RAIO-0119-64190

January 16, 2019

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 472 (eRAI No. 9445) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 472 (eRAI No. 9445)," dated May 10, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 472 (eRAI No.9445)," dated June 12, 2018

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. 9445:

- 16-44

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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9445

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9445

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

eRAI No.: 9445

Date of RAI Issue: 05/10/2018

NRC Question No.: 16-44

10 CFR 50.36(c)(1)(i)(A) requires that technical specifications include safety limits for nuclear reactors that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.

In a letter dated December 8, 2017 (ML17342B343), the applicant updated DCA Part 2, FSAR Chapters 1, 2, and 15, and Part 4, Technical Specifications (TS), with conforming changes to reflect Revision 1 of Licensing Topical Report, NuScale Power Critical Heat Flux Correlations

TR-0116-21012, which adopts a new critical heat flux (CHF) correlation, NSP4. The letter stated in part:

Note that the Technical Specification Safety Limits affected by the implementation of the NSP4 correlation were also modified to relocate the critical heat flux correlation values from the Safety Limit to the Core Operating Limits Report (COLR). The requirement for and contents of the COLR are described in Technical Specification 5.6.3. This relocation is consistent with similar approved Technical Specification changes implemented at the Farley nuclear plant (ML013400451).

The staff reviewed the safety evaluation for Amendment No. 151 to Facility Operating License No. NPF-2 and Amendment No. 143 to Facility Operating License No. NPF-8, for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively, which were issued on December 4, 2001. The amendments included a change to plant-specific TS Subsection 2.1.1, "Reactor Core SLs," that moved the curves depicting departure from nucleate boiling (DNB) criterion correlation limits to the Core Operating Limits Report (COLR), a report specified by plant-specific TS Subsection 5.6.5. This relocation was based on Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from TS," dated October 4, 1988 (ML031200485). The Farley amendments were also based on a topical report WCAP-14483-A, approved January 19, 1999 (ML020430092). The NRC staff safety evaluation for WCAP-14483 states, "Safety limits, however, may not be placed in the COLR."

Accordingly, the staff does not accept the proposed relocation of reactor core SL critical heat flux correlation values from GTS SL 2.1.1.1 to the COLR. The applicant is requested to restore these SL values to SL 2.1.1.1 and make conforming changes to the associated Bases in Subsection B 2.1.1 and the list of specifications, which reference the COLR, in Subsection 5.6.3, paragraph a.

NuScale Response:

The following clarification was received from the NRC staff during a public meeting on November 5, 2018:

> As noted, the applicant's response (ML18163A417) to RAI 472-9445, Question 16-44, does not include Griffith-Zuber CHF correlation; also, the 1.06 value for Hench-Levy is not consistent with information provided in FSAR Section 15.6.6 or the LOCA topical report. As part of the review for TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” the staff issued RAI 9536 (ML18167A015), Question 15.6.6-2, requesting that the applicant (1) submit a methodology, for NRC staff review, that describes the experimental data supporting the development of the CHFR limits for the Griffith-Zuber and Hench-Levy correlations, and that demonstrates the CHF models have sufficient validation as demonstrated through appropriate quantification of error, and (2) update the appropriate licensing documentation to consistently reflect the final CHFR limits. The response to Question 15.6.6-2 is pending.

The response to RAI 15.06.06-2 was provided on September 21, 2018 in NuScale letter RAIO-0918-61859 (ML18264A338).

Reactor core safety limits are included in the technical specifications as required by 10 CFR 50.36. The regulation requires limits be established on important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of reactivity. The correlation limits are provided to ensure that the hot fuel rod in the core does not reach or exceed critical heat flux limits that could damage the rod cladding.

Consistent with the standard technical specifications of existing plants, the safety limit in specification 2.1.1, Reactor Core SLs, are applicable in MODE 1 when the reactor is, or may be critical. The three correlation safety limits provided in the technical specifications are the safety limits that must be set to satisfy the requirements of 10 CFR 50.36 for safety limits.

In addition to the evaluation of postulated events during critical operations, the NuScale design safety analyses evaluate the effects of other, post-reactor trip transients on reactor cladding to ensure the integrity of the barrier to radioactivity release is maintained. As described in the response to RAI 15.06.06-2, the Griffith-Zuber correlation is used to evaluate CHFR in low core flow conditions that typically exist during postulated events post-reactor trip. This limit is not included in the technical specifications because it is not a process variable that is controlled by plant conditions during operations. CHF is one of many acceptance criteria applied in analyses to demonstrate that the design adequately limits the release of radioactive material after a postulated event.

In summary, the CHFR limits in specification 2.1.1.1 are met through a combination of operating limits and design criteria. The Griffith-Zuber CHFR limitation is used as an accident analysis design criteria that is not directly dependent on operating conditions or controllable by the reactor operators.

Based on this justification, the Griffith-Zuber correlation limit is not included in the technical specifications, but is used in the design and accident analyses of the plant.

Impact on DCA:

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