RAIO-0618-60409

June 12, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 235 (eRAI No. 9109) on the NuScale Design Certification Application

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 235 (eRAI No. 9109)," dated September 22, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 235 (eRAI No.9109)," dated November 17, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) supplemental response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9109:

- 06.06-1

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
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9109

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9109

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

eRAI No.: 9109
Date of RAI Issue: 09/22/2017

NRC Question No.: 06.06-1S1

10 CFR 52.47, "Contents of Applications," specifies the level of design information needed to be submitted to support design certification. 10 CFR 52.47(a)(3)(i) requires that the design information include the principle design criteria for the facility and notes that Appendix A to 10 CFR Part 50, General Design Criteria, establishes minimum requirements for the principal design criteria for watercooled nuclear power plants. Appendix A to 10 CFR Part 50 specifies, in part, the following:

- 10 CFR Part 50, Appendix A, General Design Criterion 39 requires that, "The containment heat removal system shall be designed to permit appropriate periodic inspection of important components…and piping to assure the integrity and capability of the system."
- 10 CFR Part 50, Appendix A, General Design Criterion 45 requires that, "The cooling water system shall be designed to permit appropriate periodic inspection of important components…and piping, to assure the integrity and capability of the system."
- 10 CFR Part 50, Appendix A, General Design Criterion 55 requires that for "Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment…," "Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality design, fabrication, and testing, additional provisions for inservice inspection,…"

10 CFR 50.55a(b) requires that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), with conditions on the use of ASME Code Section XI described in 10 CFR 50.55a(b)(2).

10 CFR 50.55a(g)(3)(ii) requires for design certifications under Part 52 issued on or after July 1, 1974 that ASME Code Class 2 and 3 components (including supports) must be designed and be provided with access to enable the performance of inservice examination of these components.

While ASME Code Section XI (as conditioned by 10CFR50.55a) provides reasonable assurance of leak tightness and structural integrity, several adjustments have been made to ASME Code

Section XI and to 10CFR50.55a in response to operating experience. When evaluating a new reactor design, the NRC staff therefore examines areas where any new reactor designs significantly differ from the reactors for which the NRC and ASME have a large body of experience. One area of concern the NRC staff has for applying ASME Code Section XI to the NuScale design is the inspection requirements for smaller pipe sizes, such as NPS 4 or NPS 1.

The NRC staff request additional information on the use of the NPS 4 and other size-based ISI requirements for piping in the NuScale DCD to ensure that the NuScale SMR design meets 10 CFR Part 50, Appendix A, General Design Criteria 39, 45, and/or 55, as applicable to the system. If the ISI requirements for NPS 4 and smaller piping cannot be justified, revise the DCD to propose augmented inspection requirements beyond those required by ASME Code, Section XI.

NuScale Response:

NuScale and the Nuclear Regulatory Commission (NRC) have engaged in a series of meetings to address NuScale's use of the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (BPVC) in the design and inspection of the NuScale Power Module (NPM). During the last NRC managment meeting with NuScale management, held on May 17, 2018, NuScale volunteered to provide a supplemental response to the subject request for additional Information (RAI). Note, that this supplemental response is applicable to the following RAIs:

- 9109 - 06.06-1 and 06.06-2;
- 9103 - 05.02.04-1 and 05.02.04-2;
- 9183 - 03.03-2;
- 9193 - 05.02.03-10; and
- 9358 - 03.06.02-17.

<u>Scope of Response</u>

The scope of this supplemental response is to provide further justification for the NuScale position that meeting the ASME BPVC is acceptable to satisfy General Design Criteria associated with the reactor coolant pressure boundary (RCPB). In addition, this supplemental response addresses the following three specific areas of interest raised by the NRC:

- Weld repair criteria for welds below nominal pipe size (NPS) 4 inches
- Inservice Inspection (ISI) examinations for Class 1 piping below NPS 4 does not require volumetric examinations
- Material requirements and inspection specifications for small bolting

The NRC has questioned the ability to detect flaws in the inner diameter of the Class 1 small piping (<NPS 4) inside the NuScale Power Module (NPM) containment vessel. The NRC's

concern, as expressed to NuScale, is whether NuScale's compliance with 10 CFR 50.55a requirements, Standard Review Plan (SRP) 5.2.3 guidance, and the ASME BPVC is adequate to meet General Design Criterion (GDC) 14 requirements. NRC stated that the consequence of the failure of the less than NPS 4 pipes is higher for the NuScale design than the typical large light water reactors (LLWRs), because if one of these pipes fails then the NuScale design would rely on the Emergency Core Cooling System (ECCS) to respond to the event, whereas in LLWRs, a break in a less than NPS 4 line would not initiate ECCS. Therefore, NRC is concerned that NuScale is unnecessarily challenging ECCS by not sufficiently minimizing the likelihood of flaws propagating to failure.

The NuScale design complies with NRC requirements by meeting 10 CFR 50.55a, GDC 14, SRP 5.2.3 and ASME BPVC. NuScale provided justification of meeting these requirements in the original response to this NRC request for additional information (RAI). Additional justification is provided below.

Regulatory Background

Pursuant to *Maine Yankee Atomic Power Co.* (ALAB-161, 6 AEC 1003 (1973)), as restated in the *NRC Enforcement Policy:*

> *Adequate protection is presumptively assured by compliance with NRC requirements. Circumstances may arise, however, where new information reveals, for example, that an unforeseen hazard exists or that there is a substantially greater potential for a known hazard to occur. In such situations, the NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety.*

The Statements of Consideration for the original 10 CFR 50.55a (36 Federal Register 11423, June 12, 1971) provide similarly: "It may be that the special safety significance of a particular system or component will call for supplementary measures. If analysis of the system shows that such is the case, appropriate supplementary measures are expected to be adopted by applicants and licensees, or will be required by the Commission."

Conclusion: NuScale's conformance with the ASME BPVC, pursuant to 10 CFR 50.55a, presumptively assures adequate protection with respect to RCPB integrity. To overcome that presumption and require supplementary measures, there must be demonstrated special safety significance of a particular system or component, such as an unforeseen hazard in the NPM design or a substantially greater potential for a known hazard to occur.

Unforeseen Hazard Evaluation

NuScale has not identified an unforeseen hazard from leaks and breaks in pipes smaller than NPS 4.

GDC 14 states that the RCPB should have extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The potential hazards from leaks and ruptures in the RCPB are well known — potential loss of core cooling leading to fuel damage. As stated in NUREG-0800, Section 5.2.3, Rev. 3:

> *The RCPB provides a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling. Application of GDC 14 to the RCPB materials assures that they are selected, fabricated, installed, and tested to provide a low probability of significant degradation and, which in the case of extreme degradation could cause gross failure of the RCPB resulting in substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling.* …[Emphasis added]

NuScale Class 1 piping/welding design and small bolt design includes use of: materials that are corrosion resistant and have high resistance to fast fracture materials; the latest Electric Power Reliability Institute (EPRI) chemistry controls for primary and secondary water chemistry; and a highly sensitive containment leak detection system. Each of these design features are addressed in the NuScale Final Safety Analysis Report (FSAR).

NuScale's conformance with 10 CFR 50.55a and GDC 14 assures a low probability of significant degradation of RCPB components. Additionally, were gross failure of the RCPB to occur, containment isolation and ECCS actuation retains reactor coolant inventory within the NPM and returns it to the RPV to provide continued core cooling. Chapter 15 of the NuScale FSAR evaluates postulated loss of reactor coolant inventory events and does not suggest the possibility of unforeseen hazards associated with RCPB failure. To the contrary, for the NuScale design, leak or rupture of any size pipe does not result in exceeding specified acceptable fuel design limits (SAFDLs) or uncovering the core.

Conclusion: As with all LLWRs, consequences from leaks and breaks in pipes smaller than NPS 4 are within acceptance criteria. Gross failure of the RCPB, as addressed by GDC 14, potentially involves a substantial reduction or interference with core cooling and a failure to confine fission products. For the NuScale design, leak or rupture of any size pipe does not result in exceeding specified acceptable fuel design limits (SAFDLs) or uncovering the core. Radiological consequences of such an event are minimal. Therefore, there is no unforeseen hazard in the NuScale design.

Substantially Greater Potential For a Known Hazard to Occur Evaluation

NuScale has not identified a substantially greater potential for a known hazard to occur.

- The known hazard from leaks and ruptures in the RCPB is a potential loss of core cooling resulting in core damage in the event that ECCS fails to perform its intended function. With all LLWRs, frequently relying on ECCS, i.e. “challenging ECCS,” will increase the probability of this hazard, and thus increases the overall core damage frequency (CDF) of all LLWRs.
- Like all PWRs, the NuScale design includes a nonsafety-related reactor coolant makeup system and a safety related ECCS. In a large PWR, breaks in pipes smaller than NPS 4 are larger than the normal makeup capacity relied on to mitigate the event (e.g., for Indian Point Unit 2 the limiting pipe break size for normal makeup is approximately ¾ inch). Unlike a large PWR, the NuScale chemical volume control system (CVCS) is not relied upon to mitigate small breaks; safety related isolation and ECCS assure adequate cooling for all breaks. Reliance on ECCS in the NuScale design does not increase the known hazard of loss of core cooling.
- The CDF and large release frequency (LRF) from loss of coolant events in the NPM is less than a LLWR because:

    1. In the NuScale design, ECCS is a passive activity vs. an active start of pumps and powered valves in LLWRs.
    2. The NuScale passive ECCS is more reliable than a LLWR active ECCS. Actuation of the NuScale ECCS keeps the core covered and achieves a safe, stable, long-term cooling condition without reliance on external sources of water or power. This is demonstrated by the conditional failure probability of the NuScale ECCS (4E-6) that compares favorably with the conditional failure probability of an ECCS in a typical LPWR (6E-4, NUREG-4550).
    3. Unlike a large PWR, the nonsafety-related CVCS can provide core cooling in the event of a LOCA where the NuScale ECCS fails to function. This is also true for a break in the CVCS makeup line as inventory can be added using the pressurizer spray line.
    4. Unlike large PWRs where the containment is challenged as a result of core damage, the NuScale design offers passive protection against the containment damage mechanisms of insufficient containment heat removal, basemat melt-through, steam explosion, and hydrogen combustion.

Conclusion: The CDF and LRF from loss of coolant events in the NPM is less than a LLWR. The NuScale design can mitigate any size RCPB break with only the safety-related ECCS, which is more reliable than a LLWR active ECCS. The NuScale design provides additional means to prevent and to mitigate the leak/break hazards. Therefore, there is not a substantially greater potential for a known hazard to occur.

Overall Conclusion

Compliance with 10 CFR 50.55a and the ASME BPVC in the NPM design provides adequate protection against leaks and breaks in pipes smaller than NPS 4. NuScale has demonstrated that the GDC 14 requirements, 10 CFR 50.55a requirements, SRP 5.2.3 guidance, and ASME BPVC are adequately addressed in the NPM design. The consequences of failure of RCPB pipes smaller than NPS 4 is not higher for a NuScale NPM than traditional LLWRs, so no unforeseen hazard associated with such failure has been identified. Further, in LLWRs a break in a Class 1 line less than NPS 4 will most likely initiate ECCS similar to the NuScale design, but the NuScale ECCS is more reliable. Accordingly, there is no greater potential for the known hazard of RCPB failure--core damage--to occur, Therefore, no analysis has shown there to be special safety significance of RCPB pipes smaller than NPS 4, and thus conformance with 10 CFR 50.55a assures adequate protection of public health and safety with respect to RCPB integrity.

Impact on DCA:

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