U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION REVIEW OF NUSCALE POWER, LLC

GENERIC PRESSURE-TEMPERATURE LIMITS REPORT

1.0 INTRODUCTION

By letter dated December 30, 2016, NuScale Power, LLC. (NuScale or the applicant) submitted Technical Report (TR)-1015-18177, Revision 0, "Pressure and Temperature Limits Methodology," in support of the NuScale design certification application. The purpose of this report is to provide the generic basis for the use of the pressure-temperature (P-T) limit curves found in NuScale Design Control Document (DCD) Section 5.3, "Reactor Vessel." As such, this pressure and temperature limits report (PTLR) contains an evaluation of the entire reactor vessel (RV), including beltline, closure flange, and nozzle regions. The information in this report is generic to the NuScale design and is expected to apply to all combined license (COL) applicants referencing the NuScale design certification.

The first part of the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's review was to ensure that the information in the proposed PTLR and the revised technical specification (TS) pages is in accordance with the guidance in Generic Letter (GL) 96-03, "Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996. The second part of the staff's review was to verify that the proposed P-T limits have been developed appropriately using the methodology in TR-1015-18177, Rev. 2. The report was revised and submitted on October 25, 2018, to the NRC as TR-1015-18177-P, Revision 2 (non-proprietary version is accessible at Agencywide Documents Access and Management System (ADAMS) Accession No. ML18298A304), hereafter referred to as the NuScale PTLR.

2.0 REGULATORY EVALUATION

2.1 10 CFR Part 50 Requirements for Generating Pressure-Temperature Limits and Low Temperature Overpressure Protection System Limits for Pressurized-Water Reactors

The NRC has established requirements in Appendix G, "Fracture Toughness Requirements," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Appendix G to 10 CFR Part 50 requires the P-T limits for an operating light-water nuclear reactor to be at least as conservative as those that would be generated using the methods of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). For conditions with the core critical, P-T limits must be more conservative than the limits in ASME Code Section XI, Appendix G. Table 1 of 10 CFR Part 50, Appendix G summarizes the requirements for P-T limits relative to the criteria in ASME Code Section XI, Appendix G, as well as the minimum temperature requirements for bolting up the RV during normal and pressure testing operations. Appendix G to 10 CFR Part 50 requires that applicable surveillance data from RV material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the RCPB. Appendix G to 10 CFR Part 50 also establishes conservative requirements for determining the temperature and pressure setpoints for low temperature overpressure protection (LTOP) systems. P-T limits and LTOP system limits are subject to General Design Criterion (GDC) 14, “Reactor Coolant Pressure Boundary,” GDC 15, “Reactor Coolant System Design,” GDC 30, “Quality of Reactor Coolant Pressure Boundary,” and GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50.

Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50 provides the NRC’s criteria for the design and implementation of RV material surveillance programs for operating light-water reactors. The requirements for protecting the RVs of pressurized-water reactors against pressurized thermal shock (PTS) events appear in 10 CFR 50.61, “Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events.” Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” issued May 1988, contains staff regulatory guidance related to determining the effects of radiation embrittlement on RV material parameters and P-T limit curves. Staff guidance related to the review of P-T limit curves and PTS criteria appears in Section 5.3.2, “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized, Thermal Shock,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” while NUREG-0800, Section 5.2.2, “Overpressure Protection,” contains staff guidance related to the review of LTOP system limits.

GDC 14, 30, and 31 specify the regulatory requirements for RV neutron fluence calculations. In March 2001, the staff issued RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.” The staff has approved RV neutron fluence calculation methods that satisfy the requirements of GDC 14, 30, and 31 by adhering to the guidance in RG 1.190. Neutron fluence calculations are acceptable if they are done with approved methods or with methods that are shown to conform to the guidance in RG 1.190.

The most recent version of ASME Code Section XI, Appendix G that the NRC has endorsed in 10 CFR 50.55a, “Code and Standards,” and therefore by reference in Appendix G to 10 CFR Part 50, is the 2013 Edition of the ASME Code. Additionally, 10 CFR Part 50, Appendix G imposes minimum head flange temperatures when the system pressure is at or above 20 percent of the preservice hydrostatic test pressure.

GL 96-03 addresses the technical information for a licensee to implement a PTLR. It establishes the information that should be included in (1) an acceptable PTLR methodology (with the P-T limit methodology as its subset) and (2) the PTLR itself. Technical Specification Task Force-419, “Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR,” dated September 16, 2001 (ADAMS Accession No. ML012690234), provides additional guidance that gives an alternative format for documenting the implementation of a PTLR in the “Administrative Controls” section of a facility’s TS.

2.2 Technical Specification Requirements for Pressure-Temperature Limits and Low Temperature Overpressure Protection System Limits

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include TS as part of the operating license. The Commission sets forth its regulatory requirements related to the content of the TS in 10 CFR 50.36, "Technical Specifications." That regulation requires that the TS include items in five categories: (1) safety limits, limiting safety system settings and limiting control settings, (2) limiting conditions of operation (LCOs), (3) surveillance requirements, (4) design features, and (5) administrative controls.

In 10 CFR 50.36(c)(2)(ii), the NRC requires that LCOs be established for the P-T limits and LTOP system limits because the parameters fall within the scope of Criterion 2 identified in the rule:

> Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The P-T limits and LTOP system limits are therefore ordinarily required to be included within the TS LCOs for a plant-specific facility operating license.

On January 31, 1996, the staff issued GL 96-03, which informed licensees that they may request a license amendment to relocate the actual P-T limit curves and LTOP system limit values from the TS LCOs on P-T limits and LTOP system limits and into a PTLR or other licensee-controlled document that would be administratively controlled through the administrative controls section of the TS. COL applicants referencing previously certified standard designs may include the design-limiting P-T limits, LTOP system limits, and related input parameters in a PTLR that is generic to the certified design. GL 96-03 indicated that licensees or applicants seeking to locate P-T limits and LTOP system limits for their reactors in PTLRs would need to generate their limits in accordance with an NRC-approved methodology, and that the method used to generate the limits would need to comply with the requirements of Appendices G and H to 10 CFR Part 50. Furthermore, the method used to generate the P-T limits and LTOP system limits would need to be incorporated by reference in the administrative controls section of the TS. The GL also specified that the TS administrative controls section for the PTLR should reference the staff's safety evaluation issued on the PTLR methodology and that the PTLR be defined in Section 1.0 of the TS. Attachment 1 to GL 96-03 listed the criteria that the approved PTLR methodology and PTLR application should meet.

3.0 TECHNICAL EVALUATION

3.1 NuScale Generic Technical Specification Requirements for Implementation and Control of a Pressure and Temperature Limits Report

The staff reviewed the NuScale generic TS to ensure that they contain all of the necessary provisions required for the implementation and control of a PTLR. Chapter 16 of the NuScale DCD references the NuScale generic TS (ADAMS Accession No. ML18086A198). The relevant generic TS requirements include the TS definition of the PTLR (TS Section 1.1); the TS LCOs for the reactor coolant system (RCS) P-T limits (LCO 3.4.3) and the LTOP system (LCO 3.4.11), including LCO action statements, surveillance requirements, and related applicability criteria; and the necessary administrative controls governing the PTLR content and reporting requirements (TS 5.6.4). The staff found all of the TS pages related to the implementation and control of a PTLR acceptable, when referencing an approved PTLR that is generic to the NuScale design.

3.2 Evaluation of the NuScale Generic PTLR Contents and Methodology against the Seven Criteria for PTLR Contents in Attachment 1 of GL 96-03

The NuScale PTLR provides the generic P-T limits and LTOP system limits for the NuScale RV and the methodology for their development. This report is generic for the NuScale design and is specifically referenced in Section 5.6.4 of the NuScale generic technical specifications as the controlling document governing future changes to PTLRs for NuScale plants. Accordingly, the PLTR uses generic inputs for RV beltline material chemistry, initial nil-ductility reference temperature (RT_{NDT}) values, and projected neutron fluence to determine the P-T limit curves. These generic inputs are intended to be bounding for the NuScale design; they represent the maximum allowable limits on the input parameters for any specific NuScale plant. Therefore, per DCD COL Item 5.3-2 any COL holder referencing the NuScale design will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy has been demonstrated. These procedures are to be based on the material properties of the as-built RV. This will ensure that the operating procedures are adequately bounded by the approved methodology.

Attachment 1 of GL 96-03 contains seven technical criteria to which the contents of PTLRs should conform if P-T limits and LTOP system limits are to be located in a PTLR as described in GL 96-03. The subsections that follow give the staff's evaluations of the contents of the NuScale PTLR against the seven criteria in Attachment 1 of GL 96-03.

3.2.1 *PTLR Criterion 1*

PTLR Criterion 1 states that the PTLR contents should include the neutron fluence values that are used in the calculations of the adjusted reference temperature (ART) values for the P-T limit calculations. Accurate and reliable neutron fluence values are required in order to satisfy the provisions of GDC 14, 30, and 31 of 10 CFR Part 50, Appendix A, as well as the specific fracture toughness requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.61. Section 4.1 of the NuScale PTLR, detailing the reactor vessel neutron fluence calculation method, indicates that the neutron flux calculation methodology for the NuScale RV is in accordance with RG 1.190 with exceptions as described in Technical Report TR-0116-20781, "Fluence Calculation Methodology and Results." The staff discusses its evaluation of that technical report in the safety evaluation report for NuScale DCD Section 4.3. In addition, NuScale provided peak RV neutron fluence values projected to 57 effective full-power years (EFPY) of operation in Table 4-1 of the PTLR. The staff has determined that these neutron fluence values were calculated using an NRC-approved methodology that is appropriately consistent with the guidelines in RG 1.190. The inclusion of valid peak RV neutron fluence values calculated using a neutron fluence methodology that is in conformance with RG 1.190 fulfills the provisions of PTLR Criterion 1. Therefore, the staff determined that PTLR Criterion 1 is satisfied.

3.2.2 PTLR Criterion 2

Appendix H to 10 CFR Part 50 provides the staff's requirements for designing and implementing RV material surveillance programs. It requires that RV material surveillance programs for operating reactors comply with the specifications of American Society for Testing and Materials (ASTM) E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Appendix H requires that the program design and the surveillance capsule withdrawal schedules for the programs must meet the edition of ASTM E 185 that is current on the issue date of the version of the ASME Code to which the RV was purchased, although the rule permits more recent versions of ASTM E 185 to be used, up through the 1982 version.

To ensure conformance with these requirements, PTLR Criterion 2 states that the PTLR should either provide the RV surveillance capsule withdrawal schedule or provide references, by title and number, for the documents containing the RV surveillance capsule withdrawal schedule along with how the specimen examinations shall be used to update the PTLR curves. The criterion also states that the PTLR should reference, by title and number, any applicable surveillance capsule reports that have been placed on the docket by the licensee requesting approval of the PTLR for its units. This criterion assures that the ART calculations will appropriately follow the RV material surveillance program requirements of 10 CFR Part 50, Appendix H.

Section 8 of the NuScale PTLR discusses the NuScale RV material surveillance program. PTLR Section 8 states that the surveillance capsule program adheres to all requirements of 10 CFR Part 50, Appendix H, and satisfies ASTM E 185-82. PTLR Table 8-1 gives the capsule withdrawal schedule. For a predicted transition temperature shift of less than 56 degrees Celsius (C) (100 degrees Fahrenheit (F)), Table 1 of ASTM E 185-82 requires a minimum of three surveillance capsules. For the NuScale RV, six identical surveillance capsule assemblies are provided (four capsules and archival material sufficient for two more capsules). Four primary surveillance capsules are provided for the NuScale surveillance program with three designated for removal during the 40 year license period. The staff determined that the NuScale withdrawal schedule is acceptable because it is consistent with the intent of the withdrawal schedule recommended in ASTM E 185-82.

It is stated in PTLR Section 8 that the surveillance capsule specimens will consist of "material used in the construction of the lower RPV [Reactor Pressure Vessel] shell" consistent with DCD Section 5.3.1.6 that states that,

> *all material used for specimens will be taken from the actual production forging, and from a weldment made of the same weld wire heat and flux lot combination used in the production weld.*

This is acceptable because it is in accordance with ASTM E 185-82.

In NuScale DCD Section 5.3.2.2, under COL Item 5.3-2 the applicant stated that,

> *A COL applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated. These procedures will be based on material properties of the as-built reactor vessels.*

As the surveillance program monitors the material properties of the as-built reactor, the staff concludes that when data from the surveillance capsules become available, they will be used to adjust the P-T limit curves as necessary. As stated in PTLR Section 4.2.3.2, *It should be noted that licensees of NuScale reactors are required by 10 CFR 50 Appendix H to maintain a RPV surveillance program for each RPV. The predicted ΔRTNDT values for the NuScale RPV will be verified or adjusted based on data from surveillance specimen test results ahead of RPV reaching the same embrittlement level. Systemic bias in the ΔRTNDT methodology will be detected and corrected per RG 1.99 Rev. 2 Regulatory Position 2.1 before it becomes a safety concern.*

The applicant confirms via this statement that the results of surveillance specimen testing should be used to identify and correct bias in results. This statement provides a reasonable assurance that the plant operating limits will meet the requirements of Appendix G to 10 CFR Part 50 throughout the operating life of the NuScale.

Based on the review described above, the staff determined that the provisions of PTLR Criterion 2 are satisfied. The staff notes that all provisions of PTLR Criterion 2 will remain applicable to specific plants referencing the NuScale design. As such, future NuScale plants that incorporate the NuScale PTLR will be expected to update their PTLRs in accordance with PTLR Criterion 2 as plant-specific surveillance capsule reports become available.

3.2.3 PTLR Criterion 3

The RV and connected components of the RCPB are designed to withstand the effects of neutron embrittlement from system pressure and temperature variations introduced by controlled heatup/cooldown operations and operational transients. Of the components exposed to neutron fluence, the RV is considered the most critical component susceptible to non-ductile failure because of the neutron fluence experienced over the vessel lifetime. GL 96-03 Criterion 3 states that the LTOP System lift setting limits developed using NRC approved methodologies may be included in the PTLR.

The NuScale LTOP system controls the RCS pressure at low temperatures such that the integrity of the RCPB is not compromised by the effects of neutron embrittlement. ASME, Section XI, Appendix G defines the requirements for pressure-temperature limits, derives the LTOP pressure-temperature (P-T) limits and the low temperature LTOP operational setpoint. To enforce the P-T limits and achieve overpressure protection, the LTOP system consists of three emergency core cooling system (ECCS) reactor vents valves (RVV) with each designed with sufficient relief capacity to prevent any pressure transient from exceeding the brittle fracture stress limits of the RV and RCPB pressure-retaining components when operating at low temperature conditions. The mechanical design of the RVVs prevents the valves from opening above a threshold pressure. The NuScale DCD Sections 5.2.2 and 6.3 provide a discussion of the RVV design with a summary of the design parameters in DCD Table 6.3-2.

NuScale DCD Table 5.2-10 provides the LTOP pressure setpoint values as a function of cold temperature using linear interpolation between the values up to the LTOP limit of 325 °F and Figure 5.2-4 provides a graphical profile of the variable setpoint. This temperature dependent pressure setpoint array is programmed into the module protection system (MPS) which monitors these process variables and actuates the engineered safety features actuation system (ESFAS) whenever the pressure setpoint is reached. ESFAS provides the valve control actuation to the RVVs’ pilot valve assembly when RV temperature is below the LTOP enabling temperature.

Technical Specifications LCOs 3.3.1, 3.4.3, and 3.4.10 are applicable to ensure the RVVs are operable to maintain the pressure and temperature within the limits specified in the PTLR during LTOP heatup and cooldown operation.

As described in NuScale DCD Section 6.3.2.2, "Equipment and Component Descriptions," for low temperature operations, the RVV main valve is actuated automatically by a safety function signal from the MPS that de-energizes the actuator trip valve solenoid whenever the reactor pressure reaches the setpoint value. The de-energized trip pilot valve allows the hydraulic line from the main valve control chamber to vent into the CNV allowing spring force and RCS pressure to open the main valve.

In addition, COL Item 5.2-2 states that the COL applicant that references the NuScale Power Plant design certification will provide a certified Overpressure Protection Report in compliance with the ASME Code, Section III, NB-7200 and NC-7200 to demonstrate the RCPB and secondary system are designed with low temperature overpressure protection features including a plant-specific pressure and temperature limit curves as addressed in the PTLR report, consistent with an approved methodology. Based on the above, the staff has concluded that the provisions of PTLR Criterion 3 is satisfied.

3.2.4 PTLR Criterion 4

Appendix G to 10 CFR Part 50 requires that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron embrittlement on the fracture toughness of RV beltline materials. For P-T limits, the effects of neutron embrittlement on the fracture toughness of RV beltline materials are defined in terms of the shift in the RT_{NDT} values resulting from neutron irradiation over a given period of facility operation, expressed in EFPYs. The final ART value for a material resulting from neutron embrittlement over a certain period of facility operation is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the shift in RT_{NDT} caused by irradiation (ΔRT_{NDT}), and a margin term. RG 1.99, Rev. 2 gives the staff's recommended methods for calculating ART values used for P-T limit calculations. ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the neutron fluence, and the calculational procedures. In the case of NuScale an additional temperature correction term, generally referred to as °/°, was added to account for the low T_{cold} of the NuScale design relative to the temperature zone of applicability of RG 1.99, Rev. 2. ASME Code, Section XI, Appendix G requires that licensees determine the ART at the 1/4T (one-quarter of the RV thickness) and 3/4T (three-quarters of the RV thickness) locations.

To ensure compliance with the requirements of 10 CFR Part 50, Appendix G, PTLR Criterion 4 states that PTLR contents should identify the limiting materials and limiting ART values at the 1/4T and 3/4T locations in the wall of the RV. To ensure compliance with the PTS requirements of 10 CFR 50.61, PTLR Criterion 4 also states that the PTLR contents should identify the limiting RT_{PTS} value for the RV. Section 4 of the PTLR provides the method used to determine the ART values at the 1/4T and 3/4T locations. Tables 4-1, 4-2, 4-4, and 4-5 of the NuScale

PTLR provide the inputs for the ART and RT_{PTS} calculations, including RV beltline material chemistry values, initial RT_{NDT} values, and peak RV beltline neutron fluence projections at 57 EFPY.

Additional analysis was conducted for several locations based on the unique geometry of the NuScale vessel, as illustrated in Fig. 4-1 of the NuScale PTLR. The limiting material in the NuScale RV was selected due to the combination of material properties and geometric attributes of the most limiting location analyzed from those shown in Fig. 4-1. The staff confirmed the applicant's selection of the limiting beltline material and performed a sampling review of the calculations conducted concerning the geometric discontinuities. The staff reviewed several intermediate calculations, looking for aspects such as convergence, stability, and fit to confirm that the methods used were properly chosen and applied. The staff also verified the calculation of the ART values for the beltline material using the method in RG 1.99, Revision 2. Finally the staff verified that the RT_{PTS} calculations were in compliance with the requirements of 10 CFR 50.61.

To assess the application of the °/° term the staff reviewed the available literature and operating experience. This term is briefly discussed by the NRC staff in a February 12, 1998, workshop (ADAMS Accession No. ML110070570) as appropriate for use to correct "for temperatures near 550°F." As NuScale T_{cold} is considerably lower than this, the staff compared the use of RG 1.99, Rev. 2 with and without the °/° term to materials data most similar to the NuScale conditions, those reported from Chooz-A ("Vessel Investigation Program of 'CHOOZ A' PWR Reactor after Shutdown;" Effects of Radiation on Materials: 20th International Symposium; Rosinksi, et. al.; 2001.) The staff concluded that the use of RG 1.99, Rev. 2 with the °/° term provides a reasonable approximation of the embrittlement of NuScale at neutron fluences at and below the 57-EFPY neutron fluences noted in Table 4-1 of the NuScale PTLR. The staff concludes, therefore, that the ART calculations with the °/° term are adequate.

As discussed above, the staff verified that the ART calculations were consistent with RG 1.99, Rev. 2, and the RT_{PTS} calculations were in compliance with the requirements of 10 CFR 50.61. In addition, the applicant has appropriately identified the limiting materials and limiting ART values at the 1/4T and 3/4T locations as well as the limiting RT_{PTS} value for the RV including accounting for unique geometric aspects. Therefore, the staff determined that the provisions of PTLR Criterion 4 are satisfied.

3.2.5 PTLR Criterion 5

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors and the minimum temperatures established for the stressed regions of RVs (i.e., for the RV flange and stud assemblies) be met for all conditions. Section IV.A.2 also requires that the P-T limits for operating reactors must be at least as conservative as those that would be generated if the methods of analysis in ASME Code, Section XI, Appendix G were used to generate the P-T limit curves. Table 1 of 10 CFR Part 50, Appendix G summarizes the required criteria for generating the P-T limits for operating reactors.

To ensure that PTLRs are in compliance with the above requirements, PTLR Criterion 5 states that the PTLR contents should provide the P-T limit curves for heatup and cooldown operations, core critical operations, and pressure testing conditions for operating light-water reactors.

Figures 9-1 through 9-3 give the P-T limit curves for heatup and cooldown operations, core critical operations, and hydrostatic and leak testing.

The staff reviewed the approach the applicant took to generating their analyses for the P-T curves for heatup and cooldown operations, core critical operations, and hydrostatic and pressure testing provided in the NuScale PTLR. Based on this review, the staff determined that the applicant developed its proposed P-T limits in accordance with ASME Code, Section XI, Appendix G, and that the limits therefore satisfy the requirements of 10 CFR Part 50, Appendix G. Hence, the applicant's proposed P-T limit curves are acceptable for operation of the NuScale RV. Therefore, the staff determined that the provisions of PTLR Criterion 5 are satisfied.

3.2.6 PTLR Criterion 6

Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors and the minimum temperature requirements for the highly stressed regions of the RVs (i.e., for the RV flange and stud assemblies) be met for all conditions. Table 1 of 10 CFR Part 50, Appendix G provides the criteria for meeting the minimum temperature requirements for the highly stressed regions of the RV. The applicant provided a completed Table 1 in Table 6-2 of the NuScale PTLR.

PTLR Criterion 6 states that the minimum temperature requirements of 10 CFR Part 50, Appendix G shall be incorporated into the P-T limit curves, and the PTLR shall identify minimum temperatures on the P-T limit curves such as the minimum boltup temperature and the hydrotest temperature. The staff determined that the curves were in compliance with the minimum temperature requirements of 10 CFR Part 50, Appendix G. Furthermore, the PTLR clearly identifies the minimum boltup temperature and hydrotest temperature on the P-T limit curves. Therefore, the staff determined that the provisions of PTLR Criterion 6 are satisfied.

3.2.7 PTLR Criterion 7

RG 1.99, Rev. 2 gives the staff's recommended methods for calculating the ART values for RV beltline materials. These are calculated for the 1/4T and 3/4T locations in the RV wall. ASME Code, Section XI, Appendix G and 10 CFR Part 50, Appendix G require the use of these values for the calculations of P-T limit curves for reactors. Appendix G to 10 CFR Part 50 also requires the ART values to include the applicable results of the RV material surveillance program of 10 CFR Part 50, Appendix H. ART values for ferritic RV base metal and weld materials increase as a function of accumulated neutron fluence and the quantity of alloying elements in the materials, particularly copper and nickel.

The procedures of RG 1.99, Rev. 2 specify the use of a CF as a means for quantifying the effect of the alloying elements on the ART values. Furthermore, RG 1.99, Rev. 2 specifies that a CF be calculated and input into the calculation of the final ART value for each beltline material. RG 1.99, Rev. 2 cites two possible methods for determining the CF values for the RV beltline base metal and weld materials: (1) Regulatory Position 1.1 in RG 1.99, Rev. 2 allows the licensee to determine the CF values from applicable tables in the RG as a function of copper and nickel content, or (2) Regulatory Position 2.1 allows the use of applicable RV surveillance data to determine the CF values if the base metal or weld materials are represented in a licensee's RV material surveillance program and if two or more credible surveillance data sets become available for the material in question. RG 1.99, Rev. 2 defines the criteria for determining the credibility of the RV surveillance data sets. In accordance with the requirements of 10 CFR Part 50, Appendix G, RG 1.99, Rev. 2 states that if the procedure of Regulatory Position 2.1 results in a higher ART value than that given by using the procedure of Regulatory Position 1.1, the surveillance data should be used for determining the CF and the ART. If the procedure of Regulatory Position 2.1 results in a lower value for the ART, either procedure may be used for determining the CF and the ART.

To ensure that PTLRs are in compliance with the above regulatory requirements and guidelines, PTLR Criterion 7 states that if surveillance data are used in the calculations of the ART values, the PTLR contents should include the surveillance data and calculations of the CF values for the RV base metal and weld materials as well as an evaluation of the credibility of the surveillance data against the credibility criteria of RG 1.99, Rev. 2. However, the NuScale PTLR is generic for the NuScale design and is based on bounding embrittlement correlations for which surveillance data are not yet available. Therefore, the incorporation of surveillance data and related calculations is currently not applicable to the NuScale PTLR. As previously discussed, the CF and ART values in the PTLR were determined using the procedures of Regulatory Position 1.1 in RG 1.99, Rev. 2 as adjusted by the °/° term due to the T_{cold} of the NuScale design. Therefore, the staff determined that the provisions of PTLR Criterion 7 are satisfied. However, the staff notes that the provisions of PTLR Criterion 7 will remain applicable to the review of PTLR submitted by plants referencing the NuScale design. These applicants will need to justify how their plant-specific PTLR will remain compliant with RG 1.99, Revision 2, Section 1.3.2 which requires that the necessary NuScale temperature correction factor be “justified by reference to actual data.” As such, plants that submit a NuScale PTLR based on the generic NuScale PTLR methodology will be expected to describe how they will update their PTLRs in accordance with PTLR Criterion 7 as plant-specific surveillance data become available.

4.0 CONCLUSION

The staff has completed its review of the NuScale PTLR and determined that the contents of the NuScale PTLR conform to the staff’s technical criteria for PTLRs as defined in Attachment 1 of GL 96-03. The staff has also determined that the PTLR has satisfied the requirements of 10 CFR Part 50, Appendix G. Furthermore, the staff has determined that the NuScale PTLR is compatible with the NuScale generic TS and that the PTLR-related TS provisions meet the technical criteria of GL 96-03. Based on this evaluation, the staff concludes that the latest revision of the NuScale PTLR is acceptable for generic use by NuScale COL applicants for establishing limiting P-T limit curves, LTOP system limits, and related input parameters. Pursuant to TS 5.6.4c, future NuScale COL holders will be required to provide the PTLR to the NRC upon issuance for each RV neutron fluence period and for any PTLR revision or supplement thereto. Finally, in accordance with GL 96-03, the NRC must approve any subsequent changes in the method used to develop the P-T limits.