NUSCALE
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RAIO-0519-65656

May 22, 2019

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9666 (eRAI No. 9666) on the NuScale Topical Report, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," TR-0915-17772, Revision 1

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9666 (eRAI No. 9666)," dated March 28, 2019
2. NuScale Topical Report, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," TR-0915-17772, Revision 1, dated March 2018

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

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 9666:

- 01.05-34
- 01.05-35
- 01.05-36
- 01.05-37
- 01.05-38

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9666 (eRAI No. 9666). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Steven Mirsky at 240-833-3001 or at smirsky@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. 9666, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9666, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0519-65678

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9666, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9666, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9666

Date of RAI Issue: 03/28/2019

NRC Question No.: 01.05-34

The following regulatory basis and discussion applies to all five questions in this request for additional information (RAI).

<u>Regulatory basis:</u> Emergency planning requirements are codified in 10 CFR 50.47 and 10 CFR Part 50 Appendix E. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius, or it may be determined on a case-by-case basis for reactors with an authorized power level less than 250 megawatts thermal (MWt). The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396, which was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high- consequence accidents, demonstration that the probability of exceeding a deterministic effect dose is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis, including the selection of accident sequences for source term calculations, can impact the results.

<u>Discussion</u>

NuScale Power, LLC submitted licensing topical report (LTR) TR-0915-17772-P, Revision 1, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," for review by the NRC staff. As stated in Section 3.0 of the LTR, EPZ size was optimized using a risk-informed approach and insights from the NuScale design specific probabilistic risk assessment (PRA).

<u>Question</u>

The methodology described in this LTR is based on the Combined License Applicant's (COL) PRA, but the existing review guidance for a COL PRA (Standard Review Plan Chapter 19.0 and Interim Staff Guidance DC/COL-ISG-028) is not sufficient to demonstrate the PRA is acceptable for use in risk-informed applications such as sizing the plume exposure EPZ. NuScale is asked to demonstrate how the PRA that will be used in the LTR methodology is acceptable for its intended use, including how numerical screening thresholds are affected by parameter and model uncertainties associated with a new un-built design. Staff notes that RG 1.200 provides an NRC accepted approach for determining the technical acceptability for PRA results for risk-informed activities.

NuScale Response:

Consistent with the guidance provided in Regulatory Guide (RG) 1.206, the combined license (COL) applicant should provide in Chapter 19 of the FSAR an adequate level of documentation to enable the NRC staff to determine the acceptability of the risks to public health and safety associated with operation of a proposed new plant. As stated in RG 1.206 Section 19.1.1.2.2 Risk-Informed Applications, the applicant should, “Identify and describe specific risk-informed applications being implemented during the COL application phase.” RG 1.206 Section C.I.19 provides qualitative guidance on PRA technical adequacy to support risk-informed applications in several subsections that include references to RG 1.200, SECY-98-144, and RG 1.174.

A COL applicant who wishes to apply the emergency planning zone sizing licensing topical report methodology should identify use of the PRA in a risk-informed application in the COL application. This is consistent with DCA COL Item 19.1-6: “A COL applicant that references the NuScale Power Plant design certification will specify and describe risk-informed applications during the operational phase (from initial fuel loading through commercial operation)”. The COL applicant will also need to demonstrate PRA technical adequacy (e.g., RG 1.200). Review of RG 1.200 technical adequacy expectations identfies two obvious areas as not being feasible for a new nuclear power plant design COL application: operational experience and plant walk down. A new design, by definition, will not have operational experience until it has been licensed, constructed, and operated. A plant walk down is not possible if the applicant has not yet constructed the plant. Therefore, for a new design COL risk-informed application, the applicant will need to demonstrate that the technical adequacy of the PRA is sufficient to support risk-informed decision making.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, as a result of this response.

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eRAI No.: 9666

Date of RAI Issue: 03/28/2019

NRC Question No.: 01.05-35

Question

The methodology described in this LTR uses a seismic screening threshold of 1.67 times the ground motion acceleration of the design- basis safe-shutdown earthquake while all other hazards use a screening threshold based on core damage frequency (CDF). Staff notes this PRA-based seismic margins approach was specifically approved in the SRM to SECY 93-087 for design certification and combined license applications. It is stated to be useful in developing the reliability assurance program, identifying operator training requirements, and focusing on accident management capabilities. The Commission did not approve its use in risk-informed applications such as establishing the plume exposure EPZ. Per the LTR methodology, all structures in rev. 2 of the NuScale DCA would screen out, but if the CDF screening threshold was applied using the seismic CDFs from the NuScale SAMDA analysis, seismic events would screen in. Screening out seismic risk is inconsistent with the SRM to SECY-04-0118, Phased Approach to Probabilistic Risk Assessment Quality, which states if there is a PRA standard for a hazard group, it should be used to assess risk for risk-informed applications. The staff cannot make a finding the methodology is acceptable for risk-informed applications if seismic hazards have been screened out. The staff requests that NuScale justify why a seismic PRA is not needed to determine the EPZ size for early protective actions (evacuation and sheltering) in order to provide dose savings and protect the public.

NuScale Response:

The staff has accepted the use of a PRA-based seismic margin assessment (SMA) for the design certification and combined license application as indicated in DC/COL-ISG-028.

SECY-04-118 states on page 4 that “the licensee may decide on a risk-informed activity that does not utilize all the available PRA standards and regulatory guidance”. Therefore, it is NuScale's position that using the SMA is a valid approach for an emergency planning zone (EPZ) sizing application, based on DC/COL-ISG-028 and SECY-04-118.

Screening of seismic risk should not be dependent on a seismic PRA. Seismic hazard curves are based on 300 years of reported data with informed extrapolation of any seismic frequencies of 1E-4 per year and lower based on ground liquefaction. Unlike internal hazards, the uncertainty for seismic peak ground acceleration (PGA) at frequencies less than 1E-4 per year is very large.

Instead, the screening should be based on the PGA of a potential earthquake to support pre-planned emergency planning that could be effective. At high PGAs, both the NuScale plant and the surrounding area would be affected by a similar ground motion level, including emergency response and communication equipment and evacuation routes. Due to the high seismic ruggedness of the plant (as demonstrated by the SMA), the potential incremental risk posed by the NuScale plant is a negligible contributor to public health effects associated with a severe earthquake, independent of its frequency of exceedance. As stated in NUREG-1738 (p. 3-37), “For high PGA earthquake, it was reasoned that there would be no effective evacuation and many structures would be uninhabitable.” The infrastructure relied upon for pre-planned evacuation, including roads, bridges, buildings for temporary housing, and communication equipment, is likely to be damaged. NUREG-1738 equated a high PGA earthquake to a PGA of approximately 0.5 g. While NUREG-1738 addressed spent fuel pool accident risk for decommissioned plants, the same reasoning about high PGA earthquakes applies equally to emergency planning considerations for operating plants. This is not to say that no response, such as sheltering or evacuation, should be attempted in the event of a high PGA earthquake, but rather that the pre-planned response as defined in emergency plans would be significantly less effective than an integrated national response, which assesses and accounts for damaged infrastructure over a wide area.

The NuScale EPZ methodology, provided in NuScale licensing topical report TR-0915-17772, conforms with the technical and regulatory basis for the current NRC-approved EPZ for the operating fleet, which stems from analyses in NUREG-0396 and NUREG-0654. During the development of the NuScale methodology, a detailed review of documentation supporting the current EPZ showed that seismic events, seismic PRA, and SMA were not considered in determining the 10-mile plume exposure distance for EPZ. The following facts delineate NRC policy on EPZ size and seismic risk:

- The current operating NRC-licensed fleet all have a 10-mile plume exposure EPZ regardless of safe shutdown earthquake (SSE) PGA or seismic risk at each site.Some sites have significantly higher SSEs and seismic risks than others, but all are licensed to the same plume exposure EPZ.
- Some sites have multiple co-located nuclear power plant units while others have only one nuclear power plant, but all of these sites have the same EPZ regardless of seismic risk.
- Several nuclear power plants had been granted a 5-mile EPZ by the NRC in the past. These plants were located at sites with widely differing seismic risks and SSEs, but were all operated with the same smaller 5-mile EPZ.
- Decommissioned nuclear power plants with large inventories of used nuclear fuel in their pools have all been granted a site boundary EPZ with no discernible or documented consideration of differences in seismic risk.
- Independent spent fuel storage installations (ISFSIs), either co-located with operating nuclear power plants or at orphan sites where the nuclear power plant has been decommissioned have all been licensed by the NRC with no EPZ nor any differentiation between seismic risk at different ISFSI sites.

The aforementioned licensing actions substantiate that seismic risk is not a determinant in emergency planning; that the acceptable seismic design of the plant (i.e., SSE) meets all regulatory requirements for public health and safety; and that extreme beyond-design-basis low frequency seismic events are not considered in determining an acceptable EPZ. This regulatory precedent over several decades applies equally to the NuScale EPZ methodology. NuScale has selected the SMA and our inherently high SSE as one of many criteria in justifying the EPZ. This is beyond the basis that has and is being used by the NRC.

In summary, NuScale's position is that both a PRA-based SMA and a seismic PRA are acceptable for evaluating seismic risk, and the SMA is appropriate for use in the risk-informed application of determining EPZ size. NuScale also wishes to reiterate that applications of the methodology, such as screening results based on the DCA, should not be used to evaluate the merits of the methodology.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, as a result of this response.

Response to Request for Additional Information
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eRAI No.: 9666

Date of RAI Issue: 03/28/2019

NRC Question No.: 01.05-36

Question

The Commission goals for advanced LWRs include two risk metrics, CDF and large release frequency (LRF), as specified in the SRM to SECY 90-016. The LTR methodology screens against a CDF threshold, but only appears to consider LRF if the event screens into the defense- in-depth process. Insights from the Level 2 PRA, including release timing, size of release, and risk significant structures are important and should be considered consistent with the Commission goals. NuScale is asked to justify how risk insights from the Level 2 PRA are considered in the methodology, including LRF for sequences that don't screen into the defense-in-depth process.

Additionally, LTR Section 3.8.2, Severe Accident Phenomena, concludes that severe accident phenomena do not need to be further considered in the EPZ methodology because the NuScale Design Certification Application (DCA) found them to either be not credible or to not pose a threat to containment integrity. Staff notes that in a Feb 26, 2019 supplemental response to DCA RAI 9108 (ML19057A618), NuScale revised portions of the DCA related to severe accident phenomena to clarify the presence of analysis uncertainty and remove terms such as "not physically credible." Staff requests that NuScale explain how these severe accident uncertainties are captured in the LTR methodology. Staff also requests that NuScale change the terminology in the LTR (specifically in LTR Sections 3.4.3 and 3.8.2) to be consistent with the DCA.

NuScale Response:

Large Release Frequency

The emergency planning zone (EPZ) methodology implicitly considers a criterion for large release frequency (LRF). The Commission's safety goals are a total core damage frequency (CDF) of less than 1E-4 per reactor year and a total LRF of less than 1E-6 per reactor year. NuScale's EPZ methodology initially screens at a conservative CDF of {{ }}$^{2(a),(c)}$. Considering that LRF cannot exceed CDF, a CDF limit is automatically also a screening limit on LRF. It would have been conservative and reasonable to set the CDF screening limit at 1E-6, which is two orders of magnitude below the safety goal, and a LRF screening limit of 1E-7 by incorporation of the additional goal that the conditional containment failure probability is no greater than 10 percent. By setting a screening criterion of {{ }}$^{2(a),(c)}$ for CDF and implicitly LRF, NuScale's methodology greatly exceeds the safety goals. Therefore, it would be redundant to explicitly include a separate LRF screening criterion.

Risk insights from a Level 2 PRA can include size of release and release timing, as mentioned in the RAI question, with a focus on primary vessel and containment integrity. As discussed in licensing topical report TR-0915-17772, EPZ analyses are best-estimate and are supported by an uncertainty analysis. Thus, release timing and magnitude, as well as containment status, will be simulated for any accident sequence that is screened in to the EPZ technical basis.

Severe Accident Phenomena

Section 3.8.2 of TR-0915-17772, Revision 1, was intended to explain why the severe accident phenomena were not described in the methodology report. However, as written, the section erroneously implies that the phenomena are not considered in the methodology. Due to the possibility for misinterpretation, Section 3.8.2 has been revised.

Section 3.4.3 is explicit in saying that "If containment integrity is not maintained (i.e., containment bypass loss-of-coolant accident (LOCA), containment isolation fails, or *containment is otherwise breached*) the accident sequence is classified as "more severe" and dose evaluation is performed using the methodology of Section 4.2.3" [emphasis added]. While the section does go on to discuss results from the design certification PRA, this is only to explain the use of terminology in the report, and is not intended to make a final EPZ size determination at the design certification phase. However, the methodology addresses what is done if containment integrity is lost for any reason. For clarity, minor revisions are also made to Section 3.4.3.

The supplemental RAI responses referenced by the staff support NuScale's analysis insights that containment failure due to severe accident phenomena does not occur; the qualitative terminology used in TR-0915-17772 is consistent with NuScale's analytical insights. While there is uncertainty associated with any analysis, the significant conservatisms employed in the analyses outweigh uncertainty such that conclusions are not impacted.

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, has been revised as described in the response above and as shown in the markup provided in this response.

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}}2(a),(c) For perspective, the probability associated with one in a billion years is orders of magnitude lower than the probability of civilization ending, catastrophic events such as meteor strikes. Appendix C contains an example of calculating the total CDF.

3.4.3 Final Classification of Accidents by Severity

The final step of the methodology is to classify all screened-in accident sequences by containment integrity. If the sequence does not include a loss of containment integrity (i.e., core damage is not associated with containment failure), the accident sequence is classified as "less severe" and assessed against the dose criteria of Section 3.2. The effect of an intact containment is that the only potential radionuclide release to the environment is by nominal containment leakage. The maximum allowable containment leakage is 0.2 percent volume/day, which is why intact containment can be equated with "less severe." If the methodology results in the screening out of every less severe accident sequence, the highest frequency intact containment accident sequence will be retained for source term and dose evaluation.

If containment integrity is not maintained for any module involved in an accident (i.e., containment bypass loss-of-coolant accident (LOCA), containment isolation fails, or containment is otherwise breached) the accident sequence is classified as "more severe" and dose evaluation is performed using the methodology of Section 4.2.3.

The NuScale containment has been designed to reduce the possibility of containment failure, consistent with regulatory expectations for advanced reactors. For example, there are multiple isolation valves on all containment piping penetrations. Additionally, the main steam and feedwater piping is rated for RPV design pressure up to the second isolation valve. Due to this design, it is appropriate to consider any accident sequence with a loss of containment integrity to be more severe.

~~Generally, the "more severe" sequences would also include those cases where the containment has failed as a result of severe accident phenomena. However, i~~In the NuScale design certification PRA, analyses predicted that containment integrity was not threatened by the occurrence of severe accident phenomena~~no physically credible containment failure mechanism (other than bypass) from severe accident phenomena could be identified~~. Hence in this method, "bypass" is sometimes used to denote failure of the containment function since it is the primary mechanism by which the containment function fails. However, the application of the methodology does not utilize the design certification PRA and assessment of containment integrity will be performed with the PRA which is associated with the application.

If the methodology results in the screening out of every more severe accident sequence, there is no requirement to include a more severe source term and dose evaluation.

If there is uncertainty as to the integrity of containment (e.g., a dropped module may or may not result in a breached containment), the accident sequence should be considered both less and more severe. In this case, analyses against both dose criteria shall be performed according to the following methods.

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}}2(a),(c)

3.5 Methodology for Sequence-Based Assessment of Defense-in-Depth

3.5.1 Introduction

This section addresses the methodology for an assessment of defense-in-depth for core damage accident sequences which, based on plant-specific PRA information and the

3.8.2 Severe Accident Phenomena

Relevant severe accident phenomena, including in-vessel retention, fuel-coolant interaction, hydrogen combustion in containment, and high pressure melt ejection have been assessed for the NuScale design as part of the DCA. ~~Each phenomena has been determined to either be not credible or does not pose a threat to containment integrity.~~ As such, these severe accident phenomena are not ~~considered further~~discussed in detail in the EPZ methodology. Current analyses predict that these severe accident phenomena either do not occur, or do not challenge containment integrity. However, the application of the EPZ methodology should consider the assessment of severe accident phenomena available at that time. As discussed in Section 3.4.3, if containment integrity is not maintained for any reason, including occurrence of severe accident phenomena, the accident sequence is classified as "more severe".

3.9 Security Events

Security events are addressed for completeness for EPZ.

3.9.1 Design-Basis Threat

NuScale designed the plant to address the regulatory requirements for the design-basis threat by incorporating security-by-design. The NuScale design reduces the number of safety systems, thereby reducing the number of potential targets. For the remaining safety systems, most of the safety-related components have been located below grade. As the safety systems are passive in design, there is no reliance on operator actions, electrical power, or the addition of water to maintain the safety of the reactor cores or spent fuel. The COL applicant referencing the NuScale standard plant design will build upon the enhanced security features by developing a site-specific strategy to protect against radiological sabotage, as outlined in Purpose and Scope, 10 CFR 73.1 (Reference 6.3.15).

3.9.2 Beyond-Design-Basis Events

3.9.2.1 Aircraft Impact

All new plants to be built and operated in the United States must meet the regulatory requirements for aircraft impact in Aircraft Impact Assessment, 10 CFR 50.150 (Reference 6.3.14). These regulations require that all new plant applications must:

1. Perform a design-specific assessment of the effects on the nuclear power plant facility of the impact of a large, commercial aircraft; and
2. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions:
   a. the reactor core remains cooled, or the containment remains intact; and

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NRC Question No.: 01.05-37

Question

The methodology discussed in Section 3 of the LTR classifies accidents as "less severe" if the sequence does not include containment failure and evaluates these accidents to early phase PAGs. It classifies accidents as "more severe" if containment integrity is not maintained and evaluates these accidents for early severe health effects. This approach implies more severe accidents are less frequent, which, while appropriate for the large light water reactors evaluated in NUREG- 0396, does not seem to be consistent with the unique NuScale design where the containment is not a permanent structure, but regularly transported for refueling. NuScale module drop events are the most likely cause of core damage for the NuScale design, so staff would expect this accident to be considered a more likely event and evaluated to the early phase PAGs. However, this event results in a loss of containment integrity so it is only evaluated for early severe health effects. Staff requests that Nuscale explain and justify how the methodology considers the dose consequences for each of the three module drop scenarios described in DCA Chapter 19.1.6.2, including those that are evaluated, but not quantified in the DCA. Additionally, NuScale is requested to explain and justify how the methodology evaluates the dose consequences from the DCA Chapter 19.1.7.4 module drop accident that impacts one or two other modules.

NuScale Response:

Introduction

The underlying philosophy of the plume exposure pathway (PEP) emergency planning zone (EPZ), as established in NUREG-0396, is to provide protection to the public from a range of potential nuclear power plant accidents. NuScale agrees with the philosophy of NUREG-0396, and therefore the NuScale PEP EPZ methodology provided in licensing topical report TR-0915-17772 is designed to be as consistent as possible with the original basis of establishing the PEP EPZ distance in NUREG-0396, while simultaneously being unambiguous and reproducible. NuScale's EPZ methodology is designed to be standalone and produce an EPZ distance that provides the same level of public protection as the existing EPZ basis, regardless of the probabilistic risk assessment (PRA) used as input. Therefore, the NuScale methodology is not designed around the design certification application (DCA) PRA and is not intended to be applied with the DCA PRA. Any discussion involving the DCA PRA in this response is purely to provide context of one potential application of the methodology, and not to justify a specific result or EPZ distance.

NuScale's response is separated into the following areas:

1. NuScale's evaluation of NUREG-0396 and an explanation and justification of how NuScale's EPZ methodology, including application of dose thresholds, is designed to be consistent with the original EPZ technical basis in NUREG-0396.
2. NuScale describes how our EPZ methodology would consider the different module drop scenarios in the DCA PRA and how a user of our methodology would evaluate the dose consequences of module drop scenarios involving a single module based on the DCA PRA.
3. NuScale provides an example of how a user of our methodology would evaluate the dose consequences of module drop scenarios that impact multiple modules based on the DCA PRA.

Part 1: Dose thresholds in the NuScale methodology and NUREG-0396:

Despite the discussion of the results in the main body of NUREG-0396, the evaluation method employed and documented in the appendices of NUREG-0396 does not differentiate between intact containments and failed containments in the treatment of beyond design basis core damage accidents. The first priority of the EPZ is the reduction of early severe health effects, and therefore all beyond design basis accidents were evaluated against the 200 rem dose criteria. The only differentiation made in the NUREG-0396 method was between design basis accidents (DBAs) and beyond design basis accidents, with the DBAs being evaluated against the protective action guides (PAGs). Additionally, the EPZ size was somewhat determined (note that the NUREG-0396 results are not entirely consistent with the evaluation method used) by using frequency-weighted dose consequence results that can be loosely interpreted as best-estimates (and not bounding). Any distinctions between more-severe/less-severe or more-likely/less-likely, are simply for ease of discussion and were not relevant to the actual analyses performed. Simply put, all beyond design basis core damage accidents were evaluated against the 200 rem criteria regardless of the status of the containment or accident likelihood.

Because there are no DBAs that result in core damage in the NuScale design, beyond design basis accidents with intact containment are used as surrogates to be evaluated against the more strict PAG criteria in the NuScale EPZ methodology. Beyond design basis core damage accidents with a failed containment are treated as analogous to the beyond design basis accidents in NUREG-0396 and evaluated against the 200 rem early severe health effects threshold. Consistent with NUREG-0396, the EPZ size established using the NuScale EPZ topical methodology will provide protection from a range of potential nuclear power plant accidents.

Part 2: An example of the application of the NuScale methodology with DCA PRA module drop accident involving a single module:

The NuScale EPZ methodology is intended to be used with a PRA at the COL stage; however, the PRA from the NuScale DCA is used throughout this response solely as an example. As discussed in Sections 3.4.1 and 3.4.2 of TR-0915-17772, sequences are compiled and screened for all modes and all hazards that contribute to the core damage frequency (CDF) evaluated as described in Section 3.4.2 of TR-0915-17772.

Section 19.1.6.2 of the NuScale FSAR discusses the consequences of a module drop but does not describe different module drop scenarios; however, FSAR Section 19.1.6.1.2 describes three potential module drop scenarios as initiating events: 1) a fully assembled module is dropped, 2) a partially assembled module without the lower containment vessel (CNV) is dropped, and 3) the upper vessels (i.e., the upper portions of the reactor pressure vessel (RPV) and CNV) are dropped. Because the DCA low power and shutdown (LPSD) PRA screens potential events of type 2 and 3 from consideration in the CDF, as discussed in FSAR Section 19.1.6.1.2, they would not be considered in the NuScale EPZ methodology. While the type 2 and 3 events may cause mechanical damage to some portion of the fuel assemblies in the RPV lower head, because these events maintain adequate heat removal, core damage is precluded.

A type 1 event that results in a module resting horizontally on the pool floor is conservatively assumed in the DCA PRA to result in a loss of containment integrity in a manner that results in core damage; based on this analysis, the type 1 module drop accident would be considered more severe following the TR-0915-17772 Section 3.4.3 methodology.

As discussed in part 1 of this response, a comparison to the dose threshold for early severe health effects is appropriate for all severe accidents.

<u>Part 3: An example of the application of the NuScale methodology with module drop scenarios involving multiple modules</u>

If a type 1 module drop is screened-in at the application phase, potential multi-module sequences would be identified according to Section 3.7 of TR-0915-17772.

Sequences that are screened-in are then evaluated based on their severity, as described in Section 3.4.3 of TR-0915-17772. If containment integrity is maintained in all modules, the sequence is considered less severe. If containment integrity is not maintained for any module, the sequence is considered more severe.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, as a result of this response.

Response to Request for Additional Information
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eRAI No.: 9666
Date of RAI Issue: 03/28/2019

NRC Question No.: 01.05-38

<u>Question</u>

Section 3.5 of the LTR includes an assessment of accident sequence defense-in-depth to determine if the sequence prevention and mitigation capabilities are consistent with the defense-in-depth philosophy. The methodology ranks attributes such as risk metrics or design features as having high, medium, or low defense-in-depth based on specific criteria. The low ranking is described as the lowest valuation that would be expected for advanced light water reactors (LWRs). For the staff to find that defense-in-depth has been appropriately considered, consistent with the Commission PRA policy statement (60 FR 42622, August 16, 1995), the staff requests that NuScale justify how the following definitions of low meet expectations for advanced LWRs:

1. For the "Containment Isolation Response" attribute, the definition of low as "only check valves" appears to be inconsistent with intersystem loss-of-coolant accident guidance in the SRM to SECY-93-087. SECY-93-087 expects that systems that have not been designed to full RCS pressure should include the capability for leak testing of the pressure isolation valves and valve position indication that is available in the control room when isolation valve operators are de-energized.

2. For the "Sequences LRF" attribute, the definition of low as > 1E-6 per module year is inconsistent with the SRM to SECY-91-06, which sets a LRF Goal that total LRF < 1E-6 per module year.

3. For the "Safety system response to detect and control initiating event" attribute, the definition of low as an active system with manual control seems inconsistent with the Advanced Reactor Policy Statement which expects highly reliable and less complex shutdown and decay heat

removal systems. The Commission encourages the use of inherent or passive means to accomplish this objective.

4. For the "Time to the beginning of core damage" attribute, the definition of low to be less than one hour seems inconsistent with the Advanced Reactor Policy Statement which expects longer time constants and sufficient instrumentation to allow for more diagnosis and management before reaching safety system challenges.

NuScale Response:

In the RAI, the NRC staff points out that some of the attributes for characterizing *low* defense-in-depth (DID) are inconsistent with NRC expectations; for the purpose intended for the DID evaluation in determining emergency planning zone (EPZ) size, this is intentional. Specifically, in this context, *an overall DID characterized as low would screen-in the core damage sequence into the EPZ size determination process*.

The quantitative DID portion of the method is provided as a means for validating the numerical results of the PRA and the veracity of the numerical screening that also is part of the EPZ size determination method. As stated in the second paragraph of EPZ licensing topical report TR-0915-17772 Section 3.5.1, the purpose of the DID evaluation process is to substantiate the low accident sequence frequencies being reviewed. Specifically, the DID analysis is aimed at being a substitute for a formal process for establishing the PRA technical adequacy. If it is a prerequisite that only a fully reviewed and approved PRA that has been judged to be technically adequate can be used to support the EPZ sizing method, the fundamental purpose of the DID portion of this process becomes extraneous and unnecessary. Therefore, based on NuScale's response to Question 01.05-34 of this RAI with regard to technical adequacy of the PRA within the context of RG 1.200, the DID evaluation section of TR-0915-17772 should be deleted. DID should be considered qualitatively by the NRC to the same extent that it has and is being used for all EPZ related decisions.

Further, the proposed DID evaluation process was designed to be design-neutral and not specifically tailored to the NuScale design (or to any advanced light water reactor design). Given that it is impossible to predict the details of future designs, or the details of future licensing decisions, it is imprudent to impose conservative expectations for the DID criteria (note that this applies only to the *medium* and *high* valuations since those are what determine which sequences can be safely screened-out of the EPZ size determination method). Given the potential variation in plant designs, the DID evaluation process has a limited amount of flexibility designed into it such that a single *low* valuation could be offset by two *high* valuations.

Guidance published by the NRC is provided to applicants and licensees as one potential path to satisfying rules and regulations. There could be other paths to satisfying NRC rules and regulations besides following guidance (including exemptions).

Based on the above reasoning, the entire DID methodology is judged to be acceptable without change. Additionally, due to establishing PRA technical adequacy, the DID assessment could be removed. However, each attribute identified in the RAI is given additional justification as to the “low” valuation, in the event the methodology is maintained.

Item 1 - Criterion 4 focuses only on the containment isolation valves (not the design of the connected system). In the NuScale design the connected systems are typically rated to full reactor coolant system pressure and temperature. Any downstream systems that are rated to a lower pressure or temperature would have additional potential isolation opportunities. Therefore, this attribute valuation is not inconsistent with the NRC guidance.

Item 2 - NuScale agrees that the expectation is for large release frequency (LRF) to be lower than 1E-6/year for advance reactor designs. This is why the Criterion 5 identifies a LRF greater than 1E-6/year as a *low* valuation, which for the purpose of validating the separate numerical screening, would likely result in the core damage sequence being screened-into the EPZ sizing determination.

Item 3 - As stated in the footnote to Table 3-3, the expectation is that there will likely be multiple systems available to respond to each initiating event. Each system (responding to an initiating event and appearing in a core damage accident sequence) would be evaluated separately, even those that are active systems and are actuated manually. If only one, manually controlled system were available to respond to an initiating event, in all likelihood it would screen-into the EPZ process as a result of the sequence frequency numerical screening that is a separate but integral part of the overall EPZ sizing method.

Item 4 - As noted in the NRC statement of this issue, the NRC Policy Statement states that one of the attributes that should be considered in advanced designs is: “Longer time constants and sufficient instrumentation to allow for more diagnosis and management before reaching safety systems challenge and/ or exposure of vital equipment to adverse conditions.” There is no definition of “longer time constants”, rather it is left as a matter of judgment. Criterion 4 evaluates anything less than 1 hour time to core damage as a *low* valuation. Given that there are many (typically very conservative) assumptions that factor into estimating the time to core damage, setting the *low* valuation at less than one hour, does not seem unreasonable. For example, the definition of “core damage” can vary. Additionally, providing the control room operators with one hour or more (for a *medium* valuation) would allow for thoughtful and deliberate diagnosis and management of most accident scenarios.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, as a result of this response.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0519-65678

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NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
2. I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
   a. The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
   b. The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
   c. Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
   d. The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
   e. The information requested to be withheld consists of patentable ideas.
3. Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the method by which NuScale develops its plume exposure emergency planning zones at a NuScale small modular plant site.

   NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

   The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

   If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

4. The information sought to be withheld is in the enclosed response to NRC Request for Additional Information No. 9666, eRAI No. 9666. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
   a. The information sought to be withheld is owned and has been held in confidence by NuScale.
   b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
   c. The information is being transmitted to and received by the NRC in confidence.
   d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
   e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 22, 2019.

Zackary W. Rad