A Stylized 3D PRISM-Based Nitride Fuel Configuration Benchmark Problem

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ABSTRACT

A set of 3D sodium-cooled fast reactor benchmark problems based on the General Electric Power Reactor Innovative Small Module (PRISM) for a Nitride fuel variation of the design is developed in this paper. The configurations include an uncontrolled case with all control rods withdrawn, a controlled case with all control rods inserted, and a critical case with some control rods inserted. The benchmark problems retain detailed geometric and material information of the Nitride fuel, liquid sodium coolant, duct gaps, wire spacers, Boron Carbide (B_4C) absorber, stainless steel alloy HT9 cladding for uncontrolled, critical, and controlled assembly configurations, and faithfully adheres to those design specifications that are available publicly.

Keywords: *Benchmark, Nitride Fuel, Sodium, PRISM Reactor*

1 INTRODUCTION

The Power Reactor Innovative Small Module (PRISM) [1] is a sodium cooled fast reactor (SFR) that dates back to General Electric efforts in the 1980s to develop an SFR design for commercial deployment as part of the U.S. Advanced Liquid Metal Reactor (ALMR) [2] [3] DOE program. In 2006, GE Hitachi revisited the design as part of the DOE's Global Nuclear Energy Partnership [1] as a proposed pathway to produce electricity from recycled nuclear fuel. The work in this paper is motivated by the desire to create a fast reactor core benchmark problem for evaluating particle transport methods and codes. Here, three PRISM core configurations were developed: uncontrolled (ARO), critical (SRI), and controlled (ARI). Additionally provided are heating profiles from coupled neutron-photon transport calculations using the Monte Carlo code OpenMC. Detailed core specifications and benchmark results are presented with a treatment of any design simplifications or assumptions.

The paper is organized as follows. The assumptions and considerations used for developing each benchmark problem are presented in Section 2. A detailed specification of the materials and geometry for driver fuel, blanket fuel, control and shielding assemblies as well as the full-core layout is presented in Section 3. OpenMC neutronics modelling parameters and solutions are presented in Section 4. Concluding remarks are found in Section 5.

2 ASSUMPTIONS & CONSIDERATIONS

2.1 Fuel Composition

Limited details are available about the true material composition of fuel in the driver and blanket assemblies for this Nitride configuration of the PRISM core. Isotopics used for this benchmark work are based on freshly loaded fuel compositions from references [1] and [4]. Several assumptions were made in those works, that are worth restating for completeness here. Firstly, these benchmarks model freshly loaded fuel compositions. Therefore, it is assumed that the fuel does not contain any fission products and the internal and radial blanket assemblies have the same material composition. Secondly, while many variants of the PRISM design were explored – namely low conversion ratio used nuclear fuel actinide recycle, unity conversion ratio (isobreeder), and high conversion ratio (breeder) variants [1] – for these benchmark problems we assume that minor actinide and non-fissile plutonium masses are equal, and that the ratio of non-fissile to fissile plutonium was 1:4. Thirdly, in the absence of precise minor actinide isotope fractions, we use specifications from a similar Helium-cooled fast reactor design [5]. Lastly, we assume true material densities are less than literature theoretical densities. As the exact fuel densities are unknown, a reasonable estimate of ~88% of the specified theoretical density of 13.60 g/cm^3 is used.

2.2 Containment Vessel & Secondary Systems

Rough configuration specifications for the containment vessel, heat exchanger, and secondary coolant loops are provided in literature [6]. For the purposes of this work, for which a consideration of transient analysis or thermal hydraulic coupling framework is out of scope, everything outside of the shielding assembly region is treated as vacuum. This is a valid assumption for the development of a neutron-photon transport benchmark that does not treat 1) neutronic effects of secondary systems which is negligible, or 2) an integrated neutronics thermal- fluids modelling framework for Doppler temperature feedback and varying material composition. The gas expansion module (GEM) which the PRISM design uses is modelled as a steady-state assembly.

2.3 Temperature Data

As a standalone neutronics model uncoupled from thermal fluid dependence, cell temperature for driver, blanket, reflector, shielding and GEM assemblies was estimated from [4] and [6]. There, important thermal parameters such as outlet temperatures, peak fuel temperatures for internal and radial fuel assemblies, etc. are generated for simplified and extrapolated PRISM benchmarks for transient accident scenario analysis. The temperature for each material is summarized in Table 1.

Table 1: Material Temperature Summary.

Material	Temperature (K)
Driver Fuel	968
Driver Fuel Assembly Structure Materials	791
Coolant in Driver Fuel Assembly	791
Blanket Fuel	918
Blanket Fuel Assembly Structure Materials	777
Coolant in Blanket Fuel Assembly	777
Other Materials	777

3 CORE LAYOUT

The PRISM reactor was designed to have a rated thermal power of 840 MW and an electrical output of 311 MW, with an overall thermal efficiency of 37%. The core is arranged as a series of fuel, control, reflector, and shielding hexagonal assemblies. Freshly loaded driver fuel is composed of 68.1% Uranium, 21.9% Plutonium, 4.4% Minor Actinides, and 5.6% Nitrogen with a theoretical density of $13.60\ g/cm^3$. Core parameters are summarized in Table 2. Detailed isotopic breakdown for both driver and blanket fuel is given in Table 3.

Table 2: Reactor Specification Summary.

Parameter	Value
Core Thermal Power	840 MW
Thermal Efficiency	37%
Fuel Type	Nitride
Reflector Material	HT9
Core Height	1.02 meters
Coolant	Liquid Na

3.1 Radial & Axial Layout

The core lattice consists of driver fuel assemblies, layered in rings with internal and then radial blanket fuel assemblies. Control assemblies are arranged symmetrically about the inner core, with GEM assemblies on the periphery of the fuel region. The fuel region is surrounded by two rings of reflector assemblies, and then a ring of shielding assemblies. The total number of assemblies is given in Table 3. This Nitride variant of the core, accommodating for different fuel parameters, has modified fuel assembly dimensions, and an altered core layout from the original PRISM design. Both core arrangements are depicted in Figure 1.

Table 3: Summary of Core Assemblies.

Assembly	Assembly Count	Pins-per-Assembly
Driver Fuel	114	217
Blanket Fuel	121	127
Control	12	61
Reflector	126	61
Shielding	72	7
GEM	6	N/A

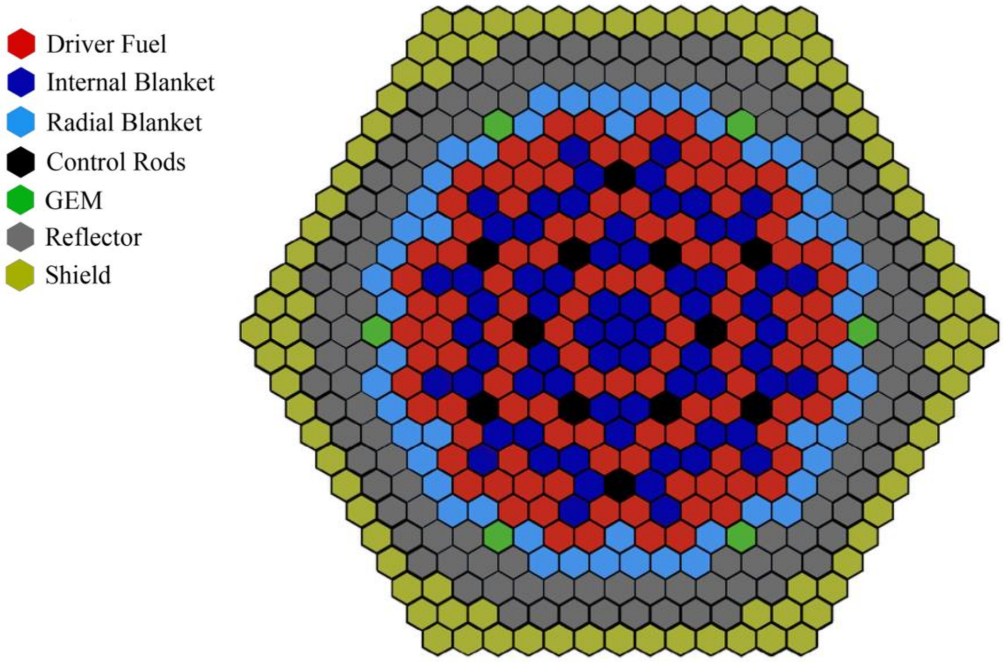

Figure 1: Radial Layout of GE PRISM Nitride core, taken directly from [4].

Two configurations of fuel assemblies are used in the Nitride PRISM design – 114 driver fuel assemblies, 73 internal blanket assemblies, and 48 radial blanket assemblies. Driver fuel assemblies have 217 pins arranged in a triangular lattice. Isotope composition for driver fuel and blanket fuel is found in Table 4. Geometrical specifications of each driver fuel pin are given in Table 5. Internal and radial blanket assemblies have 127 pins arranged in a seven-ring triangular lattice. It should be noted that freshly loaded fuel material composition is identical in internal and radial blanket fuel, before varying depletion characteristics across the core change material composition.

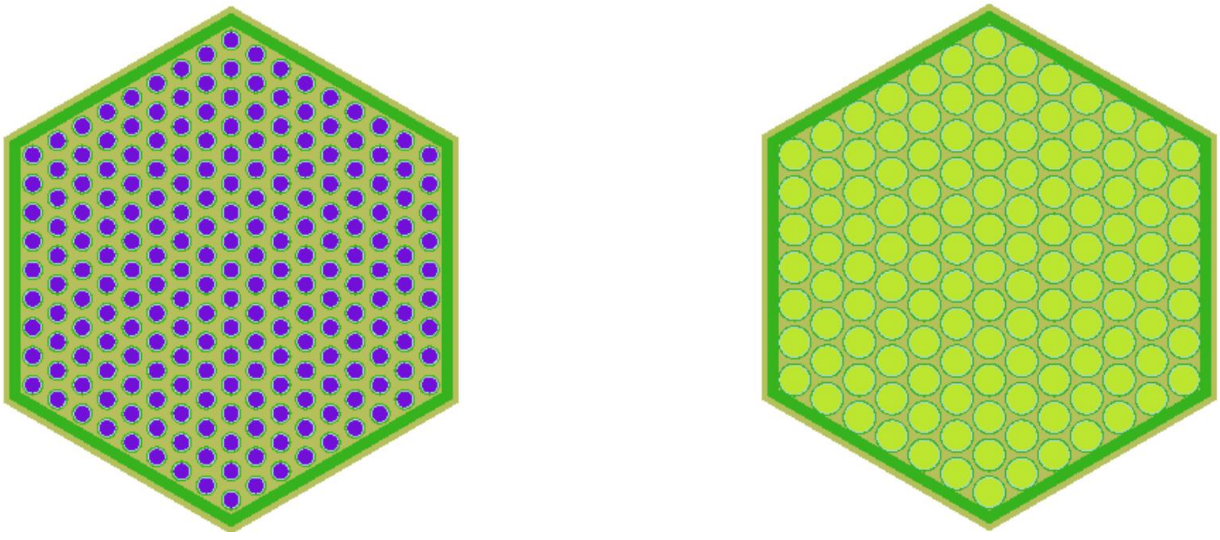

Figure 2: Driver and Blanket Fuel Assemblies shown on the left and right figures, respectively.

Table 4: Isotopic Composition of Driver and Blanket Fuel.

Nuclide	Driver Fuel Wt.%	Blanket Fuel Wt.%
U^{234}	*0.004*	*0.005*
U^{235}	*0.484*	*0.637*
U^{238}	*67.647*	*88.990*
Pu^{238}	*0.104*	*0.019*
Pu^{239}	*15.656*	*2.845*
Pu^{240}	*3.969*	*0.721*
Pu^{241}	*1.865*	*0.339*
Pu^{242}	*0.307*	*0.056*
Np^{237}	*2.191*	*0.398*
Am^{241}	*1.818*	*0.330*
Am^{243}	*0.372*	*0.068*
N^{14}	*4.989*	*4.997*
N^{15}	*0.594*	*0.595*

Freshly loaded fuel in internal and radial blankets is identical - composed of approximately natural Uranium and significantly less fertile Plutonium content compared to driver fuel. Rods consist of fuel slugs with reflector endcaps and a gas plenum which serves as buffer space for fission byproduct gas release. As several fuel designs for the PRISM reactor were developed for different purposes – actinide recycling, isobreeder and breeder configurations [1]. For this study, most data was taken from reference [4] and reasonable estimates were made for gaps such as material densities, cell temperatures, and other quantities of interest, e.g., geometry details. A schematic of a single fuel rod is given in Figure 3.

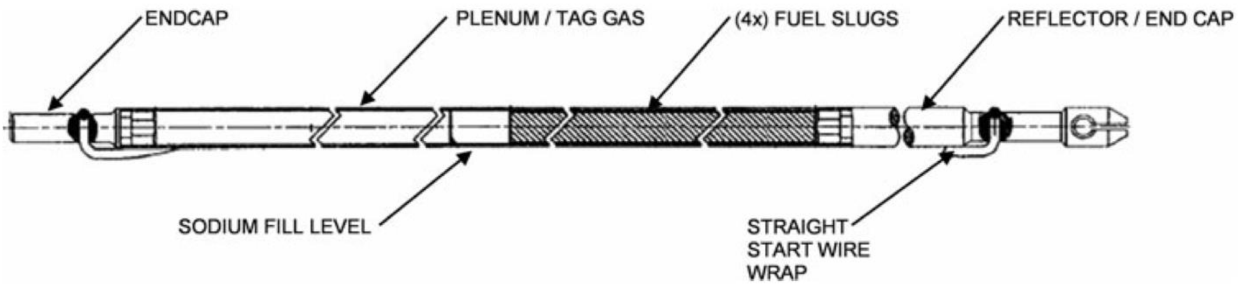

Figure 3: Schematic of Fuel Rod, taken from [1].

3.2 Control & Shielding Assembly Configuration

Reactivity control for nominal startup, reactivity control and shutdown is accomplished with B_4C control rods [1]. Control assemblies are composed of six rings of cylindrical control rods arranged in a triangular lattice surrounded by cladding and sodium coolant. For the three modelled cases, the uncontrolled case has all rods withdrawn, the controlled case has all rods inserted, and the critical case has some of the radial rods withdrawn to achieve criticality. For the critical case, all control rods in the six outer control modules (black hexagons in Figure 1) are fully inserted while control rods in the six inner control modules are fully withdrawn.

Table 5: Driver and Blanket Fuel Assembly Specification.

	Nitride PRISM Blanket Fuel Wt.%	
Parameter	Driver	Blanket
Pin Count	217	127
Pin Rings per Assembly	9	7
Assembly Pitch (cm)	16.14200	16.14200
Duct Gap (cm)	0.43200	0.43200
Duct Wall Thickness (cm)	0.39400	0.39400
Pin Pitch (cm)	1.01356	1.32542
Fuel Pin Radius (cm)	0.32735	0.50230
Gap Thickness (cm)	0.04225	0.04230
Clad Thickness (cm)	0.05590	0.05590
Clad Radius – inner (cm)	0.36960	0.54460
Clad Radius – outer (cm)	0.42550	0.60050

3.3 Gas Expansion Module (GEM)

The PRISM design makes use of GEM assemblies (see Fig. 1) arranged on the periphery of the fuel region [1]. GEM is a leakage-based reactivity control mechanism assembly filled with an inert gas. Functioning much like a piston, coolant flow pressure causes the GEM to be filled with coolant. Inversely, when coolant pressure decreases the gas expands, pushing coolant down, and thereby increasing leakage from the core. For all three benchmark cases developed in this work Helium (He) with a density of $0.0001786\ g/cm^3$ is used. As a reactivity control feature with geometry and material composition necessarily changing with coolant flow pressure during reactor operation modeling this behavior dynamically would require thermal hydraulic feedback and transient modeling significantly beyond the scope of this work. Instead, a reasonable estimate of steady state Helium density mentioned above is used to model this component within the framework of a standalone neutronics model.

4 SOLUTION TO BENCHMARK PROBLEMS

OpenMC [7] was used to model three Core configurations – uncontrolled (ARO), critical (SRI), and controlled (ARI) cases – to calculate the core eigenvalue and a fuel pin heating map of the core. Due to the symmetry of the problem, only a 1/12 radial wedge was run while imposing reflective boundary conditions on the two wedge surfaces. Coupled neutron-photon transport calculation for each benchmark was performed on a 48-core allocation on the Georgia Tech PACE HPC for eleven hours and heating in both driver and blanket fuel was tallied. Heating densities (normalized by pin volumes) are overlayed on the geometry in Figure 4.

For the uncontrolled (**ARO**) case, OpenMC calculated an average core eigenvalue of 1.0422±0.0002. Similarly, the corresponding eigenvalues for controlled (**ARI**) and critical (**SRI**) cases are 0.9611±0.0002 and 1.0005±0.0002. These results are summarized in Table 6.

Table 6: Summary of Eigenvalue Solutions.

Case	Core Eigenvalue
ARO	1.0422 ± 0.0002
SRI	1.0005 ± 0.0002
ARI	0.9611 ± 0.0002

Fission density was tallied in driver and blanket fuel pin cells. The ratio of the average fission densities (FD) in driver and blanket fuel pins is provided in Table 7 and show higher average fission density in driver fuel – as expected.

Table 7: Ratio of the Fission Densities in Driver and Blanket Fuel

Average FD in Driver Fuel	$2.9471 \times 10^{-1} \pm 0.0002$
Average FD in Blanket Fuel	$1.3255 \times 10^{-1} \pm 0.0002$
Radio (Driver Fuel/Blanket Fuel)	2.2234 ± 0.0003

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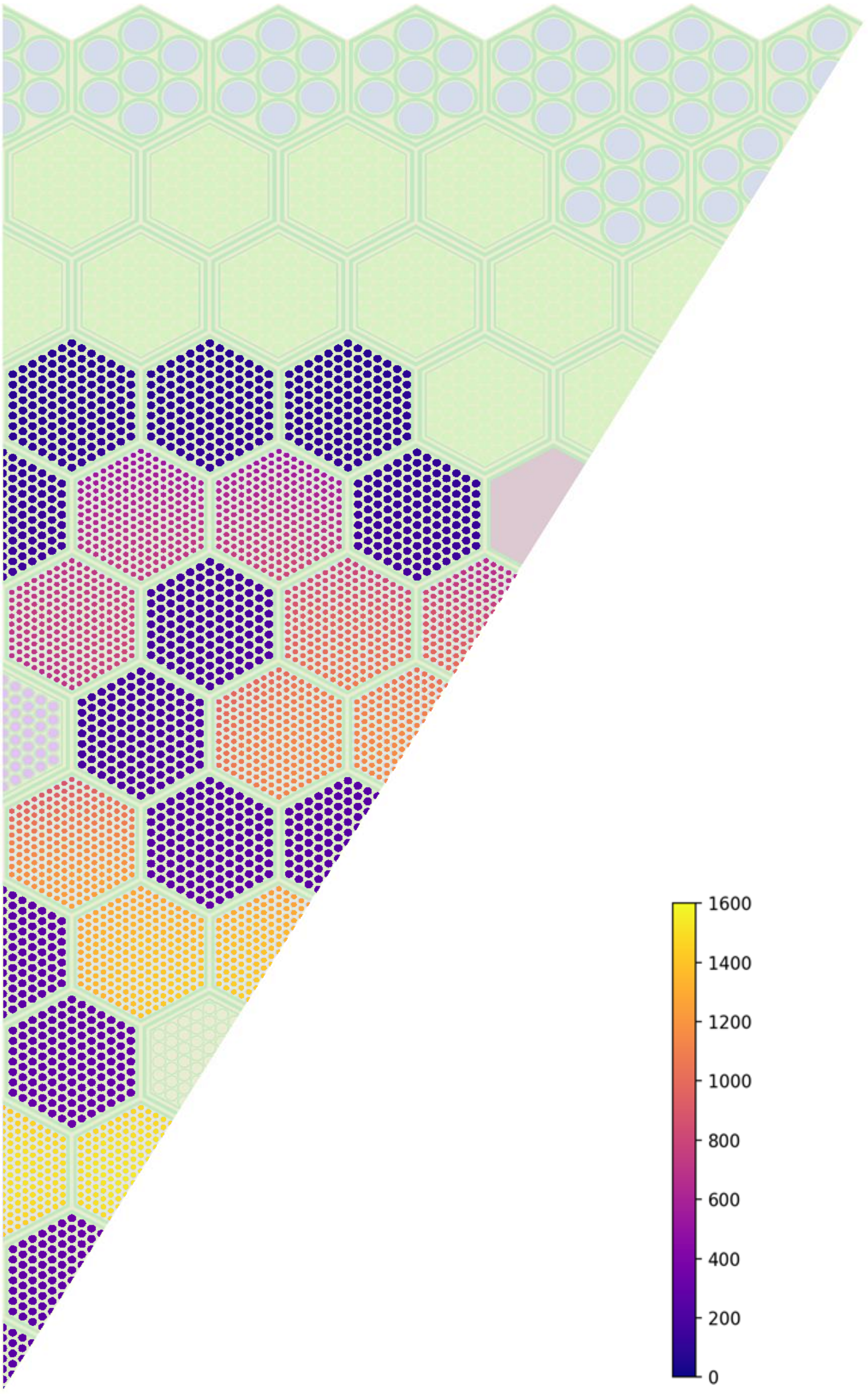

Figure 4: Fuel Pin Heating $\left(\frac{eV}{source\ particle\ cm^3}\right)$ map of Core slice.
Accounts for both fission and gamma photon heating.

The Total fuel pin power (including both neutron and photon heating in the fuel) was tallied in both driver and blanket fuel pins for the relevant cells and then plotted over the geometry. As driver and blanket fuel pins do not have the same radius, we are primarily interested in the volume averaged heating – i.e., heating density. Normalizing for cell volume shows substantially higher heating density per source particle (s.p.) in driver fuel – as expected – with peak mean heating density reaching

1550 $eV/s.p.cm^3$ in the central pins of the six most inner driver fuel assemblies. This is visualized in Figure 4 over the geometry plot.

5 CONCLUSIONS

In this work we have developed a set of stylized benchmark problems for a Nitride fuel configuration of the PRISM sodium cooled small modular reactor. This set included three cases: an uncontrolled case with all control rods withdrawn, a critical case with some rods in, and a controlled case with all control rods inserted. Publicly available geometry and material design information from references [1-4] informed much of the benchmark development. In those areas where details were unavailable, reasonable estimates were extrapolated from other sources and designs. Complete core geometrical and material parameters will be included in the upcoming special issue of *Nuclear Science and Engineering* journal on the HND 2026 conference.

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