LO-173047

August 26, 2024

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 Submittal of NuScale Technical Report, “Fluence Calculation Methodology and Results,” TR-118976, Revision 1

NuScale Power, LLC (NuScale) hereby submits Revision 1 of the technical report “Fluence Calculation Methodology and Results,” (TR-118976). The purpose of this submittal is to request that the NRC review and approve the methodology used to calculate the neutron fluence for the NuScale Power Module, reactor pressure vessel, and containment vessel as well as the associated results of applying the methodology to support the Final Safety Analysis Report Section 4.3 and Section 5.3 for the US460 standard design.

Enclosure 1 contains the proprietary version of the report entitled “Fluence Calculation Methodology and Results,” TR-118976, Revision 1. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 contains the nonproprietary version of the report.

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

If you have any questions, please contact Jim Osborn at 541-360-0693 or at josborn@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 26, 2024.

Sincerely,

Carrie Fosaaen
Vice President, Regulatory Affairs
NuScale Power, LLC

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

Enclosure 1: "Fluence Calculation Methodology and Results," TR-118976, Revision 1, Proprietary Version
Enclosure 2: "Fluence Calculation Methodology and Results," TR-118976, Revision 1, Nonproprietary Version
Enclosure 3: Affidavit of Carrie Fosaaen, AF-173048

Enclosure 1:

"Fluence Calculation Methodology and Results," TR-118976, Revision 1, 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 2:

"Fluence Calculation Methodology and Results," TR-118976, Revision 1, Nonproprietary Version

Licensing Technical Report

Fluence Calculation Methodology and Results

August 2024

Revision 1

Docket: 52-050

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

Department of Energy Acknowledgment and Disclaimer

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Abstract

This Technical Report provides the methodology developed by NuScale Power, LLC, to calculate the neutron fluence for the NuScale Power Module reactor pressure vessel (RPV) and containment vessel (CNV). Estimations of the bias and uncertainty associated with the fluence calculations, derived from benchmarking and sensitivity studies, are presented along with associated end-of-life fluence predictions for the RPV, CNV, and other locations.

NuScale's fluence methodology uses the Monte Carlo N-Particle Transport Code 6 and is based on the guidance found in Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.”. Alternatives to particular Regulatory Guide 1.190 regulatory positions are described and justified. Measured data from the Vulcain Experimental Nuclear Study 3 pressure vessel simulator benchmark are used to validate the NuScale methodology.

Executive Summary

This report provides the methodology for predicting the end-of-life fluence for the NuScale reactor pressure vessel (RPV) and containment vessel (CNV).

A best-estimate neutron fluence calculation for the Nucale Power Module (NPM) is performed using the Monte Carlo N-Particle Transport Code 6 (MCNP6) version 1.0 based on Nuclear Regulatory Commission Regulatory Guide 1.190. Alternatives to particular Regulatory Guide 1.190 regulatory positions are provided. Biases and uncertainties associated with the MCNP6 best-estimate neutron fluence model are also reported. These biases and uncertainties are established through benchmarking against the Vulcain Experimental Nuclear Study 3 experiment and NPM-specific sensitivity studies associated with key MCNP6 modeling simplifications and inputs.

The peak RPV beltline surface and CNV beltline at ¼-T fluence over a 60-year NPM operating life (assumed 95 percent capacity factor) is calculated and provides acceptable results. Neutron fluence estimates provided in this report are acceptable for supporting Final Safety Analysis Report Section 4.3 and Section 5.3 for the US460 standard design, and meet the regulatory guidance and requirements discussed in this report.

1.0 Introduction

1.1 Purpose

This report describes the methodology used to calculate the neutron fluence for the NuScale Power Module (NPM) reactor pressure vessel (RPV) and containment vessel (CNV). It also provides estimations of biases and uncertainties associated with these fluence calculations, derived from benchmarking and sensitivity studies, along with associated end-of-life fluence predictions for the RPV, CNV, and other locations.

1.2 Scope

This report provides the methodology for predicting the end-of-life fluence for the NuScale RPV and NuScale CNV as well as the associated results of applying the methodology to support the Final Safety Analysis Report (FSAR) Section 4.3 and Section 5.3 for the US460 standard design. The testing program associated with confirming these fluence predictions in the operating plant, the methodology for adjusting best-estimate fluence predictions throughout an NPM's operating life, and the effects on material properties caused by the fluence are outside the scope of this report.

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
CMS	core management software
CNV	containment vessel
LCP	lower core plate
MeV	megaelectron volt
NPM	NuScale Power Module
RG	Regulatory Guide
RPV	reactor pressure vessel
UCP	upper core plate
VENUS-3	Vulcain Experimental Nuclear Study 3

Table 1-2 Definitions

Term	Definition
Fluence	In the context of this report, the term "fluence" is taken to mean the fast neutron fluence, which is the time-integrated flux of neutrons with an energy greater than 1 megaelectron volt (MeV).

2.0 Background

Neutron fluence is known to affect the material properties of RPV materials. The extent of the effect is influenced by the magnitude of the fluence, among other factors.

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 7.1) provides guidance for calculating pressure vessel neutron fluence. NuScale's fluence calculation methodology is based on RG 1.190. Descriptions of, and justifications for, alternatives to portions of RG 1.190 regulatory positions are provided in Appendix C.

The NuScale CNV is in close proximity to the RPV compared to a typical large light water reactor and the same methodology used to calculate RPV fluence is taken to be directly applicable to calculating CNV fluence.

2.1 Regulatory Requirements

The regulatory requirements pertaining to vessel fluence analysis are:

- 10 CFR Part 50 Appendix A, General Design Criterion 14 as it relates to ensuring an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture of the reactor coolant pressure boundary, in part, insofar as it considers calculations of neutron fluence
- General Design Criterion 31 as it relates to ensuring the reactor coolant pressure boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized, in part, insofar as it considers calculations of fluence
- 10 CFR Part 50, Appendix G, as it relates to RPV material fracture toughness requirements, in part, insofar as it considers calculations of neutron fluence
- 10 CFR Part 50, Appendix H, as it relates to RPV material surveillance program requirements, in part, insofar as it considers calculations of neutron fluence
- 10 CFR 50.61 as it relates to fracture toughness criteria for pressurized water reactors relevant to pressurized thermal shock events, in part, insofar as it considers calculations of neutron fluence

The following applicable NRC acceptance criteria are listed for the vessel fluence analysis methodology:

- There is reasonable assurance that the proposed design limits can be met for the expected range of reactor operation, taking into account analysis uncertainties.
- There is reasonable assurance that during normal operation the design limits are not exceeded.
- The acceptance criteria of RG 1.190 (Reference 7.1)
- The acceptance criteria of RG 1.99 (Reference 7.2)

3.0 Analysis

3.1 Approach/Methodology

NuScale's fluence calculation methodology uses Monte Carlo N-Particle Transport Code 6 version 1.0 (MCNP6), which was released in 2013 by Los Alamos National Laboratory and merges MCNP5 and MCNPX functions. The MCNP6 is a general-purpose Monte Carlo method code used for neutron, photon, electron, or coupled neutron/photon/electron transport (Reference 7.5). The code treats an arbitrary three-dimensional configuration of materials in geometric cells. The Monte Carlo method has the advantage of allowing an exact representation of the reactor's three-dimensional geometry. In addition, the Monte Carlo method allows a continuous energy description of the nuclear cross-sections and flux solution.

NuScale calculates three-dimensional exposure and power distribution data for each fuel assembly using core management software (CMS) codes CASMO5 and SIMULATE5. CASMO5 is a lattice physics code that characterizes reactor fuel assembly designs. SIMULATE5 is a three-dimensional core simulator code for core design and core load calculations. Information from CASMO5 and SIMULATE5 is used as inputs to the MCNP6 based fluence calculation.

The variance reduction scheme used in NuScale's fluence calculation methodology is the mesh based weight window produced by Automated Variance Reduction Generator (ADVANTG) software (Reference 7.4), which is developed, maintained, and distributed by Oak Ridge National Laboratory.

3.2 Geometry

Calculations are performed using a three-dimensional MCNP6 model.

An illustration of the vertical cross-sectional view of the lower section of the NPM is shown in Figure 3-1. The vertical cross-sectional view of the MCNP6 NuScale best-estimate fluence model is presented in Figure 3-2 and the horizontal cross-sectional view is presented in Figure 3-3.

The NuScale best-estimate fluence model is representative of the US460 standard NPM design with the following general exceptions and modeling simplifications.

- The geometry is specified using cold dimensions, and thermal expansion is not modeled. Thermal expansion for hot full power dimensions is accounted for in NuScale's Studsvik Scandpower CMS codes (SIMULATE5 and CASMO5), whose outputs are used as inputs to establish the neutron source distribution in the MCNP6 model. The effect of this modeling simplification and the effect of this difference between MCNP6 and CMS treatment of cold dimensions on the fluence estimate is provided in Section B.1.3 and Section B.1.4.

- The NuScale best-estimate fluence model contains an axially homogenized representation of the active fuel region of the fuel assemblies. This modeling simplification is implemented for consistency because fuel assembly power information is taken from NuScale's SIMULATE5 model output, which is a homogenized model. A sensitivity study comparing this homogenized treatment to an MCNP6 model that explicitly models the fuel across {{ }}$^{2(a),(c)}$ is provided in Section B.1.1.
- Each fuel assembly consists of {{ }}$^{2(a),(c)}$. The active fuel pin region consists of a {{ }}$^{2(a),(c)}$. On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is negligible.
- The top nozzle skirt and upper core plate are modeled explicitly as part of the fuel assembly for assemblies that do not contain control rod assemblies. On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is negligible.
- The NuScale best-estimate fluence model accurately represents the NPM reactor pressure vessel and CNV bottom head designs, as can be seen by comparing Figure 3-1 and Figure 3-2.
- The RPV bottom core support block is not explicitly modeled. The RPV beltline region is the main region of interest for the vessel fluence estimation. On the basis of engineering judgment, the impact of this modeling simplification on the RPV beltline region fluence estimates is negligible.
- All water densities in the NuScale best estimate fluence model are {{ }}$^{2(a),(c)}$. The effect of this modeling simplification on the fluence estimate is provided in Section B.1.12.
- All temperatures of components in the NuScale best-estimate fluence model are {{ }}$^{2(a),(c)}$. On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is small relative to the effect of using a single water coolant density for the primary coolant.
- There are existing negligible differences between the calculated time weighted exposure power profiles presented in both Table 3-1 and Figure 3-5, compared with fission neutron generation probabilities entered in MCNP input files. The impact of this modeling differences on the fluence estimates is negligible.

Figure 3-1 Vertical Cross-Sectional View of the Lower Section of the NuScale Power Module

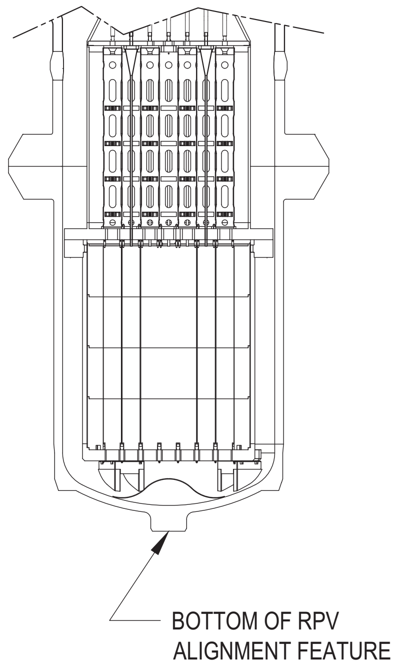

Figure 3-2 Vertical Cross-Sectional View of the Monte Carlo N-Particle Transport Code 6 Fluence Homogenized Model

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}}2(a),(c)

Figure 3-3 Horizontal Cross-Sectional View of the Monte Carlo N-Particle Transport Code 6 Fluence Homogenized Model

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}}2(a),(c)

3.3 Material Compositions

The material composition information used in the MCNP6 NuScale best-estimate fluence model is based on the typical isotopic contents associated with the materials associated with the NPM design. Cold dimensions are used and thermal expansion is not taken into account in the determination of material densities. The effect of this modeling simplification on the fluence estimate is discussed in Section B.1.3 and Section B.1.4.

The core composition of the MCNP6 base model is based on the core composition of the SIMULATE5 base model core design. The NuScale best-estimate fluence model does not contain ^{239}Pu because it is based on a fresh core (beginning of Cycle 1). A bias and uncertainty to account for the contribution of ^{239}Pu buildup to fluence is derived in Section B.1.2.

The material composition of the homogenized active fuel comprises fuel at an averaged 3.5 percent enrichment, fuel cladding, borated water, and guide tubes.

3.4 Cross-Sections

NuScale's MCNP6 based fluence calculation methodology uses the ENDF/B-VII.1 nuclear data for continuous energy cross-section libraries.

A .92c file extension is used to represent isotopic cross-section data with a temperature at {{ }}$^{2(a),(c)}$. The ENDF/B-VII.1 data libraries have cross-sections processed at selected temperatures {{ }}$^{2(a),(c)}$. The MAKXSF code is used to derive the {{ }}$^{2(a),(c)}$ library from {{ }}$^{2(a),(c)}$ and {{ }}$^{2(a),(c)}$ libraries. The {{ }}$^{2(a),(c)}$ file extension is also copied into the new data library and used for pool water at {{ }}$^{2(a),(c)}$, which has a negligible impact to vessel component fluence.

The temperature card "TMP"is used in MCNP6 to provide the time-dependent cell thermal temperatures necessary for the free-gas thermal treatment of low-energy neutron transport at the correct material temperatures. The temperature card "TMP"requires inputs to be in units of megaelectronvolts (MeV), so a conversion is performed. For example, NuScale uses {{ }}$^{2(a),(c)}$ as the averaged temperature of moderator and this temperature in K is converted to MeVs as shown in Equation 3-1.

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

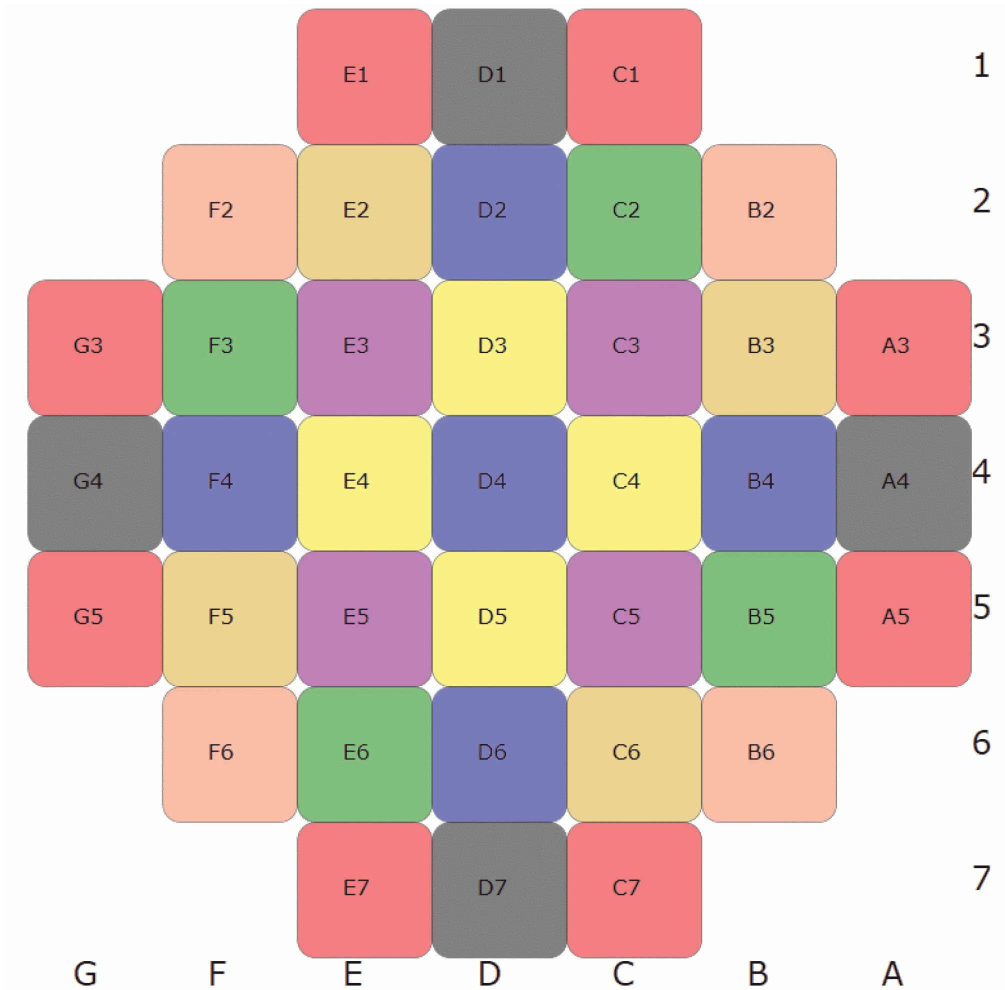
3.5 Neutron Source

For the NuScale best-estimate fluence model, the energy spectrum of the fission neutrons emitted from the fuel assemblies is taken as the Watt fission spectrum for ^{235}U. Sensitivity studies on the effect of ^{239}Pu buildup are presented in Section B.1.2.

There are no delayed neutrons separately modeled because the fission modeling is turned off by using the "NONU" card in MCNP6 input decks for neutron transport. For the purpose of the NuScale best estimate of fast neutron fluence, the delayed neutron contribution to fast neutron fluence is negligible.

For the purposes of this report, the fuel assemblies are referred to according to the naming index shown in Figure 3-4.

Figure 3-4 Fuel Assembly Naming Index



SIMULATE5 is used to calculate the core average axial power profile associated with each cycle in a lifetime refueling scheme for {{ }}[2(a),(c)]. The axial power profiles associated with each cycle are averaged to produce a lifetime exposure averaged axial power profile shown in Table 3-1. Table 3-1 is used to establish the vertical sampling of the neutron source used in the MCNP6 NuScale best-estimate fluence model. SIMULATE5 is used to calculate the assembly averaged radial power profile associated with each cycle in an 8-cycle refueling scheme. The assembly averaged radial power profile associated with each cycle are averaged to produce a lifetime exposure averaged radial power profile shown in Figure 3-5. The radial sampling of the neutron source used in the MCNP6 NuScale best-estimate fluence model is based on Figure 3-5.

Table 3-1 Lifetime Exposure Averaged Core Axial Power Profile

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}}2(a),(c),ECI

Figure 3-5 Lifetime Exposure Averaged Assembly Averaged Radial Power Profile

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}}2(a),(b),(c),ECI

MCNP6 produces flux results that are on a "per source particle" basis and part of converting to final reported results involves establishing the source intensity. The total fission neutron source intensity S (neutrons/second) in the NPM at a given power is determined by Equation 3-2:

$$S = \frac{\upsilon P \times 10^{6}\left(\frac{W}{MW}\right)}{1.602 \times 10^{-13}\left(\frac{J}{MeV}\right) K_{eff} Q_{ave}} \qquad \text{Equation 3-2}$$

where,

υ = Average number of neutrons produced per fission in NPM (neutrons/fission); calculated from results in the MCNP6 output file to be υ=2.46 at initial cycle for a fresh core with 3.5 percent ^{235}U enrichment at hot zero power,

P = Fission power (MW); taken to be 250 MW based on NPM's thermal power rating,

K_{eff} = Effective multiplication factor; taken to be 1.000 for critical light water reactor, and

Q_{ave} = The average recoverable energy per fission for all fissionable materials (MeV/fission); taken to be 198 MeV/fission as a best estimate based on other low enriched uranium systems.

The calculated fission neutron intensity for the NPM is estimated as:

$$S = \frac{2.46\frac{neutrons}{fission} * 250MW \times 10^{6}\left[\frac{W}{MW}\right]}{1.602 \times 10^{-13}\left(\frac{J}{MeV}\right) * 1.000 * 198\frac{MeV}{fission}} = 1.94 \times 10^{19}\frac{\text{neutrons}}{\text{second}} \qquad \text{Equation 3-3}$$

A factor of 1.8 x 10^9 seconds (57 effective full-power years) is then used to convert from flux to fluence based on a 60-year operating life with a 95 percent power capacity factor.

3.6 Other Modeling Considerations

There is no upper limit placed on the neutron source energy, and neutrons are treated with implicit capture in the NuScale best-estimate fluence model. A lower cut off energy of 0.9 MeV is used. Because there are no processes modeled that would result in a higher energy neutron, the implementation of the 0.9 MeV lower cut off energy makes no difference to the >1 MeV neutron fluence results.

A series of cylindrical mesh tallies are used to specify the locations of interest where fluence is calculated throughout the MCNP6 model.

Example illustrations of mesh tallies used in the calculation of RPV and CNV fluence are shown in Figure 3-6 and Figure 3-7, including naming and numbering conventions for the axial and azimuthal segments. The effect of the tally region volume impact on final fluence results is discussed in Section B.1.14.

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}}2(a),(c)

3.7 Variance Reduction Scheme and Convergence

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}}2(a),(c)

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}}2(a),(c)

Table 3-2 {{ }}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

Figure 3-6 Horizontal Cross-Sectional View of the Reactor Pressure Vessel Mesh Tally

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}}2(a),(c)

Figure 3-7 Horizontal Cross-Sectional View of the Containment Vessel Mesh Tally

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}}2(a),(c)

Figure 3-8 Y-Z Plot of the Mesh-Based Weight Window Structure

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}}2(a),(c)

Figure 3-9 Example of X-Y Plot of ADVANTG Generated Mesh-Based Weight Window

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}}2(a),(c)

Figure 3-10 X-Y Plot of the Global Fast Neutron Fluence

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}}2(a),(c),ECI

Figure 3-11 Y-Z Plot of the Global Fast Neutron Fluence

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}}2(a),(c)

Figure 3-12 X-Y Plot of the Global Statistic Check on the Fast Neutron Fluence Relative Error

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}}2(a),(c)

Figure 3-13 Y-Z Plot of the Global Statistic Check on the Fast Neutron Fluence Relative Error

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}}2(a),(c)

4.0 Bias and Uncertainty

4.1 Quantified Biases and Uncertainties

Appendix A describes the NuScale best-estimate fluence prediction benchmarking work. Appendix B describes sensitivity analysis associated with the best-estimate fluence calculation. A summary of the relevant results associated with the NuScale best-estimate fluence bias and uncertainty, and a reference to the applicable report section, are provided in Table 4-1.

Table 4-1 List of Quantified Systematic Biases and Random Uncertainties

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}}2(a),(c)

4.2 Combination of Biases

The analytical bias (also known as B_a^c per RG 1.190) is composed of known uncertainties that are biased in a certain direction compared to the best-estimate fluence calculation. For the NuScale best-estimate fluence calculation, B_a^c is calculated as the algebraic summation of systematic biases presented in Table 4-1, excluding B_b^c, as shown in Equation 4-1.

$$B_a^c = B_{homo} + B_{Pu} + B_{Pin} + B_{ax} \qquad \text{Equation 4-1}$$

A tendency for NuScale's MCNP6 based-fluence calculation methodology to {{

}}2(a),(c).

The total bias (B_T) of the best estimate fluence calculation is quantified as shown in Equation 4-2:

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}}2(a),(c)

4.3 Combination of Uncertainties

Independent random uncertainties have no specific direction associated with them with respect to their effect on the final fluence estimate. The overall uncertainty (σ^c) is established per Equation 4-3 for the NuScale best-estimate fluence MCNP6 model.

$$\sigma^c = \sqrt{\left(\sigma_a^c\right)^2} \qquad \text{Equation 4-3}$$

$$\sigma_a^c = \sqrt{\left(\sigma_b^c\right)^2 + \sigma_{Pin}^2 + \sigma_{Pu}^2 + \sigma_{water}^2 + \sigma_m^2 + \sigma_g^2 + \sigma_{ap}^2 + \sigma_{pa}^2 + \sigma_{pr}^2 + \sigma_{Boron}^2 + \sigma_{tally}^2 + \sigma_{mt}^2} \qquad \text{Equation 4-4}$$

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}}2(a),(c)

Where σ_{mt} is the relative error associated with the particular location's reported result from MCNP6 output and σ_a^c is the square root of the sum of the squares of random uncertainties in Table 4-1, as shown in Equation 4-4.

Substituting the value established for σ_a^c back into Equation 4-4 gives Equation 4-5. Equation 4-5 is used to establish overall uncertainties given in Equation 4-6.
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}}2(a),(c)

A single {{

}}2(a),(c). Section B.1.11 contains more details.

5.0 Results

5.1 NuScale Power Module Fluence Prediction Results

Table 5-1 presents the results of the best estimate fluence analysis. {{

}}2(a),(c) established in Section 4.2, to the "MCNP Calculated Neutron Fluence."

Table 5-1 Best Estimate of Fluence Expected in Various NuScale Power Module Components and Locations

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}}2(a),(c)

Table 5-1 Best Estimate of Fluence Expected in Various NuScale Power Module Components and Locations (Continued)

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}}2(a),(c)

Table 5-1 Best Estimate of Fluence Expected in Various NuScale Power Module Components and Locations (Continued)

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}}2(a),(c)

6.0 Summary and Conclusions

A best-estimate neutron fluence calculation for the NPM is performed using of the MCNP6 code based on RG 1.190. Alternatives to particular RG 1.190 regulatory positions are provided in Appendix C. Biases and uncertainties associated with the MCNP6 best-estimate neutron fluence model are reported in Table 4-1, which are established through benchmarking against the VENUS-3 experiment and NPM-specific sensitivity studies associated with key MCNP6 modeling simplifications and inputs.

The peak RPV beltline surface and CNV beltline at ¼-T fluence over a 60-year NPM operating life (assumed 95 percent capacity factor) is calculated to be
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}}2(a),(c), as reported in Table 5-1.
Neutron fluence estimates provided in this report are acceptable for supporting Final Safety Analysis Report Section 4.3 for the US460 standard design and meet the regulatory guidance and requirements discussed in Section 2.1 of this report.

7.0 References

7.1 U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, Revision 0, March 2001.

7.2 U.S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.

7.3 Los Alamos National Laboratory, Trellure, H.R. and Poston, D.I., "User's Manual, Version 2.0 for Monteburns, Version 5B," LA-UR-99-4999, Los Alamos, NM, September 1999.

7.4 Oak Ridge National Laboratory, “ADVANTG-An Automated Variance Reduction Parameter Generator" ORNL/TM2013/416, Rev. 1, Oak Ridge, TN, August 2015.

7.5 Los Alamos National Laboratory, DB Pelowitz, "Monte Carlo N-Particle Transport Code 6 Users Manual, Version 1.0," LA-CP-13-00634, Rev. 0, Los Alamos, NM, May 2013.

7.6 Oak Ridge National Laboratory, Radiation Safety Information Computational Center, "Shielding Integral Benchmark and Database," DCL-237, SINBAD-2013.12, Oak Ridge, TN, December 2013.

7.7 Organisation for Economic Co-operation and Development, Nuclear Energy Agency, Nuclear Science Committee, "Prediction of Neutron Embrittlement in the Reactor Pressure Vessel: VENUS-1 and VENUS-3 Benchmarks," OECD, 2000.

8.0 Appendices

The following Appendices are included in this report:

- Appendix A - Benchmarking Monte Carlo N-Particle Transport Code 6 for Fluence Applications
- Appendix B - NuScale Power Module Fluence Prediction Sensitivity Studies and Uncertainty Analysis
- Appendix C - Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

Appendix A Benchmarking Monte Carlo N-Particle Transport Code 6 for Fluence Applications

A.1 Vulcain Experimental Nuclear Study 3 Benchmark

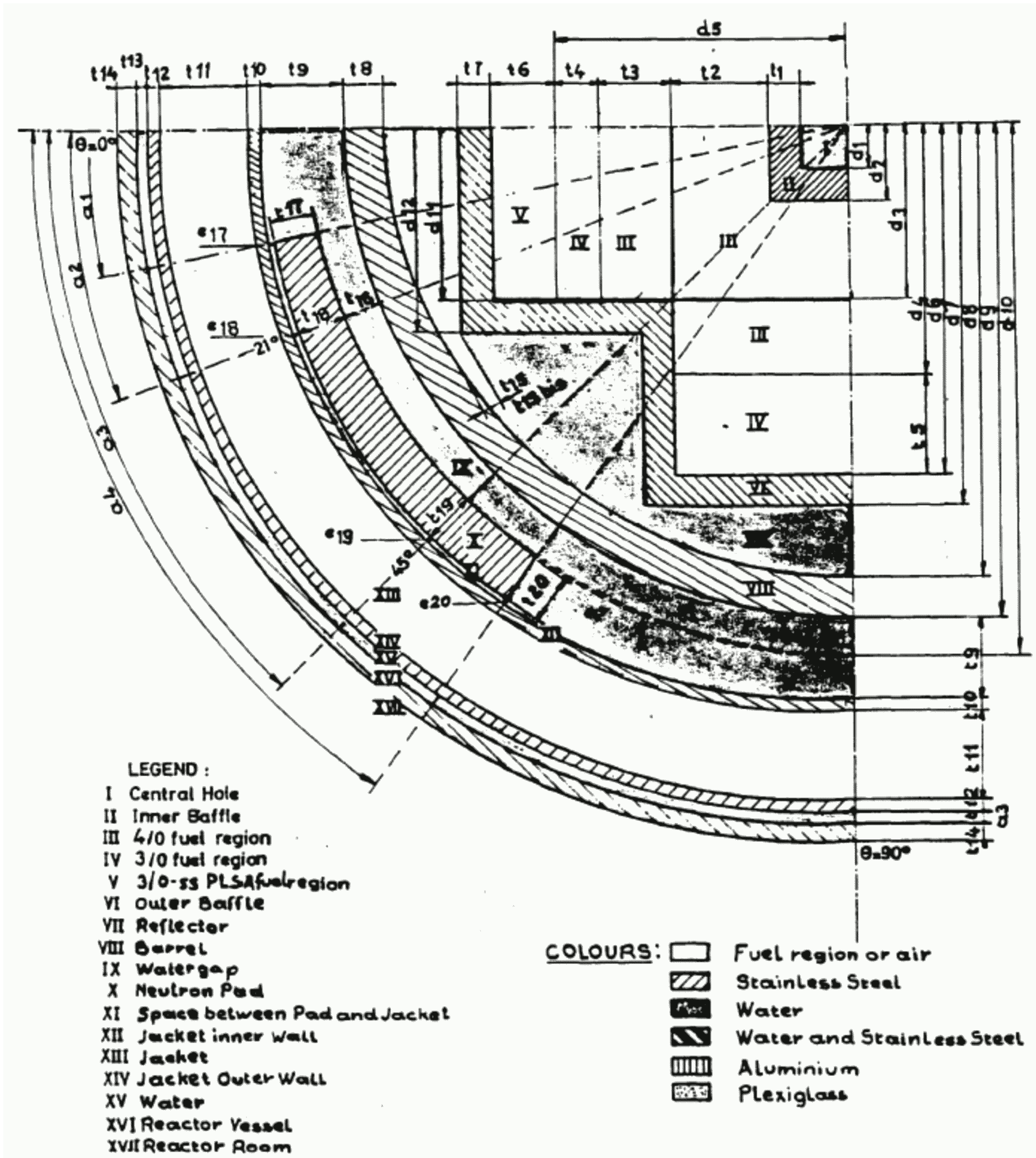
This appendix presents a description of benchmarking work performed to demonstrate that MCNP6 can perform neutron flux determinations that compare favorably with expected or experimental results. The benchmarking work shown in this appendix is also used to establish the bias and uncertainty stemming from use of the MCNP6 transport code and associated cross section data.

A.1.1 Modeling

MCNP6 code version 1.0 is used to create a model of the third configuration in the Vulcain Experimental Nuclear Study, commonly known as “VENUS-3.” The VENUS-3 pressure vessel fluence benchmark is based on documentation from the Shielding Integral Benchmark Archive and Database from the Radiation Safety Information Computational Center (Reference 7.6). The VENUS-3 benchmark provides reaction rates associated with various detector types for the core barrel of an experimental reactor setup. The VENUS-3 benchmark is considered to be generally applicable to the NPM.

The basic configuration of the VENUS-3 benchmark is shown in Figure A-1.

Figure A-1 Horizontal Cross-Sectional View of the Vulcain Experimental Nuclear Study 3 Benchmark Geometry



The MCNP6 model is based on the MCNP model supplied as part of the VENUS-3 benchmark collection in Reference 7.6, which used an earlier version of MCNP. This model is reviewed for correctness and updated as needed for use with the current MCNP version MCNP6.

The ENDF/B-VII.1 libraries associated with 293.6 K (.80c extension) are used for all materials. In addition, a light water S(α,β) library based on the ENDF/B VII.1, lwtr.20t, is used for those materials containing water. The benchmark used a ^{235}U Watt fission spectrum.

Portions of the NuScale MCNP6 model of the VENUS-3 benchmark are shown in Figure A-2 and Figure A-3.

Figure A-2 Vertical Cross-Sectional View of the Monte Carlo N-Particle Transport Code 6 Model of the Vulcain Experimental Nuclear Study 3 Benchmark

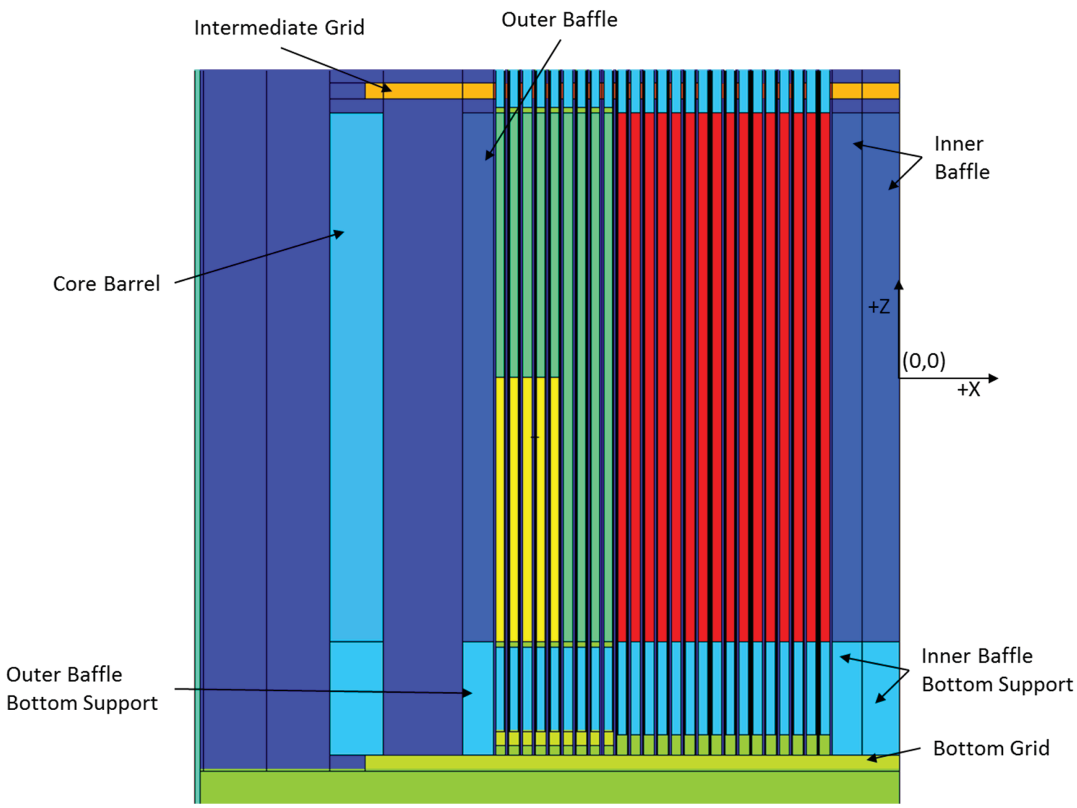
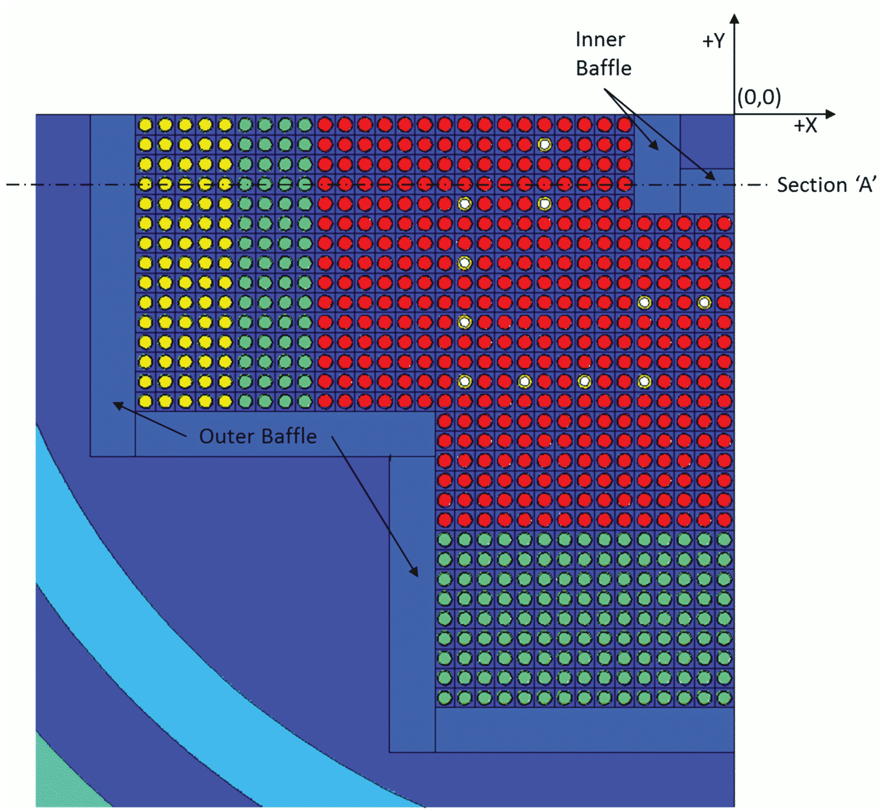


Figure A-3 Horizontal Cross-Sectional View of the Inner and Outer Baffle of the Monte Carlo N-Particle Transport Code 6 Model of the Vulcain Experimental Nuclear Study 3 Benchmark



A variety of experimental results are provided as part of the VENUS-3 collection of data, but the results of specific interest to this benchmark are the results associated with the core barrel only. These results are based on nickel, indium, and aluminum reaction rates ^{58}Ni(n,p), ^{115}In(n,n'), and ^{27}Al(n,), respectively.

Based on the energy thresholds associated with the reaction rates, the ^{115}In(n,n') reaction rates are associated with the neutron flux greater than 1 MeV, the ^{58}Ni(n,p) reaction rates are associated with neutron fluxes greater than 3 MeV, and the ^{27}Al(n,α) reaction rates are associated with neutron fluxes greater than 8 MeV. The relative experimental uncertainties for the reaction rates in the core barrel for the VENUS-3 data are reported to be 9 percent for ^{58}Ni(n,p), 7 percent for ^{115}In(n,n'), and 14 percent for ^{27}Al(n,α) in Section 6.1 of Reference 7.7.

The relative difference between the reported experimental (Exp) values for these reaction rates and the MCNP6 calculated values (Calc) is established for each data point provided in the VENUS-3 benchmark, relative to the experimental value, using

Equation A-1.The average relative difference of experimental versus calculated values and standard deviations are reported in Table A-1.

$$\textit{Relative difference (\%)} = \frac{Exp - Calc}{Exp} \times 100\% \qquad \text{Equation A-1}$$

The ^{115}In(n,n') reaction rate comparisons are judged to provide the best comparison to the overall neutron flux because it has the lowest threshold energy of ~1 MeV. The ^{58}Ni(n,p) and ^{27}Al(n,) reaction rates have higher thresholds, 3 MeV and 8 MeV, respectively. The ^{115}In(n,n') results also have the lowest experimental uncertainty associated with them. Further, the ^{115}In(n,n') results are the only results from the NuScale VENUS-3 benchmark that indicate MCNP6 has a tendency to {{

}}$^{2(a),(c)}$ compared to incorporating the ^{58}Ni(n,p) or ^{27}Al(n,α) based benchmark results.

{{

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

Table A-1 Vulcain Experimental Nuclear Study 3 Experimental Versus Calculated Results

{{

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

{{

}}2(a),(c).

The results of this benchmark demonstrate that MCNP6 can perform neutron flux determinations that compare favorably with expected or experimental results. The results show good agreement between MCNP6 and the benchmark results.

Appendix B NuScale Power Module Fluence Prediction Sensitivity Studies and Uncertainty Analysis

This appendix presents sensitivity studies and an uncertainty analysis associated with the NPM fluence prediction calculations. Appendix B results are combined with Appendix A findings in Section 4.0 of this report in order to properly present results with total uncertainty in Section 5.0 of this report.

B.1 Sensitivity Studies

B.1.1 Homogenized Fuel Model vs Explicit Fuel Model

The best-estimate fluence predictions presented in Table 5-1 are based on a homogenized fuel model. {{

}}2(a),(c).

B.1.2 Contribution of ^{239}Pu to Neutron Source

As discussed in Section 3.3, the MCNP6 NuScale best-estimate fluence model does not contain plutonium because it is based on a fresh core. {{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

B.1.3 Material Composition

The uncertainty in fluence estimates associated with differences between the as built and operating NPM material chemical compositions and densities compared to how these characteristics are modeled in the NuScale best-estimate fluence model is assumed to be {{

}}2(a),(c).

B.1.4 Geometrical Tolerances

The uncertainty in fluence estimates associated with differences between as built and operating NPM dimensions and dimensions modeled in the NuScale best-estimate fluence model is assumed to be {{

}}2(a),(c)

{{

}}2(a),(c).

B.1.5 Assembly Averaged Neutron Source Bias and Uncertainty

The MCNP6 NuScale best-estimate fluence model uses an assembly averaged pin power profile instead of an explicit pin-wise power profile.

{{

}}2(a),(c)

{{

}}2(a),(c)

Table B-1 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G4 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-2 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G5 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-3 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly F6 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-4 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly E7 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-5 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G4 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

Table B-6 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G5 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

Table B-7 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly F6 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

Table B-8 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly E7 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

B.1.6 Core Power

The uncertainty of the core power level is directly proportional to the uncertainty of the fluence estimates. {{

}}2(a),(c).

B.1.7 Radial Power Profile

Uncertainty in the radial power profile is directly proportional to the uncertainty of the fluence estimates. The radial power profile uncertainty (σ_{pr}) is estimated by

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure B-1 Time-Weighted Averages and Weighted Standard Deviations for Radial Power Profile

{{

}}2(a),(c),ECI

B.1.8 Axial Power Profile

A single, time-averaged axial profile is utilized in the MCNP6 NuScale best-estimate fluence model. Variations in the axial power profile could impact fluence estimates.
{{

}}2(a),(c)

{{

}}2(a),(c)

Table B-9 Average Axial Power Profiles

{{

}}2(a),(c),ECI

Table B-10 Variance and Weighted Standard Deviation for the Axial Power Profiles

{{

}}2(a),(c),ECI

B.1.9 Boron Concentration

The best estimate fluence prediction MCNP6 model assumed a boron concentration of {{

}}2(a),(c).

The concentration of soluble boron in the primary coolant varies over the course of the fuel cycle, in a range {{

}}2(a),(c)

B.1.10 Nuclear Cross-Section Data and Transport Code

There is uncertainty associated with the various cross sections taken from the ENDF/B-VII.1 nuclear data library and there is uncertainty associated with the use of the transport code MCNP6. {{

}}2(a),(c)

B.1.11 Monte Carlo Method

In Monte Carlo analysis, a calculational uncertainty (σ_{mt}) is introduced as a result of the finite number of particle histories sampled. The relative error (standard deviation/mean) associated with the MCNP6 results is taken to account for this uncertainty. {{

}}2(a),(c)

B.1.12 Water Density

{{

}}2(a),(c)

B.1.13 Axial Coolant Density Bias

The coolant in the MCNP6 NuScale best-estimate fluence model is modeled as {{

}}2(a),(c)

Table B-11 Coolant Water Axial Density Variations

{{

}}2(a),(c)

Table B-12 Peak Fluence Results for Axially Varied Coolant Density

{{

}}2(a),(c)

B.1.14 Tally Mesh Size

This section presents the results of the determination of the tally subdivision size uncertainty, σ_{tally}.

{{

}}2(a),(c)

Table B-13 Tally Subdivision Size Uncertainty

{{

}}2(a),(c)

Appendix C Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

RG 1.190 (Reference 7.1) provides guidance for calculating pressure vessel neutron fluence. The NuScale fluence calculation methodology described in this report used some alternative approaches to those recommended in RG 1.190. This appendix describes and justifies these alternatives in Table C-1.

The descriptions in Table C-1 are summaries or excerpts of specific portions of regulatory positions in RG 1.190.

Table C-1 Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

RG 1.190 Regulatory Position	Description of Regulatory Position	Description of Alternative and Justification
1.1.1	Regional temperatures should be included in the input data.	All materials in the NuScale best-estimate fluence model are taken to be at {{ }}2(a),(c). The effect of the latter is accounted for in Section B.1.13.
1.1.1 and 1.4.1	In the absence of plant-specific information, conservative estimates of the variations in the material compositions and dimensions should be made and accounted for in the determination of the fluence uncertainty.	Uncertainty between the “as built and operating” and “as modeled” design is accounted for {{ }}2(a),(c) estimates as discussed in Section B.1.3 and Section B.1.4.
1.1.1	The input data should account for axial and radial variations in water density.	{{ }}2(a),(c) The effect of this modeling simplification is accounted for in Section B.1.13.
1.2	The peripheral assemblies, which contribute the most to the vessel fluence, have strong radial power gradients, and these gradients should not be neglected. Peripheral assembly pin-wise neutron source distributions obtained from core depletion calculations should be used.	Assembly-averaged power profiles obtained from core depletion calculations are used in the MCNP6 NuScale best-estimate fluence model. A sensitivity study to establish the effect of this modeling simplification on the NuScale fluence estimates is discussed in Section B.1.5.

Table C-1 Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

RG 1.190 Regulatory Position	Description of Regulatory Position	Description of Alternative and Justification
1.3.2	The bias introduced by the neutron energy cutoff technique should be estimated by comparison with an unbiased calculation.	The MCNP6 NuScale best-estimate fluence model implements a cutoff energy threshold of 0.9 MeV. An additional study involving an MCNP6 model without a cutoff energy threshold is unnecessary. Because there are no processes modeled that would result in a higher energy neutron, the use of a 0.9 MeV cutoff energy threshold makes no difference to the >1 MeV fluence results.
1.3.2	Statement of 10 statistic tests provided by Monte Carlo code	{{ }}2(a),(c) as discussed in Section 3.7.
1.3.3	The capsule fluence is extremely sensitive to the representation of the capsule geometry and internal water region (if present), and the adequacy of the capsule representation and mesh must be demonstrated using sensitivity calculations.	{{ }}2(a),(c)
1.4.2	The fluence calculation methods must be validated against (1) operating reactor measurements or both, (2) a pressure vessel simulator benchmark, and (3) the fluence calculation benchmark.	The pressure vessel simulator benchmark VENUS-3 is used to validate the NuScale fluence calculation methodology (Appendix A). The VENUS-3 benchmark results are adequate to validate the NuScale fluence calculation methodology.

Enclosure 3:

Affidavit of Carrie Fosaaen, AF-173048

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 Carrie Fosaaen

I, Carrie Fosaaen, state as follows:

(1) I am the Vice President 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 report reveals distinguishing aspects about the method by which NuScale develops its Fluence Calculation Methodology and Results.

NuScale has performed significant research and evaluation to develop a basis for this methodology 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 report entitled "Fluence Calculation Methodology and Results." The enclosure contains 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 August 26, 2024

Carrie Fosaaen