NuScale Power

RAIO-0817-55577

August 22, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 78 (eRAI No. 8892) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 78 (eRAI No. 8892)," dated June 30, 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 Question from NRC eRAI No. 8892:

- 19-14

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8892

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8892

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

eRAI No.: 8892
Date of RAI Issue: 06/30/2017

NRC Question No.: 19-14

Title 10 Code of Federal Regulations (CFR) 52.47(a)(27) states that a design certification application must contain an final safety analysis report (FSAR) that includes description of the design-specific probabilistic risk assessment (PRA) and its results. In accordance with the Statement of Consideration (72 FR 49387) for the revised 10 CFR Part 52, the staff reviews the information contained in the applicant's FSAR Chapter 19, issues requests for additional information (RAIS) and conducts audits of the complete PRA (e.g., models, analyses, data, and codes) to obtain clarifying information as needed. The staff uses guidance contained in Standard Review Plan (SRP) Chapter 19.0 Revision 3, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors." In accordance with SRP Chapter 19.0 Revision 3, the staff determines whether:

> "The technical adequacy of the PRA is sufficient to justify the specific results and risk insights that are used to support the [Design Certification] DC or [Combined License] COL application. Toward this end, the applicant's PRA submittal should be consistent with prevailing PRA standards, guidance, and good practices as needed to support its uses and applications and as endorsed by the [Nuclear Regulatory Commission] NRC (e.g., [Regulatory Guide] RG 1.200)."

The staff has reviewed the information in the FSAR and examined additional clarifying information from the audit of the complete PRA and determined that it needs additional information to confirm the validity of certain assumptions used in the flooding PRAs. The supporting requirements in the American Society of Mechanical Engineers / American Nuclear Society (ASME/ANS) PRA standard include provisions for documenting sources of model uncertainties and related assumptions. Please address the following questions.

a. FSAR Table 19.1-49, "Assessment of Flood Areas Containing Equipment Modeled in the Probabilistic Risk Assessment," describes the reactor building areas that include flood protection design features to protect equipment from propagating floods. Review of supporting audit information suggests that the required level of flooding protection is determined based on the assumed time available for the operator to successfully isolate the flood source. Please confirm the staff's understanding or provide an alternative explanation.

Additionally, assuming that the staff understands correctly and considering (1) the uncertainties introduced by the current level of plant design as cited in the FSAR (such as the lack of design detail on protective and mitigative features and detailed pipe routing information) and (2) the PRA should consider scenarios beyond the design basis, please explain how operators will always successfully isolate any flood sources in the reactor building.

b. FSAR Table 19.1-48, "Internal Flooding Sources," indicates the Reactor Building Spray System as a potentially significant flood source. The staff reviewed the FSAR and associated audit documentation and was unable to locate information on potential flooding scenarios associated with this flood source. Please describe the potential flooding scenarios associated with this flood source, considering as applicable, the associated potential propagation paths, equipment damage, flooding protection and mitigation features, and operator actions.

c. FSAR Section 19.1.5.4.1 states:

"An external flood could initiate a [Loss of Offsite Power] LOOP or [Loss of Direct Current] LODC because of flooding in areas containing [highly reliable DC power system] EDSS or [13.8 kV and switchyard system] EHVS components."

This statement implies that the EDSS and the EHVS equipment is assumed to be unprotected from floods. Please discuss why flooding protection features assumed to be available for internal flooding scenarios are assumed not to be available for external flooding scenarios.

NuScale Response:

In response to the cited aspects of PRA internal flood modeling, the following information is provided:

a.) The credit taken for the flood mitigation features and operator actions in the reactor building (RXB) limited the impact of the induced initiating events postulated to be caused by potential internal flooding; specifically, flooding events in the RXB were modeled as reactor trips ("general transients") in which makeup by the chemical and volume control system (CVCS) and the containment flooding and drain system (CFDS) is unavailable.

If credit had not been taken for flood mitigation strategies, flooding could be postulated on the 75' and 86' elevations of the RXB. This could result in a loss of DC power or a spurious emergency core cooling system (ECCS) actuation.These elevations primarily contain electrical equipment. A flood originating in these areas is readily detectable because, in addition to instrumentation in the faulted system providing indications that a break or actuation has occurred, the rapid de-energization of electrical equipment in the area would be apparent.

The specific rooms containing the equipment which could result in a loss of DC power or spurious ECCS actuation contain no internal flooding hazards. The only internal flooding events which could affect equipment in these areas would have to originate in a separate flood area, such as a flood from the fire suppression system in the corridor outside of the module protection system (MPS) equipment rooms, and persist long enough to overcome the flooding protective features for the area.

Although a representative internal flooding analysis has been performed and described in FSAR Section 3.4.1, final pipe routing and the specific flood mitigation strategies that will be used to protect equipment throughout the plant have not yet been established. This representative analysis has been based in part on assumed flood volumes based on an expectation that plant personnel will eventually isolate a flood source. Accordingly, COL Item 3.4-1 and COL Item 3.4-2 have been specified to address these aspects of the as-built, as-operated plant.

Potential methods of mitigating the effects of a flooding event include the use of watertight doors, elevating equipment above flood levels, and enclosing or qualifying equipment for submersion. The method of flood protection is determined by the COL applicant to assure conformance with the design certification basis. The nature of the flood barriers used to mitigate the effects of flooding determines the specific amount of time available to mitigate the event.

The combination of the limited potential flooding hazards, detection capability, and the numerous options available for controlling flooding events indicate that PRA modeling of operator capability to isolate RXB flooding sources, before de-energization of the highly reliable DC power system (EDSS) or MPS, is reasonable.

b.) The reactor building spray system is not included within the standard NuScale design. The system has been deleted from Table 19.1-48, Table 3.2-1, and Section 19.2.5.1.

c.) The statement in FSAR Section 19.1.5.4.1 reflects a simplifying modeling assumption that the EDSS and 13.8 kV and switchyard system (EHVS) are unavailable as the result of an external flooding event. The EHVS equipment is located at grade level outside of the RXB, control building (CRB), and radioactive waste building (RWB). Because an external flooding event includes flooding that exceeds this elevation, EHVS equipment is assumed to fail. Similarly, medium voltage electrical system (EMVS), low voltage electrical system (ELVS), and backup power supply system (BPSS) equipment is also assumed to fail due to this event. A loss of these systems constitutes a complete loss of AC power. These areas were screened from the internal flooding analysis based on a lack of flood sources.

If AC power is lost, EDSS relies on battery backup power. Battery capacity is not sufficient to satisfy the 72 hour mission time used in the PRA, regardless if the EDSS equipment is directly damaged by the external flooding event. Therefore, whether a loss of DC power occurs immediately or 24 hours into the event, the accident sequence is modeled in the same way. EDSS equipment, however, is protected from the effects of internal flooding as described in FSAR Section 3.4.1, FSAR Table 3.4-2, and part (a) of this response.

Impact on DCA:

Section 19.2.5.1, Table 3.2-1, and Table 19.1-48 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 09.02.02-1, RAI 09.02.05-1, RAI 09.02.06-1, RAI 19-14, RAI 11.02-1

Table 3.2-1: Classification of Structures, Systems, and Components

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.189 or RG 1.29)
CNTS, Containment System							
All components (except as listed below)	RXB	A1	N/A	Q	None	A	I
• RXM Lifting Lugs • Top Auxiliary Mechanical Access Structure • Top Auxiliary Mechanical Access Structure Diagonal Lifting Braces	RXB	B1	None	AQ-S	• ANSI/ANS 57.1-1992 • ASME NOG-1 • NUREG-0554	N/A	I
CFDS Piping in containment	RXB	B2	None	AQ-S	None	B	II
Piping from (CES, CFDS, CVCS, FWS, MSS, and RCCWS) CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	D	I
Hydraulic Skid for valve reset	RXB	B2	None	None	None	D	III
CIV Close and Open Position Sensors: • CES, Inboard and Outboard • CFDS, Inboard and Outboard • CVCS, Inboard and Outboard PZR Spray Line • CVCS, Inboard and Outboard RCS Discharge • CVCS, Inboard and Outboard RCS Injection • CVCS, Inboard and Outboard RPV High-Point Degasification • FWS, Supply to SGs and DHR HXs FWIV • RCCWS, Inboard and Outboard Return and Supply • SGS, Steam Supply CIV/MSIVs and CIV/MSIV Bypasses	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Containment Pressure Transducer (Wide Range)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	II
• Containment Air Temperature (RTDs) • FW Temperature Transducers	RXB	B2	None	AQ-S	None	N/A	II
SGS, Steam Generator System							
• SG tubes • Feedwater plenums • Steam plenums • SG tube supports	RXB	A1	N/A	Q	None	A	I
• Steam piping inside containment • Feedwater piping inside containment • Feedwater supply nozzles • Main steam supply nozzles • Thermal relief valves	RXB	A2	N/A	Q	None	B	I
Flow restrictors	RXB	A2	N/A	Q	None	N/A	I
RXC, Reactor Core System							
Fuel assembly (RXF)	RXB	A1	N/A	Q	None	N/A	I
Fuel Assembly Guide Tube	RXB	A2	N/A	Q	None	N/A	I
Incore Instrument Tube	RXB	B2	None	AQ-S	None	N/A	I
CRDS, Control Rod Drive System							
• Control Rod Drive Shafts • Control Rod Drive Latch Mechanism	RXB	A1	N/A	Q	None	N/A	I
CRDM Pressure Boundary (Latch Housing, Rod Travel Housing, Rod Travel Housing Plug)	RXB	A2	N/A	Q	None	A	I
CRDS Cooling Water Piping and Pressure Relief Valve	RXB	B2	None	AQ-S	None	B	II
Rod Position Indication (RPI) Coils	RXB	B2	None	AQ-S	None	N/A	I
• Control Rod Drive Coils • CRDM power cables from EDN breaker to MPS breaker • CRDM power cables from MPS breaker to CRDM Cabinets	RXB	B2	None	AQ-S	None	N/A	II
• CRDM Control Cabinet • CRDM Power & Rod Position Indication Cables • Rod Position Indication Cabinets (Train A/B)	RXB	B2	None	AQ	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.189 or RG 1.29)
All components	DGB	B2	None	None	None	N/A	III
ABVS, Annex Building HVAC System							
All components	ANB	B2	None	None	None	N/A	III
FPS, Fire Protection System							
All components	Various	B2	None	None	None	N/A	III
BPDS, BOP Drain System							
All components (except as listed below)	Various	B2	None	AQ	None	D	III
Instrumentation	Various	B2	None	None	None	N/A	III
~~RBSS, Reactor Building Spray System~~							
~~All components~~	~~RXB~~	~~B2~~	~~None~~	~~AQ-S~~	~~None~~	~~N/A~~	~~II~~
EHVS, 13.8 KV and SWYD System							
All components	Various	B2	None	None	None	N/A	III
EMVS, Medium Voltage AC Electrical Distribution System							
All components	Various	B2	None	None	None	N/A	III
ELVS, Low Voltage AC Electrical Distribution System							
B6000 series Motor Control Centers	RXB	B2	None	AQ	None	N/A	III
• Motor Control Center, non-B6000 • Station Service Transformers for B6000 and non-B6000 MCCs • Load Centers (SWG) for B6000 and non-B6000 MCCs	RXB	B2	None	None	None	N/A	III
EDSS, Highly Reliable DC Power System							
• Channel A, Channel C, and Common Division I Components: - DC Bus - Switchgear - Batteries 1 and 2 - Battery Chargers 1 and 2 - Transfer Switches 1 and 2 • Channel B, Channel D, and Common Division II Components: - DC Bus - Switchgear - Batteries 1 and 2 - Battery Chargers 1 and 2 - Transfer Switches 1 and 2 • EDSS-C, Cabling • EDSS-C, Fusible Disconnects • EDSS-MS, Cabling • EDSS-MS, Fusible Disconnects	Various	B2	None	AQ-S	• 10 CFR 50.55a(1) • 10 CFR 50.55a(h) • IEEE Std. 603-1991 • Environmental Qualification • Independence • Single Failure Criterion • Common-Cause Failure • Location of Indicators and Controls • Multi-Unit Station Considerations	N/A	I
• Channel A, Channel C, and Common Division I Components: - Battery Charger Ammeters 1 and 2 - Battery Monitors 1 and 2 - DC Bus Ground Fault Relay - DC Bus Overvoltage Relay - DC Bus Undervoltage Relay • Channel B, Channel D, and Common Division II Components: - Battery Charger Ammeters 1 and 2 - Battery Monitors 1 and 2 - DC Bus Ground Fault Relay - DC Bus Overvoltage Relay - DC Bus Undervoltage Relay	Various	B2	None	AQ-S	• 10 CFR 50.55a(1) • 10 CFR 50.55a(h) • IEEE Std. 603-1991 • Environmental Qualification • Independence • Single Failure Criterion • Common-Cause Failure • Location of Indicators and Controls • Multi-Unit Station Considerations	N/A	I

RAI 19-14

Table 19.1-48: Internal Flooding Sources

System	Flooding Potential	Location
Chemical and Volume Control System	Minimal. This system does not move large volumes of water. Breaks in piping result are considered in the internal events model.	RXB
Boron Addition System	Minimal. This system does not move large volumes of water.	RXB
Module Heatup System	Minimal. This system does not move large volumes of water.	RXB
Decay Heat Removal System	Minimal. This system involves a limited inventory, the bulk of which is contained within its heat exchangers and the steam generators.	RXB
Containment Evacuation System	Minimal. During operation, this system primarily contains gases from the CNV and is judged not to have a large fluid inventory.	RXB
Containment Flooding and Drain System	Moderate potential for flooding. Although not normally in operation, this system draws suction from the UHS which contains significant water volume.	RXB
Reactor Component Cooling Water System	Minimal. This system's limited inventory may result in flooding in a small area, but it is not capable of causing widespread flooding.	RXB
Process Sampling System	Minimal. Process sampling lines are small.	RXB, TGB
Liquid Radioactive Waste Management System	Minimal. Flooding may originate from storage tanks. Small, localized flooding events may originate from breaks in other system piping.	RXB, RWB
Radioactive Waste Drain System	Minimal. This system does not normally have a fluid inventory.	RXB, RWB
Spent Fuel Pool Cooling System	Significant. This system is normally in operation and draws suction from the UHS.	RXB
Pool Cleanup System	Minimal. The majority of the piping that supports this system is associated with other systems.	RXB
Reactor Pool Cooling System	Significant. This system is normally in operation and draws suction from the UHS.	RXB
Pool Surge Control System	Minimal. This system is not normally in operation.	RXB, Yard Area
Ultimate Heat Sink	Significant. This system contains a large flooding inventory.	RXB
Pool Leakage Detection Systems	Minimal. Although this system is connected to the UHS the flow into this system is limited to leakage.	RXB
Main Steam System	Moderate. Flooding from this system could primarily occur in the form of condensation.	RXB, TGB
Condensate and Feedwater System	Significant. Although breaks in this system are intended to be isolated quickly, unisolated breaks may result in substantial flooding.	RXB, TGB
Feedwater Treatment	Minimal. The majority of the piping that supports this system is associated with other systems.	TGB
Condensate Polisher Resin Regeneration System	Minimal. The majority of the piping that supports this system is associated with other systems.	TGB
Heater Vents and Drains	Minimal. The majority of the piping that supports this system is associated with other systems.	TGB
Chilled Water System	Significant. This system moves substantial volumes of water.	RXB, RWB, CRB
Auxiliary Boiler System	Minimal. The majority of the piping that supports this system is associated with other systems.	TGB
Turbine-Generator System	Minimal. The majority of the piping that supports this system is associated with other systems.	TGB
Circulating Water System	Significant. This system moves substantial volumes of water.	TGB

Table 19.1-48: Internal Flooding Sources (Continued)

System	Flooding Potential	Location
Site Cooling Water System	Significant. This system moves substantial volumes of water.	TGB
Potable Water System	Minimal. This system does not move large volumes of water.	CRB
Utility Water System	Significant. This system moves substantial volumes of water.	RXB, RWB, CRB, Annex Building
Demineralized Water System	Significant. This system moves substantial volumes of water.	RXB
Turbine Building HVAC System	Minimal. It is assumed that the only flooding mechanism applicable to this system is through the cooling coils, which is judged to be minimal.	TGB
Security Building HVAC	Minimal. It is assumed that the only flooding mechanism applicable to this system is through the cooling coils, which is judged to be minimal.	Security Building
Diesel Generator Building HVAC	Minimal. It is assumed that the only flooding mechanism applicable to this system is through the cooling coils, which is judged to be minimal.	Diesel Generator Building
Annex Building HVAC	Minimal. It is assumed that the only flooding mechanism applicable to this system is through the cooling coils, which is judged to be minimal.	Annex Building
Fire Protection System	Significant. This system moves substantial volumes of water.	RXB, RWB, CRB
~~Reactor Building Spray System~~	~~Significant. This system moves substantial volumes of water.~~	~~RXB~~

actions to provide the necessary makeup, depending on the particular failures involved in the event, include

- manual action to open ECCS valves to allow ECCS flow between the RPV and the CNV, which allows decay heat removal to the UHS (reactor pool).
- manual initiation of makeup to the RPV through the CVCS injection line using the CVCS makeup pumps.
- manual initiation of makeup to the RPV through the pressurizer spray line using the CVCS makeup pumps.
- manual initiation of the CFDS to add water to the CNV to remove heat from the RPV through passive conduction and convection, preventing RPV over-pressurization, or when the CFDS is credited in conjunction with successful ECCS, the makeup coolant mitigates an unisolated outside-containment LOCA.

Terminate Core Damage Progression and Retain the Core within the RPV

The actions identified for prevention of core damage are also taken to arrest the progression of core damage once begun and retain the core within the RPV.

Maintaining Containment Integrity

The Level 2 PRA discussed in Section 19.1 demonstrates that physically-realistic challenges to containment are due to failure of containment isolation or containment bypass. Potential actions to maintain containment integrity, depending on the particular failures involved in the event, include

- manual action to restore containment isolation.
- isolation of an SGTF to preserve the reactor coolant pressure boundary.

Minimize Offsite Releases

The small size of an NPM core results in a correspondingly small radionuclide source term. Although not credited in the PRA, potential releases would be further minimized because

- most of the CNV is below water, thus radionuclide release due to CNV failure of the lower head would be minimized due to the scrubbing effect of the reactor pool.
- for severe accidents with CNV bypass or containment isolation failure, there is potential deposition in the bypass piping and the release would potentially be further reduced by the Seismic Category I Reactor Building.

RAI 19-14

- ~~the Reactor Building spray system would reduce a potential release to the environment.~~

19.2.5.2 Accident Management Programmatic Structure

The programmatic structure of management of severe accidents occurring in an NPM reflects lessons learned from industry experience and recent developments in severe accident response. Programmatic elements of severe accident management are: