May 15, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 404 (eRAI No. 9436) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 404 (eRAI No. 9436)," dated March 29, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9436:

- 05.03.01-5
- 05.03.01-6
- 05.03.01-7

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8G9A
Prosanta Chowdhury NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9436

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NuScale Response to NRC Request for Additional Information eRAI No. 9436

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9436
Date of RAI Issue: 03/29/2018

NRC Question No.: 05.03.01-5

Regulatory Basis:

Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," General Design Criteria (GDC) 1 requires that structures, systems, and components (SSCs) important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

GDC 4 requires that system structures and components (SSCs) important to safety shall 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 (LOCAs).

GDC 14 requires that the reactor coolant pressure boundary (RCPB) 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.

GDC 30 requires that components which are part of the RCPB 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.

GDC 31 requires that the RCPB shall 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 52.47(4) requires an applicant for a standard design certification to include, "An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility."

Below is a follow-up RAI to NuScale's letter, "NuScale Power, LLC Response to NRC Request for Additional Information No. 251 (eRAI No. 9188) on the NuScale Design Certification Application," Question 05.03.01-3 (ADAMS Accession No. ML17346A519).

The staff agrees that the revisions made to the Design Control Document (DCD) describe the locations where threaded inserts are used in the RCPB.

Question

The NuScale response states that the welding procedures and inspections will be in accordance with American Society of Mechanical Engineers (ASME) Sections III and XI, as described in DCD Tier 2, Final Safety Analysis Report (FSAR), Sections 5.3.1.4 and 5.2.3.4. The staff reviewed DCD Tier 2, FSAR, Sections 5.3.1.4 and 5.2.3.4, and while the information in the FSAR describes the welding controls, the FSAR does not describe the specific inspections that will be performed on the threaded inserts. The staff also reviewed ASME Sections III and XI and determined that the Code construction and fabrication inspection requirements are unclear on their applicability to the threaded inserts, including the seal weld.

In order to support the staff's safety finding related to the integrity of the RCPB, the staff requests that the information related to the threaded insert construction and preservice inspections that will be performed in accordance with ASME Sections III and XI be specified in the DCD. If NuScale finds that the ASME Code is not applicable to threaded inserts, then provide augmented requirements to provide a similar level of assurance based on the safety significance of the RCPB.

NuScale Response:

The American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (BPVC) Section III, paragraph NB-5271, Welded Joints of Specially Designed Seals, requires that fabrication examinations of threaded insert seal welds be performed using magnetic particle or liquid penetrant methods. ASME BPVC Section XI requires no inservice inspections for seal welds. However, the NuScale Final Safety Analysis Report (FSAR) Table 6.2-3 specifies that an augmented inservice (VT-1) examination be performed for the containment (CNV) threaded insert seal weld. FSAR Table 5.2-6 requires the same examination for the reactor pressure vessel (RPV) threaded insert seal welds. Section XI, subarticle IWB-2200 states that the threaded inserts are required to have a preservice, VT-1, examination equivalent to the inservice examination performed.

The threaded inserts are to have the appropriate material fabrication examination as defined by ASME BPVC Section III, subarticle NB-2580, Examination of Bolts, Studs and Nuts. Per the requirement of ASME BPVC Section III, paragraph NB-5282 the threaded inserts are required to have a preservice examination performed as defined by ASME BPVC Section XI, Table IWB-2500-1.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9436
Date of RAI Issue: 03/29/2018

NRC Question No.: 05.03.01-6

NuScale's response contained a general description of the threaded insert design and stated that the threads will carry the mechanical loads during normal and off-normal operations, including emergency core cooling system (ECCS) actuation. In order to maintain the RCPB, the threaded inserts need to be designed to withstand the design stresses of the RCPB. As written, the FSAR does not describe the general design of the threaded inserts or their design requirements.

In order to support the staff's safety finding on the design of the RCPB, the staff requests that the general description as well as the design requirements of the threaded inserts be included in the DCD. For example, a statement describing that the threaded inserts are externally and internally threaded, as well as a statement that the threaded inserts are designed to meet the mechanical loads from during normal and off-normal operations, including ECCS actuation.

NuScale Response:

As referenced in the NRC request for additional information (RAI), NuScale provided a description of the threaded insert design in a letter entitled, "NuScale Power, LLC Response to NRC Request for Additional Information No. 251 (eRAI No. 9188) on the NuScale Design Certification Application," dated December 12, 2017, Question 05.03.01-3 (ADAMS Accession No. ML17346A519). In this response NuScale stated that the threaded inserts used for threaded fasteners are externally threaded in addition to being internally threaded such that the inserts are threaded into the associated base metal. As such, the external threads on the inserts and internal threads in the flange bolt holes are designed to carry mechanical loads during normal and off-normal operations, including ECCS actuation.

This information has been added to the FSAR section 5.3.1.7 to address this reactor pressure vessel threaded inserts. FSAR section 6.2.1.1.2 is also modified to address the containment vessel threaded inserts. See FSAR markup provided below.

Impact on DCA:

FSAR Sections 5.3.1.7 and 6.2.1.1.2 have been revised as described in the response above and as shown in the markup provided in this response.

The capsules are inside capsule holders that are attached to the outside of the core barrel at mid-height of the core. The capsules are positioned to achieve a lead factor of approximately 2.5. The four capsules are positioned approximately 90 degrees apart around the circumference of the core support assembly. Figure 5.3-2 shows the core barrel horizontal cross-section and the location of the four capsule holders and capsule elevation on the core barrel.

RAI 05.03.02-2

The neutron flux and fluence ~~calculations~~calculation methods are consistent with the guidance of RG 1.190 ~~and are~~with exceptions as described in NuScale Technical Report TR-0116-20781, "Fluence Calculation Methodology and Results" (Reference 5.3-7).

The transition temperature upper shelf energy changes are calculated in accordance with RG 1.99, and are shown in Table 5.3-8, Table 5.3-9, and Table 5.3-10. Section 5.3.2 provides further information.

COL Item 5.3-3: A COL applicant that references the NuScale Power Plant design certification will describe the reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.

5.3.1.7 Reactor Vessel Fasteners

The RPV closure studs, nuts, and washers use SB-637 Alloy 718, instead of low alloy steels such a SA-540 Grade B23 or B24. The selection of Alloy 718 over traditional low alloy steels is to prevent general corrosion when the bolting is submerged during the plant startup and shutdown process. Because of its resistance to general corrosion, the concerns addressed by RG 1.65, Revision 1, position 2(b) do not apply to Alloy 718. Alloy 718 is an austenitic, precipitation-hardened, nickel-based alloy permitted for bolting materials by ASME BPVC Section III (Reference 5.3-1), Subsection NB-2128.

Furthermore, because Alloy 718 is not a ferritic material, the fracture toughness requirements of NB-2333 are not required. Further information is provided in Section 3.13.

RAI 05.03.01-3, RAI 05.03.01-6

Threaded inserts are used for RPV threaded fasteners except for the main RPV flange studs and steam generator inlet flow restrictor hardware. The threaded inserts used for threaded fasteners are externally threaded in addition to being internally threaded such that the inserts are threaded into the associated base metal. As such, the external threads on the inserts and internal threads in the flange bolt holes are designed to carry mechanical loads during normal and off-normal operations, including ECCS actuation. See Table 5.2-4 for threaded insert materials and applicable specifications.

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

Analyses

- enclosure of the RPV, RCS and associated components
- containment of fission product releases from the RCPB
- containment of the postulated mass and energy releases (LOCA and non LOCA) inside containment
- operation of the ECCS by the retention of reactor coolant and the transfer of sensible and core heat to the UHS.

The CNV is a compact, steel pressure vessel that consists of an upright cylinder with top and bottom head closures. The CNV is partially immersed in a below-grade reactor pool that provides a passive heat sink and is absent of internal sumps or subcompartments that could entrap water or gases. The CNV and the reactor pool are housed within a Seismic Category 1 Reactor Building (RXB).

Vertical and lateral support for the CNV is provided by the RXB via a support skirt at the bottom of the vessel. Lateral support for the CNV is provided by the reactor compartment walls through the lateral support lugs on the upper CNV shell. The CNV houses and supports the RPV and associated piping systems and valves.

RAI 05.03.01-6, RAI 06.02.01-3

Threaded inserts are used for all CNV pressure boundary threaded fasteners except for the main CNV flange studs. See Table 6.1-1 for threaded insert materials and applicable specifications. The threaded inserts used for threaded fasteners are externally threaded in addition to being internally threaded such that the inserts are threaded into the associated base metal. As such, the external threads on the inserts and internal threads in the flange bolt holes are designed to carry mechanical loads during normal and off-normal operations, including ECCS actuation. See Section 5.2.3.4 for applicable welding procedures and inspections for threaded insert welds during fabrication and installation.

Table 6.2-1 provides a list of containment design parameters, operating parameters and information relevant to the CNV. Containment general arrangement drawings are provided in Figure 6.2-2a and Figure 6.2-2b.

During normal operation, the CNV is maintained in a partially evacuated dry condition. However, there are specific operational conditions that involve the presence of water in the CNV (e.g., primary and secondary system leakage, ECCS actuation, component cooling system leakage or module disassembly and refueling).

Maintaining the containment at a vacuum has benefits for both normal operation and post mass and energy release events. A vacuum precludes the need for thermal insulation inside containment because convective heat transfer from the reactor vessel is minimized during normal reactor operation. Combustible gas control is further described in Reference 6.2-3.

The NPM is designed to be transported to and from the refueling area without loss of reactor coolant inventory, and is refueled in a partially flooded condition,

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9436
Date of RAI Issue: 03/29/2018

NRC Question No.: 05.03.01-7

NuScale's response stated that the external threads of the threaded inserts are subject to the same loads as the internal threads and associated bolting. Therefore, the stress analysis of the threaded inserts is different than the analysis of the associated bolting. The threads and thickness of the threaded insert impact their strength. During the staff's review, there is no information provided related to the stress analysis of the threaded inserts themselves.

In order to support the staff's safety finding related to the integrity of the RCPB, the staff requests the stress analysis showing that the design of the various sized threaded inserts can meet the design requirements of the RCPB. This analysis should also include information related to the installation of the threaded inserts to ensure that they are not over or under torqued. This information may be provided in the electronic reading room.

NuScale Response:

In a clarification conference call held with the NRC on March 26, 2018, NuScale described that the stress analysis for the Reactor Pressure Vessel (RPV) and Containment Vessel (CNV) threaded inserts are not currently available. NuScale also recognizes that these stress analyses are required to be performed to demonstrate compliance with the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (BPVC) design requirements. The stress analysis for the RPV and CNV threaded inserts must meet the limits specified ASME BPVC Section III, NB-3230. Bolting design for each flange meets ASME Code Class 1 design criteria. Class 1 bolted joints must meet NB-3230 which includes determination of the number of bolts and preload to maintain a tight joint (NB-3231). In addition, detailed fatigue analyses are required per NB-3232.3. To determine the minimum number of bolts and preload Nonmandatory Appendix E is used per NB-3231(a). The threaded inserts do not need to be torqued into position, but simply turned into the treaded hole. The threaded insert then has a seal weld located at the flange surface, protecting the external threaded area to prevent corrosion of the base metal from RCS fluid and preventing the threaded insert from backing out of position.

Completion of the RPV and CNV threaded insert stress analyses is a future activity that is captured in the NuScale DCA as COL item 3.9-2. COL item 3.9-2 states in part:

> *A COL applicant that references the NuScale Power Plant design certification will develop design specifications and design reports in accordance with the requirements outlined under American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (Reference 3.9-1). ....*

The stress analyses for the bolting materials is covered by the COL applicant providing the design specifications and design reports meet the requirements of the ASME BPVC, Section III. These design specifications and design reports are subject to NRC verification.

Impact on DCA:

There are no impacts to the DCA as a result of this response.