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July 15, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 516 (eRAI No. 9647) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 516 (eRAI No. 9647)," dated February 01, 2019

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

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9647:

- 15-29

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

If you have any questions on this response, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9647

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NuScale Response to NRC Request for Additional Information eRAI No. 9647

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

eRAI No.: 9647
Date of RAI Issue: 02/01/2019

NRC Question No.: 15-29

The initiating event of the rod ejection accident is the rapid ejection of a control rod (CR) caused by an assumed control rod housing failure as described in Standard Review Plan (SRP) Section 15.4.8 and Regulatory Guide 1.77. The requirement to evaluate the rod ejection accident is given by General Design Criterion (GDC) 28 which states,

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Consistent with GDC 28, SRP 15.0 classifies control rod ejection as a postulated accident as does NuScale in Final Safety Analysis Report (FSAR) Section 15.4.8.1. A conservative initial condition assumes the regulating control rods are at rod insertion limit defined by technical specifications. Following the rod ejection, a reactor trip occurs on either a high flux rate, high flux or high pressurizer pressure inserting the remaining control rods. The emergency core cooling systems (ECCS) will actuate either when the inadvertant actuation block (IAB) clears or when containment or reactor coolant system (RCS) level setpoints are reached. According to 10 CFR Part 50.46 (iii)(5),

"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the

extended period of time required by the long- lived radioactivity remaining in the core."

In addition, SRP Section 15.4.8 indicates that GDC 28 provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired by a control rod ejection accident.

As described in FSAR 15.4.8.3.2, conservative scram characteristics are applied as the, "highest worth [control rod assembly (CRA)] (other than the ejected rod) remains stuck out of the core." While the effects of the ejected rod and fully stuck rod are addressed in the short term, it is unclear if the long term effects have been evaluated.

The staff notes that NuScale has requested an exemption to GDC 27, as documented in SECY 18-0099 (ADAMS ML18065A431). SECY 18-0099 states that the staff intends to apply demonstration that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded for postulated accidents that result in a return to power as the part of the technical basis for reviewing the GDC 27 exemption. However, the return to power analysis in FSAR 15.0.6 only evaluates a single stuck rod and not the combined loss of reactivity associated with an ejected rod and stuck rod.

To demonstrate that the minimum critical heat flux (CHF) in FSAR 15.0.6 analyses (which is the long-term analysis that applies to all Chapter 15 events that can result in a return to power) bounds the scenario where ECCS actuates due to the ejected rod and stuck rod, the staff is requesting justification that the current return to power analyses in FSAR 15.0.6 bound this scenario, at any time in the cycle. The justification should address all effects that could either reduce shutdown margin or lead to a return to power, including core boron dilution (see RAI 8930, effects from soluble boron plate-out, boron lost to the lower part of containment and diluted condensate return to the reactor pressure vessel via the reactor recirculation valves, etc.).

NuScale Response:

This RAI response clarifies the NuScale design and licensing basis with respect to limiting reactivity accidents postulated pursuant to GDC 28, and provides additional technical and licensing justification for the acceptability of these licensing bases and the overall safety of the NuScale design.

NuScale's design basis assures that cold shutdown is achievable following any design basis accident with all control rods inserted, in accordance with NuScale-specific PDC 27. Additionally, pursuant to NuScale's requested exemption from GDC 27, FSAR Section 15.0.6 demonstrates that if the single highest worth control rod assembly (CRA) is assumed stuck out, that fuel clad integrity and extended core cooling would be maintained if a return to power occurred in the long-term shutdown period (i.e., holddown) following an accident.

NuScale believes the FSAR 15.0.6 analysis is acceptable with respect to a rod ejection accident (REA). Although the return to power analysis is not directed to an REA, it is bounding with respect to long-term holddown with the single control rod ejected. The return to power analysis does not bound an REA with a CRA stuck out, in addition to the ejected CRA. This licensing basis is acceptable and appropriate for the following reasons:

- GDC 28 imposes core design limits distinct from the reactivity control system capabilities addressed by GDC 27.
- GDC 27 has not previously been applied to the REA analysis required by GDC 28.
- 10 CFR 50.46 and 10 CFR 50 Appendix K also do not apply to the REA analysis.
- PDC 27 derives from GDC 27. The intent of PDC 27 viz-a-viz GDC 27 addresses the NuScale design basis for cold shutdown, which is that cold shutdown is assured with all rods fully inserted. PDC 27 is not intended to expand its applicability to the REA or any other specific event, which is not supported by precedent.
- An REA is not credible for the NuScale design. The REA is deterministically (i.e., non-mechanistically) postulated only for the purpose of evaluating the consequences of a limiting reactivity insertion event, as required by GDC 28. Therefore, the extension of PDC 27 to the REA is not warranted by unique design considerations.

The following discussion details each of these points.

GDC 28 specifies "limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can [not…] sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core." This design limit is distinct from the reactivity control system functional and performance requirements specified by GDCs 26 and 27. The safety concern underlying GDC 28 is described in RG 1.77 as follows:

> *The rate at which reactivity can be inserted into the core of a uranium oxide-fueled water-cooled power reactor is normally limited by the design of the control rod system to a value well below that which would result in serious damage to the reactor system. However, a postulated failure of the control rod system*

> *provides the potential for a relatively high rate of reactivity insertion which, if large enough, could cause a prompt power burst. For UO_2 fuel, a large fraction of this generated nuclear energy is stored momentarily in the fuel and then released to the rest of the system. If the fuel energy densities were high enough, there would exist the potential for prompt rupture of fuel pins and the consequent rapid heat transfer to the water from finely dispersed molten UO_2. Prompt fuel element rupture is defined herein as a rapid increase in internal fuel rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of fuel cladding into the coolant. This is accompanied by the conversion of nuclear energy, deposited as overpower heat in the fuel and in the coolant, to mechanical energy which, in sufficient quantity, could conceivably disarrange the reactor core or breach the primary system.*

Therefore, NuScale understands the intent of GDC 28 is to assure the core design prevents a limiting, prompt reactivity insertion from causing gross damage to the core or RCPB that would preclude accident mitigation via continued core cooling. This intent is affirmed by pending DG-1327, which states:

> *Reactivity insertion accidents, such as PWR CRE and BWR CRD, directly affect the core by challenging fuel rod bundle array geometry. Rapid local power excursions may cause gross failure of fuel rods and loss of a coolable core geometry. Furthermore, molten fuel ejected from failed rods will interact with the reactor coolant, producing a pressure pulse that may challenge the integrity of the reactor pressure boundary.*

As stated in SRP 15.4.8, avoiding damage to the RCPB and substantial disturbance of core geometry "provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired," because, as NuScale understands, these attributes ensure the capability for continued core cooling following the postulated accident. Reactor holddown capability is not identified by guidance as an applicable acceptance criterion for the postulated rod ejection accident.

GDC 27 has historically not been applied to a rod ejection accident. The rod ejection analysis is normally limited to the first few seconds of the event. GDC 27 prescribes reactivity control requirements "under postulated accident conditions," and if it were applied to an REA would require that an additional stuck rod assembly be considered in verifying "the capability to cool the core" is maintained. That is, all rod ejection analyses would be required to consider an additional stuck rod during, at a minimum, the short term transient phase ordinarily analyzed by applicants. Notwithstanding, GDC 27 is not cited in SRP 15.4.8 or RG 1.77 as an applicable regulation for the REA, and neither document addresses a deterministic stuck rod assumption.[1]

Precedent supports NuScale's understanding. With respect to GDC 27, based on precedent surveyed by NuScale, only some FSARs have assumed an additional withdrawn rod during the REA, and none have cited GDC 27 as a reason for doing so.[2] As with NuScale's 15.4.8 analysis, these analyses appear to assume the stuck rod for additional conservatism. Contrary to NuScale's analysis, this conservatism appears to materially influence the results of the analysis for some PWRs because the local reactivity insertion upon reactor trip may be affected and thus impact margin to the clad temperature and pressure boundary acceptance criteria.[3] Therefore, NuScale concludes that if it were applicable, GDC 27 would be clearly stated and universally applied as a requirement for the REA. Rather, NuScale believes that the absence of a stuck rod assumption from the requirements for REA analysis is due to the difference in intent and applicability of GDC 27 and GDC 28.

Likewise, NuScale is unable to determine that any prior applicants have been required to perform a long-term holddown analysis following an REA, with or without an additional stuck rod. While some FSARs indicate the long-term plant response to an REA is the same as that of a LOCA, this conclusion is unsupported by FSAR analyses demonstrating the rate and means of safety-related boron injection with respect to an uncertain reactor coolant system leakage rate. Therefore, NuScale concludes that if long-term shutdown margin -- which NRC Staff have determined is required by GDC 27 -- was pertinent to the REA then precedent applications would be required to analyze that phase of the accident with limiting, conservative assumptions, including the worst rod stuck out.

Similarly, 10 CFR 50.46 does not apply to an REA. The application of the 10 CFR 50.46 acceptance criteria is limited to "breaks in pipes in the reactor coolant pressure boundary" (10 CFR 50.46(c)(1)). An REA is postulated to be caused by a rupture of a CRDM pressure housing, which is a component of the RCPB but not a pipe. 10 CFR 50.46 is not cited in SRP 15.4.8 or RG 1.77 as an applicable regulation for the REA, and NuScale is unable to determine that any prior applicants have applied the 10 CFR 50.46 ECCS acceptance criteria to a REA. For instance, precedent analyses do not and could not apply the 2200 degrees F peak clad temperature limit of 10 CFR 50.46(b)(1) to the REA. This is illustrated by Watts Bar Units 1 and 2, which applied "criteria for gross damage of fuel" in the REA analysis that included a peak clad temperature of 2700 degrees F (NUREG-0847), and the AP1000 DCD calculated a peak clad temperature of 2265 degrees F for one REA case. Therefore, NuScale concludes that the LOCA-specific ECCS acceptance criteria of 10 CFR 50.46, including the long-term cooling criterion cited in the RAI (10 CFR 50.46(b)(5)), are not applicable to the REA analysis.

Therefore, based on the regulations, guidance, and precedent, NuScale's licensing and design basis for a REA are to satisfy GDC 13, GDC 28, and the applicable dose acceptance criteria. Neither SRP 15.4.8 nor the regulatory positions of RG 1.77 prescribe that a stuck control rod be deterministically assumed during the short-term transient. GDC 28 "provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired by a control rod ejection accident" by precluding core or RCPB damage extensive enough that subsequent core cooling — assured by conformance to GDCs 34 and 35 — is prevented.

Although NuScale assumes an additional stuck rod for the short-term REA analysis as a non-mechanistic, conservative assumption, this assumption is not required by regulations or guidance and does not dictate that an REA with stuck rod holddown evaluation be provided in the FSAR. As evidenced by the precedent applications and current guidance, the current regulations do not require that long-term hold-down be evaluated for an REA. NuScale's licensing basis does not alter this framework, because PDC 27 does not require that a second CRA be assumed withdrawn. As noted in the GDC 27 exemption request:

> *PDC 27 is not intended to expand the applicability of GDC 27 beyond its current scope, but only to clarify the shutdown criterion for design basis events within the scope of GDC 27. Shutdown capability after reactivity accidents, specifically rod ejection, is not addressed by GDC 27 or GDC 28. Rather than addressing reactivity insertion for reactivity control or shutdown, GDC 28 is intended to restrict the amount of positive reactivity that can be inserted from reactivity accidents, including rod ejection, and thus limit the consequences of such events.*

Moreover, NuScale does not believe new or different safety considerations justify newly imposing PDC 27 as a requirement for REA specifically for the NuScale design. NuScale's 15.4.8.3.2 analysis assumes conservative scram characteristics, including the non-mechanistic assumption of the highest worth CRA stuck out of the core, demonstrating that expected consequences from a second rod being unavailable are acceptable with respect to the RCPB and core integrity criteria of GDC 28. However, NuScale estimates the likelihood of an REA together with an additional, nonconsequential failure of a control rod to insert to be on the order of 1E-8 per year. If failure of boron addition is considered, which is also a necessary occurrence for a return to power, the likelihood of reactivity control failures that could give rise to a return to power is reduced to approximately 1E-10 per year. Neither probability accounts for the limited time in cycle where the return to power is possible. Given the low probability of the rod ejection event and by applying an acceptance criterion of protecting the fuel cladding by meeting SAFDLs, the risk to the public from such an event is extremely low. Consistent with the July 2, 2019 SRM for SECY-19-0036, NuScale believes it is appropriate to apply risk-informed principles here because "reasonable assurance of adequate protection of public health and safety" does not require "strict, prescriptive application of deterministic criteria."

Consideration of holddown for an REA with an additional stuck rod is also not warranted by mechanistic considerations, because a stuck rod resulting from an REA is not credible. As discussed in FSAR 4.6.2:

> *To prevent a mechanical failure of the CRDM housings, the CRDM nozzles are designed to be an integral part of the RPV. The CRDM pressure housings are full penetration welded to the safe ends of the CRDM nozzles. The safe-end-to-CRDM nozzle welds and safe-end-to-CRDM pressure housing welds are inspected to ASME Class 1 requirements. However, a failure of the CRDM pressure housing is postulated to provide a limiting reactivity insertion event in Section 15.4.*

The CRDM nozzles are integral parts of the reactor pressure vessel closure head forging. Whereas for the existing fleet of PWRs, the CRDM nozzles are partial penetration j-groove welded to the reactor vessel closure head. The CRDM nozzle to Alloy 690 safe-end welds are full penetration butt welds, using Alloy 52/152 weld filler materials for corrosion resistance. Therefore, the connection between the CRDM nozzles and the reactor pressure vessel closure head are more structurally robust for the NuScale design. Consistent with Chapter 3.9.7.1 of the AP1000 Design Control Document, the control rod drive mechanisms do not represent credible sources of missiles or jets due to breaks or cracks. Finally, consistent with Chapter 4.6.2 of the AP1000 Design Control Document, the pressure boundary housing of the control rod drive mechanisms is constructed to the requirements of the ASME Code and a break in this pressure boundary is not credible.

The NuScale Power Module FSAR 3.9.3.1.2 states:

> *The portions of the CRDM providing a RCPB function are ASME Code Class 1, Seismic Category I components. The CRDM pressure housing is a Class 1 appurtenance per ASME BPVC, Section III, NCA-1271. The load combinations and stress limit are presented in Table 3.9-6. The CRDM seismic supports located on both the RPV and CNV head are ASME Code Class 1, Seismic Category I component supports.*

And FSAR 3.5.1.2 states:

> *A control rod drive mechanism (CRDM) housing failure, sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core, is non-credible. The CRDM housing is a Class 1 appurtenance per ASME Section III.*

Thus, while a CRDM housing failure is postulated to cause an REA, this is a non-mechanistic assumption for the purposes of evaluating the consequences of a limiting reactivity insertion event in accordance with GDC 28, only. The failure of the CRDM housing itself is not considered credible because the CRDM housing is a Class 1 appurtenance per ASME Section III, as with other Class 1 vessels and appurtenances (e.g., the RVV and RRV valve bodies) for which gross failure is not considered credible. Because the housing failure is not credible, dynamic effects on adjacent CRDM housings need not be considered.

The last paragraph of the question requests that the effects of boron concentration be addressed. The analysis in FSAR Section 15.0.6 assumes no boron is available during the event. Therefore, none of the mechanisms addressed in RAI 8930 are applicable to this analysis.

Therefore, for the stated reasons, it is the NuScale position that the FSAR 15.0.6 analysis is acceptable with respect to a rod ejection accident (REA). Although the return to power analysis is not directed to an REA, it is bounding with respect to long-term holddown with the single control rod ejected. The return to power analysis does not bound an REA with a CRA stuck out in addition to the ejected CRA.

[1] SRP 15.4.8 provides that the regulatory acceptance criteria for a rod ejection accident are GDC 13, GDC 28, and the offsite radiation dose limits (as implemented at 10 CFR 52.47(a)(2)(iv) for DC applications). RG 1.77 provides regulatory positions and specific guidelines sufficient to meet the relevant requirements of GDC 28. DG-1327, intended to replace RG 1.77 for new applications, also cites only GDC 28 as an applicable regulation for the REA.

[2] For example, the AREVA EPR Rod Ejection methodology (ANP-10286) did not address whether an additional unavailable rod was assumed or not. In a response to RAI -7 from the NRC review, AREVA indicated an additional failed rod was not included in the trippable worth for the REA analysis, only the ejected rod and additional rod worth uncertainty:

> "The last sentence in Section 4.1.6 is intended to address the importance of trip reactivity for the ejected rod event at power even though the Phenomena Importance Ranking Tables (PIRT) analysis did not rate trip reactivity as an important parameter. The insertion of the rods may affect the timing of the severity of the departure from nucleate boiling (DNB) response as core power decreases. The trip reactivity is evaluated in TR Section 7.1.7, Table 7-3 to illustrate the difference between a nominal control rod worth and the rod worth reduced by 9 percent. The reduced scram rod worth simulates a limiting trippable rod worth by reducing the worth by the uncertainty and excluding the ejected rod. The total rod worth uncertainty reduction is 10 percent for shutdown margin calculations; however, the reactivity worth of the inserted bank from which a rod is ejected is increased by more than 15 percent…."

[3] See *Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1*, ML14188C423 ("The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip.").

Impact on DCA:

There are no impacts to the DCA as a result of this response.