December 27, 2017

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 182 (eRAI No. 9039) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 182 (eRAI No. 9039)," dated August 18, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 182 (eRAI No.9039)," dated September 20, 2017

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Questions from NRC eRAI No. 9039:

- 03.09.01-1
- 03.09.01-2

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

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

Sincerely,

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9039

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9039

NuScale Power

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

eRAI No.: 9039
Date of RAI Issue: 08/18/2017

NRC Question No.: 03.09.01-1

The staff reviewed DCD Section 3.9.1.1 to ensure that the relevant requirements of GDC 1, 2, 14, 15 and 10 CFR Part 50, Appendix S were met in regard to including a complete list of transients to be used in the design and fatigue analysis of ASME Code Class 1 and core support components, supports and reactor internals within the reactor coolant pressure boundary. The design transients define thermal-hydraulic conditions (i.e., pressure, temperature, and flow) for the NPM. Bounding thermal-hydraulic design transients are defined for components of the reactor coolant pressure boundary (RCPB).DCD Table 3.9-1, Summary of Design Transients, lists the design transients by ASME service level and includes the number of occurrences or cycles for each design transient based on a plant life of 60 years.

Load combinations and their acceptance criteria are given in Section 3.9.3 for mechanical components and associated supports and in Section 3.12 for piping systems. The Service Level A and B transients are representative events that are expected to occur during plant operation. These transients are severe or frequent enough to be evaluated for component cyclic behavior and equipment fatigue life, and the analyzed conditions are based on a conservative estimate of the frequency of occurrences as listed in Table 3.9-1 and magnitude of temperature and pressure changes. However, Table 3.9-1 does not include the seismic operating basis earthquake (OBE) load and cycles which is significant for performing a fatigue analysis. Clarify (1) whether the numbers in the last column of Table 3.9-1 is the number of occurrences or cycles for the named event, if so, the “events” should change to “Cycles,” and (2) whether a note should be added to Table 3.9-1 to address OBE for the fatigue analysis based on the discussion in Sections 3.7.3 and 3.12.4, “20 cycles of SSE” or “312 cycles of one third SSE (equivalent OBE)” to be used in lieu of one SSE and five OBE as required by fatigue analysis.

NuScale Response:

In a followup public telecon with NRC on December 12, 2017, it was requested that further clarification be provided in FSAR Table 3.9-1, Summary of Design Transients, as referenced in the response to eRAI 9039 Question 03.09.01-1, transmitted by RAIO-0917-56108, September 20, 2017.

Specifically, NuScale was asked to differentiate between Events and Cycles in the table header and to provide a brief discussion of seismic loadings consideration.

NuScale agreed to revise its RAI 9039 Question 03.09.01-1 response accordingly.

- The numbers in the last column of FSAR Table 3.9-1 are the number of cycles for the named event. For each event occurrence, the changes in hydraulic conditions that are of significance to fatigue are separated into sub-cycles and load steps, in order for the fatigue analysis to capture the local maximum and minimum stresses when stress pairs are calculated.

- FSAR Table 3.9-1 addresses the transient events that are considered for thermal fatigue analyses. The number of cycles for OBE are treated as separate loads for mechanical fatigue, for example see FSAR Table 3.12-1 for piping stress analyses. The approach for considering OBE cycles in the fatigue analysis is discussed in FSAR Sections 3.7.3 and 3.12.5 (for piping). An explanatory note has been added to FSAR Table 3.9-1.

Impact on DCA:

FSAR Tier 2 Table 3.9-1 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 03.09.01-1S1

Table 3.9-1: Summary of Design Transients

Event Name	ASME Service Level	~~Events~~Cycles for 60 Year Design Life
Reactor heatup to hot shutdown	Level A	200
Reactor cooldown from hot shutdown	Level A	200
Power ascent from hot shutdown	Level A	700
Power descent to hot shutdown	Level A	300
Load following	Level A	19,750
Load regulation	Level A	767,100
Steady-state fluctuations	Level A	5,000,000
Load ramp increase	Level A	2000
Load ramp decrease	Level A	2000
Step load increase	Level A	3000
Step load decrease	Level A	3000
Large step load decrease	Level A	200
Refueling	Level A	60
Reactor coolant system makeup	Level A	175,200
Steam generator inventory control from hot shutdown	Level A	600
High point degasification	Level A	440
Containment evacuation	Level A	66,000
Containment flooding and drain	Level A	120
Decrease in feedwater temperature	Level B	180
Increase in secondary flow	Level B	30
Turbine trip without bypass	Level B	90
Turbine trip with bypass	Level B	180
Loss of normal AC power	Level B	60
Inadvertent main steam isolation valve (MSIV) closure	Level B	30
Inadvertent operation of the decay heat removal system (DHRS)	Level B	15
Reactor trip from full power	Level B	125
Control rod misoperation	Level B	60
Inadvertent pressurizer spray	Level B	15
Cold overpressure protection	Level B	30
CVCS malfunctions	Level B	30
Spurious emergency core cooling system valve actuation	Level C	5
Inadvertent opening of a reactor safety valve	Level C	5
CVCS Pipe Break	Level C	5
Steam generator tube failure	Level C	5
Hydrogen Detonation	Level C	1
Steam piping failures	Level D	1
Feedwater piping failures	Level D	1
Control rod assembly ejection	Level D	1
Hydrogen Detonation with DDT	Level D	1
Primary hydrostatic test	Test	10
Secondary hydrostatic test	Test	10
Containment hydrostatic test	Test	10

Note:
The effects of earthquakes are not considered directly in the fluid systems design transient analyses. Where applicable, seismic loadings are considered in addition to the effects of transients in the fatigue analyses (see Table 3.9-3 to Table 3.9-14 for component load combinations and Table 3.12-1 to Table 3.12-3 for piping load combinations). Section 3.7.3.2 describes the number of seismic cycles used in fatigue evaluations of appropriate components.

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eRAI No.: 9039
Date of RAI Issue: 08/18/2017

NRC Question No.: 03.09.01-2

Generic Design Criterion (GDC) 1 to 10CFR50 requires that components important to safety be designed to high quality standard. The transient conditions selected for equipment design evaluation are based on the conservative estimates on the magnitude and frequency of temperature and pressure transients resulting from various operating conditions in the plant that may occur. The applicant is requested to provide the basis for assuming 90 cycles of turbine trip without bypass and 180 cycles of turbine trip with bypass. On each turbine trip, the stop valves close. Also confirm if the pressure transients due to turbine trips cover the transients due to stop valve closure which was considered to generate severe dynamic loads in the PWR practice.

NuScale Response:

In a followup public telecon with NRC on December 12, 2017, it was requested that further clarification be provided for the response to eRAI 9039 Question 03.09.01-2, as transmitted by RAIO-0917-56108, September 20, 2017.

Specifically, NuScale was asked to provide a basis for the number of turbine trip cycles evaluated.

NuScale agreed to revise the eRAI 9039 Question 03.09.01-2 response accordingly.

The number of cycles for the turbine trip without bypass and turbine trip with bypass transients are selected based on the predicted NuScale PRA initiating event frequencies, nuclear operating experience, and comparisons to recent design certification applications. The mean initiating event frequency for a general reactor trip is 1.3 per module critical year, per FSAR Table 19.1-8, which equates to 78 events over the 60 year design life. The event frequency for a turbine trip is included as a portion of this general reactor trip frequency. For bounding design purposes, 270 occurrences of a turbine trip are assumed.

The turbine trip with bypass is expected to be more frequent than the turbine trip without bypass, because the turbine bypass system is designed to provide 100 percent bypass flow, per FSAR Section 10.4.4, and results in minimal disruption to the NuScale power module operation. A loss of bypass flow in addition to the loss of turbine flow would require multiple equipment or control system failures, which is less likely to occur than a loss of turbine flow alone. Additionally, the turbine trip with bypass is a less significant event with respect to changes in operating conditions since the reactor does not trip. Because the turbine bypass is expected to normally operate following a turbine trip, the event is selected to occur two times more frequently than the turbine trip without bypass. Based on the bounding value of 270 total turbine trips, 180 cycles are selected for the turbine trip with bypass, and 90 cycles are selected for the turbine trip without bypass.

Dynamic fluid loads, such as those that could be generated during a rapid turbine stop valve closure, are considered in the piping load combinations, as discussed in FSAR Section 3.12.5.3, as well as in the component and component support load combinations, as discussed in FSAR Section 3.9.3.1.1.

Impact on DCA:

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