ANALYSIS OF THE PEBBLE-BED VHTR SPECTRUM SHIFTING CAPABILITIES FOR ADVANCED FUEL CYCLES

A Thesis

by

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ABSTRACT

Analysis of the Pebble-Bed VHTR Spectrum Shifting Capabilities

For Advanced Fuel Cycles. (May 2006)

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Gas-cooled nuclear reactors have been receiving specific attention for Generation IV possibilities due to desired characteristics such as relatively low cost, short construction period, and inherent safety. Attractive inherent characteristics include an inert, single phase helium coolant, refractory coated fuel with high temperature capability and low fission product release, and graphite moderator with high temperature stability and long response times. The passively safe design has a relatively low power density, annular core, large negative temperature coefficient, and passive decay heat removal system.

The objective of the U.S. DOE NERI Project is to assess the possibility, advantages and limitations of achieving ultra-long life VHTR (Very High Temperature Reactor) configurations by utilizing minor actinides as a fuel component. The present analysis takes into consideration and compares capabilities of pebble-bed core designs with various core and reflector configuration to allow spectrum shifting for advanced actinide fuels.

Whole-core 3D models for pebble-bed design with multi-heterogeneity treatments in SCALE 5.0 are developed to compare computational results with experiments. Obtained results are in agreement with the available HTR-10 data. By altering the moderator to fuel ratio, a shift in the spectrum is observed. The use of minor actinides as fuel components relies on spectrum shifting capabilities. Actinide fueled VHTR configurations reveal promising performance. With an optimized pebble-bed model, the spectrum shifting abilities are apparent and effects of altered moderator to fuel ratio, and Dancoff factor are investigated. This will lead to a facilitated development of new fuel cycles in support of future operation of Generation IV nuclear energy systems.

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1. Introduction

1.1 Research Objectives

The objective of the work is to assess the possibility and limitations of spectrum shifting in pebble-bed cores. The research effort is an integral part of the project funded by U.S. DOE to evaluate ultra-long life Very High Temperature Reactor (VHTR) cores. The present analysis takes into consideration specifically pebble-bed core design with the intention of utilizing minor actinides as a fuel component to approach the reactor lifetime long operation without intermediate refueling. [1]

The main advantages of the pebble-bed VHTR configuration are their capabilities for spectrum shifting, inherent safety features, autonomous operation, incredibly high burn-up, and higher efficiency. If successful in utilizing minor actinides from spent LWR fuel, there will be a reduced spent fuel flow and handling per unit of produced energy. If widely deployed, the developed designs would allow reducing the long term radiotoxicity and head load of high-level waste sent to a geologic repository and enable recovery of the energy contained in spent fuel.

The preliminary neutronics analysis of the ability to shift the neutron spectrum in the pebble-bed VHTR configuration is presented in this paper. The spectral shift was archived by adjusting moderator-to fuel ration of the pebble-bed core. It is shown that the minor actinides as a fuel component achieve a higher burn-up in the fast spectrum. [11] Although a high destruction rate of minor actinides is envisioned as one the inherent characteristics of ultra-long life VHTR cores considered in this study, transitions between different spectral regimes would be required to optimize use of minor actinides and to maintain self-generation during the entire reactor lifetime.

The VHTR system is one of the near-term Generation IV concepts with consideration given to both a pebble-bed and prismatic core design. The goal of the Next Generation Nuclear Plant (NGNP) Program Element is to deploy a full-scale demonstration Generation IV VHTR for low cost hydrogen production by 2017 [2]. The proposed ultra-long life VHTR configurations can dramatically improve marketability and competitiveness of the Generation IV VHTR system for hydrogen production because their implementation allows worldwide deployment including developing countries. [4-5]

1.2 Background

1.2.1 Preliminary Studies

Current recycling approaches presume actinide recycling in the conventional closed fuel cycle by reprocessing spent LWR fuel. [6] The recovered minor actinides are chemically treated and, because of the undesired long-term radiotoxicity, are stored as high level wastes. Reduction of the long-term radiotoxicity is possible but very challenging through partitioning and transmutation. [7]

1.2.2 Technical Approach using SCALE 5.0

The code used in these calculations is SCALE 5.0 which uses cross section data based on ENDF/B-V evaluation. SCALE is a modular code system for Standardized Computer Analysis for Licensing Evaluation. It has been developed by Oak Ridge National Laboratory and utilizes well-established computer codes and methods within standard analysis sequences. SCALE 5.0 has several significant new modules to allow performing calculations of continuous energy spectra for processing multigroup problem-dependent cross sections, 1D and 3D sensitivity and uncertainty analyses, and 2D flexible mesh discrete ordinates modeling. [3]

(a) Code and Module Description

The calculations in SCALE 5.0 were performed with the 238 group library. Previous modules for cross section treatment were NITAWL and BONAMI. These modules did not account for resonance overlap, thermal up-scatter, 2D effects, double heterogeneity and transport effects. The NITAWL module used Nordheim method for resolved resonance treatment and BONAMI for Bondarenko factors in unresolved resonance range, however, the new CENTRM module available in SCALE 5.0 provides problem-dependent multi-group cross sections with accuracy of continuous-energy cross sections.

It is a 1D discrete ordinates code or resonance processing which eliminates many of the limitations inherent in the Nordheim Integral Treatment for resolved resonances. CENTRM properly treats overlapping resonances, multiple fissile materials in a unit cell, anisotropic scattering, and discrete level inelastic scattering. It uses point-wise continuous energy cross-section library to produce a set of point-wise continuous energy fluxes at discrete spatial intervals for each unit cell. Using these fluxes, the auxiliary code PMC collapses the point-wise continuous energy cross sections for each nuclide in each material in the unit cell to be used by other modules. Overall, CENTRM/PMC provides multigroup cross sections with the accuracy of continuous energy data.

The TSUNAMI module is a sensitivity and uncertainty analysis code built in to SCALE 5.0. Sensitivity coefficients produced by the TSUNAMI sensitivity analysis sequences predict the relative changes in a system's calculated k-effective value due to changes in neutron cross-section data. Tsunami produces sensitivity data on a group-wise basis for each region defined in a system model.

KENO-VI is a new version of KENO Monte Carlo criticality safety code which constructs and processes geometry data as sets of quadratic equations. The code's flexibility is increased by allowing intersecting geometry regions; hexagonal as well as cuboidal arrays; regions, holes arrays, and units rotated to any angle and truncated to any desired position; the use of an array boundary that intersects the array. KENO-VI geometry can be constructed using simple geometric shapes along with more complex, tailored shapes.

These modules and sequences are managed by the main control module, CSAS6, designed for executing the criticality analysis sequences using KENO-VI. CSAS6 automatically sets up the input for each module that is called in the CSAS sequence. It allows for automated cross-section processing, simple 1D cell description, optional cell-weighting, and 3D criticality calculation using KENO-VI.

(b) Verification and Validation

The purpose of validation is to establish an acceptance criteria such that there is a high degree of confidence that a system is calculated to be subcritical is indeed subcritical. KENO-VI and CENTRM have validation reports noting that the codes were validated using the 238-group ENDF/B-V library against critical experiments. A wide range of experiments were selected, which include high-enriched, intermediate-enriched, and low-enriched uranium systems; mixed oxide and plutonium systems; and 233U systems. Most of the experiments are thermal systems, however, one set of fast- and one set of intermediate-energy critical experiments were also included. For each of these, BONAMI processed unresolved resonance region cross sections and NITAWL processed resonance region cross sections. For each problem, in addition to the system k-effective and sigma, the energy of the average lethargy of fission (EALF) is reported. [3]

These data show that KENO-VI, using BONAMI and NITAWL for cross-section data processing, can be used with confidence for the design and criticality safety analysis of a wide range of systems. KENO-IV was also previously compared with KENO V.a. for a smaller set of problems and was found to produce excellent results.

1.3 Minor Actinide Characteristics

The VHTR system configurations are being studied to understand the potential for minor actinides as a fuel component. An important aspect of any nuclear energy system is the fuel and fuel cycle. Until recently, nuclear fuel has been uranium based typically made up in UO2. Recent fuel design has allowed for low enriched uranium (LEU) and mixed oxide fuel (MOx) consisting of a blend of uranium dioxide and plutonium dioxide. These fuel components are constituents of High Level Wastes thereby posing a challenge for safe disposal. With a reduction in minor actinide and fission product components in spent fuel, the incubation time will be reduced to about 1000 years in order to attain the radioactivity of natural uranium.

Using minor actinides in fuel design requires understanding nuclear interactions and characteristics. These characteristics greatly affect production and burn-up in a nuclear reactor. There are several nuclear data libraries which are not entirely consistent with one another. This is due to the probabilistic nature of events in radionuclides. Ten nuclides including uranium, neptunium, americium, and curium are compared using characteristics such as capture cross section (σ_c), fission cross section (σ_f), neutron production per fission (ν_f), neutron production per absorption (η), capture to fission ratio (α), and delayed neutron fraction (β) for the ENDF B-VI.8.

Table I displays these characteristics as a ranking for all nuclides with one representing the largest value for that characteristic and ten as the smallest.

TABLE I

Minor Actinide Nuclear Characteristic Rankings

	σ_f	σ_c	ν_f	η	A	B (% of 238U)
^{235}U	3	6	1	4	9	38.82%
^{238}U	10	10	9	10	1	100%
^{237}Np	9	4	8	9	2	23.23%
^{241}Am	6	2	7	7	4	7.47%
^{242m}Am	1	1	6	3	8	11.97%
^{243}Am	8	7	5	8	3	13.76%
^{242}Cm	5	8	3	5	6	2.24%
^{243}Cm	4	5	4	2	7	4.97%
^{244}Cm	7	9	2	6	5	N/A
^{245}Cm	2	3	1	1	10	10.08%

The characteristics examined in Table I are supplemental to the process of shifting a pebble-bed spectrum to a desired position. When investigating the spectrum shifting possibilities, the nuclear characteristics such as fission cross section, (σ_f), and neutron capture to fission ratio, (α), are important. By altering the moderator to fuel ratio to conform to these characteristics, reactivity swings can be observed over long irradiation times. Particular nuclides can serve as fuel materials or poisons by looking at their nuclear characteristics and adjusting the spectrum.

2. Model Verification

2.1 LEU-HTR PROTEUS

PROTEUS is a zero-power low enriched uranium (LEU) research reactor characterized by a high degree of flexibility in carrying out experimental reactor physics investigations. As such, it consists of a graphite cylinder with a central cylindrical cavity. The central graphite cylinder can alter configuration based on the desired experiment specifications while containing the control rods and instrumentation.

For the HTR research program, the cavity contained a pebble-bed core and the graphite cylinder served as a reflector. To investigate other lattice designs, the central zone of the reactor cavity contains the lattice of interest. The HTR research program was carried out from July 1992 to July 1996 in the framework of an International Atomic Energy Agency (IAEA) coordinated research program with eight participating countries. Areas of special interest were the double-heterogeneity in LEU fuel, the neutron steaming effects in a pebble-bed type HTR, and the effects of accidental water ingress (since HTR systems generally tend to be under moderated, an accident of this type can lead to large positive reactivity changes, in particular in the case of LEU fuel).

HTR-10 Design Characteristics

China has a significantly substantial program for the development of high temperature test reactors with favorable features. China has recognized these features, specifically safety, and decided to develop this technology. The modular HTGR is characterized primarily by inherent safety features and objective of the HTR-10 is verify, demonstrate and establish an experimental base for developing nuclear applications. The specific aims of the HTR-10 have been defined as follows [9, 12]:

- To acquire the experience of HTGR design, construction and operation.
- To carry out the irradiation tests for fuel elements.
- To verify the inherent safety features of the modular HTGR.
- To demonstrate the electricity/heat co-generation and steam/gas turbine combined cycle.
- To develop the high temperature process heat utilization.

The HTR-10 is a pebble bed HTGR utilizing spherical fuel elements containing ceramic coated fuel particles. The reactor core has a diameter of 1.8 m, a mean height of 1.97 m and the volume of 5.0 m^3, and is surrounded by graphite reflectors. The core is composed of about 27,000 fuel elements using low enriched uranium with a designed mean burn-up of 80 000 MWd/MtU. Helium coolant temperatures at the core inlet and outlet are 250°C and 700°C respectively. The pressure of the helium coolant in the primary system is 3.0 MPa. Design characteristics of the HTR-10 are summarized in Table II.

TABLE II

HTR-10 Design Characteristics

Reactor thermal power	10 MW
Primary helium pressure	3.0 MPa
Active core volume	5 m^3
Reactor core diameter	180 cm
Average core height	197 cm
Average helium temperature at reactor outlet	700 °C
Average helium temperature at reactor inlet	250 °C
Helium mass flow rate at full power	4.3 kg/s
Main steam pressure at steam generator outlet	4.0 MPa
Main steam temperature at steam generator	440 °C
Feed water temperature	104 °C
Fuel-to-graphite ball ratio	0.57 / 0.43
Number of control rods in side reflector	10
Number of absorber ball units in side reflector	7
Nuclear fuel	UO2
Heavy metal loading per fuel element	5 g
Enrichment of fresh fuel element	17%
Number of fuel elements in equilibrium core	27,000
Fuel loading mode	multi-pass

Graphite serves as the main structural material and is located in the top, sides, and bottom reflector. The side reflectors are 100 cm thick. Cold helium flow channels are integrated into the side reflectors to allow the primary coolant to flow upward after entering the reactor vessel. To achieve a downward flow pattern into the pebble-bed, the helium reverses flow at the top. Helium then enters a hot-gas chamber in the bottom reflector after being heated in the pebble-bed and from then on to the heat exchanging components. The core is made up of the German type spherical fuel elements of radius 6.0 cm with TRISO coated fuel particles embedded in a graphite matrix.

The coated fuel particle consists of a UO_2 kernel with diameter of 0.5mm, which is coated with a buffer layer of low density pyrolytical carbon (PyC), an inner layer of high density PyC, a layer of silicon carbide (SiC), and outer high density PyC layer. Each fuel element has about 8 300 coated fuel particles and contains 5.0 g uranium with ^{235}U enrichment of 17%. Figure 1 a. and b. show the body centered cubic (BCC) lattice block. Center sphere is the fuel pebble in blue and the surrounding green corner spheres are moderator dummy pebbles. Figure 1.b has the front quarter slice cut out to shows the homogenized fuel region in pink and graphite region in blue.

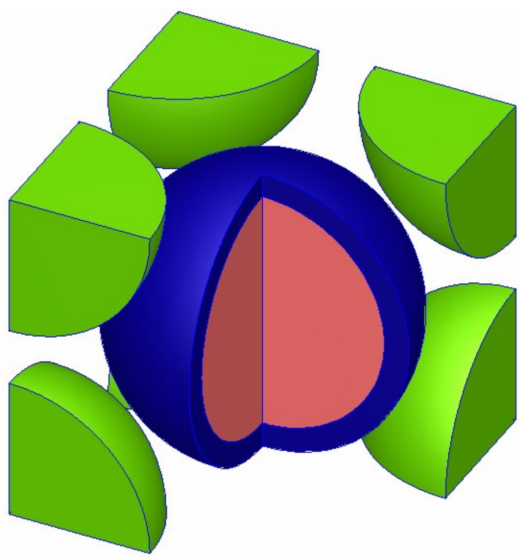

Fig. 1. KENO lattice stacking (2 x 1 x 2) of BCC blocks.

The pebble blocks as shown in Fig.1 are stacked in an array to form the cylindrical core. Fig 2. displays the cylindrical core after more fuel is added.

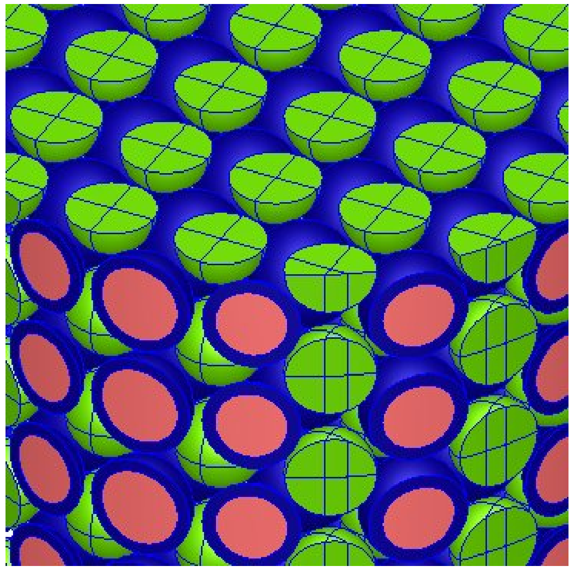

Fig 2. Stacking of BCC blocks into cylindrical HTR-10 core.

The top view of the core is in Fig 3. The white region is the homogenized fuel region which consists of UO_2 microparticles inside a graphite matrix. The blue region surrounding it is the outer graphite layer in the pebble and the yellow is the dummy ball.

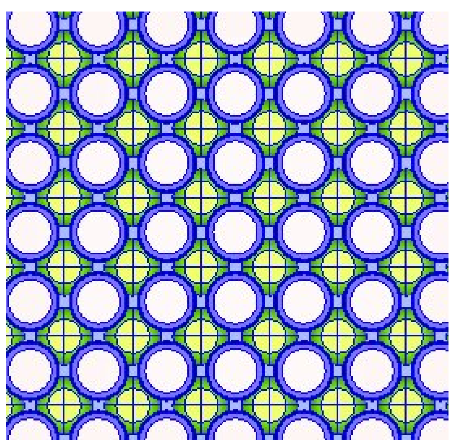

Fig. 3 Top view of HTR-10 core fuel loading.

HTR-10 Benchmark Problems

The benchmark problems performed for the HTR-10 in this report include reactor core physics evaluation for initial criticality, and initial and full core control rod worth. The core physics model includes structures only until carbon bricks. Figure 4 provides the two dimensional KENO VI reactor physics calculation model with various colors identifying different material regions.

Experimental benchmark results are available for the initial criticality problem at 288 K and code-to-code benchmark results are available for both the effective multiplication factor benchmark and the full core control rod worth.

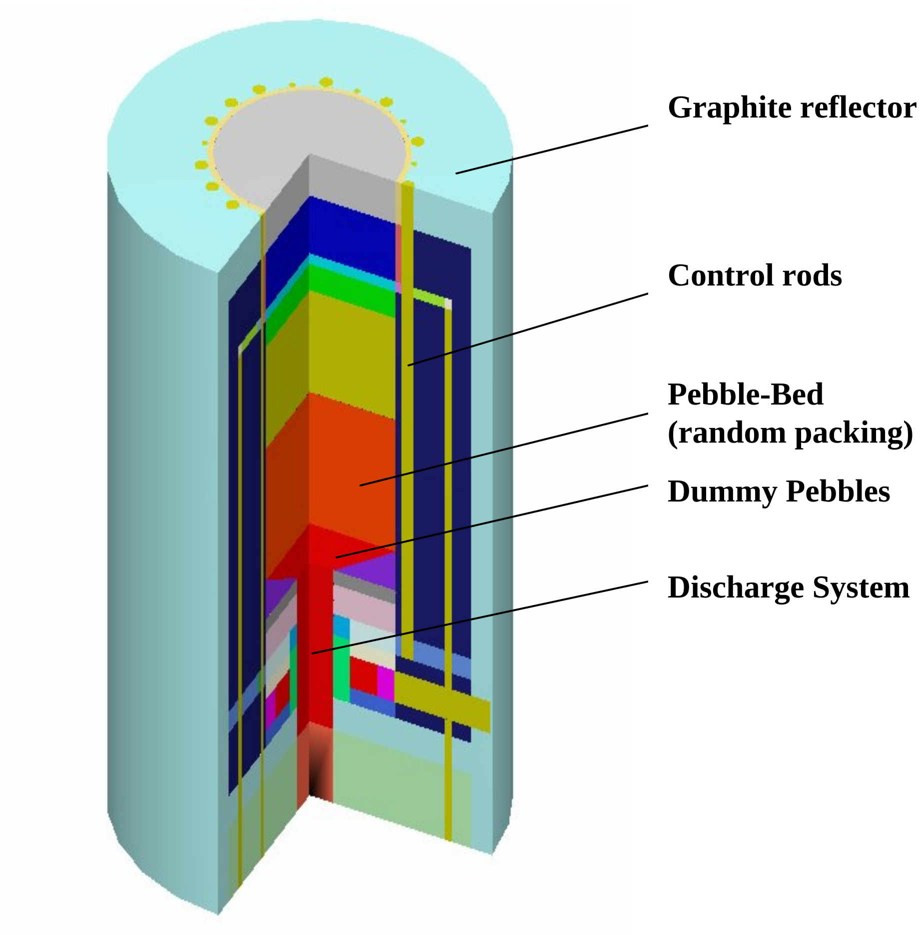

Fig. 4 KENO VI HTR-10 reactor physics calculation model for validation purposes.

Initial Criticality

The initial criticality is defined as the loading height at which criticality: $k_{eff} = 1.0$ occurs and is measured from the upper surface of the conical region. For this benchmark problem, the core is under the atmosphere of helium at 1 atm and 288 K uniform core temperature with no inserted control rods.

TABLE II

Initial Criticality Core Loading Height

Benchmark	VHTR (Heterogeneous)	VHTR (Homogeneous)	HTR-10 (experimental)	HTR-10 (China, calculated)
Critical Level (cm)	47.6	72.6	123.06	126.1

Full Core Effective Multiplication Factor

The effective multiplication factor, k_{eff}, for the full core was evaluated at 300, 393, and 523 K. For this calculation the core is under a helium atmosphere without control rods inserted.

TABLE III

Full Core Effective Core Multiplication Factors

Benchmark	VHTR (Homogeneous)	HTR-10 (China, calculated)
Fully Loaded Core (k-eff)		
Temp=20 °C	1.2558 ± 0.0018	1.1358
Temp=120 °C	1.2349 ± 0.0019	1.1262
Temp=250 °C	1.2222 ± 0.0028	1.1111

Full Core Control Rod Worth

This benchmark problem calculates the reactivity worth of the ten fully inserted B_4C control rods and one fully inserted control rod. The core is under helium atmosphere and an isothermal whole core temperature of 300 K.

3. Spectrum Shifting Through Dummy Graphite Pebble Mixing Adjustments

Pebble-bed cores exhibit possible flexibility in component configuration. This enables improvement of fissile properties of minor actinides (as shown in Table I) as a result of neutron spectrum shifting. This makes it possible for certain actinides such as neptunium, americium and curium to serve as fuel materials or burnable poisons over prolonged irradiation periods. By observing basic characteristics of actinide fuel components, the desirable spectral shifts can be visualized.

TABLE IV

Various Moderator to Fuel Ratio Results for 300, 393, 523, and 1000 K

Moderator-to-Fuel Pebble Ratio	Carbon-to-Fuel Atom Ratio	Dancoff Factor	Reactivity (%)	Average Energy Inducing Fission (eV)	Mean Free Path (cm)
Isothermal whole core criticality calculations at 300 K					
0/1 (0.000)	735.9	0.24676	+5.7404 ± 0.584	0.07224 ± 4.955E-04	3.497 ± 9.243E-03
1/3 (0.333)	976.3	0.24612	+2.4200 ± 0.712	0.06190 ± 3.557E-04	3.480 ± 9.953E-03
1/1.3 (0.750)	1277.4	0.24577	-3.1999 ± 0.485	0.05444 ± 3.482E-04	3.481 ± 7.132E-03
2/1 (2.000)	2179.9	0.24536	-19.6745 ± 0.802	0.04499 ± 3.108E-04	3.488 ± 5.233E-03
4/1 (4.000)	3623.9	0.24523	-47.645 ± 0.886	0.03952 ± 1.845E-04	3.488 ± 6.481E-03
Isothermal whole core criticality calculations at 393 K					
0/1 (0.000)	735.9	0.24676	+5.3210 ± 0.208	0.08365 ± 1.895E-04	3.458 ± 2.527E-03
1/3 (0.333)	976.3	0.24612	+0.5371 ± 0.269	0.07170 ± 1.905E-04	3.460 ± 2.902E-03
1/1.3 (0.750)	1277.4	0.24577	-4.5587 ± 0.209	0.06364 ± 1.097E-04	3.462 ± 2.312E-03
2/1 (2.000)	2179.9	0.24536	-22.4300 ± 0.331	0.05351 ± 1.164E-04	3.462 ± 2.807E-03
4/1 (4.000)	3623.9	0.24523	-51.6760 ± 0.197	0.04821 ± 6.741E-05	3.460 ± 1.999E-03
Isothermal whole core criticality calculations at 523 K					
0/1 (0.000)	735.9	0.24676	+4.3062 ± 0.191	0.10002 ± 1.986E-04	3.428 ± 2.333E-03
1/3 (0.333)	976.3	0.24612	-0.4016 ± 0.281	0.08712 ± 1.927E-04	3.434 ± 3.121E-03
1/1.3 (0.750)	1277.4	0.24577	-6.5871 ± 0.181	0.07818 ± 1.373E-04	3.435 ± 2.318E-03
2/1 (2.000)	2179.9	0.24536	-25.6913 ± 0.264	0.06701 ± 1.157E-04	3.442 ± 2.733E-03
4/1 (4.000)	3623.9	0.24523	-56.2744 ± 0.219	0.06104 ± 7.862E-05	3.436 ± 1.955E-03
Isothermal whole core criticality calculations at 1000 K					
0/1 (0.000)	735.9	0.24676	+0.299 ± 0.179	0.16297 ± 2.721E-04	3.376 ± 2.177E-03
1/3 (0.333)	976.3	0.24612	-5.2742 ± 0.221	0.14606 ± 2.601E-04	3.387 ± 2.965E-03
1/1.3 (0.750)	1277.4	0.24577	-12.9944 ± 0.215	0.13387 ± 2.087E-04	3.385 ± 2.161E-03
2/1 (2.000)	2179.9	0.24536	-37.0614 ± 0.302	0.11899 ± 1.729E-04	3.386 ± 2.232E-03
4/1 (4.000)	3623.9	0.24523	-77.3993 ± 0.231	0.11106 ± 1.143E-04	3.380 ± 1.725E-03

By mixing dummy graphite pebbles and fuel pebbles accordingly, specific spectral characteristics can be varied. Figure 5 displays the effect of varying fuel loadings on neutron flux. The inset is a close up of the fast region from $1x10^5$ eV upward.

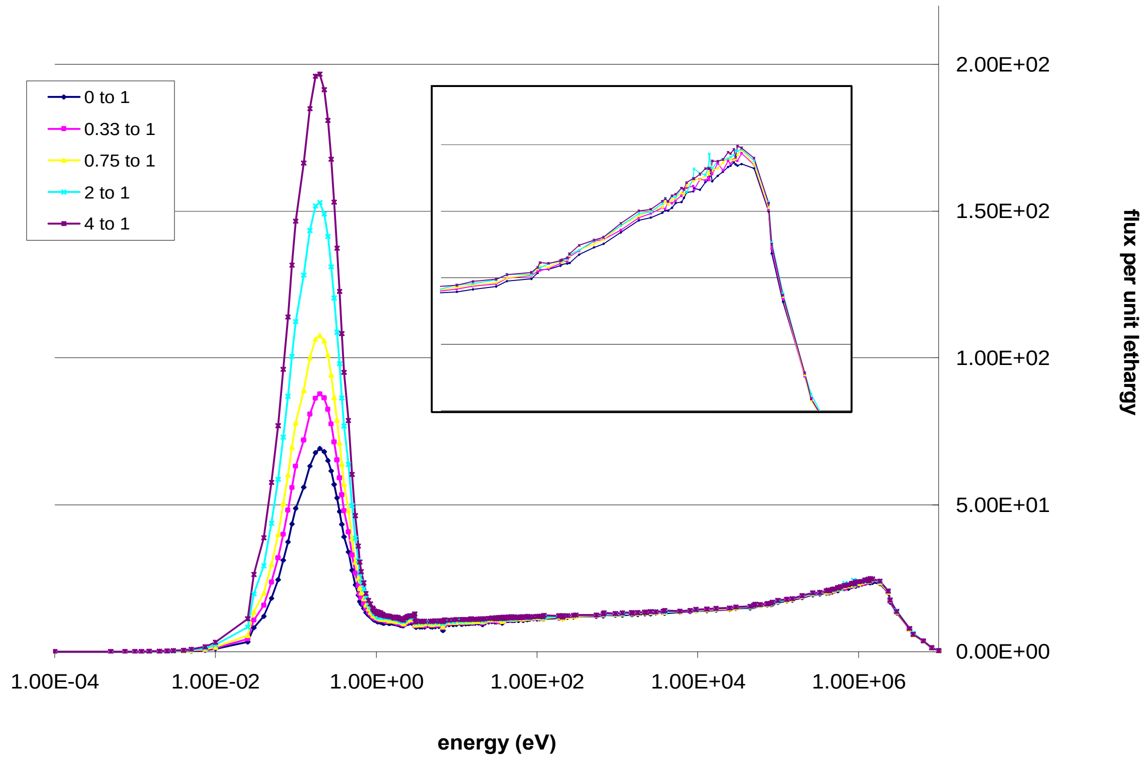

Fig. 5. Neutron flux per unit lethargy as a function of energy at 393 K for various fuel loadings.

In order to verify the use of any temperature in further calculations, the moderator to fuel ratio is fixed and the calculation is carried out at three temperatures including 300 K, 523 K, and 1000 K. Figure 6 shows the neutron flux profile for a zero to one (0/1) moderator to fuel ratio at the three specified temperatures.

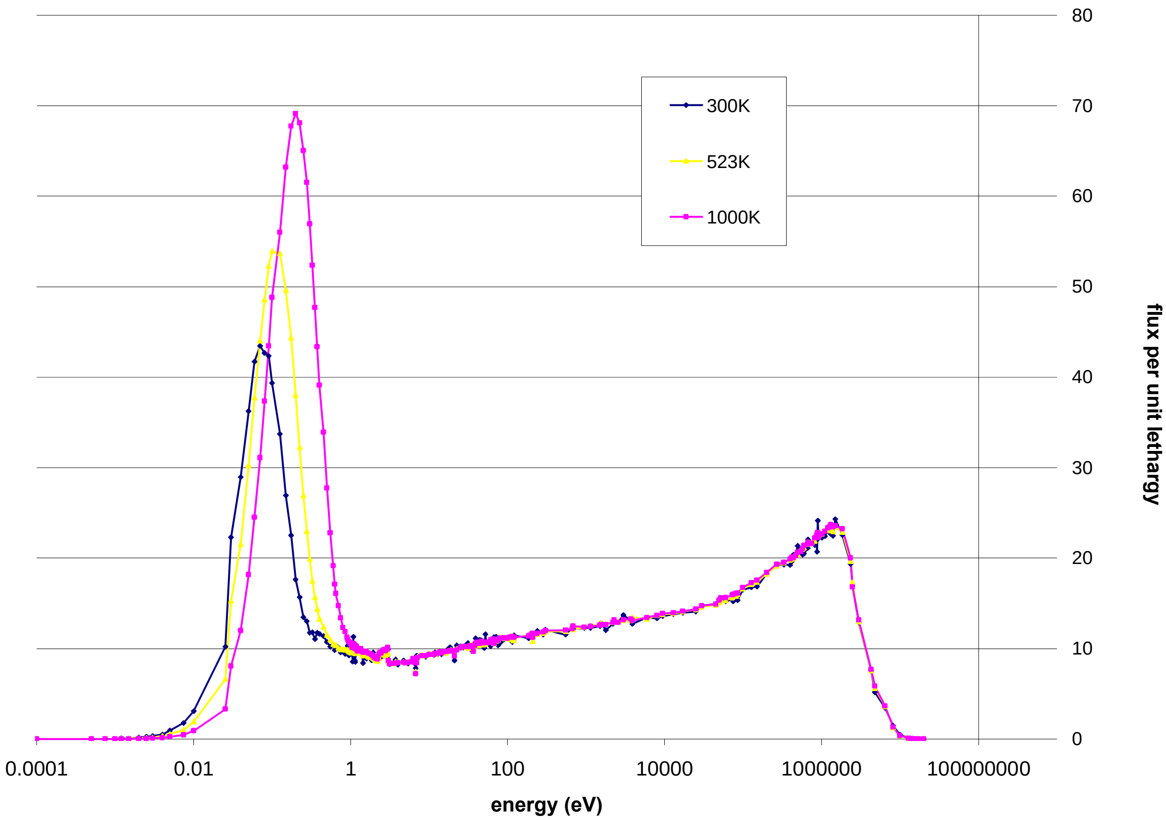

Fig 6. Neutron flux per unit lethargy as a function of energy for 0/1 moderator to fuel ratio for three temperatures.

4. Potential Applications of the Pebble-Bed VHTR Spectrum Shifting Capabilities

Minor Actinide Waste Reduction

Partitioning and transmutation of minor actinides is a difficult process but could serve as a major advance in the global nuclear fuel cycle by decreasing the amount of hazardous nuclear waste for disposal. [7] The current process of recycling minor actinides from the closed fuel cycle of the light water reactor (LWR) is a waste management strategy utilized in various countries other than the U.S. including France and Japan.

The pebble-bed reactor has unique fuel which is designed to withstand very high burn-up levels. In transmutation fuel, the larger burn-up values correspond to less reprocessing steps to achieve that high transmutation rate. If burn-up levels are high enough, a large percentage of the minor actinide components originally in the fuel are transmuted. Through spectrum shifting, which is easy and achievable in the pebble-bed configuration, the wastes of the reactor fuel can be maneuvered to fit a desired outcome. Based on nuclear characteristics, the spectrum can shifted to accommodate minimization of a less desired nuclide species. [7, 11]

5. Dancoff Factor Calculations

Purpose of Dancoff Factor

When lattice codes are applied to a unit cell in a pebble-bed high temperature reactor, special attention is required to account for double heterogeneity. [8] The unit cell consists of two regions within the pebble: the center fuel region which is itself a lattice of fuel microparticles and the outer graphite region. The outer layer of graphite is 0.5 cm thick and the center fuel region of diameter 4.5 cm consists of randomly packed UO_2 grains with a pitch of ~ 2 mm.

The pile of pebbles can be considered heterogeneous. The first heterogeneity is the fuel kernel surrounded by protective coating layers and a graphite matrix. The second heterogeneity is the inner fuel zone of the pebble ($R< 2.5$) and the outside graphite layer ($2.5 < R < 3.0$).

A single fuel region within an infinite moderator region will obviously not collide with anything but moderator. However, if the distance between the regions is not large compared to the neutron mean free path of the moderator, there is a possibility that a neutron escaping a fuel regions will collide with another fuel region without moderator interaction.

Dancoff Factor Calculation

The resonance self-shielding calculation is performed for a single fuel element located in an infinite moderator, in this case graphite, in most code systems. [8] The Dancoff correction takes into account the presence of other fuel particles not accounted for in the infinite medium.

On the smallest geometrical scale which is represented by the fuel kernel surrounded by various layers, the group cross sections are first prepared. This is then translated into a unit cell so that resonance absorptions can be calculated. The resonance absorption should be calculated with the collision probability method on the basis of microscopic lattice fuel grains, and the Dancoff factor used in resonance calculations depends on the grain-lattice geometry.

The Dancoff factor is calculated by averaging the Dancoff factors of individual fuel particles. To calculate Dancoff factors in irregular geometries, codes have been developed such as DANCOFF-MC which is a Monte Carlo based method using arbitrary arrangements of spherical fuel elements. DANCOFF-MC calculates the Dancoff factor based on its definition that a neutron emitted isotropically from the fuel lump of a fuel element enters the fuel lump of another element without a collision with any other nuclei.

Dependence of Reactor Core Multiplication Factor on Dancoff Factor

Choosing the Dancoff factor correctly for a pebble-bed configuration is extremely important. It can be seen in Figure 7 that as the Dancoff factor is altered from 0.2 to 0.8, the effective multiplication factor is altered by up to two percent. This difference in the multiplication factor displays the importance of choosing the correct Dancoff factor for code input.

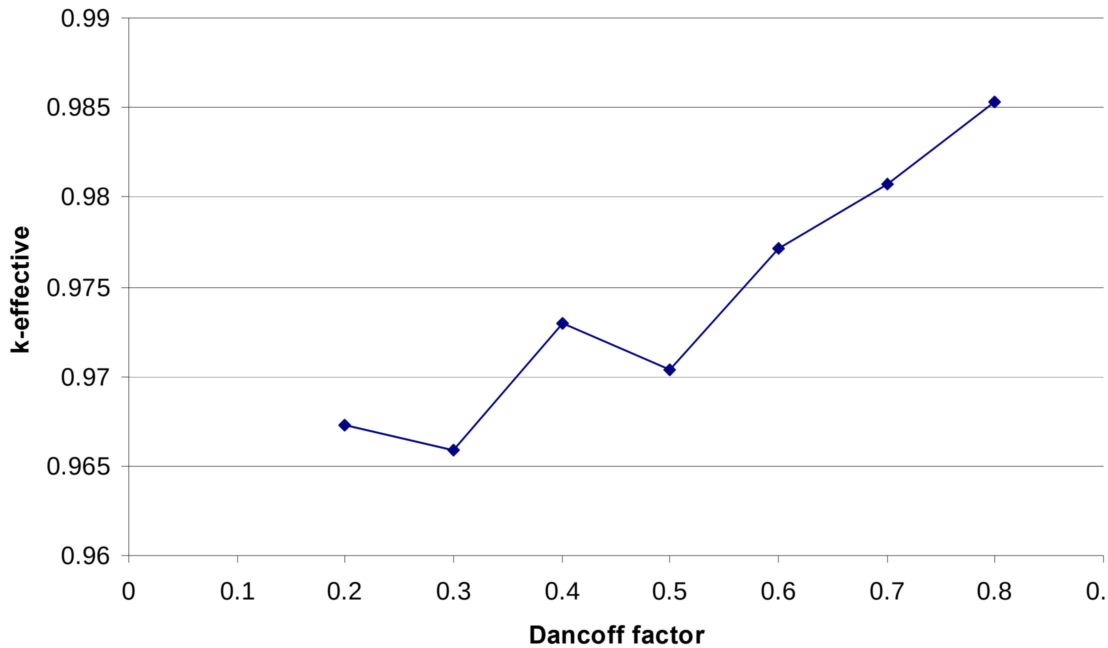

Fig. 7 Dependence of core multiplication factor in the HTR-10 core on Dancoff factor.

The double heterogeneity in the pebble creates this drastic effect on the multiplication factor from the Dancoff factor. This dependence is not as drastic in other VHTR systems such as the prismatic core configuration.

6. Conclusions

Obtained benchmark results are in agreement with the available LEU-HTR PROTEUS and HTR-10 data. Since it has been observed that, although correctly predicting reactor physics characteristics of the pebble-bed cores, the applied models underestimate the critical core height, the configurations will be investigated to evaluate the effect further.

Since the pebble-bed design allows flexibility in configuration, fuel utilization, and fuel management, it is possible to improve fissile properties of minor actinides by neutron spectrum shifting through configuration adjustments. As a result, the small reactivity swings after long periods of irradiation in the VHTR will yield high levels of burn-up. As minor actinides are converted to fissile nuclides with burn-up, the reactor is able to sustain nuclear reactions for a longer period of time.

This research effort enhances capabilities of the Generation IV VHTR and transforms it to a technology that can deliver electricity, hydrogen, and assist in spent fuel treatment while being inherently safe, environmentally friendly, and proliferation resistant. This technology also displays promising aspects in economical analyses due to high burn-up, autonomous operation, and actinide recycling.

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8. Appendix

Fuel Element Characteristics	
Diameter of ball	6.0 cm
Diameter of fuel zone	5.0 cm
Density of graphite in matrix and outer shell	1.73 g/cm³
Heavy metal (uranium) loading (weight) per ball	5.0 g
Enrichment of U-235 (weight)	17%
Equivalent natural boron content of impurities in uranium	4 ppm
Equivalent natural boron content of impurities in graphite	1.3 ppm
Volumetric filling fraction of balls in the core	0.61
Fuel kernel	
Radius of the kernel	0.025 cm
UO_2 density	10.4 g/cm³
Coatings	
Coating layer materials (starting from kernel)	PyC / PyC / SiC / PyC
Coating layer thickness (mm)	0.09 / 0.04 / 0.035 / 0.04
Coating layer density (g/cm³)	1.1 / 1.9 / 3.18 / 1.9
Dummy (no fuel) elements	
Diameter of ball	6.0 cm
Density of graphite	1.73 g/cm³
Equivalent natural boron content of impurities in graphite	1.3 ppm
Additional Reactor Core Parameters	
Density of reflector graphite	1.76 g/cm³
Equivalent natural boron impurity in reflector graphite	4.8366 ppm
Density of boronated carbon brick including B_4C	1.59 g/cm³
Weight ratio of B_4C in borated carbon brick	5%

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