August 02, 2017

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. 75 (eRAI No. 8904) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 75 (eRAI No. 8904)," dated June 28, 2017

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. 8904:

- 04.05.02-1
- 04.05.02-3

The response to RAI Question 04.05.02-2 will be provided by August 28, 2017.

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8904

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8904

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

eRAI No.: 8904
Date of RAI Issue: 06/28/2017

NRC Question No.: 04.05.02-1

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.

NuScale DCD Section 4.5.2.5 states that washers in the RVI upper riser assembly are made from Alloy 718. There is no corresponding text regarding Alloy 718 nuts. 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 and nuts in the RVI upper riser assembly and revise the DCD accordingly.

NuScale Response:

There are no Alloy 718 bolts or nuts in the Reactor Vessel Internals (RVI). The Alloy 718 washers interface with the Upper Alignment Hanger Threaded Structural Fasteners which are threaded into the Upper Riser Hanger Ring. The Upper Alignment Hanger Threaded Structural Fasteners and the Upper Riser Hanger Ring are type 304 stainless steel. The washers in the RVI upper riser assembly are are not in tension and as a result are not susceptible to stress corrosion cracking. Section 4.5.2.5 will be revised to state this as shown in the DCA markup.

Impact on DCA:

The last paragraph in Section 4.5.2.5 has been revised as described in the response above and as shown in the markup provided in this response.

content and limited to a ferrite number of 5FN to 20FN in accordance with RG 1.31 and ASME BPV Code, Section III, Paragraph NG-2433. Carbon content of austenitic stainless steel weld filler metals is limited to no more than 0.03 wt%.

Tools for abrasive work such as grinding, polishing, or wire brushing are not permitted to be contaminated by previous usage on ferritic carbon steel or other materials that could contribute to intergranular cracking or stress-corrosion cracking.

Section 5.2.3 describes the controls used to minimize the introduction of potentially harmful contaminants including chlorides, fluorides, and low melting point alloys on the surface of austenitic stainless steel components. In accordance with RG 1.44, cleaning solutions, processing equipment, degreasing agents, and other foreign materials are removed during processing prior to elevated temperature treatments. Acid pickling is avoided on stainless steel and not used on sensitized austenitic stainless steel.

4.5.2.5 Other Materials

Materials exposed to primary reactor coolant are corrosion-resistant stainless steels, nickel-based alloys, and, to a limited extent, cobalt-based alloys. These materials are selected from materials proven in light-water reactor operation and for their compatibility with the reactor coolant as specified in ASME BPV Code, Section III, Paragraph NG-2160 and Subsubarticle NG-3120.

Precipitation hardenable stainless steel 17-4 PH, Grade 630 procured in the H1100 condition is used in portions of the core support locking assemblies. The minimum precipitation hardening heat treatment maintains a minimum tempering temperature of 1100°F for at least 4 hours per ASME BPV Code, Section II, Part A, Material Specification SA-564.

A cobalt-based alloy, Stellite 3 casting, or a qualified low-cobalt or cobalt-free alloy is used on wear surfaces of the core support locking assemblies. Use of a low cobalt or cobalt-free alloy may be substituted if the wear and corrosion resistance is qualified by testing.

Nickel-based Alloy 690 is used for RVI materials as listed in Table 4.5-2 due to excellent corrosion resistance in PWR primary coolant as documented the in the EPRI materials reliability program reports for Alloy 690 (Reference 4.5-4 and Reference 4.5-5). Alloy 600 base metal and Alloy 82/182 weld metal are not used in the RVI and core support structure design.

The seismic Belleville washers that support the lower core plate are nickel-based SB-637, Grade 688, Type 3. This material is allowed for use as a core support material by Code Case N-60-5.

RAI 04.05.02-1

Washers used in the RVI upper riser assembly are nickel-based Alloy 718. These washers utilize the same final solution annealing and precipitation-hardening treatment process as used for Alloy 718 threaded fasteners. Refer to Section 3.13.1 for further discussion regarding the annealing and precipitation-hardening treatment for Alloy 718 materials. The RVI upper riser assembly washers are not in tension, and as a result, they are not susceptible to stress corrosion cracking.

4.5.3 References

4.5-1 American Society for Testing and Materials, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” ASTM A262-15, West Conshohocken, PA, September 2015.

4.5-2 American Society of Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facility Applications," ASME NQA-1-2008/1a-2009 Addenda, New York, NY.

4.5-3 Electric Power Research Institute, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)," EPRI Technical Report 1012081, Palo Alto, CA, December 2005.

4.5-4 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 Technical Report 1009801, Palo Alto, CA, U.S. Department of Energy, Washington, DC, March 2004.

4.5-5 Electric Power Research Institute, "Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloy 690 in Pressurized Water Reactors (MRP-258), EPRI Technical Report 1019086, Palo Alto, CA, U.S. Department of Energy, Washington, DC, August 2009.

NuScale Power

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

eRAI No.: 8904
Date of RAI Issue: 06/28/2017

NRC Question No.: 04.05.02-3

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.

NuScale DCD Section 4.5.2 does not address the potential for crevice corrosion. Crevice corrosion has proven a significant factor for internals aging management in the operating fleet.

The staff requests that NuScale provide any assessment pertaining to the potential for crevice corrosion in core support and reactor internal components. The staff notes that significant discussion of this topic was included in DCD Subsection 5.4.1.2 for steam generator components.

The staff further requests that a discussion of crevice corrosion be included in DCD Subsection 4.5.2 unless the applicant can establish that the potential for crevice corrosion for the subject components is too low to merit inclusion.

NuScale Response:

The potential for crevice corrosion in the NuScale design was considered. As noted in the RAI, significant discussion was provided for steam generator (SG) components. However, the discussion provided within the Final Safety Analysis Report Section 5.4.1.2 for SG components was based primarily on historical precedence for PWRs (SG tube to support crevices have historically been susceptible to crevice corrosion) and first-of-a-kind considerations (although not susceptible to crevice corrosion, the SG and SG tube support design is unique and therefore warranted additional clarification). With respect to any other reactor vessel internal components in the NuScale design, the consideration for crevice corrosion potential determined the level of risk was extremely low. Likewise, the technical basis for the extremely low risk was identical to other PWR designs, therefore consistent with other applicants, no additional discussion of crevice corrosion was provided in the FSAR. The basis for this conclusion is provided below.

The electrochemical corrosion potential (ECP) of BWR water was measured to be between 100 and 250 mV_{SHE} without hydrogen (Reference 1). Injecting small amounts of hydrogen to BWRs is highly effective in suppressing radiolysis and the ECP, thereby suppressing stress corrosion cracking (SCC) in BWR internals (References 2 and 3). Hydrogen injection, with levels below PWR primary water, has proven effective in suppressing SCC in BWR reactor internals. The dissolved hydrogen concentration in PWR water is a magnitude higher than that of BWR hydrogen water chemistry. PWR reactor internals have not experienced intergranular stress-corrosion cracking (IGSCC) like BWR reactor internals primarily due to difference in water chemistry. Without hydrogen injection, water disassociation due to radiation (radiolysis) leads to elevated level of oxygen and hydrogen peroxide, and elevated ECP in BWR water. As a result, dissolved oxygen level is undetectable in PWR reactor internals (Reference 1). The estimated ECP in PWR water is between -800 to -600 mV_{SHE}. Given the low ECP, even sensitized austenitic stainless steels next to welds in PWR reactor internals (built prior to RG 1.44 Rev 0, 1973) have not developed IGSCC after 40 years of operation. In contrast, similar locations in BWR reactor internals often showed IGSCC within a decade of startup.

NuScale is a PWR and the NuScale RCS chemistry follows Electric Power Research Institute (EPRI) primary water chemistry guidelines, as stated in FSAR Tier 2 Section 5.2.3.2.1. Per FSAR Table 5.2-5, 25-50 cc/kg hydrogen is injected to keep dissolved oxygen below 0.005 ppm; the level of chloride or fluoride is kept below 0.05 ppm. Furthermore, FSAR Section 5.2.3.2.1 states that hydrazine is added to scavenge dissolved oxygen at low temperature during plant startup.

All NuScale reactor internals materials have demonstrated satisfactory performance in PWR primary water. NuScale reactor internals materials are listed in FSAR Table 4.5-2. The base metals are Type 304, Type 304L, 17-4PH H1100, XM-19, Alloy 718, Alloy X-750, Alloy 690, and Stellite 3. The weld filler metals are SFA-5.4, SFA-5.9, and SFA-5.22 unstabilized 3XX austenitic stainless steels with their carbon content limited to a maximum of 0.03%. Any Type 304 or 304L exposed to 800-1500°F subsequent to final solution anneal are limited to 0.03% maximum carbon. The heat treatments for nickel-base Alloy X-750, Alloy 718, and Alloy 690 are optimized to maximize SCC resistance.

Crevice corrosion has not been a factor for PWR reactor internals aging management. MRP-175 "PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values" (Reference 4) was issued in 2005 to provide the screening criteria and the technical bases for age-related degradations for PWR reactor internals. Crevice corrosion is not a factor for any MRP-175 aging degradation mechanisms. MRP-175 (Section C.3) states that "*general corrosion, induced from localized environmental conditions in a crevice (e.g., a bolted connection), is not an issue in oxygen-free environments.*" MRP-227-Revision 0 (Reference 5), based on the MRP-175, was issued in 2009 to provide the aging management program for US PWR reactor internals. The NRC accepted version MRP-227-A was issued in 2011 (Reference 6).

The MRP-175 position on crevice corrosion is consistent with NUREG-1801 Revision 2 "Generic Aging Lessons Learned (GALL) Report", issued in 2010 (Reference 7). The GALL Revision 2 captured both US and international operating experience. GALL Revision 2 mentioned crevice corrosion in PWR internals in Chapter IV, Items IV.B2.RP-24, IV.B3.RP-24, and IV.B4.RP-24 for

Westinghouse, CE, and B&W designed PWRs, but credited “PWR water chemistry” program as the aging management program to prevent crevice corrosion. The water chemistry program, according to GALL Revision 2, follows EPRI PWR primary water chemistry guidelines.

Although NuScale coolant flow is lower than for other PWR designs, the flow is not stagnant and flow rates are well within the turbulent flow range; RCS flow rates are provided in Tier 2, Table 5.1-2. All PWR reactor internals or RCS systems have some lower flow areas that contain crevices. Aside from the interface between SG tubes and supports, the NuScale design does not include any geometry within the reactor pressure vessel and reactor internal components which contain crevices which significantly differs from typical PWRs. As crevice corrosion has not been an issue for PWR RCS systems, including low flow areas in reactor internals, pressurizer, CRDM housing, or valves, the same conclusion is indicated for the NuScale design. A number of bolting components made of Alloy A-286, Alloy X-750, and cold worked Type 316 or Type 316Ti have experienced SCC or irradiation assisted stress-corrosion cracking (IASCC) in PWR internals. These cracked or uncracked bolts have been removed and carefully examined. The root cause was a combination of susceptible materials and very high stress levels. Crevice corrosion was not identified in the removed parts or considered a significant factor in the failures.

In contrast to crevice corrosion in steam generators (low carbon steel exposed to PWR secondary water), there has been no discussion of crevice corrosion in reactor vessel internals (DCD Chapter 4.5) by any PWR DCAs that have been docketed with NRC or approved by NRC. This includes AP1000 DCD Revision 19, US EPR DCD Revision 5, APWR DCD Revision 4, and APR1400 DCD Revision 0. This is consistent with the industry consensus that the potential for crevice corrosion of PWR reactor internals is too low to merit inclusion in DCD Chapter 4.5. As discussed above, there is no basis to suggest that NuScale reactor internals will be any different. Therefore, the potential for crevice corrosion of reactor internals is too low to merit inclusion in NuScale FSAR Chapter 4.5.2.

References:

1. P. Scott, “A Review of Irradiation Assisted Stress Corrosion Cracking”, Journal of Nuclear Materials, 211 (1994), pp. 101-122.
2. P.L. Andresen, “Irradiation-Assisted Stress-Corrosion Cracking,” in Stress Corrosion Cracking - Materials Performance and Evaluation, ed. R.H. Jones (American Society for Metals, Metals Park, OH, 1992), pp. 182-210.
3. N. Saito, E. Kikuchi, H. Sakamoto, J. Kuniya, and S. Suzuki, “Susceptibility of Sensitized Type 304 Stainless Steel to Intergranular Stress Corrosion Cracking in Simulated Boiling-Water Reactor Environments,” Corrosion, Vol. 53, No. 7, 1997, pp. 535-545.
4. Electric Power Research Institute, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)," 1012081, 2005.
5. Electric Power Research Institute, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (MRP-227-Rev. 0), 1016596, 2008.
6. Electric Power Research Institute, "Materials Reliability Program: Pressurized Water

Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)."1022863, 2011
7. NUREG-1801, Rev 2, Generic Aging Lessons Learned (GALL) Report, December 2010.

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