Study of Helium Cooled Fast Reactor Core Design Fuelled by Thorium Carbide-Uranium Carbide with Modified Candle Axial Direction Scheme

Z Su'ud\textsuperscript{1,2,a}, N R Galih\textsuperscript{1,2b}, M Ariani\textsuperscript{3}
\textsuperscript{1} Department of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Bandung, Indonesia.
\textsuperscript{2} Nuclear Physics and Biophysics Research Division, Department of Physics, Faculty of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Bandung, Indonesia.
\textsuperscript{3} Department of Physics, Faculty of Mathematics and Natural Science, Sriwijaya University

Email: (a) szaki@fi.itb.ac.id (b) nixraragalih@gmail.com

Abstract. Human need of energy will increase time to time. Therefore, a safe, renewable, and efficient source of energy, which is Nuclear Energy, is needed. Nuclear Power Plant (NPP) is the most compatible solution to provide electricity to human race in the future. The problem that came within NPP is the danger of proliferation issues. The method that has been developed to overcome this problem is CANDLE \cite{5} and has been modified by Prof. Zaki Su’ud (Modified CANDLE scheme). This research use Axial Modified CANDLE Scheme to Helium-Cooled Fast Reactor with Natural Uranium Carbide-Thorium Carbide as fuel and applied to various size of core as optimization. Neutronic aspect such as, burn up level, multiplication factor, and conversion ratio are utilized in this paper in order to analyse the behaviour of the reactor. Other than that, percentage of Uranium has been varied to reduce power peaking. The neutronic calculation has been done using SRAC and core design calculation by FI-ITB-CH1. This research concludes that power peaking reduction is able to achieve by combining Uranium Carbide and Thorium Carbide to the fuel. The optimum reactor design reached at 360 cm of core radius and 303 cm of core height.

1. Introduction
All human activities require energy. Along with the increasing human need for energy, fossil energy sources, as the main energy source decreases. Therefore, renewable energy sources are sufficiently needed. Nuclear energy sources, being the most potential, most effective and efficient energy sources, to meet those energy needs. The problem that arises from the development of nuclear energy sources is the possibility of misuse of nuclear waste for the benefit of weapons that endanger human survival. This lead to the issuance of a non-proliferation nuclear regulation.

In order to answer the challenge, this research use Modified CANDLE burning system, a nuclear fuel combustion system on the reactor core with no prior enrichment process with the intention to forbid the radioactive waste for weapon utilization. Consider this strategy also added a good environmental and economic value. This research use Helium gas-cooled Gas Cooled Fast Reactor
(GFR) reactor with an axial directional MCANDLE system and a mixture of Uranium Carbide - Thorium Carbide fuel. 100 years operation time and decennial refuelling are required. Neutronic parameters such as burn-up level, infinite multiplication factor, effective multiplication factor, and conversion ratio are ought to be analysed.

2. Methodology

Calculations performed by SRAC (Standard thermal Reactor Analysis Code) program developed by JAERI (Japan Atomic Energy Research Institute) by adding to the reactor neutronic analysis the FITB-CH1 program is also used for MCANDLE modification.

2.1. CANDLE Burning Concept

Constant Axial Shape of Neutron Flux, Neutron Densities and Power Profile During Life of Energy Production or abbreviated as CANDLE is a fuel burn-up system that resemble the burning of a candle [2]. Fissile materials were to produce when fertile material inserted to fresh fuel region absorbs the neutron leaked from burning region. Fissile material density will continue to increase in the border of fresh fuel region and burning region, thus becomes constant when production rate is equal to reduction rate. To facilitate mathematical modelling, the core assumed to have infinite height.

However, the length of spent fuel region and fresh fuel region is shorter than burning region. In addition, although it modelled to have a candle-like flow path, burning region has the possibility to move in the opposite direction, which is bottom to top. As presented in figure 1, burning process begun as spent fuel removed and fresh fuel inserted to the burn up direction when burning region reaches the bottom of the core.

![Figure 1. CANDLE Burning Strategy [4]](image)

2.2. MCANDLE Burning Strategy

MCANDLE or modified-CANDLE is a form of development of the CANDLE burn-up strategy that introduces discrete regions [6]. MCANDLE share the same concept with CANDLE except MCANDLE use fresh fuel in the form of non-enrichment natural uranium. MCANDLE also designed to have discrete regional core division. Fresh fuel ought to be inserted to first region then transferred to second region after combustion occurred. New fresh fuel then entered first region. This sequence repeated before it reached the last region. Combustion products in last region removed from the terrace as residual combustion. In this study, reactor core has 10 axial direction regional divisions with identical region lengths as shown below.
Table 1. MCANDLE Regions

| Region | Fuel Input Description                                      |
|--------|-------------------------------------------------------------|
| 1      | Fresh fuel: natural Uranium Carbide and Thorium Carbide mixture |
| 2      | 10 years combustion product from region 1                   |
| 3      | 20 years combustion product from region 2                   |
| 4      | 30 years combustion product from region 3                   |
| 5      | 40 years combustion product from region 4                   |
| 6      | 60 years combustion product from region 5                   |
| 7      | 70 years combustion product from region 6                   |
| 8      | 80 years combustion product from region 7                   |
| 9      | 90 years combustion product from region 8                   |
| 10     | 100 years combustion product from region 9                  |

In decennial, region 1 waste transferred to region 2, region 2 to region 3, and continue until region 10. Combustion products from region 10 removed from the reactor core as final waste.

3. Reactor Design
The design of the reactor used in this study was a generation IV gas-cooled reactor. Several variations in reactor size were varied, under 90% Uranium fuel, 65% fuel volume fraction and tall typed reactor as fixed variable.
Table 2. Variation of Reactor’s Dimension

| Uranium Percentage | 90 |
|--------------------|----|
| Power (MWt)        | 550 | 550 | 550 | 550 |
| Axial Length (cm)  | 170 | 180 | 180 | 180 |
| Radial Length (cm) | 300 | 303 | 307 | 312 |
| Axial Length and Radial Reflector(cm) | 65 | 65 | 65 | 65 |
| Region wide (cm)   | 56.7 | 57.9 | 59.3 | 60.78 |
| Region Height (cm) | 17.01 | 17.38 | 17.79 | 18.23 |
| Fuel Volume Fraction (%) | 65 | 65 | 65 | 65 |
| Fuel Volume Fraction Cladding (%) | 10 | 10 | 10 | 10 |
| Fuel Volume Fraction Coolant (%) | 25 | 25 | 25 | 25 |
| Pin radial length(cm) | 0.6 | 0.6 | 0.6 | 0.6 |
| K-eff of core      | 0.99786 | 1.0310 | 1.0630 | 1.0326 |
|                    | 0.99932 | 1.0381 | 1.0711 | 1.0420 |
|                    | 1.0006 | 1.0439 | 1.0477 | 1.0496 |
|                    | 1.0016 | 1.0484 | 1.0528 | 1.0555 |
|                    | 1.0024 | 1.0520 | 1.0568 | 1.0602 |
|                    | 1.0033 | 1.0552 | 1.0605 | 1.0644 |

The most effective multiplication values are presented by yellow column in table 2. Therefore, the next research uses reactor dimensions of a core radius of 180 cm and core height of 303 cm. The width and height of each region are 57.9 cm and 17.38 cm. Thus, the full reactor specifications are presented in Table 3 below. The cell geometry used in this study is pin-shaped consist of fuel, cladding and coolant. The fuel used is natural uranium - carbide and natural thorium – carbide mixture while the coolant is Helium gas.

Considering helium has low neutron absorption tendency and less likely to be radioactive. Helium as an inert gas also prevents chemical reactions with other materials. Meanwhile, the SS316 cladding structure that is chosen appeared to be resistant to swelling, high temperatures and long-term radiation. The length of the fuel + cladding radius for this study is set at 0.6 cm.

4. Result and Analysis
The calculations in this study were carried out in two stages, namely cell neutronic calculations and reactor core calculations. Fuel cell neutronic calculations involve the calculation of fuel cell combustion for 100 years, with data collection intervals every two years. Neutronic calculations produce several neutronic parameters such as burn-up levels, infinite multiplication factors ($k_{infin}$), conversion ratio values (Inte. C.R.), and atomic density values. Meanwhile, core calculation produces several parameters such as effective multiplication factor ($k_{eff}$) of core, peak power density, and distribution of power density.
### Table 3. Reactor’s Specification

| Parameter          | Description                                                                 |
|--------------------|------------------------------------------------------------------------------|
| Power              | 550MWth                                                                      |
| Pin Cell Type      | Cylindrical cell                                                            |
| Core Geometry      | Silinder 2D                                                                   |
| Axial Region       | 10 region                                                                    |
| Refueling          | 10 tahun                                                                     |
| Fuel               | Natural Uranium Carbide – Natural Thorium Carbide                            |
| Half of core diameter | 180 cm                                                                       |
| Core height        | 303 cm                                                                       |
| Wide and height of reflector | 65 cm                                                                     |
| Region wide        | 57.95 cm                                                                     |
| Region height      | 17.38 cm                                                                     |
| Cladding           | SS316                                                                        |
| Coolant            | Helium                                                                        |
| Fuel Fraction      | 65%                                                                           |
| Coolant Fraction   | 25%                                                                           |
| Structure Fraction | 10%                                                                           |
| Pin diameter       | 1.2 cm                                                                        |
| Reflector          | SS316                                                                        |

#### 4.1. Burn Up Level

The burn-up level defined as the total energy released per unit mass of fuel as result of combustion. Has unit in MWD / TON or megawatt days per metric ton of fuel, which is the amount of mass needed to produce power per day [1]. A graph of the burn-up level (MWD / TON) value of each percentage of Uranium in the mixture obtained in this study is presented in Figure 3(a). Various variations of the percentage of Uranium are tested on core with 550MWt power.

![Figure 3(a). Burn Up Level](image1.png)

![Figure 3(b). Infinite Multiplication (K-inf)](image2.png)
4.2. Infinite Multiplication Factor

The infinite multiplication factor or k-inf shows number of neutrons in a generation compared to the number of neutrons of the previous generation, assuming there are no neutron leaks in the system. The infinite multiplication factor value is taken from the results of neutronic calculations of fuel cells through the SRAC program for reactors with 500MWt power.

It can be seen in Figure 3(b), that the value of multiplication is infinite for each percentage variation. The k-inf value shows a high increase in the first 10-year burn-up period. This relates to the placement of fuel regions in the MCANDLE system where regions containing fresh fuel (region 1) are located close to region The highest increase in the value of the effective multiplication factor is indicated by a variation of Uranium 70%. This fact is in line with the burn-up value of 70% Uranium variation which is higher than other percentage variation values. This means that the combustion of fertilizers occurs faster, and faster fissile material is produced.

4.3. Conversion Ratio

Integrated conversion ratio (Inte. C.R.) shows ratio of production rate per depletion rate of nuclides in the burn up process [1]. This value was obtained as a result of fuel cell calculation from SRAC for 550MWt core power. Integrated conversion ratio value for fuel mixtures with various Uranium percentage presented in figures 4(a).

4.4. Atomic Density

The difference in the percentage of Uranium Carbide-Thorium Carbide in this study resulted in a difference in the density of the Thorium atom at the beginning of the reactor burn-up period. The density of natural Thorium in fuel is shown by the Thorium-232 density graph in figure 4(b). This is caused by the Thorium- 232 content in natural Thorium which reaches more than 95%.

Figure 5(a) to 5(d) show a decrease in atomic density that varies according to the percentage of fuel mixture. The most drastic decline is after a 60-year burn-up period. This relates to the productivity of power by the reactor.
4.5. Effective Multiplication Factor

The value of the effective multiplication factor is obtained from the calculation of the reactor core by the FI- ITB-CH1 program, which means that this result illustrates the comparison of a generation of neutrons with previous generations of the entire reactor core. The $k_{eff}$ value is obtained for each variation of the percentage of Uranium in the fuel and the reactor power of 550MWt.

The reactor effective multiplication factor value in the graph in Figure 6 shows a different form from the infinite multiplication factor value graph in Figure 3(b). Occurred due to time scale different, where the effective multiplication factor graph was calculated for the fuel cycle time period in one region (10 years), while the infinite multiplication factor in Figure 3(b) was calculated for the 100-year burn-up period. This difference is also caused by a different calculation system, where the calculation
of the effective multiplication factor was calculated for the entire core of the reactor (by the FI-ITB-CH1 program) while the infinite multiplication factor in Figure 3(b) was calculated for only one cell (cell calculation by SRAC program).

Table 4. Effective Multiplication Factor

| Time (year)/percentage of Fuel Mixture | 100%   | 95%   | 90%   | 80%   | 70%   | 60%   | 50%   |
|---------------------------------------|--------|--------|--------|--------|--------|--------|--------|
| 0                                     | 1.0399 | 1.0355 | 1.0310 | 1.0214 | 1.0111 | 1.0006 | 0.9896 |
| 2                                     | 1.0468 | 1.0426 | 1.0381 | 1.0287 | 1.0187 | 1.0083 | 0.9975 |
| 4                                     | 1.0522 | 1.0482 | 1.0439 | 1.0349 | 1.0252 | 1.0150 | 1.0045 |
| 6                                     | 1.0565 | 1.0526 | 1.0484 | 1.0397 | 1.0303 | 1.0204 | 1.0101 |
| 8                                     | 1.0598 | 1.0560 | 1.0520 | 1.0434 | 1.0342 | 1.0246 | 1.0146 |
| 10                                    | 1.0628 | 1.0591 | 1.0552 | 1.0469 | 1.0379 | 1.0285 | 1.0187 |

As seen from the graph and table that in general the value of $k_{eff}$ for each power variation approaches the value of 1.0 and is always above the value of 1.0. This means that in general, reactors with various percentages of uranium can operate from the beginning to the end of the burn-up period. While also in general, reactors with various variations of the fuel mixture can operate at critical points with 550MWt power.

4.6. Peak Power Density

One characteristic of the CANDLE reactor is the presence of a peak value (peak-to-average power density) which is quite large. Figure 7 shows the distribution of power density for variations in Uranium 90% fuel mixture and reactor power 550MWt. These distributions can represent the distribution of variations in the percentage of other uranium for the same power.

As seen from Figure 7, the highest power density value is in the centre of the reactor. This relates to the placement of the combustion region in the MCANDLE system. The schematic division of the
combustion region in this study served in Figure 2. Meanwhile, the position of each region in the axial length of the reactor served in Table 5.

Table 5. Description of Region's Position in the Axial Length of the Core

| Description | Initial point (cm) | End Point (cm) |
|-------------|--------------------|----------------|
| reflector   | 0                  | 65             |
| region 1    | 65.1               | 82.48          |
| region 10   | 82.49              | 99.87          |
| region 9    | 99.88              | 117.26         |
| region 8    | 117.27             | 134.65         |
| region 7    | 134.66             | 152.04         |
| region 6    | 152.05             | 169.43         |
| region 5    | 169.44             | 186.82         |
| region 4    | 186.83             | 204.21         |
| region 3    | 204.22             | 221.6          |
| region 2    | 221.61             | 238.99         |
| reflector   | 239                | 304            |

Power production on the reactor core continues to increase from region 1 (the first 10 years of combustion) to its peak in region 7 (burn-up period of 60 years to 70 years), namely in the axial length of the reactor 134.66 cm to 152.04 cm. Then the power produced decreases again to region 10 (100-year burn-up period). This clarifies the phenomenon shown in the atomic density graph and infinite multiplication factors in the previous sections.

Figure 7 shows the appearance of a quarter-cross section of the reactor's power density meeting. Peak-to-average power density values that are too large are not desirable in the CANDLE reactor design [3]. This value can be levelled by combining fuel with lower reactivity in the centre of the reactor to reduce the value of the power density in the centre of the reactor.

5. Conclusions and Recommendations

5.1. Conclusions

Generation IV reactor type Gas Cooled Fast Reactor or GCFR with 550MWt power, Helium gas cooled using axial directional MCANDLE combustion scheme, and with variations of Uranium Carbide-Thorium Carbide reactor can operate at critical points with reactor radius of 180cm and height reactor 303 cm. The reactor can also operate critically at variations in the percentage of Uranium Carbide in a mixture of 70%, 80%, 90%, 95%, and 100% and has the lowest k-eff value when using a 70% Uranium mixture. The reactor is sub-critical in a reactor with a mixture of 60% and 50% Uranium-Carbide. It also has a CANDLE reactor characteristic that is peak-to-average power density in the centre of the reactor in the axial length of the reactor, which is around the seventh region.
5.2. Recommendation
The peak value of the power density can be flattened by combining fuel in the core with a lower power density value placed in the centre of the reactor. Research can be complemented by reactor optimization through variations in the volume fraction of fuel and coolant.

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