UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 29, 2020

The Honorable Kristine L. Svinicki
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE NUSCALE SMALL MODULAR REACTOR

Dear Chairman Svinicki:

During the 676th meeting of the Advisory Committee on Reactor Safeguards (ACRS), July 21-24, 2020, we completed our review of the NRC staff's advanced Safety Evaluation Report (SER) with no open items for the NuScale Power, LLC (NuScale or applicant), design certification application (DCA) and standard design approval (SDA) application for its small modular reactor. This letter report fulfills the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Sections 52.53 and 52.141 that "the ACRS shall report on those portions of the application which concern safety." During our review, we had the benefit of interactions with representatives of the NRC staff and the applicant. We also had the benefit of the documents referenced. Appendix I lists the chronology of NuScale Subcommittee and Full Committee meetings and their subjects. Appendix II lists in chronological order ACRS letter reports on DCA chapters, areas of focus, and topical reports issued by the Committee in the course of its NuScale review. Appendix III contains the list of our memoranda on advanced SER chapter reviews approved by the Committee.

CONCLUSIONS AND RECOMMENDATIONS

1. The NuScale small modular reactor is a natural-circulation, pressurized water reactor that incorporates unique design and passive safety features, providing enhanced margins of safety. There is reasonable assurance that it can be constructed and operated without undue risk to the health and safety of the public.

2. The NRC staff's final SER for the NuScale design should be issued.

3. A design certification and standard design approval for the NuScale applications should be issued, subject to the staff's proposed exclusions regarding the finality of design requirements: shield wall design, containment leakage from combustible gas monitoring, and steam generator tube structural and leakage integrity.

4. We identify in this letter report several potentially risk-significant items that are not completed at this time. We request the opportunity to review the qualification of emergency core cooling system (ECCS) valve performance, the identification of a successful recovery

strategy to prevent potential reactivity insertion accidents associated with boron dilution sequences, and the updated probabilistic risk assessment (PRA).

BACKGROUND

The NuScale design consists of up to 12 NuScale Power Modules (NPMs) in a single reactor building. The NPMs are largely immersed in a large pool of borated water, also serving as the ultimate heat sink. Each NPM is a small, integrated, natural-circulation pressurized water reactor (PWR) composed of a shrouded reactor core and riser, a pressurizer, and two helical-tube steam generators within a reactor pressure vessel, and housed integral to a high-strength, steel containment vessel that closely surrounds the reactor vessel. The design eliminates large-diameter piping to connect the steam generators and pressurizer to the reactor vessel. The reactor core uses approximately half-height, commercial fuel and control rod assemblies and is cooled by natural circulation of the primary coolant. Nominal operating conditions and fuel burnup are well bounded by those of the current operating fleet. Each NPM is rated at 160 MWt, with an output of approximately 50 MWe.

NuScale Application

By letter dated December 31, 2016, NuScale filed its application with the NRC for certification of the NuScale standard plant design. The applicant submitted Revisions 1, 2, 3, and 4 of the DCA on March 15, 2018, October 30, 2018, August 22, 2019, and January 16, 2020, respectively. In addition, the applicant submitted errata to Revision 4 of the DCA on April 1, 2020, and May 20, 2020, followed by Revision 4.1 on June 19, 2020. On July 13, 2020, the applicant submitted a request for an SDA based on the NuScale DCA. The staff has confirmed that the application contains the design information that Subpart E, “Standard Design Approvals,” of 10 CFR Part 52, requires for a standard plant design; therefore, the staff finds the applicant’s request for an SDA acceptable.

ACRS Review Approach

We conducted our initial review of the staff’s SER with open items on a traditional chapter-by-chapter basis, along with applicable topical reports (TRs), and issued letter reports accordingly. To complete our review in an expeditious manner we implemented a two-step process. Individual members conducted a detailed technical review of each chapter of the advanced SER with no open items for deliberation and approval by the Committee as a whole. We also conducted focus area reviews, in cooperation with the staff, in which we concentrated our attention on potentially safety-significant issues that were cross-cutting over multiple SER and DCA chapters. These included:

- ECCS and ECCS Valve Performance;
- Helical-Tube Steam Generator Design;
- Boron Dilution and Return to Criticality;
- Source Term; and
- PRA.

This approach provided for a more complete, in-depth review of design and operational considerations that are inherently cross-cutting regarding integrated system safety performance, allowing us and the staff to examine important technical and safety issues that affect more than a single chapter in a more efficient and effective manner than the traditional chapter-by-chapter approach.

During our areas of focus reviews, NuScale amended its DCA in “Submittal of Second Updates to NuScale Power, LLC Standard Plant Design Certification Application, Revision 4,” to address safety concerns identified in our area of focus review on boron redistribution.

DISCUSSION

The following sections discuss significant results of our focus area reviews and the important topics of electrical power and human factors.

ECCS and ECCS Valve Performance

The NuScale design has addressed many of the most risk-significant scenarios that arise with large PWRs. The final protection against challenging accidents relies on design and operational simplicity and redundancy. Rather than active systems with pumps and valves, each NPM relies on five passive ECCS valves that open when appropriate signals are received or when electric power is lost. Successful natural circulation cooling of the core is provided if one-of-three reactor vent valves open and one-of-two reactor recirculation valves open. The safety performance of this passive ECCS design is highly dependent on ensuring reliable operation of these valves.

Failures of these unique hydraulically operated ECCS valves are one of the most important risk contributors identified in the NuScale PRA. The PRA results and confidence in the safety of the design depend on ensuring that these valves operate reliably. A number of operational factors could possibly degrade performance of ECCS valves over time, leading to common cause failure of the system. Results of design demonstration testing of these valves prompted design modifications. Subsequent testing demonstrated that the modifications were successful. In addition, there were concerns about the block valve sticking shut and preventing ECCS actuation. The tests also showed that the main valve spring would open the main valve after pressures in the reactor and containment vessels equilibrate. Evaluations demonstrated that no core damage would occur in these scenarios.

NuScale will perform extensive qualification testing to provide confidence in the ability of the valves to maintain their required performance after extended periods in an operational environment. Such testing will need to consider the possibility of degradation mechanisms such as deposits, precipitates, and fouling over time in the presence of boric acid in a high temperature and radiation environment. The results of this testing should be used to confirm the realism of the PRA failure model for ECCS valves, and the validity of underlying assumptions, as stated in Combined License (COL) Item 19.1-8.

Helical-Tube Steam Generator Design

The integrity and performance of the steam generators have not yet been sufficiently validated because of uncertainties associated with unstable density wave oscillations (DWO) on the steam generator secondary side. Accelerated wear of the alloy 690TT (thermally treated) steam generator tubing material is also a potential concern. NuScale has proposed a COL item and inspections, tests, analyses and acceptance criteria (ITAAC) to address steam generator DWO, but the staff has proposed that the steam generator design not receive finality in the NuScale design certification. Successful completion of these activities should address our concerns on steam generator performance by the COL stage. We note that some uncertainty will remain until an NPM is built, tested, inspected, and operated.

Our conclusion on this topic was not unanimous. Some members consider that the applicant should have completed this work prior to receiving a design certification or SDA. All members agreed that the design must be sufficiently validated, and safety and operational concerns induced by possible DWO must be understood.

<u>Boron Dilution and Return to Criticality</u>

Boron Dilution

As part of the long-term cooling evaluation, the applicant and the staff have evaluated the impact of boron redistribution between hot and cold regions in the NPM. As the coolant boils, boron tends to concentrate in the hot regions and is diluted in the cold regions where essentially-boron-free steam condenses. We are concerned especially about boron dilution in the downcomer by steam condensation from the steam generators or the vessel wall because it may provide a source of unborated coolant that can migrate to the core following a perturbation. This could lead to a rapid return to power event (reactivity insertion accident) with the possibility of core damage.

NuScale has incorporated design and setpoint changes to the NPM to mitigate the effects of boron dilution in the downcomer for design basis uncontrolled passive cooling events and loss-of-coolant accidents (LOCAs) up to the time of ECCS actuation. The applicant has demonstrated for these scenarios, through a conservative analytical approach, that the design modifications maintain the boron concentration in the downcomer above the critical boron concentration level necessary to prevent recriticality and a return to power. The staff's evaluation confirms the applicant's analyses out to 72 hours.

However, ECCS actuation events result in water levels below the new riser holes and render them ineffective; thus, coolant in the downcomer will deborate for a range of design basis accidents, including small-break LOCAs. The estimated time for the boron concentration to drop below the critical boron concentration in the downcomer for these events is within a few hours. Operator recovery actions raise the possibility of an influx of deborated water into the core, which may result in recriticality, return to power, and the potential for core damage.

Detailed operator response and recovery procedures will be developed by the COL applicant. The staff must ensure that these recovery strategies will prevent core damage with a high degree of confidence. A focused effort by the COL applicant is needed to develop recovery strategies that will lead to effective operating procedures. Given the inability to measure the distribution of boron in the NPM during these events and other instrument uncertainties (e.g., reactor pressure vessel riser water level), these strategies should have a stronger technical basis than is currently documented that demonstrates a path to successful recovery to prevent core damage. The PRA should be updated accordingly at the COL stage to appropriately reflect the risk of boron dilution events, including associated operator actions.

Return to Criticality and Power

The NPM could regain criticality after shutdown (unrelated to boron redistribution) with the worst rod stuck out at cold end-of-cycle conditions. This has required NuScale to request an exemption to 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 27, "Combined Reactivity Control Systems Capability," that allows for return to power after shutdown when several special circumstances are present. The staff has reviewed this event in detail, and we concur with their evaluation that the GDC 27 exemption for NuScale is acceptable. The risk associated with this event is negligible because:

1. It requires: (a) the control rod with the highest worth to not insert on demand and fails to insert for at least 24 to 48 hours; (b) the boron concentration to be low, representative of end of cycle conditions, which leads to the largest moderator temperature reactivity coefficient; (c) core temperature to be cold, representative of ECCS cooling through a flooded containment (or if natural circulation is reestablished with additional assumed failures like inadvertent chemical and volume control system (CVCS) actuation); and (d) the xenon concentration to have had time to decay, which requires 24 to 36 hours after shutdown.

2. The consequences of recriticality are negligible. The core returns to a power level of at most 2%, which results in low fuel temperatures and does not challenge specified acceptable fuel design limits.

3. The progression of the recriticality event is very slow (24 to 48 hours), and there are a number of operator actions that could be credited to terminate the event, including: (a) inserting the stuck rod by repairing the failure; (b) adding boron to the vessel via the CVCS; or (c) flooding the containment with highly borated pool water.

We agree that, when operated in a passive mode (i.e., without crediting operator actions), the NPM can become critical. This is not a desirable situation; however, we concur with the staff's safety evaluation and their conclusion that the GDC 27 exemption is acceptable.

Our conclusion on this topic also was not unanimous. All members agreed that return to criticality after scram is not a desirable situation, and the NuScale GDC 27 exemption should not become a precedent for future designs. Some members consider that the applicant should have strengthened safety-grade features of the design to prevent recriticality without relying on an exemption. Most members agreed that the low risk associated with these event sequences resulting in return to criticality makes it acceptable.

<u>Source Term</u>

Alternative Source Term

The NuScale DCA develops alternative source terms for both normal operation and accident conditions to evaluate siting, safety analyses, control room and technical support center habitability, and equipment qualification and survivability. This approach aligns with Commission guidance which recognizes that design-specific source terms for light-water small modular reactors may not necessarily follow all guides that currently pertain to large light-water reactors.

The exclusion area boundary (EAB) and the low-population zone (LPZ), as defined in 10 CFR Part 100, are anticipated to be close to a NuScale plant where building wake effects become important. To properly evaluate radioactive plume dispersion and dose that may result from accident releases, NuScale has modified an NRC-sponsored computer code to more accurately evaluate doses at the EAB and LPZ. The overall approach to establish source terms for the NuScale design, as outlined in the associated topical report (TR-0915-17565, “Accident Source Term Methodology,” Revision 3) with the limitations and conditions noted by the staff, is acceptable.

Post-Accident Combustible Gas Monitoring

We are concerned that, with the proposed post-accident combustible gas monitoring system design, obtaining a sample representative of the containment atmosphere will require establishing a sizeable flow through relatively large-diameter, non-safety-grade piping outside containment, increasing the probability of contamination and H_2 detonation outside containment. Because detailed design has not yet been completed, we concur with the staff position that the combustible gas monitoring system design not receive finality in the NuScale design certification as the staff is unable to evaluate worker and offsite dose implications. We expect to have the opportunity to review the final design updates submitted by COL applicants to ensure that our concerns have been addressed and are supported by analyses.

<u>Human Factors Engineering (HFE) and Conduct of Operations</u>

NuScale has gone further than required at the DCA stage, completing many of the tasks included in their HFE program. However, we have concerns that must be addressed at the COL stage. One of these issues involves recovery actions following a boron dilution event.

NuScale and the staff have decided that HFE issues pertaining to module movement will be addressed by the reactor building crane vendor. The HFE program and the PRA, and their reviews, must be coordinated with review of the reactor building crane design features and operations in order to minimize any hazards from heavy load lifts, including module movement.

Additionally, in justification of the exclusion of errors of commission from their Chapter 18 review, the staff cites NuScale’s declaration that “infrequent operations” that could lead to safety significant errors of commission are directed by procedure, normally require a peer check, are expected to receive supervisory oversight and should be conspicuous to the operating crew. Staff should ensure that a COL applicant includes these expectations in their plant operating procedures.

<u>Electric Power</u>

The NuScale plant is designed such that it does not require onsite or offsite alternating current (AC) or direct current (DC) power to cope with design-basis events. Safety systems are not reliant on AC or DC power for activation. In their SER the staff concluded that the NuScale design conforms to the approved topical report (TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 1) on electrical systems. The NuScale plant would be the first licensed without Class 1E power.

Based on the demonstration of the conditions and limitations in the topical report and the consequent non-Class 1E classification of the onsite and offsite electric power systems, and on analysis to support the findings required by 10 CFR 50.12, the staff recommends granting the exemption requests regarding GDC 17, “Electric Power Systems,” GDC 18, “Inspection and Testing of Electric Power Systems,” and the electric power provisions of GDCs 33, “Reactor Coolant Makeup,” 34, “Residual Heat Removal,” 35, “Emergency Core Cooling,” 38, “Containment Heat Removal,” 41, “Containment Atmosphere Cleanup,” and 44, “Cooling Water.”

Probabilistic Risk Assessment

The PRA submitted in support of the NuScale DCA and SDA application is comprehensive in the scope and level of detail. The scope includes Level 1 and Level 2 PRA for internal and external initiating events, and all operating modes. The risk from seismic events was not quantified; rather, a seismic margins analysis was performed, providing confidence that the plant has margin with respect to seismic events. The PRA evaluated the risk associated with the operation of a single module. A limited evaluation of multi-module risk was also performed.

The PRA has been performed consistent with the staff guidance and was reviewed by an external expert panel. NuScale also performed a self-assessment of the PRA to evaluate conformance with industry standards. Its scope is sufficient for the discussion of risk results and insights. The level of detail provided a basis to identify design alternatives, operational vulnerabilities, and significant risk contributors.

The low calculated risk would indicate that the NuScale design meets the Commission’s Safety Goals with large margins. However, because of omissions, uncertainties, and additional concerns, discussed below, such low risk results and the completeness of the risk insights cannot be validated with confidence at this time. The most important omission in the Final Safety Analysis Report Chapter 19 is proper consideration of boron dilution events as discussed above. The current PRA model does not identify applicable boron dilution initiators, corresponding sequences, important challenges, and most importantly, the recovery actions and possible important errors of commission. After the issue was identified, the applicant performed related success criteria evaluations, supporting the conclusion that the latest design modifications have successfully mitigated boron dilution concerns, except in the cases of a few scenarios with very low frequencies.

These supporting evaluations were reviewed by the staff, discussed and approved in the Chapter 19 SER, as summarized in one of the SER conclusions: “the staff finds that the applicant has adequately addressed the impact of the boron redistribution phenomena on the DC PRA” (SER Section 19.1.4.6.4). The staff reached their conclusions based on their assessment that no core damage would occur during these events and, therefore, an update to the PRA, to include boron dilution sequences, was not required. We do not completely agree with this conclusion and recommend additional analyses be performed. The PRA should be updated at the COL stage to appropriately reflect the risk of boron dilution events, including associated operator actions.

Additional concerns we have with the PRA results and insights are related to: 1) performance uncertainties associated with the steam generator and ECCS valves, 2) the undeveloped model for the risk contributor from the reactor building crane, 3) limited PRA uncertainty and sensitivity analyses, 4) lack of information on the importance determination, and 5) risk increase to single unit operation related to multiple unit operation and buildout.

The applicant found two risk-important human actions: 1) the operator un-isolates and initiates CVCS injection into the reactor vessel, and 2) the operator un-isolates and initiates containment flooding and drain system injection into containment. These actions may also be important in boron dilution and return to criticality scenarios, and in reducing uncertainty associated with passive features. During any future COL application review, these actions and their associated systems should be reevaluated for inclusion in the reliability assurance program.

Risk insights will be better supported when the design is completed and the COL items are addressed, ITAAC items are closed, and the site and plant-specific PRA are completed before fuel load, including a human reliability analysis based on actual plant procedures and experience gained during operator training and plant simulator exercises.

SUMMARY

The NuScale small modular reactor is a natural-circulation, pressurized water reactor that incorporates unique design and passive safety features, providing enhanced margins of safety. There is reasonable assurance that it can be constructed and operated without undue risk to the health and safety of the public. The NRC staff's final SER for the NuScale design should be issued.

A design certification and standard design approval for the NuScale applications should be issued, subject to the staff's proposed exclusions regarding the finality of design requirements. We look forward to working with the staff and applicants on any future COL applications and reviewing new information on the exclusions and a number of potentially risk significant items discussed in this report.

Sincerely,

Matthew W. Sunseri Digitally signed by Matthew W. Sunseri
Date: 2020.07.29 23:02:05 -04'00'

Matthew W. Sunseri
Chairman

APPENDIX I: **Chronology of the ACRS Review of the NuScale, LLC Application for the NuScale Design Certification**

APPENDIX II: **ACRS Letter Reports on NuScale DCA Chapters, Areas of Focus and Topical Reports**

APPENDIX III: **ACRS Memoranda on Advanced SER Chapters**

REFERENCES

1. U. S. Nuclear Regulatory Commission, "NuScale Design Certification Application Advanced Safety Evaluation With No Open Items," February 11, 2020 (ML19192A011).

2. NuScale Power, "Design Certification Application, Part 2, Tier 2, 'Final Safety Analysis Report'," Revision 4, February 5, 2020 (ML20036D469).

3. U. S. Nuclear Regulatory Commission, "NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 1, 'Introduction and General Description of the Plant'," July 22, 2020 (ML20023B603).

4. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 3, ‘Design of Structures, Systems, Components and Equipment’,” July 17, 2020 (ML20178A637).

5. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 4, ‘Reactor’,” July 13, 2020 (ML20023B613).

6. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 5, ‘Reactor Coolant System and Connecting Systems’,” July 13, 2020 (ML20023B611).

7. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 6, ‘Engineered Safety Features’,” July 17, 2020 (ML20023B619).

8. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 7, ‘Instrumentation and Controls’,” July 13, 2020 (ML20023B618).

9. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 9, ‘Auxiliary Systems’,” July 13, 2020 (ML20112F156).

10. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 13, ‘Conduct of Operations’,” July 13, 2020 (ML20023B604).

11. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 15, ‘Transient and Accident Analysis’,” July 17, 2020 (ML20199M408).

12. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 16, ‘Technical Specifications’,” July 13, 2020 (ML20023B607).

13. U. S. Nuclear Regulatory Commission, “NuScale Power, LLC, Design Certification Application – Safety Evaluation With No Open Items for Chapter 19, ‘Probabilistic Risk Assessment and Severe Accident Evaluation’,” July 17, 2020 (ML20196L734).

14. NuScale Power, “Submittal of Updates to NuScale Power, LLC Standard Plant Design Certification Application,” Revision 4, April 1, 2020 (ML20092L899).

15. NuScale Power, “Submittal of Second Updates to NuScale Power, LLC Standard Plant Design Certification Application, Revision 4,” May 20, 2020 (ML20141L787).

16. NuScale Power, “Submittal of Second Updates to NuScale Power, LLC Standard Plant Design Certification Application,” Revision 4.1, June 19, 2020 (ML20198M392).

17. NuScale Power, “Submittal of Riser Flow Hole Methodology and Associated Changes to Final Safety Analysis Report Incorporating Its Use,” June 19, 2020 (ML20171A731).

18. NuScale Power, “Changes to Final Safety Analysis Report, Section 6.2, ‘Containment Systems,’ Section 6.3, ‘Emergency Core Cooling System’,” May 20, 2020 (ML20141N012).

19. NuScale Power, Licensing Technical Report TR-1116-52011, “Technical Specifications Regulatory Conformance and Development,” Revision 4, May 20, 2020 (ML20141L804).

20. NuScale Power, Licensing Technical Report TR-0516-49084, “Containment Response Analysis Methodology Technical Report,” Revision 3, May 20, 2020 (ML20141L808).

21. NuScale Power, Licensing Technical Report TR-0916-51299, “Long-Term Cooling Methodology,” Revision 3, May 20, 2020 (ML20141L816).

22. NuScale Power, Licensing Technical Report TR-0616-49121, “NuScale Instrument Setpoint Methodology Technical Report,” Revision 3, May 20, 2020 (ML20141M114).

23. NuScale Power, Licensing Technical Report TR-0316-22048, “Nuclear Steam Supply System Advanced Sensor Technical Report,” Revision 3, May 20, 2020 (ML20141M764).

24. NuScale Power, “NuScale Power, LLC Request for Standard Design Approval based on the NuScale Standard Plant Design Certification Application,” July 13, 2020 (ML20195C766).

25. NuScale Power, LLC, TR-0915-17565, “Accident Source Term Methodology,” Revision 3, April 21, 2019 (ML19112A172).

26. NuScale Power, LLC, TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 1, February 2017 (ML17048A460).

27. NuScale Power, LLC “Submittal of Standard Plant Design Certification Application,” December 30, 2016 (ML17013A229).

28. NuScale Power, LLC “Submittal of Standard Plant Design Certification Application,” Revision 1, March 15, 2018 (ML18086A090).

29. NuScale Power, LLC “Submittal of Standard Plant Design Certification Application,” Revision 2, October 30, 2018 (ML18311A006).

30. NuScale Power, LLC “Submittal of Standard Plant Design Certification Application,” Revision 3, August 22, 2019 (ML19241A315).

31. NuScale Power, LLC “Submittal of Standard Plant Design Certification Application,” Revision 4, January 16, 2020 (ML20036D336).

32. NuScale Power, LLC “Submittal of Standard Design Approval Based on the NuScale Standard Plant Design Certification Application,” July 13, 2020 (ML20195C766).

July 29, 2020

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE NUSCALE SMALL MODULAR REACTOR

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APPENDIX I

CHRONOLOGY OF THE ACRS REVIEW OF THE NUSCALE, LLC APPLICATION FOR THE NUSCALE DESIGN CERTIFICATION

The extensive ACRS review of the NuScale design and its interactions with representatives of the NRC and transcripts of the following ACRS meetings.

Subcommittee/Full Committee	Date	Subject
Future Plant Design Subcommittee	March 1, 2016	NuScale Topical Report TR-0515-13952-NP, “Risk Significance Determination”
634th ACRS Meeting	May 5-6, 2016	NuScale Topical Report TR-0515-13952-NP, “Risk Significance Determination”
NuScale Subcommittee	February 7, 2017	NuScale Topical Report TR-1015-18653-P, Revision 1, “Design of the Highly Integrated Protection System Platform”
642nd ACRS Meeting	April 6-7, 2017	NuScale Topical Report TR-1015-18653-P, Revision 1, “Design of the Highly Integrated Protection System Platform”
NuScale Subcommittee	May 24, 2017	NuScale Topical Report TR-0815-16497, Revision 1, “Safety Classification of Passive Nuclear Power Plant Electrical Systems”
645th ACRS Meeting	July 12-14, 2017	NuScale Topical Report TR-0815-16497, Revision 1, “Safety Classification of Passive Nuclear Power Plant Electrical Systems”
NuScale Subcommittee	September 20, 2017	NuScale Topical Report TR-0116-20825-P, Revision 1, “Applicability of AREVA Fuel Methodology for the NuScale Design”
647th ACRS Meeting	October 5-6, 2017	NuScale Topical Report TR-0116-20825-P, Revision 1, “Applicability of AREVA Fuel Methodology for the NuScale Design”
NuScale Subcommittee	January 23, 2018	NuScale Exemption Request From 10 CFR Part 50, Appendix A, General Design Criterion 27, “Combined Reactivity Control Systems Capability”
650th ACRS Meeting	February 8-9, 2018	NuScale Exemption Request From 10 CFR Part 50, Appendix A, General Design Criterion 27, “Combined Reactivity Control Systems Capability”
NuScale Subcommittee	May 15, 2018	NuScale Topical Report TR-0616-48793, Revision 0, "Nuclear Analysis Codes and Methods Qualification" NuScale Topical Report TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations"

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Subcommittee/Full Committee	Date	Subject
654th ACRS Meeting	June 6-7, 2018	NuScale Topical Report TR-0616-48793, Revision 0, "Nuclear Analysis Codes and Methods Qualification" NuScale Topical Report TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations"
NuScale Subcommittee	June 6, 2018	Chapter 8, “Electric Power”
NuScale Subcommittee	August 23, 2018	Chapter 7, “Instrumentation and Control”
NuScale Subcommittee	August 24, 2018	NuScale Topical Report TR-0915-17564-P, Revision 1, “Subchannel Analysis Methodology”
656th ACRS Meeting	September 6, 2018	NuScale Topical Report TR-0915-17564-P, Revision 1, “Subchannel Analysis Methodology,” Chapter 7, “Instrumentation and Control” and Chapter 8, “Electric Power”
NuScale Subcommittee	December 18, 2018	Chapter 2, “Site Characteristics and Site Parameters” and Chapter 17, “Quality Assurance and Reliability Assurance”
NuScale Subcommittee	January 23, 2019	Chapter 13, “Conduct of Operations” and Chapter 18, “Human Factors Engineering”
660th ACRS Meeting	February 6-8, 2019	Chapter 2, “Site Characteristics and Site Parameters” and Chapter 17, “Quality Assurance and Reliability Assurance”
661st ACRS Meeting	March 7-8, 2019	Chapter 13, “Conduct of Operations” and Chapter 18, “Human Factors Engineering”
NuScale Subcommittee	March 20-21, 2019	Chapter 9, “Auxiliary Systems,” Chapter 10, “Steam and Power Conversion Systems,” Chapter 11, “Radioactive Waste Management,” Chapter 12, “Radiation Protection,” and Chapter 16, “Technical Specifications”
662nd ACRS Meeting	April 4-5, 2019	Chapter 9, “Auxiliary Systems,” Chapter 10, “Steam and Power Conversion Systems,” Chapter 11, “Radioactive Waste Management,” Chapter 12, “Radiation Protection,” and Chapter 16, “Technical Specifications”

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Subcommittee/Full Committee	Date	Subject
NuScale Subcommittee	April 17, 2019	Chapter 4, "Reactor" and Chapter 5, "Reactor Coolant System and Connecting Systems"
663rd ACRS Meeting	May 2-3, 2019	Chapter 4, "Reactor" and Chapter 5, "Reactor Coolant System and Connecting Systems"
NuScale Subcommittee	May 14-16, 2019	Chapter 3, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures and Components;" Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria;" Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors;" and Chapter 21, "Multi-Module Design Considerations"
664th ACRS Meeting	June 5-7, 2019	Chapter 3, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures and Components;" Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria;" Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors;" and Chapter 21, "Multi-Module Design Considerations"
NuScale Subcommittee	July 19, 2019	NuScale Topical Report TR-0516-49417-P, Revision 0, "Evaluation Methodology for Stability Analysis of the NuScale Power Module"
NuScale Subcommittee	August 20, 2019	NuScale Topical Report TR-0716-50351, Revision 0, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces"
666th ACRS Meeting	September 4-6, 2019	NuScale Topical Report TR-0716-50351, Revision 0, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," NuScale Topical Report TR-0516-49417-P, Revision 0, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," and Proposed Focus Area Review Approach of the Advanced Safety Evaluation Report with no Open Items for the Design Certification Application of the NuScale Small Modular Reactor
NuScale Subcommittee	November 20, 2019	NuScale Topical Report TR-0915-17565, Revision 3, "Accident Source Term Methodology," and Source Term Area of Focus Review
669th ACRS Meeting	December 4-6, 2019	NuScale Topical Report TR-0915-17565, Revision 3, "Accident Source Term Methodology," and Source Term Area of Focus Review

APPENDIX I

CHRONOLOGY OF THE ACRS REVIEW OF THE NUSCALE, LLC APPLICATION FOR THE NUSCALE DESIGN CERTIFICATION

The extensive ACRS review of the NuScale design and its interactions with representatives of the NRC and transcripts of the following ACRS meetings.

Subcommittee/Full Committee	Date	Subject
NuScale Subcommittee	February 4, 2020	NuScale Area of Focus - Helical Tube Steam Generator Design
NuScale Subcommittee	February 19-20, 2020	NuScale Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology," NuScale Topical Report TR-0516-49416, Revision 2, "Non-Loss-of-Coolant Accident Analysis Methodology," NuScale Topical Report TR-0516-49422, Revision 1, "Loss-of-Coolant Accident Evaluation Model,"
NuScale Subcommittee	March 2-4, 2020	Chapters 15, Transient and Accident Analyses," Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation," and Emergency Core Cooling System and Probabilistic Areas of Focus
671st ACRS Meeting	March 5-7, 2020	NuScale Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology," NuScale Topical Report TR-0516-49416, Revision 2, "Non-Loss-of-Coolant Accident Analysis Methodology," NuScale Topical Report TR-0516-49422, Revision 1, "Loss-of-Coolant Accident Evaluation Model," and NuScale Area of Focus - Helical Tube Steam Generator Design
672nd ACRS Meeting	April 8-10, 2020	NuScale Combustible Gas Monitoring and NuScale Chapter 15: Open Item Closure and Area of Focus Reviews - Return to Criticality And Boron Distribution
673rd ACRS Meeting	May 6-8, 2020	NuScale Areas of Focus – Probabilistic Risk Assessment and Emergency Core Cooling System Valve Performance
674th ACRS Meeting	June 3-5, 2020	NuScale Area of Focus – Boron Redistribution
675th ACRS Meeting	July 8-10, 2020	NuScale Area of Focus – Boron Redistribution
676th ACRS Meeting	July 21-24, 2020	NuScale Area of Focus – Boron Redistribution and Design Certification Final Report

APPENDIX II

ACRS Letter Reports on NuScale DCA Chapters, Areas of Focus and Topical Reports

Subject	Date	ADAMS Accession Number
NuScale Power, LLC. Licensing Topical Report, “Risk Significance Determination”	May 18, 2016	ML16130A373
Safety Evaluation of the NuScale Power, LLC Licensing Topical Report TR-1015-18653-P, Revision 1, “Design of the Highly Integrated Protection System Platform”	April 24, 2017	ML17108A433
Safety Evaluation of the NuScale Power, LLC Topical Report TR-0815-16497-P, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 1	July 26, 2017	ML17205A380
Safety Evaluation of the NuScale Power, LLC Topical Report TR-0116-20825-P, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1	October 20, 2017	ML17290B057
NuScale Power Exemption Request From 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability"	February 21, 2018	ML18052A532
Safety Evaluation of the NuScale Power, LLC Topical Report TR-0616-48793, Revision 0, "Nuclear Analysis Codes and Methods Qualification" and Safety Evaluation of the NuScale Power, LLC Topical Report TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations"	June 15, 2018	ML18166A303
Chapters 7 and 8 of the NRC Staff's SER with Open Items Related to the Certification of the NuScale Small Modular Reactor	September 26, 2018	ML18270A374
Safety Evaluation of the NuScale Power, LLC Topical Report TR-0915-17564-P, Revision 1, "Subchannel Analysis Methodology"	September 26, 2018	ML18270A383
Interim Letter: Chapters 2 and 17 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the NuScale Small Modular Reactor	February 21, 2019	ML19052A046
EDO Response to ACRS Letter of September 26, 2018 on Chapters 7 and 8 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the NuScale Small Modular Reactor	March 7, 2019	ML19066A163

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ACRS Letter Reports on NuScale DCA Chapters, Areas of Focus and Topical Reports

Subject	Date	ADAMS Accession Number
Interim Letter: Chapters 13 and 18 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Design Certification Application Review of the NuScale Small Modular Reactor	March 21, 2019	ML19079A218
Interim Letter: Chapters 9, 10, 11, 12 and 16 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Design Certification Application Review of the NuScale Small Modular Reactor	April 17, 2019	ML19107A174
Interim Letters: Chapters 4 and 5 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Design Certification Application Review of the NuScale Small Modular Reactor	May 30, 2019	ML19151A306
Interim Letter – Chapter 3, Section 3.9.2, and Chapters 14, 19 and 21 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Design Certification Application Review of the NuScale Small Modular Reactor	June 19, 2019	ML19170A381
Response to Chapter 2, "Site Characteristics and Site Parameters," and Chapter 17, "Quality Assurance and Reliability Assurance" of the U.S. Nuclear Regulatory Commission Staff's Safety Evaluation Report with Open Items Related to the Certification of the NuScale Power, LLC, Small Modular Reactor	June 27, 2019	ML19171A350
Safety Evaluation of the NuScale Topical Report TR-0516-49417-P, Revision 0, "Evaluation Methodology for Stability Analysis of the NuScale Power Module"	September 20, 2019	ML19266A463
Safety Evaluation of the NuScale Topical Report TR-0716-50351, Revision 0, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces"	September 24, 2019	ML19268A109

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ACRS Letter Reports on NuScale DCA Chapters, Areas of Focus and Topical Reports

Subject	Date	ADAMS Accession Number
Proposed Focus Area Review Approach of the Advanced Safety Evaluation Report with No Open Items for the Design Certification Application of the NuScale Small Modular Reactor	September 25, 2019	ML19269B682
Safety Evaluation of the NuScale Power, LLC Topical Report TR- 0915-17565, Revision 3, "Accident Source Term Methodology," and Source Term Area of Focus Review for the NuScale Small Modular Reactor	December 20, 2019	ML19354A031
NuScale Area of Focus - Helical Tube Steam Generator Design	March 24, 2020	ML20091G387
NuScale Power, LLC, Design Certification Application - Safety Evaluation for Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 1	Mach 25, 2020	ML20085K327
Safety Evaluation Report for Topical Report TR-0516-49416, Revision 2, "Non-Loss-of-Coolant Accident Analysis Methodology"	March 25, 2020	ML20085K048
Safety Evaluation Report of the NuScale Power, LLC, Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology"	March 26, 2020	ML20086Q959
NuScale Combustible Gas Monitoring	April 28, 2020	ML20113F049
NuScale Chapter 15: Open Item Closure and Area of Focus Reviews – Return to Criticality and Boron Distribution	April 29, 2020	ML20115E403
NuScale Areas of Focus – Probabilistic Risk Assessment and Emergency Core Cooling System Valve Performance	June 1, 2020	ML20149K596
NuScale Area of Focus – Boron Redistribution	July 29, 2020	ML20210M890

APPENDIX III

ACRS Memoranda on Advanced SER Chapters

Subject	Date	ADAMS Accession Number
ACRS Review of NuScale Power, LLC, Design Certification Application - Safety Evaluation with no Open Items for Chapters 7, 10, 11, and 17	November 19, 2019	ML19297D008
ACRS Review of NuScale Power, LLC, Design Certification Application - Safety Evaluation with no Open Items for Chapters 5 and 21	December 10, 2019	ML19337B671
ACRS Review of NuScale Power, LLC, Design Certification Application - Safety Evaluation with no Open Items for Chapters 2, 8 and 12	January 8, 2020	ML20006G325
ACRS Review of NuScale Power, LLC, Design Certification Application - Safety Evaluation with no Open Items for Chapters 1, 3, 4, 6, 9, 13, 14, 15, 16, 18, 19 and 20	February 21, 2020	ML20044D595