RAIO-1217-57935

December 28, 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. 274 (eRAI No. 9137) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 274 (eRAI No. 9137)," dated November 03, 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. 9137:

- 03.12-10

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 Response to NRC Request for Additional Information eRAI No. 9137

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9137

NUSCALE POWER™

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

eRAI No.: 9137
Date of RAI Issue: 11/03/2017

NRC Question No.: 03.12-10

GDC 14 requires that the reactor coolant pressure boundary (RCPB) being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. NRC Bulletin (BL) 88-08, “Thermal Stresses in Piping Connected to Reactor Cooling Systems,” issued June 22, 1988, requests licensees to identify and evaluate the piping systems connected to the RCS susceptible to thermal stratification, cycling, and striping (TASCS) to ensure that the piping will not be subjected to unacceptable thermal stresses.

The operating experience described in the bulletin needs to be incorporated in the design in accordance with 10 CFR 52.47(a)(22). SRP Section 3.12 includes criteria related to this bulletin.

FSAR Tier 2, Section 3.12.5.7 shows that according to screening performed the NuScale piping systems connected to the RCS are not susceptible to TASCS. NuScale, piping systems connected to the RCS are not susceptible to TASCS. It is also stated that it used the screening criteria and evaluation methodology of EPRI TR-103581, "Thermal Stratification, Cycling, and Striping (TASCS)” dated July, 1999 to access unisolable piping connected to RCS to identify TASCS in the NuScale design and that it also used as a supplement “Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines," EPRI TR-1011955, June 22, 2011.

EPRI TR-103581 was an earlier EPRI report to assist utilities in addressing BL 88-08. According to EPRI TR-1011955, earlier methodology was questionable primarily because it did not exactly predict the cycling location and loadings for the Farley Unit 2 Safety Injection line leak from a circumferential crack that occurred in 1987. EPRI TR-1011955 is an updated ongoing program that licensees and applicants customarily use to manage TASCS. EPRI TR-1011955 provides a model for predicting and evaluating thermal cycling for PWR stagnant lines. The EPRI TR-1011955 model has been shown through benchmarking results to be effective in predicting the location of thermal cycling in lines attached to the RCS.

The applicant is requested to show how its model for TASCS is consistent with the EPRI TR-1011955 model and to identify the differences and provide justifications.

NuScale Response:

NUREG-0800, SRP Section 3.12, Acceptance Criterion C.ix. requests that the applicant identify and evaluate any piping connected to the RCS that is unisolable and potentially subject to temperature stratification or oscillation. If the design is vulnerable to such stratification or oscillation, then the application should describe a program meeting the criteria in NRC Bulletin 88-08 to ensure that this piping is not subjected to unacceptable thermal stresses. These actions help maintain the integrity of the reactor coolant pressure boundary in accordance with 10 CFR 50 Appendix A, General Design Criterion 14.

NuScale has performed the evaluation of unisolable piping that is part of the reactor coolant pressure boundary using EPRI TR-103581 (Reference 1), as discussed in FSAR Tier 2 Section 3.12.5.7. The FSAR section mistakenly also identified that EPRI TR-1011955 (MRP-146), “Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non Isolable Reactor Coolant System Branch Lines,” is used as a supplement in the screening process. FSAR Section 3.12.5.7 has been revised to remove this reference to EPRI MRP-146.

The development of EPRI’s Thermal Fatigue Program, which resulted in the issuance of MRP-146 and its subsequent revisions, has led to changes in the thermal fatigue screening criteria documented in EPRI TR-103581. A comparison between the TR-103581 criteria and the MRP-146 criteria is shown in Table 1, showing that the pipe size criterion is the same between the two documents but that the other criteria differ. The table compares the most basic (e.g., “Level 1”) screening criteria.

Table 1. Comparison between EPRI thermal fatigue screening criteria

TR-103581 Level 1 Screening Criteria	MRP-146 Screening Criteria
Line screens out from further consideration if any of the following are met: - pipe system maximum normal operating temperature differential is less than 220 °F - pipe size less than or equal to NPS 1 - pipe is sloped at least 45 degrees from the horizontal plane	Line screens out if - the branch line is not stagnant during normal plant operation(1) - pipe size is NPS 1 or smaller(2) - it is an up-horizontal or horizontal pipe with no potential for inleakage(1)

Footnotes:

(1) Reference 2, Slide 15

(2) Horizontal and down-horizontal lines of size NPS 1 or smaller were excluded by MRP-146, Rev. 0 (Reference 2, Slide 15). Following identification of program improvement opportunities, EPRI released interim guidance that reduced the screen-out dimension for up-horizontal piping from NPS 2 to NPS 1 (Reference 3, Attachment 1, Section 2.3). The interim guidance was incorporated into MRP-146, Rev. 2 in September 2016 (Reference 4, Slide 9).

The following lines are connected to the NuScale reactor coolant system and are evaluated for thermal stratification or cycling using EPRI TR-103581 screening criteria and evaluation methods, as described in FSAR Section 3.12.5.7:

- chemical and volume control RCS discharge piping
- chemical and volume control RCS injection piping
- pressurizer spray lines
- reactor pressure vessel high point degasification piping
- emergency core cooling system (ECCS) hydraulic lines

Per the evaluation in FSAR Section 3.12.5.7, none of these lines was determined to be susceptible to thermal stratification or cycling. Using the MRP-146 screening criteria from the Table 1 above, these lines are reevaluated in Table 2.

Table 2. MRP-146 screening

Line	Screening Outcome
RCS discharge piping	The RCS discharge piping is not stagnant during normal operation and does not require further evaluation.
RCS injection piping	The RCS injection piping is not stagnant during normal operation and does not require further evaluation.
Pressurizer spray lines	As discussed in FSAR Section 5.4.5, a reduced spray flow is continuously maintained during normal operation to minimize stresses on spray line components from thermal transients, thus the line is not stagnant. This screens out of further evaluation according to MRP-146.
Reactor pressure vessel high point degasification piping	The degasification piping is a vapor-filled, up-horizontal line with no potential for inleakage; therefore, it does not require further evaluation according to MRP-146.
ECCS hydraulic lines	The ECCS hydraulic lines are smaller than NPS 1, thus no further evaluation is required per MRP-146.

The results of the MRP-146 screening documented above are the same as the original screening in FSAR Section 3.12.5.7; therefore, there is no technical advantage to be gained by using MRP-146 instead of EPRI TR-103581 as it pertains to the NuScale design. By completing the screening documented in Section 3.12.5.7 and showing that the branch lines do not need to be part of a Bulletin 88-08 program for providing continuing assurance of acceptable thermal stresses, the operating experience in Bulletin 88-08 is incorporated into the NuScale design in accordance with 10 CFR 52.47(a)(22).

References

1. Electric Power Research Institute, "Thermal Stratification, Cycling, and Striping (TASCS)," EPRI TR-103581, March 1994, Palo Alto, CA.

2. Electric Power Research Institute, “Thermal Fatigue Management Guideline for Normally Stagnant Non-Isolable RCS Branch Lines,” presentation to the U.S. Nuclear Regulatory Commission, January 5, 2012, Agencywide Document Access and Management System (ADAMS) Accession No. ML12004A031.

3. Rudell, Bernie, Exelon, and A. Demma, EPRI, letter to Robert O. Hardies, U.S. Nuclear Regulatory Commission, “EPRI-MRP Interim Guidance for Management of Thermal Fatigue,” July 6, 2015, ADAMS Accession No. ML15189A100.

4. Electric Power Research Institute, “MRP Thermal Fatigue Program Update,” presentation at the Materials R&D Tech Exchange Meeting, May 23, 2017, ADAMS Accession No. ML17142A005.

Impact on DCA:

FSAR Tier 2 Section 3.12.5.7 and References Section 3.12.7 have been revised as described in the response above and as shown in the markup provided in this response.

Annex D of IEEE Std-344 (Reference 3.12-5). When this method is used, and if the amplitude of the vibration is taken as 1/3 of the amplitude of the SSE, then 312 fractional amplitude SSE cycles are considered.

3.12.5.6 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

Design and analysis of ASME Code Class 2 and 3 piping systems and piping components considers fatigue effects if they are subject to a total number of equivalent full temperature cycles greater than 7000 per NC-3611.2. Instead of analyzing ASME Code Class 2 and 3 components for specific cyclic loads (as for Class 1 components using cumulative usage factors), the fatigue effects are addressed by applying stress range reduction factors as provided in NC/ND-3611.2(e) to the allowable stress range for thermal expansion stresses.

3.12.5.7 Thermal Oscillations in Piping Connected to the Reactor Coolant System

The piping sections that can not be isolated and are connected to the reactor coolant system (RCS) can experience temperature stratification and oscillation due to mixing with stagnant lower temperature fluid with the higher temperature fluid at the connection interface (due to turbulent penetrating flow or leakage past an isolation component). These thermal conditions add fatigue loads to piping components due to constrained thermal deflections that must be accounted for by analysis or they can be precluded by design. Thermal oscillations in RCS connected piping were determined to be the cause of pressure boundary component failures at multiple operating nuclear plants as described in NRC Bulletin 88-08 including supplements. Therefore, unisolable sections of piping connected to the RCS of the NuScale design are evaluated for susceptibility to temperature oscillations which may affect the integrity of the components.

The screening criteria and evaluation methodology of Electric Power and Research Institute (EPRI) technical report TR-103581 (Reference 3.12-7) is used to assess unisolable piping connected to the RCS for thermal oscillations in the NuScale design. ~~EPRI TR-1011955 (Reference 3.12-6) is also used as a supplement in the screening process.~~

For thermal stratification to occur in unisolable piping connected to the RCS which could impose additional fatigue loads on pressure boundary components, the following conditions must exist:

- An isolation component (e.g. a valve) exists in the design with the potential for leakage, which separates stagnant, colder fluid from the RCS. In this configuration a pressure differential must also exist across the isolation component to drive flow through a potential leakage path.
- An unisolable section of stagnant branch piping connected to the RCS that is oriented horizontally or oriented vertically which then transitions to a horizontal run within the span of turbulent RCS penetration (from the point of interface between the branch and the RCS)

Additional fatigue loads are imposed on components when a mechanism exists to promote cycling of the stratified conditions. Depending on the mechanism a large

3.12-2 "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Other Loads and Load Combinations," NUREG 1061 Volume 4, December 1984.

3.12-3 American Society of Mechanical Engineers, Power Piping - ASME Code for Pressure Piping B31, ASME B31.1, New York, NY.

3.12-4 Welding Research Council, Inc. "Position Paper on Nuclear Plant Pipe Supports," WRC Bulletin 353, May 1990, Shaker Heights, OH.

3.12-5 Institute of Electrical and Electronics Engineers, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 344-1987, Piscataway, NJ.

3.12-6 ~~Electric Power Research Institute, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 1)," EPRI TR-1011955, June 22, 2011, Palo Alto, CA.~~Not used.

3.12-7 Electric Power Research Institute, "Thermal Stratification, Cycling, and Striping (TASCS)," EPRI TR-103581, July 7, 1999, Palo Alto, CA.

3.12-8 American National Standards Institute/American Institute of Steel Construction, "Specification for Safety-Related Steel Structures for Nuclear Facilities," ANSI/AISC N690-12, January 31, 2012, Chicago, IL.

3.12-9 American Institute of Steel Construction, "Steel Construction Manual," 14th ed. Chicago, IL.

3.12-10 American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-13) and Commentary," July 10, 2014, Farmington Hills, MI.

3.12-11 NUREG/CR-1980, Dynamic Analysis of Piping Using the Structural Overlap Method.

3.12-12 March 1, 2014 NRC white paper- Piping Level of Detail for design Certification (ML14065A067).

3.12-13 NuScale Power Module Analysis Technical Report TR-0916-51502

3.12-14 American Society of Civil Engineers, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary,” ASCE 4-98, Reston, VA.

3.12-15 MSS SP-58 - Pipe Hangers and Supports - Materials, Design, Manufacture, Selection, Application, and Installation - SP-58-2009.

3.12-16 NUREG/CR-1677, “Piping Benchmark Problems. Volume I and II.”