January 31, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC, Submittal of Changes to the Design Certification Application, Part 7, *Exemptions*, Section 16, *10 CFR 50.34(f)(2)(viii) Post-Accident Sampling*

REFERENCES:
1. Letter from NuScale Power, LLC, to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application, Revision 2," dated October 30, 2018 (ML18311A006)
2. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980

NuScale Power, LLC (NuScale) requests NRC approval of the enclosed exemption request to 10 CFR 50.34(f)(2)(vii). The effect of this exemption is to align the NuScale licensing basis with that of licensees that were granted regulatory relief from the TMI Order (Reference 2), such 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). NuScale will provide conforming changes of additional supporting materials related to the enclosed exemption request (i.e., Tier 2 impacts) by March 29, 2019.

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

Please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com with questions.

Sincerely,

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Enclosure: "Changes to the Design Certification Application, Part 7, *Exemptions*, Section 16"

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"Changes to the Design Certification Application, Part 7, *Exemptions*, Section 16"

16. 10 CFR 50.34(f)(2)(viii) Post-Accident Sampling

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

16.1 Introduction and Request

16.1.1 Summary

NuScale Power, LLC (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. The rule requires the capability to obtain and analyze samples without exceeding prescribed radiation dose limits to any individual. The underlying purpose of the rule is to ensure the capability to assess the presence and extent of core damage. The NuScale Power Plant design meets the underlying purpose of the rule by ensuring the capability to assess the presence and extent of core damage during an accident by other means, which benefits public health and safety by avoiding unnecessary operator dose, preventing the spread of contamination, and reducing the potential for radioactive leaks and spills. As a result of the exemption, performance of sampling contingency plans will not be demonstrated in terms of the criteria of 10 CFR 50.34(f)(2)(viii) and NUREG-0737 that are otherwise applied to a post-accident sampling system.

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

16.1.2 Regulatory Requirements

10 CFR 52.47(a) states, in part:

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information
>
> . . .
>
> (8) The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v);

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

10 CFR 50.34(f)(2) states, in part;

> (f) Additional TMI-related requirements.
>
> . . .
>
> (2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operation licensing stage. This information is of the type

customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.

. . .

(viii) Provide the capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials[11] without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

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

10 CFR 50.34, Footnote 11, states;

The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

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

16.1.3 Exemption Sought

Pursuant to 10 CFR 52.7, NuScale requests an exemption from the post-accident sampling requirements of 10 CFR 50.34(f)(2)(viii).

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

16.1.4 Effect on NuScale Regulatory Conformance

As a result of this exemption, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR), will not conform with the provisions of 10 CFR 50.34(f)(2)(viii). The NuScale design will include capabilities allowing a licensee to sample the reactor coolant system and containment in accordance with the licensee's contingency planning, but the application will not demonstrate the capability to perform contingency sampling, including performance within the established dose limits.

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

16.2 Justification for Exemption

16.2.1 Purpose and History of Requirement

The underlying purpose of 10 CFR 50.34(f)(2)(viii) is to ensure the capability for plant operators to assess the presence and extent of core damage following an accident. As stated in NUREG-0578, Section 2.1.8:

> The NRC staff and the ACRS have for some years emphasized the need for special features and instruments to aid in accident diagnosis and control. Although some degree of capability of this type was available at TMI-2, and exists on other plants, the TMI-2 experience shows that more is needed. The Offices of Standards Development and Nuclear Reactor Regulation have agreed to expedite revision of Regulatory Guide 1.97, which deals with this subject area… In the meantime, the following provisions are recommended for early implementation on all plants to provide a uniform, minimum capability in this area. Recommendations: a. Improved Post-Accident Sampling Capability Review and upgrade the capability to obtain samples from the reactor coolant system and containment atmosphere under high radioactivity conditions. Provide the capability for chemical and spectrum analysis of high-level samples on site.

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

Thus, improved post-accident sampling capability was specified as an interim measure to assist in accident diagnosis and control, while the NRC developed revised requirements for instrumentation capabilities that would address a similar need. While operating plants had some sampling capability, the Task Force determined additional capabilities were required, and that TMI-2 accident conditions challenged the ability to perform sampling. As stated in NUREG-0578, Section 2.1.8.a:

> Chemical and radiological analysis of reactor coolant liquid and gas samples can provide substantial information regarding core damage and coolant characteristics. . . .
>
> Timely information from reactor coolant and containment air samples can be important to reactor operators for their assessment of system conditions and can influence subsequent actions to maintain the facility in a safe condition. Following an accident, significant amounts of fission products may be present in the reactor coolant and containment air, creating abnormally high radiation levels throughout the facility. These high radiation levels may delay the obtaining of information from samples because people taking and analyzing the samples would be exposed to high levels of radiation. . . .
>
> Prompt acquisition and spectrum analysis of reactor coolant samples within several hours after the initial scram would have indicated that significant core damage had occurred; perhaps with such information, earlier remedial actions could have been taken. Similarly, analysis of an early containment air sample would have indicated the presence of hydrogen, significant core damage, and the possibility of a hydrogen explosion in the containment.

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

In sum, the Task Force found that analysis of samples could provide substantial and important information to operators, which could assist in managing the accident and in informing emergency response efforts. The Task Force determined that effective radiation protection measures were necessary to ensure that such sampling capability could be effectively used when needed, such that operators could take timely actions to manage the event.

In the years since the TMI accident, significant improvements and a considerable amount of knowledge and industry experience have been realized in the areas of understanding risks associated with plant operations and developing better strategies for managing severe accident response. Insights about plant risks and alternate severe accident assessment tools have reduced the necessity of these post-accident sampling requirements. In certain instances, the use of a post-accident sampling system (PASS) can degrade the plant emergency response by diverting resources to non-essential activities and create a radiation release pathway.

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

In 1993, during its review of licensing issues pertaining to evolutionary and advanced light water reactors (ALWRs), the NRC staff evaluated requirements for PASS specified in 10 CFR 50.34(f)(2)(viii) in developing SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-water Reactor Designs." In the SRM for SECY-93-087 (ML003708056), the Commission approved the staff's positions, with modification, relaxing some of the requirements for post-accident sampling as implemented under item II.B.3 of NUREG-0737.

In the late 1990s, Combustion Engineering Owners Group (CEOG), Westinghouse Owners Group (WOG), and the BWR Owners Group (BWROG) submitted topical reports for NRC staff's review to eliminate PASS requirements. As stated by the CEOG in CE NPSD-1157, Revision 1 (ML003699802):

> [I]ncreased knowledge of accident phenomenology and the considerable amount of operating experience that have been gained in the years since NUREG-0737 was issued have led to a better understanding of degraded core behavior and the role that a PASS would play in various accident scenarios. This better understanding supports the conclusion that PASS does not play a significant role in controlling the plant emergency management response to severe accidents. In certain instances, use of PASS can even degrade the plant emergency response by diverting limited resources to non-essential activities and/or creating a radiation release pathway into the auxiliary building. It has also been determined that the role of the PASS in emergency planning is minimal and primarily confirmatory.

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

The staff concluded that the topical reports provided adequate basis to eliminate the PASS as a required system for post-accident sampling. As discussed in the safety evaluation for CE NPSD-1157, Revision 1 (ML003734524), the NRC based its decision "on the acceptability of the proposal to eliminate PASS on the benefit that the information obtained from PASS would provide in accident management and emergency response. If this information was considered to be necessary, and therefore, planned to be obtained shortly after a severe accident, then a PASS would be prudent to ensure that samples could be taken promptly

and exposure minimized. However … the information is not considered to be beneficial for accident management or emergency response. Therefore, there is considered to be sufficient time to establish an alternate sampling capability if samples were considered to be beneficial in the longer term."

In implementing the relaxed post-accident sampling requirements via facility license amendments, the BWROG sought clarification on requirements for the "alternate" (contingency) sampling capability expected by the staff. The BWROG asked "whether licensees are obligated to demonstrate that the contingency plans for obtaining radioactive samples can be performed within the dose limits established for post-accident sampling systems" (ML020560188, p.1). The Staff responded (ML020560188, p.2) that:

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

> In calling for licensees to make a regulatory commitment to develop and maintain a contingency plan to obtain highly radioactive samples, the staff mentioned that the planning should consider needs such as sample points, shielding, and other features that would make it feasible to implement the contingency plan under severe accident conditions. The staff clarified that since the contingency plans were, at most, a supplement to other process and radiation measurements available to decision-makers during a severe accident, the plans did not need to be carried out in emergency plans or drills. In safety evaluations issued before its review and approval of NEDO-32991, the staff likewise stated that licensees do not need to demonstrate contingency plans in terms of the criteria in NUREG-0737 that had previously been applied to PASS… We nevertheless encourage licensees to perform an evaluation to provide confidence that a feasible contingency plan has been developed.

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

The current regulatory position on post-accident sampling, per NUREG-0800 Section 9.3.2, is that a dedicated PASS is not required. In lieu of the dedicated PASS, the following actions are required to qualify the normal process sampling system for taking radioactive samples without having a specific post-accident sampling capability:

- Establish the capability for classifying a fuel damage event at the alert level threshold.
- Develop contingency plans for obtaining highly radioactive samples of the reactor coolant, containment sump, and containment atmosphere.
- Determine for its own plant(s) that no decrease in the effectiveness of emergency plans will result from not having post-accident sampling system capability.
- Establish the capability to sample and analyze hydrogen in the containment atmosphere (recommended).
- Maintain offsite capability to monitor radioactivity, including radioactive iodines.

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

Accordingly, for both licensees that were granted relief from post-accident sampling requirements implemented via NUREG-0737, and for new applicants complying with 10 CFR 50.34(f)(2)(viii), a dedicated PASS is not required, but contingency sampling plans are expected. The result of this exemption is to align the NuScale licensing basis with that of

licensees that were granted regulatory relief from the TMI Order, such that sampling contingency plans developed for a NuScale licensee need not be demonstrated in terms of the dose criteria otherwise applicable under 10 CFR 50.34(f)(2)(viii). The NuScale design includes features that allow a licensee to develop contingency plans for sampling. The NuScale design also includes features that allow a licensee to continuously monitor containment hydrogen oxygen concentrations in accordance with 10 CFR 50.44(c)(4), as described in TR-0716-50424.

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

16.2.2 Technical Basis

The underlying purpose of 10 CFR 50.34(f)(2)(viii) is to ensure the capability for plant operators to assess the presence and extent of core damage following an accident. In the NuScale design, this capability is provided by radiation monitors under the bioshield and by core exit thermocouples. The NuScale design is capable of classifying a fuel damage event at the alert level threshold utilizing the radiation monitors under the bioshield and the core exit thermocouples.

The NuScale design philosophy for major accident scenarios, including core damage events, is to isolate the NuScale power module (NPM) to preserve the primary coolant inventory and contain the potential post-accident source term. The process of taking a sample from the primary coolant or containment would necessarily require the transfer of potentially radioactive post-accident material from inside containment to the outside of containment. In lieu of such a process, the NuScale design relies upon other means to indicate the presence of core damage, namely radiation monitors under the bioshield and core exit thermocouples. It is preferable to maintain the primary coolant inventory and potential accident source term radioactive materials inside the NPM if the necessary information can be obtained otherwise. This design philosophy will result in a lower potential for facility contamination and personnel radiation exposure.

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

Dissolved Gases (Including Hydrogen):

Based on the design of NuScale reactor module being insusceptible to an accumulation of non-condensable gases interfering with post-accident natural circulation, there is little benefit to requiring grab samples of post-accident reactor coolant for dissolved gas analysis for the sake of ensuring post-accident natural circulation. Therefore, the post-accident sampling of reactor coolant for dissolved gases is not required in the NuScale design.

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

Containment Hydrogen and Oxygen:

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.

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.

The NuScale design has features that support contingency post-accident sampling that also support the capability to continuously monitor hydrogen and oxygen concentrations in the containment atmosphere using process sampling system (PSS) in-line monitors during accident conditions. 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.

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

Primary Coolant Boron Concentration:

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, the boron concentration in the primary coolant will remain unchanged.

Therefore, post-accident boron samples are not necessary for the NuScale design.

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

Primary Coolant and Containment Radionuclide Concentration:

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, not sampling.

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

Radiological Exposure to Workers

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.

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

16.3 Regulatory Basis

16.3.1 Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, "consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts." The exemption requirements for 10 CFR Part 50 regulations are found in 10 CFR 50.12, and are addressed as follows:

The requested exemption is authorized by law (10 CFR 50.12(a)(1)). This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended. The NRC has authority under 10 CFR 52.7 and 10 CFR 50.12 to grant exemptions from the requirements of this regulation. Therefore, the exemption is authorized by law.

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

The requested exemption will not present an undue risk to the public health and safety (10 CFR 50.12(a)(1)). This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create new accident precursors. Therefore, the exemption does not present an undue risk to the public health and safety.

The requested exemption is consistent with the common defense and security (10 CFR 50.12(a)(1)). This exemption does not affect the design, function, or operation of structures or plant equipment necessary to maintain the secure status of the plant. The exemption has no impact on plant security or safeguards procedures. Therefore, the exemption is consistent with the common defense and security.

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

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)). The underlying purpose of 10 CFR 50.34(f)(2)(viii) is to ensure the capability for plant operators to assess the presence and extent of core damage following an accident. The NuScale design provides for core damage assessment through the use of core exit thermocouples and radiation monitors under the bioshield, as described in FSAR Section 9.3.2. Additionally, the NuScale design includes features that allow a licensee to develop such contingency plans for sampling.

Special circumstances are present (10 CFR 50.12(a)(2)(iv)) in that the requested exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption. By limiting post-accident sampling capabilities to a contingency feature, operators will not rely on sampling to assess the presence and extent of core damage following an accident. By maintaining containment isolation as the preferred accident response, the spread of potentially highly radioactive material to systems outside of the NuScale power module will be prevented, avoiding unnecessary operator dose, preventing the spread of contamination to systems outside of

the NuScale power module, and reducing the potential for leaks and spills that could result in additional dose to the public. Consistent with past determinations by licensees and NRC, there is negligible safety benefit to maintaining post-accident sampling capabilities required by 10 CFR 50.34(f)(2)(viii) where alternative instrumentation can provide the necessary information to inform operators for accident management and emergency response. Therefore, there is no identified decrease in safety as a result of this exemption.

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

16.4 Conclusion

On the basis of the information presented, NuScale requests that the NRC grant an exemption for the NuScale design certification from the requirements of 10 CFR 50.34(f)(2)(viii).