Public availability of this draft document in advance of its discussion at the January 26, 2018, ACRS subcommittee meeting is intended to inform stakeholders of the NRC staff's criteria it will use to determine if NuScale Power's request for exemption to GDC 27 is acceptable. The NRC is not currently accepting public comments on the information below. Please note that this draft document may be incomplete or in error in one or more respects and may be subject to futher revision before the NRC staff provides its formal SECY paper to the Commission.

POLICY ISSUE

(Information)

January XX, 2018 SECY-18-XXX

FOR: The Commissioners

FROM: Victor M. McCree
Executive Director for Operations

SUBJECT: NUSCALE POWER EXEMPTION REQUEST FROM 10 CFR PART 50, APPENDIX A, GENERAL DESIGN CRITERION 27,"COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY"

PURPOSE:

The purpose of this paper is to inform the Commission of (1) NuScale Power, LLC's (NuScale's) request for an exemption, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.7, "Specific exemptions," to General Design Criterion (GDC) 27, "Combined reactivity control systems capability," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and (2) the criteria that the U.S. Nuclear Regulatory Commission (NRC) staff will use to determine whether the exemption is acceptable with regard to the protection of public health and safety. If granted, this exemption would be a first of its kind for a new reactor design potentially experiencing a recriticality following a design basis event (DBE) during certain periods of plant operation while assuming one control rod assembly is stuck out of the core (i.e., failed to insert). The staff will evaluate whether granting the exemption presents an undue risk to public health and safety, and will provide its recommendation to the Commission using the established design certification rulemaking process. This paper does not address any new commitments or resource implications.

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BACKGROUND:

The GDC for nuclear power plants, as listed in 10 CFR Part 50, Appendix A, are the minimum requirements for the principle design criteria for water-cooled nuclear plants to "provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public." The NRC established the GDC based on the licensing of early commercial nuclear power plant designs, but it acknowledged that fulfillment of some of the GDC may not be necessary or appropriate for some designs. The NuScale standard plant design is modern but has many similarities to designs previously licensed by the NRC. Thus, it would be expected to meet or justify departures from the GDC as required under 10 CFR 52.47, "Contents of applications; technical information," as part of the licensing processs under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." GDC 27 states:

> *Combined reactivity control systems capability.* The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Based on historical practice and insights gained from documentation related to the development of this GDC, the staff has implemented this criterion through analysis demonstrating safe shutdown is achieved and maintained in the long term following postulated accidents (PAs). The reactor design must achieve and maintain subcriticality any time during the operating cycle using only safety-related equipment, and assuming the most reactive control rod fails to insert. To confirm this position, the staff studied previously approved pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) to assess the application of GDC 27 and found they all achieve and maintain the reactor subcritical following a PA with margin for stuck rods using only safety-related equipment.[1] The NRC has not licensed a power reactor that does not remain subcritical beyond the short term under PA conditions.

In early 2016, the NRC staff became aware that the NuScale design may return to critical following most DBEs. Specifically, the DBE analyses show that the reactor core would become critical again; that is, the shutdown reactor would, on its own, resustain the nuclear fission chain reaction. The potential DBE scenarios assume that one of the 16 control rods fails to insert and that the moderator temperature coefficient (MTC) is sufficiently negative to have recriticality occur due to decreases in reactor coolant temperature, which would be the case for much of each operating cycle. The reactor would remain critical until a nonsafety means of reactivity control becomes available for use. While these recovery actions may reflect realistic responses to an event, the subject analyses within the NuScale design and licensing basis assume no operator actions, no safety-related boron injection to mitigate the consequences of a design basis event, and no safety-related means to monitor the status of the core following the event.

On September 8, 2016, the staff documented its position on this GDC 27 issue, in a letter titled "Response to NuScale Gap Analysis Summary Report for Reactor Systems Reactivity Control Systems, Addressing Gap 11, General Design Criterion 27" (Agencywide Documents Access

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[1] In general, PWRs credit safety-related control rods and safety-related soluble boron addition, and BWRs credit safety-related control rods. For some accident scenarios in currently licensed PWRs (e.g., main steamline break), the reactor may not remain subcritical in the short term, in particular within the first few minutes of accident initiation, which is considered part of the event evolution. Beyond the short term, the PWR safety injection systems provide sufficient soluble boron addition to keep the reactor subcritical.

and Management System (ADAMS) Accession No. ML16116A083). The staff's position is that GDC 27 requires that the reactor be reliably controlled and achieve and maintain a safe, stable condition, including subcriticality, beyond the short term, using only safety-related equipment following a PA with margin for stuck rods. In this letter, the staff also informed NuScale that an exemption to GDC 27 would be required and that consideration of such an exemption entails policy issues under the purview of the Commission. NuScale responded to the NRC's letter on November 2, 2016 (ADAMS Accession No. ML16307A449), stating that the design is consistent with the regulations, and provides reasonable assurance of adequate safety, and that an exemption from GDC 27 should not be required to license the NuScale reactor design. NuScale requested that, if the staff did not agree, the white paper with NuScale's regulatory analysis be included in any staff paper to the Commission (see Enclosure 1).

DISCUSSION:

The NRC has not previously approved designs that have not achieved and maintained shutdown using only safety-related equipment in the long term following DBEs. While proceeding with the safety review of NuScale's design certification application (DCA), the staff is informing the Commission early in the review process because of the unprecedented request for a new reactor plant design potentially challenging a principal fission product barrier (i.e., fuel cladding) by remaining critical for an extended period following either an anticipated operational occurrence or PA. The staff is providing this preview of its consideration of NuScale's exemption request in its DCA and the criteria the staff will use to evaluate the acceptability of the request. The staff further notes that its evaluation may provide insights useful to support future advanced non-light-water-reactor licensing activities.

The staff will base its recommendation on the acceptance of the exemption request on the criteria in 10 CFR 50.12, "Specific exemptions." The staff will evaluate whether the NuScale design meets the underlying purpose of GDC 27 by assessing the results of NuScale's safety analyses of the events against the established specified acceptable fuel design limits (SAFDLs). In accordance with the design certification process in 10 CFR Part 52, the Commission will make the final determination on the acceptability of NuScale's proposed exemption to GDC 27 and the safety of the design based on NuScale's information and analysis and the NRC staff's review documented in the safety evaluation report.

Deterministic Design Basis Technical Review

NuScale requested an exemption from GDC 27 and proposed a design-specific principal design criterion to address the potential that the reactor could return to power under certain plant conditions if adequate passive heat removal capability exists. The staff is conducting its review consistent with the enhanced safety focused review approach (ESFRA)[2].

For a DBE, only safety-related structures, systems, and components (SSCs) are typically credited to show compliance with the regulations. Therefore, for the DBE scenarios on which the staff will make its safety finding, the following three conditions are assumed to exist:

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[2] The staff described the ESFRA for NuScale in SECY 2017-0112, "Plans for Increasing Staff Capabilities to Use Risk Information in Decision-Making Activities," consistent with Commission direction in its Staff Requirements Memorandum (SRM) dated May 11, 2011, related to SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews" (ADAMS Accession No. ML111320551). The ESFRA guides reviewers in considering plant design features in 11 key areas to help formulate the scope and depth of review activities.

(1) The highest worth control rod fails to insert (for the NuScale design, there are a total of 16 control rods and most control rods have similar worths).
(2) Alternating current (ac) power is unavailable (the NuScale design does not include safety-related ac or direct current (dc) sources).
(3) The nonsafety-related chemical and volume control system (CVCS) is unavailable.

With these assumptions, the NuScale design could experience a return to power following most design basis events during much of every operating cycle. The staff notes that with 12 modules per plant, multiple modules would likely be in this state (i.e. sufficiently negative MTC) at any given time. In the most conservative case, return to power could occur a few hours after the initiation of an event. NuScale has calculated local power levels in the area of the stuck rod to be greater than 60 percent full power, and the reactor core average power would return to approximately 10 percent full power. The energy generated would be initially removed by the decay heat removal system and may transition to cooling by the emergency core cooling system depending on the availability of the dc power system. NuScale asserts that the primary SAFDL of concern (critical heat flux) is met for both scenarios, with and without dc power. This would provide assurance that the fuel is adequately cooled near the stuck rod, the fuel cladding fission product barrier is maintained, and that these scenarios would not pose undue risks to public health and safety as a result of the release of fission products from the fuel.

The staff is performing a detailed technical review of the return to power analysis that NuScale provided in Chapter 15 "Transient and Accident Analyses," of its final safety analysis report (ADAMS Accession No. ML17013A286). The staff will assess the safety basis of NuScale's position by evaluating the postevent power level, reactor flow stability, system capacity for heat removal, and margin to fuel safety limits. The staff will make a recommendation to the Commission, as part of the design certification process in 10 CFR Part 52, on the acceptability of NuScale's GDC 27 exemption request and the overall acceptability of the design.

Regulatory Perspective and Risk Analysis

From a regulatory perspective, the staff could support the NuScale exemption provided public health and safety are maintained by sufficient core cooling during the scenario to maintain fuel cladding integrity, and provided the DBE sequence of events is not actually expected to occur during the lifetime of a module. This approach is consistent with the integrated decision-making process envisioned for new plant designs being reviewed under ESFRA. Given the unique, passive capability of the design, NuScale could demonstrate the design maintains core cooling using natural circulation without an emergency power source or operator action, following a DBE in which the core returns to a low power level. This capability does not exist in the current fleet of licensed power reactors, and the staff believes it is appropriate to consider this exemption request and potentially depart from past precedent.[3]

[3] SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs" (ADAMS Accession No. ML003708068) stated, "[t]he staff believes that [conditions other than cold shutdown] may constitute a safe shutdown state as long as reactor subcriticality, decay heat removal, and radioactive materials containment are properly maintained for the long term." In the associated SRM (ADAMS Accession No. ML003708098), the Commission approved the concept of reliance on systems that are not safety-related as support to the safety-related passive heat removal systems after 72 hours from the onset of a design basis accident.

For the NuScale design to return to criticality following a shutdown, a set of circumstances must occur simultaneously. These include a DBE concurrent with a stuck control rod assembly, loss of ac power, and the reactor operating in a portion of the cycle with a sufficiently negative MTC. Under these circumstances, NuScale estimates the probability of a return to power to be less than 1E-6 per module year. The staff will confirm that this event would not be expected to occur during the lifetime of a module.

The NuScale design also includes a nonsafety-related CVCS,[4] which could, if available and not isolated, restore the reactor to a shutdown condition. Assuming that a postevent recriticality is unlikely to occur during the life of a power module, with an adequate, highly reliable (passive) safety-related means of heat removal, and the expected availability of alternative means of ultimately achieving subcriticality (e.g., CVCS), the staff finds it appropriate to consider an alternative design criterion, protection of the SAFDLs, to govern this event.

The staff will include risk aspects in its safety evaluation of the exemption request and provide it to the Commission using the established design certification rulemaking process. The safety evaluation will also document the staff's findings on NuScale's safety analysis, performed using conservative, deterministic methods, regardless of the likelihood of the event. If NuScale demonstrates the SAFDLs are met, and the probability of a return to power is sufficiently low, the staff would find that the exemption poses no adverse risk to public health and safety. The staff is evaluating whether an existing alternative means to maintain the reactor shutdown for this scenario should be reliable and available to plant operators (e.g., CVCS).

CONCLUSIONS:

Pending satisfactory results of the detailed technical review of safety analyses, the staff anticipates recommending in its safety evaluation report on the NuScale DCA that an exemption from GDC 27 be included in the design certification rulemaking.

COORDINATION:

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

The staff discussed this issue with the Advisory Committee on Reactor Safety (ACRS). The ACRS issued a letter on February XX, 2018 (MLXXXXXX), providing its conclusions and recommendations. The staff considered its interactions with the ACRS in finalizing this paper.

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Enclosure:
NuScale Reactivity Control
Regulatory Compliance and Safety

[4] The staff is evaluating NuScale's position that the CVCS system is not risk significant and need not be included in the Design Reliability Assurance Program, in consideration of this and other events.