RAIO-1217-57679

December 14, 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. 214 (eRAI No. 8858) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 214 (eRAI No. 8858)," dated September 01, 2017
2. NuScale Power, LLC Response to NRC Request for Additional Information No. 214 (eRAI No. 8858) on the NuScale Design Certification Application, dated October 27, 2017 (ML17300B429)

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 Question from NRC eRAI No. 8858:

- 03.08.02-6

The response to twelve RAI Questions were previously provided in Reference 2. This completes all responses to eRAI No. 8858.

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 Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Marieliz Vera, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8858

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8858

NuScale Power

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

eRAI No.: 8858
Date of RAI Issue: 09/01/2017

NRC Question No.: 03.08.02-6

10 CFR 52.47 requires the design certification applicant to include a description and analysis of the structures, systems, and components with sufficient detail to permit understanding of the system designs. RG 1.216 C.1.k states that the details of the analysis and results should be submitted in report form with:

- calculated static pressure capacity;
- dynamic pressure capacity, if applicable (static pressure capacity reduced to account for dynamic amplification effects);
- associated failure modes;
- criteria governing the original design and criteria used to establish failure;
- analysis details and general results, which include (1) modeling details, (2) description of computer code(s), (3) material properties and materia modeling, (4) loading and loading sequences, (5) failure modes, and (6) interpretation of results, with all assumptions made in the analysis and test data (if relied upon) clearly stated and technically justified; and,
- appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure.

The staff is not aware of a report submitted that includes the necessary information described in RG 1.216 C.1.k and DCD Section 3.8.2.4.5 appears to only discuss the failure mode criteria of RG 1.216 C.1.k, but does not cover the other criteria. Clarify how the application is meeting RG 1.57 and RG 1.216 for analysis methods.

NuScale Response:

Technical report TR-0917-56119, "CNV Ultimate Pressure Integrity," was submitted to address the details of the predicted containment internal pressure capacity above design pressure, as specified in RG 1.216 C.1.k. FSAR Tier 2 Section 3.8.2 and Table 1.6-2 have been updated to incorporate the new technical report.

Impact on DCA:

FSAR Tier 2 Section 3.8.2 and Table 1.6-2 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 03.08.02-6, RAI 13.05.02.01-1S2

Table 1.6-2: NuScale Referenced Technical Reports

Report Number	Title	FSAR Section
TR-0116-20781	Fluence Calculation Methodology and Results	4.3, 5.3
TR-0316-22048	Nuclear Steam Supply System Advanced Sensor Technical Report	7.1, 7.2
TR-0416-48929	NuScale Design of Physical Security Systems	9.5, 13.6, 14.2, 14.3
TR-0516-49084	Containment Analysis Methodology	6.2
TR-0616-49121	NuScale Instrument Setpoint Methodology Technical Report	7.0, 7.2
TR-0716-50424	Combustible Gas Control	3.8, 6.2
TR-0716-50439	Comprehensive Vibration Assessment Program (CVAP) Technical Report TR-0716-50439	3.9, 14.2
TR-0816-49833	Fuel Storage Rack Analysis	3.7, 3.8, 9.1
TR-0816-50796	Loss of Large Areas Due to Explosions and Fires Assessment	20.2
TR-0816-50797	Mitigation Strategies for Extended Loss of AC Power (ELAP) Event	20.1
TR-0816-51127	NuFuel HTP2 Fuel and Control Rod Assembly Designs	4.2
TR-0916-51299	Long-Term Cooling Methodology	5.4, 6.2, 6.3, 15.0
TR-0916-51502	NuScale Power Module Seismic Analysis	3.7, 3.12
TR-1015-18177	Pressure and Temperature Limits Methodology	5.3
TR-1016-51669	NuScale Power Module Short-Term Transient Analysis	3.8
TR-1116-51962	NuScale Containment Leakage Integrity Assurance	6.2
TR-1116-52065	Effluent Release Methodology Technical Report	11.1, 11.2, 11.3
RP-0215-10815	Concept of Operations	18.7
RP-0316-17614	Human Factors Engineering Operating Experience Review Results Summary Report	18.2
RP-0316-17615	Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report	18.3
RP-0316-17616	Human Factors Engineering Task Analysis Results Summary Report	18.4
RP-0316-17617	Human Factors Engineering Staffing and Qualifications Results Summary Report	18.5
RP-0316-17618	Human Factors Engineering Treatment of Important Human Actions Results Summary Report	18.6
RP-0316-17619	Human Factors Engineering Human-System Interface Design Results Summary Report	18.7
RP-0516-49116	Control Room Staffing Plan Validation Results	18.5
RP-0914-8534	Human Factors Engineering Program management Plan	18.1
RP-0914-8543	Human Factors Verification and Validation Implementation Plan	18.1
RP-0914-8544	Human Factors Engineering Design Implementation Implementation Plan	18.11
RP-1215-20253	Control Room Staffing Plan Validation Methodology	18.5
TR-1117-57216	NuScale Generic Technical Guidelines	13.5
TR-0917-56119	CNV Ultimate Pressure Integrity	3.8

defined by ASME Code, Section III, Paragraph NB-1132.1(a), and use the rules of ASME Code, Section III. The lateral support lugs are constrained by corbels located on the NPM bay walls. The loads and load combination discussed in Section 3.8.2.3 are used to evaluate the CNV support lug. Stress and fatigue results are evaluated in accordance with Subarticle NB-3200 limits.

3.8.2.4.3 Containment Vessel Lower Support

The bottom of the CNV is supported vertically and laterally by the CNV support skirt. The CNV support skirt is an ASME Code Class MC support that is constructed as an ASME Code, Section III Class 1 support in accordance with the requirements of Article NF-4000. The support skirt is located below the CNV bottom head and includes two parts that are welded together: the support bearing flange and the support skirt ring. Lateral restraint is provided by contact with a metal ring called the passive skirt support, which is attached to the reactor pool floor. Vertical support is provided by bearing on the reactor pool floor. The loads and load combination discussed in Section 3.8.2.3 are used to evaluate the CNV support skirt. Stress and fatigue results are evaluated in accordance with ASME Code, Section III, Subarticle NF-3200 limits.

3.8.2.4.4 Containment Vessel Reactor Pressure Vessel Supports

Internal to the CNV, the RPV is supported by the CNV using RPV upper support ledges that are located in the CNV upper section. The RPV supports are connected to the CNV reactor pressure vessel upper support ledges located on the inner wall of the CNV by studs at each connection. The loads and load combination discussed in Section 3.8.2.3 are used to evaluate the CNV reactor pressure vessel upper support ledge. Stress and fatigue results are evaluated in accordance with ASME Code, Section III, Subarticle NF-3200 limits.

3.8.2.4.5 Containment Vessel Ultimate Capacity

RAI 03.08.02-6

A series of non-linear (plastic) 3-dimensional finite element analysis were performed to determine the ultimate pressure capacity of the CNV; the analyses conform to the guidance provided in Appendix A of NUREG/CR-6906 (Reference 3.8.2-2). The failure criteria that determine the ultimate pressure capacity of the CNV are based on guidance provided in RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure," Revision 0. Technical report TR-0917-56119, CNV Ultimate Pressure Integrity (Reference 3.8.2-7), addresses the details of the predicted containment internal pressure capacity above design pressure. The CNV is assumed to fail when one of the following criteria is met:

A. A maximum global membrane strain away from discontinuities of 1.5 percent is reached.

B. Loss of bolt preload occurs at any bolted CNV opening.

The shop hydrostatic tests of the CNV are witnessed by an authorized nuclear inspector and a NuScale inspector.

No leakage indications at the examination pressure are acceptable.

3.8.2.8 References

3.8.2-1 NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," Draft Report for Comment

3.8.2-2 NUREG/CR 6906, "Containment Integrity Research at Sandia National Laboratories - An Overview," July 2006

RAI 03.09.03-1

3.8.2-3 ~~NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36," July 1980~~ANSI N14.6-1993 "for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More"

3.8.2-4 NuScale Technical Report TR-0716-50424-P, Rev 0, “Combustible Gas Control”

3.8.2-5 ANSYS Computer Program, Release 15.0, October 2013. ANSYS Incorporated, Canonsburg, Pennsylvania

RAI 08.01-1S1

3.8.2-6 IEEE Std. 317-1983, Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations

RAI 03.08.02-6

3.8.2-7 NuScale Technical Report TR-0917-56119, Rev 0, "CNV Ultimate Pressure Integrity"