February 20, 2025 Docket No. 52-050

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 001 (RAI-10502 R1) on the NuScale Standard Design Approval Application

REFERENCE: NRC Letter to NuScale, “Request for Additional Information No. 001 (RAI-10502 R1),” dated February 13, 2025

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) responses to the referenced NRC Request for Additional Information (RAI).

The enclosures to this letter contain the NuScale responses to the following RAI questions from NRC RAI-10502 R1:

- FSR.LTR-1
- FSR.LTR-28
- FSR.LTR-41
- FSR.LTR-48

Enclosures 2, 4, and 6 are the proprietary version of the NuScale Responses to NRC RAI No. 001 (RAI-10502 R1, Questions FSR.LTR-28, FSR.LTR-41, and FSR.LTR-48). NuScale requests that the proprietary versions be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavits (Enclosures 8 and 9) support this request. Enclosure 8 pertains to the NuScale proprietary information, denoted by double braces (i.e.,“{{ }}”). Enclosure 9 pertains to the Framatome Inc. proprietary information, denoted by brackets (i.e., “[ ]”).

Enclosure 1, 3, 5, and 7 are the nonproprietary version of the NuScale responses.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Kris Cummings at 240-833-3003 or at kcummings@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 20, 2025.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
River Rohrman, Project Manager, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-1, Nonproprietary Version
Enclosure 2: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-28, Proprietary Version
Enclosure 3: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-28, Nonproprietary Version
Enclosure 4: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-41, Proprietary Version
Enclosure 5: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-41, Nonproprietary Version
Enclosure 6: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-48, Proprietary Version
Enclosure 7: NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-48, Nonproprietary Version
Enclosure 8: Affidavit of Mark W. Shaver, AF-179615
Enclosure 9: Affidavit of Morris Byram, Framatome Inc.

Enclosure 1:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-1, Nonproprietary Version

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10502
Date of RAI Issue: 02/13/2025

NRC Question No.: FSR.LTR-1

Regulatory Basis

• 10 CFR 50.68(b) provides the requirements that are necessary to prevent criticality accidents in fuel storage in lieu of a monitoring system.
• General Design Criterion (GDC) 62 requires criticality in the fuel storage and handling system to be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Issue Description

The applicant's criticality safety analysis in "NuScale US460 Fuel Storage Rack Design Topical Report," TR-145417-P, Revision 0, used TRITON/ORIGEN-S as the depletion code to estimate the post irradiated isotopic content of the fuel for utilization of burnup credit. To validate the depletion code, the applicant used 5% of the reactivity change from fresh unpoisoned fuel to the burnup of interest as the uncertainty, citing NEI 12-16, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants," Revision 4 (ML19269E069), Section 4.2.3 as justification.

NRC Regulatory Guide 1.240, "Fresh and Spent Fuel Pool Criticality Analyses," (ML20356A127) endorses, with limitations and conditions, NEI 12-16. Section 4.2.3 of NEI 12-16 states, in part, "In lieu of a formal lattice depletion validation, the licensee may apply an uncertainty equal to 5% of the reactivity decrement, if the licensee uses the lattice depletion code in a manner that is consistent with nuclear design calculations previously performed for commercial power reactor licensing. This ensures that the depletion code will produce reliable and predictable results for the intended application." The information provided by the applicant does not appear to demonstrate its use of TRITON/ORIGEN-S as the depletion code consistent with nuclear design calculations previously performed for commercial power reactor licensing. Typically, this is done by using an established code in commercial power reactor licensing or via an NRC approved topical report that establishes that the code has been benchmarked to appropriate data and produces reliable and predictable results when used with the established methodology within the prescribed limitations and conditions. There is no NRC approved topical report that establishes TRITON/ORIGIN-S as an approved code for use nor is the code in active use for commercial power reactor licensing.

Information Requested

NuScale is requested to provide equivalent justification for using the uncertainty equal to 5% of the reactivity decrement for its depletion uncertainty. Otherwise, NuScale can validate TRITON/ORIGEN-S in an alternate manner such as those described in NUREG-2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facility – Final Report,” (ML20121A190) and NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material: Final Report,” (ML20234A651).

NuScale Response:

Executive Summary

The topical report TR-145417, “NuScale US460 Fuel Storage Rack Design Topical Report,” uses the 5% of reactivity decrement to account for the uncertainty associated with depletion in accordance with Regulatory Guide 1.240 (Regulatory Position 1.e), which endorses the guidance in Nuclear Energy Institute document NEI 12-16, Revision 4 (Section 4.2.3). The 5% depletion uncertainty technical basis was established by the NRC-approved topical reports from the Electric Power Research Institute (EPRI):

- EPRI Topical Report 3002010613, “Benchmarks for Qualifying Fuel Reactivity Depletion Uncertainty — Revision 1”
- EPRI Topical Report 3002010614, “Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation — Revision 2”

The approved approach allows for the use of the 5% depletion uncertainty with “no additional work” required. - EPRI Presentation at ACRS meeting on March 3, 2021 (ML21078A131)

The SCALE code system, including TRITON, has been used extensively by both the NRC and the nuclear industry. The SCALE code system is used for a wide range of applications to determine spent fuel isotopic inventory, spent fuel decay heat, and source terms. SCALE has been validated against experimental data from radiochemical assay data, decay heat measurements, and core design data.

The 5% reactivity decrement for depletion uncertainty is justified for use because it is an NRC-approved approach, it has been accepted on past spent fuel pool criticality applications, and SCALE/TRITION has been used extensively and been found acceptable on numerous applications. The past precedent and technical basis for acceptability is provided in the detailed response.

Precedential Use of SCALE/TRITON for Spent Fuel Depletion in Criticality Analyses

The use of SCALE/TRITON as a depletion code for the calculation of the spent fuel isotopic inventory is consistent with past approvals that did not require either 1) the use of an established code in commercial power reactor licensing, or 2) an NRC-approved topical report. Table 1 lists examples of the acceptability of the use of the 5% depletion uncertainty with SCALE/TRITON, along with pertinent excerpts from the NRC Safety Evaluation Reports (SERs) for these approvals.

SCALE/TRITON – Validation Examples and Guidance

SCALE/TRITON has been validated against a large range of experiments. Table 2 lists applicable documents demonstrating that TRITON provides excellent agreement to test data. Additionally, Table 2 includes a reference to the TRITON primer, which serves as guidance to users of SCALE/TRITON.

Prior to the issuance of Regulatory Guide 1.240, multiple applicants were requested to perform validation of SCALE/TRITON as part of the approval of the application. These requests significantly extended the review time, but ultimately confirmed the conservative approach associated with the 5% depletion uncertainty. A subset of the approved applications in Table 1 included a validation of the depletion code or comparison to an alternate depletion code that was approved by the NRC.

SCALE/TRITON – General Use, Acceptability and Guidance

SCALE has the ability to generate source terms for use in radiological shielding applications, generation of decay heat, and spent fuel isotopic inventories for radiological release calculation. The use of SCALE in these applications have been found acceptable numerous times over the past 20 years. Table 3 provides a sampling of more recent approvals of the use of SCALE/TRITON and of the NRC general acceptance of the use of SCALE for these applications.

Technical Basis for 5% of Reactivity Decrement for Depletion Uncertainty – EPRI Topical Report

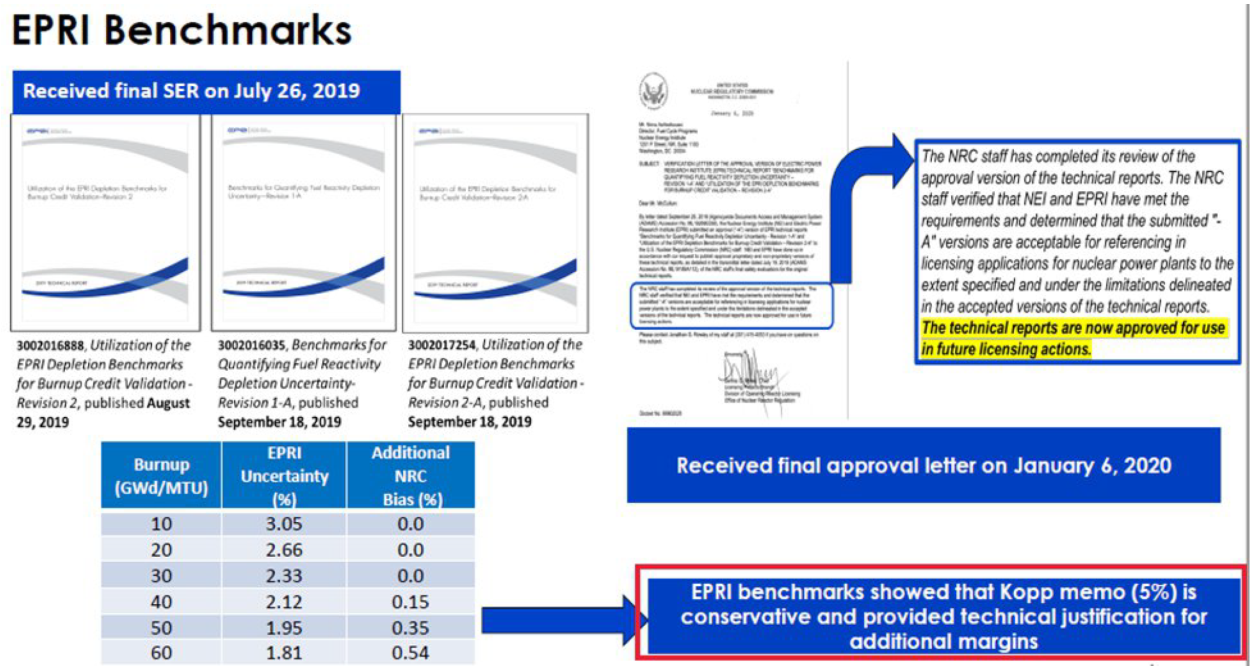
The 5% depletion uncertainty was first introduced in 1998 via an NRC memo from Senior Reactor Engineer Larry Kopp to Branch Chief Timothy Collins (i.e., the “Kopp memo”). The guidance is illustrative in its simplicity:

> "In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity to the burnup of interest is an acceptable assumption."

In 2010, the NRC updated guidance for spent fuel pool criticality analyses in DSS-ISG-2010-01. The depletion uncertainty as described in the Kopp memo was reiterated verbatim in this guidance document with additional clarification:

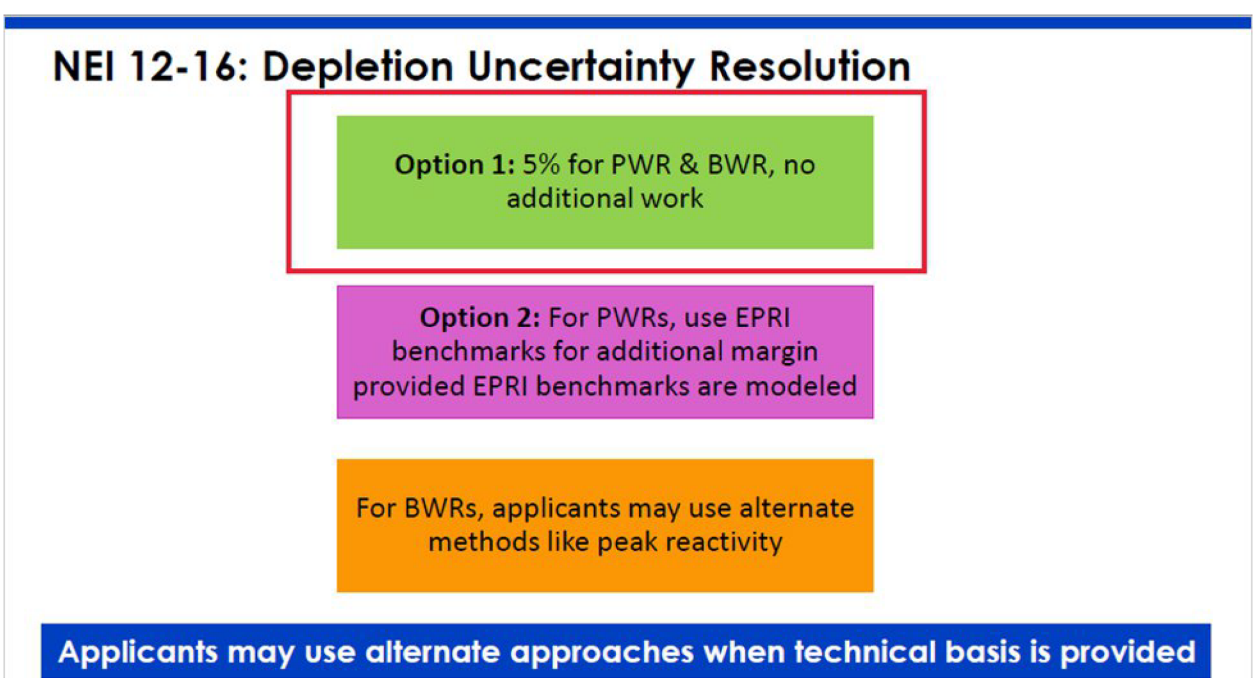
> "The staff should use the Kopp memorandum as follows:
>
> i. 'Depletion uncertainty' as cited in the Kopp memorandum should only be construed as covering the uncertainty in the isotopic number densities generated during the depletion simulations.
>
> ii. The 'reactivity decrement' should be the decrement associated with the k_{eff} of a fresh unburned fuel assembly that has no integral burnable neutron absorbers, to the k_{eff} of the fuel assembly with the burnup of interest with or without residual integral burnable neutron absorbers, whichever results in the larger reactivity decrement."

While the intent of the updated guidance was to provide additional regulatory certainty to applicants, the opposite occurred. The nuclear industry experienced a lack of regulatory certainty, in that new spent fuel pool criticality analyses were subjected to an increasing number of Requests for Additional Information, a ratcheting of requirements, and longer review times. In response, the industry, through the Nuclear Energy Institute, proposed the development of new guidance that would be more comprehensive than DSS-ISG-2010-0001 to perform BWR and PWR spent fuel pool criticality analyses. The effort to develop NEI 12-16 and receive approval was conducted over 9 years (2012 to 2021) and led to endorsement by the NRC in Regulatory Guide 1.240. Additionally, during the development of NEI 12-16, the NRC questioned the technical basis of the 5% depletion uncertainty. The Electric Power Research Institute developed a methodology to use flux measurements from critical reactor states and equate the uncertainty of those measurements to an individual fuel assembly's reactivity uncertainty due to depletion. The EPRI topical report demonstrates that the "Kopp memo (5%) is conservative and provided technical justification for additional margins." - EPRI Presentation at ACRS meeting on March 3, 2021 (ML21078A131)



Burnup (GWd/MTU)	EPRI Uncertainty (%)	Additional NRC Bias (%)
10	3.05	0.0
20	2.66	0.0
30	2.33	0.0
40	2.12	0.15
50	1.95	0.35
60	1.81	0.54

EPRI summarizes the final resolution for the depletion uncertainty as three options for applicants, which are included in NEI 12-16. NuScale chose to use Option 1, “5% for PWR & BWR, no additional work.”



Conclusion

NuScale utilizes the 5% depletion uncertainty in accordance with NEI 12-16, Section 4.2.3 and Regulatory Guide 1.240, Regulatory Position 1.e. The SCALE computer code, including TRITON, has been used by the industry and the NRC for decades for calculations to determine spent fuel isotopic inventories, decay heat and source terms. NEI 12-16 was developed to provide regulatory certainty to applicants spent fuel pool criticality analyses review. Performing the depletion calculations in accordance with NEI 12-16, Section 4.2.1 ensures that the isotopic inventories produced by SCALE/TRITION result in a conservative reactivity when used in the criticality code. The EPRI Topical reports provide the technical basis for the conservatism of the 5% reactivity decrement approach to account for depletion code uncertainty, and is acceptable for use “in lieu of a formal lattice depletion [code] validation.”[1]

[1] NEI 12-16, Revision 4, Section 4.2.3

Table 1: Past Use of 5% Depletion Uncertainty

Application Date ML#	Depletion/ Criticality Code, Nuclides	Depletion Code Validation	NRC Approval Context (emphasis added)
Surry Unit 1&2, Issuance of Amendment #314 November 2, 2023 ML23200A262	SCALE 6.2.3 238 group ENDF/B.VII CSAS/TRITON/ ORIGAMI 233 nuclides	Performed validation of depletion and criticality code against EPRI benchmarks	"The calculation depletion uncertainty is determined as a function of burnup and compared against the 5% depletion uncertainty described in the NEI 12-16. **The depletion uncertainty described in the NEI 12-16 has been previously approved by the NRC numerous times. The licensee's calculated depletion uncertainty is similar to the 5% NEI 12-16 depletion uncertainty with slight deviations due to some burnup dependence and application-specific considerations."**
North Anna Unit 1&2, FSAR Revision 56, Chapter 9 September 30, 2020 ML20309A593 North Anna Units 1&2 Issuance of Amendment July 27, 2018 ML18180A197 North Anna Units 1&2 Criticality Safety Evaluation Report November 20, 2016 ML17129A452	SCALE 6.0 T5-depl option of TRITON ENDF/B-VII 251 nuclides	Comparison to CASMO "For example, the licensee demonstrated an acceptable use of the SCALE TRITON depletion sequence for this application."	"The criticality analysis used the SCALE 6.0 computer code package and the ENDF/B-VII cross section library. Criticality calculations are done with the CSAS5 module and the **depletion analysis is done with the t5-depl option of TRITON. Conservative depletion conditions, including RCS temperature, fuel temperature, soluble boron, and burnable absorber loading were used to maximize spent fuel reactivity in the SFP.** TRITON depletion input variables were chosen to ensure proper convergence of the calculated depleted fuel isotopic content. **Uncertainty in the depleted fuel content is covered by the 5% of depletion reactivity worth.** Bounding analysis values include, but are not limited to -fuel density, fuel dimensions, depletion fuel temperature, depletion moderator temperature, depletion cycle average soluble boron, burnable poison loading, and cycle average RCCA insertion history. **If a bounding composite fuel design value or depletion characteristic should be exceeded, then evaluation under the provisions of 10 CFR 50.59 is required to ensure the analysis and loading curves remain bounding. Identified retained margin was included in the analysis for this purpose."**

Table 1: Past Use of 5% Depletion Uncertainty (Continued)

Application Date ML#	Depletion/ Criticality Code, Nuclides	Depletion Code Validation	NRC Approval Context (emphasis added)
Millstone Unit 3, Amendment No. 273, Revised Criticality Analysis for SFP May 28, 2019 ML19126A000 ———————— Submittal Millstone Unit 3, Amendment No 273 May 3, 2018 ML18128A049 ML18340A028 ML19092A332 ML19135A067	SCALE 6.0, T5-DEPL TRITON SCALE CSAS5 KENO V.a 238 group ENDF/B.VII 251 isotopes	"Code validation is consistent with the DSS-ISG-2010 [4] and NUREG/CRs 6698 [12] and 7109 [13]." Comparison against CASMO4, CASMO5. Reference to previous comparisons for Millstone Unit 2 and North Anna. Applicabl e because of similar fuel design type (17x17).	"In a previous application from the licensee, the NRC questioned the applicability of the Kopp depletion uncertainty methodology to the SCALE 6.0 T5-DEPL sequence.[1] **"The licensee provided sufficient information for the NRC staff to accept the use of the Kopp depletion uncertainty methodology to its analysis. In its LAR, the licensee cited prior approvals as precedent and justified its applicability to this request.** **The NRC staff finds that the licensee's analysis provides reasonable assurance that previous guidance regarding the depletion validation is applicable to this Millstone 3 LAR. Consistent with the guidance provided in the Kopp memo, the licensee's analysis has incorporated an uncertainty equal to 5 percent of the reactivity decrement to cover lack of validation of fuel composition calculations. This uncertainty was calculated by the licensee and applied correctly."** [1] NRC-issued License Amendment No. 327 for Millstone, Unit No. 2, dated June 23, 2016 (ADAMS Accession # ML16003A008)

Table 1: Past Use of 5% Depletion Uncertainty (Continued)

Application Date ML#	Depletion/ Criticality Code, Nuclides	Depletion Code Validation	NRC Approval Context (emphasis added)
Indian Point Unit 2 – Issuance of Amendment 290 Sept. 4, 2019 ML19209C966 ——————— NET-28091-003-01, Revision 0 Criticality Safety Analysis for the Indian Point Unit 2 SFP with no Absorber Panel Credit November 28, 2017 ML17354A015	SCALE 6.1.2 t5-depl in TRITON CSAS5 238 group ENDF/B-VII.0 388 nuclides	Comparison of TRITON to CASMO for a limited set of cases	"In response to the NRC staff's RAI regarding this matter, the licensee provided information that compared the SFP Keff results of T5-DEPL TRITON depletion sequence in SCALE 6.1.2 and CASM0-5. **That comparison indicates that, within the confines of this analysis, using TRITON is conservative.** **"The NRC staff evaluated the licensee's use of the T5-DEPL TRITON depletion sequence from SCALE 6.1.2 to perform its depletion step in Section 3.4.1.1 and finds it acceptable.** Because the licensee demonstrated its use of the T5-DEPL TRITON depletion sequence from SCALE 6.1.2 is conservative relative to CASM0-5, which is an approved NRC depletion and reactor operation simulation code and method, the NRC staff considers it acceptable for the licensee to use the Kopp memo depletion uncertainty determination method in the confines of this analysis."
APR1400 Spent and New Fuel Storage Racks Criticality Analysis May 2018 ML18214A561	SCALE 6.1.2 – TRITON ORIGEN-S KENO-V.a 238-group ENDF/B-VII Nuclides: 12 actinides, 16 fission products from ISG-8 (Casks)	No depletion code validation 5% of reactivity decrement used (Kopp)	**"In lieu of formally validating the depletion code sequence, the applicant used the guidance in the Kopp memo, as discussed in Section 9.1.1.4.7 of this report. This approach, described further in Interim Staff Guidance (ISG) DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," is consistent with what the staff documented in several recently issued SERs for SFP criticality analyses and is therefore acceptable."**

Table 2: SCALE/TRITON Validation

Document Date ML#	SCALE/TRITON version	Cross-sections or Nuclides	Use and Context (emphasis added)
NUREG/CR-7303 Validating Actinides and Fission Products for Burnup Credit Criticality Safety Analyses September 2023 ML23254A400	SCALE 6.2.4 - TRITON/ORIGEN	252-group ENDF/B-VII.1 28 burnup credit nuclides	Burnup credit calculations for transportation casks – limited set of f.p. and actinide nuclide set NRC-funded
NUREG/CR-7284, SCALE 6.2 Lattice Physics Performance Assessment" March 2023 ML23076A034	SCALE 6.2 rev19189 (pre-release of 6.2.1) SCALE 6.2.2 SCALE 6.2.4 (App F)	ENDF/B-VII.1 continuous energy and 252 group libraries	Table 6-1 summarizes the results of benchmarking SCALE 6.2 Lattice codes for the 14 test suites. **"Nearly all cases passed the acceptance accuracy criteria, and the majority of cases passed the more stringent target accuracy criteria."** "…[T]he use of the new SCALE 6.2.4 version for the reevaluated test suites **did not lead to a significant change in the statistics shown in this report. The conclusions documented in this report are still valid."**
NUREG/CR-7041,SCALE/TRITON Primer: A Primer for Light Water Reactor Lattice Physics Calculations November 2012 ML12338A215	SCALE 6.1	N/A	Starting point for the reactor engineer who uses SCALE/TRITON for lattice physics

Table 2: SCALE/TRITON Validation (Continued)

Document Date ML#	SCALE/TRITON version	Cross-sections or Nuclides	Use and Context (emphasis added)
Assessment of Core Physics Characteristics of Extended Enrichment and Higher Burnup LWR Fuels using the Polaris/PARCS Two Step Approach – Volume 1: PWR Fuel, ORNL/TM-2022/1831 June 2022 ML23012A122	SCALE 6.2.4, Polaris, ORIGEN	56-group cross-section library based on ENDF/B-VII.1 data	Fuel Depletion LEU and LEU+ cores at various reactor operating conditions. Funded by the Department of Energy Polaris is a more recent transport solver developed for the SCALE code package, but uses ORIGEN for cross-section development and depletion. PARCS is a core simulator.
NUREG/IA-0529 Simulations of the BEAVRS PWR with SCALE and PARCS December 2022 ML22339A240	SCALE 6.1.2 PARCS 3.2 Core Nodal Simulator	ENDF/B-V (44 group) and ENDF/B-VII (238 group)	“Hot Zero Power core physics and Hot Full Power operation simulations for the 1st fuel cycle were performed using the PARCS core simulator.” “In the calculations, the older SCALE 6.1.2 was applied, and it is **recommended to use the newer SCALE 6.2 with TRITON** or the new POLARIS transport solver. It is also customary to use newer PARCS versions.” **“The Hot Full Power fuel cycle results for 238 groups are in very good agreement with the available BEAVRS data.”**

Table 3: Application of SCALE/TRITON in Other Nuclear Applications (Source Term, Decay Heat, Isotopic Source Term)

Document Date ML#	SCALE/TRITON version	Cross-sections or Nuclides	Use and Context (emphasis added)
Final SER for HI-2210161, "Radiological Fuel Qualification Methodology for Dry Storage Systems" September 2023 ML23234A158 ――――――――― HI-2210161, Radiological Fuel Qualification Methodology for Dry Storage Systems," Revision 4 April 13, 2023 ML23104A379	SCALE 6.2.1, TRITON/ORIGAMI	ENDF/B-VII.1	Source term, fuel depletion "The Topical Report states in section 3.1 that **using the modules associated with newer versions of SCALE are acceptable** as long as the newer codes demonstrate that for a small set of BECTs the dose rate results are within 5% of those calculated with the SCALE 6.2.1 system. **The staff found this acceptable based on the rigorous testing procedures developed by ORNL for the SCALE code system.** Updates that cause a significant change from a previous version would not be incorporated unless it was well understood that the change resulted in more accurate physics. In addition, newer versions of the ORIGAMI module are not likely to have an effect on spent fuel nuclides for LWR fuel as updates to the code would likely be made to add features needed for depletion calculations of advanced reactor fuel. Newer nuclear data, such as ENDF/B-VIII, if incorporated, would likely have a larger effect, however, reactor libraries generated with the newer data would not be incorporated until ORNL was able to determine that the reactor libraries and data were appropriate."
Model No BRR Transportation Package, BEA Research Reactor Package SER September 29, 2022 ML22266A278	TRITON in SCALE 6.2.3	252 multi-group – ENDF/B-VII All nuclides available for depletion	NRC Staff use of code for confirmatory calculations of source term Isotopic content upto 75 GWD/MTU NUREG/CR-6701used as guidance

Table 3: Application of SCALE/TRITON in Other Nuclear Applications (Source Term, Decay Heat, Isotopic Source Term) (Continued)

Document Date ML#	SCALE/TRITON version	Cross-sections or Nuclides	Use and Context (emphasis added)
TN Eagle CoC No 9382 SER October 23, 2023 ML23275A040	TRITON and ORIGEN-ARP in SCALE 6.0	NRC: 238-group – ENDF/B-VII Applicant: 252-group ENDF/B-VII	Gamma and neutron source term from fission products - shielding NUREG/CR-6802 – TRITON acceptable for use **“The staff found the use of the TRITON code to generate cross sections and ORIGEN-ARP for calculating depletion decay as acceptable codes.** ORIGEN-ARP is discussed in NUREG/CR- 6802 and has been found acceptable by the staff for this purpose. **TRITON is a newer code and has much more detailed physics and is acceptable to the staff for modeling high burnup fuel.**”
ORNL/SPR-2022/2692 Review of SCALE Validations Applicable to Spent Nuclear Fuel Shielding Calculations December 2022 ML23003A167	SCALE 6.2.4 – TRITON SCALE 6.1 – TRITON	ENDF/B-VII.1 ENDF/B-VII.0	Isotopic benchmarking against radiochemical assay data for spent fuel shielding validation “The nuclide concentration bias and bias uncertainty that are based on validation against RCA measurement data are affected by **uncertainties in the measurement data, uncertainties in the modeling data (e.g., fuel operating history and assembly characteristics), uncertainties associated with nuclear cross-section data**, and intrinsic uncertainties and approximations associated with the computational methods used for simulations” The associated bias uncertainty values are less than **approximately 11% for all nuclides of interest** except for 106Ru, 241Am, and 246Cm. For these nuclides, the bias uncertainty values vary from 20% to approximately 26%.

Table 3: Application of SCALE/TRITON in Other Nuclear Applications (Source Term, Decay Heat, Isotopic Source Term) (Continued)

Document Date ML#	SCALE/TRITON version	Cross-sections or Nuclides	Use and Context (emphasis added)
Model No. TN-LC, CoC No. 9358, Rev. 6 SER May 20, 2022 ML22131A322	SCALE 6.1.3 - STARBUCS - TRITON	238 group ENDF/B-VII.0	**“The codes and cross section data used by the applicant for the criticality analysis of the TN-LC package are standards in the industry and are acceptable to staff for evaluating the package.”**
Susquehanna Unit 1&2 Issuance of Amendment 276 and 258 Modification of Design Basis LOCA Analysis October 8, 2020 ML20199G749	SCALE 6.2.3 TRITON/ORIGEN-ARP	Not specified	**“Consistent with RG 1.183, Regulatory Position 3.1, the proposed ATRIUM 11 core source term was developed using TRITON/ORIGEN-ARP from SCALE 6.2.3**.” “The NRC staff finds that the use of the updated version of the ORIGEN code is acceptable.”

Enclosure 2:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-28, Proprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 3:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-28, Nonproprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10502
Date of RAI Issue: 02/13/2025

NRC Question No.: FSR.LTR-28

Regulatory Basis

• 10 CFR 50.68(b) provides the requirements that are necessary to prevent criticality accidents in fuel storage in lieu of a monitoring system.
• General Design Criterion (GDC) 62 requires criticality in the fuel storage and handling system to be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Issue Description

NuScale US460 Fuel Storage Rack Design Topical Report, TR-145417-P, Revision 0, Section 6.3.4, "KENO-V.a Criticality Model" discusses the axial burnup profiles used to represent irradiated fuel in the spent fuel pool (SFP) model. Section 6.3.4 states, in part, "…For all assemblies the burnup distribution is normalized and grouped by average exposure at intervals of 5 GWD/MTU. For each group the minimum normalized value is chosen for the top and bottom 5 nodes. For the middle 15 nodes the average value is chosen. The resulting distribution is not renormalized, resulting in average normalized values below 1.0. …" NRC Regulatory Guide 1.240 endorses, with limitations and conditions, NEI 12-16, Revision 4. During the audit, the applicant indicated that they are following Section 5.1.4 of NEI 12-16; however, the averaging of the middle 15 nodes is not part of the guidance in Section 5.1.4 of NEI 12-16 Revision 4. While the applicant has stated the composite axial burnup shapes were not renormalized, the resulting average burnup in most of the composite axial burnup shapes is essentially 1.0. Therefore, it is not clear that the guidance was followed, since there is little to no margin with respect to the composite shape's average burnup.

Information Requested

NuScale is requested to:
a) Indicate where the values in TR-145417-P, Revision 0, Table 6-6 come from.
b) Indicate how many different profiles were evaluated in each burnup group.
c) Explain how those profiles differ from each other in how they were generated.
d) Indicate whether transition cycles were included in determining the composite axial burnup shapes.
e) Describe the analysis that was performed, including any calculations, to demonstrate the method used in determining that the composite axial burnup shapes were conservative relative to actual axial burnup shapes.

NuScale Response:

The calculations in the whole pool model are performed with both a composite axial burnup distribution and an axially uniform burnup profile, in accordance with Section 5.1.4 of NEI 12-16. Compared with {{ **[**

] }}$^{2(a),(c)}$ Therefore, **[**

]

a) The axial burnup distributions in TR-145417-P, Revision 0, “NuScale US460 Fuel Storage Rack Design Topical Report,” Table 6-6 are developed in CASMO/SIMULATE {{ **[**

] }}$^{2(a),(c)}$ Assemblies are sorted into a burnup group corresponding to the nearest exposure increment of 5 GWD/MTU, resulting in approximately 3,300 total axial shapes. {{ **[**

] }}$^{2(a),(c)}$

b) Following the method defined in a) of this audit response, the assembly average exposures are normally distributed. Out of approximately 3,300 total axial shapes, {{ **[**

] }}$^{2(a),(c)}$

c) The process described in the response to part a) is used to develop a large number of axial distributions. Each axial burnup distribution corresponds to a single assembly in the core at a given burnup. The equilibrium core contains a mix of hot and cold assemblies in accordance with a typical core design configuration.

d) **[**

] However, given the binning of assemblies into exposure steps, the conservative post-processing of shapes described in b), and the constraints of axial buckling on axial power distributions for a given operational domain, **[** **]** Specifically, these distributions are generic in nature, as such they do not require core-specific checks for unique core designs. Only significant changes to operations (such as core load follow) or fuel design (such as different burnable poisons or axially distributed enrichments) require updated axial distributions and further evaluation.

e) The determination of the composite axial burnup distributions used in the criticality analysis followed the method described in Option 2 in Section 5.1.4 of NEI 12-16, Revision 4. Because the composite axial burnup distributions are determined in accordance with the guidance document, no additional calculations are required to demonstrate the conservatism of the composite axial burnup distributions.

Enclosure 4:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-41, Proprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 5:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-41, Nonproprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10502
Date of RAI Issue: 02/13/2025

NRC Question No.: FSR.LTR-41

Regulatory Basis

• 10 CFR 50.68(b) provides the requirements that are necessary to prevent criticality accidents in fuel storage in lieu of a monitoring system.
• General Design Criterion (GDC) 62 requires criticality in the fuel storage and handling system to be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Issue Description

The applicant's criticality safety analysis is contained in TR-145417-P, Revision 0, Section 6.5.3.2, "System Bias for Thermal Expansion in TRITON/ORIGEN-S Depletion," Tables 6-19 and 6-20 have the biases, while Tables 6-20 and 6-22 have the standard deviations. In some cases, {{ **[** **]** }}$^{2(a),(c)}$ Additionally, the {{ **[** **]** }}$^{2(a),(c)}$ in TR-145417-P, Section 6.5.1.1. {{ **[** **]** }}$^{2(a),(c)}$

Information Requested

Explain and justify {{ **[** **]** }}$^{2(a),(c)}$ in TR-145417-P, Revision 0, Section 6.5.3.2.

NuScale Response:

As stated in Section 6.1.2 of the topical report, Criticality Analysis, KENO V.a cases passed the source convergence tests, the standard deviation of k-effective is acceptably low, and k-effective passed the chi-square test for normality. These results provide confidence that the KENO V.a values of k-effective and the standard deviation are converged.

As stated in Section 6.5.1.1, uncertainty factors for pin pitch and clad outside diameter tolerance factors have little variation with burnup. This conclusion is based on perturbed and unperturbed KENO V.a cases that obtained their fuel compositions from the same nominal TRITON/ORIGEN-S cases, which include the effects of thermal expansion. In these cases the perturbation to the pin pitch and clad outside diameter are introduced in the KENO V.a input. Table 6-7 contains the uncertainty factors for these two parameters, **[** **]**

The bias factors for thermal expansion are based on separate TRITON/ORIGEN-S cases. The unperturbed case includes the effect of thermal expansion, and the perturbed case uses cold dimensions and material densities. The KENO V.a perturbed and unperturbed cases have the same cold dimensions and material dimensions while obtaining fuel compositions from the appropriate perturbed or unperturbed TRITON/ORIGEN-S case. Thus, the thermal expansion bias factors only account for the depletion effects of thermal expansion. Consequently, **[** **]** and explains the difference in results from thermal expansion versus pin pitch and cladding outside diameter. The standard deviations shown in Tables 6-20 and 6-22 **[** **]** the standard deviation of the combined uncertainty factors shown in Tables 6-13 and 6-14, providing confidence in the convergence of the individual KENO V.a calculations.

Enclosure 6:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-48, Proprietary Version

Enclosure 7:

NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-48, Nonproprietary Version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket: 052000050

RAI No.: 10502
Date of RAI Issue:

NRC Question No.: FSR.LTR-48

Regulatory Basis

• 10 CFR 50.68(b) provides the requirements that are necessary to prevent criticality accidents in fuel storage in lieu of a monitoring system.
• General Design Criterion (GDC) 62 requires criticality in the fuel storage and handling system to be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Issue Description

The applicant's criticality safety analysis in TR-145417-P, Revision 0, Section 6.5.11, "Multiple Misloaded Fuel Assemblies" {{

}}2(a),(c) During the audit, the applicant cited NEI 12-16, Revision 4 Section 6.3.5 as a reference for its approach. NEI 12-16, Revision 4 Section 6.3.5, states that it is important to have a multi-tier defense-in-depth program to prevent or mitigate a multiple fuel misload scenario. The applicant does not explicitly define the administrative controls that will prevent multiple fuel misload.

Information Requested

Provide an explanation of the administrative controls that would prevent multiple fuel misload. Otherwise, provide an explanation and justification for not modeling multiple fresh fuel assemblies misloaded into Zone 1. The discussion should be detailed and use NEI 12-16, Revision 4 Section 6.3.5 as guidance. Include all requirements that future combined license applicants will have to adopt to be consistent with TR-145417-P, Revision 0.

NuScale Response:

Barrier 1 - Licensee controlled procedures and administrative programs

An applicant that references the NuScale US460 Fuel Storage Rack (FSR) Design Topical Report must incorporate into the plant programs and procedures the acceptable storage locations for fresh and irradiated fuel shown in Figure 6-6 of the topical report, Nominal Fuel Loading Map (Map Option 5), and the rules of fuel assembly placement outlined in Section 6.6.1, Fuel Storage Rack Assembly Map:

- Rule 1: Zone 1 fuel may be placed anywhere, even in Zone 2 locations.
- Rule 2: Zone 2 fuel in the outer rows of an individual FSR that face the pool walls must be separated by at least one storage cell.
- Rule 3: Zone 2 fuel in the outer rows on an individual FSR that faces another FSR:
  - Must be separated by at least two storage cells.
  - Cannot be directly across from a Zone 2 FA in an adjacent FSR.
  - A “knights move” (two spaces in one direction and one space in the perpendicular direction) is allowed at the corner of an FSR.
- Rule 4: At the interface between Rule 2 and Rule 3, Zone 2 fuel may be diagonally adjacent.

Additionally, the licensee will develop a licensee controlled administrative program in accordance with Section 6.3.5.1 of NEI 12-16.

Barrier 2 - Analysis of multiple misload scenarios

The nominal fuel loading map {{ }}[2(a),(c)] For this reason, NuScale does not consider loading multiple fresh fuel {{ }}[2(a),(c)] a credible scenario that needs to be evaluated as described in Section 6.3.5.5 of NEI 12-16, Multiple Misload Analysis. NuScale does model the misloading of multiple fresh fuel assemblies into Zone 1, as depicted in Figure 6-23, Fuel Loading Map for Multiple Misloaded Fuel Assemblies (Map Option 13). This analysis bounds scenarios {{ }}[2(a),(c)]

Barrier 3 - Fuel handling instructions

Training and instruction for fuel handling provides an additional barrier against misplacement of multiple fresh fuel assemblies in adjacent storage locations. The spent fuel pool allowable loading configuration restricts fresh fuel assemblies to the periphery of the spent fuel storage modules. The training must include instructions {{ }}[2(a),(c)] Additionally, the fuel handling instructions must include post-movement verification as described in Section 6.3.5.3 of NEI 12-16.

Barrier 4 - Fuel handling machine digital control system

In addition to defense-in-depth strategies outlined in NEI 12-16, NuScale uses a digital control system design as part of the fuel handling machine (FHM) to minimize the possibility for human error. The fuel handling machine, as described in SDAA Section 9.1.4, Fuel Handling Equipment, includes automatic and semi-automatic modes of operation, which reduce the probability of operator error while operating the FHM. The FHM has an operator control station to allow monitoring of the system while in these modes.

Enclosure 8:

Affidavit of Mark W. Shaver, AF-179615

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the response by which NuScale develops its NuScale Power, LLC Responses to NRC Request for Additional Information (RAI-10502 R1, Questions FSR.LTR-28, FSR.LTR-41, and FSR.LTR-48) on the NuScale Standard Design Approval Application.

NuScale has performed significant research and evaluation to develop a basis for this response and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed responses to NRC Request for Additional Information RAI-10502 R1, Questions FSR.LTR-28, FSR.LTR-41, and FSR.LTR-48. The enclosures contain the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 20, 2025.

Mark W. Shaver

Mark W. Shaver

Enclosure 9:

Affidavit of Morris Byram, Framatome Inc.

A F F I D A V I T

1. My name is Morris Byram. I am Product Manager, Licensing & Regulatory Affairs for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Enclosure 2 entitled "NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-28," and Enclosure 4 entitled "NuScale Response to NRC Request for Additional Information RAI-10502 R1, Question FSR.LTR-41" to the NuScale Power, LLC letter Number LO-179614, with subject "NuScale Power, LLC Response to NRC Request for Additional Information No. 001 (RAI-10502 R1) on the NuScale Standard Design Approval Application," and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) “Trade secrets and commercial or financial information.”

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome’s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraph 6(d) and 6(e) above.

7. In accordance with Framatome’s policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: (2/19/2025)

BYRAM Morris Digitally signed by BYRAM Morris
Date: 2025.02.19 14:48:44 -08'00'

(NAME)

Email: morris.byram@framatome.com
Phone: 434-221-1082