Fluence Calculation Methodology and Results

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Abstract

The methodology developed by NuScale Power, LLC, to calculate the neutron fluence for the NuScale Power Module reactor pressure vessel and containment vessel is provided by this Technical Report. Estimations of the bias and uncertainty associated with these fluence calculations, derived from benchmarking and sensitivity studies, are presented along with associated end of life fluence predictions for the NuScale reactor pressure vessel, containment vessel, 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" (Reference 7.1). Alternatives to particular Regulatory Guide 1.190 regulatory positions are described and justified. Measured data from the VENUS-3 pressure vessel simulator benchmark is used to validate the NuScale methodology.

1.0 Introduction

1.1 Purpose

The purpose of this report is to provide the methodology used to calculate the neutron fluence for the NuScale Power Module (NPM) reactor pressure vessel (RPV) and containment vessel (CNV). This report also provides the estimations of the bias and uncertainty associated with these fluence calculations, derived from benchmarking and sensitivity studies, along with associated end of life fluence predictions for the NuScale RPV, CNV, and other locations.

1.2 Scope

This report covers 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 Section 4.3 of the NuScale Design Certification Application (DCA). 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 of the scope of this report.

1.3 Abbreviations, Acronyms, and Definitions

Table 1-1 Abbreviations and Acronyms

Term	Definition
BN	bottom nozzle
CMS	core management software
CNV	containment vessel
CRA	control rod assembly
DCA	Design Certification Application
MCNP	Monte Carlo N-Particle Transport Code
MeV	megaelectronvolt
NPM	NuScale Power Module
RG	Regulatory Guide
RPV	reactor pressure vessel
TN	top nozzle
VENUS-3	Vulcain Experimental Nuclear Study 3

Table 1-2 Definitions

Term	Definition
Fluence	In the context of this report, the term "fluence" is always taken to mean the fast neutron fluence, which is the time integrated flux of neutrons with an energy greater than one megaelectronvolt (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,” 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 certain applicable 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

This report in conjunction with the NuScale Pressure and Temperature Limits Methodology Technical Report (Reference 7.5) and Final Safety Analysis Report Sections 4.3 and 5.3 address the regulatory requirements pertaining to vessel fluence analysis and surveillance.

The regulatory requirements pertaining to vessel fluence analysis and surveillance are as follows:

- 10 CFR Part 50 Appendix A, General Design Criterion 14 as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of 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 that the reactor coolant pressure boundary will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized, in part, insofar as it considers calculations of fluence.

- Appendix G, to 10 CFR Part 50, as it relates to RPV material fracture toughness requirements, in part, insofar as it considers calculations of neutron fluence.

- Appendix H, to 10 CFR Part 50, 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.

Standard Review Plan Section 4.3 and Design Specific Review Standard Sections 5.3.2 and 5.3.3 provide the following applicable NRC acceptance criteria 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 will not be exceeded.

- The acceptance criteria of RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

- The acceptance criteria of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

3.0 NuScale Power Module Fluence Prediction Methodology

3.1 Overview

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 code is a general-purpose Monte Carlo method code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. 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 follow calculations. Information from CASMO5 and SIMULATE5 is used as inputs to the MCNP6 based fluence calculation.

3.2 Geometry

Calculations are run on 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 standard NPM design submitted as part of the DCA 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 discussed 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 was implemented for consistency because fuel assembly power information was taken from NuScale's SIMULATE5 model's 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)] nodes is presented 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 TN skirt and upper core plate are modeled explicitly as part of the fuel assembly for assemblies that do not contain control rod assemblies (CRAs). {{ }}2(a),(c). On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is negligible.

- The calculation of the homogenized plenum's composition is based on the cross-sectional area within the guide and instrument tube, rather than the actual volume of the tube. This difference will lead to a higher zircalloy fraction versus water fraction. 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 except for the bottom alignment feature. The NuScale best estimate fluence model features a previous design of the bottom alignment feature compared to the design submitted as part of the DCA, as can be seen by comparing Figure 3-1 and Figure 3-2. On the basis of engineering judgment, the impact of this modeling inconsistency on the vessel fluence estimates is negligible.

- The RPV bottom core support block was not explicitly modeled and RPV flange bolts are not 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 these modeling simplification on the RPV beltline region fluence estimates is negligible.

- The weld between the {{ }}2(a),(c),ECI. On the basis of engineering judgment, the impact of this modeling inconsistency on the 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 discussed 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.

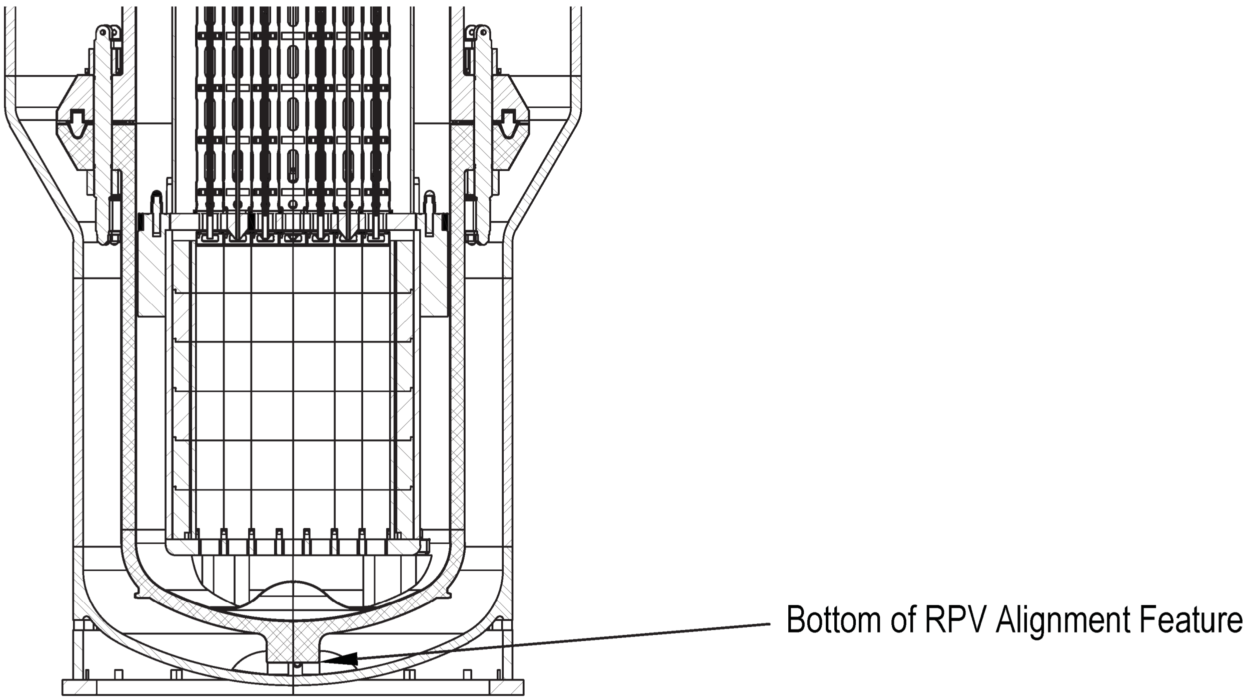

Figure 3-1 Vertical cross-sectional view of the lower section of the NuScale Power Module

{{

}}2(a),(c)

Figure 3-2 Vertical cross-sectional view of the Monte Carlo N-Particle Transport Code 6 fluence homogenized model

{{

}}2(a),(c)

Figure 3-3 Horizontal cross-sectional view of the Monte Carlo N-Particle Transport Code 6 fluence homogenized model

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 any Pu^{239} since it is based on a fresh core (beginning of cycle of Cycle 1). A bias and uncertainty to account for the contribution of Pu^{239} buildup to fluence is derived in Section B.1.2.

The material composition of the homogenized active fuel is composed of 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 was 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 was used to derive the {{ }}$^{2(a),(c)}$ library from {{ }}$^{2(a),(c)}$ and {{ }}$^{2(a),(c)}$ libraries. The {{ }}$^{2(a),(c)}$ 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 that are 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 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 megaelectronvolts as shown in Equation 3-1.

{{ }}$^{2(a),(c)}$ Eq. 3-1

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 U^{235} . Sensitivity studies on the effect of Pu^{239} buildup are presented in Section B.1.2.

There are no delayed neutrons modeled since 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 refered to according to the naming index shown in Figure 3-4.

{{

}}2(a),(c)

Figure 3-4 Fuel assembly naming index

SIMULATE5 was used to calculate the core average axial power profile associated with each cycle in an 8-cycle refueling scheme for {{ }}2(a),(c). The axial power profiles associated with each cycle were averaged to produce the 8-cycle exposure averaged axial power profile shown in Table 3-1. Table 3-1 was used to establish the vertical sampling of the neutron source used in the MCNP6 NuScale best estimate fluence model.

Table 3-1 8-cycle exposure averaged core axial power profile

{{

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

SIMULATE5 was 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 were averaged to produce the 8-cycle exposure averaged radial power profile shown in Table 3-2. The radial sampling of the neutron source used in the MCNP6 NuScale best estimate fluence model is based on Table 3-2.

Table 3-2 8-cycle exposure averaged assembly averaged radial power profile

{{

}}2(a),(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:

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

where,

υ = Average number of neutrons produced per fission in NuScale module (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 U^{235} enrichment at hot zero power,

P= Fission power (MW); taken to be 160 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:

$$S = \frac{2.46 \frac{neutrons}{fission} * \ 160MW * 10^6 \left[\frac{W}{MW}\right]}{1.602 \times 10^{-13} \left[\frac{J}{MeV}\right] * 1.000 * 198 \frac{MeV}{Fission}} = 1.24 \times 10^{19} \frac{\text{neutrons}}{\text{second}}$$

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

No lower cut off energy is utilized, 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 series of cylindrical mesh tallies and surface 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 Figures 3-5 through Figure 3-7, including naming and numbering conventions for the axial and azimuthal segments. The cylindrical mesh tallies were defined as {{ }}$^{2(a),(c)}$. A sensitivity study on the effect of the tally region volume's impact on final fluence results was performed by implementing a finer, {{ }}$^{2(a),(c)}$. The difference between results produced by the {{ }}$^{2(a),(c)}$. Note that the difference in results produced by the finer mesh is generally less than the MCNP6 calculated relative error; therefore, no term accounting for the uncertainty in the fluence results associated with variances caused by mesh size selection was added to Table 4-1.

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

{{

}}2(a),(c)

Figure 3-5 Vertical cross-sectional view of the reactor pressure vessel mesh tally

{{

}}2(a),(c)

Figure 3-6 Horizontal cross-sectional view of the reactor pressure vessel mesh tally

{{

}}2(a),(c)

Figure 3-7 Horizontal cross-sectional view of the containment vessel mesh tally

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, is provided in Table 4-1.

Table 4-1 List of quantified systematic biases (+ or -) and random uncertainties(+/-)

{{

}}2(a),(c)

{{

}}2(a),(c)

4.2 Combination of Biases

The analytical bias (also known as B_a^c per RG 1.190 terminology) is comprised of all 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 all systematic biases presented in Table 4-1, excluding B_b^c, as shown in Equation 4-1.

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

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:

{{
}}2(a),(c) Eq. 4-2

4.3 Combination of Uncertainties

Independent random uncertainties are all uncertainties that 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.

$$\sigma^c = \sqrt{\sigma_{mt}^2 + (\sigma_a^c)^2} \qquad \text{Eq. 4-3}$$

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 all random uncertainties in Table 4-1, excluding σ_{mt}, as shown in Equation 4-4.

$$\sigma_a^c = \sqrt{\sigma_{Pu}^2 + \sigma_g^2 + \sigma_m^2 + \sigma_{ap}^2 + \sigma_{pr}^2 + \sigma_{pa}^2 + \sigma_{Boron}^2} \qquad \text{Eq. 4-4}$$

{{

}}2(a),(c)

Substituting the value established for σ_a^c back into Equation 4-3 gives Equation 4-5. Equation 4-5 is used to establish overall uncertainties given in Table 5-1.

{{ }}2(a),(c) Eq. 4-5

5.0 NuScale Power Module Fluence Prediction Results

Table 5-1 presents the results of the best estimate fluence analysis. These results are of the highest fluence found on the particular component after comparing results across various axial and radial locations. The uncertainties associated with each location based on Section 4.3 are also presented. Note that in Table 5-1 the "Best Estimate Neutron Fluence" is the result of applying the total bias (B_T) of {{ }}2(a),(c), established in Section 4.2, to the "MCNP Calculated Neutron Fluence."

The five representative statistical tests used to ensure tally convergence discussed in RG 1.190 were satisfied for the results in Table 5-1, with the exception that fluence estimates with relative errors greater than 0.1 (10 percent) or total uncertainties greater than 20 percent are noted in Table 5-1.

Table 5-1 Best estimate of fluence expected to be experienced in various NuScale Power Module components and locations

{{

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

{{

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

{{

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

{{

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

(1) Result had a relative error greater than 0.1, but the result is still seen as a reliable fluence estimate.
(2) Result had an uncertainty greater than 20 percent, but the result is still seen as a reliable fluence estimate.

{{

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

6.0 Summary and Conclusions

A best estimate neutron fluence calculation for the NPM was performed through the use 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 were 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 cladding surface and CNV beltline at ¼-T fluence over a 60-year NPM operating life (assumed 95 percent capacity factor) was calculated to be {{

}}2(a),(c),ECI as reported in Table 5-1. Neutron fluence estimates provided in this report are acceptable for supporting Final Safety Analysis Report Section 4.3 of the NuScale DCA 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, March 2001.

7.2 D.I. Poston and H.R. Trellue, "User's Manual, Version 2.0 for Monteburns, Version 5B," Los Alamos National Laboratory, 1999.

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

7.4 NEA Nuclear Science Committee, “Prediction of Neutron Embrittlement in the Reactor Pressure Vessel: VENUS-1 and VENUS-3 Benchmarks,” OECD, 2000.

7.5 NuScale Power, LLC, "Pressure and Temperature Limits Methodology Topical Report," TR-1015-18177, Revision 0.

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 was 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 7.3). 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.

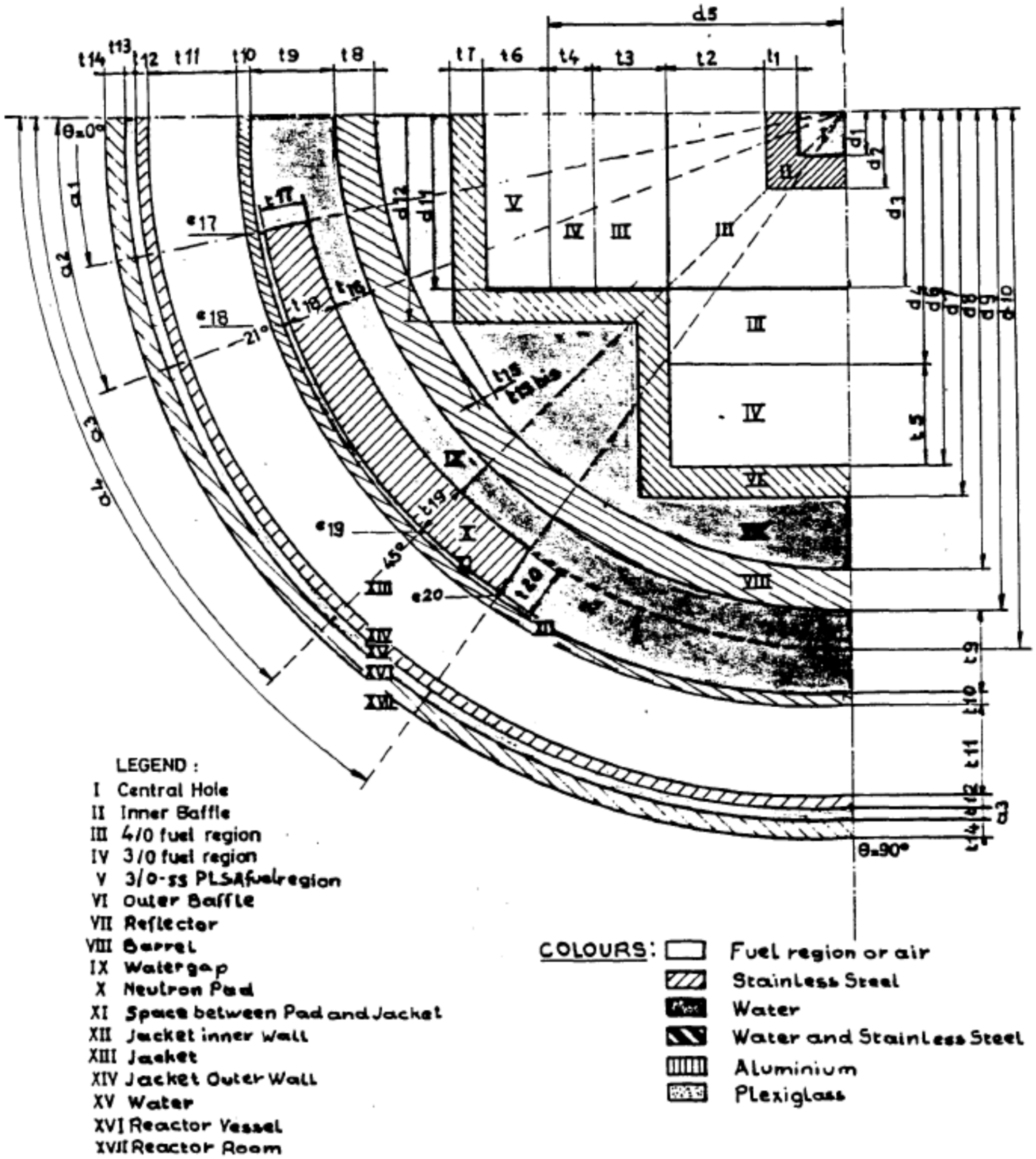

Figure A-1 Horizontal cross-sectional view of the Vulcain Experimental Nuclear Study 3 benchmark geometry

The MCNP6 model is based on the MCNP model supplied as part of the VENUS-3 benchmark collection in Reference 7.3, which used an earlier version of MCNP. This model was 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 degrees K (.80c extension) were 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 U^{235} Watt fission spectrum.

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

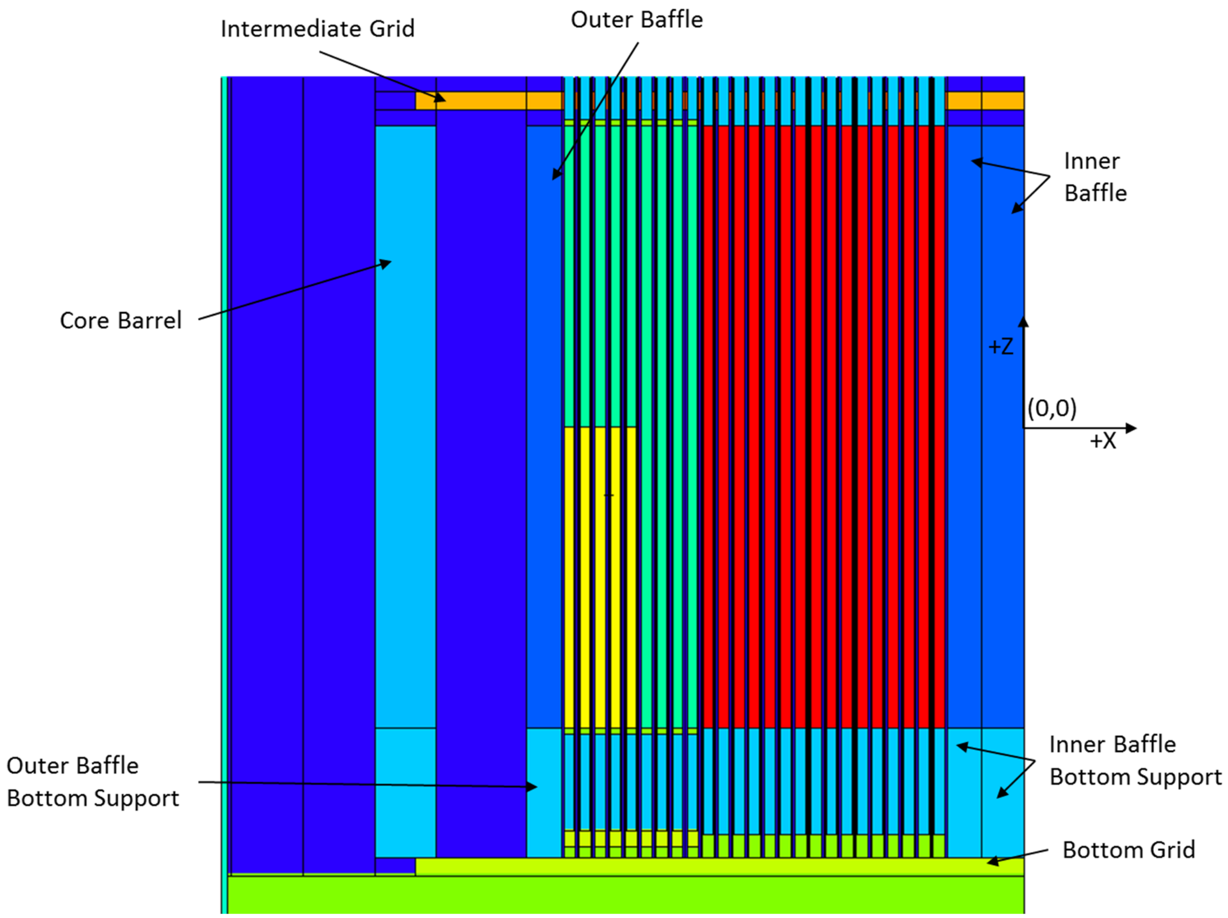

Figure A-2 Vertical cross-sectional view of the Monte Carlo N-Particle Transport Code 6 model of the Vulcain Experimental Nuclear Study 3 benchmark

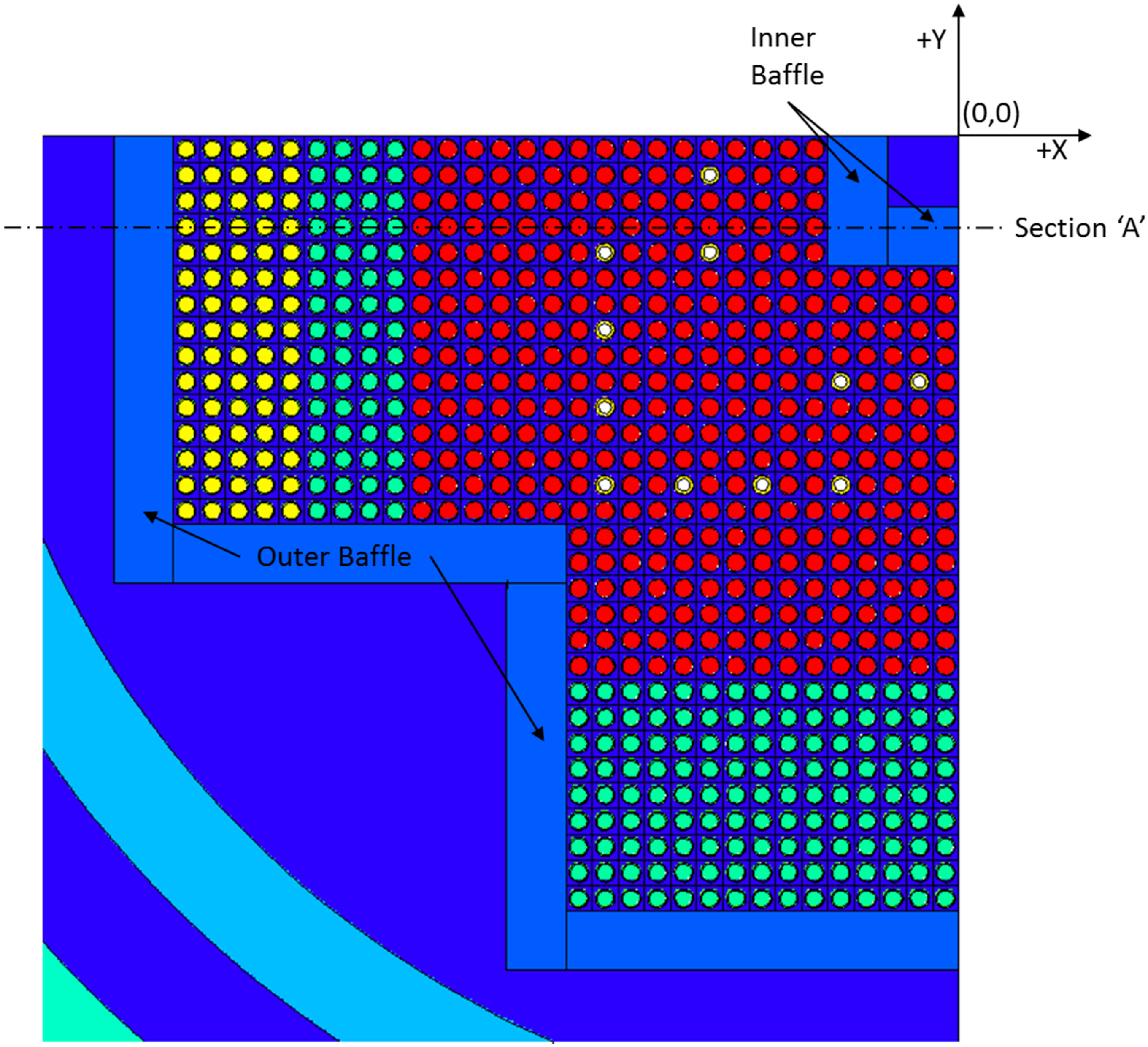

Figure A-3 Horizontal cross-sectional view of the inner and outer baffle of the Monte Carlo N-Particle Transport Code 6 model of the Vulcain Experimental Nuclear Study 3 benchmark

A variety of experimental results were 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 $Ni^{58}(n,p)$, $In^{115}(n,n')$, and $Al^{27}(n,\alpha)$, respectively.

Based on the energy thresholds associated with the reaction rates, the $In^{115}(n,n')$ reaction rates are associated with the neutron flux greater than 1 MeV, the $Ni^{58}(n,p)$

reaction rates are associated with neutron fluxes greater than 3 MeV, and the $Al^{27}(n,\alpha)$ 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 were reported to be 9 percent for $Ni^{58}(n,p)$, 7 percent for $In^{115}(n,n')$, and 14 percent for $Al^{27}(n,\alpha)$ in Section 6.1 of Reference 7.4.

The relative difference between the reported experimental (Exp) values for these reaction rates and the MCNP6 calculated values (Calc) was 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 all experimental versus calculated values and standard deviations are reported in Table A-1.

$$Relative\ difference\ (\%) = \frac{Exp - Calc}{Exp} \qquad \text{Eq. A-1}$$

The $In^{115}(n,n')$ reaction rate comparisons were judged to provide the best comparison to the overall neutron flux since it has the lowest threshold energy of ~1 MeV. The $Ni^{58}(n,p)$ and $^{27}Al(n,\alpha)$ reaction rates have higher thresholds, 3 MeV and 8 MeV, respectively. The $In^{115}(n,n')$ results also have the lowest experimental uncertainty associated with them. Further, the $In^{115}(n,n')$ 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 $Ni^{58}(n,p)$ or $Al^{27}(n,\alpha)$ 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. This will be 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 were based on a homogenized fuel model. {{

}}2(a),(c).

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

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

}}2(a),(c)

{{

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

Table B-1 Interpolated fast neutron response function as a function of ν/Q_{ave} and fission weighting fraction

{{

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

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 were 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)

B.1.5 Pin Power Profile

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)

Table B-2 The averaged fast neutron flux in pin lattice of fuel assembly A4, cycle 8 (units of $10^{13}/cm^2s$)

{{

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

{{

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

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

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

}}2(a),(c).

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 cycle averaged radial power distribution was used in the NuScale best estimate fluence model. {{

}}2(a),(c).

B.1.8 Axial Power Profile

Uncertainty in the axial power profile is directly proportional to the uncertainty of the fluence estimates. The cycle averaged axial power distribution was used in the NuScale best estimate fluence model. {{

}}2(a),(c).

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 will vary over the course of the fuel cycle. {{

}}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. {{

}}2(a),(c)

B.1.12 Water Density

{{

}}2(a),(c).

Appendix C. Regulatory Guide 1.190 Alternatives

Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” provides guidance for calculating pressure vessel neutron fluence. The NuScale fluence calculation methodology described in this report utilized 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 particular regulatory positions of RG 1.190.

Table C-1 Alternatives to particular 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 was accounted for in Section B.1.12.
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 was accounted for {{ }}2(a),(c) as discussed in Sections B.1.3 and 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.1.12.

RG 1.190 Regulatory Position	Description of Regulatory Position	Description of Alternative and Justification
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 were 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.
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 (see Appendix A). The VENUS-3 benchmark results are adequate to validate the NuScale fluence calculation methodology.
1.4.3	The fluence accuracy requirements are generally application specific; however, a vessel fluence uncertainty of 20 percent (1 sigma) is acceptable for RT_{PTS} and RT_{NDT} determination.	{{ }}2(a),(c)