RAIO-0318-59156

March 15, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 195 (eRAI No. 9060) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 195 (eRAI No. 9060)," dated August 25, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 195 (eRAI No.9060)," dated October 24, 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 Question from NRC eRAI No. 9060:

- 06.02.04-7

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,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9060

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9060

NuScale Power™

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

eRAI No.: 9060
Date of RAI Issue: 08/25/2017

NRC Question No.: 06.02.04-7

The provisions contained in 10 CFR 50.34(f)(2)(xiv) require containment isolation systems, which in subparagraph (E), "Include automatic closing on a high radiation signal for all systems that provide a path to the environs." NuScale requests an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the containment evacuation system (CES). NuScale contends that the NuScale containment design has a small volume, which provides increased sensitivity of pressure changes within containment, and therefore, the CES isolation upon high containment pressure provides appropriate isolation for the CES system, considering the operating experience that led to the development 10 CFR 50.34(f)(2)(xiv) (see NUREG-0578, NUREG-0660, and NUREG- 0737). NuScale provided no quantitative justification supporting its exemption request. The staff observes that NuScale's containment sensitivity to pressure changes may be dampened by operation (or maloperation) of the containment evacuation system pumps (e.g., pumps remove gases and steam), which are non-safety related. Therefore, the staff requests that the applicant provide quantitative justification for the exemption request including a description of the events considered, the methodology used to evaluate the events, the evaluation/acceptance criteria used to determine that containment pressure provides appropriate isolation and that a high-radiation isolation signal is unnecessary, and how uncertainties were considered. The events considered should include design basis events, and degraded-core and core-melt accidents (per NUREG-0660 objective for the provisions that eventually would become 10 CFR 50.34(f)(2)(xiv)).

NuScale Response:

During a January 16, 2018 eRAI 9060 response follow-up call, the staff indicated that the information added to DCA Part 7, Chapter 13 by the initial response to the RAI includes information that is not in the FSAR. Specifically, the FSAR does not clearly indicate that the design of the NPM itself ensures generation of a low pressurizer level signal prior to core uncovery, since the pressurizer is an integral part of the reactor vessel, located well above the top of the core. Therefore, the comments in Table 1.9-5 concerning the departure basis for 10 CFR 50.34(f)(2)(xiv) are revised to align with the discussion in Table 1.9-3 and Part 7, Chapter 13. Additionally, FSAR Section 9.3.6 is also revised to align with the discussion in Table 1.9-3

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and Part 7, Chapter 13.

Impact on DCA:

Table 1.9-5 and Section 9.3 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 06.02.04-7S1, RAI 06.02.04-9, RAI 06.02.04-9S1, RAI 08.01-1, RAI 08.02-4, RAI 08.02-6, RAI 08.03.02-1, RAI 09.02.06-1

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(1)(i)	Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8)	Partially Conforms	Design certification will address reliability of core and containment heat removal systems, with an update required by COL applicant to reflect site-specific conditions.	19.0 19.1 19.2
50.34(f)(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (II.E.1.1)	Not Applicable	This rule requires an evaluation of proposed PWR auxiliary feedwater (AFW) systems. The NuScale plant design does have an AFW system like a typical LWR. Neither the literal language nor the intent of this rule applies to the NuScale design.	Not Applicable
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA (II.K.2.16 and II.K.3.25)	Not Applicable	The NuScale reactor design differs from large PWRs because the NuScale design does not require or include reactor coolant pumps. Rather, the NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power-operated relief valve (PORV) (II.K.3.2)	Not Applicable	This guidance is applicable only to PWRs that are designed with power-operated pressurizer relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and reactor core isolation cooling system initiation levels (II.K.3.13)	Not Applicable	This requirement applies only to BWRs.	Not Applicable
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves (II.K.3.16)	Not Applicable	This requirement applies only to BWRs. Regardless, the issue contemplated by this requirement was related to power-operated relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(2)(xiv)	Provide containment isolation systems that (A) ensure all non-essential systems are isolated automatically ; (B) ensure each non-essential penetration (except instrument lines) have two isolation barriers in series; (C) do not result in reopening of the containment isolation valves on resetting of the isolation signal; (D) use a containment set point pressure for initiating containment isolation as low as is compatible with normal operation; and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs (II.E.4.2)	Departure	~~The containment evacuation system has the potential for an open path from containment to the environs but is isolated upon a high containment vessel pressure signal, a low-low pressurizer level signal, a low alternating current voltage signal, or high under-the-bioshield temperature. Additionally, the CES discharge is re-directed into the gaseous radioactive waste system upon a high radiation signal.~~The NuScale design differs from that of a traditional large water reactor design of a TMI-era vintage because reactor core uncovery, and resulting core damage, cannot occur without reaching the low low pressurizer level containment isolation setpoint. The pressurizer is an integral part of the reactor vessel, located well above the reactor core, and not connected to the reactor core by piping. Design basis events meet their thermal and hydraulic acceptance criteria without reliance on isolating the CES in a high radiation signal. No design basis event results in degraded or damaged core conditions. Section 19.2 analyses demonstrate severe accident conditions, with resultant core damage, also result in generation of reliable containment isolation signals, without reliance on isolation on high containment radiation. An in-containment event resulting in core damage or degradation also results in containment isolation on low low pressurizer level and high containment pressure. An event that leads to core damage or degradation also results in containment isolation on low low pressurizer level. These features provide a reliable alternative means to prevent radiological release from the CES to the environs.	5.2.5 6.2.4 7.1.5 7.2.13 9.3.6

Consistent with GDC 5, structures, systems, and components shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Consistent with GDC 30, components that are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. General Design Criterion 30 was considered in the design of the CES and the CFDS. The CES provides three methods to detect and quantify leakage into the CNV. See Section 9.3.6 for a description of the three leakage detection methods.

Consistent with GDC 60, the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. General Design Criterion 60 was considered in the design of the CES and the CFDS. Liquid effluents are returned to the reactor pool or, as necessary, routed to the RWDS. Gaseous effluents are filtered and routed to the Reactor Building HVAC system (RBVS) or, when specified limits are exceeded, to the GRWS, where they are contained, monitored, and processed for release to the environment.

Consistent with GDC 64, means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents. General Design Criterion 64 was considered in the design of the CES and the CFDS. Liquid effluents are returned to the reactor pool or, as necessary, routed to the RWDS. Gaseous effluents are filtered and routed to the Reactor Building HVAC system (RBVS) or, when specified limits are exceeded, to the GRWS, where they are contained, monitored, and processed for release to the environment.

Consistent with 10 CFR 20.1101(b), the CES and the CFDS have provisions for draining and flushing piping and major components to the RWDS prior to maintenance or inspection activities to reduce dose to onsite personnel to as low as reasonably achievable (ALARA). Consistent with 10 CFR 20.1406, the CES and the CFDS have provisions for draining and flushing piping and major components to the RWDS prior to maintenance or inspection activities to reduce generation of radioactive waste and minimize contamination of the facility and the environment.

RAI 06.02.04-7S1

The NuScale design supports exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES. ~~The CES will have radioactivity monitors on the discharge path of the system; however, they will not trigger a safety-related containment isolation signal in the event of a high radioactivity signal. Rather, upon a high radioactive effluent signal on the gaseous discharge path, the system automatically diverts the gaseous effluent from the RBVS to the GRWS using nonsafety-related signals and equipment.~~Design basis events meet their thermal and hydraulic acceptance criteria without reliance on isolating the

CES in a high radiation signal. No design basis event results in degraded or damaged core conditions. Section 19.2 analyses demonstrate severe accident conditions, with resultant core damage, also result in generation of reliable containment isolation signals, without reliance on isolation on high containment radiation. An in-containment event resulting in core damage or degradation also results in containment isolation on low low pressurizer level and high containment pressure. An event that leads to core damage or degradation also results in containment isolation on low low pressurizer level. These features provide a reliable alternative means to prevent radiological release from the CES to the environs. Refer to Part 7, Chapter 13 for further details.

9.3.6.2 System Description

9.3.6.2.1 Containment Evacuation System

General Description

Each NPM is supported by a dedicated, nonsafety-related CES. The CES establishes and maintains a vacuum in the CNV by removing water vapor and non-condensable gases from the CNV using a vacuum pump that draws gases from the top of the CNV and discharges the gases to the CES condenser as shown in Figure 9.3.6-1.

Condensate from the CES condenser is gravity drained to a sample vessel before being gravity drained to the RWDS. Samples of the non-condensable gases are directed to the process sampling system sample panel for analysis.

The CES is operated from the main control room (MCR) using the module control system (MCS) that provides both automatic and operator control of key CES functions, including valve alignment, vacuum pump speed, and purge gas flow. The MCS provides

- indication, alarms, and interlocks for CES flow, temperature, pressure, radioactivity level, humidity, and valve position
- alarms and required automatic actuation for off normal conditions

Electrical power for the CES vacuum pumps and valves is provided by the low voltage AC electrical distribution system (ELVS). Electrical power for the CES instrumentation and control equipment is provided by normal DC power system (EDNS).

Component Descriptions

Vacuum Pump

The CES for each NPM includes two parallel, 100-percent capacity, mechanical vacuum pumps. Redundant pumps are provided to allow maintenance activities during normal operation. The vacuum pumps are 100-percent duty cycle, variable-speed pumps with dry process-side internals. The vacuum pumps are cooled by the reactor component cooling water system (RCCWS).