Pressure and Temperature Limits Methodology

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Department of Energy Acknowledgement and Disclaimer

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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) set point for the NuScale Power, LLC (NuScale) standard plant. Plant operation within these limits protects the reactor coolant pressure boundary (RCPB) from non-ductile fracture.

The P-T limits developed in this report are based on the requirements and the methodologies in 10 CFR 50, Appendix G and ASME Section XI, Appendix G, and account for vessel embrittlement due to neutron fluence in accordance with RG 1.99, Revision 2. Representative P-T limits for the NuScale standard plant are presented as tables and figures displaying maximum allowable reactor coolant system (RCS) pressure as a function of RCS temperature.

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

Representative limits developed in this report are based on the projected 57 effective full-power years (EFPY) neutron fluence over the 60-year design life. The P-T limits and LTOP setpoints applicable to operating units are unit-specific based on material properties of as-built reactor vessels. These limits will be provided by plant licensees and can be based on the methods provided in this report.

The NuScale reactor vessel contains samples of the material used in the construction of the lower reactor pressure vessel (RPV). These samples are located inside the reactor vessel adjacent to the vessel wall at the beltline elevation, which is the level at which the greatest neutron exposure is expected. The samples are exposed to approximately the same temperature as the vessel beltline. Owing to their location, the samples accumulate neutron damage at a faster rate than does the reactor vessel, and can therefore be used as predictors of neutron damage to the vessel. Future changes in material properties of the vessel can be estimated by removing and testing a representative number of these samples. This report summarizes the reactor vessel material surveillance program.

Executive Summary

There are a number of U.S. Nuclear Regulatory Commission (NRC) regulations related to reactor coolant pressure boundary (RCPB) integrity, including general design criterion (GDC) 31, GDC 32, 10 CFR 50.60, 10 CFR 50 Appendix G, and 10 CFR 50 Appendix H. Collectively, these regulations require a licensee to

- ensure the RCPB is designed with 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, which are limitations on reactor operating pressure as a function of reactor coolant temperature for various operating conditions.
- develop and maintain a surveillance program to monitor reduction in material toughness over the life of the reactor vessel.

This report presents a methodology to demonstrate compliance with these requirements and provides a representative set of calculations and results for the NuScale standard reactor vessel.

Historically, P-T limits were included in the plant's technical specifications. 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 they are needed over the life of the plant. Moving the P-T limits to the PTLR requires the licensee to develop methods and programs to address each of the following aspects:

1. neutron fluence calculation method
2. adjusted reference temperature (ART) calculation method to account for the effects of neutron embrittlement
3. limiting ART
4. minimum temperature requirements for the reactor vessel during various operational and testing modes
5. reactor vessel surveillance program (RVSP)
6. low temperature overpressure protection (LTOP) setpoint calculation method

This report addresses each of these topics. A licensee may utilize the methods found in this report to develop a PTLR rather than maintaining P-T limits in the plant's technical specifications.

1.0 Introduction

1.1 Purpose

The purpose of this report is to describe the methodology used to develop the NuScale Power, LLC (NuScale) standard plant heatup and cooldown curves (pressure-temperature curves) and low temperature overpressure protection (LTOP) set points. 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 11.4), and describes the reactor vessel surveillance program (RVSP) to be developed and implemented by the licensee.

1.2 Scope

This report provides a methodology for development of pressure and temperature limits for the NuScale standard plant reactor coolant pressure boundary (RCPB) including:

- heatup and cooldown curves and pressure-temperature (P-T) limits for normal operation
- the P-T limits for in-service leak and hydrostatic tests
- the LTOP set points

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.

The report does not provide pressure and temperature limits for use in an as-built plant; these limits must be created on a unit-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 unit-specific pressure-temperature limits report (PTLR), or may choose to develop an alternative methodology.

In accordance with GL 96-03 (Reference 11.5), this report addresses the following six methodology aspects:

1. neutron fluence calculation method
2. ART calculation method accounting for neutron embrittlement in accordance with Radiation Embrittlement of Reactor Vessel Materials, Regulatory Guide 1.99
3. limiting ART
4. minimum temperature requirements based on Appendix G to 10 CFR 50 (Reference 11.1)
5. RVSP
6. the LTOP setpoint calculation method

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
ART	adjusted reference temperature
CNV	containment vessel
EFPY	effective full-power years
GDC	General Design Criterion
ID	inside diameter
ISLH	inservice leak and hydrostatic
LTOP	low temperature overpressure protection
MCNP	Monte Carlo N–Particle Transport Code
NRC	U.S. Nuclear Regulatory Commission
OD	outside diameter
P-T	pressure-temperature
PTLR	pressure-temperature limits report
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RPV	reactor pressure vessel
RT_{NDT}	reference temperature for nil-ductility transition
ΔRT_{NDT}	transition temperature shift in RT_{NDT}
RVSP	reactor vessel surveillance program
SIF	stress intensity factor

Table 1-2 Definitions

Term	Definition
Beltline	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. This includes all regions of the reactor vessel for which the end of life fluence of neutrons with energy greater than 1 MeV is greater than 10^{17} n/cm^2. Beltline is further defined in Section 3 of this document.
End of life	For purposes of this report, end of life of the reactor vessel is 57 effective full-power years.
RT_{NDT}	The reference temperature for a reactor vessel material, under any conditions. For the reactor vessel beltline materials, reference temperature for nil-ductility transition (RT_{NDT}) must account for the effects of neutron radiation.
ΔRT_{NDT}	The transition temperature shift, or change in RT_{NDT}, due to neutron radiation effects, that is evaluated as the difference in the 30 ft-lb (41 J) index temperatures from the average Charpy curves measured before and after irradiation.
Steady-state ISLH testing	Inservice leak and hydrostatic (ISLH) testing performed after a one-hour soak period, during which the reactor coolant system (RCS) T_{AVG}, T_{HOT} and T_{COLD} do not vary more than +/- 5 degrees F from the mean.

2.0 Regulatory Considerations

2.1 General Design Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

General Design Criterion (GDC) 31 requires the RCPB to be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions: (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. Changes in material properties must account for service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining: (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

2.2 General Design Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

GDC 32 requires licensees to develop and maintain a material surveillance program for the RCPB.

2.3 10 CFR 50.60 – Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation

10 CFR 50.60 requires licensed light water reactors to meet the fracture toughness and material surveillance program requirements for the RCPB set forth in Appendices G and H of 10 CFR 50. Proposed alternatives to the described requirements in Appendices G and H or portions thereof may be used when an exemption is granted by the U.S. NRC under § 50.12.

2.4 10 CFR 50 Appendix G – Fracture Toughness Requirements

This regulation 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, including anticipated operational occurrences and system hydrostatic tests, that the pressure boundary may be subjected to over its service lifetime.

2.5 10 CFR 50 Appendix H – Reactor Vessel Material Surveillance Program Requirements

This regulation requires licensees to 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 that result from exposure of these materials to neutron irradiation and the thermal environment.

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

GL 96-03 provides information that is needed to describe the methodology that licensees will use to create PTLRs, and therefore directly applies to this report.

2.7 Regulatory Guide 1.99 – Radiation Embrittlement of Reactor Vessel Materials

RG 1.99 provides general procedures for calculating the effects of neutron radiation embrittlement of low-allow steels used for light-water reactor vessels.

3.0 Design Inputs

In accordance with 10 CFR 50, Appendix G, the calculations in this report bound the ferritic materials for pressure-retaining components of the RCPB.

3.1 Reactor Pressure Vessel Beltline and Beltline Materials

3.1.1 Definition of Reactor Pressure Vessel Beltline per 10 CFR 50

10 CFR 50 Appendix H provides the following threshold fluence before an RVSP is required:

> *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 all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 1E+17 n/cm^2, E > 1 MeV.*

10 CFR 50 Appendix G provides the following definition of 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.*

When the RVSP requirements in 10 CFR 50 Appendix G and Appendix H are taken together, the RPV beltline refers to the portion of the RPV whose design life peak fluence exceeds 1E+17 n/cm^2, E > 1 MeV. That is, the RPV ferritic materials (ferritic base metal, weld metal, and heat affected zone) within this RPV beltline must be monitored by a RVSP per 10 CFR 50 Appendix H for irradiation effect on RT_{NDT} and Charpy upper shelf energy by 10 CFR 50 Appendix G.

3.1.2 NuScale Reactor Pressure Vessel Beltline Extent and Materials

Figure 3-1 illustrates the lower RPV configuration. The lower RPV has no vertical welds and one circumferential weld between the lower RPV shell and RPV bottom head. The lower RPV includes the following three items:

- lower RPV shell
- lower RPV weld
- RPV bottom head

Using the definition of beltline developed in Section 3.1.1, the extent of NuScale RPV beltline can be delineated by the 57-EFPY RPV fluence values. The 57-EFPY fluence represents the end-of-design-life fluence because NuScale has a 60-year design life and an assumed 95 percent capacity factor.

Table 3-1 shows that the peak 57-EFPY fluence at the RPV bottom alignment feature is above 1E+17 n/cm^2, E > 1 MeV. Therefore, the entire RPV bottom head is within the RPV beltline.

Table 3-1 lists the following 57-EFPY peak fluences, E > 1 MeV for delineating NuScale beltline extent; the bounding 57-EFPY fluences used for predicting RPV neutron embrittlement per RG 1.99 are listed in Table 4-1 and Appendix A:

a) {{

}}$^{2(a),(c),ECI}$

Table 3-1 End of life neutron fluence for defining RPV beltline

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}}$^{2(a),(c),ECI}$

Lower Reactor Pressure Vessel Geometry

Figure 3-1 illustrates the NuScale lower RPV, which is the region of the reactor vessel exposed to most neutron fluence. The lower RPV has no vertical weld and only one circumferential weld. Lower RPV geometry is described in Table 3-2 and material descriptions are provided in Table 3-3.

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

Figure 3-1 Lower reactor pressure vessel

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

Table 3-2 Lower reactor pressure vessel geometry

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

3.2 Materials

Table 3-3 Reactor pressure vessel beltline material chemistry

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

3.3 Minimum Pool and Containment Vessel Flooding Water Temperature

The minimum pool and containment vessel (CNV) flooding water temperature is assumed to be 40 degrees F.

3.4 Heatup Transient

The heatup transient assumed in this report is shown in Figure 3-2. In addition, a bounding heatup rate of 100 °F/hr is analyzed in this report.

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

Figure 3-2 Transient temperature for heatup

3.5 Cold Shutdown Valve Alignment Temperature

The RCS cooldown transient assumed in this report is shown in Figure 3-3. The maximum RCS temperature is 350 degrees F when CNV flooding is initiated. This analysis assumes that flood water from the reactor pool is 40 degrees F. The temperature transient for water in the CNV is shown in Figure 3-4.

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

Figure 3-3 Coolant temperature for cooldown

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

Figure 3-4 Containment vessel water temperature during cooldown flooding

3.6 Preservice Hydrostatic Test Pressure and Temperature

In accordance with the requirements of ASME Section III, NB-6000 (Reference 11.8), a preservice hydrostatic test pressure will be performed at no less than 1.25 times the system design pressure, 2100 psia. Therefore, the minimum preservice hydrostatic test pressure is 2625 psia. The minimum specified hydrostatic test temperature of 70 degrees F is used in this report.

3.7 Inservice Leak and Hydrostatic Test Pressure

The ISLH test pressure is assumed to be 2035 psia, which is ten percent above the normal operating pressure of 1850 psia as required by ASME Section XI (Reference 11.9), Table IWB-5230.

4.0 Components of the Pressure-Temperature Calculations

4.1 Neutron Fluence

A Monte Carlo method using Monte Carlo N–Particle Transport Code (MCNP) 6.1 (Reference 11.10) was chosen to develop the fluence profile. MCNP6.1 is a general-purpose Monte Carlo N-Particle code which can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three dimensional configuration of materials in geometric cells bounded by first and second degree surfaces and some special fourth degree surfaces. ENDF-B/VII.I pointwise (continuous energy) cross-section data are available with the MCNP6.1 package used in this analysis.

The neutron flux in in the RPV in units of n/cm^2-sec is calculated by using MCNP cylindrical mesh tallies of the neutron flux by using a defined mesh structure superimposed over the region of interest. Each data block provides the mesh central coordinates in three dimensions, as well as tally results and its relative error.

The fast neutron (E > 1 MeV) fluence (n/cm^2) in the RPV is calculated using flux tallies with a 1 MeV energy cutoff. Neutron fluence is evaluated at several locations, including the surveillance capsules, by using MCNP mesh tallies.

The total fission neutron source intensity S in the NuScale module at a given power is determined by the following equation.

$$S = \frac{\upsilon P * 10^6\left[\frac{W}{MW}\right]}{1.602\times10^{-13}\left[\frac{J}{MeV}\right] * K_{eff} * Q_{ave}} \qquad \text{Eq. 4-1}$$

where,

ν: Average number of neutrons produced per fission,

P: Fission power defined (MW),

K_{eff}: Effective multiplication factor of 1.000 for critical light water reactor, and

Q_{ave}: The average recoverable energy per fission for all materials (MeV).

Using the above equation, the calculated fission neutron intensity of the NuScale module operating at 160 MW is 1.23×10^{19} n/s at the beginning of cycle 1 in watt fission energy spectrum distribution. The fission neutron spatial distribution in the NuScale module base model is assumed homogeneous in both axial and radial direction. Exposure averaged profiles over the multiple cycles were used in the best estimate calculation.

The conversion of neutron flux to an accumulated 57-EFPY fluence (in units of n/cm^2) is completed by multiplying the calculated MCNP neutron flux at different operational cycles (in units of n/cm^{2}-sec) with the time constant of each cycle for a total of 57 EFPY.

Each mesh tally is mapped into specific zones of the RPV components, with segments along radial direction of R, axial direction of Z and azimuthal direction of Theta. The maximum neutron exposure is calculated and represents the exposure of a specific material at a specific location.

Fluences were determined at the surfaces using the Monte Carlo method described above. At depths of ¼ and ¾ the thickness of the material, fluence was determined using the method of RG 1.99:

$$f = f_{surf} * \exp(-0.24x) \qquad \text{Eq. 4-2}$$

where "x" is the depth (in inches) below 0-T.

Table 4-1 Fast neutron (energy > 1 MeV) fluence in reactor pressure vessel at 57 EFPY

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

4.2 Evaluation of Adjusted Reference Temperature

In order to demonstrate that the reactor vessel will not become embrittled beyond acceptable limits, the adjusted reference temperature (ART) is calculated and compared against the acceptance criterion found in RG 1.99.

4.2.1 Adjusted Reference Temperature Definition

The ART is defined by the methodology in regulatory position 1.1 of RG 1.99:

$$\mathrm{ART} = \mathrm{Initial\ RT_{NDT}} + \Delta \mathrm{RT_{NDT}} + \mathrm{Margin} \qquad \text{Eq. 4-3}$$

Each term is discussed below.

4.2.2 Acceptance Criterion

Regulatory position 3 of RG 1.99 has the following requirement for new plants:

> *"The copper content should be such that the calculated adjusted reference temperature at the ¼-T position in the vessel wall at end of life is less than 200 degrees F."*

4.2.3 Initial Reference Temperature for Nil-Ductility Transition

Initial RT_{NDT} is the unirradiated RT_{NDT}. A set of initial RT_{NDT} values for the standard plant is shown in the following table.

Table 4-2 Reactor pressure vessel beltline material properties

Location	Material	Initial RT_{NDT}
Lower RPV Shell	SA-508 Grade 3 Class 1	-10°F max
Lower RPV Weld	Low Alloy Steel Weld	-20°F max
RPV Bottom Head	SA-508 Grade 3 Class 1	-10°F max

4.2.4 Transition Temperature Shift in Reference Temperature for Nil-Ductility Transition

The ΔRT_{NDT} calculation method of RG 1.99 assumes a nominal irradiation temperature of 550 degrees F, and states that irradiation below 525°F should be considered to produce greater embrittlement. The RPV irradiation temperature is approximately T_{COLD}, which is approximately 496 degrees F at 100% power. Irradiation temperature is not an input to the ΔRT_{NDT} equation by RG 1.99. Because no embrittlement data from NuScale reactors is available to allow referencing actual data, alternative ΔRT_{NDT} methods are also used to account for NuScale's lower irradiation temperature.

The following two RPV embrittlement methodologies use irradiation temperature directly, to estimate ΔRT_{NDT}:

- ASTM E900-15 (Reference 11.6)
- 10 CFR 50.61a

ΔRT_{NDT} was calculated by the methods of RG 1.99, ASTM E900-15, and 10 CFR 50.61a. The method of 10 CFR 50.61a consistently resulted in the highest shift, and was therefore conservatively used to reflect the NuScale T_{COLD}. Details of the ΔRT_{NDT} calculations are included in Appendix A and summarized in Table 4-3 and Table 4-4.

Table 4-3 ¼-T Summary of ΔRT_{NDT} (°F) calculations

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

Table 4-4 ¾-T Summary of ΔRT_{NDT} (°F) calculations

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

4.2.5 Margin

The margin term is calculated per the following RG 1.99 equation:

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

where,

σ_U is the standard deviation for initial RT_{NDT} and is assigned 0 degrees F when the initial RT_{NDT} is measured from actual RPV materials. This is appropriate because the measured initial RT_{NDT} will be available for NuScale RPV materials.

σ_Δ is the standard deviation for ΔRT_{NDT}. Per RG 1.99, σ_Δ is assigned 17 degrees F for base metal and 28 degrees F for weld material, and does not need to exceed one-half of ΔRT_{NDT}.

4.2.6 Adjusted Reference Temperature Calculation

The ART at depths of ¼ T and ¾ T and an exposure of 57 EFPY were calculated and compared to the RG 1.99 acceptance criterion, and are tabulated in Table 4-5 and Table 4-6. The ΔRT_{NDT} values used were the limiting values from Table 4-3 and Table 4-4.

Table 4-5 ¼-T adjusted reference temperature results at 57 effective full power years fluence

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

Table 4-6 ¾ -T adjusted reference temperature results at 57 effective full power years fluence

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

*As mentioned in the discussion of margin, this is the lesser of σ_{Δ} or $\Delta RT_{NDT}/2$.

4.2.7 Comparison of Adjusted Reference Temperature to Acceptance Criterion

As shown in Table 4-5, the predicted ART at a depth of ¼-T and a fluence of 57 EFPY is less than the 200 degrees F maximum allowed by RG 1.99 for new plants.

4.3 Analysis of Cracks

The methods of ASME Code, Section XI, Appendix G for protection against failure of the vessel postulate the existence of a sharp surface crack in the RPV that is normal to the direction of the maximum stress. The crack depth is one-fourth of the RPV wall thickness, and the crack length is 1.5 times the wall thickness as specified in ASME Section XI, paragraph G-2120. Both inside and outside surface cracks in axial and circumferential directions are evaluated.

Figure 4-1 illustrates the postulated crack locations, and are numbered to correspond with Table 4-7. Typical circumferential and axial cracks are shown in Figure 4-2 and Figure 4-3.

For the RPV shell, axial cracks are postulated on both the inside and outside surfaces. For the circumferential weld connecting the RPV shell and the RPV bottom head, circumferential cracks are postulated on both the inside and outside surfaces. For locations with geometric discontinuity such as the RPV wall connection to the RPV flange, core support blocks, and RPV bottom head alignment feature, inside or outside cracks are postulated accordingly. Table 4-7 lists the postulated cracks along with the initial RT_{NDT} and ART values from Table 4-5 and Table 4-6.

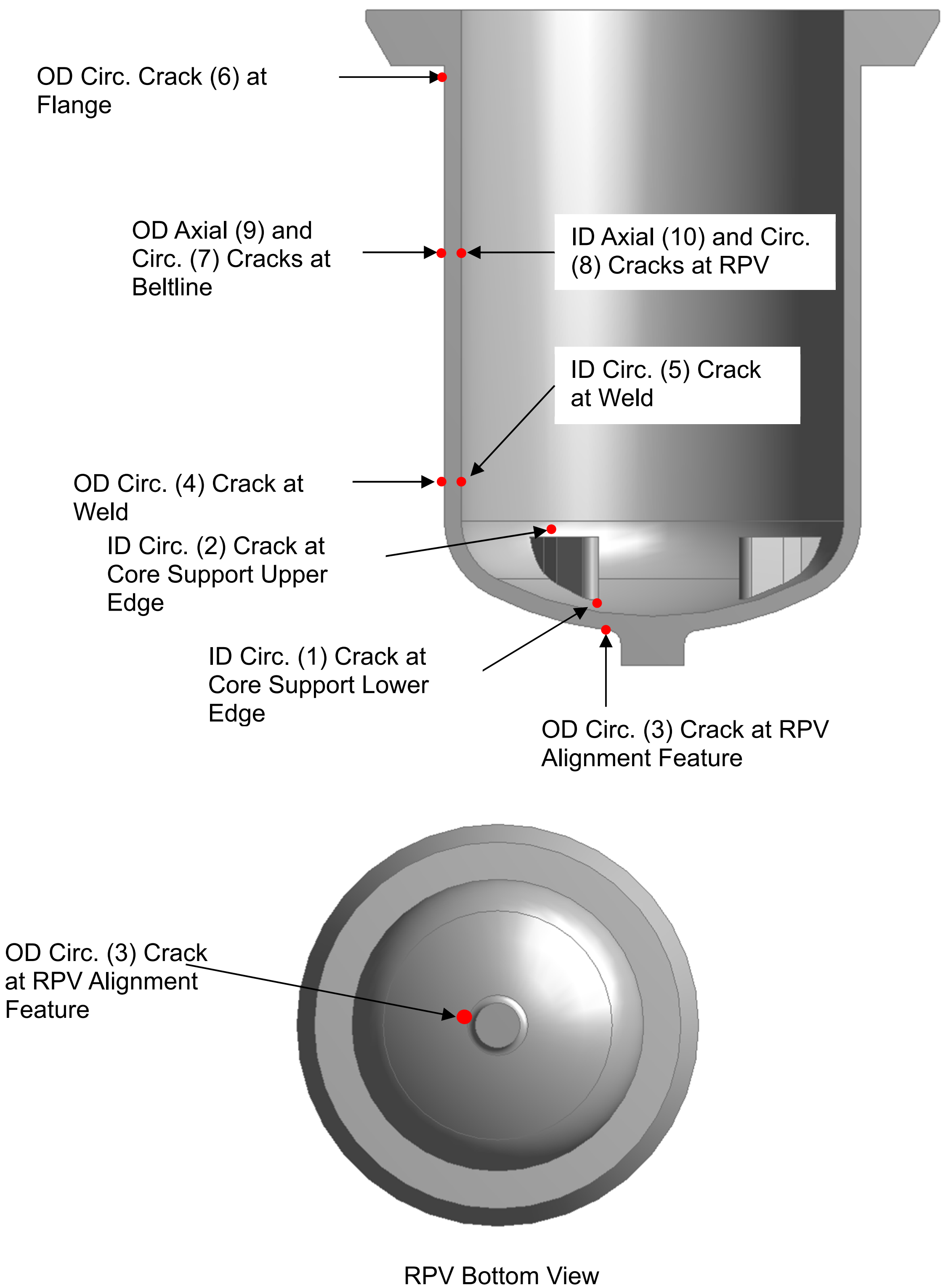

Figure 4-1 Postulated crack locations

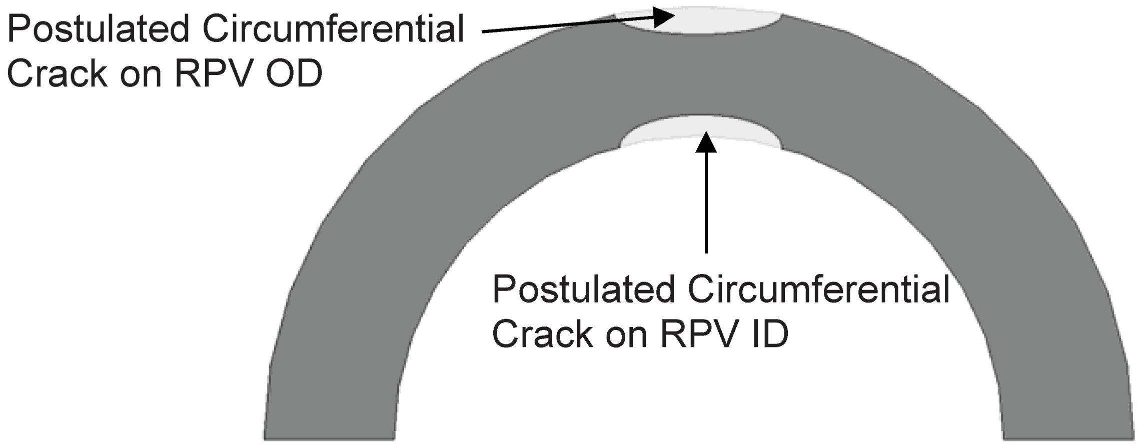

Figure 4-2 Postulated semi-elliptical circumferential cracks in reactor pressure vessel wall

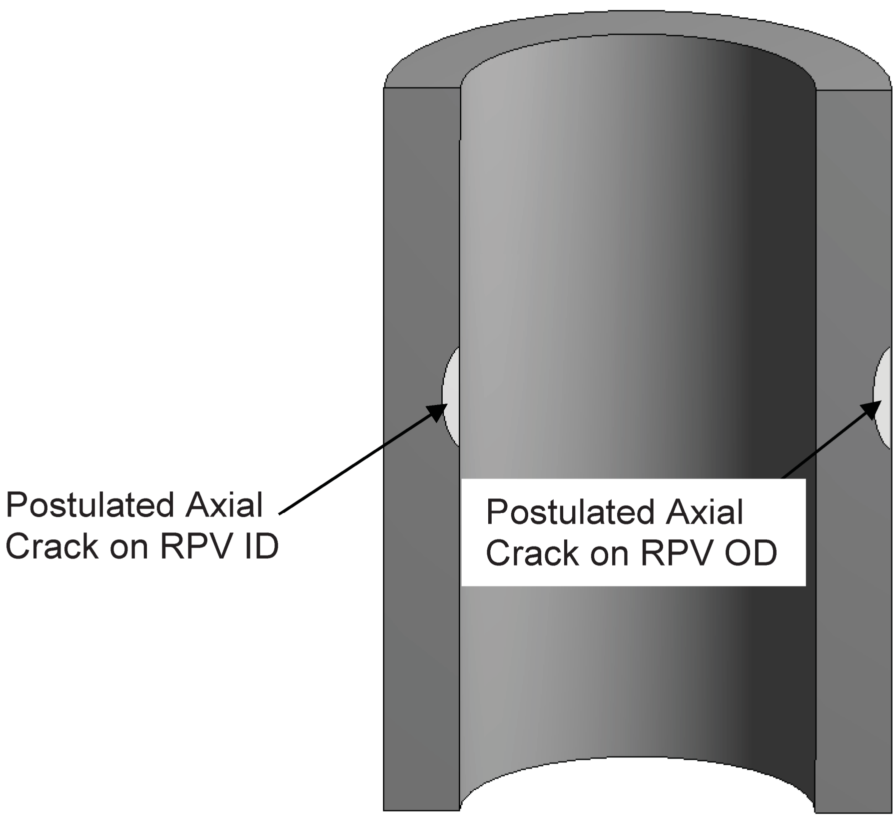

Figure 4-3 Postulated semi-elliptical axial cracks in reactor pressure vessel shell

Table 4-7 Postulated cracks and applicable RT_{NDT}

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

4.4 Stress Analysis

4.4.1 Thermal Stress Analysis

A thermal transient analysis is conducted first for each transient. The temperature field from the thermal transient analysis is then applied to the static structural model for stress analysis. The hoop and axial stresses at the crack locations are curve-fit to 3rd order polynomial functions that are used to calculate thermal stress intensity factors (SIFs) K_{IT}. The format of the polynomial function is:

$$\sigma = c_0 + c_1\left(\frac{x}{a}\right) + c_2\left(\frac{x}{a}\right)^2 + c_3\left(\frac{x}{a}\right)^3 \qquad \text{Eq. 4-4}$$

where c_0, c_1, c_2 and c_3 are coefficients, and

σ = hoop stress or axial stress used to calculate the SIF for postulated axial or circumferential crack,

a = crack depth, and

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

4.4.2 Fracture Toughness

ASME Section XI, Appendix G requires application of the critical SIF K_{IC}, defined by Eq. 4-5, in P-T limit calculations.

$$K_{IC} = 33.2 + 20.734\exp[0.02(T - RT_{NDT})] \qquad \text{Eq. 4-5}$$

where,

K_{IC} = Critical SIF measuring fracture toughness, $ksi \cdot in^{0.5}$,

T = Temperature at crack tip, and

RT_{NDT} = Reference temperature for nil-ductility transition.

Consistent with industry practice, the upper shelf fracture toughness K_{IC} from Eq. 4-5 is conservatively limited to an upper bound value of 200 $ksi \cdot in^{0.5}$. The crack-tip temperature needed for these fracture toughness calculations is obtained from transient thermal analysis as described below.

4.4.3 Calculation of Stress Intensity Factors due to Internal Pressure

ASME Section XI, paragraph G-2214.1 (Reference 11.89) provides a method to calculate K_{im} corresponding to membrane tension for postulated axial and circumferential cracks that can be used for locations away from geometry discontinuity where hoop stress and axial stress can be calculated directly through an influence coefficient M_m (M_{m_axial} for axial cracks, and M_{m_circ} for circumferential cracks).

For postulated axial cracks,

$$K_{Im\_axial} = M_{m\_axial}(pR_i/t) \qquad \text{Eq. 4-6}$$

on inside surface:

$$M_{m\_axial} = \begin{cases} 1.85 & for\ t < 4\ in \\ 0.926\sqrt{t} & for\ 4\ in \leq t \leq 12\ in \\ 3.21 & for\ t > 12\ in \end{cases} \qquad \text{Eq. 4-7}$$

on outside surface:

$$M_{m\_axial} = \begin{cases} 1.77 & for\ t < 4\ in \\ 0.893\sqrt{t} & for\ 4\ in \leq t \leq 12\ in \\ 3.09 & for\ t > 12\ in \end{cases} \qquad \text{Eq. 4-8}$$

and for postulated circumferential cracks on inside or outside surface:

$$K_{Im\_circ} = M_{m\_circ}(pR_i/t) \qquad \text{Eq. 4-9}$$

$$M_{m\_circ} = \begin{cases} 0.89 & for\ t < 4\ in \\ 0.443\sqrt{t} & for\ 4\ in \leq t \leq 12\ in \\ 1.53 & for\ t > 12\ in \end{cases} \qquad \text{Eq. 4-10}$$

where,

p = internal pressure, ksi,

R_i = vessel inner radius, inches, and

t = vessel wall thickness, inches.

For cracks postulated at locations with geometric discontinuity, the above equations are not valid. A finite element analysis crack model is used to calculate the SIFs due to pressure. A unit pressure (1 psi) is applied to the lower RPV inside surface. 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 2 through 5 is the maximum SIF (K_{Im}). The influence coefficient M_m can be calculated for both axial and circumferential cracks, as:

$$M_m = K_{Im} t / R_i \qquad \text{Eq. 4-11}$$

Note that the above method to calculate SIF due to pressure can also capture any contribution from primary bending stresses at locations of geometric discontinuity.

4.4.4 Calculation of Stress Intensity Factor K_{IT} due to Thermal Stress

ASME Section XI, paragraph G-2214.3(b) provides 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 SIFs are calculated by the following equations.

For an inside surface crack during a cooldown transient,

$$K_{IT} = (1.0359c_0 + 0.6322c_1 + 0.4753c_2 + \ 0.3855c_3)\sqrt{\pi a} \qquad \text{Eq. 4-12}$$

for an outside surface crack during a heatup transient,

$$K_{IT} = (1.043c_0 + 0.630c_1 + 0.481 + \ 0.401c_3)\sqrt{\pi a} \qquad \text{Eq. 4-13}$$

where a is the crack depth, and c_0, c_1, c_2 and c_3 are coefficients of the 3rd order polynomial equation for hoop or axial stresses calculated in Section 4.4.1.

A finite element analysis crack model is used to calculate the SIFs due to transient thermal stresses by the superposition principle. To do so, a unit pressure ($1\ \mathrm{psi}$) is applied to the crack top face and crack bottom face in four separate steps:

Step 1: Constant unit pressure, set $c_0 = 1, c_1 = 0, c_2 = 0$ and $c_3 = 0$ in Eq. 4-4. The calculated SIF is $K_{It\_C_0}$.

Step 2: Linear pressure along the crack depth direction, set $c_0 = 0, c_1 = 1, c_2 = 0$ and $c_3 = 0$ in Eq. 4-4. The calculated SIF is $K_{It\_C_1}$.

Step 3: Quadratic pressure along the crack depth direction, set $c_0 = 0, c_1 = 0, c_2 = 1$ and $c_3 = 0$ in Eq. 4-4. The calculated SIF is $K_{It\_C_2}$.

Step 4: Cubic pressure along the crack depth direction, set $c_0 = 0, c_1 = 0, c_2 = 0$ and $c_3 = 1$ in Eq. 4-4. 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 2 through 5 is the maximum SIF. The proposed crack-specific equation to calculate SIFs for any axial or circumferential, inside or outside surface cracks is

$$K_{IT} = c_0 K_{It\_C_0} + c_1 K_{It\_C_1} + c_2 K_{It\_C_2} + c_3 K_{It\_C_3} \qquad \text{Eq. 4-14}$$

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, a zero value is used conservatively in the allowable pressure calculation.

Similarly, the above method to calculate SIF due to thermal transients can also capture any contribution from secondary bending stresses at locations of geometric discontinuity.

4.4.5 Stress Intensity Factors due to Unit Internal Pressure

The SIFs due to unit internal pressure (i.e., membrane tension), denoted by K_{Im}, and the components of the calculation of K_{IT}, denoted by $K_{It\_C_0}$, $K_{It\_C_1}$, $K_{It\_C_2}$ and $K_{It\_C_3}$, are calculated using the crack models for the postulated cracks in Table 4-7. Table 4-8 lists these values, which will be used to calculate thermal SIFs and allowable pressures in the following sections. Also shown in Table 4-8 are the results for Crack #2 with finer mesh, indicating good agreement with the coarser mesh.

Table 4-8 Stress intensity factors due to unit loads, $psi \cdot \sqrt{in}$

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

For the cracks in the RPV shell sufficiently far away from geometric discontinuities, the SIF K_{Im} can also be calculated using the formulation provided in ASME Section XI, paragraph G-2214.1, as shown in Section 4.4.3 . A comparison of the calculated K_{Im} is provided in Table 4-9, showing good agreement between the finite element analysis method as summarized in Table 4-8 and the formulation methods.

Table 4-9 Comparison of K_{Im} by formulation and finite element analysis, $psi \cdot \sqrt{in}$

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4.4.6 Transient Thermal Stresses

The transient thermal stresses are calculated for the heatup and cooldown transients with the appropriate film coefficients on the RPV inside diameter (ID) and OD. The transient temperature results at selected time points of the thermal analysis are then applied to the static structural analysis to calculate the normal stresses in hoop and axial directions. The time points are selected based on the coolant temperatures that are used to construct the P-T limit curves, and also the time points that give maximum thermal gradient during heatup and cooldown transients.

Figure 4-4(a) shows the transient temperature distribution at 38,100 seconds for the heatup transient. Figure 4-4(b) shows the transient temperature distribution at 19,020 seconds for the cooldown transient, about half an hour after CNV flooding is initiated.

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

Figure 4-4 Temperature distribution in the lower reactor pressure vessel

Figure 4-5(a) shows the hoop stress distribution at 38,100 seconds for the heatup transient. Figure 4-5(b) shows the hoop distribution at 19,020 seconds for the cooldown transient.

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

Figure 4-5 Thermal hoop stress distribution in the lower reactor pressure vessel

The axial and hoop stresses are extracted from the stress analysis. The stresses in OD and ID ¼ T are then calculated as a function of the radial distance from appropriate surfaces. The radial distance is normalized so that the value is 0 at the surface, and 1 at the crack's deepest point (the ¼ T location). The stresses are then curve-fit to the third-order polynomial Eq. 4-4.

4.4.7 Stress Intensity Factor K_{IT} due to Thermal Stress

Using the SIFs due to unit loads from Table 4-8, the SIF K_{IT} due to transient thermal stresses are calculated using Eq. 4-14.

5.0 ASME Code Section XI Appendix G Limits

This section documents the ASME Section XI, Appendix G methodology for calculating the RPV allowable pressure for preservice hydrostatic test, normal heatup and cooldown transients, and ISLH conditions. A representative set of P-T calculations is developed.

The ASME code allowable pressure is part of the 10 CFR 50 Appendix G requirements, as summarized in Table 5-1 Except for the preservice hydrostatic test, the only requirement of 10 CFR 50, Appendix G 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:

Eq. 5.1

$$K_{I\,applied} = K_{IC}$$

where K_{IC} is the lower bound crack initiation fracture toughness for the material as represented in Eq. 4-5, and $K_{I\,applied}$ is the SIF due to pressure and thermal gradient loads, at the tip of the ¼ T postulated cracks,

Eq. 5-2

$$K_{I\,applied} = SF \cdot M_m \cdot (pR_i/t) + K_{IT}$$

where SF is the required structural factor applied to the pressure loading and, dependent upon which P-T curve is being evaluated, M_m is the influence coefficient given in Section 4.4.3, and K_{IT} is calculated by the methodology discussed in Section 4.4.4.

The allowable pressure associated with a specified temperature along a P-T limit curve is given by:

Eq. 5-3

$$P = \frac{(K_{IC} - K_{IT})t}{SF \cdot M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{SF \cdot K_{Im}}$$

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

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

  Eq. 5-4

  $$P = \frac{K_{IC} t}{M_m \cdot R_i} = \frac{K_{IC}}{K_{Im}}$$

  The allowable pressure is calculated for the crack with highest M_m that bounds the other cracks. The preservice limiting pressure is based on NUREG-0800, Section 5.3.2 (Reference 11.2).

- For the heatup and cooldown transients, the thermal SIF K_{IT} is calculated at selected time points, and the required structural factor $SF = 2$.

$$P = \frac{(K_{IC} - K_{IT})t}{2M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{2K_{Im}} \quad \text{Eq. 5-5}$$

- For ISLH tests, the SIF K_{IT} from heatup and cooldown transients are conservatively applied for the most limiting crack, and the required structural factor $SF = 1.5$.

$$P = \frac{(K_{IC} - K_{IT})t}{1.5M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{1.5K_{Im}} \quad \text{Eq. 5-6}$$

5.1.1 Preservice Hydrostatic Test

The allowable pressure during the preservice hydrostatic test is calculated using Eq. 5-4. The fracture toughness K_{IC} is calculated for initial RT_{NDT} provided in Table 4-7 and temperature of 70 degrees F. The K_{Im} in Table 4-8 is used for each postulated crack. The limiting allowable pressure is {{ }}2(a),(c),ECI psi for the preservice hydrostatic test (based on fracture toughness only; other requirements may apply to lower the limiting pressure).

Table 5-1 Allowable pressure for preservice hydrostatic test

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

5.1.2 Allowable Pressure for Normal Heatup and Cooldown

The allowable pressure for normal heatup, postulated heatup at 100 degrees F/hr, and cooldown transients is calculated using Eq. 5-5. The fracture toughness K_{IC} is calculated for 57-EFPY RT_{NDT} provided in Table 4-7 and the transient temperature at the postulated crack tip. The bounding curves for heatup and cooldown transients are plotted in Figure 5-1.

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

Figure 5-1 Bounding allowable pressure for heatup and cooldown transients

5.1.3 Allowable Pressure for Inservice Leak and Hydrostatic Tests

The allowable pressure for ISLH tests is calculated using Eq. 5-6. The bounding curves for heatup and cooldown transients in Figure 5-1 are scaled by the ratio of the safety factor for the heatup and cooldown conditions to that for the ISLH conditions. The allowable pressure for ISLH tests is shown in Figure 5-2. The transient ISLH curve is the lower bounding of the ISLH heatup and cooldown curves.

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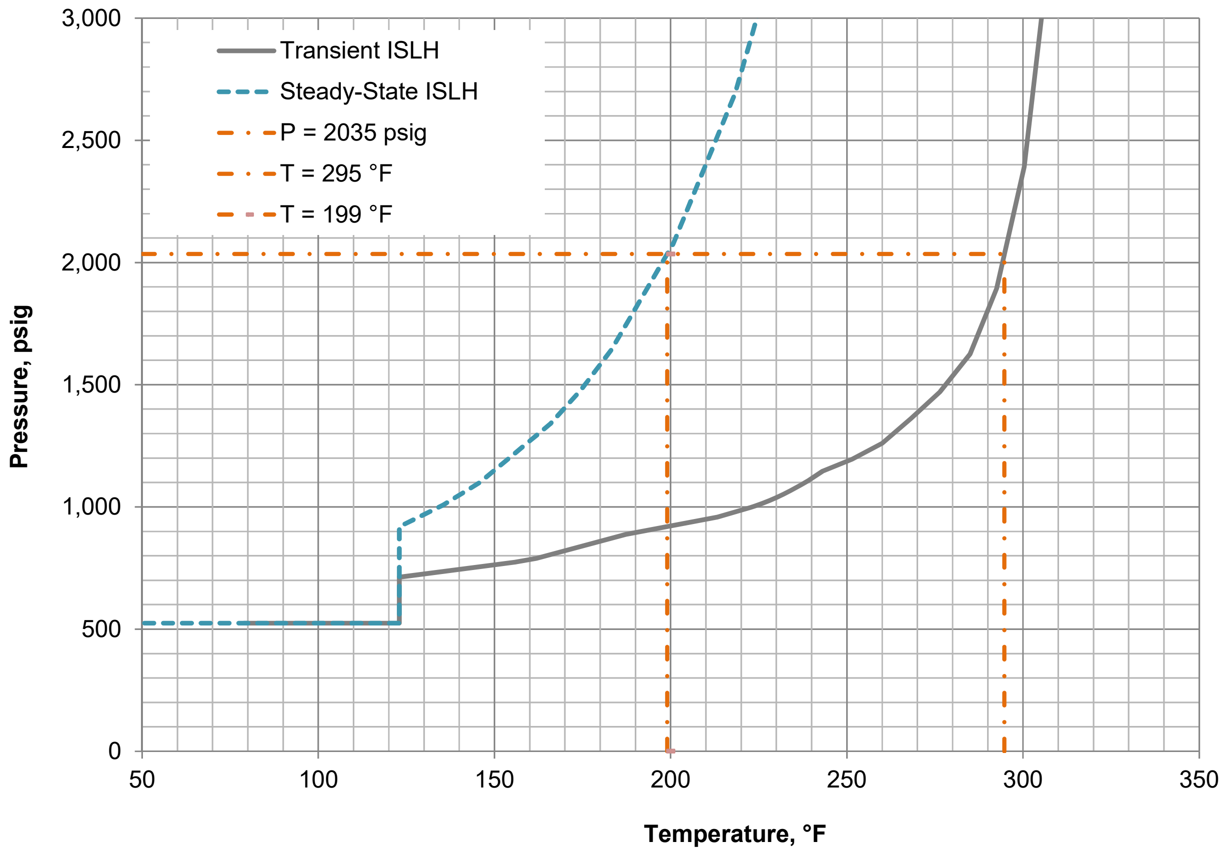

Figure 5-2 Allowable pressure for inservice leak and hydrostatic test

6.0 10 CFR 50 Appendix G Pressure and Temperature Limits

Appendix G to 10 CFR Part 50 requires that the P-T limits be at least as conservative as limits obtained by following the ASME Section XI, Appendix G methods presented in Section 5.0. In addition, Appendix G to 10 CFR Part 50 specifies P-T limits and minimum temperatures for operation of a reactor vessel, dependent upon pressure, criticality, and the presence or absence of fuel. The requirements applicable to the NuScale design are presented in Table 6-1.

Table 6-1 Pressure and temperature requirements for the reactor pressure vessel

Operating Condition	Vessel Pressure (1)	P-T Limits	Minimum Temperature Requirements
Hydrostatic Pressure and Leak Tests (Core is not critical)			
Fuel in the vessel	≤ 20%	ASME Appendix G Limits	(2)
Fuel in the vessel	> 20%	ASME Appendix G Limits	(2) + 90°F
No fuel in the vessel (Preservice Hydrotest)	ALL	Not Applicable	(3) + 60°F
Normal Heatup and Cooldown, including CNV Flooding during Cooldown			
Core not critical	≤ 20%	ASME Appendix G Limits	(2)
Core not critical	> 20%	ASME Appendix G Limits	(2) + 120°F(5)
Core critical	≤ 20%	ASME Appendix G Limits + 40°F	The maximum of ((4)) or ((2)+40°F(5))
Core critical	> 20%	ASME Appendix G Limits + 40°F	The maximum of ((4)) or ((2)+160°F(5))

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 leak and hydrostatic 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.

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

The application of Table 1 of Appendix G to 10 CFR, Part 50 to NuScale RPV is presented in Table 6-2. The uncorrected P-T limit curves are presented in Section 9.0.

Table 6-2 Pressure-temperature limits for NuScale reactor pressure vessel per 10 CFR 50, Appendix G

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

7.0 Low Temperature Overpressure Protection Limits

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Table 7-1 Variable LTOP pressure setpoint as a function of RCS cold temperature

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In summary, the LTOP pressure setpoint is a function of wide range RCS cold temperature as defined in Table 7-1. Linear interpolation is used for temperatures between table entries. If the wide range RCS pressure measurement is larger than the interpolated LTOP pressure setpoint, the module protection system opens the RVVs. The LTOP enable temperature is 325 degrees F.

8.0 Surveillance Program

Each NuScale reactor module contains specimens of the material used in the construction of the lower RPV shell. These specimens are located inside the vessel at the vessel beltline, and are therefore exposed to a higher neutron flux and approximately the same temperature as is the lower RPV shell.

Specimens are periodically removed and tested in order to monitor changes in fracture toughness in accordance with ASTM E185-82 (Reference 11.7) as required by 10 CFR 50, Appendix H or in accordance with NUREG-1801 (Reference 11.3). The estimated withdrawal schedule for the surveillance capsules is shown in Table 8-1.

The specimen lead factor is defined as the ratio of average fluence of specimens inside the capsule to the RPV inside surface peak fluence. The specimen lead factor is 4.3, which exceeds the 1 to 3 range recommended by ASTM E185-82. A lead factor of 4.3 is below the maximum 5 allowed by ASTM E185 revisions subsequent to E185-82, hence is within the current industry consensus of acceptable lead factor range.

Table 8-1 Estimated surveillance capsule withdrawal schedule

Sequence	ASTM E185-82[(a)] or NUREG-1801 Rev. 2 [(b)]	Estimated Withdrawal
1st [(a)]	Whichever comes first • 6-EFPY • Capsule fluence > 5E+18 n/cm^2, E > 1 MeV • Highest predicted ΔRT_{NDT} > ~50°F of all encapsulated materials	3.5 EFPY for capsule fluence to reach 5E+18 n/cm^2, E > 1 MeV. • From Table 4-1, 57-EFPY peak RPV inside surface fluence is 1.91E+19 n/cm^2, E > 1 MeV • (5E+18/1.91E+19)*(57 EFPY/4.3) = 3.5 EFPY
2nd [(a)]	Whichever comes first • 15 EFPY • Capsule fluence > peak 32-EFPY RPV inside surface fluence	7.4 EFPY for capsule fluence to reach peak 32-EFPY RPV inside surface fluence • 32 EFPY/4.3 = 7.4 EFPY
3rd [(a)]	Capsule fluence is between 1 and 2 times of peak 32-EFPY inside surface fluence	Between 7.4 EFPY and 14.9 EFPY • 32 EFPY/4.3 = 7.4 EFPY • 64 EFPY/4.3 = 14.9 EFPY
4th [(b)]	Capsule fluence is between 1 and 2 times of peak 57-EFPY inside surface fluence	Between 13.3 EFPY and 26.5 EFPY • 57 EFPY/4.3 = 13.3 EFPY • 114 EFPY/4.3 = 26.5 EFPY

[(a)] The withdrawal schedule for the first three capsules is in accordance with ASTM E185-82 for the initial 40-year operating license period.

[(b)] The 4th capsule is used to cover the last 20 years of a 60-year design life, and its withdrawal schedule is based on NUREG-1801 Rev. 2 for license renewal period.

9.0 Pressure-Temperature Curves

Using the methodology provided in ASME Code, Section XI, Appendix G, and the requirements in 10 CFR 50, Appendix G, a representative set of P-T limits at 57-EFPY fluence was developed for various conditions. The P-T limits for normal heatup (core critical and core not critical), normal cooldown and ISLH tests are provided in Figure 9-1, Figure 9-2, and Figure 9-3, respectively. The corresponding numerical values are listed in Table 9-1 and Table 9-2. These P-T curves meet the pressure and temperature requirements listed in Table 1 of 10 CFR 50, Appendix G.

The RCS pressure should be maintained below the limit of the P-T limit curves to ensure protection against brittle failure. Acceptable pressure and temperature combinations for RCPB operation are below and to the right of the applicable P-T curves. These P-T curves do not include location correction or instrument uncertainty. For the purpose of location correction, the allowable pressure in the P-T curves can be taken as the pressure at the RPV bottom. The reactor is not permitted to be critical until the P-T combinations are to the right of the criticality curve shown in Figure 9-1.

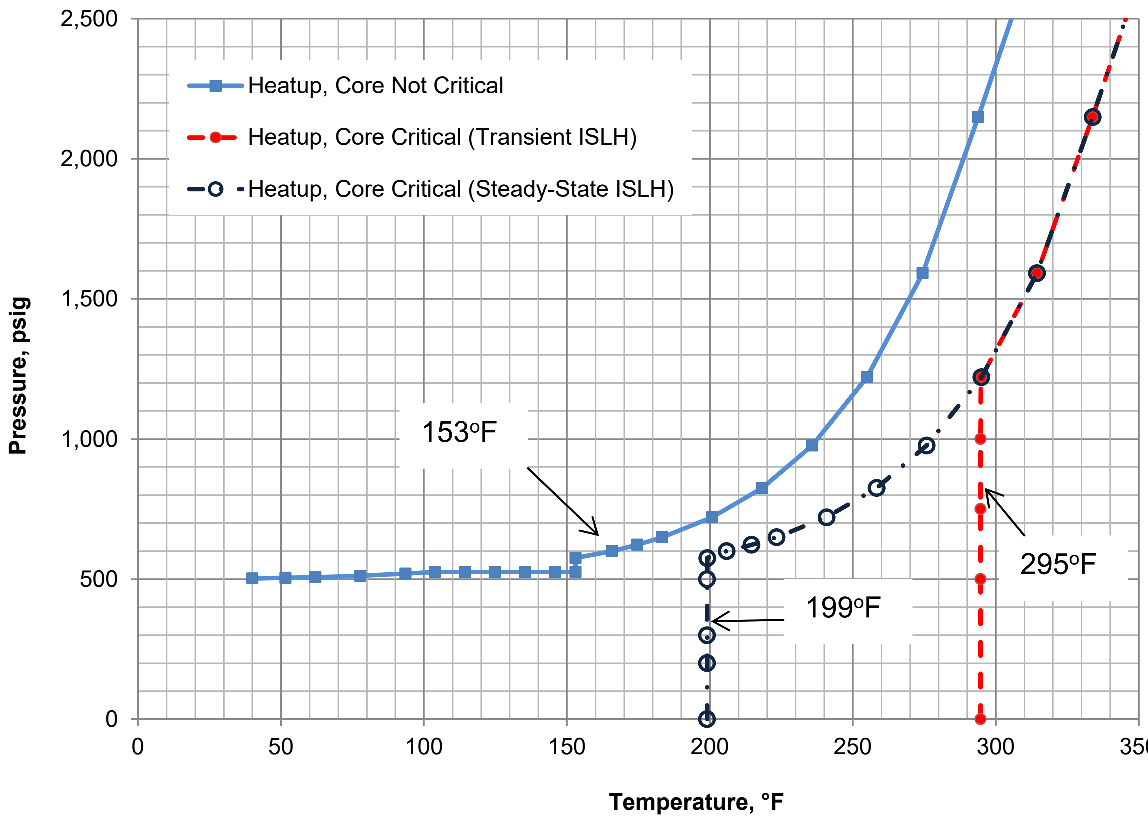

Figure 9-1 Pressure-temperature limits for normal heatup and criticality limit

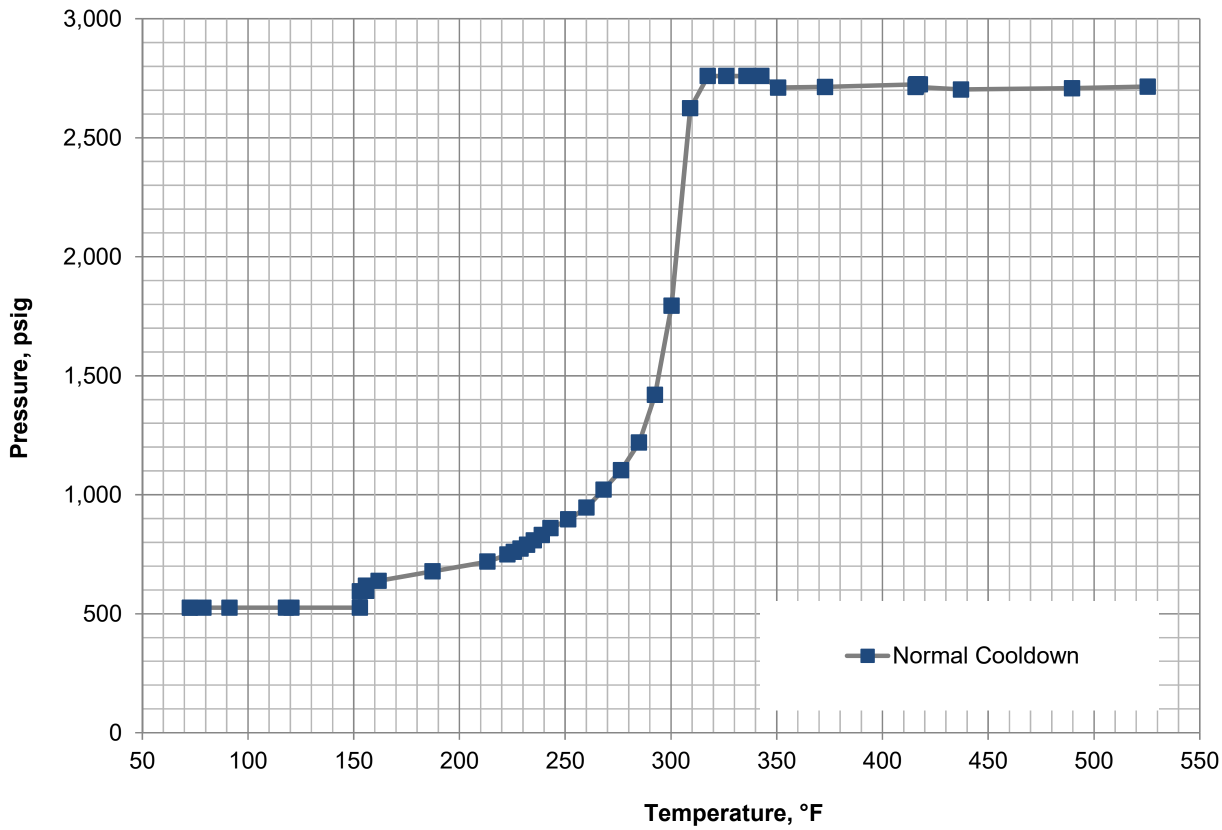

Figure 9-2 Pressure-temperature limits for normal cooldown with decay heat removal system and containment vessel flooding

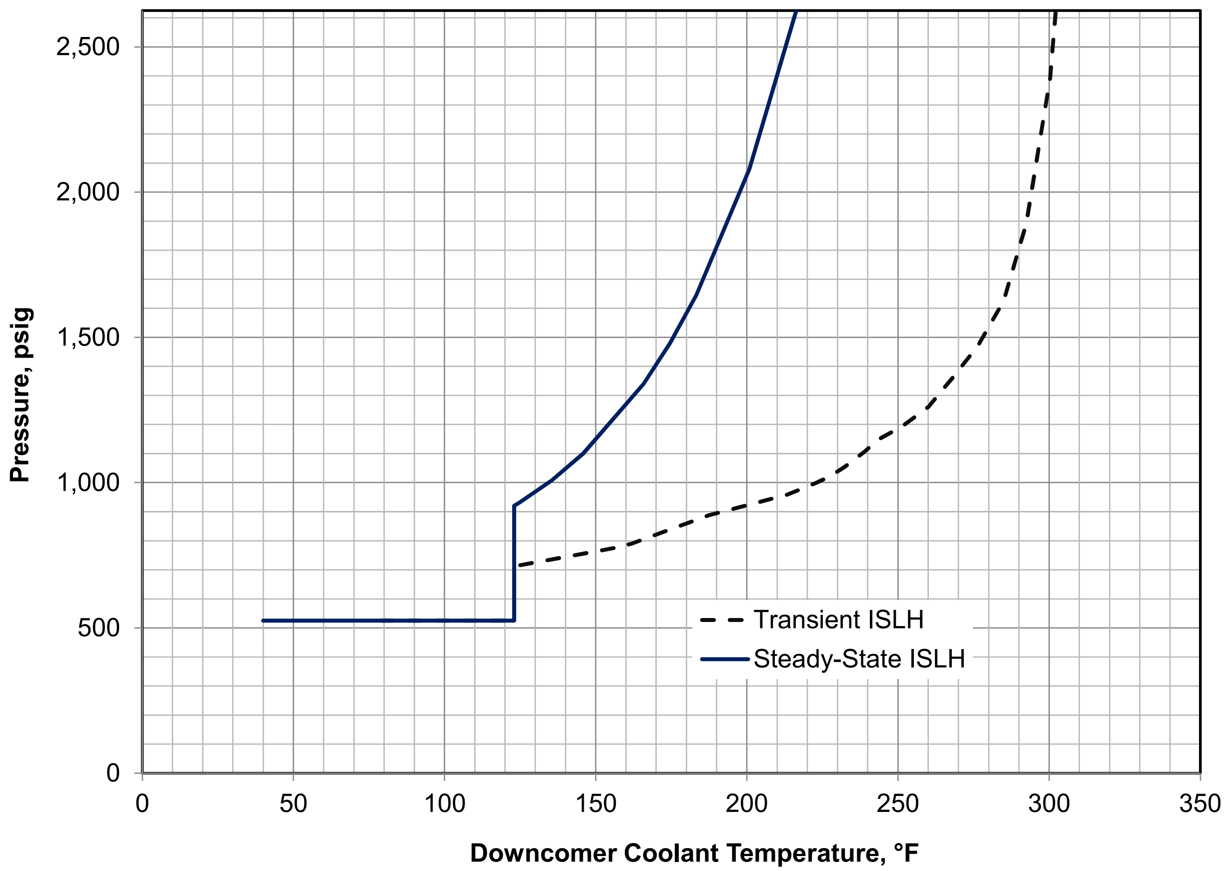

Figure 9-3 Pressure-temperature limits for transient inservice leak and hydrostatic (composite of heatup and cooldown) and steady-state inservice leak and hydrostatic test

Table 9-1 Pressure-temperature limits for normal heatup and cooldown

Normal Heat up (Core Not Critical)		Normal Heat up (Core Critical) (Minimum core critical temperature determined from the transient ISLH curve)		Normal Heat up (Core Critical) (Minimum core critical temperature determined from the Steady-State ISLH curve)		Normal Cooldown	
Fluid Temp. °F	Press. psig	Fluid Temp. °F	Press. psig	Fluid Temp. °F	Press. psig	Fluid Temp. °F	Press. psig
40	502	(Reactor is not permitted to be critical below 295°F if ISLH testing is performed at Heat up/Cooldown transient conditions)		(Reactor is not permitted to be critical below 199°F if ISLH testing is performed at steady-state conditions.)		309	2623
52	504					300	1794
62	507					292	1420
78	511					285	1220
94	519					276	1103
104	525					268	1022
114	525			199	0	260	945
125	525			199	200	252	896
135	525			199	300	243	859
146	525			199	500	239	830
153	525			199	575	235	808
153	575			206	600	232	789
166	600			215	622	229	774
175	622	295	0	223	649	226	760
183	649	295	500	241	721	223	748
201	721	295	750	258	825	213	719
218	825	295	1000	276	977	187	678
236	977	295	1217	295	1221	162	638
255	1221	295	1221	314	1592	156	617
274	1592	314	1592	334	2149	156	596
294	2149	334	2149	354	2752	153	594
314	2752	354	2752	374	2750	153	525
334	2750	374	2750	393	2748	121	525
353	2748	393	2748	413	2746	118	525
373	2746	413	2746	426	2744	91	525
386	2744	426	2744	440	2742	79	525
400	2742	440	2742	454	2739	73	525
414	2739	454	2739	462	2735	73	525

Table 9-2 Pressure-temperature limits for inservice leak and hydrostatic test

ISLH for Heat up Transient		**ISLH for Cooldown Transient**		**Transient ISLH (Bounding of Heat up and Cooldown)**		**Steady-State ISLH**	
Fluid Temp. °F	Press. psig	Fluid Temp. °F	Pressure psig	Fluid Temp. °F	Pressure psig	Fluid Temp. °F	Pressure psig
79	525	300	2392	79	525	40	525
91	525	292	1893	91	525	52	525
118	525	285	1626	118	525	62	525
121	525	276	1471	121	525	78	525
123	525	268	1362	123	525	94	525
123	712	260	1260	123	712	104	525
156	775	252	1195	156	775	114	525
156	775	243	1146	156	775	123	525
162	790	239	1107	162	790	123	920
187	887	235	1077	187	887	125	933
213	1060	232	1053	213	958	135	1008
223	1150	229	1032	223	998	146	1100
226	1187	226	1013	226	1013	166	1340
229	1223	223	998	229	1032	175	1479
232	1260	213	958	232	1053	183	1645
235	1296	187	904	235	1077	201	2079
239	1356	162	851	239	1107	218	2693
243	1427	156	823	243	1146		
252	1569	156	795	252	1195		
260	1757	123	765	260	1260		
268	1964	123	525	268	1362		
276	2200	121	525	276	1471		
285	2527	118	525	285	1626		
292	2813	91	525	292	1893		
300	3127	79	525	300	2392		

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

10.0 Summary and Conclusions

A methodology based on 10 CFR 50, Appendix G and ASME Section XI, Appendix G has been presented for the calculation of P-T limits (P-T curves) applicable to the NuScale RCPB, and a set of P-T curves applicable to the NuScale standard plant was developed using these methods. These limits account for the effects of neutron-induced embrittlement up to an exposure of 57 EFPY. Curves developed include:

- normal heatup
  - core critical
  - core not critical
- normal cooldown
- in-service leak and hydro tests

In addition, the pressure was established for the preservice hydrostatic leak test.

Calculations were performed to predict ART of reactor vessel at 57-EFPY fluence. These calculations, based on the methodology of RG 1.99, were modified to account for the lower temperature at which a NuScale reactor vessel may operate. The results indicate that the NuScale standard reactor vessel ¼-T ART has substantial margin to the RG 1.99 acceptance criteria of 200 degrees F max for new plants at the end of design life.

The LTOP limits were developed for the NuScale standard plant, and a description of the reactor vessel material surveillance program was provided.

The licensee will develop a reactor vessel embrittlement surveillance program for each NuScale reactor vessel, and may use the methods described in this report provided as-built material properties are used in the analyses.

11.0 References

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

11.2 U.S. Nuclear Regulatory Commission, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock", NUREG-0800, Section 5.3.2, Revision 2, March 2007.

11.3 U.S. Nuclear Regulatory Commission, "Generic Aging Lessons Learned (GALL) Report", NUREG-1801, Revision 2, December 2010.

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

11.5 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.

11.6 ASTM International, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials", ASTM E900-15, West Conshohocken, PA.

11.7 ASTM International, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels", ASTM E185-82, West Conshohocken, PA.

11.8 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section III, Rules for Construction of Nuclear Facility Components, New York, NY.

11.9 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, New York, NY.

11.10 RSICC CODE PACKAGE CCC-810, Monte Carlo N–Particle Transport Code System Including MCNP6.1, MCNP5-1.60, MCNPX-2.7.0 and Data Libraries, Los Alamos National Laboratory.

Appendix A. Calculation of ΔRT_{NDT}

Three methods of calculating $\Delta \mathrm{RT}_{\mathrm{NDT}}$ are shown in the following sections.

A.1 RG 1.99 Rev 2 Methodology

From RG 1.99, the mean value of the adjustment in reference temperature due to irradiation is calculated as:

$$\Delta \mathrm{RT}_{\mathrm{NDT}} = \mathrm{CF} * \mathrm{FF}$$

The chemistry factor CF is selected from RG 1.99, Tables 1 and 2, using the Cu and Ni contents in Table 3-3.

The fluence factor FF is calculated per the following RG 1.99 equation:

$$\mathrm{FF} = \mathrm{f}^{(0.28-0.10*\mathrm{logf})}$$

where,

“f” is the attenuated fluence at ¼-T and ¾-T , and is in units of 1E+19 n/cm^2, E > 1 MeV.

Table A-1 Calculation of ΔRT_{NDT} per RG 1.99

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

A.2 ASTM E900-15 Methodology

The following equations are used by ASTM E900-15 to calculate ΔRT_{NDT}:

$$TTS = TTS1 + TTS2$$

TTS is defined as transition temperature shift at 30 ft-lb Charpy energy level, which is identical to the ΔRT_{NDT} definition in RG 1.99.

$$TTS1 = A\left(\frac{5}{9}\right)(1.8943 \times 10^{-12})(\emptyset^{0.5695})\left(\frac{1.8 \cdot T + 32}{550}\right)^{-5.47}\left(0.09 + \frac{P}{0.012}\right)^{0.216}\left(1.66 + \frac{Ni^{8.54}}{0.63}\right)^{0.39}\left(\frac{Mn}{1.36}\right)^{0.3}$$

$$A = \begin{pmatrix} 1.011\ for\ forgings \\ 0.919\ for\ welds \end{pmatrix}$$

$$TTS2 = \left(\frac{5}{9}\right) \cdot max[min(Cu, 0.28) - 0.053, 0] \cdot M$$

$$M = B \cdot max\left\{min\left[113.87\left(ln(\emptyset) - ln(4.5 \times 10^{20})\right), 612.6\right], 0\right\} \cdot \left(\frac{1.8 \cdot T + 32}{550}\right)^{-5.45}\left(0.1 + \frac{P}{0.012}\right)^{-0.098}\left(0.168 + \frac{Ni^{0.58}}{0.63}\right)^{0.73}$$

$$B = \begin{pmatrix} 0.738\ for\ forgings \\ 0.968\ for\ welds \end{pmatrix}$$

In the above ASTM E900-15 equations:

- TTS (ΔRT_{NDT}), TTS1, TTS2, and T are in degrees C. T is the irradiation temperature, which is approximately the T_{COLD}.
- Cu, Ni, P, and Mn are in wt percent
- Φ is neutron fluence in n/m^2, E > 1 MeV.

The ΔRT_{NDT} calculated per ASTM E900-15 is summarized in Table A-2.

Table A-2 Calculation of ΔRT_{NDT} per ASTM E900-15

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

A.3 10 CFR 50.61a Methodology

The following equations are used by 10 CFR 50.61a to calculate ΔRT_{NDT}:

$$\Delta T_{30} = MD + CRP$$

- ΔT_{30} is defined as transition temperature shift at 30 ft-lb Charpy energy level, which is identical to the ΔRT_{NDT} definition in RG 1.99.
- MD and CRP terms refer to embrittlement attributed to matrix damage and copper rich precipitate respectively, and are defined by the equations below.

$$MD = A \times (1 - 0.001718 \times T_C)(1 + 6.13 \times P \times Mn^{2.471})(\varphi t_e)^{0.5}$$

$$CRP = B \times (1 + 3.77 \times Ni^{1.191}) \times f(Cu_e, P) \times g(Cu_e, Ni, \varphi t_e)$$

$$f(Cu_e, P) = 0, for\ Cu\ \leq 0.072\%$$

- Because Cu is restricted to 0.06 percent max by NuScale the $f(Cu_e, P)$ term is zero, causing the CRP term to be zero. Therefore, only MD inputs are listed below.

$$A = \begin{pmatrix} 1.140 \times 10^{-7} for\ forgings \\ 1.417 \times 10^{-7}\ for\ welds \end{pmatrix}$$

$$\varphi t_e = \begin{pmatrix} \varphi t\ for\ \varphi\ \geq 4.39 \times 10^{10} \dfrac{n}{cm^2 \cdot sec} \\ \varphi t \times \left(\dfrac{4.39 \times 10^{10}}{\varphi}\right)^{0.2595}\ for\ \varphi\ < 4.39 \times 10^{10} \dfrac{n}{cm^2 \cdot sec} \end{pmatrix}$$

In the above 10 CFR 50.61a equations:

- ΔT_{30}, MD, and CRP are in degrees F
- T_C is cold leg temperature in degrees F
- Cu, Ni, P, and Mn are in wt percent
- t is time in seconds
- φ is flux in n/cm^2/sec, E > 1 MeV
- φt and φt_e are fluence in n/cm^2, E > 1 MeV

Neutron flux is calculated by the following equation:

$$Flux = \frac{57\ EFPY\ Fluence}{57 * 365.25 * 24 * 3600\ sec} = \frac{32\ EFPY\ Fluence}{32 * 365.25 * 24 * 3600\ sec}$$

The ΔRT_{NDT} calculated per 10 CFR 50.61a is summarized in Table A-3.

Table A-3 Calculation of ΔRT_{NDT} per 10 CFR 50.61a

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

A.4 NuScale Methodology

The ΔRT_{NDT} calculated by the RG 1.99, ASTM E900-15, and 10 CFR 50.61a methods are now compared. Of the three methods, 10 CFR 50.61a consistently predicts the highest shift for all locations of interest. Compared to RG 1.99, ASTM E900-15 predicts a higher shift for the base metal, but lower shift for the weld metal.

The ΔRT_{NDT} methodology for NuScale RPV is to use the highest predicted shift from the three methods to bound the RG 1.99 prediction. This adjustment is considered appropriate to account for the lower irradiation temperature for the NuScale RPV based on the following considerations:

- The RG 1.99 method requires a correction factor for temperatures below 525 degrees F justified by reference to actual data. However, currently there is no embrittlement data from NuScale reactors to allow correction based on actual data. However, licensees of NuScale reactors are required by 10 CFR 50, Appendix H to maintain a RPV surveillance program for each RPV. The predicted ΔRT_{NDT} values for the NuScale RPV will be verified by surveillance specimens ahead of RPV reaching the same embrittlement level. Systemic bias in the ΔRT_{NDT} methodology will be detected and corrected such as per RG 1.99 Regulatory Position 2.1 before it becomes a safety concern.
- The ASTM E900-15 and 10 CFR 50.61a embrittlement models have incorporated light water RPV surveillance data not available when RG 1.99 was issued. The 10 CFR 50.61a embrittlement model was based on U.S. RPV surveillance data through approximately 2002 while the recent ASTM E900-15 included RPV surveillance data from 13 countries with western-designed light water reactors. These two are the best known peer-reviewed RPV embrittlement models since RG 1.99 was issued.
- The embrittlement models used by ASTM E900-15 and 10 CFR 50.61a allow direct use of irradiation temperature to estimate the shift. In addition to Cu and Ni content, these two methods also take into account the embrittlement effect due to P and Mn content that are not considered by RG 1.99.

The ΔRT_{NDT} predicted by 10 CFR 50.61a and adopted for NuScale is higher than that predicted by RG 1.99 for all locations of interest, reflecting the effects of irradiation at temperatures below the typical range assumed by RG 1.99. The adopted ΔRT_{NDT} for NuScale is also higher (more conservative) than the ASTM E900-15 prediction using identical inputs.