RAIO-0318-59237

March 21, 2018

Docket No. 52-048

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

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 11 (eRAI No. 8759)," dated April 25, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 11 (eRAI No. 8759)," dated June 19, 2017

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

The Enclosure to this letter contains NuScale's supplemental 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
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8759

Enclosure 1:

NuScale Supplemental 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:

This supplement provides a revised response to RAI 12.02-1, submitted on June 19, 2017 in letter RAI-0617-54545 (ML17170A366).

NuScale proposed Generic Technical Specification 3.7.8, "Reactor Coolant System Specific Activity," has been revised so that the limits are based on a fuel defect level of 0.066% as assumed by the NuScale design basis source term.

Impact on DCA:

The Technical Specifications have been revised as described in the response above and as shown in the markup provided in this response.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Specific Activity

LCO 3.4.8 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > ~~0.2~~3.7E-2 μCi/gm.	---------------------NOTE-------------------- LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ ~~12~~2.2 μCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 > ~~60~~10 μCi/gm.	---------------------NOTE-------------------- LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > ~~12~~2.2 μCi/gm.	C.1 Be in MODE 2.	6 hours
	AND	
	C.2 Be in MODE 3.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ ~~60~~10 μCi/gm.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ ~~0.2~~3.7E-2 μCi/gm.	In accordance with the Surveillance Frequency Control Program AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% of RTP within a 1 hour period

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

The limits on RCS specific activity ensure that the doses due to postulated accidents are within the doses reported in FSAR Chapter 15.

The RCS specific activity LCO limits the allowable concentration of iodines and noble gases in the reactor coolant. The LCO limits are established based on~~to be intentionally conservative when compared with~~ a fuel defect level of 0.066~~28~~% assumed by the NuScale operating source term and to ensure that unit operation remains within the conditions assumed for Design Basis Accident (DBA) release analyses.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to limit the doses due to postulated accidents to within the values calculated in the radiological consequences analyses (as reported in FSAR Chapter 15).

APPLICABLE SAFETY ANALYSES

The LCO limits on the reactor coolant specific activity are a factor in accident analyses that assume a release of primary coolant to the environment either directly as in a small line break outside containment or indirectly by way of LEAKAGE to the secondary coolant system and then to the environment (the Steam Line Break).

The events which incorporate the LCO values for primary coolant specific activity in the radiological consequence analysis include the following:

- Steam generator tube failure,
- Control rod ejection,
- Steam Line Break (SLB), and
- Small line break outside containment

The limiting event for release of primary coolant activity is the small line break. The small line break dose analysis considers the possibility of a pre-existing iodine spike (in which case the maximum LCO of ~~12~~2.2 μCi/gm DOSE EQUIVALENT I-131 is assumed) as well as the more likely initiation of an iodine spike due to the reactor trip and depressurization. In the latter case, the LCO of ~~0.2~~3.7E-2 μCi/gm DOSE EQUIVALENT I-131 is assumed at the initiation of the accident, but the primary coolant

BASES

APPLICABLE SAFETY ANALYSES (continued)

specific activity is assumed to increase with time due to the elevated iodine appearance rate in the coolant. The reactor coolant noble gas specific activity for both cases is assumed to be the LCO of ~~60~~10 μCi/gm DOSE EQUIVALENT XE-133.

The LCO limits ensure that, in either case, the doses reported in FSAR Chapter 15 remain bounding.

The RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to ~~0.2~~3.7E-2 μCi/gm DOSE EQUIVALENT I-131, and the specific noble gas activity is limited to ~~60~~10 μCi/gm DOSE EQUIVALENT XE-133. These limits ensure that the doses resulting from a DBA will be within the values reported in FSAR Chapter 15.

The accident analyses (Ref. 1) show that the offsite doses are within acceptance limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of small line break accident, lead to doses that exceed those reported FSAR Chapter 15.

APPLICABILITY

In MODES 1 and 2, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity are necessary to contain the potential consequences of applicable safety analysis events to within the calculated site boundary dose values.

For operation in MODES 3, 4, and 5, the release of radioactivity in the event is limited by the reduced pressures and temperatures in the primary and secondary systems.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that DOSE EQUIVALENT I-131 is ≤ ~~12~~2.2 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to normal within 48 hours. If the concentration cannot be restored to within the LCO limit in 48 hours, it is assumed that the LCO violation is not the result of normal iodine spiking.

BASES

ACTIONS (continued)

A Note to the Required Action of Condition A states that LCO 3.0.4.c is applicable. This exception allows entry into the applicable MODE(S) when an allowance is stated in the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a small line break occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES, relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to, power operation.

C.1 and C.2

If a Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > ~~12~~2.2 μCi/gm, the reactor must be brought to MODE 2 within 6 hours and MODE 3 within 36 hours. The allowed Completion Times are reasonable, based on operating requirements, to reach the required unit conditions from full power conditions in an orderly manner.