Modeling a Thorium Fueled Molten Salt Breeder Reactor

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INTRODUCTION

In the 1960s, Oak Ridge National Laboratory (ORNL) conducted a series of experiments related to the design of a molten salt breeder reactor (MSBR). Using the results of the molten salt reactor experiments, a team of students from the University of Tennessee, Knoxville is creating and optimizing a model of a thorium-fueled MSBR. This work was completed as a part of their senior design project for the course "Nuclear System Design II."

BACKGROUND

Molten salt reactors (MSRs) offer a number of unique advantages over traditional light water reactors. MSRs are able to operate at higher temperatures and thermal efficiencies than today's reactors, while the fluid nature of their fuel makes them immune to meltdown and allows for online refueling and fission product removal. For these reasons, MSRs are a good candidate for breeder reactors, notably for a thorium fuel cycle, and are one of the reactor types being investigated for Generation IV.

From 1965-1969, Oak Ridge National Laboratory operated an experimental molten salt reactor known as the MSRE. Through the 1970s, ORNL continued the development of MSBR designs to breed ^{233}Pa, which decays to the fissile isotope ^{233}U, from fertile ^{232}Th. Some of the later designs from this research used a hexagonal-pitched graphite-moderated core with fluid channels for the molten salts [1], which was used as a basis for the design project completed by the students from the University of Tennessee.

DESCRIPTION

The reactor core consists of a number of cells similar to the one depicted in Fig. 1. Each cell contains one central and six auxiliary fuel salt channels. The "fuel salt" consists of ^{233}U dissolved in a mixture of lithium fluoride and beryllium fluoride salt, known as FLiBe. Surrounding each cell is the "blanket" of FLiBe salt containing the fertile ^{232}Th.

In order to produce as much fuel as it consumes during operation, the reactor would need to have a breeding ratio (BR) of 1. The breeding ratio was approximated as the ratio of the number neutron capture reactions by ^{232}Th and ^{234}U to the number of their fission reactions (Eq. 1):

$$BR = \frac{^{232}\mathrm{Th}_{capt} - {}^{233}\mathrm{Pa}_{capt} + {}^{234}\mathrm{U}_{capt}}{^{232}\mathrm{Th}_{fiss} + {}^{233}\mathrm{Pa}_{fiss} + {}^{234}\mathrm{U}_{fiss}} \quad (1)$$

Should ^{233}Pa capture a neutron, it will decay to ^{234}U instead of the desired ^{233}U isotope, necessitating the subtraction of its capture term from the numerator.

MCNP5 calculations were run in order to find an optimum cell geometry that would produce this ratio [2]. For the optimization, the hexagonal cell was divided into twelve symmetric unit cells reflected infinitely, as shown in Fig. 2. The geometry of the unit cell was a function of three parameters: the pitch p between cells, the ratio of the fuel salt to moderator ("fuel salt fraction", or FSF), and thickness of the blanket salt surrounding each cell ("slit width", or s).

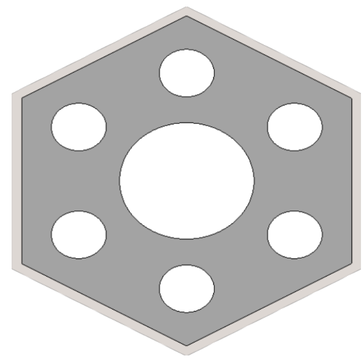


Fig. 1: Example of a hexagonal fuel cell. (White: fuel salt; Light gray: blanket salt; Dark gray: graphite)

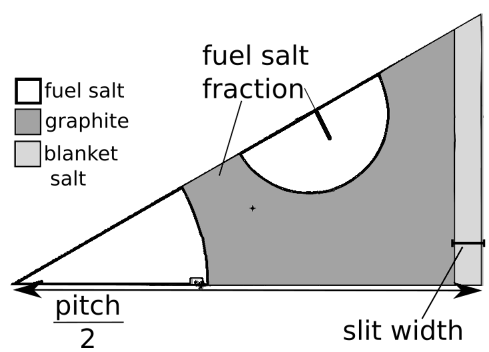


Fig. 2: Triangular unit cell for MCNP optimization.

Initially, MCNP calculations were run for a coarse mesh of the three parameters in order to establish reasonable ranges for each. When the results were restricted to cases with k_{eff} and BR within 3% of their desired values, they formed a 3D surface, as depicted in Fig. 3. This surface was then parameterized to find the required s as a function of FSF and p for criticality and fuel breeding, as shown in Fig. 4.

MCNP simulations were then run for thousands of combinations of pitch, from 5 cm to 60 cm; fuel salt fraction, from 2% to 54%; and slit width at its calculated value s as well as $0.9s$ and $1.1s$. Cases containing the highest k_∞ at a BR of greater than 0.98 were evaluated. The geometries with the ten highest values of k_∞ are listed in TABLE I in the results.

For an individual cell, it was hypothesized that the breeding ratio could be less than unity. In ORNL's design, the core

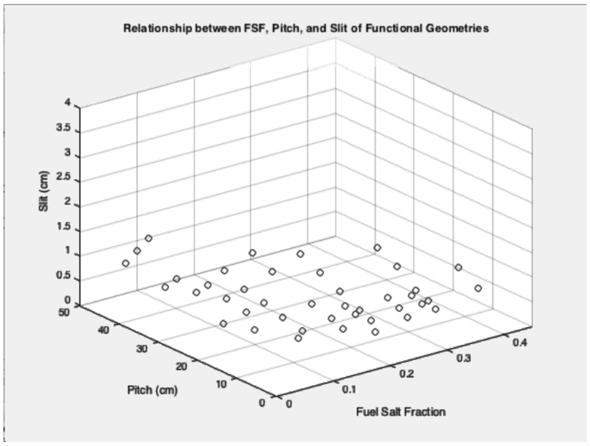


Fig. 3: Initial cases for a coarse mesh of FSF, p, and s.

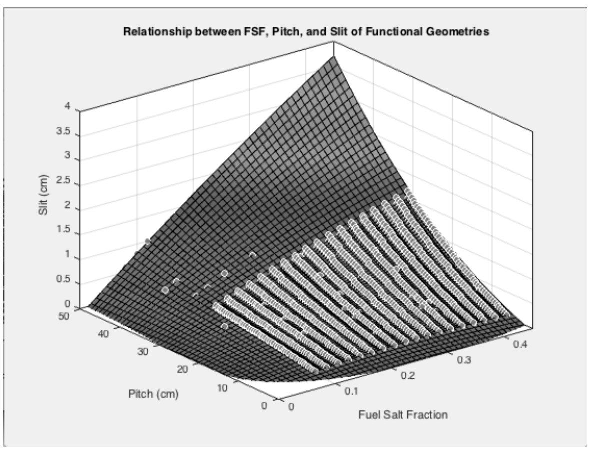


Fig. 4: Sample cases determined by the parameterized surface.

was surrounded by thick axial and radial blanket salt reflectors, as shown in Fig. 5. The blanket salt was further surrounded by several inches of a a graphite reflector. It is thought that any the vast majority of neutrons leaking from any fuel cell will end up in either adjacent cells or the blanket salt reflector, increasing the breeding ratio of the reactor as a whole. The contribution of these leaked neutrons to breeding was arbitrarily assumed to be 2% (for a BR of ≥ 0.98). Determining the validity of this assumption awaits the completion of the 3D full-core model.

One some representative dimensions for the hexagonal cells were established, development of the full-core model began. Serpent 2 was chosen over MCNP for this model due to its burnup calculation capabilities and the availability of a hexagonal lattice geometry [3]. A 2D model with a hexagonal core (shown in Fig. 6) has been created, and a 3D model with a circular core similar to the one in Fig. 5 is currently under construction.

When the model is completed, the circular core layout will be optimized to include controls and to produce the required thermal power. A power level in the range of 500 MWt is desired to comply with the NRC's definition of a small modular reactor (300 MWe or less).

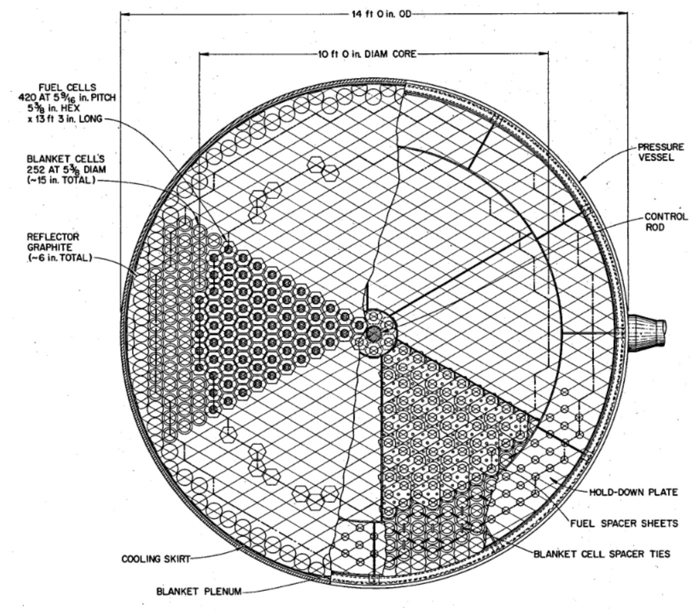


Fig. 5: Horizontal cross section of the reactor vessel from ORNL-4528 [1]. Note the radial reflector.

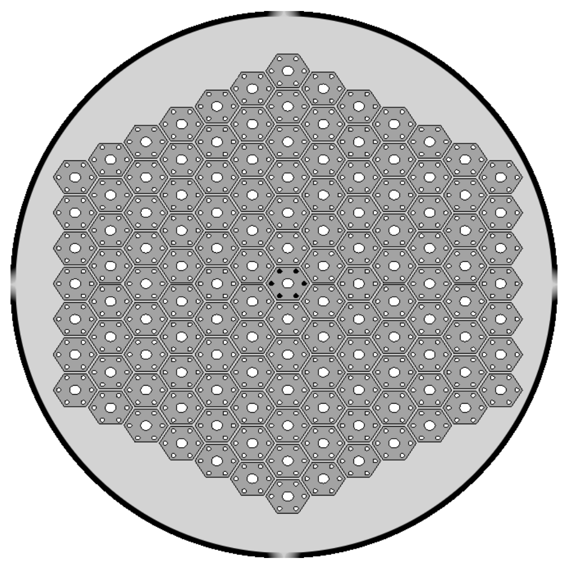


Fig. 6: Sample 2D lattice for the full-core Serpent model.

PRESENT WORK

At print, the 3D full-core Serpent model was nearly complete. This model will be used to perform burnup calculations in order to determine the steady-state isotopic composition accounting for the online refueling and fission product removal.

Once this steady-state model is complete, it is planned to be enhanced with the addition of control rod calculations and with thermal hydraulic feedback effects. Currently, the core is modeled at a fixed temperature of 900 K, which is a physically inaccurate approximation. The team is planning to couple the Serpent model with OpenFOAM to determine the temperature gradient and properly account for the thermal feedback effects.

There is also some flexibility with the exact geometry of the fuel cells used in the core. TABLE I lists the cases with the highest calculated k_{eff}. In a solid-fueled reactor, it is generally desirable to maximize this value to have the greatest period between refueling. However, since MSRs are designed for online refueling, maximizing k_{eff} is not necessarily the only goal of the optimization. It may be more economical to choose a geometry with a lower k_{eff} and simply increase the refueling rate.

Materials exposed to high neutron flux, such as in a reactor core, will experience some degradation. Increasing the pitch and decreasing the FSF of the fuel cells will result in a larger quantity of graphite moderator in the core, and consequently less degradation per unit mass. A larger pitch will therefore make the necessity of replacing graphite less frequent or unnecessary. Additionally, due to the high cost of FLiBe, a lower fuel salt fraction will make the reactor cheaper to initially construct. These factors are yet another consideration when choosing the cell's ultimate dimensions.

Furthermore, once results from the the full-core model's criticality calculations are available, the cutoff for BR will almost certainly change from 0.98. This change may either make better geometries available or render some of the existing choices unsuitable.

RESULTS

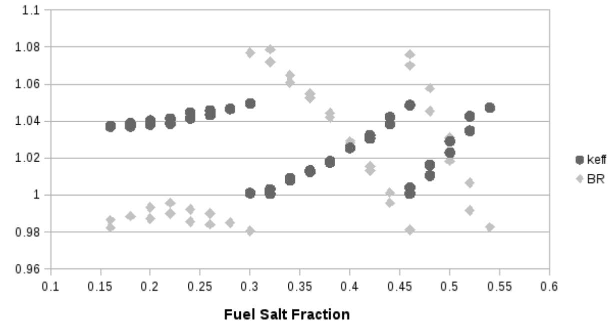
Building on research originally conducted at ORNL, the MSR project design team from the University of Tennessee was able to produce over 1,000 fuel cell geometries that appear to be a good start toward a feasible reactor design.

FSF	pitch	slit	k_∞	BR
0.34	9.5	0.274	1.0500 ± 0.0006	0.9806
0.30	11.5	0.323	1.0495 ± 0.0006	0.9807
0.32	10.5	0.300	1.0488 ± 0.0007	0.9822
0.46	12.0	0.419	1.0486 ± 0.0007	0.9812
0.48	13.0	0.469	1.0485 ± 0.0006	0.9800
0.54	12.0	0.438	1.0472 ± 0.0008	0.9826
0.44	10.5	0.349	1.0470 ± 0.0008	0.9839
0.52	14.5	0.552	1.0469 ± 0.0007	0.9810
0.48	20.0	0.804	1.0468 ± 0.0006	0.9810
0.38	22.0	0.815	1.0468 ± 0.0007	0.9807
0.28	11.5	0.312	1.0466 ± 0.0006	0.9850
0.54	15.0	0.582	1.0464 ± 0.0007	0.9806

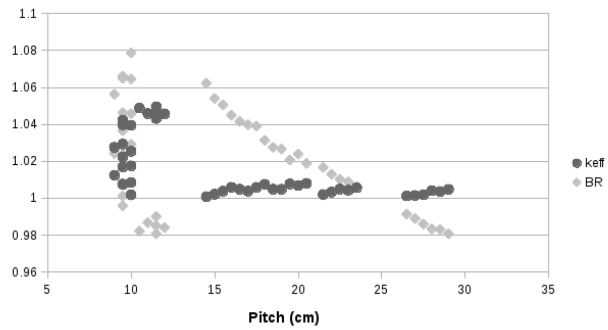
TABLE I: Top ten cases for k_∞

Based on considerations other than k_∞, it may be necessary to fine-tune the final fuel cell dimensions. Figs. 7a-c depict the relationships of s, p, and FSF with k_{eff} and the breeding ratio as determined by the MCNP model. Each plot is a function of one variable with another held constant and the third restricted to a small range of values.

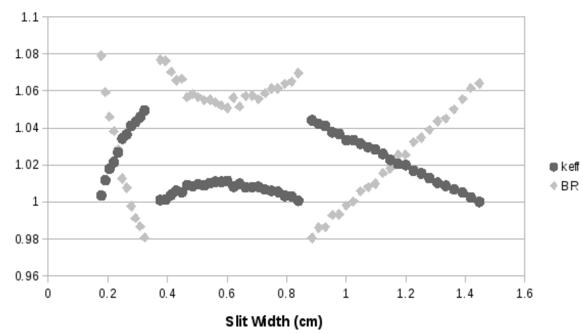
This project is still in progress. Although much work still remains to be done at the time of submission, the team has



(a) Fuel salt fraction: constant p



(b) Pitch: constant s



(c) Slit width: constant FSF

Fig. 7

so far been successful in using modern engineering tools to optimize and model a promising Generation IV reactor design.

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