NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Tuesday, January 09, 2018 3:34 PM
To: RAI@nuscalepower.com
Cc: NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Dudek, Michael; Lavera, Ronald; Markley, Anthony
Subject: Request for Additional Information No. 332 RAI No. 9245 (12.3)
Attachments: Request for Additional Information No. 332 (eRAI No. 9245).pdf

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk.
The NRC Staff recognizes that NuScale has preliminarily identified that the response to one or more questions in this RAI is likely to require greater than 60 days. NuScale is expected to provide a schedule for the RAI response by email within 14 days.

If you have any questions, please contact me.

Thank you.

Gregory Cranston, Senior Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-0546

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Request for Additional Information No. 332 (eRAI No. 9245)

Issue Date: 01/09/2018
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 12.03-12.04 - Radiation Protection Design Features
Application Section: 12.2, 12.3, 4.5.1, 4.6, 9.2.2, 3.11

QUESTIONS

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 200 ºF. 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.

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.