December 12, 2017

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 259 (eRAI No. 9138) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 259 (eRAI No. 9138)," dated October 13, 2017

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 Question from NRC eRAI No. 9138:

- 19-33

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9138

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NuScale Response to NRC Request for Additional Information eRAI No. 9138

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

eRAI No.: 9138
Date of RAI Issue: 10/13/2017

NRC Question No.: 19-33

10 CFR 52.47(a)(23) states that a design certification (DC) application must contain a final safety analysis report (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 provide the additional information requested below.

a. FSAR Section 19.2.3.3.5 describes a thermodynamic analysis of the energy release from a hypothetical in-vessel steam explosion (i.e., within the reactor pressure vessel (RPV)) which concludes that the energy released is insufficient to challenge RPV integrity. The staff requests additional information to understand how the analysis relates to past Nuclear Regulatory Commission and industry studies of in-vessel steam explosion including "An Assessment of Steam-Explosion-Induced Containment Failure. Part 1: Probabilistic Aspects," Theophanous, T. G., Najafi B., and Rumble, E., *Nuclear Science and Engineering*: 97, 259-281 (1987) and "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues; Second Steam Explosion Review Group Workshop," NUREG-1524, August 1996. For example, what did the applicant assume for the conversion ratio versus thermal energy in the mixed melt? What did the applicant assume for the net energy in the head after dissipation in upper internals versus slug energy? Please provide any benchmarking of the applicant's analysis against past studies in terms of energy release and mechanical loads, considering the similarities in phenomenology and the design differences (NuScale vs. large light-water reactors)?
b. For in-vessel steam explosion, the applicant's analysis addressed different initial reactor coolant system (RCS) hole sizes and emergency core cooling system (ECCS) failure modes by using as input the MELCOR predictions for sequences with different initial RCS hole sizes and ECCS failure modes. However, the NRC staff could not find where the applicant's analysis addressed uncertainties in the modeling of physical phenomena in its MELCOR simulations, such as uncertainties in in-vessel melt progression modeling (e.g., modeling of core heat-up, collapse, and formation of molten pool). Such uncertainties have

the potential to result in a different energy of corium relocating to the water in the RPV lower plenum. An example of consideration of such uncertainties is given in “State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses,” Draft Report, April 2016 (ADAMS Accession Number ML17156A255). The staff requests additional information to understand how the applicant’s analysis addressed uncertainties in the modeling of physical phenomena.

c. FSAR Section 19.2.3.3.5 states that an ex-vessel steam explosion (i.e., within the containment vessel (CNV)) is judged to be physically unrealistic based on the size of the NuScale core, the physical dimensions of the CNV, the proximity between the RPV and CNV, the associated potential drop height for fuel between the two, and thermal-hydraulic conditions within the CNV in the postulated condition that the RPV were breached. The applicant is requested to provide quantitative justification for this judgment, including addressing the potential for an ex-vessel steam explosion to cause the CNV to move sufficiently to induce a CNV structural failure. For example, AP1000 calculations in NUREG/CR-6849 indicated the potential for large impulse loads on the cavity and the RPV (and subsequently the containment penetrations). The phenomena occurring inside the NuScale CNV with a high water level appears to be similar to the AP1000 analysis.

NuScale Response:

Item a.

Fuel-coolant interactions (FCI), or steam explosions, within the NuScale reactor pressure vessel (RPV) are evaluated using the Hicks-Menzies thermodynamic model (cited in FSAR Reference 19.2-19) to determine if sufficient energy exists in an energetic interaction of fuel materials and coolant to induce RPV failure and subsequent alpha-mode containment failure. This model calculates the theoretical thermodynamic maximum work potential of expanding steam after an interaction with hot fuel material. The model is adiabatic. As such, the energy from the expansion process is used as a conservative estimate of the work capable of challenging RPV integrity.

Input parameters are determined from MELCOR accident sequence simulations presented in FSAR Section 19.2. To ensure a bounding analysis, parameters are selected from a set of MELCOR simulations such that the work performed during the expansion and propagation phase of the model is maximized. Only MELCOR simulations involving an intact containment and core relocation are used to inform the FCI analysis. Simulations with a breached or bypassed containment already contain a release pathway from the containment vessel (CNV); as a result, the alpha-mode of containment failure would be of no additional consequence.

The model analyzes FCI with an isochoric fuel and coolant mixing process where fuel and coolant reach thermodynamic equilibrium (equilibrium state) and an isentropic fluid expansion process where a pressurized fluid expands to a larger volume while performing work on the vessel (final state). To demonstrate RPV integrity, the energetic expansion of coolant is

compared to (and must be less than) the energy of a pressurized fluid within the RPV at the ultimate failure pressure of the vessel. This comparison results in a calculated capacity of the RPV with respect to an energetic interaction of fuel materials and liquid coolant within the lower head of the RPV. Furthermore, because the Hicks-Menzies model considers a thermodynamic maximum conversion ratio of thermal energy to mechanical work (approximately 20% for NuScale), the evaluation of in-vessel FCI for the RPV is inherently conservative.

To account for alternative initial configurations of coolant (saturated liquid versus a vapor coolant mixture), NuScale has performed three studies considering alternative initial compositions of the water pool in the lower head of the RPV:

- water pool has an initial 10% void fraction,
- water pool is initially saturated liquid, and
- water pool has an initial 50% void fraction.

The sensitivity case with the initial condition of saturated liquid bounds the other cases in terms of conversion ratio and expansion work energy. All three cases result in significantly less mechanical work than the capacity of the RPV against expansion work. Further, because the calculation assumes a 100% efficiency between the release of mechanical expansion energy, upward liquid slug energy, and the net energy in the upper head of the RPV after dissipation in the upper vessel internals, the loading to the upper head of the RPV is conservative and the energy available to produce an alpha-mode containment failure is maximized. A comparison of NuScale conditions to information provided in, “An Assessment of Steam-Explosion-Induced Containment Failure. Part 1: Probabilistic Aspects,” by Theofanus, Najafi, and Rumble (Figure 16 ID-6, Figure 17 ID-7), suggests that a release of mechanical energy similar to that predicted by the NuScale analysis with the Hicks-Menzies model produces an upward slug energy of less than the expansion work with a conversion ratio of thermal energy to mechanical work of approximately 20%. Then, based on the upward slug energy, the net energy in the vessel head after dissipation in the upper vessel internals would be nearly zero. As a result, work performed by expanding coolant on the RPV is insufficient to challenge vessel integrity and induce an alpha-mode containment failure. Additionally, “Evaluation of Dynamic Pressures from Steam Explosions Applied to Advanced Light Water Reactors,” by Corradini, Blanchard, and Martin provides an analysis of steam explosions in advanced light water reactors. This analysis is supported by a study using preliminary data from NuScale-specific severe accident simulations and early design information to postulate a condition resulting in an in-vessel FCI. The study concluded that a steam explosion-induced dynamic pressure pulse applied to the RPV structure is insufficient to compromise wall integrity.

The in-vessel FCI analysis performed for the NuScale RPV described in FSAR Section 19.2.3.3.5 conservatively assumes conditions for evaluating the potential consequences of an energetic steam explosion which are not predicted by MELCOR simulations. As an example, the thermal hydraulic analyses described in FSAR Section 19.2.3.2 predict that the core support structure fails prior to significant core melting. This means that the core relocates to the RPV lower head before the core actually melts. As such, there is limited potential for a suspended

molten fuel mass to interact with a liquid water pool below the fuel or for the solid fuel configuration to fragment during a drop into a water pool in the lower RPV head. As a result, rapid heat transfer between solid fuel and coolant is difficult. Further, because a solid fuel mass has a lower bulk temperature than a molten mass, there is less thermal energy available for transfer to a water pool. In conjunction with the limited thermal load of a small NuScale core relative to large reactors, an energetic in-vessel FCI is not credible.

FSAR Section 19.2.3.3.5 has been modified to include additional detail and references related to the quantitative analysis performed for in-vessel FCI.

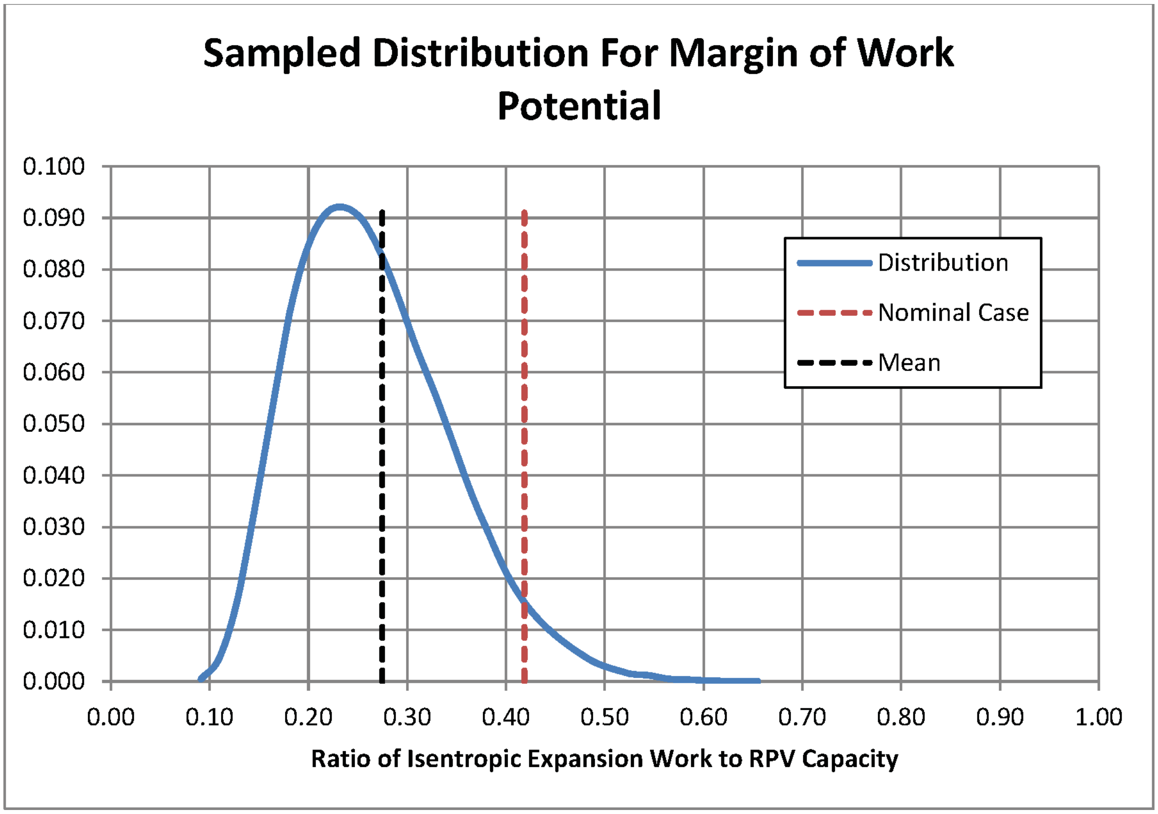
Item b.

The analysis of in-vessel FCI is performed considering input parameters from the MELCOR accident sequence simulations presented in FSAR Section 19.2. While the FCI analysis does not explicitly consider alternative reactor coolant system (RCS) hole sizes or other changes in reactor configuration, several severe accident sequences were evaluated for potential FCI effects, as summarized in FSAR Table 19.2-10. Because each accident sequence results in different estimates for core damage timing, core relocation timing, mass of relocated core materials, relocated material temperatures, etc., inputs to the Hicks-Menzies model reflect uncertainty in accident phenomena.

To inform the uncertainty analysis for in-vessel FCI, input values to the thermodynamic FCI model are acquired from each MELCOR accident simulation; this range of values is then used to develop a probability distribution for each input. The minimum and maximum observations are assumed equivalent to the minus and plus one standard deviation values of the distribution, respectively, and a uniform distribution is assumed between those values. A lognormal distribution is assigned for the tails of each input parameter distribution to allow for the sampling of extreme values not predicted within MELCOR simulations. Random samples for each input are then acquired by Monte Carlo sampling and used to evaluate the Hicks-Menzies model for the expansion work performed on the RPV. The result of each sampling iteration is a comparison of the expansion work to the capacity of the RPV, presented as a ratio between the two (note that a ratio greater than 1.0 constitutes failure).

The in-vessel FCI uncertainty analysis results are reproduced in the following figure. In the figure, the sampled distribution is in blue, the mean value of the sampled distribution is the black dashed line, and the value assuming a bounding combination of input parameters from MELCOR is the red dashed line. Based on the maximum sampled ratio (approximately 0.67), significant margin exists between the RPV capacity for coolant expansion and expansion work determined by the Hicks-Menzies model.

FSAR Section 19.2.3.3.5 has been modified to include additional detail related to consideration of uncertainty in the modeling of in-vessel FCI.



Item c.

As discussed in FSAR Section 19.2.3.2.1, failure of the RPV after a core damage accident involving an intact containment is not physically realistic. For such situations, a significant volume of liquid water will be present in the annular region between the CNV and the RPV to provide a cooling pathway from the core region to the reactor building pool. This volume of water prevents failure of the RPV lower head. As a result, an instantaneous interaction of fuel materials inside of the RPV and liquid coolant in the CNV is not credible and a quantitative analysis postulating such conditions was not performed.

However, from the perspective of demonstrating defense-in-depth, several fundamental characteristics affecting the possibility of an ex-vessel FCI are addressed in FSAR Section 19.2.3.3.5. These characteristics were evaluated in terms of NuScale design features which preclude the possibility of an ex-vessel FCI and are discussed in more detail below:

- Considering a situation with an intact containment, RCS water relocated to the CNV, and a failed RPV lower head, the distance between the bottom of the lower head of the RPV and the CNV is small and the MELCOR accident sequences in FSAR Section 19.2 with an intact containment predict that this space would be occupied by a water pool. An

energetic FCI requires space between molten fuel materials, if present, and a water pool to promote material breakup. Breakup helps to create a large total surface area for contact with liquid coolant, thereby increasing rapid heat transfer. Because accident sequences with an intact containment contain a significant amount of liquid coolant in the annular region between the RPV and CNV, there is no available space between a failed RPV lower head and the water pool beneath to foster material breakup needed to promote an energetic transfer of heat to the water pool in the CNV.

- The CNV is not large enough to allow for a relocation of all core materials from the RPV to the CNV. Because of the limited space between the RPV and the CNV, a significant portion of the fuel material will remain backfilled within the RPV above a fuel mass in the CNV. This prevents all fuel material from interacting with a water pool in containment. Coupled with the small size of the NuScale core, a relocation of fuel materials from the RPV to the CNV will involve less material than a similar FCI within the RPV, further limiting the potential energy transference necessary for an energetic ex-vessel steam explosion.

- Because of the large water pool predicted to reside in the containment annulus, the resultant conversion ratio for an ex-vessel FCI will be significantly less than the predicted ratio using the Hicks-Menzies thermodynamic model of an in-vessel FCI (which was shown to not challenge RPV integrity), thereby limiting the potential for work to be performed on the CNV by expanding coolant.

For these reasons, efficient transfer of thermal energy between fine fuel materials and coolant cannot take place and the conditions required for an energetic ex-vessel FCI are not established. As a result, an ex-vessel FCI is not physically credible and therefore, has not been quantified

FSAR Section 19.2.3.3.5 has been modified to include additional discussion of ex-vessel FCI.

Impact on DCA:

FSAR Sections 19.2.3.3.5 has been revised as described in the response above and as shown in the markup provided in this response.

triggering stage of fuel-coolant interaction. The resistance to film collapse impedes fuel-coolant interaction.

- a steam explosion is more difficult to trigger spontaneously when the system pressure is high because the stability of the vapor film increases with pressure. Additional energy relative to lower pressure situations is required to collapse this vapor film.
- a "melt pour" type of interaction (i.e., corium poured into a water pool) bounds the energetics associated with a "stratified" type of interaction (i.e., water flooding a corium debris bed).

The potential for in-vessel and ex-vessel steam explosions are discussed in more detail below.

In-Vessel Steam Explosion

The "alpha mode" of containment failure is considered for its potential in the NuScale design. In the alpha mode, the concern is that a steam explosion inside the RPV could induce a water slug which could impact the uppermost structures of the RPV or induce significant dynamic loading challenging the integrity of the RPV. If such an event were to occur, a sudden increase in energy within the RPV could challenge RPV section and bolted interfaces, potentially compromising vessel integrity. If this were to happen, it could cause the rupture of the bolts that hold the upper head in place, potentially resulting in containment failure. The issue is described in NUREG-1524 (Reference 19.2-19).

For an in-vessel steam explosion, the body of molten corium is in the core region above the core support plate, while the water pool is below the plate. An in-vessel steam explosion is evaluated to be physically unrealistic based on the size of the NuScale core, physical dimensions of the RPV, and thermal-hydraulic conditions within the RPV.

- The amount of melt available for steam explosion is small. The thermal-hydraulic analyses, as described in Section 19.2.3.2, conclude that the core support structure is expected to fail prior to significant core melting. Thus, there is limited potential for interaction between a significant amount of suspended molten corium mass and a water pool within the RPV.
- Fuel materials are predominantly solid, rather than molten. As such, debris fragmentation following a core relocation event is unlikely. Without a breakup of core materials, rapid thermal-energy transfer between fuel and coolant is difficult.
- Water volume and associated water mass in the RPV lower plenum is small. Small dimensions limit the potential for corium fragmentation and inhibit energy transfer to existing coolant.

RAI 19-33

To provide additional insight into the potential for a steam explosion to damage the RPV (and subsequently induce an alpha-mode containment failure), an analysis is performed postulating the occurrence of an in-vessel steam explosion. To

understand the release of energy as a result of the process, ~~a~~the Hicks-Menzies thermodynamic model of a steam explosion is used ~~that conservatively~~(Reference 19.2-19). The Hicks-Menzies model represents a thermodynamic maximum of the work potential of an expanding fluid. As such, calculated core debris energy and coolant expansion work is inherently conservative. The model assumes fuel and coolant achieve thermodynamic equilibrium in an adiabatic and isochoric process. Energy is then released as a result of fluid expansion during an isentropic process. ~~In the first process, coolant receives heat from the fuel debris and reaches its highest energy level. In the second process, energized coolant mass expands isentropically to fill the RPV volume. Throughout this process~~During expansion, coolant internal energy decreases as work is performed on the RPV. ~~The transition of thermal energy to mechanical energy between the equilibrium and final (full RPV volume) states is the work performed as a result of the expansion process.~~To demonstrate RPV integrity following a steam explosion, the energy associated with coolant expansion is compared to, and must be less than, the energy of a pressurized fluid within the RPV at the ultimate failure pressure of the vessel. This comparison serves as the basis to describe the capacity of the RPV against in-vessel FCI expansion work.

RAI 19-33

To evaluate the thermodynamic model of this phenomenon, ~~conservative values were selected for parameters related to~~parameter values were selected for corium mass, corium temperature, initial RPV system pressure, and coolant mass so ~~to ensure a bounding result with respect to expansion work potential. System configurations resulting in a non-conservative result (like venting through an RPV break or open valve) are ignored and assigned a configuration resulting in a bounding result (no venting through the RPV, for example).~~that the conversion of thermal energy in the fuel to mechanical work by the expanding coolant is maximized.

RAI 19-33

Severe accident sequences with an intact containment (i.e., break inside containment or break outside containment, but isolated) and core relocation are of interest for the evaluation of ~~a potential~~an in-vessel steam explosion. ~~Sequences with unisolated outside containment breaks result in lower RPV pressures and have, at most, an equivalent amount of coolant within the RPV. As such, these sequences are not bounding relative to the isolated containment sequences for an evaluation of steam explosion expansion work potential.~~Sequences with a breached or bypassed containment already contain a release pathway. As such, the alpha-mode containment failure is of no additional consequence. MELCOR cases satisfying this criterion are provided in Table 19.2-10. Within this table, each case is summarized and key characteristics, parameters, and initial state values relevant to the steam explosion analysis are provided.

From MELCOR results, observations of the NPM design were confirmed, including relocated core material configuration and the available coolant inventory. Because core relocation occurs as a result of support plate failure at temperatures less than the melting temperature of oxidic fuel materials, relocated corium within the RPV is largely a solidified mass, and not molten. Furthermore, limited coolant inventory

within the RPV reduces the potential for energetic steam explosions to perform work on the RPV during fluid expansion.

RAI 19-33

The result of the Hicks-Menzies thermodynamic analysis shows a high conversion ratio of thermal energy in the fuel to mechanical energy as coolant expands to fill the full volume of the RPV. All expansion energy is assumed to transfer to an upward liquid water slug and no energy loss is assumed for dissipation in the upper internals of the RPV. As a result, the energy applied to the upper head of the RPV is conservative. Furthermore, because the NuScale core is small, relocated fuel materials contain a relatively small amount of initial thermal energy. This limits the potential for coolant to perform work on the RPV during the expansion process. Consequently, work performed on the RPV by expanding coolant is insufficient to challenge vessel integrity.

RAI 19-33

Uncertainty in input parameters is considered by using the results of MELCOR accident simulations to inform the analysis of in-vessel FCI. A probability distribution for each input parameter is created using the minimum and maximum values from the MELCOR accident simulations presented in Table 19.2-10. A uniform distribution is applied between those bounds with a lognormal distribution describing extreme values of each distribution which are beyond the extreme values predicted by MELCOR simulations. Random samples for each input parameter are then acquired via Monte Carlo sampling and used to evaluate the Hicks-Menzies model. No combination of sampled input parameters result in failure of the RPV as a result of an in-vessel FCI.

RAI 19-33

With a significant amount of solid (versus molten) material in ~~the falling debris and the limited water in the RPV lower head~~relocated core debris, the potential for an in-vessel steam explosion is ~~remote~~highly unlikely. Further, a conservative thermodynamic analysis found that the energy released from a hypothetical steam explosion is insufficient to challenge RPV integrity. As a result, the alpha-mode of containment failure is not credible relative to an in-vessel FCI.

Ex-vessel Steam Explosion

RAI 19-33

~~For an ex-vessel steam explosion, the body of molten corium is in the RPV lower head, while the water pool is in what is referred to as the "reactor cavity." An ex-vessel steam explosion is judged to be physically unrealistic based on the size of the NuScale core, the physical dimensions of the CNV, the proximity between the RPV and CNV, the associated potential drop height for fuel between the two, and thermal-hydraulic conditions within the CNV in the postulated condition that the RPV were breached. If a failure to the RPV lower head were to occur (melted lower head), hot debris would immediately contact water in a shallow pool in the lower head of the CNV. For this relocation, there is insufficient fall height for the debris to achieve breakup and fragmentation upon entry into the shallow pool. As a result,~~

~~efficient transfer of thermal energy between fine fuel materials and coolant cannot take place and the conditions necessary for an energetic steam explosion are not established.~~As discussed in FSAR Section 19.2.3.2.1, failure of the RPV after a core damage accident involving an intact containment is not physically realistic. For such situations, a significant volume of liquid water will be present in the annular region between the CNV and the RPV to provide a cooling pathway from the core region to the reactor building pool. This volume of water prevents failure of the RPV lower head. As a result, an instantaneous interaction of fuel materials inside of the RPV and liquid coolant in the CNV is not credible and a quantitative analysis postulating such conditions was not performed.

RAI 19-33

However, from the perspective of demonstrating defense-in-depth, several aspects of the NuScale design preclude the possibility of an ex-vessel FCI:

RAI 19-33

- Considering a situation with an intact containment, the RCS water relocated to the CNV, and a failed RPV lower head, the distance between the bottom of the lower head of the RPV and the CNV is small and MELCOR accident sequences with an intact containment predict this space to be occupied by a water pool. An energetic FCI requires space between molten fuel materials, if present, and a water pool to promote material breakup. Breakup helps create a larger total surface area for contact with liquid coolant, thereby increasing rapid heat transfer. Because accident sequences with an intact containment contain a significant amount of liquid coolant in the annular region between the RPV and CNV, there is no available space between a failed RPV lower head and the water pool beneath to foster material breakup needed to promote an energetic transfer of heat to the water pool in the CNV.

RAI 19-33

- The CNV is not large enough to allow for a relocation of all core materials from the RPV to the CNV. Because of the limited space between the RPV and the CNV, a significant portion of the fuel material will remain backfilled within the RPV above a fuel mass in the CNV. This prevents fuel material from interacting with a water pool in containment. Coupled with the small size of the NuScale core, a relocation of fuel materials from the RPV to the CNV will involve less material than a similar FCI within the RPV, further limiting the potential energy transference necessary for an energetic ex-vessel steam explosion.

RAI 19-33

- Because of the large water pool predicted to reside in the containment annulus, the resultant conversion ratio for an ex-vessel FCI will be significantly less than the predicted ratio using the Hicks-Menzies thermodynamic model of an in-vessel FCI (which was shown to not challenge RPV integrity), thereby limiting the potential for work to be performed on the CNV by expanding coolant.

RAI 19-33

For these reasons, efficient transfer of thermal energy between fine fuel materials and coolant cannot take place and the conditions required for an energetic ex-vessel FCI are not established.

19.2-26 Investigation of Fire and Explosion Accidents in the Chemical, Mining, and Fuel-Related Industries-A Manual, Bulletin 680, Bureau of Mines, US Department of Interior, 1985.

19.2-27 M. G. Zabetakis, Flammability characteristics of combustible gases and vapors, Bulletin No. 627, Washington, U.S. Dept of the Interior, Bureau of Mines, 1965.

19.2-28 Z. M. Shapiro, T. R. Moffette, Hydrogen Flammability Data and Application to PWR Loss-of-Coolant Accident, Report WAPD-SC-545, 13th Edition, Bettis Plant, 1957.

19.2-29 NUREG/CR-3468, Hydrogen: Air: Steam Flammability Limits and Combustion Characteristics in the FITS Vessel, R3, December 1986.

19.2-30 NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories - An Overview," U.S. Nuclear Regulatory Commission, July 2006.

RAI 19-33

19.2-31 Theofanous, et al., "An Assessment of Steam-Explosion-Induced Containment Failure. Part 1: Probabilistic Aspects," Nuclear Science and Engineering, 97 (1987).

RAI 19-33

19.2-32 Corradini, et al., "Evaluation of Dynamic Pressures from Steam Explosions Applied to Advanced Light Water Reactors," Nuclear Science and Engineering, 168 (2011).

RAI 19-33

Table 19.2-10: MELCOR Cases for Fuel-Coolant Interaction (Steam Explosion)

MELCOR Case ID	Initiating Event	Corium Mass (lbm)	Corium Temperature (°F)	RPV Pressure (psia)	Coolant Liquid Volume (ft^3)	~~Coolant Vapor Mass (lbm)~~
TRN-07T	General reactor trip	12,300	2318	232.8	49.3	~~72.2~~
LEC-06T	Spurious opening of a single RVV	9,620	2822	136.9	50.9	~~146.9~~
LCC-05T-01	CVCS charging line LOCA inside containment	9,078	2782	143.5	52.8	~~112.0~~
LCC-05T-02	CVCS charging line LOCA inside containment	9,063	3133	187.8	48.5	~~253.5~~
LCC-05T-03	CVCS charging line LOCA inside containment	9,456	2770	205.6	51.1	~~110.6~~

Notes:
1. Three sensitivities involving the LCC-05T MELCOR case were used to define bounding input parameters for FCI analysis.
2. Parameters were taken at a time equal to or after core relocation for each MELCOR case to produce a bounding result.