NuScale Power

RAIO-0318-59166

March 16, 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. 339 (eRAI No. 9275) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 339 (eRAI No. 9275)," dated January 18, 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 Question from NRC eRAI No. 9275:

- 12.03-13

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,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9275

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9275

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

eRAI No.: 9275
Date of RAI Issue: 01/18/2018

NRC Question No.: 12.03-13

Regulatory Basis

10 CFR 50.34.f(2)(xxvi) [NUREG- 0737 III.D.1.1] “Additional TMI-related requirements,” requires leakage control and detection for systems outside containment that might contain highly radioactive fluids, and requires applicants to submit a leakage control program, including an initial test program and a schedule for retesting systems.

Appendix A to 10 CFR Part 50— “General Design Criteria for Nuclear Power Plants,” Criterion (GDC) 61 “Fuel Storage and Handling and Radioactivity Control,” requires that new and spent fuel storage facilities include provisions for inspection and testing are necessary to verify that there is no corrosion of the spent fuel pool liner.

10 CFR 52.47(a)(6) requires compliance with the requirements of 10 CFR 20.1406 “Minimization of contamination,” which requires a description in the DCD 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.

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,” state 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.

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 guidance provided in NuScale DSRS Section 12.3-12.4 “Radiation Protection Design Features,” and Standard Review Plan (SRP) Section 9.1.2 “New and Spent Fuel Storage,” are consistent with and support the review of the design features provided for satisfy these regulatory requirements.

Background

The guidance in Regulatory Guide (RG) 1.68 “Initial Test Programs for Water-Cooled Nuclear Power Plants,” to meet the initial test requirements stated in 10 CFR 50.34.f(2)(xxvi) and NUREG- 0737 III.D.1.1. The guidance contained in NUREG-0737 III.D.1.1, specifically identifies systems that should have initial leakage rate tests performed, including the chemical and volume control system (CVCS), the plant sampling system (PSS) and the gaseous radioactive waste system (GRWS).

Key Issue 1

DCD Tier 2 Revision 0 Table 14.2-36, “Gaseous Radioactive Waste System Test # 36,” does not contain an initial leakage test consistent with 10 CFR 50.34.f(2)(xxvi) and NUREG- 0737 III.D.1.1. DCD Table 14.2-38, “Chemical and Volume Control System Test # 38,” does not contain an initial leakage test consistent with 10 CFR 50.34.f(2)(xxvi) and NUREG- 0737 III.D.1.1. DCD Table 14.2-53, “Process Sampling System Test # 53,” does not contain an initial leakage test consistent with 10 CFR 50.34.f(2)(xxvi) and NUREG- 0737 III.D.1.1.

Question 1

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions with respect to demonstrating compliance with the requirements of 10 CFR 50.34.f(2)(xxvi) and consistent with the guidance in NUREG- 0737 III.D.1.1, the staff requests that the applicant:

- Justify/explain how the proposed NuScale testing program meets the requirements of 10 CFR 50.34.f(2)(xxvi), for the CVCS, PSS and GRWS,
- As necessary, revise DCD Section 14.2 to include the pre-operational leakage tests for the CVCS, PSS and GRWS

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

As described in Tier 2, Section 9.3.4.2.3, the chemical volume and control system (CVCS) is designed for leakage detection and monitoring to minimize the leakage from those portions of CVCS outside of the containment that contain or may contain radioactive material following an accident consistent with 10 CFR 50.34(f)(2)(xxvi).

The CVCS circulates pressurized reactor coolant outside of containment during normal operation, and provides automated functions to detect and mitigate system leakage. As detailed in Section 9.3.4.2.3, a system differential mass flow rate greater than the high setpoint generates an alarm in the main control room (MCR). A system differential mass flow rate greater than the high-high setpoint generates an alarm in the main control room, and initiates closure of the CVCS module injection isolation valve and the CVCS module discharge isolation valve. The CVCS module isolation valves are located outboard of the safety-related containment vessel CIVs. This is a non-safety function and is intended to quickly isolate smaller CVCS leaks that would potentially take longer for the safety-related CIVs to close and isolate at the high-high setpoint.

The logic for the CVCS leakage detection is contained in the module control system (MCS). The MCS logic will be tested as part of the MCS factory and site acceptance tests. Tier 2, Figure 7.2-2, NuScale System and Software Technical Development Life Cycle Processes, identifies the MCS test phases. CVCS Test #38, component-level test i., requires verification that all CVCS remotely-operated valves can be operated from the MCR, thereby ensuring the proper response of the valve from the leakage detection automatic isolation signal.

For the process sampling system (PSS), design details for the leakage control and detection are described in Section 9.3.2. An operator can identify leakage in the PSS by PSS flow and pressure indication. The PSS does not contain any automatic operation for leak detection. The operator can isolate a PSS leak manually by remotely closing a PSS isolation valve from the MCR. A leak in the PSS can also be identified by the CVCS because the PSS sample line is downstream of the CVCS isolation valves. The automatic isolation of CVCS due to leakage is described above in the discussion of the CVCS leak detection system. Testing as described in Tier 2, Table 14.2-53, Process Sampling System Test #53, includes verifying the remote operation of each PSS remotely-operated valve. This satisfies the requirements of 10 CFR 50.34.f(2)(xxvi) for leakage control.

The gaseous radioactive waste system (GRWS) does not play a role, and is not credited, in post-accident operations, including post-accident sampling. During normal operations, the GRWS receives input streams from the liquid radioactive waste system (LRWS) degasifier and from the containment evacuation system (CES). In accident scenarios, the GRWS is isolated from the NuScale power module (NPM). For a post-accident primary coolant liquid sampling operation, the LRWS degasifier is not involved, therefore there would be no gaseous waste sent to the GRWS from LRWS. For a post-accident containment gas sampling operation, the gas sample is extracted from the gaseous process stream in the CES, which is returned to containment, therefore the CES gaseous stream is not sent to GRWS. In conclusion, the GRWS will not contain highly radioactive fluids in a post-accident situation, and is therefore not required be included in the leakage control program, as described in 10 CFR 50.34(f)(2)(xxvi).

Impact on DCA:

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