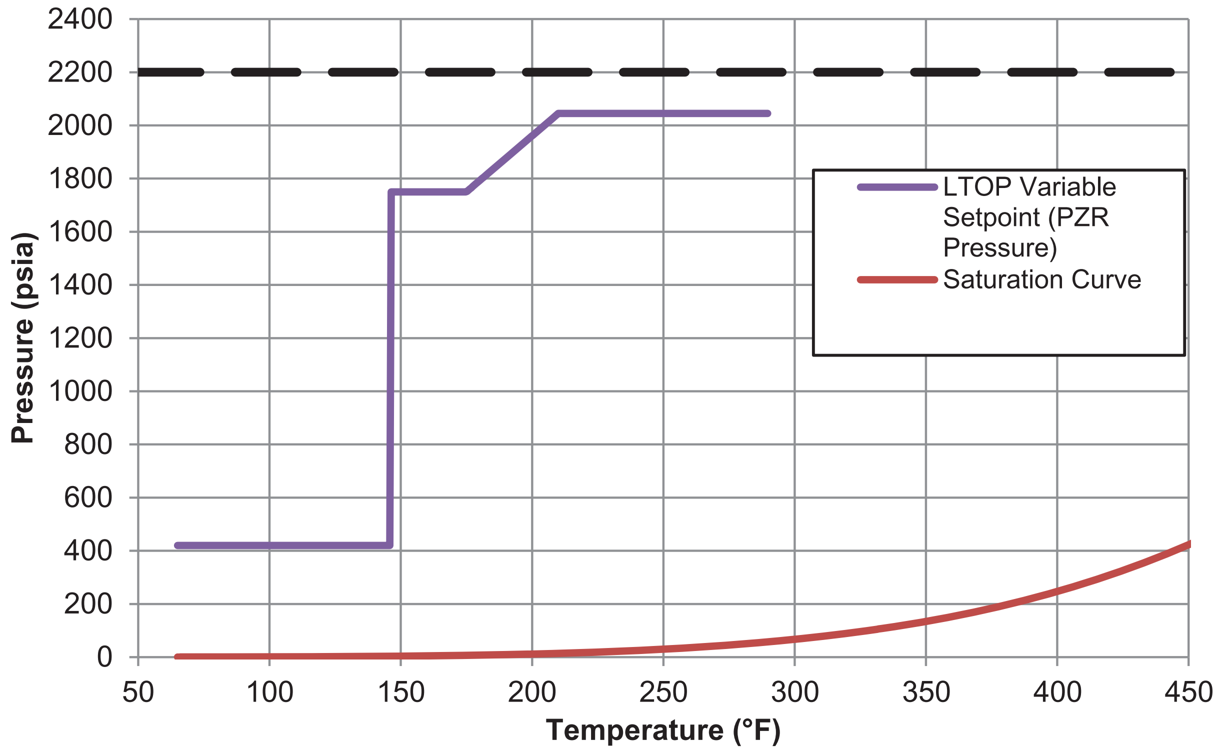
The pressurizer pressure determines the recommended LTOP setpoint; the LTOP setpoint has a conservative bias for the elevation head to the bottom of the RPV. Table 4-4 shows the recommended LTOP pressure setpoint as a function of reactor coolant system (RCS) cold temperature.

Table 4-4 Recommended Low Temperature Overpressure Protection Pressure Setpoint as a Function of Cold Temperature

T_{cold} (degrees F)	Pressurizer Pressure (psia)
<146.0	420.0
146.0	1750.0
175.0	1750.0
210.0	2025.0
290.0	2025.0
>290.0	LTOP not enabled

Figure 4-8 shows the recommended LTOP setpoint along with the saturation pressure curve.

Figure 4-8 Recommended Low Temperature Overpressure Protection Setpoint



4.3.1 Pressurized Thermal Shock Screening

The $RT_{NDT(U)}$ is the RT_{NDT} before service (unirradiated condition) established by impact testing per NB-2331 of Reference 6.1.6.

The 10 CFR 50.61(b)(2) (Reference 6.1.3) acceptance criteria for passing the PTS screening are as follows: RT_{PTS} not to exceed 270 degrees F for plates, forgings, and axial welds; and RT_{PTS} not to exceed 300 degrees F for circumferential welds.

Per NB-2311 of Reference 6.1.6, austenitic stainless steels are exempt from impact test requirements and therefore are exempt from RT_{NDT} requirements of NB-2331 of Reference 6.1.6. While 10 CFR 50.61 (Reference 6.1.3) does not specifically state that it applies only to ferritic materials, the chemistry factors in Table 1 and Table 2 of 10 CFR 50.61 (Reference 6.1.3) were derived for ferritic materials. Therefore, the PTS screening requirements in 10 CFR 50.61 (Reference 6.1.3) do not apply to the austenitic stainless steel used in the lower RPV of the NPM (Table 3-1). While there are ferritic materials in the upper RPV of the NPM, the 57 EFPY design life peak fluence for the top surface of the lower RPV flange is {{ }}[2(a),(c),ECI]. Hence, the design life peak fluence for the upper RPV shell is below the Appendix H of Reference 6.1.3 threshold value of 1E+17 n/cm^2, E > 1 MeV for the RPV beltline. Therefore, PTS screening of the upper RPV is not required, and PTS screening does not apply to the lower RPV shell.

5.0 Summary and Conclusions

This report contains methodology based on Appendix G of Reference 6.1.3 and Appendix G of Reference 6.1.5 for the RCPB and P-T limits applicable to the NPM. An example set of P-T curves applicable to the NPM included in this report use these methods. These limits account for the effects of neutron-induced embrittlement up to an exposure of 57 EFPY fluence. Curves developed include

- transient ISLH.
- steady-state ISLH.
- bounding heatup and power ascent transient ISLH.
- bounding power descent and cooldown transient ISLH.
- bounding normal heatup and power ascent.
- bounding normal power descent and cooldown.
- core critical.

The NPM design does not necessitate an RVSP to ensure adequate fracture toughness.

This report contains the LTOP limits and methodology for the NPM.

Using the material properties of an as-built reactor vessel, the licensee may use the methods developed in this report to develop their P-T limits and LTOP setpoints.

The PTS screening requirement of 10 CFR 50.61 (Reference 6.1.3) does not apply to the NPM reactor pressure vessel.

6.0 References

6.1 Source Documents

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

6.1.2 U.S. Nuclear Regulatory Commission, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," Generic Letter 96-03, January 1996.

6.1.3 U.S. Code of Federal Regulations, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy," (10 CFR 50).

6.1.4 U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, Revision 2, June 1987.

6.1.5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, New York, NY.

6.1.6 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Rules for Construction of Nuclear Facility Components, New York, NY.