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Conceptual Reactor Core Design of a Heat Pipe Cooled Reactor using MCS

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1. Introduction

Reliable and sustainable power sources are essential for remote locations such as lunar outposts, isolated cities, small industrial bases, and military installations, where continuous power supply is required despite extreme environmental conditions and limited maintenance capabilities [1]. Conventional energy sources, including solar and battery systems, often struggle to meet these demands due to their dependence on sunlight, low energy density, limited operational lifespan, and maintenance challenges. To address these challenges, nuclear power has emerged as a viable solution, offering high energy density, long operational lifetimes, and minimal reliance on external resources.

The heat pipe cooled reactor (HPR) presents a highly promising technology for such applications. HPR is a compact and durable nuclear reactor that utilizes heat pipes to efficiently transfer heat from the reactor core to an external heat exchanger, minimizing the need for mechanical pumps or other active cooling components [2]. This passive cooling mechanism significantly enhances system reliability by reducing failure risks associated with moving parts. Furthermore, the heat pipe system operates effectively in low-gravity environments, making it well-suited for lunar outpost [2]. At the same time, its scalability and robustness enable its application in remote cities, small industrial facilities, and military installations where consistent and independent power generation is essential.

Recently, the STRIMS (Sejong TRISO-fueled Molten Salt filled) HPR concept has been proposed by the Sejong University [3]. In this paper, the STRIMS HPR core design optimized for lunar exploration, isolated cities, remote infrastructure, and military applications is introduced. Additionally, the k_{inf}, neutron flux spectrum, and radial power distribution of the developed HPR core, calculated using MCS, are presented. The proposed HPR employs tristructural isotropic (TRISO) fuel and FLiBe (Li_2BeF_4) fluoride salt coolant, based on the design principles of the generic pebble bed fluoride-salt-cooled high-temperature reactor (gFHR) developed by Kairos Power [4]. MCS, a high-fidelity Monte Carlo simulation code developed and maintained by the Ulsan National Institute of Science and Technology (UNIST), has been enhanced to support the modeling and simulation of TRISO fuel and pebble bed reactor configurations [5].

2. Modeling

This section describes the full-core STRIMS HPR core model designed in this study. In this reactor core model, pebbles and TRISO particles are randomly distributed at each location, and MCS can simulate this distribution.

2.1 Reactor Core Design

Table I presents the design parameters of STRIMS HPR core. The core radius and height are 52 cm and 152 cm, respectively, with a thermal power of 0.145 MW. The core uses U-235 enrichment of 19.5 wt.% and FLiBe fluoride salt as the coolant. Inside the heat pipes, which have an inner diameter of 2.6 cm, sodium exists in both liquid and vapor phases. To utilize HPR for a lunar exploration, reducing its weight is essential due to the extreme difficulty of transportation. The nuclear fuel used in this core weighs 41.35 kg.

Table I: STRIMS HPR core design parameters

Parameter	Value
Cycle Length [years]	10
Power [MWt]	0.145
Power Density [W/g]	3.5
Core Radius [cm]	50
Core Height [cm]	152
Fuel Material	UO_2
U-235 Enrichment [wt.%]	19.5
Fuel Mass [kg]	41.35
Pebble Radius [cm]	2.0
Fluoride Salt Coolant	FLiBe
Average Coolant Temperature [K]	923.15
Number of Heat Pipes	69
Heat Pipe Fluid Material	Sodium
Average Heat Pipe Fluid Temperature [K]	898.15
Inner Diameter of Heat Pipe [cm]	2.6
Outer Diameter of Heat Pipe [cm]	3.0

Fig. 1 shows the radial and axial layout of STRIMS HPR core modeled using MCS. As shown in the radial layout, pebbles are randomly loaded in the region excluding the heat pipes, with a pebble packing fraction of 0.35. The space excluding the heat pipes and pebbles is filled with FLiBe coolant. A total of 69 heat pipes are symmetrically arranged. Some pebbles overlapping with the heat pipes are partially truncated. This issue is planned to be addressed in the future by improving the MCS to enable more accurate modeling. Since the appropriate reflector material and thickness for HPR have not been determined, the outer edge of the core is defined as a reflective boundary for all neutrons.

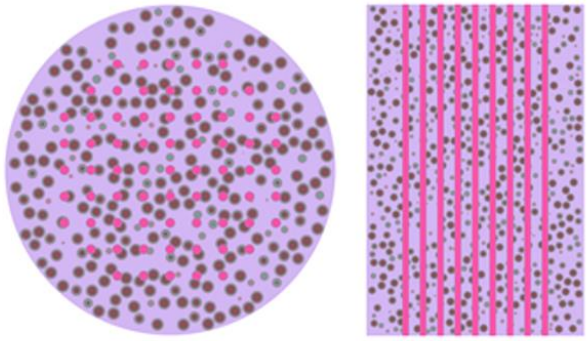

Fig. 1. Reactor core model in the radial layout (left) and axial layout (right)

Refueling nuclear fuel in the lunar environment, isolated cities and military installations is challenging, so the cycle length of the STRIMS HPR core is set to a target of 10 years or more. In the reactor core simulation, thermal-hydraulic calculations are not performed. Therefore, the temperatures of the FLiBe coolant and the sodium fluid inside the heat pipe are set to average values of 923.15 K and 898.15 K, respectively.

Table II compares the power densities in units of W/g and W/cc of the gFHR, the STRIMS HPR developed in this study, pebble bed modular reactor (PBMR) and high temperature enginee-ring test reactor (HTTR), which are types of high temperature gas cooled reactor.

Table II: Comparison of the power density of reactor cores

Reactor Core Model	Power Density	
	W/g	W/cc
gFHR	332.88	20.00
PBMR	0.91	4.78
HTTR	33.35	2.49
STRIMS HPR	3.5	0.13

2.2 TRISO Fuel and FLiBe Coolant

The geometric structure and material composition of the TRISO and the pebble used in this STRIMS HPR core are shown in Table III and Table IV, respectively. As shown in Fig. 2, the TRISO are randomly distributed in the fuel layer region of the pebble. The volume fraction occupied by the TRISO within the fuel layer region of the pebble is 22 %, i.e., the packing fraction of the TRISO is 0.22. Furthermore, the volume fraction occupied by the pebble within the reactor core region is 35 %, i.e., the packing fraction of the pebble is 0.35. TRISO and pebble have a radius of 0.04275 and 2.0 cm, respectively. The carbon present in TRISO and pebble functions as a neutron moderator and has high thermal conductivity, effectively transferring the heat generated in the nuclear fuel to the coolant.

Table III: Geometric structure and material composition of the TRISO

TRISO			
Parameters	Radius [cm]	Material	Density [g/cm3]
Kernal (fuel)	0.02125	UO_2	10.50
Graphite Buffer	0.03125	Carbon	1.05
Inner PyC	0.03525	Carbon	1.90
SiC	0.03875	SiC	3.18
Outer PyC	0.04275	Carbon	1.90
Packing Fraction: 0.22			

Table IV: The geometric structure and material composition of the pebble

Pebble			
Parameters	Radius [cm]	Material	Density [g/cm3]
Inner Shell	1.38	Carbon	1.41
Fuel Layer	1.80	TRISO and Carbon	1.74
Outer Shell	2.00	Carbon	1.74
Packing Fraction: 0.35			

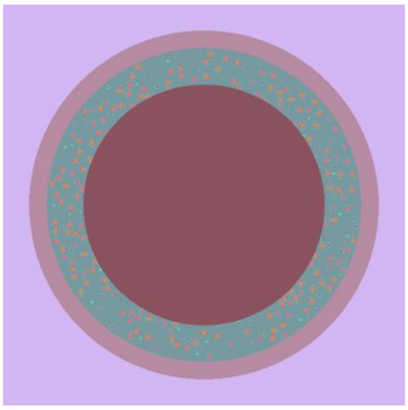

Fig. 2. Single pebble containing TRISO

TRISO nuclear fuel is designed for Generation-IV reactors like high-temperature gas-cooled reactors. It is made up of minuscule spherical fuel particles, each encased in several protective layers. These layers effectively contain fission products, drastically minimizing the risk of radioactive leaks. TRISO fuel also exhibits superior stability, even at extremely high temperatures (over 1800 K), compared to traditional nuclear fuels [6]. The TRISO fuel used in the gFHR comes in spherical form with a radius of 0.02125 cm and uses uranium-235 enriched to 19.75 wt.%. The fuel particles are coated with multiple layers of carbon and silicon carbide (SiC) for enhanced protection. These TRISO fuel particles are incorporated into a fuel layer region of the pebble. This pebble has a core of low-density carbon surrounded by a fuel layer. This fuel layer consists of a blend of TRISO fuel particles and carbon. The entire pebble is then encased in a final outer shell of carbon. The low-density carbon core is crucial for ensuring the pebble's overall density is similar to that of the FLiBe coolant used in the reactor. This prevents the pebble from floating or sinking in the FLiBe coolant.

Naturally occurring lithium consists of two stable isotopes, lithium-7 (approximately 92.41%) and lithium-6 (approximately 7.59%). However, because lithium-6 has a significantly larger neutron absorption cross-section compared to lithium-7, FLiBe coolant uses lithium enriched to 99.995% lithium-7 to minimize neutron losses. FLiBe is a molten salt that offers several attractive features as a coolant, particularly in advanced reactor core designs like molten salt reactors. Its high boiling point allows for high-temperature operation without the need for pressurization, leading to increased thermal efficiency and simpler reactor designs. Additionally, FLiBe coolant is chemically stable, has a low neutron absorption cross-section, and possesses a high heat capacity, making it a promising coolant for Generation-VI reactors [4, 7]. In other words, the FLiBe coolant serves the dual purpose of moderating neutrons generated by nuclear fission and transferring the heat produced in the pebbles to the heat pipes.

3. Results

This section describes the burnup simulation results using MCS for the STRIMS HPR core model developed in this study. The standard deviation of MCS simulation results is approximately 30 pcm for k_{inf}.

3.1 HPR Burnup Simulation Results using MCS

Fig. 3 shows the k_{inf} of the STRIMS HPR core model over a 12-year burnup.

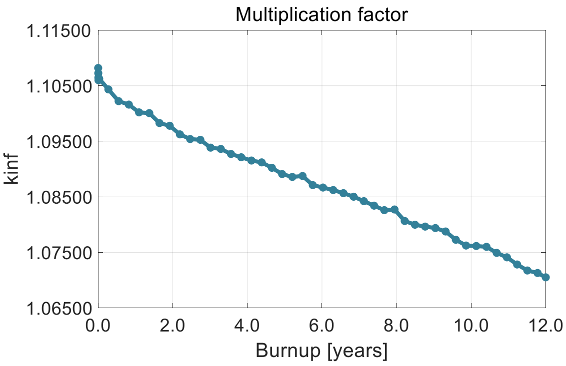

Fig. 3. k_{inf} during burnup simulation for STRIMS HPR core model

The excess reactivity at beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) is 10819, 8667, and 7051 pcm, respectively. It is important to note that in the simulation performed in this study, the outer edge of the core is defined as a reflective boundary for all neutrons. With the addition of a reflector, a slight decrease in the k_{inf} and a shortening of the cycle length will be observed.

Table V presents the change in the average U-235 number density across the entire fuel region at BOC, MOC, and EOC. During a 12-year burnup simulation, the average uranium-235 number density decreases by 9.67 % relative to its initial value at BOC.

Table V: U-235 number density change with burnup simulation

Depletion Point	U-235	
	Average Number Density [#/barn-cm]	Rate of Change [%]
BOC (0 years)	4.62314E-03	-
MOC (6 years)	4.39375E-03	-4.96
EOC (12 years)	4.17603E-03	-9.67

In Fig. 4, the orange and gray lines represent the normalized neutron flux spectrum of the STRIMS HPR core and the OPR1000, respectively. The relative standard deviation of the neutron flux spectrum calculated by MCS is approximately 0.1%. The STRIMS HPR core has a higher neutron distribution in the thermal energy region compared to a commercial pressurized water reactor (PWR). This is primarily due to the effective moderation of fast neutrons by the carbon surrounding the UO_2 fuel in TRISO and pebbles, as well as the FLiBe coolant, as shown in Table III and IV. A higher neutron distribution in the thermal energy region provides several advantages, such as improved neutron economy, enhanced safety and more effective reactivity control using control rods.

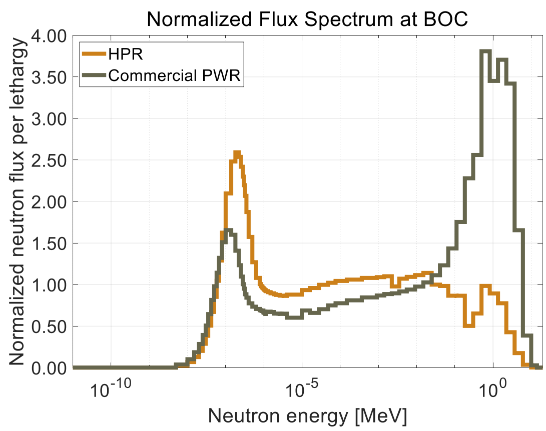

Fig. 4. Normalized flux spectrum for STRIMS HPR core model at BOC

The difference in neutron flux spectrum between the STRIMS HPR and OPR1000 also affects neutron leakage and reflector efficiency. Since thermal neutrons have lower diffusion lengths compared to fast neutrons, neutron losses at the core boundary may be lower in the HPR core. This could reduce the need for extensive neutron reflectors, contributing to a more compact reactor system. Future optimizations may figure out the use of alternative reflector materials or structural modifications to enhance neutron economy and power flattening.

Fig. 5 illustrates the radial power distribution at BOC. The core is divided into 20 equally spaced rings, and the power distribution values are normalized so that the average is 1.0. The relative standard deviation of the radial power distribution calculated by MCS is approximately 0.5% at every ring. The outer region without heat pipes exhibits higher power, and this distribution trend remains consistent throughout BOC, MOC, and EOC.

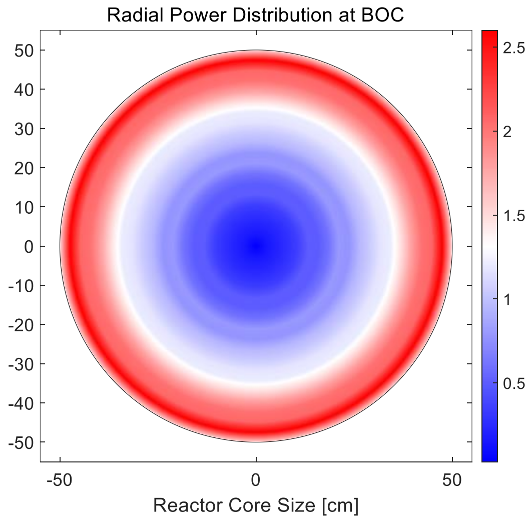

Fig. 5. Radial power distribution for STRIMS HPR core model at BOC

As shown in Fig. 5, the arrangement of heat pipes affects the radial power distribution. Therefore, when designing the heat pipe layout, it is important to consider the radial power distribution to ensure that the power peaks do not become excessively high.

4. Conclusions

In this study, a conceptual STRIMS HPR core was designed to be suitable for applications in lunar bases, remote cities, small industrial bases, and military installations. The HPR core was designed with a radius of 50 cm and a height of 152 cm, generating a thermal power of 0.145 MW. The core adopted TRISO and pebble nuclear fuel, as used in the gFHR developed by Kairos Power, and employed FLiBe fluoride salt as the coolant. In addition, 69 heat pipes with an inner diameter of 2.6 cm are arranged in the core, and liquid or gaseous sodium is present inside the heat pipes.

A 12-year burnup simulation was conducted using MCS, and key reactor characteristics were analyzed. The simulation results included the k_{inf}, the normalized neutron flux spectrum, the radial power distribution, and changes in U-235 number density. The results showed that the reactor maintained sufficient reactivity over the operational period, with gradual fuel depletion reflected in the reduction of U-235 number density. Additionally, the neutron flux spectrum indicated effective moderation by the carbon-based fuel structure and the FLiBe coolant. The radial power distribution demonstrated a stable and predictable power profile throughout the burnup cycle.

For future research, further design improvements will focus on the incorporation of a neutron reflector, radiation shielding, and control rods to enhance the safety and operational efficiency of the HPR system. These additions will play a crucial role in controlling excess reactivity and ensuring long-term reliability for deployment in extreme environments. Moreover, further investigations into alternative fuel compositions, such as uranium-carbide (UC) fuels, could provide additional benefits in terms of higher thermal conductivity and improved neutron economy.

Acknowledgments

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