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INTEGRATED RADIATION TRANSPORT AND NUCLEAR FUEL PERFORMANCE FOR ASSEMBLY-LEVEL SIMULATIONS

Steven Hamilton, Kevin Clarno, Bobby Philip, Mark Berrill,
Rahul Sampath and Srikanth Allu*
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6285
hamiltonsp@ornl.gov

ABSTRACT

The Advanced Multi-Physics (AMP) Nuclear Fuel Performance code (AMPFuel) is focused on predicting the temperature and strain within a nuclear fuel assembly to evaluate the performance of existing and advanced nuclear fuel bundles within nuclear reactors. AMPFuel was extended to include an integrated nuclear fuel assembly capability for (one-way) coupled radiation transport and nuclear fuel assembly thermo-mechanics. This capability is the initial step toward incorporating an improved predictive nuclear fuel assembly modeling capability to accurately account for source-terms (i.e. neutron flux distribution, coolant conditions, and assembly mechanical stresses) of traditional nuclear fuel performance simulations.

A novel scheme is introduced for transferring the power distribution from the SCALE/Denovo (Denovo) radiation transport code (structured, Cartesian mesh with smeared materials within each cell) to AMPFuel (unstructured, hexagonal mesh with a single material within each cell), allowing the use of a relatively coarse spatial mesh for the radiation transport and a fine spatial mesh for thermo-mechanics with very little loss of accuracy.

With this new capability, AMPFuel was used to model an entire 17×17 pressurized water reactor fuel assembly with many of the features resolved in three dimensions (for thermo-mechanics and/or neutronics). A full assembly calculation was executed on Jaguar using 40,000 cores in under 10 hours to model over 160 billion degrees of freedom for 10 loading steps. The single radiation transport calculation required about 50% of the time required to solve the thermo-mechanics with a single loading step, which demonstrates that it is feasible to incorporate, in a single code, a high-fidelity radiation transport capability with a high-fidelity nuclear fuel thermo-mechanics capability and anticipate acceptable computational requirements.

The results of the full assembly simulation clearly show the axial, radial, and azimuthal variation of the power, temperature, and deformation of the assembly, highlighting behavior that is neglected in traditional axisymmetric fuel performance codes that do not account for assembly features, such as guide tubes and control rods.

Key Words: multiphysics, nuclear fuel performance, thermo-mechanics

1. INTRODUCTION

The ability to predict the behavior of nuclear fuel, at a continuum scale, during irradiation in an operating reactor poses many significant challenges, including prediction of the effects of the

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microstructural evolution of the materials, the uncertainty and variability in the materials, and the impact of external source-terms and boundary conditions. These external factors exist in the form of the power distribution, coolant conditions (temperature and pressure), mechanical stresses (debris, bowing, and fretting), and chemistry effects (oxidation and CRUD). This report documents the initial developments in the Advanced Multi-Physics (AMP) Nuclear Fuel Performance code (AMPFuel) [1] to incorporate an improved predictive nuclear fuel assembly modeling capability by incorporating a detailed physics-based power distribution using the radiation transport capability from SCALE/Denovo (Denovo) [2]. Previous studies have shown that uncertainty in the power distribution can account for significant uncertainty in the final fuel performance simulation [3].

This paper describes an initial simulation integrating high-fidelity neutronics (capable of predictive neutron flux distributions) and fuel thermo-mechanics for assembly-level fuel performance analyses. Section 2 offers motivation for predictive multiphysics calculations. Section 3 describes the software and numerical methods used in this study. A description of the problem and results of the full assembly simulation are found in Section 4. Section 5 offers some concluding remarks and comments on the focus of future efforts.

2. MOTIVATION

The need for simulations accounting for multidimensional, multiphysics effects has been seen in many operating nuclear reactors [4], perhaps most notably in a set of fuel failures in Unit 1 of the Hatch Nuclear Power Plant [5]. In Cycle 21, during a routine control blade movement, six fuel rods failed (due to pellet-clad interaction) in four fuel bundles that were located in three out of four symmetric control cells. All of the failed rods were adjacent to the control blades. Many analyses have been performed to understand the cause of these failures, including the effects of local pin power peaking due to burnup gradients, fuel channel bowing, and transient thermo-mechanics [5]. Although the cause is still not fully explained, simulations have shown that simplifications in the power distribution and irradiation history underpredicted the average and within-pin power distributions, and the azimuthally symmetric fuel performance simulations were unable to properly account for the multidimensional thermo-mechanical response of the fuel rod.

2.1. Traditional Neutronics Source-Term

Traditional fuel performance codes, such as FRAPCON [6], model a single fuel pin in isolation and require the user to input the average pin power and axial pin power distribution for a given time step in piece-wise constant "fuel regions" as a function of time, producing region-averaged power as a function of time. Within each "fuel region," effective (one energy group) spatially averaged neutron cross sections have been defined for the key transuranic isotopes along with a spatially flat (or radially varying) effective neutron flux and radially varying effective uranium-238 (U238) capture cross section. These quantities are based on a series of MCNP simulations [7]. The fast neutron flux, which affects material models in the cladding through the fluence (time-integrated neutron flux), is specified by means of a user-defined magnitude or ratio of fast flux to average power [8–10]. This suite of approximations has been empirically and computationally tuned to provide sufficient accuracy for nominal operation with UO_2 and mixed oxide (MOX) in well-defined problems, but it lacks the extensibility and generality required for

predictive simulation. Significant limitations restricting the predictive nature of this approach include the inability to account for variations in the neutron flux distribution due to changes in isotopics, coolant, temperature, or geometry (including radial and azimuthal variations introduced by neighboring fuel pins and assembly structure).

3. SOFTWARE AND METHODOLOGY

In this section we describe the numerical methods and software to be used in this study. The power distribution to be used is computed using the SCALE/Denovo radiation transport solver using cross sections generated with the SCALE/CSASI sequence of the SCALE package. The thermo-mechanics portion of the calculation is performed with the Advanced Multi-Physics (AMP) Nuclear Fuel Performance code. Denovo has been configured within AMPFuel, and the entire system is executed in parallel.

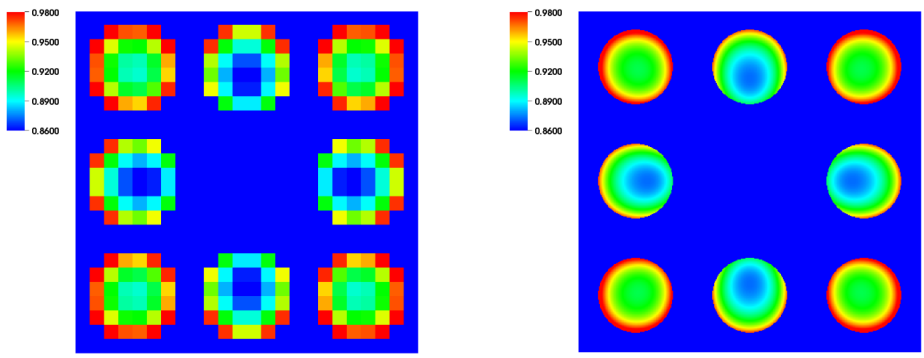
3.1. Radiation Transport Using Denovo

Denovo is a framework for solving the Boltzmann radiation transport equation. It has the capability to solve the two-dimensional (2D) and 3D deterministic multigroup discrete-ordinates equations. Denovo uses the Koch-Baker-Alcouffe (KBA) [11] parallel spatial decomposition on nonuniform Cartesian grids. Volume-weighted subcell mixing of materials is used to offset geometric difficulties associated with the use of Cartesian meshes. The accuracy of Denovo has been verified for nuclear reactor simulation through the use of standard numerical benchmarks for radiation transport [12]. Additional parallelism in the form of an energy decomposition is also available, extending the parallel scalability to hundreds of thousands of processors [2,13]. Numerous spatial discretizations are available, as well as a variety of angular quadrature sets and boundary condition options. Advanced Krylov subspace solvers from several Trilinos [14] packages can be used to solve both the fixed-source and k-eigenvalue formulations of the transport equation. Further details of the parallel algorithms and solvers can be found in Refs. [2] and [13].

3.1.1. Solution transfer from smeared Cartesian power to resolved unstructured meshes

Denovo computes a flux (power) distribution on a Cartesian mesh that will frequently be significantly coarser than the meshes used for other components of a multiphysics simulation. Directly mapping coarse Denovo powers onto a fine unstructured mesh may result in a "staircase" effect by which the coarse power distribution produces highly nonphysical asymmetries in the temperature and mechanics solutions. To counteract such behavior, a smoothing operation has been introduced in the form of a polynomial expansion of the power within each fuel pin. In particular, we make use of the Zernike polynomials in the $x - y$ plane and the Legendre polynomials in z . The Zernike polynomials form a complete set of orthogonal functions on the unit disk and are denoted $Z_n^m(x, y)$, where n is the degree and m is the order of the polynomial. The Legendre polynomials form a complete set of orthogonal polynomials on a line segment and are denoted $P_\ell(z)$, where ℓ is the degree of the polynomial. Combining these polynomials gives a complete expansion for a function on a cylinder, that is,

$$F(x, y, z) = \sum_{\ell=0}^{L} \sum_{n=0}^{N} \sum_{m=-n}^{n} C_{n,m}^{\ell} Z_n^m(x, y) P_\ell(z), \tag{1}$$



(a) Coarse-Mesh Denovo Power (b) Zernike-Smoothed Power from Coarse-Mesh

Figure 1. Normalized power distribution before and after smoothing from a coarse-mesh Denovo calculation.

where $C^{\ell}_{n,m}$ is an expansion coefficient, L is the prescribed degree of the Legendre expansion, and N is the degree of the Zernike expansion. Evaluating these coefficients is accomplished by exploiting the orthogonality of the polynomials, producing the expression

$$C^{\ell}_{n,m} = \frac{(n+1)(2\ell+1)}{(1+\delta_{m,0})\pi} \iint_{x^2+y^2\leq 1} dx\, dy \int_{-1}^{1} dz\, F(x,y,z) Z^m_n(x,y) P_\ell(z), \tag{2}$$

where $\delta_{n,m}$ is the Kronecker delta function. Once these coefficients have been evaluated for a particular flux or power distribution on the Denovo mesh (a separate set of coefficients will be computed for each pin), Eq. (1) can be used to map the distribution onto the desired locations on an unstructure mesh. The effect of this polynomial smoothing on a two-dimensional, 3×3 "mini-assembly" consisting of eight UO_2 fuel rods surrounding a B_4C control rod is shown in Fig. 1. Fig. 1a shows the result of a Denovo calculation on a coarse mesh (8×8 per pincell), and Fig. 1b is the result of mapping this coarse mesh solution onto a much finer mesh (128×128 per pincell) using the polynomial smoother. Previous studies have demonstrated that such a process can not only produce a smoother solution but also increase the numerical accuracy [15].

3.2. Thermo-Mechanics Using AMPFuel

AMPFuel is a framework for solving coupled multiphysics problems for the evaluation of nuclear fuels. Early work has primarily focused on the development of strategies for solving coupled heat transfer and structural mechanics problems on nuclear fuel pellets and cladding.

In normal operation, for nuclear fuel assemblies held together with a grid spacer (as opposed to wire wraps), the fuel pins are physically connected to the assembly structure only at grid-spacer springs and dimples, imposing very little stress on the fuel pin. The fuel pins are set just above the bottom assembly end fitting, with no weld to enforce contact, and have a "shoulder-height" separation from the bottom of the top assembly end fitting. Therefore, the fuel pins are isolated mechanically from one another and relatively isolated from the assembly structure.

Because there is negligible heat loss through the grid spacer, the fuel pins are thermally coupled to other fuel pins only through conduction and convection in the coolant. Because the flow is predominantly in the axial direction, there is relatively little heat transfer between "sub-channels" (regions between fuel pins) except near the grid-spacers and mixing vanes. Therefore, for this initial assembly simulation, the thermo-mechanics of the nuclear fuel pins are numerically approximated to be thermally and mechanically isolated.

3.2.1. Heat transfer within the fuel, gap, and cladding

In this study, the quasi-static conjugate heat transfer is modeled using nonlinear thermal conduction within each volume (fuel region and clad of each pin) and a Robin boundary condition to account for the transfer of heat between volumes (fuel-to-fuel and fuel-to-clad) and from the cladding to the coolant, as described by the equations

$$-\nabla \cdot [k\nabla T] = \rho Q_m \quad \text{in } \Omega, \tag{3}$$

$$\text{and } -k\nabla T + hT = hT' \quad \text{on } \partial\Omega. \tag{4}$$

In these equations T is the temperature, k is the thermal conductivity, ρ is the density, Q_m is the specific power, h is the heat transfer coefficient of the gap or coolant, and T' is a projected map of the temperature of the surface of a neighboring volume ($\partial\Omega'$) onto the surface of the current volume ($\partial\Omega$). The coolant, like the gap, is not modeled as an explicit volume. The thermal conduction component of AMPFuel uses a standard continuous finite-element discretization of each volume. Specific power $\left(\frac{W}{kg(\mathrm{UO_2})}\right)$ rather than power density $\left(\frac{W}{m^3}\right)$ is used as an input in AMPFuel because specific power is based on the initial geometry and can therefore be more easily mapped onto the AMPFuel mesh (even when the mesh is moving).

3.2.2. Mechanical deformation of the fuel and cladding

The mechanics component of AMPFuel approximates the governing equation that describes a body, Ω, undergoing an infinitesimal strain and displacement. At static mechanical equilibrium, the body satisfies the condition

$$\nabla\, \sigma + \rho b = 0 \quad \text{in} \quad \Omega. \tag{5}$$

Here, σ, ρ, and b are the symmetric Cauchy stress tensor, density, and body forces, respectively, and σ is a function of the displacements, u; the precise form of this function depends on the material used in the body. This is solved with a standard finite-element methodology.

The mechanics material models provide an estimate of stress (σ_{ij}) at each Gauss point subjected to some external strain (ϵ_{ij}), based on the deformation mechanisms (e.g., thermal expansion, swelling, densification) causing the displacement of the fuel pellet and the clad. These material models also return the constitutive matrix (C_{ijkl}) at each Gauss point, which relates the strain with the stress:

$$\sigma_{ij} = C_{ijkl}\epsilon_{kl}. \tag{6}$$

Because this problem is based on fresh fuel, small strain conditions are assumed and the densification and swelling are included, but negligible. Thus the total strain (ϵ_{ij}^{tot}) can be written in

additive form:

$$\epsilon_{ij}^{tot} = \epsilon_{ij}^{e} + \epsilon_{ij}^{p} + \epsilon_{ij}^{th} + \epsilon_{ij}^{sw} + \epsilon_{ij}^{de} + \epsilon_{ij}^{c} + \epsilon_{ij}^{re}, \tag{7}$$

where ϵ_{ij}^{e} is the elastic strain; ϵ_{ij}^{p} is the plastic strain; ϵ_{ij}^{th} is the strain due to thermal expansion; ϵ_{ij}^{sw} is the swelling strain; ϵ_{ij}^{de} is the densification strain; ϵ_{ij}^{c} denotes the strain due to temperature, irradiation and stress-induced creep; and ϵ_{ij}^{re} signifies the relocation strain. All strains are computed implicitly, and most properties are a function of temperature.

3.2.3. AMPFuel solution strategy

For each fuel pin, the thermo-mechanics problem is solved by first solving the thermal problem in a parallel, fully coupled approach using a Jacobian-Free Newton-Krylov solver (using PETSc for the inexact Newton iteration and a flexible GMRES solver for the solution of linear systems). The thermal preconditioner (using Trilinos for algebraic multi-grid) for the heat-transfer block is a parallel block-Jacobi (additive Schwarz) preconditioner in which each block involves inverting a thermal operator over a single volume (fuel region or cladding). The block Jacobi approach is a sufficiently strong preconditioner for this problem and allows all blocks to be inverted asynchronously, so that the time spent in preconditioning does not increase with the number of fuel regions. The preconditioner (also using Trilinos algebraic multi-grid solvers) for the mechanics simulation is a physics-based extension of a parallel block-Gauss Seidel (multiplicative Schwarz) algorithm that accounts for the upward movement of fuel regions due to the axial strain of the regions below. The clad mechanics preconditioner (also using a physics-based approach) operates asynchronously with the preconditioner for the fuel regions.

AMPFuel uses a geometry-aware parallel spatial decomposition that places different fuel pins as well as different individual volumes (fuel regions and cladding) within a fuel pin on disjoint sets of processors, thereby minimizing extraneous communication as the number of volumes is increased and ensuring that no communication is required during the computationally expensive block-diagonal preconditioning portion of the heat transfer solve. Although the current mechanics preconditioning approach does require communication between volumes, the flexibility built into AMPFuel will allow rapid experimentation with various forms of preconditioning. Even when the pins are modeled as physically coupled in the future (through multidimensional flow modeling or fuel assembly distortion), we expect that the current approach will prove to be a highly efficient and scalable preconditioning strategy.

4. PWR ASSEMBLY SIMULATION

The development of an integrated neutronics and nuclear fuel performance capability for assembly-level simulations is relevant to many reactor and fuel concepts. In this work, the capability is demonstrated on a full pressurized water reactor (PWR) assembly using Denovo to provide a power distribution to the thermo-mechanics of AMPFuel.

4.1. Problem Description

The geometry of the problem consists of a 17×17 array of pin cells with 264 locations containing fuel pins, 1 instrumentation tube, and the remaining 24 containing guide tubes, as shown in Fig. 2.

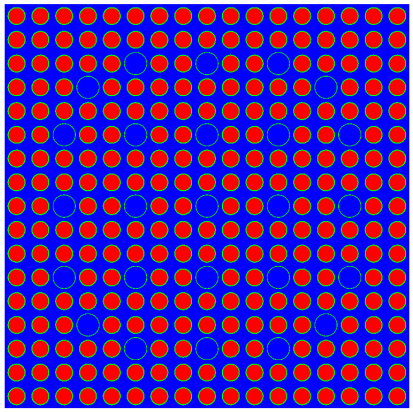
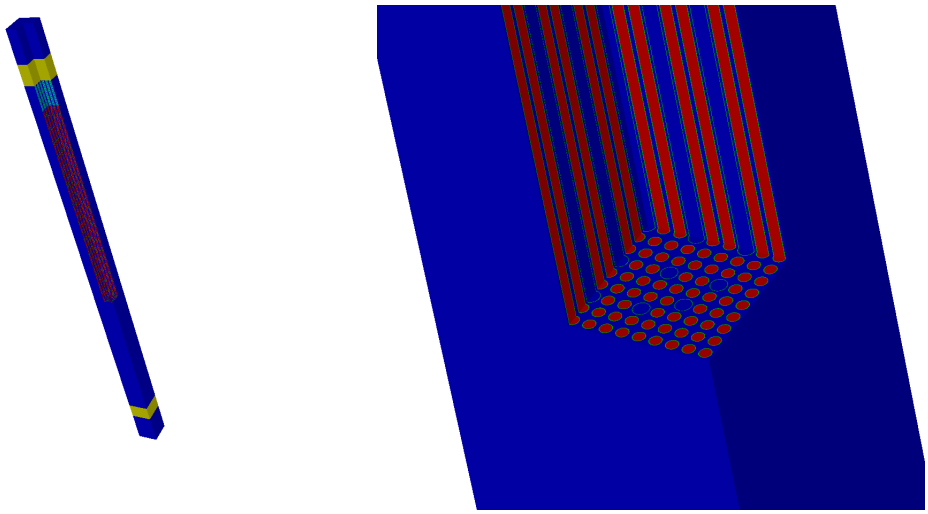


Figure 2. Planar slice of the materials in the assembly.



(a) Full Height (b) Detail

Figure 3. Denovo geometry with materials. Blue is moderator, red is fuel, green is clad, cyan is the plenum, and yellow is the smeared end fitting. The gap between the pellet and clad is modeled but not visible due to the small distance.

A detailed description of the dimensions of the assembly is found in Table I. The top and bottom assembly end fittings are included in the Denovo model but are neglected in the AMPFuel thermo-mechanics calculation. The full layout of (the Denovo model of) the assembly is illustrated in Fig. 3.

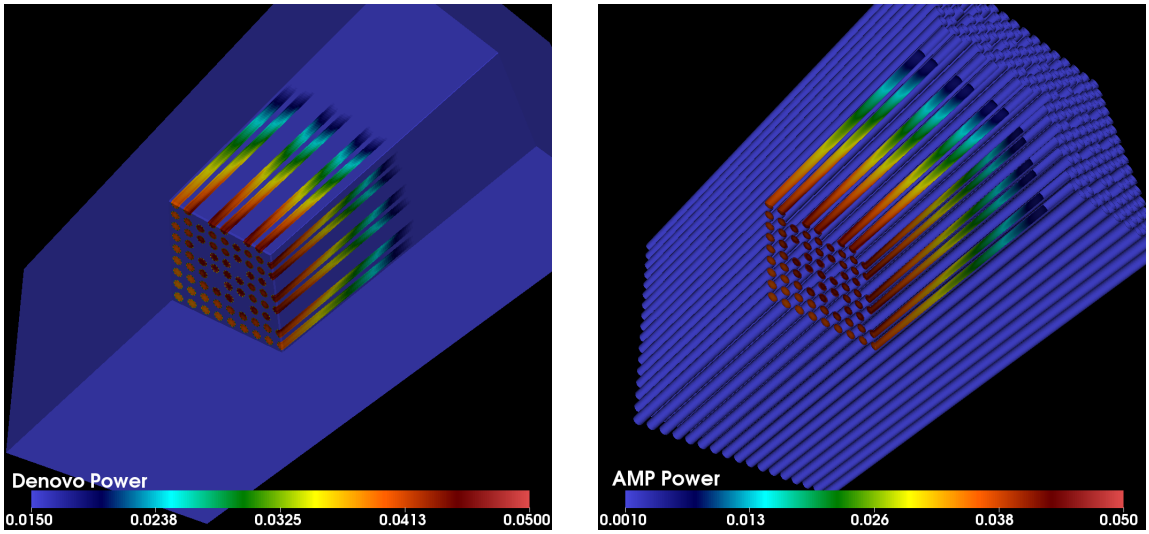
Table I. Dimensions of the PWR assembly

Region	Distance (cm)	Distance (in)
Fuel pellet radius	0.4095	0.1612
Clad inner radius	0.418	0.1646
Clad outer radius	0.475	0.187
Guide tube inner radius	0.5715	0.225
Guide tube outer radius	0.6120	0.2409
Pincell width (pitch)	1.26	0.4961
Active fuel height (360 pellets)	365.76	144
Upper plenum height	28.448	11.2
Total fuel pin height	396.24	156.0
Bottom reflector height	32.512	12.8
Bottom end fitting height	15.240	6.0
Top end fitting height	15.240	6.0
Upper reflector height	32.512	12.8
Total model height	491.744	193.6

The fuel is composed of 5% enriched UO_2 at 10.4 g/cc; the clad, guide tube, and top/bottom end fitting are composed of Zircaloy-4 at 6.55 g/cc; the gap is filled with pure helium at 0.0002 g/cc; and the coolant is H_2O with 1000 ppm natural boron.

The AMPFuel material coefficients for heat transfer (k, ρ, h) from Eqs. (3) and (4) are based on FRAPCON [6]. The thermal conductivity of Zircaloy-4 and density of UO_2 (assuming the temperature is below the melting point) depend only on temperature; the thermal conductivity of UO_2 is a function of temperature, burnup, porosity, and gadolinia content. The heat transfer coefficient (h) accounts for conduction, convection, radiation, and discontinuities at materials interfaces, which is a function of contact pressure. The fuel-region-to-fuel-region ($f2f$) and clad-to-water ($c2w$) heat transfer coefficients take the fixed values $h[f2f] = 10$ and $h[c2w] = 11,000$, respectively. The model that provides the fuel-to-clad ($f2c$) heat transfer coefficient is nonlinearly dependent upon the gap thickness, the temperature of the inner surface of the clad, and the temperature of the outer surface of the fuel. Futher details of these material models can be found in Ref. [6].

The cold zero power condition (which is the reference state for thermal expansion) consists of all materials at a temperature of 500 K; hot zero power (the state for which the thermo-mechanics solution is desired) consists of a coolant temperature of 555 K and an operating power level of 20,000 W/kg(UO_2) or 0.02 GW per metric tonne of UO_2. Only fresh fuel is considered for this



(a) Denovo Power Distribution (b) AMPFuel Power Distribution

Figure 4. Power $\left(\frac{GW}{MT(UO_2)}\right)$ distribution showing the 3D variation.

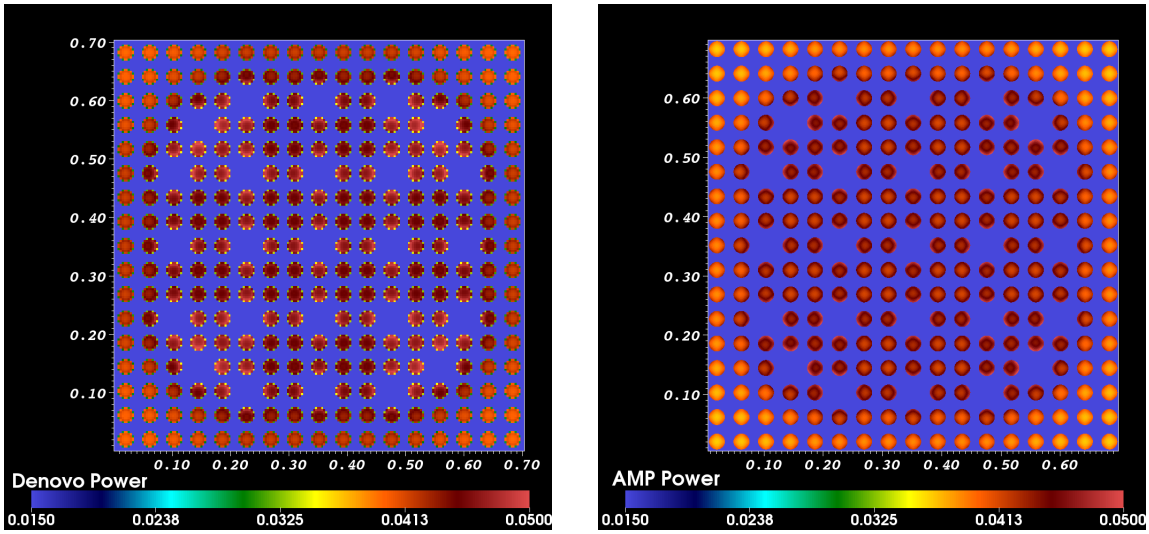
calculation.

The Denovo calculation contained slightly over 10 million cells, 128 angles and 44 energy groups for a total of around 56 billion degrees of freedom. The AMPFuel model contained approximately 12 million spatial elements per fuel pin and around 208,000 elements per clad. With 264 fuel pins, the total element count for the AMPFuel simulation was approximately 3.3 billion. The finite element models used eight Gauss points per element, and the thermo-mechanics calculation entails four unknowns per point (one for thermal and three for mechanics), bringing the total degrees of freedom to over 105 billion for the AMPFuel portion of the calculation.

4.2. Results

The full PWR assembly simulation was completed using 40,000 cores of the Jaguar (Cray XK6) supercomputer at the Oak Ridge Leadership Computing Facility (OLCF). The entire calculation required 10 hours of wall clock time. Of this, around 45 minutes was spent in the Denovo radiation transport calculation, 15 minutes in the heat transfer calculation, and the remaining 9 hours in the mechanics portion of the calculation (distributed between 10 loading steps).

The power distribution is shown in Fig. 4a as computed by Denovo and in Fig. 4b after the mapping onto the AMPFuel mesh. In this plot (and subsequent 3D plots), the geometry has been compressed by a factor of 10 in the axial direction to more clearly show the 3D variation. Fig. 5 shows an axial slice of the power distribution on each mesh 5 feet above the bottom of the active fuel region. The coarseness of the radiation transport calculation is evident in the Denovo computed power (as low power regions around the edge of the fuel pins as a result of homogenization), but the polynomial smoothing operation described in Section 3.1.1 eliminates



(a) Denovo Power Distribution (b) AMPFuel Power Distribution

Figure 5. Power $\left(\frac{GW}{MT(UO_2)}\right)$ distribution within the assembly 5 feet above the bottom of the active fuel region.

such artifacts on the AMPFuel mesh. In addition to the inter-pin effects of fuel pins neighboring guide tubes having a higher overall power than pins near the periphery of the assembly, within-pin variations can also be seen as gradients across the pins near guide tubes. The reason that the magnitude of the power on the Denovo mesh is slightly different than the magnitude on the AMPFuel mesh is due to the polynomial smoothing of the power in the axial direction. Although this process can result in a slightly different power magnitude at a particular axial level, the average value of the power over a fuel region is strictly conserved.

The temperature distribution resulting from this power distribution is shown in Fig. 6 with slices at two different axial levels also shown in Fig. 7. Although azimuthal variations in the temperature are not readily evident in these figures, such variations do exist as will be more evident from the results of the mechanics calculation. Because no temperature feedback to the cross sections is included, this simulation represents a one-way coupling between the radiation transport and heat transfer components.

Fig. 8 shows the 3D distribution of the axial component of the mechanical strain. An axial slice 6 feet above the bottom of the active fuel is provided in Fig. 9. The maximum axial strain at the top of the highest temperature fuel pins is approximately 1.3 cm (0.35% of the total fuel height), a value consistent with expectations for this problem. This plot clearly demonstrates the effects of the asymmetries in the power distribution; in many of the pins the displacement is larger on one side of the pin that the other due to a higher temperature in the vicinity of the guide tubes. This variation in strain within a pin causes a bending, or bowing, of the pin away from the guide tube locations. Although this effect is physically justified, it is unknown to what extent approximations made in the AMPFuel model (smearing of fuel regions, boundary conditions between pellets, etc.) have impacted this result.

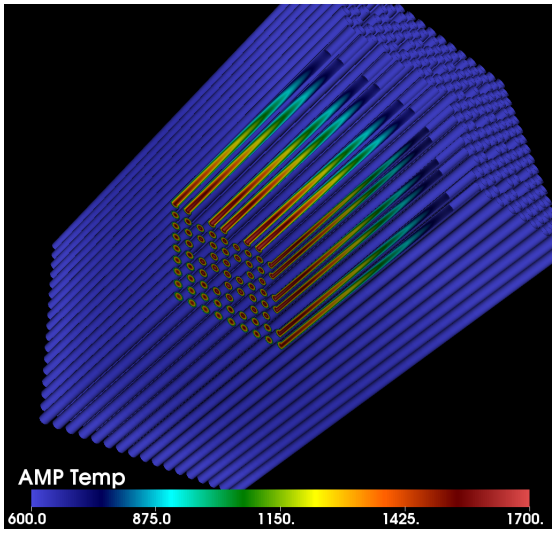
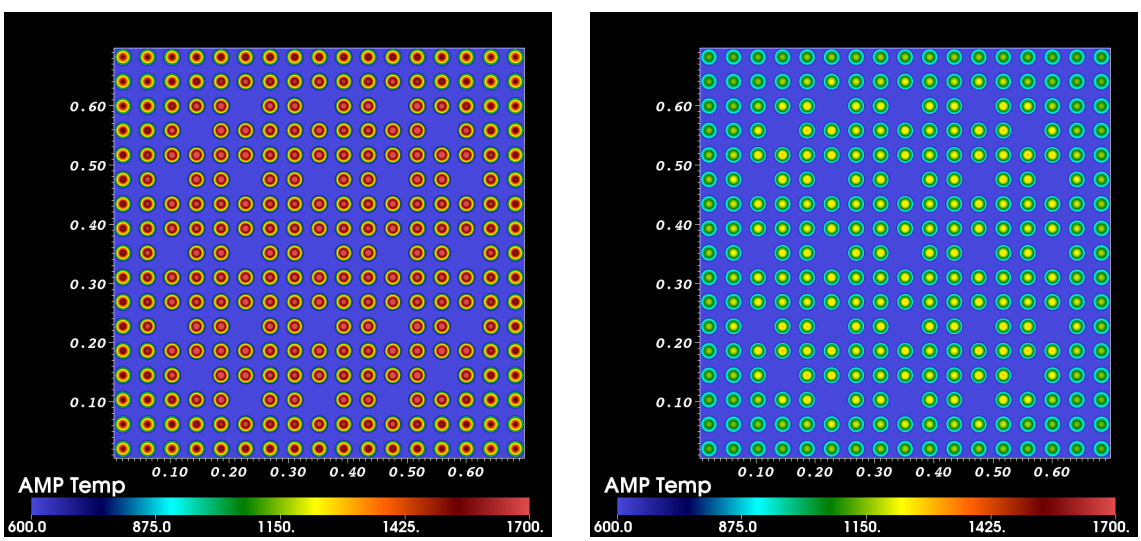


Figure 6. Temperature (Kelvin) distribution within the assembly.



(a) 5 feet (b) 9 feet

Figure 7. Temperature (Kelvin) distribution within the assembly for various heights above the bottom of the active fuel region.

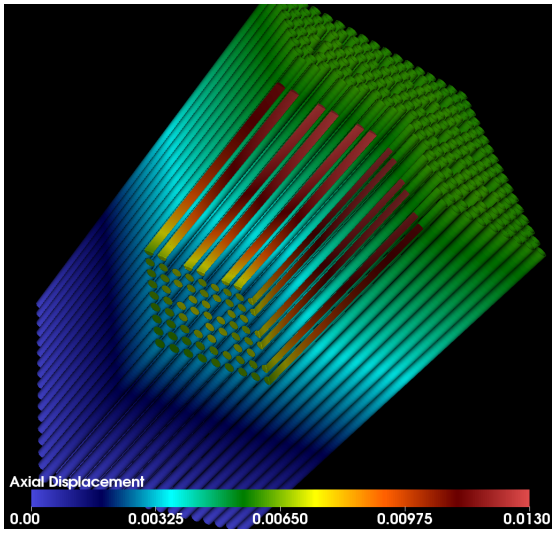


Figure 8. Axial displacement (meters) on the initial AMPFuel mesh.

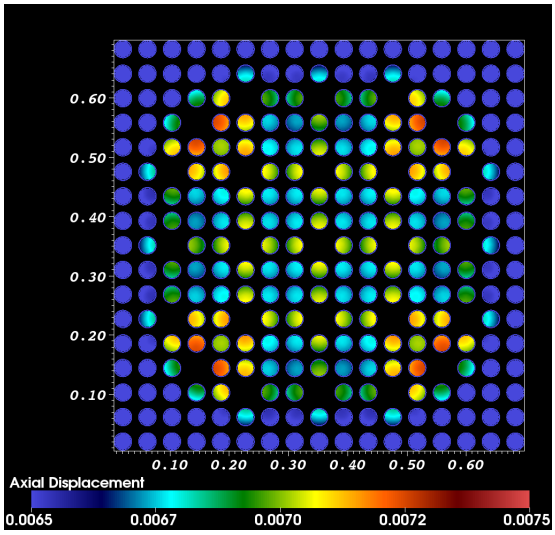


Figure 9. Axial displacement (meters) at the assembly mid-plane.

5. CONCLUSIONS

The capability of the Advanced Multi-Physics (AMP) Nuclear Fuel Performance code (AMPFuel) has been extended to allow the use of detailed power distributions computed by solving the Boltzmann radiation transport equation using the SCALE/Denovo code for use in nuclear fuel thermo-mechanics simulations. This capability allows AMPFuel to explore assembly-level effects due to asymmetries in the power distribution that would be neglected in traditional fuel performance codes. A new scheme for transferring a power distribution from a coarse Cartesian Denovo mesh to an unstructred AMPFuel mesh has been developed based on a cylindrical polynomial expansion. This method significantly reduces the impact of discontinuities in the computed power distribution and allows for a much coarser radiation transport problem to be used with relatively little loss in accuracy. The reduction in computational cost is likely to be significant compared to solving the radiation transport equation on the same mesh as the thermo-mechanics calculation. Both the heat transfer and mechanics portions of the AMPFuel thermo-mechanics calculations use a fully coupled Jacobian-Free Newton Krylov solver. New physics-based preconditioning strategies were developed for solving each of these components using a geometry-aware decomposition to minimize communication between domains combined with algebraic multi-grid solvers operating on individual domains.

This new capability was used to model a full 17×17 PWR fuel assembly. The model contained over 160 billion degrees of freedom (radiation transport and thermo-mechanics combined) using over 40,000 cores on the Jaguar supercomputer. The results of this calculation clearly illustrate axial, radial, and azimuthal variations in the computed quantities, including an artificial rod bowing effect in the fuel pins closest to the assembly guide tubes. Such effects cannot be accounted for in traditional 1D fuel performance codes.

Because it is based on 3D radiation transport, rather than calibrated models, the approach described in this paper is extensible to incorporate additional assembly features, such as grid spacers and part-length rods, that create additional heterogeneity in the flux and power distribution. As this problem is extended to include more assembly features, different reactor and fuel types, reduced approximations in the thermo-mechanics, coupling of the physics, and isotopic depletion, we will be able to develop a more comprehensive understanding of the physics and models at a resolution that was previously unattainable and develop approximations that can improve the accuracy of reactor simulations (which use lower-fidelity fuel models) and single-pin fuel simulations (which use lower fidelity neutronics models).

6. ACKNOWLEDGMENTS

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limited distribution, through the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory as BCC-001.

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