NuScaleDCRaisPEm Resource

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Sent:	Friday, September 01, 2017 2:23 PM
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Subject:	Request for Additional Information No. 209, RAI 9112 (19)
Attachments:	Request for Additional Information No. 209 (eRAI No. 9112).pdf

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

Gregory Cranston, Senior Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-0546

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Request for Additional Information No. 209 (eRAI No. 9112)

Issue Date: 09/01/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS

19-29

Regulatory Basis

10 CFR 52.47(a)(27) states that a Design Certification (DC) application must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific Probabilistic Risk Assessment (PRA) and its results. 10 CFR 52.47(a)(23) states that a DC application for light-water reactor (LWR) designs must contain an FSAR that includes a description and analysis of design features for the prevention and mitigation of severe accidents (e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure melt ejection, hydrogen combustion, and containment bypass). For staff to make a finding that the applicant has performed an adequate evaluation of the risk from severe accidents in accordance with Standard Review Plan (SRP) 19.0, the applicant is requested to respond to address the issues below.

Request for Additional Information

The FSAR presents MELCOR simulations for a range of accidents that show the pressure difference between the reactor and containment is below 200 psi by the time core damage starts. The FSAR cites NRC studies as the basis for a 200 psi criterion below which high pressure melt ejection (HPME) is not a threat to containment. The FSAR states that HPME challenges to containment include a) dispersed debris causing rapid heating of the containment atmosphere and b) direct contact of the dispersed debris with the metal containment itself.

The NRC direct containment heating (DCH) issue resolution studies do not claim that 200 psi is a pressure below which HPME challenges to containment are not credible. On the contrary, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, December 1990, states that, while attempts have been made to define a cutoff pressure, the technical basis for a cutoff pressure is weak. Also, more recent studies (e.g., “Direct Containment Heating Experiments at Low Reactor Coolant System Pressure in the Surtsey Test Facility,” NUREG/CR-5746, July 1999; “Direct Containment Heating Integral Effects Tests in Geometries of European Nuclear Power Plants,” *Nuclear Engineering and Design*, April 2009) suggest that a pressure of 145 psi or lower could threaten containment depending on the lower head hole size.

The applicability of these studies to the NuScale design is unclear, because the studies were for large LWRs and required plant-specific analysis of containment loads supported by experiments, including consideration of phenomenological uncertainties during late-phase melt-progression. Containment loads for large LWRs are influenced by containment structures and compartmentalization (e.g., cavity, instrument tunnel, seal table room) which can reduce the containment overpressure challenge by trapping corium ejected from the reactor. Containment loads for large LWRs also can be influenced by debris water interactions (e.g., steam spike when debris enters water in the cavity) and atmosphere-structure heat transfer. The studies used codes validated with geometry-specific tests for Surry, Zion, and Calvert Cliffs that were generally representative of cavity and lower containment designs for operating PWRs using appropriate modeling parameters.

As part of the RAI response, the applicant is requested to describe the NuScale Power Module's susceptibility (or resistance) to both over-pressure and over-temperature challenges associated with high pressure melt ejection including the potential vulnerability of the unwetted section of the containment (i.e., above the reactor pool water line) to contact with core debris ejected from the reactor following reactor vessel lower head failure.