July 30, 2018

Docket No: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Topical Report TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," Revision 1

REFERENCE: Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Topical Report TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," Revision 0, dated August 30, 2016 (ML 16243A517).

The NuScale Power, LLC (NuScale) Nuclear Analysis Codes and Methods Qualification Topical Report TR-0616-48793 was submitted to the NRC in the reference and was presented to the Advisory Committee on Reactor Safeguards (ACRS) NuScale subcommittee on May 15, 2018, and to the ACRS full committee on June 7, 2018. Revision 1 to this topical report incorporates changes made as a result of an NRC request for additional information, declassification of some proprietary information, minor revision of information in a couple of tables, and a change to the document date in a reference. The changes are indicated by revision bars in the right hand column.

Enclosure 1 contains the proprietary version of the revised topical report entitled "Nuclear Analysis Codes and Methods Qualification". 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.

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

Please contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
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NuScale Power, LLC

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Enclosure 1: TR-0616-48793-P, “Nuclear Analysis Codes and Methods Qualification,” Revision 1, proprietary version
Enclosure 2: TR-0616-48793-NP, “Nuclear Analysis Codes and Methods Qualification,” Revision 1, nonproprietary version
Enclosure 3: Affidavit of Thomas A. Bergman, AF-0618-60702

Enclosure 1:

TR-0616-48793-P, “Nuclear Analysis Codes and Methods Qualification,” Revision 1, proprietary version

Enclosure 2:

TR-0616-48793-NP, “Nuclear Analysis Codes and Methods Qualification,” Revision 1, nonproprietary version

Nuclear Analysis Codes and Methods Qualification

July 2018

Revision 1

Docket: PROJ0769

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Abstract

This NuScale topical report describes the methodology for design and steady-state analysis of the NuScale reactor core. The nuclear design process comprises the utilization of mechanical properties as design input to the nuclear code system, the application of the associated methodology for core design, analysis, and operational support, and the development of Nuclear Reliability Factors (NRFs) for key reactor physics parameters. This report describes the application of the nuclear design process to the NuScale reactor core design and analysis methodology using the Studsvik Scandpower Core Management Software version 5 (CMS5) suite. The CMS5 suite includes the CASMO5, CMSLINK5, and SIMULATE5 computer codes.

This report addresses code qualification and validation using benchmarking data to develop a set of base NRFs to be used when generating key physics parameters for core design, startup, operation, and safety analysis. NuScale requests approval of the nuclear analysis methodology and applications, the nuclear reliability factors developed for application to the NuScale design, and the methodology for updating them once operating data are available.

Executive Summary

This report describes the NuScale Power Plant nuclear design and steady-state analysis methodology. This report provides an overview of the nuclear code system employed by NuScale for reactor core physics design and analysis, and a description of the defining characteristics of the NuScale fuel and reactor core to which the code system is applied. Application of the code system, in addition to design and analysis, includes methodology and model development for core design activities and reactor physics calculations to support safety analysis, startup physics testing, core predictions for operations, and core follow predictions.

The methodology uses the Studsvik Scandpower CMS5 software to perform reactor physics core design and analysis calculations. The CMS5 software used in this methodology includes the lattice physics code, CASMO5; the cross section library generation linkage code, CMSLINK5; and the reactor simulator code, SIMULATE5. CASMO5 generates cross sections as a function of various reactor state points. Input to CASMO5 includes fuel rod and assembly geometry and compositions, and core operating data. CMSLINK5 reads CASMO5 card image files to produce a cross section library for use in SIMULATE5. Input to SIMULATE5 includes the cross section library generated by CMSLINK5, core geometry and compositions, core operating configurations and data, and fuel and control rod assembly axial compositions. SIMULATE5 is used to perform calculations to determine various reactor neutronics parameters at different times in cycle life, including power distributions, peaking factors, critical boron concentration, reactivity coefficients, and control rod assembly worth. Results from SIMULATE5 are used to determine core designs, provide input to safety analysis, provide data for startup physics testing and operations, and for core follow.

Qualification of the CMS5 codes is accomplished by performing higher-order code comparison benchmarks, empirical benchmarks against experimental reactors, and benchmarks to commercial reactor data. Because no operational core physics data specific to the NuScale design are currently available to support code qualification, the higher-order code-to-code benchmarks are important for demonstrating code capability specifically applicable to the NuScale design. Uncertainty factors are developed from the benchmarking results and are used to derive nuclear reliability factors (NRFs). These NRFs are applied conservatively to CMS5-calculated neutronics parameters. Following startup and operation of the initial NuScale modules, physics data collected will be used to confirm these NRFs, or provide the basis for updating the NRFs.

Core design and analysis is performed using CASMO5/SIMULATE5 to model the mechanical and operational characteristics of the NuScale reactor core to calculate physics parameters that are compared against pre-established design targets. Core physics parameters are an input to the safety analysis of the NuScale reactor, and are calculated conservatively in order to bound expected core operational behavior.

SIMULATE5 is also used to calculate neutronic parameters for comparison to measurements during reactor startup after fuel loading, predict physics parameters for operation, and for core follow. These comparisons provide assurance that the core will operate as designed and ensure acceptable core performance during the entire operating cycle.

This report meets the regulatory requirements for nuclear methods and design in accordance with 10 CFR 50 Appendix A, General Design Criteria 10, 11, and 12, and is consistent with the guidance of NUREG-0800, Section 4.3. This topical report requests NRC approval of the following:

1. the application of the CMS5 nuclear analysis code suite and methodology to generate nuclear physics parameters for the core design, safety analysis, and plant startup and operations of the NuScale design;

2. a base set of NRFs shown in Table 1-2 to be applied to the CMS5 calculated physics parameters for the NuScale design; and

3. the methodology for updating NRFs when operating data are available for a NuScale reactor following initial plant startup.

1.0 Introduction

This report describes the nuclear analysis codes and methods to be used for the NuScale Power Plant reactor design. NuScale utilizes the Studsvik Scandpower CMS5 code suite with CASMO5 and SIMULATE5 as the key neutronics codes for steady state nuclear analysis. Previous versions of CASMO/SIMULATE have been widely used in the industry and accepted by the NRC for steady state reactor physics applications.

This topical report justifies the use of the CMS5 code suite to perform the neutronics modeling for the NuScale design. The justification includes code validation with benchmarking to higher-order codes, critical and empirical experiments, and commercial operating reactors. As a result of the benchmarks, a series of biases and NRFs are generated for use. This topical report also describes a methodology for updating NRFs once measured data are available for the NuScale design. Finally, the report describes the application of the neutronics parameters developed from the CMS5 code suite in the plant operations and safety analysis.

CASMO-4 and SIMULATE-3 have an extensive licensing history for use in core physics calculations. CASMO5 and SIMULATE5 are the current generation of this line of neutronics software developed by Studsvik Scandpower. This software has significant modeling enhancements from the previous code versions. Details of the enhancements are provided in Section 3.0 of this report.

1.1 Scope

This report is applicable to the key NuScale fuel characteristics and limits provided in Table 1-1.

Section 2.0 of this report describes the reactor design, fuel design and material properties of the NuScale module. Section 3.0 provides the details of the CMS5 code methods, model descriptions, and software qualification process. Section 4.0 discusses the statistical methodology. Sections 5.0 and 6.0 discuss validation of the CMS5 code suite using a series of benchmarks including code-to-code benchmarks, critical benchmarks, experimental reactor benchmarks, commercial reactor benchmarks, and comparisons to industry standards. Section 7.0 uses the results of these benchmarks to develop a set of base NRFs to be used for the NuScale design. Section 7.0 also discusses the methods by which these NRFs will be validated and updated once measurements are performed during startup and operation of the initial NuScale modules. Finally, the intended application of this methodology for generating neutronics parameters for core design, safety analysis, and plant operations is described in Section 8.0. The flow from benchmarking and validation through NRF development into code applications is shown in Figure 1-1.

This topical report requests NRC approval of the following:

1. the application of the CMS5 nuclear analysis code suite and methodology to generate nuclear physics parameters for the core design, safety analysis, and plant startup and operations of the NuScale design;

2. a base set of NRFs derived in Section 7.0 and shown in Table 1-2 to be applied to the CMS5 calculated physics parameters for the NuScale design; and
3. the methodology for updating NRFs when operating data are available for a NuScale reactor following initial plant startup.

Table 1-1. NuScale Fuel Material Characteristics Applicability Limits

Parameter	Applicability Limits
Fuel Lattice Geometry	17x17
Number of Fuel Assemblies in Core	37
Fuel Composition	Uranium Dioxide (UO_2) (≤ 4.95% ^{235}U enrichment)
Fuel Rods per Assembly	264
Guide Tubes/Instrument Tubes per Assembly	24/1
Fuel Clad Material	Zirconium-based alloy
Guide Tube Material	Zirconium-based alloy
Integral Burnable Poison Type and Weight Percent	Gadolinium Oxide (Gd_2O_3) (≤ 8.0 wt%)
Control Rod Assembly Material	Silver-Indium-Cadmium (AIC), Boron Carbide (B_4C)
Maximum Rod Burnup	≤ 62 Gigawatt days per Metric Ton of Uranium (GWd/MTU)

Table 1-2. Base Nuclear Reliability Factors

Parameter	Lower NRF	Upper NRF
Critical Boron Concentration	{{	
Differential Boron Worth		
Isothermal Temperature Coefficient and Moderator Temperature Coefficient		
Power Coefficient and Fuel Temperature Coefficient		
Control Rod Assembly Bank Worth		}}^{2a,c}
Assembly Radial Peaking	N/A	{{
F_Q Pin Peaking	N/A	
$F_{\Delta H}$ Pin Peaking	N/A	}}^{2a,c}
Axial Offset	{{	
Kinetics		}}^{2a,c}

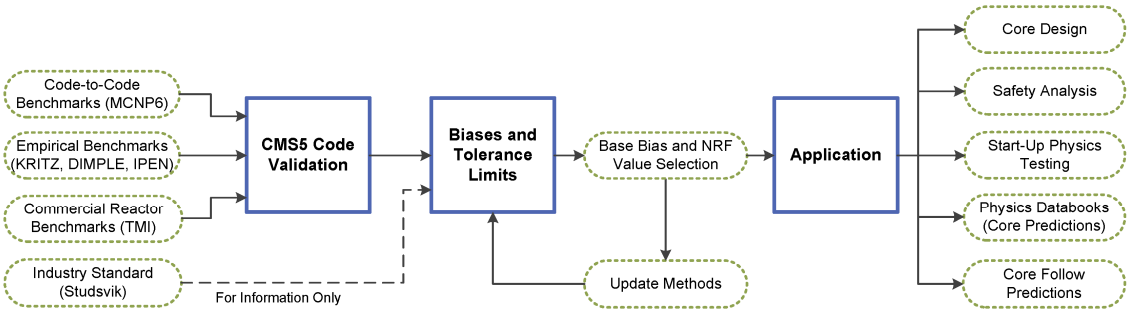

Figure 1-1. Code Validation and Benchmarking Flowchart

1.2 Regulatory Requirements

The following regulatory requirements and guidance documents are relevant to the codes and methods qualification described in this report:

- 10 CFR 50, Appendix A, General Design Criterion (GDC) 10, *Reactor Design*
- 10 CFR 50, Appendix A, GDC 11, *Reactor inherent Protection*
- 10 CFR 50, Appendix A, GDC 12, *Suppression of Reactor Power Oscillations*
- NUREG-0800 Standard Review Plan (SRP), Section 4.3, *Nuclear Design*
- NUREG-0800 SRP, Section 15.0.2, Review of Transient and Accident Analysis Methods
- Regulatory Guide (RG) 1.203, *Transient and Accident Analysis Methods*

This report documents the qualification of the nuclear analysis codes and methods used to determine the nuclear design of the NuScale reactor. The nuclear methods described in this report ensure the nuclear design of the NuScale reactor complies with the applicable GDCs listed in SRP, Section 4.3, and are acceptable to perform those calculations aimed at assessing the safety of the nuclear design of the NuScale reactor core. Additionally, SRP Section 15.0.2 and RG 1.203 describe the requirements and guidance for analytical models and computer codes used to analyze accident and transient behavior. The nuclear analysis codes and methods presented in this report are primarily for design purposes. However, overlap exists between design methods, and transient and accident analysis methods, so SRP Section 15.0.2 and RG 1.203 are applicable to this report.

This topical report is expected to support Section 4.3 of the NuScale FSAR by describing the steady state nuclear analysis methodology for nuclear design and analysis of the NuScale reactor core. The nuclear design process consists of the compilation of the mechanical properties used as nuclear design input, application of the nuclear code system and associated methodology for core design, analysis and operational support, and the development of biases and NRFs for key reactor physics parameters.

This topical report is intended for use by Combined License applicants and licensees to implement core design methodology for their safety analysis calculations and operational support.

1.3 Abbreviations and Definitions

Table 1-3. Abbreviations

Term	Definition
°F	degrees Fahrenheit
λ	delayed neutron decay constant
%Δ	percent difference or percent change
2D	two dimensional
3D	three dimensional
AIC	silver-indium-cadmium
AO	axial offset
ARO	all rods out
B_4C	boron carbide
β_{eff}	effective delayed neutron fraction
BOC	beginning of cycle
BP	burnable poison
CBC	critical boron concentration
CHF	critical heat flux
cm	centimeters
CMS5	Core Management Software, version 5
CRA	control rod assembly
CVCS	chemical volume control system
DBW	differential boron worth
ENDF	Evaluated Nuclear Data File
EOC	end of cycle
eV	electron volt
$F_{\Delta H}$	enthalpy rise hot channel factor
F_Q	heat flux hot channel factor (total peaking factor)
FSAR	Final Safety Analysis Report
FTC	fuel temperature coefficient
g/cc	grams per cubic centimeter
Gd_2O_3	gadolinium oxide or gadolinia
GDC	general design criterion
GWd/MTU	gigawatt days per metric ton of uranium
HFP	hot full power
HZP	hot zero power
ID	inner diameter
in	inches

Term	Definition
k_{eff}	effective neutron multiplication factor
ITC	isothermal temperature coefficient
LTL	lower tolerance limit
LWR	light water reactor
MCNP5	Monte Carlo N-Particle, version 5
MCNP6	Monte Carlo N-Particle, version 6
MeV	mega electron volt
MOC	middle of cycle
MTC	moderator temperature coefficient
MWd/MTU	megawatt days per metric ton of uranium
MWt	megawatts thermal
N/A	not applicable
NRF	nuclear reliability factor
OD	outer diameter
pcm	percent millirho
PDIL	power dependent insertion limit
PLHGR	peak linear heat generation rate
ppm	parts per million
PWR	pressurized water reactor
QA	quality assurance
RAP	relative assembly power
RCS	reactor coolant system
RG	Regulatory Guide
RLP	relative linear power
RMS	root mean square
RPP	relative pin power
SRP	Standard Review Plan
TH	thermal hydraulics
TMI-1	Three Mile Island Unit 1
UO_2	uranium dioxide
UTL	upper tolerance limit
V&V	verification and validation

The definitions included in Table 1-4 are provided to explicitly define core physics terms of importance that are used in this topical report.

Table 1-4. Definitions

Core Physics Term	Definition
Assembly Radial Peaking	The ratio of the peak assembly power to the average assembly power.
Axial Offset (AO)	The ratio of the difference between the power in the top and bottom halves of the core to the total core power.
Burnable Poison (BP)	An integral burnable poison is a material with a large neutron absorption cross section that is mixed homogeneously with the fuel material in select fuel rods, and is intended to compensate for excess reactivity.
Critical Boron Concentration (CBC)	The concentration of boron in the water moderator of a reactor core that allows the reactor to be exactly critical (i.e., Effective neutron multiplication factor (k_{eff}) = 1.0).
Control Rod Assembly (CRA) Worth	A measure of the potential change in reactivity of the core that is represented by the insertion or removal of the CRA.
Control Rod Assembly Bank Worth	A measure of the potential change in reactivity of the core that is represented by the insertion or removal of a CRA bank.
Differential Boron Worth (DBW)	The change in reactivity associated with a change in the boron concentration in the water moderator. The reactivity change is measured in units of percent millirho (pcm) and the change in boron concentration is in parts per million (ppm).
Fuel Temperature Coefficient (FTC), or Doppler Temperature Coefficient	The change in reactivity associated with a change in fuel temperature. The reactivity change is measured in units of pcm and the change in fuel temperature is in degrees Fahrenheit.
$F_{\Delta H}$	The enthalpy rise hot channel factor, defined as a ratio of the enthalpy rise of the hot channel to the average channel enthalpy rise of the core.
F_Q	The heat flux hot channel factor, defined as a ratio of the maximum heat flux in the hot channel to the average heat flux in the core.
Isothermal Temperature Coefficient (ITC)	The change in reactivity associated with the combined change in both the moderator and the fuel temperature in a reactor in which the temperature is uniform across the core. The reactivity change is measured in units of pcm and the change in temperature is in degrees Fahrenheit.
Kinetics Parameters	The key parameters that characterize the time-dependent nature of a neutron population. Kinetic parameters of specific interest are the effective delayed neutron fraction (β_{eff}) and the delayed neutron decay constant (λ). These parameters are typically determined based on six energy groups and may be presented for each group and as a sum over all groups. The groups are based on the approximate half-life of the delayed neutrons.
Moderator Temperature Coefficient (MTC)	The change in reactivity associated with a change in moderator temperature. The reactivity change is measured in units of pcm and the change in temperature is in degrees Fahrenheit.
NRF	The tolerance applied to the calculated value of a parameter that accounts for uncertainties and ensures conservatism in the safety-related application of the parameter.
Peak Linear Heat Generation Rate (PLHGR)	The maximum rate of heat flow per unit length of the fuel rod. PLHGR is calculated from the maximum total peaking factor (F_Q) multiplied by the average linear heat generation rate.
Pin-to-Box Ratio	The ratio of the peak pin power in the assembly to the average pin power in the assembly.

Core Physics Term	Definition
Pin Power	The amount of power generated by a single pin, expressed either as an absolute value or relative to an assembly- or core-average value.
Pin Peaking	Defined by determining the worst case for a ratio of local conditions to average conditions; pin peaking is frequently characterized by the F_Q and $F_{\Delta H}$ peaking factors.
Power Coefficient	The change in reactivity associated with a percent change in power; the power coefficient is the combined effect of the moderator, fuel, and void coefficients. The reactivity change is measured in units of pcm and the power change is measured by change in percent of rated thermal power.
Relative Assembly Power (RAP)	The ratio of the relative power in a fuel assembly to the average power in the core.
Relative Linear Power (RLP)	The ratio of the linear power in an axial node of a pin to the average linear power.
Relative Pin Power (RPP)	The ratio of the power in a pin to the average pin power.
Void Coefficient	The change in reactivity associated with a percent change in void volume. The reactivity change is measured in units of pcm and the percent change in void volume is unitless.

2.0 Reactor Design

The CMS5 code suite is used to model and perform steady state nuclear analysis of the NuScale reactor. The mechanical design properties, described in Table 2-1, are input to the code models including descriptions of the fuel assemblies, fuel rods, CRAs, integral BP, and heavy reflector.

Table 2-1. Nominal Reactor Mechanical Design Parameters

Core Description	
Power Level, MWt	160
Number of Fuel Assemblies	37
Number of CRAs	16
Fuel Assembly Pitch, in. (cm)	8.466 (21.504)
Fuel Assembly Description	
Fuel Rod Array	17 x 17
Fuel Rod Pitch, in. (cm)	0.496 (1.26)
Number of Guide Tubes	24
Number of Instrument Tubes	1
Guide Tube Inner Diameter (ID), in. (cm)	0.45 (1.143)
Guide Tube Outer Diameter (OD), in. (cm)	0.482 (1.224)
Guide Tube Material	Zircaloy-4
Fuel Rod Description	
Material	UO_2
Maximum Enrichment, wt%	4.95
Nominal Stack Height Density, g/cc	10.4
Pellet Diameter, in. (cm)	0.3195 (0.8115)
Clad Material	M5®
Clad ID, in. (cm)	0.326 (0.828)
Clad OD, in. (cm)	0.374 (0.95)
Clad Thickness, in. (cm)	0.024 (0.061)
Active Fuel Length, in. (cm)	78.74 (200)
Control Rod Assembly Description	
Clad Material	Stainless Steel
Clad Thickness, in. (cm)	0.0185 (0.047)
Clad OD, in. (cm)	0.381 (0.968)
Poison Material	AIC, B_4C
AIC Poison Material Length, in. (cm)	11.8 (30.0)
B_4C Poison Material Length, in. (cm)	62.0 (157.5)
Integral Burnable Poison Description	
Absorber Material	Gd_2O_3
Loading, wt%	{{
Number of BP Rods Per Fuel Assembly	}}^{2a,c}

2.1 Reactor Core Description

The reactor core is designed to produce 160 megawatts thermal (MWt) at full power. The core contains thirty-seven (37) fuel assemblies as shown in Figure 2-1. Each fuel assembly is configured in a square 17x17 array consisting of two hundred sixty-four (264) fuel rods, twenty-four (24) guide tubes, and one instrumentation tube. The fuel assembly structure consists of a top nozzle, a bottom nozzle, and five (5) axially positioned spacer grids. An example fuel assembly and its lattice structure are shown in Figure 2-2.

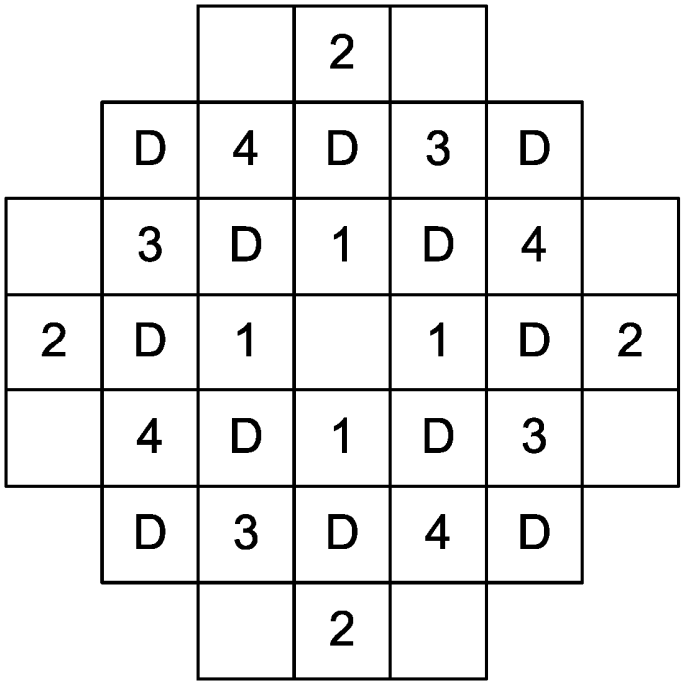

Figure 2-1. Control Rod Assembly and In-Core Instrument Locations

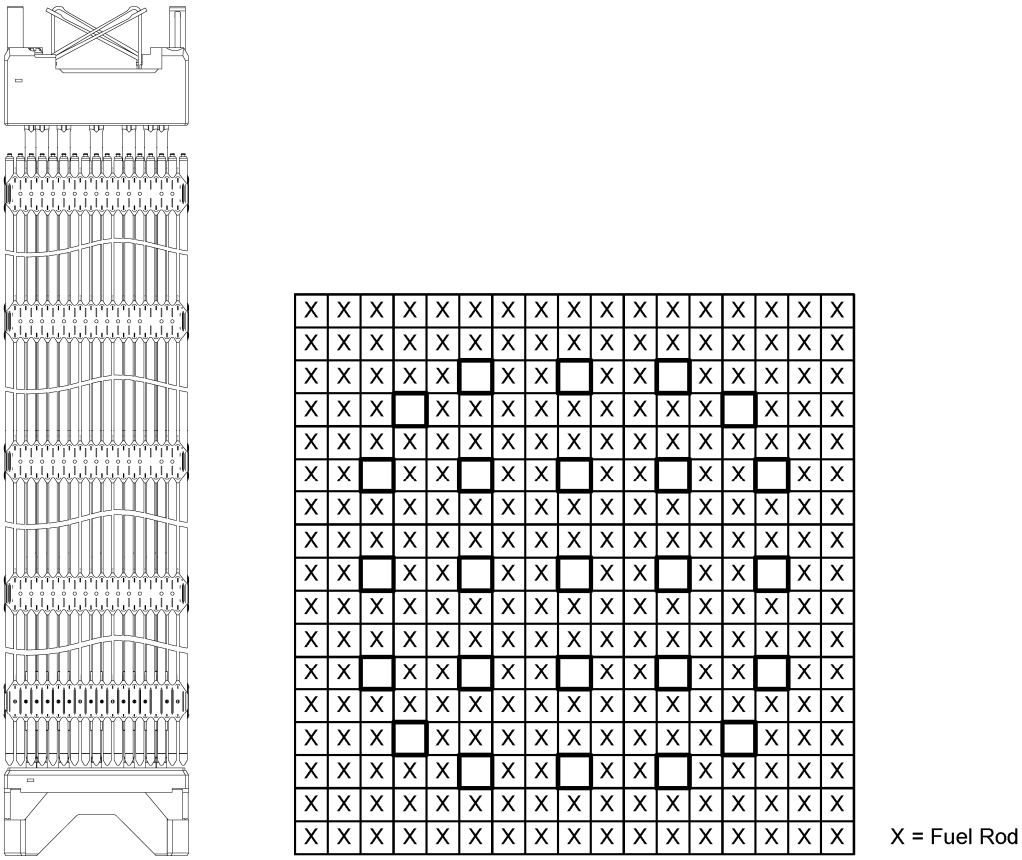

Figure 2-2. Example Fuel Assembly and 17x17 Lattice Design

2.2 Fuel Rod Description

The fuel rod consists of a column of stacked, cylindrical, ceramic pellets of up to 4.95 percent enriched UO_2 encapsulated in M5® (a zirconium-based alloy) cladding. The guide tubes are composed of Zircaloy-4 (a zirconium-based alloy). The active fuel length is 2.0 meters. The fuel rods are pressurized with helium gas. The fuel rods may contain distributed BP in the form of Gd_2O_3 and may be axially zoned with varying UO_2 enrichment or BP loading to improve neutron economy and the axial power distribution. The peak rod burnup will not exceed 62 GWd/MTU (Reference 10.1.13). An example fuel rod is depicted in Figure 2-3.

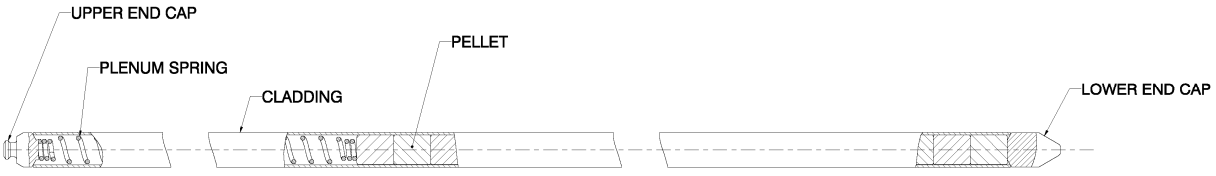

Figure 2-3. Example Fuel Rod

2.3 Control Rod Assembly Description

The reactor contains sixteen (16) CRAs. Along with soluble boron, CRAs serve as one of the two methods to control core reactivity. The design function of the CRAs is to provide the ability to shut down the reactor and perform power control. The NuScale design incorporates two CRA banks, as shown in Figure 2-1. The shutdown bank, which is composed of three sub-groups, is used for reactor shutdown; the regulating bank is used for power maneuvering and to maintain the AO within the licensing limits during normal operation.

Each CRA contains twenty-four (24) rods fastened at the top end to a common hub or spider. The rods contain two neutron absorber materials, AIC at the bottom of the rod, and B_4C. The CRA rods are clad with stainless steel. An example CRA is shown in Figure 2-4.

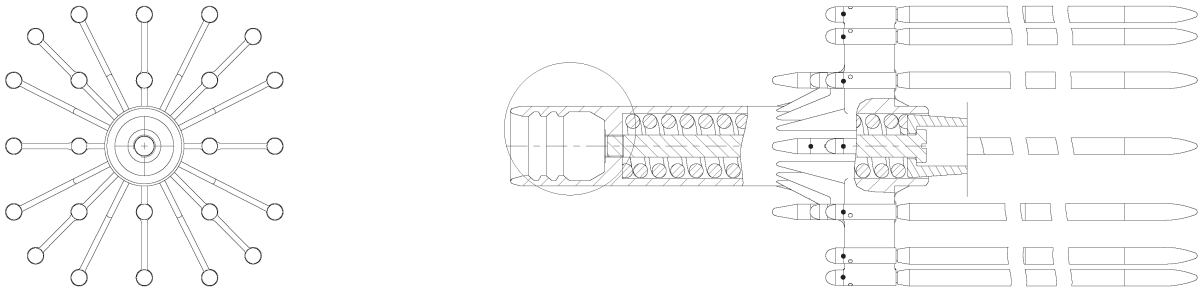

Figure 2-4. Example Control Rod Assembly

2.4 Integral Burnable Poison Description

Burnable poisons are loaded into the reactor core to compensate for the excess reactivity needed for full power operation over the entire cycle. The design uses a combination of chemical shim (soluble boron) in the coolant, and poisons distributed in the fuel to help compensate for the excess reactivity of the fuel. However, for discussion purposes, BPs mentioned in this report refers to the fuel-distributed poisons and not the chemical shim.

Select assemblies and fuel rods in the core designs use Gd_2O_3 (gadolinia) loadings ranging from {{ }}^{2a,c}, which is homogenously combined with the UO_2 in the fuel pellet. The number of gadolinia-containing fuel rods in an assembly range from {{ }}^{2a,c}, increasing by multiples of four for the purpose of maintaining symmetry for assembly loading.

Although Gd_2O_3 is physically compatible with UO_2, its addition to the fuel degrades some of the material properties of the UO_2. Specifically, the thermal conductivity and the melting temperature of the mixture are lower than those of pure UO_2. Thus, fuel containing gadolinia is limited to a lower linear heat generation rate than fuel only containing UO_2 by considerations of centerline melting. To compensate, the ^{235}U

enrichment in the gadolinia-containing fuel rods is lower than the UO_2 only fuel rods in order to prevent a gadolinia-containing fuel rod from being the limiting case.

2.5 In-Core Instrumentation

The in-core instrumentation system for power distribution measurement consists of twelve (12) strings of self-powered neutron detectors. The location of the in-core detectors in the reactor core is shown in Figure 2-1.

2.6 Reflector Description

Water and stainless steel baffle reflectors, typically one to two centimeters thick, are commonly used in experimental and commercial Light Water Reactors (LWRs). To reduce neutron leakage and more evenly distribute power, the reactor core is surrounded by a stainless steel heavy reflector that varies in radial thickness nominally from ten to thirty centimeters (10 cm - 30 cm). The heavy reflector design is composed of stainless steel with coolant channels throughout to allow primary coolant to absorb the heat generated by gamma radiation (Figure 2-5).

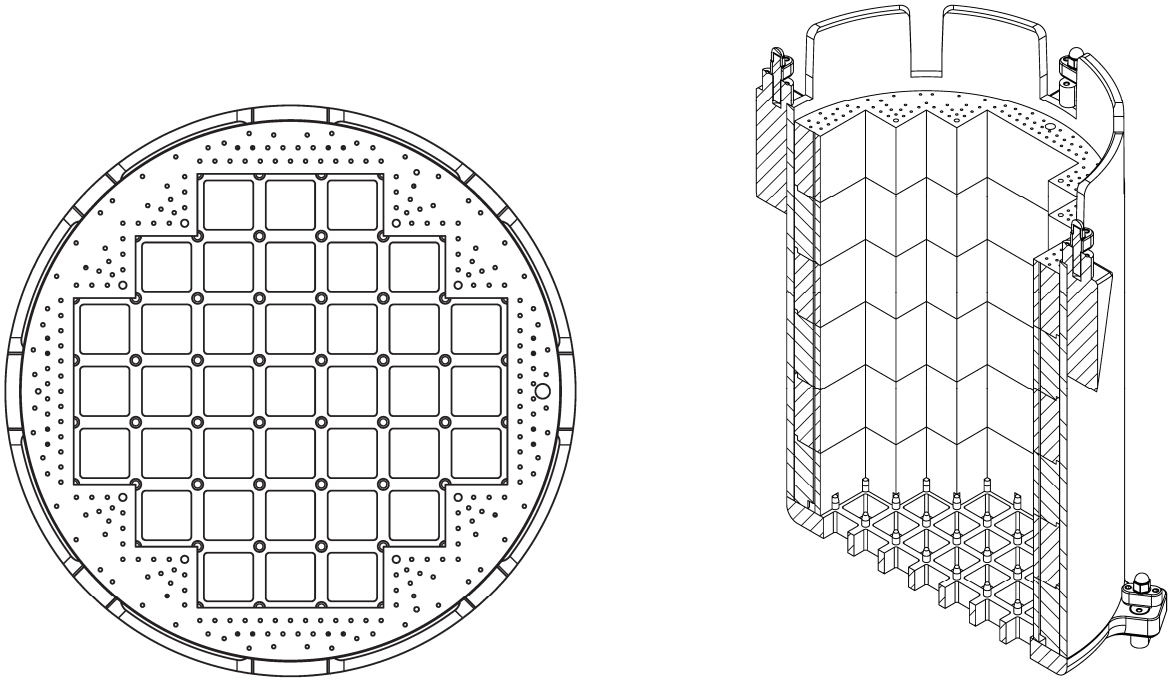

Figure 2-5. Radial Reflector

3.0 Nuclear Code System Description

Studsvik Scandpower developed and maintains the CMS5 code suite, which is an industry standard computer code package for comprehensive neutronic simulation of LWRs. This code package forms the basis of the methodology in the nuclear design and analysis of the NuScale design. The suite is comprised of the lattice physics code CASMO5, the linkage code CMSLINK5 for nuclear data library generation, and the core simulator code SIMULATE5.

3.1 CASMO5

CASMO5 (References 10.1.5 and 10.1.6) is a multi-group two-dimensional (2D) transport theory code used to generate pin cell or assembly lattice physics parameters for LWRs. These parameters include cross sections, nuclide concentrations, pin power distributions, and other nuclear data used for core performance analysis, providing input into the core simulator code SIMULATE5.

CASMO5 solves the 2D neutron transport equation by the Method of Characteristics and can be executed in a number of different energy group structures. The code produces a 2D transport solution based upon heterogeneous model geometry, incorporating the direct microscopic depletion of burnable absorbers, such as gadolinium, into the main calculation. The calculation of isotopic depletion as a function of burnup is performed for each fuel rod and for each region containing a burnable absorber. The Method of Characteristics has been established as a methodology well-suited to solve these types of problems and is incorporated in many of the latest generation lattice physics codes (e.g. CASMO5, HELIOS-2, APOLLO-2, LANCER-02, and WIMS-10).

The CASMO5 nuclear data library consists of 586 energy groups covering a range from 0 to 20 mega electron volts (MeVs). Macroscopic cross sections are directly calculated from the geometries and material properties provided from the code input. Resonance integrals are used to calculate effective absorption and fission cross sections for each fuel rod in the assembly. Dancoff factors are calculated to account for the shadowing effect between different rods in the assembly.

The CASMO5 geometrical configuration consists of a square pitch array containing cylindrical fuel rods of varying composition. Burnable absorbers such as gadolinium, erbium, or integral fuel burnable absorber can be modeled in the code. The lattice structure can be input as a full assembly (symmetric or non-symmetric), or based on octant, quarter, or half-core symmetry. The code input may include burnable absorber rods, cluster control rods, in-core instrument channels, water gaps, and cruciform control rods in the regions separating the fuel assemblies depending on the details of the assembly lattice design.

CASMO5 models the reflector regions directly. The code performs a multi-group neutron transport calculation for a fuel assembly with a reflector region positioned adjacent to one edge of the assembly. CASMO5 calculates cross sections and discontinuity factors for the homogenized reflector regions defined in the model. The cross sections are flux-volume weighted over the reflector region. Discontinuity factors are computed by

calculating the homogenous flux distribution in the reflector, and serve to preserve the group-wise core leakage and the absorption in the reflector. SIMULATE5 uses the reflector cross sections and discontinuity factors to account for neutron leakage and thermalization outside the reactor core.

CASMO5 runs a series of depletions and branch cases to off-nominal conditions in order to generate a neutronic data library for SIMULATE5. These calculations are known as the "SIMULATE5 Case Matrix", which functionalize boron concentration, moderator temperature, fuel temperature, shutdown cooling (isotopic decay between cycles or over long outage times), and CRA positioning with respect to exposure.

CASMO5 is the most recent version of the CASMO lattice code from Studsvik Scandpower. The previous version, CASMO-4, has been approved and used throughout the nuclear industry for many years. Enhancements made to CASMO5 compared to CASMO-4 include:

- Inclusion of the Evaluated Nuclear Data File (ENDF) ENDF/B-VII Cross Section Library. The library update improves accuracy and enhances resonance treatments. A nuclear data library comparison between CASMO5 and CASMO-4 is provided in Table 3-1.
- Enhanced scattering kernel that leads to more accurate Doppler coefficient predictions.
- Quadratic gadolinia depletion model that allows for larger depletion step sizes while maintaining accuracy.
- Improved energy release model that explicitly computes the isotopic energy yields as lattice compositions evolve (compared to fixed energy release models).

Table 3-1. CASMO Library Comparison

	CASMO-4 Library	**CASMO5 Library**
Evaluation	ENDF/B-IV	ENDF/B-VII
Number of Neutron Groups	{{	
Number of Resonance Groups		
Number of Thermal Groups		
Number of Nuclides/Materials		
Number of Actinides		
Number of Fission Products		
Number of Nuclides/Materials with Resonance Data		}}^{2a,c}

CASMO5 output generates edits for the eigenvalue, power distributions, reaction rates, gamma fluxes, gamma detector response, and few-group parameters for use in core calculations. CASMO5 data is processed by CMSLINK5 for use in SIMULATE5 core simulations. This data includes macroscopic cross sections, fluxes, volumes and surface currents, isotopic number densities and microscopic cross sections for the isotopes most important to reactivity. CASMO5 output also includes pin-by-pin power peaking data,

detector cross sections and discontinuity factors, and kinetics parameters for use by SIMULATE5.

3.2 CMSLINK5

CMSLINK5 (Reference 10.1.7) is a data linkage code that processes data from CASMO5 for use by SIMULATE5. The primary purpose of CMSLINK5 is to read the CASMO5 ASCII card image file, functionalize key neutronic variables versus important independent variables, and produce a binary master cross section library for use in SIMULATE5. The code collects the following data from CASMO5 and creates a binary library for SIMULATE5:

- multigroup macroscopic and microscopic nodal cross sections
- multigroup submesh macroscopic cross sections
- discontinuity factors
- fission product data (fission yields and microscopic cross sections)
- detector data (in-core instrument response)
- pin power reconstruction data
- kinetics data (betas, lambdas, and neutron velocities)
- isotopic data
- spontaneous fission data/Alpha-n sources
- decay heat data

Neutronic data are generated by performing CASMO5 fuel assembly base depletions for nominal conditions and several branch depletion cases for off-nominal conditions, where one or multiple state point parameters is changed from the base depletion. Branch cases include changes in soluble boron, moderator temperature, fuel temperature, insertion and removal of BP rods, insertion of CRAs, and isotopic decay after shutdown. The case matrix covers the operating range from room temperature to full power operation. These calculations provide the data needed to functionalize neutronic parameters versus the following independent variables:

- burnup
- moderator density and density history
- soluble boron and soluble boron history
- fuel temperature and fuel temperature history
- control rod assembly type and control rod assembly history
- control rod assembly depletion fraction
- xenon number density

- spacer grid type
- detector type
- shutdown cooling time
- removable burnable absorber

CMSLINK5 functionalizes the data from CASMO5 into a series of single or multi-dimensional table sets (for each unique fuel type). This functionalization allows for linear interpolation in SIMULATE5 to accurately evaluate neutronics data for each node in the reactor core for the appropriate reactor conditions. The functionalization of data in CMSLINK5 depends upon the detailed branch and depletion data generated in CASMO5 that is controlled by the automated and predefined case matrices. The output of the code is a summary of card image file content, segments present in the library, the case matrix functionalization, and tables of the infinite neutron multiplication factor (k-infinity).

3.3 SIMULATE5

SIMULATE5 (References 10.1.8 and 10.1.9) is a three-dimensional (3D), steady-state, nodal diffusion theory, reactor simulator code. The previous code version, SIMULATE-3, has been approved by the NRC and used throughout the nuclear industry for many years. SIMULATE5 is a more detailed and more accurate code. It solves the multi-group nodal diffusion equation (compared to two-group), and contains enhanced capabilities that improve the ability of the code to model axial and radial heterogeneities, and provide more detailed depletion and pin power reconstruction models.

SIMULATE5 employs a hybrid microscopic-macroscopic cross section model that accounts for depletion history effects. The code tracks approximately 60 nuclides during the depletion process; the nuclides have been chosen based on impact to reactivity and safeguard interests. Heterogeneities in the axial direction of an assembly are treated explicitly. A fuel channel is divided into nodes of approximately 10-15 cm. Each node may contain material discontinuities due to enrichment and burnable poison loading, CRAs, spacer grids, etc. To correct for node heterogeneity, the nodes are further divided into sub-nodes, such that each sub-node is materially homogenous. Burnup and nuclide data are stored per sub-node. The one-dimensional multi-group diffusion equation is solved to produce discontinuity factors and flux weighting factors to homogenize cross sections in the axial direction for the larger nodes. The axial homogenization computes averaged cross sections and axial discontinuity factors to reproduce the correct heterogeneous nature of the assembly.

Radially, the assembly is divided into heterogeneous sub-meshes. The submesh model consists of 2D planes comprised of all fuel assemblies and reflector regions. Each plane is divided into NxN rectangular sub-meshes (typically N=5). Each sub-mesh has uniform cross sections that have been homogenized in the axial direction (see above). The output of the sub-mesh calculation is radially re-homogenized cross sections, radial discontinuity factors, and pin power form functions. These derived parameters are then employed in the 3D solution scheme of SIMULATE5.

The reactor power, coolant density, and fuel temperature distributions are coupled in SIMULATE5. The code performs a coupled neutronics/thermal-hydraulics (TH) iteration to determine these distributions. During the iterative process, the reactor power distribution can be considered known, and the coupled problem is reduced temporarily to determining the coolant density and fuel temperature distributions for a fixed power distribution. SIMULATE5 includes a four-equation TH model. Each fuel assembly has an active channel and a number of parallel water channels. In each axial node of a channel, the total mixture mass, steam mass, mixture enthalpy, and mixture momentum balance equations are solved and void fractions are determined by a drift flux model. The pressurized water reactor (PWR) core treats assembly cross-flow by solving the axial and lateral momentum equations. Outside of the core, the TH are calculated from the lower to upper nozzles. All thermodynamic quantities are evaluated by using the modern NIST/ASME steam/water function library.

Previous code versions of SIMULATE used tabulated temperature data to determine nodal fuel temperatures. SIMULATE5 determines the 3D fuel temperature distribution by solving the one dimensional, annular heat conduction equation for the average fuel rod of each node. This calculation is part of the TH module of SIMULATE5. The model calculates the temperature distribution within the fuel rod and the transport of heat from the fuel into the coolant. Feedback from the average fuel temperature is used to calculate cross sections and nodal fuel enthalpy. The fuel temperature model has been validated by the code vendor (Reference 10.1.10).

Output of SIMULATE5 includes nuclear analysis predictions such as CBC, boron worth, reactivity coefficients, CRA worth, shutdown margin, power distributions, and peaking factors.

3.4 CMS5 Model Descriptions

Figure 3-1 depicts the reactor analysis modeling process. The figure describes the inputs and outputs for each code in the modeling process, and the interfaces between the codes. The CMS5 code system is used to numerically model and simulate the reactor core.

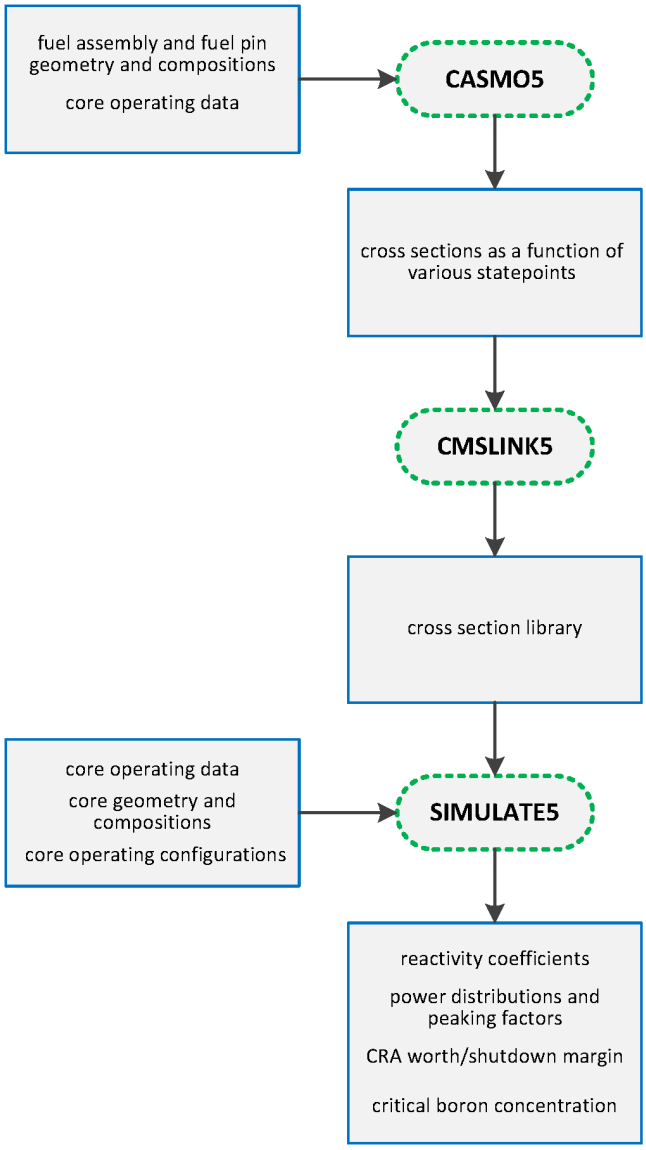

Figure 3-1. CMS5 Code Flow Chart

3.4.1 CASMO5 Assembly Models Description

CASMO5 models are 2D. However, the axial composition of the fuel assemblies and CRAs are modeled in 3D in SIMULATE5 by assigning the appropriate cross sections generated in CASMO5 based on composition within each axial zone. Therefore, cross sections must be determined for all fuel zones in all fuel assembly types (including variations in enrichment and burnable poison concentration), all control rod assembly types, the top and bottom reflectors, and the radial reflector. Fuel compositions consist of the fuel density, the enrichment, and any potential burnable poison loading. Spacer grids are defined by their mass, length, and material composition. The guide and instrumentation tubes and any water gap between fuel assemblies in the reactor core

are included in the CASMO5 model. Axial zoning of the fuel rods is accounted for when generating cross sections in CASMO5. The different axial zones consist of a bottom and top reflector, the central region that is enriched for each particular fuel type, and two blanket regions that contain reduced ^{235}U enrichment compared to the central region. The central region may contain enriched UO_2 or UO_2-Gd_2O_3 fuel pellets.

CASMO5 input includes reactor core operating data as well as fuel assembly, fuel rod geometry, and compositions (Section 2.0). Core operating data input includes, but is not limited to, core average power density, system operating pressure, average fuel temperature, moderator temperature (inlet and average), and core average boron concentration. The geometrical configurations of the fuel rods and CRAs are specified by inputting the radial dimensions of the different parts of those components. Each radial section is then provided a material composition that completely defines each component in 2D. Figure 3-2 depicts the radial composition of a typical NuScale fuel rod. In CASMO5, the fuel rod would be defined by the radius of the pellet along with the composition of the pellet (fuel – UO_2 or UO_2-Gd_2O_3), the inside radius of the clad along with the composition of the area between the pellet and the clad (gap – helium gas), and the outside radius of the clad along with the composition of the clad (clad – M5®).

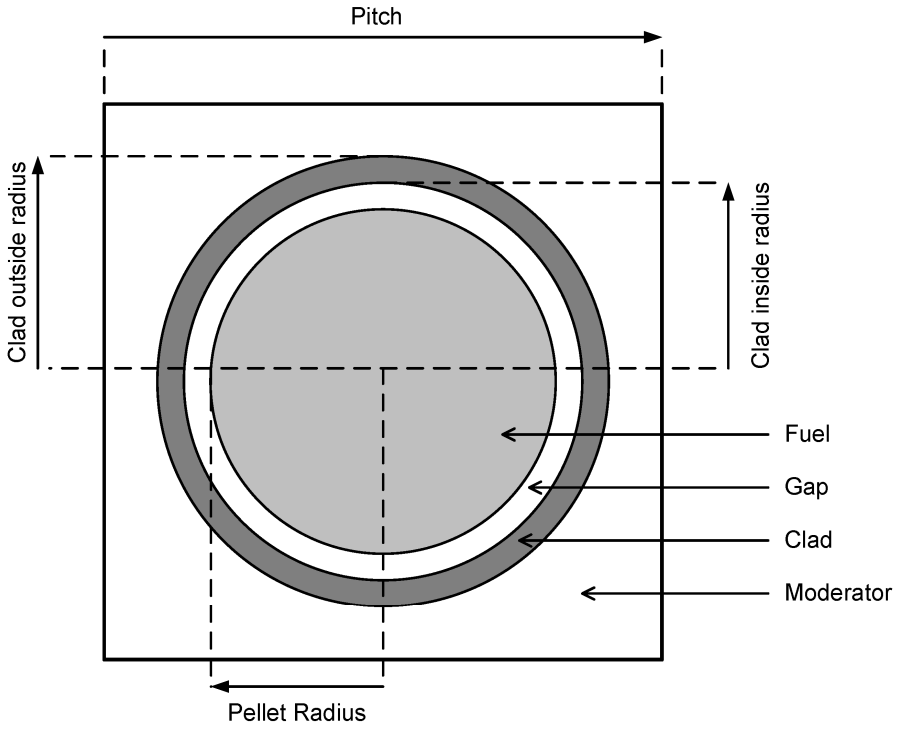

Figure 3-2. Fuel Rod Radial Composition

3.4.2 CASMO5 Reflector Models Description

The top, bottom, and radial regions surrounding the fuel are modeled separately as reflector segments in CASMO5. The CASMO5 reflector model is 2D. SIMULATE5 is a

3D code, where the top and bottom reflectors segments are used to model the material directly above and below the fuel assemblies, and the radial reflector segments span the entire length of the model, as shown in Figure 3-3.

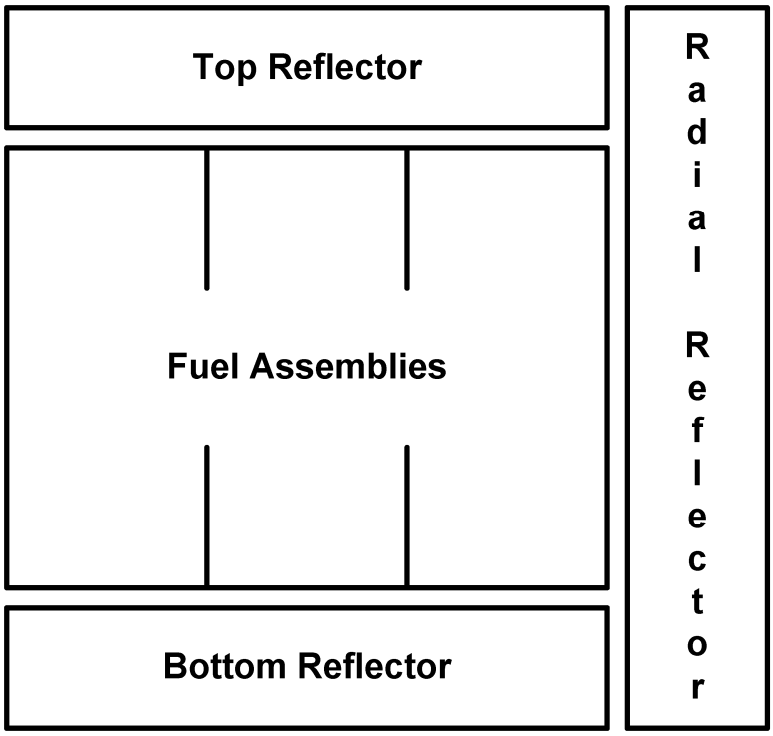

Figure 3-3. CMS5 Three-Dimensional Reflector Model Representation

The reflector is subdivided into segments consisting of a square geometry with sides equal to an assembly pitch. Each of the reflector segments uses the neighboring assembly lattice to drive the flux spectrum in the reflecting region. The layout of the reflector segment is modeled as slabs placed adjacent to the neighboring assembly lattice (Figure 3-4). The 2D reflector in CASMO5 is defined by the slab thicknesses and material compositions that make up the reflector. Reflector compositions are homogenized to account for all material quantities within the individual slab geometries. The neighboring lattice is representative of the fuel in the region immediately adjacent to the reflector. In 2D, the reflector segment boundary opposite the fuel segment is modeled as a black boundary; the reflector segment boundaries adjacent to the fuel/reflector boundary are modeled as mirror boundaries.

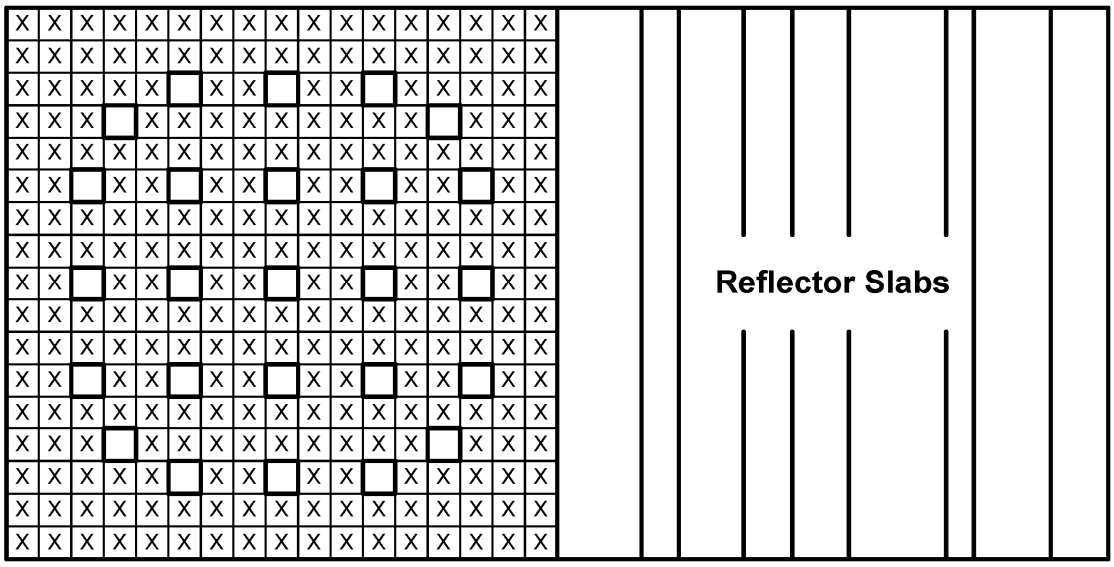

Figure 3-4. CASMO5 Two-Dimensional Reflector Model Representation

The CASMO5 top reflector is modeled from the axial composition of materials in the upper internals and includes the material located above the active fuel. Materials in this region include the fuel rod cladding and guide tubes, the top end plugs, the spacer grid, helium fill gas, the fuel rod internal spring, the top nozzle, the upper core plate, and the coolant. The CASMO5 lower reflector is modeled from the axial composition of materials in the lower internals and includes the material located below the active fuel. Materials in this region include the bottom end plugs, the fuel rod cladding and guide tubes, the bottom nozzle, lower core plate, and coolant.

The NuScale reactor incorporates a stainless steel reflector with cylindrical coolant channels placed throughout. Due to the symmetry of the model, six unique radial reflector segments are modeled in CASMO5 to capture the various components of the reflector region contained within the assembly pitch segment of interest. The modeling scheme used to identify the six unique radial reflector segments is provided in Figure 3-5.

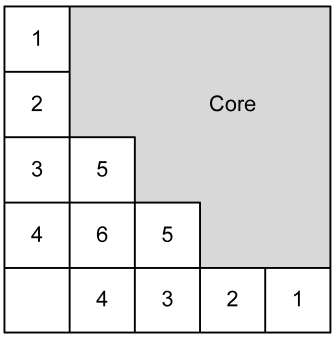

Figure 3-5. SIMULATE5 Radial Reflector Region Diagram

Each of the six radial reflector model segments is defined by thickness and specific homogenized material definitions. Reflector slabs that contain only stainless steel (no coolant) are defined using the stainless steel material composition identifier in CASMO5, and slabs that contain both stainless steel and moderator are defined as a homogenized material composition (by volume fraction) in the code.

Cross sections are generated for the reflector regions with CMSLINK5. The unique reflector segments are then utilized in the SIMULATE5 model when defining the 3D location of the various fuel and reflector segments.

3.4.3 CASMO5 Case Matrix

As discussed in Section 3.1, the CASMO5 case matrix runs a series of depletions and branch cases in order to generate a neutronic data library for SIMULATE5. Multiple depletions are needed to generate cross section data for each fuel assembly type. First, the fuel assembly is depleted at hot full power (HFP) with nominal reactor conditions (base depletion). Moderator temperature, fuel temperature, and soluble boron concentration are set to constant nominal values for the complete depletion. Depletion calculations are also performed at higher and lower moderator temperatures, corresponding to the expected inlet and outlet temperatures of the reactor. For these cases, the fuel temperature and soluble boron concentration are kept at the constant HFP nominal conditions. Next, assembly depletion is performed at HFP nominal conditions, but with a constant soluble boron concentration equal to twice the nominal concentration from the base depletion. Fuel temperature depletion is then performed with a fuel temperature equal to the moderator inlet temperature. The last depletion case is for rodded depletion. Each fuel assembly is depleted up to sixty (60 GWd/MTU) assembly average burnup using the CASMO5 default depletion steps.

Branch cases are permutations to various nominal inputs (e.g., primary conditions or control rod position) that are performed to calculate instantaneous effects. These effects are individually calculated and combined together later to recreate the proper fuel assembly cross sections. The branch cases are executed from the depletion cases at burnups ranging from zero to sixty (0 to 60) GWd/MTU, with the option to deplete to higher burnups when necessary. Branch cases are run for off-nominal moderator temperatures (room temperature to saturation temperature), fuel temperatures (room temperature to fuel melting temperature), soluble boron concentrations one tenth to twenty-four hundred (0.1 to 2400) ppm, and CRA insertions. The CASMO5 case matrix covers the full range of conditions, from startup to normal operations including limiting conditions.

3.5 CMSLINK5 Model

The CMSLINK5 computer program generates multi-group fuel assembly and reflector cross section tables for SIMULATE5. For each fuel assembly and reflector type, data from the CASMO5 card image file are processed into a binary cross section library for input to SIMULATE5.

3.6 SIMULATE5 Model

SIMULATE5 performs 3D model simulations of the reactor core. Code input includes the nuclear data library produced by CASMO5 and CMSLINK5, reactor core configurations, core geometry and material compositions, and core operating data.

In SIMULATE5, the core is broken into 25 axial nodes and a 2x2 nodal mesh for each assembly. The stainless steel heavy reflector is modeled around the fuel (described in Section 3.4.2). Each fuel assembly is given a unique identifier corresponding to the fuel batch type, and the assemblies are arranged according to the core loading pattern design.

Geometrically, the core model is defined by the placement of the assemblies and their axial composition (including top and bottom reflector), the position and composition of the radial reflector, the placement of the CRAs and their axial composition, and the composition and axial position of the spacer grids. Cross sections from the nuclear data library prepared by CMSLINK5 are assigned axially, per node, for each assembly type. Each assembly is assigned cross sections for the bottom reflector region, the active fuel region, and the top reflector. The radial reflector surrounding the fuel is also assigned cross sections from the nuclear data library. CRAs are modeled according to their step spacing and number of steps for full withdrawal, their axial composition, and their placement. CRA axial position can be specified by bank or by individual rod. Spacer grids are defined by their composition and axial position in the core. Core operating data input includes the thermal power output, flow density, system pressure, and core average temperature.

Input to the TH model includes mechanical design data, spacer grid area and flow losses, core bypass flow fraction, and a temperature versus power curve. The mechanical design data include assembly, fuel rod, and guide tube properties (geometry, loss coefficients). The reactor core is modeled axially from the bottom of the lower core plate to the top of the upper core plate. The axial zones are chosen so that each axially unique region of the assemblies has its own axial zone. The TH and mechanical design data are used to determine the active coolant flow and flow area, flow rates, and pressure drops. These data are also used in the energy construction equations, and the determination of nodal fuel temperature distributions.

Detailed 2D and 3D edits are available from SIMULATE5 output. The SIMULATE5 model is used to perform base depletion calculations and predict cycle length, assembly and pin-by-pin power distributions, peaking factors, critical boron concentration, reactivity coefficients, CRA worth, shutdown margin, and other physics parameters that may be used for nuclear design and analysis.

3.7 CMS5 Software Configuration Control

The software development of the CMS5 code suite was performed by Studsvik Scandpower and was delivered to NuScale as a compiled, commercial software package. Verification and validation (V&V) activities were performed by the code developer prior to delivery of the software package to demonstrate that the codes can

correctly perform the functions intended and accurately predict results. The software package delivery was accompanied by installation test cases and user manual, methodology, and version change documentation, as applicable to each code in the software package. Upon delivery, configuration control is initiated and the software package was subjected to appropriate controls within the NuScale Quality Assurance (QA) program (Reference 10.1.32). In addition to the V&V activities performed by Studsvik Scandpower, software validation is performed for applications and use specific to NuScale. The QA program is compliant with references 10.1.11 and 10.1.12. The QA program governs activities associated with acquisition of commercial grade software, configuration control, validation, and dedication of the CMS5 code suite.

The commercial software was placed under configuration control and installation testing was performed using the test case inputs and reference solutions included with the software delivery. This installation testing ensures that the software has been installed properly by comparing solutions of the test case inputs to the reference solutions and ensuring there are no unexpected differences in the results. After successful installation, validation and benchmarking demonstrates the codes perform the functions intended for NuScale applications. All software used to support this topical report has been appropriately controlled under the NRC approved NuScale QA program.

4.0 Code Validation Statistical Methodology

Statistical methods were used in the determination of the bias and tolerance limit associated with the ability of the CMS5 code system to predict reactor core physics parameters. The data evaluated consisted of absolute and/or relative differences between experimental, operational, or Monte Carlo N-Particle, version 6 (MCNP6) calculated data (collectively referred to as benchmark values), and CMS5 calculated data.

The CMS5 bias and tolerance limit determinations utilized statistical methods as part of a systematic process. This process involves:

1. data processing
2. normality testing
3. bias and tolerance limit determination

4.1 Data Processing

For each parameter being examined, the CMS5 results and measured data are collected. Depending on the number of data points and nature of the parameter, the data may be formed into one or more groups. Data grouping is further explained when discussing the analysis of data.

The data used for statistical analysis is the difference between measured values (experimental or MCNP6, referred to collectively as benchmark data) and the CMS5 values. For a positive parameter, a negative absolute difference value implies that CMS5 is over predicting the parameter while a positive value implies CMS5 is under predicting; the opposite is true for a negative parameter. In many cases a relative difference is developed and is expressed as a percent change relative to the CMS5 value.

4.2 Normality Testing

The differences between the benchmark data values and the CMS5 results were evaluated to determine normality. Multiple tests were employed, including both graphical and analytical techniques as given in the following list.

- histogram of the data
- normality plot of the data
- Shapiro-Wilk normality test
- D'Agostino D' ("D prime") normality test
- Anderson-Darling normality test

The normality plot is a graphical technique that involves the plotting of ordered data against the normal order statistics (Reference 10.1.16). For normally distributed data,

the plotted data approximates a straight line on a normality plot. These plots are useful in identifying various types of deviations from normally distributed data.

Three normality tests were selected to ensure that at least two tests can be applied to each dataset. The Shapiro-Wilk is a common normality test, but is restricted to sample sizes of 50 or less and it may exhibit abnormal behavior when the data includes duplicates. The D'Agostino D' normality test is a modification of the Shapiro-Wilk test that is not restricted by the size of the dataset. The Anderson-Darling test is also commonly used for normality testing.

4.3 Bias and Tolerance Limit Determination

For normally distributed data, the bias is represented by the average deviation between the benchmark data and CMS5. The bias may be presented as either the absolute difference or fractional difference. The tolerance limit for normally distributed data is represented by the uncertainty on the average deviation between the benchmark data and CMS5. This limit is determined based on the method discussed in Section 4.3.1. This method is a one-sided tolerance limit method.

For data sets that do not demonstrate a normal distribution, a non-parametric method is used to determine one-sided tolerance limits. These limits on the deviation between benchmark data and CMS5 are determined based on the method discussed in Section 4.3.2.

4.3.1 One-Sided Tolerance Limit for a Normal Distribution

For normally distributed data, the bias and tolerance limit are determined based on a one-sided tolerance bound taken from Section 4.6.3 of Reference 10.1.18. This technique determines a lower or upper bound based on the sample mean and standard deviation. This lower or upper limit is given by the following equation:

$$T = \bar{x} \pm k\sigma \qquad \text{Equation 1}$$

Where,

T = tolerance limit

$\bar{x}$ = mean of the observations

σ = standard deviation of observations

k = tolerance factor for one-sided tolerance limit

These factors are approximated by the following formulas given in Section 7.2.6.3 of Reference 10.1.16 for N is greater than 4:

$$k = \frac{z_p + \sqrt{z_p^2 - ab}}{a} \qquad \text{Equation 2}$$

Where,

$$a = 1 - \frac{z_\gamma^2}{2(N-1)}$$ Equation 3

$$b = z_p^2 - \frac{z_\gamma^2}{N}$$ Equation 4

z_γ = critical value for normal distribution and confidence γ

z_p = critical value for normal distribution and proportion p

N = number of observations

4.3.2 Non-Parametric Tolerance Limits

If the data set does not demonstrate a normal distribution, a non-parametric or distribution-free technique is used to determine the one-sided tolerance limits. This technique is based on the distribution-free tolerance limits discussed in Section 7.2.6.4 of Reference 10.1.16 and further detailed in Section 5.3 of Reference 10.1.18.

This technique involves ordering the data from smallest to largest and then selecting the value data from the sample value that represents an appropriate limit. The selection from the sample is based on a desired proportion of the population to which the limit is applicable, the desired confidence, and the sample size. This selection involves determining a number of extremes (largest and smallest values in a sample) to consider.

For a lower tolerance limit (LTL), the value of ν identifies the number of the ordered sample to use to represent the lower tolerance limit. For an upper tolerance limit (UTL), the value from an ordered sample to be used as the UTL is the n-ν+1 value.

These subroutines utilize the following basic equation:

$$1 - \alpha = B_{cdf}(n - \nu, n, p)$$ Equation 5

Where,

1 – α = Confidence on the tolerance limit for given proportion p of the population for sample size *n* and number of extremes *ν*.

B_{cdf} = Cumulative binomial distribution at *n-ν* for n trials and a probability of success *p*.

Equation 5 can be rearranged based on an inverse binomial function to give the value of ν directly:

$$\nu = n - InvB_{cdf}(n, p, 1 - \alpha)$$ Equation 6

Where,

$InvB_{cdf}$ = Inverse cumulative binomial distribution for n trials, a probability of success *p*, and confidence 1 - α.

In those instances that the underlying data has uncertainties, two times the uncertainty of a selected ordered data based on the value of ν will be applied in the conservative direction.

4.4 Use of Measurement and Monte Carlo Uncertainties

The data utilized in the bias and tolerance limit determinations have associated uncertainties in the form of experimental uncertainties, such as measurement errors, or uncertainties associated with the Monte Carlo method used in MCNP6. These uncertainties are addressed in the bias and tolerance limit determinations. Determination of uncertainties depends on whether or not the data set is normally distributed. When data are not normally distributed, the uncertainties of the selected bounding values (from the non-parametric method of determining the tolerance limits) are used directly in the tolerance limit determination as discussed at the end of Section 4.3.2. When the data are normally distributed, uncertainties will be determined using weighting factor on the determination of the mean. The use of a pooled variance accounts for the impact of these uncertainties on the standard deviation and tolerance limit.

When the data set is normally distributed, the bias and tolerance limit are based on the mean and standard deviation of the data. Therefore, the uncertainties on the data are accounted for in the determination of these quantities. For this application, uncertainties in the data are per Reference 10.1.20. The measured and/or calculation uncertainties of the individual values are utilized as weights to modify the usual mean and standard deviation formulas to the following:

The weighted mean is given by:

$$\bar{x} = \frac{\sum \frac{x_i}{\sigma_i^2}}{\sum \frac{1}{\sigma_i^2}} \qquad \text{Equation 7}$$

Where,

$\bar{x}$ = Weighted mean of the x_i values

x_i = the i^{th} value of x

σ_i = Uncertainty on the value of x_i

The weighted standard deviation is given by:

$$\sigma_w = \sqrt{\frac{\frac{1}{n-1}\sum_{i=1}^{n}\frac{(x_i-\bar{x})^2}{\sigma_i^2}}{\frac{1}{n}\sum_{i=1}^{n}\frac{1}{\sigma_i^2}}} \qquad \text{Equation 8}$$

Where,

σ_w = weighted standard deviation

n = number of x_i values in dataset

In addition, the uncertainties on the x_i values are accounted for using a pooled variance that is determined based on the following formulas.

The average total uncertainty is given by:

$$\bar{\sigma} = \sqrt{\frac{n}{\sum\frac{1}{\sigma_i^2}}} \qquad \text{Equation 9}$$

The square root of the pooled variance is given by:

$$S_P = \sqrt{\sigma_w^2 + \bar{\sigma}^2} \qquad \text{Equation 10}$$

The above evaluations are performed as a means of capturing the effect of the measurement or calculation uncertainties in the final bias and tolerance limit determinations. These quantities are not utilized when performing other statistical analyses described in Sections 4.1 and 4.2.

5.0 CMS5 Code Validation Benchmarking

Software validation of CMS5 is composed of benchmarking against measured data from experimental reactors, commercial reactors, and simulations using higher-order codes (also known as code-to-code benchmarking). The collective benchmarks are used to evaluate the ability to accurately predict key physics parameters used in the design and analysis of the reactor core. Comparisons with experimental reactors provide validation of basic reactor parameters, such as core reactivity and pin fission rates or relative power. Commercial reactor comparisons validate the broader set of key physics parameters, including core reactivity, critical boron concentration, power distribution, CRA worth, and reactivity feedback coefficients. The empirical data benchmarks are selected to have similar variables that are important to neutronic characteristics of the reactor core such as lattice configurations and operating conditions. Some key characteristics of the NuScale design not found in the empirical data benchmarks are core size and a heavy reflector. Because no data exist for comparison of small reactor designs similar to the NuScale reactor core, CMS5 is benchmarked against the higher-order code MCNP6. In this way, the core geometry and conditions within the intended range of operation can be validated for the NuScale design.

5.1 Higher-Order Code Benchmarking with MCNP6

A variety of critical experiments, experimental reactors, and commercial reactors were benchmarked to collectively address the primary features of the NuScale design. However, the individual sets of empirical data available do not fully represent the NuScale design. Consequently, higher-order code benchmarking was performed to further assess the ability of the CMS5 code suite to perform relevant reactor physics calculations for the NuScale design. MCNP6, the latest release of MCNP, was selected as the higher-order code. MCNP6 is an improvement upon MCNP5 with enhancements including the addition of the Evaluated Nuclear Data File, Version B-VII.1 (ENDF/B-VII.1) data library and the ability to model a wider range of particles and energies (Reference 10.1.19).

A series of higher-order benchmarks (henceforth referred to as code-to-code benchmarks) was performed comparing results obtained in MCNP6 with that of CMS5. A variety of analyses were selected including k_{eff}, CBC, DBW, MTC, ITC, CRA worth, CRA bank worth, and power distribution (including pin peaking, assembly radial peaking, AO, and axial power shape). Each of these analyses was performed for both hot zero power (HZP) and HFP conditions. The HZP cases consider an isothermal Cycle 1 clean core. The HFP cases considered Cycles 1 through 4 at multiple exposure steps within the cycle.

5.1.1 MCNP6 Software Configuration

The software development of the MCNP code was performed by Los Alamos National Laboratory and was delivered to NuScale as a commercial software package. V&V activities were performed by the code developer prior to delivery of the software package to demonstrate that the code can correctly perform the functions intended and accurately predict results. The software package delivery was accompanied by installation test

cases, a user manual, version change documentation, and other relevant reference documents. Upon delivery, the code was placed under configuration control and the software package was subjected to appropriate controls within the NuScale QA program (Reference 10.1.32), compliant with references 10.1.11 and 10.1.12. The QA program governs activities associated with acquisition of commercial grade software, configuration control, validation, and dedication of the MCNP6 code.

Once under configuration control, installation testing was performed using the test case inputs and reference solutions included with the software delivery. This installation testing ensures that the software has been installed properly by comparing solutions of the test case inputs to the reference solutions and ensuring there are no differences in the results.

5.1.2 MCNP6 Model Description

The MCNP6 code-to-code benchmarking models are divided into three groups: the HZP model, the HFP explicit fuel isotopics model, and the HFP assembly average isotopics models. The HZP model is developed for Cycle 1 at isothermal beginning of cycle (BOC) conditions. The HFP explicit fuel isotopics model is developed for Cycle 1 at BOC. The HFP assembly average isotopics models are developed for Cycles 1 through Cycle 4 at multiple exposure steps throughout each cycle. The vast majority of the MCNP6 model is identical between the HZP and HFP models; however some of the simplifications of the Cycle 1 BOC explicit models do not apply to the HFP assembly average isotopics models, which include complexities such as depletion and assembly shuffling.

5.1.2.1 MCNP6 Hot Zero Power Model Description

The HZP model is developed consistent with the CMS5 model, and serves as the starting point for all subsequent model variations.

The geometry and material compositions defined in the MCNP6 model represent an explicit 3D rendering of the actual dimensions and materials comprising the lower portion of the NuScale reactor module as described in Section 2.0. Where modeling simplifications are required, every reasonable effort is made to ensure the material mass is conserved and the neutronic impact of the simplification is minimal. The primary components of the MCNP6 model are shown in Figure 5-1 and Figure 5-2.

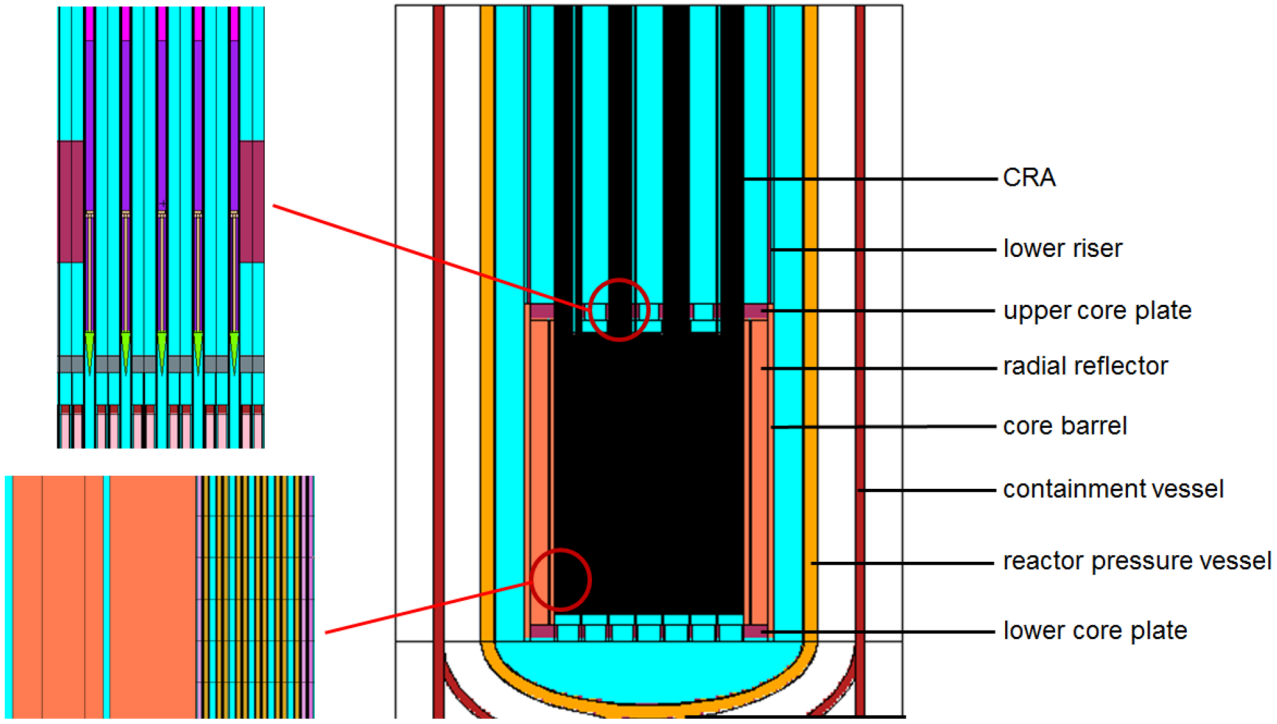

Figure 5-1. MCNP6 Model Elevation View

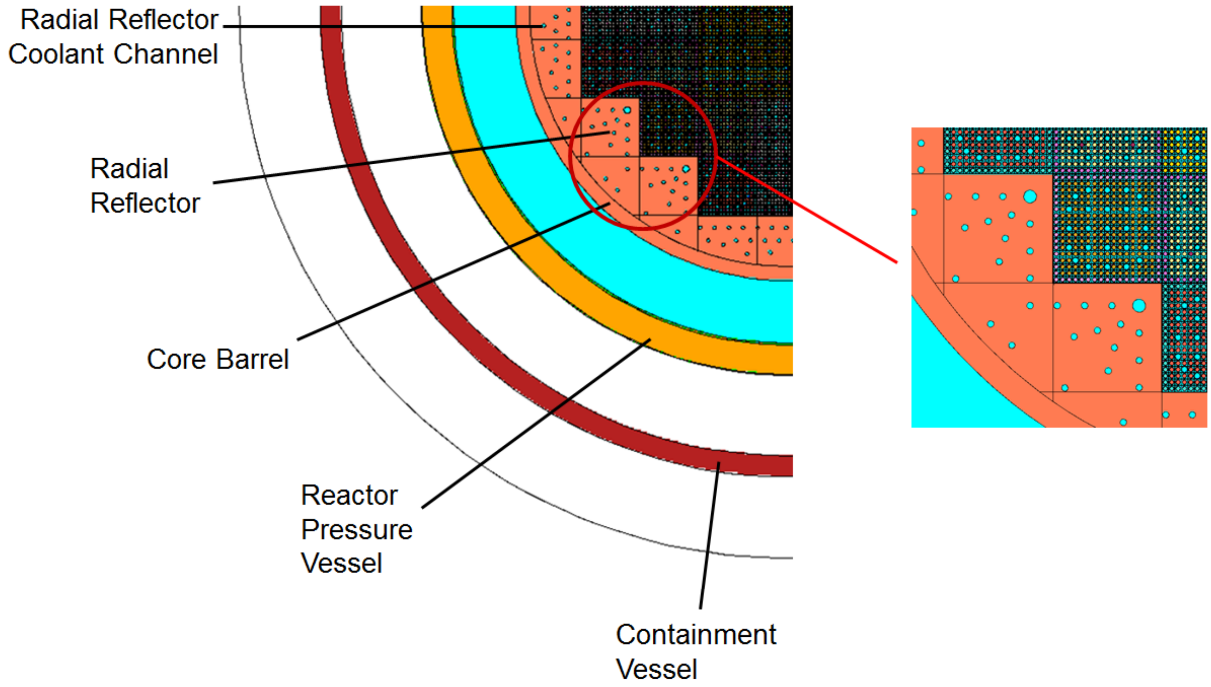

Figure 5-2. MCNP6 Model Top View

The temperature effects in the model are accounted for in the cross section library evaluation temperature, free-gas thermal treatment temperature, and the S(α,β) thermal scattering data temperature. The MCNP6 standard data libraries (ENDF/B-VII.1) have cross sections processed at selected temperatures; therefore it was necessary to generate a temperature-dependent cross section library at HZP temperature conditions.

5.1.2.2 MCNP6 Hot Full Power Explicit Fuel Isotopics Model Description

The HFP explicit fuel isotopics MCNP6 model is an adaptation of the HZP MCNP6 model described in Section 5.1.2.1. The HFP model is developed for Cycle 1 BOC and only differs from the HZP model in the temperature effects in the model and critical boron concentration. The average moderator density, average moderator temperature, and average fuel temperature are defined consistent with the corresponding CMS5 model. The temperature effects in the model are accounted for in the cross-section library evaluation temperature, free-gas thermal treatment temperature for each cell, and the S(α,β) thermal scattering data temperature for hydrogen in light water.

5.1.2.3 MCNP6 Hot Full Power Assembly-Averaged Fuel Isotopics Models Description

The HFP assembly-averaged smeared fuel isotopics MCNP6 models are an adaptation of the HZP MCNP6 model described in Section 5.1.2.1. The HFP smeared models are developed for Cycle 1 through Cycle 4 at multiple exposure steps throughout each cycle. Some modeling simplifications are necessary in the HFP smeared models to ensure a proper code-to-code comparison due to the intrinsic complexities in the models related to depletion, radial and axial temperature variation, and assembly shuffling.

Each of the fuel assemblies in the MCNP6 model is defined independently in order to accurately track isotopic and fuel depletion characteristics with consideration to assembly shuffling and discharge. Furthermore, each fuel rod is divided into twenty-five (25) axial nodes, corresponding to the axial nodalization of the SIMULATE5 model, to accommodate the axial isotopic changes in the fuel. Figure 5-3 shows a detailed representation (not to scale) of the various components of the fuel rod and the axial nodalization of the fuel.

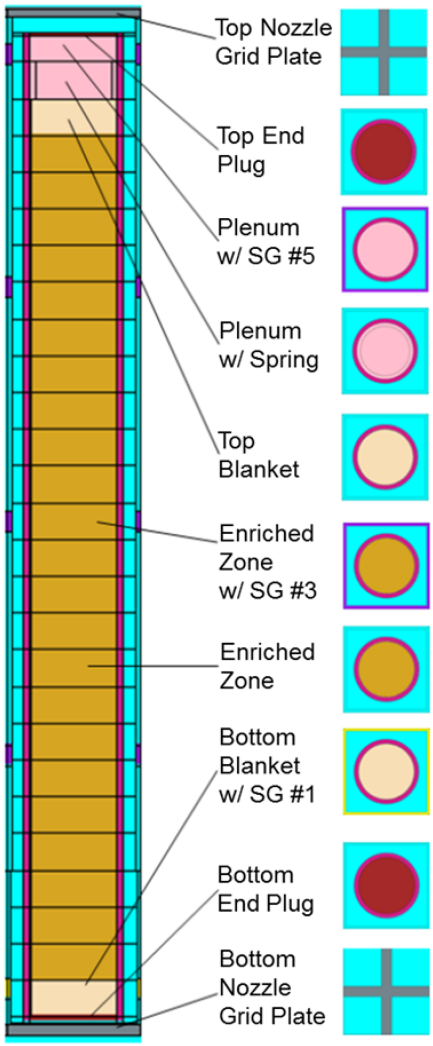

Figure 5-3. MCNP6 Model Fuel Rod Detail

MCNP6 has no straight-forward method of determining the changes to the fuel material as a function of exposure throughout the cycles examined in the HFP analyses. For the purpose of this code-to-code benchmark evaluation and to ensure a consistent basis for the MCNP6 and CMS5 models, the isotopic depletion and buildup data provided by CMS5 is utilized as the basis for the fuel materials utilized in the MCNP6 model.

The isotopic data are defined on an assembly average basis for each of the fuel assembly locations for twenty-five (25) axial nodes. The smeared assembly average isotopic data for a given node are applied to each fuel rod cell, with the exception of fuel rod locations that contain burnable poison. The integral burnable poison fuel rod locations are assigned explicit fuel material compositions to ensure the explicit location of the BP within a given assembly is properly modeled between the two codes. This modeling methodology does not capture the full effects of the power or exposure gradients across a fuel assembly, or the resulting variation of isotopic composition from

one pin to the next. Parameters that are more directly influenced by the ability of the model to capture the radial variation of fuel isotopics within an assembly are therefore more appropriately modeled using an explicit isotopic representation. However, the HFP assembly-averaged smeared isotopic fuel models are a reasonable approximation and provide a basis for additional benchmarking comparisons of many parameters.

For each case, a corresponding CBC, average moderator density, average moderator temperature, and average fuel temperature are defined consistent with the corresponding CMS5 model. The temperature effects in the model are accounted for in the cross section library evaluation temperature, free-gas thermal treatment temperature for each cell, and the S(α,β) thermal scattering data temperature for hydrogen in light water. These temperature effects are applied to the model on a case-by-case basis for the HFP temperature conditions representative of the given cycle and exposure step of interest.

The MCNP6 HFP smeared benchmark models rely on the CMS5 model for fuel isotopics; use of the same material composition definitions between the two codes is an important component of ensuring consistent MCNP6 and CMS5 models for direct code comparison purposes.

5.1.3 Higher-Order Benchmark CMS5 Model Description

The CMS5 model used for the code-to-code comparisons borrows from the CMS5 base model (described in Sections 3.4 through 3.6) making specific modifications to match the MCNP6 comparison model and perform the calculation of interest. For the CMS5 HZP model, the TH conditions are assumed to be isothermal. For the CMS5 HFP models, the TH modeling options are turned off in order to match the MCNP6 models, and the fuel temperature, moderator temperature and moderator density are explicitly defined to be consistent with what is modeled in MCNP6 for each case.

Additionally, the fission product concentration is set to zero for all HZP model comparisons and for HFP model comparisons where the fuel composition is defined explicitly in the MCNP6 model. For the majority of the HFP model comparisons where the fuel composition in the MCNP6 model is defined using smeared assembly average data, the fission product concentration in the CMS5 model is locked at the HFP conditions, all rods out (ARO), equilibrium fission product concentration, which is used in the corresponding HFP MCNP6 model. Finally, for power distribution comparisons only, the heat produced from fission is assumed to be deposited 100 percent locally in the fuel rod to allow for direct comparison between the MCNP6 and SIMULATE5 predicted powers.

5.1.4 Results

The code-to-code benchmarking results are presented in the following section. The results are separated into two groups: standard statistical results, and bias and tolerance limit results. The standard statistical results presented in Section 5.1.4.1 represent a summary of the basic mean, standard deviation, minimum, and maximum values for all of the parameters of interest. The bias and tolerance limit results presented in

Section 5.1.4.2 are limited to parameters for which adequate data is available and of interest to perform a bias and tolerance limit analysis.

5.1.4.1 Standard Statistical Results

The standard statistical results for the code-to-code benchmark comparison are summarized in this section. The dataset analyzed for the HZP model is limited in comparison to that of the HFP models. As stated previously, the HZP model is restricted to Cycle 1 BOC only to allow for explicit fuel isotopic definitions. In contrast, the HFP smeared models include analyses performed for Cycle 1 through Cycle 4 at multiple exposure steps as allowed by utilizing smeared fuel isotopic data from the CMS5 model.

The complete set of code-to-code benchmark comparison results is summarized quantitatively in Table 5-1 and Table 5-2. The information presented in Table 5-1 and Table 5-2 is the difference between the MCNP6 values and the CMS5 values. Therefore, for a positive parameter a negative value implies that CMS5 is over-predicting the value while a positive value implies CMS5 is under-predicting; the opposite is true for a negative parameter. In some instances the relative difference is examined and is expressed as a percent difference. In some instances the data set is limited to a single point. In these instances, the value presented in the column labeled “mean” represents the value of that singular point; consequently no standard deviation is provided. For parameters better presented graphically, figures are provided.

Table 5-1. HZP Code-to-Code Benchmark Comparative Results

Value	HZP Model	
	Mean	Standard Deviation
CBC	{{	
DBW		
CRA Worth		
CRA Bank Worth		
Assembly Radial Peaking		
AO		
Axial Root Mean Square (RMS) Error		}}^{2a,c}
Pin-to-Box Ratio	{{	}}^{2a,c}

The data summarized in Table 5-2 utilize the HFP assembly-averaged smeared isotopic fuel models (with the exception of the pin-to-box data that utilizes the HFP explicit fuel isotopics model). The assembly-averaged smeared isotopic modeling methodology does not capture the full effects of the power or exposure gradients across a fuel assembly, or the resulting variation of isotopic composition from one pin to the next. The HFP assembly-averaged smeared isotopic fuel models are a reasonable approximation and provide a basis for additional benchmarking comparisons of many parameters. The full set of data is provided for reference; however, not all data are utilized as input into the MCNP6 tolerance limits described in Section 5.1.4.2.

Table 5-2. HFP Code-to-Code Benchmark Comparative Results

Value	HFP Model	
	Mean	**Standard Deviation**
k_{eff}	{{	
CBC (ppm)		
DBW		
MTC (pcm/°F)		
ITC (pcm/°F)		
CRA Worth		
CRA Bank Worth		
Assembly Radial Peaking		
AO		
Axial RMS Error		
Pin-to-Box Ratio		}}^{2a,c}

The code-to-code benchmark comparison of axially averaged relative power distribution results are presented in Figure 5-4 and Figure 5-5. The figures display the excellent agreement between the two codes.

{{

}}2a,c

Figure 5-4. Code-to-Code Power Distribution Results – Cycle 1 BOC 2D Top View

{{

}}2a,c

Figure 5-5. Code-to-Code Power Distribution Results – Cycle 1 BOC 2D Side View

The code-to-code benchmark comparison of axial power shape results for Cycle 1 at BOC, Middle of Cycle (MOC), and End of Cycle (EOC) is presented in Figure 5-6. The axial power shape results in Figure 5-6 utilize the HFP assembly-averaged smeared isotopic fuel model. Although the smeared modeling methodology does not fully capture radial isotopic variation within a given assembly, the methodology does explicitly capture axial isotopic variation within the assembly. Therefore, the axial power shape results presented in Figure 5-6 represent a reasonable approximation. The figure displays agreement between the two codes. The two codes display the same axial power shape and shift in power toward the top half of the core with longer exposure.

{{

}}2a,c

Figure 5-6. Code-to-Code Axial Power Results – Cycle 1

5.1.4.2 Bias and Tolerance Limit Results

The following sections present the results of a bias and tolerance limit analysis of the code-to-code comparison results. The statistical analysis performed is described in Section 4.0 and results in the determination of bias and tolerance limits for each examined parameter.

5.1.4.2.1 Differential Boron Worth

The MCNP6 and CMS5 DBWs were calculated and a comparison was performed to determine a bias and tolerance limits for CMS5.

A total of {{ }}2a,c calculated DBWs were used for the determination of DBW bias and tolerance limits for CMS5. The MCNP6 and CMS5 DBW results are presented in Table 5-3.

The %ΔDBW data from Table 5-3 demonstrated normality when tested as noted in Section 4.2. Based on this validation, the bias and tolerance limits are determined

utilizing the technique applicable to normally distributed data described in Section 4.3.1. These determinations used a weighted mean and weighted standard deviation per Section 4.4. These values are shown in Table 5-4 along with the bias and tolerance limit determinations for the %ΔDBW values from Table 5-3. The results give a bias of {{ }}^{2a,c}. This bias represents the relative difference between MCNP6 and CMS5. The uncertainty on the bias was determined to be {{ }}^{2a,c}.

Table 5-3. MCNP6 and CMS5 DBW comparison data

Initial Boron Concentration (ppm)	Burnup (GWd/MTU)	Cycle	MCNP6 DBW (pcm/ppm)	CMS5 DBW (pcm/ppm)	Relative Difference(1) (%)
{{					
					}}2a,c

Notes: (1) The percent deviations are multiplied by negative one in order to preserve the correct implied under/over prediction of the CMS5 values in comparison to MCNP6. This sign change must be reversed when the bias or tolerance limits are applied to a CMS5-calucated value. The number of decimal places given for the results in this table has been reduced for presentation purposes. The resulting relative difference may not be calculated exactly as given in this table as a result of this rounding.

Table 5-4. Bias and tolerance limit for MCNP6 to CMS5 %ΔDBW MCNP6 to CMS5 comparison

Parameter	Value
Bias (Weighted Mean)	{{
Weighted Standard Deviation	
LTL	
UTL	}}^{2a,c}

5.1.4.2.2 Isothermal Temperature Coefficient

The MCNP6 and CMS5 ITCs were calculated and a comparison was performed to determine a bias and tolerance limits for CMS5. The ITC was determined by calculating the reactivity change between two overall core temperatures. A 50 degree F temperature change was used in these determinations.

A total of {{ }}^{2a,c} calculated ITCs were used for the determination of ITC bias and tolerance limits for CMS5. The MCNP6 and CMS5 ITC results are presented in Table 5-5. The MCNP6 uncertainties on the ITC were {{ }}^{2a,c} and results in corresponding uncertainties on the ITC difference (ΔITC).

The ΔITC information from Table 5-5 was evaluated using the normality tests noted in Section 4.2. The results demonstrated that the data are normally distributed. Based on this result, the bias and tolerance limits are determined utilizing the technique described in Section 4.3.1. These determinations used a weighted mean and weighted standard deviation based on Equation 7 through Equation 10 in Section 4.4. These values are shown in Table 5-8 along with the bias and tolerance limit determinations for the ΔITC values from Table 5-7. The results indicate that CMS5 has a tendency to {{ }}^{2a,c} based on a bias of {{ }}^{2a,c}. This bias represents the absolute difference between the MCNP6 and CMS5 ITC. The uncertainty on the bias was determined to be {{ }}^{2a,c}.

Table 5-5. MCNP6 and CMS5 ITC comparison data

Initial Temperature (°F)	Burnup (GWd/MTU)	Cycle	MCNP6 ITC (pcm/°F)	CMS5 ITC (pcm/°F)	Absolute Difference(1) (pcm/°F)
{{					
					}}2a,c

Notes: (1) The number of decimal places given for the results in this table has been reduced for presentation purposes. For some values, the resulting absolute difference may not be calculated exactly as given in this table as a result of this rounding.

Table 5-6. Bias and tolerance limit for MCNP6 to CMS5 ΔITC comparison

Parameter	Value
Bias (Weighted Mean)	{{
Weighted Standard Deviation	
LTL	
UTL	}}^{2a,c}

5.1.4.2.3 Moderator Temperature Coefficient

The MCNP6 and CMS5 MTCs were calculated and a comparison was performed to determine a bias and tolerance limits for CMS5.

A total of {{ }}^{2a,c} calculated MTC values were used for the determination of MTC bias and tolerance limits for CMS5. The MCNP6 and CMS5 MTC results are presented in Table 5-7.

The difference in the MTC data (ΔMTC) from Table 5-7 was evaluated using the normality tests noted in Section 4.2. The results of these tests demonstrate that the data are normally distributed. Based on this result, the bias and tolerance limits are determined utilizing the technique described in Section 4.3.1. These determinations used a weighted mean and weighted standard deviation based on Section 4.4. These values are shown in Table 5-8 along with the bias and tolerance limit determinations for the ΔMTC data from Table 5-7. The results indicate that CMS5 has a tendency to {{ }}^{2a,c} based on a bias of {{ }}^{2a,c}. This bias represents the absolute difference between the MCNP6 and CMS5 MTC. The uncertainty on the bias was determined to be {{ }}^{2a,c}.

Table 5-7. MCNP6 and CMS5 MTC comparison data

Initial Temperature (°F)	Burnup (GWd/MTU)	Cycle	MCNP6 MTC (pcm/°F)	CMS5 MTC (pcm/°F)	Absolute Difference(1) (pcm/°F)
{{					
					}}2a,c

Notes: (1) The number of decimal places given for the results in this table has been reduced for presentation purposes. For some values, the resulting absolute difference may not be calculated exactly as given in this table as a result of this rounding.

Table 5-8. Bias and tolerance limit for MCNP6 to CMS5 ΔMTC comparison

Parameter	Value
Bias (Weighted Mean)	{{
Weighted Standard Deviation	
LTL	
UTL	}}^{2a,c}

5.1.4.2.4 Peak Relative Linear Power

In order to determine a CMS5 bias and tolerance limit applicable to the peaking factor F_Q, a comparison was performed between MCNP6 and CMS5 RLP values. This comparison examined the differences with RLP values that were within {{ }}^{2a,c} of the peak CMS5 RLP value. The RLP values derived are specific to the NuScale reactor design at HFP conditions at the beginning of Cycle 1 where the MCNP6 results are based on an explicit representation of the fuel isotopics for each fuel rod. This resulted in {{ }}^{2a,c} RLP within {{ }}^{2a,c} of the peak CMS5 RLP of {{ }}^{2a,c}. Given the size of this dataset these values are not tabulated here, but the CMS5 values and the corresponding MCNP6 RLP values are presented in Figure 5-7. The relative difference of the peak RLP values (%ΔRLP) were used to determine relative bias and tolerance limit values for the CMS5 peak RLP based on the MCNP6 RLP.

The %ΔRLP data were evaluated using the normality tests noted in Section 4.2. The results of these tests demonstrate that the data set is normally distributed. As such, the bias and tolerance limits are determined based the technique described in Section 4.3.1. These determinations utilized a weighted mean and weighted standard deviation based on Section 4.4. These values are shown in Table 5-9 along with the bias and tolerance limit determinations for the %ΔRLP data. The results indicate that CMS5 has a tendency to {{ }}^{2a,c} based on the bias of {{ }}^{2a,c}. This bias represents the relative difference in the peak RLP. The uncertainty on the bias was determined to be {{ }}^{2a,c}.

{{

}}2a,c

Figure 5-7. CMS5 versus MCNP6 Peak RLP

Table 5-9. Bias and tolerance limit results for peak RLP data

Parameter	Value
Bias (Weighted Mean)	{{
Weighted Standard Deviation	
LTL	
UTL	}}2a,c

5.1.4.2.5 Peak Relative Pin Power

In order to determine a CMS5 bias and tolerance limit applicable to the peaking factor $F_{\Delta H}$, a comparison was performed between MCNP6 and CMS5 RPP. This comparison examined the differences with RPP that were within {{ }}2a,c of the peak CMS5 RPP. The RPP derived are specific to the NuScale reactor design at HFP conditions at BOC Cycle 1 where the MCNP6 results are based on an explicit representation of the fuel isotopics for each fuel rod. This resulted in {{ }}2a,c RPP values within {{ }}2a,c of the peak CMS5 RPP of {{ }}2a,c. Given the size of this dataset, these values are not tabulated here, but the CMS5 RPP and the corresponding MCNP6 RPP are plotted in Figure 5-8. The relative difference of the peak RPP values (%ΔRPP) were used to determine a relative bias and tolerance limits for the CMS5 peak RPP based on the MCNP6 RPP values.

The %ΔRPP data were evaluated using the normality tests noted in Section 4.2. The results demonstrate that the data are normally distributed. As such, the bias and tolerance limits are determined based on the technique described in Section 4.3.1. These determinations utilized a weighted mean and weighted standard deviation based on Section 4.4. These values are shown in Table 5-10 along with the bias and tolerance limit determinations for the %ΔRPP data. The results indicate that CMS5 has a slight tendency to {{ }}[2a,c] based on the bias of {{ }}[2a,c]. This bias represents the relative difference in the peak RPP values. The uncertainty on the bias was determined to be {{ }}[2a,c].

{{

}}[2a,c]

Figure 5-8. CMS5 versus MCNP6 Peak RPP

Table 5-10. Bias and tolerance limit results for %ΔRPP data

Parameter	Value
Bias (Weighted Mean)	{{
Weighted Standard Deviation	
LTL	
UTL	}}[2a,c]

5.1.4.2.6 Peak Relative Assembly Power

Bias and tolerance limits were determined for CMS5 calculated peak relative assembly power (the peak RAP) values by using a comparison between MCNP6 and CMS5. The RAP values derived are specific to the NuScale reactor core at HFP conditions for Cycle 1 BOC where the MCNP6 model utilizes explicit fuel isotopics. A total of {{ }}^{2a,c} RAP values were included in the bias and tolerance limit determination. The %ΔRAP from Table 5-11 were used to determine a relative bias and tolerance limits for the CMS5 peak RAP.

The %ΔRAP data were evaluated using the normality tests noted in Section 4.2. The results demonstrate that the data are normally distributed. As such, the bias and tolerance limits are determined based on the technique described in Section 4.3.1. These determinations utilized a weighted mean based on Section 4.4 and are shown in Table 5-12 along with the tolerance limit determinations. The tolerance limits are based on the smallest and largest %ΔRAP from Table 5-11. The results indicate that CMS5 has a slight tendency to {{ }}^{2a,c} values based on the bias of {{ }}^{2a,c}. This bias represents the relative difference in the peak RAP.

Table 5-11. MCNP6 and CMS5 RAP comparison at BOC

MCNP6 RAP	CMS5 RAP	Relative Difference (%)
{{		
		}}[2a,c]

Table 5-12. Bias and tolerance limit results for MCNP6 to CMS5 %ΔRAP comparison data

Parameter	Value
Bias (Weighted Mean)	{{
N (Values Used)	
LTL	
UTL	}}^{2a,c}

5.1.4.2.7 Predicted Pin Power

Ideally, CMS5 pin powers would be compared directly to measured values in order to determine bias and tolerance limits for the NuScale reactor. However, no data exist to provide measured pin power data for the NuScale reactor core. Therefore, the ability of CMS5 to perform pin power reconstruction was demonstrated by comparing CMS5 peak RLP and peak RPP to MCNP6 predictions for the NuScale reactor core (Sections 5.1.4.2.4 and 5.1.4.2.5). These predictions provide an explicit comparison of the peaking factors, peak F_Q and peak $F_{\Delta H}$, so the tolerance limits developed in Sections 5.1.4.2.4 and 5.1.4.2.5 are directly applicable to CMS5 peaking factors.

Since no method exists for the NuScale reactor core to directly measure pin power, the determination of pin powers and peaking factors during operations is based on synthesized assembly power measurements (from detector signals) and CMS5 predicted pin-to-box ratios (peak pin power in an assembly to average pin power in an assembly). The uncertainty in the peak pin power is then a combination of the uncertainty determined for the assembly power predictions and the uncertainty determined for the pin-to-box ratio predictions. The uncertainty associated with predicted assembly power and the method for updating that uncertainty is provided in Section 7.6. Uncertainty in the pin-to-box ratio predictions is based on the MCNP6 to CMS5 peak RPP comparisons (Section 5.1.4.2.5), which is a determination of the peak pin power in the core to the average pin power in the core. This comparison is conservative for the pin-to-box ratio uncertainty, since it is determining the peak pin power relative to all the pins in the core, not just the pins in an assembly.

5.2 Empirical Benchmarks

Empirical benchmark comparisons of critical experiments and experimental reactors are used to support validation of the CMS5 code suite. The reactor cores in these benchmarks are typically small cores with fewer fuel rods than in the NuScale reactor core, but with the same or similar lattice geometries and materials. Critical experiments are cold, zero-power configurations that serve for validation of core reactivity comparisons. Experimental reactors are also typically cold and zero-power cores, but may be at elevated coolant temperatures or very low power levels. Experimental reactors also serve for core reactivity comparisons and for reaction rate benchmarking for power distribution. For the critical reactors and experimental reactors benchmarked, the evaluations are performed in 3D; cross section data for each segment is calculated

in 2D (radially) in CASMO5, compiled to a library in CMSLINK5, and modeled with the axial dimension in SIMULATE5.

5.2.1 Critical Experiments

Critical experiment benchmarking was performed against the IPEN/MB-01 experimental reactor (References 10.1.21 through 10.1.25). This experiment was selected for the similarity of many geometry and material features with the NuScale reactor core including the fuel rod geometry, lattice spacing, fuel enrichment, and CRA absorbing materials.

The critical configurations for the IPEN/MB-01 experimental reactor provide a source of comparison for k_{eff} and demonstrate code capability. The IPEN/MB-01 research reactor is a zero power critical facility that has been utilized for basic reactor physics research and as an instructional laboratory. The maximum power is 100 watts, controlled by two CRA banks that are composed of AIC and B_4C rods. The fuel is composed of square-pitched lattices with low-enriched uranium (nominally 4.35% ^{235}U), clad in stainless-steel, and contained in a water-moderated tank configuration. For each test configuration, criticality was achieved either as a function of the number of fuel rods or reflector plates in specific geometric configurations, such that the system was brought as close as possible to a critical state, with some configurations incorporating blank or absorbing rods.

5.2.2 Experimental Reactors

Experimental reactor benchmarking was performed by comparisons with the KRITZ (KRITZ-2:1 and KRITZ-2:13) and DIMPLE (DIMPLE-S01 and DIMPLE-S06) experimental configurations. The experimental reactor systems are described in the paragraphs below.

The objective of KRITZ-2:1 and KRITZ-2:13 experiments was to attain criticality, by adjusting water level and soluble boron concentration, of a rectangular array of Zircaloy-2 clad, 1.86% ^{235}U enriched UO_2 fuel rods in light water. Criticality was achieved for isothermal conditions at room temperature and at elevated temperatures. Typical water level at criticality was below the top of the fuel, so the top portions of the fuel rods extended in the air and steam region. The CBCs and water level were determined and the relative rod powers (fission rate distributions) were measured for selected fuel rods (References 10.1.26 and 10.1.27).

The DIMPLE-S01 and DIMPLE-S06 experimental programs considered room temperature light water moderated critical experiments with steel-clad, low enriched (3.0 wt% ^{235}U) UO_2 fuel rods. The array simulated the rectangular corner configuration of a PWR and effectively represented twelve PWR fuel assemblies. All configurations were evaluated at various fuel rod locations for radial and axial reaction rate distributions. This information included ^{235}U fission, ^{238}U fission, ^{239}Pu fission, ^{238}U capture and reaction rate ratios for the fast fission ratio [^{238}U fission over ^{235}U fission], ^{239}Pu fission over ^{235}U fission, and the relative conversion ratio [^{238}U capture over ^{235}U fission] (References 10.1.28 and 10.1.29).

5.2.3 Results

The following sections present the results from the experimental reactor benchmarks pertinent to the determination of NRFs, including comparison data for k_{eff} values and radial fission rates.

5.2.3.1 Effective Neutron Multiplication Factor

The experiments described in Sections 5.2.1 and 5.2.2 included determinations for the k_{eff}. A comparison between the benchmark and CMS5 k_{eff} was performed for the three experimental reactor setups for a total of fifty-eight (58) critical configurations. Forty-eight (48) tests are part of a series of experiments involving the IPEN experimental reactor (References 10.1.21 through 10.1.25) and six tests are based on experiments involving the DIMPLE experimental reactor (References 10.1.28 and 10.1.29). The remaining four tests are based on experiments involving the KRITZ experimental reactor (References 10.1.26 and 10.1.27). The k_{eff} results are presented in Table 5-13 along with the associated absolute differences in k_{eff} (Δk_{eff}). The uncertainties on the Δk_{eff} values ranged from approximately 0.0004 to 0.003. These results show that CMS5 has a small tendency to {{ }}^{2a,c} based on the small weighted mean of the Δk_{eff} values of {{ }}^{2a,c}. The small weighted standard deviation of {{ }}^{2a,c} shows that CMS5 consistently and accurately calculates k_{eff}.

Table 5-13. k_{eff} comparison data from IPEN, KRITZ, and DIMPLE experiments

ID	Experiment k_{eff}	CMS5 k_{eff}	Absolute Difference (Δk_{eff})
{{			
			}}[2a,c]

ID	Experiment k_{eff}	CMS5 k_{eff}	Absolute Difference (Δk_{eff})
{{			
			}}[2a,c]

5.2.3.2 Radial Fission Rates

Relative radial fission rates from two experimental reactors (KRITZ and DIMPLE) were compared with those determined by CMS5. The following present the results of these comparisons along with the basic statistics.

The relative fission rate is a measure of the fission rate in a fuel rod relative to the measured fission rate of a reference fuel rod. The KRITZ experimental results for relative fission rates were compared with CMS5 for two configurations. The KRITZ-2:13 experiment consisted of two cases that varied only by temperature. The results of these two cases are grouped together.

The relative fission rates used are benchmark values from the KRITZ-2:1 and KRITZ-2:13 experiments (References 10.1.26 and 10.1.27). The basic results for the relative fission rates are presented in Table 5-14 and Table 5-15. These results include the determined weighted mean, weighted standard deviation, minimum, and maximum for the relative difference of relative fission rate values.

The DIMPLE experimental results for relative fission rates were compared with CMS5 for two configurations. The results for these two configurations are presented in Table 5-16 (DIMPLE-S01A) and Table 5-17 (DIMPLE-S06A). These tables include the determined weighted mean, weighted standard deviation, minimum, and maximum for the relative difference of relative fission rate values.

A total of four sets of relative fission rate data were evaluated. Two sets from the KRITZ series of experiments and two sets from the DIMPLE series of experiments. The basic statistics for the relative difference of relative fission rate values are summarized in Table 5-18. In addition, the overall basic statistics are included in Table 5-18 that is done assuming all the percent difference of relative fission rate values from the four sets are a single set of data.

Table 5-14. Relative fission rate comparison data from KRITZ-2:1 experiment

ID	Experiment	CMS5	Relative Difference[(1)] (%Δ)
{{			
			}}[2a,c]

Notes: [(1)] The number of decimal places given for the results in this table has been reduced for presentation purposes. For some values, the resulting relative difference may not be calculated exactly as given in this table as a result of this rounding.

Table 5-15. Relative fission rate comparison data from KRITZ-2:13 Experiment

ID	Experiment	CMS5	Relative Difference (%Δ)
{{			
			}}[2a,c]

ID	Experiment	CMS5	Relative Difference (%Δ)
{{			
			}}[2a,c]

Table 5-16. Relative fission rate comparison data from DIMPLE-S01A Experiment

Unique ID	CMS5	Experiment	Relative Difference(1) (%Δ)
{{			
			}}2a,c

Unique ID	CMS5	Experiment	Relative Difference(1) (%Δ)
{{			
			}}2a,c

Notes: (1) The number of decimal places given for the results in this table has been reduced for presentation purposes. For some values, the resulting relative difference may not be calculated exactly as given in this table as a result of this rounding.

Table 5-17. Relative fission rate comparison data from DIMPLE-S06A Experiment

Unique ID	CMS5	Experiment	Relative Difference(1) (%Δ)
{{			
			}}2a,c

Unique ID	CMS5	Experiment	Relative Difference(1) (%Δ)
{{			
			-
			}}2a,c

Notes: (1) The number of decimal places given for the results in this table has been reduced for presentation purposes. For some values, the resulting relative difference may not be calculated exactly as given in this table as a result of this rounding.

Table 5-18. Summary of basic statistics for the %Δ relative fission rate results

Experiment	Weighted Mean	Weighted Standard Deviation	Minimum	Maximum
{{				
				}}[2a,c]

6.0 Operational Data Benchmarking

Comparison with operational data is provided in Section 6.1 through benchmarking against actual operating plant data to demonstrate the ability to accurately predict a range of typical PWR neutronic parameters using the CMS5 code suite. In addition, in Section 6.2 industry-developed uncertainties determined from comparison with operating reactor data, experimental reactors, and other code benchmarks is provided for information only.

6.1 Three Mile Island Unit 1 Reactor Description

Three Mile Island Unit 1 (TMI-1) is a PWR with a design thermal power level of 2535 MWt. The reactor coolant system is composed of two loops with one hot leg and two cold legs per loop. The reactor core contains 177 fuel assemblies each consisting of 208 fuel rods, 16 CRA rod guide tubes, and one in-core instrument guide tube. Reactivity is controlled by 61 full-length AIC CRAs and soluble boron. Eight part-length AIC CRAs are used to control axial power distribution. In Cycle 1 only, 68 discrete burnable poison rod assemblies were inserted in the core. Benchmarks against TMI-1 data were performed for Cycle 1 and Cycle 2. Comparisons were made with data collected throughout these operating cycles for multiple state points and cycle burnups (Reference 10.1.30). This reactor is modeled in full 3D geometry using CASMO5 and SIMULATE5.

6.1.1 Results

Measured values of multiple key core physics parameters are taken from operating data at various burnup points throughout the cycle. The uncertainty in the calculation of burnup is inherently included in the uncertainty determined through comparison between the measured and calculated values of each key core physics parameter. The benchmarking results for a number of neutronic parameters are given in the following sections based on modeling TMI-1 Cycles 1 and 2 in CASMO5 and SIMULATE5.

6.1.1.1 Critical Boron Concentration

The CBC values used in the benchmarking are measured values from TMI-1 Cycles 1 and 2. The CMS5 and TMI-1 measured values for CBC results are presented in Table 6-1.

The CBC versus burnup data are also presented in Figure 6-1 and Figure 6-2 for the Cycle 1 and Cycle 2 data, respectively. These figures show a reasonably consistent {{ }}^{2a,c} of the measured data by CMS5 of approximately {{ }}^{2a,c} throughout the examined burnup range. This consistency is a strong indication that CMS5 is properly performing the isotopic depletion and ingrowth calculations needed to accurately predict the reactivity of the system throughout a given burnup range.

For the purpose of determining bias and tolerance limits, the data from Table 6-1 were maintained as a single group. The absolute difference for each pair of critical boron concentrations were determined and are shown in Table 6-1.

The ΔCBC values from Table 6-1 were evaluated using the normality tests noted in Section 4.2. The results demonstrate that the data are normally distributed. As such, the bias and tolerance limits are determined utilizing the technique described in Section 4.3.1. A summary of the bias and tolerance limit determinations for the TMI-1 ΔCBC data from Table 6-1 is provided in Table 6-2. The results indicate that CMS5 has a tendency to {{ }}^{2a,c} the CBC values based on the bias of {{ }}^{2a,c}. This bias represents the absolute difference in the CBC data. The uncertainty on the bias was determined to be {{ }}^{2a,c}.

Table 6-1. CBC data used for bias and tolerance limit determinations

Burnup (MWd/MTU)	Core Power		CBC (ppm)		Absolute Difference (ppm)
	MWt	% Full	Measured	CMS5	
Cycle 1 Data					
{{					
					}}^{2a,c}

Burnup (MWd/MTU)	Core Power		CBC (ppm)		Absolute Difference (ppm)
	MWt	% Full	Measured	CMS5	
{{					
					}}2a,c
Cycle 2 Data					
{{					
					}}2a,c

Burnup (MWd/MTU)	Core Power		CBC (ppm)		Absolute Difference (ppm)
	MWt	% Full	Measured	CMS5	
{{					
					}}[2a,c]

{{

}}[2a,c]

Figure 6-1. Critical boron concentration versus burnup for TMI-1 Cycle 1 data

{{

}}2a,c

Figure 6-2. Critical boron concentration versus burnup for TMI-1 Cycle 2 data

Table 6-2. Bias and tolerance limit results for ΔCBC for TMI-1 data

Parameter	Value
Bias (Mean)	{{
Standard Deviation	
LTL	
UTL	}}2a,c

6.1.1.2 Differential Boron Worth

Four TMI-1 DBW measurements were evaluated. Three values were associated with TMI-1 Cycle 1 data and one was from the TMI-1 Cycle 2 data. These values are listed in Table 6-3 along with the CMS5 calculated values. These data are not sufficient to perform a detailed statistical analysis, so a basic comparison is performed. The results in Table 6-3 show that the CMS5 calculated values are in good agreement with the measured values.

Table 6-3. Differential Boron Worth Results

Observation	Cycle	Boron Concentration (ppm)	DBW (pcm/ppm)		Relative Difference (%)
			Measured	CMS5	
{{					
					}}[2a,c]

6.1.1.3 Isothermal Temperature Coefficient

A total of eight ITC values were reported. Of these, six were at HZP and two were at HFP. This small number of values is not sufficient to perform a detailed statistical analysis. These values are all listed in Table 6-4 along with the CMS5 calculated values and absolute differences between the benchmark and CMS5 calculated values.

Table 6-4. Isothermal Temperature Coefficient Results

Observation	HZP/HFP	Cycle	Boron Concentration (ppm)	ITC (pcm/°F)		Absolute Difference (pcm/°F)
				Measured	CMS5	
{{						
						}}[2a,c]

6.1.1.4 Power Coefficient

Only two power coefficient values were reported in Reference 10.1.30, one for Cycle 1 and one for Cycle 2. This small number is not sufficient to perform a detailed statistical analysis. These values are listed in Table 6-5 along with the CMS5 calculated values and the relative difference between the benchmark and CMS5 calculated values.

Table 6-5. Power Coefficient Results

Observation	Cycle	Boron Concentration (ppm)	PC (pcm/%P)		Relative Difference (%)
			Measured	CMS5	
{{					
					}}2a,c

6.1.1.5 Control Rod Assembly Group Worth

A total of nine CRA group worth values were reported based on data from TMI-1 Cycles 1 and 2 at HZP (Reference 10.1.30). Of these, eight are shown in Table 6-6 along with the CMS5 calculated values and the differences between the benchmark and calculated values.

The value associated with Group 8 from the Cycle 1 data was not included in the presented results. The Group 8 part-length design is not representative of the NuScale CRA design, and it is not included as part of this review of the CRA group worth results.

Table 6-6. Control Rod Assembly Group Worth Results

Observation	Cycle	CRA Group(s)	CRA Group Worth (pcm)		Relative Difference (%)
			Measured	CMS5	
{{					
					}}2a,c

Notes: * at 50% withdrawn

6.1.1.6 Peak Relative Assembly Power

The RAP data used in the benchmarking are measured values from TMI-1 for Cycles 1 and 2 (Reference 10.1.30). The specific values used are those {{ }}2a,c and are presented in Table 6-7. The table also provides the relative difference between the benchmark and CMS5 calculated values.

The percent difference of peak RAP values from Table 6-7 were ordered by value. These ordered values were used to produce a normality plot. This figure shows that the data is normally distributed. The {{ }}2a,c %ΔRAP from Table 6-7 were evaluated for normality using the tests noted in Section 4.2. The results demonstrate that the data are normally distributed. As such, the bias and tolerance limits are determined utilizing

the technique applicable to normally distributed data described in Section 4.3.1. A summary of the bias and tolerance limit determinations for the TMI-1 percent difference of RAP from Table 6-7 is provided in Table 6-8. The results indicate that CMS5 has a tendency to {{ }}[2a,c] based on the bias of {{ }}[2a,c]. This bias represents relative difference in the peak RAP. The uncertainty on the bias was determined to be {{ }}[2a,c].

Table 6-7. TMI-1 peak RAP

Assembly (Row – Column)	Cycle	Burnup (MWd/MTU)	CMS5	Measured	Relative Difference (%)
{{					
					}}2a,c

Table 6-8. Bias and tolerance limits results for peak RAP for TMI-1 data

Parameter	Value
Bias (Mean)	{{
Standard Deviation	
LTL	
UTL	}}^{2a,c}

6.2 Industry Standard Values

Commercial reactor operating data for the NuScale reactor core are not yet available. In order to provide an overall sense of the fidelity of the aforementioned benchmarks, the NRFs determined by Studsvik with the CMS5 code package are provided in Table 6-9 (Reference 10.1.31) and in discussions within this report will be referred to as industry standard values. These values provide an indication that the NuScale values are reasonable and consistent with CMS5 derived factors for current operating PWRs and are strictly for information only and specific approval of this data for use in the methodology is not requested.

Table 6-9. Industry Standard Nuclear Reliability Factors (For Information Use)

Physics Parameter	Upper NRF	Lower NRF
Core Reactivity (Critical Boron Concentration)	{{	
Integral Control Rod Bank Worth (Individual Bank and Total of All Banks)		
Isothermal and Moderator Temperature Coefficient		
$F_{\Delta H}/F_r$ (Movable and Fixed In-Core Detectors)		
F_Q (Movable and Fixed In-Core Detectors)		
Differential Boron Worth		
Doppler Temperature / Power Coefficient		
Kinetics Parameters (Delayed Neutron and Prompt Neutron Lifetime)		}}^{2a,c}

7.0 Code Biases and Nuclear Reliability Factors

This section describes the biases and tolerance limits associated with the key core physics parameters that are developed based on the benchmarking results presented in Sections 5.0 and 6.0. Using this information, NRFs are developed and implemented during startup and operation of the reactor core. Since actual operating data are not yet available for the NuScale reactor core, this report provides base values and a methodology for verifying and updating the values when actual plant data are available.

The core physics parameters in this section are discussed with respect to biases and tolerance limits for the generation of NRFs that may be based on the following:

- code-to-code benchmarking against the higher-order code MCNP6 (Section 5.1),
- empirical benchmarking against the IPEN, DIMPLE, and KRITZ experimental reactors (Section 5.2), and
- benchmarking against operating data from TMI Unit 1 (Section 6.1).

In addition, industry standard values for the CMS5 code suite are presented that demonstrate consistency of the benchmarking presented in this report (Section 6.2).

7.1 Critical Boron Concentration

The CBC represents the concentration of boron in the water moderator of an reactor core that allows the reactor to be exactly critical (i.e., k_{eff} = 1.0).

7.1.1 Base Nuclear Reliability Factors

Table 7-1 presents a summary of the CBC comparison results between CMS5 and source values. These results are based on the absolute difference between the source value and CMS5 calculated values. A summary of the data and determinations that form the basis of these results are provided in Section 6.1.1.1. Code-to-code data are not evaluated in the determination of this NRF and empirical data are not available for this parameter.

Table 7-1. Critical boron concentration comparative results

Value	CBC Absolute Difference (ppm)			
	Code-to-Code	Empirical	Operating	Industry[(1)]
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}^{2a,c}
[(1)]Units are converted from pcm to ppm using a representative differential boron worth of {{ }}^{2a,c}.				

The operating data and associated uncertainty and tolerance limits for CBC are based on the TMI-1 reactor, which differs from the NuScale design. However, this benchmark of operating data can be used to determine a bounding base NRF value to be applied to the NuScale design until sufficient measurement data can be acquired after startup and operation to refine the NRFs. The base CBC NRFs for both HZP and HFP are:

Lower NRF = {{ }}^{2a,c}

Upper NRF = {{ }}^{2a,c}

The total tolerance band of the NRFs is reasonable with respect to the industry standard values given in Table 7-1.

7.1.2 NRF Update Methodology

During the initial cycle startup of a module and for every successive module startup, the CBC is measured at HZP and HFP, and then HFP CBC measurements will be made at periodic intervals during the operating cycle. The following methodology will be used to update the CBC NRFs:

- At HZP, CBC measurements will be made at startup of each module. A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.1.1; the values will be assessed to account for the available measured data. When updated, the new NRFs will replace the base values for HZP for initial and reload cycles.
- At HFP, CBC measurements will be made during the initial startup and at periodic intervals during the cycle for each module. A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.1.1; the values will be assessed to ensure that the NRFs remain conservative. When updated, the new NRFs will replace the base values for HFP for initial and reload cycles.
- At HZP and HFP, CBC NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.

7.2 Differential Boron Worth

The DBW represents a measure of the change in reactivity associated with a change in the boron concentration in the water moderator. The reactivity change is measured in units of pcm and the change in boron concentration is in ppm.

7.2.1 Base Nuclear Reliability Factors

Table 7-2 presents a summary of DBW worth comparison results between CMS5 and various source values. These results are based on the relative difference between the source value and CMS5 calculated values. A summary of the data and determinations that form the basis of these results is provided in Sections 5.1.4.2.1 and 6.1.1.2. No empirical data are available for DBW.

Table 7-2. Differential boron worth comparative results

Value	DBW Relative Difference			
	Code-to-Code	**Empirical**	**Operating**	**Industry**
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}^{2a,c}

The code-to-code data are representative of the fidelity of the CMS5 prediction for this parameter. The operating data and associated uncertainty for differential boron worth are based on the TMI-1 reactor, which differs from the NuScale design. The code-to-code and operating data individually are not applied to the NuScale design, however, these benchmarks can be used collectively to determine a base NRF value to be applied to the NuScale design until sufficient measurement data can be acquired after startup and operation to refine the NRFs.

Since the operating data are limited, a more detailed statistical evaluation is only performed on the code-to-code data for this parameter. Using the results shown in Table 7-2, values with sufficiently large conservatism are chosen for the base NRFs. The base DBW NRFs are:

Lower NRF = {{ }}^{2a,c}

Upper NRF = {{ }}^{2a,c}

These NRFs are more conservative than the industry standard values shown in Table 7-2.

7.2.2 Nuclear Reliability Factor Update Methodology

During the initial cycle startup of a module, and for every successive module startup, the DBW will be measured at HZP. The following methodology will be used to update the DBW NRFs:

- HZP DBW measurements will be made at startup of each module. A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.2.1; the values will be assessed to account for the available measured data. When updated, the new NRFs will replace the base values for initial and reload cycles.
- DBW NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.

7.3 Isothermal Temperature Coefficient and Moderator Temperature Coefficient

Given the similarities between the ITC and MTC, both are discussed together in the following subsections.

The ITC is the change in reactivity due to a combined change in both the moderator and the fuel temperature in a reactor where the temperature is uniform across the core. Similarly, the MTC is the change in reactivity due to a change in moderator temperature. Since ITC is the quantity that will be measured, the biases and tolerance limits for ITC will be representative of MTC.

7.3.1 Base Nuclear Reliability Factors

Table 7-3 and Table 7-4 present a summary of ITC and MTC comparison results between CMS5 and various source values. The values presented are related to the absolute difference between the source value and the CMS5 calculated value. A summary of the data and determinations that form the basis for these results is provided in Sections 5.1.4.2.2, 5.1.4.2.3, and 6.1.1.3. No empirical data are available for ITC or MTC. Operating data are also not available for MTC.

Table 7-3. Isothermal temperature coefficient comparative results

Value	ITC Absolute Difference (pcm/°F)			
	Code-to-Code	Empirical	Operating	Industry
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}^{2a,c}

Table 7-4. Moderator temperature coefficient comparative results

Value	MTC Absolute Difference (pcm/°F)			
	Code-to-Code	Empirical	Operating	Industry
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}^{2a,c}

The code-to-code data are representative of the fidelity of the CMS5 prediction for these parameters. The operating data and associated uncertainty for ITC are based on the TMI-1 reactor, which differs from the NuScale design. The code-to-code and operating data individually are not applied to the NuScale design; however, these benchmarks can be used collectively to determine a base NRF value to be used for the NuScale design until sufficient measurement data can be acquired after startup and operation to refine the NRFs.

Since the operating data are limited, a more detailed statistical evaluation is only performed on the code-to-code data for these parameters. The base NRFs are chosen to have sufficiently large conservatism relative to the code-to-code and operating data results shown in Table 7-3 and Table 7-4. The base ITC and MTC NRFs to be used for HZP and HFP are:

Lower NRF = {{ }}^{2a,c}

Upper NRF = {{ }}^{2a,c}

These NRFs are more conservative than the industry standard values shown in Table 7-3 and Table 7-4.

7.3.2 Nuclear Reliability Factor Update Methodology

During the initial cycle startup of a module and for every successive module startup, the ITC measurement is performed at HZP. Typical measurements are done with a heatup and cooldown of approximately 5 degrees F. The following methodology will be used to update the ITC NRFs:

- HZP ITC measurements will be made at startup of each module. A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.3.1; the values will be assessed to account for the available measured data. When updated, the new NRFs will replace the base values for initial and reload cycles.
- ITC NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.
- Any new NRFs determined for ITC will apply equally to MTC.

7.4 Power Coefficient and Fuel Temperature Coefficient

The power coefficient is the change in reactivity due to a change in power. The reactivity change is measured in units of pcm and the change in power is in percent power. The FTC (also referred to as the Doppler coefficient) is the change in reactivity due to a change in the fuel temperature. The reactivity change is measured in units of pcm and the change in temperature is in degrees Fahrenheit. Since the power coefficient is the quantity that will be measured, the biases and tolerance limits for the power coefficient will be representative of FTC.

7.4.1 Base Nuclear Reliability Factors

Table 7-5 presents a summary of power coefficient comparison results between CMS5 and source values. The values presented are related to the relative difference between the source value and the CMS5 calculated value. A summary of the data and determinations that form the basis of these results is provided in Section 6.1.1.4. No code-to-code data or empirical data are available for power coefficient.

Table 7-5. Power coefficient comparative results

Value	PC Relative Difference			
	Code-to-Code	**Empirical**	**Operating(1)**	**Industry**
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}2a,c

Notes: (1) Only two values available. Each is listed and no mean or standard deviation is determined.

A power coefficient is rarely measured and the operational data are very limited (two data points). As such, no statistical evaluation is performed for this parameter. Based on the limited operating data results shown in Table 7-5, representative values that envelop these results are chosen for the base NRFs. The base power coefficient and FTC NRFs are:

Lower NRF = {{ }}2a,c

Upper NRF = {{ }}2a,c

These NRFs are more conservative than the industry standard values shown in Table 7-5.

7.4.2 Nuclear Reliability Factor Update Methodology

Power coefficient is typically not measured; however, if measurements are made the following methodology will be used to update the power coefficient and FTC NRFs:

- A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.4.1. The values will be assessed to account for the available measured data.
- Power coefficient NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.
- Any new NRFs determined for power coefficient will apply equally to FTC.

7.5 Control Rod Assembly Bank Worth

Control rod assembly bank worth is the potential change in reactivity of the core that is represented by the insertion or removal of a CRA bank. It is representative of insertion of individual CRAs and CRA banks. The reactivity worth is provided in units of pcm.

7.5.1 Base Nuclear Reliability Factors

The worth of a control rod assembly, group, or bank can vary significantly based on the design of the CRAs, number of CRAs being evaluated, and their location in a core. The comparisons provided in Table 7-6 are associated with the relative differences between the CMS5 calculated values and the source values. A summary of the data and determinations that form the basis of these results is provided in Section 5.1.4.1 and Section 6.1.1.5. Empirical data are not available for this parameter.

Table 7-6. Control Rod Assembly Bank Worth comparative results

Value	CRA Bank Worth Relative Difference			
	Code-to-Code Bank / Single	**Empirical**	**Operating**	**Industry**
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}^{2a,c}

The operating data and associated uncertainty for CRA worth are based on code-to-code benchmarking and on the TMI-1 reactor. This TMI-1 reactor differs from the NuScale design, however, this benchmark of operating data can be used to support determination of a base NRF value to be applied to the NuScale design until sufficient measurement data can be acquired after startup and operation to refine the NRFs.

Since the operating data and code-to-code data are limited, no detailed statistical evaluation is performed for this parameter. Based on the code-to-code and operating data results shown in Table 7-6, reasonable values are chosen for the base NRFs. The base CRA bank worth NRFs are:

Lower NRF = {{ }}^{2a,c}

Upper NRF = {{ }}^{2a,c}

These NRFs are more conservative than the industry standard values shown in Table 7-6.

7.5.2 Nuclear Reliability Factor Update Methodology

During the initial cycle startup of a module and for every successive module startup, CRA worth measurements will be taken during startup testing. The following methodology will be used to update the CRA bank worth NRFs:

- Measurements will be made during low power physics testing at startup of each module.

- A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.5.1; the values will be assessed to account for the available measured data.
- CRA worth NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.

7.6 Assembly Radial Peaking

Assembly radial peaking is a relative quantity representative of the ratio of the peak assembly power to the average assembly power.

7.6.1 Base Nuclear Reliability Factors

The following table presents a summary of comparison results between CMS5 and various source values. These results are based on the relative difference between the source values and the CMS5 values. A summary of the data and determinations that form the basis for the results presented in Table 7-7 is provided in Sections 5.1.4.2.6 and 6.1.1.6. No empirical data are available for this parameter.

Table 7-7. Assembly radial peaking comparative results

Value	Assembly Radial Peaking Relative Difference			
	Code-to-Code(1)	Empirical	Operating	Industry
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}2a,c

(1)The code-to-code results are for all assemblies in a core configuration and not just the peak assembly locations.

The code-to-code data are representative of the fidelity of the CMS5 prediction for this parameter. The operating data and associated uncertainty and tolerance limits for assembly power peaking are based on the TMI-1 reactor, which differs from the NuScale design. The code-to-code and operating data individually are not applied for the NuScale design, however, these benchmarks can be used collectively to determine a base NRF value to be applied to the NuScale design until sufficient measurement data can be acquired after startup and operation to refine the NRFs.

Assembly radial peaking is only significant when CMS5 under-predicts peaking. Therefore, only the upper tolerance limit is significant for assembly radial peaking. The base NRFs are chosen to have sufficiently large conservatism relative to the code-to-code and operating data results shown in Table 7-7. The assembly radial peaking NRFs are:

Lower NRF = N/A

Upper NRF = {{ }}^{2a,c}

7.6.2 Nuclear Reliability Factor Update Methodology

During the initial cycle startup of a module and for every successive module, initial power ascension assembly relative powers will be measured at intermediate power level and at full power to determine assembly peaking factor measurements for the overall peak power (assembly F_Q) and the radial peak power (assembly $F_{\Delta H}$). Thereafter, full power measurements will be made at periodic intervals during the operating cycle. The following methodology will be used to update the assembly radial peaking NRFs:

- At an intermediate power level, assembly radial peaking factor measurements will be made during the initial power ascension of each module.
- At full power, assembly radial peaking factor measurements will be made during the initial startup and at periodic intervals during the cycle for each module.
- For intermediate and full power levels, a comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.6.1; the values will be assessed to account for the available measured data.
- Assembly radial peaking NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.

7.7 Pin Peaking

Pin power is used to describe the variations in the power of a core both radially and axially. Pin power is expressed as a relative quantity with the absolute power of a given portion of the reactor (pin, pin segment, or assembly) divided by a reference value, typically an average value for the entire core. For example, relative pin power is defined as the power from one pin divided by the average power of all the pins. The quantities of specific interest to the NuScale design are RLP and RPP.

7.7.1 Base Nuclear Reliability Factors

Table 7-8 presents a summary of comparison results between CMS5 calculated 3D peaking (representing F_Q) and radial pin peaking (representing $F_{\Delta H}$) with measured and calculated assembly radial peaking from various sources. These results are based on the relative difference between the source values and the CMS5 values. A summary of the data and determinations that form the basis of these results is provided in Sections 5.1.4.2.4, 5.1.4.2.5, and 5.2.3.2. Operating data are not available for this parameter.

Table 7-8. Pin peaking comparative results

Value	Pin Peaking Relative Difference					
	Code-to-Code		Empirical	Operating	Industry	
	F_Q	$F_{\Delta H}$	Relative Fission Rate(1)		F_Q	$F_{\Delta H}$
Bias (Mean)	{{					
Standard Deviation						
LTL						
UTL						}}^{2a,c}

Notes: (1) This included multi-groups of data that could not be reasonably pooled. The presented mean and standard deviation are for the combined group of all results.

The code-to-code data are representative of the fidelity of the CMS5 prediction for this parameter. The empirical data are based on relative fission rates determined for a sampling of fuel rods within a core. As such, the data are only presented as a demonstration of the fidelity of CMS5 in predicting values related to pin power. For the initial NRF determination, the code-to-code and empirical data individually are not applied to the NuScale design, however these benchmarks can be used collectively to determine base NRF values to be applied to the NuScale design until sufficient measurement data can be acquired after startup and operation to refine the NRFs.

The measurement of pin power peaking is only significant when CMS5 under-predicts peaking. Therefore, only the upper tolerance limit is significant for these parameters. Using the code-to-code and empirical data results shown in Table 7-8, values with sufficient conservatism are chosen for the base NRFs. The $F_{\Delta H}$ peaking NRFs are:

$$F_{\Delta H} \text{ Lower NRF} = N/A$$

$$F_{\Delta H} \text{ Upper NRF} = \{\{ \quad \}\}^{2a,c}$$

Because the F_Q peaking tolerance limit is higher than the radial peaking tolerance limit, a higher NRF is chosen for the F_Q peaking. The F_Q peaking NRFs are:

$$F_Q \text{ Lower NRF} = N/A$$

$$F_Q \text{ Upper NRF} = \{\{ \quad \}\}^{2a,c}$$

These NRFs are more conservative than the industry standard values shown in Table 7-8.

7.7.2 Nuclear Reliability Factor Update Methodology

The measurement of pin power peaking is a construction composed of assembly radial peaking and assembly pin-to-box ratios. Assembly power will be a measurable quantity

for the NuScale reactor, but pin power will not be measurable. As such, the process used for updating peaking factor uncertainty will be based on the following steps:

1. Assembly radial peaking NRF will be updated as described in Section 7.6.2.
2. Based on the discussion provided in Sections 5.1.4.2.5 and 5.1.4.2.7, a standard {{ }}^{2a,c} pin-to-box ratio uncertainty (rounded from {{ }}^{2a,c} $F_{\Delta H}$ upper relative tolerance limit) will be assumed.
3. The pin peaking factor upper NRFs will be determined using the square root of the sum of the squares from the assembly radial peaking NRFs with the {{ }}^{2a,c} pin-to-box ratio uncertainty from Step 2 above.

7.8 Axial Offset

Axial offset is a measure of the ratio of the difference between the power in the top and bottom halves of the core to the total core power.

7.8.1 Base Nuclear Reliability Factors

Table 7-9 presents a summary of axial offset comparison results between CMS5 and code-to-code benchmarking. The values presented are related to the absolute difference between the higher-order code and the CMS5 calculated value. A summary of the data and determinations that form the basis of these results is provided in Section 5.1.4.1. Empirical and operating data are not available for this parameter.

Table 7-9. Axial Offset comparative results

Value	AO Absolute Difference			
	Code-to-Code	Empirical	Operating	Industry
Bias (Mean)	{{			
Standard Deviation				
LTL				
UTL				}}^{2a,c}

The code-to-code data are representative of the fidelity of the CMS5 prediction for this parameter. The code-to code data are limited and as such, no detailed statistical evaluation is performed for this parameter; therefore, based on the code-to-code data in Table 7-9, reasonable values are chosen for the base NRFs. The base axial offset NRFs are:

Lower NRF = {{ }}^{2a,c}

Upper NRF = {{ }}^{2a,c}

7.8.2 Nuclear Reliability Factor Update Methodology

During the initial cycle startup of a module and for every successive module, AO will be measured at an intermediate power level and at full power. Thereafter, full power measurements will be made at periodic intervals during the operating cycle. The following methodology will be used to update the AO NRFs:

- At an intermediate power level, AO measurements will be made during the initial power ascension of each module.
- At full power, AO measurements will be made during the initial startup and at periodic intervals for each module.
- A comparison of measured and predicted values will be used to evaluate the base NRFs from Section 7.8.1; the values will be assessed to account for the available measured data.
- AO NRFs will be updated with measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected.

7.9 Kinetics

Kinetic parameters of specific interest are the effective delayed neutron fraction (β_{eff}) and the delayed neutron decay constant (λ). These parameters may be presented for each group or as a sum over all groups, where the groups are based on the approximate half-life of the decay of delayed neutrons precursor fission products.

The key kinetics parameters used in the nuclear physics analyses are the neutron lifetime, the delayed neutron fractions, and the decay constants. The neutron lifetime is the time between the neutron generation by fission and the neutron absorption or leakage. The neutron lifetime includes the slowing down time and the diffusion time. The delayed neutron fraction is the fraction of fission neutrons that are emitted with a slight delay following fission as a result of the decay of some fission products. The delayed neutron fractions are divided into groups according to their decay constants.

7.9.1 Base Nuclear Reliability Factors

The neutron lifetime is not a measured parameter by itself. Because neutron lifetime is proportional to the soluble boron worth, the NRFs are taken to be the same as that for the DBW. Therefore, the base NRFs for neutron lifetime are:

$$\text{Lower NRF} = \{\{\quad\}\}^{2a,c}$$

$$\text{Upper NRF} = \{\{\quad\}\}^{2a,c}$$

The uncertainty in the delayed neutron fraction is inherently included in the uncertainty of other reactivity parameters with which it may be associated; so no explicit uncertainty is applied. However, when β is used in an independent manner, the kinetics NRF is applied.

7.9.2 Nuclear Reliability Factor Update Methodology

Because the neutron lifetime NRFs are the same as the DBW values, the NRFs for neutron lifetime will be updated when the DBW is updated per Section 7.2.2.

8.0 Application

NuScale intends to use the CMS5 code suite to perform nuclear analysis for core design, input to safety analysis, startup physics testing, core follow predictions, and operations support. The details of these applications are described in the following subsections. This topical report is also intended for use by Combined License applicants and licensees to implement core design methodology for their safety analysis calculations and operational support.

An analysis which considers the presence of surrounding modules demonstrates that there is no neutronic impact due to module proximity. As a result, multi-module effects do not need to be further considered in the application of this methodology.

8.1 NuScale Reactor Core Design Methodology

CMS5 is used to determine loading patterns to meet the NuScale energy requirement and calculate core physics parameters that meet design limits to ensure the core will meet safety analysis requirements.

8.1.1 NuScale Fuel Rod and Assembly Lattice Configuration

To meet design constraints, core designs may employ fuel rod and assembly enrichment loading schemes that control the radial and axial power distribution in the core and are designed to help limit power peaking.

NuScale loading pattern designs may include assembly radial enrichment zoning to lower the pin-to-box ratio, and help reduce the overall radial peaking factor in the core. Assembly radial enrichment zoning consists of placing fuel rods of different enrichments in specific lattice locations within an assembly. The fuel rods containing the lower enrichment are generally placed in the high flux regions of the lattice that may include the corners, along the outside edge of the assembly, or around the instrumentation or guide tubes.

Fuel rod axial enrichment zoning is the use of different enrichments within the pellet stack of a fuel rod. Fuel rod axial enrichments are utilized to lower the axial leakage and to shape the axial power profile (reduce axial peaking). “Blankets” are areas of lower pellet enrichment placed at the top and bottom of a pellet stack to reduce leakage. These blankets, in effect, act as a reflector and improve fuel utilization. “Cutback” regions are areas in a BP fuel rod that do not contain BP. The cutback region does not necessarily contain a reduction in the ^{235}U enrichment as is done for the region containing the BP (Section 2.4). Cutback regions are used to shape the axial power shape of fuel rods containing BP. The NuScale reactor core may employ blankets and cutback regions in the fuel.

Cross sections for all assembly lattice configurations and fuel rod loading configurations necessary for the SIMULATE5 model are generated in CASMO5.

8.1.2 NuScale Fuel Management Strategy

The typical fuel management scheme utilizes placement of fresh fuel at the periphery of the core with burned fuel loaded in the middle; an out-in approach. This out-in fuel management scheme lowers the power peaking, maximizes the thermal margin, and provides a flatter radial power distribution across the core, compared to a low leakage core where fresh assemblies are loaded in the interior of the core. The out-in approach also provides for a straightforward analysis and operational approach to the NuScale design. Specifically, this strategy minimizes the BP loading requirement because of the inherently flatter power distribution, and results in the maximum boron concentration and power peaking occurring at the BOC. Core loading patterns aim for a monotonically decreasing boron concentration from BOC to the EOC. This loading pattern results in bounding physics parameters to most likely occur at BOC or EOC, thus simplifying the analysis process.

NuScale loading patterns are typically quadrant-symmetric with respect to core sub-batch types. Maintaining quadrant symmetry helps maintain core symmetric power distributions, helps prevent power tilt, and simplifies the analysis process.

8.1.3 Core Loading Pattern Development

The CMS5 code suite is used to determine NuScale core loading pattern designs through an iterative process, whereby input parameters (control variables) are varied until output parameters (state variables) meet desired design constraints. CMS5 input parameters may include new fuel enrichment, placement of fresh and burned assemblies, BP strength and location, and enrichment zoning. CMS5 output parameters may include power distributions and power peaking, CBC, burnup, cycle length, CRA worth, and reactivity coefficients.

The initial step in the core design process is determination of the energy requirement for a particular cycle. The energy requirement is dependent on the reactor rated thermal power, the plant operating schedule, the capacity factor, and the planned outage length. Once a preliminary pattern has met the energy requirements of the cycle, the core designer adjusts the control variables in order to optimize the pattern and meet the pre-determined core design constraints to produce a final loading pattern for the cycle.

8.1.4 Core Design Constraints

CMS5 is used to calculate several physics parameters to compare against pre-determined design constraints to determine acceptable loading pattern designs. Design targets are developed from design limits that are specified in the Technical Specifications or developed from licensing limitations. When a design target is met for a particular parameter, the core designer has reasonable assurance that the design limits for that particular parameter will not be exceeded during the safety analysis of that loading pattern design. Design targets typically include uncertainty and margin for CMS5 calculated parameters.

Table 8-1 presents preliminary core design constraint parameters and the associated core condition for calculation to ensure the design target is met throughout the cycle. A discussion of each parameter follows the table.

Table 8-1. Core Constraint Parameters

Section #	Parameter	Conditions
8.1.4.1	Max ARO HFP $F_{\Delta H}$	Cycle Max, HFP, ARO
8.1.4.2	Max ARO HFP F_Q	Cycle Max, HFP, ARO
8.1.4.3	Peak Linear Heat Generation Rate (kW/m)	Cycle Max, HFP, ARO
8.1.4.4	Peak Rod Burnup (GWd/MTU)	Cycle Max, HFP, ARO
8.1.4.5	Maximum HFP Boron Concentration (ppm)	Cycle Max, HFP, ARO
8.1.4.6	Minimum Refueling Boron Concentration (ppm)	BOC
8.1.4.7	Most Positive MTC at HZP (pcm/°F)	BOC, HZP
8.1.4.7	Most Positive MTC at multiple power levels (pcm/°F)	BOC, startup
8.1.4.7	Most Positive MTC at HFP (pcm/°F)	Cycle Max, HFP, ARO
8.1.4.8	Most Negative MTC at HFP (pcm/°F)	EOC, HFP, ARO
8.1.4.9	Minimum HFP Shutdown Margin (critical eigenvalue)	N-1, HFP
8.1.4.9	Minimum HZP Shutdown Margin (critical eigenvalue)	N-1, HZP

8.1.4.1 Maximum All Rods Out Hot Full Power $F_{\Delta H}$

The enthalpy rise hot channel factor, $F_{\Delta H}$, is the ratio of the enthalpy rise of the hot channel to the average channel enthalpy rise of the core. The limit on $F_{\Delta H}$ is established to ensure that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. This limit ensures that the design basis value for the critical heat flux (CHF) ratio is met for normal operation, operational transients, and any accident event of moderate frequency. The $F_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise.

Control of the core power distribution with respect to the peaking factors, $F_{\Delta H}$ and F_Q, ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analysis.

$F_{\Delta H}$ is calculated in SIMULATE5 during the cycle depletion for all times in life, at HFP and ARO conditions. The NRF for the $F_{\Delta H}$ peaking factor (Section 7.7) is applied to increase the calculated value, resulting in a conservative prediction of $F_{\Delta H}$.

8.1.4.2 Maximum All Rods Out Hot Full Power F_Q

The heat flux hot channel factor (or total peaking factor), F_Q, is the ratio of maximum local heat flux in the hot channel to the average fuel rod heat flux. The maximum F_Q value is used to calculate the PLHGR (Section 8.1.4.3). The limit on F_Q is established to

ensure none of the fuel design criteria are exceeded and the assumptions made in the accident analysis remain valid.

Control of the core power distribution with respect to the peaking factors, $F_{\Delta H}$ and F_Q, ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analysis.

F_Q is calculated in SIMULATE5 during the cycle depletion for all times in life, at HFP and ARO conditions. The NRF for the F_Q peaking factor (Section 7.7) is applied to increase the calculated value, resulting in a conservative prediction of F_Q.

8.1.4.3 Peak Linear Heat Generation Rate

The PLHGR is the maximum rate of heat generation per unit length of the fuel rod. PLHGR is calculated from the maximum total peaking factor (F_Q) multiplied by the average linear heat generation rate.

The constraint on PLHGR helps ensure that no fuel performance limitations are exceeded. The PLHGR constraint limits the fuel temperature so it will remain below the centerline melt criterion, limits the peak cladding temperature so cladding-coolant chemical interactions remain in the acceptable range, and limits the cladding strain to ensure cladding integrity is maintained.

The NRF for the F_Q peaking factor (Section 7.7) is applied to the maximum F_Q calculated by SIMULATE5 during the cycle depletion to increase the calculated value for a conservative determination of PLHGR.

8.1.4.4 Peak Rod Burnup

Fuel burnup is a measure of fuel depletion that represents the integrated energy output of the fuel in GWd/MTU and is a useful means for quantifying fuel exposure criteria. Peak rod burnup is defined as the fuel rod with the maximum axially integrated burnup across the entire core.

The constraint on peak rod burnup represents the licensed peak rod burnup for the NuScale fuel rods. The peak rod burnup is determined at the EOC (maximum cycle length) from the SIMULATE5 cycle depletion at HFP and ARO. No bias or reliability factor is applied to this term.

8.1.4.5 Maximum Hot Full Power Boron Concentration

Boron is a strong absorber of thermal neutrons and is dissolved in the primary coolant for reactivity control. Soluble boron, along with the CRAs, comprise the two independent systems used to control core reactivity.

The maximum HFP CBC is the maximum value calculated in SIMULATE5 during the cycle depletion for all times in life, at HFP and ARO conditions. An uncertainty is applied

such that the determination of the maximum HFP CBC is conservative. The upper (positive) NRF for the CBC (Section 7.1) is applied, resulting in a conservative prediction of the maximum HFP boron concentration.

8.1.4.6 Minimum Refueling Critical Boron Concentration

The minimum refueling boron concentration ensures that all reactor coolant system filled structures with access to the reactor vessel during refueling maintain a minimum boron concentration to keep the reactor core subcritical during refueling operation. The boron concentration is specified to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with CRAs and fuel assemblies assumed to be in the most adverse configuration allowed by plant procedures.

The minimum boron concentration is based on the core reactivity at the beginning of each fuel cycle (the end of refueling). The lower (negative) NRF for CBC (Section 7.1) is applied, resulting in a conservative prediction of the minimum refueling boron concentration.

8.1.4.7 Most Positive Moderator Temperature Coefficient

The reactor core and its interaction with the reactor coolant system must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases. The MTC is an input to the safety analysis and is used in modeling the reactor response during accidents and transients. The design constraints on positive MTC ensure overall negative reactivity feedback characteristics of the reactor core during normal operation. Therefore, it is necessary to know the MTC over a range of moderator temperatures and core power conditions.

The MTC is determined from SIMULATE5 calculations at various core conditions including HZP, and multiple power levels at BOC, and during the SIMULATE5 depletion calculation for all times in life at HFP and ARO. The MTC is determined by changing the moderator temperature from the reference temperature, causing a change in the reactivity. The upper (positive) NRF for MTC (Section 7.3) is applied, resulting in a conservative prediction for the most positive MTC.

8.1.4.8 Most Negative Moderator Temperature Coefficient

For normal operations, a negative MTC is desired to prevent core reactivity insertions in response to an increase in the moderator temperature. However, for accident scenarios where the reactor coolant system undergoes a rapid cool-down, a negative MTC will add reactivity to the core and could challenge the core safety limits. The core design constraint for negative MTC is established to help mitigate the consequences of a cool-down event.

The MTC is determined from SIMULATE5 calculations at EOC HFP and ARO conditions. The lower (negative) NRF for MTC (Section 7.3) is applied, resulting in a conservative prediction for the most negative MTC.

8.1.4.9 Minimum Shutdown Margin

Shutdown margin is the instantaneous amount of reactivity that the reactor is subcritical by or would be subcritical from its present condition assuming all CRAs (shutdown and regulating) are fully inserted, except for the single CRA of highest reactivity worth, which is assumed to be fully withdrawn. The core design constraint for shutdown margin assures that there is sufficient SCRAM worth following a reactor trip, under all credible operating conditions, to shut the reactor down and prevent safety limits from being exceeded.

8.2 Physics Input to Safety Analysis

Core physics parameters are calculated as part of the safety analysis of the NuScale reactor core in order to analyze FSAR Chapter 15 transient and accident scenarios. FSAR Chapter 15 scenarios must be conservatively analyzed to ensure that fuel design limits, system overpressure design limits, and dose consequences are not exceeded. The transient and accident analyses are performed using a set of assumptions, which are combined in a consistent and conservative manner to produce conservative results. In this manner, the safety analysis bounds the expected operating conditions of the reactor core, and therefore the actual core response to a transient or accident scenario does not exceed the response predicted by analysis.

8.2.1 Determination of Safety Analysis Physics Parameters

Core physics parameters are calculated as part of the transient or accident analysis to provide input about the initial conditions and transient response of the core. These parameters are calculated conservatively to provide input to safety analysis that bounds expected core operational behavior. The relative importance of an individual physics parameter varies between transients; however, it is possible to identify for each event, a set of physics parameters that are significant and directly affect the results of the analysis. These parameters, which directly impact the course or consequence of the FSAR Chapter 15 analyses, are designated as safety analysis physics parameters. Once these key parameters have been determined, then the impact to these parameters due to a change in the core loading pattern or operating history can be assessed. A conservative value can then be selected for analysis, or several combinations can be analyzed to ensure the transient response is bounded.

8.2.2 Calculation Methodology of Safety Analysis Physics Parameters

SIMULATE5 3D models are used to calculate core physics parameters and power distributions for input to the safety analysis. The models used to calculate these parameters are based on the loading pattern and operational history of previous cycles. Section 8.1 detailed the NuScale core design process, where certain core physics parameters have been chosen as design criteria to establish the likelihood of acceptable loading pattern designs for safety analysis evaluation. Detailed core physics calculations follow the core design process to establish the physics parameter values to be used in the safety analysis. For each physics parameter of interest, calculations are performed using conservative assumptions and core modeling options to determine conservative

parameters and biases, and NRFs are applied to these calculated values. These parameters are then evaluated to determine if they are bounded by the values used in the reference safety analysis. If they are not bounded, then these parameter values are passed to safety analysis for evaluation.

8.2.2.1 Initial Core Power Distribution

The safety analysis assumes that any core power distribution permitted within normal operating limits is a valid initial condition for a transient or accident scenario. Licensing limitations require that core power distribution remains within prescribed limits during power operation. Power distributions are calculated for nominal conditions and anticipated transient or accident conditions. For transient or accident scenarios where the initial power distribution has a significant impact on the event, the power distribution is manipulated within the operating limits, including the limiting conditions, to provide conservative results. Power peaking limits are provided as limits to F_Q and $F_{\Delta H}$. The NRF for the peaking factors (Section 7.7) is applied to increase the calculated values, resulting in a conservative prediction of the peaking factors for use in safety analyses.

8.2.2.2 Reactivity Coefficients

Reactivity coefficients describe the dynamic behavior of the reactor during reactor transients, including power maneuvers and accident conditions. Reactivity coefficients describe the change in reactivity caused by a change in a reactor parameter, such as power, moderator density, fuel temperature, or boron concentration. Reactivity coefficients have a dependence on core exposure, so they are calculated at multiple exposure points during cycle lifetime. The state points that reactivity coefficients are evaluated at are chosen to ensure that the assumptions made in a particular transient or accident analysis remain bounded.

8.2.2.3 Moderator Temperature Coefficient

The MTC is defined as the change in core reactivity resulting from a change in moderator temperature. The NuScale reactor core is designed with a negative MTC at nominal power conditions for all times in life to help protect the reactor core from rapid increases in reactivity. The SIMULATE5 core model is used to calculate bounding (least negative or most positive, and most negative) coefficients for the NuScale reactor core. The particular transient or accident scenario determines which bounding MTC to use for that analysis. For example, events that increase core power and moderator temperature would use the bounding least negative (or most positive) MTC, and events that decrease moderator temperature would use the most negative MTC.

The MTC is calculated by changing the moderator temperature from the reference temperature, causing a change in the reactivity. The NRF for MTC (Section 7.3) is applied, resulting in a conservative prediction for MTC. For events that require the most negative MTC, the lower (most negative) NRF is used, and for events that require the most positive MTC, the upper (least negative or most positive) NRF is used.

8.2.2.4 Fuel (Doppler) Temperature Coefficient

The Doppler or FTC is defined as the change in core reactivity resulting from a change in fuel temperature. The 3D SIMULATE5 core model is used to calculate bounding coefficients for the NuScale reactor core. The FTC is calculated over a range of fuel temperatures through variations in the power level, and at multiple exposure points. At each state point considered, power is varied, which causes the fuel temperature to change, and SIMULATE5 calculates the FTC by changing the fuel temperature from the reference temperature at the specified power level, causing a change in reactivity.

The NRF for FTC (Section 7.4) is applied, resulting in a conservative prediction for FTC. For events that require the most negative FTC, the lower NRF is used, and for events that require the most positive FTC, the upper NRF is used.

8.2.2.5 Kinetics Parameters

The reactor response to transient and accident scenarios is determined, predominantly, by the kinetics properties of the reactor. The delayed neutron parameters are important during rapid reactivity excursion events. The minimum β_{eff} is conservative for analyzing a rapid reactivity insertion event such as the rod ejection accident. However, β_{eff} is not significant for analyzing transient or accident scenarios that are not characterized by a rapid reactivity excursion. The prompt neutron lifetime (ℓ^*) is also important for transient and accident scenarios with rapid changes to reactivity.

The SIMULATE5 core model is used to calculate the kinetics parameters for transient and accident analysis. Delayed neutron fractions and decay constants are calculated for six effective delayed neutron groups. The total β_{eff} is the sum of the six group effective fractions. Calculations are performed at BOC and EOC with nominal conditions, and bounding values are used for the transient or accident scenario analyzed. The kinetic parameter calculations provide the necessary output to construct a point reactor kinetics model with six delayed neutron groups, which can be used to calculate the rate of change in power from a reactivity insertion event.

8.2.2.6 Control Rod Assembly Worth

The primary purpose of the CRAs is to provide the ability to shut down the reactor core during normal operations and accident conditions. The NuScale reactor incorporates two CRA banks. The shutdown bank and the regulating bank are used to help shut down the reactor. However, during normal operation, the shutdown bank is positioned in the ARO configuration. The regulating bank can be used to offset reactivity changes, to conduct power maneuvers, and to maintain AO within the licensing limits during normal operation. During normal operation, the regulating bank may be positioned in the core to the power dependent insertion limit (PDIL). However, the regulating bank is typically kept in the ARO configuration for normal operation.

SIMULATE5 is used to calculate CRA worth at limiting power levels, including HFP and HZP, and limiting times in life, including BOC and EOC. The most adverse CRA configuration (from ARO to the PDIL) is determined for each event. Adverse axial power

shapes and limiting moderator temperatures are used to maximize the CRA worth. The NRF for CRA bank worth (Section 7.5) is applied depending on the application, such that the predicted worth is conservative (upper or positive NRF to maximize worth, and lower or negative NRF to minimize worth).

8.2.2.7 Shutdown Margin

Shutdown margin is defined as the instantaneous amount of reactivity that the reactor is subcritical by or would be subcritical from its present condition assuming all CRAs are fully inserted except for the single CRA of highest reactivity worth that is assumed to be fully withdrawn. The shutdown margin is calculated as the difference between the available worth and the required worth to shut down.

8.2.2.8 Critical Boron Concentration

The NuScale reactor core uses a combination of gadolinia-loaded fuel and soluble boron to control excess reactivity. The NuScale chemical volume control system (CVCS) controls the soluble boron concentration to compensate for fuel depletion and xenon burnout reactivity changes. The SIMULATE5 core model is used to predict the CBC as a function of reactor power, exposure, moderator temperature, and CRA positioning. The CBC is calculated for various core operating conditions and core configurations for safety analysis transient and accident scenarios. The NRF for CBC (Section 7.1) is applied depending on the application, such that the predicted boron concentration is conservative (upper or positive NRF to maximize boron concentration and lower or negative NRF to minimize boron concentration).

8.2.2.9 Boron Worth

The DBW is the change in core reactivity resulting from a change in soluble boron concentration. The DBW is dependent on the reactor power, exposure, moderator temperature, and CRA positioning. The DBW is calculated using the SIMULATE5 core model at various core exposure points, including BOC, MOC, and EOC. SIMULATE5 calculates the DBW by changing the boron concentration from the reference boron concentration at the specified power level. The NRF for DBW (Section 7.2) is applied depending on the application, such that the predicted boron worth is conservative (upper or positive NRF to maximize boron worth and lower or negative NRF to minimize boron worth).

8.2.3 Conservative Parameters

The relative importance of various physics parameters and the sensitivity to variations in the values of the parameters varies between transients. However, it is possible to identify for each event, a set of physics parameters that are significant and directly affect the results of the analysis. Table 8-2 provides a list of the FSAR Chapter 15 events relevant to the NuScale plant design along with the key physics parameters associated with each event. The application of the parameters is also included in the table. The NRFs determined in Section 7.0 are applied to each calculated parameter, such that the predicted value is conservative for the accident or transient scenario under analysis

(upper or positive NRF to maximize the predicted value and lower or negative NRF to minimize the predicted value).

Table 8-2. Key Physics Parameters for Transient and Accident Analysis

Key Parameter	Event	Physics Application
Power Distribution	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Maximum peaking
	Steam System Piping Failure	Maximum peaking
	Dropped CRA or Dropped CRA Bank	Maximum change - $F_{\Delta H}$
	Statically Misaligned CRA	Maximum change - $F_{\Delta H}$
	Single CRA Withdrawal	Maximum change - $F_{\Delta H}$
	CRA Ejection	Maximum total peak - F_Q
MTC	Decrease in Feedwater Temperature	Most negative
	Increase in Feedwater Flow	Most negative
	Increase in Steam Flow	Most negative
	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Most negative
	Steam System Piping Failure	Most negative
	Loss of External Load	Most positive
	Turbine Trip	Most positive
	Loss of Normal Feedwater Flow	Most positive
	Uncontrolled CRA Bank Withdrawal at Power	Bounding BOC and EOC conditions
	Uncontrolled CRA Withdrawal at Subcritical or Low Power	Most positive
	Dropped CRA or Dropped CRA Bank	Bounding BOC and EOC conditions
	Statically Misaligned CRA	Bounding BOC and EOC conditions
	CVCS Malfunction that Increases Reactor Coolant Inventory	Most positive
	Inadvertent Opening of a Pressurizer Safety Relief Valve	Most positive
	Feedwater System Pipe Break	Most positive
	Steam Generator Tube Rupture	Most positive
	CRA Ejection	Most positive
	Single CRA Withdrawal	Most positive
FTC	Decrease in Feedwater Temperature	Most positive
	Increase in Feedwater Flow	Most positive
	Increase in Steam Flow	Most positive
	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Most positive
	Steam System Piping Failure	Most positive
	Loss of External Load	Most positive
	Turbine Trip	Most positive
	Loss of Normal Feedwater Flow	Most positive
	Uncontrolled CRA Bank Withdrawal at Power	Bounding BOC and EOC conditions
	Uncontrolled CRA Withdrawal at Subcritical or	Most positive

Key Parameter	Event	Physics Application
	Low Power	
	Dropped CRA or Dropped CRA Bank	Bounding BOC and EOC conditions
	Statically Misaligned CRA	Bounding BOC and EOC conditions
	CVCS Malfunction that Increases Reactor Coolant Inventory	Most positive
	Inadvertent Opening of a Pressurizer Safety Relief Valve	Most positive
	Feedwater System Pipe Break	Most positive
	Steam Generator Tube Rupture	Most positive
	CRA Ejection	Most positive
	Single CRA Withdrawal	Most positive
Reactivity Insertion Rate	Uncontrolled CRA Bank Withdrawal at Power	Minimum to maximum
	Uncontrolled CRA Withdrawal at Subcritical or Low Power	Maximum
CRA Worth	Dropped CRA or Dropped CRA Bank	Maximum - dropped rod
	Statically Misaligned CRA	Maximum - stuck rod
	Inadvertent Decrease in Boron Concentration in RCS	Maximum - stuck rod
	CRA Ejection	Maximum - ejected rod
	Single CRA Withdrawal	Maximum - single CRA
CBC	Inadvertent Decrease in Boron Concentration in RCS	Maximum
Initial Boron Concentration	Inadvertent Decrease in Boron Concentration in RCS	Closest to CBC
Delayed Neutron Fraction	CRA Ejection	Minimum
Neutron Lifetime	CRA Ejection	Minimum

8.3 Start-Up Physics Testing

The purpose of startup physics testing (low power testing and power ascension testing) is to measure certain neutronic characteristics of a newly loaded core and compare those measurements with predictions to verify that the reactor core is operating as designed, validate the analytical models, and verify the correctness or conservatism of assumptions used in the safety analyses. NuScale expects to use CMS5 to perform startup physics testing predictions. If the measurements agree with the CMS5 predictions, then there is reasonable assurance that the reactor core will operate as designed. Also, as detailed in Section 7.0 comparisons between measured and calculated values during start-up testing may be used to assess the continued applicability of the NRFs, or the data may be used to update the NRFs depending on the agreement between, or any trends in the data between the measured and calculated values.

There are five characteristics that must be confirmed for each newly loaded core:

- reactivity balance

- reactivity control
- power distribution
- shutdown capability
- shutdown requirement

The reactivity balance is determined by the measurement of the HZP ARO boron concentration. Agreement between the measured value and the predicted value means that the total amount of fissile material and absorbing material in the core is consistent with the design. Reactivity control is determined by measurement of the HZP ITC and comparison to the predicted value. Agreement means that the behavior of the core to temperature changes is consistent with the design. Power distributions are confirmed by measuring the neutron flux throughout the core at low, intermediate, and higher power levels and comparing to design predictions. The power distribution at lower power levels must be confirmed before escalating to higher power levels. Control rod assembly worth measurements confirm the capability of the core to be shut down, and the shutdown requirement is confirmed by measuring the power defect (reactivity difference between zero power and full power). If the results of these measurements agree with the design predictions, then there is reasonable assurance that the newly loaded as-built core will operate as designed.

Design calculations using CMS5 will be used to generate predictions for physics parameters for comparison to measured values. The model for start-up testing will be the model used for the design analysis, which is based on the current as-built core design. Calculations are performed at BOC, modeling the core conditions and configurations expected for start-up testing, and various physics parameters are determined to compare against measured values. At a minimum, the following parameters will be measured and compared to predicted values during startup physics testing:

- critical boron concentration
- isothermal temperature coefficient
- control rod assembly worth
- differential boron worth
- power tilt (flux/power distribution symmetry)
- core power distributions (low, intermediate, and full powers)

8.4 Core Predictions for Operations Support

CMS5 is used to generate physics data to support core operations during the cycle. The model used for the design analysis that is based on the as-built core design will also be used to calculate physics parameters to support operations. The code is used to predict physics parameters from BOC to EOC as a function of varying reactor conditions, including power level and moderator temperature. This information assists operations in

understanding reactor behavior, and helps determine the response to plant maneuvers. The following parameters may be generated to support operations:

- CBC at various plant conditions as a function of core average burnup;
- minimum boron concentration to assure reactor shutdown margin is provided at various moderator temperatures and burnups;
- DBW as a function of power level, moderator temperature, and burnup;
- CRA bank worth as a function of power level, core average moderator temperature, and burnup;
- the change in reactivity due to changes in power, moderator temperature, and fuel temperature as a function of power level, burnup, and moderator temperature;
- axial and radial power distributions at HFP as a function of burnup and power level;
- the xenon and samarium worth as a function of power level and burnup; and
- information pertaining to reactor kinetics parameters and startup reactivity as a function of burnup and power level.

The appropriate NRFs are applied to the CMS5 core physics information provided to support operations.

8.5 Core Follow Predictions

CMS5 is used to generate reactor physics parameters for comparison to measured values during the operating cycle. The measured reactor physics core performance parameters are compared with design depletion predictions to ensure that the reactor core is performing as designed. SIMULATE5 calculations are performed at various depletion points throughout the cycle and key physics parameters are determined to compare against measured values. The model for core follow predictions will be the model used for the design analysis, which is based on the as-built core design, and will take into account the operating history of the reactor. The model will be configured in a manner reflecting the state of the core at the time the physics parameters are measured. Core follow comparisons may be used to assess the continued applicability of the NRFs. The data may be used to update the NRFs depending on the agreement of the data or presence of any trends in the data between the measured and SIMULATE5 calculated values.

At a minimum, the following parameters are measured during the cycle and compared to predicted values:

- critical boron concentration
- total peaking factor , F_Q
- enthalpy rise hot channel factor, $F_{\Delta H}$
- axial offset

- power tilt (flux/power distribution symmetry)
- radial power distribution
- axial power distribution

9.0 Summary and Conclusions

This report describes the validation and benchmarking performed using the CMS5 code system for application to the NuScale reactor core design. This effort consisted of higher-order code validation using MCNP6, empirical benchmarks to experimental reactors, and comparisons of calculated physics parameters to operating data. Although this CASMO5 and SIMULATE5 code suite do not have a commercial power licensing history, the predecessors CASMO-4 and SIMULATE-3 have an extensive licensing history for use in core physics calculations.

Based on the results submitted in this report, the CMS5 code methodology applies to all steady-state reactor physics calculations for the NuScale fuel and core design. The accuracy of the methodology along with the NRFs described in this report are sufficient to be used in licensing applications for core design, safety analysis inputs, startup and core follow predictions, and operations support.

This report meets the nuclear method and design regulatory requirements as described in 10 CFR 50 Appendix A General Design Criteria and the guidance of SRP, Section 4.3.

10.0 References

10.1.1 10 CFR 50, Appendix A. General Design Criteria for Nuclear Power Plants.

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10.1.3 NUREG-0800 SRP, Section 15.0.2, “Review of Transient and Accident Analysis Methods,” Revision 0, 2005.

10.1.4 Regulatory Guide 1.203, Transient and Accident Analysis Methods. December 2005.

10.1.5 SSP-08/405 Rev 2 CASMO5 Methodology Manual. March 2014.

10.1.6 SSP-07/431 Rev 7. CASMO5 User’s Manual, December 2013.

10.1.7 SSP-10/437 Rev 3. CMSLINK5 User’s Manual. December 2013.

10.1.8 SSP-10/465 Rev 3. SIMULATE5 Methodology. January 2014.

10.1.9 SSP-10/438 Rev 4. SIMULATE5 Advanced Three-Dimensional Multigroup Reactor Analysis Code. December 2013.

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10.1.11 ASME NQA-1-2008, Quality Assurance Requirements for Nuclear Facility Applications, 2008 Edition.

10.1.12 ASME NQA-1a-2009 Addenda, Quality Assurance Requirements for Nuclear Facility Applications.

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10.1.15 NUREG-0085, The Analysis of Fuel Densification, June 1976.

10.1.16 *NIST/SEMATECH e-Handbook of Statistical Methods*, http://www.itl.nist.gov/div898/handbook/, October 2013.

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10.1.24 NEA/NCS/DOC(95)03/IV, LEU-COMP-THERM-077. Critical Loading Configurations of the IPEN/MB-01 Reactor. International Handbook of Evaluated Criticality Safety Benchmark Experiments.

10.1.25 NEA/NCS/DOC(95)03/IV, LEU-COMP-THERM-091. Critical Loading Configurations of the IPEN/MB-01 Reactor with UO_2, Stainless Steel and Gd_2O_3 Rods. International Handbook of Evaluated Criticality Safety Benchmark Experiments.

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10.1.32 NP-TR-1010-859-NP-A, NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant, Revision 3, October 2016.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0618-60702

NuScale Power, LLC
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www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, 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 topical report reveals distinguishing aspects about the method by which NuScale develops its nuclear analysis of the reactor core.

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 "Nuclear Analysis Codes and Methods Qualification." 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.

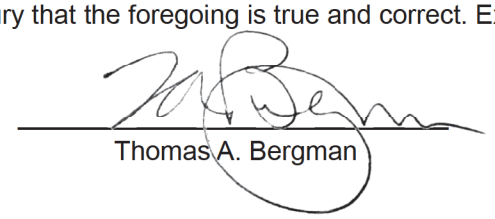
(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, 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 July 30, 2018.



Thomas A. Bergman