NuScale Power

LO-1116-51829

November 2, 2016

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of White Paper Entitled “NuScale Reactivity Control Regulatory Compliance and Safety,” Revision 0 (NRC Project No. 0769)

REFERENCE: F. Akstulewicz, U.S. Nuclear Regulatory Commission, letter to T. Bergman, NuScale Power, LLC, September 8, 2016, ADAMS Accession No. ML16116A083.

In the reference letter, NRC staff responded to Gap 11 identified in the NuScale “Gap Analysis Summary Report,” concerning General Design Criterion (GDC) 27 of 10 CFR Part 50, Appendix A. In consideration of information provided by NuScale concerning a potential for return to low power conditions following shutdown under certain conservative assuptions and conditions, NRC staff addressed the originally identified GDC 27 gap as well as the postulated return to power event. With regard to the latter, NRC Staff stated “the staff’s current view is that GDC 27 requires that the reactor be reliably controlled and that the reactor achieve and maintain a safe, stable condition, including subcriticality beyond the short term, using only safety related equipment following a postulated accident with margin for stuck rods.” The Staff further indicated that NuScale would need to seek an exemption on the basis of the staff’s interpretation.

NuScale appreciates Staff’s receptiveness to additional information and alternative perspectives on this topic, as noted in their letter. NuScale has carefully considered NRC Staff’s current view, further evaluated the design, and conducted a review of regulatory history, regulatory basis, NRC policy and guidance, and licensing precedent. Attachment 1 summarizes the results of that review and describes NuScale’s basis for compliance with the meaning and intent of GDC 27, and similar provisions of GDC 26. Because NuScale believes the design is consistent with regulation and provides reasonable assurance of adequate safety, NuScale does not believe that technical changes are warranted.

While NuScale concurs with Staff’s view that current, approved light water reactors “achieve a safe, stable condition with appropriate margin for stuck rods,” NuScale disagrees that the approach followed by current, approved light water reactors is required by regulation. By expecting NuScale to meet GDC 27 in the same manner as prior designs, Staff’s position appears contrary to the plain language of GDC 27 itself, as well as NuScale’s understanding of the intent of the General Design Criteria overall—to provide engineering goals rather than precise tests or methodology by which reactor safety can be gauged. Furthermore, although guidance and precedent provides an acceptable means for meeting GDC 27, it is not a necessary means for compliance.

The NuScale design provides safe and reliable short term and long term reactivity control. The safety-related, seismically qualified control rods—alone—will maintain shutdown from hot zero power to a conservative minimum core temperature. Under normal and accident conditions, with a postulated limiting stuck rod and with conservative Chapter 15 assumptions, a low-level return to power is predicted during the long term response. Under these conditions, safety-related, passive cooling systems will maintain adequate core cooling to safely cool the core and prevent radiological release. As further described in Attachment 1, such a condition is safe and the probability for such a condition is insignificant.

Accordingly, NuScale respectfully requests NRC Staff consider the perspective of Attachment 1 with respect to the current position. Specifically, NuScale has the following requests.

- That NRC Staff conclude that the proposed design basis, as demonstrated and validated by the Final Safety Analysis Report, would comply with the meaning and intent of the GDCs as written. Under this approach, NuScale would submit its design certification application (DCA) as presently drafted (no exemption request). This is NuScale's preferred approach.
- Should NRC Staff disagree, NuScale requests NRC Staff to consider that for reactor designs not "similar in design" to plants licensed as of 1971, 10 CFR Appendix A "provides guidance to applicants in establishing principal design criteria" (10 CFR 52.47(a)(3)(iii)). Thus, NuScale believes Staff should consider that NuScale's unique design basis results from an advanced Small Module Reactor design relying only on passive, inherently safe reactor cooling systems that reduce core temperature well below a hot shutdown state, such that it is appropriate to treat GDCs 26 and 27 as guidance for the NuScale design. Under this approach, NuScale would modify its DCA to provide the basis in this paragraph as justification for design-specific Principal Design Criteria 26 and 27, without exemption to or modification of the language in the GDCs.
- If, after considering Attachment 1, Staff continue to believe an exemption from GDCs 26 and 27 is required, NuScale asks Staff to include Attachment 1 in any Policy Paper prepared on this topic for Commission consideration. As part of any such paper, NuScale requests the Staff seek Commission approval that the information in Attachment 1 provides sufficient basis for granting an exemption. This would provide regulatory certainty for this approach early in the DCA review.

In the event a Staff or Commission decision is not provided by December 10, 2016, NuScale requests Staff agreement that the DCA submitted as presently drafted is sufficient for docketing with respect to GDCs 26 and 27. This timeframe is necessary to allow NuScale time to incorporate changes to its DCA without impacting the submittal schedule if the staff does not accept NuScale's recommended approach.

This letter and its attachment make no regulatory commitments and no revisions to any existing regulatory commitments.

Please feel free to contact Gary Becker, Regulatory Affairs Counsel at (541) 360-0549 or at gbecker@nuscalepower.com if you have any questions.

Sincerely,

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Attachment 1: "NuScale Reactivity Control Regulatory Compliance and Safety," Revision 0

NUSCALE REACTIVITY CONTROL
REGULATORY COMPLIANCE AND SAFETY

1.0 Introduction

NuScale believes that the proposed[1] design basis for the NuScale Reactor complies with regulatory requirements and provides sufficiently safe and sufficiently reliable shutdown capability to provide reasonable assurance of adequate protection of public health and safety.

This paper first sets forth NuScale's interpretation of General Design Criteria (GDCs) 26 and 27. NuScale interprets the pertinent provisions of GDC 26 and GDC 27[2] to require adequate reactivity control capability to protect the core under normal and anticipated operational occurrence (AOO) conditions and support core coolability under design basis accident (DBA) conditions. In other words, the protection function of these GDCs pertains to reactivity reduction to the extent necessary to prevent or mitigate radiological release. Thus, these provisions do not require shutdown per se; rather shutdown is addressed only by the last provision of GDC 26.

Next, this paper summarizes the NuScale design bases for reactivity control to explain how the design bases relate to and will satisfy the GDCs and other regulatory requirements.

Finally, this paper summarizes the NuScale shutdown capabilities, and demonstrates that such capabilities are sufficiently safe and sufficiently reliable, such that adequate protection of public health and safety is assured.

Technical information presented in this paper is preliminary and may differ from technical information which will be provided in the NuScale Design Certification Application. Any differences should not be significant with respect to the policy and technical issues discussed in this paper.

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[1] Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission (U.S. NRC), "NuScale Power, LLC Submittal of Presentation Materials Entitled "Reactivity Control, NuScale Design and Licensing Basis," Revision 0, PM-0416-48560 for Use during a Closed Meeting on May 19, 2016 (NRC Project No. 0769)," LO-0416-48923, May 17, 2016 (Accession No. ML16145A500).

[2] The provisions of focus here are the second sentence of GDC 26 and GDC 27. These provisions are relevant in consideration of NRC's September 8, 2016 Gap Analysis Response letter (ML16116A083), which provided an interpretation of GDC 27 that differs from NuScale's. The second sentence of GDC 26 is similar to that at issue in GDC 27, but relates to normal and AOO conditions rather than accidents.

2.0 Regulatory Interpretation

2.1 Summary

NuScale interprets GDCs 26 and 27 to require two separate reactivity control functions, as relevant to the scope of this paper:

1) Protection function: Sufficient control of reactivity changes to assure fuel protection (normal operation and AOOs) or core coolability (DBAs). Due to the safety significance of this function, it shall be assured with sufficient reliability and sufficient margin (worst rod stuck out). The protection function is addressed by the second sentence of GDC 26 and by GDC 27.

2) Shutdown function: Capability to hold the core subcritical under cold conditions. The shutdown function is addressed by the fourth sentence of GDC 26.

Because the protection function is concerned with preventing and mitigating radiological release, shutdown is not required per se. Accordingly, a safety analysis of a transient that predicts a return to power, without violating the transient acceptance criteria, would satisfy those provisions.

2.2 Plain Language

The relevant portion of GDC 26 states: “One of the [reactivity control] systems shall use control rods… and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.”

GDC 27 states: “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.”

Under a plain reading of these GDCs, the acceptance criterion of each provision requires that reactivity control ensure protection of the core—e.g., “reliably controlling reactivity changes <u>to assure</u>… specified acceptable fuel design limits are not exceeded.” Thus, NuScale understands that controlling reactivity changes is not an independent requirement, but rather is required to the extent necessary to protect fuel or ensure core coolability.

Additionally, “controlling reactivity changes” is not equivalent to assuring shutdown. Shutdown may be a sufficient means of controlling reactivity changes, but it is not necessarily required under a plain reading. The focus on reactivity <u>changes</u> relates to the protection function, and implies that reactivity in its own right is not the issue, but rather the issue is controlling changes in reactivity during a transient to prevent unacceptable consequences. As explained below, the Atomic Energy Commission

(AEC) demonstrated in the GDC rulemaking that the choice of terminology was intentional by removing "shutdown" from these provisions in the final GDCs.

2.3 Context

The role of GDCs 26 and 27 within the context of the remaining GDCs supports the plain reading that the provisions at issue are concerned with protecting the core and mitigating accidents, not ensuring shutdown. At a high level, the GDCs "establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public."[3] NuScale's reading of GDCs 26 and 27 is consistent with assuring public health and safety, as it would prevent or mitigate radiological release. In the case of the NuScale design, no radiological release is predicted due to a return to low power, as significant margin exists to maintain fuel clad integrity.

GDC 26 contains a separate provision squarely aimed at shutdown: "One of the systems shall be capable of holding the reactor core subcritical under cold conditions." Notably, this provision does not specify margin for stuck rods. To read the subject provisions of GDC 26 and 27—which cover normal, AOO, and accident conditions—to also prescribe shutdown (with margin for stuck rods) would render this provision superfluous.

GDC 20 sets forth the protection system safety functions, and requires a protection system "to (1) initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety" (emphasis added). GDC 29 states "The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences." Thus, GDCs 20 and 29 reinforce the relationship between the protection and reactivity control systems to prevent and mitigate accidents. In other words, the primary safety function of the reactivity control systems is to accomplish the public health and safety objective of preventing or mitigating radiological release. Shutdown capability, on the other hand, is addressed only by the last sentence of GDC 26.

Moreover, strict application of the GDCs would be inconsistent with the role they serve in the overall regulatory framework. As the Commission has stated:

> *General design criteria (GDC), as their name implies, are "intended to provide engineering goals rather than precise tests or methodologies by which reactor safety [can] be fully and satisfactorily gauged." They are cast in broad, general terms and constitute the minimum requirements for the principal design criteria of water-cooled nuclear power plants. There are a variety of methods for*

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[3] 10 CFR pt. 50, Appendix A (emphasis added).

demonstrating compliance with GDC. Through regulatory guides, standard format and content guides for safety analysis reports, Standard Review Plan provisions, and Branch Technical Positions, license applicants are given guidance as to acceptable methods for implementing the general criteria. However, applicants are free to select other methods to achieve the same goal.… Even if there is nonconformance with the staff's guidance to licensees, the GDC may still be met.[4]

Because the GDCs were intended to be flexible, strict application of past licensee implementation should not be the test for compliance. Rather, NuScale should be able to demonstrate compliance through other methods, within the constraints established by the plain language of the GDCs and consistent with their intent.

2.4 Regulatory History

The current GDCs 26 and 27 originated from four draft criteria in the proposed criteria[5] (draft GDCs, or DGDCs).

DGDCs 28, 29, and 30 comprise what became the provisions of final GDCs 26 and 27 at issue here.[6] Each of these three DGDCs addressed “shutdown” or “holddown” capability, and each explicitly required that the core be made “subcritical” under various conditions. Final GDCs 26 and 27 eliminated the term “subcritical,” except in the last sentence of GDC 26.

A complete history of the considerations the Atomic Energy Commission (AEC) made in proposing, revising, and adopting the GDCs is not readily available, but public comments that the AEC received and considered provide some insight.

DGDC 28 and DGDC 29 both concerned <u>fast shutdown</u> capability. DGDC 28 would have required at least two systems that could make and hold the core subcritical from normal operating conditions. DGDC 29 would have required that one of the systems make (not hold) the core subcritical under normal and AOO conditions, with shutdown margins greater than the worst rod stuck out (WRSO). Several commenters[7] indicated that redundant fast shutdown capability per DGDC 28 was inconsistent with current practice and only one reactivity control system was needed for these conditions. The AEC agreed that only one means of “fast shutdown” was needed, [8] but in so doing also removed the term shutdown (apparently in response to other comments, see below).

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[4] *In the Matter of Petition for Remedial Action*, 7 N.R.C. 400, 406 (1978), quoting *Nader v. NRC*, 513 F.2d 1045, 1052 (1975) (citations omitted).
[5] General Design Criteria for Nuclear Power Plant Construction Permits, 32 Fed. Reg. 10,213 (1967).
[6] DGDC 27 became the first sentence of GDC 26.
[7] *E.g.*, Letter from G. J. Stathakis, General Electric Company to Secretary, U.S. Atomic Energy Commission, Sept., 5, 1967 (Accession No. 9210120281); Letter from W. B. Behnke Jr., Commonwealth Edison Company, to W.B. McCool, Secretary, U.S. Atomic Energy Commission, Sept. 8, 1967 (Accession No. 9210130236); Letter from J. J. Flaherty, Atomics International Division, to Secretary, U.S. Atomic Energy Commission, Sept. 25, 1967 (Accession No. ML003726564).
[8] U.S. Atomic Energy Commission, SECY R-143, Amendment to 10 CFR 50 – General Design Criteria for Nuclear Power Plants, Jan. 28, 1971 (Accession No. ML072420278).

The final requirement, formerly for fast shutdown, was consolidated as the second sentence of GDC 26.

The original intent of DGDC 30 is unclear. It related to reactivity "holddown," rather than fast shutdown, and would have required one system "capable of making and holding the core subcritical under any conditions with appropriate margins."[9] Numerous commenters interpreted the AEC's intent here—given the coverage of normal operations and AOOs by DGDC 28 and 29—to concern accident conditions, not "any" conditions.[10] Whether that was the original intent, AEC eventually adopted GDC 27 specifically to address reactivity control for accident conditions (while addressing cold shutdown in the last sentence of GDC 26). In so doing, AEC appears to have heeded comments by the Atomic Industrial Forum that a return to low power may be acceptable with respect to public health and safety.[11]

The final GDCs appear to reflect a recommendation to "distinguish more clearly between functions of reactivity control; namely, the dynamic reactivity reduction process and the static holddown functions."[12] Comments from Oak Ridge National Laboratory (ORNL) stated that the dynamic reactivity reduction function "must be performed at such times as in power transients and loss-of-coolant accidents with the objective of preventing exceeding 'acceptable fuel damage limits.'" AEC distinguished those functions in the final GDCs, with the last sentence of GDC 26 being the static holddown requirement. ORNL further stated "The reliability with which each function [dynamic reactivity reduction versus static holddown] must be carried out depends upon the seriousness of the consequences of failure of that function." In the final GDCs, the dynamic reactivity reduction requirements of GDCs 26 and 27—the provisions at issue here—require reliability and sufficient margin, whereas the static holddown requirement (last sentence of GDC 26) does not address either.

In summary, whereas there was overlap and interrelationship between DGDCs 28, 29, and 30, ultimately they were restructured such that GDC 26 deals exclusively with normal and AOO conditions, and GDC 27 deals with accident conditions. In both cases, the final GDCs eliminated the term "shutdown" in the provisions at issue. This approach seems to reflect the considerations that reactivity "reduction" for the objective of

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[9] General Design Criteria, 32 Fed. Reg. 10,213 (emphasis added).
[10] Letter from J. C. Rengel, Westinghouse Electric Corporation to Secretary, U.S. Atomic Energy Commission, Sept. 8, 1967 (Accession No. 9210130230); Behnke Jr., *supra* n. 7; Letter from E. Wiggin, Atomics Industrial Forum to Secretary, U.S. Atomic Energy Commission, Oct. 2, 1967 (Accession No. ML003674718).
[11] Wiggin, *supra* n. 10 ("'The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the-health and safety of the public.' … Further, the public health and safety will not be compromised by a return to low power." *See also* Rengel, *supra* n. 10 ("We have inserted the words "in a timely fashion" to clarify the criterion so that it does not arbitrarily rule out a short return to criticality during the shutdown transient.") While Rengel was concerned with only short term return to power, the AEC's deletion of the term shutdown altogether demonstrates a preference for greater flexibility.
[12] Letter from W.B. Cottrell, Nuclear Safety Information Center, Oak Ridge National Laboratory to H.L. Price, Director of Regulation, U.S. Atomic Energy Commission, Sept. 6, 1967 (Accession No. ML003726522) (emphasis added). ORNL generally sought increased requirements for shutdown capability, namely two fast shutdown systems for AOO and accident conditions where containment integrity depended on shutdown. The final GDCs do not reflect those comments.

protecting the core and mitigating accidents was the pertinent safety function with respect to transient events. On the other hand, the shutdown function provision of GDC 26 lacks requirements for reliability and margin.

2.5 Implementation and Precedent

Generally, reactor designers have ensured shutdown with an assumed WRSO in response to AOOs and accidents as a means of ensuring safety and meeting GDCs 26 and 27, and all currently licensed PWRs would remain subcritical beyond the short term relying only upon safety-related SSCs. However, NuScale believes that while this is an acceptable manner of meeting the GDCs, it is not the only way. NRC implementation and precedent supports this interpretation.

First, SRP Chapter 15.0 does not address reactor shutdown as an analysis acceptance criterion for AOOs or accidents.[13] GDC 26 is listed as an applicable requirement "as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded even during AOOs," and GDC 27 is listed "as [it relates] to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained." NuScale recognizes the importance NRC has ascribed to long term safe shutdown using safety-related means through guidance documents such as RG 1.139 and BTP 5-4.[14] Those expectations were developed as guidance, however, suggesting that it is not strictly a requirement of the GDCs.

That PWRs can incur return to power in the short term further supports this point. Of note, Generic Letter 85-16 encourages the removal or decommissioning of Boron Injection Tanks (BITs) in Westinghouse PWRs due to problems associated with them. In the Westinghouse report supporting BIT elimination at Indian Point 3 (typical of others), Westinghouse states:

> *The credible steamline break analysis was performed using a revised criterion whereby the plant may return to criticality but no damage may occur to the fuel. This constitutes a relaxation of the conservative internal Westinghouse criterion for Class II events. This criterion is in compliance with the criteria used by the NRC and ANS, which require that releases during steamline break accidents remain within the limits set forth in 10CFR Part 20. This limit could be met with a return to criticality if it is assured that there is no consequential fuel damage.*[15]

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[13] Other SRP Sections do address GDCs 26 and 27 within the context of shutdown. *See, e.g.*, U.S. NRC, NUREG-0800,*Standard Review Plan*, § 4.3 Nuclear Design (March 2007) ("The nuclear design review verifies that the RCSs provide a movable control rod system and a liquid poison system and that the core has sufficient shutdown margin assuming a stuck rod."). As discussed *infra*, NuScale believes the technical basis underlying this guidance is not applicable to the NuScale design, as a calculated return to low power under conservative assumptions does not pose a safety concern. The Standard Review Plan is not a substitute for the regulations, and compliance with it is not required. 10 CFR 52.47(a)(9).

[14] *See* n. 18, *infra*.

[15] Westinghouse, *Revised Feasibility Report for BIT Elimination for Indian Point 3* (1988) (Accession No. 9305110283).

This was a change from an internal “conservative” Westinghouse criterion that there be “no return to criticality” for the credible steamline break. The credible steamline breaks at issue were Class II events such as a relief valve failing open; in other words, an AOO. The resulting analysis shows that following BIT removal, return to power would occur approximately 600 seconds following a steam generator safety valve opening, criticality would last until approximately 1000 seconds, and the power level would peak at approximately 1 percent.[16] Likewise, for the hypothetical steam line break (a condition IV event, or design basis accident), the Westinghouse acceptance criteria considered only radiological dose, pursuant to 10 CFR Part 100 limits. The resulting analysis shows that following BIT removal, return to power would occur approximately 30 seconds following the event for steamline breaks, would continue beyond the analyzed time period (700 seconds), and the power level would peak at approximately 17 percent before achieving a steady 5 percent.[17]

NuScale recognizes that this and other licensing precedent related to transient return to power relates to what may be considered “short term” return to power. That said, from the plain language of GDCs 26 and 27, NuScale is unable to discern a legal distinction between a return to power 10 minutes after an event versus several hours after an event.

Furthermore, in light of PWR precedent concerning “long-term” shutdown with an assumed WRSO in meeting GDCs 26 and 27, NuScale believes that the safety basis underlying such an approach does not apply to the NuScale design. The safety issue that underpins guidance concerning a safety related means to achieve cold shutdown (RG 1.139 and BTP 5-4) is the potential to challenge heat removal capabilities when power produced after a failure to shutdown exceeds decay heat levels. [18] Similarly, specific guidance for achieving safe shutdown after a LOCA is given in SRP 6.3 with respect to an acceptable means for complying with GDC 27, also to address the potential challenge to heat removal. Per SRP 6.3, Technical Rationale 5, “GDC 27 is applicable because upon actuation the ECCS in PWRs provides rapid injection of borated water to ensure reactor shutdown and adequate core cooling with appropriate margins for stuck control rods. Injection of borated water provides negative reactivity to <u>reduce reactor power to residual levels and ensures sufficient cooling flow to the core</u>” (emphasis added). For currently licensed PWRs, ECCS long-term heat removal is sized to remove decay heat. Therefore, shutdown is evaluated with margin for stuck control

[16] *See id.* at Figure 10 and Figure 11 (data points are estimated from graphs of low quality reproduction).
[17] *See id.* at Figure 2 and Figure 3 (data points are estimated from graphs of low quality reproduction).
[18] U.S. NRC, Regulatory Guide 1.139, Guidance for Residual Heat Removal (1978) (“[I]n the event of a plant trip even with a successful operation of the [reactor protection system], systems or equipment failures that led to the inability to remove decay heat resulted in a higher probability of a core melt than that predicted for a large LOCA for both PWRs and BWRs”); U.S. NRC, NUREG-0800, Standard Review Plan, Branch Technical Position 5-4: Design Requirements of the Residual Heat Removal System (Rev. 4, 2007) (related to RHR system design and capability, pursuant to GDC 34). RG 1.139 was withdrawn in 2008 “because it describes an overly conservative and prescriptive method for complying with [GDC 34s and 39].” Withdraw of Regulatory Guide 1.139, 73 Fed. Reg. 32,750 (June 10, 2008). *See also* U.S. NRC, SECY-94-084, Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, March 28, 1994 (citing GDC 34 in addressing the shutdown cooling capability of passive ALWR designs).

rods, in order to ensure capability to reduce power to decay heat levels such that adequate heat can be removed and fuel integrity can be maintained.

As discussed below, the NuScale design provides inherent reactivity control capability and passive, safety-related means for adequate long-term heat removal such that fuel integrity is not challenged, even in the unlikely event of a return to low power with an assumed WRSO. Therefore, preventing long term return to power with WRSO to satisfy the GDCs does not present the same safety benefit to the NuScale design that has previously been considered. Accordingly, while long-term shutdown with an assumed WRSO has been an acceptable means of meeting GDCs 26 and 27, NuScale believes it is not necessary in this instance.

3.0 Design Basis in Relation to Regulatory Requirements

Under NuScale's interpretation of GDCs 26 and 27, the protection function of the reactivity control systems can be met even with a long-term return to power, so long as the fuel is protected or core coolability ensured. Pursuant to 10 CFR 52.47(a)(3)(ii), the NuScale FSAR would reflect the relation of the facility design bases to the principal design criteria as follows:

- GDC 26: "Two independent reactivity control systems of different design principles shall be provided." The NuScale design basis incorporates two independent reactivity control systems—the control rods and Chemical and Volume Control System (CVCS) boron addition.

- GDC 26: "One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded." Under the NuScale design basis, during normal operation sufficient negative reactivity is maintained (instantaneous shutdown margin) to ensure that the fuel design limits will be protected by rapid control rod insertion with an assumed WRSO.

- GDC 26 "The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded." Under the NuScale design basis, CVCS is used during normal operation to adjust the target boron concentration during power changes to maintain shutdown margin (instantaneous shutdown margin used in safety analysis) and rod insertion limits prior to an AOO, to protect fuel design limits.

- GDC 26 "One of the systems shall be capable of holding the reactor core subcritical under cold conditions." Under the NuScale design basis, control rods, with all control rods inserted, can maintain the reactor shutdown under cold conditions (long term shutdown margin).

- GDC 27 "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." Under the NuScale design basis, during normal operation sufficient negative reactivity is maintained (instantaneous shutdown margin) to ensure that the capability to cool the core is maintained under accident conditions, by rapid control rod insertion with an assumed WRSO.

Accordingly, Chapter 15 of the NuScale FSAR will provide a safety analysis to demonstrate that the NuScale SSCs will perform their design basis functions under analyzed events. The safety analysis would demonstrate that the second sentence of GDC 26 and GDC 27 are met by providing sufficient reactivity control such that the specified acceptable fuel design limits are not exceeded under AOO conditions and core coolability is maintained under accident conditions. This Chapter 15 safety analysis, using conservative and bounding assumptions, including WRSO, may yield the possibility of return to low power in the long-term period following an AOO or DBA.[19] Maintaining a shutdown condition is not necessary to support safety system functions, as the NuScale safety systems would continue to adequately remove reactor heat to ensure fuel protection is not further challenged. Under NuScale's interpretation, this design basis would satisfy the relevant provisions of GDCs 26 and 27.

NuScale's design basis separately satisfies the shutdown provision of GDC 26 by providing control rods with sufficient negative reactivity to shut down the reactor and maintain it in a shutdown condition indefinitely. Control rods alone will maintain the core subcritical to a conservative minimum RCS temperature. This function of GDC 26 does not prescribe WRSO margin, and NuScale would satisfy it with all rods inserted. While this seems to create incongruence with the analysis described above, NuScale believes that the shutdown requirement is distinct and serves a different purpose than the protection function. Given the public health and safety implications of the reactivity control protection function, the WRSO margin and other conservative assumptions provide margin to ensure fuel protection or core coolability. Cold shutdown capability under the fourth sentence of GDC 26, however, can be demonstrated with all rods inserted.

As to safety classification, the GDCs do not prescribe safety-related or seismic classification, which is determined by applying the seismic criteria of 10 CFR 100 Appendix A and safety-related criteria of 10 CFR 50.2. NuScale will apply those criteria commensurate with the facility design basis. NuScale determined that the control rods would be Seismic Category I and safety-related, while boron addition (CVCS) would not:

[19] Relying on heat removal through the reactor vent valves and reactor recirculation valves, long term power after a return to power will be limited to less than 0.1% of rated power due to negative reactivity feedback from voiding. Realistically, a return to power cannot occur during a DHRS cooldown prior to transitioning to heat removal through the reactor vent valves and reactor recirculation valves. Preliminary conservative analysis of a return to power for a DHRS cooldown in Chapter 15 results in a peak power of 8% of rated power after two hours and a long term power level of less than 2%.

- Under the second criterion, control rods, alone, provide the capability to shut down the reactor and maintain it in a safe shutdown condition. This application follows from NuScale's interpretation of GDCs 26 and 27—because the second sentence of GDC 26 and GDC 27 do not require shutdown per se, the control rods would not be relied on to provide shut down capability for transient conditions. The Seismic I control rods would, however, maintain the reactor shutdown under a safe shutdown earthquake, where it is not required to assume a failure of the control rods.

- Under the third criterion, the control rods, alone, are relied on to prevent or mitigate the consequences of accidents. This is the function required by GDC 27, which the control rods perform, with WRSO, by ensuring core coolability and maintaining offsite doses within acceptable limits.

Accordingly, NuScale's design basis would comply with the relevant regulatory requirements.

4.0 Safety and Reliability of NuScale Capability for Long term Shutdown

NuScale has previously addressed the adequacy of short-term reactivity control capability to explain that the AOO and accident acceptance criteria of the GDCs and SRP chapter 15 would be fully met. The following summarizes the basis for NuScale's determination that the long term shutdown capability is sufficiently safe and sufficiently reliable to provide reasonable assurance of adequate protection of public health and safety.

4.1 Description of Long Term Shutdown Capability

Subsequent to a reactor trip, control rods with all control rods inserted provide sufficient negative reactivity to shut down the reactor and maintain it in a shutdown condition indefinitely. In the event of a control rod that failed to insert, soluble boron can be added to the reactor coolant system using the nonsafety-related chemical volume and control system (CVCS), if needed. Soluble boron can also be added to the containment and subsequently to the RCS through natural circulation when the reactor vent valves (RVVs) and reactor recirculation valves (RRVs) are opened, using the nonsafety-related containment flood and drain system (CFDS). A potential return to low power can occur subsequent to a reactor trip only under all of the following conditions:

- the trip occurred late in the fuel cycle
- the highest worth control rod is assumed stuck out
- decay heat levels are very low, and
- a nonsafety-related means for boration is not available.[20]

[20] A return to power will be prevented early in cycle for the equilibrium core design due to a favorable moderator temperature coefficient. A return to power is also prevented late in cycle when high decay heat levels prevail; with sufficient decay heat present, negative reactivity from voiding will prevent a return to power. For a normal fuel cycle, high decay heat levels at the end of cycle will prevent a return to power for 100 days after a reactor trip. After a return

4.2 Basis for Shutdown Capability That is Sufficiently Safe

The shutdown capability for the NuScale design is sufficiently safe because fuel is protected and heat is removed through passive means, even in the unlikely event of a return to low power. In the event of a return to low power, fuel is protected through inherent physical processes that control reactivity, limit power, and cool the core. The inherent means for limiting the power is dependent on the heat removal system used.

For design basis events that rely on heat removal using natural circulation flow through the RVVs and RRVs, the heat produced from a return to power will be limited to less than 100 kW[21] (0.06% of rated power) by negative reactivity feedback from voiding. If decay heat exceeds 100 kW, the reactor will be maintained subcritical even with the WRSO because of negative void reactivity. Therefore, the heat produced after a return to power with natural circulation through the RVVs and RRVs is non-limiting and is bounded by maximum decay heat. Consequently, the maximum decay heat curve, with the reactor subcritical, is used to demonstrate maintaining fuel integrity and to demonstrate emergency core cooling system (ECCS) heat removal capability during long term residual and decay heat removal.

For design basis events that rely on heat removal using the decay heat removal system (DHRS), heat produced after a return to low power with a stuck control rod will be limited by negative moderator temperature feedback. A return to power for a generic cooldown transient with DHRS is evaluated and included in Chapter 15 of the NuScale FSAR to demonstrate that fuel cladding integrity is maintained. The time to a return to power and the peak power level attained in the evaluation is based on conservative assumptions for the purpose of demonstrating fuel protection and is not an indication of realistic shutdown capability.[22] Without AC power, all events that rely on DHRS for heat removal will transition to heat removal using natural circulation flow through the RVVs and RRVs after 24 hours. Subsequently the fission power level from a return to power will be limited to less than a 100 kW by negative reactivity feedback from voiding.

4.3 Basis for Shutdown Capability That is Sufficiently Reliable

The shutdown capability for the NuScale design is sufficiently reliable because the probability for a return to power is insignificant. A bounding probability for a return to power is calculated to be less than 1E-6 per reactor year. Calculation of the bounding probability takes into account the reliability of reactivity control systems and the likelihood that the reactor is in a state that can subsequently lead to a return to power. Using a trip frequency of one trip per reactor year, the contributions to a bounding probability for a return to power are as follows:

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to power, long term power will be limited to less than 0.1% of rated power due to negative reactivity feedback from voiding.

[21] Based on current best estimate calculation for the NuScale equilibrium core design.

[22] Realistically, a return to power cannot occur during a DHRS cooldown; the NuScale Power Module will transition to heat removal through the reactor vent valves and reactor recirculation valves prior to a return to power. However, preliminary conservative analysis of a return to power for a DHRS cooldown results in a peak power of 8% of rated power after two hours and a long term power level of less than 2%.

- The probability that one out of 16 control rods fails to insert is 2E-4 per demand (1.3E-5 per demand per control rod x 16). This probability is bounding as only some of the control rods with a higher rod worth would lead to a return to power when stuck out. Further, industry experience with stuck control rods involved control rods that did not fully insert rather than stuck in the fully withdrawn position. The contribution of external initiating events to the reliability of control rod insertion is negligible because control rods are designed to insert under conditions during and after external initiating events.

- The probability of a CVCS failure to insert soluble boron is 8E-3 per demand. This probability is a bounding probability for a failure to insert soluble boron as it does not take into account alternative means for increasing soluble boron using the CFDS. Using the CFDS, soluble boron can be added to the containment and subsequently to the RCS through natural circulation when the reactor vent valves (RVVs) and reactor recirculation valves (RRVs) are opened. The contribution of external initiating events to the unavailability of these systems is small because of the low probability of external initiating events.

- The probability that the reactor is in a state which could result in a return to power with a WRSO is 4E-2 to 1E-1 per year.[23] This probability is bounding as it assumes that a return to power can occur shortly after a restart at any time during the fuel cycle, if limited decay heat is produced due to limited time at power. In reality, a return to power will be prevented early in cycle for the equilibrium core design due to a favorable moderator temperature coefficient. A return to power is also prevented late in cycle when high decay heat levels prevail; with sufficient decay heat present, negative reactivity from voiding will prevent a return to power.

Therefore, a bounding probability for a return to power is calculated to be between 7E-8 to 2E-7 per reactor year. A further implication of the low probability is that nonsafety-related systems do not need to be designed for external events in order to ensure a sufficiently low probability for a return to power.

The NRC and the nuclear industry have not established a goal for acceptable shutdown reliability. However, in the following instances, NRC has considered whether the probability for an inadvertent return to power is acceptable, and whether the risk from an anticipated transient without scram (ATWS) is acceptable:

- GSI-22, Inadvertent Boron Dilution Events. The probability for an inadvertent return to power due to boron dilution during a shutdown or refueling was calculated to have a probability 2E-4 per reactor year. A 3E-5 per reactor year probability was calculated for offsite radiological release of gap activity from any leak already present in the fuel. Given the low probability and low consequences from such an event, NRC staff concluded that boron dilution events did not constitute a significant risk to

[23] This assumption bounds future core designs. Future core designs may have conditions that can lead to a return to power which differ from the equilibrium core design used for design certification.

the public.[24] GSI-22 was closed with no new requirements and without implementing potential plant changes.

- NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States. The potential for core damage due to rapid boron dilution may occur with a frequency on the order of 1E-5 per reactor year. This assessment was performed using conservative system parameters without which core damage would be avoided. No further regulatory requirements were developed to reduce the potential for rapid boron dilution.

- GSI-185, Control of Recriticality Following Small-Break LOCAs in PWRs. The probability for core damage due to inadvertent boron dilution during a small-break LOCA transient was calculated to be 3E-8 for B&W designs. Due to the low risk associated with this issue GSI-185 was closed with no changes to existing regulations or guidance

- 10 CFR 50.62 ATWS. The intent of the ATWS rule is to reduce the CDF contribution from ATWS to less than 1E-5 per reactor year.[25] A failure to scram after an anticipated transient, which was calculated to occur between 1E-4 to 1E-5 per reactor year for operating LWRs, is considered a beyond design basis event. The potential for ATWS events are, however, evaluated as part of the plant design basis in order to ensure that the ATWS CDF contribution is acceptably low.

The bounding probability for a return to low power for the NuScale design is significantly smaller than what was considered for GSI-22 and NUREG-1449. Because no fuel damage or offsite release is predicted due to a return to low power in the NuScale reactor, there are no radiological consequences and thus no risk. Therefore, design or operational requirements to further reduce the low frequency or reduce the consequences of a return to power are not warranted as part of the NuScale design basis.

Further, the probability for a stuck control rod and a failure on demand of the CVCS system is less than 1E-5 per reactor year. Therefore, the occurrence of a reactor trip with a stuck control rod coincident with a failure of the CVCS could be considered a beyond design basis event (BDBE), considering that ATWS events with a similar

[24] U.S. NRC, NUREG-0933, Resolution of Generic Safety Issues: Issue 22: Inadvertent Boron Dilution Events, Rev. 2, GSI-22 concluded: “Based on the low value/impact score and low public risk reduction associated with an inadvertent criticality, DST/NRR concluded that boron dilution events did not constitute a significant risk to the public and recommended that the issue be dropped from further consideration. However, DSI/NRR disagreed with this evaluation and obtained permission from the NRR Director to pursue the issue further. As a result of DSI's work, it was determined that the consequences of an unmitigated boron dilution event, although undesirable, were not severe enough to warrant backfit of additional protective features at operating plants.… Thus, this issue was RESOLVED and no new requirements were established.”
[25] *See* U.S. NRC, SECY 83-293, "Amendments To 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," July 19,1983.

frequency are classified as beyond design basis events. A threshold of 1E-4 has been proposed for event classification of BDBEs of previous advanced designs.[26]

Regulation of low probability events with no safety consequence, such as the unlikely potential for a benign return to power in the NuScale design, would be inconsistent with past resolution of issues that have a low probability and potential for low to significant consequences, such as rapid boron dilution, which the NRC has closed with no changes to existing regulations or guidance.

5.0 Conclusion

In summary, NuScale interprets GDCs 26 and 27 to require two separate and distinct functions: a protection function and a shutdown function. Because of the importance of the protection function to public health and safety, capability for rapid reduction of reactivity must be sufficiently reliable and with margin for malfunctions such as stuck rods. The shutdown function does not prescribe those additional conditions. Based on this interpretation, NuScale's design basis would fulfill the GDCs by addressing the protection function with a stuck rod assumption, and show conservatively that safety functions are performed and the applicable analysis acceptance criteria are met. NuScale's design basis would fulfill the shutdown function with all rods inserted.

NuScale's control rods will be Seismic category I and safety-related, and with all rods inserted provide long-term shutdown capability consistent with seismic category I criteria (10 CFR 100 Appendix A) and safety-related criteria (10 CFR 50.2).

The probability of a stuck rod occurring coincident with other reactor conditions necessary to cause a return to power is less than 1E-6 per reactor year. In the unlikely event of a return to low power, fuel is protected through inherent physical processes that control reactivity, limit power, and cool the core, such that no core damage and no offsite release is predicted under such an occurrence.

Therefore, the control rods provide sufficiently safe and sufficiently reliable long-term shutdown capability to meet both the protection function and the shutdown function requirements. Additional regulatory requirements or guidance to further reduce the probability for a return to power are not warranted, given that the probability for such an event is small and the consequences are benign.

[26] *See* Letter from J.S. Armijo, U.S. NRC Advisory Committee on Reactor Safeguards to R.W. Borchardt, U.S. NRC, "Next Generation Nuclear Plant (NGNP) Key Licensing Issues," May 15, 2013 (Accession No. ML13135A290).