17 QUALITY ASSURANCE AND RELIABILITY ASSURANCE

Introduction

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 17, "Quality Assurance and Reliability Assurance," of the NuScale Power, LLC (hereinafter referred to as "NuScale" or "the applicant"), US460 Power Plant Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report (FSAR)." The staff based its regulatory findings documented in this report on Revision 2 of the SDAA, dated April 09, 2025 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML25099A237). The precise parameter values, as reviewed by the staff in this safety evaluation (SE), are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this SE to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

FSAR Chapter 17 discusses quality assurance (QA) during the design phase, QA during the construction and operations phases, the reliability assurance program (RAP), and the quality assurance program description (QAPD) for the NuScale Power Plant US460 standard design. It also discusses a program for implementing Title 10 of the *Code of Federal Regulations* (10 CFR) 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (the Maintenance Rule), in Section 17.6 "Maintenance Rule." The following sections of FSAR Chapter 17 address different areas of the quality assurance program: Section 17.1, "Quality Assurance During the Design Phase;" Section 17.2, "Quality Assurance During the Construction and Operation Phases;" Section 17.3, "Quality Assurance Program Description;" and Section 17.5, "Quality Assurance Program Description;" these apply to QA during the standard design approval application (SDAA) phase for NuScale standard plant design activities. The reliability assurance program described in Section 17.4 applies to those structures, systems, and components (SSCs) identified as being risk significant or important contributors to plant safety.

17.0 Quality Assurance

Licensing Topical Report (LTR) MN-122626-A, Revision 2, "NuScale Power, LLC Quality Assurance Program Description (QAPD)," dated March 11, 2025 (ML25070A195), describes the applicant's quality assurance program supporting the NuScale US460 Power Plant SDAA. The staff approved this QAPD on March 21, 2025 (ML25080A178). The QAPD is based on the applicable portions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA) Standards NQA-1-2008, "Quality Assurance Requirements For

Nuclear Facility Applications," and NQA-1a-2009, "Quality Assurance Requirements For Nuclear Facility Applications Addenda 1a."

17.1 Quality Assurance during the Design Phase

FSAR Section 17.5 addresses the NuScale QAPD for the NuScale Power Plant US460 standard design. The staff reviewed FSAR Section 17.5 in accordance with Section 17.5, Revision 1, "Quality Assurance Program Description—Design Certification, Early Site Permit and New License Applicants," dated August 2015 (ML15037A441), of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP). The staff includes that evaluation in Section 17.5 of this report.

17.2 Quality Assurance during the Construction and Operations Phases

The review of QA during the construction and operations phases is not applicable to the SDAA. A combined license (COL) applicant that references the NuScale Power Plant US460 standard design will describe the quality assurance program applicable to site-specific design activities and to the construction and operations phases.

17.3 Quality Assurance Program Description

Section 17.5 of this report addresses the QAPD.

17.4 Reliability Assurance Program

17.4.1 Introduction

FSAR Section 17.4, "Reliability Assurance Program," describes NuScale's reliability assurance program (RAP). The program applies to safety-related and non-safety-related SSCs identified as risk significant. NuScale stated that probabilistic risk assessment (PRA), deterministic, and other methods of analysis are used to determine the risk significance of SSCs.

17.4.2 Summary of Application

SDAA Part 2 (FSAR): In FSAR Section 17.4, NuScale stated the objectives of the RAP and described a two-stage implementation process for the program. The first stage of the program encompasses the reliability assurance activities that occur during detailed design of the plant before initial fuel load (i.e., design reliability assurance program (D-RAP)). The second stage consists of the operational phase of the plant's life to ensure that the reliability of the SSCs within the scope of the RAP is maintained during operations. Consistent with this approach, NuScale included action items in the SDAA for combined license (COL) applicants referencing the NuScale Power Plant US460 standard design.

ITAAC: There are no ITAAC for this section.

Technical Specifications: There are no technical specifications for this section.

Technical Reports: There are no technical reports for this section.

Topical Reports: FSAR Section 17.4.3.1, “Structures, Systems, and Components Classification and Categorization Process,” references the staff approved NuScale topical report, TR-0515-13952-NP-A, Revision 0, “Risk Significance Determination.” (ML16284A016). Section 3.0, “Analysis/Methodology,” under Section D of TR-0515-13952-NP-A, is partially incorporated by reference in FSAR Table 1.6-1, “NuScale Referenced Topical Reports.”

17.4.3 Regulatory Basis

The following NRC regulation contains the relevant requirements for this review:

- In Title 10 of the *Code of Federal Regulations* (10 CFR) 52.137(a)(9), the NRC requires an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission’s regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance with the SRP is not a requirement.

The guidance in SRP Section 17.4, Revision 1, “Reliability Assurance Program,” issued May 2014, lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections. The guidance addresses the Commission policy stated in item E of SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084),” dated May 22, 1995, and approved in the staff requirements memorandum for SECY-95-132, “SECY-95-132—Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084),” dated June 28, 1995.

17.4.4 Technical Evaluation

The staff reviewed FSAR Section 17.4. Additionally, the staff audited non-docketed information during an audit to support the review (ML24211A089). The staff based its evaluation on the guidance in SRP Section 17.4, which implements the Commission policy approved in the staff requirements memorandum for SECY-95-132.

FSAR Table 1.9-2, “Conformance with Regulatory Guides,” describes the application’s conformance with regulatory guides in effect 6 months before the docket date, and it states that the application partially conforms with Regulatory Guide (RG) 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance.” The applicant stated that it applied the guidance in RG 1.201 and Nuclear Energy Institute (NEI) 00-04 to the extent appropriate given the baseline risk metrics for the NuScale US460 reactor design. The staff reviewed NuScale’s RAP using the guidance in SRP Section 17.4; it did not review it against the voluntary requirements of 10 CFR 50.69 or guidance in RG 1.201. As such, the evaluation contained below does not constitute approval of

SSC categorization for alternative treatment requirements or any uses of the PRA beyond those described in FSAR Table 19.1-1, “Use of Probabilistic Risk Assessment at the Design Phase.”

17.4.4.1 Description of the Design Reliability Assurance Program

The staff verified that the applicant’s description in FSAR Section 17.4 included the details of the RAP that will be implemented during design and construction activities preceding initial fuel load. The staff confirmed that the scope, purpose, objectives, framework, and activities of the applicant’s D-RAP are consistent with those described in Section I of SRP Section 17.4. The staff also confirmed that the applicant established an appropriate COL item (COL Item 17.4-1) to provide assurance that a COL applicant that references the NuScale US460 standard design will propose a process for integrating the RAP into operational programs. Based on its review as described above, the staff finds that the description of the RAP is acceptable because it is consistent with guidance in SRP Section 17.4.

17.4.4.2 Programmatic Controls of Design Reliability Assurance Program

The staff verified that the applicant established the appropriate D-RAP programmatic controls to support design phase activities. Consistent with guidance in SRP Section 17.4, the programmatic controls established by the applicant address organization responsibilities, design control activities, procedures and instructions, records, corrective actions, and assessment plans. The latter four controls were developed by the applicant in accordance with the applicable provisions of the Quality Assurance Program Description (QAPD), which is addressed in SER Section 17.5. The staff finds that the applicant established programmatic controls for the RAP consistent with guidance in SRP Section 17.4.

The staff also confirmed that the applicant established an appropriate COL item (COL Item 17.4-3) to provide assurance that a COL applicant referencing the NuScale US460 standard design will identify the QA controls for the RAP SSCs during site-specific design, procurement, fabrication, construction, and preoperational testing activities. This provides assurance that the latter four controls listed above will be in place during the COL design and construction phases. The staff finds this COL item to be clear and to provide an appropriate level of assurance.

17.4.4.3 Methodology for Identifying Structures, Systems, and Components within the Scope of the Design Reliability Assurance Program

FSAR Figure 17.4-1, “Structures, Systems, and Components Classification Process Flow Chart,” describes the applicant’s process for determining the risk significance of SSCs and indicates that insights from the PRA and the importance of an SSC with respect to defense in depth are factors considered in identifying SSCs within the scope of the D-RAP. The process identifies safety analysis, quantified PRA results, RTNSS, and other considerations (e.g., Fukushima and multiple modules) as inputs to SSC classification and identifies operating experience, PRA and severe accident insights and assumptions, defense in depth, and systems interactions as additional considerations by the expert panel.

The staff verified that the methodology is based on a combination of probabilistic, deterministic, and other methods of analysis, and that the methodology accounted for multimodule considerations in the development of the list of D-RAP SSCs.

The applicant identified candidate risk-significant SSCs using the criteria in FSAR Table 19.1-19, “Criteria for Risk Significance.” As discussed in SER Section 19.1.4.4.8, “Quantification and Risk Insights,” these criteria are acceptable for use for the NuScale US460 SDAA. Based on the staff’s approval of the risk significance criteria in FSAR Table 19.1-19; the acceptability of TR-0515-13952-NP-A, Revision 0, with the limitations and conditions addressed in the NuScale US460 SDAA described in SER Section 19.1.4.4.8; NuScale’s justification for the sliding Fussel-Vesely threshold, described in SER Section 19.1.4.4.8; and the expert panel’s consideration of deterministic and other qualitative considerations in its selection of SSCs for the D-RAP list discussed above; the applicant’s methodology is acceptable because it is consistent with guidance in SRP Section 17.4.

17.4.4.4 Expert Panel

The applicant’s D-RAP includes use of an expert panel to confirm which SSCs in the design should be considered risk significant, RTNSS, or non-safety-related with augmented requirements. The application describes the technical disciplines, roles and responsibilities, and the qualification requirements of the expert panel.

FSAR Section 17.4.4, “Expert Panels,” states that the expert panel members must have an accredited 4-year degree in engineering, science, or other related field with a minimum of 5 years of experience in one or more of the following areas:

- PRA or risk and reliability analysis, including 3 years of PRA experience on small modular reactor design
- safety analysis expertise with U.S. nuclear regulatory guidelines
- licensing
- power plant operations, maintenance, previous commercial senior reactor operator license
- design integration or systems engineering
- design engineering (mechanical, electrical, instrumentation and controls, structural, civil)

During the audit (ML24211A089), the staff confirmed the range and level of expertise described in the SDAA. The range and level of expertise stated in the SDAA are consistent with guidance in SRP Section 17.4 and are, therefore, acceptable.

17.4.4.5 Structures, Systems, and Components within the Scope of the Design Reliability Assurance Program

The staff reviewed the list of SSCs required to perform risk-significant system functions in FSAR Table 17.4-1, “Design Reliability Assurance Program Structures, Systems, and Components Functions, Categorization, and Categorization Basis.” Notably, the staff focused on the following insights and SSCs:

- The augmented DC power system (EDAS) holds the reactor vent valves (RVVs) closed during normal operations and, therefore, contributes to defense in depth in the design. The RVVs do not include an inadvertent actuation block valve that was present in the NuScale US600 design.

- The safety-related passive autocatalytic recombiner (PAR) is designed to maintain the containment atmosphere inert (i.e., less than 4 percent oxygen by volume) during design-basis events and severe accidents. The PAR is a new component in the NuScale US460 standard design.

- The safety-related steam generator system (SGS), including the steam generator tubes, steam generator tube supports, feedwater plenums, and integral steam plenums, is not identified as risk-significant in FSAR Table 17.4-1; this system performs the same system functions in the US600 design and was identified as risk significant in the NuScale US600 Design Certification Application (DCA).

- Safety-related components in the control rod drive system (CRDS), including the control rod drive shafts and control rod drive latch mechanism, are not identified as risk significant in FSAR Table 17.4-1; these components perform the same system functions in the US600 design and were identified as risk significant in the NuScale US600 DCA.

The staff reviewed the SSC classification results and met with the expert panel during the NRC staff audit of FSAR Section 17.4 to gain a more in-depth understanding of how the expert panel evaluated deterministic and defense-in-depth considerations during the staff’s audit of non-docketed information (ML24211A089).

The staff requested additional information on the classification of the SGS and CRDS components (Request for Additional Information (RAI)-10199-R1 (ML24146A001)) and the classification of the PAR (RAI-10185-R1 (ML24131A116)). In the audit meeting with the expert panel and its response to RAI-10199-R1 (ML24271A282), the applicant clarified the information in FSAR Section 17.4 and described that a subject matter expert provides a risk significance classification recommendation based on available design and analysis information, including but not limited to thermal-hydraulic simulations, severe accident simulations, the list of candidates for risk significance from the PRA, PRA insights, and sensitivity analyses, for confirmation by the expert panel. The applicant also pointed to FSAR Table 19.1-2, “Design Features/Operational Strategies to Reduce Risk,” for examples where defense in depth was considered in the design of the plant. The applicant also stated that the detailed deliberations during the expert panel meetings were not captured in NuScale records, so the specific considerations and dispositions for functions categorized as not risk-significant were not available.

In its response to RAI-10185-R1 (ML24327A149), the applicant stated that the PAR is safety-related and that, as a safety-related SSC, additional controls (e.g., 10 CFR Part 50, Appendix B design control measures), programs (e.g., environmental qualification and Technical Specifications), and requirements (e.g., ITAAC) provide assurance that the PAR will perform its function in maintaining the containment atmosphere inert. The applicant described that the PAR includes the following design requirements because it is safety-related: (1) it conforms with General Design Criterion 4; (2) it is designed in accordance with the relevant requirements of American Society of Mechanical Engineers (ASME) AG-1-2019, which includes analysis of prescribed load conditions including dynamic loads such as jet impingement; (3) it is a Seismic Category I component; and (4) it is included in the environmental qualification program. The applicant also included the PAR within the scope of ITAAC because it is a safety-related component within the NuScale US460 power module.

The applicant stated that the D-RAP did not classify the PAR as risk significant because hydrogen combustion is not a safety concern in the context of the PRA. Specifically, the operation or failure of the PAR has no impact on the likelihood of core damage. The applicant clarified that the loss of the PAR and assumed subsequent containment failure does not lead to (1) core damage or (2) a large release in accident sequences that do not involve core damage.

The staff reviewed the set of SSCs within the scope of the D-RAP in accordance with the applicant's criteria and considered the applicant's rationale for excluding EDAS, the PAR, SGS, and CRDS components from the scope of the D-RAP.

The staff's evaluation of EDAS is documented in SER Sections 8.3 and 19.1. Based on these evaluations, the staff finds that the augmented design requirements for EDAS are comparable with the design requirements for SSCs identified as risk significant under D-RAP and, therefore, acceptable. The staff's evaluation of the PAR is in SER Section 6.2.5 and concludes that the safety classification of the PAR is acceptable. The applicant provided the safety classification of the SGS and CRDS components, respectively, in FSAR Table 5.4-9, "Classification of Structures, Systems, and Components," and FSAR Table 3.9-19, "Classification of Structures, Systems, and Components." The staff finds that the SGS and components of CRDS are safety-related and, therefore, subject to the requirements of the QAPD TR described in FSAR Section 17.5. The staff's review and acceptance of the applicant's QAPD TR is in SER Section 17.5. The staff finds that: (1) the design and quality requirements for EDAS, the PAR, SGS, and the safety-related CRDS components are consistent with the guidance in SRP Section 17.4 and meet the intent of the Commission policy stated in item E of SECY-95-132 for the NuScale US460 standard design, and (2) the design and quality requirements resulting from the classification of SSCs are, therefore, acceptable.

The staff reviewed COL Item 17.4-2, which provides assurance that a COL applicant referencing the NuScale US460 standard design will identify site-specific SSCs within the scope of the RAP. The COL item is clear, complete, and therefore, acceptable.

17.4.4.6 Process for Determining Dominant Failure Modes

The staff verified that the application describes a proposed process for determining the dominant failure modes of RAP SSCs. The applicant stated that this process incorporates industry experience and analytical methods. Analytical methods for identifying dominant failure modes include PRA importance analysis, root cause analysis, fault trees, and failure modes and effects analysis. The process described by the applicant is consistent with the guidance in SRP Section 17.4 and is therefore acceptable.

17.4.4.7 Quality Assurance Associated with Design Activities

The staff verified that the applicant specified the following QA controls in FSAR, Section 17.4.7, "Quality Assurance Applicable to Reliability Assurance Program Activities":

- The QAPD in FSAR, Section 17.5, includes QA controls applicable to the D-RAP process during the design activity phase. The staff's review and acceptance of the applicant's QAPD TR is in SER Section 17.5.

- COL Item 17.4-3 requires COL applicants that reference the NuScale US460 standard design to identify the QA controls for the RAP SSCs during site-specific design, procurement, fabrication, construction, and preoperational testing activities.

These controls are consistent with the staff's expectations for QA controls described in SRP Section 17.4 and are therefore acceptable.

17.4.5 Combined License Information Items

Table 17.4-1 lists COL information item numbers and descriptions related to the RAP from FSAR, Section 17.4. The staff finds the COL information items to be reasonable and, therefore, acceptable.

Table 17.4-1 NuScale Combined License Information Items for FSAR Section 17.4

Item No.	Description	FSAR Section
COL Item 17.4-1	An applicant that references the NuScale Power Plant US460 standard design will describe the Reliability Assurance Program conducted during the operations phases of the plant's 60-year design life.	17.4
COL Item 17.4-2	An applicant that references the NuScale Power Plant US460 standard design will identify site-specific structures, systems, and components within the scope of the Reliability Assurance Program.	17.4.1

Item No.	Description	FSAR Section
COL Item 17.4-3	An applicant that references the NuScale Power Plant US460 standard design will identify the quality assurance controls for the Reliability Assurance Program structures, systems, and components during site-specific design, procurement, fabrication, construction, and preoperational testing activities.	17.4.7

17.4.6 Conclusion

The NRC staff finds that the applicant has fully addressed the required information relating to the RAP at the standard design approval phase and concludes that it is acceptable for this SDAA.

17.5 Quality Assurance Program Description—Design Certification, Early Site Permits, and New License Applicants

17.5.1 Introduction

MN-122626-A, Revision 2, provides the QAPD for the NuScale SDA. The QAPD incorporates the requirements of ASME NQA-1-2008 and NQA-1a-2009, as endorsed by NRC Regulatory Guide 1.28, Revision 4, “Quality Assurance Program Criteria (Design and Construction),” issued June 2010 (ML100160003).

The QAPD submitted by NuScale addresses the design activities affecting the quality and performance of items and services provided by NuScale supporting the SDA and customer contracts. The NuScale QAPD does not address construction and design QA activities that occur once construction begins. A COL applicant that references the NuScale Power Plant US460 standard design will describe the quality assurance program applicable to these specific activities.

17.5.2 Summary of Application

FSAR, Chapter 17, Section 17.5, states the following:

> The Quality Assurance Program Description (QAPD) for the NuScale Power Plant US460 standard design is provided in the topical report, “NuScale Power, LLC Quality Assurance Program Description.”

ITAAC: The applicant did not include any inspections, tests, analyses, and acceptance criteria (ITAAC) for the QAPD.

Technical Specifications: There are no technical specifications (TS) for this area of review.

Topical Reports: MN-122626-A, Revision 2, “NuScale Power, LLC Quality Assurance Program Description,” dated March 11, 2025 (ML25070A195).

17.5.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion 1, “Quality standards and records,” requires that a QAP be established and implemented.

- 10 CFR 52.137(a)(19) requires that an SDA applicant include a QAPD that satisfies the applicable requirements of Appendix B to 10 CFR Part 50.

The guidance in SRP Section 17.5 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

17.5.4 Technical Evaluation

The NRC staff reviewed MN-122626-A, Revision 2. The staff’s SE documents the evaluation of the NuScale QAPD. Specifically, the NRC staff evaluated the NuScale QAPD and verified that it meets NRC regulations by following the guidance in SRP Section 17.5. The staff verified that FSAR, Revision 2, Section 17.5, incorporates MN-122626-A, Revision 2, without exception, for the control of activities affecting quality during the development of the NuScale SDA and is therefore acceptable.

Staff Inspection of NuScale QAPD Implementation

Appendix B to 10 CFR Part 50 sets forth the requirements for quality assurance programs for nuclear power plants. According to 10 CFR 52.137(a)(19), the description of the quality assurance program for a nuclear power plant shall discuss how the applicable requirements of Appendix B to 10 CFR Part 50 were satisfied. NRC Inspection Manual Chapter 2508, “Construction Inspection Program: Design Certification,” contains staff guidance on performing a postdocketing quality assurance program inspection. This postdocketing quality assurance program inspection provides the staff with reasonable assurance that the quality assurance program has been adequately implemented. This inspection is consistent with the regulations that govern all stages of the licensing process and allows the staff to verify whether activities affecting quality are conducted under the appropriate provisions of Appendix B to 10 CFR Part 50. Effective implementation of the quality assurance program provides reasonable assurance that SSCs will perform their intended safety function. The NRC staff conducted a postdocketing quality assurance program inspection on February 26 through March 1, 2024 (ML24099A129).

17.5.5 Combined License Information Items

The staff notes that FSAR Section 17.5 did not provide any COL information items. However, the NuScale’s QAPD (MN-122626-A, Revision 2) does not cover QA for the site-specific design activities and construction and operations phases. Therefore, consistent with 10 CFR 52.79(a)(25), any COL applicant referencing the NuScale Power Plant US460 standard design shall describe the quality assurance program applicable to site-specific design activities and to the construction and operations phases.

17.5.6 Conclusion

As discussed above, the NRC staff completed its review of FSAR Section 17.5. The staff used the requirements of Appendix B to 10 CFR Part 50 and 10 CFR 52.137(a)(19), and the guidance in SRP Section 17.5, as the bases for evaluating the acceptability of NuScale's QAPD (MN-122626-A, Revision 2). The staff concludes that NuScale's QAPD has established an acceptable quality assurance program in accordance with applicable NRC regulations and industry standards for SDA activities. The NRC staff conducted a postdocketing QAP inspection on February 26 through March 1, 2024 (ML24099A129).

17.6 Maintenance Rule

17.6.1 Introduction

FSAR Section 17.6, "Maintenance Rule," addresses the NuScale Maintenance Rule program.

17.6.2 Summary of Application

SDAA Part 2 (FSAR): FSAR Section 17.6 states that this section addresses an operational program that is the responsibility of an applicant and is not applicable to new plant designs.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: There are no TS for this area of review.

Technical Reports: There are no technical reports for this area of review.

17.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.65
- 10 CFR 52.79(a)(15), which requires that a COL FSAR describe the program, and its implementation, for monitoring the effectiveness of maintenance necessary to meet the requirements of 10 CFR 50.65

The guidance in SRP Section 17.6, "Maintenance Rule," lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

17.6.4 Technical Evaluation

SRP Section 17.6, “Maintenance Rule,” addresses the Maintenance Rule program based on the requirements of 10 CFR 50.65 and the guidance in Nuclear Management and Resources Council (NUMARC) 93-01, as endorsed by Regulatory Guide (RG) 1.160.

The staff has reviewed FSAR Section 17.6 in accordance with the guidance in SRP Section 17.6. Per the SRP guidance, the Maintenance Rule program is an operational program addressed in a COL application, and no outstanding information is expected in the SDAA related to this program.

The NRC staff agrees that the plan or process for implementing the Maintenance Rule program and describing that plan or process in the COL FSAR are the responsibilities of the COL applicant referencing the NuScale Power Plant US460 standard design. The COL applicant shall implement the Maintenance Rule program, at the latest by fuel load (i.e., by the time the Commission makes the finding required in 10 CFR 52.103(g)). The applicant may implement an acceptable Maintenance Rule program in advance of the Commission’s 10 CFR 52.103(g) finding, with components being monitored or tracked as they become available.

17.6.5 Combined License Information Items

FSAR Section 17.6 does not contain any COL information items.

17.6.6 Conclusion

The NRC staff confirmed that the applicant has addressed the information relevant to the Maintenance Rule program at the SDA phase. The NRC staff agrees with the SDA application that the COL applicant is responsible for developing and implementing the Maintenance Rule program (under the requirements of 10 CFR 52.79(a)(15) and 10 CFR 50.65). Thus, the staff concludes that the Maintenance Rule information presented in FSAR Section 17.6 is acceptable.