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Subject: Request for Additional Information No. 47, RAI 8820
Attachments: Request for Additional Information No. 47 (eRAI No. 8820).pdf

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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Request for Additional Information No. 47 (eRAI No. 8820)

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Docket No. 52-048
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QUESTIONS

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.

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 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 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.

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.

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.