RAIO-0617-54545

June 19, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 11 (eRAI No. 8759) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 11 (eRAI No. 8759)," dated April 25, 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 Question from NRC eRAI No. 8759:

- 12.02-1

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8759

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8759

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

eRAI No.: 8759
Date of RAI Issue: 04/25/2017

NRC Question No.: 12.02-1

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced during operations and the means for controlling and limiting radiation exposures. 10 CFR Part 20 requires the use of engineering features to control and minimize the amount of radiation exposure to members of the public and occupational workers, from both internal and external sources. 10 CFR 50.49(e)(4) requires applicants to identify the type of radiation and the total dose expected during normal operation over the installed life of the equipment. General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50 requires applicants to ensure that structures, systems, and components (SSC) important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation.

Design Specific Review Standard (DSRS) section 12.2 Acceptance Criteria states in part, that the shielding and ventilation design fission product source terms will be acceptable if developed using the bases of 0.25-percent (%) fuel cladding defects (aka design basis failed fuel fraction value) for pressurized-water reactors (PWRs) or the reactor coolant system (RCS) isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to operation for a full fuel cycle at the technical specification (TS) limits for halogens (I-131 dose equivalent) and noble gases (Xe-133 dose equivalent). DSRS Chapter 11, Sections 11.1, 11.2, and 11.3; NUREG-0800 “Standard Review Plan” (SRP) Chapter 3, Section 3.11; and Branch Technical Positions (BTPs) 11-5 and 11-6 also provide guidance on fuel leakage (failed fuel fraction) assumptions.

NuScale Design Control Document (DCD), Tier 2, Revision 0, Chapter 11, Table 11.1-2: “Parameters Used to Calculate Coolant Source Terms,” shows that the design basis failed fuel fraction value is 0.028% (which is proposed as the basis for determining plant radiation shielding, zoning, ventilation design, equipment qualification (EQ) dose calculations, etc). NuScale Technical Report TR-1116-52065 Revision 0 “Effluent Release (GALE Replacement) Methodology and Results,” describes the derivation of the 0.028% value. In essence, the method determined the **average** fuel failure fraction over a multiple year period (0.0028%),

and then multiplied that value by a factor of 10, to arrive at the proposed 0.028% failed fuel fraction value to be used for shielding and ventilation system design, etc. The value of 0.028%, is less than one tenth of the TSs section 3.4.8 RCS Specific Activity Limit of 0.2 micro Ci/gram Dose Equivalent Iodine (DEI).

NuScale used data from "Benchmarking of GALE-09 Release Predictions Using Site Specific Data from 2005 to 2010," PNNL- 22076, dated November 2012, to determine a realistic fission product source term to be used for the evaluation of normal effluent releases. This report examined the reported average fuel failure fraction from 2005 to 2010. DCD subsection 11.1 states "(t)he design basis source term assumes a conservative value of equivalent fuel defects an order-of-magnitude greater than the realistic coolant source term. This results in a design basis failed fuel fraction that is ten times greater than the realistic failed fuel fraction." In the view of the staff, an empirical survey of operational experience regarding failed fuel experiences at operating reactor facilities does not constitute a safety case for the proposed NuScale failed fuel fraction of 0.028%. Further, the use of an average failed fuel experience, without a corresponding Technical Specification limit for DEI and Dose Equivalent Xenon, does not comport with established licensing practices used by the staff for evaluating the acceptability of the proposed design bases for shielding and ventilation systems. The value proposed by NuScale is not a bounding design bases value, since it is about a factor of 5 less than the amount of fuel cladding defects that occurred at one plant in 2009, which had been designed for normal system operation with clad defects in fuel rods generating 1% of rated core thermal power. The NuScale proposed value is comparatively less than that used by other plant designs, including passive designs, and is less than the historically used 0.25% failed fuel fraction, is not bounding with respect to operational data within the stated time frame, and is not conservative. Based on the information provided, this lower failed fuel fraction would not meet the acceptance criteria in NuScale DSRS Chapter 11, Sections 11.1, 11.2, 11.3, and Chapter 12, Section 12.2; NUREG-0800 Chapter 3, Section 3.11; and BTPs 11-5 and 11-6. (Note: for shielding and ventilation system design, the value is about a factor of 10 less than that used by the NRC to license plants under 10 CFR Parts 50 and 52, and about a factor of 10 less than the value currently specified in NuScale DCD TS Part 4, Volume 1, Section 3.4.8.)

Based upon the docketed information, the NRC staff is unable to determine whether the NuScale design is adequate to protect members of the public and occupational workers from exposure to radiation and protection of SSCs important to safety. The staff requests the applicant to provide additional information (e,g., requisite analyses and safety margin evaluations) that clearly demonstrates, through the implementation of the shielding and ventilation system acceptance criteria stated in DSRS 12.2 and DSRS 11.1, that the NuScale design provides reasonable assurance that the public and occupational workers will be protected from exposure to radiation. Explain why the proposed assumed failed fuel fraction is appropriately conservative for the purposes of evaluating personnel doses, radiation protection design features, radwaste handling system capacities, and equipment qualification

analyses. Explain why adopting a technical specification limit that bounds the newly proposed failed fuel fraction is not warranted as discussed in the DSRS acceptance criteria stated above.

NuScale Response:

In the subject RAI the NRC Staff states that in their view, an empirical survey of operational experience regarding failed fuel experiences at operating reactor facilities does not constitute a safety case for the proposed NuScale failed fuel fraction of 0.028%.

Further, the use of an average failed fuel experience, without a corresponding Technical Specification limit for DEI and Dose Equivalent Xenon, does not comport with established licensing practices used by the staff for evaluating the acceptability of the proposed design bases for shielding and ventilation systems.

The value proposed by NuScale is not a bounding design bases value, since it is about a factor of 5 less than the amount of fuel cladding defects that occurred at one plant in 2009, which had been designed for normal system operation with clad defects in fuel rods generating 1% of rated core thermal power.

The NuScale proposed value is comparatively less than that used by other plant designs, including passive designs, and is less than the historically used 0.25% failed fuel fraction, is not bounding with respect to operational data within the stated time frame, and is not conservative.

The staff requests the applicant to provide additional information (e.g., requisite analyses and safety margin evaluations) that clearly demonstrates, through the implementation of the shielding and ventilation system acceptance criteria stated in DSRS 12.2 and DSRS 11.1, that the NuScale design provides reasonable assurance that the public and occupational workers will be protected from exposure to radiation.

Explain why the proposed assumed failed fuel fraction is appropriately conservative for the purposes of evaluating personnel doses, radiation protection design features, radwaste handling system capacities, and equipment qualification analysis.

NuScale Response (part 1):

NuScale assumes a normal operation failed fuel fraction of 0.028%. In determining a reasonable failed fuel fraction, NuScale first evaluated the regulatory basis and history for the failed fuel fraction values cited in Design Specific Review Standard (DSRS) 12.2, Standard Review Plan (SRP) 11.1, 11.2, and 11.3 as well as Branch Technical Positions (BTPs) 11-5 and

11-6. SRPs 11.1, 11.2, 11.3 and 12.2 were first issued in 1975 with the earliest revision available on ADAMS from 1978. The current revisions of these SRPs were issued in 2016. BTPs 11-5 and 11-6 were first issued in 2007 with the current revisions dated 2016. The 0.25% and 1.0% failed fuel fraction cited in the SRP regulatory documents has not changed since the 1978 editions. As shown in Figure 5-2 and Table A-6 of the NuScale Effluent Release Technical Report, TR-1116-52065, zirconium alloy fuel failure rates were in the range of 0.02 to 1 percent during the 1970s. Data cited in the technical report for the time period up to 1985 shows an average fuel failure rate of 0.026%. Therefore, a factor of 10 applied to this average result is approximately the 0.25% value cited in regulatory guidance. There were individual nuclear power plants in the 1970s that experienced greater than 0.25% fuel failure, but the selection of 0.25% by the NRC was intended to conservatively encompass most expected operation fuel failure rates at that time. The 0.25% was a licensing basis value using the data at that time as regulatory basis.

All the data presented in NuScale TR-1116-52065, both as plots in Figure 5-2 and tabulated in Table A-6 consistently demonstrate that the fuel failure rates have dropped by over an order of magnitude between the 1970s and 2010 based on NRC, NEI, EPRI, GNF, ANS, and IAEA references. NuScale calculated the minimum, average, and maximum annual fuel failure rates for three different time periods (1996-2000, 2001-2005, and 2005-2010). Over this 15 year time period, NuScale selected the year with the highest annual average fuel failure rate of 0.0028% and, using the same margin of conservatism as reflected in selecting 0.25% from 1970s data, multiplied this highest annual average value by ten, resulting in 0.028%. The 0.028% value that NuScale utilizes comes from the highest (maximum) annual value (from a single year - 2001) within the date range considered (1996-2010). Although the data supporting the 0.0028% value is an average for 2001, it is not an average fuel failure fraction over a multiple year period.

The NRC presented fuel performance information in 2005 (ML050560020, page 14 in Oversight and Guidance, Akstulewicz, F.) which lists the PWR and BWR fuel failure rates as 0.00067% and 0.00043%, respectively. With regard to margin of safety, if an assumed 0.25% fuel failure rate (NUREG-0800) was appropriate and acceptable when the industry average fuel failure rate was 0.026% (industry average prior to 1985), then a similar margin of safety assuming 0.028% when the average fuel failure rate is 0.0025% (1990's) is appropriate. The margin is even greater when compared to the more recent industry data since the year 2000. The NuScale methodology is in conformance with the similar margin of conservatism for selecting 0.25% failed fuel in 1978.

The NuScale design uses natural circulation to drive primary coolant flow. This mitigates some of the major fuel failure mechanisms. The technical report shows that the two major industry fuel failure mechanisms are grid-to-rod fretting and debris. The driving force behind these failure mechanisms is the forced primary coolant flowrate (typically around 15--16 feet per second for a large PWR), and represents about 90% of the industry historical fuel failures.

The NuScale primary coolant flowrate through the reactor core using natural circulation is sufficiently slow (approximately 2.7 feet per second), making these failure mechanisms unlikely. Therefore, the use of existing industry fuel failure data is conservative.

The regulatory guidance has not been increased because of any individual outlier in operational experience over any given year or for any individual plant. Therefore, citation of one specific instance of a plant experiencing greater than 0.028% fuel failures in 2009 should not influence the NuScale value any more than an instance where greater than 0.25% fuel failure occurring during the history of U.S. nuclear power have has changed the NRC regulatory guidance value of 0.25%.

To protect the health and safety of the public and workers during normal operations, there are the required operational programs that measure and control the amounts of radiation to which people are exposed. These programs include the ODCM and RP programs, which are COL items. Should the fuel failure rate for a NuScale RXM be higher than that assumed in the design, these operational programs will protect the health and safety of workers and the public by restricting their radiation exposure. These operation programs provide reasonable assurance that the health risk due to higher fuel failure rates will be low thus mitigating the probability of human radiation exposures greater than that allowed by regulation. The NRC has affirmed the use of radiation protection programs for compliance with public and occupational exposure limits per 10 CFR 20 and ALARA design objectives per 10 CFR 50 Appendix I in a 2005 Nuclear Fuel Performance presentation (ML050560020 page 13 in Oversight and Guidance, Akstulewicz, F.).

A design basis value of 0.028% fuel failure rate for the NuScale design is appropriate, reasonably conservative and conforms to a similar margin of conservatism in selecting 0.25% from fuel failure data in the 1970s. The 0.028% value conservatively accounts for large improvements in fuel design, operation, and integrity that have occurred over the last 30 years. Significant improvements in fuel performance and a large reduction in fuel failure rates have been acknowledged by the NRC, EPRI, GNF, NEI, and the IAEA since the 1990s. Principal fuel failure mechanisms are mitigated in the NuScale design due to natural circulation, low coolant flow velocity in the reactor core. As affirmed by the NRC, NuScale uses radiological control programs to ensure normal operation occupational and public exposure meets all regulatory limits while maintaining ALARA principles. As affirmed by the NRC, the NuScale technical specification on primary coolant activity maintains design basis accident limits commensurate with site boundary and control room dose consequences.

The RAI requests that NuScale explain why adopting a technical specification limit that bounds the newly proposed failed fuel fraction is not warranted as discussed in the DSRS acceptance criteria.

NuScale Response (part 2):

A technical specification limit that encompasses the NuScale failed fuel fraction is not warranted because the RCS activity is a process variable that is an assumed initial condition of a design basis accident. It is appropriate to include an RCS activity limit within the Technical Specifications that is equal to that value assumed in the design basis accident analyses, per 10 CFR 50.36. For NuScale, this value is equivalent to 0.2 uCi/gram DEI-131, which supports the protection of the health and safety of the public. The use of technical specifications to limit coolant activity for design basis accident dose consequences is affirmed in a 2005 NRC presentation on Nuclear Fuel Performance (ML050560020, page 12 in Oversight and Guidance, Akstulewicz, F.).

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