POLICY ISSUE
(Information)

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FOR: The Commissioners

FROM: Margaret M. Doane
Executive Director for Operations

SUBJECT: STAFF APPROACH TO EVALUATE ACCIDENT SOURCE TERMS FOR THE NUSCALE POWER DESIGN CERTIFICATION APPLICATION

PURPOSE:

The purpose of this paper is to inform the Commission about 1) NuScale Power LLC's (NuScale's) proposed novel approach to addressing accident source terms in its design certification application, and 2) the U.S. Nuclear Regulatory Commission (NRC) staff's plan for evaluating the use of different source terms within the NuScale design certification review. This paper is consistent with SECY-10-0034,[1] for the staff to inform the Commission of this novel technical issue concerning accident source terms. This paper does not address any new commitments or resource implications.

SUMMARY:

The Commission expects that power reactors reflect, through design, construction, and operation, an extremely low probability of accidents.[2] A key feature of the NRC's regulatory framework for power reactors is the evaluation of radiological impacts on members of the public, plant personnel, and plant equipment. To support these evaluations, applicants and staff make assumptions about the specific composition and progression of radionuclide sources that can cause these impacts.

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[1] *See* SECY-10-0034, *Potential Policy, Licensing, and Key Technical Issues for Small Modular Reactor Designs*, at 2 (March 28, 2010) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML093290268).

[2] *See* Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(1)(ii),10 CFR 52.47(a)(2).

Using these source terms,[3] applicants then ensure that plant equipment and structures needed for accident prevention and mitigation can perform their intended functions; health and safety objectives for offsite public are met; and control room operator radiological doses are maintained within appropriate limits. As discussed further in this paper, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" require evaluations of, and in some cases design features to address, consequences derived from source terms that are based on a range of accident scenarios.

To demonstrate compliance with requirements associated with offsite dose consequences, control room habitability, and environmental qualification, applicants have historically used conservative bounding source terms that were developed based on NRC guidance.[4] These bounding source terms were based on considering a range of accident-specific source terms and have reflected a core melt accident.[5] Although the NRC has considered the use of source terms for advanced non-light-water power reactors that do not reflect a core melt accident, the Commission has not directly addressed the application of a source term that does not consider core melt in past regulatory applications for light-water power reactors.[6]

NuScale does not intend to use a core damage source term (CDST) to support its evaluation of certain radiological impacts associated with its light-water reactor design. For example, in the course of addressing staff questions associated with its source term assumptions, NuScale identified testing and analysis challenges associated with verifying the environmental qualification of certain accident mitigation equipment and features under a source term that includes consideration of a core melt accident. Some NRC regulations, including those associated with environmental qualification of electrical equipment important to safety and control room habitability, do not refer to use of a core melt accident in determining the requisite source term. Therefore, in its design certification application, NuScale plans to evaluate different accident source terms for the different analyses required by NRC regulations. This approach would result in the use of non-core melt source terms for regulatory applications that have historically considered core damage in the development of the associated accident source terms.

In general, the staff finds that using different accident source terms to satisfy separate regulatory requirements would be acceptable with adequate justification. Although applicants in the past have generally considered a CDST to address various radiological impact analyses, using different accident source terms to satisfy specific regulatory requirements is a more flexible approach that is permitted by the regulations. As discussed further in this paper, the

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3 "Source term" is defined in 10 CFR 50.2 as "the magnitude and mix of radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release." The staff's use of this term throughout the paper is consistent with this definition.

4 *See* RG 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors* (July 2000) (ADAMS Accession No. ML003716792); RG 1.195, *Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors* (May 2003) (ADAMS Accession No. ML031490640).

5 The term "core melt accident" is used in this paper to denote an accident involving significant core damage and fission product release from an overheating core, which may or may not involve significant core melting. NuScale's current white paper submittal WP-0318-58980, *Accident Source Terms Regulatory Framework*, Rev. 1 (January 31, 2019) (ADAMS Accession No. ML19032A146), uses the phrase "core damage source term." The staff considers the terms "core melt" and "core damage" to be equivalent for the purposes of this discussion.

6 *See infra* p. 10.

staff finds that NuScale's proposal to use an accident source term without consideration of a core melt for environmental qualification, if technically justified, is consistent with 10 CFR 50.49 and presents an acceptable regulatory approach to addressing accident source term. The staff has not yet made a technical determination regarding the adequacy of NuScale's proposed approach to environmental qualification accident source term. The staff's ultimate findings and technical conclusions of NuScale's proposed approach to environmental qualification accident source term will be described in its safety evaluation report and will be based on NuScale's revision of its Topical Report (TR) on this issue, TR-0915-17565, and subsequent revisions to the NuScale design certification application.[7]

NuScale has also proposed using a source term that does not consider core melt for other regulatory applications, such as certain aspects of the control room habitability analysis. Specifically, NuScale has proposed using a source term that does not consider core melt to demonstrate compliance with General Design Criterion (GDC) 19, "Control Room," access and occupancy requirements. Additionally, NuScale has proposed using source terms that do not consider core melt and a source term that does consider core melt, to assess offsite dose consequences under 10 CFR 52.47(a)(2)(iv). NuScale categorizes a core melt accident as a "beyond-design-basis" event in its submissions and states that "[t]he classification of [a core damage event] CDE as beyond-design-basis is appropriate in order to consistently define the design basis of the plant."[8] As discussed further in this paper, the applicable NRC regulations do not require classification of source terms as "design basis" or "beyond-design-basis" to demonstrate compliance with the requirements. Therefore, the staff has determined the classification of a core melt accident as a "beyond-design-basis" event for the NuScale design is not material to the staff's findings under these regulations and could lead to confusion about the underlying licensing basis for the design. However, based on the information that NuScale intends to provide to demonstrate compliance with the control room habitability requirements and the offsite dose consequence analyses, the staff believes that it has a path forward to make a regulatory finding under GDC 19 and 52.47(a)(2)(iv) if appropriate technical detail is included.

<u>BACKGROUND</u>:

NuScale has developed and shared with staff several proposals to address accident source terms for its small modular reactor design.[9] Beginning in January 2018, the NRC has held public meetings to discuss various options NuScale has proposed to satisfy NRC regulations related to accident source terms.[10]

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[7] NuScale submitted Revision 3 of TR-0915-17565, "Accident Source Term Methodology" on April 21, 2019. The submittal is currently under staff review. (ADAMS Accession No. ML19112A172).

[8] *See* Revision 3 of TR-0915-17565, "Accident Source Term Methodology" at 134 (April 21, 2019) (ADAMS Accession No. ML19112A172). In NuScale's final safety analysis report, it classifies the core damage event used to evaluate radiological consequences under 10 CFR 52.47(a)(2)(iv) as a special event, which includes "those beyond design basis events that are explicitly defined by regulation." *See Final Safety Analysis Report Chapter Fifteen, Transient and Accident Analyses*, Rev. 2 (October 2018) (ADAMS Accession No. ML18310A337).

[9] WP-0318-58980, *Accident Source Terms Regulatory Framework* (May 15, 2018) (ADAMS Accession No. ML18136A850); TR-0915-17565, *Accident Source Term Methodology*, Rev. 1 (April 8, 2016) (ADAMS Accession No. ML16099A394); WP-0318-58980, *Accident Source Terms Regulatory Framework*, Rev. 1 (January 31, 2019) (ADAMS Accession No. ML19032A146); TR-0915-17565, *Accident Source Term Methodology*, Rev. 2 (September 11, 2017) (ADAMS Accession No. ML17254B068).

[10] January 23, 2018 (ADAMS Accession No. ML18022A140); June 7, 2018 (ADAMS Accession No. ML18173A260); June 27, 2018 (ADAMS Accession No. ML18206A933); August 9, 2018 (ADAMS Accession No. ML18240A210); August 29, 2018 (ADAMS Accession No. ML18249A261); December 12, 2018 (ADAMS Accession

There are differences in the language associated with source terms within NRC regulations, as discussed below. Accident source term evaluations pertinent to the staff's review of NuScale's design certification application include:

- Offsite radiological analysis: 10 CFR 52.47(a)(2)(iv) requires an applicant assume a "fission product release from the core" and demonstrate that specified doses would not be exceeded.

- Additional Three Mile Island (TMI)-related requirements: 10 CFR 50.34(f)(2)(vii) requires design for access and equipment protection from the radiation environment associated with an accident based on the consideration of "accident source term radioactive materials."

- Environmental qualification of electric equipment important to safety: 10 CFR 50.49(e)(4) requires environmental qualification[11] of safety related structures, systems, and components to address a radiation environment based on the "most severe design basis accident during or following which the equipment is required to remain functional."

- Control room habitability: Part 50, Appendix A, Criterion 19, as incorporated by reference in 10 CFR 52.47(a)(3)(i), requires radiation protection for the control room so that operators do not exceed 5 rem total effective dose equivalent under "accident conditions, including loss-of-coolant accidents."

The footnotes associated with 10 CFR 50.34(f)(2)(vii), 10 CFR 52.47(a)(2)(iv), and other regulations[12] state that the fission product release source term should be "based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events." These footnotes go on to state that these "accidents have generally been assumed to result in substantial meltdown of the core" with subsequent release of "appreciable quantities of fission products." Although 10 CFR 50.49 does not include a similar footnote, all power reactor license applicants to date have considered a core melt accident source term for both the 10 CFR 50.49 environmental qualification evaluation and those associated with GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 52.47(a)(2)(iv).

No. ML18351A110); February 14, 2019 (ADAMS Accession No. ML19065A124); February 27, 2019 (ADAMS Accession No. ML19065A124); April 9, 2019 (ADAMS Accession No. ML19134A363).

[11] RG 1.89, Section B states: "for the purposes of this guide, "qualification" is a verification of design limited to demonstrating that the electric equipment is capable of performing its safety function under significant environmental stresses resulting from design basis accidents in order to avoid common-cause failures." Note that in the context of RG 1.89, the term "qualification" refers to "environmental qualification". RG 1.89, Rev. 1, *Environmental Qualification of Certain Electric Equipment important to Safety for Nuclear Power Plants* (June 1984) (ADAMS Accession No. ML003740271).

[12] *See* 10 CFR 50.34(f)(2)(vii) n.11, 10 CFR 52.47(2)(iv) n.3. The language referenced here that appears in 10 CFR 50.34(f)(2)(vii) n.11, 10 CFR 50.67(b) n.1, and 10 CFR 100.11(a) n.1, was added for consistency as part of the rulemaking associated with Part 52. *See* 2007 Licenses, Certifications, and Approvals for Nuclear Power Plants, 72 Fed. Reg. 49,352, 49,369 (August 28, 2007) (final rule). The footnote language was changed in Part 52 itself to omit the phrase "that would result in potential hazards not exceeded by those from any accident considered credible" and states that "accidents have generally been assumed to result in substantial meltdown of the core with subsequent release *into containment* of appreciable quantities of fission products." (Emphasis added). Due to the similarity of the pertinent language and lack of discussion in the statements of consideration regarding the language differences, the staff treats the footnote as equivalent for the purposes of this evaluation and considers 10 CFR 52.47(a)(2)(iv) linked to the analyses in §§ 50.34, 50.67, and 100.11.

The staff discussed this history with NuScale and noted the differences in the language associated with source terms within NRC regulations during a public meeting on December 12, 2018.[13] Following the public meeting, NuScale submitted a white paper on January 31, 2019, referencing the difference in the language used in 10 CFR 50.49(e)(4) and the footnotes associated with 10 CFR 50.34(f)(2)(vii) and 10 CFR 52.47(a)(2)(iv). In the white paper, NuScale proposes to apply a source term release inside containment derived from consideration of postulated accidents that do not result in core melt to the environmental qualification analysis under 10 CFR 50.49 and the control room habitability analysis under GDC 19. In addition, NuScale proposes to evaluate offsite dose consequences under 10 CFR 52.47(a)(2)(iv) by applying a source term that does not consider core melt in addition to using a core melt accident source term.. NuScale characterizes the source term that considers core melt to be "beyond-design-basis" for its design.

DISCUSSION:

NuScale's proposed approach to environmental qualification differs from past practice but is consistent with the language of 10 CFR 50.49 and compatible with past Commission direction.[14] NuScale submitted an update to the TR and relevant portions of the design certification application final safety analysis report to address the technical details supporting its approach. This submittal is currently under review by the NRC staff.

NuScale's Approach

NuScale proposes to demonstrate compliance with the offsite dose evaluation required under 10 CFR 52.47(a)(2)(iv) using source terms that do not consider core melt. NuScale also proposes to evaluate offsite dose using a core melt accident source term based on beyond-design-basis events. This offsite dose evaluation will be explicitly defined as "beyond-design-basis" for the NuScale design.

NuScale also proposes to use a source term without consideration of core melt to demonstrate compliance with GDC 19 for control room habitability. However, NuScale stated that it would also evaluate control room dose consequences based on the radiological consequences of an accident involving core melt to address TMI requirements 10 CFR 50.34(f)(2)(vii) and 50.34(f)(2)(xxviii).

NuScale states in its January 31, 2019, white paper that "[b]ased on the safety characteristics of the NuScale design, a severe accident scenario resulting in core damage is a beyond-design-basis event" and "it is unnecessary to apply traditional safety-related requirements to the design and qualification of [structure, system, and component] functions credited to mitigate the core damage event."[15] As a result, NuScale's proposal states that the environmental qualification evaluation under 10 CFR 50.49 in the TR will be based on a radiological environment caused by a bounding source term that is relevant to the equipment being evaluated but does not include consideration of a traditional core melt accident. In particular, NuScale states that a bounding

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[13] ADAMS Accession No. ML18351A110.

[14] *See infra* p. 9-10.

[15] WP-0318-58980, *Accident Source Terms Regulatory Framework*, Rev. 1, at 1 (January 31, 2019) (ADAMS Accession No. ML19032A146).

source term for equipment inside containment will be based on the design basis accident[16] that is projected to result in the highest radiation environment in the containment. In determining the source terms associated with design basis accidents, NuScale proposes crediting only safety-related structures, systems, and components that affect the event progression and those that mitigate the associated radionuclide release. NuScale's approach is consistent with staff guidance in the Standard Review Plan, NUREG-0800, as it applies to the crediting of safety-related structures, systems, and components for design basis accident analyses.[17]

NuScale has proposed that in circumstances where a structure, system, or component is used to mitigate consequences of a core melt accident, equipment survivability will be demonstrated in accordance with NRC staff positions in SECY-90-016 and SECY-93-087 and the Commission direction in the associated Staff Requirements Memoranda (SRM).[18] As discussed in SECY-90-016, the term "equipment survivability" is used to describe equipment functionality under temperature, pressure, and radiation conditions associated with a severe accident. The staff in SECY-90-016 noted that "[i]n instances where safety related equipment, (which is provided for design bases accidents) is relied upon to cope with severe accidents situations[,] there should also be a high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function." NuScale's high-level approach to demonstrating equipment survivability is described below.

For structures, systems, and components that must remain functional during both a design basis event and a beyond-design-basis event, NuScale proposes that an evaluation under 10 CFR 50.49 will be performed for design basis event conditions and an equipment survivability analysis will be performed for conditions resulting from a beyond-design-basis event. For those structures, systems, and components that must remain functional during beyond-design-basis events only, such as equipment used for post-accident monitoring under 10 CFR 50.34(f)(2)(xix) and 10 CFR 50.44(c)(4), NuScale intends to only perform an equipment survivability analysis. In its white paper, NuScale proposes to demonstrate equipment survivability for radiation through radiation testing or analysis with a dose equivalent to a core melt accident radiation environment for a 24-hour duration following the onset of core damage based on the features of the NuScale design. After 24 hours, NuScale states that it will qualitatively assess equipment survivability based on industry data and vendor experience with

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[16] Although the term "design basis accident" is not explicitly defined in the regulations, the NRC Staff's Standard Review Plan defines design basis accidents as unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit) that are used to set design criteria and limits for the design and sizing of safety-related systems and components. *See* NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition — Transient and Accident Analysis*, Section 15.0, "Introduction – Transient and Accident Analyses," Rev. 3, at 14 (March 2007) (ADAMS Accession No. ML070710376). The staff will evaluate NuScale's revised TR to determine whether NuScale's use of the term "design basis accident" is consistent with the definition above.

[17] *See* NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition — Transient and Accident Analysis*, Section 15.0, "Introduction – Transient and Accident Analyses," Rev. 3, at 14 (March 2007) (ADAMS Accession No. ML070710376).

[18] SECY-90-016, *Evolutionary and Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements*, (January 12, 1990) (ADAMS Accession No. ML003707849); SRM-SECY-90-016, *SECY-90-016 Evolutionary and Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements*, (June 26, 1990) (ADAMS Accession No. ML003707885); SECY-93-087, *Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs*, (April 2, 1993) (ADAMS Accession No. ML003708021); SRM-SECY-93-087, *SECY-93-087 – Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs*, (July 21, 1993) (ADAMS Accession No. ML003708056).

known degradation mechanisms. NuScale further states that this evaluation of the post-accident monitoring equipment "provides the necessary capabilities to enable operators to assess the presence and extent of core damage during a severe accident, thereby fulfilling the underlying intent of post-accident sampling per 10 CFR 50.34(f)(2)(viii)." NuScale stated that it plans to provide details about its approach in an updated TR and in associated revisions to its design certification application. This TR submittal was received on April 21, 2019 is currently under staff review.[19]

Staff's Approach and Considerations

1) Environmental Qualification

Based on NuScale's proposal regarding the use of a core melt source term for evaluation of environmental qualification, the staff has determined that NuScale's proposed approach is consistent with 10 CFR 50.49, compatible with prior Commission direction regarding accident source terms, and would be acceptable to the NRC staff with adequate technical justification. The staff intends to evaluate NuScale's technical justification of its proposed approach during the review of the NuScale TR revision and proposed revisions to the NuScale design certification application. Additionally, NuScale has submitted an exemption request for the post-accident sampling requirement of 10 CFR 50.34(f)(2)(viii). The staff will evaluate NuScale's exemption request concurrently with its review of the NuScale TR.[20] An analysis of the relevant regulatory requirements, a discussion of relevant Commission direction, and the staff's plan for evaluating NuScale's novel regulatory approach to meet environmental qualification requirements is provided below.

Regulatory Language Analysis of 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"

The Commission directed the staff in CLI-80-21 to initiate a rulemaking to address environmental qualification of safety-grade electrical equipment.[21] The resulting regulation, 10 CFR 50.49, required an environmental qualification program for electric equipment that is based in part on "the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional." The Statements of Consideration for 10 CFR 50.49 referenced NUREG-0588, the Division of Operating Reactors – "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors, dated November 13, 1979," Guidelines, and RG 1.89 as consistent with the intent of the rule and its requirements. In these documents, the staff described the accident analysis in terms of core melt damage.[22] This guidance suggests at that

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[19] *See* Revision 3 of TR-0915-17565, "Accident Source Term Methodology" (April 21, 2019) (ADAMS Accession No. ML19112A172).

[20] *See* NuScale Power, LLC, *Submittal of Changes to the Design Certification Application, Part 7, Exemptions, Section 16, 10 CFR 50.34(f)(2)(viii) Post-Accident Sampling* (January 31, 2019) (ADAMS Accession No. ML19031C975).

[21] *See Petition for Emergency and Remedial Action*, CLI-80-21, 11 NRC 707, 712 (1980).

[22] *See* Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, 48 Fed. Reg. 2,729 2,730 (January 21, 1983) (final rule); RG 1.89, *Qualification of Class IE Equipment for Nuclear Power Plants* (November 1974) (ADAMS Accession No. ML012880422); NUREG-0558, Rev.1, *Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment* (July 1981).

time the staff anticipated "the most severe design basis accident" would involve a core melt accident with associated radioactive release to the environment.

Despite these references associated with the implementation of the rule and the likely understanding of the term at the time, "design basis accident" is not defined in NRC regulations.

Section 50.49(b) includes safety-related electrical equipment within the scope of the environmental qualification requirements and specifies "this equipment is that relied upon during and following design basis events." Section 50.49(b)(1)(ii) states that design basis events include "conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena." As defined in 10 CFR 50.2, the design bases of a plant include "information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design." Such values may be "requirements derived from analysis . . . of the effects of a postulated accident for which a structure, system, or component must meet its functional goals." As a function of these regulatory definitions, safety-related structures, systems, and components are not required to be capable of mitigating CDEs if CDEs are not considered to be design basis events for that structure, system, or component.[23]

In addition, the staff in SECY-90-016, notes that for new reactors "features provided for severe-accident protection (prevention and mitigation) only (not required for design basis accidents) need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements, (b) all aspects of 10 CFR Part 50, Appendix B quality assurance requirements, or (c) 10 CFR Part 50, Appendix A redundancy/diversity requirements."[24] The staff further noted in this paper that the "reason for this judgment is that the staff does not believe that severe core damage accidents should be design basis accidents (DBA) in the traditional sense that DBAs have been treated in the past." This reasoning was extended in SECY-93-087 to new reactor passive plant design features provided only for severe accident mitigation.[25]

The use of the term "design basis accident" elsewhere in the regulations is inconsistent. For example, although 10 CFR 50.49(e)(1) also references "the most severe design basis accident" for temperature and pressure effects, the staff has not used a core melt accident to define these parameters. Conversely, 10 CFR 50.67, "Accident source term," uses the term "design basis accidents" but includes a footnote stating that "[s]uch accidents have generally been assumed to result in substantial meltdown of the core." The Statements of Consideration for

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[23] The staff notes that the term "design basis event" connotes a particular set of events uniquely determined for each plant design and can vary. NuScale's identification of safety related equipment was provided based on the analyses of design basis events in Chapter 15 of NuScale's Final Safety Analysis Report. *See Final Safety Analysis Report Chapter Fifteen, Transient and Accident Analyses*, Rev. 2 (October 2018) (ADAMS Accession No. ML18310A337). Consistent with NuScale's licensing approach, none of these design basis events result in core damage. Therefore, safety related structures, systems, and components in the NuScale design are not required to mitigate core melt accidents.

[24] SECY-90-016, *Evolutionary and Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements*, (January 12, 1990) (ADAMS Accession No. ML003707849); SRM-SECY-90-016, *SECY-90-016 Evolutionary and Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements*, (June 26, 1990) (ADAMS Accession No. ML003707885).

[25] SECY-93-087, *Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs*, (Apr. 2, 1993) (ADAMS Accession No. ML003708021); SRM-SECY-93-087, *SECY-93-087 – Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs*, (July 21, 1993) (ADAMS Accession No. ML003708056).

10 CFR 50.67(b) discuss the original development of the accident source term under 10 CFR 100.11(a), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," and the past practice of assuming significant core damage, which was typically presumed to occur with a postulated loss of coolant accident initiating event combined with multiple additional failures, even though emergency core cooling systems are designed to prevent significant core damage.[26]

Although the meaning of the term "design basis accident" in the context of 10 CFR 50.49 is undefined, this language is notably different from the language used in 10 CFR 50.34(f)(2)(vii) and 52.47(a)(2)(iv) and the associated footnotes. This difference suggests that it would be appropriate to draw a corresponding distinction in the underlying analyses. From a regulatory interpretation standpoint, it is significant that 10 CFR 50.49(e)(4) does not share the footnote that appears in 10 CFR 50.34(f)(2)(vii) and 52.47(a)(2)(iv). Like 10 CFR 50.49(e)(4), the in-text language that appears in 10 CFR 50.34(f)(2)(vii) and 52.47(a)(2)(iv) does not explicitly require the analyses to include a core melt accident. However, the language in the footnote associated with these sections states that the accidents analyzed for these evaluations "have generally been assumed to result in substantial meltdown of the core." Although this footnote language still leaves room to use an accident analysis that does not include a core melt accident, it more clearly articulates the severity of the accident that should be considered. This same language was incorporated into several different places in the regulations, but it was not added to 10 CFR 50.49(e)(4).[27] These differences can be used to draw a distinction between the analysis in 10 CFR 50.49(e)(4) and the analyses in 10 CFR 50.34(f)(2)(vii) and 52.47(a)(2)(iv).

Consistency with Commission Direction Regarding Source Term Development

The staff's past communications with the Commission in the advanced non-LWR power reactor area demonstrate an openness to approving a design-specific source term analysis that does not solely depend on deterministic values. In SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements" the staff recommended, and the Commission approved, an approach that included the use of scenario-specific mechanistic source terms for siting advanced reactors.[28] Subsequently, the staff sought clarification from the Commission in SECY-03-0047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs," regarding whether to use a deterministic bounding source term based on the conservative assumption of core damage and fission product release in the evaluation of source terms for advanced reactors, or to retain the guidance associated with using scenario-specific mechanistic source terms discussed in SECY-93-092.[29] The staff recommended that the Commission retain the guidance

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[26] *See* Use of Alternative Source Terms at Operating Reactors, 64 Fed. Reg. 71,990, 72,001 (December 23, 1999) (final rule).

[27] The footnote language was added to 10 CFR 50.34(f)(2)(vii) in 1982, relatively close in time to the promulgation of 10 CFR 50.49 in 1983. *See* Licensing Requirements for Pending Construction permit and Manufacturing License Applications, 47 Fed. Reg. 2,286, 2,301 (January 15, 1982); Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, 48 Fed. Reg. 2,729 (January 21, 1983).

[28] *See* SECY-93-092, *Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements*, (April 8, 1993) (ADAMS Accession No. ML040210725); SRM-SECY-93-092, *SECY-93-092 Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements*, (July 30, 1993) (ADAMS Accession No. ML003760774).

[29] SECY-03-0047 *Policy Issues Related to Licensing Non-Light-Water Reactor Designs*, (March 28, 2003) (ADAMS Accession No. ML030160002).

and position in SECY-93-092, "provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering [judgment] is used to bound uncertainties." In an accompanying footnote, the staff states that "[t]his represents a fundamental change in practice from that used on LWRs, in that the source term used for siting considerations may not be that associated with a core melt accident."[30] This recommendation was accepted by the Commission in SRM-SECY-03-0047.[31]

This discussion, albeit in the context of advanced non-LWRs, demonstrates that the concept of using source terms based on accidents that do not involve core melt has been considered and accepted previously by the Commission. Neither the staff's paper nor the Commission's determination limits the applicability of this approach to any specific regulation that evaluates source terms. Therefore, such reasoning would reasonably support use of a non-core melt accident in the evaluations under 10 CFR 50.34(f)(2)(vii) and 52.47(a)(2)(iv). However, the Commission has not directly addressed this question.

Referencing these past communications with the Commission, SECY-16-0012 discusses siting issues associated with advanced reactors and carries forward the use of mechanistic source terms to address such issues.[32] SECY-16-0012 informed the Commission of the expansion of using a mechanistic source term as discussed in SECY-93-092 and SECY-03-0047 for small modular reactors, which would include the NuScale design.[33] Although SECY-16-0012 is focused on siting issues, it is reasonable that a similar approach to source term development could be used in the evaluation for environmental qualification. [34]

Staff's Review Framework

Although the staff has determined the approach provided by NuScale for environmental qualification may be permissible within the current NRC regulatory structure, NuScale's January 2019 white paper only includes a high-level description of the proposed analysis methodology and does not include sufficient details to demonstrate the technical viability of its novel approach. Additional details in the following areas will need to be clearly addressed in the NuScale TR update, which is currently being reviewed by the staff, in order for the staff to

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[30] SECY-03-0047 *Policy Issues Related to Licensing Non-Light-Water Reactor Designs*, Attachment 5, n.2. (March 28, 2003) (ADAMS Accession No. ML030160002).

[31] *See* SRM-SECY-03-0047, *Staff Requirements – SECY-03-0047 - Policy Issues Related to Licensing Non-Light-Water Reactor Designs*, (June 26, 2003) (ADAMS Accession No. ML031770124).

[32] *See* SECY-16-0012, *Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors*, (February 7, 2016) (ADAMS Accession No. ML15309A319).

[33] SECY-16-0012 was sent to the Commission as an information paper and as such, no Commission action on it was taken.

[34] Prior to the staff's shift towards the use of mechanistic source terms for advanced reactors in SECY-93-092 and SECY-03-0047 and small modular reactors in SECY-16-0012, source terms for siting and environmental qualification were based predominantly on deterministic and conservative assumptions of significant core damage. *See* SECY-94-302, *Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs*, (December 19, 1994) (ADAMS Accession No. ML003708141) (stating that "[f]or the purpose of radiological assessments, the staff proposes to define the [design basis accident] radiation environment as that resulting from fission-product releases from coolant activity release, gap release, and in-vessel release." The staff explains that "[t]hese source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel" and reflect "the most severe scenarios from which a plant could be expected to return to a safe shutdown condition.")

complete its technical review of the proposed approach and ensure compliance with applicable regulatory requirements and Commission policy:

- the technical basis for equipment survivability determinations or evaluations including the chosen duration for equipment survivability following an accident and the supporting testing and/or analysis information;
- the hazards considered in the equipment survivability evaluation methodology including temperature, pressure, humidity, chemical environment, and radiation environment;
- the characteristics of the proposed bounding source term to be used in the environmental qualification evaluation; and
- the consideration of appropriate margin and a demonstration that synergistic effects are being appropriately considered consistent with the requirements of 10 CFR 50.49.

To conduct its technical and safety review, the staff will assess whether NuScale's approach demonstrates that NRC regulations addressing environmental qualification and equipment survivability are satisfied. The staff intends to conduct the review of the Topical Report (TR) update concurrently with its review of the revised design certification application, consistent with the process outlined below.[35]

First, the staff will evaluate the design certification application to determine whether NuScale has identified an appropriate spectrum of design basis accidents as part of the review conducted under Standard Review Plan 15.0, "Introduction - Transient and Accident Analyses." As part of this evaluation, the staff will verify whether the completed categorization of events in the Final Safety Analysis Report is consistent with the approach described in NuScale's revised TR on accident source term methodology.

Second, the staff will evaluate NuScale's methodology for developing and applying the source terms used in the environmental qualification evaluations. NuScale's proposal defines the bounding design basis accident radiological source term for environmental qualification evaluations.[36] Because all environmental qualification assessments to date considered a core melt accident, there is no specific guidance on NuScale's proposed determination of an environmental qualification radiological source term for an event that does not result in core melt. Therefore, the source term used for environmental qualification will be evaluated to ensure that it is appropriate and complete (e.g., all relevant radionuclides important to equipment total integrated dose have been considered). The staff will make a technical determination regarding NuScale's proposed source term for the environmental qualification evaluation to ensure that structures, systems, and components will be qualified for design basis accident conditions in accordance with 10 CFR 50.49.

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[35] The staff is currently evaluating Revision 3 of TR-0915-17565 as part of its review process and has not yet made a determination on the technical justification provided by NuScale to support its approach to environmental qualification. *See* Revision 3 of TR-0915-17565, "Accident Source Term Methodology" (April 21, 2019) (ADAMS Accession No. ML19112A172).

[36] Based on the approach described in the NuScale white paper, this would be based on non-core melt accidents described in NuScale's Final Safety Analysis Report Chapter 15 adjusted to reflect short term fuel behavior following a transient (e.g. iodine spiking factor). *See Final Safety Analysis Report Chapter Fifteen, Transient and Accident Analyses*, Rev. 2 (October 2018) (ADAMS Accession No. ML18310A337).

Third, the staff will analyze NuScale's approach to addressing the equipment survivability evaluation. In May 2019, NuScale submitted its equipment survivability evaluation for equipment to be relied upon in a beyond design basis event, per the guidance in SECY-90-016, SECY-93-087, and their associated SRMs.[37] NuScale's simplified and passive design appears to include fewer components subject to equipment survivability than previous new reactor designs. NuScale has currently identified under-the-bio-shield radiation monitors,[38] electrical penetration assemblies, and containment isolation valves as components that would be impacted by the change in approach and would be evaluated under equipment survivability. For previous new reactor applications, the equipment survivability evaluation relied, in part, on portions of the environmental qualification evaluation for the radiation environment assumed for core melt accident conditions. The survivability of equipment that was not adequately addressed by the environmental qualification evaluation was addressed on a case-by-case basis using the guidance in SECY-90-016, SECY-93-087, and their associated SRMs. Consistent with the design certification application review process for a probabilistic risk assessment, the staff will make a technical determination regarding the survivability of structures, systems, and components needed to mitigate severe accident conditions for the period that is needed to perform the intended functions. This will include staff consideration of the appropriate methodology (i.e., testing, analysis, or some combination thereof) used to demonstrate equipment survivability of structures, systems, and components for core melt accidents.

2) Offsite Dose Consequence Analysis and Control Room Habitability

In addition to NuScale's proposed approach to address accident source terms for environmental qualification, NuScale also proposed in its January 2019 white paper using a source term that does not consider a core melt accident to demonstrate compliance with GDC 19 for control room habitability. Specifically, NuScale states in its white paper submittal that "[t]he GDC 19 dose criteria are not directly applicable to a [beyond-design-basis event] BDBE like the core damage source term (CDST). However, certain TMI requirements relate to control room habitability under severe accident conditions."[39] NuScale has not provided sufficient technical justification to support its conclusion that GDC 19 need not consider a CDST. Furthermore, the staff disagrees with NuScale's application of the term "beyond-design-basis" in this context. As noted earlier, the "design bases" for a facility include "the specific values or ranges of values chosen for controlling parameters as reference bounds for design."[40] To the extent a source term that considers a core damage accident is used to establish reference bounds for a design,

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[37] *See* WP-0318-58980, *Accident Source Terms Regulatory Framework*, Rev. 1, at 8, 18 (January 31, 2019) (ADAMS Accession No. ML19032A146).

[38] The under-the-bio-shield monitors perform an important post-accident monitoring function in the NuScale design and may be needed after the initial occurrence of core damage. 10 CFR 50.49(b)(3) designates certain post-accident monitoring equipment as electric equipment important to safety and subject to the requirements of the section. This regulation includes a footnote reference to Revision 2 of RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident." (December 1980) (ADAMS Accession No. ML060750525). This guidance states that it is conservative to select certain variables for accident monitoring that are functionally significant to "be monitored by instruments qualified to more stringent environmental requirements and with ranges that extend well beyond that which the selected variables can attain under limiting conditions."

[39] *See* WP-0318-58980, *Accident Source Terms Regulatory Framework*, Rev. 1, at 15 (January 31, 2019) (ADAMS Accession No. ML19032A146).

[40] *See* 10 CFR 50.2

this source term would be considered to be within the "design bases" for the facility. While the staff recognizes that a CDST may be based on accident events that are more severe than the spectrum of design basis events used to identify safety-related systems, structures, and components, such a source term would still be within the "design bases" for the facility because of its use in other evaluations. Therefore, the staff believes that NuScale's characterization of a source term that is within a facility's "design bases" to be characterized as "beyond-design-basis" can lead to confusion regarding the actual design and licensing basis for the facility.

However, the staff understands that NuScale intends to evaluate control room habitability with a source term that considers a core melt accident for TMI requirements 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) while applying the 5 rem dose criterion contained in GDC 19. If NuScale provides sufficient information to demonstrate that the analysis performed to meet the TMI requirements associated with control room habitability appropriately bound the accessibility and occupancy requirements of GDC 19, the staff believes that it can make a regulatory finding for control room habitability under 10 CFR 50.34(f)(2)(vii), 10 CFR 50.34(f)(2)(xxviii), and GDC 19. Therefore, the staff plans to evaluate compliance with the applicable control room habitability requirements based on the information submitted by NuScale.

Regarding the offsite dose consequence analysis, NuScale similarly categorizes a core melt accident as a beyond design basis event and proposes to meet 10 CFR 52.47(a)(2)(iv) by analyzing source terms that consider a core melt accident as well as those that do not. Specifically, NuScale states that "the [design basis source terms] DBSTs will be evaluated to demonstrate acceptable offsite doses and serve as [design basis events] DBEs for associated regulatory requirements. The CDST will also be evaluated to demonstrate acceptable offsite doses but will be treated as a beyond design basis accident."[41]

NuScale explains further in its TR submittal that "[t]he classification of the CDE as beyond-design-basis is appropriate in order to consistently define the design basis of the plant. Treatment of the CDE as a design-basis accident for some purposes and not others would create an inconsistency in the implementation of the regulations."[42]

Although the staff does not agree with NuScale's categorization of accident source terms, the staff understands NuScale's proposal to be that it intends to perform the offsite dose consequence analysis using a source term that considers core melt, as well as source terms that do not consider core melt. The staff further interprets the use of a source term that considers core melt as part of NuScale's demonstration that the offsite dose consequence requirements of 10 CFR 52.47(a)(2)(iv) are met. Therefore, a source term that considers core melt for the purposes of the offsite dose consequence analysis would be considered within the design and licensing bases for the NuScale design certification. On this basis, the staff believes that it will be able to make its regulatory finding regarding the requirements of 10 CFR 52.47(a)(2)(iv) if NuScale provides the appropriate technical information.

Potential Applicability of the Approach

As discussed above, there are differences in the language associated with source terms within NRC regulations that may provide enough flexibility for a current licensee or reactor applicant to

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[41] *See* WP-0318-58980, *Accident Source Terms Regulatory Framework*, Rev. 1, at 10 (January 31, 2019) (ADAMS Accession No. ML19032A146).

[42] *See* Revision 3 of TR-0915-17565, "Accident Source Term Methodology" at 134 (April 21, 2019) (ADAMS Accession No. ML19112A172).

present a submittal that does not use a core melt accident source term for environmental qualification or for other analyses. Although the staff does not offer judgment as to the success of such an approach, the staff recognizes that some of the reasoning proposed by NuScale could be extended to apply to other, related parts of the regulations in the future to address specific features of advanced reactor designs. The staff further notes that current licensees may choose to propose a similar approach to environmental qualification of equipment, control room habitability, and/or offsite dose consequences analysis, potentially in the context of license renewal. Such approaches in other applications will be considered by the staff on a case-by-case basis.

CONCLUSION:

Based on the history and the language in 10 CFR 50.49 as described above, the staff has determined that NuScale's proposed approach of using radiological source terms derived from design basis accidents without consideration of core melt is consistent with 10 CFR 50.49 and compatible with prior Commission direction on this subject. In view of the extensive history and past practice associated with using a core melt accident to determine source term in the 10 CFR 50.49 evaluation, the staff determined it would be appropriate to communicate this new approach to the Commission. The staff intends to move forward with evaluating NuScale's proposed approach and will ultimately determine its technical viability after review and evaluation of the revised TR-0915-17565 and proposed revisions to the NuScale design certification application. As stated above, the staff believes that it will be able to make a regulatory finding for control room habitability if NuScale provides appropriate technical information to show that compliance with 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) using a source term that considers core melt appropriately addresses the access and occupancy requirements of GDC 19. Furthermore, given that NuScale intends to demonstrate compliance with the regulatory requirements in 10 CFR 52.47(a)(2)(iv) using a source term that considers a core melt accident, the staff believes that it will be able to make a regulatory finding for the offsite dose consequence analysis if appropriate technical information is submitted. The staff will continue to inform the Commission on this issue, as appropriate, if new issues arise.[43]

[43] NuScale submitted Revision 3 of the TR, along with the majority of associated changes to the NuScale Final Safety Analysis Report on April 21, 2019 (ADAMS Accession Nos. ML19112A172 and ML19112A220). NuScale's revisions to the Final Safety Analysis Report Chapter 19 analysis were submitted on May 22, 2019 (ADAMS Accession No. ML19142A397). The staff anticipates that there may be challenges to the review schedule but remains committed to meeting the overall 42-month review schedule.

<u>COORDINATION</u>:

The Office of the General Counsel has reviewed this paper and has no legal objection.

Margaret M. Doane
Margaret M. Doane
Executive Director
for Operations

SUBJECT: STAFF APPROACH TO EVALUATE ACCIDENT SOURCE TERMS FOR THE NUSCALE POWER DESIGN CERTIFICATION APPLICATION
DATE: AUGUST 16, 2019

ADAMS Accession No.: ML19107A455 **SECY-012**

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