May 28, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Summary of Impacts to eRAI 8930 Response and Discussion on the Exemption from General Design Criterion 33

REFERENCES:

1. Nuclear Regulatory Commission, "Request for Additional Information No. 484 (eRAI No. 8930)," dated May 29, 2018 (ML18149A640)
2. NuScale Response to NRC "Request for Additional Information No. 484 (eRAI 8930)," dated September 14, 2018 (ML18257A308)
3. NuScale Supplemental Response to NRC "Request for Additional Information No. 484 (eRAI 8930)," dated July 18, 2019 (ML19199A117)
4. NuScale Supplemental Response to NRC "Request for Additional Information No. 484 (eRAI 8930)," dated November 27, 2019 (ML19332A120)
5. Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "Submittal of Second Updates to NuScale Power, LLC Standard Plant Design Certification Application, Revision 4," dated May 20, 2020 (ML20141L787)

During public teleconferences on April 1 and April 14, 2020 with NRC staff, including reviewers from Reactor Systems Branch, NuScale Power, LLC (NuScale) discussed potential updates to the actuation of the emergency core cooling system described in the Final Safety Analysis Report (FSAR). As a result of these discussions, NuScale provided changes to the FSAR in Reference 5.

Attachment 1 to this letter provides a summary on the impacts of this update to NuScale's previous response to eRAI 8930 (Reference 4) due to the design changes discussed in Reference 5.

Attachment 2 to this letter provides a discussion of analyses performed to support the exemption requested from General Design Criterion 33 in Part 7 of the NuScale Design Certification Application, for events which led to the design changes provided in Reference 5.

While the discussion in Attachment 2 shows that NuScale was able to demonstrate the events addressed were acceptable using deterministic assumptions and methods, NuScale would like to recognize that an opportunity was lost to take a risk-informed approach in lieu of performing these analyses. Specifically, as summarized in Attachment 2, implementation of

Commission direction in “Staff Requirements - SECY-19-0036 – Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” dated July 2, 2019 would seem to have resolved this issue. Although the opportunity to apply the Staff Requirement Memorandum (SRM) was missed in this case, we look forward to continue working with the NRC Staff to implement the Commission’s SRM in the most effective manner for the scope and content of this application, and future NRC staff reviews.

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

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Attachment 1: ECCS Setpoint and Riser Design Change Impact on NuScale’s Response to eRAI 8930
Attachment 2: Review of the NuScale Design Certification Application Exemption Request from General Design Criterion 33 (GDC 33)

ECCS Setpoint and Riser Design Change Impact on NuScale's Response to eRAI 8930

The RAI 8930 supplement 2 response submitted in NuScale letter RAIO-1119-68093 and dated November 27, 2019 (ML19332A120) completely replaced the previous supplemental response and the original response submitted in RAIO-0719-66323, dated July 17, 2019 (ML19199A117), and RAIO-0918-61810, dated September 15, 2018 (ML18257A308), respectively. The RAI 8930 supplement 2 response is not impacted by the emergency core cooling system (ECCS) setpoint changes or the riser design change to add flow holes as discussed below.

The RAI 8930 response evaluated the boron distribution in the NuScale Power Module (NPM), with focus on boron transport after ECCS actuation, and concluded that the boron concentration in the core remained above the initial concentration for 72 hours after event initiation. Three events were evaluated: RCCW (Reactor Core Cooling Water) line break, RRV (Reactor Recirculation Valve) inadvertent opening, and an RVV (Reactor Vent Valve) inadvertent opening. The RRV inadvertent opening event was found to be limiting for boron distribution analysis.

A long-term cooling technical report (LTC) (TR-0916-51299) assessment compared an RVV steam space break (five percent and full), an injection line liquid break (five percent and full), and decay heat removal system (DHRS) cooldown events' progressions with and without the ECCS setpoint changes and riser hole addition. The full injection line liquid break behaves similarly to the RRV inadvertent opening event. The low reactor coolant system (RCS) pressure ECCS signal was found to have no impact on LTC analyses since all loss-of-coolant accident (LOCA) cases assume loss of DC power, so that ECCS actuation occurs as soon as the inadvertent actuation block (IAB) release setpoint is reached. In addition, the signal is designed to prevent actuation during non-LOCA events. Lowering the ECCS actuation setpoint on containment vessel (CNV) level has no impact for similar reasoning, and therefore the changes to the ECCS setpoints have no impact on the LTC analyses. For the limiting boron dilution analysis presented in response to RAI 8930, likewise, the ECCS setpoint changes will have no impact, since all ECCS valves are open very soon after event initiation at about the same time as seen in the full injection line liquid break in the assessment.

As shown in the assessment, the riser holes do not significantly impact the long-term module cooldown. For LOCA events, the riser holes will be uncovered after ECCS actuation and have no impact long term. For small breaks, a small difference in RCS depressurization rates can impact the timing of ECCS actuation on IAB release pressure, but long-term conditions trend toward the same value regardless of differences in timing. For the five percent high point vent line break, core inlet temperature and RCS pressure merged at seven hours, and there was no deviation in these parameters for the full break. For the injection line break, both the five per cent and full breaks exhibited essentially no difference in core inlet temperature and RCS pressure.

The RAI 8930 response evaluated:

1. boron transport between the CNV, downcomer, and core;
2. boron volatility in the steam generated in the core and riser;
3. boron loss mechanisms;

4. core and riser boron mixing;
5. critical heating length; and
6. reactivity balance in the core

An evaluation for the boron distribution was also performed for an extended period of time out to 7 days, with more realistic assumptions for the limiting RRV inadvertent opening event, and is discussed in the RAI 8930 supplemental response.

Prior to ECCS actuation, the riser holes have negligible impact on the RCS thermal-hydraulic response, and provide a benefit to the boron transport behavior for smaller breaks before ECCS actuation. The LTC assessment showed negligible changes in the RCS thermal-hydraulic response and, by extension, negligible changes in the boron transport behavior following ECCS actuation. There may be local flow distribution changes due to the addition of the riser holes, however the impact is limited given the holes are relatively small, and only impact the liquid flow for a short period of time in the initial transient phase.

It is concluded that core and riser mixing assumptions are not adversely impacted by the addition of riser holes due to minimal changes in the RCS thermal-hydraulic response and the holes are uncovered following ECCS actuation. Additionally, it is concluded that the core inlet conditions are not impacted such that the “critical boiling length” calculation, demonstrating that the conservatism of the return to power analysis, is not adversely affected.

Volatility and boron loss mechanisms are likewise not negatively impacted due to the minimal changes in the RCS thermal-hydraulic response. Volatility is mainly dependent on the core and riser boron concentration and thermal hydraulic conditions. As volatility is a relatively smaller boron loss term, and the most significant loss term (boron in the CNV liquid below the RRV elevation) is not dependent on the design changes, boron loss mechanisms are not significantly impacted.

For the extended 7 day evaluation, the thermal hydraulic conditions beyond 72 hours are quasi-steady-state and core heat removal is performed by ECCS. The RCS thermal hydraulic response in the LTC evaluation showed that the events evaluated with and without the riser holes produced identical results beyond about seven hours after event initiation. This comparison can be extended to the end of the 7 day period due to the RCS quasi steady-state conditions. Therefore, the conclusions of the 7 day analysis discussed in the last supplement to the RAI 8930 response is still applicable.

In conclusion, the prior discussion demonstrates that the ECCS setpoint changes and the riser hole addition have minimal impact on the RAI 8930 response. In addition, it is noted that no credit has been given for the riser hole addition in the RAI 8930 transient cases to mitigate the adverse impacts of boron redistribution prior to riser hole uncovery.

Review of the NuScale Design Certification Application Exemption Request from General Design Criterion 33 (GDC 33)

As part of the NuScale Design Certification Application (DCA), an exemption was requested from GDC 33 and described in DCA Part 7. The Final Safety Analysis Report (FSAR) provides analyses that conservatively demonstrate that the automatic emergency core cooling system (ECCS) functions ensure that specified acceptable fuel design limit (SAFDL) violations will not occur for events resulting in design basis event leakage, i.e., that fall within the defined scope of applicability and criteria in 10 CFR 50.46.

Additionally, NuScale has performed analyses that demonstrate automatic ECCS functions (safety-related actuations or inherent design features) will ensure no SAFDL violations will occur for leak rates below that scope with assumed initial conditions beyond those expected or likely to occur during normal operations. These conditions conservatively overlay potential normal operational coolant leaks described in the technical specification limits.

Coolant loss and effects during normal operation are required to be identified and responded to by operator actions specified in the plant operating license as technical specification requirements. These actions are required to be taken and subject to control at the same level as other limits established in the plant technical specifications.

Review of Analyses

The following descriptions summarize the analyses provided for the consideration of boron redistribution and potential for core dilution.

Non-Loss-of-Coolant Accident (Non-LOCA) Transients
Non-LOCA transients were evaluated, including a spectrum of primary and secondary inventory conditions (reflecting, for example, a chemical and volume control system (CVCS) break outside containment, or increase in feedwater flow transients). Results showed acceptable boron distribution results that preclude core dilution concerns by ensuring the downcomer concentrations remain conservatively above critical boron concentration (CBC), accounting for uncertainties, for up to a 7-day period.

Loss-of-Coolant Accident (LOCA) Transients
LOCA transients were analyzed for the full LOCA spectrum, as defined by CVCS makeup capacity, to demonstrate sufficiently early ECCS actuation prior to any significant downcomer dilution occurring. Due to the rapid nature of ECCS actuation and low CBCs at the time of ECCS actuation, these scenarios are well bounded by extended decay heat removal system (DHRS) operating scenarios.

Reactor Coolant System (RCS) Leakage
RCS leak rate from a range of approximately 6gpm up to 40gpm was also evaluated in support of the NuScale GDC 33 exemption request. The analysis concludes that early ECCS actuation on low RCS

pressure can be expected for all leakage, except in the case of very cold initial pool temperature conditions. For these cold initial pool temperature cases it is expected that the RCS pressure will drop sufficiently for the ECCS valves to open themselves passively within approximately 20 hours. If this ECCS valve function is ignored by analysis, the riser mixture level is conservatively calculated to provide sufficient flow through the riser holes to maintain the downcomer concentration above the critical boron concentration for a period of 24 hours. At 24 hours, ECCS would be actuated for loss of AC power scenarios. If AC power is available, a valid ECCS actuation signal would not be expected within a 72 hour period due to the lack of containment (CNV) level accumulation.

While it is not deterministically considered, if the possibility of ECCS actuation were assumed at any point after 24 hours and before 72 hours, there is a possibility that the riser mixture level could drop below the riser holes which could generate a boron gradient between the core and downcomer. The analysis for the core concentration when ECCS is actuated at 24 hours indicates that a later ECCS actuation would not generate a core dilution concern even if the downcomer is diluted. This is because during the ECCS transient, there is a net mass flow out of the core due to the increased mixture level during the depressurization period. Eventually, pressures and fluid levels stabilize, at which point diluted coolant from the CNV enters the downcomer through the RRVs, which is a condition that was analyzed in detail in the final supplemental response to RAI-8930 (ML19332A120).

From a risk perspective, RCS leakage events, in which operators hypothetically fail to operate the plant within Technical Specifications that transition safely to ECCS, are attributed a frequency of 6E-7 per year and would not result in a core damage. The likelihood of a common cause failure of all 5 ECCS valves to passively open on low differential pressure is significantly less likely (5E-12 per module-critical-year) and also would not result in a core damage event. The more probable scenario of a subset of the ECCS valves not opening correctly is already captured in the probabilistic risk assessment (PRA) as an incomplete ECCS actuation with a frequency of approximately 7E-12 per year and results in subsequent core damage, as RCS makeup would also be nominally available, but is assumed to fail in this scenario. Because the events of interest include an isolated containment, there is no large release frequency and thus no significant radiological consequences. This provides additional support to conclude that the events postulated above are not risk significant.

Requirement for Action by the Plant Operating Staff

Technical specifications are incorporated into the operating license of nuclear power plants as Appendix A. They describe requirements that meet the criteria in 10 CFR 50.36 that help ensure the plant will be operated in accordance with the safety analyses that protect the health and safety of the public.

The technical specifications include requirements regarding the use and application of the requirements specified within them. These are specified in Chapter 1, “Use and Application,” with additional requirements for implementing Limiting Conditions for Operation specified in Section LCO 3.0, “Limiting Condition for Operation (LCO) Applicability,” and SR 3.0, “Surveillance Requirements (SR) Applicability.”

The former apply to all sections of the specifications and define terms used throughout the document (and presented in ALL CAPS), as well as the meaning of formatting elements used in the document. The latter, sections 3.0 specify requirements and rules applied to actions and surveillance requirements. For example, LCO 3.0.1 states that "LCOs shall be met during MODES or other specified conditions…," and LCO 3.0.2 states "[u]pon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met…." [1]

Controls and requirements are also provided by the surveillance testing required by the technical specifications. The applicability rules in SR 3.0 provide requirements for performance of testing and actions to be taken if a failure to meet a surveillance occurs. Specifically, SR 3.0.1 states

> *Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be a failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO…*

If such a failure occurs, the requirements of LCO 3.0.2 apply.

Technical specification 5.2 specifies the required staff for the plant, including the manager responsible for overall safe operation of the plant, and the facility staff required to be at the facility to implement the requirements of the operating license, including the technical specifications.

Plant operations are required to be conducted in accordance with written procedures as specified in technical specification 5.4, "Procedures."

In summary, the Commission has issued regulations describing the required content of those specifications as 10 CFR 50.36 and 10 CFR 50.36a. The staff and industry have developed guidance and expectations regarding the format and use of the technical specifications. NuScale has proposed technical specifications in conformance with these requirements and guidance documents. And the technical specifications include requirements for the plant to be operated by qualified staff in accordance with written procedures that will require LCOs to be met, or upon discovery of a failure to meet an LCO the specified Required Actions must be taken.

These requirements are the specific basis for the operating license requirement that the operating staff take action in accordance with written procedures if an LCO or SR is not met.

<u>RCS Leakage Limits and Required Actions</u>

Technical specifications require that RCS Operational LEAKAGE be limited to less than the values listed in LCO 3.4.5. As noted in the Bases for LCO 3.4.5,

[1] The LCO 3.0 and SR 3.0 sections include exceptions and allowances that address other, special, and limited circumstances that occur during implementation of the requirements. They have evolved over decades of use, however the discussion provided is the fundamental requirements of the subsections quoted. For additional details see the NuScale, or generic technical specifications (NUREG-1431, etc.) LCO 3.0 and SR 3.0 sections and associated bases.

Except for primary to secondary LEAKAGE, the safety analyses do not address RCS Operational LEAKAGE. However, other forms of RCS Operational LEAKAGE are related to the safety analyses for LOCA. The amount of LEAKAGE can affect the probability of such an event.

LCO 3.4.5 includes limits on pressure boundary, unidentified, identified RCS, and primary to secondary LEAKAGE. These limits are structured similar to those in large pressurized water reactors, however the NuScale design makes identification of leakage challenging and in most conditions the 0.5 gpm unidentified LEAKAGE maximum is likely to be the limit applied during the conditions of applicability.

If the identified RCS or unidentified LEAKAGE limits are exceeded, Condition A of LCO 3.4.5 will apply and the Required Action is that the LEAKAGE must be restored to within limits within 4 hours, as required by LCO 3.0.2.

If the LEAKAGE cannot be restored to within limits within four hours, or pressure boundary or primary-to-secondary LEAKAGE limits are not met, then Condition B must be entered as required by LCO 3.0.2. Required Actions B.1 and B.2 require the operators to take action in accordance with LCO 3.0.2 for the plant to be in MODE 2 (subcritical) within 6 hours and to be in MODE 3 with the RCS hot temperature below 200 ºF within 48 hours.

These actions are required by LCO 3.0.2, and designed to verify leakage rates and either identify unidentified LEAKAGE or reduce RCS Operational LEAKAGE to within limits before the reactor must be shut down. The completion times for Condition B are designed to provide a reasonable opportunity to reach the required unit conditions from full power conditions in an orderly manner.

RCS Leakage Instrumentation Requirements

GDC 30 of Appendix A to 10 CFR 50 requires means for detecting and, to the extent practical, identifying the source of RCS LEAKAGE. Regulatory Guide 1.45 describes acceptable methods for selecting LEAKAGE detection systems.

LEAKAGE detection systems must have the capability to detect significant reactor coolant pressure boundary degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE. As noted in the Bases of LCO 3.4.5 RCS LEAKAGE is related to the safety analyses for LOCA. The amount of LEAKAGE can affect the probability of such an event.

The NuScale design includes three distinct methods for detection and monitoring of RCS leakage as described in FSAR section 5.2.5.1.

- Containment Evacuation System (CES) Collected Condensate
- Containment Pressure
- Radioactivity Monitoring

All three methods utilize features of the CES that is used to maintain the containment vessel at a very low pressure during operations.

All three methods are required to be OPERABLE as specified in LCO 3.4.7, RCS Leakage Detection Instrumentation. The LCO prescribes the requirements that each of the instrumentation methods be available when the reactor coolant system (RCS) is above 200 ºF, with the ECCS valves closed and the containment not flooded.

The first two of the three methods described above include two channels each of instrumentation used to measure the relevant parameters, i.e., there are two channels for measuring condensate collection, and two channels for measuring containment pressure. One channel of radioactivity monitoring is available in addition to the other two methods. Each of the methods and channels are subjected to CHANNEL CHECKS, CHANNEL OPERATIONAL TESTS, and CALIBRATIONS in accordance with the surveillance requirements in LCO 3.4.7.

Plant procedures described in FSAR chapter 13, and required by technical specification 5.4, will describe the means and methods the operating staff will use to detect and respond to leakage from the RCS. These will include alarms, indications, and surveillance testing results. Each of the three methods provide indication to the plant operators of the plant compliance with the specified limits. Detailed descriptions of the functions, instrumentation, capability, and required conditions to ensure OPERABILITY are described in FSAR 5.2.5.1.

LCO 3.4.7 includes Required Actions that must be taken if the LCO cannot be met in accordance with LCO 3.0.1 and LCO 3.0.2. The Conditions are more complex than typical PWR specifications because of the design of the NuScale leakage detection instrumentation.

In accordance with the requirements of LCO 3.0.2, if only one channel of the first two required methods is inoperable, operations are allowed to continue for up to 14 days if a surveillance requirement of the RCS leakage LCO is performed every 24 hours. This is acceptable because three methods of RCS leakage remain OPERABLE, albeit with a single channel OPERABLE for the affected method. The required surveillance performs an RCS water inventory balance providing further assurance of leakage detection with an inoperable channel.

If both channels of a method with two channels, or the single-channel radioactivity monitor method, are inoperable then at least one channel must be restored for functions with two channels, or the radioactivity monitor must be restored to OPERABLE within 72 hours, as required by LCO 3.0.2. If the inoperable function is not restored, then Condition C is required to be implemented and the plant must be placed in MODE 2 (subcritical) within 6 hours, and be in MODE 3 with the RCS hot temperature below 200 ºF within 48 hours. The completion times are designed to provide a reasonable opportunity to reach the required unit conditions from full power conditions in an orderly manner.

Operational Implementation of Leakage Requirements

The operations staff in the control room are provided with extensive capability to monitor the RCS leakage and assure the requirements of LCO 3.4.5 and 3.4.7 are met. Technical specification 5.2 requires minimum staffing in the plant control room.

Plant procedures required by technical specification 5.4 will ensure that appropriate operational awareness of the plant conditions is maintained by the required operators.

- COL Item 13.5-1 requires a COL applicant to describe the site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance provided in Regulatory Guide 1.33, Revision 3.
- COL Item 13.5-2 requires a COL applicant to describe the site-specific procedures that operators use in the main control room and locally in the plant, including normal operating procedures, abnormal operating procedures, and emergency operating procedures.
- COL Item 13.5-3 requires a COL applicant to describe the site-specific maintenance and operating procedures, including calibration and test procedures, and maintenance procedures.

These COL items and the programs and procedures required to be prepared by a COL applicant will ensure the operating staff in the control room maintains continuous awareness of the plant status. The design of the systems ensure that operators are notified if parameters are trending toward or exceed limits, and response procedures that implement the technical specification requirements will ensure operators take action.

The available leakage monitoring capabilities are described in detail in FSAR 5.2.5.1 and include:

Condensate Collection

- Collection vessel level instrumentation includes quantification and trending of leak rate changes.
- Indication and alarm in the control room to monitor and trend liquid leakage into the containment atmosphere.
- Control room alarm before reaching the maximum allowable pressure-temperature to ensure method functionality.
- Capability to detect an RCS leak rate of less than 1 gpm within one hour.
- Minimum detectable leak rate of less than 0.05 gpm.

Containment Pressure

- The inlet to the containment evacuation system includes pressure sensors that monitors containment pressure.
- Control room indication of pressure and capability to alarm and trend changes in pressure.
- Control room alarm before reaching the maximum allowable pressure-temperature to ensure method functionality.
- Capable of detecting an increase of 0.1 psi in less than one minute of a 1 gpm leak from the RCS.
- Capable of detecting a minimum leak rate from the RCS of 0.007 gpm.

Radioactivity Monitor

- Monitors both gaseous and liquid effluent removed from the containment atmosphere.
- Capability to evaluate isotopic makeup (including tracer gases if used) to identify leakage source.
- Indication, alarm, and capability to trend radioactivity levels in the control room.

The NuScale design is inherently more sensitive to detecting RCS leakage than other designs. The containment vessel is operated at an actively maintained low pressure that minimizes the presence of any gas or liquid. This sensitivity makes any change caused by postulated leakage more apparent and readily quantifiable than previous designs.

Additionally, the OPERABILITY of two leak detection functions that depend on the containment evacuation system requires that the pressure in the containment be maintained below the saturation pressure of the surrounding pool water as described in FSAR section 5.2.5.1. This pressure is alarmed and can be continuously monitored in the control room if the pressure approaches that which would render the detection functions inoperable. As shown in FSAR Figure 5.2-3, the nominal UHS pool temperature of 100 ºF will result in an alarm before the containment pressure approaches about 0.8 psia.

In the NuScale design the containment pressure, i.e., vacuum, becomes a key operating parameter being maintained and monitored by the plant staff using the containment evacuation system described in FSAR section 9.3.6.2.1. While existing large PWRs monitor containment pressure and incorporate alarms and actuations, the large volume of those containment structures typically result in slower pressure variations and no sensitivity to liquid leaks into the containment volume.

Based on the above, the NuScale design ensures that any leakage into the containment will be rapidly detected, evaluated, and acted upon in accordance with the requirements of the technical specifications limits and actions required to be implemented by the unit operating license.

Risk Informed Summary

Although in this case NuScale is able to demonstrate the events addressed in this attachment were acceptable using Chapter 15 assumptions and methods, an opportunity was lost to take a risk-informed approach in lieu of these analyses, which would have resulted in less time and cost, while still providing reasonable assurance of adequate protection of public health and safety. Specifically, as summarized below, implementation of Commission direction in “Staff Requirements - SECY-19-0036 – Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” dated July 2, 2019 would seem to have resolved this issue. In that staff requirements memorandum (SRM), the Commission required that, “… the Staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria … is unnecessary to provide for reasonable assurance of adequate protection of public health and safety.”

The strict, prescriptive application of deterministic criteria imposed in this case were that, for the duration of the analyses:

- AC and DC power were assumed available, and
- Operators were assumed to not take any actions, including those required by the operating license via technical specifications.

Under these assumptions and using Chapter 15 methods, ECCS actuation would be indefinitely delayed for very small (6-12 gpm) reactor coolant system leaks. Thus, Staff questioned whether this hypothetical scenario would lead to an analyzed plant condition outside the NuScale licensing basis. Because it is plausible that the NuScale plant may be less safe with electric power available in certain rare circumstances, performing analysis assuming electric power is available is normally an appropriate approach for Chapter 15. However, coupling that assumption with an assumption that operators violate technical specification required actions (which in all cases would have mitigated the events within 10 hours) was unnecessary in this case.

The NuScale plant is designed to remain safe indefinitely, with a complete loss of all AC and DC power, a situation that is catastrophic for most designs whether analyzed under Chapter 15 or more realistically under Chapter 19 methods. While the NuScale design is safe without AC and DC power, it should be obvious that the plant will be even safer when AC and DC power are available, regardless of conclusions from Chapter 15. This is evident from the Probabilistic Risk Analysis documented in this attachment and in Chapter 19. For the small RCS leak sequence postulated here, the risk of core damage is several orders of magnitude below the NRC's core damage frequency (CDF) goal, and a large release is precluded by an intact containment. Therefore, applying risk-informed principles would have shown that strict, prescriptive application of deterministic criteria was unnecessary in this case to provide for reasonable assurance of adequate protection of public health and safety. Pursuant to the Commission's direction in SRM-SECY-19-0036, the deterministic analysis described in this attachment was unnecessary.

Although the opportunity to apply the SRM was missed in this case, it is an important lesson-learned to carry forward to the SDA application. Going forward, NuScale recommends that pre-application engagement with the NRC on our planned Standard Design Approval (SDA) application focus on identifying areas where the SRM may be implemented to yield a more safety-focused application and review. We look forward to working with the NRC staff to implement the Commission's SRM in the most effective manner for the scope and content of the application, and the NRC staff review.