NuScaleDCDocsPEm Resource

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MCB Issue List NuScale, DCD Section 4.5.2 – Reactor Internals and Core Support Structure Materials

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria 1 and 10 CFR Part 50.55a require that structures, systems, and components (SSCs) important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed.

In order for the staff to determine whether the NuScale design control document (DCD) meets these criteria with regard to reactor internals materials, the staff is requesting the following information.

Issue #1

The DCD includes limited schematical diagrams concerning the reactor vessel internals (RVIs). The staff requires more comprehensive and detailed diagrams of the RVIs in order to disposition the adequacy of the information provided in DCD Section 4.5.2. To assess the adequacy of material selection and fabrication, the staff needs to thoroughly understand the geometry, construction, and interfaces of the RVIs and the assessment of potential wear surfaces, crevice locations, and material compatibilities. The staff currently lacks sufficient visual information pertaining to the nature, size, interfaces, and geometry of the RVIs.

Several examples of the level of detail needed by the staff can be found in publically available applications such as the Donald C. Cook, "Transmittal of Reactor Vessel Internals Aging Management Program," Agency Documents Access and Management System Accession No. ML12284A320:

https://adamswebsearch2.nrc.gov/webSearch2/main.jsp?AccessionNumber=ML12284A320

Figure 2-1 in the above is an example of the level of detail and clarity the staff requests be added to the figures in the DCD. This figure provides a volumetric sense of the components with indication of interfaces, water space, etc. All major components are to be present and scaled roughly accurately. The staff requests that the DCD be revised to include diagrams suitable to the task as noted above.

The figures in Appendix E of the Cook submittal present detailed and accurately scaled information for individual reactor internal components including information concerning the locations of welds and interfaces. While such detailed figures do not warrant inclusion in the DCD, the staff requires such figures to appropriately understand and disposition information concerning the reactor internals and core supports. The staff requests diagrams be provided featuring all components listed in Table 4.5.2-2 with emphasis on interfaces between components.

Issue #2

NuScale DCD Section 4.5.2.1 states that the materials for core support structures and threaded structural fasteners conform to the requirements of ASME Code, Section III, Subsubarticle NG-2120, and Tables 2A, 2B, and 4 of ASME Code, Section II. The DCD does not, however, clarify what the "remaining portions of the RVI" conform to. NUREG 0800, Section 4.5.2, "Standard Review Plan - Reactor Internal and Core Support Structure Materials" states that both core support and reactor internals are to comply with the ASME Code items cited above.

The staff requests that the applicant clarify that all core support and reactor internal materials shall comply with ASME Code, Section III NG-2120 and ASME Code, Section II Tables 2A, 2B, and 4 as they pertain to material specifications and revise the DCD accordingly. Alternatively the staff request that the applicant justify their position and revise the DCD accordingly.

Issue #3

NuScale DCD Section 4.5.2.4 states the processing and welding of unstabilized AISI Type 3XX austenitic stainless steel shall comply with Regulatory Guide (RG) 1.44, "Control of the Processing and use of Stainless Steel." This is further clarified by stating that such materials, when rapidly cooled by means other than water quenching, are to be subjected to Practice A or E of ASTM A262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels." The staff notes that this position is not entirely consistent with RG 1.44, which states,

> 3. Nonsensitization of the material[3] should be verified using ASTM A262, Practices A or E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steel. Test specimens should be selected from material subjected to each different heat treatment practice and from each heat.
> …
>
> [3] Material of product forms with simple shapes not subject to distortion during heat treatment, such as plate, sheet, bars, pipe, and tubes, need not be tested provided that water quenching follows the solution heat treatment.

As stated in Note 3, quoted above, a larger set of materials should be subjected to Practice A or E of ASTM A262 than stated in the DCD to comply with RG 1.44.

The staff requests that the applicant revise the DCD to be fully compliant with RG 1.44 as written. Alternatively the staff request that the applicant justify their position and revise the DCD accordingly.

Issue #4

NuScale DCD Section 4.5.2.5 states that washers in the RVI upper riser assembly are made from Alloy 718. The section refers to DCD Section 3.13.1 for discussion concerning the annealing and precipitation hardening treatment for Alloy 718 materials. DCD Section 4.5.2.5 does not address the potential for stress corrosion cracking (SCC) for Alloy 718 materials exposed to reactor coolant. DCD Section 3.13 does not address the potential for SCC in components exposed to high temperatures and reactor coolant.

The staff requests that the applicant address the potential for SCC of Alloy 718 washers in the RVI upper riser assembly and revise the DCD accordingly.

Issue #5

NuScale DCD Section 4.5.2 cites Fig. 3.9-5. This figure does not exist.

The staff requests that the applicant correct the citation by revising the DCD.

Issue #6

NuScale DCD Section 4.5.2.1 states that,

> Neutron irradiation-induced degradations such as irradiation-assisted stress corrosion cracking, void-swelling, stress-relaxation, and irradiation embrittlement have been evaluated using material aging degradation mechanism screening criteria of the Electric Power Research Institute (EPRI) materials reliability program (Reference 4.5-3).

The applicant then states that,

> The components meeting the screening criteria are the upper CRDS (control rod drive system) supports, ICI (in-core instrumentation) guide tube supports, CRDS alignment cones, CRA (control rod assembly) lower flange, CRA cards, fuel pins, shared fuel pins, fuel pin caps, upper support blocks, lower core plate alignment pins, lower core support lock insert, core support lock plate assembly, and the lock plate. In addition, components identified as susceptible to irradiation-induced stress relaxation are also included for potential wear due to loosening.

No discussion is presented regarding what, if anything, is done regarding the components that meet the criteria, or which criteria were met for which components. The staff notes that although MRP-175, "PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," was used to provide the criteria for the NuScale evaluation, the results of applying MRP-175 based criteria do not stand alone from how the evaluation was conducted for NuScale specifically. The staff has been provided with no basis with which to verify the evaluation and no consequent actions for components which were deemed susceptible.

The staff requests that, (a) the applicant provide the evaluations that were performed, and (b) a discussion of the compensatory actions applied (or to be applied) to the components that screened in. If no compensatory actions are required, the staff request that this be justified in the response to (b). The staff further request that the response to (b) be added to the DCD.

MCB Issue List NuScale, DCD Section 5.3.1 – Reactor Vessel Materials

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 1 and 30 require that the reactor vessel shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. 10 CFR 50, Appendix A, GDC 4 requires that the reactor vessel be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. 10 CFR 50, Appendix A, GDC 14 requires that the vessel be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. 10 CFR 50, Appendix A, GDC 31 requires that the reactor vessel be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. 10 CFR 50, Appendix A, GDC 32 requires that the reactor vessel be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel. Conformance with 10 CFR 50.55a, 50.60, Appendix B, Appendices G & H, 10 CFR 52.47(b)(1) and 10 CFR 52.80(a) is required as necessary to comply and ensure adequacy with regards to the GDCs listed above and licensing requirements.

In order for the staff to determine whether the NuScale design meets these criteria with regard to reactor vessel materials, the staff is requesting the following information.

Issue #1

NuScale DCD Section 5.3.1 does not explicitly require or note that the manufacturer or installer is required to certify, by application of the appropriate Code Symbol and completion of appropriate data report in accordance with ASME Code Section III, Article NCA-8000, that the material used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000. The staff notes that DCD Section 5.3.1 traces the application of the ASME Code at a very high level making verification of adequacy difficult.

The staff requests that the applicant revise the section to more clearly and logically delineate ASME Code requirements such as noted above. A level of citation detail commensurate with the level of detail found in NUREG-0800, Section 5.3.1, "Standard Review Plan – Reactor Vessel Materials," would be considered appropriate. Traceability and verification of materials is critical to ensure the adequacy of the final vessel in addition to supporting emergent needs of the licensee post-installation.

Issue #2

NuScale DCD Section 5.3.1.2 states that austenitic stainless steel material quenched by means other than water quenching shall be subjected to Practice A or E of ASTM A262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels." The staff notes that this position does not comply with RG 1.44, "Control of the Processing and use of Stainless Steel," which states,

> 3. Nonsensitization of the material[3] should be verified using ASTM A262, Practices A or E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steel. Test specimens should be selected from material subjected to each different heat treatment practice and from each heat.
> …
> [3] Material of product forms with simple shapes not subject to distortion during heat treatment, such as plate, sheet, bars, pipe, and tubes, need not be tested provided that water quenching follows the solution heat treatment.

As stated in Note 3, quoted above, a larger set of materials should be subjected to Practice A or E of ASTM A262 than stated in the DCD to comply with RG 1.44.

The staff requests that the applicant revise the DCD to be fully compliant with RG 1.44 as written. Alternatively the staff request that the applicant justify their position and revise the DCD accordingly.

Issue #3

NuScale DCD Section 5.3.1.3 references examinations for SG tubing. As the SG is not integral to the reactor vessel (rather is a component internal to the reactor vessel), the staff requests that this text be removed from DCD Section 5.3.1.3. The staff notes that this text is duplicative to DCD Section 5.4.1.4.

Issue #4

NuScale DCD Section 5.3.1.4 contains no discussion of abrasive work controls for austenitic stainless steel surfaces to avoid contamination and surface cold-work which may promote stress corrosion cracking.

The staff requests that the applicant revise DCD Section 5.3.1.4 to address abrasive cold work controls, consistent with the discussion in DCD Section 5.2.3.4.1.

Issue #5

NuScale DCD Section 5.3.1.5 references Table 5.3-3 as providing “1/4T-adjusted reference temperature and upper shelf energy projections using RG 1.99.“ The values reported in DCD Table 5.3-3 are not acceptable as addressed directly in the staff review of TR-1015-18177, “Pressure and Temperature Limits Methodology,” from which these values derive.

The staff requests that the applicant revise DCD Table 5.3-3, and any other locations containing or referencing these values, to maintain consistency with any revisions to TR-1015-18177.

Issue #6

NuScale DCD Section 5.3.1.6 and Table 5.3-5 contain language concerning license renewal.

The staff requests that the applicant either remove this text, or make it clear that this text is for information only and is subject to any and all regulatory requirements such as they exist at the time of a potential future license renewal application.

Issue #7

NuScale DCD Section 5.3.1.6 does not contain or reference a COL item to submit their final surveillance program plan consistent with NUREG 0800, Section 5.3.1, “Standard Review Plan – Reactor Vessel Materials,” II.6.C.

The staff requests that the applicant revise DCD Section 5.3.1.6 to add a COL item consistent with the above or justify why a COL applicant does not need to submit their operational program as part of the COL. This revision should clarify the COL specific actions necessary to establish an acceptable program.

Issue #8

NuScale DCD Section 5.3.1.6 cites reactor vessel surveillance capsule lead factors ranging between 1.5 to 4.5. This range of lead factors is inconsistent with ASTM E-185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactors Vessels,” which permits lead factors between 1 and 3. In addition all lead factors used in DCD Table 5.3-5 are 4.3 implying that all capsules could or should be positioned to achieve the same high lead factor.

Historically a range of lead factors was implemented with the highest lead factor capsules being removed first to provide early access to measured irradiated material properties of the actual vessel. Lower lead factor capsules then ensured capture of data based on a longer period of operational conditions. The incorporation of some surveillance capsules with lower lead factors ensures that data on reactor vessel embrittlement is being gathered from samples which represent the actual reactor vessel material conditions as closely as practicable.

The staff requests that the applicant justify the expanded range of lead factors cited in DCD Section 5.3.1.6 and revise the DCD accordingly. In addition the staff request that the applicant address the suitability of a 100% high lead factor program.

Issue #9

NuScale DCD Sections 5.3.1.2 and 5.3.1.3 echo information concerning cladding that is also included in NuScale DCD Section 5.2.3. The staff has requested, or will request clarification concerning cladding in an Issue List related to DCD Section 5.2.3.

The staff requests that the applicant replace the cladding information in DCD Section 5.3.1 with references to the appropriate subsections of DCD Section 5.2.3.

MCB Issue List NuScale, DCD Section 5.3.2 – Pressure Temperature Limits, PTS, USE

Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 14 requires that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. 10 CFR Part 50, Appendix A, GDC 31 requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of 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.

In order for the staff to determine whether the NuScale design control document (DCD) meets these criteria with regard to pressure temperature limits, pressurized thermal shock (PTS), and upper shelf energy (USE), the staff is requesting the following information.

Issue #1

NuScale DCD Section 5.3.2 references Limit Curves (DCD Section 5.3.2.1) and adjusted reference temperature information (DCD Section 5.3.1.5) based on reference TR-1015-18177, “Pressure and Temperature Limits Methodology.” This methodology includes methods unacceptable to the staff for this application, namely 10 CFR 50.61a, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” and ASTM E900-15, “Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials.”

The use of 10 CFR 50.61a is restricted as noted therein,

> The requirements of this section apply to each holder of an operating license for a pressurized water nuclear power reactor whose construction permit was issued before February 3, 2010 and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Vessel Code, 1998 Edition or earlier.

This is due to the fact that the analytical basis (e.g., thermohydraulic transient types and frequencies, etc.) underlying 10 CFR 50.61a would not apply to a design like the NuScale SMR. Similarly the use of E900-15, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," would require substantial analysis and justification to be used in concert with 10 CFR 50.61. Consequently neither of these methodologies are acceptable as presented to the staff for this application.

The staff request that the applicant revise the associated information in the DCD to conform to any and all necessary revisions of TR-1015-18177. The staff has identified that Figures 5.3-3 through 5.3-5; and Tables 5.3-6, 5.3-7, 5.3-8, 5.3-9, and 5.3-10 contain information requiring update due to necessary revision of TR-1015-18177 results. Any and all information linked to, based on, or echoing any of the information stemming from the unacceptable methods used in TR-1015-18177 prior to its revision should be updated.

<u>Issue #2</u>

DCD Section 5.3.2.2 COL Item 5.3-2 does not clearly require that the applicant base their operating procedures on actual plant properties as opposed to the design properties presented in DCD Section 5.3 and TR-1015-18177, "Pressure and Temperature Limits Methodology."

The staff requests that the applicant provide sufficient clarification in DCD Section 5.3.2, in COL Item 5.3-2 to make it clear that the applicant must verify that their plant-specific properties are bounded by the values provided in the DCD and TR-1015-18177 or produce appropriate limits based on actual plant properties.

MCB Issue List NuScale, PTLR – Pressure and Temperature Limits Methodology

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. 10 CFR Part 50, Appendix G, requires that the P-T limits for an operating light-water nuclear reactor 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 (ASME) Boiler and Pressure Vessel Code (ASME Section XI, Appendix G). For conditions with the core critical, P-T limits must be more conservative than the ASME Section XI, Appendix G limits. Table 1 of 10 CFR Part 50, Appendix G, provides a summary of the requirements for P-T limits relative the ASME Section XI, Appendix G, criteria, as well as the minimum temperature requirements for bolting up the reactor vessel (RV) during normal and pressure testing operations. 10 CFR Part 50, Appendix G, also 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. The rule 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 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 14, 15, 30, and 31.

Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” addresses the technical information necessary for a licensee to implement a PTLR. GL 96-03 establishes the information that must be included in (1) an acceptable PTLR methodology (with the P-T limit methodology as its subset) and, (2) the PTLR itself. TSTF-419 (ADAMS Accession No. ML012690234) provides additional guidance which provides an alternative format for documenting the implementation of a PTLR in the “Administrative Controls” section of a facility’s technical specifications (TS).

In order for the staff to determine whether the NuScale technical report (TR) meets the criteria of 10 CFR Part 50, Appendix G and Generic Letter 96-03 the staff requires the following information.

Issue #1

NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” Rev. 0 - The staff requires clarification regarding the status of information which is inconsistently marked between the TR and the DCD. This information is inconsistent with precedent from operating reactors and previous design certification reviews, the staff requests further information to understand why such a designation should apply to the NuScale design.

Issue #2

NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” Rev. 0 identifies that T_{COLD} for the NuScale design falls outside of the applicability of Regulatory Guide 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials.” The TR utilizes the methodologies of ASTM E900-15, “Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials,” and 10 CFR 50.61a, “Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” to calculate necessary embrittlement results.

The use of 10 CFR 50.61a is restricted as noted therein,

> The requirements of this section apply to each holder of an operating license for a pressurized water nuclear power reactor whose construction permit was issued before February 3, 2010 and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Vessel Code, 1998 Edition or earlier.

This is due to the fact that the analytical basis (e.g., thermohydraulic transient types and frequencies, etc.) underlying 10 CFR 50.61a would not apply to a design like the NuScale SMR. Similarly the use of E900-15, “Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials,” would require substantial analysis and justification to be used in concert with 10 CFR 50.61. Consequently neither of these methodologies are acceptable as presented to the staff for this application.

The staff requests that the applicant revise TR-1015-18177 with a staff approved methodology to account for embrittlement at the presented T_{COLD}. The staff has approved of methodologies which account for T_{COLD} temperatures outside of the range explicitly addressed in RG 1.99, Rev. 2. An example of an approved methodology can be found in a staff presentation accessible using ADAMS Accession No. ML110070570, and at:

https://www.nrc.gov/docs/ML1100/ML110070570.pdf

Issue #3

The staff requests that the applicant provide several example calculations, including all intermediate calculations and their values, for representative points on the curves provided in NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” once revised to include a staff acceptable correction factor for T_{COLD}. This information is necessary for the staff to confirm the accuracy of the P-T Limits provided in the report.

Issue #4

Please confirm that all neutron fluence related information presented in the NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” is consistent with the information and methodology presented in TR-0116-20781, “Fluence Calculation Methodology and Results.” If this is not the case, justify that the approach used to calculate the fluence values used in TR-1015-18177 is acceptable per Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” or other staff acceptable basis.

Issue #5

NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” Section 8.0 cites a RPV surveillance capsule specimen lead factor of 4.3. This is not compliant with ASTM E185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels” as required by 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.” Citation of an unapproved edition of ASTM E185, namely E185-15, as justification for lead factors up to 5 as being “within the current industry consensus of acceptable lead factor range” is not a sufficient basis.

The staff requests that the applicant justify the expanded range of lead factors by citing a publically available scientific or technical basis in the DCD and TR-1015-18177 and revise the TR accordingly.

Issue #6

NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” does not contain clear prescriptive requirements to monitor, and update if necessary, the pressure temperature limits as surveillance data becomes available consistent with Generic Letter (GL) 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits.” Criterion 7 from GL 96-03 states,

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR
Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature	Describe how the data from multiple surveillance capsules are used in the ART calculation.	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation.

(RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2.) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + 2.), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Describe procedure if measured value exceeds predicted value. WHEN OTHER PLANT DATA ARE USED 1. - Identify the source(s) of data when other plant data are used. 2.a - Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability. OR 2.b - Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.	Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.

The closest language in the TR is in TR Appendix A,

> 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.

This language implies that surveillance data will be used to correct the ΔRT_{NDT} calculations. This is not clearly in accordance with Generic Letter 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits,” Criterion 7.

The staff requests that the applicant revise the TR to include clear prescriptive requirements to monitor, and update if necessary, the pressure temperature limits and associated inputs as surveillance data becomes available in accordance with Generic Letter 96-03, Criterion 7.

Issue #7

NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” contains language concerning license renewal, in particular citing NUREG-1801, “Generic Lessons Learned Report,” Rev. 2.

The staff requests that the applicant either remove this text, or make it clear that this text is for information only and is subject to any and all regulatory requirements such as they exist at the time of a potential future license renewal application.

Issue #8

NuScale TR-1015-18177, “Pressure and Temperature Limits Methodology,” contains no discussion of geometric discontinuities above the lower RPV flange, for example nozzles. The staff requests that the applicant clarify how and why they determined that all discontinuities are appropriately bounded by the analysis. If the applicant relies on citation of ASME Code Section XI, Appendix G, G-2223(c) the applicant is further requested to justify that this position developed for large light water reactor applications appropriately applies to the NuScale design.