PHYSOR 2012 – Advances in Reactor Physics – Linking Research, Industry, and Education
Knoxville, Tennessee, USA, April 15-20, 2012, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2012)

COMPARISON OF A NUSCALE SMR CONCEPTUAL CORE DESIGN USING CASMO5/SIMULATE5 AND MCNP5

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ABSTRACT

A key issue during the initial startups of new Small Modular Reactors (SMRs) is the lack of operational data for reactor model validation. To help better understand the accuracy of the reactor analysis codes CASMO5 and SIMULATE5, higher order comparisons to MCNP5 have been performed. These comparisons are for an initial core conceptual design of the NuScale reactor. The data have been evaluated at Hot Zero Power (HZP) conditions. Comparisons of core reactivity, fuel temperature coefficient (FTC), and moderator temperature coefficients (MTC) have been performed.

Comparison results show good agreement between CASMO5/SIMULATE5 and MCNP5 for the conceptual initial core design.

Key Words: SMR, CASMO5, SIMULATE5, NuScale, MCNP5

1 INTRODUCTION

The development of SMR's presents unique issues for reactor vendors and potential operators. A key issue addressed in this paper is the lack of operating history for SMRs. This lack of operating history creates a challenge in validating predictions of neutronic parameters during the startup of this new design.

The NuScale reactor is analyzed here using current state-of-the-art lattice physics code, CASMO5 [1] and reactor simulator, SIMULATE5 [2]. The reference solution produced for validation uses a higher order computational method. In this case it is the continuous energy Monte Carlo code MCNP5 [3].

The neutronic parameters that are required to be validated can be parsed into two groups, global parameters and local parameters.

Global reactor parameters are those that are derived from examining the state of the entire reactor core. These include reactivity coefficients such as moderator temperature, fuel

temperature (Doppler), and boron. As well as the reactivity associated with control rods. The combinations of these parameters defines the operational characteristics of the reactor and are a majority of the quantities measured during start-up physics testing.

Local reactor parameters are those calculated over small geometric regions at a given core state. The parameters include those that are associated with the Specified Acceptable Fuel Design Limits (SAFDLs). These include pin power quantities that are derived from the local reaction rate distributions. The resulting integrated quantities provide fuel pin exposure, local linear heat generation rate (Fq) and enthalpy rise hot channel factors ($F_{\Delta}h$) as examples.

As a first step this paper focuses on global neutronic parameters that are typically verified during the start-up physics testing. These tests are a component of what is used to verify that core design predictions are sufficiently accurate to allow the unit to operate within the domain analyzed by the safety analysis.

In the future the next steps of this validation will focus on examining local reaction rate distributions.

2 REACTOR OVERVIEW

The NuScale SMR is a scalable modular integral pressurized water reactor (iPWR) design. Each reactor module has its own turbine generator set that is capable of producing 45MWe. A scalable plant is capable of supporting 12 modules within a single reactor building for an overall capacity of 540 MWe. The independent modularity of each reactor represents a reduced operational risk for electricity generation and a new way of performing refueling and maintenance.

The passively safe design of the reactor uses natural circulation as the method to transfer heat from the nuclear fuel to the steam generator as well as for the various safety systems. The integral nature of the design allows for all of the primary components to be contained within the main pressure vessel. These include the core, pressurizer, and steam generators. The control rod drive mechanisms are fixed to the top of the reactor vessel.

The NuScale reactor employs several unique passive safety systems: the high-pressure steel containment, the decay heat removal system (DHRS), and the containment heat removal system (CHRS). The high-pressure containment houses and protects the nuclear steam supply system (NSSS), control rod drive mechanisms, and associated safety-related systems from external hazards. This containment is submerged in the reactor pool, which provides a passive heat sink under loss of coolant accident (LOCA) conditions. The DHRS provides core cooling when normal feedwater is not available, and removes residual heat generated by fission product decay from the reactor core. The CHRS rapidly reduces the containment pressure and temperature and maintains them at acceptably low levels for extended periods of time after a LOCA event.

The conceptual layout of a single module is shown visually in Figure 1.

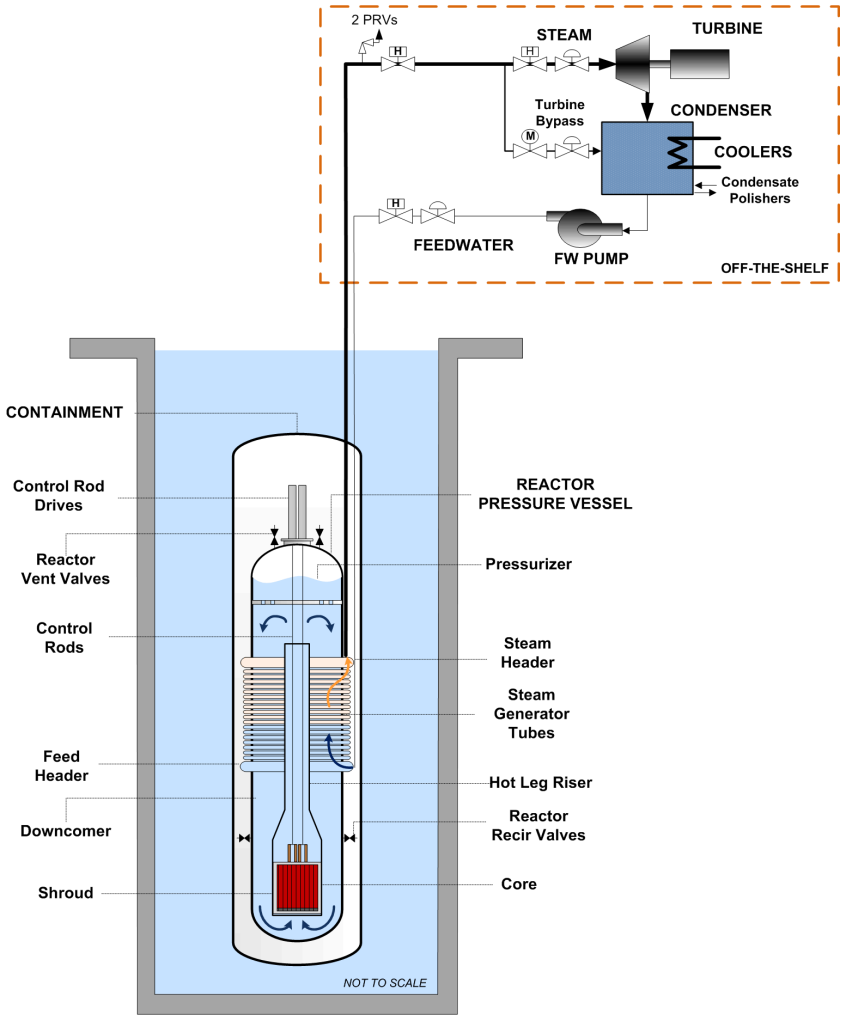

Figure 1: NuScale Passively Safe Reactor Concept

2.1 Reactor Core

The NuScale reactor core operates at a reduced power density, approximately 50% , compared to a conventional PWR. Current cycle length design target is 24 months. Geometrically the core consists of 37 fuel assemblies. The fuel assemblies are arranged to approximate a right cylinder with seven assemblies comprising the widest dimension. The reactor core is controlled like a conventional PWR with a combination of soluble boron and control rods. The design currently contains 16 control rods arranged into two banks. Bank 1 is designated for reactivity control during startup and operation. Bank 2 is designated as a shutdown bank for use when the reactor receives a SCRAM signal. The lower right quarter core layout is shown below in Figure 2 where the control rod positions are also indicated.

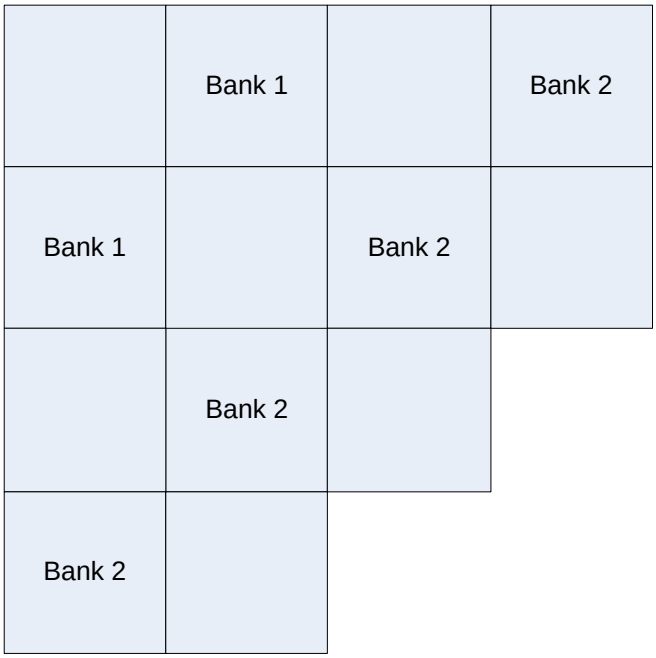

Figure 2: NuScale Reactor Quarter Core Layout

2.2 Fuel Assemblies

The NuScale fuel assemblies are similar in design to conventional PWRs. The main difference is a reduced active height of approximately 50% compared to a conventional assembly. The fuel assembly lattice is 17x17 (Figure 3) with 264 fueled locations, 24 guide tubes, and one central instrument tube. The fuel is conventional Zircaloy clad UO_2 ceramic with enrichments below 5 w/o U^{235}. An integral burnable absorber of Gadolina ($Gd_2O_{3)}$ is used in some assemblies to control beginning of cycle moderator temperature coefficient (MTC) and power peaking.

The initial conceptual core design consists of four different enrichments and five assembly types. Of the five assembly types, two contain Gadolinia burnable absorber in some of the fuel pins. All the fuel assemblies contain low enriched axial blankets to minimize leakage.

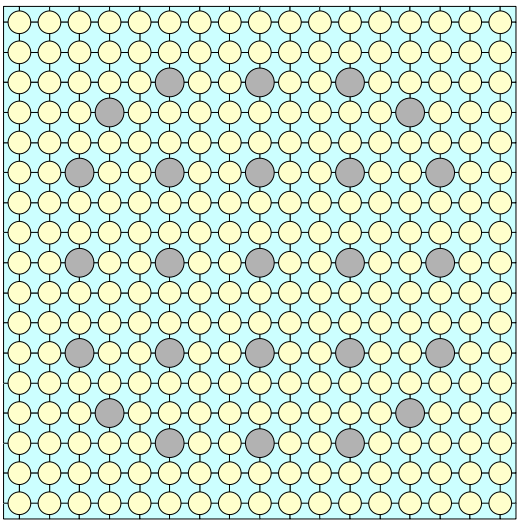

Figure 3: NuScale 17x17 Fuel Lattice

2.3 Control Rods

The control rods are similar to a conventional PWR design where the 24 individual rodlets attach to a central spider assembly and hub. The 24 rodlets are inserted into the guide tubes shown in Figure 3. The central instrument tube is empty in assemblies containing control rods. The control rod material for the conceptual design is Silver, Indium, Cadmium (AgInCd) alloy.

3 MODEL DESCRIPTION

There were two independent models generated for this benchmark. One created for the Monte Carlo code MCNP5 and the second created for deterministic CASMO5/SIMULATE5. The key assumptions and features are described below.

3.1 MCNP5 Model

The MCNP5 model of the NuScale reactor module comprised several units: the core, reflector, barrel, and reactor pressure vessel. The core was modeled as an 11 x 11 lattice, and each core lattice element is a 17x17 fuel/fuel-control assembly, an assembly (no fuel or control segment) with only upper and lower core plates, a reflector or coolant. Figure 4 shows both plan and elevation views of the MCNP5 model of one quarter of the NuScale reactor module.

The MCNP5 model of the 17x17 fuel assembly consisted of the lattice cells for fuel, control rods, instrument and guide tubes, top nozzle, bottom nozzle, and spacer grids. The units that make up the lattice cell/fuel pin were: top end plug, plenum, plenum spring, top blanket, enriched zone, bottom blanket, bottom end plug, fuel-cladding gap, and cladding. The repeated-structures capability of MCNP5 was used to build the lattice cells that comprise the UO_2 or

UO_2+Gd_2O_3 fuel elements at the beginning of cycle (BOC), control rod elements, instrument tubes, and guide tubes. . Material compositions used in the MCNP5 model were matched to those specified for the CASMO5 case matrix.

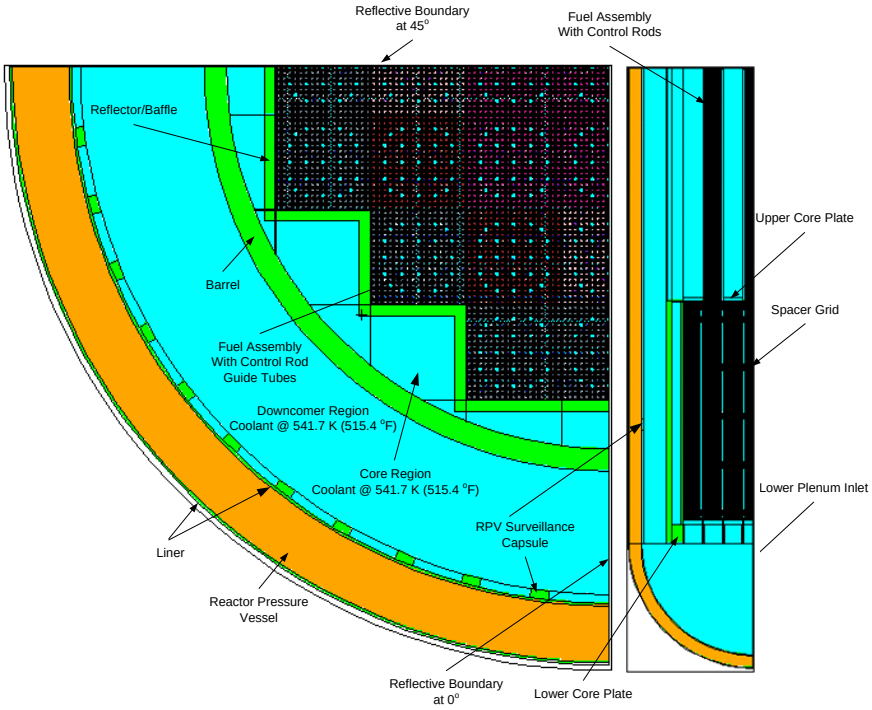

Figure 4: MCNP5 model of the NuScale reactor module shows plan and elevation views of the core lattice, reflector, barrel, downcomer and pressure vessel. Temperature reflects Hot Zero Power condition.

The MCNP5 code version 1.51 was used to model the NuScale reactor module core configuration with continuous energy neutron cross sections data evaluations from ENDF/B-VII.0 [5] libraries. The free gas scattering kernel model was chosen, except for hydrogen in light water for which the appropriate S(α,β) data [3] at the appropriate temperatures were used to account for chemical molecular binding and crystalline effects that become important as the neutron's energy becomes sufficiently low—below 4 eV. Doppler broadened cross sections were processed, using the cross sections processing code Makxsf [4], at 541.7 K (515.4 ^{o}F) for HZP isothermal calculations

For MTC calculations, the perturbed moderator temperature was processed at 552.8 K (535.4 ^{o}F); for ITC calculations, cross sections for all materials in the model were were processed at 552.8 K (535.4 ^{o}F) for the perturbed condition. For FTC two perturbed calculations were performed: one with fuel and cladding cross sections processed at 552.8 K (535.4 ^{o}F) and the other with only the fuel cross sections processed at 552.8 K (535.4 ^{o}F). The average of the two results was approximated to be the FTC best estimate value. Ideally cross sections for the

cladding should be processed so as to reflect the true temperature profile since the FTC is very sensitive to small temperature variations.

In the MCNP5 criticality calculations 80,000 starting fission neutrons were used, and 150 cycles were skipped, to improve the spatial distribution of the source in the core. The criticality calculations were run for a quarter core configuration with reflective boundary conditions at 0 and 45 degrees (see Figure 4). An initial guess of k-effective =1.0 was used, and a total of 2,000 k-effective cycles were run for all calculations.

3.2 CASMO5/SIMULATE5 Model

The CASMO5/SIMULATE5 model consisted of two main parts. These are the 2D lattice segment models constructed in CASMO5 and the 3D core simulation model constructed in SIMULATE5.

The recommended default input values for use in conventional PWRs were used. This was a choice in this initial benchmark of the codes. This may be reexamined in the future if the operating domain or characteristics of the NuScale reactor warrant it.

3.2.1 CASMO5

The lattice calculations were performed for any unique combination of fuel enrichment and burnable absorber that may comprise an axial zone in a fuel assembly. For the NuScale core design there are six unique lattice types. For each lattice a series of calculations were performed to cover the potential operating domain of the reactor. In this model the spacer grids were treated explicitly with branch calculations.

Three reflector segments were also run to determine the correct behavior at the boundaries of the reactor. The three segments included a radial reflector to represent the core periphery, and a top and bottom reflector. The radial reflector represented the explicit geometry of the boundary fuel assembly and adjoining baffle, water, and barrel regions as needed. The top and bottom reflectors were three zone regions that contain a homogenization of the appropriate materials such as moderator, springs, cladding, endplugs, and nozzles.

The geometry was specified using cold dimensions and the thermal expansion model in CASMO5 was deactivated to be consistent with the MCNP5 model. Additionally the resonance upscatter model was turned off to be consistent with the MCNP5 modeling. The standard case matrix, needed to generate all the cross sections and derivatives, was run. The cross section library for CASMO5 was developed from ENDF/B-VII.0 [5].

3.2.2 SIMULATE5

The 3D simulator model was constructed using 25 axial zones and four radial zones per fuel assembly. Each fuel assembly type was modeled and placed within the core model. The control rods were modeled as the two banks shown in Figure 3. The spacer grids that were present in the

active core region were modeled explicitly. The recommended default of four energy groups was used in this calculation.

The thermal hydraulic detail in this model was minimal. The focus of the comparisons was on isothermal zero power conditions.

4 VALIDATION RESULTS

To capture the data needed to simulate start-up physics testing several calculations were performed. They can be easily divided into two categories: the first, core reactivity predictions, the second, reactivity coefficients.

The core conditions for each calculation compared are shown in Table 1 and Table 2. All calculations began at zero power isothermal conditions with All Rods Out (ARO) and a boron concentration (BC) predicted by SIMULATE5. The resulting eigenvalue predictions and coefficients were compared between the two codes.

Table 1: Core Reactivity Calculation Conditions

Calculation #	Description
ARO	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = All Rods Out
Bank 1@ 50%	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = Bank 1 @ 50% Withdrawn Bank 2 @ 100% Withdrawn
Bank 1@ 0%	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = Bank 1 @ 0% Withdrawn Bank 2 @ 100% Withdrawn

Table 2: Reactivity Coefficient Calculation Conditions

Coefficient	Initial Condition	Perturbed Condition
MTC*	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = All Rods Out	Power = 0.0% **$T_{Moderator}$ = 535.4 °F, 552.8 K** T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = All Rods Out
ITC*	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = All Rods Out	Power = 0.0% **$T_{Moderator}$ = 535.4 °F, 552.8 K** **T_{Fuel} = 535.4 °F, 552.8 K** Boron Conc. = 1303 ppm Rod Position = All Rods Out
FTC	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = All Rods Out	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K **T_{Fuel} = 535.4 °F, 552.8 K** Boron Conc. = 1303 ppm Rod Position = All Rods Out
Boron	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K Boron Conc. = 1303 ppm Rod Position = All Rods Out	Power = 0.0% $T_{Moderator}$ = 515.4 °F, 541.7 K T_{Fuel} = 515.4 °F, 541.7 K **Boron Conc. = 1313 ppm** Rod Position = All Rods Out

* For SIMULATE5 the ITC and MTC calculations are performed at the midpoints between cross-section library temperatures (603.7, 567.4, 531.1, 469.1°F). This is done to eliminate the cross-section interpolation noise. The calculated value presented in Table 4 is linearly interpolated from the two nearest calculation points. A linear interpolation was deemed appropriate as the MTC behavior in that region displayed linear behavior.

The core reactivity prediction results are shown in Table 3. The results compared between the two code systems show very good agreement for rodded and unrodded conditions.

Table 3: Core Reactivity Calculation Results

Calculation	CASMO5/SIMULATE5	MCNP5	Difference
ARO	0.99999	1.00014 ± 0.00005	-15 pcm
Bank 1@ 50 %	0.98940	0.98956 ± 0.00006	-16 pcm
Bank 1@ 0 %	0.97463	0.97493 ± 0.00006	-30 pcm

The reactivity coefficient prediction results are shown in Table 4. The results compare well between the two codes.

The results for the reactivity perturbations in the model, boron concentration, ITC, MTC and FTC, show very good agreement. However, these differences lie slightly outside the range of the uncertainty in the Monte Carlo calculations. The root cause of these differences could be attributed to several factors. One reason is how the temperature of the different cells (fuel, moderator, etc.) within the MCNP model was treated. While thermal scattering law data is used for the bound hydrogen in the water moderator. Free gas scattering is used to model the other materials. The temperature of the free gas treatment for these materials is specified using the temperature card (TMP), in units of MeV. In this study the TMP card is not used and the temperatures default to 293K.

The MCNP5 calculations for some of the perturbations have small resulting reactivity changes. Here the resulting the uncertainties in the MCNP5 results begin to contribute significantly compared to the magnitude of the coefficients. For example in the MTC calculation with a perturbation of 20°F the corresponding coefficient value is -2.50 pcm/°F. If we look at the range of values based on the 99 percent keff (0.99949 to 0.99978) confidence interval for MTC, the results range from -1.8 to -3.25 pcm/°F. While the first thought may be to perform a larger perturbation there is a limit as water density becomes much less linear with the approach to saturation.

Given some of these modeling choices and uncertainty limitations the results still agree well.

Table 4: Reactivity Coefficient Calculation Results

Coefficient	CASMO5/SIMULATE5	MCNP5	Difference
MTC	-2.96* pcm/°F	-2.50 pcm/°F ($T_{Moderator}$ = 515.4°F, keff = 1.00014 ± 0.00005; $T_{Moderator}$ = 535.4°F, keff = 0.99964 ± 0.00005)	-0.46 pcm/°F
ITC	-4.62* pcm/°F	-4.00 pcm/°F (T_{Fuel}, $T_{Moderator}$ = 515.4°F, keff = 1.00014 ± 0.00005; T_{Fuel}, $T_{Moderator}$ = 535.4°F, keff = 0.99934 ± 0.00005)	-0.62 pcm/°F
FTC	-1.65 pcm/°F	-1.53** pcm/°F =-(1.4+1.65)/2 (TFuel = 515.4°F, keff = 1.00014 ± 0.00006; TFuel = 535.4°F, TCladd = 535.4°F keff = 0.99981 ± 0.00005); TFuel = 535.4°F, TCladd = 515.4°F keff = 0.99986 ± 0.00005)	-0.12 pcm/°F
Boron	-11.03 pcm/ppm	-11.91 pcm/ppm (1303 ppm, keff = 1.00014 ± 0.00005; 1313 ppm keff = 0.99895 ± 0.00005)	0.88 pcm/ppm

* Interpolated value see note from Table 2

** Average value of two FTC calculations: one with the fuel and cladding at 535.4°F and the other the fuel is 535.4°F and cladding at 515.4°F.

5 CONCLUSIONS

The calculations presented form the first step in assessing state-of-the-art production core design codes CASMO5 and SIMULATE5 for the NuScale SMR. The comparisons to the continuous energy Monte Carlo code MCNP5 showed good agreement for the global core wide reactivity parameters considered. Additional work is needed to quantify the local parameters and the uncertainties over a wider range of core design parameters.

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