ANNOTATED COMMENT SUBMISSIONS ON PROPOSED RULE

NUSCALE SMALL MODULAR REACTOR DESIGN CERTIFICATION

(NRC-2017-0029; RIN 3150-AJ98)

June 2022

PUBLIC SUBMISSION

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Docket: NRC-2017-0029
Design Certification for the NuScale Small Modular Reactor Design Certification

Comment On: NRC-2017-0029-0001
NuScale Small Modular Reactor Design Certification

Document: NRC-2017-0029-DRAFT-0001
Comment on FR Doc # 2021-13940

Submitter Information

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General Comment

I am a radiation safety professional, and as such, it is my opinion that the NuScale design is a safe and effective reactor configuration. However, it is also my opinion that all nuclear power plants should incorporate onsite backup generators. There is no reason (other than cost) not to equip these reactors with backup generation. Even if the plant itself can withstand the accident conditions, why force operators to deal with an accident in the absence of onsite power? The midst of a serious problem is not the time to be managing basic issues such as power. Yes, utilities could install backup power if they wished, but the only way to ensure it is to require it.

1-1

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Docket: NRC-2017-0029
Design Certification for the NuScale Small Modular Reactor Design Certification

Comment On: NRC-2017-0029-0001
NuScale Small Modular Reactor Design Certification

Document: NRC-2017-0029-DRAFT-0002
Comment on FR Doc # 2021-13940

Submitter Information

Name: James Hoerner
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General Comment

I strongly support the design certification of the NuScale SMR.

The NRC has conducted an extremely extensive review, and the safety benefits of NuScale's design are clear, with a low power density core design and walk-away passive safety.

2-1

The innovative design will also play an important role in providing relatively clean, safe, reliable, and cost-competitive base-load electricity and ensuring America remains a leader global nuclear technology.

Sincerely,
James A. Hoerner
Nuclear Fuel Engineer

PUBLIC SUBMISSION

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Design Certification for the NuScale Small Modular Reactor Design Certification

Comment On: NRC-2017-0029-0001
NuScale Small Modular Reactor Design Certification

Document: NRC-2017-0029-DRAFT-0003
Comment on FR Doc # 2021-13940

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General Comment

3-1 See attached file(s)
I do not consent! Shut down the corruption! No more nukes! FOCUS on our WATER

Attachments

NUKIE

Abolish
Price-Anderson
#nuclearban
#NoNUKES

From:	O"NEILL, Martin
To:	RulemakingComments Resource
Cc:	UHLE, Jennifer; Andrukat, Dennis; Lauron, Carolyn
Subject:	[External_Sender] Docket ID: NRC-2017-0329 [RIN 3150–AJ98] -- NEI Comments in Response to Proposed Rule – NuScale Small Modular Reactor Design
Date:	Thursday, October 14, 2021 6:54:18 PM
Attachments:	NEI Comments on Proposed NuScale SMR Design Certification Rule_PDF (10-14-2021).pdf

Dear NRC Rulemakings and Adjudications Staff,

On behalf of Dr. Jennifer Uhle, Vice President, Generation and Suppliers, Nuclear Energy Institute (NEI), please find attached to this email NEI's comments submitted in response to the NRC's NuScale Small Modular Reactor Design Certification; Proposed Rule, 86 Fed. Reg. 34999 (July 1, 2021), for which the comment period was extended to October 14, 2021 by *Federal Register* Notice dated August 24, 2021 (86 Fed. Reg. 47251). Please confirm receipt of these comments.

Please feel free to contact Dr. Uhle or me by email or phone if you have any questions regarding this submittal. NEI appreciates the opportunity to submit comments.

Regards,

Martin O'Neill

NUCLEAR ENERGY INSTITUTE

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NUCLEAR ENERGY INSTITUTE

October 14, 2021

Secretary of the Commission
ATTN: Rulemakings and Adjudications Staff
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Submitted Via Rulemaking.Comments@nrc.gov

Subject: NEI Comments in Response to Proposed Rule – NuScale Small Modular Reactor Design Certification [Docket ID: NRC-2017-0329] [RIN 3150–AJ98]

On July 1, 2021, the U.S. Nuclear Regulatory Commission (NRC) published in the *Federal Register* a proposed rule that would amend 10 CFR Part 52 to certify the NuScale standard design for a small modular reactor (SMR).[1] The proposed rule, which would add a new Appendix G to Part 52, is the culmination of a robust design certification review process that commenced with the NRC's docketing of NuScale's SMR design certification application on March 30, 2017. In August 2020, the NRC staff completed its review and issued the final safety evaluation report (FSER) for the NuScale standard plant design application, and subsequently recommended that the Commission approve the proposed design certification rule (DCR) for public comment.[2] The Commission approved publication of the proposed DCR by a Staff Requirements Memorandum dated May 6, 2021 (ML21126A153).

The Nuclear Energy Institute (NEI)[3] is providing these comments in response to the proposed design certification rule.[4] As explained below, NEI's comments, while not necessarily seeking modifications to the proposed rule, request clarification regarding a regulatory interpretation issue identified by

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1 NuScale Small Modular Reactor Design Certification; Proposed Rule, 86 Fed. Reg. 34,999 (July 1, 2021).

2 *See* SECY-21-0004, "Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98) (NRC-2017-0329)" (Jan. 14, 2021) (ML19353A003).

3 The Nuclear Energy Institute (NEI) is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include entities licensed to operate commercial nuclear power plants in the U.S., nuclear plant designers, major architect/engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations and entities involved in the nuclear energy industry.

4 The public comment period originally was scheduled to close on August 30, 2021. The NRC extended the comment period by 45 days to October 14, 2021 to allow more time for members of the public to develop and submit their comments. *See* 86 Fed. Reg. 47,251 (Aug. 24, 2021).

former Commissioner Caputo in her comments on SECY-21-0004.[5] Specifically, Commissioner Caputo stated, in pertinent part:

> In preparing the draft proposed rule, the staff used language that has been consistently used in design certification rulemakings since the initial issuance of the design certificate rule for the U.S. Advanced Boiling Water Reactor in 1997. Changes to the design certification process over the years, however, have led to a mismatch between the definition of "backfitting" in the Backfit Rule, 10 CFR 50.109, and the backfitting provisions for generic technical specifications and other operational requirements in paragraph VIII.C.1 of this proposed rule. To ensure consistency and clarity in the application of this section in future design certifications, section VIII.C.1 should continue to be interpreted such that changes to generic technical specifications and approved operational requirements would be subject to the Backfit Rule.
>
> We expect future advanced reactor applicants to address additional operational requirements in design. As a result, there needs to be regulatory predictability in how backfitting or issue finality works in this area. *The staff should therefore memorialize in a durable, publicly available document the interpretation of how changes to generic technical specifications and approved operational requirements would be subject to the Backfit Rule*.[6]

Commissioner Caputo proposed some related revisions to the "Operational Requirements" discussion in the Statements of Consideration (SOC) of the draft proposed rule.

NEI does not view this fact as constituting a major flaw in the proposed NuScale DCR. Nonetheless, in the interest of promoting greater regulatory clarity and certainty, NEI agrees that the backfitting/issue finality issue identified by Commissioner Caputo warrants clarification. NEI therefore suggests that the NRC's response to these comments serve as the "durable, publicly available document" that memorializes the NRC's interpretation of how changes to generic technical specifications and approved operational requirements would be subject to the Backfit Rule. Given the generic nature or implications of this issue (it involves certain NRC Part 52 regulations and language that appears in multiple design certification rulemakings), the NRC also might consider addressing the issue within the context of a broader generic rulemaking; e.g., the ongoing Part

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[5] NEI notes that former NRC Chairman Kristine Svinicki raised a similar issue in her comments (ML21012A364) on SECY-20-0112, in which the NRC staff sought Commission approval to publish a direct final rule to renew the certification for the U.S. Advanced Boiling Water Reactor (ABWR) standard design.

[6] Commissioner Caputo's Comments on SECY-21-0004: Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0329) (Apr. 15, 2021) (ML21109A238) (emphasis added).

50/52 lessons learned rulemaking, for which the NRC staff has issued a draft regulatory basis document.[7]

The specific clarifications sought by NEI involve the interrelationships among several discrete provisions within Part 52 and the proposed NuScale DCR. They include:

- **10 CFR 50.109(a)** ("Backfitting"), which states, in relevant part:

  (a)(1) Backfitting is defined as the modification of or addition to systems, structures, components, or design of a facility; or the *design approval* or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission's regulations or the imposition of a regulatory staff position interpreting the Commission's regulations that is either new or different from a previously applicable staff position . . . . (Emphasis added).

- **10 CFR 52.63(a)(5)** ("Finality of standard design certifications"), which states in full:

  (5) Except as provided in 10 CFR 2.335, in making the findings required for issuance of a combined license, construction permit, operating license, or manufacturing license, or for any hearing under § 52.103, the Commission shall treat as resolved those matters resolved in connection with the issuance or renewal of a design certification rule.

- **Section VI.C** under Section VI ("Issue Resolution") of the proposed NuScale DCR (Appendix G), which states in full:

  C. The Commission does not consider operational requirements for an applicant or licensee who references this appendix to be matters resolved within the meaning of § 52.63(a)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

- **Section VIII.C.1** ("Operational Requirements") under Section VIII ("Processes for Changes and Departures") of the proposed NuScale DCR (Appendix G), which states in full:

  1. Changes to NuScale design certification generic TS and other operational requirements that were completely reviewed and approved in the design certification rule and do not require a change to a design feature in the generic DCD are *governed by the requirements in 10 CFR 50.109*. Changes that require a change to a

[7] *See* "Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing, Regulatory Basis for Public Comment," RIN No. 3150-AI66, Docket ID No. NRC-2009-0196 (Jan. 15, 2021) (ML20149K680). For instance, the NRC could amend 10 CFR 50.109(a) to explicitly reference design certifications.

> design feature in the generic DCD are governed by the requirements in paragraphs A or B of this section. (Emphasis added).

- **Section VIII.C.4** ("Operational Requirements") under Section VIII ("Processes for Changes and Departures") of the proposed NuScale DCR (Appendix G) which states, in relevant part:

> 4. An applicant who references this appendix may request an exemption from the generic TS or other operational requirements.

Application of these regulations may engender some potential confusion so as to warrant clarification by the NRC. First, 10 CFR 50.109(a) refers explicitly to design approvals but not to design certifications. The NRC appears to have addressed this issue through the above-quoted language of Section VIII.C.1 of Appendix G, which provides that the backfitting requirements in 10 CFR 50.109 apply to changes to NuScale design certification generic technical specifications and other operational requirements that were completely reviewed and approved in the DCR and do not require a change to a design feature in the generic DCD. The SOC to the NuScale DCR elaborates on this point:

4-1

> The process in paragraph VIII.C.1 for making generic changes to the generic technical specifications in Chapter 16 of the DCD or other operational requirements in the generic DCD *would be accomplished by rulemaking and governed by the backfit standards in § 50.109*. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rule would be based upon the extent to which the NRC reached a safety conclusion in the [FSER] on this matter. If a technical specification or operational requirement was completely reviewed and finalized in the design certification rule, *then the requirement of § 50.109 would apply because a position was taken on that safety matter*. Generic changes made under paragraph VIII.C.1 would be applicable to all applicants or licensees (refer to paragraph VIII.C.2), unless the change is irrelevant because of a plant-specific departure.[8]

Thus, while Section 50.109(a)(1) does not mention design certifications specifically, the language of Section VIII.C.1 of Appendix G expressly incorporates "the requirements in 10 CFR 50.109" into the change process described therein. In essence, Section VIII.C.1 could be viewed as, or at least akin to, a rule of particular applicability, the promulgation of which is well within the NRC's rulemaking authority under the Atomic Energy Act (AEA) and the Administrative Procedure Act (APA).[9] That is,

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[8] 86 Fed. Reg. at 35,010 (emphasis added).

[9] *See* 5 USC 551(4) (defining a "rule" as "the whole or a part of an agency statement of general or particular applicability and future effect designed to implement, interpret, or prescribe law or policy"); *NLRB v. Wyman-Gordon Co.*, 394 U.S. 759, 772 (1969) (Black, J., concurring) ("[S]o long as the matter involved can be dealt with in a way satisfying the definition of either 'rule making' or 'adjudication' under the Administrative Procedure Act, that Act . . . should be read as conferring upon the [agency] the authority to decide, within its informed discretion, whether to proceed by rule making or adjudication.").

while Section VIII.C.1 does apply to all applicants/licensees who reference Appendix G, the change process described therein is specific to the NuScale DCR. NEI requests that the NRC confirm that NEI's understanding is consistent with the NRC's intent in Section VIII.C.1 of Appendix G.

A second and related issue on which NEI seeks clarification concerns the NRC's intent in Section VI.C of Appendix G, which provides that "[t]he Commission does not consider operational requirements for an applicant or licensee who references this appendix to be matters resolved within the meaning of § 52.63(a)(5)." This provision, at least on its face, appears to be in tension with Section VIII.C.1, which, as discussed above, affords backfit protection to certain NuScale design certification generic technical specifications and other operational requirements that were completely reviewed and approved in the DCR, and Section VIII.C.4, which requires an exemption for an applicant referencing the appendix to depart from the operational requirements imposed by the DCD. That is, an interpretation of the rule that would withhold issue resolution, but grant backfit protection and require exemptions for departures from unresolved matters, seems inconsistent.

4-2

NEI believes that this ostensible inconsistency is resolved by reading Section VI.C to apply to operational requirements that were not completely reviewed and approved in the NuScale DCR, and which the NRC, in its sound technical discretion, may conclude are necessary to impose on an applicant or licensee who references Appendix G in a future application. This interpretation appears to be consistent with statements in the SOC for the NuScale DCR. For example, the SOC notes that Section VI.C "reflects the fact that only some operational requirements, including portions of the generic technical specification in Chapter 16 of the DCD, were completely or comprehensively reviewed by the NRC in this design certification proposed rule proceeding."[10] The SOC further explains that:

> Also, paragraph VI.C allows the NRC to impose future operational requirements (distinct from design matters) on applicants who reference this design certification. License conditions for portions of the plant within the scope of this design certification (e.g., startup and power ascension testing) are not restricted by § 52.63. The requirement to perform these testing programs is contained in the Tier 1 information. However, ITAAC cannot be specified for these subjects because the matters to be addressed in these license conditions cannot be verified prior to fuel load and operation when the ITAAC are satisfied. In the absence of detailed design information to evaluate the need for and develop specific post-fuel load verifications for these matters, the NRC is reserving the right to impose, at the time of COL issuance, license conditions addressing post-fuel load verification activities for portions of the plant within the scope of this design certification.[11]

Thus, while Section VI.C and Section VIII.C.1 of Appendix G both refer to "operational requirements," they appear to address different circumstances and serve disparate functions, with

[10] 86 Fed. Reg. at 35008.

[11] *Id.*

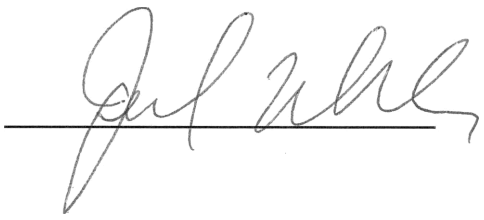
the latter provision providing backfit protection for generic operational requirements that were completely reviewed and approved in the design certification rule.

The NuScale design certification addresses many more "operational requirements" than were considered in the early design certifications. For example, important operational requirements such as minimum operator staffing and containment leakage rate testing were completely reviewed and approved in the NRC staff's FSER. Applicants expend significant resources addressing these operational requirements and, as Commissioner Caputo noted, it is reasonable to expect "future advanced reactor applicants to address additional operational requirements in design." Therefore, it is imperative that design certification and COL applicants understand the applicability of the issue resolution, finality, and change and departure provisions of a design certification rule. NEI believes that issue resolution should be afforded where the NRC staff have completed their safety review and the public has been afforded an opportunity to comment. If the NRC disagrees with NEI's above interpretation that issue resolution should be afforded in these circumstances, then it should document its conclusion in response to this letter. However, the NRC should revisit these provisions for operational requirements on a generic basis, and in a manner that does not impact the NuScale DCR schedule.

In conclusion, NEI believes the clarifications sought herein are warranted, particularly given the potential for future advanced reactor developers to address additional operational requirements (e.g., emergency planning, physical security) as part of the design process. This underscores the need for regulatory clarity and certainty with regard to how backfitting or issue finality works in this area.

Please contact me or Martin O'Neill, NEI Associate General Counsel (mjo@nei.org), if you have any questions regarding these comments.

Sincerely,



Jennifer L. Uhle

C: Dennis Andrukat, NMSS/REFS/RRPB
Carolyn Lauron, NRR/DNRL/NRLB

From:	Edwin Lyman
To:	RulemakingComments Resource
Subject:	[External_Sender] Union of Concerned Scientists comments on the NuScale Design Certification Proposed Rule, NRC-2017-0029
Date:	Thursday, October 14, 2021 11:32:08 PM
Attachments:	ucs comments nuscale proposed rule 10 14 21.pdf

Please find attached comments by the Union of Concerned Scientists on the proposed rule.

Sincerely,

Edwin Lyman
Director of Nuclear Power Safety
Union of Concerned Scientists
1825 K St, NW Ste. 800
Washington, DC 20006 USA
elyman@ucsusa.org

Union of Concerned Scientists Comments on The NuScale Design Certification Proposed Rule, NRC-2017-0029

Edwin S. Lyman, PhD
Director of Nuclear Power Safety
October 14, 2021

The Union of Concerned Scientists has identified the following serious safety issues with the NuScale design certification proposed rule:

1) Failure to consider severe accident mitigation design alternatives (SAMDAs) associated with the potential for boron redistribution/dilution transients that could lead to core damage.

In the July 6, 2020 report supporting his non-concurrence from the staff's approval of the NuScale FSER (ML20232D086), Dr. Shanlai Lu writes that "the [NuScale] reactor could reach fuel failure and prompt criticality condition for a wide range of initial conditions. The CDF could be between 0.33E-4 to 0.33E-6 without any other new design changes or analyses to justify otherwise."

Instead of heeding Dr. Lu's stark warning and taking the necessary time to fully assess his claims, NRC management dismissed and papered over his non-concurrence in order to meet a self-imposed artificial deadline for completing the FSER, In our view, this was a major mistake that undermines the integrity of the NRC's review process of the NuScale DCA and leaves unresolved serious safety questions about the NuScale design. 5-1

Specifically, the NRC's environmental assessment (EA) referenced in the draft rule improperly fails to evaluate potential SAMDAs that could reduce the risk of core damage and radiological release associated with the boron redistribution events that so concern Dr. Lu. He writes further that "as the result of Chapter 15 deficiencies, the ECCS design is incomplete. The latest NuScale design changes have improved the boron mixing prior to the ECCS actuation. However, **additional design modifications are needed** [emphasis added] for NuScale to mitigate post ECCS actuation boron dilution and demonstrate that the system capabilities to bring the system back to normal with no adverse impacts on the core cooling."

Dr. Lu points out that "once NuScale recognizes the significant deficiencies of its design to handle boron dilution with those non-safety injection system and the current ECCS post actuation settings, there are always many design options for NuScale to consider. The current NuScale innovative passive design features allows the development of a passive boron addition system as a natural extension of its passive design features." This indicates that there are SAMDAs that could potentially mitigate the risks discussed by Dr. Lu and should be evaluated in the NuScale Environmental Report (ER) and the staff's EA.

However, in NuScale's letter to the NRC on July 10, 2020 (ML20192A326), NuScale writes that "the ER is unaffected in terms of … addition or removal of postulated severe accident management design alternatives." The NRC EA asserts that the staff "verified" this conclusion

(presumably without including Dr. Lu.) However, there is no indication that the staff considered Dr. Lu's proposal for a passive boron injection system or other design modifications that could mitigate the boron dilution transients of concern in evaluating the potential need for SAMDAs. This deficiency should be corrected in the final rule by including a complete analysis of SAMDAs that could address this problem, as well as implementation of any SAMDAs that are cost-beneficial.

2) Failure to consider SAMDAs for a cask drop during refueling

A fundamental aspect of the NuScale design is the co-location of all reactor modules within a common water-filled pool. A NuScale plant will have also have much more frequent refueling activities, as each module will require refueling on a 24-month cycle. Consequently, refueling activities for one module will occur while the other modules are operating, raising the potential for a serious accident associated with refueling. The NuScale PRA has identified a cask drop during refueling as the internal initiating event with the highest frequency of core damage: on the order of $1x10^{-6}$/plant-year for a 12-module plant. Nevertheless, the NRC, despite being unable to "reach a finding" (whatever that means) on SAMDAs associated with a cask drop during refueling (Release Category 8 in the staff EA), the NRC approved the NuScale ER on the basis that any SAMDA addressing this risk would be associated only with improvements to the reactor building crane, which "is not considered part of the design certification."

5-2

This is false on a number of levels. First, given the crane has a critical function in the operation of the plant, and plays an outsized role in the plant risk, it should be considered a fundamental part of the design and thoroughly evaluated in the design certification. Second, it is highly likely that other SAMDAs could be identified to help mitigate the risk of a cask drop. Therefore, the NRC's failure to consider SAMDAs associated with RC 8 is a significant deficiency in the EA.

3) Illogical designation of the active inadvertent block valves (IABs) as passive components.

UCS strongly agrees with the NRC staff's recommendation in SECY-19-0036 and Commissioner Baran's dissenting vote to reject NuScale's assertion that the critically important IABs, which must "close rapidly and fully seal to prevent premature opening of the main ECCS valve" should be regarded as "passive" components that are not subject to the single failure criterion. The Commission's majority vote to accept NuScale's illogical contention is irresponsible, dangerous, violates common sense, and should be overturned in the final rule.

5-3

LO-108050

October 13, 2021

Dennis Andrukat, Office of Nuclear Material Safety and Safeguards
Carolyn Lauron, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

SUBJECT: NuScale Power, LLC Comments on *NuScale Small Modular Reactor Design Certification*, Docket ID NRC-2017-0029

REFERENCES: 1) *NuScale Small Modular Reactor Design Certification*, 86 Fed. Reg. 34,999, July 1, 2021.

2) *NuScale Small Modular Reactor Design Certification*, 86 Fed. Reg. 47,251, August 24, 2021.

3) Standard Design Approval for the NuScale Power Plant Based on the NuScale Standard Plant Design Certification Application, September 11, 2020 (ML20247J564).

In Reference 1, the U.S. Nuclear Regulatory Commission (NRC) issued for public comment the proposed rule to certify the NuScale standard design. In Reference 2, the NRC extended the comment period to October 14, 2021.

On behalf of NuScale Power, LLC (NuScale), I commend the NRC Staff for their efforts in reviewing the NuScale design certification application, issuing the NuScale Standard Design Approval (Reference 3), and promulgating this rule. This design certification is the culmination of NRC's comprehensive safety review, which involved over a quarter million review hours, about two million pages of documentation available for review or audit, and about 100 gigabytes of test data. NRC Staff thoroughly examined the design's innovative and robust safety features and addressed numerous regulatory, policy, and technical issues associated with the design and its licensing basis.

In an effort to clarify the final rule for NuScale, other potential users, and interested members of the public, NuScale has developed the comments attached to this letter for your consideration.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@NuScalePower.com.

Carrie Fosaaen
Acting Vice President, Regulatory Affairs

Attachment: Comments and Review of U.S Nuclear regulatory Commission Docket ID NRC 2017-0029: NuScale Small Modular Reactor Design Certification, 86 Fed. Reg. 34,999

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Comments and Review of U.S. Nuclear Regulatory Commission Docket ID NRC 2017-0029 *NuScale Small Modular Reactor Design Certification*, 86 Fed. Reg. 34,999			
No.	**Page/Section/Paragraph**	**Comment**	**Proposed Resolution**
1	Statements of consideration § III.C.3; proposed 10 CFR 52 Appendix G § IV.A.2.i	The proposed rule states that the requirements of 10 CFR Part 20 have not been demonstrated with respect to steam generator tube integrity. The radiation protection standards of Part 20 pertain to doses to plant workers and members of the public as a result of expected plant operations. Failure of steam generator tubes is an accident condition, as noted in the statements of consideration ("The failure of multiple steam generator tubes resulting from failure of an inlet flow restrictor has not been included within the scope of the NuScale accident analyses in DCA Part 2, Tier 2, Chapter 15.").	Delete references to 10 CFR Part 20 requirements with respect to steam generator integrity.
2	Statements of consideration § IV.F	This discussion identifies FSER sections 12.2, 12.3, 3.11 and 15.0.3 as discussing TR-0915-17565. Section 15.0.2 also discusses that report.	Add FSER section 15.0.2 to list of sections that discuss TR-0915-17565.
3	Statements of consideration § IV.A	This discussion identifies FSER Chapter 3 as "Design of Structures, Components, Equipment, and Systems." The title of FSER Chapter 3 is "Design of Structures, Systems, Components, and Equipment."	Correct the title of FSER Chapter 3.
4	Statements of consideration § IV.A	The discussion states "With the exception of the steam generator tube and inlet flow restrictor issue discussed previously…" Identifying that previous discussion as Section III.C.3 would increase clarity.	Replace "previously" with "in Section III.C.3."
5	Statements of consideration § III.C.2	This discussion states the combustible gas monitoring leakage issue "may be resolved by performing radiation dose calculations and demonstrating that doses would remain within applicable dose limits in 10 CFR part 20…" As the preceding sentence notes, "this issue does not affect normal plant operation or non-core damage accidents." The dose limits of 10 CFR Part 20 apply to normal plant operations. FSER Section 12.3.4.1.3 invokes the control room habitability assessment of 10 CFR 50.34(f)(2)(xxviii) and "important area" access requirement of 10 CFR	Delete reference to 10 CFR part 20 as an applicable requirement for a COLA applicant to resolve the combustible gas monitoring leakage issue.

6-1
6-2
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			50.34(f)(2)(vii) as relevant to the potential onsite doses associated with this core damage accident-related release; it cites the accident dose limits of 10 CFR 52.47(a)(2)(iv) as applicable to offsite doses.	
6-6	6	Statements of consideration § IV.C	The discussion states that the inadvertent actuation block valve is "safety-significant." In this context "safety significant" is an undefined term and creates ambiguity. NuScale has not undertaken risk-informed categorization of SSCs pursuant to 10 CFR 50.69, which categorizes SSCs for their safety significance. Risk insights indicate the IAB is not risk significant.	Delete phrase "safety-significant" from statements of consideration.
6-7	7	Statements of consideration § IV.H; proposed 10 CFR 52 Appendix G § V	The statements of consideration and proposed rule do not address the inapplicability of 10 CFR 50 Appendix J to a licensee referencing the NuScale design. NuScale DCA Part 7, Section 7, sought an exemption from GDC 52 for the NuScale design and an exemption from Appendix J Type A testing for licensees referencing the NuScale design (similar to the control room staffing requirements of 10 CFR 50.54(m), the Appendix J testing requirements are applicable to a licensee and not a design certification applicant). Staff's FSER Section 6.2.6.4 approved both requests. Neither the statements of consideration nor the "applicable regulations" portion of the proposed rule discuss the exemption for licensees from the Type A testing requirements of Appendix J. Also see comment 14 concerning the exemption to 10 CFR 50.54(m), which is applicable for the Appendix J exemption.	Add Appendix J Type A testing to the list of exemptions granted by the final rule.
6-8	8	Statements of consideration § V.B	The discussion states that for the plant-specific DCD "a COL applicant may also have to include considerations for multi-module facilities in the plant-specific DCD that were not previously evaluated as part of the design certification rule." It is unclear what the NRC intends by this statement. The NuScale FSAR is based on a 12-module plant, so multi-module aspects have been addressed for the to-be-certified design.	Clarify or delete statement that a COL applicant may need to address additional multi-module considerations in the plant-specific DCD.

6-9	9	Statements of consideration §§ V.B, V.F, V.H	The statements of consideration in several locations state that the generic technical specifications for the NuScale design are in Chapter 16 of the generic DCD. Chapter 16 of the NuScale FSAR describes the process for developing the technical specifications, but the generic technical specifications are found in Part 4 of the DCA.	Correct references to Chapter 16 of the DCD to instead refer to DCA Part 4.
6-10	10	Statements of consideration §XI	The discussion states that "site-specific SAMDAs, multi-unit aspects, procedural and training SAMDAs, and the reactor building crane design would need to be assessed when a specific site is proposed for constructing and operating a NuScale power plant." The term "multi-unit" in the context of a multi-module reactor design is ambiguous, as each reactor module could be considered a unit. The environmental assessment considered multi-module aspects; it appears this phrase was meant to instead refer to multi-plant aspects (i.e., more than one 12-module facility at a site).	Replace "multi-unit" with "multi-plant."
6-11	11	Proposed 10 CFR 52 Appendix G § II.A	The Generic DCD is defined as "the document containing" Tier 1 information, Tier 2 information, and generic technical specifications. This definition may cause confusion because the NuScale DCA does not include a discrete document containing that information; the generic technical specifications are in Part 4 of the DCA.	Revise so that the final definition reads "...means the Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1) and generic technical specifications that are incorporated by reference into this appendix."
6-12	12	Proposed 10 CFR 52 Appendix G § II.C	The plant-specific DCD definition is defined to include "plant-specific changes to generic DCD information." Under design certification rule nomenclature, "changes" are generic while "departures" are plant-specific.	Replace "changes" with "departures."
6-13	13	Proposed 10 CFR 52 Appendix G § IV.A.2.g	This rule provision would require the COLA to include shielding design information to meet the radiation zones specified in DCA Part 2, Tier 2, Figure 12.3-1. This requirement effectively controls that Tier 2 radiation zone map equivalently to Tier 1 information, because a COLA applicant would have no ability to depart from the radiation zone map without	Revise this provision to refer to the plant-specific DCD radiation zone map instead of the DCA radiation zone map.

		first getting an exemption from this requirement. In other words, if a COLA applicant were to depart from the radiation zone map in a manner otherwise acceptable under the Tier 2 departure provisions (because it meets the 50.59-like criteria), the applicant would still need an exemption from this provision because they would not provide shielding satisfying the generic DCD's radiation zone map. This is an unnecessary new control on Tier 2 information. The regulatory history of Tier 1, standardization, and the change control provisions does not support an exemption requirement for this radiation zone map. For example, SECY-92-287 states that Tier 1 information includes "important design information that was relied upon as the fundamental bases for the staff's safety review, such as the key assumptions in the safety analyses and in the bases for the technical specifications." This radiation zone map, while supporting the operational dose limits and equipment qualification, does not rise to the level of a fundamental basis for the Staff's review and is not essential to standardization of the plant design, and thus does not justify an exemption requirement for departures from it. The COLA applicant can adequately address NRC's expectation to address shielding of major penetrations by providing the shielding details necessary to meet the radiation zones specified in their plant-specific DCD; the applicant then maintains the ability to depart from the generic DCD radiation zone maps to the same extent they otherwise would be able to if the shielding details were provided in the generic DCD.	
14	Proposed 10 CFR 52 Appendix G § V	The 10 CFR 50.54(m) exemption is listed amongst exemptions for "the NuScale design." This exemption is not applicable to the design, but rather to a licensee referencing the design certification. Listing this exemption with the design-related exemptions may cause confusion. Separately identifying this exemption for licensees referencing the NuScale DC, for example in a new section V.C, would make the final rule clearer. The exemption from 10 CFR Appendix J for licensees	Consider clarifying the 10 CFR 50.54(m) exemption for licensees by addressing separately from the design exemptions. Consider clarifying discussion of this and

6-13

6-14

			referencing the NuScale DC would also be listed there (see previous comment 7). The corresponding discussion on "applicable regulations" (statements of consideration Section V.D) may also warrant a brief discussion of this new approach to exemptions from 10 CFR 50.54(m) and Appendix J for licensees referencing the NuScale design certification.	the Appendix J exemption in the statements of consideration.
6-15	15	Proposed 10 CFR 52 Appendix G §VI.B.1.d	This provision states that GDC 10 is applicable to the steam generator integrity issue, implying that the COLA must demonstrate conformance with GDC 10 to resolve Staff's concerns. Two other provisions of the proposed rule addressing steam generator integrity do not cite GDC 10. As GDC 10 concerns the reactor design it is not relevant to steam generator integrity and is not cited by the FSER in this respect.	Delete references to GDC 10 with respect to steam generator integrity.
6-16	16	Proposed 10 CFR 52 Appendix G § VI.B	The list of matters resolved does not include referenced information in public documents. Nuclear safety and safeguards issues associated with referenced information intended as requirements in nonpublic reports are explicitly resolved, but not safety issues in public reports. Several of the reports referenced in the generic DCD are exclusively public reports, with no equivalent nonpublic report that would be within scope of issue resolution. While issue resolution for the FSER, Tier 2, and the rulemaking record implies resolution of referenced public reports, 10 CFR Part 52 Appendix E (ESBWR DC) includes the 20 documents approved for incorporation by reference by the Director of the Office of the Federal Register (i.e., the public documents) within the scope of Issue Resolution paragraph B.1. A clearer approach for the NuScale DC may be to revise paragraph B.2 or include a new paragraph.	Revise issue resolution provisions to include nuclear safety issues associated with referenced information in public documents which, in context, are intended as requirements in the generic DCD for the NuScale design.
6-17	17	Proposed 10 CFR 52 Appendix G § VII	The proposed rule provides a 15 year duration "from October 29, 2021." Other proposed DC rules (aside from the direct final rule approach for the APR1400) have included a placeholder for the final rule effective date; NuScale wants to call attention to ensure that the final rule includes the correct duration start date.	Revise the duration provision to begin with the effective date of the final rule.

6-18

18	Proposed 10 CFR 52 Appendix G § VIII.B.5.g	The proposed rule states that in making a contention on compliance with the Tier 2 departure provisions, the intervenor "must demonstrate that the change stands on an asserted noncompliance with an ITAAC acceptance criterion…" It is unclear what it means for a change to "stand on" an asserted ITAAC noncompliance. Previous DC rules have used the term "bears on," which appears correct in this context. However, replacing "change" with "departure" would enhance clarity.	Revise provision to state "Further, the petition must demonstrate that the departure bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a § 52.103 preoperational hearing…"

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Comment on FR Doc # 2021-13940

Submitter Information

Email: gwaites@odonoghuelaw.com
Organization: United Association of Plumbers and Pipe Fitters and the Mechanical Contractors Association of America (joint submission)

General Comment

Please find attached joint comments for this proceeding on behalf of the United Association of Plumbers and Pipe Fitters and the Mechanical Contractors of America. If you have any questions regarding this submission, please contact me. Gerard Waites, O'Donoghue & O'Donoghue, LLP 301-523-6599. Thank you for your attention in this matter.

Attachments

UA MCAA Comments on NuScale NRC Rule 10-14-2021-FINAL

October 13, 2021

Submitted to: Rulemaking.Comments@nrc.gov
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

Re: Docket ID NRC–2017–0029

Dear Sir or Madam:

The following joint comments are being submitted in support of the requested certification in the above-referenced proceedings on behalf of the United Association of Plumbers and Pipefitters (UA) and the Mechanical Contractors Association of America (MCAA). The UA represents 360,000 highly skilled workers in the piping industry, a substantial number of which are employed in the nuclear industry. The UA's partner, the MCAA is the leading trade association for mechanical contractors in the country and represents over 2,600 high quality companies serving all major construction and maintenance markets, including those in the nuclear industry.

In this proceeding, the U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to certify the design for the NuScale standard small modular reactor (SMR) developed by NuScale Power, LCC. Strongly supported by the U.S. Department of Energy (DOE) as an essential step in developing the next generation of nuclear reactors, the NuScale reactor will be the first SMR to receive NRC certification and is vital for meeting our nation's increasingly critical clean energy needs. As the DOE has stressed:

> "The NRC's [previously issued] final safety evaluation report on NuScale Power's small modular reactor (SMR) design. This accomplishment is the first of its kind for a SMR and puts NuScale on track to receive a full design certification . . . . *The milestone is the direct result of more than $400 million in funding by . . . [DOE] since 2014 to accelerate the development and deployment of SMRs . . . . DOE is proud to support the licensing and development of NuScale's Power Module and other SMR technologies that have the potential to bring clean and reliable power to areas never thought possible* by nuclear reactors in the U.S., and soon the world.[1]

7-1 As with other advanced reactors, the NuScale SMR is designed to provide a safer, more cost-effective clean option for meeting future energy needs and is particularly well suited to replacing aging U.S. coal plants. Moreover, "[t]hese advanced reactors, envisioned to vary in size from tens of megawatts up to hundreds of megawatts, *can be used for power generation, process heat, desalination, or other industrial uses*." *Id*. (emphasis added.)

[1] Advanced Small Modular Reactors, *NRC Approves First U.S. Small Modular Reactor Design*, DOE, Office of Nuclear Energy (last visited 10/12/21), https://www.energy.gov/ne/articles/nrc-approves-first-us-small-modular-reactor-design (emphasis added).

There is little dispute that the ever-increasing U.S. demand for energy must be met by new clean sources. Nuclear power, especially in the form of SMRs and other advanced reactor systems, presents one of the most viable alternatives for providing safe, reliable, affordable power for electricity as well as other energy needs. While major strides have been made in developing renewable energy, these sources have substantial limitations in meeting the demand for clean power. All renewable energy sources combined provide only *12 percent* of our nation's energy, according to the latest federal data.[2] Moreover, wind and solar sources, the two renewable alternatives receiving the most attention and support in recent years, still have significant reliability limitations due to their intermittent nature.

For these reasons, nuclear power is essential to meeting the challenge of maximizing clean energy production in the shortest time-frame possible. The nation's leading energy authority, DOE, recognizes these facts as it views nuclear power as a major component in the sensible "*All the Above*" strategy it advocates for meeting future energy needs.[3] As noted, it has also specifically embraced NuScale SMRs. Another compelling point that should be considered when assessing the value and benefits of nuclear power generally and NuScale's advanced system specifically is *JOBs*! While the economic impact and jobs factor is not an issue typically reviewed in the instant proceeding, we submit it is important in considering the overall impact and benefits from the NuScale technology.

The reality is that the current wholescale transformation of the U.S. energy industry to clean sources could potentially result in the loss of millions of jobs. These are critically needed *good* middle-class jobs that have been steadily declining in our country. New nuclear facilities, including NuScale, require large industrial processes. Thus, unlike many renewable sources, which create relatively few jobs, these sources generate substantial employment and other major economic benefits in affected communities. These include both construction and operation jobs critical to providing just transition opportunities to workers losing lifelong stable employment from the clean energy revolution.

7-1 For these reasons, we support NRC approval of proposed certification in the instant proceeding. If you have any questions or should need further information, please let us know.

Respectfully submitted,

Mark McManus
UA General President

Armand Kilijian
MCAA President

[2] U.S. Energy Information Administration, Monthly Energy Review, Table 1.3 and 10.1, April 2021, preliminary data; (last visited 10/12/21), https://www.eia.gov/energyexplained/us-energy-facts

[3] DOE, Office of Energy Efficiency and Renewable Energy (last visited 10/12/21), https://www.energy.gov/science-innovation/clean-energy

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Submitter Information

Email: adam@thebreakthrough.org
Organization: The Breakthrough Institute

General Comment

See attached file(s)

Attachments

Comment on Docket NRC-2017-0029-BTI

October 14, 2021

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Comment on NuScale SMR Design Certification Rulemaking

(Docket-NRC-2017-0029)

This letter and its enclosure provide the perspective of the Breakthrough Institute on the NuScale SMR Design Certification Rulemaking.

The Breakthrough Institute is an independent 501(c)(3) global research center that identifies and promotes technological solutions to environmental and human development challenges. We advocate appropriate regulation and licensing of advanced nuclear reactors to enable the commercialization of innovative and economically viable emerging nuclear technologies, which we believe to represent critical pathways to climate mitigation and deep decarbonization. The Breakthrough Institute does not receive funding from industry.

Sincerely,

Adam Stein, Ph.D.
The Breakthrough Institute

Sola Talabi, Ph.D.
Pittsburgh Technical

Enclosure: Attachment A - Comment on NuScale SMR Design Certification Rulemaking

Attachment A:

Comment on NuScale SMR Design Certification Rulemaking

Preface

This report provides a summary of the review of the NuScale Small Modular Reactor (SMR) and Design Certification Application (DCA) and associated Nuclear Regulatory Commission (NRC) review process. It also provides a review of the proposed rulemaking, which will allow licensees seeking to build the NuScale SMR to do so, by referencing the rule. The report provides the required information to enable an adequate understanding of NuScale as a company and the SMR design, in particular. It also provides an overview of the regulatory review process and associated documents. It further provides a set of recommendations for reactor vendors, regulators and policymakers.

1. Background and Ownership of NuScale

NuScale Power is a privately owned nuclear reactor design vendor, which was formed in 2007. Fluor Corporation is the majority and strategic investor in NuScale, while minority investors include Sargent and Lundy, ARES corporation, Oregon State University, ENERCON, DOOSAN Heavy Industries and ARES Corporation and ULTRA Electronics are minority investors. NuScale has locations in the United States and United Kingdom, with over 400 employees.

1.1 NuScale Technology Development

As of the date of this report, NuScale has currently designed a Small Modular Reactor (SMR) plant and is working on the design of a microreactor. [1] The NuScale SMR plant is constituted of several power modules, with each module producing up to 77Mwe. It should be note that at the time of submission of the DCA, the NuScale power module output was specified as 50Mwe, but has since been uprated by NuScale to 77Mwe. Hence, the total SMR plant size depends on the number of modules in the plant. The plant power can be scaled up to match demand by including more modules. The proposed plant sizes range from 4 modules producing 308Mwe, up to 12 modules producing 924Mwe. The NuScale SMR received Design Approval in August, 2020, making it the first SMR to receive NRC design approval.

1.2 NuScale Small Modular Reactor History

NuScale Power was initially formed in 2007 for the purpose of commercializing the basic SMR technology, which had been developed at Oregon State University through the US Department of Energy funded Multi-Application Small Light Water Reactor (MASLWR) program. Fluor Corporation acquired a lead investment position in 2011 and NuScale won a competitively awarded US DOE grant of approximately $300 million. The total public and private investment in NuScale technology as of the time of this report is over $1billion. The design development efforts have resulted in several peer reviewed academic studies and over 530 worldwide patents.

1.3 Overview of the NuScale SMR Design

The NuScale SMR is constituted of multiple NuScale Power Modules (NPM). The NPM is an integral package that includes the reactor vessel, steam generators, pressurizer, and containment vessel. The reactor within the NPM is 65 feet in height and 9 feet in diameter, and sits in a containment vessel measuring 76 feet in

[1]It should be noted that the NuScale design had been completed to an extent that it could be reviewed for regulatory approval, which is a level of design completion that is focused on safety functionality and not on performance or design optimization. As an example, the Westinghouse AP1000 design was submitted for NRC review in 2002, and the initial design certification was received in 2004, however the AP1000 design effort continued until 2014. Hence, it should be noted that the NuScale SMR design may still be undergoing design modifications to incorporate additional safety features beyond standards required by the regulator, and other changes for improved performance.

height x 15 feet in diameter. The reactor and containment vessel operate inside a water-filled pool that is built below grade. The reactor operates using the principles of buoyancy driven natural circulation; hence, no reactor coolant pumps are needed to circulate water through the reactor, as in LLWRs. Water passes over the core, gets heated and rises through a riser located at the center of the vessel. The heated water exits the at the top of the riser and is drawn back downward by water that is cooled passing through the steam generators. The water is then pulled by gravity back down to the bottom of the reactor where it again flows over the core. Water in the reactor system is kept separate from the water in the steam generator system to prevent contamination. As the hot water passes over the hundreds of tubes in the steam generator, heat is transferred through the tube walls and the water inside the tubes turns to superheated steam. The steam is channeled to a small turbine that is attached by a single shaft to an electrical generator. After passing through the turbine, the steam loses its energy and is cooled back into liquid form in the condenser and pumped by the feed water pump back to the steam generator where it begins the cycle again.

The NPM is a simplified version of existing Large Light Water Reactors (LLWR), in that it eliminates reactor coolant pumps, large pipes and other major components found in LLWRs. These components have been eliminated by demonstrating that safety objectives are achieved without their inclusion. Specifically, certain active safety systems such as reactor coolant pumps and containment sprays have been replaced by passive safety systems such as natural convective cooling and natural aerosol deposition phenomena respectively. These passive systems are enabled by thermal hydraulic conditions enhanced by geometric configurations and placement of the NPM in a pool of water to serve as an ultimate heat sink. The geometric configuration is such that the NPM reactor core is 1/20th the size of LLWR cores. The integrated design allows the elimination of certain postulated accidents such as large pipe breaks, as the need for piping between major components is eliminated.

The NuScale power module design has multiple features that differ from current large light-water reactors. Due to these differences the NuScale power module design was granted 17 exemptions to certain requirements under 10 CFR 50. Some of the most significant features that diverge from typical light-water reactors are:

1.3.1 **No AC or DC Power for Safe Shutdown and Cooling:** Current LLWRs require AC or DC power for safe shutdown. The NPM however does not require

AC or DC power, and no operator or computer aided action, it only requires the existing inventory of water within the pool it is submerged in.

1.3.2 **Helical Coil Steam Generators (HCSG):** The NPM is able to achieve compactness by innovative geometric configurations such as the helical steam generator tubes, which allow for a large heat transfer surface area in a small volume. Given the circulation force is natural convective flow, the HCSG provides a relatively low pressure drop, which does not impede the flow. The once-through counter-flow design enables the generation of superheated steam and good thermal efficiency using natural circulation flow.

1.3.3 **High Strength Steel Containment Immersed in a Cooling Pool:** The containment vessel serves as a heat exchanger and transfers reactor heat to the pool water, which limits the containment pressure, which eliminates the need for containment spray systems for cooling.

1.3.4 **Maintaining containment in a vacuum limits heat exchange during normal operation:** This minimizes reactor vessel heat loss, limits oxygen content, and prevents component corrosion, eliminating the requirement for physical reactor vessel insulation and hydrogen recombiners.

1.3.5 **Small and Efficient Core Design Limits Source Term:** The NPM has 1/20 of the nuclear fuel of a large-scale reactor. Its small decay heat, inherent stability, and reactor physics eliminates fuel damage in all postulated design basis events, including those with failure of all control rods to insert. For postulated beyond design basis events, radionuclide particle transport is limited due to the lower starting inventory and geometric and thermal hydraulic conditions that assist with deposition of the radionuclide particles. Hence, radiation from fuel damage is well below regulatory limits at the plant site boundary.

1.3.6 **Digital Instrumentation & Control (I&C):** A field programmable gate array digital I&C system provides comprehensive monitoring and control of all plant systems in a single control room.

2. NuScale SMR Design Certification Application Summary

This section summarizes the major elements of the NuScale SMR DCA, which include the schedule, contents, and major NRC communication documents.

2.1 Schedule

The following major activities and dates were associated with the Nuclear Regulatory Commission (NRC) review of the NuScale DCA:

a. Acceptance Review

i. Issue acknowledgement letter to applicant - 01/18/17
ii. Publish Federal Register Notice of receipt of application - 02/15/17
iii. Issue acceptance letter to applicant - 03/23/17
iv. Publish Federal Register Notice of docketing of application - 03/20/17

b. Safety Review

i. Phase 1 – Preliminary Safety Evaluation Report (SER) and Requests for Additional Information - 04/16/18
ii. Phase 2 – SER with Open Items - 07/12/19
iii. Phase 3 - ACRS Review of SER with Open Items - 07/12/19
iv. Phase 4 – Advanced SER with No Open Items - 12/12/19
v. Phase 5 – ACRS Review of Advanced SER with No Open Items - 07/31/20
vi. Phase 6 – Final SER with No Open Items - 08/28/20

2.2 Major Documents and Contents of the NuScale DCA

The NuScale DCA contained over 12,000 pages and was submitted in the following 10 parts listed in *Table I.* The DCA also included 14 associated topical reports. This list is provided for the reader's reference, to allow further review of specific topics as desired. The documents are publicly available and may be accessed from the NRC's website.

Table I: Major Documents and Contents of the NuScale DCA:

Part	Title
1	General and Financial Information
2	Final Safety Analysis Report
	Certified Design Descriptions and Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)
	Introduction and General Description of the Plant
	Site Characteristics and Site Parameters
	Design of Structures, Systems, Components and Equipment
	Reactor
	Reactor Coolant System and Connecting Systems
	Engineered Safety Features
	Instrumentation and Controls
	Electric Power
	Auxiliary Systems
	Steam and Power Conversion System
	Radioactive Waste Management
	Radiation Protection
	Conduct of Operations
	Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria
	Transient and Accident Analyses
	Technical Specifications
	Quality Assurance and Reliability Assurance
	Human Factors Engineering
	Probabilistic Risk Assessment and Severe Accident Evaluation
	Mitigation of Beyond-Design-Basis Events
	Multi-Module Design Considerations
3	Applicants Environmental Report - Standard Design Certification
4	Generic Technical Specifications
5	Emergency Plans
6	Security Plans

Part	Title
7	Exemptions
8	License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC)
9	Withheld Information
10	Quality Assurance Program Description

2.3 Major NRC Review and Response Documents Related to the NuScale DCA

The NRC periodically provided several documents in support of the NuScale DCA review, which are provided in. This list is provided for the reader's reference, to allow further review of specific topics as desired. The documents are publicly available and may be accessed from the NRC's website.

Table II: NRC Review Documents

Date	Description
05/22/17	Review Schedule for the NuScale Power, LLC, Standard Design Certification of a Small Modular Reactor
03/23/17	NuScale Power, LLC - Acceptance of an Application for Standard Design Certification of a Small Modular Reactor
03/20/17	Federal Register Notice on Acceptance of NuScale Power, LLC Application for Standard Design Certification of a Small Modular Reactor (82 FR 15717 (FR DOC # 2017-06309); March 30, 2017)
02/15/17	Federal Register Notice on Availability of the NuScale Design Certification Application (82 FR 11372 (FR DOC # 2017-03438); February 22, 2017)
01/18/17	Acknowledgment of Receipt of the NuScale Design Certification Application

3. Comments and Recommendations Based on a Review of the NuScale DCA Proposed Rulemaking

Comment 1: Acknowledgement of Rigorous and Complete Review by the NRC

The NRC and NuScale should be commended for performing an extensive and rigorous review of the NuScale DCA. The review process was clear and well communicated in a manner that provides a high level of public confidence.

8-1

Recommendation 1: Document Lessons Learned from the NuScale DCA for Posterity and Knowledge Sharing

Considering the first-of-a-kind nature of the NuScale design, and certain aspects of the DCA submission, several lessons were learned. These lessons should be documented and disseminated for general knowledge and improvement of future DCA submissions. It will also assist COL applicants to proceed more effectively with their applications.

Comment 2: Prescriptive Nature of Current Regulations Required NuScale to Seek Exemptions and Retains Regulatory Uncertainty for COL Applicants.

Current regulations are written in a specific, prescriptive manner, which is based on large light water reactor operational experience and incidents. As an example, 10CFR part 50 "Domestic Licensing of Production and Utilization Facilities", provides control room staffing requirements based on a set of assumptions applicable to LLWRs. This will require a COL applicant that seeks to deploy the NuScale reactor to seek exemptions if they wish to use the number of operators recommended by NuScale.

8-2

The prescriptive nature of the regulations also required NuScale to seek exemptions from a standard, rather than simply describing how safety objectives are met by the NuScale design. NuScale was required to apply for a total of 17 exemptions, which were granted by the NRC based on technical justifications provided by NuScale.

Recommendation 2: Allow the Implementation of the Proposed Risk-Informed Technology Inclusive Regulatory Framework Approach for COL Applicants Referencing the NuScale DCA.

The current proposed rulemaking for a Risk-Informed Technology Inclusive regulatory framework includes elements that would improve regulatory certainty for COL applicants. Specifically, the framework includes provisions for performance-based demonstrations that would enhance the ability of COL applicants to demonstrate that safety objectives have been met, without seeking exemptions.

Comment 3: Unresolved Technical Issues Present a Regulatory Risk to COL Applicants

The NRC identified three issues that were unresolved open items in the DCA. These items are the Shielding Wall Design, Containment Leakage from the combustible gas monitoring system and steam generator stability during density wave oscillations. These unresolved issues create regulatory uncertainty for COL applicants.

8-3

Recommendation 3: Genericize the Outstanding Issues to Allow Effective Resolution

Considering the DCA has been approved with these items being outstanding, and with the NRC identifying the COL applicants as potentially being responsible for dispositioning the items, the nuclear industry should consider genericizing the issues. This would allow the issues to be addressed through various mechanisms and allow the research community to assist in retiring these technical issues. There is also a provision for COL applicants to include in the plant-specific DCA, multi-module considerations that were not included in the NuScale DCA. The NRC should clarify what the potential outstanding multi-module considerations are and provide guidance on how they may be resolved. The industry can then genericize the issues and allow them to be dispositioned by the research community.

4. Conclusion

The Breakthrough Institute concludes that approval of the NuScale DCA by the NRC is appropriate. This is based on our independent review of the contents of the DCA and the rigorous review processes undertaken by the NRC and supported by NuScale.

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NuScale Small Modular Reactor Design Certification

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Comment on FR Doc # 2021-13940

Submitter Information

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General Comment

The Federal Register Notice, on page 35,001, summarizes NRC's position that "there was insufficient information available regarding NuScale combustible gas monitoring system and the potential for leakage from this system outside containment." NRC was "unable to determine whether this leakage could impact analyses performed to assess main control room dose consequences, offsite dose consequences to members of the public, and whether this system can be safely re-isolated..."

I've read the Staff's safety evaluation and NuScale's take on this issue and I believe the NRC conclusions are mistaken. This issue comes down to leakage from a system (combustible gas monitoring) that is provided under 10 CFR 50.44 for the express purpose of monitoring combustible gases in a beyond design basis core damage event. This type of beyond design basis events are not required to meet the offsite dose criteria of 10 CFR 52.47(a)(2)(iv). No design basis event in the NuScale design damages fuel cladding, let alone severe core damage! NRC seems to be mixing the design basis offsite dose requirement--which includes a *hyothetical* major fission product release inside containment postulated only for that purpose—with the functional requirement for combustible gas monitoring under real (although extremely unlikely) core damage scenarios, which are beyond design basis.

9-1

NRC Staff also try to point to TMI rules for their position for worker doses. These rules don't seem to support the Staff's position. The TMI rules do address beyond design basis accidents, but they do so by requiring additional functions to help mitigate those events, not by imposing dose limits. Nuscale addressed these rules in their "Lessons-Learned from the Design Certification Review of the NuScale Power, LLC Small Modular Reactor" (ML21050A431). As they note there, 50.34(f)(xxvi) does not apply a dose limit for leakage control, just that leakage be as low as practical. 50.34(f)(2)(vii) is explicit (in NUREG-0737) that it does not adress leakage from systems outside containment, because those systems already have leakage "as low as practical" under (xxvi). Lastly, 50.34(f)(2)(xxviii) doesn't require the control room habitability to address new beyond design basis events; it just required licensees to re-verify

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their control rooms for the Chapter 15 events.

All in all, NRC seems to be combining the combustible gas monitoring requirement with other unrelated rules to yield a result that, for the first time, applies dose criteria to beyond design basis events. This is akin to requiring a plant to analyze doses for a station blackout or ATWS event. Therefore, I do not think NRC's "issue not resolved" position is correct and this would set a bad precedent for future applicants. I agree with NuScale's position in their "Lessons Learned" letter and believe the final rule should consider this issue fully resolved.

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