Response to SDAA Audit Question

Question Number: A-15.4.7-4S2

Receipt Date: 08/01/2024

Question:

Audit question A-15.4.7-4 was received by NuScale on April 15, 2024. NuScale provided a response on May 7, 2024. On May 30, 2024, NuScale received writen feedback from the NRC. NuScale provided a supplemental response on July 12, 2024. On August 1, 2024, the following written feedback from the NRC:

The information provided in the response is insufficient for the staff to make technical finding on the acceptability of the revised fuel misload calculation.

The applicant revised its calculation supporting FSAR Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in and Improper Position" (misload) to consider an uncertainty of 20% in the assembly power {{

}}2(a),(c) The staff considers this assumption reasonable based on open publications on self-powered neutron detectors and necessary algorithms for gamma compensation (References 1, 2) and engineering judgement on core power synthetization. However, the staff notes that the audit response states: "With more cases screened in, the analysis uses a higher misload peaking factor of 1.15 (previously 1.10) to bound the peaking associated with operation of these misloads." Based on Section 4.7.4 of TR-0915-17564-P, "Subchannel Analysis Methodology" (ML18305B221) and ER-A021-3589, "Nuclear Analysis Methodology," Revision 4, the staff understands that this statement refers to an adjustment in the misload augmentation factor for $F_{\Delta H}$, which is defined as the ratio of the maximum $F_{\Delta H}$ of the core for any time in cycle following a misload and the maximum $F_{\Delta H}$ of the as-designed core for any time in cycle. TR-0915-17564-P states: "The subchannel analysis evaluates the maximum allowed augmentation factor that results in MCHFR at the CHF analysis limit." The TR further states: "The key result of the misload subchannel analysis is the maximum allowed radial peaking augmentation factor. Each cycle-specific nuclear analysis confirms that the maximum augmentation factor calculated is not violated." {{ }}2(a),(c)

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}}2(a),(c) Please clarify how the $F_{\Delta H}$ augmentation factor is determined and used and revise the FSAR to clarify any inconsistency.

Please also provide the acceptance criterion, together with the technical basis, for selecting the $F_{\Delta H}$ augmentation factor and explain why the selected $F_{\Delta H}$ augmentation factor of 1.15 is appropriate and conservative. {{

}}2(a),(c) Also, please provide for NRC staff audit the assemblywise and pinwise power distribution that shows the maximum power differences between the designed and actual power distributions for the "undetected" misload resulting in the highest $F_{\Delta H}$ augmentation factor.

In addition, the staff notes that the change in peak linear heat generation rate shown in the markup to Table 15.4-18 appears smaller than would be expected from the change in $F_{\Delta H}$ augmentation factor. Please provide an updated revision of EC-115225, "Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position" or other calculation documents for subchannel analyses of misloads that are not detectable with the 20% uncertainty.

References:

1. Wanno Lee et. al., "A Study on the Sensitivity of Self-Powered Neutron Detector (SPND)", Nuclear Science Symposium, 1999. Conference Record. 1999 IEEE Volume: 2
2. Todt, William H. Sr. "Characteristics of self-powered neutron detectors used in power reactors." Switzerland: European Nuclear Society, 1998.

Response:

This response is a supplement to the April 15, 2024 and July 12, 2024 responses to address the feedback from August 1, 2024; the original April 15, 2024 response and July 12, 2024 supplemental response remain unchanged in the electronic reading room (eRR).

To address the detailed NRC second round of feedback, each aspect of the feedback is addressed by providing the text of the feedback in indented, italicized text, followed by the NuScale response in un-indented, regular text.

> *"The applicant revised its calculation supporting FSAR Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in and Improper Position" (misload) to consider an uncertainty of 20% in the assembly power {{*
>
> *}}2(a),(c) The staff considers this assumption reasonable based on open publications on self-powered neutron detectors and necessary algorithms for gamma compensation (References 1, 2) and engineering judgement on core power synthetization."*

NuScale agrees.

> *"However, the staff notes that the audit response states: "With more cases screened in, <u>the analysis uses a higher misload peaking factor of 1.15 (previously 1.10) to bound the peaking</u> associated with operation of these misloads." Based on Section 4.7.4 of TR-0915-17564-P, "Subchannel Analysis Methodology" (ML18305B221) and ER-A021-3589, "Nuclear Analysis Methodology," Revision 4, the staff understands that this statement refers to an adjustment in the misload augmentation factor for $F_{\Delta H}$, which is defined as the ratio of the maximum $F_{\Delta H}$ of the core for any time in cycle following a misload and the maximum $F_{\Delta H}$ of the as-designed core for any time in cycle."*

NuScale agrees. The misload augmentation factor for $F_{\Delta h}$ used as input to the subchannel analysis was increased to 1.15. The factor is determined based on output of the nuclear analysis.

> *"TR-0915-17564-P states: "<u>The subchannel analysis evaluates the maximum allowed augmentation factor that results in MCHFR at the CHF analysis limit</u>." The TR further states: "<u>The key result of the misload subchannel analysis is the maximum allowed radial peaking augmentation factor. Each cycle-specific nuclear analysis confirms that the maximum augmentation factor calculated is not violated.</u>"* {{ }}2(a),(c)

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}}2(a),(c)

The NRC statements reflect a misunderstanding of the quoted statements from TR-0915-17564-P-A, Revision 2, “Subchannel Analysis Methodology.” TR-0915-17564-P-A Section 4.7.4, which includes the quoted statements above, also includes the statement that “The spectrum of potential misload scenarios for the core and corresponding **augmentation factor is determined by nuclear analysis**” (emphasis added). Additionally, Section 4.7.4 provides the equation for calculating the misload augmentation factor with the nuclear analysis (i.e., not the subchannel analysis). The misload augmentation factor is then used as an input to the subchannel analyses that determine minimum critical heat flux ratio (MCHFR).

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}}2(a),(c) TR-0915-17564-P-A clearly identifies that the misload augmentation factor is an output of the nuclear analysis and an input for the subchannel analysis. There has been no change in the use of the TR-0915-17564-P-A methodology with respect to the misload augmentation factor from the previously approved Design Certification Application (DCA). Changes to TR-0915-17564-P-A to provide further clarification of the quoted statements are not necessary because the topical report is approved by the NRC and has been used consistently by NuScale in both the DCA for the US600 and the Standard Design Approval Application (SDAA) for the US460.

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}}2(a),(c)

NuScale agrees. The misload augmentation factor is an input to the subchannel analysis. The misload augmentation factor is determined based on output of the nuclear analysis. The nuclear analysis determines the misload augmentation factor for each specific core design. {{

}}2(a),(c) The assumption is validated by the core-specific nuclear analysis results.

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}}2(a),(c)

NuScale agrees. {{

}}2(a),(c) the misload augmentation factor is due to the revised nuclear analysis using a detection threshold of ±20 percent instead of ±6 percent based on the NRC audit questions.

> *"Please clarify how the $F_{\Delta H}$ augmentation factor is determined and used and revise the FSAR to clarify any inconsistency."*

The misload augmentation factor is determined and used as described in the NuScale responses above. The current discussion of the misload augmentation factor in FSAR Section 15.4.7.3.2, as revised in the markup with the July 12 supplemental response, states that "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 cross-core misload. This limiting misload power peaking augmentation factor is bounded by the analysis value of 1.15." The FSAR Section 15.4.7.3.2 discussion is accurate and consistent with the information in this response: the value of 1.15 is bounding of the limiting results from the nuclear analysis and is used as an input to the subchannel analysis. No FSAR revisions are required because there are no inconsistencies between the FSAR and the supporting analyses.

> *"Please also provide the acceptance criterion, together with the technical basis, for selecting the $F_{\Delta H}$ augmentation factor and explain why the selected $F_{\Delta H}$ augmentation factor of 1.15 is appropriate and conservative."*

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}}2(a),(c) This demonstrates that the value of the misload augmentation factor of 1.15 used in the subchannel analysis, and identified in FSAR Section 15.4.7.3.2, is conservative with respect to the nuclear analysis. The value of misload augmentation factor of 1.15 results in an MCHFR of 1.53, demonstrating adequate margin to the limit of 1.43. This demonstrates that the value of the misload augmentation factor of 1.15 used in the subchannel analysis is not overly conservative. Therefore, the value of 1.15 is technically justified and its use provides acceptable and conservative results.

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}}2(a),(c)

There are no deviations from the methodology described in the revised response to audit question A-15.4.7-1 (identified in SDAA Audit Chapter 15 eRR as "A-15.04.07-01 Audit

Response R01”). There are no deviations from the methodology described in Sections 6.3.9 and A.8 of ER-A021-3589, Revision 4, “Nuclear Analysis Methodology.”

> *“Also, please provide for NRC staff audit the assemblywise and pinwise power distribution that shows the maximum power differences between the designed and actual power distributions for the “undetected” misload resulting in the highest $F_{\Delta H}$ augmentation factor.”*

{{

}}2(a),(c)

> *“In addition, the staff notes that the change in peak linear heat generation rate shown in the markup to Table 15.4-18 appears smaller than would be expected from the change in $F_{\Delta H}$ augmentation factor. Please provide an updated revision of EC-115225, “Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position” or other calculation documents for subchannel analyses of misloads that are not detectable with the 20% uncertainty.”*

EC-115225, Revision 3, “Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position,” is provided in the eRR. Revision 3 of EC-115225 is the version that supports the changes indicated in the markup associated with the July 12 supplemental response. Note that Revision 2 of EC-115225 was the version previously provided in the eRR. The following information in Revision 3 of EC-115225 is relevant to this response:

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}}2(a),(c)

In preparing the supplemental response above, it was identified that the markup previously provided with the July 12 supplemental response was missing one additional change. Specifically, the MCHFR limit of 1.45 in Table 15.4-18 should be 1.43 as indicated in EC-115225, Revision 3, Table 5-2. The MCHFR limit change is not related to this audit question. The development of the MCHFR limit is still consistent with the NRC approved methodology. This change to the limit applies to other Chapter 15 events and will be incorporated in other FSAR sections in future changes. An updated markup that revises the MCHFR limit, but is otherwise identical to the markup from the July 12 supplemental response, is provided with this supplemental response.

Markups of the affected changes, as described in the response, are provided below:

(ITC) and comparison to the predicted value. Agreement means the response of the core to temperature changes is consistent with the design. Power distributions are confirmed by measuring the neutron flux throughout the core at low, intermediate, and higher power levels and comparing the measurements to design predictions. The power distribution at lower power levels must be confirmed before increasing to higher power levels. Control rod worth measurements confirm the capability of the core to be shut down, and the shutdown requirement is confirmed by measuring the power defect (reactivity difference between zero power and full power).

Additional detail on startup physics testing is provided in Reference 4.3-2.

4.3.2.2.9 Monitoring

The ICIS continuously monitors core neutron flux distribution and core inlet and outlet temperatures. The core neutron flux information is provided to the module control system to ensure power distribution is in agreement with predictions. The core inlet and outlet temperature information is provided to the module protection system to ensure adequate core cooling is being provided for post-accident conditions.

The ICIS is used to synthesize core-wide three-dimensional power distributions. These power distributions are compared to predicted core power distributions to verify the core is operating as designed. Axial power distributions are continuously monitored to validate the AO operating window and actions required by the technical specifications are initiated based on this information. Power distributions from the ICIS are used to calibrate the ex-core neutron flux detectors. When the rod position indication system is not working properly, the ICIS has the capability to determine the relative position of a stuck or misaligned control rod.

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During startup, the core inlet and core exit thermocouples are calibrated against the reactor coolant system narrow- and wide-range temperature measurements. The synthesized power distribution is compared against analytical predictions to verify proper fuel loading, calibrate the ex-core detectors, measure core peaking factors, and confirm core behavior. A fuel assembly misload with a synthesized power 20 percent or more different than its expected power is required to be detected by the ICIS, including consideration of uncertainties. This capability of the ICIS and core monitoring system to identify fuel assembly misloads based on the ±20 percent detection threshold is assumed in the analysis in Section 15.4.7.

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During normal power operation, the ICIS is used to measure and synthesize the power distribution to ensure peaking factors are within limits.

predicted and radionuclide barriers maintain integrity during a decrease in boron concentration event. The results for the non-power modes of operation show that shutdown margin is maintained for a decrease in boron concentration event. The normal leakage related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.4.6.5 Conclusions

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The NPM conditions after the decrease in boron concentration cases during Mode 1 operation are bounded by the uncontrolled CRA withdrawal from a subcritical or low power startup condition and uncontrolled CRA withdrawal at power analyses presented in Section 15.4.1 and Section 15.4.2, respectively. Shutdown margin is also maintained for Mode 1 cases. The results for the non-power modes of operation (Modes 2 and 3) show that shutdown margin is maintained for a decrease in boron concentration event. The results of pool dilution during Mode 5 operation demonstrate there is no potential for a boron dilution of the RCS resulting in a loss of shutdown margin. Based on these results, it is concluded that the SRP 15.4.6 acceptance criteria are met.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

An inadvertent loading and operation of a fuel assembly in an improper position is an event that could affect the power distribution and power peaking of the reactor core. If undetected, a fuel assembly loading error could lead to a reduced CHF ratio and reduced margin to fuel centerline melt.

Fuel loading controls are established to prevent a fuel assembly loading error. The fuel loading operation is conducted in accordance with detailed approved procedures. The fuel loading safety measures and procedures are discussed in Section 14.2.

An inadvertent loading and operation of a fuel assembly in an improper position is classified as an infrequent event.

15.4.7.2 Sequence of Events and Systems Operation

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The core monitoring system detects a fuel loading error if it causes a relative power shape deviation larger than a detection threshold. The overpower fraction detection threshold is 1.20~~6~~ and the underpower fraction detection threshold is 0.80~~94~~. These fractions mean that if an assembly is ~~6~~20 percent above its predicted power or ~~6~~20 percent below its predicted power, it is detected by the core monitoring system. ~~Some~~The in-core instrumentation system, described in Section 4.3.2 and Section 7.0.4, supports the assumed detection thresholds, including consideration of uncertainties. Because fuel assembly misloads resulting in powers outside the detection threshold are assumed to be detected and corrected prior to full power operation, subchannel analyses of detectable fuel assembly misloads are not performed. Undetectable fuel assembly misloads (i.e., those that result in powers not outside the detection threshold) are further analyzed. Note that some assembly misloads can be detected before these power fraction detection thresholds are reached although this is not credited in the analyses. The entire spectrum of potential shuffle and rotational fuel assembly misloads are considered.

Shuffle Misloads

The spectrum of potential fuel assembly misloads for the reactor core is examined to assess the impacts of an undetectable fuel assembly misload being present during normal operations. Figure 15.4-17 shows the full spectrum of fuel misloads considered.

The fuel assembly misloads are evaluated in the three categories shown in Figure 15.4-17: 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. For the three categories, there are 231 potential misloads that are analyzed.

The center assembly is a fresh assembly or previously irradiated in the center location. 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 are consistent with those performed in a single quadrant.

Rotational Misloads

Section 4.2 describes the design features that prevent fuel assembly rotational misloads. Nevertheless, 180 degree rotational misloads are examined.

15.4.7.3 Thermal Hydraulic and Subchannel Analyses

15.4.7.3.1 Evaluation Model

There is no NPM thermal hydraulic analysis required for the fuel assembly misloads given the steady-state nature of the NPM thermal hydraulic conditions of the events. SIMULATE5 is used to determine the limiting undetectable fuel misload, and to provide peaking factors to the subchannel analysis. A discussion of SIMULATE5 is provided in Section 4.3.

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. Section 15.0.2 includes 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 reactor pressure vessel (RPV). The power distribution of the equilibrium core analysis is discussed in Section 4.3.

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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 cross-core misload. This limiting misload power peaking augmentation factor is bounded by the analysis value of 1.15. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1 and Reference 15.4-2.

15.4.7.3.3 Results

The limiting undetectable fuel misloading event results in an MCHFR that is above the CHF analysis limit. Fuel temperature margin to centerline melt is calculated for the worst case fuel assembly misloading event. The calculated value of LHGR for the worst misload is below the limiting LHGR. These results are provided in Table 15.4-18. 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

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Table 15.4-18: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.43~~5~~	1.53~~61~~
Peak LHGR	14.2 kW/ft	11.7~~5~~ kW/ft