RAIO-0717-55158

August 01, 2017

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 47 (eRAI No. 8820) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 47 (eRAI No. 8820)," dated June 02, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 8820:

- 03.09.06-1
- 03.09.06-2
- 03.09.06-3
- 03.09.06-4

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Marieliz Vera, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8820

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NuScale Response to NRC Request for Additional Information eRAI No. 8820

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8820
Date of RAI Issue: 06/02/2017

NRC Question No.: 03.09.06-1

NuScale Final Safety Analysis Report (FSAR) Tier 2, Section 6.3, “Emergency Core Cooling System,” specifies that the emergency core cooling system (ECCS) serves three fundamental purposes: (1) to function as part of the reactor coolant pressure boundary (RCPB); (2) to cool the reactor core in situations when it cannot be cooled by other means, such as a loss of coolant accident (LOCA) inside the containment vessel (CNV); and (3) to provide low temperature overpressure protection (LTOP) for the reactor pressure vessel (RPV). The ECCS valves with their first-of-a-kind (FOAK) design consist of three reactor vent valves (RVVs) at the top of the RPV, and two reactor recirculation valves (RRVs) on the side of the RPV above the active fuel level. NuScale FSAR Tier 2, Section 6.3 indicates that each ECCS valve consists of 4 distinct valve components connected by several feet of tubing that contains borated reactor coolant used as the valve hydraulic fluid as follows: (a) the main valve that is held closed by hydraulic force from reactor coolant pressure in the main valve control chamber, and is opened by spring force assisted by reactor coolant pressure when the main valve control chamber is vented by tubing through the inadvertent block (IAB) feature and trip valve into the CNV; (b) the solenoid-operated trip valve located outside the CNV in the cooling pool that is normally closed and is de-energized to open to vent borated reactor coolant from the main valve control chamber (provided the IAB feature allows passage of reactor coolant); (c) the solenoid-operated reset valve located outside the CNV in the cooling pool that is normally closed and is energized to open to pressurize the main valve control chamber with borated reactor coolant from an outside source (provided the IAB feature allows passage of reactor coolant) to close the main valve against spring force; and (d) the IAB feature located between the main valve and solenoid-operated valves that uses spring force to overcome the differential pressure between the RPV and CNV to retract a block valve to allow reactor coolant (hydraulic fluid) to be supplied to or vented from the main valve control chamber through the applicable solenoid-operated valve. NuScale FSAR Tier 2, Section 5.2.2.2.2, “Low Temperature Overpressure Protection System,” specifies that the RVVs are designed with sufficient capacity to prevent RCPB pressure from exceeding the limiting pressure when below the LTOP enabling temperature such that the RPV is maintained below brittle facture stress limits during operating, maintenance, testing, or postulated accident conditions. During a public telephone conference on April 19, 2017, NuScale representatives stated that the design and qualification of the ECCS valves and their valve components are not complete.

The NRC regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. With respect to the ECCS, General Design Criterion (GDC) 35, “Emergency core cooling,” in 10 CFR Part 50, Appendix A, requires, in part, that a system to provide abundant emergency core cooling be provided and its 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 37, “Testing of emergency core cooling system,” in 10 CFR Part 50, Appendix A, requires, in part, that the ECCS shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

For a nuclear reactor design certification application, the NRC regulations in 10 CFR Part 52, Section 47, “Contents of applications; technical information,” require, in the introduction statement, the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. This regulation specifies that the information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The regulation indicates that the NRC will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

Among the specific requirements for a design certification application, the NRC regulations in 10 CFR 52.47(a)(2) require, in part, that the application contain an FSAR that describes the facility, presents the design bases and limits on its operation, and presents a safety analysis of the SSCs and of the facility as a whole, and must include a description and analysis of the SSCs with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The NRC regulations in 10 CFR 52.47(a)(3) specify, in part, that the FSAR describe the design of the facility including (i) the principal design criteria for the facility with reference to 10 CFR Part 50, Appendix A; (ii) the design bases and the relation of the design bases to the principal design criteria; and (iii) information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with an

adequate margin for safety. The NRC regulations in 10 CFR 52.47(b)(1) specify that the application must contain proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the NRC rules and regulations. The NRC regulations in 10 CFR 52.47(c)(2) require, in part, that an application for certification of a nuclear power reactor design that uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design except for site-specific elements; and must meet 10 CFR 50.43(e), which specifies either (1) demonstration of each safety feature, demonstration of acceptable interdependent effects among safety features, and sufficient data exist on safety features to assess the analytical tools for the safety analyses, or (2) acceptable testing of a prototype plant.

Based on the above regulations and the incomplete status of the design and qualification of the ECCS valves and their valve components, the NRC staff requests that the NuScale design certification applicant provide the following information (or the schedule for its availability) either in the FSAR or in documentation for NRC audit:

a. NuScale FSAR Tier 2, Section 6.3 does not describe the specific design aspects of the ECCS valves and their individual valve components, or the connecting hydraulic fluid tubing. In accordance with 10 CFR Part 50, Appendix A, GDC 35 and 37, and 10 CFR 52.47, introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe the specific design of the ECCS valves in the form of design drawings or specifications, including an evaluation of the adequacy of these valves (and their materials) to perform their safety functions over the full range of operating conditions with debris-laden fluid up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients.
b. NuScale FSAR Tier 2, Section 6.3 states that after actuation, the ECCS is a passive system. NuScale FSAR Tier 2, Section 6.3 does not justify the classification of the ECCS valves and their individual valve components as active or passive. In accordance with 10 CFR Part 50, Appendix A, GDC 35 and 37, and 10 CFR 52.47, introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant justify the classification of the ECCS valves and their valve components as active or passive, based on the provisions for operation of the ECCS valves in NuScale FSAR Tier 2, Section 6.3, including the module protection system (MPS) logic providing an actuation signal to open the RVVs as part of the safety-related function for LTOP.
c. The NRC regulations in 10 CFR 50.55a incorporate by reference the ASME *Boiler and Pressure Vessel Code* (BPV Code) with regulatory conditions for the construction and inservice inspection (ISI) of components in nuclear power plants. NuScale FSAR Tier 2, Section 6.3 states that the components of the ECCS valves (valves, hydraulic lines, and actuator assemblies) are Quality Group A, Seismic Category I components

designed to the requirements of ASME BPV Code, Section III, Subsection NB, 2013 Edition. NuScale FSAR Tier 2, Section 6.3, does not provide the level of design information sufficient to reach a conclusion that the NuScale ECCS valves will satisfy the ASME BPV Code of record. In accordance with 10 CFR Part 50, Appendix A, GDC 35 and 37, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe the plans and schedule to implement the requirements in the ASME BPV Code of record as incorporated by reference in 10 CFR 50.55a with regulatory conditions for the ECCS valves and their valve components to satisfy the applicable provisions for material; design; fabrication and installation; examination; testing; overpressure protection; and nameplates, stamping, and reports. Among the specific design aspects to address are (1) the capacity certification for the various fluid conditions (such as steam, liquid, and steam-water transitions) that will be experienced by the ECCS valves over their full range of operating conditions including debris-laden fluid up to and including design-basis accident conditions (such as LOCA and LTOP scenarios); (2) the structural capability of the ECCS solenoid-operated valves to serve as part of the RCPB inside the CNV and the cooling pool based on the statement in Section 6.3 that the ECCS does not extend beyond the CNV boundary; and (3) the structural integrity of the minimally supported hydraulic tubing connecting the ECCS valve components.

d. NuScale FSAR Tier 2, Section 3.9.6.1, “Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints,” specifies that the functional design and qualification of safety-related valves is performed in accordance with ASME Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” as endorsed in NRC Regulatory Guide (RG) 1.100, Revision 3, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” with clarifications as described in Section 3.10.2, “Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation.” NuScale FSAR Tier 2, Section 3.10.2 states that ASME QME-1-2007 is used with the exceptions noted in RG 1.100 (Revision 3) for the qualification of active mechanical equipment. However, the design and qualification of the ECCS valves are not complete. In accordance with 10 CFR Part 50, Appendix A, GDC 35 and 37, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe the specific performance requirements for the ECCS valves that will be demonstrated during the qualification process; and the Qualification Plan including schedule and location for the test program and procedures in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) to demonstrate the seismic/dynamic, environmental, and functional capability of the ECCS valves and their valve components to perform their safety functions over the full range of operating conditions with debris-laden fluid up to and including design-basis accident conditions (such as LOCA and LTOP scenarios) within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. Among the performance characteristics to evaluate are (1) environmental transients,

such as caused by steam leaks or condensation, in the areas surrounding the ECCS valves and their valve components; and (2) the debris limits as specified in NuScale FSAR Tier 2, Section 6.3 that were developed to address Generic Safety Issue GSI-191.

e. Table 6.3-3, "Emergency Core Cooling System Failure Modes and Effects," in NuScale FSAR Tier 2 provides a general summary of a failure modes and effects analysis (FMEA) of the ECCS valves. The summarized table does not address the specific aspects of the design of the ECCS valves and their valve components, and their potential failure modes, causes, and resulting effects. In accordance with 10 CFR Part 50, Appendix A, GDC 35 and 37, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant provide its detailed FMEA for the ECCS valves and their valve components to support their reliability and failure rate assumptions in light of common cause failure considerations such as the following:

   (1) the potential for performance issues to occur during operation of the installed ECCS valves and their four valve components that were not identified as part of their initial design, qualification, and testing in light of the lack of operating and maintenance experience with this FOAK design;
   (2) the potential for incorrect spring force determinations or adjustments for the four valve components of each ECCS valve over their full range of environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients, including improper sealing force in the IAB feature that might result in premature or delayed venting of the hydraulic fluid from the main valve control chamber;
   (3) the potential for blockage of flow through the small tubing connecting the four valve components of each ECCS valve, and the narrow passages in those valve components, as a result of boron precipitation from the reactor coolant used as the hydraulic fluid over the range of credible pool temperatures during the initial reactor module operation up to a full complement of reactor modules, based on the statement in Section 6.3 that analyses show that boron precipitation does not occur at temperatures greater than 80 °F;
   (4) the potential for loss of structural integrity or flow path capability of the small tubing connecting the four valve components of each ECCS valve that might prevent the supply or venting of the main valve control chamber in a timely manner;
   (5) the potential impact on the performance of the ECCS valves and their four valve components and their materials based on the effects of expansion, contraction, and mechanical property changes over the range of environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients;
   (6) the potential for a loss of integrity of the RCPB through the ECCS solenoid-operated valves and their interior parts containing borated reactor coolant extending into the cooling pool;
   (7) the potential for debris-laden reactor coolant to cause adverse flow conditions

through the ECCS valves; and
(8) the potential for incorrect information on the status of core cooling to influence ECCS valve actuation either manually or automatically based on the statement in Section 6.3 that control room indication is provided for a diverse selection of monitored parameters.

f. NuScale FSAR Tier 1, Section 2.1, "NuScale Power Module," specifies proposed ITAAC for the ECCS valves. For example, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," includes ITAAC 14 that requires ECCS valves to change position during a test under design differential pressure; and ITAAC 19 that requires the ECCS valves to fail to their safety-related position during a loss of electrical power test under design differential pressure. NuScale FSAR Tier 1, Section 2.8, "Equipment Qualification," addresses equipment qualification of safety-related electrical and mechanical equipment located in harsh environments and digital instrumentation and controls equipment in mild environments. Table 2.8- 2, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria," includes ITAAC 6 for the functional design and qualification of safety-related valves within the scope of Section 2.8 and references the Functional Qualification Report of ASME QME-1-2007. NuScale FSAR Tier 2, Section 14.3, "Certified Design Material and Inspections, Tests, Analyses, and Acceptance Criteria," Table 14.3-1, "Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference," also includes these proposed ITAAC related to the ECCS valves. With their reference to only differential pressure, the proposed ITAAC in NuScale FSAR Tier 1, Section 2.1 do not provide assurance of the functional capability qualification of the ECCS valves to perform their safety functions over the full range of operating conditions with debris-laden fluid up to and including design-basis accident conditions (such as LOCA and LTOP scenarios) within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. In addition, the NuScale ITAAC do not appear to specify preoperational testing of the ECCS valves that can be correlated to the capability of the valves to perform their safety functions over the full range of operating conditions with debris-laden fluid up to and including design-basis accident conditions (such as LOCA and LTOP scenarios) within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. In accordance with 10 CFR 52.47(b)(2), the NRC staff requests that the NuScale design certification applicant propose appropriate ITAAC to address type testing, preoperational testing, and loss of motive force testing of the ECCS valves.

g. The NRC regulations in 10 CFR 50.55a incorporate by reference the ASME *Operation and Maintenance of Nuclear Power Plants* (OM Code) with regulatory conditions for the inservice testing (IST) of components in nuclear power plants. The NRC regulations in 10 CFR 50.55a(f) require that valves must be designed and provided with access to enable the performance of inservice testing of valves for assessing operational readiness set forth in the ASME OM Code (or NRC accepted ASME OM

Code Cases), incorporated by reference in 10 CFR 50.55a. NuScale FSAR Tier 2, Section 3.9.6.1 indicates that safety-related valves are designed and provided with access to enable the performance of inservice testing to assess operational readiness in accord with the ASME OM Code and as defined in the IST program. NuScale FSAR Tier 2, Section 3.9.6.3, "Inservice Testing Program for Valves," provides a general description of the IST program for valves in the NuScale Power Plant. NuScale FSAR Tier 2, Section 3.9.6.5, "Relief Requests and Alternative Authorizations to the OM Code," states that no relief requests to the ASME OM Code are anticipated for the NuScale Power Plant design. NuScale FSAR Tier 2, Section 6.3 states that preoperational testing of the ECCS function is conducted to ensure that the specified design functions are met during any condition of normal operation, anticipated operational occurrences (AOOs), or postulated accident conditions. Section 6.3 also states that the ECCS valves are power-actuated relief valves (OM Category B/C) that are tested during refueling outages under conditions colder than would exist for a required actuation of the ECCS valves and at a low differential pressure. In accordance with 10 CFR 50.55a(f), the NRC staff requests that the NuScale design certification applicant describe its plans to satisfy the 10 CFR 50.55a regulatory requirements for design and accessibility to perform the preservice and inservice testing specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a to demonstrate the operational readiness of the ECCS valves to perform their safety functions, such as justification for testing these valves under environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) that are less severe than the design-basis accident conditions.

h. NuScale FSAR Tier 2, Section 6.3 describes several instances of operation of the ECCS valves. For example, Section 6.3 states that the MPS logic provides an actuation signal that opens the RVVs in describing the LTOP feature. Later, Section 6.3 states that the ECCS design does not require alternating current (AC) or direct current (DC) power to effectively cool the core. Section 6.3 states that the ECCS does not require operator action or nonsafety-related system support for operation although manual actuation is possible from the control room. Section 6.3 states that the operator can take action to change the position of a malfunctioning valve to its demand position. Section 6.3 states that preoperational testing of the ECCS function is conducted to ensure that the specified design functions are met during any condition of normal operation, AOOs, or postulated accident conditions. These descriptions of the operation of the ECCS valves in NuScale FSAR Tier 2, Section 6.3 do not discuss the IAB feature that can prevent the main control chamber from being pressurized or vented for operation of the main valve. In addition, the statement regarding the absence of a need for AC or DC power is not clear in comparison to other statements in the FSAR regarding ECCS operation. In accordance with 10 CFR Part 50, Appendix A, GDC 35 and 37, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe its plans to open or close the ECCS valves during or following postulated plant events, including consideration of the IAB feature and the need for electric power.

NuScale Response:

Question 03.09.06-1(a): The NRC Staff requests a description of the specific design of the ECCS valves in the form of design drawings or specifications, including an evaluation of the adequacy of these valves (and their materials) to perform their safety functions over the full range of operating conditions with debris-laden fluid up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients.

NuScale response to Question 03.09.06-1(a): The design of the ECCS is described in three ways: an ASME design specification, detailed drawings and the ECCS System Design Description. The ASME design specification establishes the design and operational requirements that the equipment vendor must meet to provide an acceptable ECCS valve design. The ASME Design Specification provides the basis for the licensing commitments found in the DCA regarding ECCS valve performance. It also provides design basis loads, environmental conditions, and seismic classification. Detailed drawings and the ECCS System Design Description describe the requirements for the ECCS valve design. Additional detailed design drawings and documents are currently being developed to clarify the physical valve design and operational details. NuScale has previously provided the ASME design specification and ECCS System Design Description, for NRC staff audit. The additional design drawings are expected to be available for audit in October, 2017.

NuScale test programs described in FSAR Tier 2, Section 1.5 verify safety analysis models for use in LOCA, non-LOCA and long-term cooling applications and demonstrate the performance of unique design features and components. The safety analyses model the ECCS valves in accordance with specified design attributes described in FSAR Tier 2, Section 6.3.2.2. These attributes ensure the ECCS adequately perform functions modeled by safety analyses. The ECCS valves themselves are hydraulically closed, spring-assist to open power-actuated relief valves. Qualification testing performed in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) will verify that the installed ECCS valves are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions and temperature conditions, up to and including DBA conditions (Refer to Table 14.3-1, ITAAC No. 02.08.06). In addition, NuScale plans to perform testing and analyses to provide reasonable assurance that the ECCS valves perform as specified. A qualification plan which will address the specific performance requirements for the ECCS valves is expected to be available for audit in December 2017.

NuScale expects to complete the initial tests and analyses of the ECCS valves by December, 2019. Test plans for this initial testing are expected to be available for NRC staff audit by May, 2018. The scheduling of the testing and test plans are tentative pending issue of the associated contracts.

The NuScale design allows for strict specification of containment component materials such that

there are no credible debris sources within the containment or RCS that could impact the operation of the ECCS system. A detailed assessment of potential debris loads and analysis of their potential impact on ECCS system operation is provided in Section 6.3.3.1 of the NuScale DCD.

Question 03.09.06-1(b): The NRC Staff requests justification of the classification of the ECCS valves and their components as active or passive, based on the provisions for operation of the ECCS valves in NuScale FSAR Tier 2, Section 6.3, including the module protection system (MPS) logic providing an actuation signal to open the RVVs as part of the safety-related function for LTOP.

NuScale response to Question 03.09.06-1(b): The ECCS valve is considered an active mechanical component subject to active failure consideration, as discussed by FSAR Section 15.0.0.5. No single failure prevents the ECCS from performing its safety function, including single failures in electrical power (single failures in onsite power and offsite power, busses, electrical and mechanical parts, cabinets and wiring), initiation logic, and single active or passive component failure (refer to Section 15.0.0.6.4). With either a valid actuation signal or if electrical power is unavailable, the ECCS valves open once RCS pressure goes below the inadvertent actuation block (IAB) pressure locking threshold. The IAB is considered a passive mechanical component. Justification for IAB classification is discussed in the response to RAI 15-2.

Question 03.06.09-1(c): The NRC Staff requests a description of the plans and schedule to implement the requirements in the ASME BPV Code of record as incorporated by reference in 10 CFR 50.55a with regulatory conditions for the ECCS valves and their valve components to satisfy the applicable provisions for material; design; fabrication and installation; examination; testing; overpressure protection; and nameplates, stamping, and reports. Among the specific design aspects to address are (1) the capacity certification for the various fluid conditions (such as steam, liquid, and steam-water transitions) that will be experienced by the ECCS valves over their full range of operating conditions including debris-laden fluid up to and including design-basis accident conditions (such as LOCA and LTOP scenarios); (2) the structural capability of the ECCS solenoid-operated valves to serve as part of the RCPB inside the CNV and the cooling pool based on the statement in Section 6.3 that the ECCS does not extend beyond the CNV boundary; and (3) the structural integrity of the minimally supported hydraulic tubing connecting the ECCS valve components.

NuScale response to Question 03.0.09-1(c): The ECCS valves are ASME Code Class 1 components designed to be constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 2013 Edition. NuScale has developed an ASME design specification for use in developing the ECCS valve design and the NuScale design certification application.

A COL applicant that references the NuScale Power Plant design will develop design reports in accordance with the requirements outlined under ASME BPVC, Section III for the ECCS valves (Reference COL Item 3.9-2). The ASME design reports developed per this COL Item provide

assurance that the ECCS valves as designed meet the applicable requirements for material; design; fabrication and installation; examination; testing; overpressure protection; and nameplates, stamping, and reports. Approval of the final ASME design report will include addressing specific aspects such as valve capacity in terms of meeting the required performance criteria for Cv and Xt and the structural capability of the ECCS main valves and their supporting trip and reset solenoid-operated assemblies to serve as part of the RCPB and CNV boundary, respectively.

The hydraulic tubing between the ECCS main valve and the trip and reset assembly is small bore tubing and is covered by a separate ASME design specification and will be addressed in a separate design report. The design of this tubing in accordance with the appropriate ASME Code provides reasonable reassurance that it will maintain its structural integrity. The final design report for this tubing will be developed per COL item 3.9-2.

NuScale has provided the current ASME design specification for the ECCS valves for NRC staff audit.

Question 03.09.06-1(d): The NRC Staff requests a description of the specific performance requirements for the ECCS valves that will be demonstrated during the qualification process; and the Qualification Plan including schedule and location for the test program and procedures in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) to demonstrate the seismic/dynamic, environmental, and functional capability of the ECCS valves and their valve components to perform their safety functions over the full range of operating conditions with debris-laden fluid up to and including design-basis accident conditions (such as LOCA and LTOP scenarios) within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. Among the performance characteristics to evaluate are (1) environmental transients, such as caused by steam leaks or condensation, in the areas surrounding the ECCS valves and their valve components; and (2) the debris limits as specified in NuScale FSAR Tier 2, Section 6.3 that were developed to address Generic Safety Issue GSI-191.

NuScale response to Question 03.09.06-1(d): A qualification plan which will address the specific performance requirements for the ECCS valves is expected to be available for audit in December 2017. The qualification plan will describe the specific performance requirements and environmental conditions for qualification in accordance with QME-1-2007 as accepted in RG 1.100 (Revision 3). Implementation of the qualification plan is the responsibility of the COL applicant. Therefore, the qualification plan that is provided for audit will not include a test schedule and location.

ITAAC No. 02.08.06 will verify that the ECCS valves are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions and temperature conditions, up to and including DBA conditions (Refer to Table 14.3-1). Flow capacity certification will be included in the qualification report. The report will qualify the valves for service in design basis conditions, including environmental transients under design basis fluid conditions.

Question 03.09.06-1(e): The NRC Staff requests NuScale to provide its detailed FMEA for the ECCS valves and their valve components to support their reliability and failure rate assumptions in light of common cause failure considerations such as the following:

e(1) Previously unidentified failure mechanisms:

e(2) Incorrect spring force determinations;

e(3) Tubing flow blockage;

e(4) Loss of tubing structural integrity or flow capability;

e(5) Effect of material property changes;

e(6) Pressure boundary failures;

e(7) Adverse flow conditions due to debris laden reactor coolant fluid;

e(8) Incorrect status indication adversely impacting ECCS valve manual or automatic operation.

NuScale response to Question 03.09.06-1(e): A subcomponent level FMEA for the ECCS valves is expected to be available for audit in December, 2017. The subcomponent level FMEA will identify failure modes for each subcomponent of the valve assembly. Each failure mode will be analyzed for its effect on the valve system (not its effect on the reactor system). If a failure mode is determined to be not credible, then justification will be provided.

Question 03.09.06-1(f): The NRC Staff requests that appropriate ITAAC be proposed to address type testing, preoperational testing, and the loss of motive force testing of the ECCS valves.

NuScale response to Question 03.09.06-1(f): Tier 1 Table 2.8-2, Equipment Qualification ITAAC currently contains the following ITAAC to satisfy the NRC request to provide ITAAC for type testing of ECCS valves. This ITAAC is consistent with Standardized DCA ITAAC number Q06 contained in a NRC letter to NuScale Power, entitled “Transmittal of Draft Standard Inspections, Tests, Analysis and Acceptance Criteria”, dated April 8, 2016 (ML16096A132), as modified by NuScale letter to the NRC, entitled “NuScale Power, LLC Submittal of Additional Meeting Material for Use During Public Meeting on July 20, 2016”, dated July 13, 2016 (ML16195A391).

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.	The safety-related valves are functionally designed and qualified to perform their safety-related function under the full range of fluid flow, differential pressure, electrical conditions, and temperature conditions up to and including DBA conditions.	A type test or a combination of type test and analysis will be performed of the safety-related valves.	A Functional Qualification Report exists and concludes that the safety-related valves listed in Table 2.8-1 are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions, and temperature conditions up to and including DBA conditions.

Tier 1 Table 2.1-4, NuScale Power Module Inspections, Tests, Analysis and Acceptance Criteria currently contains the following ITAAC to satisfy the NRC request to provide ITAAC for loss of motive force testing of ECCS valves. This ITAAC is consistent with Standardized DCA ITAAC number M08 contained in a NRC letter to NuScale Power, entitled “Transmittal of Draft Standard Inspections, Tests, Analysis and Acceptance Criteria”, dated April 8, 2016 (ML16096A132), as modified by NuScale letter to the NRC, entitled “NuScale Power, LLC Submittal of Additional Meeting Material for Use During Public Meeting on July 20, 2016”, dated July 13, 2016 (ML16195A391).

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19.	The ECCS safety-related RRVs and RVVs fail to their safety-related position on loss of electrical power to their corresponding trip valves under design differential pressure.	A test will be performed of the ECCS safety-related RRVs and RVVs.	Each ECCS safety-related RRV and RVV listed in Table 2.1-2 fails open on loss of electrical power to its corresponding trip valve.

Establishing ECCS component adequacy during design bases conditions is completed via qualification testing in accordance with QME-1-2007 and the ASME OM Code Testing Program. Design bases conditions for the ECCS cannot be replicated during preoperational testing. Preoperational and inservice testing as prescribed by the NuScale design certification provide reasonable assurance of the valves to function as designed when called upon.

Question 03.09.06-1(g): The NRC Staff requests a description of the plans to satisfy the 10 CFR 50.55a regulatory requirements for design and accessibility to perform the preservice and inservice testing specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a to demonstrate the operational readiness of the ECCS valves to perform their safety

functions, such as justification for testing these valves under environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) that are less severe than the design-basis accident conditions.

NuScale response to Question 03.09.06-1(g): The ECCS valves are ASME Section III, Class 1 valves that are designed and provided access to enable the performance of inservice testing to assess operational readiness in accordance with the ASME OM Code as defined in the inservice testing program (refer to Section 3.9.6.1). Working platforms are provided to facilitate preservice and inservice testing of the ECCS valves. Containment drawings depicting these platforms, demonstrating accessibility, will be made available for NRC staff audit.

As described in Tier 2, Chapter 6.3 and 3.9.6, the ECCS valves are designed in accord with the ASME BPV, will be qualified to ASME QME-1 and periodically tested in accord with the ASME OM Code as endorsed in 10CFR50.55a. NuScale has not taken any exception to the BPV, OM Code or 10CFR50.55a and thus satisfies the regulatory requirements for design and accessibility to perform preservice and inservice testing of the ECCS valves. The design, preoperational and inservice testing as described in the NuScale design certification provide reasonable assurance of the valves to function as designed when called upon.

Question 03.09.06-1(h): Descriptions of the operation of the ECCS valves in NuScale FSAR Tier 2, Section 6.3 do not discuss the IAB feature that can prevent the main control chamber from being pressurized or vented for operation of the main valve. In addition, the statement regarding the absence of a need for AC or DC power is not clear in comparison to other statements in the FSAR regarding ECCS operation. The NRC Staff requests a description of plans to open or close the ECCS valves during or following postulated plant events, including consideration of the IAB feature and the need for electric power.

NuScale response to Question 03.09.06-1(h): The IAB feature is described separately in section 6.3.2.2 and is designed to reduce the frequency of inadvertent operation (opening) during power operations. The IAB consists of a block valve with a spring-loaded disc that functions to block venting of the main valve control chamber when RCS pressure is significantly higher than CNV pressure. The IAB does not rely on AC or DC power to operate. The function of the IAB feature is passive and not required to be analyzed as a single active failure in conjunction with other event initiations, as discussed by the response to eRAI 8815, Question 15-2.

For all cases other than those where an inadvertent ECCS valve opening is the initiating event, the IAB feature of the ECCS will function to maintain the main valves in the closed position until the differential between RCS and CNT pressure is below the release point. For all cases described in this question, opening of the ECCS valves is accomplished by removing power from the associated trip valve, which vents the downstream side of the IAB, which in turn will allow the spring forces of the valve and the RCS system pressure to open the valve once below the release point. The LTOP function of the ECCS valves is maintained by selecting an IAB setpoint pressure that does not impede the ECCS valves' ability to provide adequate low pressure protection.

The design does not require the ECCS valves to be closed during or following any postulated

plant events. The ECCS valves are only required to be maintained closed during normal operation in order to maintain a pressure boundary between the RCS and CNT. After opening, the only time during which the ECCS valves must be reclosed is to support plant startup operations. This includes during normal cold shutdown and refueling operations, during which the ECCS valves remain open.

Impact on DCA:
There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8820
Date of RAI Issue: 06/02/2017

NRC Question No.: 03.09.06-2

Section 6.2.4, "Containment Isolation System," in the NuScale FSAR Tier 2 for the containment system (CNTS) states that the containment boundary is formed by the containment vessel (CNV) and by the containment isolation valves (CIVs) and the passive containment isolation barriers that are used to prevent releases through the penetrations in the CNV. Section 6.2.4 indicates that there are eight mechanical penetrations through the CNV top head with two hydraulically operated primary system containment isolation valves (PSCIVs) in series outside of the CNV in lines connected to the reactor coolant pressure boundary (RCPB) or open to the atmosphere inside of the CNV. Section 6.2.4 also indicates that there are four mechanical penetrations through the CNV top head with a single hydraulically operated secondary system containment isolation valve (SSCIV) in lines outside of the CNV for piping inside of the CNV for a closed piping system and not connected to the RCPB or the atmosphere inside of the CNV. Section 6.2.4 states that the PSCIVs have a design with a configuration of two valves (with separate actuators and ball-valve obturators) contained in a single body. Section 6.2.4 indicates that the PSCIVs will include a design feature to allow excess pressure caused by heatup of fluid between its two valves to be released into the CNV. The SSCIVs use a single ball-valve design. The main steam isolation valves (MSIVs) are specified as single SSCIVs. The Feedwater Isolation Valves (FWIVs) are specified as SSCIVs, but also have a feedwater isolation check valve housed in the same valve body. Figure 6.2-6b, "Feedwater Isolation Valve with Nozzle Check Valve and Actuator Assembly," shows a nozzle check valve in the same valve body with the FWIV. Section 6.2.4 states that hydraulic actuators with nitrogen gas cylinders are used to operate both the PSCIV and SSCIV designs. It is the NRC staff's understanding that the design and qualification of the CIVs with their first-of-a-kind (FOAK) design features have not been completed.

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. With respect to CIVs, General Design Criterion (GDC) 54, "Piping systems penetrating containment," in 10 CFR Part 50, Appendix A, requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy,

reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. GDC 54 also requires that such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits. GDC 55, “Reactor coolant pressure boundary penetrating containment,” in 10 CFR Part 50, Appendix A, requires that each line that is part of the RCPB and that penetrates primary reactor containment shall be provided with CIVs as specified in this GDC, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. GDC 56, “Primary containment isolation,” in 10 CFR Part 50, Appendix A, requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with CIVs as specified in this GDC, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. GDC 57, “Closed system isolation valves,” in 10 CFR Part 50, Appendix A, requires that each line that penetrates primary reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere shall have at least one CIV which shall be either automatic, or locked closed, or capable of remote manual operation.

For a nuclear reactor design certification application, the NRC regulations in 10 CFR Part 52, Section 47, “Contents of applications; technical information,” require, in the introduction statement, the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. This regulation specifies that the information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The regulation indicates that the NRC will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

Among the specific requirements for a design certification application, the NRC regulations in 10 CFR 52.47(a)(2) require, in part, that the application contain an FSAR that describes the facility, presents the design bases and limits on its operation, and presents a safety analysis of the SSCs and of the facility as a whole, and must include a description and analysis of the SSCs with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The NRC regulations in 10 CFR 52.47(a)(3) specify, in part, that the FSAR describe the design of the facility including (i) the principal design criteria for the facility with reference to 10 CFR Part 50, Appendix A; (ii) the design bases and the relation of the design bases to the principal design criteria; and (iii) information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with an adequate margin for safety. The NRC regulations in 10 CFR 52.47(b)(1) specify that the

application must contain proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the NRC rules and regulations. The NRC regulations in 10 CFR 52.47(c)(2) require, in part, that an application for certification of a nuclear power reactor design that uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design except for site-specific elements; and must meet 10 CFR 50.43(e), which specifies either (1) demonstration of each safety feature, demonstration of acceptable interdependent effects among safety features, and sufficient data exist on safety features to assess the znalytical tools for the safety analyses, or (2) acceptable testing of a prototype plant.

Based on the above regulations and the incomplete status of the design and qualification of the CIVs, the NRC staff requests that the NuScale design certification applicant provide the following information (or the schedule for its availability) either in the FSAR or in documentation for NRC audit:

a. NuScale FSAR Tier 2, Section 6.2.4 does not describe the specific design aspects of the CIVs. In accordance with 10 CFR Part 50, Appendix A, GDC 54, 55, 56, and 57, and 10 CFR 52.47, introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe the specific design of the CIVs in the form of design drawings or specifications, including an evaluation of the adequacy of these valves (and their materials) to perform their safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients.
b. The NRC regulations in 10 CFR 50.55a incorporate by reference the ASME *Boiler and Pressure Vessel Code* (BPV Code) with regulatory conditions for the construction and inservice inspection (ISI) of components in nuclear power plants. NuScale FSAR Tier 2, Section 6.2.4 states that PSCIVs in lines that directly contact the reactor coolant during normal operation are designed and constructed in accordance with the ASME *Boiler and Pressure Vessel Code* (BPV Code), Section III, Class 1, Subsection NB, Quality Group A, and Seismic Category I criteria. The PSCIVs in the other lines are designed and constructed as Class 1; however, these valves are classified the same as the lines, which are not part of the RCPB inside of containment. These PSCIVs are Quality Group B components and are in lines designed and constructed in accordance with ASME BPV Code, Section III, Class 2, Subsection NC, Quality Group B, and Seismic Category I criteria. The SSCIVs are designed to the ASME BPV Code, Section III, Class 2, Subsection NC, Quality Group B, and Seismic Category I criteria. NuScale FSAR Tier 2, Section 6.2.4 does not provide the level of design

information sufficient to reach a conclusion that the NuScale CIVs will satisfy the ASME BPV Code of record. In accordance with 10 CFR Part 50, Appendix A, GDC 54, 55, 56, and 57, and 10 CFR 52.47, introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe the plans and schedule to implement the requirements in the ASME BPV Code of record as incorporated by reference in 10 CFR 50.55a with regulatory conditions for CIVs to satisfy the applicable provisions for material; design; fabrication and installation; examination; testing; and nameplates, stamping, and reports. Among the design aspects to address are (1) the dual ball-valve design of the PSCIVs in a single valve body; (2) the design feature of the PSCIV to allow excess pressure caused by heatup of fluid between its two valves to be released into the CNV; and (3) the FWIV design with a ball valve and nozzle-check valve in the same valve body.

c. NuScale FSAR Tier 2, Section 3.9.6.1, "Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints," specifies that the functional design and qualification of safety-related valves is performed in accordance with ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as endorsed in NRC Regulatory Guide (RG) 1.100, Revision 3, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with clarifications as described in Section 3.10.2, "Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation." NuScale FSAR Tier 2, Section 3.10.2 states that ASME QME-1-2007 is used with the exceptions noted in RG 1.100 (Revision 3) for the qualification of active mechanical equipment. However, the design and qualification of the CIVs are not complete. In accordance with 10 CFR Part 50, Appendix A, GDC 54, 55, 56, and 57, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe the specific performance requirements for the CIVs that will be demonstrated during the qualification process; and the Qualification Plan including schedule and location for the test program and procedures in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) to demonstrate the seismic/dynamic, environmental, and functional capability of the CIVs to perform their safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. Among the performance requirements specified in NuScale FSAR Tier 2, Section 6.2.4, to be addressed are (1) the PSCIVs will be designed to stop line break flow within a 5-second valve stroke time; (2) the MSIVs will be capable of stopping fully developed steam line break flows of 100% and 4% steam conditions within

a 5-second valve stroke time while the main steam isolation bypass valve is capable of closure within 10 seconds of receipt of a closure signal or loss of power; (3) the FWIVs will be capable of stopping fully developed feedwater line break flows of 200% in the forward direction and together with the internal safety-related check valve are capable of closure within 1 second on fully-developed reverse flow; and (4) the containment parameters in Table 6.2-1, "Containment Design and Operating Parameters," which does not appear to include data for all of the listed parameters.

d. NuScale FSAR Tier 2, Section 6.2.4 does not describe a failure modes and effects analysis (FMEA) of the CIVs to address the specific aspects of various CIV designs, and their potential failure modes, causes, and resulting effects. In accordance with 10 CFR Part 50, Appendix A, GDC 54, 55, 56, and 57, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant provide its FMEA for the various CIV designs to support their reliability and failure rate assumptions in light of common cause failure considerations such as the following:

   (1) the potential for performance issues to occur during operation of CIVs that were not identified as part of their initial design, qualification, and testing for their FOAK features, such as the PSCIVs with dual ball-valve actuators and obturators in a single valve body, and those SSCIVs that include a ball valve and a nozzle check valve in a single valve body;
   (2) the potential for the solenoid valves used to vent the hydraulic fluid to a reservoir to not provide closure of the CIVs in a timely manner;
   (3) the potential for improper operation of the design feature of each PSCIV that is intended to allow excess pressure caused by heatup of fluid between its two valves to be released into the CNV, such as inadequate release of valve interior pressure or inadequate isolation with radioactive release outside of the CNV; and
   (4) the potential for the CIV nitrogen gas cylinders to not provide CIV closure within the required time (such as 5 seconds for the MSIVs) with the specified fluid conditions and steam content, and environmental conditions; or to not adequately maintain the CIVs closed for the full duration of postulated plant events.

e. NuScale FSAR Tier 1, Section 2.1, "NuScale Power Module," specifies proposed ITAAC for the CIVs. For example, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," includes ITAAC 8 for CIV closure time, ITAAC 13 for CNTS valve testing under design differential pressure, ITAAC 18 for CNTS valve loss of power testing under design differential pressure, and ITAAC 21 for CNTS check valve testing under design differential pressure and flow. NuScale FSAR Tier 1, Section 2.8, "Equipment Qualification," addresses equipment qualification of

safety-related electrical and mechanical equipment located in harsh environments and digital instrumentation and controls equipment in mild environments. Table 2.8-2, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria," includes ITAAC 6 for the functional design and qualification of safety-related valves within the scope of Section 2.8, and references the Functional Qualification Report of ASME QME-1-2007. NuScale FSAR Tier 2, Section 14.3, Table 14.3-1, "Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference," also includes these proposed ITAAC related to the CIVs. The ITAAC proposed in NuScale FSAR Tier 1, Section 2.1 do not provide assurance of the functional capability qualification of the CIVs to perform their safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. In addition, the NuScale ITAAC do not appear to specify preoperational testing of the CIVs that can be correlated to the capability of the valves to perform their safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. In accordance with 10 CFR 52.47(b)(2), the NRC staff requests that the NuScale design certification applicant propose appropriate ITAAC to address type testing, preoperational testing, and loss of motive force testing of the various CIVs.

f. The NRC regulations in 10 CFR 50.55a incorporate by reference the ASME *Operation and Maintenance of Nuclear Power Plants* (OM Code) with regulatory conditions for the inservice testing (IST) of components in nuclear power plants. The NRC regulations in 10 CFR 50.55a(f) require that valves must be designed and provided with access to enable the performance of inservice testing of valves for assessing operational readiness set forth in the ASME OM Code (or NRC accepted ASME OM Code Cases), incorporated by reference in 10 CFR 50.55a. NuScale FSAR Tier 2, Section 3.9.6.1 indicates that safety-related valves are designed and provided with access to enable the performance of inservice testing to assess operational readiness in accord with the ASME OM Code and as defined in the IST program. NuScale FSAR Tier 2, Section 3.9.6.3, "Inservice Testing Program for Valves," provides a general description of the IST program for valves in the NuScale Power Plant. NuScale FSAR Tier 2, Section 3.9.6.5, "Relief Requests and Alternative Authorizations to the OM Code," states that no relief requests to the ASME OM Code are anticipated for the NuScale Power Plant design. NuScale FSAR Tier 2, Section 6.2.4 states that the periodic testing program meets the ASME OM

Code in accordance with 10 CFR 50.55a, but does not demonstrate that all ASME OM Code provisions can be achieved for the CIVs with their FOAK design features. In accordance with 10 CFR 50.55a(f), the NRC staff requests that the NuScale design certification applicant describe its plans to satisfy the 10 CFR 50.55a regulatory requirements for design and accessibility to perform the preservice and inservice testing specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a to demonstrate the operational readiness of the CIVs to perform their safety functions.

g. NuScale FSAR Tier 2, Section 6.2.4 states that hydraulic actuators with nitrogen gas cylinders are used to operate the PSCIV and SSCIV designs. Section 6.2.4 also indicates that dual solenoid valves are positioned in the supply side of each hydraulic line, and are de-energized to vent the hydraulic fluid in a supply line to a reservoir which depressurizes the valve hydraulic supply and allows the stored energy in the nitrogen cylinders to close the valve. Section 6.2.4 does not discuss subsequent operation of the CIVs during or following postulated plant events. In accordance with 10 CFR Part 50, Appendix A, GDC 54, 55, 56, and 57, and 10 CFR 52.47 introduction, (a)(2), (a)(3), and (c)(2), the NRC staff requests that the NuScale design certification applicant describe any plans to close and later re-open CIVs during or following postulated plant events; and the design, qualification, and testing to demonstrate the functional capability qualification of the CIVs to perform those safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients.

NuScale Response:

Question 03.09.06-2(a): The NRC Staff requests a description of the specific design of the CIVs in the form of design drawings or specifications, including an evaluation of the adequacy of these valves (and their materials) to perform their safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients.

NuScale response to Question 03.09.06-2(a): The design of the containment isolation valves (CIVs) is represented by a design specification, component drawings, and design documents. The design specification establishes the requirements that the vendor must meet to provide an acceptable CIV design. The design specification also provides the basis for the licensing commitments found in the DCA regarding CIV performance. It also provides design basis loads, environmental conditions, and seismic classification. It provides requirements used to generate an ASME Design Report. Component drawings and additional design documents describe the

CIV design and how it meets the requirements set forth in the design specification. These detailed design drawings and engineering documents are currently being developed. The ASME design specification for the CIVs has been previously made available for NRC staff audit. Component drawings are expected to be available for audit in January, 2018. Other design documents are expected to be available for NRC staff audit in January, 2018.

The primary system CIVs are dual ball valve, single valve body design and the secondary system CIVs are single ball valve, single body design, as discussed in FSAR Section 6.2.4.2.2. Qualification testing performed in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) will verify that the installed CIVs are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions and temperature conditions, up to and including DBA conditions (Refer to Table 14.3-1, ITAAC No. 02.08.06). In addition, NuScale plans to perform testing and analyses to provide reasonable assurance that the CIVs perform as specified. A qualification plan which will address the specific performance requirements for the CIVs is expected to be available for audit in the third quarter of 2018.

NuScale expects to complete the initial tests and analyses of the CIVs in December, 2019. Test plans for this initial testing are expected to be made available for NRC staff audit in December, 2018. The scheduling of the testing and test plans are tentative pending issue of the associated contracts.

Question 03.09.06-2(b): The NRC Staff requests a description of the plans and schedule to implement the requirements in the ASME BPV Code of record as incorporated by reference in 10 CFR 50.55a with regulatory conditions for CIVs to satisfy the applicable provisions for material; design; fabrication and installation; examination; testing; and nameplates, stamping, and reports. Among the design aspects to address are (1) the dual ball-valve design of the PSCIVs in a single valve body; (2) the design feature of the PSCIV to allow excess pressure caused by heatup of fluid between its two valves to be released into the CNV; and (3) the FWIV design with a ball valve and nozzle-check valve in the same valve body.

NuScale response to Question 03.09.06-2(b): The CIVs are ASME Code Class 1 components designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 2013 Edition. The Design specification dictates the loads, load combinations, and environmental conditions for component analyses to be included in the ASME design report. NuScale has developed an ASME design specification for use in developing the CIV design and the NuScale design certification application.

A COL applicant that references the NuScale Power Plant design will develop design reports in accordance with the requirements outlined under ASME BPVC, Section III for the CIVs (Reference COL Item 3.9-2). The ASME design reports developed per this COL provide assurance that the CIVs as designed meet the applicable requirements for material; design; fabrication and installation; examination; testing; overpressure protection; and nameplates, stamping, and reports. Approval of the final ASME design report will include addressing specific aspects such as the structural capability of the CIVs to serve as part of the CNV boundary.

NuScale has provided the current ASME design specification for the CIVs for NRC staff audit.

Question 03.09.06-2(c): The NRC Staff requests a description of the CIV performance requirements that will be demonstrated during the qualification process and the Qualification Plan including schedule and location for the test program and procedures in accordance with ASME QME-1-2007 as accepted in RG 1.100 (Revision 3) to demonstrate the seismic/dynamic, environmental, and functional capability of the CIVs to perform their safety functions over the full range of operating conditions up to and including design-basis accident conditions within their required operating times under the applicable environmental conditions (such as pressure; temperature; humidity; steam or liquid; and radiation) and their potential transients. Among the performance requirements specified in NuScale FSAR Tier 2, Section 6.2.4, to be addressed are (1) the PSCIVs will be designed to stop line break flow within a 5-second valve stroke time; (2) the MSIVs will be capable of stopping fully developed steam line break flows of 100% and 4% steam conditions within a 5-second valve stroke time while the main steam isolation bypass valve is capable of closure within 10 seconds of receipt of a closure signal or loss of power; (3) the FWIVs will be capable of stopping fully developed feedwater line break flows of 200% in the forward direction and together with the internal safety-related check valve are capable of closure within 1 second on fully-developed reverse flow; and (4) the containment parameters in Table 6.2-1, “Containment Design and Operating Parameters,” which does not appear to include data for all of the listed parameters.

NuScale response to Question 03.09.06-2(c): A qualification plan which will address the specific performance requirements for the CIVs is expected to be available for audit in the third quarter of 2018. The qualification plan will describe the specific performance requirements and environmental conditions for qualification in accordance with QME-1-2007 as accepted in RG 1.100 (Revision 3). Implementation of the qualification plan is the responsibility of the COL applicant. Therefore, the qualification plan provided for audit will not include a test schedule and location.

ITAAC No. 02.08.06 will verify that the CIVs are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions and temperature conditions, up to and including DBA conditions (Refer to Table 14.3-1). The report will qualify the valves for service in design basis conditions, including environmental transients under design basis fluid conditions.

Question 03.06.09-2(d): The NRC Staff requests NuScale to provide its detailed FMEA for the CIVs and their valve components to support their reliability and failure rate assumptions in light of common cause failure considerations such as the following:

1) The potential for performance issues to occur that were not previously identified;

2) The potential for solenoid valves to not provide CIV closure in a timely manner;

3) The potential for improper operation due to excessive heatup of fluid between the two valves;

4) The potential for the CIV nitrogen gas cylinders to not provide CIV closure within the required time.

NuScale response to Question 03.09.06-2(d): A subcomponent-level FMEA for the CIVs is expected to be available for audit by January, 2019. A subcomponent level FMEA identifies failure modes for each subcomponent of the valve assembly. Each failure mode will be analyzed for its effect on the valve system (not its effect on the reactor system). If a failure mode is determined to be not credible then justification will be provided.

Question 03.09.06-2(e): The NRC Staff requests that appropriate ITAAC be proposed to address type testing, preoperational testing, and the loss of motive force testing of the various CIVs.

NuScale response to Question 03.09.06-2(e): Tier 1 Table 2.8-2, Equipment Qualification ITAAC currently contains the following ITAAC to satisfy the NRC request to provide ITAAC for type testing of the CIVs. This ITAAC is consistent with Standardized DCA ITAAC number Q06 contained in a NRC letter to NuScale Power, entitled “Transmittal of Draft Standard Inspections, Tests, Analysis and Acceptance Criteria”, dated April 8, 2016 (ML16096A132), as modified by NuScale letter to the NRC, entitled “NuScale Power, LLC Submittal of Additional Meeting Material for Use During Public Meeting on July 20, 2016”, dated July 13, 2016 (ML16195A391).

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.	The safety-related valves are functionally designed and qualified to perform their safety-related function under the full range of fluid flow, differential pressure, electrical conditions, and temperature conditions up to and including DBA conditions.	A type test or a combination of type test and analysis will be performed of the safety-related valves.	A Functional Qualification Report exists and concludes that the safety-related valves listed in Table 2.8-1 are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions, and temperature conditions up to and including DBA conditions.

Tier 1 Table 2.1-4, NuScale Power Module Inspections, Tests, Analysis and Acceptance Criteria currently contains the following ITAAC to satisfy the NRC request to provide ITAAC for loss of motive force testing of the CIVs. This ITAAC is consistent with Standardized DCA ITAAC number M08 contained in a NRC letter to NuScale Power, entitled “Transmittal of Draft Standard Inspections, Tests, Analysis and Acceptance Criteria”, dated April 8, 2016

(ML16096A132), as modified by NuScale letter to the NRC, entitled “NuScale Power, LLC Submittal of Additional Meeting Material for Use During Public Meeting on July 20, 2016”, dated July 13, 2016 (ML16195A391).

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18.	The CNTS safety-related hydraulic operated valves fail to their safety-related position on loss of electrical power under design differential pressure.	A test will be performed of the CNTS safety-related hydraulic-operated valves.	Each CNTS safety-related hydraulic-operated valve listed in Table 2.1-2 fails to its safety-related position on loss of motive power.

Establishing CIV component adequacy during design bases conditions is completed via qualification testing in accordance with QME-1-2007 and the ASME OM Code Testing Program. Design bases conditions for the CIVs cannot be replicated during preoperational testing. Preoperational and inservice testing as prescribed by the NuScale design certification provide reasonable assurance of the valves to function as designed when called upon.

Question 03.09.06-2(f): The NRC staff requests that the NuScale design certification applicant describe its plans to satisfy the 10 CFR 50.55a regulatory requirements for design and accessibility to perform the preservice and inservice testing specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a to demonstrate the operational readiness of the CIVs to perform their safety functions.

NuScale response to Question 03.09.06-2(f): The CIVs are ASME Section III, Class 1 valves that are designed and provided access to enable the performance of inservice testing to assess operational readiness in accordance with the ASME OM Code as defined in the inservice testing program (refer to Section 3.9.6.1). Working platforms are provided to facilitate preservice and inservice testing of the CIVs. Containment drawings depicting these platforms, demonstrating accessibility, will be made available for NRC staff audit.

As described in Tier 2, Chapter 6.2 and 3.9.6, the CIVs are designed in accord with the ASME BPV, will be qualified to ASME QME-1 and periodically tested in accord with the ASME OM Code as endorsed in 10CFR50.55a. NuScale has not taken any exception to the BPV, OM Code or 10CFR50.55a and thus satisfies the regulatory requirements for design and accessibility to perform preservice and inservice testing of the CIVs. The design, preoperational and inservice testing as described in the NuScale design certification provide reasonable assurance of the valves to function as designed when called upon.

Question 03.09.06-2(g): The NRC Staff requests a description of any plans to close and later re-open the CIVs during or following postulated plant events and the design, qualification and testing to demonstrate the capability of the valves to perform those safety related functions over their full range of operating conditions.

NuScale response to Question 03.09.06-2(g): NuScale has no plans to close and later re-open the CIVs to mitigate any design basis event. Therefore, the CIVs have no safety related function to re-open following any postulated plant event.

Impact on DCA:
There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8820
Date of RAI Issue: 06/02/2017

NRC Question No.: 03.09.06-3

In response to performance issues with power-operated valves at operating nuclear power plants as a result of inadequacies in the design and qualification process for those valves to perform their safety functions, the NRC has required that the process for qualifying the capability of power-operated valves to perform their safety functions described in design certification applications to be specified as Tier 2* information for specific new reactor designs to prevent modifications to the valve qualification process without prior NRC review. NuScale FSAR Tier 2, Section 3.9.6 specifies that safety-related valves will satisfy the qualification provisions of ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as endorsed in NRC Regulatory Guide (RG) 1.100, Revision 3, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," with clarifications as described in Section 3.10.2, "Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation." NuScale FSAR Tier 2, Section 3.10.2 states that ASME QME-1-2007 is used with the exceptions noted in RG 1.100 (Revision 3) for the qualification of active mechanical equipment. However, NuScale FSAR Tier 2, Section 3.9.6 does not identify this provision for the functional capability qualification of safety-related valves as a Tier 2* requirement. The Commission approved the NRC staff use of the Tier 2* designation in a Staff Requirements Memorandum dated June 30, 1994 (ADAMS Accession No. ML003708098) in response to Commission Paper SECY-94-084, dated May 31, 1994 (ADAMS Accession No. ML003708079), which characterized Tier 2* information to be appropriate as Tier 1 information if the Tier 2* designation was not applied. Based on the safety significance of the proper performance of power-operated valves, the NRC staff considers the process to demonstrate the functional capability of safety- related power-operated valves in the NuScale Power Plant to be appropriate as a Tier 1 requirement or Tier 2* information that should not be modified without prior NRC review. Therefore, the NRC staff requests that NuScale specify the provision to implement the ASME Standard QME-1-2007 as accepted in RG 1.100 (Revision 3) for the functional capability qualification of safety-related valves for the NuScale Power Plant as an ITAAC requirement in NuScale FSAR Tier 1, or designate this provision as Tier 2* information in NuScale FSAR Tier 2.

NuScale Response:

In Tier 1, Reference is not made to ASME Standard QME-1 to prevent it from becoming Tier 1 information. The principle to not include NRC regulatory guidance and industry standards in Tier 1 was applied to all Tier 1 material.

However, ASME Standard QME-1 is referenced in FSAR Tier 2, Table 14.3-1 to describe the implementation of equipment qualification i.e. ITAAC 02.08.01, 02.08.02, 02.08.03, 02.08.06 and 03.14.01

In addition to FSAR Tier 2, Sections 3.9.6, 3.10 and Table 14.3-1, QME-1 is addressed in:

FSAR Tier 2, Chapter 3.11.6 Qualification of Mechanical Equipment

FSAR Tier 2, Chapter 3, Appendix 3C.4 states:

“Mechanical equipment that performs an active design function related to safety during or following exposure to harsh environmental conditions is qualified in accordance with ASME QME-1, Appendix”

FSAR Tier 2, Chapter 3, Appendix 3C.5, Design Specification

NuScale, in both the FSAR and in its design specifications, has committed to the use of QME-1 to qualify all safety-related valves.

The NuScale approach to referencing QME-1 in Tier 2 is consistent with industry guidance. Both the FSAR and Design Specifications provide reasonable assurance that all safety-related valves will be qualified in accordance with ASME Standard QME-1 as stated in RG 1.100. No changes to either Tier 1 or Tier 2 are required.

Impact on DCA:
There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8820
Date of RAI Issue: 06/02/2017

NRC Question No.: 03.09.06-4

For a nuclear reactor design certification application, the NRC regulations in 10 CFR Part 52, Section 47, “Contents of applications; technical information,” require, in the introductory statement, that the application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. This regulation specifies that the information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The regulation indicates that the NRC will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination. In accordance with 10 CFR 52.47, the NRC staff requests that NuScale design certification applicant provide its schedule for completing the information normally contained in the procurement specification, and construction and installation specifications, for safety-related valves (such as those in the emergency core cooling system and containment isolation system); and for making this information available for audit to allow the NRC to reach a safety determination on the NuScale design certification application.

NuScale Response:

As discussed during a public phone call on May 2nd, 2017 the type of information requested is contained in the ASME design specification for the ECCS valves, which is currently available for audit. The ASME design specifications include requirements for design, analysis, materials of construction, fabrication, inspection and examination, testing, preparation, shipment, delivery, owner and supplier responsibilities, and environmental control during fabrication. This information is sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC and procurement specifications and construction and installation specifications by an applicant, as required by 10 CFR 52.47.

Impact on DCA:
There are no impacts to the DCA as a result of this response.