NuScale Power™

LO-1216-52419

December 31, 2016

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application (NRC Project No. 0769)

REFERENCES: 1. NRC Letter to NuScale, "Detailed Pre-Application Readiness Assessment Observations of the NuScale Power, LLC Design Certification Application," dated November 3, 2016 (ML16287A718).

NuScale Power, LLC (NuScale) is pleased to submit to the U. S. Nuclear Regulatory Commission (NRC) Revision 0 of the NuScale Standard Plant Design Certification Application (DCA) in accordance with the requirements of 10 CFR 52, Subpart B, Standard Design Certifications. NuScale requests NRC review, approval, and granting of standard design certification for the NuScale Standard Plant Design.

The NuScale DCA is the first such application for a Small Modular Reactor (SMR). It describes in detail the unique performance and safety attributes of the NuScale design. The reactor coolant system uses simple properties of physics: convection, conduction and gravity to provide unprecedented protection for the public and the environment.

A nuclear power plant using NuScale's technology comprises a number of individual NuScale Power Modules™, each producing 50 megawatts of electricity (gross) with its own factory-built, combined containment vessel and reactor vessel, and its own dedicated turbine-generator set. A power plant can include as many as 12 NuScale Power Modules to produce as much as 600 MWe, gross (570 net, nominal, after house loads). NuScale power plants are scalable - additional modules can be added incrementally over time to match electrical load growth.

The NuScale DCA is the culmination of efforts by more than 800 highly trained, specialized staff and over 50 vendor partners. It is the result of two million staff hours invested in design, testing and licensing activities spanning 8 years. During this time NuScale has placed over 1,000 documents on its docket with the NRC and leveraged over 40,000 NRC staff hours to discuss NuScale design information and resolve technical and policy issues so the NRC would be ready to receive the first SMR design certification application for review.

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NRO

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

The NuScale DCA includes the following parts:

Part 1, "General and Financial Information" – Part 1 provides general and financial information in accordance with the requirements of 10 CFR 52.46 and pursuant to 10 CFR 50.33(a) through (c) and (j).

Part 2, Tier 1 and 2, "Final Safety Analysis Report" – Part 2, Tier 1 and 2 provide the NuScale Final Safety Analysis Report in accordance with 10 CFR 52.47(a).

Part 3, "Applicant's Environmental Report - Standard Design Certification" – Part 3 provides the NuScale Environmental Report in accordance with 10 CFR 52.47(b)(2) as required by 10 CFR 51.55.

Part 4, "Technical Specifications" – Part 4 provides the NuScale Technical Specifications and Bases in accordance with 10 CFR 52.47(a)(11); the Technical Specifications and Bases are referenced in Part 2, Tier 2 Chapter 16.

Part 5, "Emergency Plans" – Emergency plans are not applicable to design certification applications and, therefore, not provided in this application.

Part 6, "Security Plans" – Security plans are not applicable to design certification applications and, therefore, not provided in this application.

Part 7, "Exemptions" – Part 7 provides exemption requests necessary to accommodate the unique safety features of the NuScale SMR.

Part 8, "License Conditions; Inspections, Tests, Analyses & Acceptance Criteria" – License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC) have been developed in accordance with 10 CFR 52.47(b)(1). The ITAAC are located in Chapters 2 and 3 of Part 2, Tier 1 and are based on information found in Part 2, Tier 2. The description of the process used to develop the ITAAC is provided in Part 2, Tier 2, FSAR Section 14.3. License conditions are not applicable to the design certification applications.

Part 9, "Withheld Information" – Part 9 identifies the location of security-related information within the FSAR that is to be withheld from public disclosure in accordance with 10 CFR 2.390.

Part 10, "Quality Assurance Program Description" – In accordance with 10 CFR 52.47(a)(19), NuScale Topical Report NP-TR-1010-859-A, Revision 3, "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant (QAPD)" is referenced in Part 2, Tier 2, FSAR Section 17.5.

In support of the DCA, NuScale has developed Topical and Technical Reports that provide supplementary information, data and analyses. These reports are referenced in NuScale Final Safety Analysis Report (FSAR). Attachment 1 to this letter provides a comprehensive list of referenced Topical and Technical reports.

In addition, from September 19 to September 29, 2016, the NRC performed a pre-application readiness assessment (readiness assessment) of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal to and subsequent review by NRC Staff (Reference 1). NuScale is providing information in the DCA that: 1) closes gaps identified between the draft DCA and technical content generally expected by the NRC; and 2) resolves identified technical or policy issues which might have adversely impacted acceptance, docketing, or technical review of the application.

Attachment 2 to this letter provides a summary of the disposition of each of the potential docketing items identified during the readiness assessment.

The DCA is being submitted electronically, as Enclosures 1 and 2, on two digital versatile discs (DVDs).

- Enclosure 1 (DVD 1) includes certain information, designated in accordance with NRC guidance as sensitive unclassified nonsafeguards information (SUNSI) and referred to as security-related information (SRI), that is to be withheld from public disclosure under 10 CFR 2.390.
- Enclosure 2 (DVD 2) omits the SUNSI/SRI mentioned above and is suitable for public disclosure.

Please contact Thomas A. Bergman, Vice President, Regulatory Affairs at 541-360-0740 if additional information is necessary for successful completion of NRC's technical review of the NuScale Standard Plant Design Certification Application.

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

Sincerely,

Dale K. Atkinson
Chief Operating Officer/Chief Nuclear Officer
NuScale Power, LLC

Thomas A. Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC

Attachment 1:	Topical and Technical Reports Supporting the NuScale DCA
Attachment 2	Itemized Disposition of NRC Readiness Assessment Observations
Enclosure 1:	DVD 1, NuScale Standard Plant Design Certification Application | Non-Public Version–Withhold Under 10 CFR 2.390
Enclosure 2:	DVD 2, NuScale Standard Plant Design Certification Application | Public Version – NuScale Nonproprietary

cc: Vonna Ordaz, Acting Director Office of New Reactors (NRO)

Attachment 1: Topical and Technical Reports Supporting the NuScale DCA

NuScale Topical Reports		
Report #	**Report Title**	**Submittal Date**
NP-TR-1010-859	NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Reactor, Rev. 3	03/24/2016
TR-0515-13952	Risk Significance Determination	07/30/2015
TR-0815-16497	Safety Classification of Passive Nuclear Power Plant Electrical Systems	10/29/2015
TR-0915-17565	Accident Source Term Methodology, Rev. 1	04/08/2016
TR-0116-20825	Applicability of AREVA Fuel Methodology for the NuScale Design, Rev. 1	07/01/2016
TR-1015-18653	Design of the Highly Integrated Protection System Platform Topical Report, Rev. 1	11/04/2016
TR-0516-49417	Evaluation Methodology for Stability Analysis of the NuScale Power Module	07/31/2016
TR-0616-48793	Nuclear Analysis Codes and Methods Qualification	08/30/2016
TR-0716-50351	NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	09/30/2016
TR-0116-21012	NuScale Power Critical Heat Flux Correlation NSP2	10/05/2016
TR-0915-17564	Subchannel Analysis Methodology	10/31/2016
TR-0516-49422	Loss-of-Coolant Accident Evaluation Model	12/30/2016
TR-0716-50350	Rod Ejection Accident Methodology	12/30/2016
TR-0516-49416	Non-Loss-of-Coolant Accident Analysis Methodology	01/10/2017*

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Technical Reports		
Report #	**Report Title**	**Submittal Date**
TR-0816-50796	Loss of Large Areas Due to Explosions and Fires Assessment	12/30/2016
TR-0116-20781	Fluence Calculation Methodology and Results	12/30/2016
TR-0616-49121	NuScale Instrument Setpoint Methodology	12/30/2016
TR-0816-50797	Mitigation Strategies for Extended Loss of AC Power Event	12/30/2016
TR-0716-50439	NuScale Comprehensive Vibration Assessment Program	12/30/2016
TR-0316-22048	Nuclear Steam Supply System Advanced Sensor	12/30/2016
TR-0816-49833	Fuel Storage Rack Analysis	12/30/2016
TR-1015-18177	Pressure and Temperature Limits Methodology	12/30/2016
TR-1016-51669	NuScale Power Module Short-Term Transient Analysis	12/30/2016
TR-1116-51962	NuScale Containment Vessel Integrity Assurance	12/30/2016
TR-1116-52065	Effluent Release (GALE Replacement) Methodology and Results	12/30/2016
TR-0716-50424	Combustible Gas Control	12/30/2016
TR-1116-52011	Technical Specifications Regulatory Conformance and Development	12/30/2016
TR-0816-51127	NuFuel-HTP2™ Fuel and Control Rod Assembly Designs	01/10/2017*
TR-0916-51299	Long Term Cooling	01/10/2017*
TR-0516-49084	Containment Response Analysis	01/10/2017*
TR-0916-51502	NuScale Power Module Seismic Analysis	01/10/2017*

*scheduled submittal date

Attachment 2: Disposition Summary of Potential Docketing Issues

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
1	Tier 2 13.6.1	Reference technical report (TR)-0416-48929, "NuScale Design of Physical Security Systems," (SGI) does not provide design descriptions information sufficiently detailed to determine how design requirements will be met to comply with requirements of 10 CFR 52.48 and 52.47(b)(1)....	TR-0416-48929 has been updated to include additional detail addressing this item. Specific design details have been verified to be the responsibility of the COL applicant.
2	Table 14.2-2-14.2-33, 14.3-2	DCA Tier 2 Section 14.2 did not include information on the test abstracts for ITAAC that are appropriate and necessary for verifying physical security ITAAC.	Additional detail has been added to Tier 1 Chapter 3 "Physical Security" ITAAC which provides information relative to how the ITAAC will be verified.
3	3.2	The DCA did not include the SSC classification for the SDIS. NuScale needs to provide the SSC classification for Post-Accident Monitoring Systems and SDIS in Table 3.2-1.	The SDIS classification has been added to Table 3.2-1.
4	3.13, 4.5.1, 4.5.2, 5.2.4	The staff could not determine whether NuScale has evaluated the requirements of ASME Section XI to ensure that the examination requirements are applicable, adequate, and sufficient for the NuScale design, in particular as applied to ASME Code Class 1 piping, components, bolting, etc., and reactor internals....	Additional material was added to address this as follows: Class 1 piping and components are addressed in Table 5.2-6. (5.2.4) Reactor Vessel Internals are addressed in Table 5.2-7. (4.5.1 and 4.5.2) Bolting is addressed in FSAR Section 5.2.4.1 and Table 6.2-3. (3.13)
5	3.5.1.3	Other than a COL Item 3.5-1... NuScale did not provide a description and analysis of how SSCs important to safety (including RTNSS) will be protected from turbine missiles...to meet the requirements of GDC 4 and DSRS 3.5.1.3...	A turbine rotor integrity program has been outlined in FSAR Section 10.2.3 which will protect SSCs from turbine missiles. FSAR Section 3.5.1.3 and the COL item were revised.
6	3.6.2 3.9.1 3.9.2.5 3.9.5 App 3A 3.10 3.12 3.14 17.4.7	Level of detail and references are not adequate. NuScale will need to provide the following information in order for the staff to determine the sufficiency of the DCA for acceptance: • An outline of information needs to be included in the pipe break hazards analysis report • Additional level of detail in Section 3.9.5 and Appendix 3A. These sections should include information such as descriptions of the derivation of inputs used in the analysis, load combinations, acceptance criteria and basis, assumptions used in modeling along with the justifications for the assumptions, detailed information and sketches, descriptions of fabrication, and a description of the ASME Section III NG analysis, with a summary of results, etc. • In Section 3.12, "Description of graded approach," NuScale should address the Graded Approach and the scope for piping and list in detail what piping systems are chosen for design completion, the technical basis (screening criteria, safety significance etc.) for choosing only these systems and a discussion of the analysis results. Also discuss preliminary evaluations and to what extent these were carried out (methodology, design inputs etc.). • In Sections 3.12 and 3.10, specify the RG 1.92 revision and method for combining modal responses • Include references to applicable reports that were performed to analyze the containment vessel, reactor vessel, and reactor internals • Include D-RAP list in DCA Tier 2, Section 17.4.7 • Identify reference to all computer codes used in development of DCA information. SRP section 3.9.1 review will not be able to be completed unless this information is included...	FSAR Section 3.6.2 text was revised to include a discussion of the Pipe Break Hazards Analysis. Two new Technical Reports for a) Dynamic Analysis of the NPM and b) Short Term Transient Analysis of the NPM were developed. Additional information to provide more details was added to FSAR Section 3.9.5 and Figure 3.9.4 was revised. Graded approach is discussed in FSAR Section 3.12.1. The technical basis for choosing the systems for analysis and discussion about the results, and preliminary analyses evaluations are included in this section. FSAR Section 3.12.3 addresses the methods for combining modal responses and provides details for the application of appropriate revisions (revision 1 or revision 3). Appendix 3A summarizes the analyses for NPM (CNT, RPV, RVI). All computer codes used are identified in FSAR Section 3.9.1. FSAR Section 3.9.6 has been revised to address the NRC concerns. The FSAR Section is consistent with RIS 2000-03. FSAR Section 3.9 has a COL item for an OM Code relief request to be made within 18 months of initial fuel load, which is the OM requirement. FSAR Section 3.9.6 has been revised to discuss ASME OM Code reliefs and alternatives. FSAR Section 5.2.2.4 describes the design with basis for deviation from ASME Code. FSAR Section 5.2.2.10 discusses IST testing and reference is made to FSAR Section 3.9.6.
7	3.6.2.3 3.6.2.4	The DCA describes the Integral Shield Restraint (ISR) function as to hold the pipe in place. As required by GDC 4/SRP 3.6.2, NuScale should address the dynamic jet effect from the pipe break and the impact of the ISR on the piping analysis. NuScale should also validate that the ISR will perform its intended function (e.g., through a proof-of-concept testing)...	Additional detail included in FSAR Section 3.6.5 to address the design of the NuScale ISR. The ISR design discussion includes a summary of the qualification CFD analysis addressing the dynamic jet effect from the pipe break and impact of the ISR on piping analysis. An outline of the ISR test program has also been provided in FSAR Section 3.6.5.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
8	3.12 3.10	DCA 3.12 shows that piping utilizes damping per RG 1.61 Revision 1, 2007, which for SSE recommends 4% damping. Combination of modal responses is done either per RG 1.92 Revision 3 or RG 1.92 Revision 1 (with missing mass consideration). If more than 2% damping is used, methods of RG 1.92 Revision 1 that only combine modal responses of closely spaced modes within 10% are not appropriate...	FSAR Section 3.12.3 was revised to provide requested information and describe the use of applicable revisions (Revision 1 or 3) of RG 1.92 for combination of modal responses. FSAR Section 3.12.3.2.4 provides discussion of closely spaced modes and related requested information. FSAR Section 3.10 has been updated to identify Revision 3 of RG 1.92.
9	Tier 1	Tier 1 includes the term Class 1E and a Type A test for the containment, but Tier 2, Section 3.1 discusses exemptions related to these terms. Specifically an exemption was identified to performing the Type A test in Tier 2, yet there is an ITAAC for the Type A testing....	The information in Tier 1 has been verified to be consistent with the information in Tier 2.
10	3.7.2.11 3.8.4 3.8.5	5% additional accidental torsion was not considered in the design. NuScale should provide a quantitative basis that shows that the effect is insignificant to the design of structural members.	FSAR Section 3.7.2.11 has been updated to include a discussion of how accidental torsion was included in the analysis.
11	3.7.2, 3.7.2.9.1	Section 3.7.2 indicates that this section provides discussion about the effect on the design of operations with less than the full complement of 12 NPM models. Sufficient details should be provided in the DCA to include the results of the sensitivity studies consistent with the potential operational configurations....	FSAR Section 3.7.2.9.1 has been updated to include a discussion about operation with less than 12 modules.
12	3.7.2	NuScale needs to provide a summary in the DCA of the V&V that demonstrates appropriateness of the numerical analysis software for NuScale applications. The demonstration should test those characteristics of the software that mimic the physical conditions, the material properties and physical processes that represent the NuScale design in numerical analysis. The details of such verification can be presented via technical reports as completed to expedite review or available for audit at a later date. Some of the parameters to be addressed during this V&V process are: (a) the selection of aspect ratios, of the finite elements used in the models, such that the accuracy of the calculated results remain unaffected. Provide the maximum aspect ratios of the plate and solid finite elements used to model the structure, basemat, below-grade exterior walls, and excavated volume mesh of the embedded structures including the back fill...	FSAR Section 3.7 has been updated to address this item. The validation performed is addressed in FSAR Section 3.7.5. Information about the size of elements is provided in the "cut-off frequency" section of FSAR Section 3.7.2.1.1.3. Typical and maximum aspect ratios are provided in Table 3.7.2-1 and Table 3.7.2-9. Poisson's ratio, which is a limit for SSI and SASSI2010 is addressed in FSAR Section 3.7.1.3.2. Capacity of SASSI2010 versus our analytical needs is addressed in FSAR Section 3.7.5. Acceptability of mesh size for passing frequency is addressed in FSAR Section 3.7.2.1.1.3 "cut-off frequency".
13	3.7.2.1.1.3	NuScale needs to provide a summary in the DCA of the Benchmarking analysis that demonstrate adequacy of the use of the extended subtraction method....	FSAR Section 3.7.2.1.1.3 has been updated to provide the modeling approach and benchmarking of ESM in "Modeling Approach".
14	3.7.2.4	NuScale needs to provide a summary discussion of their review of the transfer functions, at critical sections, in the DCA...	FSAR Section 3.7.2.1.1.3 has been updated to provide the modeling approach and benchmarking of ESM. The use of transfer functions to identify anomalies is discussed in subsection entitled "Ensuring accurate results".
15	3.7.2	The narrative in the DCA should be enhanced to include the following topics for the staff to complete the safety evaluation: (a) basis of the vertical depth of the SSI model; (b) how was the effect of soil wall separation and potential uplift of the mat considered and included in the design basis-if it was not considered, what is the justification for not doing so?; (c) How was the added mass for the internal walls and columns submerged in the pool considered in the design & analysis; (d) summarized results (acceleration, displacements, soil pressure, ISRS, etc.)....	FSAR Section 3.7 has been updated to address this item. "Model dimensions" is a subsection of FSAR Section 3.7.2.1.1.3. Soil separation is a subsection of FSAR Section 3.7.2.1.1.3. How water was addressed is described in FSAR Section 3.7.2.1.2.4. Summarized results items (d), (e), and (f) are covered by FSAR Appendix 3B.
16	3.7.3.12	DCA should establish the design basis of any Category I or Category II tunnels in close proximity of the safety related structures...	FSAR Section 3.8.4.1.2 has been updated to describe the Control Building tunnel.
17	TR-0816-49833-P	The basis for concluding that any fuel assembly can be placed in any location at any time is not provided in the seismic and thermal analysis of fuel racks presented in the report....	TR-0816-49833-P was updated to provide further details on the administrative controls on the loading patterns for fuel with the same characteristics evaluated in the analysis.
18	TR-0816-49833-P	In the dynamic analysis model, the fuel assembly is not modeled explicitly rather it is lumped with the rack model......	The information provided in the TR-0816-49833-P was reviewed and verified to be a sufficient summary. Detailed information is available in the supporting calculations available for audit.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
19	TR-0816-49833-P	Fuel irradiation embrittlement effects on the material properties have not been considered and accounted for the fuel rack seismic model and in establishing the material allowable limits.	Details have been added to the technical report to clarify the properties used. FSAR Section 3.1.6.6.6 of the technical report addresses the fuel assembly forces.
20	TR-0816-49833-P	The narrative in the DCA need only summarize information from the technical report. The following observations are for the technical report: Sufficient details of the mathematical model, including a description of how the important parameters are obtained, are not provided. The details should include the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and the effect of submergence on the mass, the mass distribution, and the effective damping of the fuel bundle and the fuel racks. The effect of gaps, sloshing water, and increase of effective mass and damping resulting from submergence in water should be quantified.	The "whole pool" analysis section of TR-0816-49833-P includes information to address this item, as follows: Sloshing assumptions - Section 3.1.4.2, Assumption 3. Gaps between racks - Section 3.1.4.4, bullet 2. Effects of being submerged in water - Section 1.1, Section 3.1.4., Section 3.1.4.6.3. Damping - Section 3.1.4.1. Mathematical model, major assumptions and development of the whole pool analysis model - Sections 3.1.4.1, 3.1.4.2, 3.1.4.3. Figure 3-125 and 3-126 provide model views. 3.1.4.4 provides the initial conditions for the mathematical model. Section 3.1.4.6 provides analysis approach.
21	TR-0816-49833-P	Evaluation of the fuel pool structure, including the pool slab and fuel pool liner is not provided for accident load combinations which include the impact of the spent fuel cask, the heaviest postulated load drop, and/or accidental drop of the fuel assembly from the maximum height.	FSAR Sections 9.1.5.2.3 and 9.1.2.2.2 addresses heavy load drops in the reactor and spent fuel pools including the heaviest postulated load.
22	TR-0816-49833-P	NuScale did not describe materials, quality control procedures, and any special construction techniques; the sequence of installation of the new fuel racks... NuScale did not describe any COLA items. The staff may seek confirmation that there are no COLA action items.	Quality control procedures are defined by NQA-1 in report Section 1.2 and there are no special construction techniques. No controls are needed for rack installation as identified in report Section 1.2. Re-racking may be addressed by a COL applicant if considered.
23	9.1.2	The DSRS states that a thermal-hydraulic analysis of the flow through the spent fuel racks shall demonstrate adequate decay heat removal from the spent fuel assemblies during all anticipated operating and accident conditions that prevent nucleate boiling for all fuel assemblies....	FSAR Section 9.1.2.3 was revised to describe the analysis that shows no nucleate boiling occurs because there is adequate cooling water flow through the fuel storage racks. A reference was added to the detailed analysis provided in the Technical Report TR-0816-49833.
24	9.2.5 9.1.3	The DSRS states that the design shall include a seismic Category I, Quality group C makeup system and an [equally qualified] backup system. The minimum makeup capacity of each system must exceed the larger of the pool leakage rate assuming SFP liner perforation resulting from a dropped fuel assembly or the maximum evaporation rate. The DCA only includes discussion of one makeup line that meets the criteria...	FSAR Section 9.1.3 was revised to explain the two seismic Category I makeup supply systems for the spent fuel pool (SFP) that are provided in the design. The water stored in the seismic Category I ultimate heat sink (UHS) pools is passively available for SFP makeup. The flow rate available to the SFP through the weir between the refueling pool and SFP exceeds what could be provided by a typical SFP makeup line. A separate seismic Category I and Quality Group C emergency makeup supply line provides the redundant flow path. FSAR Section 9.1.3 explains the adequacy of the makeup capacity.
25	9.1.5	Section 9.1.5.2.2, "Special Lifting Devices," describes the use of slings for handling heavy loads in the vicinity of the reactor modules or the spent fuel pool (SFP) without citing any specific application/configuration details and its associated rated loads. As a part of the components within the heavy load handling system, relevant information should be discussed...	Information on typical loads handled by the crane was added to address the item under FSAR Section 9.1.5.2.3 "Refueling Operations". Additional information was added on how the MLA meets single-failure-proof criteria in FSAR Section 9.1.5.2.2 under "Module Lifting Adapter." There is also a statement that bounds special lifting devices that have not been designed in this section (this covers items such as slings).
26	9A	The Fire Hazard Analysis (FHA) is missing the identification of the safe shutdown (SSD) equipment affected by fire, and a description of the SSD methodology or mitigation actions credited to achieve post-fire safe shutdown for a postulated fire in each fire area.	FSAR Section 9A has been updated to include a list of SSD equipment in each fire area.
27	9A	The FHA did not include the assessments for the Turbine Buildings and the outside Yard Area adjacent to the Reactor Building (RBX) and the Control Building (CRB). A major fire in these areas may potentially challenge the external walls and spread fire, smoke, and hot gas products into the RBX and CRB where SSD equipment are located.	FSAR Chapter 1.2.1 describes structures that are considered to be conceptual design information, including the Turbine Generator Building. FSAR Section 9A.3.3 provides information on the protection provided to the RWB, RXB and CRB against External Fire Exposures.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
28	10.1	DCA Tier 2, Section 10.1... the stretch power (VWO) heat balance should also be included ...	FSAR Section 10.1 has been updated to include a "valves wide open" heat balance figure.
29	10.2	The information on turbine overspeed protection in DCA Tier 2, Section 10.2, is insufficient to conduct a review. Deficiencies include: • Did not identify or discuss turbine over speed trip set points • Did not describe how diversity and independence of the overspeed is achieved...	Additional turbine overspeed information added to FSAR Section 10.2; including description of the diversity, redundancy and independence of the two overspeed protection systems and the associated setpoints.
30	19.5	The documentation in the application of the Aircraft Impact Assessment required by 10 CFR 50.150 is incomplete because: • It does not clearly state which acceptance criteria of 50.150(a) is committed to by NuScale... • Some key design features are missing adequate descriptions or pointers to other DCA sections (containing the description).... • Section 19.5 credits both the Main Control Room (MCR) and the Remote Shutdown Station (RSS); however, there is no description or pointer to a DCA section and/or figures that describe the characteristics of the MCR and RSS credited in the assessment... • Section 19.5.3.2 is inconsistent in its description of "structural analysis" and the corresponding intervening structure screening....	FSAR Chapter 19 has been updated to address this item. FSAR Sections 19.5.1 and 19.5.7 have been updated to clearly state that all 4 criteria are met for the NuScale design. Design features are discussed in 19.5.5.1 - additional text and pointers were added. FSAR Sections 19.5.5.7 and 19.5.6 were updated to describe physical separation of MCR and RSS. FSAR Section 19.5.3.2 was updated to explain that the RWB is credited in the structural analysis, but not the heat removal analysis. FSAR Section19.5.5.6 was updated to include additional detail and pointers. FSAR Section19.5.4.3 was updated to include additional detail and pointers. FSAR Section19.5.5.6 was updated to include discussion that the crane accommodates the impact and identifies parking location (although no admin controls are included).
31	multiple	The NRC staff identified gaps where significant information was missing or incomplete, such as missing figures and tables with entries of "TBD" and several hundred "OI" designations...	Additional content has been added to address the content that was not available at the time of the readiness review. There is no information identified as TBD or as an "Open Item" in Revision 0 of the NuScale DCA. FSAR Section 9.1.5 has been updated to address the specific examples identified in this docketing item.
32	multiple	Referenced documents, such as Technical Reports, were not included or were not referenced with the specificity needed for use in a design certification, in accordance with the requirements of 10 CFR Section 52.47, "Contents of applications; technical information."...	The list of technical reports which impose requirements on the design have been added to FSAR Section 1.6 as reports incorporated by reference.
33	Exemptions 4.3.1.5 15.0	NuScale does not provide a sufficient basis to meet 10 CFR 50, Appendix A, GDC 27, Combined reactivity control systems capability, nor did the DCA indicate an exemption request is expected to be provided....	A request for exemption to GDC 27 has been included. The NuScale design and justification is described in FSAR Section 4.3 and 15.0. FSAR Section 15.0.6 addresses return to power. Table 15.0-8 (previously Table 15.0-6) is complete and addresses moderator temperature coefficients (MTC) and Doppler coefficients. FSAR Section 15.0.4 addresses safe, stabilized condition; FSAR Section 15.0.5 addresses long term decay and residual heat removal; and FSAR Section 15.0.6 discusses the evaluation of a return to power.
34	4.3.2	No information was provided describing the methods used to develop the limits provided in Figures 4.3-3.	FSAR Section 4.3 has been updated to include additional information that addresses this item.
35	4.3.2	There is no discussion of the safety implications of ... or alternately, a statement confirming load following is not being considered for the design certification.	FSAR Section 4.3 and 4.4 address this item by indicating that load follow is not anticipated.
36	4.4	No basis is provided for establishing the CHF safety limit...	The basis is explained in the NuScale Subchannel topical report (TR-0915-17564). Additional information has been added to FSAR Section 4.4 along with a reference to the topical report.
37	4.4.2.10	This section clarifies that the limiting radial peaking factor used in thermal-hydraulic analyses... The methodology for developing this value is not provided.	The basis is explained in the NuScale Subchannel topical report (TR-0915-17564). Additional information has been added to FSAR Section 4.4 along with a reference to the TR-0915-17564.
38	4.4.3.3	In DCA Section 4.4.3.3, the methodologies for setting the operating limits are not provided ...	FSAR Section 4.4 has been updated with information explaining how these curves were developed.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
39	4.6.2	GDC 4 requires that the CRDS perform the required safety-related functions in adverse environmental conditions. There is no description (or cross-reference to the description) of analysis or testing, or other justification...	FSAR Section 4.6 has been updated to explain that the upper portion of the CRA drive shaft penetrates the pressurizer and is exposed to 625 degrees F steam, the remainder of the drive is exposed to 590 degrees F liquid, and that the drive shaft is designed for 650 degrees F.
40	4.6.2	Jet impingement loads from the discharge associated with opening the RSV and ECCS RVVs have not been evaluated in detail. There is no analysis or detailed description of why RSV and ECCS valves opening do not challenge CRDM...	FSAR Section 4.6.2 has been updated to describe the use of integral shield restraints at pipe weld sections and fluid diffusers at the outlet of RSV and RVVs.
41	5.2.2	GDC 31 is not addressed.	FSAR Section 5.2.2 has been updated to reference GDC 31.
42	5.2.2	...Figure 5.1-2 is only a RCS simplified diagram, and does not provide sufficient detail for staff to perform its review.	FSAR Section 5.1 has been updated to include an updated figure that depicts the detail requested.
43	5.4.3	In Section 5.4.3, Figure 5.4.7 does not provide sufficient detail for staff review...	FSAR Figure 5.4.7 (now figure 5.4.8) has been revised to add requested detail.
44	5.4.3	No information is provided on the passive condenser or orifice components of the DHRS.	FSAR Section 5.4.3 has been updated to include information on the passive condenser and orifice.
45	5.4.4	In the exemption request related to 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) ... the exemption request points to Section 6.2 of the DCA ...Section 6.2 does not go into detail describing the effect of ECCS operation under increasing/limiting amounts of non-condensable gases.	FSAR Section 6.2.2.3 has been updated to include a statement indicating that release of non-condensable gases in the pressurizer do not adversely impact long term core cooling. This statement supports the exemption request.
46	9.1.1	SRP Section 9.1.1 specifies criticality analysis of the spent fuel pool. There is no justification provided for the assertion that input assumptions 5 and 7 of TR-0816-49833-P ...	TR-0816-49833 has been updated to support assumptions 5 and 7.
47	15.0	There is no justification or description in the DCA of the NuScale position regarding event escalation for Chapter 15 events...	FSAR Chapter 15 has been updated to include a description of why AOOs transitioning from DHRS to natural circulation through the RVVs and RRVs is not an escalation of the event.
48	15.0.0.5	There is no justification detailed for the crediting of non-safety related backup valves in event mitigation (such as TS LCO, surveillances, etc.).	FSAR Section 15.0.0.6.6 has been updated to describe what is credited. A reference to Section 3.2, Table 3.2-1 was added for classification information.
49	15.0.0.6.5	Statements are incomplete or missing regarding the power scenarios that are considered for Chapter 15 events both within 15.0 and within each individual Chapter 15 event sections.	FSAR Section 15.0.0.6.5 has been updated to include the information. Individual event discussion sections were also expanded to include further detail.
50	15.0.2 6.3	DCA Section 15.0.2 does not describe the evaluation model used to analyze the potential for boron precipitation...	FSAR Section 15.6.5 has been updated to include further detail on boron precipitation.
51	15.0.2	DCA states analysis of infrequent events may be performed using best estimate analysis....	FSAR Section 15.0.2 has been revised. Infrequent events are no longer included in the best-estimate discussion.
52	15.2.7 15.2.8	There is no justification for the 30% uncertainty that is added to the SG heat transfer when it appears to be more conservative to take a penalty to heat transfer. This issue is common throughout 15.2.	The methodology for which uncertainties (high, low or no bias) are used is discussed in the NuScale Non-LOCA Topical Report (TR-0516-49416). FSAR Chapter 15 states what uncertainties are used for each limiting case and the magnitude, where appropriate.
53	15.4.8	SRP Section 15.4.8 references RG 1.77 (see page 2, paragraph 2) regarding the need to evaluate a rod ejection event initiating from a Control Rod Drive Mechanism housing failure, which causes a concurrent Loss of Coolant Accident. Regulatory Guide 1.77 specifies the need to consider the effects from the loss of primary system integrity...	FSAR Section 15.4.8 has been updated to include a statement that the containment response of an REA is bounded by an RVV opening. The basis for not analyzing CRDM housing failure for CNV response has been included in TR-0516-49084.
54	15.4.8	...The staff did note this DCA section is currently incomplete with numerous NuScale open items, but none of the open items would appear to address the criteria of SRP 4.2, Appendix B.	FSAR Section 15.4.8 has been updated to include information that addresses this specific item. There are no open items in Revision 0 of the DCA.
55	15.6.6	Section 15.6.6, "Inadvertent Operation of Emergency Core Cooling System," did not provide any justification for several assumptions...	FSAR Section 15.6.6 has been revised and includes information on inputs and assumptions and justification for the assumptions.
56	16	Technical Specifications not provided.	Technical Specifications are provided in Part 4 of the DCA, as described in Chapter 16 of the FSAR.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
57	6.2.1.1.2	DCA 6.2.1.1.2 states that "Maintenance of a vacuum also reduces the time to accumulate combustible levels of oxygen and hydrogen by post-LOCA metal-water and radiolysis process. Combustible gas control is further described in Section 6.3." NuScale does not provide analysis results that would substantiate this statement...	FSAR Section 6.2.1.1 has been revised and no longer indicates that vacuum conditions in containment prolong the time to accumulate combustible quantities of hydrogen and oxygen. FSAR Section 6.2.5.3 indicates that the integrated rate of energy deposition into water and the rate of combustible gas generation by radiolysis are contained in Reference 6.2-3 (TR-0716-50424).
58	6.2.5.2	...NuScale has not shown the results of an evaluation of radiolytic generation of hydrogen and oxygen. Radiolysis should be accounted for an "extended period of time required by the long-lived radioactivity remains in the core." (10 CFR 50.46(b)(5))	FSAR Section 6.2.5 has been updated. FSAR Section 6.2.5 and the NuScale Combustible Gas Control Technical Report (TR-0716-50424) describe long term radiolysis up to 72 hours. Mitigating actions are assumed beyond 72 hours as described in TR-0716-50424. TR-0716-50424 includes Figures depicting combustible gas generation by radiolysis for 72 hours. FSAR Section 6.2.5.3 references TR-0716-50424 for this information.
59	6.2.5.3	DCA 6.2.5.3 states that "For design basis accidents, hydrogen release to the containment is limited to that produced from radiolysis of the reactor coolant which is shown to result in an accumulated hydrogen concentration below the threshold for combustion during the first 24 hours." Radiolysis is not limited to the first 24 hours...	FSAR Section 6.2.5.3 has been updated and no longer includes the statement indicating accumulated hydrogen is below the combustion level for 24 hours. FSAR Section 6.2.5 and the TR-0716-50424 evaluate combustible gas generation by radiolysis for the first 72 hours. Mitigating actions are credited after 72 hours.
60	6.2.1.1	For the NuScale NPM to meet GDCs 16/50 and GDC 38, its containment structure and associated systems need to be designed to withstand the peak pressure and temperature conditions without exceeding the design leakage rate and with sufficient margin. NuScale DCA Section 6.2 refers to "sufficient margin" for the containment to meet GDCs 16/50 and GDC 38, but their values are neither quoted nor discussed vis-à-vis the pertinent DSRS acceptance criteria...	NuScale Containment Response Analysis Technical Report (TR-0516-49084) has been updated. TR-0516-49084 provides details of the limiting CNV pressure/temperature case results and M&E methodology. Figures demonstrating compliance with the DSRS acceptance criteria are provided. The technical report and FSAR Section 6.2.1 provide the results and margin.
61	6.2.1.1	...Even though DCA Table 6.2-7 does provide information on the containment heat transfer correlations, no technical information is included or referenced in DCA Section 6.2 or in the Containment Response Analysis Methodology TeR to qualify those correlations for modeling the novel NuScale containment design...	TR-0516-49084 references the NuScale LOCA EM and non-LOCA Topical Reports for validation of the heat transfer correlations for NPM analysis and describes pertinent NIST-1 validation testing. The NuScale Long Term Cooling Technical Report (TR-0916-51299) also demonstrates that core uncover does not occur and describes NIST-1 testing assessing NRELAP5 predictions of pertinent LTC parameters. FSAR Sections 6.2.1 and 6.2.2 reference these reports.
62	6.2.1.1	... NuScale design has sufficient margin to withstand the maximum external pressure.	FSAR Section 6.2.1 has been updated to include a statement providing the basis for the external design pressure.
63	6.2.1.1	One of the DSRS Section 6.2.1.1.A regulatory requirements is 10 CFR 50.34, "Contents of Applications; Technical Information," paragraph (f)(3)(v)(A)(1), "Additional TMI Related Requirements," as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100-percent fuel-clad metal-water reaction accompanied by hydrogen burning...	FSAR Section 6.2.5 and TR-0716-50424 describe an analysis demonstrating that the atmosphere is inert with the hydrogen generated from a 100% fuel-clad interaction. The analysis instead determined combustion loading at the most limiting possible combustion case. The full range of possible fuel clad interaction scenarios was considered, up to 100%.
64	6.2.1.1.A	Neither DCA Section 6.2 nor the Containment Response Analysis Methodology TR addresses how the NuScale design would meet the GDC 5 requirements for sharing SSCs important to safety among its nuclear power modules....	FSAR Section 9.2.5 has been updated and addresses GDC 5 conformance indicating that the UHS is a shared system capable of dissipating heat from one unit and simultaneously supporting safe shutdown of the other units.
65	6.2.1.3	...no tabulations of the transient M&E data are provided in the DCA or in the Containment Response Analysis TR...	TR-0516-49084 has been updated to include transient M&E data.
66	6.2.1.4	... Section 6.2.1.4 does not provide any details about the sources of energy used for the secondary mass and energy (M&E) release analysis.	TR-0516-49084 identifies sources of energy considered for secondary break M&E release analysis. A table providing limiting secondary release M&E data has been developed and is added to the technical report.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
67	6.2.2	The DCA section is missing references to analyses and/or reports that support the stated heat removal capability of the containment....	See response to item 61. The first item has been addressed through the existing validation work completed for containment heat removal (i.e. NIST testing). TR-0916-51299 evaluates all postulated conditions that that could affect long term core cooling and demonstrates the fuel remains covered. TR-0916-51299 includes figures depicting collapsed liquid levels, temperatures and pressures for each of these conditions. FSAR Section 6.2.2.3 references the TR-0916-51299 for these results.
68	6.2.2	Numerous tables do not have complete design information, including: Table 6.2-1 and Table 6.2-2.	FSAR Section 6.2.2 has been updated including the completion of Tables 6.2-1 and 6.2.2.
69	6.2.2 RG 1.82 GSI-191	The design basis for the long term cooling analysis (e.g., fuel) appears to be missing references to reports and analyses that support the stated post-LOCA debris generation and associated debris limits...	FSAR Section 6.3.3 has been updated to include Section 6.3.3.1 detailing the GSI-191 evaluation.
70	6.2.2.3 RG 1.82 GSI-191	...Design information that shows how NuScale addresses staff guidance (RG 1.82 R4) needs to be documented in the DCA...	FSAR Section 6.3.3 has been updated to include Section 6.3.3.1 detailing the GSI-191 evaluation. The discussion identifies the bases, assumptions and considerations for the GSI-191 evaluation. The evaluation demonstrates that debris build-up have no effect on long term cooling capability.
71	6.2.5 19.2.3	Section 6.2.5 does not provide sufficient detail to allow stand alone review to demonstrate the design meets the acceptance criteria for 10 CFR 50.44(c) conclusions...	FSAR Section 6.2.5 references TR-0716-50424 which addresses compliance with 10 CFR 50.44(c) acceptance criteria. An exemption request is provided in Part 7 of the DCA to address 10 CFR 50.44(c)(2).
72	6.2.6	In Part 7.7 of the DCA (Exemption Requests), NuScale seeks an exemption from GDC 52 capability for containment leakage rate testing (CILRT) at design pressure. NuScale states that a CILRT (also known as a "Type A" test) is not required to meet underlying purpose of the rule. The technical basis provided is not sufficient to support docketing the application....	NuScale developed the Containment Vessel Integrity Assurance Technical Report (TR-1116-51962) to detail the technical basis for the subject exemption. FSAR Section 6.2.6 indicates that the design justifies an exemption from the integrated leak (Type A) testing specified by GDC 52.
73	19.2.3.3.2	There are insufficient reference(s) to explain and support results reported...	FSAR Section 19.2 text has been updated to summarize the analyses method and results, including appropriate industry references. The hydrogen analysis has been linked to the FSAR Chapter 6 analysis. NuScale internal calculations are not explicitly referenced within the FSAR, but are available for NRC audit as part of the review.
74	19.1 19.3	The DCA does not provide a sufficiently comprehensive RTNSS evaluation in accordance with SRP 19.0 and SRP 19.3...	FSAR Section 19.3 (RTNSS) has been revised to include additional details on the RTNSS evaluation process.
75	19.1	Numerous tables, figures, and portions of text do not have complete information. For example, there is incomplete information related to PRA insights, key assumptions, and multi-module and module drop risk evaluations. The DCA should include a COL item for COL applicant/holder to ensure that the key assumptions are validated (see below on COL Items). CCFP is listed as TBD...	FSAR Section 19.1 has been updated to include this information. Additional information has been added to address insights, assumptions, multi-module and module drop evaluations. The source terms for the two release categories RC1 and RC2 have been specified. Also, clarification was provided that the module drop accident is assumed to result in only a small fraction of LRF, when pool scrubbing is credited. Composite CCFP has been added. COL item was added for validation of assumptions.
76	19.1 19.2 ER 4.0, 5.7	DCA does not sufficiently explain or substantiate the approaches and assumptions used to address various PRA and severe accident analysis issues, in light of the NuScale's novel design features. If used, underlying supporting documents (e.g., PRA documents, technical/topical reports) should be appropriately referenced in the DCA...	Additional content has been added in the identified sections of the FSAR to provide a basis for assumptions and summarize the results of analyses, including relevant technical/topical reports submitted with the application. Internal calculations are available for audit by the NRC. The following FSAR Sections have been updated to address the specific examples in this item; Section 19.1.4.1.1.5, Section 19.1.4.1.1.2, Section 19.1.6.1.2, and Section 19.2.
77	11.2 11.3	As partially identified by NuScale in its Readiness Assessment documents (DCA) as open items, the DCA is incomplete with missing information as listed below...	The applicable FSAR Sections have been updated to close all open items and to address the specific examples included in this item. Access to areas post-accident is addressed in Item 79 below.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
78	11 12 3.11	The plant radiation shielding design, radiation zoning, ventilation system design, equipment qualification analysis, tank failure analysis, etc., are all based on an assumed 0.028 failed fuel percentage (DCA Chapter 11 references TR-0916-51298, "Effluent Release (GALE Replacement) Methodology" which proposes realistic and design basis failed fuel fractions of 0.0028% and 0.028%, respectively). In comparison, this is inconsistent with DSRS Sections 11.1 and 12.2 which recommend 0.25% for the radiation shielding and zoning and RG 1.89 which recommends 1% for equipment qualification and other NRC guidance. The DSRSs for Chapter 11 and 12 indicate that if an alternative to the recommended failed fuel percentage is used it should be based on a limit provided in the technical specifications, however, the technical specifications have not been provided and NuScale provides no indication that the 0.028% is the limit allowed by the technical specifications...	FSAR Appendix 3C describes the use of operating experience to derive dose estimates used in the NuScale OE program. NuScale Accident Source Term Methodology Topical Report (TR-0915-17565) describes the methodology for determining the accident design basis source term. FSAR Section 11.1 and Technical Report TR-1116-52065 describe the basis and justification for the realistic failed fuel fraction (0.0028%) used for normal effluent release calculations to demonstrate compliance. A Technical Specification controls the RCS specific activity that conservatively bounds the assumptions used for failed fuel as input to the accident analysis in Chapter 15.
79	12.3-12.4	10 CFR 52.47(a)(8) requires that a design certification application contain the information necessary to demonstrate compliance with any technically relevant portions for the TMI requirements and 10 CFR 50.34(f)(2)(vii) requires that the applicant perform radiation and shielding design reviews of spaces around systems that may contain an accident source term and design as necessary to permit adequate access to important areas. While NUREG-0737, "Clarification of TMI Action Plan Requirements," Section II.B.2 indicates that the ability to take and analyze samples must be an area that is designed and analyzed to meet this criteria, the DCA provides no information regarding this analysis and NuScale indicates that the information is not necessary because they believe that it shouldn't be required. Also there is insufficient discussion regarding why no other vital areas are identified, such as those recommended in NUREG-0737...	FSAR Section 12.3.1.3.2 describes design reviews of spaces around systems that may contain an accident source term and design as necessary to permit adequate access to important areas in accordance with 10 CFR 50.34(f)(2)(vii); results are contained in FSAR Section 3.11. A radiation and shielding design review has been performed of spaces around systems that may contain accident source term materials to support qualification of equipment. FSAR Section 15.0.3 describes the post-accident operator dose assessments for the main control room and the technical support center. FSAR Section 12.4.1.8 has been updated to include dose assessment for post-accident contingency actions for RCS sampling, as described in FSAR Section 9.3.2.
80	DSRS 12.2 DSRS 12.3 iv.G	The DCA makes, without justification, several non-conservative assumptions involving information that affect the source term development and, consequently would preclude the staff from starting its assessment of the adequacy of radiation protection and radioactive waste management design features. This results in an incomplete Chapter 12 with respect to description and justification of the source terms such that the staff cannot start the assessment for equipment qualification, shielding, and occupational dose exposure. Some of these issues are identified below: ➢ NuScale secondary coolant activity calculations use a value of 3.53 lbm/day, much less than the 75 lbm/day primary to secondary leakage usually used. This apparently is based on the smaller steam generators and their derived value is scaled from the thermal power. However, there are no Technical Specifications that reflect corresponding limitations on Secondary Coolant activity, or Technical Specification limitations on allowable primary to secondary leakage. No technical justification is provided. There is a related issue of ensuring that the radiation monitoring required by the EPRI Primary to Secondary Leakage Guidelines, which are referenced by NuScale design documents, can be satisfied. ➢ The NuScale design documents state that neutron activation of materials in the Reactor Building pool water is insignificant without justification...	FSAR Section 11.1 describes analyses assuming 75 lb/day (multiplied by 12 NPMs) for design basis calculations related to shielding, system design and equipment qualification, consistent with NUREG-0017. FSAR Section 11.1 describes effluent releases assuming 3.53 lb/day primary to secondary leakage based on a linear thermal power scaling of the values in NUREG-0017. This is discussed in NuScale Effluent Technical Report TR-1116-52065. Technical Specification LCO 3.4.5 establishes a limit on primary to secondary leakage, consistent with initial conditions assumed in safety analyses. This is discussed in FSAR Section 11.2 and 11.3. FSAR Section 12.2 has been updated to clarify how the neutron flux outside an NPM is calculated. FSAR Section 12.2 has been updated to clarify the basis for the assumption of 50% resin line fill. FSAR Section 12.2 has been updated to clarify the analyses used to determine that the secondary coolant purification equipment accumulates less than 100 mCi's of activity based on a consideration of design basis secondary coolant.
81	15.0.3	The information provided about the maximum hypothetical accident (MHA) radiological consequences analysis in draft DCA Chapter 15 (Section 15.0.3) is technically insufficient....	FSAR Section 15.0.3.9 has been updated to include additional information about the maximum hypothetical accident (MHA) radiological consequences analysis.
82	15 15.0.3	Not all dose analysis assumptions are clearly noted in the draft DCA Chapter 15, and it is unclear that cross-referencing information in other DCA sections would provide all the information related to the dose analyses...	FSAR Chapter 15 has been updated to include additional information regarding dose analysis assumptions.

Item	DCA Section	Brief Description of the Docketing Issue (truncated*)	Disposition
83	14.2.3.3	Section 14.2.3.3 provides limited information on first of a kind (FOAK), first plant only and first three plant tests. FOAK tests which are very significant for proving the NuScale design are incompletely defined and no definition for first plant only and first three plant tests is provided....	FSAR Section 14.2.3.3 has been updated to provide information regarding FOAK testing, discussion of and reference to Comprehensive Vibration Assessment Program (CVAP) technical report TR-0716-50439, and completed test abstracts.
84	14.2.1.3.3	Section 14.2.1.3.3 provides a description for Low – Power Testing that includes confirmation of operability and design features that could not be tested previously due to lack of adequate heat source for reactor coolant and main steam systems. There is no detail included...	FSAR Section 14.2.10.1 has been updated to include additional detail as well as the test abstracts which provides the noted information pertaining to technical specification compliance and system operability.
85	Tables 14.2-1 - 14.2-41	Insufficient detail for acceptance criteria – when applicable include what is the fail-safe position, include values for pump/fan speed/flow/DP, provide values for setpoints for actuations (start, stop, alarm, etc.), for timed actuations provide time requirements, for grab samples provide a minimum quantity, etc...	FSAR Chapter 14 has been updated to complete all test abstracts and to include additional detail in the test abstracts relative to development of the acceptance criteria.

*expanded NRC observations of potential docketing issues are available in Reference 1

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Enclosure 1: DVD 1, NuScale Standard Plant Design Certification Application | Non-Public Version– Withhold Under 10 CFR 2.390

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Enclosure 2: DVD 2, NuScale Standard Plant Design Certification Application | Public Version – NuScale Nonproprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com