Response to SDAA Audit Question

Question Number: A-5.3.2.3-1

Receipt Date: 10/20/2023

Question:

SDAA Section 5.3.2.3 states that because 10 CFR 50.61 is not applicable to the SDAA design consequently no consideration of pressurized thermal shock is necessary. The description implies, but does not state, that due to the minimal embrittlement in the upper RV, and the selection of austenitic material for the lower RV, the subject RV material cannot credibly be challenged by pressurized thermal shock transients.

The staff cannot make a finding without this information in the SDAA. Clarify how NuScale confirmed that pressurized thermal shock, or other similar transient, will not credibly challenge the reactor power module integrity. Alternatively, provide a supplemental explanation.

Follow-up question received on January 19, 2024:

The staff requested that NuScale clarify in the SDAA how NuScale confirmed that pressurized thermal shock (PTS), or other similar transient, will not credibly challenge reactor power module integrity. NuScale response stated that the PTS screening methodology in 50.61 does not apply to the RPV beltline material because the lower RPV is not made of ferritic materials. The staff concur that the limiting material/location in the NPM is not the in the beltline. Consequently, to make a finding regarding PTS the staff require a verifiable docketed basis that the ferritic materials in the upper NPM will not be inappropriately challenged by PTS. The applicant did not provide any revisions to the SDAA section 5.3.2.3 to describe why PTS will not challenge NPM integrity. Please describe in the SDAA how NuScale confirmed that the RV material cannot be challenged by PTS transients.

Response:

Regulation 10 CFR 50.61 requires that, "...the licensee shall have projected values of RT_{PTS} or RT_{MAX-X}, accepted by the NRC, for each reactor vessel beltline material." Standard Design Approval Application (SDAA) Section 5.3.2.3 addresses the fact that pressurized thermal shock

(PTS) screening methodology is not applicable to the NuScale Power Module (NPM-20) reactor pressure vessel (RPV) beltline material because the lower RPV is not made of ferritic materials. Section 4.3.1 of TR-130877-P, “Pressure and Temperature Limits Methodology,” addresses PTS screening for the NPM-20, which supports the statements in SDAA Section 5.3.2.3.

Note that the applicability of PTS to the RPV does not exempt the RPV from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III qualifications. Analysis of the structural integrity of the RPV is in accordance with ASME BPVC Section III. Preliminary evaluations show that fatigue damage caused by the expected transients is very low in comparison to conventional pressurized water reactors due to the low flow rates in the NPM-20 due to natural circulation flow rather than pump-driven flow. NuScale will complete ASME BPVC Section III, Appendix G evaluations later in the design process because the large margins on the preliminary ASME BPVC Section III evaluation and mild nature of the NPM-20 transients shows this area to be of low risk of structural integrity issues.

There is an update to the Standard Design Approval Application (SDAA) Section 5.3.2.3 attached to this response clarifying how the US460 design meets this requirement.

Markups of the affected changes, as described in the response, are provided below:

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

Table 1.6-2: NuScale Referenced Technical Reports

Report Number	Title	FSAR Section
TR-118318, Revision 1	NuScale Design of Physical Security Systems	9.5, 13.6, 14.2
TR-122844-P, Revision 0	NuScale Instrument Setpoint Methodology Technical Report	7.0, 7.2
TR-121353-P, Revision 0	NuScale Comprehensive Vibration Assessment Program Analysis Technical Report	3.9, 14.2
TR-121515-P, Revision 0	NuScale Power Module Seismic Analysis	3.7, 3.12
TR-130877-P, Revision 0	Pressure and Temperature Limits Methodology	5.2, 5.3
TR-121517-P, Revision 0	NuScale Power Module Short-Term Transient Analysis	3.9
TR-130721-P, Revision 0	Use of Austenitic Stainless Steel for US460 Standard Design Reactor Pressure Vessel	5.2, 5.3
TR-121354-P, Revision 0	NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report	3.9, 14.2
TR-121507-P, Revision 0	Pipe Rupture Hazards Analysis	3.6

– GDC 14 and 31. Design and fabrication of the RPV and pressure retaining components associated with the RCPB ensure sufficient margin such that the RCPB behaves in a non-brittle manner and minimizes the probability of rapidly propagating fracture and gross rupture of the RCPB (Reference 5.2-10).

- Criterion XIII of 10 CFR 50, Appendix B. The Quality Assurance Program requires procedures for the control of the on-site cleaning of RPV and the RCPB during construction.

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

- Appendix G to 10 CFR 50. The RPV ferritic pressure retaining and integrally attached materials meet applicable fracture toughness acceptance criteria (Reference 5.2-10 and Reference 5.2-11). The design supports an exemption from the requirements of 10 CFR 50.60 which invokes compliance with 10 CFR 50, Appendix G. ~~Section 5.3.1.6~~Section 5.3.1.5 provides further details.

5.2.3.1 Material Specifications

The materials for the Class 1 components and supports that comprise the RCPB, including the RPV and SGs, are in Table 5.2-3. Table 5.2-3 also includes materials and specifications associated with the RPV attachments and appurtenances. The table lists the grade or type, as applicable, of the ferritic low alloy steels, austenitic stainless steels, and nickel-based alloys specified for the RCPB. Except where noted in Table 5.2-3, the associated ASME BPVC material specification provides the final metallurgical condition. Further discussion of the materials associated with the RPV is in Section 5.3, Reactor Vessel.

The RCPB surface materials in contact with reactor coolant or in contact with pool water during refueling, including welds, are corrosion resistant alloys or clad with austenitic stainless steel or nickel-based alloy. The SG tubesheet bores are the exception, the SA-508 tubesheet bores do not have corrosion resistant cladded surface. The SG tubes expand into the tubesheet bore to provide corrosion protection in the crevice between the SG tube and tubesheet.

Processing and welding of unstablized American Iron and Steel Institute Type 3XX series austenitic stainless steels for pressure retaining components comply with RG 1.44 to prevent sensitization caused by chromium depletion at the grain boundaries during welding and heat treatment operations. For unstablized American Iron and Steel Institute Type 3XX series austenitic stainless steel subjected to sensitizing temperatures subsequent to solution heat treatment, the carbon content is no more than 0.03 weight percent.

Processing and welding of American Iron and Steel Institute Type 2XX series austenitic stainless steels for pressure retaining components comply with ASME BPVC paragraph NB-2433 and RG 1.31 for delta ferrite composition. The ferrite number are in the range of 5 FN to 16 FN. The carbon content of the weld filler materials is restricted to 0.04 percent maximum.

Nickel-based Alloy 690 is a base metal in RCPB components and structures along with Alloy 52/152 cladding and weld metals and similar alloys developed for improved weldability. Alloy 690 and 52/152 have a high resistance to general

5.2.6 References

5.2-1 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-P, Revision 4.

5.2-2 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3.

5.2-3 Electric Power Research Institute, "Pressurized Water Reactor Primary Water Chemistry Guidelines," EPRI #3002000505, Revision 7, Palo Alto, CA, 2014.

5.2-4 American Society of Mechanical Engineers, "*Quality Assurance Requirements for Nuclear Facility Applications*," ASME NQA-1-2015, New York, NY.

5.2-5 Electric Power Research Institute, "Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors (MRP-111)," EPRI #1009801, Palo Alto, CA, 2004.

5.2-6 Electric Power Research Institute, "Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloy 690 in Pressurized Water Reactors (MRP-258)," EPRI #1019086, Palo Alto, CA, 2009.

5.2-7 Electric Power Research Institute, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines," EPRI #1013706, Revision 7, Palo Alto, CA, 2007.

5.2-8 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," New York, NY.

5.2-9 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.

5.2-10 NuScale Power, LLC, "Use of Austenitic Stainless Steel for NPM Reactor Pressure Vessel," TR-130721-P, Revision 0.

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

5.2-11 NuScale Power, LLC, "Pressure and Temperature Limits Methodology," TR-130877-P, Revision 0.

Section 4.5.1, Control Rod Drive - Materials Specifications, addresses the use of cold-worked austenitic stainless steel.

5.3.1.5 Fracture Toughness

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. However, the design supports an exemption from the requirements of 10 CFR 50.60, which includes an exemption from the requirements of 10 CFR 50, Appendix G for the lower RPV. The materials used for the lower RPV are not ~~applicable~~subject to the fracture toughness analyses required by 10 CFR 50, Appendix G~~, and the upper RPV does not meet the neutron fluence levels to be assessed for the effects of neutron embrittlement~~.

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

10 CFR 50, Appendix G, requirements apply to ferritic materials of pressure-retaining components of the RCPB of light water nuclear power reactors. The NPM uses austenitic stainless steel materials (Table 5.2-3) in the lower RPV shell. The requirements of 10 CFR 50, Appendix G, rely on impact testing data performed in accordance with ASME BPVC Section III, Paragraph NB-2331. ~~However,~~Paragraph NB-2331 follows NB-2311, which does not require impact testing of austenitic stainless steel.

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

~~The fluence values for the upper RPV shell do not exceed 1.0E+17 n/cm² (E > 1 MeV), which is the peak neutron fluence at the end of the design life of a reactor vessel that requires an assessment of the effects of neutron embrittlement as specified in 10 CFR 50, Appendix H. Therefore, the upper RPV, which is made of ferritic steel, does not require an assessment for the effects of neutron embrittlement and, there are no considerations of adjustments for embrittlement necessary using the RG 1.99 methodology.~~

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

~~The neutron flux and fluence calculation methods follow the guidance of RG 1.190 with exceptions as described in the NuScale Technical Report "Fluence Calculation Methodology and Results" (Reference 5.3-7).~~ Reference 5.3-7 provides further details regarding the fracture toughness capabilities of the austenitic stainless steel material used in the lower RPV. Reference 5.3-6 provides the methodology used for derivation of the pressure-temperature limits for the RPV, which ensures that the upper RPV meets GDC 14, GDC 15, GDC 31, and 10 CFR 50 Appendix G.

5.3.1.6 Material Surveillance

10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” applies to ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. However, the design supports

The welding of the stud to the cladding requires a cladding preservice liquid penetrant exam, per ASME BPVC Section III, paragraph NB-5272, Weld Metal Cladding. The welding of the stud to the cladding also complies with ASME BPVC Section III, paragraph NB-4435, Welding of Nonstructural Attachments.

There are no inservice exam requirements for the lock plate stud welds or the lock plates.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

The information in this section describes the bases for setting operational limits on pressure and temperature for the RCPB. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The design supports an exemption from the requirements of 10 CFR 50.60, which includes an exemption from the requirements of 10 CFR 50, Appendix G, and 10 CFR 50, Appendix H, for the lower RPV. The design supports an exemption from the requirements of 10 CFR 50.61. Reference 5.3-9 provides further details regarding austenitic stainless steel used in the lower RPV, which is resistant to the effects of neutron and thermal embrittlement.

5.3.2.1 Limit Curves

The calculation of a generic set of pressure-temperature limits at 57 EFPY uses the methodology provided in ASME BPVC Section XI, Appendix G, and the applicable limits provided in 10 CFR 50, Appendix G, as described below. Consideration of only the initial RT_{NDT} temperature is necessary because the lower portion of the RPV is not a ferritic material, and the peak fluence for the upper portion of the RPV shell is less than the 10 CFR 50, Appendix H, criteria (1.0E+17 n/cm^2(E > 1 MeV)). Therefore, no adjustment is necessary to account for fluence embrittlement effects (Reference 5.3-5). For conservatism, the 10 CFR 50, Appendix G, Table 1, limits have been applied to the final pressure-temperature limits.

The pressure-temperature limits for normal heatup and criticality conditions, normal cooldown, and inservice leak and hydrostatic (ISLH) tests including transient conditions are in Figure 5.3-2, Figure 5.3-3, and Figure 5.3-4, respectively. The corresponding numerical values are in Table 5.3-2 and Table 5.3-3. RCS pressure maintained below the limit of the pressure-temperature limit curves ensures protection against non-ductile failure. Acceptable pressure and temperature combinations for reactor vessel operation are below and to the right of the applicable pressure-temperature curves. These pressure-temperature curves include neither location correction nor instrument uncertainty. For the purpose of location correction, the allowable pressure in the pressure-temperature curves is the pressure at the RPV bottom. The reactor is not permitted to be critical until the pressure-temperature combinations are to the right of the criticality curve shown in Figure 5.3-2.

Further information on the methodology used to develop the limits is in the NuScale Technical Report, "Pressure and Temperature Limits Methodology" (Reference 5.3-6).

5.3.2.2 Operating Procedures

Section 13.5, Plant Procedures, addresses development of plant operating procedures that ensure pressure-temperature limit compliance. These procedures ensure compliance with the technical specifications during normal power operating conditions and anticipated transients.

COL Item 5.3-1: An applicant that references the NuScale Power Plant US460 standard design 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.

5.3.2.3 Pressurized Thermal Shock

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The design supports an exemption from the requirements of 10 CFR 50.61 due to the use of austenitic stainless steel in the lower RPV. The methodology described in 10 CFR 50.61 determines RT_{PTS}, which is the RT_{NDT} evaluated for the end of design life peak fluence for each beltline material. Because the lower RPV material is austenitic stainless steel, this material is exempt from impact test requirements per ASME BPVC Section III, NB-2311. As a result, the PTS screening methodology in 10 CFR 50.61 is not applicable to RPV beltline materials. ~~Further, 10 CFR 50.61 is not applicable to the upper RPV shell. The end of design life fluence value for the upper RPV shell does not exceed 1.0E+17 n/cm2 (E > 1 MeV). Therefore, 10 CFR 50.61 is not applicable to this area of the RPV.This fluence means that the entire upper RPV shell is outside the RPV beltline region per 10 CFR 50.61. Therefore, 10 CFR 50.61 PTS screening is not required for the upper RPV shell.~~ Reference 5.3-7 provides further details regarding the effects of neutron and thermal embrittlement on the austenitic stainless steel material used in the lower RPV. The requirements of 10 CFR 50.61 are addressed in the NuScale Technical Report, “Pressure and Temperature Limits Methodology” (Reference 5.3-6).

5.3.2.4 Upper-Shelf Energy

Audit Question 5.3.1.5-1, Audit Question A-5.3.2.3-1, Audit Question A-5.3.2.4-1

The evaluation of effects of neutron embrittlement on RPV materials uses Charpy ~~U~~upper-~~S~~shelf ~~E~~energy. A decrease in Charpy ~~U~~upper-~~S~~shelf ~~E~~energy level as defined in ASTM E 185-82 occurs based on fluence levels and copper content in the material. The design does not require this evaluation because the lower RPV shell is not a ferritic material. The upper RPV is ferritic and meets the requirements of ASME BPVC Section III, Subsection NV, Paragraph NB-3210, which requires a minimum Charpy upper-shelf energy of 50 ft-lb.~~, and the fluence levels for the upper RPV shell are less than the peak neutron fluence at the end of the design life of 1.0E+17 n/cm^2 (E>1 MeV) (Reference 5.3-5).~~ Reference 5.3-7 provides further details regarding the resistance to the effects of neutron and thermal embrittlement of the austenitic stainless steel material used in the lower RPV.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

Section 5.3.1, Reactor Vessel Materials describes the compatibility of the RPV design with established standards. Section 5.2.4, RCPB Inservice Inspection and Testing, and Section 5.3.1, Reactor Vessel Materials describes how the basic design of the RPV establishes compatibility with required inspections.

5.3.3.2 Materials of Construction

Section 5.2.3, RCPB Materials, and Section 5.3.1, Reactor Vessel Materials describe the reactor vessel materials of construction.

5.3.3.3 Fabrication Methods

Section 5.2.3, RCPB Materials, and Section 5.3.1, Reactor Vessel Materials describe the fabrication methods used in the construction of the reactor vessel.

5.3.3.4 Inspection Requirements

Section 5.3.1, Reactor Vessel Materials describes the nondestructive examinations performed.

5.3.3.5 Shipment and Installation

Section 5.2.3.4.2 describes the packaging, shipment, handling, and storage of the RPV.

A dry environment is maintained for RPV surfaces, both primary and secondary sides, by an installed non-chloride, non-corrosive desiccant. Humidity indicators covering a suitable range of moisture content are shipped with the RPV. Both the primary and secondary sides of the RPV ship under positive pressure. The internal atmosphere on both sides of the SG tubes are evacuated to eliminate residual moisture and filled with nitrogen having a dew point less than -20 degrees F.

In preparation for shipping the RPV, the fabricator takes appropriate foreign material exclusion measures.

There are cleanliness and contamination controls in place during handling, storage, shipping, and during installation of the RPV. Section 5.2.3.4.2, Cleaning