VERA-CS Validation of Critical Experiments

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ABSTRACT

VERA is a suite of multiphysics codes which uses MPACT to model neutron transport in light water reactors (LWRs) [1]. In this paper, we validate MPACT by modeling critical experiments conducted at the IPEN/MB-01 and B&W facilities. We modeled critical loading experiments with a variety of different fuel pins and materials placed in the core. The experiments were modeled in two dimensions using MPACT and an axial buckling term. Default mesh parameters exist in MPACT for modeling larger reactor cores, and a mesh convergence study was performed to find appropriate mesh parameters for modeling the smaller critical reactors. The k_{eff} results show a consistent bias and small standard deviation for the IPEN/MB-01 reactor and a small bias and small standard deviation for the B&W facility. Overall, the results show that MPACT performs well for modeling small critical reactors.

KEYWORDS: IPEN,B&W,VERA,CASL,MPACT,Validation

1. INTRODUCTION

The Consortium for the Advanced Simulation of Light Water Reactors (CASL) has developed a suite of coupled multiphysics computer codes called VERA [1]. One component of VERA is the core simulator, VERA-CS, which uses MPACT as the transport solver to model LWRs. In this paper, we seek to validate MPACT by modeling small critical reactors using MPACT, utilizing the VERA common input [2], and comparing the results to experimentally measured benchmark data. Specifically, we model in two-dimensions experiments conducted at the IPEN/MB-01 reactor in São Paulo, Brazil and the Babcock and Wilcox (B&W) critical facility located at the Lynchburg Research Center in Virginia, USA. Validating MPACT for small critical reactors is an important step in the CASL Verification & Validation (V&V) program [3].

2. MPACT

MPACT uses the Method of Characteristics (MoC) to calculate the neutron transport in two-dimensional planes. Multigroup cross sections are generated using a subgroup method for resonance self-shielding and a 51-group ENDF/B-VII.1 cross section library.

2.1. Mesh Settings

A mesh convergence study was performed for the IPEN/MB-01 reactor to determine the appropriate mesh settings for small critical reactors. In VERA, the default mesh settings for the MoC solver suffices for larger reactors, since the buckling of these reactors is smaller. In order to properly model a reactor with larger bucklings, a finer mesh is necessary to properly account for the larger leakage. We performed a mesh convergence study by adjusting the mesh settings one by one until the change in k_{eff} between subsequent mesh parameter values was small, about 10 pcm. We used the results from the IPEN mesh convergence study in the B&W models as well. Table 1 summarizes the results from this study, with a description of each mesh parameter given in the VERA Input Manual [2].

Table 1: MPACT mesh convergence results

Parameter	Default Value	Small Critical
Ray spacing	0.05	0.01
Polars angles per octant	2	3
Azimuthal angles per octant	16	32
Number of FSR per radial rings in the fuel[a]	3	10
Number of FSR azimuthal sections in the fuel[a]	8	32

[a] Flat source region (FSR)

2.2. Axial Buckling

To model the three-dimensional core configurations in two-dimensions, an axial buckling is used to calculate the axial leakage from the core. The axial buckling used for the IPEN/MB-01 reactor model was calculated by Hykes using results from 3D MCNP models [4]. We followed the same methodology used by Hykes to interpolate the axial buckling value at different temperatures for the IPEN cases. The axial buckling used in the B&W reactor configurations was the same as used in previous investigations [5,6] and is based on a historical value adjusted for the critical water height.

3. Experimental Results

3.1. IPEN/MB-01

The IPEN/MB-01 facility is a zero power test reactor used for a litany of reactor physics experiments. The experiments modeled in MPACT were critical loading of the IPEN reactor with: gadolinia fuel, steel and copper rods, and varying moderator temperature. The facility consists of a reactor with 30×30 fuel rod locations with fuel loaded on a 28×26 sub-grid. The fuel was UO_2 enriched to 4.3486 weight% and had stainless steel (SS-304) clad. The core is submerged in a tank of light water held at a constant temperature. The number densities for each of the experimental data sets are given in §3 *Benchmark Specifications* of [7–9], along with the benchmark k_{eff}'s and temperatures.

LEU-COMP-THERM-044 (LCT-044) explored critical loadings of the IPEN/MB-01 reactor with steel and copper rods. The experimental set contained 10 critical loading configurations of the reactor conducted at room temperature, 21°C. Each configuration contained 12 copper rods placed into the reactor core at different positions, and the number of steel rods and locations varied for each case. The results for LCT-044 show an average bias of +221 $\pm$ 34 pcm from the experimental k_{eff}. The MPACT model used the benchmark corrected moderator temperature of 20.5°C and the number densities provided in the benchmark for the fuel and reactor materials [7]. The moderator density was calculated using the IAPWS-IF97 water property tables. Table 2 shows the results for each of the cases. Figure 1 shows a typical core loading patterns for this experimental data set.

Table 2: LEU-COMP-THERM-044 Results

Case #	Calculated k_{eff}	Benchmark k_{eff}	Difference (pcm)
1	1.00255	1.00050	205
2	1.00269	1.00030	239
3	1.00241	1.00050	191
4	1.00242	1.00030	212
5	1.00250	1.00030	220
6	1.00240	1.00040	200
7	1.00258	1.00040	218
8	1.00232	1.00030	202
9	1.00251	1.00040	211
10	1.00351	1.00040	311

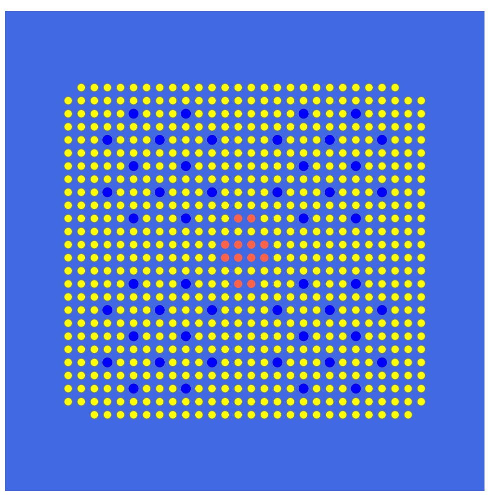

Figure 1: LEU-COMP-THERM-044 Case 01

LEU-COMP-THERM-046 (LCT-046) explored critical loadings of the IPEN/MB-01 reactor with a variation in moderator temperature, from 14°C to 85°C. The experiments also explored loadings with steel and copper rods, along with gadolinia fuel rods. There were 22 critical loading patterns presented in this data set, with cases 1-11 only having UO_2 fuel rods loaded in the core, cases 12-17 having gadolinia rods, and cases 17-22 having steel and copper rods. The results for LCT-046 shows an average bias of +216 ± 19 pcm from the experimental k_{eff} provided in the experimental benchmark [8]. Table 3 shows the results for each of the cases. Figure 2 shows a typical core loading pattern for this experiment set.

Table 3: LCT-046 Results

Case #	Calculated k_{eff}	Benchmark k_{eff}	Difference (pcm)	Moderator Temperature (°C)
1	1.00291	1.00039	252	24.05
2	1.00283	1.00040	243	37.41
3	1.00277	1.00039	238	42.43
4	1.00273	1.00039	234	46.96
5	1.00271	1.00038	233	51.93
6	1.00274	1.00039	235	59.53
7	1.00276	1.00039	237	14.22
8	1.00266	1.00039	227	42.91
9	1.00255	1.00039	216	57.27
10	1.00255	1.00036	219	64.44
11	1.00255	1.00037	218	78.99
12	1.00241	1.00038	203	30.60
13	1.00237	1.00039	198	39.95
14	1.00228	1.00038	190	48.95
15	1.00236	1.00045	191	55.24
16	1.00226	1.00037	189	65.26
17	1.00240	1.00039	201	85.31
18	1.00245	1.00043	202	14.29
19	1.00252	1.00040	212	31.27
20	1.00251	1.00039	212	39.27
21	1.00254	1.00039	215	49.97
22	1.00229	1.00039	190	64.07

LEU-COMP-THERM-054 (LCT-054) explored critical loadings of the IPEN/MB-01 reactor with gadolinia fuel rods. The experiments were conducted at room temperature, 21°C. The reactor was loaded with 4 or 6 gadolinia pins in addition to the UO_2 fuel rods. There are 8 critical configurations in this experiment set. The results from LCT-054 shows an average bias of +224 ± 18 pcm from the experimental results. The simulations were conducted at the benchmark corrected moderator temperature of 20.5°C using the number densities provided in the experiment report [9]. Table 4 shows the results for each of the cases. Figure 3 shows a typical loading pattern for this experiment set.

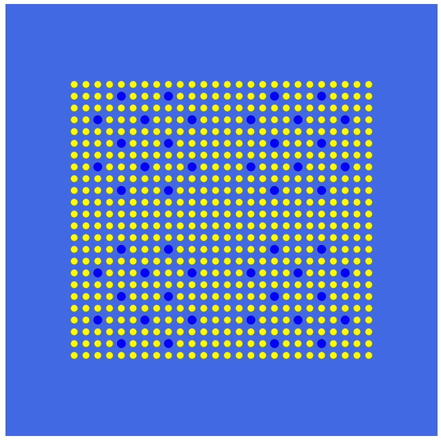

Figure 2: LEU-COMP-THERM-046 Case 01

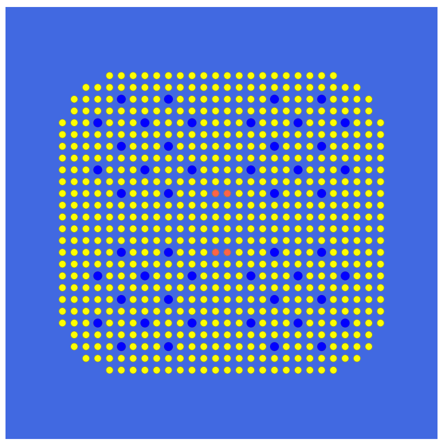

Figure 3: LEU-COMP-THERM-054 Case 01

Table 4: LCT-054 Results

Case #	Calculated k_{eff}	Benchmark k_{eff}	Difference (pcm)
1	1.00315	1.00070	245
2	1.00298	1.00050	248
3	1.00283	1.00040	243
4	1.00271	1.00060	211
5	1.00253	1.00050	203
6	1.00240	1.00020	220
7	1.00261	1.00050	211
8	1.00231	1.00020	211

3.2. Babcock & Wilcox Facility

The B&W facility is a core tank facility in which critical experiments were conducted. The core is set in a tank of light water. There are numerous experimental sets conducted at the facility, however some of the experiment sets included an asymmetric reflector blanket which cannot be modeled in MPACT. We modeled core patterns from the 1484 and 1810 experimental data sets [10,11]. For the experiments modeled, there were two different fuel types utilized, a low and high enriched fuel, with aluminum (Al-6061) and steel (SS-304) clad respectively. The different fuel types also had slightly different dimensions. In addition, boron was added to the moderator in the B&W cases, unlike the IPEN/MB-01 cases.

For the 1484 experiment, fuel with Al-6061 was used to investigate critical configurations for loading of LWR fuel in storage. While there exist 21 different experiments in the 1484 report, only Cores I-III are modeled due to a limitation in modeling asymmetric cores. Core III had configurations a-g differing in moderator temperature and boron concentration for each case. The results from the 1484 experimental set show an average bias -70 ± 65 pcm from critical. Table 5 shows the results for each of the cases. Figure 4 shows a typical core loading pattern.

Table 5: 1484 Results

Case ID	Calculated k_{eff}	Δk (pcm)	Moderator Temperature (°C)	Boron Concentration (ppm)
Core 1	0.99809	-190.8	21.0	0
Core 2	0.99867	-133.3	18.5	1037
Core 3a	0.99987	-13.3	18.0	769
Core 3b	1.00019	18.8	18.0	764
Core 3c	0.99970	-30.1	18.0	762
Core 3d	0.99953	-47.1	18.5	753
Core 3e	0.99941	-59.2	18.0	739
Core 3f	0.99935	-65.2	18.0	721
Core 3g	0.99886	-114	18.5	702

The B&W 1810 experiment set explored critical loading configurations using fuel with different enrichments and gadolinia burnable absorber rods at a constant temperature of 25°C. We modeled 23 critical configurations labeled 1-20 with configurations 5A, 5B, and 6A. The results from the 1810 experimental set show an average bias of -51 ± 58 pcm from critical. Cases 1-10 contain only low enriched fuel pins and cases 11-20 contain a mixture of both low and high enriched fuel pins. Table 6 shows the results for each of the cases with Figure 5 showing a typical core loading pattern [6].

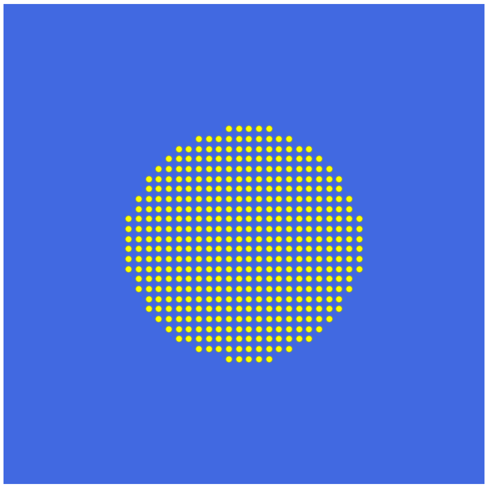

Figure 4: 1484 Core 1

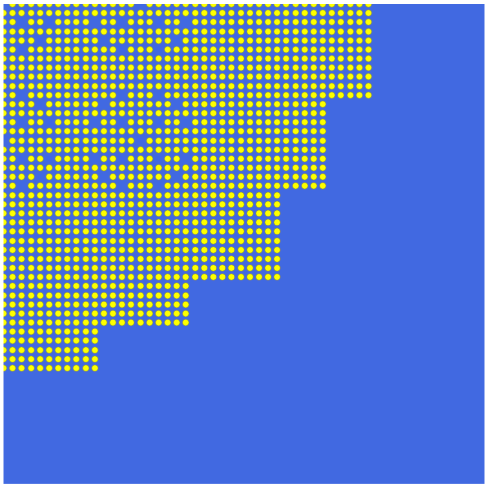

Figure 5: 1810 Core 1

Table 6: 1810 Results

Case ID	Calculated k_{eff}	Δk (pcm)	Boron (ppm)
Core 01	0.99962	-37.8	1337.9
Core 02	0.99963	-36.8	1250.0
Core 03	0.99918	-82	1239.3
Core 04	1.00020	20.1	1171.7
Core 05	0.99884	-115.8	1208.0
Core 05A	0.99873	-127.2	1191.3
Core 05B	0.99890	-110.1	1207.1
Core 06	0.99940	-60.3	1155.8
Core 06A	0.99934	-66	1135.6
Core 07	0.99901	-98.9	1208.8
Core 08	0.99894	-105.9	1170.7
Core 09	0.99911	-88.6	1130.5
Core 10	0.99875	-125.2	1177.1
Core 12	0.99965	-35.1	1899.3
Core 13	0.99999	-0.6	1635.4
Core 14	0.99935	-65.5	1653.8
Core 15	0.99984	-15.9	1479.7
Core 16	0.99934	-66.5	1579.4
Core 17	0.99946	-54.4	1432.1
Core 18	1.00077	76.5	1776.8
Core 19	1.00048	48.4	1628.3
Core 20	1.00031	31.1	1499.8

4. CONCLUSIONS

This paper shows the validation results of using the CASL core simulator MPACT to calculate the k_{eff} of several small critical configurations. The results over all 40 IPEN cases show a bias of +219 pcm and a small standard deviation of 23 pcm. The results over all 31 of the B&W cases show a bias of -57 pcm and a small standard deviation of 60 pcm. Overall, the results are very good and show that MPACT can accurately model a wide range of small experimental configurations.

These results are used in the CASL Verification & Validation (V&V) program [3] and provide support that the neutron transport and multigroup library processing in MPACT can accurately model a wide variety of critical experimental configurations. Additional validation work includes modeling full-core PWR reactor problems (including critical boron, reaction rates, and startup physics testing) and comparison of depletion isotopics to post irradiation experiments (PIEs). Verification activities include a wide variety of code verification and solution verification.

Future work in this area includes modeling additional critical reactors and expanding the comparisons to include other measured data such as fission rates and kinetic parameters.

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