RAIO-0719-66444

July 26, 2019

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. 523 (eRAI No. 9682) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 523 (eRAI No. 9682)," dated June 06, 2019

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. 9682:

- 12.03-64
- 12.03-65

The response to question 12.03-66 will be be provided by August 26, 2019 and the response to question 12.03-67 will be provided soon.

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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9682

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NuScale Response to NRC Request for Additional Information eRAI No. 9682

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

eRAI No.: 9682
Date of RAI Issue: 06/06/2019

NRC Question No.: 12.03-64

Regulatory Basis:

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body. Materials to be analyzed and quantified include certain radionuclides, hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

Background:

On January 31, 2019, NuScale submitted an exemption request from 10 CFR 50.34(f)(2)(viii). The cover letter indicates that the exemption is requesting, "that sampling contingency plans for a NuScale Power Plant need not be demonstrated in terms of the dose criteria otherwise applicable under 10 CFR 50.34(f)(2)(viii)." However, the summary of the exemption specifies that NuScale, "requests an exemption from 10 CFR 50.34(f)(2)(viii), requiring capability for post-accident sampling of the reactor coolant system and containment." This summary appears to indicate that it is a request to be exempt from 10 CFR 50.34(f)(2)(viii) in its entirety.

On March 29, 2019, NuScale provided proposed FSAR markups associated with the exemption request. These FSAR markups included an update to FSAR Table 1.9-5, "Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)" which was revised to indicate that NuScale is taking a "departure" from 10 CFR 50.34(f)(2)(viii). The comment column of the table states, "Per Design Specific Review Standards (DSRS) 9.3.2, post-accident sampling is a contingency plan to be developed by a COL applicant (COL Item

9.3-2). The NuScale design supports an exemption from the portions of 10 CFR 50.34(f)(2)(viii) related to demonstrating the personnel radiation exposures." Finally, Section 12.4.1.8 of the proposed FSAR markups also indicates that the exemption is from exceeding prescribed radiation dose limits.

Issue:

The staff seeks clarification on the scope of the proposed exemption. For example, is the exemption request (1) a request for a full exemption from 10 CFR 50.34(f)(2)(viii) (i.e. a request to not be required to take post-accident samples at all); (2) a request to be exempt from the 5 rem design criteria for post-accident sampling in 10 CFR 50.34(f)(2)(viii); or (3) an exemption from 10 CFR Part 20 or other requirements.

Requests:

1. Please clearly specify the full scope of the exemption request and ensure that the enclosed details and FSAR appropriately reflect the intended scope of the exemption.

2. A technical justification is needed for why NuScale does not need to promptly obtain and analyze samples from the reactor coolant system and containment for radionuclides, hydrogen, dissolved gases, chlorides, and boron concentration. Please provide a detailed technical justification regarding why post-accident sampling is not needed for each of the sample types individually. If NuScale is requesting a full exemption from post-accident sampling, the response should consider any potential consequences of not sampling following both core damage and non-core damage accidents.

3. Section 9.3.2 of the DSRS is draft and has not been formally approved by NRC. Please correct this reference.

NuScale Response:

The scope of the subject exemption request has been modified to address 10 CFR 50.34(f)(2)(viii) in its entirety; approval of the exemption request would eliminate the requirement to provide a design capability for taking post-accident samples for accidents. The NuScale design maintains the capability to obtain samples as described in FSAR Section 9.3.2. The justification for the exemption request is that circumstances necessitating such a sample are unexpected

and of low probability, and that there are other indications available for the necessary parameters, such as the radiation monitors under the bioshield to detect core damage. The NuScale design philosophy and accident mitigation strategy is centered around maintaining coolant inventory by maintaining containment isolation. This strategy also prevents the spreading of contamination into other plant systems and reducing the risk of unnecessary radiation exposures to workers and to the public. Unisolating containment post-accident would be a restricted manual action.

As described in the exemption request, DCA Part 7, Section 16.2.2, the technical bases for not needing post-accident samples of various parameters are provided, as follows:

> **Radionuclides:** The purpose of sampling the post-accident reactor coolant for radionuclide content is to verify that the integrity of the fuel rod cladding has not been breached during an accident. The capability to measure reactor coolant radionuclides also supports the Emergency Action Level (EAL) classification in the Site Emergency Plan.
>
> The NuScale design utilizes radiation monitors under the bioshield and core exit thermocouples to assess core damage, therefore post-accident sampling of the reactor coolant for radionuclide content is not necessary in the NuScale design.
>
> **Hydrogen:** The purpose of sampling the containment atmosphere for hydrogen concentration post-accident is to both (1) help establish the degree of core degradation, and (2) assess the potential for containment failure due to hydrogen combustion.
>
> The NuScale design utilizes radiation monitors under the bioshield and core exit thermocouples to assess core damage. Therefore, the NuScale design does not require post-accident containment atmosphere grab samples to establish the degree of core degradation.
>
> NuScale has requested an exemption from 10 CFR 50.44(c)(2) "Combustible Gas Control," by complying with the underlying purpose of 10 CFR 50.44, which is to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event. The NuScale design accomplishes this purpose by withstanding a postulated worst-case hydrogen ignition during the first 72 hours of a design basis or beyond design basis event.
>
> This exemption does not impact the ability of a licensee to establish combustible gas monitoring following a design basis or beyond design basis event as described in TR-0716-50424, if needed. Therefore, post-accident containment atmosphere grab sampling is not necessary to meet the intent of the post-accident sampling requirements of 10 CFR 50.34(f)(2)(viii).

Dissolved Gases: The NuScale reactor module design differs from typical large light water reactors in that natural circulation core coolant cannot be inhibited in the reactor module by the accumulation of non-condensable gases, whether such accumulation is in the reactor vessel head or other systems (e.g., emergency core cooling). Emergency Core Cooling System (ECCS) actuation vents the non-condensable gas and steam from the integral reactor vessel and pressurizer gas space to the surrounding containment vessel through the ECCS vent valves. This continuous venting of the reactor vessel is maintained until ECCS is terminated preventing the formation of a gas barrier that could inhibit the natural circulation. Condensation of the steam occurs in the surrounding containment vessel and the liquid is returned to the reactor vessel through the ECCS recirculation valves to maintain circulation and keep the core covered. Therefore, there is no reasonable likelihood that an accumulation of non-condensable gases could interfere with post-accident natural circulation or otherwise inhibit long-term cooling following an accident. Based on the design of NuScale reactor module being insusceptible to an accumulation of non-condensable gases interfering with post-accident natural circulation, grab samples of post-accident reactor coolant for dissolved gas analysis is not necessary in the NuScale design.

Chlorides: The purpose of sampling the post-accident reactor coolant for chlorides is to ensure that chloride-induced stress corrosion cracking of stainless steel components and piping inside containment will not occur in the long term. The NRC policy, as expressed in SECY-93-087 on the design of the ALWRs, does not require the capability to take reactor coolant chloride measurements for the ALWR based on several plant design features that reduces the potential for chloride concentration in contact with stainless steel components. Materials for NuScale's major reactor components are not Alloy 600, which subsequently reducing the risk of stress corrosion cracking. Additionally, presence of chloride is minimized by the design of components inside the containment (minimal chlorinated cable insulation) and by monitoring chloride concentration in makeup inventory source during normal operation. Therefore, the post-accident sampling of reactor coolant for chloride is not necessary in the NuScale design.

Boron: The purpose of sampling the reactor coolant for boron is to ensure that there is adequate shutdown margin in the RCS to enable safe shutdown to be achieved. The capability to ascertain the RCS boron concentration is an important long term issue when water, other than the original reactor coolant inventory, will be used to refill the reactor vessel or to flood the containment. Because there is no automatic coolant makeup or safety injection capabilities in the NuScale design, the boron concentration in the primary coolant will remain unchanged for design basis event. The CVCS or CFDS can be manually initiated to provide coolant makeup to the reactor module or containment for beyond design basis event. However, because makeup source of manual injection is known, boron concentration can be determined prior to injecting water into the reactor module or containment.Therefore, post-accident boron samples are not necessary for the NuScale design.

The reference to DSRS Section 9.3.2 in DCA Table 1.9-3 and Table 1.9-5 has been corrected to reference SRP 9.3.2.

Impact on DCA:

FSAR Table 1.9-3 and Table 1.9-5 have been revised as described in the response above and as shown in the markup provided in this response.

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
~~DSRS~~SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.1	Sampling Capability	Partially Conforms	This acceptance criterion is applicable except for aspects that are BWR-specific, or not part of the NuScale design (e.g., refueling water storage tank, pressurizer relief tank, and containment sump).	9.3.2
~~DSRS 9.3.2, Rev 3: Process and Post-Accident Sampling Systems~~	~~II.2~~	~~10 CFR 20.1406. Minimization of contamination~~	~~Conforms~~	~~None.~~	~~9.3.2~~
~~DSRS~~SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.2~~3~~	Technical Specifications	Not Applicable	This was addressed in NRC-approved TSTF 366-A and is no longer applicable.	Not Applicable
~~DSRS~~SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.3~~4~~	Process Sampling System Functional Design	Conforms	None.	9.3.2
~~DSRS~~SRP 9.3.2, Rev 3: Process and Post-Accident Sampling Systems	II.4~~5~~	Seismic Design and Quality Group Classification	Conforms	None.	9.3.2
~~DSRS 9.3.2, Rev 3: Process and Post-Accident Sampling Systems~~	~~II.6~~	~~ITAAC~~	~~Conforms~~	~~None.~~	~~9.3.2~~
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.1	Protection Against Natural Phenomena	Conforms	None.	9.3.3
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.2	Environmental and Dynamic Effects	Conforms	None.	9.3.3
SRP 9.3.3, Rev 3: Equipment and Floor Drainage System	II.3	Control of Releases of Radioactive Material to the Environment	Conforms	No portions of the NuScale drain system penetrate the containment barrier.	9.3.3
DSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.1	CVCS Functional Performance during Adverse Environmental Phenomena; Pumping Capacity; and defense-in-depth RCS makeup	Partially Conforms	The only CVCS safety-related function precludes inadvertent boron dilution of the reactor coolant system.	9.3.4
DSRS 9.3.4, Rev 0: Chemical and Volume Control System (PWR) (Including Boron Recovery System)	II.2	Single Failure Criteria and GDC 5	Conforms	The single-failure criteria apply only to the two safety-related demineralized water isolation valves provided to preclude an inadvertent boron dilution of the reactor coolant system.	9.3.4

RAI 03.09.06-11S1, RAI 06.02.04-4S1, RAI 06.02.04-4S2, RAI 06.02.04-7S1, RAI 06.02.04-9, RAI 06.02.04-9S1, RAI 08.01-1, RAI 08.02-4, RAI 08.02-6, RAI 08.03.02-1, RAI 09.02.06-1, RAI 12.03-31, RAI 12.03-32, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35, RAI 12.03-36, RAI 12.03-37, RAI 12.03-39, RAI 12.03-40, RAI 12.03-64

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(1)(i)	Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8)	Partially Conforms	Design certification will address reliability of core and containment heat removal systems, with an update required by COL applicant to reflect site-specific conditions.	19.0 19.1 19.2
50.34(f)(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (II.E.1.1)	Not Applicable	This rule requires an evaluation of proposed PWR auxiliary feedwater (AFW) systems. The NuScale plant design does have an AFW system like a typical LWR. Neither the literal language nor the intent of this rule applies to the NuScale design.	Not Applicable
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA (II.K.2.16 and II.K.3.25)	Not Applicable	The NuScale reactor design differs from large PWRs because the NuScale design does not require or include reactor coolant pumps. Rather, the NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power-operated relief valve (PORV) (II.K.3.2)	Not Applicable	This guidance is applicable only to PWRs that are designed with power-operated pressurizer relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and reactor core isolation cooling system initiation levels (II.K.3.13)	Not Applicable	This requirement applies only to BWRs.	Not Applicable
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves (II.K.3.16)	Not Applicable	This requirement applies only to BWRs. Regardless, the issue contemplated by this requirement was related to power-operated relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation (II.K.3.18)	Not Applicable	This requirement applies only to BWRs.	Not Applicable

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(2)(viii)	Provide capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials (II.B.3)	Departure~~Partially Conforms~~	The NuScale design does not rely on primary coolant or containment samples to assess the extent of potential core damage. The NuScale design relies upon radiation monitors under the bioshield and core exit temperature indications for this assessment. Per SRP 9.3.2, post-accident sampling is a contingency plan to be developed by a COL applicant (COL Item 9.3-2). The NuScale design supports an exemption from the portions of 10 CFR 50.34(f)(2)(viii) related to demonstrating the personnel radiation exposures.~~As described by SRP 9.3.2, I.6, and RG 1.206, C.I.9.3.2, a post-accident sampling system is not required provided that the guidance provided in SRP 9.3.2 for utilizing the normal process sampling system (post-accident) has been satisfied.~~	9.3.2 11.5 12.4
50.34(f)(2)(ix)	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction (II.B.8)	Not Applicable	Pursuant to 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), Paragraph (f)(2)(ix) is excluded from the information required to be included in an application for a design certification.	Not Applicable
50.34(f)(2)(x)	Provide a test program and associated model development, and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves (II.D.1)	Partially Conforms	This requirement is applicable to the DCA except for aspects specifying PORV block valve testing and consideration of ATWS conditions in the testing program. The NuScale design does not use power-operated relief valves. The ATWS provision is not technically relevant to the NuScale design. This aspect of the regulation relates to reactor designs that rely on the relief and safety valves to mitigate the consequences of an ATWS event. The NuScale design supports an exemption from 10 CFR 50.62(c)(1) because the NuScale design relies on protection system diversity to prevent an ATWS, rather than design features to mitigate the condition. As a result, the module response to an ATWS is not analyzed in FSAR Section 15.8, such that the performance of the relief and safety valves is not relied upon to meet the ATWS safety criteria. Therefore, consideration of ATWS conditions in the relief and safety valve test program is not necessary to ensure acceptable performance.	5.2.2

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

eRAI No.: 9682
Date of RAI Issue: 06/06/2019

NRC Question No.: 12.03-65

Regulatory Basis:

10 CFR 50.34(f)(2)(viii) requires that the design provides the capability to promptly obtain and analyze samples from the reactor coolant system and containment without exceeding 5 rem.

10 CFR Part 20 establishes standards for protection against ionizing radiation resulting from activities conducted under licenses issued by the Nuclear Regulatory Commission. In addition, 10 CFR Part 20, Subpart C, establishes the occupational dose limits.

Background:

On January 31, 2019, NuScale submitted an exemption request from 10 CFR 50.34(f)(2)(viii). In Section 16.2.2 of the exemption request, under "Radiological Exposure to Workers," the applicant states the following:

"As a result of this exemption, a licensee need not demonstrate sampling contingency plans in terms of the dose criteria otherwise applicable under 10 CFR 50.34(f)(2)(viii). A licensee will be required by 10 CFR 50.47(b)(11) to establish means for controlling radiological exposures to workers in an emergency, which will include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides. Therefore, emergency workers will be protected from undue radiological exposure during emergency conditions that may necessitate obtaining post-accident samples or monitoring of containment hydrogen and oxygen."

While the requirements of 10 CFR Part 20 are not intended to limit actions that may be necessary to protect health and safety, the regulations of 10 CFR Part 20 do still apply during post-accident conditions. This is supported by the statements of consideration for 10 CFR Part 20 (FRN Vol. 56, No. 98, Tuesday, May 21, 1991), which state that, "[t]he Commission believes that the dose limits for normal operation should remain the primary guidelines in emergencies. However, the Commission also recognizes that, in an emergency, operations that do not conform to the regulations may have to be carried out to achieve the high-priority tasks of worker, public, and facility protection." It also states, "In evaluating any ensuing violations and their severity, the Commission will consider on a case-by-case basis any extenuating circumstances."

Issue:

Based on the regulations of 10 CFR Part 20 and the statements of consideration associated with 10 CFR Part 20, the regulations of 10 CFR Part 20 apply during post-accident conditions. As a result, the staff is seeking additional information on the regulatory basis for replacing 10 CFR Part 20 with the EPA Emergency Worker and Lifesaving Activity Protective Action Guides (EPA PAGs).

Requests:

1. Please clarify the scope of NuScale's exemption request intent in specifying that the exposure guidelines provided in the EPA PAGs will be used for controlling worker exposure when collecting post-accident samples.

2. If NuScale's exemption request retains the ability to perform post-accident sampling, explain how Part 20 exposure limits will be met by the proposed approach and provide appropriate revisions/markups of applicable documentation.

3. It appears that NuScale is indicating that the exposure guidelines provided in the EPA PAGs will be used for controlling worker exposure when monitoring containment hydrogen and oxygen. Please clarify if this is the intent.

NuScale Response:

The scope of the NuScale exemption request in Part 7, Chapter 16, is to be exempt from the entirety of 10 CFR 50.34(f)(2)(viii), based on the NuScale design not needing the information from post-accident primary coolant samples (reference response to RAI 12.03-64). Therefore, because no post-accident samples are required to be taken, there is no operator dose assessment required. NuScale will remove the discussion related to operator dose from the exemption request.

Hydrogen and oxygen monitoring is outside of the scope of this exemption request. Hydrogen and oxygen monitoring is unchanged and unaffected by the 10 CFR 50.34(f)(2)(viii) exemption request, and the discussion of hydrogen and oxygen monitoring has been removed from the exemption request. The revised exemption request will be provided with DCA Revision 3.

Impact on DCA:

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