Non - Proprietary
Discussion Topics for April 9, 2019, Public Meeting with NuScale

A. AST Whitepaper

The staff observed that the justifications provided for NuScale's planned use of a non-core melt source term for GDC-19 and use of a core melt source term for offsite doses as a "defense in depth" measure are not as substantive as the justifications provided in other portions of the white paper. The staff is concerned that the discussion in these areas of the white paper does not properly justify a departure from past practice.

B. 10 CFR 50.34(f)(2)(viii) Post-Accident Sampling Exemption Request

1. As stated in the exemption request, the NuScale design has features that support contingency post-accident sampling, how will NuScale ensure that an applicant will develop a contingency plan?
2. When will the contingency plan be developed?
3. Which sampling will be included in the contingency plan?
4. How will the contingency plan be verified for adequacy?
5. On Page 16-7 of the exemption request, NuScale appears to indicate that if during an accident the licensee decided to take post-accident samples using contingency plans that the licensee will use the EPA Emergency Worker and Lifesaving Activity Protective Action Guides for controlling the radiological exposures to workers in an emergency. In other portions of the exemption, NuScale appears to only be requesting an exemption from demonstrating compliance with the 10 CFR 50.34(f)(2)(viii) dose limit as part of the application (and not a request to exceed dose limits in the event that post-accident sampling actually occurred). Please clarify whether NuScale's exemption request's that the dose limit in 10 CFR 50.34(f)(2)(viii) not apply if a decision was made to conduct post-accident sampling (using the contingency plans) or is the request only that compliance with the dose limit need not be demonstrated in the application? Of note, in a letter to the Boiling Water Reactor Owners Group (BWROG) (ML020560188), regarding contingency plans for post-accident sampling, the NRC staff clarified that the governing regulations are those in 10 CFR Part 20, including provisions for keeping doses as low as is reasonably achievable.
6. NuScale's exemption request references Combustion Engineering Owners Group, Westinghouse Owners Group, and BWR Owners Group topical reports and related documents associated with post-accident sampling system (PASS) relaxations. NuScale appears to imply that their request is similar to requests made by these licensees, however, for these requests it appears that staff only approved relaxation from PASS requirements for these other plants (such as removal of technical specifications, relaxation from needing to provide a dedicated PASS, allowance of using contingency plans, etc.). These past approvals do not appear to grant an exemption from 10 CFR 50.34(f)(2)(viii) nor do they allow for exceedance of dose limits. Does NuScale have an example where an exemption from 10 CFR 50.34(f)(2)(viii) has been granted? If NuScale is requesting that dose limits not apply if post-accident sampling is conducted during actual plant operation, does NuScale have an example where an allowance to exceed dose limits during sampling was granted?

7. In the response to RAI 8775, Question 12.03-1, the applicant indicated that the NuScale design does not have any credited post-accident operator actions or vital missions under 10 CFR 50.34(f)(2)(vii). However, NuScale indicates that part of the basis for an exemption from post-accident sampling under 10 CFR 50.34(f)(2)(viii), is that they can perform hydrogen monitoring. While the full scope of the actions required to perform hydrogen monitoring is unclear, it appears that various operator actions are required both inside and outside of the control room. The staff, thus, seeks clarification of why these actions associated with hydrogen monitoring do not constitute a vital mission under 10 CFR 50.34(f)(2)(vii).
   a. Is the applicant's position that there are no 10 CFR 50.34(f)(2)(vii), vital missions, associated with the NuScale design? If so, provide justification for why 10 CFR 50.34(f)(2)(vii) would not apply to hydrogen monitoring?
   b. If the actions associated with hydrogen monitoring are considered a vital mission, please discuss all actions associated with performing hydrogen monitoring. Include in the discussion all actions necessary and the locations of the actions (whether in the control room or otherwise). Also include any necessary actions associated with isolating other systems in order to prevent the inadvertent spreading or release of radioactive material (such as the reactor building ventilation system and the gaseous waste management system). Discuss relevant radiation dose and shielding considerations.
   c. After hydrogen monitoring has been initiated, discuss if a licensee would expect to keep the containment un-isolated for the remainder of the accident or would the containment eventually be re-isolated? Has the potential need to re-isolate containment been evaluated?
8. 10 CFR 50.34(f)(2)(xxvi) requires the provision of leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. Since NuScale states that they will meet the regulatory requirement for monitoring the containment atmosphere for hydrogen following an accident, and since NuScale Technical Specifications does not contain provisions for a leakage control program for systems outside of containment that contain (or might contain) accident source term radioactive materials, please describe how NuScale plans to meet this requirement. Please describe the systems that contain (or might contain) accident source term radioactive materials, including interface systems. Describe any necessary proposed changes to the Initial Test Program.

C. NuScale Response to RAI 9464 on Containment Leak Rate

[ ]