September 19, 2017 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. 146 (eRAI No. 9028) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 146 (eRAI No. 9028)," dated August 05, 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. 9028:

- 19-24

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9028

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NuScale Response to NRC Request for Additional Information eRAI No. 9028

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

eRAI No.: 9028
Date of RAI Issue: 08/05/2017

NRC Question No.: 19-24

Regulatory Basis

The information requested is needed to evaluate the applicant's assessment against criterion B for regulatory treatment of non- safety systems in accordance with guidance in Standard Review Plan Section 19.3, "Regulatory Treatment of Non-safety Systems for Passive Advanced Light Water Reactors," and ensure that safety functions are met in the extended period between 72 hours and seven days following an accident.

Background

In Section 19.3.2.2 of the Final Safety Analysis Report (FSAR), and related to regulatory treatment of non-safety-related systems (RTNSS) criterion B for establishing nonsafety-related structures, systems and components (SSCs) requiring regulatory treatment, the applicant states that the safety analyses, probabilistic risk assessment (PRA) insights (including seismic margins analysis), and expert panel considerations (discussed in Chapter 15, Section 19.1 and Section 17.4, respectively) did not reveal any non-safety-related SSCs relied on to perform a backup to passive safety functions (i.e., to ensure long-term safety) in the period of 72 hours to seven days following an accident or credited for seismic margins analysis (SMA). Therefore, no non-safety-related SSCs meet the RTNSS B criteria.

Request

Since no nonsafety-related SSCs were identified as being necessary for achieving safety functions in the period between 72 hours and seven days following an accident, it appears that the functions of core cooling and containment cooling are achieved in this extended period using only passive safety systems. Please explain how passive safety systems perform the safety functions of core cooling and containment cooling in the extended period between 72 hours and seven days following an accident. Please describe the capability of the heat sink(s) credited for the extended period and the extent to which operator action is needed to achieve the safety functions.

NuScale Response:

In the NuScale design, safety systems passively perform the safety-related functions of core cooling and containment cooling in the extended period between 72 hours and 7 days following an accident. These functions are automatically established and then passively maintained using only safety-related equipment.

Each NuScale Power Module operates partially immersed in the large, safety-related, pool of water forming the ultimate heat sink (UHS). The reactor pressure vessel (RPV) is housed in a steel containment vessel (CNV) that transfers sensible and core decay heat through the CNV walls to the UHS which provides an effective passive heat sink for both short and long-term heat removal. The containment isolation valves (CIVs) are designed to close upon receipt of a signal, or to fail closed on a loss of power, to perform the containment isolation function. Automatic actuation of the CIVs to close maintains reactor coolant system (RCS) inventory and automatic actuation of the decay heat removal system (DHRS) valves to open establishes natural circulation flow. Core cooling is provided by heat transfer through the DHRS heat exchangers which are submerged in the UHS. The emergency core cooling system (ECCS) valves automatically open to establish natural circulation flow of reactor coolant between the RPV and the CNV and to allow heat transfer from the fuel to the UHS. Peak temperature and pressure in the CNV are controlled passively by the CNV being partially immersed in the UHS, cooling the outer surface of the CNV.

By 72 hours following a design basis event, the safety-related systems described above are performing their safety-related functions of core cooling and containment cooling passively and continue to do so between 72 hours to 7 days following the event. As described in Section 9.2.5, the UHS has a cooling capability that extends well beyond this 7 day period; Table 9.2.5-2 shows that sufficient water is available in the UHS to cool the plant for more than 30 days with no makeup water, no active cooling systems, and no operator actions.

FSAR Section 19.3.2.2 has been revised as shown in the attached to summarize that safety-related systems perform their functions automatically, without operator action, and nonsafety-related structures, systems, and components are not relied on to perform a RTNSS B function by providing a backup to a passive safety function in the period of 72 hours to 7 days following an accident to ensure long-term safety.

Impact on DCA:

FSAR Section 19.3.2.2 has been revised as described in the response above and as shown in the markup provided in this response.

during the normal coping strategy, the SBO analysis described in Section 8.4 also demonstrates that core cooling and containment integrity are successfully maintained with only safety-related systems and no reliance on DC power systems. As such, there are no SSC for mitigating SBO that meet RTNSS criteria.

Since the issuance of SECY-95-132 that revised portions of SECY-94-084, the NRC has not identified any additional beyond design basis deterministic requirements within the scope of RTNSS A SSC (in addition to those for ATWS and SBO discussed above).

Based on the consideration of beyond design basis deterministic NRC performance requirements for ATWS and SBO, there are no SSC that meet the RTNSS A criteria.

19.3.2.2 RTNSS B

Nonsafety-related SSC functions identified through the D-RAP process in Section 17.4 are evaluated to determine whether they are relied upon to:

- provide a long term nonsafety-related back-up to passive system functional capability and for a period after 72 hours up to 7 days following an accident.
- meet the acceptance criteria for the seismic margins analysis (SMA).

RAI-19-7

The safety analyses, PRA insights ~~(including SMA)~~, and expert panel considerations (discussed in Chapter 15, Section 19.1, and Section 17.4, respectively) did not identify any nonsafety- related SSC relied on to perform a backup to passive safety functions (i.e., ensure long term safety) in the period of 72 hours to 7 days~~, nor credited for SMA~~.

RAI 19-24

The NuScale Power Modules are partially immersed in the reactor pool and protected using safety-related SSC. The reactor pressure vessel is housed in a steel containment vessel (CNV) that transfers sensible and core decay heat through the CNV walls to the ultimate heat sink which provides an effective passive heat sink for both short and long-term heat removal. The functions of core cooling and containment cooling are performed by safety-related SSC that operate automatically without operator action, fail-safe on a loss of power, and are passively maintained for extended periods following an accident. Therefore, nonsafety-related SSC are not relied on to perform a RTNSS B function for a period after 72 hours up to 7 days following an accident to ensure long-term safety.

RAI-19-7

The RTNSS B evaluation process also considered if any nonsafety-related SSC were candidates for additional regulatory oversight from seismic considerations.

RAI-19-7

As described in Section 19.1.5.1, both active and passive, nonsafety-related SSC are modeled in the SMA. None of the active, nonsafety-related SSC in the SMA are critical to a success path that averts core damage or a large release. These SSC are in the SMA model, but are modeled with high failure rates so there is limited credit for success