RAIO-0318-59021

March 08, 2018

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. 332 (eRAI No. 9245) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 332 (eRAI No. 9245)," dated January 09, 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. 9245:

- 12.03-7
- 12.03-8

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,

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. 9245

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 9245

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NuScale Power™

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

**eRAI No.:** 9245
**Date of RAI Issue:** 01/09/2018

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**NRC Question No.:** 12.03-7

**Regulatory Basis**

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. Appendix A to Part 50—General Design Criteria (GDC) for Nuclear Power Plants, Criterion 4 requires applicants to ensure that structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation and postulated accidents.

NuScale DSRS 12.2 and DSRS 3.11 Acceptance Criteria state that the applicant should describe the radiation fields in sufficient detail for evaluating the inputs to shielding codes, and determination of radiation dose to electrical equipment important to safety as described in 10 CFR 50.49, and GDC 4.

10 CFR 50.46 (b)(5) and GDC 35 requires providing long term emergency core cooling. The guidance of Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," provides guidance for minimizing the potential for debris introduction into containment that could impact the ability to cool the core. As noted in RG 1.82, the debris may be generated as a result of the post-accident environment.

**Background**

NuScale DCD, Tier 2 Revision 0, Table 3C-6, "Normal Operating Environmental Conditions," states that the 60 Years Integrated N Dose (Rads) for the area outside of the top of the pressurizer is 6.00E7 rads (120 rads/hour). NuScale Technical Report TR-0116-20781-P Rev. 0, "Fluence Calculation Methodology and Results," Table 5-1 "Best estimate of fluence expected to be experienced in various NuScale Power Module components and locations," describes the neutron fluence to the reactor vessel and containment vessel, in the vicinity of the core, but does not provide any neutron flux or spectrum information for the area above the pressurizer. The Control Rod Drive Mechanisms (CRDM) are located in the area above the pressurizer and inside the containment vessel.

The acceptance criteria of NuScale DSRS 12.2 and DSRS 3.11 state that the source descriptions should include all pertinent information required for input to shielding codes used in the design process, establishment of related facility design features, and determination of radiation dose to electrical equipment important to safety as described in 10 CFR 50.49, and GDC 4.

Based on information made available to the staff as a result of the RPAC Chapter 12 Audit, and RPAC participation in the Control Rod Drive Mechanism (CRDM) Audit, the staff became aware that there were a number of B2 components (i.e. non-safety related and non-risk significant components) that were located outside of the reactor coolant system pressure boundary, but within the Containment Vessel that were not included in the EQ program described in DCD Section 3.11. For example, information reviewed by the staff during these audits, specified the use of flexible metal hoses between the Reactor Closed Cooling Water (RCCW) system and the CRDM magnet cooling coils. The hoses are classified as B2 items. When asked as part of the audits, NuScale stated that the hoses were rated for 200 ºF. When asked about the condition of the hose following actuation of the reactor recirculation valves (RRV) and/or the reactor vent valves (RVV), they stated that the RCCW system was not required to be operational following actuation of the RRVs and RVVs (i.e., post-accident). However, the post-accident conditions inside of the containment vessel (CNV) far exceed 200ºF. A similar discussion was held regarding some RCCW Thermal Relief valves, again located inside of the containment vessel and outside of the RV.

Key Issue:

It is unclear to the staff that if the non-safety related equipment located inside of containment were to degrade, as a result of the normal or post-accident environmental conditions (e.g., radiation,), that the safety-related SSCs would still be able to carry out their safety related function, such as coolant recirculation through the reactor core.

**Question Q-31009**

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions with respect to radiation exposures, the staff requests that the applicant:

- Explain/justify how non-safety related components located inside of the CNV but outside of the reactor vessel, are evaluated as it relates to meeting the requirements of 10 CFR 50.49(e)(4), GDC 4 and 10 CFR 50.46 (b)(5) and GDC 35.
- As necessary, revise DCD 3.11 to include any non-safety related equipment located inside of the CNV but outside of the reactor vessel that should be included in the DCD Section 3.11,

OR

Provide the specific alternative approaches used and the associated justification.

**NuScale Response:**

To address the 50.49(e)(4) and GDC 4 requirements, the RCCW thermal relief valve and the CRDM hoses are included in the environmental qualification program. Each of the respective components are included in Table 3.11-1, CRDM System, as part of the "CRDS Cooling Water Piping and Pressure Relief Valve" line entry. These components will be mechanically qualified and are considered EQ category B.

To address 10CFR50.46(b)(5) and GDC 35 requirements, per FSAR Section 6.2.2.2, "The generation of post-accident debris from coatings, insulation, latent debris, and chemical effects is considered along with debris transport and downstream effects in the CNTS design. Components located within the CNV do not contribute to the postaccident debris load and are not permitted to use fibrous or organic insulation materials except when encapsulated in a manner that prevents debris generation (i.e. conduit or sheathed). The materials selected for the components within containment take into consideration the anticipated water chemistry conditions. Protective coatings are not allowed within the CNV and susceptible materials (cables, etc.) are designed to withstand anticipated accident conditions within containment. Mineral insulated, metallic sheathed (MI) cable is used and does not contain fibrous insulation so it does not contribute to debris loading. Cable that is not MI cable is installed inside conduit.

Thermal insulation is not used inside containment. Section 6.3.2.5 describes conformance with RG 1.82 and the approach used to address Generic Safety Issue 191 (GSI-191), Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance.

NuScale design specifications ensure that these design requirements are in place. Additionally, programmatic controls are in place to ensure these requirements are not removed.

**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.:** 9245
**Date of RAI Issue:** 01/09/2018

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**NRC Question No.:** 12.03-8

**Regulatory Basis**

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radiation exposures within the limits of 10 CFR Part 20.

10 CFR 20.1101(b) and 10 CFR 20.1003 require the use of engineering controls to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical. The acceptance criteria of NuScale DSRS section 12.3, “Radiation Protection Design Feature,” states that radiation protection features should be incorporated into the design including design measures to reduce the production, distribution, and retention of activated corrosion products (e.g., material selection), including those resulting from direct neutron activation.

10 CFR 20.1406 requires applicants to describe in the application how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. The acceptance criteria of NuScale DSRS Section 12.3-12.4, “Radiation Protection Design Features,” states that the applicant is to describe how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

**Background**

The design documents reviewed by the NRC staff during the CRDM Audit, indicated that due to the length of the control rod drive shaft and the expected deceleration forces expected during control rod dropping, that there was a potential for increased flexure at the control rod drive shaft to control rod assembly junction. The applicant stated that the design of the CRDMs was not complete, so future design work and testing of this junction was expected. Since the applicant is currently unable to provide additional information regarding how increased flexure of this junction could affect cobalt introduction rates, the staff was not able to identify how the expected extra wear at this junction was factored into estimating the introduction of cobalt-containing wear products into the reactor coolant system.

NuScale DCD Tier 2, Revision 0 Section 12.3.1.1.13, “Material Selection,” states that proper material selection is an important factor to balance component performance while reducing the amount of corrosion and activation products generated. The use of materials containing cobalt is minimized to reduce the quantity of activation products. DCD Table 12.3-4, “Typical Cobalt Content of Materials,” states that the Maximum Weight Percent (w/o) of Cobalt in the CRDM internals springs in contact with primary coolant (Inconel X-750) is 1.00 w/o, and the cobalt content of other small components in contact with primary coolant, is not limited.

DCD Section 4.2.2.8, “Control Rod Assembly Description,” states that the top ends of the control rods are fastened to a spider using a threaded and pinned joint. The upper end plug is designed with a flex joint which provides the ability to accommodate misalignment between the control rods and the fuel assembly. DCD Tier 2 Revision 0 Section 4.5.1.3 “Other,” states that nickel- chromium based alloy X-750 is used for the CRDM springs and cobalt-based alloys Haynes 25 and Stellite 6 are used for wear- resistant parts as identified in Table 4.5-1, “Control Rod Drive Mechanism Materials.” DCD Tier 2, Revision 0, Table 4.5-1, “Control Rod Drive Mechanism Materials,” states that Stellite 6 may be used for Hard facing for latch arm tips, and the control rod remote disconnect expansion plugs use Haynes Alloy 25.

Industry material specification data shows that for Stellite 6 and Haynes Alloy 25, over 50% of the base metal consist of cobalt. Industry literature shows that for Alloy X-750 (UNS N07750) the cobalt impurity is limited to 1%. Due to the relatively high neutron absorption cross section of cobalt, the relatively high resultant specific radioactivity and the quantity and energy of the emitted photons when cobalt 60 (Co-60) decays, wear of components containing high cobalt content is important for evaluating compliance with 10 CFR 20.1101(b).

**Key Issue**

Since the applicant is currently unable to provide additional information regarding how increased flexure of this junction could affect cobalt introduction rates, and because cobalt is a major source of radiation exposure in operating nuclear power plants, increased wear of cobalt containing material will increase operational radiation exposure, contrary to the requirements of 10 CFR 20.1101(b) and fails to minimize contamination in accordance with 10 CFR 20.1406.

**Question**

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions with respect to potential Co-60 contamination from the CRDM, the staff requests that the applicant:

1. Explain/justify the amount of allowable flexure for the control rod drive shaft, including the limiting number of cycles, the basis for the stated flexure value, and the expected material wear rates,
2. Explain/justify the testing that will be performed to assess the actual amount of control rod

drive shaft flexure,

3. As necessary, revise and update section 12.3 of the NuScale DCD to specify the design features of the control rod drive shaft provided to minimize the introduction of cobalt due to flexure,

OR

Provide the specific alternative approaches used and the associated justification.

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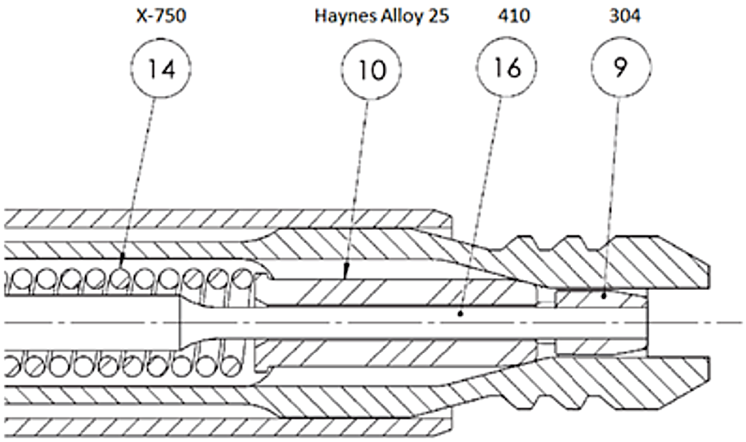
**NuScale Response:**

The flexure referenced in this question comes from a review of the buckling analysis in the vendor summary report for the CRDM. The purpose for this analysis was to demonstrate a large buckling margin. The deflections are a result of applying a very large load sufficient to create a buckling condition, and in many of the cases without credit for lateral supports. The smallest buckling load was approximately 5 times greater than the largest expected load on the control rod drive shaft, and this was without the horizontal guide tube (card) supports . The depicted flexures cannot actually occur and the maximum allowable flexure will be limited by the control rod assembly (CRA) card s. The design specification defines the maximum number of SCRAMs and latching cycles for this component.

Figure 2-17, in TR-0716-50439, shows the configuration of the control rod guide tubes. The CRA cards have a clearance hole that is currently 1/8 inch larger than the diameter of the control rod drive shaft, and limits the displacement at these locations to a maximum of 1/8 inch . The support from these CRA cards was not included in the buckling analysis to make it a very conservative analysis.

The Cobalt item, at the base of the control rod drive shaft, is the coupling expansion plug, item 10, in the figure below.The design of this latching interface is similar in form and in material to the control rod latching mechanisms in service in the existing fleet. In these existing applications Haynes Alloy 25 has proved to be the appropriate material for these applications. Since this item is inside the control rod drive shaft there will be very little motion due to any shaft bending (flexure) motion during a SCRAM. The vertical motion is also expected to be similar to that of similar mechanisms in the existing fleet, verification of this is one of the items that will be evaluated in the qualification testing. The wear characteristics will also be evaluated to see if an alternate material can be used, however industry experience has shown that Haynes Alloy 25 is the appropriate material for this application.




**Impact on DCA:**

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