Validation and Application of the 3D Neutron Transport Code MPACT within CASL VERA-CS

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ABSTRACT

MPACT is a new high-fidelity neutron transport code designed to provide an advanced pin-resolved transport capability for the VERA (Virtual Environment for Reactor Applications) which is the end-user reactor simulation tool being developed for the Consortium for the Advanced Simulation of Light Water Reactors (CASL). The transport methods currently implemented in MPACT are based on the 2-D/1-D method of characteristics (MOC) capabilities to provide a pin-resolved solution of the neutron flux and power throughout the reactor. The cross section resonance treatment utilizes the subgroup method and thermal-hydraulic feedback capability within MPACT includes a simplified T/H feedback model, as well as coupling to the subchannel code CTF and ongoing work to couple MPACT to Computational Fluid Dynamics (CFD). Other features include full core depletion and parallel execution enabling efficient scaling up to $O(10^5)$ processors. This capability has been validated for several benchmark problems and several cycles of operating Pressurized Water Reactors (PWR).

KEYWORDS
CASL, Neutronics, MOC, VERA-CS

1. INTRODUCTION

The 3-D pin resolved reactor analysis transport code MPACT has been under development at the University of Michigan (UM) since the Fall of 2011 [1]. In 2013, Oak Ridge National Laboratory (ORNL) began to collaborate in the development of MPACT to support the needs of the Consortium for the Advanced Simulation of Light Water Reactors (CASL), the U.S. Department of Energy's Nuclear Modeling and Simulation Energy Innovation Hub. MPACT is now co-owned by UM and ORNL and is leveraged by other research projects outside of CASL at both institutions. Within CASL, MPACT provides capabilities for neutron transport using the 2D/1D technique based on the method of

characteristics (MOC), time-dependent transport, and a traditional lattice physics capability as part of the VERA-CS (Virtual Environment for Reactor Applications) [2].

As a reactor analysis tool MPACT is capable of performing high fidelity whole core simulations providing pin-resolved neutron fluxes. Whole core simulations are achieved by solving the 2-D/1-D form of the Boltzmann Transport equation in which the radial plane is solved using the well-established Method of Characteristics (MOC) and the less heterogeneous axial solution is obtained using a lower order transport method. For the cross section temperature feedback, currently the subgroup method is used to provide resonance self-shielding in MPACT. Temperature feedback is provided in MPACT using either a simplified internal mode or linkage to external T/H codes such as the subchannel code CTF for whole core calculations or CFD codes for special applications. Other features include nuclide depletion and parallel execution allowing for scaling up to $O(10^5)$ processors.

These capabilities were demonstrated for several benchmark problems to include a realistic full core PWR. The primary purpose of this paper is to provide an overview of the methodology of the MPACT code and describe some of the code validation. A secondary purpose is to describe the role of MPACT within CASL as a component within the VERA (Virtual Environment for Reactor Applications). The following section will first provide an overview of the VERA-CS and the role of MPACT as the core neutronics simulator. The subsequent sections will then review the methods in MPACT and summarize some of the code verification and validation.

2. CASL CORE SIMULATOR VERA-CS

VERA-CS is a suite of simulation capabilities as shown in Figure 1 which are being developed in order to address the CASL Challenge Problems in PWRs [2]. VERA-CS has been designed to be broadly applicable for other PWR analyses problems and to be extensible to allow future application to other reactor types and to challenges beyond normal operation. The Physics Integration (PHI) Focus Area (FA) is responsible for overall VERA development, and for the development of the subchannel thermal-hydraulics code, COBRA-TF. The Radiation Transport Methods (RTM) FA is responsible for the development of radiation transport codes in VERA, which includes the pin-resolved deterministic transport code MPACT. COBRA-TF and MPACT were the principal codes used in the work reported here to demonstrate the operational reactor depletion capability in VERA-CS.

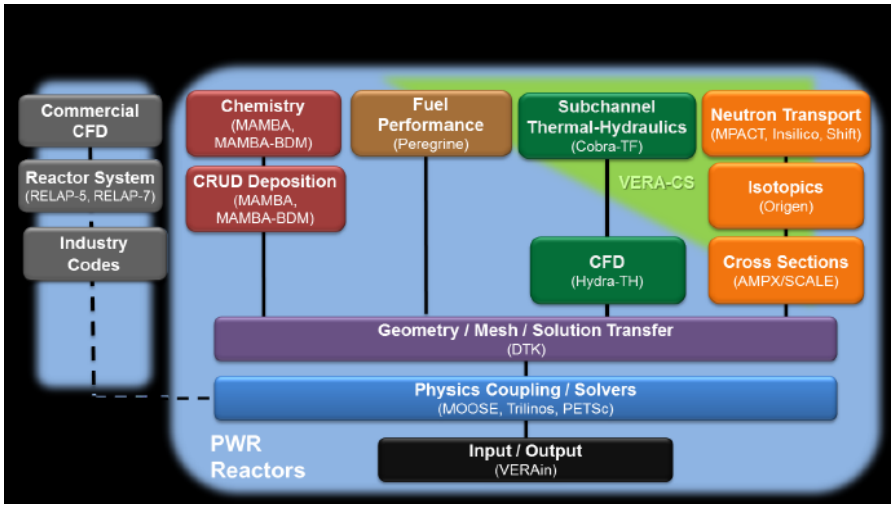

Figure 1 VERA-CS Code Suite

CTF is a thermal-hydraulic simulation code designed for Light Water Reactor (LWR) analysis) [3]. CTF has a long lineage that goes back to the original COBRA program developed in 1980 by Pacific Northwest Laboratory under sponsorship of the Nuclear Regulatory Commission (NRC). The original COBRA began as a thermal-hydraulic rod-bundle analysis code, but versions of the code have been continually updated and expanded over the past several decades to cover almost all of the steady-state and transient analysis of both PWR's and BWR's. CTF is currently being developed and maintained by the Reactor Dynamics and Fuel Management Group (RDFMG) at the Pennsylvania State University. CTF includes a wide range of thermal-hydraulic models important to LWR safety analysis including flow regime dependent two-phase wall heat transfer, inter-phase heat transfer and drag, droplet breakup, and quench-front tracking. CTF also includes several internal models to help facilitate the simulation of actual fuel assemblies. These models include spacer grid models, a fuel rod conduction model, and built-in material properties for both the structural materials and the coolant (i.e. steam tables). CTF uses a two-fluid, three-field representation of the two-phase flow. Also more recently, several improvements were made to CTF to improve performance and parallelism [4]. Some of the reasons for selecting CTF as the primary T/H solver in the VERA core simulator include the reasonable run-times compared to CFD (although CFD will be available as an option) and the ability to support future applications of VERA such as transient safety analysis, BWR, and SMR applications.

3. MPACT METHODS

This section will provide a brief overview of the essential methods in MPACT since a detailed description is provided in [5]. MPACT is based on the method of characteristics (MOC) which has received widespread acceptance over the last several years and has been implemented in most of the popular 2-D lattice physics codes. It is generally accepted that MOC has become the most popular transport method within the industry for routine 2-D assembly level analysis to generate cross sections for full core LWR simulations using the current generation of lattice codes and nodal core simulators such as CASMO-5/SIMULATE-5 [22]. It is worthwhile to briefly review the innovation in methods that have made it computationally possible to extend MOC methods to full core analysis.

The governing equation for neutron transport within a nuclear reactor, and the starting point for the derivation is the 3-D steady-state Boltzmann neutron transport equation. The common multi-group form of the equation is given as:

$$\vec{\Omega}\cdot\nabla\varphi_g\left(\vec{r},\vec{\Omega}\right)+\Sigma_{t,g}\left(\vec{r}\right)\varphi_g\left(\vec{r},\vec{\Omega}\right)= \frac{\chi_g}{4\pi k_{eff}}\sum_{g'=1}^{G}\nu\Sigma_{f,g'}\left(\vec{r}\right)\int_0^{4\pi}\varphi_{g'}\left(\vec{r},\vec{\Omega}'\right)d\Omega'+\sum_{g'=1}^{G}\int_0^{4\pi}\Sigma_{s,g'\to g}\left(\vec{r},\vec{\Omega}'\cdot\vec{\Omega}\right)\varphi_{g'}\left(\vec{r},\vec{\Omega}'\right)d\Omega', \quad (1)$$

where the typical notation, $\vec{r}$ and $\vec{\Omega}$ is used for space and angle, respectively. The subscript g denotes the neutron energy group index. φ, Σ represent the angular neutron flux, and cross sections, respectively. For the cross sections the subscript indicates the reaction type where t is for total, s is for scattering, and f is for fission. χ is the normalized fission spectrum and k_{eff} is the effective neutron multiplication factor, or eigenvalue, of the system. The MOC solution is a general mathematical technique for solving partial differential equations. For the transport equation a coordinate transformation is applied to Eq. (1) to yield a first order ordinary differential equation for the solution along the characteristic direction. However, the solution of the 3D MOC equations for a full core problem is not computationally possible, even on computers with $O(10^5)$ processors. The following section will describe an innovative 2-D/1-D approximation which has reduced the computational burden and made it possible to apply MOC to full core problems.

3.1 2-D/1-D Approximation of the 3-D Transport Equation

Since most 3-D transport methods are computationally too intensive to be practical, an alternative numerical method colloquially referred to as "2-D/1-D" was implemented in MPACT. The 2-D/1-D approach is a numerical method for whole-core transport solutions pioneered largely by the DeCART code [6]. The original implementation in DeCART was motivated by the observation that most reactor designs, and in particular light water reactors have only mild variation in their design in the axial direction. Therefore, it is reasonable to assume that the solution of the transport equation should also have mild variation in the axial direction. However, in DeCART the method did not have a rigorous mathematical foundation and was known to have stability issues for certain kinds of meshing, in particular for thin axially discretized regions. Recent work [7] was able to provide a systematic derivation of the discretized 2-D/1-D equations and an accompanying stable iteration scheme. This section summarizes the essential elements of their method.

The 2-D/1-D equation follows from the 3-D transport equation by modifying the streaming operator with the following approximation:

$$\vec{\Omega}\cdot\nabla\varphi_g\left(\vec{r},\vec{\Omega}\right)\approx\left(\Omega_x\frac{\partial\varphi_g\left(\vec{r},\vec{\Omega}\right)}{\partial x}+\Omega_y\frac{\partial\varphi_g\left(\vec{r},\vec{\Omega}\right)}{\partial y}\right)-\frac{\partial}{\partial z}\frac{D}{4\pi}\frac{\partial\phi_g\left(\vec{r}\right)}{\partial z}. \tag{2}$$

Thus, Eq. (1) becomes:

$$\left(\Omega_x\frac{\partial\varphi_g\left(\vec{r},\vec{\Omega}\right)}{\partial x}+\Omega_y\frac{\partial\varphi_g\left(\vec{r},\vec{\Omega}\right)}{\partial y}\right)-\frac{\partial}{\partial z}\frac{D_g\left(\vec{r}\right)}{4\pi}\frac{\partial\phi_g\left(\vec{r}\right)}{\partial z}+\Sigma_{t,g}\left(\vec{r}\right)\varphi_g\left(\vec{r},\vec{\Omega}\right)=q_g\left(\vec{r},\vec{\Omega}\right). \tag{3}$$

To obtain the respective 2-D and 1-D discretized equations, Eq. (3), is integrated in the transverse directions. The 2-D equation is obtained by integration of Eq. (3) over a finite interval in z and the 1-D axial equation is obtained by integrating Eq. (3) over a finite interval in the x-y plane. The 2-D equation is given as:

$$\left(\Omega_x\frac{\partial\varphi_g\left(x,y,\vec{\Omega}\right)}{\partial x}+\Omega_y\frac{\partial\varphi_g\left(x,y,\vec{\Omega}\right)}{\partial y}\right)+\Sigma_{t,g,z}\left(x,y\right)\varphi_g\left(x,y,\vec{\Omega}\right)=q_{g,z}\left(x,y,\vec{\Omega}\right)-\frac{L_{z,g}\left(x,y\right)}{\Delta_z}, \tag{4}$$

where L_z is the *axial transverse leakage* given by the finite difference approximation of the axial derivative integrated over the interval [z-1/2,z+1/2].

$$\frac{L_{g,z}\left(x,y\right)}{\Delta_z}\approx\left(\frac{D_{g,z+1/2}\left(x,y\right)}{\Delta_{z+1/2}}\left(\phi_{g,z+1}\left(x,y\right)-\phi_{g,z}\left(x,y\right)\right)-\frac{D_{g,z-1/2}\left(x,y\right)}{\Delta_{z-1/2}}\left(\phi_{g,z}\left(x,y\right)-\phi_{g,z-1}\left(x,y\right)\right)\right) \tag{5}$$

Here the subscript z is to denote the quantity has been averaged over the interval [z-1/2,z+1/2]. Similarly, the 1-D equation is:

$$-\frac{\partial}{\partial z}D_{g,x,y}\left(z\right)\frac{\partial\phi_{g,x,y}\left(z\right)}{\partial z}+\Sigma_{t,g,x,y}\left(z\right)\phi_{g,x,y}\left(z\right)=q_{g,x,y,0}\left(z\right)+L_{g,x,y}\left(z\right) \tag{6}$$

where $L_{x,y}$ is the *radial transverse leakage* given by:

$$L_{g,x,y}\left(z\right)=\bar{J}^{net}_{g,x-1/2}\left(z\right)+\bar{J}^{net}_{g,x+1/2}\left(z\right)+\bar{J}^{net}_{g,y-1/2}\left(z\right)+\bar{J}^{net}_{g,y+1/2}\left(z\right) \tag{7}$$

The iteration scheme of 2-D/1-D involves alternating between the solution of Eq. (4) and Eq. (6) with the solutions of each equation coupled through the transverse leakages of Eq. (5) and Eq. (7). In previous implementations the iteration scheme was observed to become unstable for some applications. The primary reason for this was that the source term on the right hand side of Eq. (4) could become negative because the transverse leakage part of the source is proportional to the inverse of Δ_z, and thus refining the mesh in the *z*-direction would cause the iteration scheme to become unstable. A detailed assessment of the performance of the 2D-1D methods in MPACT is provided in [8].

3.2 CMFD Acceleration

A key aspect of solving the transport equation efficiently is the use of an effective acceleration technique. The primary objective is to minimize the amount work needed to reach convergence which usually is achieved by minimizing the number of iterations or transport sweeps required to converge. One such acceleration method for *k*-eigenvalue problems that exhibits these qualities is the *coarse mesh finite difference* method (CMFD) that was originally developed as a technique for the nodal diffusion based methods used in reactor analysis [23]. However, its fundamental concept applies equally as well to transport methods, and has been shown to be very effective at accelerating 2-D MOC transport methods where it has been used extensively. In general CMFD can be thought of as a non-linear coarse mesh diffusion synthetic acceleration scheme, in which a correction factor, $\hat{D}_{j,g,s}$, is then computed as shown by Eq. (8), where the fine mesh transport method computes the neutron net current along the surfaces of the coarse mesh. Consequently this correction factor along with cross section homogenization creates equivalence between the solution of the fine mesh MOC equations and the coarse mesh diffusion equations at convergence. The homogenization process in general preserves all the node volume integrated quantities based on the fine mesh solution, specifically the node average reaction rates. The correction factor of Eq. Eq. (8) allows the low order system to also preserve the node surface integrated quantities of the fine mesh solution, and specifically the average leakage. Because of this equivalence, the multiplication factor, k_{eff}, of the CMFD linear system is the same as that of the fine mesh transport method computed from source iteration when $\hat{D}_{j,g,s}$ is converged.

$$\hat{D}_{j,g,s} = \frac{\bar{J}^{net}_{j,g,s} + \tilde{D}_{j,g,s}\left(\bar{\phi}_{j,g} - \bar{\phi}_{j,g,s}\right)}{\left(\bar{\phi}_{j,g} + \bar{\phi}_{j,g,s}\right)}. \tag{8}$$

3.3 Cross Section Resonance Self-Shielding

The MPACT code utilizes the subgroup method for resonance self-shielding calculations based on multi-group libraries with subgroup parameters generated by ORNL [9]. As the code has been designed for easy extension of new libraries by implementing a small set of interfacing functions associated with the library, it is very straightforward to add new multi-group libraries to perform the resonance and transport calculations, such as when a code user may want to use a library containing proprietary data. Presently, the microscopic cross section library used by MPACT is processed from ENDF/B-VII.0 data using sequences of SCALE-6.2 code system [9]. Research is currently underway to add a new self-shielding method to MPACT based on the Embedded Self-Shielding Method (ESSM) [26].

3.4 Fuel Depletion

The depletion capability in MPACT is largely based on previous work reported in Refs [10] [11] which included the coupling of MPACT to the ORIGEN-API [17]. The depletion chain includes 510 nuclides and the point depletion calculation solves the matrix exponential for long-lived nuclides using a Taylor series expansion method and Gaussian elimination for short-lived nuclides that are assumed to be in secular equilibrium. The reactor depletion problem is governed by a system of first order ordinary differential equations with constant coefficients for nuclide transmutation and decay. In principle, each azimuthal and radial flat flux region within the fuel pin can be depleted but because of current computation time and memory limitations depletion within MPACT is currently only performed for each radial ring within the pin.

3.5 Thermal-Hydraulic Feedback

For the T/H feedback in MPACT, the T/H solution is currently applied at the pin-cell level as illustrated in Figure 2. Additionally, there is the consideration for the other various regions within the core such as the fuel assembly. In the region below the active fuel, the inlet T/H conditions are user specified, as are regions that are radially outside of the active fuel, such as the baffle and reflector. The T/H and neutronics coupling is illustrated in Figure 3.

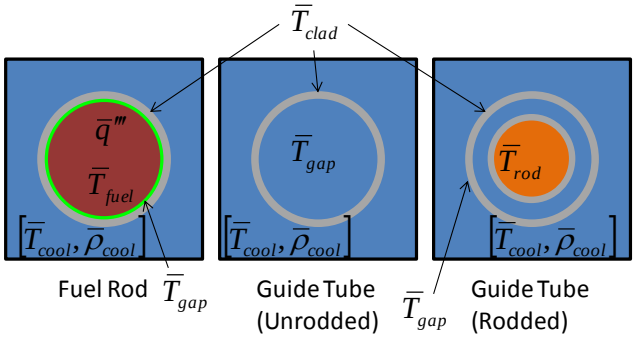

Figure 2 Pin Cell Coupling Models used in MPACT [11]

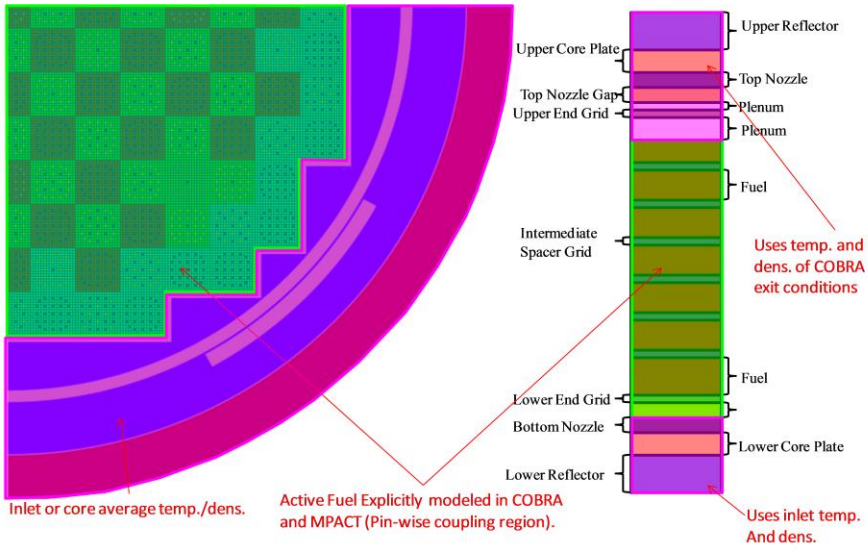

Figure 3 T/H Coupling of Core Regions in MPACT [11]

The local temperature and density conditions are used to update the macroscopic cross sections. The resonance self-shielding methodology in MPACT treats the explicit geometry and local T/H conditions directly. When the T/H solution is updated, a resonance calculation is performed to update the equivalence cross section and the macroscopic cross sections are then recomputed from the microscopic cross sections based on the local T/H conditions and updated resonance self-shielding factors. MPACT currently obtains feedback using either an internal simplified TH model or coupling to the subchannel code CTF. The coupling to CTF employs the iteration scheme depicted in Figure 4.

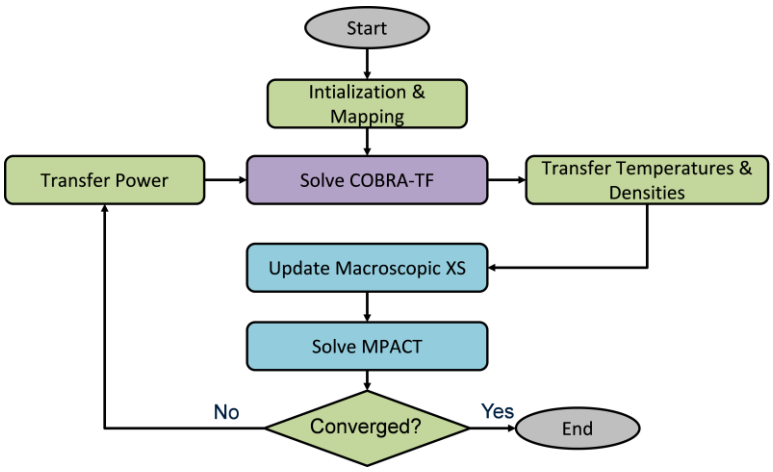

Figure 4 Flow chart illustrating the fixed point iteration between MPACT and COBRA [11]

An internal simplified TH model in MPACT is used primarily for QA purposes and model development, and employs a 1-D convection solution for each assembly and assumes every assembly in the core has an equal mass flow rate. The effects of pressure change are ignored and the equations are reduced to a simple energy balance for each axial node. Given as:

$$h_{out,z} = h_{in,z} + \frac{\bar{q}_{asy,z}}{\dot{m}} \tag{9}$$

The enthalpy is linearly interpolated to obtain the node average coolant enthalpy which is related to coolant temperature through a closure model. Once the coolant temperature for each axial location is obtained, a 1-D radial conduction model is used for every fuel pin with its specific power calculated by MPACT. The boundary condition for the fuel pin is set using a heat transfer coefficient obtained from the modified Dittus-Boelter correlation.

$$q''_{pin} = \bar{h}_{DB}\left(T_{s,clad} - T_{fluid}\right) \tag{10}$$

$$\bar{h}_{DB} = \frac{k_{fluid}}{D_h}\left(0.042\frac{P_{pin}}{D_{clad}} - 0.024\right)\mathrm{Re}^{0.8}\,\mathrm{Pr}^{0.4} \tag{11}$$

The correlation used to compute the thermal conductivity of the cladding is the same as the one specified in the MATPRO materials properties library [24]. The heat transfer through the fuel-clad gap is also treated with the 1-D conduction solver using a lumped conduction model where the heat transfer coefficient for the gap is a user specified input. The 1-D conduction problem is solved using a finite difference formulation and the Thomas' algorithm for Gaussian elimination of tri-diagonal linear systems.

All of the thermo-physical properties for the fluid were functionalized for only temperature (or enthalpy) using a quadratic fit for normal PWR operating conditions. Therefore the effects of pressure are neglected and all values are generated assuming a system pressure of 2250 psi. This is generally sufficient for normal steady-state PWR operating conditions.

4. MPACT VERIFICATION

Over the past few years, the MPACT code has matured and the confidence level has increased in the ability of MPACT to model an operational Pressurized Water Reactor to calculate the steady-state reactor core neutron distribution for operating nuclear power plant (PWR) conditions across multiple fuel cycles [21]. A key part of the confidence in MPACT as a core simulator has been the development and implementation of Code V&V plan [18] which is one part of the overall software life cycle for all CASL codes. The overarching objective of code verification in MPACT has been to establish that a model implemented in the code accurately represents the developer's conceptual description and the solution of the model. The verification activities in MPACT have been designed to address this general objective and have encompassed both the verification of the source code itself as well as the verification of the solution. Source code verification activities in MPACT have been achieved by establishing a comprehensive software testing practices, to include code unit testing and code regression analysis. The current MPACT Code Testing Statistics are shown in Table 1.

Table 1 MPACT Code Testing Statistics

Metric	M_libs[1]	M_Drivers[2]	Total
Unit Tests	123	4	127
Regression Tests	0	159	159
Coverage	80.17%	67.24%	79.69%
Lines of Soure Code	91,006	3,446	94,452

[1]M_libs are the functional subpackages comprising the majority of the code base
[2]M_Drivers are the top level main programs which provide the code execution flow

The principal focus of solution verification activities within MPACT has been to evaluate the numerical error in the solution. The initial focus of solution verification was to perform mesh convergence analysis in support of the initial code validation activities in MPACT, however, a more comprehensive and thorough solution verification has been initiated based on the Method of Manufactured Solutions (MMS).

MPACT validation work has been ongoing in both the areas of measured data from critical experiments as well as measured data from operating nuclear power plants. Both of these areas have been supplemented with calculated quantities on fine scales from continuous energy (CE) Monte Carlo methods. A roadmap has been established by the VERA-CS Validation Plan [20] to guide the efforts during Phase II and to further increase the validation base of the code. Increased emphasis during the next few years is being placed on the validation of MPACT depletion with measured isotopics, implementation of a formal data uncertainty quantification (UQ) protocol in MPACT, the addition of problems to validate the pin resolved capability in MPACT, and a coordinated effort within VERA-CS to ensure that the validation needs identified for the CASL challenge problems are covered sufficiently by the MPACT validation suite.

5. APPLICATION OF MPACT WITHIN VERA-CS

As noted in section 2, one of the primary drivers for MPACT development has been CASL for which MPACT is a key component in the VERA Core Simulator, which also consists of the T/H subchannel code COBRA-TF, CASL-BISON for fuel performance, and parts of SCALE for cross section generation and nuclide depletion with ORIGEN. Because MPACT provides the pin-resolved deterministic transport capability, it is important for several of the CASL challenge problems such as those modeling CRUD induced localized corrosion (CILC) and CRUD induced power shift (CIPS), as well as others which model reactivity insertion accidents and pellet clad interactions (PCI). These challenge problems drive the VERA-CS simulation capability. MPACT enables the simulation of hot full power conditions through multiple cycles which provides the starting conditions for these challenge problems. The core state for all of these applications depends on the ability to accurately deplete the reactor to the appropriate point in the cycle burnup. The following section will describe the application and validation of MPACT and VERA-CS by depleting the Watts Bar reactor.

5.1 Full Core PWR Cycle Depletion: Watts Bar

The full core PWR corresponds to the public Watts Bar Unit 1 Cycle 1 model defined in the VERA Benchmark Progression Problems published by CASL [12]. The full-core contains 3 enrichment zones (2.1, 2.6, and 3.1% U-235) and several configurations of Pyrex burnable absorber rods in the assemblies. The loading pattern is shown on the left hand side of Figure 5. The detailed descriptions for the pyrex insert configurations, control rod bank locations, and in-core detector locations can all be found in [12] and are omitted here for brevity.

In the discretization of the operational reactor, the problem was divided into 58 axial planes and included the upper and lower assembly nozzles and core plate/reflector. For the transport solution each fuel pin was divided into 3 radial rings and 8 azimuthal slices. The fuel-clad gap was explicitly modeled, however this gap thickness is assumed constant for the duration of the cycle. The fuel-clad gap model will improved in the future. The grid spacers were also modeled by preserving their mass and homogenizing the material into the coolant over the height of the actual grid spacer. In MPACT the 2-D/1-D method was used to obtain the solution to the transport equation with ray spacing of 0.05 cm and the Chebyshev Gauss-Legendre product quadrature with 16 azimuthal angles and 4 polar angles per octant of the unit sphere. The 1-D axial solver used was the nodal expansion method [NEM] using the P3 transport approximation and the T/H calculation was performed using CTF.

Because of wall time administrative constraints, the simulation was performed using both TITAN [13] and the EOS computers [25] with multiple restarts. A total of 4176 processors was used with 72 MPI processes for COBRA-TF and 522 MPI processes for MPACT, each with 8 threads. This resulted in an approximate breakdown of runtime of 25% for CTF and 75% for MPACT. A total of 19 statepoints were performed during the depletion requiring a total execution time on TITAN and EOS of about 40 hours. However, the same cycle depletion with more statepoints is currently being performed on TITAN in less than a day [21].

The measured critical boron concentration during the cycle is compared to the value predicted by the simulation in Figure 5 (RHS). The results in the Figure indicate that the predicted criticality of the is in reasonable agreement with the plant measurement since the critical boron concentrations are within 50 ppmB of the measurement. However, future results are expected to be in even better agreement since the differences are mostly due to the prediction of the fuel temperatures, and in particular the simplified gap conductance model which is currently being improved in MPACT. Slight deviations between the simulated model and the physical model of the reactor when the measurement was taken (e.g. slightly different rod position, operating power, homogenized grid spacers, etc.) can also account for some

differences in reactivity. A principal area of future work has been to improve the overall model, in particular treating the operating history more explicitly and improving the fuel temperature and gap conductance model to account for the local burnup conditions. MPACT has recently been applied to the first several cycles of Watts Bar [21] in support of the CASL Challenge Problems using these improved modeling features.

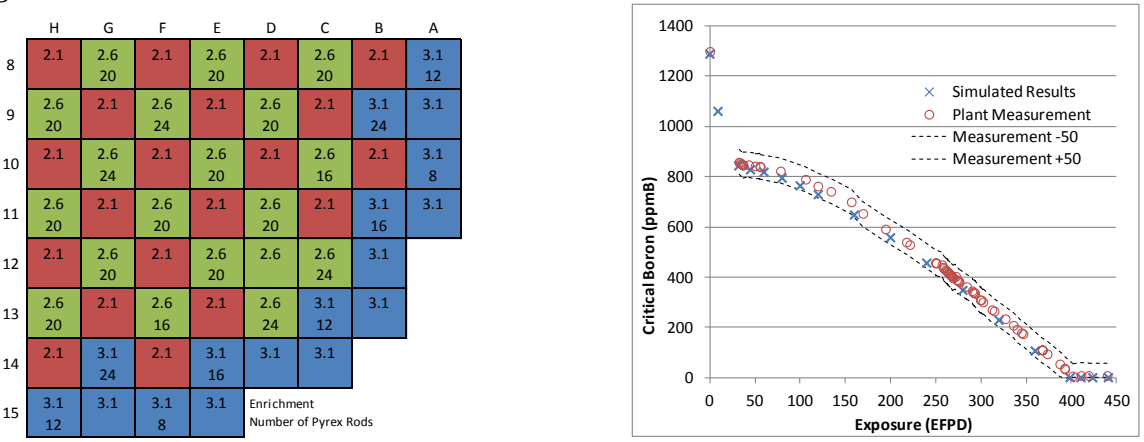

Figure 5 Core fuel and poison loading pattern for Watts Bar Unit 1 cycle 1 (LHS) MPACT Prediction of Watts Bar Critical Boron Concentration (RHS)

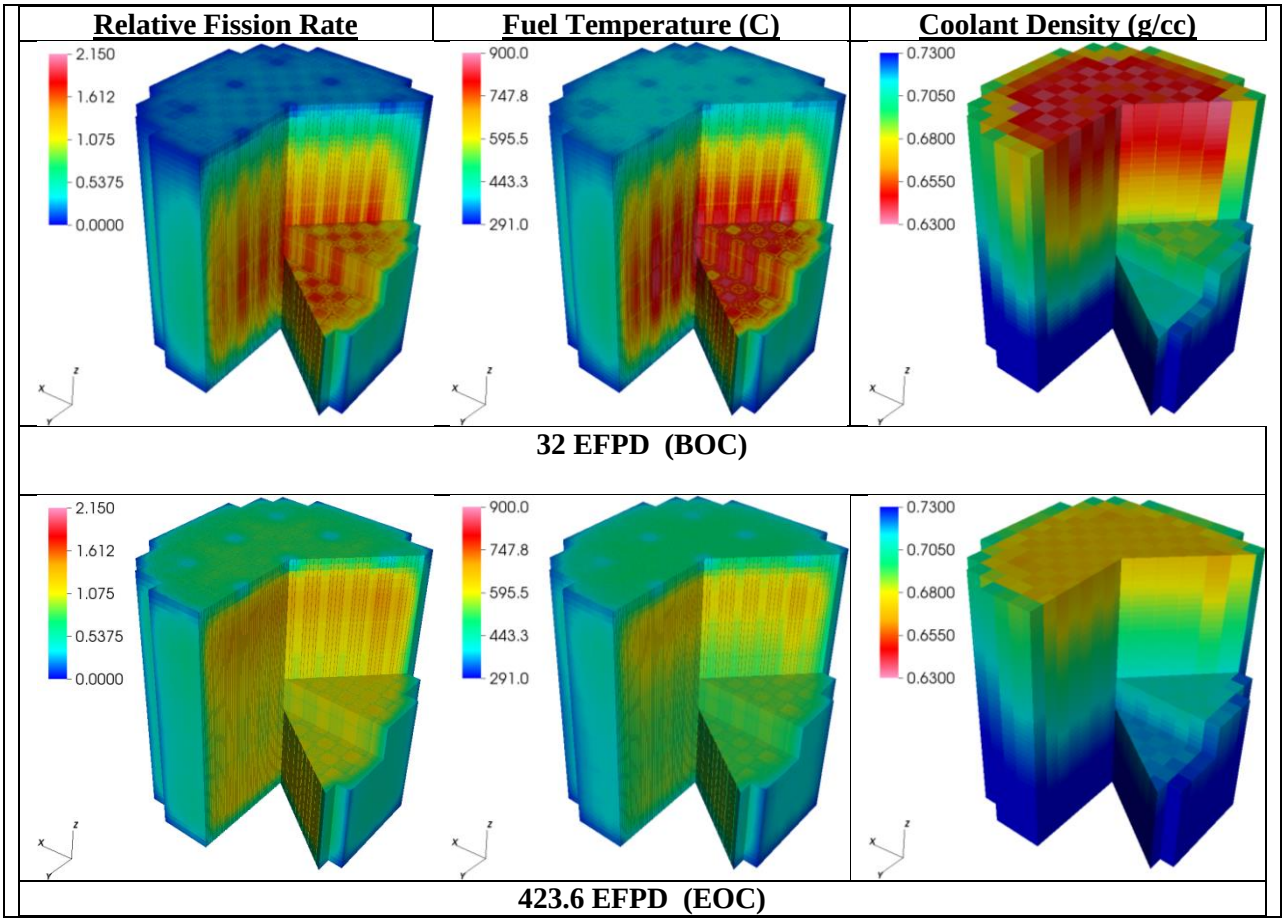

Figure 6 Core Fission Rate, Temperature, and Coolant Density Distributions for Watts Bar

5.2 Control Rod Worth Analysis: AP1000

A second recent application of VERA-CS was to the analysis of the first core of the WEC AP1000 PWR [14]. As shown in Figure 7, the first cycle of the core has a low-leakage 18-month cycle featuring five fuel regions with intra-assembly enrichment zoning and a combination of burnable absorbers including the Westinghouse Integral Fuel Burnable Absorber (IFBA) a ZrB_2 coating on the pellet surface, and Wet Annular Burnable Absorber (WABA), an insert employed at selected guide thimble locations. The core loading pattern is depicted in Figure 7 which also shows the assembly loading pattern for Region D fuel, featuring radial enrichment zoning (3 enrichments), 68 IFBA rods, 8 "long" and 4 "short" WABA inserts. The long and short WABA inserts differ in the axial length (longer or shorter) of the poisoned-bearing region, with the top plena and lower Zr spacer varying accordingly. Region E fuel assemblies are similarly characterized by enrichment zoning, IFBA rods and, when present, axially heterogeneous WABA inserts. Lower-enriched axial blankets are employed for Regions C, D and E; the blankets for the IFBA rods consist of annular fuel with a central void to accommodate He release from ^{10}B neutron absorptions in ZrB_2. The AP1000 PWR operates following the MSHIM™ core control strategy, an advanced operational strategy that entails operation with multiple control rod banks inserted in the core, including light tungsten banks and standard Ag-In-Cd banks [16]. The AP1000 PWR advanced core design and operational features make application of an advanced core simulator like VERA-CS especially relevant for the analysis.

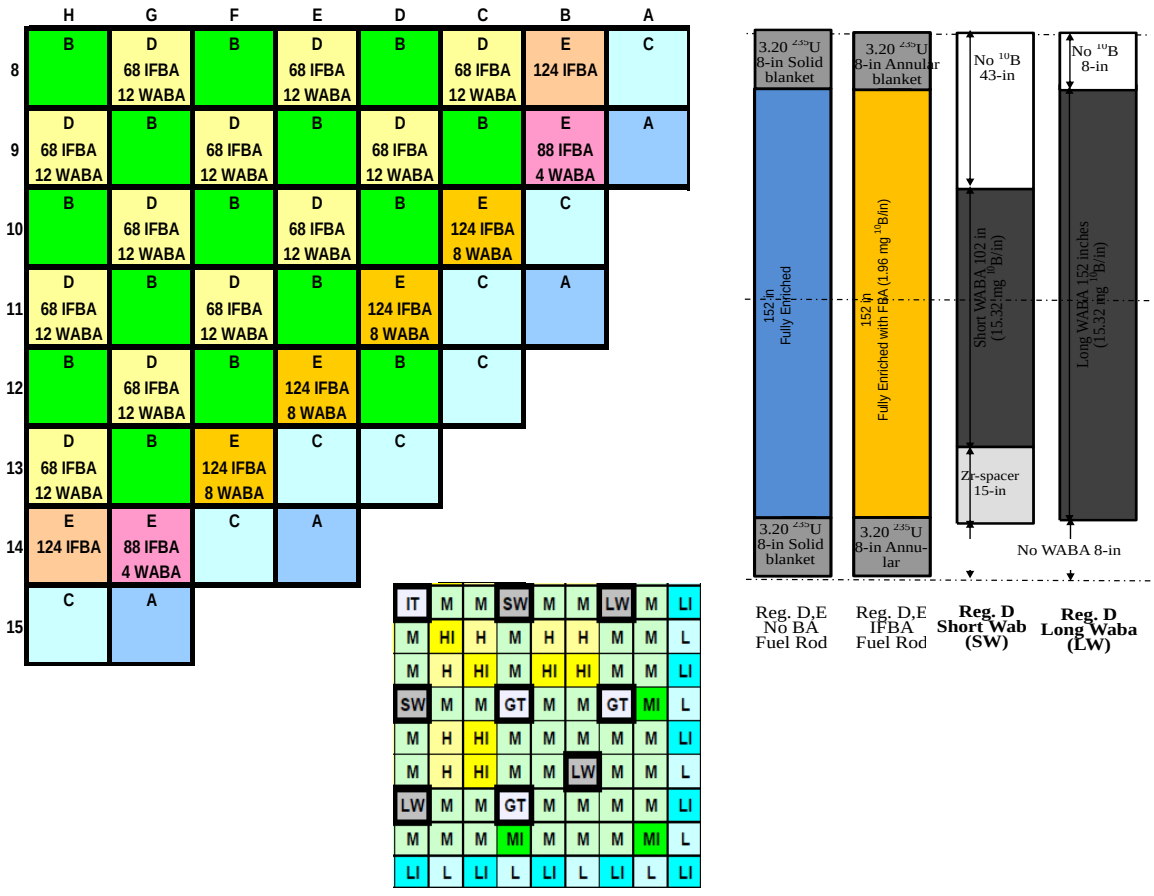

Figure 7 AP1000 PWR First Core – Fuel Loading Pattern (quarter core radial map) and Region D Fuel Assembly design (quarter assembly radial map) (note: the letters denote different fuel enrichments and IT/GT is Instrument/Guide Tube) [16]

The accuracy of the MPACT flux solution for the AP1000 was evaluated by comparison with the normalized fission rate from the Monte Carlo code KENO [15] using 25B particle histories for this calculation, with a less than 0.1% reported average power uncertainty. A 2D radial core slice with a 1.5-in stainless steel baffle radial reflector surrounded by water was simulated. The differences in the normalized fission rates are depicted on the left hand side of Figure 8. As indicated, there is very good agreement between MPACT and KENO: pin ΔP RMS of 0.7%, and max pin ΔP of 1.4 % located on the low power pins on the core periphery. The control rod worths were then evaluated for the rod positions shown on the right hand side of Figure 8. As shown in Table 1, the MPACT results are in excellent agreement with the Monte Carlo results.

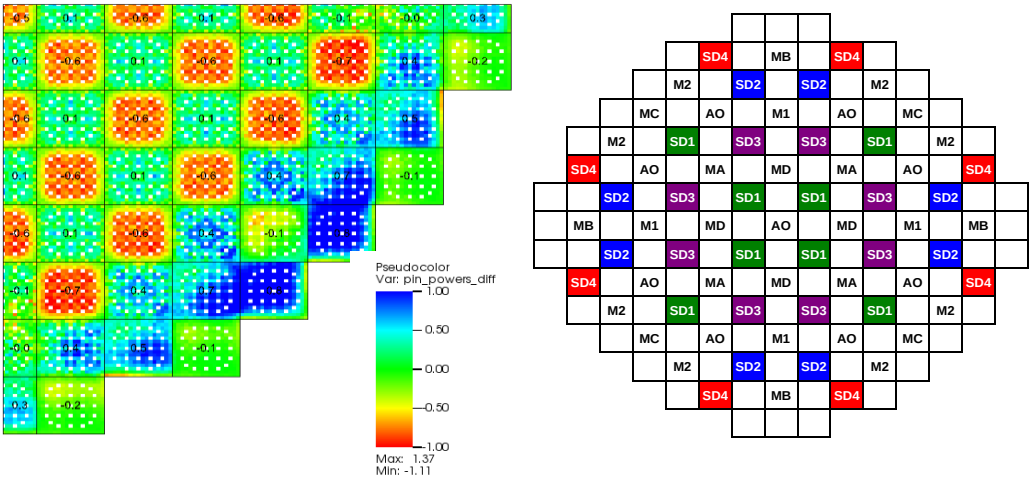

Figure 8 2D Radial Power Distribution Differences ΔP (MPACT-KENO)x100 (LHS) AP1000 PWR Control Banks Configuration (RHS)

Table 2 Control Rod Worth Results MPACT vrs. KENO for AP1000 [16]

		KENO	MPACT	
	Material	Worth (pcm)	ΔWorth (pcm)	ΔWorth (%)
MA	Tungsten	258	259	0.5%
MC	Tungsten	188	189	0.4%
MD	Tungsten	234	237	1.3%
M1	Ag-In-Cd	651	643	-1.2%
M2	Ag-In-Cd	887	893	0.7%
AO	Ag-In-Cd	1635	1624	-0.7%
S1	Ag-In-Cd	1079	1080	0.1%
S2	Ag-In-Cd	1096	1085	-1.0%
S3	Ag-In-Cd	1124	1125	0.1%
S4	Ag-In-Cd	580	578	-0.4%

6. SUMMARY AND CONCLUSIONS

An overview of the methods and capabilities of MPACT were presented along with some details about code performance for practical reactor analysis problems. The methods implemented in MPACT include 2-D/1-D and 3-D transport methods based on the characteristics form (MOC) of the neutron transport equation and detailed cross section resonance treatments. The application of MPACT to the depletion of the Watts Bar Cycle 1 depletion and to the BOC startup of the AP1000 showed good agreement of the MPACT predictions to the measured plant data and Monte Carlo code predictions, respectively. Work is ongoing to deplete the AP1000 and to perform multi-cycle analysis of Watts Bar. An important new functionality during Phase II of CASL will be the extension of MPACT to the Boiling Water Reactor.

The collaborative development of MPACT by the University of Michigan and ORNL is continuing during Phase II of the CASL project with an important focus being the application of MPACT to a series of practical "challenge problems" for which accurate pin-resolved isotopics and fission rates will be an important requirement within the CASL core simulator VERA-CS.

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