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Modeling and Analysis of a Micro-reactor using the DireWolf Code Suite

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ABSTRACT

Micro-reactors, specifically those cooled with heat pipes, are tightly coupled systems which require multi-physics tools to accurately model both steady-state and transient behavior. Idaho National Laboratory's DireWolf code suite is tailor-built to model heat-pipe reactors. DireWolf's ability to model the coupled steady-state behavior of a heat-pipe micro-reactor are demonstrated herein. The DireWolf modeling methodology is described, and a micro-reactor design is introduced. A full-core model is developed, and the steady-state coupled neutronics-thermal analysis is performed. Results of the analyses, including flux, power, and temperature distributions are presented and discussed.

KEYWORDS: Direwolf, micro-reactor, multi-physics analyses

1. INTRODUCTION

A micro-reactor, specifically one cooled with heat pipes, is intrinsically a tightly coupled system. Traditional loosely coupled, single-physics models are not sufficient to accurately predict the system behavior; a true multi-physics approach is required. To this extent, Idaho National Laboratory has taken existing MOOSE [1]-based, finite-element computational tools and packaged them into a code suite specifically tailored for heat pipe reactors, called DireWolf [2]. DireWolf includes Griffin [3] for neutronics modeling, Bison [4] for thermal-mechanical analysis, and Sockeye [5] for heat pipe modeling.

This paper will demonstrate DireWolf's ability to model the multi-physics behavior of a heat-pipe micro-reactor. A brief description of each individual software is presented along with DireWolf's coupling methodology. The micro-reactor core design will be introduced, and a Direwolf model is presented. This model includes a mesh for both neutronics and thermal analysis, as well as a cross-section library required for the neutronics evaluation. A steady-state DireWolf simulation is performed and results, including flux, power, and temperature distributions, are presented. The analysis methodology will be briefly discussed (details will be deferred to the references). Finally, a refined model is presented to be used in subsequent transient scenarios.

2. MODEL DEVELOPMENT

Analyzing a transient event in DireWolf first requires the development of a suitable model which provides adequate level of detail but can be solved in a reasonable amount of time. As DireWolf is a finite element code, the primary component of the model is the mesh. In addition to the mesh, it is also necessary to provide nuclear data in the form of a cross-section library. This section will describe the micro-reactor core design and the model development.

2.1. Core Design

The micro-reactor design presented here generates 6 MWth at full power. Heat is removed from the reactor core and transferred to the power conversion system (PCS) using heat pipes. A heat pipe is, very simply, a sealed tube with a working fluid and a "wick" inside. There is a hot side and a cold side. On the hot side the pipe is heated such that the working fluid vaporizes; this vapor travels to the cold side where it releases latent heat. The fluid then travels back to the hot side via the wick through capillary action. In this micro-reactor, the core serves as the evaporator and the PCS serves as the condenser and the heat pipes passively remove heat from the core to the PCS. In this reactor, the PCS is an open-air Brayton cycle. The PCS can maintain the heat pipes at an average temperature of 800 °C and the average fuel temperature is 900 °C.

The reactor is fueled with 19.75 wt% enriched High-Assay Low-Enriched Uranium (HALEU) in the form of Tri-structural Isotropic (TRISO) fuel compacts. The TRISO particles are compliant with AGR-2 licensed topical report [6]. The core lifetime is approximately 3 effective full power years, by which time this reactor achieves a core-average burnup of ~20 GWd/MTU.

The reactor core is comprised of a graphite monolith with channels for fuel, heat pipes, and discrete moderator rods. The monolith is surrounded by a radial reflector which is embedded with 12 control drums. The control drums are made of an absorbing (B_4C) segment and reflector material. The drums can rotate the B_4C segment toward or away from the core, hence providing a reactivity control mechanism. With all drums fully rotated inward, the core is in a subcritical configuration. The core also contains locations for transportation/shutdown rods which provide a redundant means for shutdown and additional negative reactivity to hold the core subcritical during a hypothetical criticality accident scenario (e.g., flooding). The radial reflector is surrounded by an inner cannister, then a gamma shield, neutron shield, and an outer canister. Figure 1 shows the radial core configuration, Figure 2 provides a more detailed view of the core. Table I provides a range of core dimensions. The core dimensions were iterated on, within the ranges provided in Table I, until a core was achieved which met design objectives including size and mass limits, power output, and core lifetime.

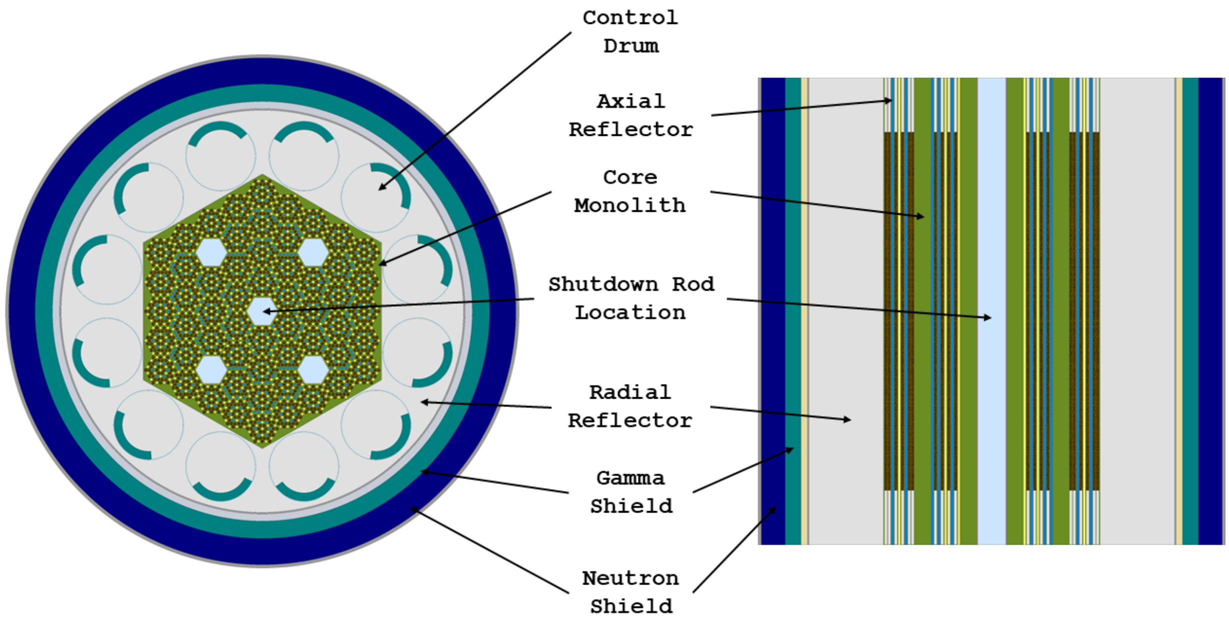


Figure 1. Micro-reactor Core Layout

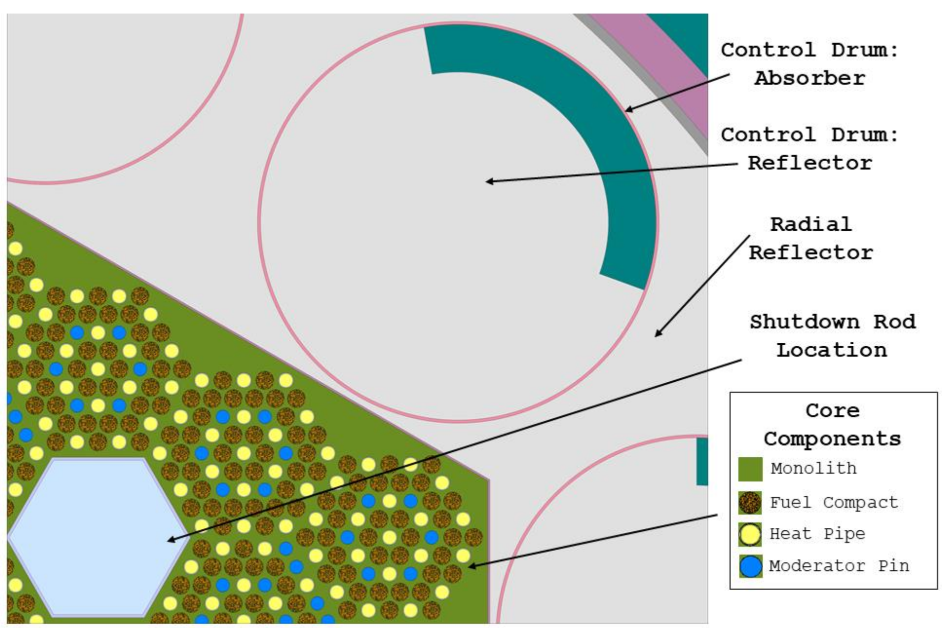


Figure 2. Detailed View of Micro-reactor Core

Table I. Micro-reactor Core Dimensions

Parameter	Range of Dimension, cm
Fuel Compact Diameter	1.6 – 2.0
Active Fuel Height	148.0 – 182.0
Heat Pipe Outer Diameter	1.30 – 1.60
Moderator Rod Diameter	1.20 – 1.50
Pin Pitch	1.90 – 2.30
Core Flat-to-Flat	122.0 – 150.0
Control Drum Outer Diameter	35.0 – 44.0
Control Drum B_4C Thickness	4.0 – 5.5
Radial Reflector Outer Diameter	200.0 – 250.0
Gamma Shield Thickness	13.0 – 17.0
Neutron Shield Thickness	9.0 – 11.0

2.2. Mesh Generation

In the mesh generation stage, there is a decision that needs to be made regarding the level of detail in the mesh. The core can be modeled heterogeneously, explicitly resolving each pin, or it can be homogenized. The latter will yield much faster run times; however, it will not provide detailed pin-wise information, which is often required for safety analyses. An additional consideration is whether symmetry can be applied to reduce the extent of the model, which will also reduce computation time. Additionally, a different mesh may be used for different physics modeling. For this analysis a 3D assembly-homogenized mesh is selected for neutronics modeling in Griffin (Figure 3), and a 3D pin-wise mesh is used for thermal modeling in Bison (Figure 4). Both meshes are generated using Cubit along with a few custom-built helper scripts [7,8].

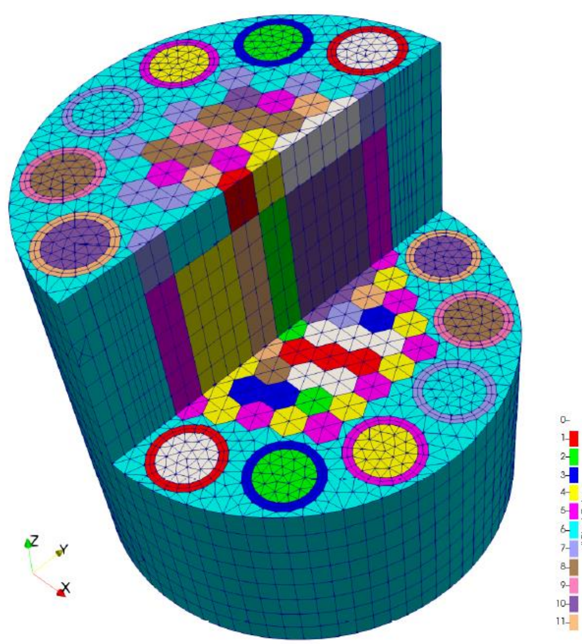


Figure 3. 3D Full-core Assembly-homogenized Neutronics Mesh for Griffin

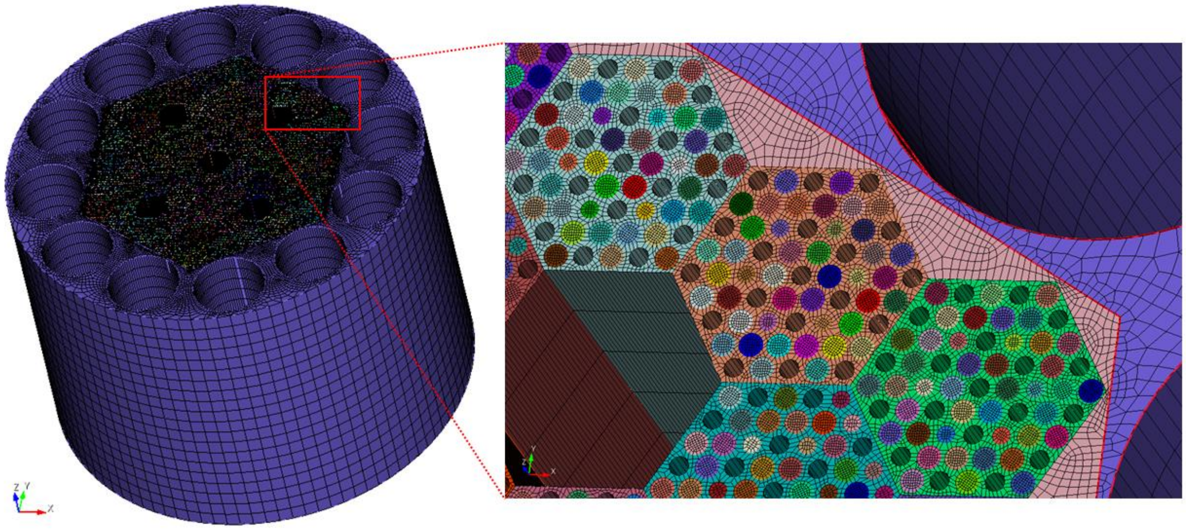


Figure 4. 3D Full-core Pin-wise Thermal Mesh for Bison

2.3. Cross-Section Library

Energy-group dependent neutron cross-section data are required to perform the neutronics calculation with Griffin. There are numerous variables to consider when developing a cross-section library for a coupled multi-physics calculation, especially when modeling transient events. Variables which impact nuclear data are individual component temperatures, control drum positions, fuel burnup, and component irradiation. It is easy to see that the cross-section library can quickly balloon in size. It is necessary to evaluate which parameters are of highest importance and what level of resolution is required in the data structure to obtain reliable results. For this analysis a cross-section library with 108 grid points is selected: 4 control drum positions (0,60,120, and 180 degrees), 3 fuel temperatures (27, 500, and 1300 °C), 3 reflector temperatures (27, 500, and 1100 °C), and 3 moderator/monolith/heat pipe temperatures (27, 527, and 827 °C). An 11-group energy structure is used, Table II, which was derived from the SCALE 44-group library. Further study is required to evaluate the quality of this structure, especially with respect to kinetics parameters.

Table II. Cross-section Library Energy Group Structure.

Group	Lower Energy (MeV)	Upper Energy (MeV)
1	2.35E+00	2.00E+07
2	9.00E-01	2.35E+00
3	1.00E-01	9.00E-01
4	5.50E-04	1.00E-01
5	1.00E-05	5.50E-04
6	3.00E-06	1.00E-05
7	6.25E-07	3.00E-06
8	3.50E-07	6.25E-07
9	2.00E-07	3.50E-07
10	1.00E-07	2.00E-07
11	1.00E-11	1.00E-07

The Serpent Monte Carlo code [9] has a built-in capability to calculate multigroup cross sections. A 3D full-core Serpent model is developed and used to generate the necessary multi-group cross sections, including the homogenized fuel-moderator-heat pipe regions. The Serpent calculation uses the ENDF/B-VII.1 continuous-energy nuclear data [10].

3. ANALYSIS & RESULTS

3.1. Methodology

DireWolf is comprised of three distinct tools for modeling the neutronics, thermomechanics, and heat pipe behavior. Each of these codes are coupled through the MOOSE framework. Details of each physics model are described, as well as the coupling approach. Note that, as the analyses presented in this paper only focus on the coupling of thermal and neutronics, only those physics models are discussed below.

3.1.1. Neutronics

Griffin is the particle transport tool built specifically for multi-physics coupling. Griffin solves the multigroup transport equations using finite element methods to calculate steady-state, transient, and eigenvalue problems. Griffin includes many different algorithms for solving the discretized transport equations algorithms. For this problem, we are interested in calculating the eigenvalue (k) and the continuous finite element scheme (CFEM-Diffusion) is used to do so. Griffin also includes an algorithm to determine the "critical drum position," i.e., the rotation angle of the drums such that k equals 1.0. Griffin will obtain 3D full core fluxes and power distributions in the mesh shown in Figure 2. Vacuum boundary conditions are imposed at all boundaries. To correct for the error associated with using a coarse, assembly-homogenized mesh, the Super Homogenization (SPH) method, spatially restricted to the fuel regions, is used [11, 12]. No void treatment is performed as the spatial restriction of the SPH procedure to all the fuel regions can preserve the integrated leakage out of these regions [12]. In fact, the Griffin calculation can match the Serpent eigenvalues and assembly powers at state points. Between state points, the Griffin control drum rotation treatment [13] is applied to avoid cusping effects and match the Serpent critical angle within a few degrees.

3.1.2. Thermal

Bison calculates the thermal and mechanical response, by solving the fully coupled thermomechanical and species diffusion equations in finite element form. Bison is most often used for fuel performance calculations, but in this application, it is being used to evaluate full core temperature distribution. Calculations were performed with and without mechanics modeling, i.e., thermal expansion and displacement, and it was shown that mechanical feedback did not have significant impact on reactivity or power distribution. Mechanics modeling requires the use of a fine mesh, therefore significant computational resources. As the impact of these calculations are small, mechanics are not modeled. We are only modeling the heat conduction from the fuel rods through all core components to calculate the temperature distribution. Pin-power reconstruction is not currently used, the assembly-averaged power is smeared over all pins within an assembly – this will cause an underestimation of peak system temperature. Bison will calculate a full 3D temperature distribution for the mesh shown in Figure 3. As the heat pipe behavior is not explicitly calculated in this analysis, heat pipes are represented as a convection boundary condition. The periphery of the radial reflector is assigned a heat flux, representing convective heat loss to the outside.

3.1.3. Heat Pipes

Sockeye is the tool used to explicitly model heat pipe performance. It is designed primarily for liquid-metal filled cylindrical tube type, with an internal annular wick, heat pipes in a microreactor environment.

Sockeye utilizes 1D approximations to model the two-phase flow of the working fluid during normal operations; it additionally calculates 2D conduction through the heat pipe walls. It will explicitly calculate the thermodynamic state, velocities, and volume fractions of each phase, and temperature and heat flux at the heat pipe walls. For steady-state calculations, heat pipes are modeled as a convective heat flux boundary condition with a temperature dependent heat transfer coefficient and constant bulk temperature. For transients, particularly heat pipe related ones, Sockeye will be used to explicitly model heat pipes.

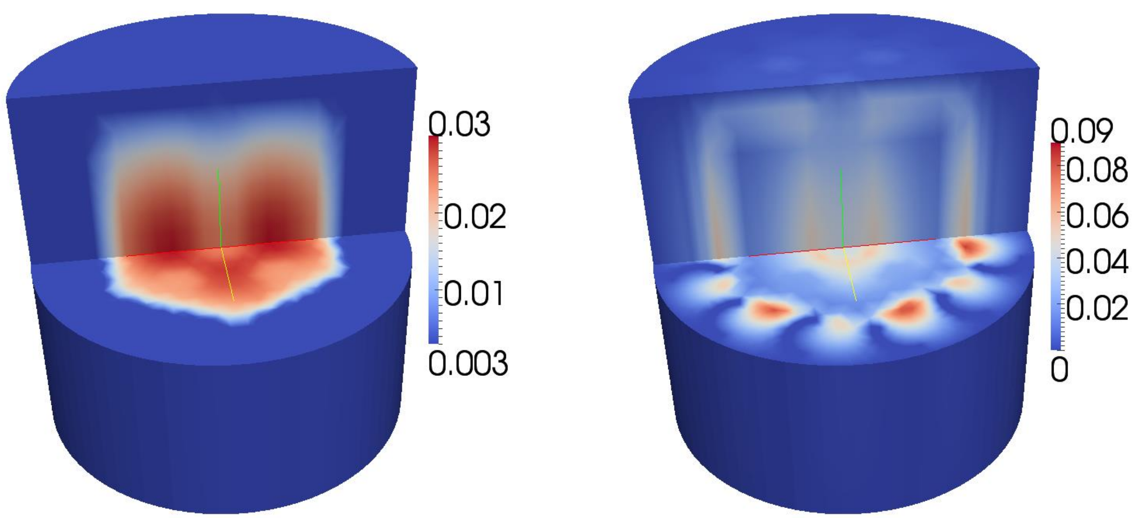
3.1.4. Code Coupling

For this analysis, we are considering only a coupled thermal and neutronic simulation. DireWolf will first use Griffin to calculate the neutronics solution, i.e., flux and power distribution, using the user-input initial temperature conditions. The power generated in each fuel pin is transferred to Bison and used to calculate the full 3D temperature distribution. The temperature distribution is fed back to Griffin, cross-sections are updated based on interpolation of the cross-section library and a subsequent neutronics solution is calculated. This process is iterated on until a converged temperature and power distributions are obtained.

3.2. Steady State Solution

The steady state solution is obtained in a two-step approach. First a stand-alone Griffin calculation is performed to determine the critical drum position. Then this critical drum position is provided as an input to the neutronic-thermal coupled calculation. The calculated critical drum position is 79.83 degrees (where 0 degrees represents all drums out).

Figure 5 shows the fast (a) and thermal (b) neutron flux distribution. The fast flux is peaked, as expected, in the fuel region. The thermal flux shows a local peak in the center of the core, due to a higher concentration of moderator pins. The thermal flux also shows local peaks in the reflector part of the control drums, where the B_4C absorber pads are located.



(a) Fast (>2.35 MeV) (b) Thermal (<0.1 eV)

Figure 5. 3D Steady-state Normalized Neutron Flux Distribution

Figure 6 presents the power density distribution, which shows a very centrally peaked distribution, as expected.

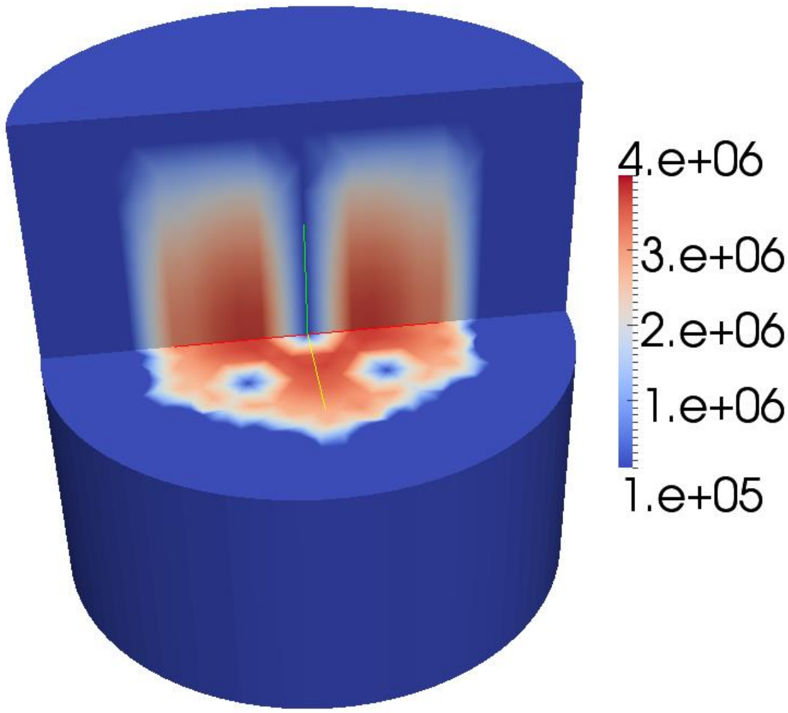


Figure 6. 3D Steady-state Power Density Distribution (W/m^3)

Figure 7 shows the temperature distribution and Table III provides the average, maximum, and minimum steady-state component temperatures. The radial temperature distribution shows some asymmetries, which are expected due to the somewhat asymmetric distribution of fuel, heat pipes, moderator rods.

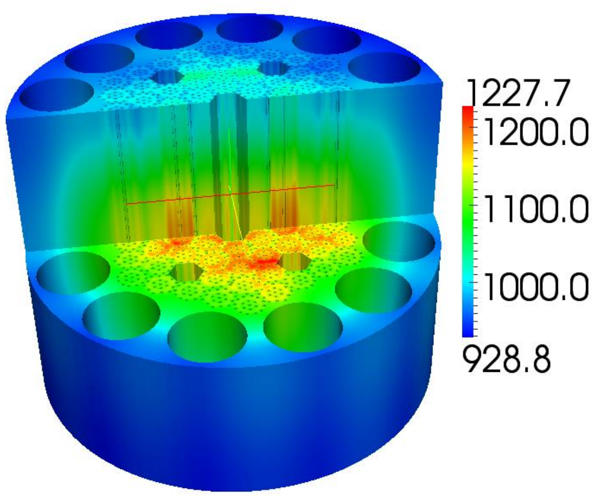


Figure 7. 3D Steady-state Temperature Distribution (K)

Table III. Steady-state Component Temperatures, °C.

Component	Average	Max	Min
Fuel	802.1	953.9	682.4
Moderator	801.7	945.7	682.4
Monolith	790.7	950.3	678.5
Reflector	706.7	847.7	656.0
Heat Pipe*	788.6	677.6	927.0

*Heat pipe surface temperature.

To understand the dynamic behavior of this reactor it is useful to look at temperature reactivity coefficients and kinetics parameters. The total temperature reactivity coefficient is calculated by running a perturbation with temperatures at ±10 °C. Kinetics parameters are automatically determined from an adjoint calculation, using the initial temperature profile. These parameters are approximate during the transient calculations, especially if the temperature profile drastically changes.

Table IV presents the temperature reactivity coefficient and kinetics parameters. The negative temperature reactivity coefficient ensures that the core reactivity will decrease with an increase in temperature. A negative temperature coefficient ensures that this core will inherently shut down in the event of a power or temperature excursion. The delayed neutron fraction and decay constant are larger than a typical light water reactor. This means that, in a reactivity insertion transient, there is a larger margin to prompt criticality and that the transient will progress slower than a typical light water reactor. The combination of the negative reactivity coefficient and the larger delayed neutron fraction and decay constant show that this reactor provides inherent protection against safety limit violations during hypothetic accident scenarios.

Table IV. Steady-state Component Temperatures, °C.

Parameter, Unit	Value
Temperature Reactivity Coefficient, pcm/°C	-4.26
Delayed Neutron Fraction (β_{eff})	0.00712
Delayed Neutron Decay Constant (λ), s^{-1}	0.472

4. CONCLUSION

This paper shows the successful use of the DireWolf code suite to model the steady-state behavior of a heat-pipe cooled micro-reactor. A detailed model of the reactor is presented, including neutronic and thermal mesh and nuclear cross-sections. DireWolf was used to determine critical drum position at beginning of life and a coupled thermal-neutronic solution was obtained. Results including fast and thermal flux, power distributions, and temperature distributions are presented.

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