Innovative Fuel Designs for Pebble Bed Reactors

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INTRODUCTION

This paper focuses on the analysis of alternative pebble bed reactor fuel to improve primarily by utilizing decommissioned plutonium and other non-uranium-based fuels. Nuclear reactors, especially pebble bed reactors, offer a reliable source of clean energy. Pebble bed reactors use graphite spheres embedded with uranium particles, known as tri-structural isotropic (TRISO) fuel, to moderate and sustain nuclear fission. The fuel kernel within the TRISO particles is the main variable analyzed in this project. Effective improvements to the fuel design can increase energy output and reduce the time needed to store radioactive waste. [1]

The goal of this project is to investigate alternative pebble fuel configurations that could enhance nuclear efficiency and safety. Key objectives include evaluating new fuel types, such as plutonium, mixed oxide (MOX), and inert matrix fuels (IMF), for better burnup rates and reduced waste management times. The team will analyze the heat transfer properties of different pebble sizes and designs, maintaining criticality and similar or improved thermal performance compared to current pebble designs.

The research scope includes simulating various fuel types using the Monte Carlo N-Particle 6.3 (MCNP) [2] software to determine their neutronic characteristics. Heat transfer analysis will be performed for each fuel type to maintain or improve the current pebble designs. The results will guide recommendations for improved fuel configurations, which can extend the viability of nuclear power in the clean energy landscape. This work supports the global push towards more sustainable and safer nuclear energy solutions. The push for more sustainable and better nuclear fuel aligns with the broader international efforts that can be seen optimizing and reducing nuclear waste. This shows firsthand how essential these innovations are around the world.

CONCEPT SELECTION

The concept selection was based largely on the collaboration that the team had with the research client, as well as the needs and requirements that must be met for the commissioning source to be satisfied. The design concept ultimately selected was to further develop the Inert Matrix Fuel (IMF) TRISO configuration. The decision to pursue this concept was made by first evaluating whether or not this project would align with the objectives and constraints brought to us by the client. To help the researchers identify that this was the best concept to pursue, a qualitative systematic decision-making process was used. The criteria that went into this consists of the concept's safety and thus the fission products released in accident scenarios, as well as the concept's efficiency which would be assessed by looking at the burnup and thermal conductivity. The concept must also meet cost criteria based on aspects like the cost per pebble. Other criteria like the waste reduction, which is seen through the amount of long-lived radionuclides generated, as well as the scalability and manufacturing realities for the end product were considered before finally arriving at our selected IMF TRISO design.

COMPUTATIONAL METHODOLOGY

Due to the nature of this project, no physical prototypes were created. Instead of verifying the final physical product by testing it in a real-world application, the aforementioned MCNP was used. The validation process largely involved coming to a deep understanding of the underlying principles that govern the design, and verifying that they match the conclusions that were arrived at. This means a lot of the validation comes from logically examining the systems and equations that govern the behavior of our simulations. Data is also verified by analyzing the data collected by other researchers and scientists in the field. If the results match similar studies and experiments, the researchers are qualifying that as verification and can thus gain confidence in their conclusions.

Initial data collection used an already existing MCNP model of a standard X-energy type pebble fuel that was created by former VCU students [3]. MCNP's KCODE feature was utilized to simulate reactor conditions and analyze the neutron flux and burnup characteristics. The KCODE function can assess the moderator temperature coefficient (MTC) and fuel temperature coefficient (FTC) This required simulation to be run with different set temperatures for each of the different materials. This then allowed for the calculation of the coefficients. The temperature was changed by altering the selected nuclear interaction data for the fuel and moderator material. These selections could be made independently of each other, meaning that the temperature of the fuel could be changed without affecting the temperature of the moderator. This is essential to accurately isolate the effect of the FTC and MTC on k-infinity independently. The next step was adding tally cards in our simulations to depict neutron flux for different energy ranges. Neutron flux graphs were created for all of the tested fuel materials.

Lastly, the fuel burnup simulation was carried out to track the behavior and performance of the IMF TRISO fuel during the reactor operation and cooling period. Each fuel material was irradiated for just over 3 years followed by a waste storage period of 100,000 years to track radioactive decay. A key factor that the researchers were interested in was the amount of fissile material present in the fuel as the reactor is operated. The burnup feature of MCNP tracks the change in nuclide concentration throughout a simulation, which allows for the summations of the amount of fissile material present at different stages of the fuel's lifetime. During the waste storage period, MCNP calculates the radioactive decay of all of the radionuclides present as well as the total radioactivity of the fuel. This allows for the estimation of the amount of time needed for storage.

PEBBLE BED REACTOR FUEL MODEL

The following fuel compositions were chosen based on the needs of the client and common proposed fuel types. Table 1 depicts the proposed fuel types and their isotopic compositions of interest. Note that the two enrichments for uranium are based on 5%, the current standard and ~18% which has been proposed for longer reactor run times and better fuel performance. The two plutonium compositions were chosen based on 70% ^{239}Pu which is considered reactor grade [4], and 93% ^{239}Pu, which is the approximate composition of weapon-grade plutonium [4]. The likely source for this plutonium will be from decommissioned nuclear weapons.

Two special fuel types were also considered, MOX, and IMF. MOX fuel is created by mixing plutonium with the depleted uranium. A mixture of 8% ^{239}Pu corresponds to an enrichment of 5% ^{235}U [4]. IMF is created by mixing minor actinides with fissionable materials like plutonium. The benefit of IMF is its ability to help transmute plutonium and minor actinides into shorter lived waste products thus helping minimize the current excess of reactor spent fuel waste [5]. It's important to note that at the current stage of the project MOX and IMF have not been analyzed. Future reports will include this analysis.

Table 1: Proposed fuel types and enrichments

Uranium Oxycarbide (UCO)	5% ^{235}U and 18% ^{235}U
Uranium Dioxide (UO_2)	5% ^{235}U and 18% ^{235}U
Plutonium Oxycarbide (PuCO)	70% ^{239}Pu and 93% ^{293}Pu
Plutonium Dioxide (PuO_2)	70% ^{239}Pu and 93% ^{293}Pu
MOX	8% ^{239}Pu with ^{238}U
IMF	25%, 50%, 75% and 100% Inert TRISO particles

With the background research complete the MCNP models were then ready to be modified. A new model was created for each of the fuel types found above, with each model only altering the fuel material card. Figure 1 is taken from MCNP's visual plotter and shows a cross-section of the inside of a pebble. The TRISO particles can be seen as small black and red dots scattered throughout the pebble with blue representing graphite.

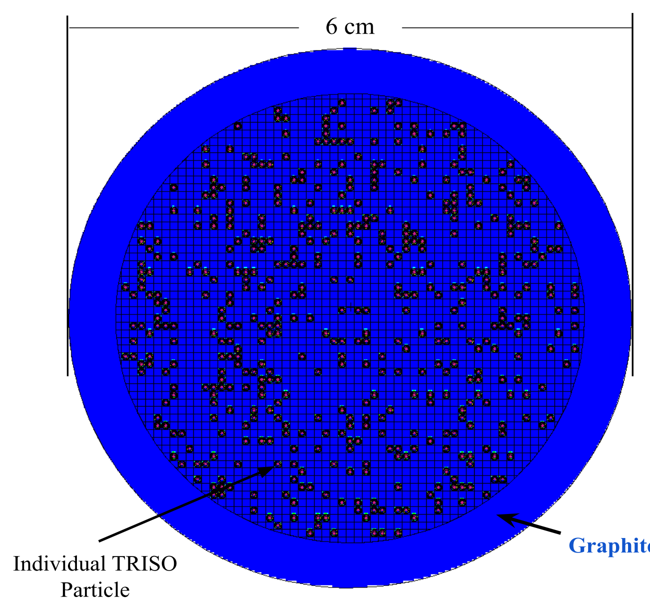

Fig. 1. MCNP model of a reactor pebble. Blue represents reactor-grade graphite, and red and black dots represent TRISO fuel particles.

Figure 2 was created from MCNP's visual plotter to show the cross-section of an individual TRISO particle. The outer shells are pyrolytic carbon (light blue), silicon carbide (green), inner pyrolytic carbon (yellow), and an inner porous carbon buffer (dark blue). The center of the TRISO is the fuel kernel (magenta). This fuel kernel is what was changed from model to model for each fuel type.

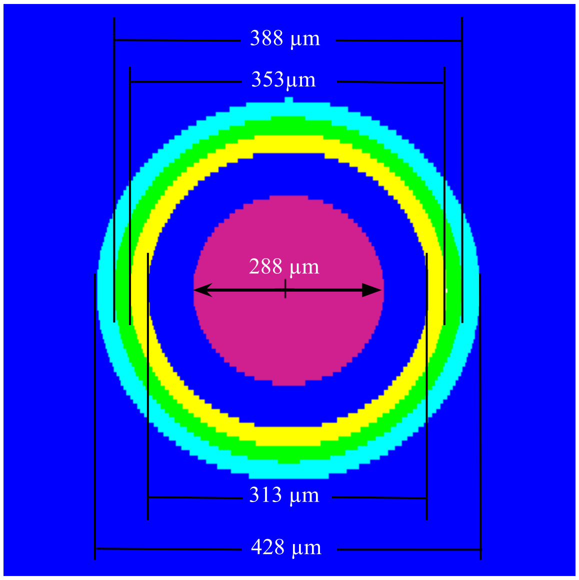

Fig. 2. MCNP model of TRISO particle. Outer pyrolytic carbon (light blue), silicon carbide (green), inner pyrolytic carbon (yellow), inner porous carbon buffer (dark blue), and fuel kernel (magenta).

RESULTS

One of the first experiments to be performed was an analysis of the FTC for the various fuel types. The MCNP KCODE function was used to calculate k_∞. The cross-section data used for the fuel material was simulated at different temperatures from 600 K to 2,500 K. The FTC can be calculated using Eq. 1. In this equation k_∞ represents the calculated k_∞ given from the MCNP Kcode, and T_{fuel} represents the temperature of the cross section data the fuel was used with.

$$FTC = \frac{\Delta\rho}{\Delta T} = \frac{((k_\infty - 1)/k_\infty)_2 - ((k_\infty - 1)/k_\infty)_1}{T_{Fuel,2} - T_{Fuel,1}} \quad (1)$$

The FTCs were then averaged over all ranges to create a metric to compare the fuel behavior at all temperature ranges. Table 2 shows the results of these calculations.

TABLE 2: Calculated FTC averaged over the range of temperatures tested.

Fuel Type	Average FTC (600 K - 2500 K), PCM/K
UCO- 18% ^{235}U	-7.426
UO_2 - 18% ^{235}U	-7.41
PuCO - 70% ^{239}Pu	-2.55
PuO_2 - 70% ^{239}Pu	-3.17
PuCO - 93% ^{239}Pu	-2.40
PuO_2 - 93% ^{239}Pu	-3.2

The results shown in Table 2 indicate that the uranium-based TRISO fuels have a much better FTC. This fact means that to implement plutonium TRISO fuels, these factors will need to be accounted for in the design of the reactor. This reflection of data signifies that the procedures used for determining the safety and efficiency of these nuclear fuels are meaningful.

Another experiment performed in MCNP was a burnup simulation on the different fuel types. These simulations build on techniques already in practice by Oak Ridge National Labs [6]. Here, each of the reactor fuel types and compositions were simulated in reactor-like conditions for just under 3.5 years, following this burn, a decay period of 100,000 years was simulated to examine the longer-lived waste products.

One of the first analyses conducted was to determine the percentage of fissile material present in the reactor with time. This metric represents the longevity of the fuel and helps compare the overall reactor burn time available for each fuel type. Due to the time steps chosen in the MCNP input file, data points are available roughly every month of the 3-year burn time. Figure 3 depicts the change in the fissile material percentages of UCO, PuCO 70%, and PuCO 93% over time. It can be seen that a significantly higher percentage of fissile material is present in plutonium-based pebbles after three years of burn time. This result suggests that the plutonium fuels can be burned for much longer to show their full life span potential.

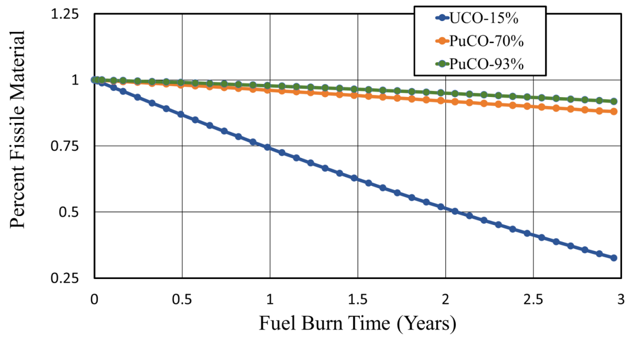

Fig.3. Percentage of fissile isotopes remaining during reactor operation.

As shown in Figure 3, the composition of plutonium has a positive correlation with the percentage of fissile material remaining. One could also note from Figure 3 that plutonium fuels could have been burned longer to fully utilize their capacity. A future simulation is planned in which plutonium will be allowed to burn for a longer fuel lifetime.

The waste products from the same burnup experiment were analyzed. However, no meaningful comparison can currently be made between the two fuel types until they reach a similar burnup level. The advantage of plutonium-based fuels in radioactive waste management primarily stems from their longer burn time, which allows for the destruction of more fission products. Until a burnup simulation is performed to test the full fuel lifetime of plutonium, no definitive conclusions can be drawn. Future papers and presentations will include comparisons of the results from these longer burn times.

For each of the fuel types, the neutron flux was analyzed with the use of MCNP's neutron tally capability. The results comparing UCO and PuCO of both plutonium compositions can be seen in Figure 4. When analyzing the comparison between UO_2 and PuO_2 it was found that the flux shapes were identical to that of the UCO and PuCO. Because of this, flux plots of the dioxide fuels are not shown.

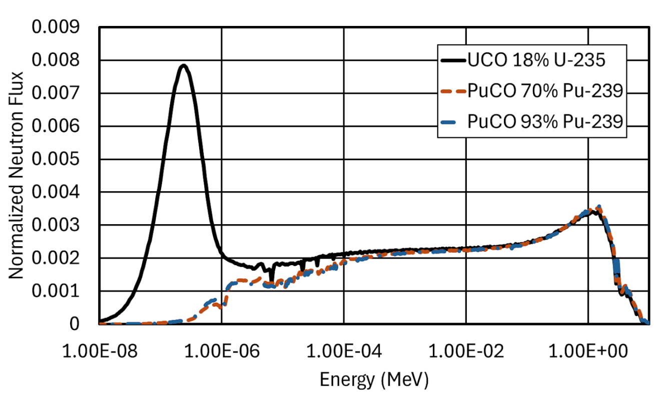

Fig.4. Neutron flux of UCO and PuCO pebbles.

Based on the results, we can see a notable lack of a thermal region neutron flux for the plutonium-based fuels. Due to this feature, it has been proposed that the effects of creating a fast reactor plutonium-based pebble should be analyzed as part of the project. To model this fast pebble, the graphite will be replaced with either a stainless steel or zirconium metal. Future papers and presentations will include this analysis.

FUTURE WORK

Much more work is needed to complete the initial goals laid out at the start of this project. In the current state of the project, no analysis work has been conducted on MOX or IMF fuels. However, an MCNP model with these fuel types has been created based on background research. The next step for these two fuels is to complete MCNP simulations and analyze the data. Work still needs to be performed on the heat transfer. Further work might feature alternative moderators or alternative pebble geometry. Analysis of a fast pebble bed reactor pebble will also be investigated. Extended irradiation testing of IMF configurations might also offer valuable insight in the future.

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