UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 15, 2022

Ms. Carrie A. Fosaaen
Director, Regulatory Affairs
NuScale Power, LLC
1100 Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: PREAPPLICATION READINESS ASSESSMENT REPORT OF THE NUSCALE, POWER, LLC STANDARD DESIGN APPROVAL DRAFT APPLICATION

Dear Ms. Fosaaen:

On October 21, 2022, members of the staff of the U.S. Nuclear Regulatory Commission (NRC) staff completed a preapplication Readiness Assessment (hereinafter "Readiness Assessment") of the draft application and supporting documentation that NuScale Power, LLC (NuScale), intends to submit as part of its Standard Design Approval Application (SDAA). NuScale's letter requesting the Readiness Assessment can be found at Agencywide Documents Access and Management System (ADAMS) Accession No. ML22145A460. NuScale requested the Readiness Assessment in order to (1) identify any required information that is missing from the proposed SDAA and (2) identify technical or regulatory issues that may complicate the acceptance or technical reviews of the SDAA. The staff's readiness assessment plan used to conduct the review of NuScale's proposed SDAA can be found at ML22178A254.

The staff conducted the readiness assessment via NuScale's Electronic Reading Room in accordance with NRC's Office Instruction LIC-116, "Pre-application Readiness Assessment" (ML20104B698). The readiness assessment is not part of the NRC's official acceptance review process; however, the staff performed the Readiness Assessment as part of an approved preapplication activity that allowed the staff to gain an understanding of any significant issues or information gaps between the draft application and the technical content required to be included in the final application submitted to the NRC.

The enclosed Observation Report provides the NRC staff observations of NuScale's SDA draft application. The staff observed and recognized that the NuScale draft SDAA is a work in progress and several chapters are incomplete or have not been updated to reflect the US460 design. Additionally, there were several documents referenced by the draft SDAA (i.e., Topical Reports, Technical Reports, calculations, technical tables, references, probabilistic risk analysis information, etc.) that were not made available to the staff during the Readiness Assessment. As such, the NRC has not observed the entirety of the proposed SDAA application and thus cannot provide a complete assessment of the draft application. Further, the staff's Observation Report does not include the information known to NuScale as missing or incomplete and the observations (included) do not predetermine whether the SDA application will be accepted for review.

In conducting the Readiness Assessment, the staff used a phased approach with groupings of relevant chapters and sections. The collection of observations in the report is organized by chapter and any chapters that did not have observations are absent from the report. Overall, the staff has identified several challenging and/or significant issues that could be focus areas for the SDAA acceptance and/or safety review. While there has been some early engagement on these topics, the staff would encourage continued engagement on these topics until the SDAA's submittal. These topics include:

- Augmented Direct Current (DC) Power System (EDAS) Safety Classification
- Comprehensive Vibration Assessment Plan and Steam Generator Tube Support
- Density Wave Oscillation Analysis
- Containment Vessel Material/Reactor Vessel Material
- Loss-of-Coolant Accident Analysis
- SDA "Optimization"

Enclosed are two Observation reports – non-proprietary version (Enclosure 1) and proprietary version (Enclosure 2). The complete ADAMS package can be found under ML22305A518. The observations included are those that the staff considered to be significant. Additional less significant observations were identified and communicated to NuScale throughout the review. The staff recommends that NuScale consider the entirety of the observations while finalizing the application and application submission date based on your evaluation of the time needed to address the observations.

If you have any questions, please contact Getachew Tesfaye, Senior Project Manager, at (301) 415-8013 or Getachew.Tesfaye@nrc.gov.

Sincerely,

/RA/

Brian Smith, Division Director
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

Docket No. 99902078

Enclosures:
1. Observations Report
2. Observation Report, Proprietary

cc: w/encl.: DC NuScale Power LLC Listserv

SUBJECT: PREAPPLICATION READINESS ASSESSMENT REPORT OF THE NUSCALE, POWER, LLC STANDARD DESIGN APPROVAL DRAFT APPLICATION NOVEMBER 15, 2022

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ADAMS Accession Nos.:
PKG: ML22305A518
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PROP: ML22305A519 (Enclosure 2 Proprietary Summary) *via email NRR-106

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NuScale Standard Design Approval Application Readiness Assessment Observations Report – Non-Proprietary

Chapter	Section	Observation
All	All	The staff found that the "optimization" effort for the Standard Design Approval (SDA) application may have removed information needed to support a reasonable assurance of public health and safety determination.
3	3.2.1.4	The submittal should explain what the new design will entail given that the following paragraphs were removed from Section 3.2.1.4: *There are no Seismic Category I Structures, Systems and Components (SSC) that have RG 1.143 design requirements. There is one Seismic Category II SSC that does. The Radioactive Waste Building is Seismic Category II due to its proximity to the Reactor Building, and it is RW-IIa due to its design radioactive material content.* *RG 1.143 specifies that RW-IIa SSC are designed to withstand ½ of the SSE. As such, the Radioactive Waste Building is designed to both remain intact (satisfying Seismic Category II) when subjected to a full SSE; and intact and functional (satisfying RW-IIa) when subjected to an earthquake with half the force of the SSE.* *All other radioactive waste SSC are sufficiently separated from Seismic Category I SSC that they are Seismic Category III.* *RG 1.143 classification is included in Table 3.2-1 within the Quality Class column. SSC that are classified as RW-IIb and RW-IIc are designed to industry codes and standards, which conforms with Seismic Category III.*
3	3.2.2	The submittal should explain what criteria will be used for the instrument sensing lines given that the following paragraph was removed from Section 3.2.2: *Safety-related instrument sensing lines are designed and constructed in accordance with ANSI/ISA-67.02.01-1999 (Reference 3.2-2) as described in RG 1.151. The standard ANSI/ISA-67.02.01-1999 establishes the applicable code requirements and code boundaries for the design and installation of instrument sensing lines interconnecting safety-related piping and vessels with both safety-related and nonsafety-related instrumentation. This is further discussed in Section 7.2.2.*
3	3.2.2	The submittal should explain what design criteria will be used given that the following paragraph was removed from Section 3.2.2: *The reactor vessel internals (see Section 3.9.5) and steam generator supports, and tube supports (see Section 5.4.1.5) comply with the design and construction requirements of Subsection NG of Section III, Division 1 of the ASME BPVC (Reference 3.2-1).*

3	3.2-1	Per Standard Review Plan (SRP) Section 3.2.1, "Seismic Classification," Revision 3, staff reviews the seismic classification design criteria of those structure, system, and components (SSCs) that are important to safety and are specified as seismic Category I by the applicant's safety analysis report (SAR) and designed to withstand, without loss of function, the effects of a safe shutdown earthquake (SSE). The review also covers identification of SSCs that are not required to remain functional following a seismic event, but whose failure could reduce the functioning of any seismic Category I SSCs (seismic Category II). For a SSC, the delineations between seismic Category I and Category II need to be identified. As an example, SDA Table 3.2-1 Seismic Classification of Building Structures lists the reactor building (RXB) as seismic Category I (as expected). However, the document in the Electronic Reading Room titled, "Design Changes Roadmap-7.15.pdf" reclassifies some floor and roof slabs above grade as seismic Category II. If the RXB contains seismic Category II structures, the entire RXB should not be identified in Table 3.2-1 as seismic Category I without identifying the exceptions or providing justification (e.g., Seismic II/I interactions). The RXB discussion above is one example identified by the staff and it is recommended that the extent of condition is investigated because the staff do not have all the SAR chapters. The staff also factors this seismic information into the Chapter 19 evaluation of the expected probabilistic risk assessment (PRA) based seismic margins assessment.
3	3.9.2	Provide design of steam generator tube supports and demonstrate how this design will avoid damage to the tubes. Also provide the final design of the steam generator tube inlet flow restrictors, and any testing plans to ensure leakage flow mechanisms will not damage the restrictors in both normal and reverse flow conditions.
3	3.4	Provide flood analysis methodology, assumptions, and SSC subject to flood protection for RXB Flood Analysis and control building (CRB) Flood Analysis.
3	3.5.1.3	Describe CRB as related to its missile protection barrier function or explain the reason for the elimination of this function being taken credit of in the design certification application (DCA).
3	3.5.1.3	Identify all the SSC subject to missile protection according to Regulatory Guide (RG) 1.115.

3	3.5.1.3	Understood that this section is still in development (Sections 3.5.1.3.3.1 and 3.5.1.3.3.2). However, the NRC staff would note that the information in Section 3.5.1.3 should be consistent with design control document, including using the same methodology for calculating missile, weight, speed, acceptance criteria, etc. In addition, the NRC staff notes that specific information has been deleted in this version that would be needed to come to a reasonable assurance finding, including the following: • Section 3.5.1.3 needs to include a description on the barriers approach NuScale will use as in DCA. Specific wording quoted from RG 1.115 need not be included, but a description of how barriers will be used and acceptance criteria to be met should be included. • Section 3.5.1.3.2 does not specify bounding missile speed. • Combined License (COL) Item 3.5-2 was deleted and should be included in 3.5.1.3.4 to address turbine missiles from nearby or co-located facilities. • Section 3.5.1.3.5 does not demonstrate that all essential SSCs are protected from postulated low-trajectory missiles as was previously done. Information needs to be included. • No information provided in Figures 3.5-1 and 3.5-2 for missile trajectory essential SSCs, etc. • Table 3.5-2 does not have any information on NuScale turbine missile.
3	3.8.2.3.9	In Subsections 3.8.2.3.9 provides discussion on the load combinations for the containment vessel (CNV). It is not clear why combinations of classified stresses (primary, secondary, peak) for comparison to specified code limits were not considered as provided in Figure XIII-2100-1 of American Society of Mechanical Engineers (ASME) Section III, Appendix XIII.
3	3.8.2.4 and 3.8.2.4.1	In Subsections 3.8.2.4 and 3.8.2.4.1, the last sentence in second and third paragraphs, respectively, states, *"Alternatively, limit analyses to determine lower bound limit buckling loads may be employed in lieu of Code Case N-759-2."* It is not clear which provision(s) in ASME Section III, Division 1, Subsection NB provides the design requirements for determining; the lower bound limit buckling loads using limit analyses.
3	3.8.2.3.6	In Subsection 3.8.2.3.6, identified "aircraft hazard," and "explosion pressure waves" as the external environmental loadings. It is not clear why "aircraft hazard," and "explosion pressure waves" were considered under the external environmental loadings.
3	3.8.4.1.4.2	In Subsection 3.8.4.1.4.2, middle of the first paragraph states, *"The liner is 304L, or equivalent, stainless steel that is 0.25 in. thick in most locations and covers the pool floor."* The requirements for designing SC walls are summarized in TR-0929-71621 with carbon-steel (CS) plates only and is described in Subsection 3.8.4.6.1.3. The current design standard/guidance may not support the construction of the RXB pool SC walls with 0.25-inch-thick stainless steel (SS) liner plate (0.25-inch-thick SS liner is for maintaining the pool water inventory only, not providing structural integrity of the SC wall).

3	3.8.4.6.1.3	In Subsection 3.8.4.6.1.3, first sentence states, *"The requirements for designing SC walls are summarized in TR-0929-71621."* The TR provides a design methodology of SC walls constructed using carbon-steel (CS) plates. The SC walls may also be constructed with SS pool liner and the CS plates and the tie-bars. This will create a detrimental degrading condition where the SS material will be the "noble-material," and the CS material will be the "sacrificial material." Furthermore, it is not clear, what would be the percent (%) of critical damping value for this SC configuration and there might be an issue related to the thermal expansions of SS vs. CS under accident conditions. Finally, the current design standard/guidance may not support the application of dissimilar materials to construct the SC walls.
3	3.8.5.6.5	In Subsection 3.8.5.6.5, "Thermal Loads," no information was provided.
3	3.8.5.6.4	In Subsection 3.8.5.6.4, the allowable limits for vertical displacements, differential settlements (tilting settlements) of the RXB and CRB foundations were provided. However, the allowable limits for foundation settlements between structures (e.g., RXB and CRB, RXB and RWB, etc.) were not provided. It is also not clear whether any comparative analytical assessments of compounded settlements between structures were performed to ensure the structural integrity of tunnels between buildings. Furthermore, the limitations and assessments for angular-distortions in foundations of buildings may also be provided and considered (justification(s) may be provided for not determining the angular-distortions in building foundations).
3	3.3.1	Clarify the maximum wind speed. The staff noted the wind speed 190 mph in Section 3.3.1.1 and the wind speed 145 mph in Section 3.3.1.2.
3	3.9.6	Section 3.9.6 and Table 3.9-18 should be reviewed to update the discussion and relief/alternative requests as necessary to reflect that the ASME Operation and Maintenance (OM) Code of record for the NuScale SDA is the 2017 Edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The ASME OM Code of record for the NuScale DCA was the 2012 Edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. Therefore, the relief/alternative requests in Section 3.9.6 for the NuScale SDA might need to be revised or supplemented to address the implementation of the 2017 Edition of the ASME OM Code as incorporated by reference in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a. The justification for the alternative request for inadvertent actuation block (IAB) valve testing in Section 3.9.6 should be reviewed considering the revised emergency core cooling system (ECCS) design with IAB valves not included in the reactor vent valve (RVV) system. The testing provisions specified in Table 3.9-18 should be reviewed for any necessary changes to reflect the requirements in the 2017 Edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The notes at the end of Table 3.9-18 should be updated to be consistent with the ECCS design changes, and to reflect that the implementation of 2017 Edition of the ASME OM Code will be a regulatory requirement for the NuScale US460 SDA.

4	4.4	All critical heat flux (CHF) correlations used in the application should be described in 4.4. The applicability ranges and limits should be included for the CHF correlations.
4	4.4	Subsection 4.4.4.5.2 states that "conservatisms in the modeling of the subchannel analysis methodology have been determined to bound the effects of crud." However, there is no reference for where this determination was made and is not included in the subchannel analysis TR.
4	4.3	The following information needed to evaluate the core design is missing from Section 4.3: -Fuel loading pattern -Peaking factor (FdH and FQ) curves -Differential boron worth curves -Radial relative power within fresh, once-, and twice-burned fuel assemblies -Power coefficient of reactivity -Assumed xenon concentration in refueling boron concentration calculation Additionally, some entries in Table 4.3-2 are blank.
4	4.3	Provide control rod worth (total for each bank, total from power-dependent insertion limit to full insertion, and differential) at the different operating modes and stages of the reactor and the associated uncertainty information to allow reviewers to make a determination regarding the adequacy of the shutdown margin at different stages of the reactor operations, e.g. beginning of cycle, emergency operations center and middle of the cycle. Because the control rod worth is dependent of the reactor core design, a new set of control rod worth for each bank should be provided for the new core design.
4	4.3	The DCA figure displaying power defect (i.e., Figure 4.3-15) was removed from the SDA. This information is related to the integrated power coefficient over the range of the power change. Reviewers will need this information to evaluate the shutdown margin.
4	4.3	The role of the supplemental boron addition system in assuring long-term shutdown should be discussed.

5	5.3	SS generally have superior ductility relative to low-alloy steels and are less subject to embrittlement effects at similar thermal and neutron flux conditions. Never-the-less, the lack of operating experience and materials data regarding use of SA-965, FXM-19 (and potentially other SS materials used for pressure boundary applications) for reactor vessel construction requires consideration. The staff will require a technical basis supporting that the requirements of General Design Criteria (GDC) 14, 15, and 31 will be met within the SDA that includes the proposed material selections.
5	5.3	10 CFR 50.60 – Acceptance Criteria for Fracture Prevention The rule does not specify application only to ferrite or low-alloy steel; however, reference is made to Appendices G and H. Alternatives may be granted to Appendices G and H through 50.60(b). As neither 10 CFR Appendix G nor H provide support for SA-965, FXM-19 material, exemptions and proposed alternative treatments will be necessary via 10 CFR 50.60(b). 10 CFR 50.61 – Fracture toughness requirements for protection from pressurized thermal shock events The rule does not specify application only to ferritic or low-alloy steel; however, the embrittlement trend curves within the rule are fit to low-alloy data that does not apply to the proposed SA-965, FXM-19 material. Exemption would be needed, and a technical basis provided addressing the topics handled for ferritic steels in 10 CFR 50.61. Note – 50.61a may not be used due to 50.61a(b). Appendix G – Fracture Toughness Requirements Appendix G explicitly relates to ferritic materials and does not support determinations concerning fracture toughness in GDC. Stainless steels have superior ductility and are also subject to some thermal and neutron embrittlement effects. A technical basis considering thermal and embrittlement effect will be necessary for the staff to make determinations regarding whether the SDA meets GDC 14, 15, and 31. Appendix H – Reactor Vessel Materials Surveillance Program Requirements Appendix H explicitly relates to ferritic materials and does not support determinations made concerning material condition for stainless steels. An alternative should be provided. SSs, have superior ductility, and are subject to some thermal and neutron embrittlement effects. A technical basis considering thermal and embrittlement effect will be necessary for the staff to make determinations regarding whether the SDA meets GDC 14, 15, and 31.
5	5.3.1	Section 5.3.1.5 contains no comparison of SS properties and the low-alloy steel components. Discussion should indicate how SS portion of reactor pressure vessel (RPV) is appropriately bounded by (alternative?) Appendix G and low-alloy requirements. This is necessary to demonstrate compliance with GDCs.
5	5.3.1	Section 5.3.1.6 indicates that no surveillance program is necessary as NRC requirements only specify requirements for ferritic materials. This does not appear to be correct as 10 CFR 50.60 does not specify only ferritic materials, and GDCs cannot be met without some form of justification and an exemption. Operating experience concerning the aging (thermal and neutron induced) of SA-965 FXM-19, for example, is extremely limited.

5	5.3.2	Section 5.3.2 only addresses requirements for ferritic materials and makes no justification that these will adequately bound the SS portions and any potential thermal aging.
5	5.3.2	Section 5.3.2.3 contains no comparison of SS properties and the low-alloy steel components. Discussion should indicate how SS portion of RPV is appropriately bounded by 10 CFR 50.61 ferritic requirements and whether thermal aging considerations should apply or have been considered. This is necessary to demonstrate compliance with GDCs.
5	5.3.2	Section 5.3.2.4 does not contain any discussion of thermal effects of aging on components and provides no basis to establish the conclusions therein for SS components. A basis is necessary to demonstrate compliance with the GDCs.
6	6.1.1	The NuScale SDA US460 Final Safety Analysis Report (FSAR) Section 6.1.1 indicates that F6NM (Type 415) Martensitic SS is used to fabricate the lower CNV shell above the RPV flange elevation, the lower flange, upper flange, upper shell, and top head. This material is permitted by way of ASME Code Case N-774 Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4540 kg) and Otherwise Conforming to the Requirements of SA-336/ SA-336M for Class 1, 2 and 3 Construction Section III, Division 1. Code Case N-774 is listed in RG 1.84, Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Revision 39, December 2021, as acceptable without conditions. This is a substantial change from low-alloy steel used in the previously approved NuScale design. The NRC staff is unaware of any pressure vessels fabricated from martensitic SS being used in any nuclear power plant in the United States or abroad, in safety-related or nonsafety-related systems. In addition, this material is not typically used to fabricate large pressure vessels in any industry although there may be certain applications that the NRC staff is not aware of. There is no nuclear operating experience with this material in its intended application and environment. The draft FSAR Section 6.1.1 does not provide any description of the acceptability of this material for its intended application. In addition, due to the potential difficulty in fabricating pressure components using this material, the staff is concerned about through thickness impact properties of weld heat affected zones. The use of this material would represent manufacturing challenges as it can be difficult to weld and is highly sensitive to hydrogen cracking and thus special controls during welding will be needed (Appendix B to 10 CFR Part 50). The applicant needs to provide information regarding the justification for the use of this material in its intended application, a description of the special controls that will be used during welding that will ensure that excessive repairs will not be required which could results in potential degradation issues during the life of the CVN, and a description of how through thickness impact properties will be assured. The NRC staff notes that there are no exemptions for post-weld heat treatment, by way of using preheat, regardless of the thickness of the weld. As a result, ALL welds regardless of size require a preheat and postweld heat treatment to be performed.

6	6.3	Supporting information for the Emergency Supplemental Boron (ESB) addition system related to performance and operability is missing from this section. Boron dissolution testing and operating requirements related to the ESB should be presented or referenced in this section. Section 6.3 is missing a detailed description of the ESB system design (not just the ESB function description) or reference to the system design description.
6	6.3	Section 6.3 is missing discussion or reference to information related to the operability of ESB considering potential dose to operators during outages and impact of dose on the system. The boron pellets of ESB need to be reloaded during each outage when the containment water is drained below the basket elevation. The drainage of water in the containment could result in higher dose rate than that with a flooded containment.
6	6.3	The ECCS description provided in SDA Section 6.3 is missing inclusion of the decay heat removal system (DHRS) being part of the ECCS, and associated portions of the secondary side, although it is relied upon to mitigate the consequences of a loss-of-coolant accident (LOCA) through removing decay heat and maintaining core cooling. FSAR Table 3.2-2 of the SDA has not been updated to classify DHRS and relevant secondary side steam system as ASME Code Class 1 components consistent with all other ECCS components. If the DHRS system function is as described above, it needs to meet General Design Criteria 2, 14, 27, 30, 31, 50, 51, 52, and 53. The relevant information would need to be added in Sections 3.2.2, 3.5, 3.6, 3.7, and 4.6.2.
6	6.2.1/ 6.2.2	Even though the draft SDAA identifies the design-basis events conditions, the modeling details of the limiting events are referenced to the LOCA methodology topical report cited as Reference 6.2-2. Now, Tables 6.2-6 and 6.2-7 have credited DHRS actuation in the sequence of both the peak containment pressure and wall temperature events, but no further information is provided on the DHRS T/H now credited to the revised safety analyses. The staff expects the relevant DHRS modeling and testing details to be available in the future licensing topical reports, as appropriately revised for the SDAA. The staff will also review the potential deterioration of DHRS performance due to the transport of radiolytic gases from the core and accumulation in the DHRS, and its adverse impact on in-tube condensation.
6	6.2.1	Table 6.2-1 has documented the revision of the containment internal design pressure from 1050 to 1200 psia, and the containment design temperature from 550 F to 600 F. Due to the revision of these most important containment design parameters by the containment material change to Code Case N-774 SA-336, Gr F6NM, a reference needs to be included in Chapter 6 for the [technical/topical] report where the technical justifications will be provided for revising these containment design parameters. The staff would also need the material and thermophysical properties of the revised material, as used by the applicant in the updated design and safety analyses.
6	6.2.1	Section 6.2.1.3.5 has a brief description of the long-term cooling (LTC) model. It states, “The previously described methodology and model are utilized for this purpose. This demonstrates adequate long-term containment removal.” The applicant needs to include either a justification or a [technical/topical] report reference in Chapter 6 for making this conclusion.

6	6.2.1/ 6.2.2	The SDAA CNV is a passive design that relies on a passive autocatalytic recombiner (PAR) to preclude combustible gas mixtures by limiting the oxygen concentration in the containment atmosphere to a level (<4% by volume) that does not support combustion for at least 72 hours. The prescribed 4% oxygen limit would also correspond to an 8% volume concentration limit for radiolytic hydrogen, which means an overall 12% non-condensable presence by volume in the containment during the event. The staff recognizes that oxygen and hydrogen generation from radiolysis is a slow process that may not impact short term CNV peak pressure calculations since CNV peak pressure occurs early in the event. However, NuScale would need to demonstrate the LTC capacity of the containment in the presence of at least 12% non-condensables due to radiolytic gases.
6	6.2.1/ 6.2.2	The staff understands that Containment Response Analysis Methodology will now be merged into the LOCA topical report. Section 6.2.1.3 recognizes the distinct nature of containment conservatism for safety analysis and identifies its various elements for maximizing the mass and energy release and minimizing the performance of the containment heat removal systems. The staff expects the consolidated LOCA topical report to appropriately capture the different levels of conservatisms for the containment and RPV safety analyses, which would result into two different sets of NRELAP5 decks. The staff also expects NuScale to have justifications for its proposed approach in the draft NuScale SDAA Chapter 6.
6	6.1.1	Section 6.1.1 does not provide any detail regarding dissimilar metal welds. Dissimilar metal welds (DMWs) are more susceptible to degradation inservice than similar metal welds. The staff would most likely request additional information on this subject if the information is not included in the FSAR, particularly for DMWs involving F6NM martensitic SS. The staff would expect description of all DMWs.
6	6.2.7	Section 6.2.7 includes discussion presuming that the upper CNV is ferritic, however F6NM as proposed in Section 6.1.1 is a martensitic specification. Consequently, text in Section 6.2.7 does not appear to have been updated consistent with Section 6.1.1. Consequently, the staff have no basis to provide substantive comments on Section 6.2.7.
6	6.2.7	Reference is made to Section 3.1.5 in Section 6.2.7 regarding adequacy of fracture toughness margin; however, Section 3.1.5 was not made available to the staff for the preapplication review. Consequently, the staff have no alternative basis to provide substantive comments on Section 6.2.7.
6	6.2.1.1.3/ 6.2.2	Section 6.2.1.1.3 states that both containment limiting peak pressure and temperature calculations assumed “no single active failure.” However, Section 6.2.2 states that the evaluation of mass and energy released into the containment following a postulated main steam-line break or feedwater line break considered “the automatic isolation of the main steam and feed water lines and single failures.” Please clarify considering that Principal Design Criterion 35 requires “Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.”

6	6.3	The SDAA should include drawings of the new key design features (e.g., ECCS flow restricting venturis, ESB feature) depicting details for location, layout and configuration.
6	6.3	SDAA Section 19.3 states that the passive safety functions of the nuclear power modules (NPMs) are relied on for 7 days following a design-basis event as justification to not need regulatory treatment of nonsafety systems (RTNSS) B SSCs. Accordingly, this 7-day design feature of the ECCS should be defined and described within SDAA Section 6.3.
9	All	The application makes many references to other sections that are not available for the staff to review and confirm conformance. For example, reference to Section 9.2.5 for the design of the spent fuel pool (SFP) level instrument, for the safety water levels for shielding and cooling, and accident scenario thermal evaluation.
9	9.1.3	Section 9.1.3.3.4, "Residual Heat Removal," presents how the cooling and purification system conforms with GDC 61. This section includes a discussion of different heat load calculations. The staff identified that most of the information provided is not applicable to SDAA but is copied from the DCA. For example, the section includes the exact same heat loads discussed in the DCA, references to 12 NPMs, and the calculations of rate of water loss based on the DCA ultimate heat sink (UHS) water volume and heat loads and not the SDAA, which was not provided.
9	9.1.2	SRP Section 9.1.2(iii)(1), which indicates that the minimum SFP storage capacity should equal or exceed the amount of spent fuel from 5 years of operation at full power plus one full-core discharge. The application indicates that the SFP has capacity for 10 years, but there is no minimum number of assemblies to achieve this capacity or maximum capacity credited in the thermal analysis.
9	9.2	Section 9.2.2.1, "Design-Basis," of the SDA does not specify or identify any specific functions that the system is designed to perform. In the DCA it stated that the reactor component cooling water system (RCCWS) is designed to remove the heat load from the control rod drive mechanisms (CRDMs), the chemical and volume control system (CVCS) nonregenerative heat exchangers, the containment evacuation system (CES) condensers and vacuum pumps, and the process sampling system (PSS) coolers and temperature control units (TCUs) during normal plant operation.

9	9.2	In Section 9.2.2.2, "System Description," no description of the system is provided. The system was described in the DCA, but the SDA does not contain a description of the system. In the DCA, system reference was made to the system diagram and major system components were identified and discussed. The RCCWS is a shared system and information on what SSCs are common to the shared system, and which are module specific should be identified. Also, the capability to isolate modules or portion of the system from unaffected portions of the system in the case of accidents or leaks should be included.
9	9.2	The RCCW system drawing (Figure 9.2.1) and the RCCW Equipment Design Data (Table 9.2.2-1) included in the DCA, were not included in Section 9.2.2 of the SDA. The RCCW system is located inside the RXB, and provides cooling for the CRDMs, CVCS heat exchanger, CES condenser and vacuum pumps, and the PSS coolers and analyzer cooler TCUs. Because the RCCW system interfaces with systems like the CVCS and CRDM, pipe breaks and system failures could impact certain Chapter 15 events (See FSAR Section 15.1.6). The RCCW design parameters in Table 9.2.2.1 are used as the bases for the input parameters and initial conditions in FSAR Section 15.1.6. Removal of this information would result in information gaps including unsupported inputs into a chapter 15 event evaluation.
9	9.2.5	The different thermal evaluations that determine key design parameters of the UHS have not been finalized. until the thermal evaluations are completed and reviewed by the staff, they cannot determine that the SDAA adequately addresses all key design parameters.
9	9.2.7	The site cooling water system (SCWS) performs many of the functions that are performed by service water systems at other pressurized-water reactors. As such, the staff review of the SCWS uses relevant guidance in SRP Section 9.2.1. In accordance with SRP Section 9.2.1, Section 1.2.A, the design is reviewed for the capability to detect, control, and isolate system leakage including the capability to detect and control radioactive leakage into and out of the system and prevent accidental releases to the environment. Draft SDAA Sections 9.2.7.1, “Design-Basis,” and 9.2.7.2, “System Description,” do not identify the plant auxiliaries cooled by the SCWS nor do they provide information on the functional arrangement of the SCWS. This information was included in the DCA and used to verify statements concerning compliance with GDC 2, 4, 60, and 64.
9	9.2.7	In the DCA the spent fuel pool and reactor pool heat exchanger are cooled by the SCWS. When the reactor is shutdown, the decay heat from the DHRS heat exchangers is deposited in the reactor pool. Information related to the role that the SCWS plays in normal operation and safe shutdown (SSD) of the plant was discussed in Section 9.2.7.2 of the DCA but was not included in the SDA. This information should be included in the SDAA so that the staff can review and verify it.
9	9.5	Revised layout and location of CRB and radioactive waste building (RWB). Does this impact firefighting plan or post-fire SSD access/egress route?

9A	9A.5.4.1	Removal of the remote shutdown station - Clarify statement that for safe shutdown “no operator actions are required outside of the [main control room] (MCR)” (i.e., no need to monitor or control anything once the manual switches in the MCR are activated?)
9	9.1.5	In Section 9.1.5.2.3 of the SDA, it states that Heavy Load Exclusion Zones are defined in operating procedures and equipment drawings. Physical Limits and Administrative Controls are included to ensure safe handling of critical loads. In Section 9.1.5.2.3, the staff did not find any discussion of the physical limits that reduce the chance of operator errors such as over lift, over travel, over speed, etc. In the DCA, FSAR Table 9.1.5-1: Heavy Load Handling Equipment Design Data documents boundaries for the RXB crane. There was also a discussion of limit switches in DCA FSAR Section 9.1.5. In this SDA application, the staff noted that this RXB Crane is still nonsafety-related but risk significant.
9	9.3.4	Paragraph on Module Heatup System (MHS) to Auxiliary Boiler System (ABS) leakage radiation monitors has been deleted. In the DCA, on a high-high alarm the monitor initiated a control room alarm and closure of the valves for the affected MHS heat exchanger. Note that Table 11.5-1 still discusses the monitors, but it is unknown if the table in Section 11.5 is accurate. Section 11.5 does not discuss the monitors purpose or function. Section 10.4.8, which discusses the ABS, states that Section 11.5 discusses radiation monitoring in the ABS but does not provide any additional information regarding the monitors. If the monitors remain in the design, a description of their purpose and functions should be provided. If they are no longer included; a description of how the design prevents/limits contamination of the ABS should be provided.
10	ALL	It was noted from discussions with NuScale that the SDAA will be a standalone application, and that much of the documentation made available was being further developed. During the Readiness Assessment review of Chapter 10 systems the NRC staff found the information regarding the design and operation of the systems was insufficient to permit understanding of the system design and the potential impact of the system operation or failure on plant safety or the potential for radiological release. Chapter 10 covers the steam and power conversion system, which differs appreciably from the NuScale DCA in that the system now must support a nuclear module that has significantly higher energy output (54 percent higher than currently approved design), and incorporates air-cooled condensers into the system design, which is the first such application of an air-cooled condenser to serve as the main condenser for a nuclear power plant. The information provided to NRC staff for the Readiness Assessment for the air-cooled condenser system (ACCS) in Section 10.4.1 does not sufficiently define the ACCS design that is to be used in plant’s power conversion system. Design information (system design requirements, drawings, design data and performance requirements) for the ACCS was not included in Section 10.4.1. While detailed information on the condenser design is not needed, general information such as a simplified ACCS diagram, design information on the major components, and design performance requirements should be included. Additionally, the information provided for Turbine Bypass System, Condensate and Feedwater System, and ABS does not sufficiently define the systems and it is noted that design information for these systems was included in the DCA, but information was removed in the current SDAA.
10	10.1	Information in the DCA that provided an understanding in broad terms of the steam and power conversion system design and operation is not in the current SDAA. The information in Table 10.1 is available in Section 10.3, but heat balances diagrams (Figure 10.1-2 in DCA) should be included in Section 10.1 of the US460 SDAA.

10	10.3-1	Table 10.3-1 removed design information (including secondary main steam isolation and main steam bypass valves closure times). Although not safety-related, the secondary main steam isolation valves (MSIV) are credited as backup protection for the safety-related MSIVs and were included in the technical specifications (TSs) in the DCA.
10	10.3.2.1	Is the information for the condensate drain still valid for the air-cooled condenser which will be located outside the turbine building?
10	10.3	It appears all discussion of extraction steam has been removed. Extraction steam is the heat source for the feedwater heaters. What is the reason for removal?
10	10.4	Section 10.4 gives the design-basis for the air-cooled condensers which will be located outside the turbine building. The turbine exhaust will be transported to the condenser, and condensate returned from the condenser via piping external to the turbine building. GDC 4 is the only GDC identified. Failure of the system or its components could lead to the release of radiation directly to the atmosphere, GDC 2, 4, and 64 may need to be considered, as well as 10 CFR 20.1406.
10	10.4.6.5	There is no mention of temperature control in the section on instrumentation. As with the system components in general, this makes it hard to conclude the system is designed to protect the resin from high-temperature.
10	10.4.7.3	It appears that GDC 60, 64, and 10 CFR 20.1406 were removed from the design-basis. In general, GDC 60 and 64 may not be applicable to a condensate and feedwater systems enclosed in the turbine building, however parts of the condensate and feedwater system might be external to the Turbine Building and GDC 60 and 64 should be considered, or an explanation why they do not apply should be included. 10 CFR 20.1406 applies to the condensate and feedwater system and should also be addressed in the SDA. These items were previously addressed in the DCA.

10	10.4.9.3	Section 10.4.9.3 only addresses startup. Should it also address operations? If not, why not?
11	11.4	Is storage of packaged radioactive waste in the RWB in the above grade portions of the building? If not, where will packaged radioactive waste be stored?
11	11.4	The draft SDAA removed some design feature discussions related to Branch Technical Position 11--3. Examples are: (1) Components and piping that contain radioactive slurries should have flushing connections and piping runs that minimize the number of bends and traps that may retain radioactivity and lead to increased ambient external radiation exposure rates; and (2) Discussions related to the venting of storage tanks. Will this information be in Chapter 12?
11	11.4	Section on component descriptions was removed from the draft SDAA. Table 11.4-1 and draft SDAA 11.4 figures inform the reader of the tanks and pumps part of the solid radioactive waste system, but not having descriptions of each component causes draft SDAA Section 11.4 not to address some of the as low as reasonably achievable (ALARA) design features involved with the design of the cubicles the tanks are in, features of pumps as they relate to the movement of resin between tanks and storage containers, as well as any accompanying assumptions that would be used for a dose analysis such as decay times prior to packaging.
11	11.4	Not many details are provided on the dewatering system. The document stated that it is a modular system, but details on the purpose of this system should be stated consistently as information stated in the DCA. Will it be a part of the SDAA design?
11	11.4	Instrumentation details are removed from draft SDAA Section 11.4. There are Section 11.5 pointers back to this Section 11.4 but no information on the monitoring equipment is provided. Will this information be provided in Section 11.4 in the SDAA?

11	11.5	DCA Section 11.5 provided: (1) information on the sources of radioactivity being monitored, (2) locations of monitors, and (3) purposes of monitors. In draft SDAA Section 11.5 these details are either removed or moved to another section that are not available at this point of the readiness assessment. Draft SDAA Section 11.5 figures and tables are not available for review currently, although information on ranges, locations and function were previously found in Tables 11.5 for the DCA.
11	11.2	Information related to the components details that make up the Liquid Radioactive Waste System (LRWS) are not present. Details related to what could be understood as ALARA design features are no longer discussed. Examples of this would be information on having sufficient volumes in tank cubicle rooms to prevent unmitigated release, tank venting when overfilling the tank, valves and pipes with backflow protections, pumps being in shielded cubicles and being well ventilated.
11	11.2	Discussion of the LRWS in Section 11.2 is not as detailed as what was provided in the DCA and not as detailed as the discussion of the Gaseous Radioactive Waste System. Details that were provided in DCA Section 11.2 on the cleanup functions of the Low Conductivity Waste and High Conductivity Waste processing skids are not in the draft SDAA. Limited details are provided on how NuScale plans to process liquid waste (for example - will NuScale use Cation, Anion, Mixed Bed, and Cesium demineralizers?). Only brief mentions of the skid-based processing equipment are made in draft SDAA Section 11.2.2. More information is needed on how NuScale plans to process liquid waste.
11	11.2, 11.3, 11.4	Based on the information currently provided, one cannot determine what is above and below grade for the RG 1.143 assessments. The purpose of this would be to identify the stated boundaries of each system and where they are in the RWB. This comment is applicable to draft SDAA Sections 11.2, 11.3, and 11.4.
11	11.2	Not clear about the actual changes to the radwaste processing equipment. Cannot tell if the SDAA 11.2 changes have an impact on the liquid effluent wastes that will be released. Process flow figures were removed from SDAA Section 11.2. Will system diagrams be in the official submittal?
12	12.4	Table 12.4-7. The dose estimates for refueling activities were already low in the DCA and the doses got even lower in the SDAA, and it is not clear what caused the significant decrease in dose. It states that the dose for disassembly, lower containment work, refuel work, and reassembly are all 0.0 man-rem and that there is no exposure time for these activities. Staff understands that some activities can be performed remotely, but all of these items resulted in some dose in the DCA. It is unclear if there are any design changes resulting in lower expected doses during refueling activities. In addition, it is unclear how there can be no dose for some of these activities. For example, how can work be performed on the lower containment without receiving any expected dose?

15	15.0.5	{{ }}
15	15.9	{{ }}
15	15.0.5	{{ }}
15	15.0.5	{{ }}
15	15.0.5	{{ }}
15	15.0.5	{{ }}

15	15.4.6	{{ }}
15	15.6.5	{{ }}
15	15.0.0.6.5	{{ }}
15	15	{{ }}

15	15	{{ }}
17	17.4	As discussed in SRP Section17.4 “Reliability Assurance Program (RAP),” Revision 1, the RAP applies to those SSCs, both safety-related and nonsafety-related, identified as risk significant (or significant contributors to plant safety). Section 17.4 of the SDA references the staff approved TR-0515-13952-NP-A “Risk Significance Determination,” Revision 0. To use a TR, an applicant needs to demonstrate that all conditions of use are either met or justify deviations. Therefore, the SDA needs to include a discussion on how each condition of use for the TR is met, including Condition 2, which identifies the consideration of PRA uncertainties and sensitivity assessment in risk significance determination. This discussion can be either in SDA Section 17.4 or elsewhere with a pointer to the location in SDA Section 17.4. The staff provided a similar comment during the June 2022 public meeting on the use of this TR for the SDA application (ML22195A049).
19	19.1	In SDAA Section 19.1.4.1.9, Risk Significance Determination, the section references the staff approved topical report, NuScale Power, LLC, "Risk Significance Determination," TR-0515-13952-NP-A, Revision 0. Section 6 of the TR includes limitations and conditions on its use, including Condition 2, which states (emphasis added), “the ultimate determination of risk significance shall be based on the specific application, with appropriate consideration of uncertainties, sensitivities, traditional engineering evaluations and regulations, and maintaining sufficient defense-in-depth and safety margin.” a. To use a TR, an applicant needs to demonstrate that all conditions of use are either met or justify deviations. SDAA Chapter 19 does not address how all the limitations and conditions on the topical report are met for this application. Therefore, the SDAA needs to include a discussion on how each condition of use for the TR is met. A similar observation was provided as part of the readiness assessment for Section 17.4. b. Per a (closed) public meeting in June 2022, NuScale has changed the numerical criteria to identify candidate risk significant SSCs for the SDAA from the approved topical report. Further, NuScale considers this change to be consistent with Limitation and Condition 3 on the TR. This represents a significant change to the SDAA compared to the DCA. The SDAA does not provide any: discussion of the approach for the changes to the numerical criteria, justification for the changes to the numerical criteria, and consistency of the changes with Limitation and Condition 3 on the TR. Such information is necessary to determine the technical acceptability of changed numerical criteria and its consistency with the approved TR.
19	19.1, 19.2	Consistent with §52.137(25) and the 1985 Commission Severe Accident Policy Statement, the staff reviews the results and insights of the PRA to assure no undue risk to public health and safety. SDAA Chapter 19 provides scant and distributed information regarding the ECCS supplemental boron injection system that actuates within 4 hours of ECCS actuation and is referenced in SDAA Section 6.3.2.2.1. This system represents a significant change to the SDAA compared to the DCA. SDAA Section 19.2.3.1 mentions that following an ECCS actuation, boron in the CNV is recirculated in the reactor core to assure shutdown margin under cold conditions. On SDAA page 19.1-40, it is mentioned that return to power only occurs once after passive heat transfer to the UHS is established. No further context or information is provided. The staff could not find an assessment of this system in Chapter 19, including its risk significance determination. This supplemental boron injection system is referenced in TSs under Limiting Condition for Operation 3.5.4. The staff could not find a reference to this system in Chapter 17.4, the design reliability assurance Program (DRAP). Because the supplemental boron injection system represents a significant change to the SDA design compared to the DCA design and impacts the progression of certain scenarios, the SDA needs to include a discussion of boron redistribution following ECCS actuation and the risk significance of the supplemental boron injection system in Chapter 19. It is recommended that either a similar discussion is provided in Section 17.4 of the SDAA or a pointer to Chapter 19.

19	19	There are 10 references to the IAB valves in Chapter 19 which often do not differentiate between ECCS RVVs, which have the IABs removed, and the Reactor Recirculation Valves, which still have the IABs, according to SDAA Section 6.3. Chapter 19 of the SDAA needs to identify the design changes to the ECCS valves clearly and explicitly to support the staff's review of the impacts of these changes on the risk insights and results in Chapter 19.
19	19.2, 6.2.5	52.137(a)(23) requires light-water reactor designs to provide a description and analysis of design features for the prevention and mitigation of severe accidents. SDAA FSAR, Revision A, Section 6.2.5 includes selected updates related to equipment survivability and hydrogen combustion. Observations on both are provided below. <u>Equipment survivability</u> The post-accident monitoring (PAM) function to monitor the hydrogen and oxygen concentrations in containment will no longer be based on utilizing the CES and the containment flooding and drain (CFD) systems as described in the NuScale DCA. SDAA FSAR, Revision A, §6.2.5, Combustible Gas Control, explains that the PAM function using CES and CFD is to be removed by an Exemption Request. Since the CES and CFD would no longer be needed for their severe accident PAM mitigation function, SDAA Section 19.2.3.3.8, Equipment Survivability, including Table 19.2-10, should be updated to reflect this design change. In addition, the topical or technical reports which support the SDAA Equipment Survivability design should be identified, and their current applicability confirmed. <u>Hydrogen combustion</u> SDAA FSAR, Revision A, Section 6.2.5.1, Combustible Gas Control, states that the NuScale design adds PARs which support exemptions from (i) the hydrogen control requirement of 10 CFR 50.44(c)(2) to limit oxygen concentrations in containment to less than 4 percent, (ii) from 10 CFR 50.44(c)(4)(ii) for the hydrogen and oxygen monitoring equipment to be functional and capable of continuous monitoring following a significant beyond design-basis accident, and (iii) from 10 CFR 50.44(c)(5) to provide a hydrogen burn containment analysis. It is unclear whether DCA Technical report TR-0716-50424-P, "Combustible Gas Control," is still applicable. If it is, the technical report should be revised to reflect the SDAA fuel design and added PARs. If the technical report is not applicable, the SDAA should include discussion of the technical evaluation using PARs and not carry references to the technical report.
19	19, 17.4, 3, 16	Consistent with §52.137(25) and the 1985 Commission Severe Accident Policy Statement, the staff reviews the results and insights of the PRA to assure no undue risk to public health and safety. The staff could not find a reference in Chapter 19 for the PARs which is a new system in the SDAA compared to the certified design. SDA Section 6.2.5 states that the PARs are designed to limit oxygen concentrations such that combustion does not occur following a severe accident that releases an equivalent amount of hydrogen following a 100% fuel clad- coolant interaction. At a minimum, a discussion of the PARs evaluation should be presented in: a. The Level 2 PRA to reflect the capability of the PARs to minimize the potential for hydrogen deflagration and detonation. b. Chapter 17.4 for risk significance under the DRAP. c. Chapter 19.3 under the RTNSS program. d. Criterion 4 of 10 CFR 50.36 to see if the PARs should be considered for TSs (under risk significance or operating experience). e. Chapter 3 of the SDAA for the appropriate seismic classification.

19	19, 6	10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” specifies that this includes Type A testing. NuScale DCA requested and received an exemption to not perform a Type A leakage test, and to not require design capability to conduct this periodic leakage test. The staff’s approval of this exemption was significantly based on the results in NuScale technical report, TR-1116-51962-P, Revision 1, “NuScale Containment Leakage Integrity Assurance.” A significant increase in the containment design pressure is among the SDAA design changes. If the SDAA includes an Exemption Request similar to that identified above and the basis for the exemption relies on NuScale technical report, TR-1116-51962-P, Revision 1, the application needs to (i) justify that the technical report and its results are applicable in their entirety to the SDAA design, including relevant design changes compared to the DCA, and (ii) justify that all limitations and conditions of use on the technical report are met for the SDAA design. If the SDAA includes an Exemption Request similar to that identified above and the basis for the exemption does not rely on NuScale technical report, TR-1116-51962-P, Revision 1, a discussion of the new basis should be provided in the SDAA, with detailed information available for staff audit.
19	19.3	Section 19.3.2.2 states the core cooling and containment functions are performed by the safety-related SSCs for an extended period following an accident. Section 19.3 should include, or provide a reference to, the evaluation that demonstrates the capability of the facility to sustain all design-basis events with onsite equipment and resources for 7 days consistent with SECY-96-128.
19	19.2	The power uprate to 250 MWth from 160 MWth and other design changes is expected to impact all stages of the Severe Accident Evaluation, from identified accident sequences, derived source terms, and radiological consequence analyses. However, it appears all Section 19.2.3.2, “Severe Accident Progression,” analyses remain unchanged (also see applicable corresponding Tables 19.2-1 through 19.2-10 and Figures 19.2-4 through Figures 19.2-7).