NUSCALE POWER, LLC
SAFETY EVALUATION FOR TOPICAL REPORT TR-0815-16497,
REVISION 0, "SAFETY CLASSIFICATION OF PASSIVE NUCLEAR POWER PLANT ELECTRICAL SYSTEMS"
(CAC. NO. RQ6002)

1.0 Introduction

By letter dated October 29, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15306A263), NuScale Power, LLC (the applicant or NuScale), submitted Topical Report (TR)-0815-16497, Revision 0, "Safety Classification of Passive Nuclear Power Plant Electrical Systems." Topical Report Section 1.1, "Purpose," states the purpose of the submittal and describes the review and approval that the applicant seeks from the U.S. Nuclear Regulatory Commission (NRC) staff, as follows:

> *The purpose of this topical report (TR) is to request Nuclear Regulatory Commission (NRC) review and approval of what are termed herein as "conditions of applicability," and the methodology and bases used in their development. The conditions of applicability comprise a set of passive reactor plant design and operational attributes that, if met in full by a reactor design or license applicant, justify the applicant's determination that none of the plant electrical supply systems fulfill functions that per the regulatory definitions of "safety-related" and "Class 1E" would warrant a Class 1E classification.*
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> *This topical report also seeks NRC review and approval of augmented design, qualification, and quality assurance (QA) provisions that are an extension of the "conditions of applicability." The augmented provisions are described in Table 3-2. These augmented design, qualification, and QA provisions would be applied as minimum requirements to electrical systems that have been determined to be non-safety-related but yet are essential to the post-accident monitoring of Type B and Type C variables. Provided the "conditions of applicability" are fully satisfied, the approved augmented provisions would represent an acceptable alternative to the portion of Regulatory Guide 1.97, Revision 4, that specifies a Class 1E power source for instrumentation associated with Type B and Type C variables.*

Upon the successful demonstration by the applicant that all the conditions of applicability and augmented provisions are met, an applicant could use this TR as part of the supporting basis for demonstrating the acceptability of a non-Class 1E classification of its electrical supply systems. The Institute of Electrical and Electronics Engineers (IEEE) Std 323-1974 "IEEE Standard-for Qualifying Class IE Equipment for Nuclear Power Generating Stations," (ML032200206) gives the following definition for the term "Class 1E": The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or otherwise are essential in preventing significant release of radioactive material to the environment.

Based on its review of the TR, the NRC staff issued requests for additional information (RAI) related to passive nuclear power plant electrical systems, in particular, the direct current (dc) equipment and system, post-accident monitoring, and reactor coolant pressure boundary integrity and safe shutdown (ADAMS Accession No. ML16281A298). In response to the NRC staff's request, NuScale provided supplemental information in a letter dated December 5, 2016 (ADAMS Accession No. ML16340D339).

2.0 Regulatory Requirements

The NRC staff notes that the electrical power systems in a nuclear power plant do not establish any classification requirements on the systems that they support. As a support system, it is the loads that determine classification requirements for the power system. Therefore, a reactor plant design, with no safety-related equipment that is dependent on electrical power to perform its safety function, would not need Class 1E alternating current (ac) or dc power systems.

In TR Section 3.1, "Methodology Used to Develop Conditions of Applicability," the applicant stated that the application of augmented provisions is consistent with the process established in the NRC's regulations for treatment of nonsafety-related structures, systems, and components that are determined to have risk significance, which includes SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 (ADAMS Accession No. ML003708068), and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," dated May 22, 1995 (ADAMS Accession No. ML003708005).

In TR Table 3-2, "Augmented Design, Qualification, and Quality Assurance Provisions," the applicant listed the regulatory requirements and guidance documents that a future passive plant applicant would need to apply or consider for the augmented design, qualification, and QA provisions to the non-Class 1E electrical systems (referred to as "highly reliable" dc electrical systems) for powering the post-accident monitoring instrumentation for Type B and Type C variables and for the plant emergency lighting systems.

The NRC staff evaluated the conditions of applicability in Table 3-1 of the TR by first identifying the functions performed by safety-related structures, systems and components (as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.2) that are required by regulation and then ensuring that these required functions are addressed by the conditions of applicability in Table 3-1. These functions that are specified by the general design criteria (GDC) of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 are summarized below:

- GDC 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems be provided with appropriate margin to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effect of anticipated operational occurrences (AOO).

- GDC 13, "Instrumentation and Control," requires, in part, that the applicant provide instrumentation to monitor variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions as appropriate to assure adequate safety.

- GDC 15, "Reactor Coolant System Design," requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs.

- GDC 16, “Containment Design,” requires that the reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

- GDC 19, “Control Room,” requires, in part, that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.

- GDC 20, “Protection System Functions,” requires, in part, that the protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of AOOs.

- GDC 26, “Reactivity Control System Redundancy and Capability,” requires, in part, that the control rods be capable of reliably controlling reactivity changes to assure that, under conditions of normal operation, including AOOs, and with appropriate margin for stuck rods, SAFDLs are not exceeded.

- GDC 27, “Combined Reactivity Control Systems Capability,” requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 34, “Residual Heat Removal,” requires, in part, that a residual heat removal system be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded.

- GDC 35, “Emergency Core Cooling,” requires, in part, that a system to provide abundant core cooling be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

- GDC 38, “Containment Heat Removal,” requires, in part, the provision of a system to remove heat from the reactor containment. The system safety function shall be to rapidly reduce, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and to maintain them at acceptably low levels.

- GDC 41, “Containment Atmosphere Cleanup,” requires, in part, systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

- GDC 50, "Containment Design Basis," requires, in part, that the reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

- GDC 54, "Piping Systems Penetrating Containment," requires, in part, that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities that have redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

- GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," requires, in part, that each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves.

- GDC 56, "Primary Containment Isolation," requires, in part, each line that connects directly to the containment atmosphere and penetrates the primary reactor containment shall be provided with containment isolation valves.

- GDC 57, "Closed System Isolation Valves," requires each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere to have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

- GDC 61, "Fuel Storage and Handling and Radioactivity Control," requires, in part, that fuel storage and handling, radioactive waste, and other systems which may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions. This criterion specifies that such systems shall be designed to include appropriate containment, confinement, and filtering systems.

- GDC 63, "Monitoring Fuel and Waste Storage" requires, in part, appropriate systems in fuel storage and radioactive waste systems and handling areas to detect conditions that may cause a loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions.

- GDC 64, "Monitoring Radioactive Releases," requires, in part, the means for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released as a result of postulated accidents.

The NRC staff also determined that the following regulatory requirements and guidance documents are applicable to the review of this TR:

- 10 CFR 50.34(f)(2)(xix) requires the applicant to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

- 10 CFR 50.34(f)(2)(xx) requires the applicant, in part, to provide power supplies for pressurize level indicators such that pressurizer level indicators are powered from vital buses and electric power is provided from emergency power sources.

- 10 CFR 50.44, "Combustible gas control for nuclear power reactors," requires, in part, that an applicant must perform an analysis that demonstrates containment structural integrity. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by the hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as it pertains to the acceptance criteria and requirements of the evaluation model used to demonstrate performance of the emergency core cooling system.

- 10 CFR 50.55a(h) approves for incorporation by reference the 1991 version of Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," including the correction sheet dated January 30, 1995.

- 10 CFR 50.63, "Loss of all alternating current power," in part, requires that the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration.

- 10 CFR 100.21, "non-seismic siting criteria," as it relates to, in part, radiological dose consequences of postulated accidents.

- 10 CFR 50.34(a)(1)(ii)(D) as it relates to the assessment of radiological consequences of accidents.

- 10 CFR 52.47(a)(2)(iv) as it relates to the assessment of radiological consequences of accidents.

- Regulatory Guide (RG) 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," issued June 2006, which endorses (with certain clarifying regulatory positions specified in Section C of the guide) the Institute of Electrical and Electronics Engineers standard (IEEE Std.) 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations."

- Branch Technical Position (BTP) 7-10, “Guidance on Application of Regulatory Guide 1.97.”

- SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs,” dated March 28, 1994 (ML003708068) and associated Staff Requirements Memorandum (SRM), June 30, 1994 (ML003708098)

- SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs,” dated May 22, 1995 (ML003708005), and associated SRM, June 28, 1995 (ML003708019).

3.0 Staff Evaluation

Topical Report Section 1.2, “Scope,” provides the scope of review specific to the safety classification of plant electrical systems for which the conditions of applicability and augmented provisions apply, as follows:

- offsite and onsite ac electrical power systems.

- onsite dc electrical power systems.

NuScale stated that the above scope does not include instrumentation and control (I&C) equipment and circuits, which include both Class 1E and non-Class 1E systems, that serve to monitor and control power to and operation of safety-related and nonsafety-related loads.

The TR contains four appendices:

(1) Appendix A, “Example Overview of Electrical Systems and I&C Systems Design.”

(2) Appendix B, “Example Safety Classification Assessment for Electrical Systems.”

(3) Appendix C, “Example Failure Modes and Effects Assessment for the Highly Reliable DC Power Systems.”

(4) Appendix D, “Example Safety Analysis Results.”

The information in the appendices provides examples of how future applicants (including NuScale) can implement the TR. As part of the scope of this TR, NuScale is not seeking NRC approval of the information contained in the appendices. NuScale further stated that its design certification application (DCA) will present the final design information and that the DCA will confirm that the final design meets the conditions of applicability described in TR Table 3-1, which lists the attributes to be satisfied as conditions of applicability.

The TR Table 3-1 has two sections described by NuScale as:

(1) Section I contains the specific conditions that, if fully met, would adequately justify that no Class 1E electrical supply systems (power sources) are required.

(2) Section II contains additional conditions to be applied (after meeting Section I).

One of the conditions in Section II requires augmented design, qualification, and QA provisions described in TR Table 3-2. The provisions in TR Table 3-2 are the minimum requirements to be applied to non-Class 1E electrical system(s) (termed as “highly reliable” dc electrical systems) that will be used to power post-accident monitoring instrumentation for Type B and Type C variables and to power the plant emergency lighting system. Topical Report Tables 3-1 and 3-2 contain proprietary information provided by NuScale. If a passive nuclear plant can meet all the conditions listed in TR Table 3-1, without the need for any electrical power, Class 1E ac or dc power supply systems may not be necessary. This is subject to satisfying the capability to achieve safe shutdown, core cooling, and containment and reactor coolant pressure boundary integrity and to maintain these conditions for a minimum of 72 hours.

The four appendices to the TR describe the methodology/procedures to be applied to an example power system design to ensure that a dc power system design can be “highly reliable.”

- Appendix A to the TR provides an overall description of an onsite power system that could serve a passive plant design that meets the conditions of applicability. In addition, Appendix A includes a set of typical one-line diagrams to facilitate an overall understanding of the concepts as applied to a passive plant electrical system.

- Appendix B to the TR describes how a hypothetical complete loss of all electrical power (both ac and dc) would impact the various safety functions and explains how the applicant can satisfy the attributes of conditions of applicability.

- Appendix C to the TR provides an example failure modes and effects analysis (FMEA) of the example onsite dc power system described in Appendix A. The effects of failure modes and mechanisms for components in the example FMEA establish that no single failure exists that could prevent safety-related functions from being achieved and maintained.

- Appendix D to the TR provides example safety analysis results of a passive plant that has the design attributes described in Appendices A and B. The analysis shows that, in each postulated design-basis event analyzed, none of the systems credited for mitigating the event require electrical power nor operator action.

The NRC staff did not review TR Appendices A–D for approval of design specifics. NuScale provided the appendices as an example overview for a passive plant that future applicants could refer to. Therefore, the NRC staff limited its review to the main body of the TR and focused on the design criteria considered in the conditions of applicability and not an actual design.

<u>Concept of “Highly Reliable” Non-Class 1E Direct Current System</u>

With regard to a fully non-Class 1E dc power system for a completely passive nuclear power plant design, the NRC staff was concerned whether the dc power system would have high reliability. More specifically, the NRC staff was concerned that the valve-regulated lead-acid (VRLA) battery life could be seriously and suddenly reduced by prolonged high temperatures, the magnitude and frequency of discharge cycles, or overcharging. The NRC staff devised a three-pronged review approach (performance, QA, and quantification) to determine the relative reliability of the conceptual dc power system design (presented in Appendix A to the TR) in comparison to a Class 1E dc power system.

To determine the reliability of a dc system for which usage of a VRLA battery system is offered for a passive plant, the NRC staff requested additional information. To date, conventional large, light-water nuclear power plants have not used VRLA batteries for onsite power. Therefore, the NRC staff requested information on battery life, QA, performance, qualification, and reliability.

<u>RAI 08.03.02-01</u>

In letter dated December 5, 2016 (ADAMS Accession No. ML16340D339), NuScale acknowledged the NRC staff's concerns with VRLA battery life and stated that these effects can be mitigated by following the recommendations provided in the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 1187-2013, "IEEE Recommended Practice for Installation Design and Installation of Valve-Regulated Lead-Acid Batteries for Stationary Applications," and IEEE Std. 1188-2005 (R2010), "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Valve-Regulated Lead- Acid (VRLA) Batteries for Stationary Applications," as noted in TR Table 3-2. Additionally, IEEE Std. 1187-2013 refers to IEEE Std. 1491-2012, "IEEE Guide for Selection and Use of Battery Monitoring Equipment in Stationary Applications," and IEEE Std. 1635-2012, "IEEE/ASHRAE Guide for the Ventilation and Thermal Management of Batteries for Stationary Applications."

In addition to the use of the industry standard procedures mentioned above for design, testing and implementation of the VRLA battery-powered dc system, the applicant stated the following:

- *The backup power supply system delivers backup power to heating, ventilation, and air conditioning systems serving the battery and associated charger rooms to avoid prolonged periods of high ambient temperature.*

- *For design consideration for magnitude and frequency of discharge cycle related monitoring, the applicant will follow the guidance in IEEE Std. 1187-2013, IEEE Std. 1188-2005, and specifically IEEE Std. 1491-2012, which provides criteria to detect and monitor a battery for degradation.*

- *The batteries will not be overcharged, following the guidance in IEEE Std. 1187-2013 as supplemented by IEEE Std. 1491-2012, to allow detection of instances of potential overcharging before a battery is degraded to a point at which it is not able to perform its intended function.*

The electrical power system presented in TR Appendix A depicts an onsite power system design with no Class 1E power sources assuming the reactor design does not require any safety-related electrical loads to support the safety analyses. The NRC staff reviewed the RAI response and determined that the use of VRLA batteries in a nonsafety dc power system design for a passive nuclear power plant, construction, and monitoring will follow the guidance in IEEE Stds. 1187-2013 and 1188-2005, as supplemented by IEEE Std. 1491-2012, and IEEE Std. 1635-2012. These IEEE Standards provide widely established industry guidance for design, testing and performance of VRLA batteries. The NRC staff determined that based upon the IEEE Standards mentioned above, the design will provide reasonable assurance that a VRLA-battery-provided dc power system will be prevented from prolonged periods of exposure to high temperature, will be monitored for potential overcharging, and will be monitored for magnitude and frequency of discharge cycles that may degrade the battery performance. For the reasons discussed above, the NRC staff concludes that, for a nonsafety dc system that uses VRLA batteries, the applicant's response provides reasonable assurance that the dc system will be monitored for degradation and will not adversely impact its intended function. The staff requested the applicant to include its response to the NRC staff's RAI 08.03.02-01 in the next revision to the TR. **This is confirmatory item 6.1 in Section 6.0 of this SER.**

RAI 08.03.02-02

In TR Table 3-2, NuScale stated that a graded QA program will be applied to the dc electrical system, which will meet or exceed the augmented QA guidance in Appendix A, "Quality Assurance Guidance for Non-Safety Systems and Equipment," to RG 1.155, "Station Blackout." The NRC staff asked NuScale to describe the proposed QA program in sufficient detail to enable the staff to verify whether it meets or exceeds the guidance in RG 1.155.

In its RAI response letter dated December 5, 2016, NuScale stated that a combined license (COL) applicant that references TR-0815-16487 will be required to follow the guidance in RG 1.155, Appendix A, "Quality Assurance Guidance for Non-safety Systems and Equipment." The NRC staff finds NuScale's response reasonable. The NRC staff has placed a condition in Section 4.0, "Additional Conditions on Applicants or Licensees Referencing this TR," Condition 4.1, of this SER to ensure that all future applicants that reference TR-0815-16497 address the guidance in RG 1.155, Appendix A, in sufficient detail to verify whether the relevant QA program would meet or exceed the guidance in RG 1.155.

RAI 08.03.02-03

In TR Table 3-2, under the provision "Batteries," NuScale stated that the VRLA batteries have augmented design, QA, and qualification provisions. The NRC staff asked NuScale to describe the methods and processes that a passive reactor nuclear power plant will use to verify that VRLA batteries will perform their intended function(s) during normal operation, anticipated operational occurrences, and postulated design-basis events.

In a letter dated December 5, 2016 (ADAMS Accession No. ML16340D339), NuScale stated that the VRLA batteries used in a passive reactor nuclear power plant design are not credited for use in mitigating the consequences of postulated design-basis events. NuScale also stated that an applicant using this TR shall implement a testing and monitoring program, as described in IEEE Std. 1188-2005 and IEEE Std. 1491-2012, to ensure that VRLA batteries will perform their intended function(s) when called upon. These standards provide for a wide variety of operating parameters to be monitored on a continuous basis, including cell-specific parameters.

Furthermore, NuScale stated that applicants would be required to environmentally qualify their VRLA batteries in accordance with IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," and to seismically qualify their batteries in accordance with IEEE Std. 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," to provide further assurance that the batteries will perform their intended functions.

The NRC staff also asked NuScale to provide the industry standards or applicable references that will be used for verification purposes. NuScale provided the following industry standards:

- IEEE Std. 323-1974, as endorsed by RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” for harsh environments.

- IEEE Std. 323-2003 for mild environments.

- IEEE Std. 344-2004, as endorsed by RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants.”

- IEEE Std. 1188-2005.

- IEEE Std. 1491-2012.

The NRC staff reviewed the applicant’s response to RAI 08.03.02-03 and determined that the design of the VRLA batteries used as a non-Class 1E dc power source in a passive reactor nuclear power plant design, in accordance with the following widely accepted industry practices: IEEE Std. 1188-2005 and IEEE Std. 1491-2012 for testing and monitoring, IEEE Std. 323-1974 and IEEE Std. 323-2003 for environmental qualification, and IEEE Std. 344-2004 for seismic qualification, provide reasonable assurance that the VRLA batteries will perform their intended functions.

The NRC staff concludes that NuScale’s response is acceptable with regard to the methods and processes used to verify that the VRLA batteries will perform as intended. However, an applicant using the TR must provide a qualification testing plan that includes an environmental and seismic qualification, and also a technical functional requirement for VRLA batteries in order to provide reasonable assurance that VRLA batteries will perform their intended functions. For this reason, the staff has established Condition 4.2 on the TR for the applicant or licensee to confirm that the VRLA batteries and their structures are Seismic Category I. To provide reasonable assurance that the VRLA batteries will perform as intended, the applicant or licensee that references the TR shall provide a COL action item to support that the VRLA batteries and their structures are Seismic Category I. A qualification testing plan includes environmental and seismic qualification and a technical functional requirement for VRLA batteries to show they can perform as intended.

<u>RAI 08.03.02-04</u>

In the TR, NuScale has described its dc power system as “highly reliable” and substantially equal in reliability to that of an analogous Class 1E dc power system. However, the TR did not fully justify these statements. Therefore, to complete its review, the NRC staff asked the applicant to provide additional quantitative information. Specifically, the staff asked the applicant to describe the methodology that it will use to compare the highly reliable dc system that it will describe in its DCA to a Class 1E dc power system to show that the highly reliable dc system is substantially equal in reliability to a typical Class 1E dc power system.

NuScale provided a two-part response. The first part describes the methodology in the TR that DC applicants would use to perform a quantitative analysis. This methodology comprises the following five steps needed to compare the reliability of the highly reliable dc system to that of a typical Class 1E dc power system:

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

The second part of NuScale's response provided the results of its comparative analysis using the above methodology. NuScale indicated that its results were favorable in that the augmented non-Class 1E design indicated a factor of approximately 5 times higher reliability than that of the Class 1E design. In its response, NuScale further concluded that amending the TR to include the methodology presented is not necessary.

Based on the review of this response, the NRC staff concludes that the five-step process outlined in the applicant's response provides an acceptable approach for demonstrating the relative reliability of a non-Class 1E system with that of an analogous Class 1E system. In addition, as part of the review, the staff identified two items for clarification with respect to the response.

NuScale and the NRC staff held a conference call on January 6, 2017, to address the NRC staff's questions. First, the NRC staff requested clarification on whether NuScale's referenced probabilistic risk analysis (PRA) model included common-cause failures among each of the two-battery-in-parallel configurations. NuScale stated that the model included common-cause failure of the two-battery configurations. The concern was that any battery operating in parallel could experience certain common-cause events. The NRC staff finds NuScale's answer acceptable in the context of its review of the TR. Any further questions on PRA methodology would be part of the PRA review of the referencing DCA or COL application.

Second, the NRC staff requested clarification about the statement at the end of the response that the response does not require a revision to the licensing document (i.e., TR-0815-16497). The NRC staff questioned this statement in that TR-0815-16497 is a methodology document and the response to RAI 08.03.02-04 provides additional methodology necessary for use of the TR by any applicant referencing it. During the conference call, NuScale committed to supplement the TR accordingly. **This is confirmatory item 6.2 in Section 6.0 of this SER.**

Based on the discussion of the applicant's responses to RAI 08.03.02-04, the NRC staff's concerns for the dc power system to act as a highly reliable system are adequately addressed.

3.1 Post-accident Monitoring

The primary purpose of post-accident monitoring instrumentation is to display plant variables that provide information required by the control room operator during and after an accident. The GDC 13, 19, 64, 10 CFR 50.34(f)(2)(xix), 10 CFR 50.34(f)(2)(xx), and 10 CFR 50.55a(h) contain regulatory requirements governing post-accident monitoring instrumentation. The NRC provides the primary guidance for implementing these regulatory requirements in RG 1.97, which describes a method acceptable to the staff for complying with the Commission's regulations to provide instrumentation for monitoring plant variables and systems during and after an accident. RG 1.97, which endorses IEEE Std. 497-2002, with certain clarifying regulatory positions specified in Section C of the guide, specifies that a Class 1E electrical system should be provided to supply the instrumentation that monitors Type A, B, and C variables under post-accident conditions. Under 10 CFR 50.34(f)(2)(xx), the NRC requires that electrical power for pressurizer level indicators must be powered by vital buses.

RG 1.97 defines Type A, B, and C variables, as follows:

- Type A: Variables that provide the primary information required to allow main control room operators to take manual actions for which no automatic control is provided.

- Type B: Variables that provide primary information to the control room operators to assess the plant safety functions.

- Type C: Variables that provide primary information to the control room operators to indicate the potential for breach or the actual breach of fission product barriers (e.g., fuel cladding, reactor coolant system pressure boundary, and containment pressure boundary).

IEEE Std. 603-1991, paragraph 5.8.1, "Displays for Manually Controlled Actions," specifies that monitoring instrumentation be part of the safety systems and meet the requirements of IEEE Std. 497-2002. For monitoring instrumentation used for these operations, IEEE Std. 603-1991 and IEEE Std. 497-2002 specify a Class 1E electrical power supply. [redacted]

During its review, the NRC staff considered whether the Commission's regulations requiring safety-system designs to provide accident monitoring instrumentation also require that instrumentation to be powered by a Class 1E electrical system (i.e. Type B and C variables). The staff determined the following:

- Regulatory requirements in GDC 13, 19, and 64 are applicable to postulated DBEs (which do not include core damage) and do not specify a Class 1E electrical power supply. Thus, the beyond-design-basis events (BDBEs) are considered outside the scope of the regulatory findings for GDC 13, 19, and 64.

- The regulation in 10 CFR 50.34(f)(2)(xix) requires the monitoring instrumentation to be adequate for monitoring plant conditions following an accident that includes core damage. Because 10 CFR 50.34(f)(2)(xix) is a Three Mile Island (TMI)-related requirement, the staff views core damage in this regulation as the type experienced in the TMI accident which was far beyond what is allowed for DBA. Further, 10 CFR 50.34(f)(2)(xix) does not address the quality of electrical power supply. For these reasons the staff does not view this regulation as a requirement for a Class 1E electrical power supply.

- The regulation in 10 CFR 50.34(f)(2)(xx) requires, in part, that the safety-system design provide power supplies for pressurizer level indicators such that pressurizer level indicators are powered from vital buses and electric power is provided from emergency power sources. NUREG-0737 states that the pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available. Because vital buses and emergency power sources are not Class 1E, this regulatory requirement does not specify a Class 1E electrical power supply.

- Clause 5.8.2 of IEEE Std 603-1991 states, in part, that the display instrumentation provided for safety system status indication need not be part of the safety systems. Because the display instrumation need not be part of the safety system, this regulatory requirement does not require a Class 1E electrical power supply.

Even if the regulatory requirements for post-accident monitoring do not require a Class 1E electrical system, Type B and Type C accident monitoring instrumentation is required to perform its intended function under postulated accident conditions. As such, the reliability of the electrical power supply for these instruments should be substantially similar to that of a Class 1E electrical system (see Section 3.0 of this SER).

In TR Appendix B, Section B.2.7, “Post-Accident Monitoring,” the applicant provided an alternative to RG 1.97 that uses a highly reliable direct current (dc) power system in lieu of a Class 1E electrical system to supply electrical power to the post-accident monitoring instrumentation. When performing this review, the NRC staff considered the electrical system reliability of the highly reliable dc electrical system. The staff established a three-pronged approach to establishing whether or not the highly reliable dc electrical system provides a substantially equal reliability to that of a Class 1E design. The three-pronged approach consisted of (1) evaluation of the augmented design, qualification, and QA provisions, (2) consideration of the rigor of the highly reliable dc power system as demonstrated by the failure modes and effect analysis, and (3) quantification via fault tree analysis to compare the NuScale design with an approved passive pressurized-water reactor dc system design. The staff discusses its evaluation of the electrical system reliability of the highly reliable dc power system in Section 3.0 of this SE. Based on its evaluation of the electrical system reliability, the staff concluded that the highly reliable dc electrical system provides a substantially equal reliability to that of a Class 1E design, and thus it provides additional assurance that post-accident monitoring capability is maintained during and following a DBE.

Based on the NRC staff's review of the TR and the regulatory requirements governing accident monitoring instrumentation, the staff found that the augmented design, qualification, and QA provisions of the power sources for Type B and Type C variables represent an acceptable alternative to the guidance in RG 1.97. Although the applicant stated that no Type A variables have been identified in its design, the staff has established Condition 4.3 in the SER Section 4.0 for the applicants or licensees referencing this SE to confirm that operator actions are not necessary to ensure safety-related functions for any postulated DBE (i.e., the design does not include Type A variables as defined in IEEE Std. 497-2002, as modified in RG 1.97, Regulatory Position C.4). Condition 4.3 ensures that the requirements of GDC 13, 19, and 64, 10 CFR 50.34(f)(2)(xix), 10 CFR 50.34(f)(2)(xx), and 10 CFR 50.55a(h) are met.

Spent Fuel Pool Considerations

The spent fuel pool (SFP) has the safety function of maintaining the spent fuel assemblies in a safe and subcritical array during all credible storage conditions. GDC 63 for spent fuel storage facilities requires monitoring systems to (1) detect conditions that may cause the loss of residual heat removal capability and excessive radiation levels, and (2) indicate when to take action to initiate appropriate safety actions.

In TR Appendix B, Section B.2.2, "Fuel Assembly Cooling—Spent Fuel and Module Core Refueling," the applicant described [redacted]

The TR Table 3-1, "Conditions of applicability," items 3 and 4, specify that in order for the TR to be applicable to a design, the applicant must demonstrate that:

[redacted]

[redacted]

The staff determined that Conditions of Applicability 3 and 4, as stated above, are consistent with the staff guidance in standard review plan (SRP) 19.3 "Regulatory Treatment of Non-Safety Systems For Passive Advanced Light Water Reactors," and therefore, if a design met these conditions, then Class 1E power would not be required for monitoring SFP conditions.

3.2 Safe Shutdown, Core Cooling, and Reactor Coolant Pressure Boundary Integrity

NRC staff used the review guidance in the SRP to identify the Commission's regulations associated with safe shutdown, core cooling, and reactor coolant pressure boundary integrity.

From the acceptance criteria provided in SRP Sections 4.2, 4.3, 4.4, 5.4.7, 6.3, and Chapter 15, NRC staff identified GDC 10, GDC 15, GDC 20, GDC 26, GDC 27, GDC 34, and 10 CFR 50.46 as being associated with safety-related structures, systems and components, in accordance with the definition in 10 CFR 50.2, that need to be addressed by the conditions of applicability provided in Table 3-1 of the TR.. As described in 10 CFR Part 50, Appendix A, the GDC established the minimum requirements for the principal design criteria for water-cooled nuclear power plants that are similar in design and location to plants for which the Commission has issued construction permits. The GDCs are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units. Therefore, the NRC staff established Condition 4.4 on the TR to require an applicability determination.

The NRC staff acknowledges that Condition of Applicability No. I.1.a, [redacted] and Condition of Applicability No. I.1.c., [redacted] require, in part, that the reactor trip and core cooling systems be capable of automatically initiating in the absence of electrical power. Therefore, the NRC staff finds these requirements to be consistent with GDC 20. Accordingly, the NRC staff finds that Conditions of Applicability No. I.1.a and No. I.1.c are necessary and sufficient only for determining that no Class 1E power is required to satisfy GDC 20.

Condition of Applicability No. I.1.b states, [redacted] SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plants," dated March 28, 1994 (ADAMS Accession No. ML003708068), clarifies the conditions that constitute a safe-shutdown condition as reactor subcriticality, decay heat removal, and radioactive material containment. Additionally, SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," states that an appropriate safety analysis can be used to demonstrate passive system capabilities to bring the plant to a safe, stable condition and to maintain this condition. The TR contained clarifying examples in Appendix B, "Example Safety Classification Assessment for Electrical Systems," and Appendix D, "Example Safety Analysis Results" to illustrate how the conditions of applicability can be demonstrated. The examples provided did not include a quantitative safety analysis to demonstrate the ability to insert sufficient negative reactivity during and following a design-basis event to achieve and maintain safe shutdown. This caused the NRC staff to question the interpretation of safe shutdown as applied to Condition of Applicability No. I.1.b. Accordingly, the NRC staff issued RAI 08 03 02-05, dated October 7, 2016 (ADAMS Accession No. ML16281A298), asking the applicant to (1) specify the criteria that constitute a safe shutdown as applied to Condition of Applicability Item I.1.b, and (2) describe how a future applicant for a passive plant will demonstrate that electrical power is not necessary to achieve and maintain a safe shutdown for a minimum of 72 hours.

The applicant's response, provided in a letter dated December 5, 2016 (ADAMS Accession No. ML16340D339), stated that, "The criteria that constitute a safe shutdown are sub-criticality and decay heat removal in order to maintain fuel clad integrity (radioactive material containment)." The NRC staff finds this response acceptable because it is more restrictive than the criteria provided in SECY-94-084.

The applicant's response to RAI 08.03.02-05 further discussed an approach to demonstrating Condition of Applicability No. I.1.b, namely, *an applicant [or licensee] will evaluate the reactivity control systems to ensure sufficient shutdown function capability and evaluate the decay heat removal system to ensure sufficient heat removal capability. To ensure that safe shutdown capability is sufficient to address the safety issue of heat removal reliability, a probabilistic risk assessment is used to ensure that the reliability of systems used to achieve and maintain safe shutdown support[s] conformance to the commission's safety goal guidelines.*

The applicant further explained that safety analyses of design-basis events (as typically presented in Chapter 15 of the final safety analysis report) may not be suitable for demonstrating the ability to achieve and maintain a safe shutdown following a design-basis event. Specifically, it stated,

Conservative assumptions are applied to Chapter 15 safety analysis of DBEs [design-basis events] appropriate for the intended purpose of ensuring appropriate margins to protect fuel integrity and core coolability. Although these safety analyses can be used to demonstrate adequate shutdown capability per SECY-94-084, application of the same conservative assumptions may lead to excessive margin with respect to shutdown capability.

The NRC staff previously communicated positions on shutdown margin during and following design-basis events in letters discussing GDC 26 and GDC 27, dated December 5, 2016 (ADAMS Accession No. ML16292A589), and September 8, 2016 (ADAMS Accession No. ML16116A083), respectively. These letters clarify that shutting down the reactor and maintaining a subcritical reactor are safety functions considered in GDC 26 and GDC 27, both of which require margin for malfunctions such as stuck rods. In the letter addressing GDC 27, the NRC staff stated,

... the staff's current view is that GDC 27 requires that the reactor be reliably controlled and that the reactor achieve and maintain a safe, stable condition, including subcriticality beyond the short term, using only safety related equipment following a postulated accident with margin for stuck rods.

Based upon the shutdown margin requirements of GDC 26 and GDC 27, the NRC staff established Condition 4.6 to require a demonstration or appropriate justification of shutdown margin. Based upon the applicant's criteria for safe shutdown and pursuant to Condition 4.6, the NRC staff finds that Condition of Applicability No. I.1.b is necessary and sufficient only for determining that no Class 1E power is required to satisfy GDC 26 and GDC 27.

The NRC staff acknowledges that Condition of Applicability No. I.1.c is a high-level requirement that partially addresses the safety functions discussed in GDC 10, GDC 34, GDC 35, and 10 CFR 50.46. The TR contained clarifying examples in Appendix B and Appendix D. However, the applicant provided this information for clarification purposes only, and the NRC staff has no findings about the technical sufficiency of the information contained in Appendix B or Appendix D to the TR. In addressing Condition of Applicability No. I.1.c, an applicant or licensee referencing this TR would be required by 10 CFR 50.34, "Contents of Applications; Technical Information," 10 CFR 52.47, "Contents of Applications; Technical Information," and 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report," to (1) provide a description and analysis of the safety-related systems, structures, and components credited to perform core cooling functions, with emphasis upon performance requirements, and (2) perform evaluations to show that safety functions will be accomplished in the absence of electrical power. Based upon the discussion in this paragraph, the NRC staff finds that Condition of Applicability No. I.1.c is necessary and sufficient only for determining that Class 1E power is not required to satisfy GDC 10, GDC 34, GDC 35, and 10 CFR 50.46.

Condition of Applicability No. I.1.g states, [redacted] This statement supports Condition of Applicability No. I.1, which states, [redacted] The TR contained clarifying examples in Appendix B and Appendix D to illustrate how the conditions of applicability can be demonstrated. The example safety analysis in Appendix D shows that the example passive plant response to an AOO includes establishing a direct coolant flowpath between the reactor core and the containment, thereby removing a fission product barrier. This caused the NRC staff to question whether Condition of Applicability No. I.1.g is sufficient for demonstrating RCPB integrity. Accordingly, the NRC staff issued RAI 08.03.02-06, dated October 7, 2016 (ADAMS Accession No. ML16281A298), asking the applicant to (1) specify the criteria that constitute RCPB integrity as applied to Condition of Applicability No. I.1, and (2) explain why the removal of a fission product barrier during an AOO is not considered an event escalation.

The applicant's response, provided in a letter dated December 5, 2016 (ADAMS Accession No. ML16340D339), stated that a loss of RCPB integrity involves a mechanical failure in an RCPB component, but it does not include the opening of a valve. The applicant further stated that considering the RCPB to be lost when a valve opens is problematic because (1) it would preclude advanced designs that offer improvements in safety by relying on valves to depressurize the reactor coolant system for safe shutdown, (2) it is not consistent with the licensing basis for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), as these designs rely on safety relief valves for overpressure protection, and (3) the GDC address maintaining structural integrity of RCPB components rather than preventing the opening of valves to allow fluid to pass into or out of the RCPB.

Additionally, the applicant stated that opening a valve to depressurize the reactor coolant system and establish long-term cooling is not considered a removal of a fission product barrier, and thus not event escalation, because the functions of the reactor coolant system barrier are not lost. The applicant further stated that events that do not result in unacceptable consequences or significantly increase the risk for radiological release do not challenge the intent of the non-escalation criterion specified in Section 15.0, "Introduction—Transient and Accident Analyses," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."

The NRC staff's evaluation of the applicant's response considered the examples from operating PWRs and BWRs. The applicant's response included examples in which valves connected to the reactor coolant system opened and allowed fluid to pass through the RCPB. The NRC staff finds these examples to differ from the scenario that was the basis for RAI 08.03.02-06. In particular, the staff identifies that establishing a direct coolant flowpath between the reactor core and the containment, in a manner similar to an emergency depressurization of the reactor coolant system (1) can result in a significant pressurization of the containment, (2) requires the containment to perform an AOO mitigation function by establishing a coolant return path to the reactor pressure vessel, (3) can result in a significant tensile stress on the fuel cladding, and

(4) may not be terminated through the closure of the open valve. The AOO scenario, provided in Appendix D to the TR, appears to rely upon the containment to retain the reactor coolant necessary to ensure fuel cladding integrity during an AOO. Because an AOO, by definition, is expected to occur one or more times during the life of the nuclear power plant, the NRC staff is concerned that such reliance upon the containment may not be consistent with the underlying defense-in-depth purpose of GDC 15. Accordingly, the NRC staff established Condition 4.5 on the TR to address reliability requirements for the system(s) necessary to retain reactor coolant within the RCPB. Based upon the overpressure protection of the RCPB and pursuant to Condition 4.5, the NRC finds that Condition of Applicability I.1.g is necessary and sufficient only for determining that Class 1E power is not required to satisfy GDC 15.

3.3 Containment Isolation

The TR Condition of Applicability No. I.1.d specifies for [redacted]. The NRC staff finds this condition to be consistent with the containment isolation provisions in GDCs 54–57, because these GDCs, in part, require containment isolation capabilities which reflect the importance to safety of isolating piping systems in response to a design basis event. The provision that the safety function can be achieved and maintained with no reliance on electrical power (e.g., electrical power is not necessary during and following a design basis event to ensure performance of the containment isolation function) for a minimum of 72 hours is also consistent with SECY-94-084 as the minimum duration that passive systems should be able to perform their safety functions, independent of electrical power, operator action, or offsite support, after an initiating event. Because the condition is consistent with regulatory requirements and guidance, the staff finds that the condition is necessary and sufficient only for determining that Class 1E electrical power is not required to satisfy GDC's 54-57.

3.4 Containment Integrity

The TR Condition of Applicability No. I.1.e. specifies for [redacted] The NRC staff finds this condition to be consistent with containment provisions in GDCs 16, 38, 41, and 50, because these GDCs require, in part, that the containment safety function can be achieved and maintained during design basis events. In addition, the staff finds this condition to be consistent with 10 CFR 50.44, because this requirement addresses the control of combustible gases in the containment. The provision that the safety function can be achieved and maintained with [redacted] is also consistent with SECY-94-084 as the minimum duration that passive systems should be able to perform their safety functions, independent of electrical power, operator action, or offsite support, after an initiating event. Because the condition is consistent with regulatory requirements and guidance, the staff finds that the condition is necessary and sufficient only for determining that Class 1E electrical power is not required to satisfy GDCs 16, 38, 41, 50, and 10 CFR 50.44.

3.5 Fission Product Control

The TR Condition of Applicability No. I.1.f specifies for [redacted]

[redacted] The staff finds this condition would be necessary to assess GDC 41 because GDC 41, in part, addresses fission product control systems. The staff finds it acceptable to require a condition that ensures compliance with the applicable guideline exposures cited above without reliance on active systems (e.g., those requiring electrical power) to control fission products, because the condition must meet regulatory requirements associated with exposure. Because the condition is consistent with regulatory requirements, the staff finds that the condition is necessary and sufficient only for determining that Class 1E electrical power is not required to satisfy GDC 41.

3.6 Control Room Habitability

The TR Condition of Applicability I.5 specifies that electrical power is not necessary [redacted] The staff finds this requirement to be consistent with GDC 19, because GDC 19 requires, in part, that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. The provision that the safety function can be achieved and maintained with no reliance on electrical power [redacted] is also consistent with SECY-94-084 as the minimum duration that passive systems should be able to perform their safety functions, independent of electrical power, operator action, or offsite support, after an initiating event. Because the condition is consistent with regulatory requirements and guidance, the staff finds that the condition is necessary and sufficient only for determining that Class 1E electrical power is not required to satisfy GDC 19.

3.7 Cooling for Building Areas Containing Safety-Related Equipment

The TR Condition of Applicability I.6 specifies that [redacted] The NRC staff finds this requirement to be consistent with 10 CFR 50.63 because 10 CFR 50.63, in part, requires that the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The provision that safety functions [redacted] is also consistent with SECY-94-084 as the minimum duration that passive systems should be able to perform their safety functions, independent of electrical power, operator action, or offsite support, after an initiating event. Because the condition is consistent with regulatory requirements and guidance, the staff finds that the condition is necessary and sufficient only for determining that Class 1E electrical power is not required to satisfy 10 CFR 50.63.

3.8 Building Ventilation

The TR Condition of Applicability I.7 specifies that [redacted] The staff finds this condition would be necessary to address GDC 61 because GDC 61, in part, requires that fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. The staff finds it acceptable to require a condition that complies with the applicable guideline exposures cited above without reliance on active systems (e.g., those requiring electrical power), because the condition must meet the regulatory requirements associated with exposure. Because the condition is consistent with regulatory requirements, the staff finds that the condition is necessary and sufficient only for determining that Class 1E electrical power is not required to satisfy GDC 61.

3.9 Emergency Lighting

Section 3.2.2 of the TR states that portions of the emergency lighting system are powered from the highly reliable DC electrical system, and is classified as non-Class 1E. Additionally, TR Condition of Applicability II.3 (Section II of Table 3-1) specifies that the applicant's emergency lighting capability [redacted] The staff finds that the TR Condition of Applicability II.3 is consistent with the NRC staff's guidance regarding the classification of the emergency lighting system as non-Class 1E and, therefore, acceptable.

4.0 <u>Limitations and Conditions</u>

If an applicant with a passive design chooses to reference TR-0815-16497 as part of its application, the applicant or licensee must demonstrate that the reactor design meets all of the conditions of applicability in the TR Table 3-1, and all of the augmented design, qualification, and quality assurance provisions in the TR Table 3-2.

Additionally, an applicant or licensee referencing this TR must:

4.1 Address the guidance in RG 1.155, Appendix A, in sufficient detail to enable the staff to verify that the relevant QA program would meet or exceed the guidance in RG 1.155.

4.2 Confirm that the VRLA batteries and their structures are Seismic Category I. To provide reasonable assurance that the VRLA batteries will perform as intended, the applicant or licensee that references the TR shall provide a COL action item to support that the VRLA batteries and their structures are Seismic Category I. A qualification testing plan includes environmental and seismic qualification and a technical functional requirement for VRLA batteries to show they can perform as intended.

4.3 Demonstrate that operator actions are not necessary to ensure the performance of safety-related functions for any postulated DBE (i.e., the design does not include Type A variables as defined in IEEE Std. 497-2002, as modified in RG 1.97, Regulatory Position C.4), as presented in Chapter 15 of its final safety analysis report (FSAR), and the human factors analysis in Chapter 18 of its FSAR.

4.4 Demonstrate that the conditions of applicability given in Table 3-1 of the TR are consistent with the functional requirements contained in the principal design criteria for the nuclear power plant.

4.5 Demonstrate that system(s) necessary to retain reactor coolant within the RCPB are designed with sufficient reliability such that a challenge to containment does not occur with the frequency of an AOO. Alternatively, an applicant or licensee referencing the TR can demonstrate that a failure of the containment would not impede the ability to maintain decay heat removal and radioactive material containment for the long term.

4.6 Demonstrate, or otherwise justify, that the reactor can be brought to a safe shutdown using only safety-related equipment in the absence of electrical power following a DBE, with margin for stuck rods.

5.0 Conclusions

The NRC staff approves the use of the NuScale TR-0815-16497 as a reference document subject to the conditions and limitations specified in Section 4.0, and implementation of Confirmatory Items as listed in Section 6.0, “Summary of Confirmatory Items,” of this SER. Specifically, based on its review of TR-0815-16497, the staff finds that if a reactor design can meet the conditions of applicability and the augmented design, qualification, and QA provisions, Class 1E power sources would not be necessary. This approval of the concepts discussed in the TR does not constitute approval of any specific design.

6.0 Summary of Confirmatory Items

As a result of its review of the TR, including additional information submitted to the staff, the staff identified the following confirmatory items (CIs). An item is considered confirmatory if the staff and the applicant have reached a satisfactory resolution, but the resolution has not been received by the staff or placed on the applicant’s docket. The staff will close these CIs once future revisions to the TR incorporate the requested information.

CI 6.1: The staff requested the applicant to include its response to the staff’s RAI 08.03.02-01 in the next revision to the TR to state that future applicants referencing this TR must follow the guidance in IEEE Stds. 1187-2013 and 1188-2005, as supplemented by IEEE Std. 1491-2012, and IEEE Std. 1635-2012.

CI 6.2: NuScale and the NRC staff held a conference call on January 6, 2017, to further clarify NuScale’s response to RAI 08.03.02-04. During the conference call, the NRC staff requested clarification regarding NuScale’s written response to the RAI 08.03.02-04 that the response does not require a revision to the licensing document TR-0815-16497. The NRC staff questioned this statement in that TR-0815-16497 is a methodology document and the response to RAI 08.03.02-04 provides additional methodology necessary for use of the TR by any applicant referencing it. During the conference call, NuScale committed to supplement the TR accordingly. The staff requested that NuScale update the TR to include the requested information.