Response to NuScale Technical Report Audit Question

Question Number: A-5.PTLR-10

Receipt Date: 10/20/2023

Question:

NuScale provided EC-107452 R1 in response to audit Questions A-5.PTLR-2, 3, 4, and 5. Based on the information provided in EC-107452 R1:

Page 29: Geometry – Please describe how potential changes in the preliminary geometry could impact the result of the P-T limit analysis. Describe how uncertainties were factored into the analysis, and any applied conservatisms to account for the uncertainties in the preliminary geometry.

Response:

Original Response:

The response to A-5.PTLR-9 answers this question.

Updated Response based on February 8th, 2024 Clarification Call:

Combined license (COL) item 5.3-1 was updated to state the following:

> "An applicant that references the NuScale Power Plant US460 standard design will choose the final transients to generate the as-built reactor vessel pressure-temperature limits report and limiting conditions for operation. Operating procedures will ensure that pressure-temperature limits for the as-built reactor vessel are not exceeded. These procedures will be based on the limits defined in the pressure-temperature limits report and material properties of the as-built reactor vessel."

This updated wording instructs the applicant that references the NuScale Power Plant US460 standard design, that they will have to choose the final transients to generate the as-built reactor vessel pressure-temperature limits report (PTLR) and limiting conditions for operation. The revised wording ensures that the as-built vessel is used by the applicant in preparing the PTLR for the as-built reactor.

In section 5.3.2.1 of the Standard Design Approval Application (SDAA), the word “generic” was replaced by “sample” in the first sentence describing the set of calculated P-T limits. This is to clarify that the P-T limits present in Chapter 5 of the SDAA are only for illustrative purpose of the P-T methodology and do not necessarily apply to applicants.

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

Table 1.8-1: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.13-1:	An applicant that references the NuScale Power Plant US460 standard design will provide an inservice inspection program for American Society of Mechanical Engineers Class 1, 2, and 3 threaded fasteners. The program will identify the applicable edition and addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.	3.13
COL Item 4.2-1:	An applicant that references the NuScale Power Plant US460 standard design and wishes to utilize non-baseload operations will provide justification for the fuel performance codes and methods corresponding to the desired operation.	4.2
COL Item 5.2-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a certified Overpressure Protection Report in compliance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Subarticles NB-7200 and NC-7200 to demonstrate the reactor coolant pressure boundary and secondary system design contains adequate overpressure protection features, including low temperature overpressure protection features.	5.2
COL Item 5.2-2:	An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Strategic Water Chemistry Plan. The Strategic Water Chemistry Plan will provide the optimization strategy for maintaining primary coolant chemistry and provide the basis for requirements for sampling and analysis frequencies, and corrective actions for control of primary water chemistry consistent with the latest version of the Electric Power Research Institute Pressurized Water Reactor Primary Water Chemistry Guidelines.	5.2
COL Item 5.2-3:	An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for subsequent service, monitoring of the containment atmosphere for evidence of reactor coolant system leakage, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.	5.2
COL Item 5.2-4:	An applicant that references the NuScale Power Plant US460 standard design will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the American Society of Mechanical Engineers Operations and Maintenance Code, and will establish implementation milestones. If applicable, an applicant that references the NuScale Power Plant US460 standard design will identify the implementation milestone for the augmented inservice inspection program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a.	5.2
COL Item 5.2-5:	An applicant that references the NuScale Power Plant US460 standard design will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.	5.2
COL Item 5.3-1:	An applicant that references the NuScale Power Plant US460 standard design will choose the final transients to generate the as-built reactor vessel pressure-temperature limits report and limiting conditions for operation. Operating procedures will ensure that pressure-temperature limits for the as-built reactor vessel are not exceeded. These procedures will be based on the limits defined in the pressure-temperature limits report and material properties of the as-built reactor vessel.~~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

Audit Question A-5.PTLR-6
Audit Question A-5.PTLR-9
Audit Question A-5.PTLR-10

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

Audit Question A-5.PTLR-6, Audit Question A-5.PTLR-9, Audit Question A-5.PTLR-10

The calculation of a sample~~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.

Audit Question A-5.PTLR-6, Audit Question A-5.PTLR-9, Audit Question A-5.PTLR-10

COL Item 5.3-1: An applicant that references the NuScale Power Plant US460 standard design will choose the final transients to generate the as-built reactor vessel pressure-temperature limits report and limiting conditions for operation. Operating procedures will ensure that pressure-temperature limits for the as-built reactor vessel are not exceeded. These procedures will be based on the limits defined in the pressure-temperature limits report and material properties of the as-built reactor vessel.~~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).