NUSCALE POWER™

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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. 449 (eRAI No. 9497) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 449 (eRAI No. 9497)," dated May 01, 2018

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 Questions from NRC eRAI No. 9497:

- 15.04.07-1
- 15.04.07-2

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 Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9497

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NuScale Response to NRC Request for Additional Information eRAI No. 9497

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

eRAI No.: 9497
Date of RAI Issue: 05/01/2018

NRC Question No.: 15.04.07-1

General Design Criterion 10, "Reactor design," in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). GDC 13 requires that instrumentation be provided to monitor variables and systems over their anticipated ranges of normal operation, including the effects of AOOs, and appropriate controls to maintain listed variables and systems within prescribed operating ranges.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," provides the staff guidance in reviewing potential fuel loading errors. The areas of review under SRP Section 15.4.7 include identification of the worst situation undetectable by incore instrumentation and the resulting changes in the power distribution.

To allow the staff to ascertain that the most limiting misload has been identified, provide the following details regarding the worst-case misload, and update FSAR Section 15.4.7 to include this information:

- Location of the worst-case misload
- Whether the worst-case misload involves rotation of an assembly
- The resulting change in the power distribution (such as power peaking augmentation factors as input to the subchannel analysis)

NuScale Response:

FSAR Section 15.4.7.2 describes the potential mechanisms by which a fuel assembly may be inadvertently loaded in an improper position: characterized as either a shuffle misload, where two assemblies are swapped, or a rotational misload, where a single assembly is rotated in place. All possible combinations of shuffle and rotational misloads are evaluated for detectability of the deviation between measured versus predicted power distribution.

The limiting undetectable misload is a swap of two adjacent assemblies in the 10 and 13 locations as seen in the 'Quarter Core' portion of Figure 15.4-28. This misload results in a power peaking augmentation factor of 1.189. This information was added to FSAR Section 15.4.7.

In general, for the NuScale reactor core, limiting undetectable misloads are a result of a shuffle swap of two fuel assemblies with substantially different burnups (i.e. assemblies from different batches). An in-place rotational misload tends to be non-limiting because the intra-assembly burnup gradients are not significant enough to result in limiting deviations of the radial core power distribution.

Impact on DCA:

FSAR Section 15.4.7 has been revised as described in the response above and as shown in the markup provided in this response.

potentially be shuffled into assembly locations 13 through 18, resulting a total of 72 potential fuel misloads to be considered on a half-core basis. There are 231 potential misloads that are analyzed.

The center assembly in the equilibrium cycle is a fresh assembly. Therefore, exchanges of the center assembly are only examined on a quarter-core basis because the faces of the center assembly do not have a different depletion history than each other and exchanges in the other quadrants will be consistent with those performed in a single quadrant.

Rotational Misloads

The fuel assembly top nozzle has two holes that mate with pins in the upper core plate, and a third alignment hole that mates with the fuel handling equipment (Section 4.2). These features collectively, prevent fuel assembly rotational misloads. Nevertheless, 180 degree rotational misloads are conservatively examined.

15.4.7.3 Core and System Performance

15.4.7.3.1 Evaluation Model

The design and analysis of the NuScale Power Module reactor core is performed with the Studsvik Scandpower Core Management Software suite of reactor simulation tools. A discussion of the analysis tools and analytical methods is provided in Section 4.3.3.

RAI 15.04.07-1

SIMULATE5 is an advanced three-dimensional (3D), steady-state, multi-group nodal reactor analysis code capable of multi-dimensional nuclear analyses of reactors. SIMULATE5 is used to determine the limiting undetectable fuel misload, and to provide peaking factors to the subchannel analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 and Section 4.4 for a discussion of the VIPRE-01 code and evaluation model.

15.4.7.3.2 Input Parameters and Initial Conditions

The fuel misload event changes the power distribution of the core, but the thermal hydraulic boundary conditions remain the same. Therefore, there is no need for an NRELAP5 analysis to ensure that the RCS pressure remains below the design limit of the RPV. The power distribution of the equilibrium core analysis is discussed in Section 4.3.

RAI 15.04.07-1

The power peaking augmentation factors for the limiting undetectable misload are calculated using SIMULATE5, and provided as input to the steady-state subchannel analysis to determine the MCHFR for this event. The limiting undetectable misload is a swap of two adjacent assemblies in the 10 and 13 locations as seen in the 'Quarter Core' portion of Figure 15.4-28. This limiting misload power peaking factor augmentation is 1.189. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1.

15.4.7.3.3 Results

RAI 15.04.07-1

The limiting undetectable fuel misloading event results in an MCHFR~~,~~ which is above the 95/95 CHFR limit. Fuel temperature margin to centerline melt is calculated for the worst case fuel assembly misloading event. The calculated value of Linear Heat Generation Rate (LHGR) for the worst misload is below the limiting LHGR. These results are provided in Table 15.4-20. Because MCHFR is above the limit and fuel centerline melting is not expected to occur, no fuel damage is expected. These events change the power distribution within the core, not overall core power. Therefore, there is no power increase associated with the fuel misloading events that could challenge the radionuclide barriers.

15.4.7.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design basis accident analyses presented in Section 15.0.3.

15.4.7.5 Conclusions

The results from the evaluation of the limiting undetectable fuel misloading events show that no fuel damage is expected. There is no pressure transient associated with this event, so the RCS pressure boundary is not challenged. With no fuel damage and no challenge to radionuclide boundaries, the normal leakage related radiological consequences of this event are bounded by the design basis accident analyses in Section 15.0.3. Therefore, all SRP 15.4.7 acceptance criteria are met.

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

A postulated failure of the CRDM pressure housing could cause a control rod to be ejected from the core. The unexpected and rapid increase in positive reactivity demonstrates the effects of a limiting reactivity insertion event.

The power spike resulting from the CRA ejection is quickly countered by the fuel reactivity feedback as the fuel temperature begins to increase. The sudden increase in power is detected by the MPS, resulting in a reactor trip. The sudden ejection of a CRA adds positive reactivity to a localized region of the core in a very short period of time. This CRA ejection results in a power excursion in the region near the affected fuel assembly and results in a highly asymmetric power distribution in the radial dimension. This adverse power distribution subsequently leads to overheating of the affected fuel assemblies and possible fuel damage.

NuScale Power

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

eRAI No.: 9497
Date of RAI Issue: 05/01/2018

NRC Question No.: 15.04.07-2

GDC 13 requires that instrumentation be provided to monitor variables and systems over their anticipated ranges of normal operation, including the effects of AOOs, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges.

The staff requests further clarification or correction of the following statements in FSAR Tier 2, Section 15.4.7:

- FSAR Section 15.4.7.2: "The overpower fraction detection threshold is 1.44 and the underpower fraction detection threshold is 0.65. These fractions mean that an assembly could be as much as [emphasis added] 44 percent above its predicted power or 35 percent below its predicted power to be detectable by the core monitoring system." However, the staff understands that the deviation must be higher than the detection threshold. Therefore, it seems that an assembly must be at least 44 percent above its predicted power or 35 percent below its predicted power to be detectable by the core monitoring system.
- FSAR Section 15.4.7.3.3 states that the results for the limiting undetectable fuel misloading event are in FSAR Table 15.4-15, but the results are shown in FSAR Table 15.4-14.

The information in the design certification application that demonstrates how GDC 13 is met needs to be precise and consistent so the staff is able to make a reasonable assurance finding. Regarding the above items, either (1) update the FSAR to correct them or (2) justify why the information is correct as currently written.

NuScale Response:

FSAR Section 15.4.7.2 describes the power fraction detection thresholds for misloaded fuel assemblies. There can be some misloaded assemblies with power fractions that do not exceed these thresholds that will still be detected. However, any assembly with a power fraction exceeding these thresholds will be detected. The text in FSAR Section 15.4.7.2 has been updated to clarify these conditions as shown in the markup provided with this response.

FSAR Section 15.4.7.3.3 correctly states that the results for the fuel misloading event are provided in Table 15.4-20.

Impact on DCA:

FSAR Section 15.4.7 has been revised as described in the response above and as shown in the markup provided in this response.

15.4.7.2 Sequence of Events and Systems Operation

RAI 15.04.07-2

The core monitoring system detects a fuel loading error if it causes a relative power shape deviation higher than a detection threshold. The overpower fraction detection threshold is 1.44 and the underpower fraction detection threshold is 0.65. These fractions mean that if an assembly is~~could be as much as~~ 44 percent above its predicted power or 35 percent below its predicted power it will~~to~~ be detected~~able~~ by the core monitoring system. Some assembly misloads can be detected before these power fraction detection thresholds are reached. Fuel assembly manufacturing practices and quality assurance techniques ensure that assemblies containing un-prescribed enrichments or burnable poison loadings are not available on site for initial loading or reload of the core. Therefore, no misloads related to un-prescribed enrichments or burnable poisons loadings for a given core loading plan are considered. The entire spectrum of potential shuffle and rotational fuel assembly misloads are considered.

Shuffle Misloads

The spectrum of potential fuel assembly misloads for the NuScale reactor core is examined to assess the impacts of an undetectable fuel assembly misload being present during normal operations. Figure 15.4-28 shows the full spectrum of fuel misloads considered.

The fuel assembly misloads are evaluated in the three categories shown in Figure 15.4-28: quarter-core, half-core, and cross-core. Each of the assembly locations that are evaluated as misloads are numbered for each of the three categories. For the quarter-core misloads, assembly locations 1 through 13 could potentially be shuffled into any of the other numbered assembly locations, which results in a total of 78 potential fuel misloads to be considered on a quarter-core basis. For the half-core misloads, assembly locations 1 through 9 could potentially be shuffled into assembly locations 10 through 18, a total of 81 potential fuel misloads to be considered on a half-core basis. For the cross-core misloads, assembly locations 1 through 12 could potentially be shuffled into assembly locations 13 through 18, resulting a total of 72 potential fuel misloads to be considered on a half-core basis. There are 231 potential misloads that are analyzed.

The center assembly in the equilibrium cycle is a fresh assembly. Therefore, exchanges of the center assembly are only examined on a quarter-core basis because the faces of the center assembly do not have a different depletion history than each other and exchanges in the other quadrants will be consistent with those performed in a single quadrant.

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