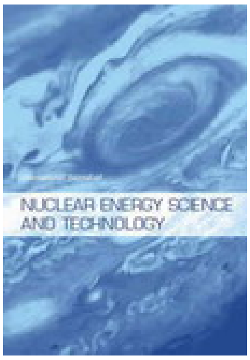

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A comparison of neutronic studies on 400 MW thermal fast reactor with modified CANDLE burn-up schemes using helium gas, lead bismuth eutectic and liquid sodium coolants

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A comparison of neutronic studies on 400 MW thermal fast reactor with modified CANDLE burn-up schemes using helium gas, lead bismuth eutectic and liquid sodium coolants

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Abstract: One of the most essential technical challenges in fast reactor design is coolant selection. This paper investigates neutronic studies on a 400 MW thermal Modified CANDLE burnup scheme in a fast reactor using helium gas, lead-bismuth eutectic, and liquid sodium coolants. The core has been separated into ten regions with the equivalent volume in radial direction and recharging every ten years. Natural uranium was loaded in the first region, and after ten years, it was transported to the second region. These movements have taken place in all regions and fuel in tenth region got out of core. The neutronic features such as burnup level, integral conversion ratio, relative power density, flux level, infinite multiplication factor and effective multiplication factor were carried out. According to the results of the multiplication factor, liquid sodium attained criticality faster than helium gas and lead-bismuth eutectic coolants. This makes it to be the best coolant.

Keywords: fast reactor; coolant; modified CANDLE; SRAC; JENDL 4.0; effective multiplication factor.

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1 Introduction

Water-cooled reactors generate around 95% of the world's nuclear power (Atomic and Agency, 1992). However, water-cooled nuclear power reactor experience demonstrates that even under normal operating conditions, some undesirable consequences such as corrosion and erosion on heat transfer surfaces can occur (Atomic and Agency, 1992). The Physics and Power-Engineering Institute advanced a suggestion and, in collaboration with other organisations, began work on fast reactors for nuclear power with sodium coolant, nuclear power with sodium-potassium eutectics coolant for use in space and nuclear power with Lead-Bismuth Eutectics (LBE) coolant for use in nuclear-powered submarines (Shimkevich, 2005; Wolniewicz, 2012). Toshinsky and Petrochenko (2012) first used lead-bismuth coolant in fast reactors in 1950 while evaluating the feasibility of building a breeder reactor. Because of its physicochemical and thermodynamic properties, the LBE is a primary coolant for nuclear reactors, allowing it to meet the requirements of nuclear-powered submarine reactors in terms of size, weight and safety (Troyanov et al., 2022). However, employing LBE in nuclear technology has been challenging due to that LBE damages the environment, its limited heat transfer and its corrosivity to most metals. Thus, sodium was chosen as the coolant in fast breeder reactors when they were constructed (Hosemann, 2008; Toshinsky and Petrochenko, 2012). Because of its higher power density and lower doubling time. In the 1960s, sodium became the favoured choice of coolant (IAEA, 2012). The coolant used is chosen with the goal of introducing the least amount of absorption and moderation (Van Rooijen, 2009). A liquid metal, most typically sodium, is preferred, but a gas coolant is also an alternative. Helium, CO_2 and steam are the most often used in gas-cooled fast reactors (Waltar and Reynolds, 1981). Compared to sodium, gas coolants have advantages for fast reactors such as being chemically compatible with water, having optical transparency, supporting fuel shuffling operations and inspection, lowering the positive void effect, lowering the potential for reactivity swings under accidental situations, allowing a harder neutron spectrum, enhancing the reactor's breeding potential and permitting a harder neutron spectrum (Waltar and Reynolds, 1981). Helium gas coolant technology is based on its successful application to numerous fission reactors, notably Peach Bottom in the USA (Wong et al., 1994). In this work helium gas, LBE and liquid sodium coolants have been selected to employ in modified CANDLE burnup due to that helium has high thermal conductivity, lower neutron absorption low boiling point and its chemical inertness with other chemical reactions (Su'ud and Galih, 2021). Owing to those properties, helium has been proved to substantially enhance the performance of fast reactors when used as a coolant. Meanwhile, LBE has been selected as coolant due to its physicochemical and thermodynamic properties such as its low-neutron absorption cross-section, compatibility with cladding, high-thermal conductivity, high-boiling point and its chemical inertness while contacting with air and water (Widiawati et al., 2022). The use of LBE has been proposed to achieve a lower melting point (Widiawati et al., 2021a). Moreover, the utilisation of sodium as coolant has been selected due to its low pressure and providing high-power density with low-coolant volume (Widiawati et al., 2022, 2021a).

The modified CANDLE burn-up strategy has an economic benefit in that it saves the expense of uranium enrichment because natural uranium is required. It also eliminates the problem of nuclear proliferation owing uranium reprocessing or enrichment is unnecessary (Nguyen et al., 2020; Su'ud et al., 2017). Therefore, the aim of this study is to investigate the neutronic studies on 400MWt fast reactor with modified CANDLE burn-up scheme employing helium gas, LBE and liquid sodium coolant. In this work, an achievable design study of a 400 MWt modular modified CANDLE fast reactor that can employ natural uranium as fuel has been conducted, and it requires recharging every ten years of burnup. The core has been separated into 10 regions with the equivalent volume in radial direction. A fuel was loaded in the first region, and after 10 years, it was transported to the second region, and 10 later, it was transported to the third region. These movements have taken place in all regions and fuel in tenth region got out of the core (Irka et al., 2023). Integral conversion ratio, k_{inf}, burn-up level, K-effective, uranium-238 atomic density and plutonium-239 atomic density neutronic parameters have been investigated for each coolant type. A standard reactor analysis code (SRAC) was used to perform the calculation. Cell calculations were performed using SRAC's collision probability method (PIJ), and core design calculations were carried out using SRAC's CITATION module, based on the fourth version of the Japanese Evaluated Nuclear Data Library (JENDL-4). According to the results of the *K*-effective, sodium coolant reached criticality before lead-bismuth eutectic and helium gas coolants.

2 The design concept

2.1 The CANDLE burnup

CANDLE means *C*onstant *A*xial Shape of *N*eutron Flux, Nuclide Densities and Power Shape *D*uring *L*ife of *E*nergy Production) burnup was suggested by Su'ud and Sekimoto (2012); Widiawati et al. (2021b) and Hong et al. (2016). One advantage of this burn-up approach, it can employ natural uranium as fuel. The core is separated into three parts: spent fuel, burning fuel and fresh fuel. Figure 1 depicts CANDLE burn-up strategy while Figure 2 depicts CANDLE burnup and refuelling strategy. The fission reactions take place in the burning region and produce fast neutrons. The fast neutrons produced penetrate into the fresh fuel region, causing fertile materials (U-238) to convert into fissile material (Pu-239) (Nguyen et al., 2020). Fissile materials decrease at the top of the burning region, whereas fission products accumulate. This allows the burning region to move from top to bottom during burnup while maintaining power density distribution, nuclide densities and the shape of the neutron flux (Kheradmandsaadi, 2013). Unlike conventional reactor designs, there is no need for movable components to manage burnup, even if the fuel is fixed in the core. To clarify the burn-up strategy characteristics, the core assumed to be extremely long. In the actual core, the combination of the length of the spent and fresh fuel parts is typically significantly shorter than the burning part. In Figure 1, the burning region moves from top to bottom, however it can also move from bottom to top. When the burning region attains the end of the core, the fuel should be changed by taking out the spent fuel region and refuelling the burn-up region with fresh fuel, as seen in Figure 2 (Yan and Sekimoto, 2008; Widiawati et al., 2021a). This burning scheme has numerous potential benefits in terms of non-proliferation of nuclear materials, waste reduction, safety, fuel sustainability and safeguarding.

Figure 1 CANDLE burnup (see online version for colours)

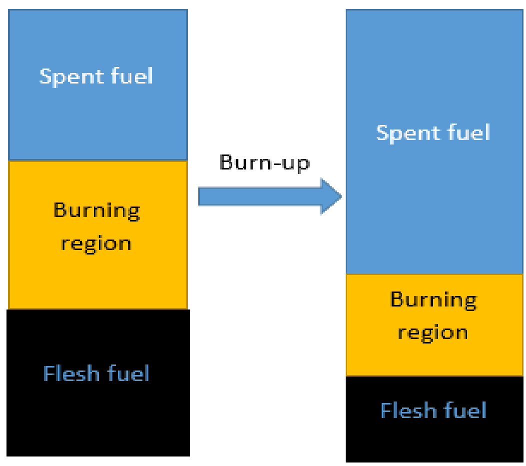
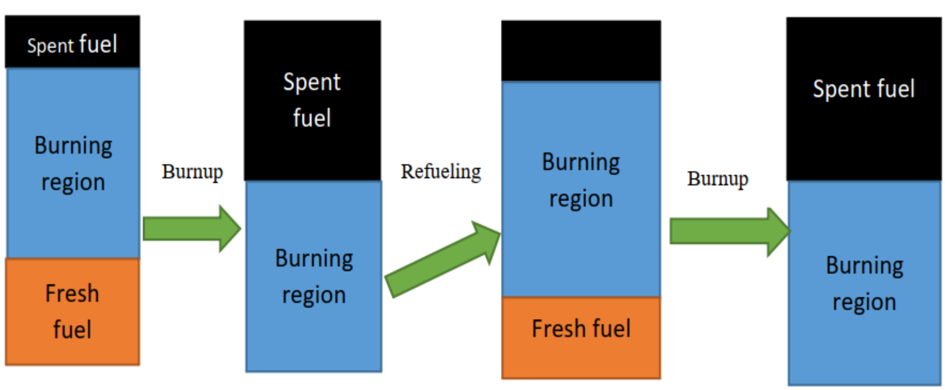


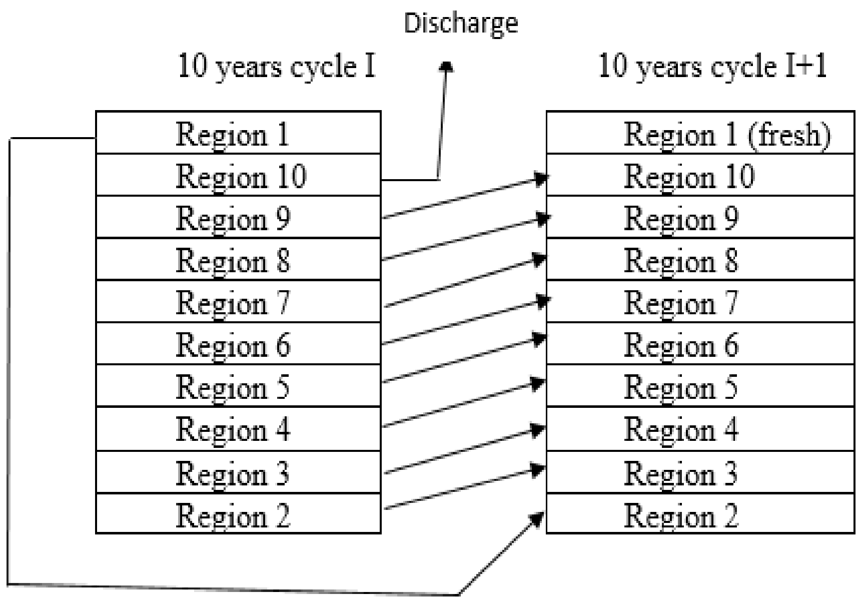
Figure 2 CANDLE burnup and refuelling strategy (see online version for colours)



2.2 Modified CANDLE (MCANDLE) burnup scheme

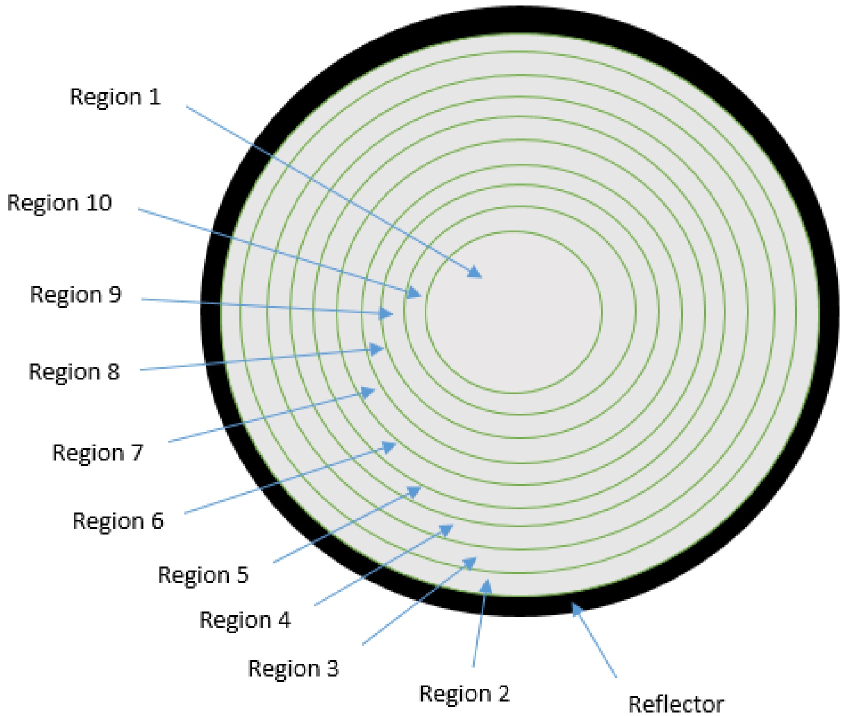
The MCANDLE burnup is a variation of the CANDLE burn-up method that adds discrete regions, elaborated by Widiawati et al. (2021b; Su'ud and Galih, 2021). The cores are separated into multiple zones of identical volume in either the axial or radial axes. Firstly, the first zone is filled with natural uranium. It is relocated to the second zone after one cycle of 10 years, and the first zone is loaded with fresh natural uranium (Su'ud and Sekimoto, 2010; Su'ud et al., 2018; Yanti et al., 2009; Widiawati et al., 2021b). This technique applies to all regions. The MCANDLE burn-up method is presented in Figure 3.

Figure 3 MCANDLE scheme in the axial direction



Several researchers at Institut Teknologi Bandung (ITB) have widely done the MCANDLE burnup development in the axial direction under the supervision of Prof. Zaki Su'ud. In addition, radial direction techniques have been investigated. The essential process remains the same as in the axial direction, except that the active core is separated into multiple radial regions. Figure 4 is an example of the MCANDLE in the radial direction. The core is separated into ten radial and then shielded with a reflector.

Figure 4 MCANDLE burn-up scheme in the radial direction (see online version for colours)



3 Methodology

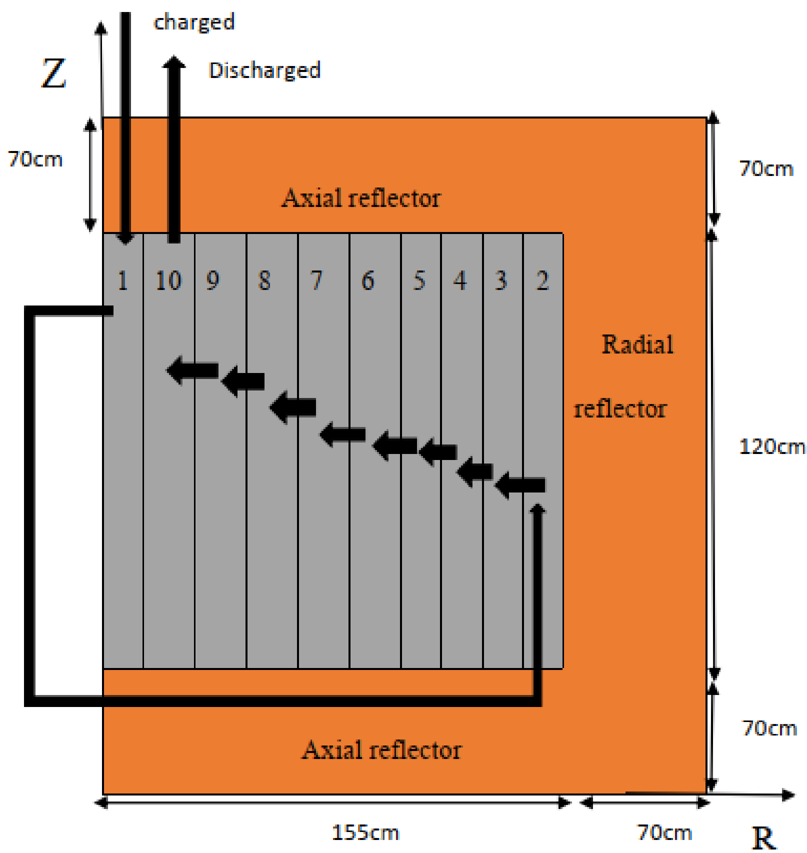
3.1 Description of reactor design

Table 1 displays the reference core's design specifications. The thermal power reactor is 400 MWt with a refuelling process every 10 years of burnup. Natural uranium (U-238) and enriched nitride (^{15}N) have been employed as fuel. The coolants employed during this study, are liquid sodium LBE and Helium gas. The cladding was HT-9 steel, while the reflector material was lead-bismuth eutectic. The fuel cell and core have cylindrical geometry. The volume fractions are 60%, 10% and 30%., fuel, cladding and coolant, respectively. The core reactor has a 120 cm axial width and 155 cm radial width. Figure 5 depicts the reactor core design and arrow shows refuelling process.

Table 1 Sample design parameters

Parameters	*Value (Description)*
Thermal power (MWt)	400
The number of identical volumes in the core	10
Refuelling	10 years
Fuel type	$U^{15}N$ (99% -^{15}N enriched)
Fuel volume percentage	60%
Coolant volume percentage	30%
Cladding volume percentage	10%
Coolant type	Na, LBE, He
Effective core axial width	120 cm
Effective core radial width	155 cm
The axial and radial widths of the reflector	70 cm
Material for reflectors	Pb-Bi
The reactor's operational life	100 years
Material used for cladding	Stainless steel HT9

Figure 5 Reactor core design (see online version for colours)



3.2 Methodology for neutronic analysis

SRAC code systems were used for the neutronic analysis (Okumura, 2007; Tsuchihashi, 1983). The collision probability (PIJ) method was used to calculate cell burnup while JENDL-4.0 was employed as a nuclear data library (Jima, 2012).

The power density in each region was taken for granted and then burnup and multi-group diffusion calculations were performed. The core has been separated into 10 regions with the equivalent volume in radial direction. Fuel was loaded in the first region, and after 10 years, it was transported to the second region, and ten years later, it was transported to the third region. These movements have taken place in all regions and fuel in tenth region got out of core. At each step of the burnup, we evaluated the fuel's macroscopic cross-section value. The outcomes were applied to solve the multigroup diffusion equation in the 2D cylinder (R-Z) geometry core of the SRAC2006 CITATION module. The average power density in each region obtained from the diffusion calculation is inserted back into SRAC code for cell burnup calculations. This iteration is performed until equilibrium is achieved (Su'ud and Galih, 2021). With 15 overlapping energy groups, the SRAC 2006 module has 107 groups of energy grouped into 74 fast energy ranges and 48 thermal energy ranges. These 74 fast energy groups have been reduced into 8 energy groups, as shown in Table 2.

Table 2 Structure of neutron groups of energy

Groups of energy	*Range of energy (eV)*	
	Upper	*Lower*
1	1.0000E+07	1.3534E+06
2	1.3534E+06	1.8316E+05
3	1.8316E+05	2.4788E+04
4	2.4788E+04	3.3546E+03
5	3.3546E+03	4.5400E+02
6	4.5400E+02	6.1442E+01
7	6.1442E+01	8.3153E+0
8	8.3153E+0	4.1399E-01

4 Calculation results analysis

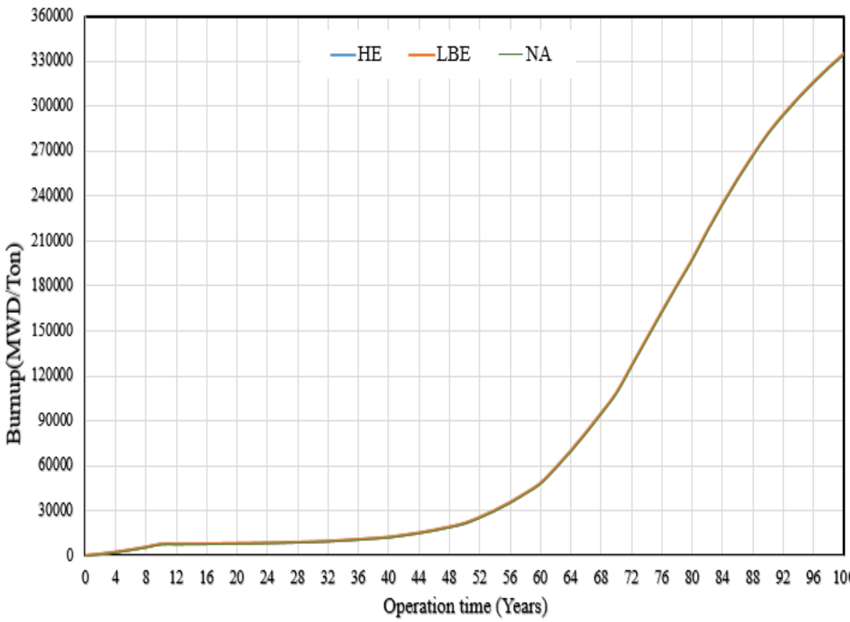
The calculations were evaluated in two steps: Calculation for the reactor core and calculation for the fuel cell. Fuel cell calculations give the k_{inf}, burn-up level, integral conversion ratio values and the atomic densities of U-238, and Pu-239 neutronic parameters. While the reactor core calculation yields various parameters, *K*-eff and relative power density.

4.1 Burn-up level

The availability of analysed burn-up data is necessary to enhance the economic of the reactor design, safety and performance (Atomic and Agency, 1992; Stacey, 2007). The quantity of energy extracted per mass of original fuel loaded is referred to as fuel burnup.

Its unit is MWd/kg or MWd/Ton (Waltar and Reynolds, 1981). Figure 6 shows the burn-up level in MWd/ton during operation time. It shows that lead-bismuth eutectic, liquid sodium and helium gas coolants cannot significantly differ in burn-up level because the power density levels assumed, are the same for all coolants during the conducting of fuel burnup. According to Figure 6, the burn-up level increases sharply at the Beginning of Life (BOL). This is because the 1st zone is close to the 10th zone, which is the most active zone. After 10 years, the burnup level gradually increases until it reaches 50 years. After 50 years, the burn-up level increases significantly. This is because the fuel entered the breeding regions. The burn-up level at the End of Life (EOL) reached up to 330GWd/ton or 33% HM, which is very high given the presence of a fast reactor. The cladding employed here is HT9, which is capable of withstanding a neutron fluence of 5×10^{23} n/cm^2 (Su'ud, Z. and Sekimoto, 2013). Because of the high burn-up level, special materials are required for each batch of shuffling or recladding techniques must be implemented.

Figure 6 Burn-up level (see online version for colours)

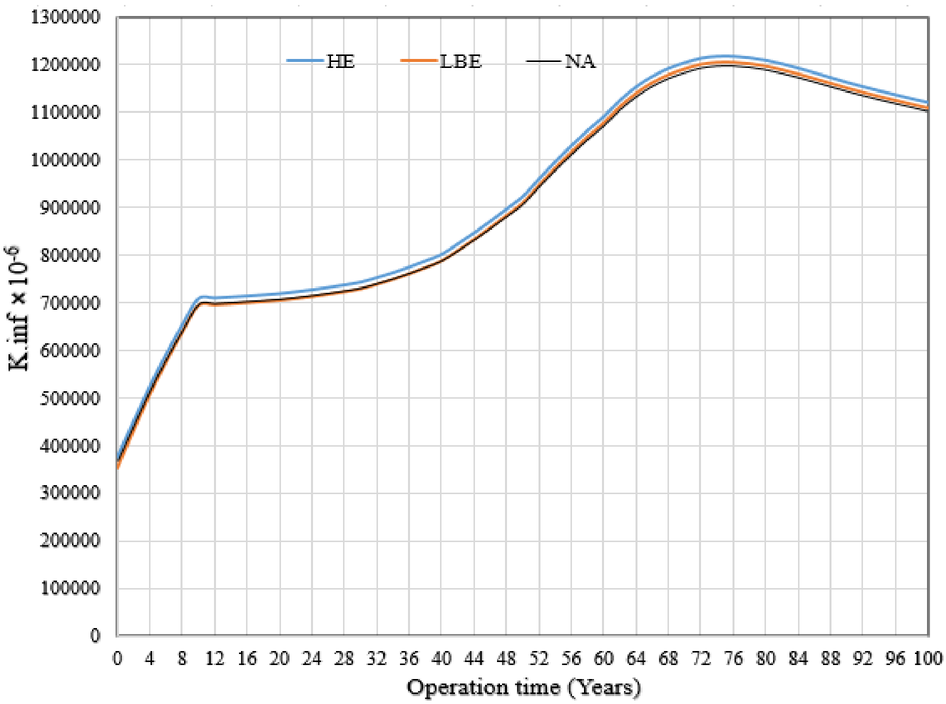


4.2 K_{inf} *or infinite multiplication factor*

The ratio of neutrons created by fission in one neutron generation to neutrons lost through absorption in the previous neutron generation, ignoring neutron leakage is called k_{inf} (James and Duderstadt, 1976). Figure 7 shows the k_{inf} changes during burnup for various coolants (helium, LBE and liquid sodium). It shows that k_{inf} changes during burnup increases sharply at the Beginning of Life (BOL). This is because the 1st zone is close to the 10th zone, known as the most active region. The k_{inf} increases dramatically after 10 years of burnup and reaches its highest value at about 74 years. It then lowers significantly because of the large collection of plutonium-239 and the large reduction of

U-238. The k_{inf} for helium gas coolant is higher than that of lead-bismuth eutectic and liquid sodium. This is because helium is an inert gas, which protects against chemical interactions with materials (Su'ud and Galih, 2021).

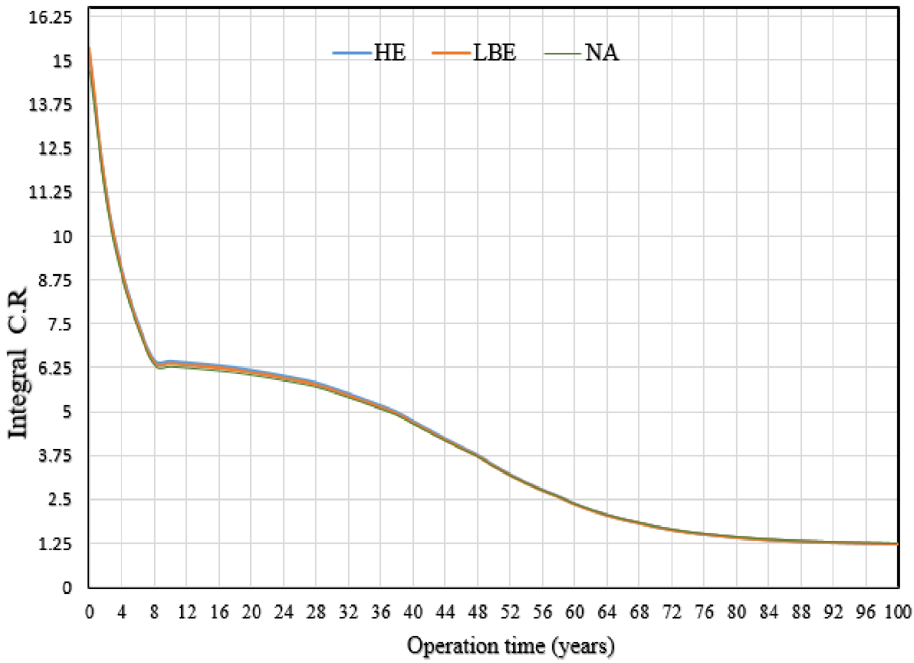
Figure 7 Infinite multiplication factor changes during burnup (see online version for colours)



4.3 Integral conversion ratio (integral CR)

The integral CR is the proportion of fissile material generated to fissile material consumed through fission or absorption (Waltar and Reynolds, 1981). Figure 8 shows the integral conversion ratio changes during burnup for various coolants (helium, LBE and sodium). The conversion ratio is shown to be significantly lowering at the BOL due to the accumulation of plutonium (Pu-239) and the reduction of natural uranium. This is because the 1st zone is close to the 10th, most active zone. After 10 years of burnup, the integral CR slowly decreases until it attains approximately 70 years of burnup history. After 70 years of burnup, the conversion ratio also decreases. This is also because of the considerable reduction of natural uranium and the accumulation of Pu-239. The figure shows that the significant difference in the integral conversion ratio caused by the coolants appears in a range of 10 to 40 years of operation time.

Figure 8 Conversion ratio changes during burnup (see online version for colours)



4.4 *Atomic density*

The number of atoms or nuclides per cm^3 is referred to as atomic density and its unit is atoms/cm^3, or atoms per 10^{24} barn cm (atom / barn cm) (Bala et al., 2018). The atomic density is a crucial aspect in determining the accuracy of the reactor core design's neutronic calculations. U-238's atomic density is depicted in Figure 9. It slowly decreases at the beginning, but after 10 years, it rapidly decreases. This is because the fuel is placed in the most active zones. There is the accumulation of plutonium-239 and a reduction of U-238. There is no significant difference in atomic density U-238 for all coolants because the power density levels assumed, are the same for all coolants.

Figure 10 depicts Pu-239's atomic density. It is demonstrated that the atomic density of plutonium-239 increased significantly at the beginning of life and gradually increased after ten years of burnup to 40 years because the 1st zone is close to the 10th, most active zone. Consequently, in the beginning (first 10 years), the accumulation of Pu-239 is high compared to the next period (10 to 40years), after 40 years of burnup, Pu-239 atomic density increased sharply until it peaked at about 76 years of burnup, then decreased because of considerable reduction of atomic density of U-238. From the figure, we see that there is no significant difference in Pu-239 atomic density for lead-bismuth eutectic coolant and that of liquid sodium coolant, however, there is a significant difference compared to that of helium coolant.

This is due to the chemical and physical properties of those coolants. The degree of coolant taking place is proportional to the atomic mass and the density (Waltar and Reynolds, 1981). LBE is heavier than sodium, and sodium is heavier than helium.

Figure 9 The atomic density of U-238 (see online version for colours)

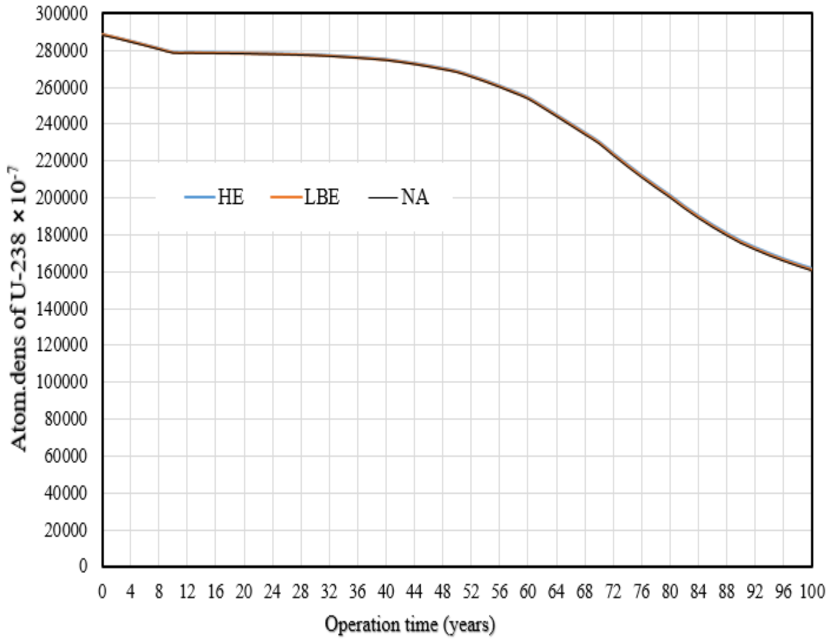
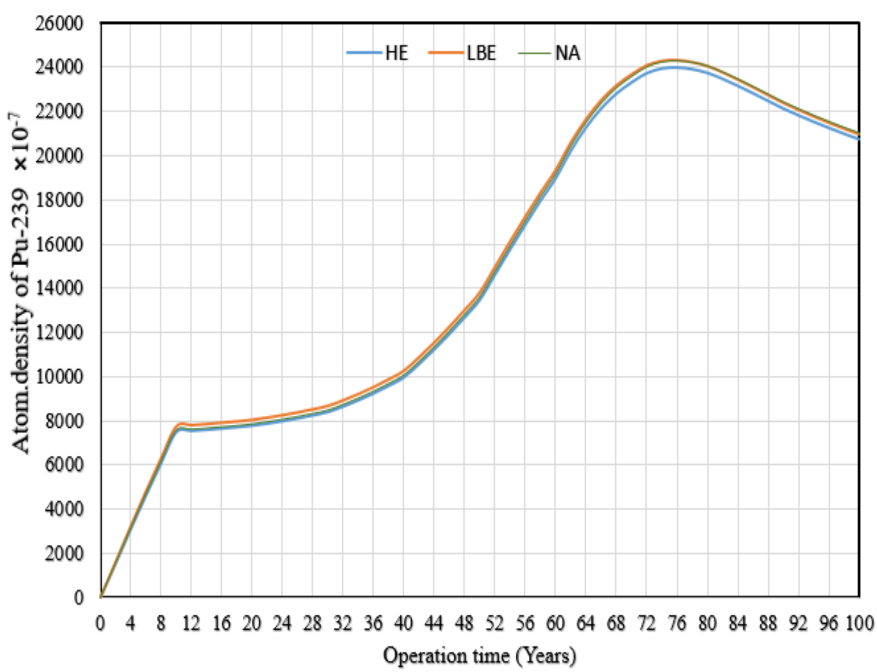


Figure 10 The change of Pu-239 atomic density (see online version for colours)



The atomic density of U-235 is depicted in Figure 11. Because the first zone is situated close to the most active zones, the U-235 burnt quickly within the first 10 years. However, after 10 years, the burning procedure of U-235 is slower until about half of its life (50 years), Following that, the rate of U-235 burning is gradually raised until it reaches around 72 years of operation time, at which point the atomic density of U-235 is reduced.

Figure 11 U-235 atomic density during operation time (see online version for colours)

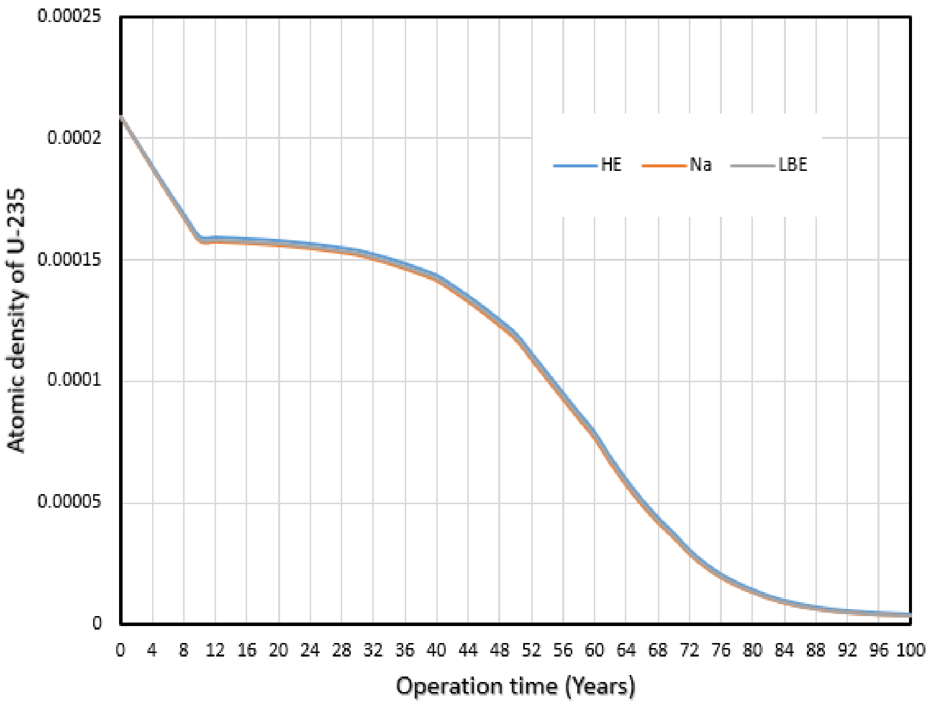


Figure 12 presents the atomic density of Pu-240. During the first 10 years, the atomic density of Pu-240 increased sharply. After 10 years of burnup until 50 years of operation time, the accumulation of the atomic density of Pu-240 is relatively slow, however, after 50 years the accumulation of the atomic density of Pu-240 grows very fast due to the diminishing of fuel. The atomic density of Pu-240 for sodium is greater than that of LBE and helium because of its ability to cool while that of helium is less than sodium and LBE because helium is an inert gas.

4.5 Neutron flux level

The total distance travelled by neutrons per unit of time in $cm^{-2}s^{-1}$ is referred to as neutron flux. Figure 13 presents the neutron flux level during operation time. The difference in flux level for helium (coolant and LBE coolant is not significant however, it is significant compared with the neutron flux level of sodium coolant. At the beginning, neutron flux level is larger than that of the period between 10 years and 50 years of operation time due to that during the first 10 years of burnup the fuel is placed in the most active zones at the right-hand side of the reactor core. It reaches the maximum value during 80 years of burnup and then decreases due to the fuel being placed in the least active regions.

Figure 12 Atomic density of Pu-240 (see online version for colours)

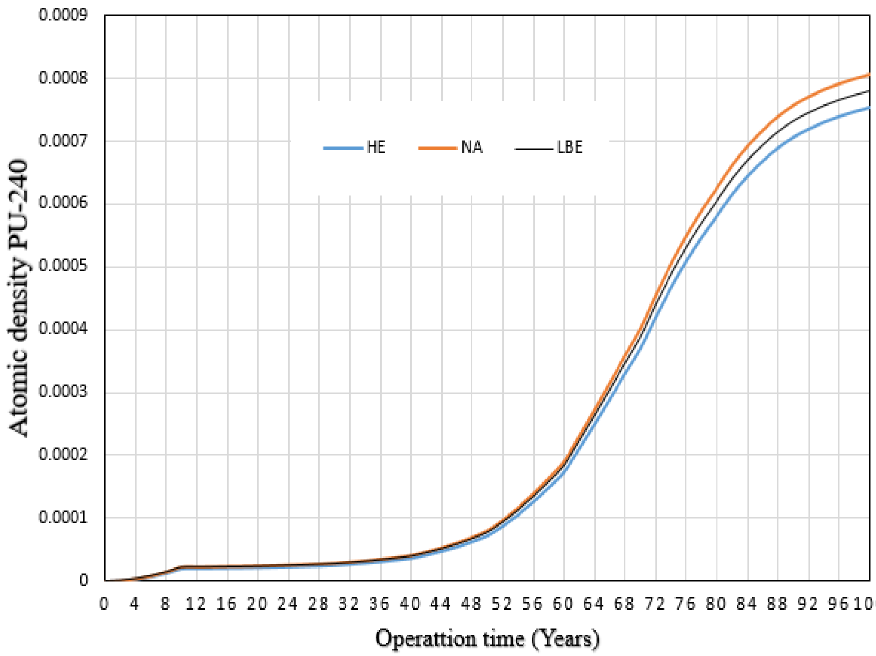
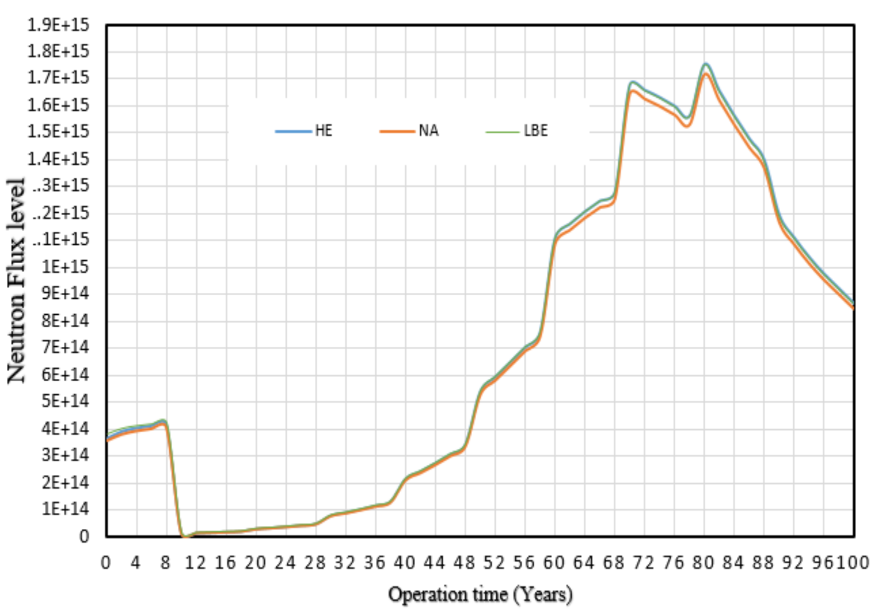


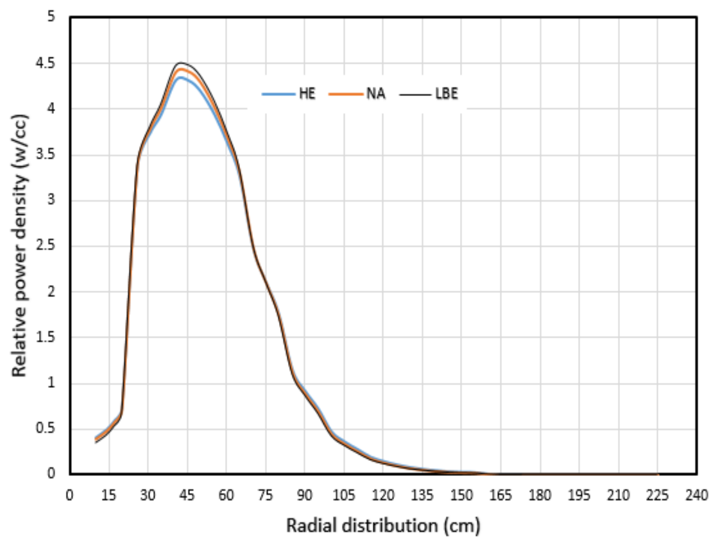
Figure 13 Neutron flux level changes during operation time (see online version for colours)



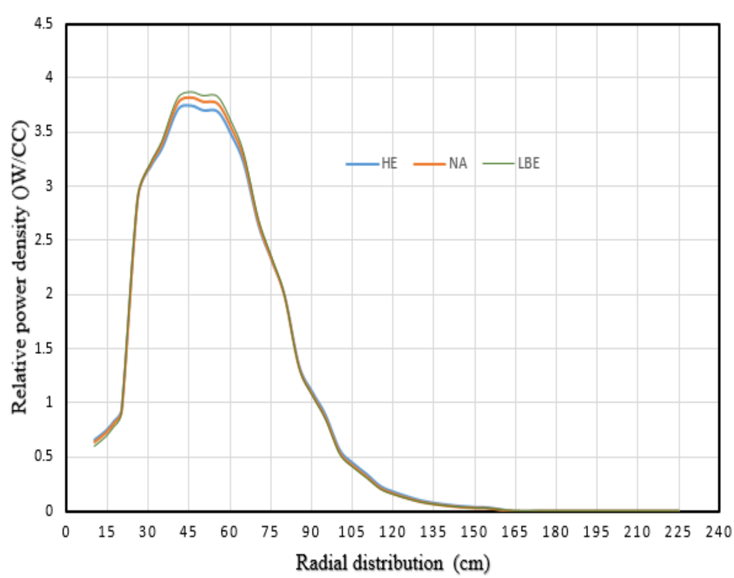
4.6 The relative power density

The quantity of energy deposited in a fissile material per unit volume per unit time is referred to as power density. Figure 14 shows the relative power density: (a) beginning of cycle, (b) middle of cycle and (c) end of cycle. Figure 14 illustrates that the relative power density of helium gas coolant is lower than that of sodium and lead bismuth eutectic (LBE) coolants.

Figure 14 Relative power density: (a) Beginning of cycle, (b) Middle of cycle and (c) End of cycle (c) (see online version for colours)

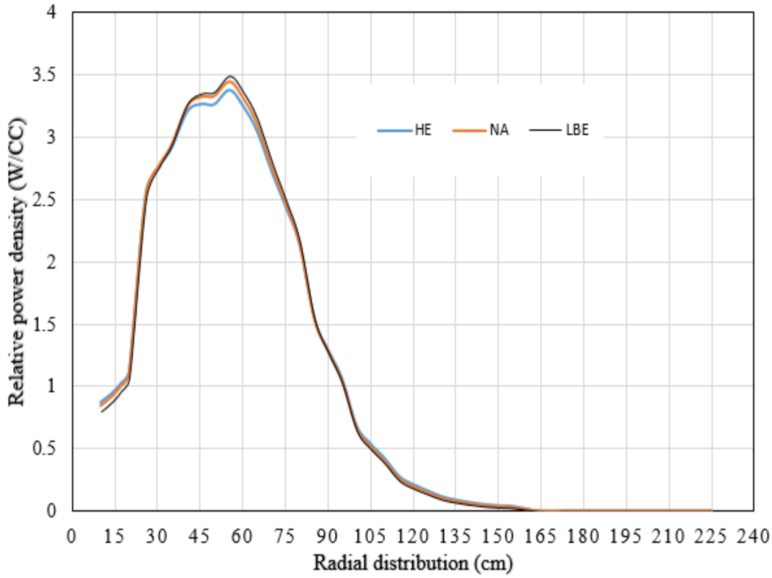


(a)



(b)

Figure 14 Relative power density: (a) Beginning of cycle, (b) Middle of cycle and (c) End of cycle (c) (see online version for colours) (continued)



(c)

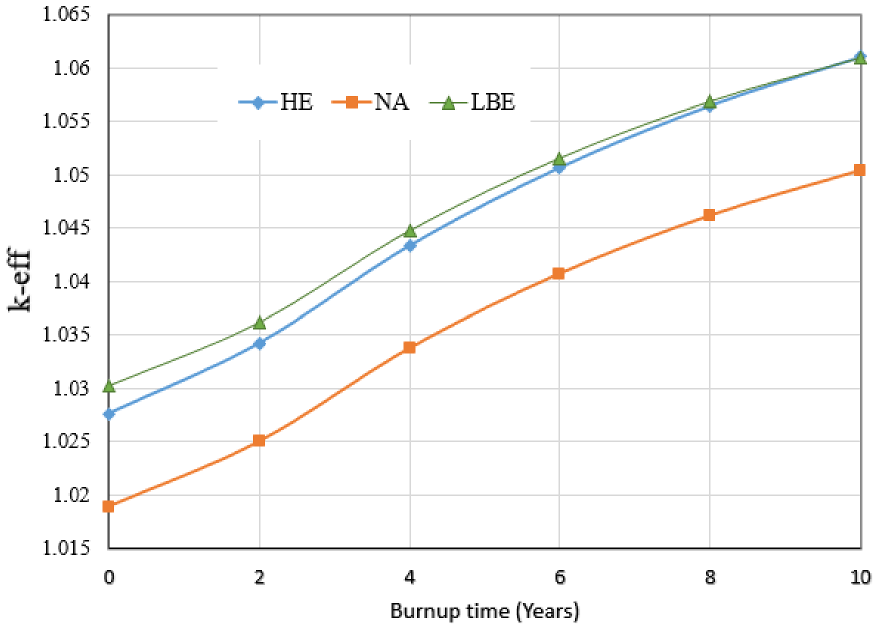
4.7 The effective multiplication factor (K-eff)

The ratio of neutrons created through fission in one generation to the amount of neutrons lost due to absorption and leakage in the previous generation is referred to as K-eff (James and Duderstadt, 1976). Figure 15 depicts the changes in K-eff during burnup. The graph shows that the *K*-eff increases monotonically. The K-eff of a LBE coolant is greater than that of a helium coolant or a liquid sodium (NA) coolant. The difference between LBE and helium becomes smaller at the EOL, while the difference between helium and sodium becomes bigger at the end of life. Table 3 displays the *K*-eff at the beginning of life. *K*-eff varies depending on the coolant used. The differences are mostly attributable to the coolants' neutron economies. The maximum *K*-eff is found in LBE whereas the lowest is found in sodium. However, the change is insignificant. Sodium outperforms the other coolants used in this study in terms of cooling performance. This is owing to its enhanced cooling ability and compatibility with stainless steel HT-9. It can boost power density while shortening the doubling time (Sekimoto, 2010).

Table 3 *K*-eff in the beginning of life given by each coolant

Coolant material	*Sodium*	*Lead Bismuth Eutectic*	*Helium*
K-eff	1.024971	1.036133	1.034226

Figure 15 Effective multiplication factor changes during burnup (see online version for colours)



5 Conclusion

This study compared the neutronics of a fast reactor with the MCANDLE burn-up scheme employing helium gas, LBE, and liquid sodium coolants. A comparison was investigated on a reactor of 400MWth power. To explore reactor behaviour, neutronic features such as burn-up level, integral conversion ratio, relative power density, flux level, infinite multiplication factor and *K*-eff were compared. The highest *K*-eff is found in the LBE, whereas the lowest is found in sodium. However, the differences are insignificant. The cooling performance of sodium is superior to that of LBE and helium coolants. This is owing to its enhanced cooling ability and compatibility with stainless steel HT-9 cladding. It can increase power density while decreasing doubling time.

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