INDUSTRY USE OF CASL TOOLS

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ABSTRACT

This paper focuses on the specific industry applications of the Consortium for the Advanced Simulation of LWRs (CASL) technology. CASL is a multi-year program that is funded by the US Department of energy (DOE) with the express purpose of developing multiphysics applications for LWR simulations. As a key industry partner of CASL, Westinghouse has been deeply involved in the development of the CASL methods. The Westinghouse focus has been in areas where multiphysics modelling and simulation (M&S) advances will have a significant benefit in improving the economics and operation of the existing fleet of reactors. Additionally, it is also recognized that advanced M&S techniques can inform decisions for the next generation of advanced fuel designs and new generation reactors. The key areas that are discussed are advances in predictive nuclear design methodology, prediction of the crud induced power shift phenomena, and more realistic prediction of limiting reactor constraints such as Departure from Nucleate Boiling (DNB). Included are the recent modelling advances that are needed for predicting Accident Tolerant Fuel (ATF) behavior.

1. Introduction

CASL's mission is to provide coupled, higher-fidelity modeling and simulation capabilities needed to address Light Water Reactor (LWR) operational and safety performance-defining phenomena. Since its inception, CASL's Virtual Environment for Reactor Applications (VERA) has advanced the state-of-the-art for commercial reactor simulations. Recent developments have improved accuracy, reduced computational requirements, and expanded capabilities. Westinghouse's strong engagement with the CASL program has led to application of CASL tools to several different problems, providing user feedback to developers along the way, resulting in further improvements with more accurate and usable tools.

Existing methods for reactor core analysis are limited in fidelity and usually rely on couplings to coarse Thermal-Hydraulic (T/H) mesh followed by separate (and sometimes inconsistent) bounding fuel performance analyses. The methods are not necessarily difficult to use, but require conservative modeling assumptions, different inputs and technical knowledge for each physics package. Most notably the core simulation is a multi-stage approach using several levels of condensation, resulting in an increase in computation speed but a loss of accuracy at the local rod level. The T/H and fuel performance analyses supporting reactor core reload and design are not typically a part of the neutronics core simulator, and therefore are not directly used for reactor simulation. VERA has been developed to address these shortcomings for improved accuracy on smaller spatial scales. It is based on advanced physics and mathematical methods and limits the number of stages and approximations made between user input and solution. The individual methods employed by VERA are each

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an advancement in technology beyond analysis methods currently used by the industry, either in methodology and/or spatial scale and refinement. For instance both the neutronics and T/H methods in VERA solve directly on the fuel rod level as opposed to a lumped 2x2 per assembly scale. The individual physics codes in VERA are directly coupled, and multi-physics problems are fully solved iteratively to convergence. The results of this process are always best-estimate, and therefore the coupling is, in itself, and advancement beyond current practice.

Figure 1 shows the current components of VERA. Items inside the blue box show different functional areas, including neutronics, thermal-hydraulics, coolant chemistry, and fuel performance. For each of these functional areas, different codes provide multiple levels of fidelity. While Westinghouse's main focus been in the performance-based components (e.g., MPACT, Shift, CTF, etc…) for solving large, realistic problems, tools outside the blue box provide the necessary accuracy on small scale problems to improve the accuracy from "production" tools (e.g., STAR-CCM+).

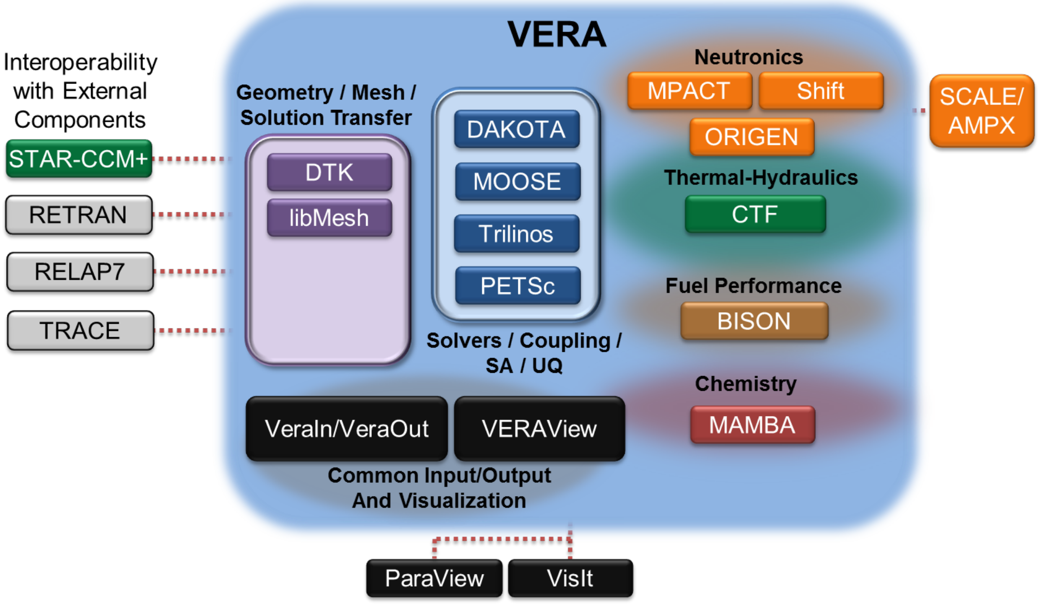

Figure 1. Components of VERA

The key areas that are discussed in the following sections are advances in reactor analysis methodology for more realistic prediction of limiting reactor constraints such as DNB, and prediction of the crud induced power shift phenomena. Included are the modelling advances that are needed for predicting Accident Tolerant Fuel (ATF) behavior. Examples are provided where the developing CASL technology has already been evaluated and used by Westinghouse, as well as the future expectations for the industry use of the finished CASL products.

2. Nuclear Design & Multiphysics Applications of CASL Tools

2.1. AP1000® PWR Core Modelling and Simulations

VERA-CS (VERA-Core Simulation) was applied to the core physics analysis of the AP1000 PWR [1,2,3], which features an advanced first core with radial and axial heterogeneities and at-power control rods insertion to perform the MSHIM™ advanced operational strategy. This advanced core design relies on low enrichment fresh fuel in the core periphery to simulate burnt fuel in a typical low leakage core pattern. The benefit of this type of loading is that it saves many millions of dollars when compared to traditional first core designs that have been utilized in the past. However, this loading challenges the current industry code sets that use diffusion theory to model core design. The first of a kind AP1000 first core design requires a higher fidelity application to calibrate the accuracy of current design tools, and this makes the application of VERA-CS to the AP1000 PWR first core especially relevant. Initial calculations focused on the qualification efforts at hot zero power conditions with established reference Monte-Carlo solutions. The comparison of both global core parameters (e.g. critical boron concentration, rod worth and reactivity coefficients) and fine-mesh fission rate spatial distribution indicated excellent numerical agreement between VERA-CS, Monte Carlo predictions, and Westinghouse's in-house core physics package, reinforcing confidence in the startup predictions. Analyses were then extended to Hot Full Power (HFP) conditions [4]. Lattice depletion simulations were performed with VERA-CS, PARAGON2 Westinghouse's state-of-the-art lattice code with ultra-fine (6064 energy groups) energy resolution [5], and SERPENT continuous energy Monte Carlo [6], which showed excellent agreement among all codes. Core depletion simulations were performed with VERA-CS and the Westinghouse core physics code system ANC/PARAGON2 [7], demonstrating excellent agreement in the key core operational parameters, and confirming the Westinghouse design values for the AP1000 PWR first core. This analysis demonstrated an excellent application of VERA physics predictive capabilities – namely, that the availability of higher physics resolution tools is especially important when the use of current methods may be under doubt to address first of a kind situations.

2.2. Watts Bar Unit 2 Startup Simulations

In the summer of 2015, CASL successfully modeled the operating history of TVA's Watts Bar Nuclear Unit 1 with VERA [8]. Results compared well with historical measurements including critical boron concentrations, control bank reactivity worths, and in-core measured neutron flux distributions. This was the first true industry-grade validation of VERA against past fuel cycle operations since the program's inception, and the highest resolution plant benchmark ever performed. With the startup of TVA's Watts Bar Unit 2 in 2016, there has been a unique opportunity to apply VERA's high-fidelity multi-physics capability in advance of the first commercial reactor startup in the United States in two decades. With CASL partners TVA and Westinghouse, and with the assistance of the INL High Performance Computing Facility, calculations were prepared in advance of the startup which corroborated the results from the design licensed methods. Additionally, with support from the Oak Ridge Leadership Computing Facility, the VERA tools has been utilized to follow the entire initial power ascension procedure, through all power ramps, load reductions, and shutdowns, with comparisons of measured core reactivity and in-core power distributions. This unique activity represents the largest single simulation for CASL to date, and an excellent benchmarking opportunity, during a special and critical time for TVA, Westinghouse, and the entire nuclear power industry. With the help of VERA results, Westinghouse was able to refine its in-house models (e.g., detailed explicit WABA, reflector constants, cross-section library based on ENDF/B-VII.1 data, etc.)

2.3. The AP1000 Reactor Control Rod Ejection Accident

Control rod ejection events were simulated for the AP1000 reactor core at the end of cycle hot zero, hot full and part power conditions using VERA-CS [10]. The core was first depleted to end of Cycle 1 and then calculations were restarted to simulate the rod ejection events in full core geometry. These simulations were performed using MPACT coupled to CTF within VERA-CS. Figure 2 shows the core-wide radial power distribution close to the top of the core along with the axial pin power distribution and the hot assembly pin power distributions at the peak of the power pulse. As the figure indicates, the power distribution is highly asymmetric; higher peaking factors are observed in and around the ejected rod location and lower power rods are observed away from the ejected rod location. With this study, it was demonstrated that the VERA-CS code system was capable of simulating full core geometry rod ejection events in a stable manner with the expected power pulse resulting from a superprompt critical reactivity insertion and the resulting negative Doppler reactivity feedback. With high-fidelity simulation capabilities, VERA-CS is positioned to support the industry on analyzing reactivity initiated accidents, and to assist in responding to regulatory rule changes.

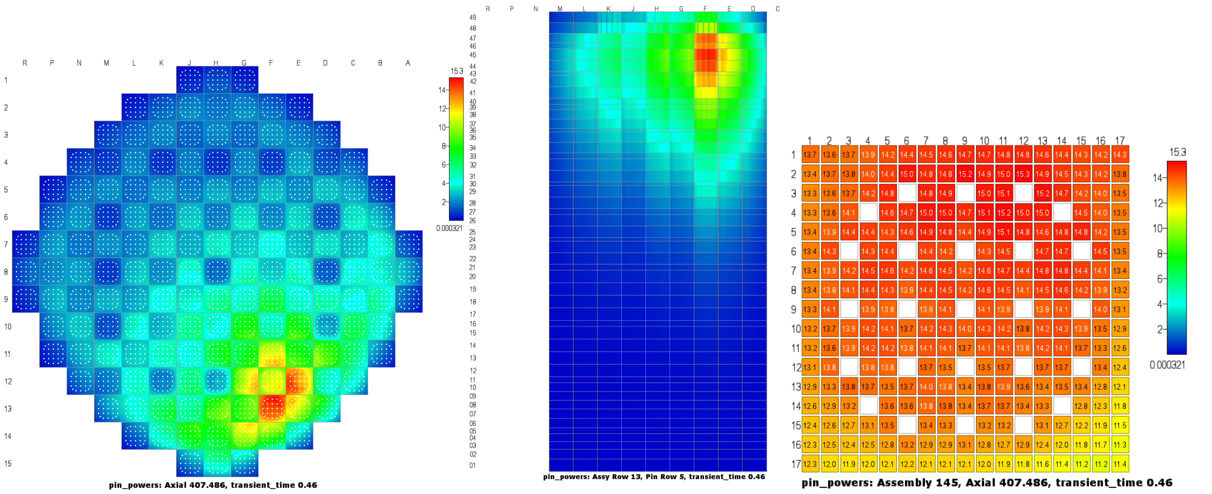

Figure 2. Core-Wide and Assembly Pin Power Distribution at the Peak of Power Pulse for HZP Rod Ejection

2.4. DNB Evaluation of PWR Steamline Break Event

Reactor core response with respect to Departure from Nucleate Boiling (DNB) at the limiting time step of a 4-loop PWR postulated main steamline break (MSLB) event initiated at HZP was evaluated using both CASL's VERA-CS and the Westinghouse coupled code system ANCKVIPRE [11], which links the ANC code with the VIPRE-W code. ANC is Westinghouse's multi-dimensional nodal diffusion code for all nuclear core design calculations and VIPRE-W is the Westinghouse version of Electric Power Research Institute's VIPRE-01 T/H subchannel code developed for light water reactor core design applications. For this study, two MSLB scenarios were evaluated for the 4-loop core, in which either offsite power was available and all coolant pumps remained in operation (i.e., the high-flow case), or without offsite power and the reactor core was cooled through natural circulation (i.e., the low-flow case) [12,13]. Results from the ANCKVIPRE simulations were compared to those from VERA-CS. Simulations with ANCKVIPRE and VERA both used reactor system state-points generated as the core boundary condition at the DNB limiting time step from the HZP MSLB transient calculation using the system transient code RETRAN-02 [14] and core inlet temperature and flow distributions using the STAR-CCM+ computational fluid dynamics (CFD) code [15] at the DNB limiting time step. In order to capture the asymmetric power distribution in the reactor core (Figure 3) due to the broken steam pipe in one loop and the stuck Rod Cluster Control Assembly (RCCA), both the ANCKVIPRE and VERA models were set up for the whole reactor core, and the VERA-CS model had the higher resolution on a rod-by-rod and subchannel basis.

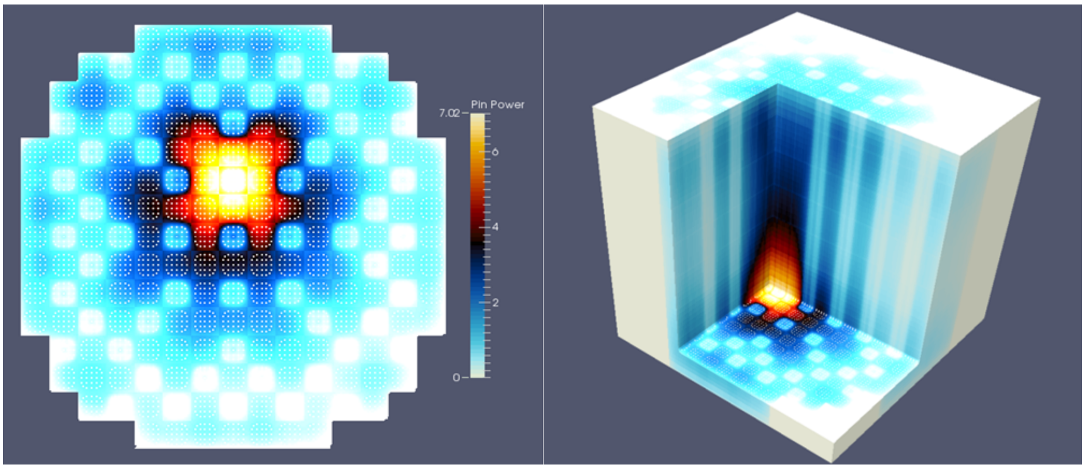

Figure 3. Whole-Core Pin Power Distribution for MSLB High-Flow Case

Coupled neutronic and T/H code calculations were performed using the two coupled codes to predict the quasi-steady state core response such as the power distribution and local fluid conditions as well as DNB calculations using the available correlations in VIPREW and CTF. The results from the two coupled multi-physics code systems, as seen in Table 1, confirmed that the HZP MSLB case with offsite power available (the high-flow case) is more limiting with respect to the DNB acceptance criterion, consistent with the limiting case analyzed in a 4-loop plant Safety Analysis Report.

Table 1. Comparison of Minimum DNBR Values for MSLB Low- and High-Flow Cases

Case	P (psia)	h_{inlet} (BTU/lbm)	$\dot{m}_{inlet}$ (Mblbm/hr-ft2)	q' (BTU/s-ft)	MDBNR
Low-flow	808.2	377.70	0.275	0.489	9.616 (W3)
High-flow	460.1	398.90	3.000	1.069	3.624 (W3)

3. Thermal-Hydraulic Applications of CASL Tools

3.1. PWR DNB Margin Improvement

CASL and Westinghouse are developing a new application of the coupled VERA-CS code system to modeling and simulation of PWR reactor core response during accident conditions for thermal (DNB) margin improvement. The idea here is to provide more plant operational margin, which can be used in many different ways by utilities to reduce operating costs. This application focuses on predicting core power distributions that affect input to the reactor protection system such as the Over-Temperature Delta-T (OTΔT) and the Over-Power Delta-T (OPΔT) trip. The more realistic predictions of the core power distributions from the high-resolution VERA-CS code system provide better quantification of the DNBR margin retained in the existing trip setpoints based on the traditional safety analysis approach. The VERA-CS application will help the industry to reallocate the design margins to enhance reactor safety and to reduce costs in fuel reloads and/or plant operations. The application is part of the CASL DNB Challenge Problem implementation plan.

The OTΔT trip protects the core against low DNBR and trips the reactor when the loop temperature measurements reach the trip setpoint which contains a penalty function, f(ΔI), to account for the effect of axial power distributions that are more adverse to DNBR margin than the reference axial power distribution used for the core safety limit. The penalty is a function of the indicated difference between the top and bottom detectors of the power range neutron ion chambers (ΔI). The input to calculation of the f(ΔI) function for DNB protection is often defined as difference between the core inlet temperature for the core safety limit and the calculated temperature with the axial power shape versus ΔI. The ΔTin versus ΔI limit indicates that reductions of the core inlet temperature for axial power distributions skewed to the bottom or top of the reactor core, in order to preserve the core safety DNBR limit.

VERA-CS includes coupled neutronic, thermal-hydraulic, and fuel temperature components with an isotopic depletion capability. The neutronic capability employed is based on MPACT, a three-dimensional (3-D) whole core transport code. The thermal-hydraulic and fuel temperature models are provided by the COBRA-TF subchannel code. The isotopic depletion is performed using the ORIGEN code system. A Westinghouse-designed 4-loop PWR operating cycle, with its core containing the 17x17 VANTAGE 5H fuel assemblies, was chosen as the reference reactor core for the study using the VERA-CS code system. Various cases were selected to define the core conditions in the operating region at different depletion steps during the cycle, simulating DNB limiting conditions of RCCA malfunction events mitigated by the OTΔT/OPΔT trip protections. Ranges of the dropped control rod worth in pcm and the differential rod worth in pcm per unit length are defined based on the plant analysis record.

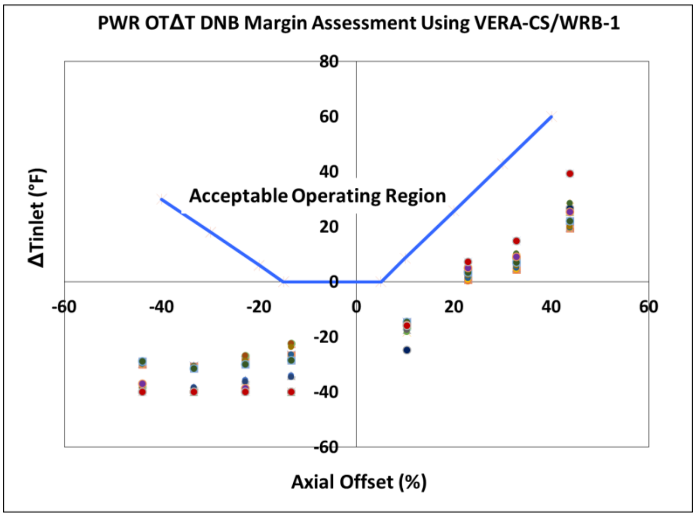

Figure 4. PWR OTΔT DNB Margin Assessment based on VERA-CS

The VERA-CS calculations provide the power distributions in the subchannels as input to DNBR predictions using the CTF code and the WRB-1 DNB correlation. The 3D power distributions from the VERA-CS simulations show substantial margins to the existing core and axial offset limits for the OTDT trip setpoints, which were generated conservatively based on the 1-D axial shapes and the hottest fuel rod at the radial peaking power limit. The calculated results are compared to the ΔTin-ΔI limit in Figure 4. Each point in the figure 5 represented the required inlet temperature change for the accident-specific core power distribution to meet the DNBR limit based on the VERA-CS result. The comparison indicates significant margin about 20°F on average between the calculated T_{inlet} based on the VERA-CS power distributions and the ΔT limit. The 20°F temperature difference is about 20% DNBR margin. The VERA-CS 3D results show that no reduction of the trip setpoint was needed for the bottom skewed axial power distribution with negative ΔI values. The VERA-CS results also indicate that the ΔT-ΔI limit or the f(ΔI) setpoint could be less restrictive with the top skewed axial power distribution with positive ΔI values.

3.2. CFD Based Thermal-Hydraulic Applications

Current industrial predictions of Departure from Nucleate Boiling (DNB) are performed using one-dimensional thermal-hydraulics codes such as VIPRE-W and COBRA-TF (CTF), which rely on empirically-derived parameters or correlations to model the spacer grid effects, and to specify the conditions for DNB will occurrence. Computational fluid dynamics (CFD) simulations of DNB promise to eliminate these limitations and have received increased attention in recent years.

Since modeling the behavior of individual bubbles is not yet feasible, these CFD boiling simulations employ the Eulerian multiphase framework which relies on various physics-based boiling closure models to account for subgrid-scale effects. Most industrial CFD predictions of DNB have been performed using commercial codes which typically use similar boiling closure models. The CASL program is developing improvements to the existing CFD boiling closure models used in DNB prediction.

Recently as part of the CASL program, Westinghouse has performed evaluations of existing boiling closure models to provide a baseline of the predictive accuracy for the improved models being developed by CASL. CFD predictions of DNB were compared with experimentally-measured values for two Westinghouse DNB test grid designs under a variety of pressure, inlet temperature and mass flow rate conditions [16,17]. The STAR-CCM+ CFD code with a selected set of boiling closure models was used for the DNB simulations.

Figure 5 shows one of these spacer grid designs, referred to as the 4-Vane (4V) design, which demonstrates that a CFD-based modeling approach is capable of modeling very complex geometries. Some DNB prediction results on the 4V grid design are compared with test data in Figure 6. The results show that the selected boiling closure models are capable of predicting qualitative trends for CHF, but there is still room for improvement regarding the quantitative prediction of DNB. Ongoing developments within CASL are expected to address the limitations of the existing boiling closure models. The improved models for CFD-based DNB modeling being developed by CASL will form the basis of a DNB prediction tool for new and modified spacer grid and fuel assembly designs at Westinghouse.

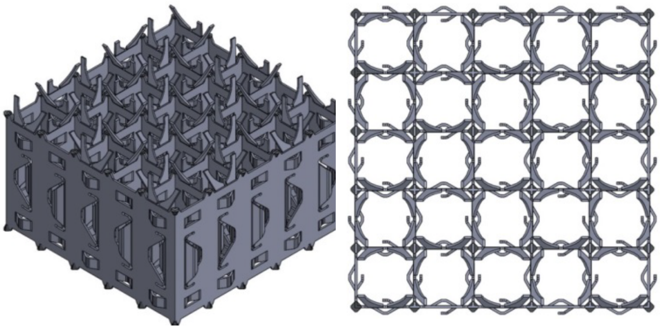

Figure 5. CAD Model of 4-Vane (4V) Grid – Isometric View (left) and Top View (right)

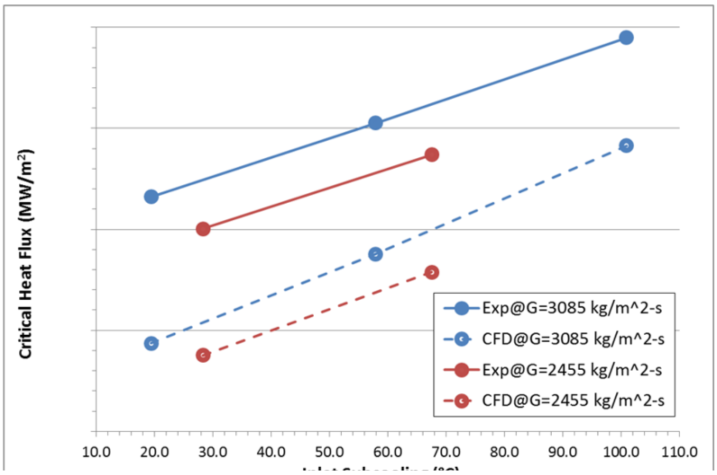

Figure 6. 4-Vane CHF Comparison for Nominal System Pressure of 16.0 MPa at Different Mass Fluxes and Inlet Subcooling Levels.

While CFD-based models offer the ability to model detailed geometry from first-principle physics, this comes at the expense of greater computational cost. An alternative to the direct use of CFD to model thermal-hydraulics is its use to improve faster, lower-fidelity modeling techniques, such as the one-dimensional codes mentioned earlier. This use of higher-fidelity codes to inform and calibrate the results of lower-fidelity codes is often called a High-to-Low (Hi2Lo) strategy. The Hi2Lo approach takes advantage of the granular, detailed information provided by the high fidelity code, and the faster run times of the low fidelity code. The Hi2Lo methodology for nuclear thermal-hydraulics analysis is being developed under the CASL program using Dakota, a software suite that enables design exploration, model calibration, risk analysis, and quantification of margins and uncertainty with computational models.

The CASL Hi2Lo methodology has recently demonstrated improvement of the spacer grid modeling capabilities of the CTF subchannel thermal hydraulic code using the high-fidelity STAR-CCM+ commercial CFD code and available experimental data from a 5x5 rod bundle containing Westinghouse mixing vane spacer grids [18,19]. CTF uses a two-phase turbulent mixing coefficient parameter (β), to represent the effects of the inter-channel mixing induced by the spacer grids, and was therefore used as the calibration parameter for the analysis. As shown in Figure 7, it was found that the CTF-predicted subchannel temperatures showed a sensitivity to β, and the Hi2Lo procedure with STAR-CCM+ was used to find an optimal value of $\beta = 0.037$. This Hi2Lo process may also be used to improve the values of other parameters used by CTF, such as those used in the prediction of DNB, for example.

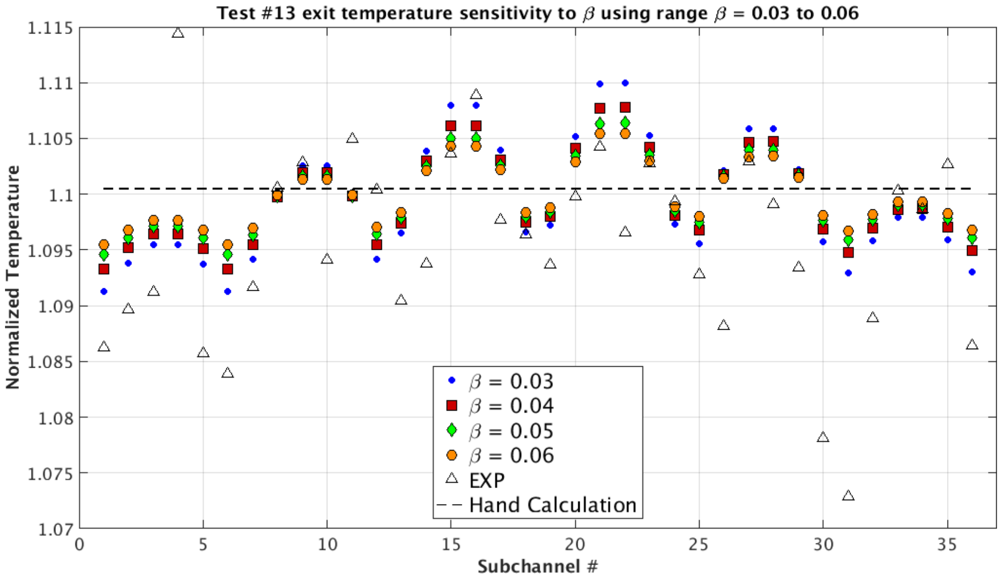

Figure 7. Sensitivity study on the mixing coefficient β

4. Fuel Rod Applications of CASL Tools

The primary tool used for fuel rod design and fuel performance analyses in Westinghouse is the PAD code (latest NRC approved version is PAD5 [20]). The PAD5 code incorporates a set of interrelated fuel and cladding performance models that are calibrated to best predict PWR fuel rod performance parameters such as fuel temperatures, rod internal pressure, and fuel and clad dimensional changes as a function of time and burnup. Key performance models for complex behaviors such as fission gas release, fuel densification and swelling, and cladding creep and growth use phenomenological based model forms with empirically derived model coefficients to best predict measured data for the range of performance parameters defined by the available measurement database. The scope of the calibration and validation database for measured fuel rod performance parameters defines the range of applicability for PAD5.

In VERA-CS, the BISON code [21] is used for fuel rod design and performance analyses. It provides the fuel temperature input to MPACT for feedback calculations. It can be run in a coupled mode within VERA or in a standalone mode. BISON is based on the Multiphysics Object-Oriented Simulation Environment (MOOSE) framework developed by Idaho National Laboratory (INL). MOOSE [22] is a finite element based framework for the solution of non-linear differential equations adapted to model fuel rod performance. It incorporates a diverse set of models, or kernels, which can be selected to calculate fuel performance for a broad range of different fuel types including particle fuel, metallic rod and plate fuel and standard PWR UO_2 fuel rods. BISON can model a variety of mesh geometries (e.g., 1.5D, 2D R-Z, and 3D). High definition material property and fuel performance models for a multiple fuel and cladding materials, based on first principals to the extent possible, are built into BISON, and the modular architecture supports the addition of new models as needed.

BISON is under active development. The industry need for advanced fuel and cladding is driving efforts to add and/or update models for ATF materials. For fuel materials such as U_3Si_2, when data is limited, multiscale modeling is being done to define basic material behaviors on an atomistic scale. Using intermediate scale modeling, these fundamental atomistic scale models are translated into engineering scale models for incorporation into BISON (Figure 8). Complex models such as fission gas transport and release, gas bubble formation and gas bubble swelling can be derived on this basis and then extended to incorporate the effects of time and burnup. With the completion of these models, BISON will be capable of predicting ATF performance over the full range of planned operation and will be capable of extending the understanding of expected behaviors beyond the range of available measured data.

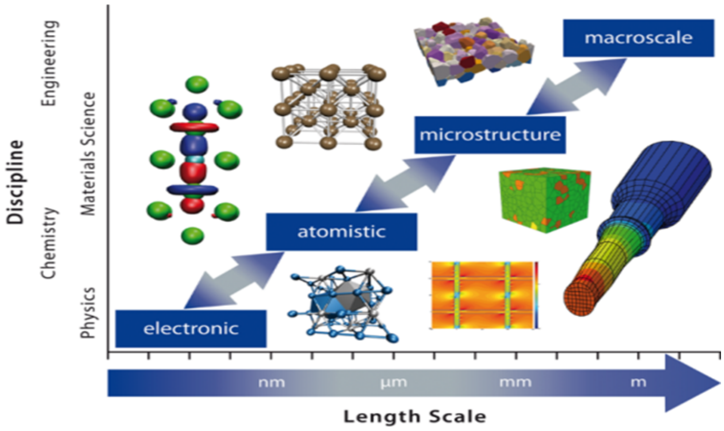

Figure 8. Multiscale Mechanistic Modelling Process

An important attribute of BISON is its flexibility in modeling a range of fuel and cladding types and geometries, including new fuel and cladding materials such as U_3Si_2 fuel pellets, coated zirconium based and silicon carbide (SiC) composite claddings. The high fidelity solution methods in BISON are capable of predicting behaviors over a wide range of operating conditions, including transients. As a result, it provides an important tool for evaluating ATF concepts where empirically based codes are limited in scope because of the limited availability of measured data.

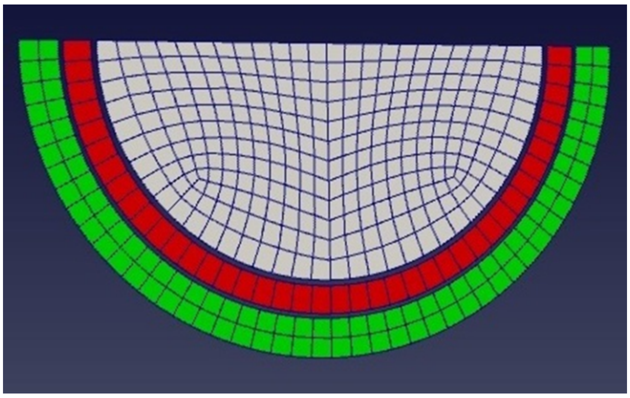

Figure 9. Double Encapsulated Test Fuel Rod Concentric Case

While BISON models for ATF are not yet complete, the current code can be used to evaluate expected ATF performance for designs that are outside the range of the normal fuel rod design code capability. An example of BISON application to ATF is an assessment of the impact of eccentricity on fuel and cladding temperatures for a double encapsulated test rod with U_3Si_2 fuel pellets. Double encapsulation may provide additional assurance that the fuel failure will not occur under demanding test irradiation. The inner capsule contains the fuel pellets, spacer and plenum spring and is capped and sealed with typical helium pre-pressure. This capsule is then inserted into a fuel tube which is also capped, sealed and pre-pressurized. Figure 9 illustrates the BISON mesh for a concentric system with the outer clad in green, the inner clad in red and the fuel pellet in gray.

Two eccentric loading cases were also considered: Case A has both the pellet and the inner clad shifted to contact on one side; Case B has the inner clad shifted to the contact with the outer clad on one side while the pellet is shifted to contact with the inner clad on the opposite side. The pellet mid-point displacement is fixed.

Using a power history ramping linearly from 0 to 35 kW/m in the first 7000 seconds of operation and then held constant at 35 kW/m to 1E8 seconds (27778 hours), BISON was run for each case to assess peak fuel and inner clad temperatures, with results shown in Figure 10.

Peak fuel temperatures occur when power reaches 35 kW/m at 7000 seconds, after which they reduce with time as the gaps begin to close with fuel swelling and cladding creep. The concentric geometry produces the maximum fuel temperatures with the two eccentricity cases having similar peak temperature profiles.

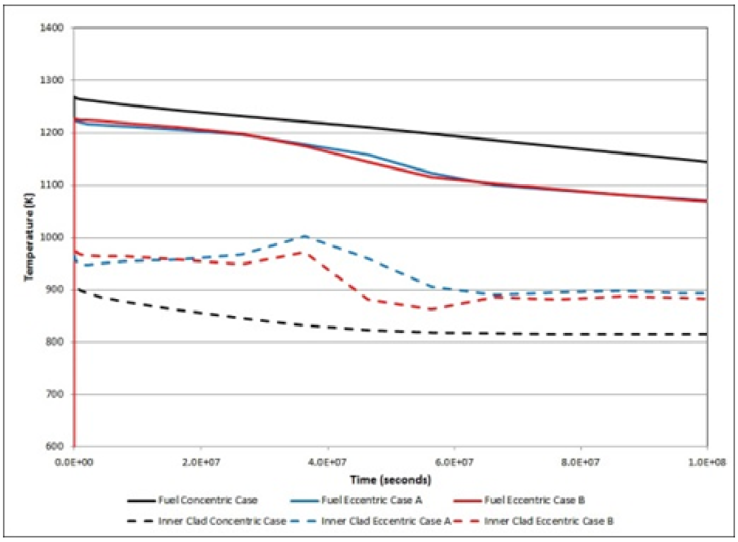

Figure 10. Peak Fuel and Inner Clad Temperature versus Time

In the concentric case, the inner clad temperature also peaks at 7000 seconds, consistent with the peak fuel temperatures. However, with eccentricity, the inner cladding temperature peaks later, at about 3.7E7 seconds. At that time, Case A peak clad temperature is about 170K greater than the reference concentric case and Case B peak clad temperature is about 140K greater. The Case B peak clad temperature at 3.7E7 seconds exceeds the maximum clad temperature predicted for the concentric geometry at 7000 seconds by about 100K. The temperature distributions for the concentric case peak fuel temperature at 7000 seconds and for eccentric Case A at 3.7E7 seconds are illustrated in Figure 11 and Figure 12.

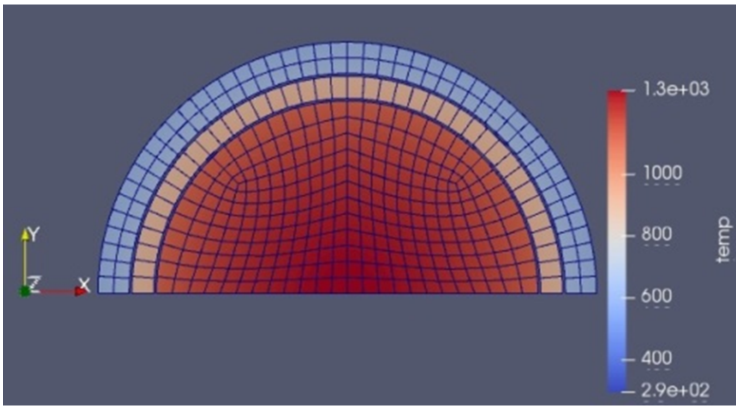

Figure 12. Concentric Case Temperature Distribution at 7000 Seconds

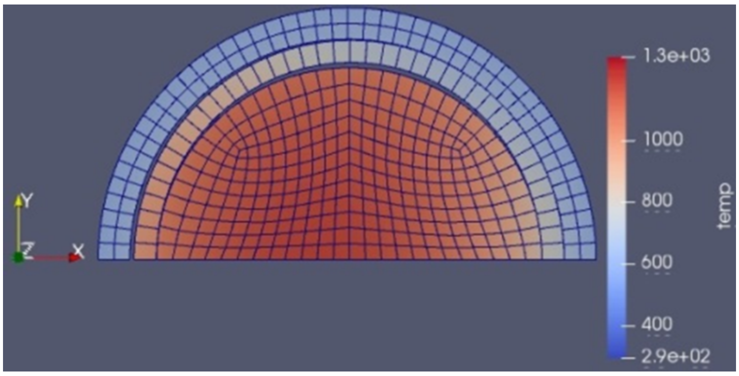

Figure 11. Eccentric Case A Temperature Distribution at 3.7E7 Seconds

These analyses could also be performed with other finite element codes such as ANSYS, but what is unique to BISON is the ability to perform these analyses with integrated fuel and cladding performance models so that fuel rod specific impacts such as fission gas release and can be inherently accounted for. Advanced multiscale modeling techniques provide first principles based engineering scale models in BISON for evaluation of new fuel and clad materials with complex geometry that cannot be readily modeled in other tools. In the simple example above, BISON is already using these models for U_3Si_2 fission gas release and swelling.

These models will be improved as atomistic analyses of burnup affects are completed and updated models are incorporated into BISON and adjusted to best predict available measured data. With these first principles based modeling capabilities, BISON will provide a sound basis for evaluation of performance trends outside the range of the measured data base. These analytical extrapolations of the database will provide added confidence in the expected performance of ATF designs such as the Westinghouse EnCore® advanced ATF.

Coupled with advanced core physics modeling with VERA, BISON analyses will confirm the expected behaviors of lead test rods (LTR.) and lead test assemblies (LTAs). Another CASL tool, DAKOTA, can be used to perform automated analysis of potential uncertainty impacts on fuel and cladding performance.

The CASL program, with the development of the VERA suite of advanced, integrated, high fidelity core and fuel rod modeling tools, including BISON, provides timely support to the industry demand for advanced accident tolerant fuel.

5. Advanced Analyses of CRUD

The VERA toolset includes an advanced tool for CRUD growth on the fuel cladding known as MAMBA [23]. This capability is directly coupled to the neutronics and thermal-hydraulics solvers providing direct feedback to important quantities such as peak clad temperature, power distribution, and shutdown margin.

MAMBA is an advanced surface chemistry solver for accumulation and erosion of CRUD on each fuel rod surface in the VERA model. Microstructural chemistry and heat transfer is solved in three dimensions in explicit meshes on the outside of the cladding. Soluble boron from the reactor coolant is captured in the CRUD layer under sub-cooled boiling conditions and subsequently dissolves when reactor powers and temperatures are reduced. MAMBA can also be informed by CFD simulations for very high-resolution predictions of CRUD striping upstream of spacer grids. Finally, MAMBA provides engineering-scale models for CRUD source terms and mass balance in the reactor coolant system, as well as supports the shuffling and cleaning of CRUD during refueling outages.

The coupling of MAMBA to the other advanced tools in VERA is represented in Figure 13. MAMBA leverages the critical boron concentration from MPACT and the coolant and cladding temperatures from CTF to calculate the axial and azimuthal distribution of CRUD and boron mass for every fuel rod in the simulation. This boron mass is then directly included in the neutronics transport solution and the CRUD provides an increased thermal resistance to CTF. These calculations are performed for each iteration until convergence between all three is achieved, and then repeated for each depletion step in simulation of plant operation.

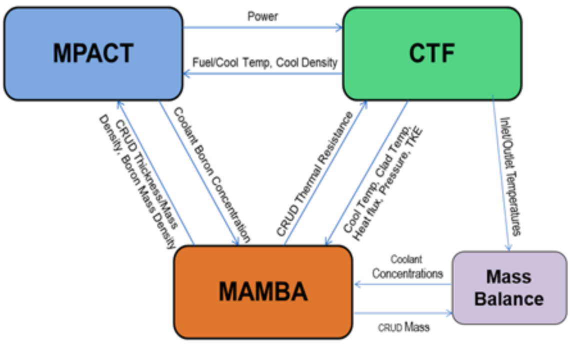

Figure 13. Multi-way Coupling of MAMBA within VERA

The CRUD solution methodology in VERA provides a one-of-a-kind, state-of-the-art analysis tool for CIPS and CILC calculations. Current industry methods provide a qualitative risk-based approach for making conservative core design decisions for avoiding the onset of CIPS and CILC. This has been successful at avoiding these issues for decades, but comes with significantly increased fuel costs as loading patterns are often de-optimized for conservatism. VERA enables a more direct quantitative approach to this analysis, which will allow core designers to better under the existing margin to CIPS and CILC, to be able to precisely calculate the effects of increased CRUD on vital core design parameters such as axial offset and shutdown margin, and possibly to implement advanced monitoring schemes that would allow operators to identify the early onset of CIPS from detailed local in-core data. Ultimately, VERA will enable more confidence in core designs resulting in significant fuel cost savings over many fuel cycles.

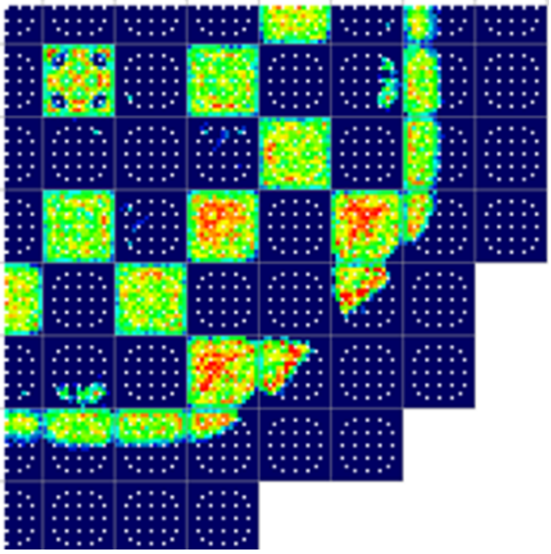

Figure 14. Heterogeneous Boron Distribution in a Radial Slice of a Proposed Catawba 2 Loading Pattern

To demonstrate these potential savings, an early CIPS application using a previous development version of MAMBA was performed with the cooperation of Duke Energy. VERA was utilized to simulate three proposed core designs for Catawba 2 Cycle 22 [24]. Though the most expensive, most conservative design with the least predicted boron uptake was selected, simulations with VERA showed that the effects of CRUD on the axial power shape and shutdown margin between the three patterns was relatively insignificant. The fuel cost savings in this case would have been between $125K and $425K, depending on market conditions. As MAMBA matures, and core designers begin to utilize this advanced capability going into the loading pattern optimization process, the potential savings may be much larger in some cases. Figure 14 shows the heterogeneous boron deposition in the CRUD layers for a radial slice of one of the Catawba 2 models.

6. Summary and Conclusions

This paper focused on the specific industry applications of the CASL technology, including predictive nuclear design methodology, such as reactivity and power distributions, prediction of the CIPS phenomena, more realistic prediction of limiting reactor constraints such as DNB, as well as the recent modelling advances needed for predicting ATF behavior. Examples were provided where the developing CASL technology had already been evaluated and used by Westinghouse, as well as the future expectations for the industry use of the finished CASL products. With the help of the CASL tools, it is recognized that advanced M&S techniques can inform decisions for the next generation of advanced fuel designs and new generation reactors, as well as help in the resolution of any anomalous core behavior in existing LWRs.

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