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The MOX Fuel Performance Analysis of ALFRED Core Using MCNPX Transport Code

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Abstract

The (**300** MW_{th}) pool type Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED) is an essential step in the evolution of the LFR technology developed to demonstrate the viability of the European Lead Fast Reactor (ELFR) of Generation-IV for use in the future commercial power plant. The ALFRED core consists of two radial zones, inner zone (IZ) and outer zone (OZ), with closed hexagonal Fuel Assemblies (FAs). The fuel is composed of MOX pellets with different fissile plutonium enrichment for inner and outer zones. Using MCNPX transport code, 3D model of ALFRED core has been designed to study the MOX fuel performance analysis of the ALFRED core and to simulate its operation during a fuel cycle of a duration of 1825 effective full power days. The obtained results show good agreement with previous studies and confirm the capability of the reactor ALFRED to achieve sustainability.

Keywords

ALFRED, MCNP, MOX

1. Introduction

In 2002, the Fourth Generation International Forum (GIF) regarded fast spectrum reactors utilizing lead as a coolant as one of the three advanced systems within the category of fast spectrum nuclear reactors (DOE,2002). Hence, fast spectrum reactors stand out as the most efficient technology for maximizing the utilization of uranium resources. This is attributed to their capability to recycle the uranium found in spent nuclear fuel, generating energy. As a result, they offer a practical pathway towards achieving sustainable energy practices aligned with environmental stewardship. This involves the implementation of advanced technology to effectively manage long-lived minor actinides and nuclear fission products (Waltar, 2011). Among the various innovative concepts within the fourth generation (Gen-IV) of advanced nuclear technology, the lead-cooled fast reactor (LFR) stood out as a highly promising option. It was recognized for its potential to demonstrate excellence across four main domains: sustainability, economic viability, safety and reliability, and resistance to proliferation while ensuring physical protection (Zohuri, 2020). The lead-cooled fast reactor is characterized by its capacity for high-temperature operation, low-pressure conditions, a closed fuel cycle, and the ability to function flexibly in two modes. It can serve as a breeder, facilitating the net creation of fissile fuel, or as a transmuter, converting long-lived minor actinides and other radioisotopes into shorter-lived ones. Due to its distinct safety advantages compared to other fast reactors, the lead-cooled fast reactor holds promising prospects for global development. Currently, the Generation-IV International Forum (GIF) System Research Plan (SRP) centers around Europe's ELFR lead-cooled system (Alemberti, 2011), Russia's BREST-OD-300 (Zrodnikov, 2011), and the US-designed SSTAR system concept (Smith, 2008). These projects serve as the foundation for the GIF's exploration of lead-cooled fast reactor system concepts. ALFRED (Advanced Lead-cooled Fast Reactor European Demonstrator) stands as a cutting-edge initiative in Europe, serving as a demonstrative model for the ELFR (European Lead Fast Reactor) concept. Its primary objective is to validate safety and reliability under various operational conditions, with operations anticipated to commence in 2025 (Grasso et al., 2014). Three critical stages in the execution of this project include choosing the fuel, creating the design, and optimizing the process. The objective of this study is to employ the MCNPX stochastic transport code for assessing the performance of MOX fuel in the ALFRED core, along with analyzing its isotopic transmutation over varying burn-up times.

2. ALFRED Design Parameters

ALFRED, a pool-type reactor with a small footprint, features a core comprised of 171 hexagonal fuel assemblies (FAs) arranged in a cylindrical configuration. Additionally, there are sixteen positions designated for twelve control rods (CRs) and four safety rods (SRs). Surrounding the core are two rows of 108 dummy elements serving as a reflector. The 171 FAs are radially divided into an inner zone consisting of 57 FAs and an outer zone comprising 114 FAs, each with distinct plutonium enrichments. Each FA includes a triangular lattice housing 127 fuel pins and is cooled by pure lead. **Figure 1** and **Figure 2** illustrates the primary system and core layout of ALFRED respectively. Essential parameters of the ALFRED core are summarized in **Table 1.** The fuel consists of MOX pellets with a theoretical density of 95% and an Oxygen-to-Metal ratio (O/M) of 1.97. The U and Pu isotopic compositions are determined based on recovered U and Pu from LWR spent fuel burnt up to 45 GWd/t, with a 4.5% initial enrichment in ^{235}U, after 15 years of cooling, including 4 years post-reprocessing. **Table 2.** details the fuel isotopic composition of uranium and plutonium. The fuel pin design, depicted in **Figure 3**, incorporates inner and outer radii of 1 and 4.5 mm, respectively, with an active length of 60 cm (Grasso et al., 2013). (Grasso et al., 2014), as outlined in **Table 3.** B_4C, with 90 at.% of ^{10}B and a density of 2.2 g/cm^3, is the chosen absorber material for control and safety rods.

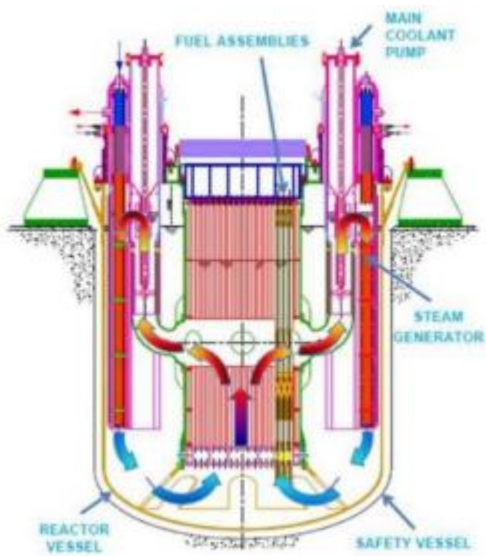

Figure 1. ALFRED primary system (Luzzi, Lelio, et al.,2014) (Ibrahim et al., 2021).

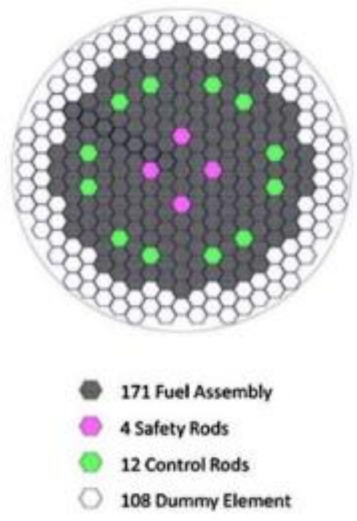

Figure 2. ALFRED core layout (Luzzi, Lelio, et al.,2014) (Ibrahim et al., 2021).

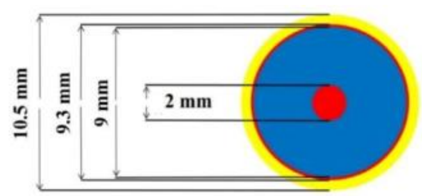

Figure 3. Fuel pin cross-section of ALFRED core layout (Ibrahim et al., 2021).

Table 1. ALFRED main parameters (Grasso et al.,2013) (Ibrahim et al., 2021).

Main parameters	MCNPX
Thermal power (MW) 300	300
Fuel assembly concept	Closed hexagonal
Coolant	Lead
Inlet temperature of coolant (oC)	400
Outlet temperature of coolant (oC)	480
Mass flow rate of coolant (kg s^{-1})	26000
Maximum velocity of coolant (m s^{-1})	< 2.0
Cladding	15-15 Ti
Cladding maximum temperature at nominal conditions (°C)	550
Peak burn-up (MWd/kg)	100

Table 2. The isotopic vectors of uranium and plutonium (Sobolev et al., 2009) (Ibrahim et al., 2021).

Plutonium isotopes	Fraction (wt.%)
^{238}Pu	2.332
^{239}Pu	56.873
^{240}Pu	26.997
^{241}Pu	6.105
^{242}Pu	7.693
Uranium isotopes	Fraction (wt.%)
^{234}U	0.003
^{235}U	0.404
^{236}U	0.010
^{238}U	99.583

Table 3. Design parameters of ALFRED fuel pin (Ibrahim et al., 2021).

Main parameters	MCNPX
Type of fuel	MOX
Fill gas	He
Inner diameter of fuel pellet (mm)	2.0
Outer diameter of fuel pellet (mm)	9.0
Inner diameter of cladding (mm)	9.3
Outer diameter of cladding (mm)	10.5
Length of lower plenum (mm)	550
Length of upper plenum (mm)	120
Length of Active zone (mm)	600

3. Modeling and Simulation Tools

The ALFRED core's neutronic behavior and fuel transmutation were investigated using the MCNPX computational model. All simulations considered operational conditions, with control and safety rods in their parking positions (safety rods above the fuel region and control rods below the fuel region). The MCNPX simulations employed 1000 source histories per cycle to simulate neutron transport in the model. The skip cycles were set at 55, and the total cycles amounted to 550. **Table 4.** presents a comparison of the neutronic characterization of the ALFRED core with the reference values from (Grasso et al., 2014).

Table 4. Neutronic characterization of the ALFRED core.

Parameter	MCNPX	Refrence value
k_{eff}	1.02956	
(Δk_{eff} / k_{eff}) swing in 1 year (pcm)	- 1740	- 2580
Fuel burn-up (GWd/t)	71.6	73.3
Average neutron flux (cm^{-2} s^{-1})	1.53×10^{15}	1.5×10^{15}
Max power in FA (MW)	2.2	2.21
Total worth of 12 CRs (pcm)	- 8890	- 8500
Total worth of 4 SRs (pcm)	- 2890	- 3300
βeff (pcm)	410	336
Doppler constant (pcm)	- 481	- 555
Conversion factor after 5 years	84.2%	

4. Results and Discussion

This analysis utilized the ENDF/B-VI.2 cross-sections library from a neutronic perspective. The library comprises cross-sectional data for various nuclides, including fuel, reflector materials, clad, and coolant, provided at different temperatures (e.g., 1200 K, 900 K, 600 K, and 293 K). Simulating the burn-up of MOX fuel within the assembly spanned a total cycle length of 5 years, equivalent to 1825 Effective Full Power Days (EFPD). The accumulation of U/Pu/MA isotopes' masses was tracked at various irradiation time intervals to compare fuel transmutation results between 3D-homogeneous and 3D-heterogeneous models. The simulation employed a Dell Precision T5610 with two processors (Intel Xeon processor E5-2600 V2) and 32GB ECC RDIMM memory at 1866MHz to run the MCNPX code.

4.1. k_{eff} and fuel burn-up of the ALFRED core

The graph in **Figure 4** illustrates how the k_{eff} value of the ALFRED core changes over irradiation time. The k_{eff} curve demonstrates a minimal reactivity swing of (0.017 Δk/yr) over a one-year burn-up period, indicating a slow reduction in total fissile material throughout the burn-up duration. Additionally, **Figure 5** depicts that the discharged fuel burn-up of the ALFRED core at the conclusion of the irradiation period (1825 EFPD) has reached 71.6 GWd/t. Furthermore, according to MCNPX burn-up results, neutrons in the thermal, intermediate, and fast neutron ranges contribute 0.29%, 39.61%, and 60.10%, respectively, to the fission reaction on average. This level is deemed acceptable for the generation of new fissile isotopes, where the quantity of neutrons generated per fission exceeds 2. Hence, thermal neutrons make negligible contributions to the fission power, while only fast and intermediate neutrons play a role in generating fission power. Additionally, the mean energy of neutrons causing fission was determined to be 0.73 MeV, and the average lifetime of these neutrons is $11.9 \times 10^{-6} \pm 5 \times 10^{-8}$ seconds, significantly shorter than that of thermal (slow) neutrons, whose lifetime in thermal reactors can be as long as 10^{-3} seconds (Rugh, 1996). The average number of neutrons generated per fission was established to be 2.938 neutrons.

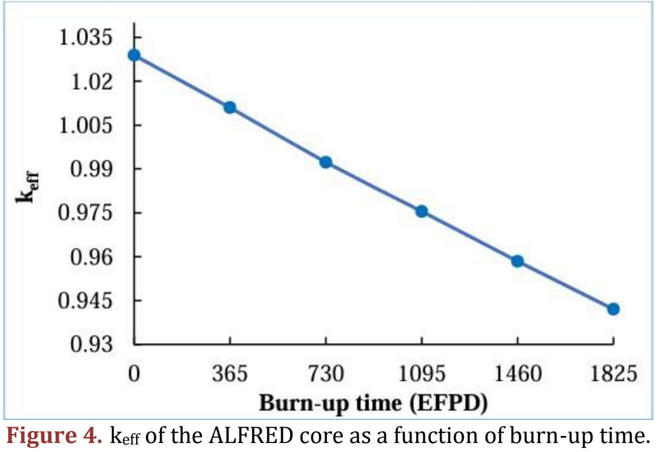

Figure 4. k_{eff} of the ALFRED core as a function of burn-up time.

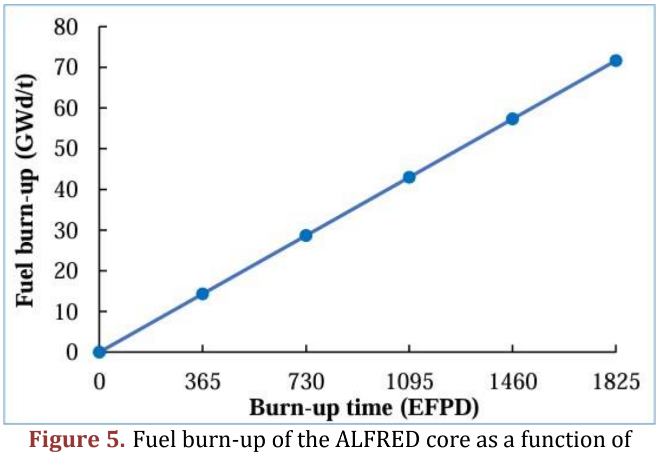

Figure 5. Fuel burn-up of the ALFRED core as a function of burn-up time.

4.2. ALFRED Radial Flux Distribution

In **Figure 6**, the radial distribution of neutron flux (measured in $cm^{-2}\ s^{-1}$) in the ALFRED reactor is depicted. The measurements were taken at the midpoint between the upper and lower extents of the axial active core. These values were computed at the onset of core life (BOL) using F4 and FM tallies within the MCNP5 code (X-5 Monte Carlo Team, 2003). The neutron flux per starting neutron history obtained through the F4 tally in MCNP is appropriately normalized to yield the total neutron flux corresponding to the overall neutrons generated for the entire reactor power.

4.3. ALFRED Radial Power Distribution

The power distribution in the ALFRED core at the beginning of life (BOL), represented in **Figure 7**, illustrates power levels in megawatts (MW). The analysis reveals that the maximum power within the fuel assembly (FA) reaches 2.2 MW. Additionally, the overall power distribution remains relatively uniform throughout the core, with no peak occurring at the core's midpoint; instead, the peak is situated at the boundary between the two fuel zones.

4.4. Control Rods Worth (CRW)

Different insertion distances of control rods (CSDs) were investigated to quantify their total worth and to make

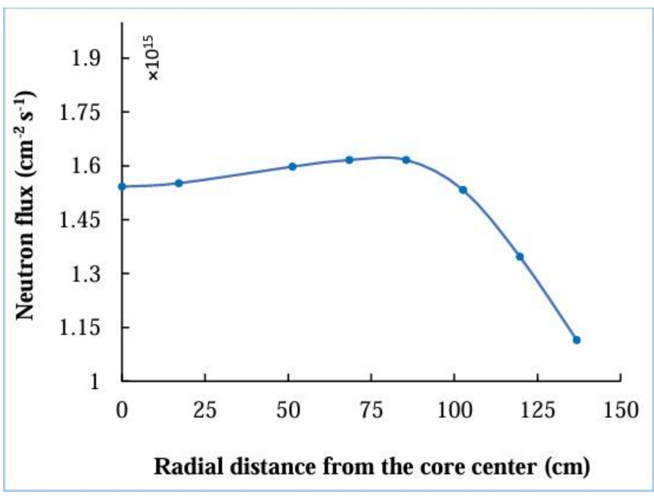

Figure 6. Radial neutron flux distribution of ALFRED core.

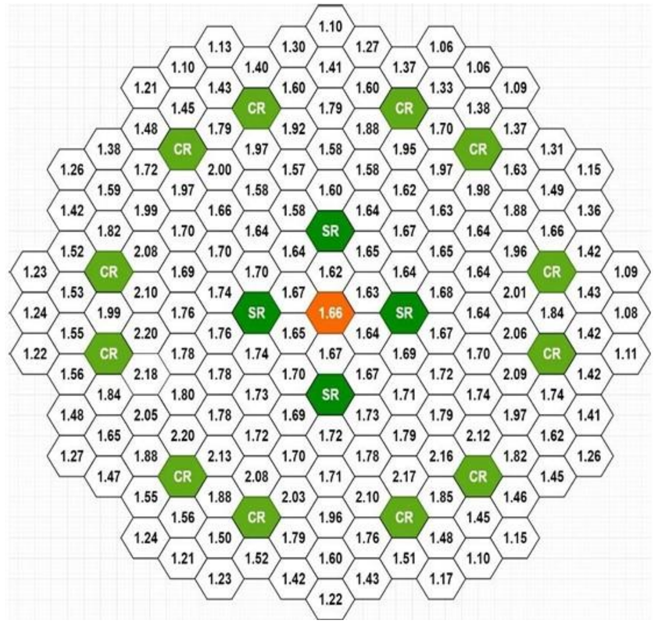

Figure 7. Radial power distribution (MW) of ALFRED core.

sure that they can introduce sufficient negative reactivity to shut down the reactor in case of accidents. The control rod worth (CRW) was estimated using the following equation,

$$CRW = k2 - k1\ k2 \times k1 \quad (4.1)$$

Where, k1 and k2 are the keff values of the core, being calculated without and with the insertion of control rods respectively. To determine the exact effect of the insertion and withdrawal of control rods on core reactivity, integral rod worth has been investigated for the core model. The integral rod worth was calculated for CSDs rods as shown in **Figure 8**. The slope of the integral rod worth curve ($\Delta\rho/\Delta x$), and therefore the integrated reactivity introduced per unit insertion of CSDs is greatest when the control rod is midway in the core. This occurs because the region of greatest neutron flux is near the center of the core and thus the amount of change in neutron absorption is the greatest in this area.

Summary and Conclusion

In summary, we looked closely at how the ALFRED nuclear concept behaves when it comes to its neutrons – those tiny particles that power the reactor. Using a computer program called MCNPX, we made a 3D model of the

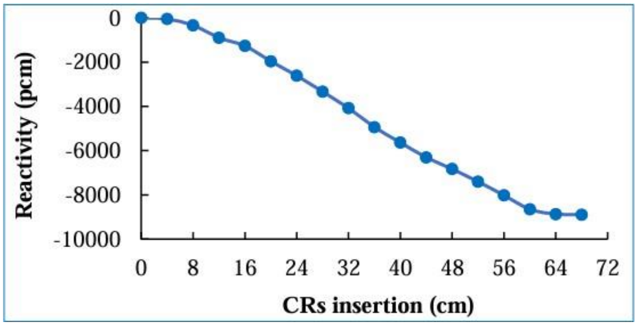

Figure 8. The CSDs integral rod worth curve of ALFRED core.

ALFRED core to see how it works from the start to the end of its fuel cycle. We found out that the core has safety features to keep it in check during accidents, like shutting down if needed. Also, we discovered that even if something goes wrong and the coolant is lost, the reactor won't go out of control thanks to certain safety measures. Plus, we learned that the core can keep running for a long time without needing a lot of new fuel. It can still have about 97% of its starting fuel even after a long time, showing that it can keep going without needing a lot of extra stuff. These findings show that the ALFRED design is strong and can be a good option for clean energy in the future.

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