Feasibility Study of Solid Matrix Fuels for Space Power Reactors

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Abstract. Recent advances in nuclear fuel fabrication allow for new types of fuels with superior power density, safety, fabrication cost, and other traits. Spark plasma sintering (SPS) is one such technology that is currently being explored. The University of Florida and the Center for Space Nuclear Research have successfully demonstrated the technology for nuclear applications. SPS is unique in its ability to sinter disparate materials together allow for new types of cermets, heterogeneous metal alloys, and ceramic-ceramic composite fuels. SPS has been successfully demonstrated with fuels such as UO_2-W cermet matrix fuels and UO_2 with high thermal conductivity additives.

One of the fundamental requirements for space nuclear fuels is a high temperature capability. UO_2 is a high temperature fuel. However, its thermal conductivity is very low. SPS opens the door to creating composite materials with high thermal conductivity allowing the fuels to reach much higher power densities and lower temperature peaking factors. In addition, safety can be improved by utilizing small fuel pellets sintered to a fuel matrix which can retain fission products, possibly eliminating the need for fuel cladding.

This research focuses on exploring the design options for matrix composite fuels. Matrix material such as W, SiC, Be, BeO, and graphite are explored with UO_2, UN, and UC fuels. This analysis investigates neutronic properties, thermal conductivity of the composite matrix fuels.

Keywords: Spark Plasma Sintering, Space, Composite Fuels, Material Properties

INTRODUCTION

High temperature capability and high thermal conductivity are key attributes for high performance nuclear fuels. In space high temperatures are especially important as minimize the mass of space power systems; first by increasing the thermodynamic efficiency but more predominately by decreasing the size of the heat rejection system. A high thermal conductivity is necessary to reduce the temperature peaking in the fuel and protect the nuclear fuel from melt and creep.

Safety is a primary concern for nuclear fuels. In this research, analysis is directed toward encapsulated fissile fuel forms where fission products are kept within the fuel by a non-fissionable matrix material, adding a highly engineered barrier to fission product release. This research can also be applied to safety tolerant nuclear fuels for terrestrial reactors as high temperature capability and high thermal conductivity provide resistance to reactor accidents.

High temperatures place increasing demands on the materials for nuclear fuels and cause many traditional reactor components such as water to be incompatible. In this paper several materials were identified which hold promise to be used as high performance space nuclear fuels. Table 1 lists the fissile fuel and matrix materials that were analyzed in this study and Figure 1 lists some of their properties.

TABLE 1: Materials Analyzed [1-4].

Material	Type	Melting Point [K]	More Information
Graphite	Moderator Matrix	4000	Well Known, Moderating Properties
W	Structural Matrix	3695	Radiation Resistance, Strength, Thermal Conductivity, High Neutron Absorption
W-184	Structural Matrix	3695	Lower neutron absorption cross section, Must be enriched
Be	Moderator Matrix	1560	Well Known, Moderating Properties
BeO	Moderator Matrix	2780	Well Known, Moderating Properties
SiC	Structural Matrix	3000	Strength
(B-11)C_4	Moderator Matrix	2718	Moderating Properties
Nb	Structural Matrix	2741	Strength, Thermal Conductivity, Low Neutron Absorption
Mo	Structural Matrix	2893	Radiation Resistance, Strength, Thermal Conductivity, Low Neutron Absorption
Mo-92	Structural Matrix	2893	Corrosion Resistance, Radiation Resistance, Strength, Thermal Conductivity, Extremely Low Neutron Absorption
UO_2	Fissile	3140	Standard Fuel Form, Low Thermal Conductivity
UN	Fissile	3000	High Thermal Conductivity
UC	Fissile	2638	High Thermal Conductivity, High Uranium Density

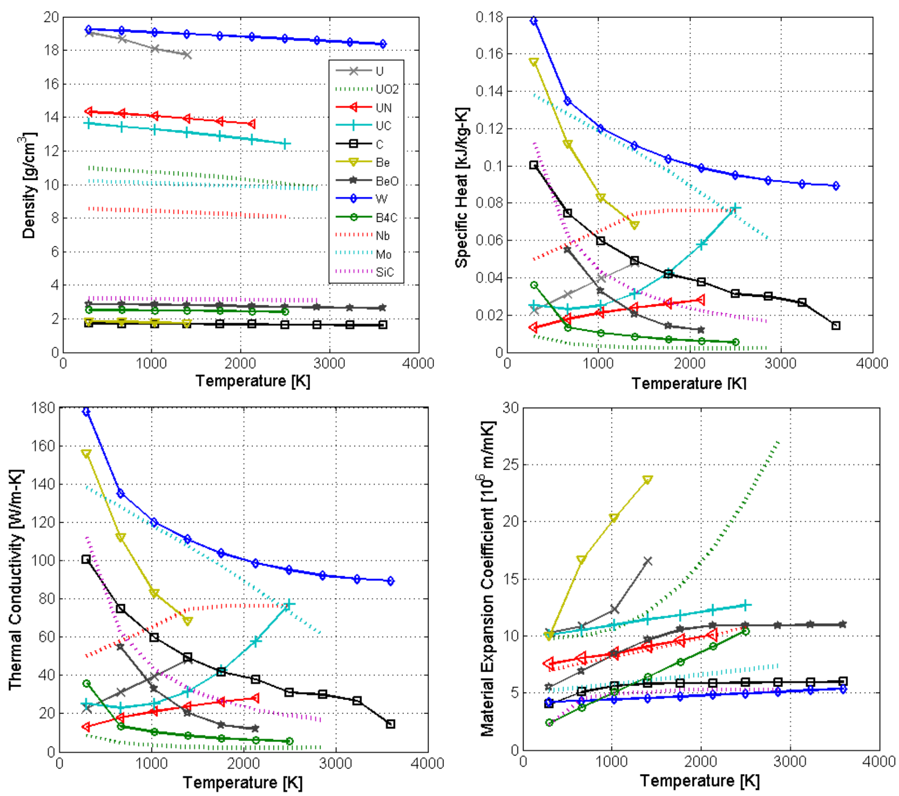

FIGURE 1: Material Properties

COMPOSITE MATRIX INFORMATION

Composite matrix solid nuclear fuels are composed of two distinct heterogeneously mixed materials. The matrix of the composite surrounds the nuclear fuel. Figure 2 below depicts a hypothetical cross section of the composite fuel. The nuclear fuel is represented as discrete spherical yellow dots in surrounded by the blue continuous matrix material.

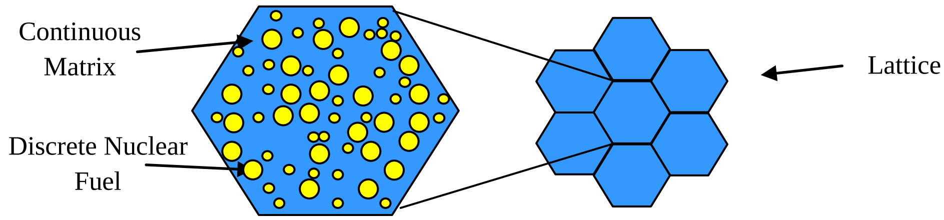

FIGURE 2. Hypothetical Composite Material Fuel Block

The discrete fissile fuel material is encapsulated by the continuous material. The matrix will keep the fission product waste contained inside the composite fuel block and the matrix material can be used to complement the shortcomings of the discrete nuclear fuel material providing superior composite fuel block properties.

$$\phi_f = \frac{V_f}{V_f + V_m} \tag{1}$$

The fissile fuel volume fraction, ϕ_f describes the relationship of the matrix material to the fuel material. The packing of the discrete phase within the matrix sets an upper bound for matrix-type compose materials. For ordered spherical fuel particles, the maximum fuel volume fraction is 74 percent. For random spherical packing, the maximum fuel volume fraction is approximately 63 percent.

Matrix Composite Material Properties

Thermal conductivity is a key in determining the fuel centerline temperature and the power density achievable. The temperature increase of a hot spot in a fuel element is roughly inversely proportional to the thermal conductivity. The thermal conductivity of a composite material can be approximated by the Maxwell theoretical model for composite thermal conductivity show in Equation 2 [5].

$$k_{composite} \backsimeq k_{matrix}\left[\frac{k_{fuel} + 2k_{matrix} + 2\phi_{fuel}(k_{fuel} - k_{matrix})}{k_{fuel} + 2k_{matrix} - \phi_{fuel}(k_{fuel} - k_{matrix})}\right] \tag{2}$$

The multiplication of the density and specific heat determine the thermal inertia of the reactor. A larger thermal capacity is advantageous for reactor control and balancing out transients. The density of composite materials can be determined by a volume weighted average. The composite specific heat can be calculated by a mass weighted average.

$$\rho_{composite} = \phi_{fuel}\rho_{fuel} + (1 - \phi_{fuel})\rho_{matrix} \tag{3}$$

$$C_{p_{composite}} = \frac{\phi_{fuel}\rho_{fuel}C_{p_{fuel}} + (1 - \phi_{fuel})\rho_{matrix}C_{p_{matrix}}}{\rho_{composite}} \tag{4}$$

The linear coefficient of expansion is an important factor to match between the matrix and fuel. A large mismatch will cause internal stresses in the composite matrix during temperature swings. The net composite in a matrix-type arrangement can be expressed in the relationship in Equation 6 [2].

$$\alpha_{composite} = \phi_{fuel}(\alpha_{fuel} - \alpha_{matrix}) + (1 - \phi_{fuel})\alpha_{matrix} \tag{6}$$

Neutronic Properties of Reactor Core with Matrix Composite Fuel

Due to the heterogeneous configuration of a typical reactor core design, core neutronic properties are dependent upon the neighboring geometries and materials. Those neighboring geometries may be comprised of composite fuel elements, moderating elements, coolants, reflectors, and other parts.

The disparate geometry and materials are simplified by looking at reactor systems composed of an infinite number of fuel elements, moderator elements, and coolant in an infinitely repeated grouping called a lattice. An infinite lattice is useful for approximating a finite reactor core's neutronic properties and can determine if a fuel is capable of going critical.

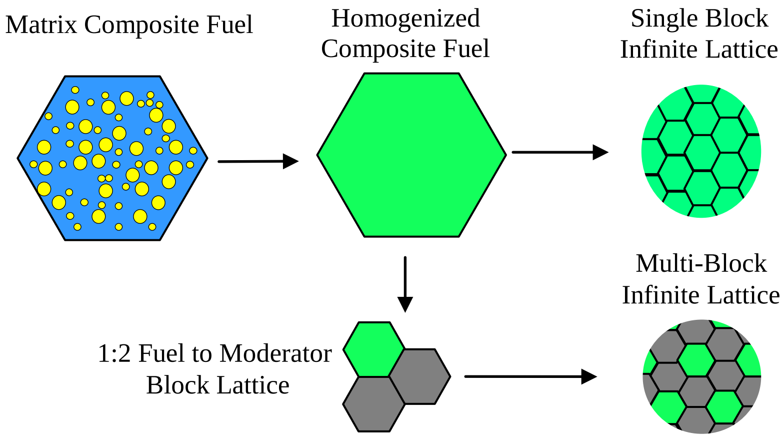

FIGURE 3: Infinite Lattice

Figure 3 depicts the process of infinite lattice creation. In many cases it is desirable to make a lattice with two block types. This applies directly to W, Nb, and Mo matrix fuel blocks which do not contain a moderating material. To effectively form a critical lattice these fuel forms typically require a moderator block and forms a multi-block lattice

Infinite lattice analysis misses one crucial aspect of reactor core design, the critical size of the reactor. A finite radius search explores the physical size of the reactor. A radius search with a defined reflector is also a useful gauge, as modern reactors typically have a reflector. Figure 4 depicts a criticality search.

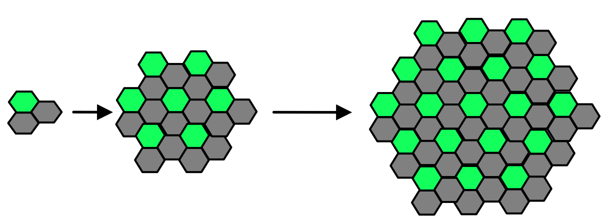

FIGURE 4: Finite Ring Criticality Search.

Reactor Core Design

Nuclear reactor design is an optimization process involving with three major analyses: neutronic, thermal-hydraulic, and material analyses. The neutronic analysis determines the minimum size of the reactor. The thermal-hydraulic analysis determines the maximum temperature and power of the reactor. The material analysis determines the ability of the reactor to resist damage. These three performance areas are coupled by each other. An unsatisfactory performance in one area requires a core redesign which is often achieved by changing the repeating lattice. Changes core geometry will affect the other areas performance areas. Figure 5 illustrates this process.

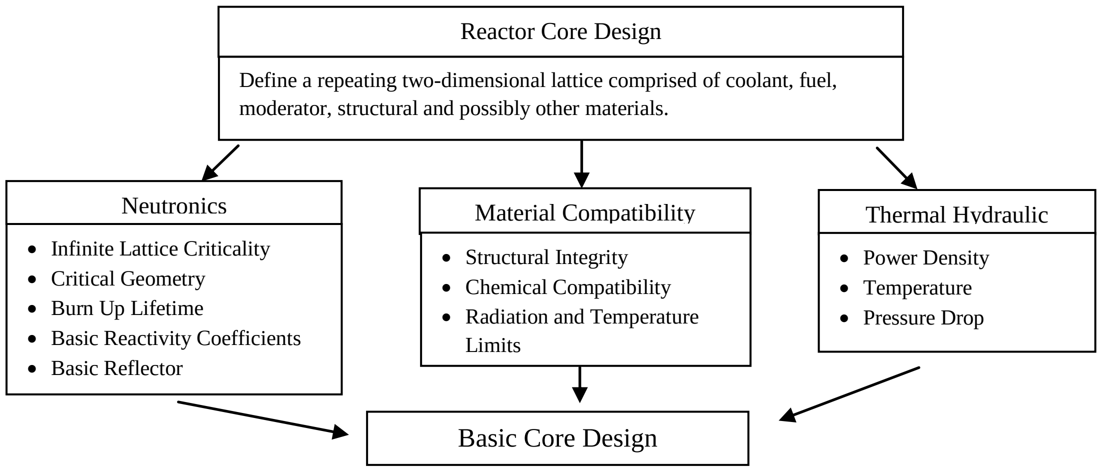

FIGURE 5: Basic Reactor Design Process

ANALYSIS AND DISCUSSION

Nuclear Properties

Figure 7 below depicts the results of the k-infinite analysis for homogenous mixtures of UO_2 combined with various matrix materials. On the x-axis the UO_2 fuel volume fraction is listed, and on y-axis the k-infinite value is given. On the left side, the fuel volume fraction is logarithmic going from 0.01 percent to 100 percent. Fast reactors exist in fuel volume fractions above 10 percent and thermal reactors in regions below 2 percent fuel volume fraction. The thermal region begins to taper around 0.1 percent as the fuel density becomes too small. A linear graph of the fast reactor domain is shown on the right.

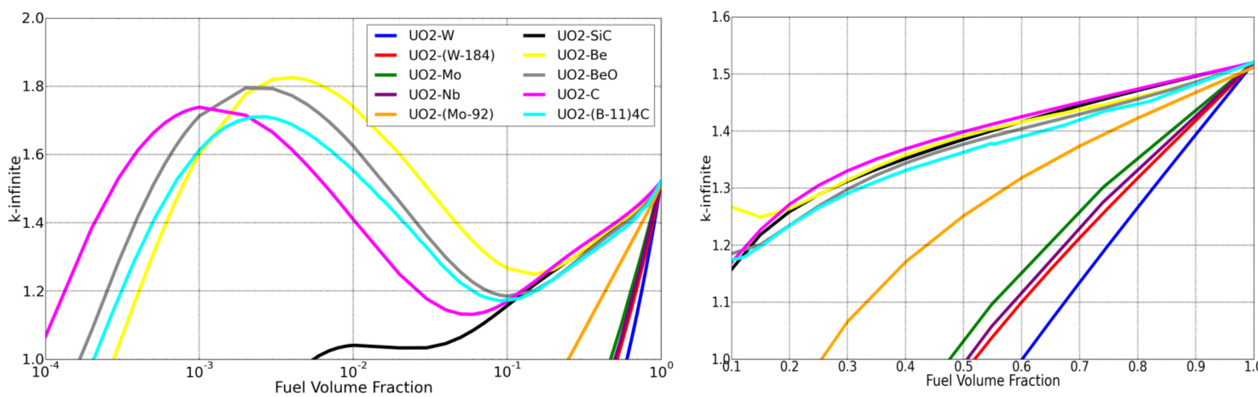

FIGURE 6: UO_2 Fuel Volume Fraction vs. k-infinite for a 300 K Temperature and 20 Percent Enrichment

Other fissile fuels such as UN and UC follow the same trend as UO_2 with slightly higher k-infinite values in the fast reactor range because of the higher Uranium density. In the thermal reactor, k-infinite is slightly shifted based on the absorption cross section of the nitrogen for UN and carbon for UC. Higher enrichments raise the value of k-infinite. Higher temperatures tend to decrease the k-infinite because of increased resonance absorption from Doppler broadening.

Figure 7 gives the critical radius of a reactor. On the x-axis the fuel volume fraction of UO_2 is given, and on the y-axis the critical radius in cm is given. The critical radius in this context is a cylinder's radius with a height of twice the radius size. On the left hand side a bare unreflected critical core radius is given. On the right side the critical radius has been reduced by surrounding the cylindrical reactor with a 30 cm axial and radial reflector.

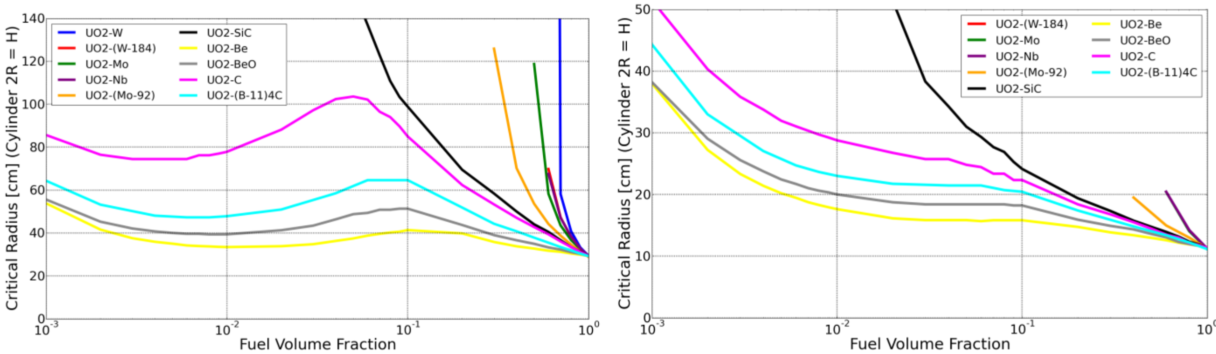

FIGURE 7: UO_2 Fuel Volume Fraction vs. Critical Radius for a 300 K Temperature and 20 Percent Enrichment
Left: Bare Core Right: 30 cm Be Axial and Radial Reflector

The reflected core differs significantly from the bare core. The reflected core has a smaller critical radius. In addition the hump associated with the epithermal resonance region shown for the bare core was smoothed by the thermalization effect of the reflector in the reflected core.

Several matrix materials including Mo, Nb and W are non-moderating and become subcritical below 50 percent volume fraction and above. These matrix materials cannot effectively make fast reactors because of their large fast neutron absorption cross section. These fuels require a multi-block infinite lattice containing moderating blocks to form effective critical geometries. The non-moderating matrix material's primary purpose is to provide a structural material to encapsulate the fuel. A multi-block lattice structure could be developed to improve their criticality by adding a moderator block to shift the neutron spectrum into the thermal range.

A multi-block lattice cell was defined for Figure 8. The fuel block was given a one-to-one matrix to fuel ratio. The moderator fuel block is composed of BeO moderator. Figure 8 depicts the k-infinite and bare reflected core nuclear properties of the multi-block lattice with 20 percent enrichment at 300 K. The x-axis lists the fuel block volume fraction. One minus the fuel block volume fraction yields the moderator block fraction.

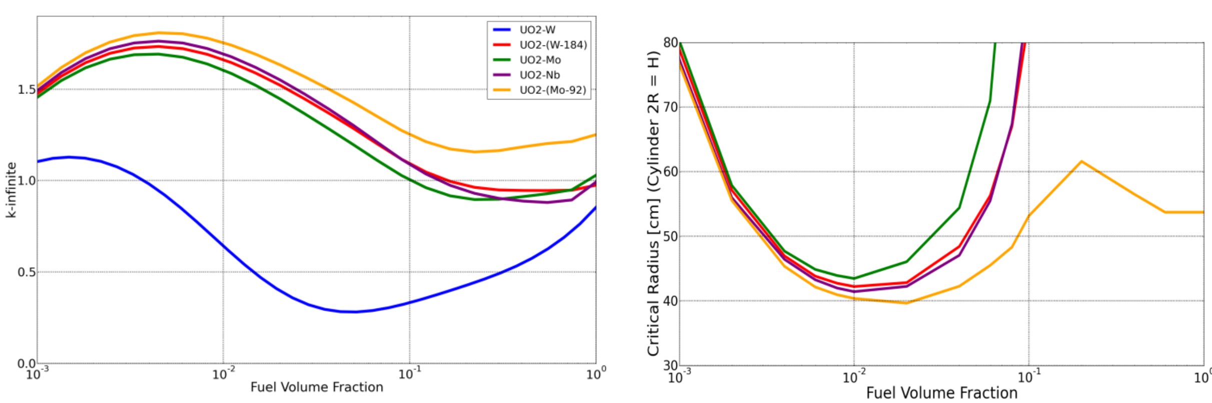

FIGURE 8: Multi-block lattice 1:1 UO_2 to Matrix for the Fuel Block and BeO for the Moderating Block. Left: k-infinite calculation. Right: Unreflected core critical radius.

The structural matrix materials shown in Figure 8 with the exception of Mo-92 can only exist as thermal reactors. The natural W lattice is almost incapable of being utilized as a thermal reactor because of its relatively large thermal neutron absorption cross section.

The trends for neutronic properties of composite fuels are dominated by the matrix material. The fissile fuel forms shift k-infinite and the critical radius but only to a small degree. Table 2 below summarizes matrix components and their respective neutronic traits.

TABLE 2: Neutronic Conclusions

Material	Type	Thermal Spectrum	Fast Spectrum	Other Notes
Graphite	Moderator Matrix	Strong	Strong	Larger Thermal Critical Radius
W	Structural Matrix	Very Weak	No	Challenging Criticality Requirements
W-184	Structural Matrix	Fair	No	
Be	Moderator Matrix	Strong	Strong	Small Thermal Critical Radius
BeO	Moderator Matrix	Strong	Strong	Small Thermal Critical Radius
SiC	Structural Matrix	No	Strong	
$(B\text{-}11)C_4$	Moderator Matrix	Strong	Strong	
Nb	Structural Matrix	Fair	No	
Mo	Structural Matrix	Fair	No	
Mo-92	Structural Matrix	Strong	Very Weak	Best Performing Refractory Metal

Thermal Conductivity

Thermal conductivity of composite material is given in Figure 9 for UO_2 and UC composites. The composite fuels contain a 50 percent fuel volume 50 percent matrix volume.

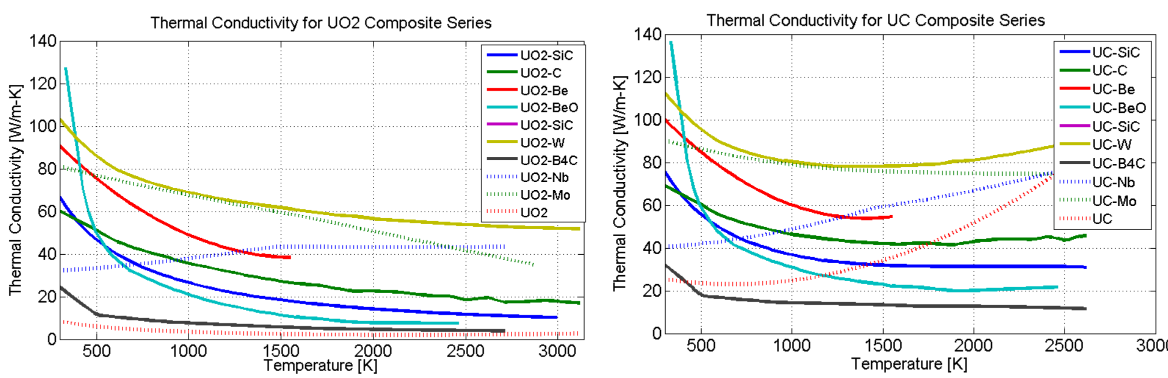

FIGURE 9: Composite Matrix Thermal Conductivity with a 50 Percent Fissile Fuel Volume Fraction Left: UO_2 Right: UC

For thermal conductivity both the matrix and fissile fuel components have a large impact upon the composite thermal conductivity. UO_2 has an extremely low thermal conductivity and its thermal conductivity greatly augmented with any of the matrix material analyzed. UC and to a lesser degree UN have significant thermal conductivity and form superior thermal conductivity composites mainly with the metal matrix materials.

CONCLUSION

In designing a nuclear fuel there are a series of key design questions. How much heat can be effectively removed, what temperature range can the fuel operate over, what are the critical configurations, and how much burn up can be achieved?

Composite materials allow nuclear fuel designers to combine positive traits of the constituent components to meet design and safety requirements. An intrinsic benefit of composite matrix fuels is that the matrix serves as a barrier to fission product release for improved safety. In this paper analysis was completed to explore the neutronic and thermal conductivity of various promising composite fuels. The data in this paper are not conclusive. There are many options for fuel design and the options can be explored by viewing the graphics in this paper.

This paper has focused (though not exclusively) on 20 percent enrichment, UO_2 fissile fuel, solid fuel forms, and high temperatures. This focus could be rescoped easily to accommodate, for example, liquid core reactors or high enrichments.

In future work, burn up analysis is planned to explore the question of fuel lifetime. A higher fidelity material compatibility analysis is planned. The eventual goal is to build upon this data a systematic process to complete full core reactor design.

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