September 16, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 525 (eRAI No. 9705) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 525 (eRAI No. 9705)," dated August 23, 2019

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 Question from NRC eRAI No. 9705:

- 19.02-1

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 Rebecca Norris at 541-602-1260 or at rnorris@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9705

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9705

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

eRAI No.: 9705
Date of RAI Issue: 08/23/2019

NRC Question No.: 19.02-1

Regulatory Basis

10 CFR 52.47(a)(2)(iv) states, in part, that the application for a design certification must contain an FSAR that includes the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents.

SECY-90-016, Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements, states that mitigation features must be designed so there is reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Equipment survivability expectations should include consideration of the environment (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function.

SECY-90-016, also discusses that the containment should maintain its role as a reliable leak tight barrier and should provide a barrier against the uncontrolled release of fission products. This review should be informed by SECY 19-0047, "Containment Performance Goals for The NuScale Small Modular Reactor Design," and its acceptance criteria.

10 CFR 50.44 Combustible Gas Control for Nuclear Power Reactors, *50.44(c)(3) Equipment Survivability, states, in part,* Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to

be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.

10 CFR 50.44(c)(4) Monitoring, states, in part, that equipment must be provided for monitoring hydrogen and oxygen in the containment. Equipment for monitoring hydrogen and oxygen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen and oxygen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

Background

LO-0519-65662 NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report, Section 19.2, "Severe Accident Evaluation", 22 May 2019

NuScale TR-0716-50424-P, Revision 1 "Combustible Gas Control," ML19091A235

NuScale ER-P020-7109, Rev 0, 16 May 2019, "Equipment Survivability Methodology and Results" , located in the NuScale electronic reading room.

Issue

For the NuScale design, the severe accident mitigation functions are maintaining containment integrity and providing post-accident monitoring of hydrogen and oxygen in containment.

In order to demonstrate reasonable assurance that equipment required to mitigate severe accidents is shown to meet the requirements of SECY-90-016 and the staff review criteria described in SECY 19-0047, severe accident mitigation equipment and its required functions must be identified. The time duration and the environmental conditions of pressure, temperature, humidity, and radiological dose for which this function is required must also be identified. These conditions also include exposure to the environmental conditions created by the burning of hydrogen. The means to demonstrate that the equipment would perform its required function under post-accident conditions, whether by testing, analysis or a combination, must also be identified.

NuScale has performed such an evaluation in its report ER-P020-7109, Rev 0, 16 May 2019, "Equipment Survivability Methodology and Results". The purpose of this document is to identify the equipment survivability expectations in the context of radiation dose due to a core damage accident. As specified in SECY-90-016, "mitigation features must be designed so that there is reasonable assurance that they will operate in the severe- accident environment for which they are intended and over the time span for which they are needed." In order for staff to make a

safety finding that the NuScale severe accident features will operate in the radiological dose environment identified in NuScale report ER-P020-7109, Rev 0, please submit this information in the FSAR. Additionally,

1. As specified in SECY-90-016, "mitigation features must be designed so that there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed." Since NuScale specifies that the Containment Evacuation System (CES) and the Containment Flooding and Drain (CFDS) Containment Isolation Valves (CIVs) will be re-opened no later than 72 hours after a severe accident, staff requests that NuScale revise the duration of the radiological dose to these CIVs to at least 72 hours. NuScale should provide a technical basis as to why a longer time span is not needed.

2. In order to meet the regulation 10 CFR 50.44(c)(4) to provide hydrogen and oxygen monitoring post- accident, NuScale will utilize the CES, CFDS and Process Sampling System (PSS) systems. This monitoring must be established by 72 hours post-accident, but could be aligned earlier, as long as the containment pressure is less than 250 psia, in order not to exceed the CES design pressure. For this reason, NuScale is requested to revise the duration for the post-accident monitoring variable of Wide Range Containment Pressure to at least 72 hours. NuScale should provide a technical basis as to why a longer time span is not needed.

3. NuScale report ER-P020-7109, Rev 0, 16 May 2019, section 3.0, Methodology, states that equipment survivability will be assured either through dose comparisons, qualitative assessment, and/or additional testing or analysis. NuScale further states that qualitative assessment is a justification of survivability based on existing industry or vendor data. Staff requests that NuScale elaborate on their basis for concluding that vendor data or test results relevant to the NuScale containment high radiation dose post-accident atmosphere exist. Staff requests that NuScale discuss how the vendor data and/or test results would be assessed, and which acceptance criteria would apply.

4. NuScale is requested to submit a COL Item in FSAR chapter 19.2 which addresses the following:

   The COL applicant that references the NuScale Power Plant design certification should submit a full description of the Equipment Survivability Program in accordance with the

scope and methodology described in FSAR Section 19.2.3.3.8. Milestones and completion dates for program implementation should also be included.

5. In order to demonstrate reasonable assurance that equipment required to mitigate severe accidents is shown to meet the requirements of SECY-90-016 and the staff review criteria described in SECY 19-0047, severe accident mitigation equipment and its required functions must be identified. The time duration and the environmental conditions of pressure, temperature, humidity, and radiological dose for which this function is required must also be identified. In order to meet 10 CFR *50.44(c)(3) Equipment Survivability,* containments must be able to establish and maintain safe shutdown and structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. NuScale is requested show that the scope, durations, and methodology of ER-P020-7109, Rev 0, 16 May 2019 apply to the environmental conditions of pressure and temperature following the combustion of hydrogen in containment.

NuScale Response:

1. As stated in FSAR Revision 3 Section 19.2.3.3.8, equipment necessary for continuous combustible gas monitoring is reasonably assured to survive for at least 72 hours. For consistency with other components used for combustible gas monitoring, NuScale concurs with the staff that 72 hours is an appropriate survivability duration for the containment evacuation system (CES) and containment flooding and drain system (CFDS) containment isolation valves (CIVs) as well as the associated hydraulic skids. The CES and CFDS CIVs are required to open no later than 72 hours to allow sampling from the containment vessel (CNV) gas space. After the valves are opened, there is no further valve movement required to support monitoring. Therefore, a survivability duration in excess of 72 hours is not justified.

The revised survivability duration of the CIVs and hydraulic skids is reflected in a new table, Table 19.2-11, which has been added to the FSAR. Table 19.2-11 identifies the scope of equipment considered in the survivability evaluation as well as the associated function and survivability duration.

2. As stated in TR-0716-50424-NP, Revision 1, “Combustible Gas Control” (ML19091A232), the hydrogen and oxygen monitoring flow path can be established when containment pressure is below 250 psia, which is the design pressure of the monitoring loop. NuScale severe accident simulations presented in FSAR Section 19.2.3.2 show that condensation and passive heat transfer to the reactor pool cause the CNV pressure to decrease, ultimately dropping below 250 psia before core damage occurs. After core damage and potential relocation, the CNV pressure remains below 250 psia, even considering hydrogen produced in a 100 percent fuel-clad-coolant interaction, as described in FSAR Section 19.2.3.3.2, and the steam produced due to fuel relocation. Based on these simulations, which model the associated pressurization phenomena, combustible gas monitoring can be initiated at any time after core damage without over-pressurizing the monitoring loop; therefore revising the survivability duration of the containment pressure variable is not justified to support combustible gas monitoring.

3. Specific components have not been selected at the design certification stage, therefore an assessment of the applicability of available testing data, with associated acceptance criteria, is not performed. When components are selected, they will be qualified per the environmental qualification programs, as indicated by COL Items 3.11-1 through 3.11-4. As stated in FSAR Section 19.2.3.3.8, for components in which the predicted severe accident dose exceeds the environmental qualification dose, “qualitative assessments, testing, or additional analyses” are needed to provide reasonable assurance of equipment survivability.

Using test data is one option that may be used to demonstrate reasonable assurance of survivability during a qualitative assessment. The methodology stated in FSAR Section 19.2.3.3.8 does not assume that relevant vendor data will exist that encompass the severe accident environment. If such data do not exist, the methodology stated in the FSAR allows for alternate means to be used to provide reasonable assurance of survivability.

4. As indicated in NuScale’s response to RAI 9151 (Q19-31) provided in letter RAIO-1117-57303, dated November 21, 2017 (ML17325B728) and supplemented in letter RAIO-0318-58993, dated March 02, 2018 (ML18061A147), the combined license (COL) items in Chapter 19 are written at a summary level to identify the major COL activities associated with developing a PRA to support an as-built/as-operated facility. COL Items 19.1-1 through 19.1-8 address responsibilities of a COL applicant that references the NuScale Power Plant design certification from the application phase through commercial operation. COL Item 19.1-5 states that the applicant “will specify and describe the use of the probabilistic risk assessment in support of licensee programs during the operational phase”.

In addition to the specific COL items identified in the FSAR, development of an acceptable combined license application is based on regulatory requirements, such as 10 CFR 52.79(46),

regulatory guidance documents, such as DC/COL-ISG-28, and industry standards, such as the ASME/ANS PRA Standard.

Combined license evaluation of equipment survivability, as described in FSAR Section 19.2.3.3.8, is considered in the same manner as other updates of the PRA for an as-built/as-operated plant. The existing COL items support the transition from the design certification PRA to an as-built/as-operated PRA; thus, a unique COL item for equipment survivability evaluation is not required.

5. FSAR Section 19.2.3.3.8 addresses equipment survivability for the environmental conditions of pressure and temperature following the combustion of hydrogen in containment. The purpose of the equipment survivability assessment is to determine the subset of equipment which must be considered for survivability and the necessary durations in which the equipment is required to survive. The duration is determined based on function, and must be valid for environmental conditions in combination (i.e., temperature, pressure, humidity, and radiological dose). The NuScale internal report cited in the RAI question assesses the radiological dose for use in the survivability assessment; additional NuScale analyses assess equipment survivability considering the conditions of temperature and pressure. FSAR Section 19.2.3.3.8 summarizes the results of these relevant NuScale analyses. For clarity, FSAR Section 19.2.3.3.8 has been revised to clarify the scope of equipment survivability and Section 19.2.4 has been revised to reference Section 19.2.3.3.8 for equipment survivability with respect to combustion. As noted in the response to Question 1, Section 19.2.3.3.8 has also been revised to include a table of all components and variables considered within the survivability evaluation.

Impact on DCA:

Table 19.2-11 has been added to the FSAR; also FSAR Sections 19.2.3.3.8 and 19.2.4 have been revised as described in the response above and as shown in the markup provided in this response.

based, in part, on environments predicted for severe accidents as modeled in the NuScale PRA. This approach provides confidence that the equipment needed for severe accident mitigation and monitoring survives over the time span in which it is needed. Equipment survivability in a radiation environment is first evaluated by comparing the severe accident dose to the environmental qualification design-basis dose. The severe accident dose is based on the core damage source term described in Section 15.10. For cases in which the environmental qualification dose is larger, survivability is assured. For cases in which the severe accident dose is larger, qualitative assessments, testing, or additional analyses are performed to assure survivability.

RAI 19.02-1

Table 19.2-11 summarizes the evaluation of equipment for survivability; the table identifies each component or variable, its function, and the duration over which it is needed. Post-accident temperature and pressure conditions, including those that result from hydrogen combustion, are discussed with regard to containment integrity and post-accident monitoring capabilities as follows.

Containment Integrity

Containment integrity is the only safety function relied upon for severe accident mitigation. The function is ensured through successful closure of the containment isolation valves and ensuring that the CNV, including penetrations and seals, remains intact. Given how early a containment isolation signal is generated following postulated PRA initiating events, containment isolation valves are expected to reach the desired position well before core damage occurs.

Simulation results confirm the NPM remains below CNV temperature and pressure limits for all accident sequences considered in the PRA. The two most challenging transients with respect to CNV temperature and pressure loads are the CNV response to an ultimate failure of the RPV due to overpressurization and the CNV response to an adiabatic complete combustion of the hydrogen conditions described in Section 19.2.3.3.2. Thermal-hydraulic results show that even if the RPV were to fail due to overpressurization, the CNV ultimate failure pressure would not be exceeded and any RPV-CNV pressure differential would subside well before core damage. A conservative thermodynamic analysis of a complete combustion of the hydrogen/oxygen inventory described in Section 19.2.3.3.2 imparting all energy adiabatically and directly into the exposed CNV steel (exposed on the inside-surface) confirms that the steel temperature would rise less than 75F, remaining well below the CNV design temperature.

NuScale's unique and robust design has reduced or eliminated many of the traditional failure mechanisms that challenge containment integrity once it is successfully isolated. Section 19.2.3.3 further discusses a module's response to such challenges.

Combustible Gas Control

RAI 19.02-1

Containment performance is ensured also by achieving combustible gas control. During normal plant operation, combustible gas control is achieved by maintaining a near vacuum in the CNV by the CES. As discussed in Section 19.2.3.3.2, during severe accident conditions combustible gas control is provided initially by the steam-inert environment and later by the large production of hydrogen that reduces oxygen concentration below combustible limits. Additionally, an adiabatic isochoric complete combustion analysis was performed to evaluate the ability of the CNV to cope with combustible gases generated by radiolysis occurring for weeks after a severe accident. The analysis showed the resulting containment pressure was calculated to be below the CNV design pressure which demonstrates that hydrogen combustion does not pose a credible risk to the NuScale CNV. A listing of SSC that are required to survive~~remain functional~~ following a hydrogen combustion event to support containment integrity and post-accident monitoring~~core cooling~~ is included in Section 19.2.3.3.8.~~provided in Section 3.3.5 of TR-0716-50424, Revision 0, "Combustible Gas Control" Technical Report (Reference 6.2-3).~~

Summary of Containment Performance

Consistent with SECY-93-087, deterministic and probabilistic evaluations of containment capability have been performed. The deterministic evaluation of containment capability in comparison to potential severe accident challenges confirms that the CNV is a leak-tight barrier for a period of at least 24 hours following the onset of core damage for the most-likely severe accident sequences. The probabilistic evaluation demonstrates that the reliability of containment isolation in response to severe accident meets the safety goal, as confirmed by the composite CCFP.

19.2.5 Accident Management

Accident management refers to the actions taken during the course of a beyond design basis accident by the plant operating and technical staff to:

- prevent core damage
- terminate the progress of core damage if it begins and retain the core within the RPV
- maintain containment integrity as long as possible
- minimize offsite releases

The inherent design characteristics (e.g., fail-safe equipment position and design simplicity) and thermal-hydraulic characteristics (e.g., passive cooling) of the NuScale design are such that there are no operator actions required to place an NPM in a safe configuration for postulated design basis accidents. That is, operator actions during postulated accidents are associated with monitoring the module or providing backup in the event of multiple component failures. Section 19.2.5.1 summarizes the capability of the NuScale design with respect to the different stages of a postulated accident. Section 19.2.5.2 summarizes the programmatic structure for accident management.

RAI 19.02-1

Table 19.2-11: Equipment Survivability List

Component/Variable	Function	Duration
CIVs	Close to maintain containment integrity	1 hour after transient
CNV (including Closure Flanges and Bolting)	Maintain containment integrity	24 hours after core damage
Electrical Penetration Assemblies	Maintain containment integrity	24 hours after core damage
ECCS Trip and Reset Valves	Maintain containment integrity	24 hours after core damage
CES and CFDS CIVs	Open to allow combustible gas monitoring	72 hours
CIV Hydraulic Skids	Combustible gas monitoring	72 hours
Combustible Gas Monitors	Combustible gas monitoring	72 hours
Containment Gas Sample Pump	Combustible gas monitoring	72 hours
CIV Position	PAM variable	1 hour after transient
Narrow Range Containment Pressure	PAM variable	1 hour after transient
Reactor Trip Breaker and Pressurizer Heater Trip Breaker Position Feedback	PAM variable	1 hour after transient
Under-the-Bioshield Temperature	PAM variable	1 hour after transient
Neutron Flux	PAM variable	Until core damage
Core Inlet and Exit Temperatures	PAM variable	Until core damage
Wide Range RCS T_{HOT}	PAM variable	Until core damage
Wide Range RCS Pressure	PAM variable	Until core damage
RPV Riser Level	PAM variable	Until core damage
Wide Range Containment Pressure	PAM variable	Until core damage
Containment Water Level	PAM variable	Until core damage
ECCS Valve Position (including Trip Valve)	PAM variable	Until core damage
Reactor Pool Temperature (Operating Bay)	PAM variable	Until core damage
DHRS Valve Position	PAM variable	Until core damage
Secondary MSIV and MSIV Bypass Valve Position	PAM variable	Until core damage
FWRV Position	PAM variable	Until core damage
Main Steam Temperature and Pressure (DHRS Inlet Temperature and Pressure)	PAM variable	Until core damage
DHRS Outlet Temperature and Pressure	PAM variable	Until core damage
RCS Flow	PAM variable	Until core damage
Demineralized Water Supply Isolation Valve Position	PAM variable	Until core damage
Control Room Habitability System (CRHS) Valve Position (Air Supply Isolation and Pressure Relief Isolation)	PAM variable	Until core damage
Control Room HVAC System (CRVS) Damper Position (Supply Air, Smoke Purge Exhaust, General Exhaust, and Return Air)	PAM variable	Until core damage
Inside Bioshield Area Radiation Monitor	PAM variable	24 hours after core damage
Spent Fuel Pool Water Level	PAM variable	24 hours after core damage
EDSS-MS / EDSS-C Bus Voltage	PAM variable	24 hours after core damage