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Neutronic analysis of sodium-cooled fast reactor design with different fuel types using modified CANDLE shuffling strategy in a radial direction

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Summary

This study compares three fuel types using neutronic analysis for use in a sodium-cooled fast reactor (SFR) design with a modified CANDLE (Constant Axial shape of Neutron flux, nuclide densities, and power shape During Life of Energy production) radial shuffling strategy. SFR is one type of generation IV reactor that is currently under investigation for commercial implementation. In this study, the SFR design utilizes natural uranium as the fuel input. The designed reactor core has a two-dimensional cylindrical geometry for each fuel mixed oxide (MOX), uranium-plutonium nitride ([U-Pu]N), and uranium-plutonium zirconium ([U-Pu]Zr). A radial shuffling strategy, using natural uranium as the fuel input, is applied to the SFR to manage the nuclear fuel burn-up process of the long-life reactor. This strategy is called the modified CANDLE burn-up scheme. The reactor core is divided into 10 regions with equal volume to represent the 10 years that the reactor operates without refueling. Initially, the first (innermost) region of the reactor core is filled with natural uranium fuel. The result of the burn-up from the first region is then shuffled into the second region. The third region is the result of the burn-up that is shuffled from the second region, and so on. This mechanism only requires natural uranium as the input for each 10-year fuel cycle. In this study, the fuel movement scheme is examined for three fuels. Global neutronic parameters, such as the multiplication factors and burn-up analyses, are observed and optimized. Overall, for an output power of 500 MWth and an active core radius of 110 cm and height of 210 cm, the study indicates that (U-Pu)N is the optimal fuel to be applied in the SFR with a modified CANDLE burn-up scheme. (U-Pu)Zr reaches a critical condition at an output power of 500 MWth with an active core radius of 130 cm and a height of 210 cm; whereas criticality for MOX is achieved at an output power of 1500 MWth with an active core radius of 210 cm and a height of 210 cm.

KEYWORDS

burn-up analysis, fuel type, modified CANDLE, multiplication factor, SFR, shuffling strategy

1 | INTRODUCTION

The nuclear community is working to improve the safety and efficiency of nuclear reactor systems. To achieve this goal, the physical phenomena in the reactor must be identified and investigated as the basis for the design and methodology of new reactors.[1] The sodium-cooled fast reactor (SFR) is one of the six types of Generation IV reactors recommended by the Generation-IV International Forum (GIF) as candidates for future reactors. The SFR is the most promising reactor type because it can achieve the Generation IV goals in a reasonable amount of time, based on experience accumulated over the years.[2] Generation IV reactors have the potential to ensure safety, economic competitiveness, reduction in environmental burden, efficient utilization of resources, resistance to nuclear proliferation, and enhanced physical protection.[3,4] Recently, design and construction work have started on active advanced SFR research projects, such as the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) in France and Prototype Generation-IV Sodium-cooled Fast Reactor (PGSFR) in South Korea.[5] The commercial Japan Sodium-cooled Fast Reactor (JSFR) was included officially in their national plan with a start-up target date of 2025[3] and operation target date of 2050.[6]

Neutronic, thermal-hydraulic, and safety analyses are required in nuclear reactor design. Neutronic review is the most crucial aspect in nuclear reactor design and simulates the actual neutron behavior inside the reactor core. The behavior of the neutron, as they react with materials, can be expressed as the neutron flux distribution, whose mathematical representation is given by the neutron transport equation.[7,8] Neutron transport involves the distribution, movement, and interaction of the neutron with the nucleus in the reactor core. The problem of neutron transport must be solved to determine the distribution of neutrons as a function of time, position, and energy. The distribution of neutrons is very influential in reactor power production. In addition to the flux distribution characteristics and neutron analysis also cover the fuel burn-up process. This burn-up study describes an effective and efficient nuclear fuel management strategy, and studies the lifetime of the nuclear reactor.

One of the approaches to improve reactor fuel efficiency is to innovate the refueling process with a shuffling strategy.[9,10] The aim of this study is to design an SFR that has the ability to operate without refueling[11] for different types of fuels over a long period of time. The modified Constant Axial shape of Neutron flux, nuclide densities, and power shape During Life of Energy production (CANDLE) burn-up shuffling strategy is a modification of the original CANDLE[10] burn-up, where the burning region moves along the direction of the core axis without changing the spatial distribution of density, nuclides, neutron flux, and power density.[12] The modified CANDLE radial shuffling strategy is implemented to regulate the use of natural uranium as a nuclear fuel so that the reactor can operate without fuel enrichment. This strategy is similar to the breed and burn reactor, also known as a traveling wave reactor (TWR).[13] Reactor core performance shuffling strategies have also been implemented along the axial,[10] radial,[13-15] and mixed axial-radial[9] directions. The burn-up process of radial fuel shuffling with natural uranium as applied to SFR technologies has been carried out in previous studies for TWRs.[13,16]

This study is conducted on an SFR design with a cylindrical core, based on a design with an output power of 500 MWth and a 10-year refueling period for an operating time of 100 years using the Standard Reactor Analysis Code (SRAC) system.[17] The effects of neutronic analysis parameters, such as the effective and infinitive multiplication factor, burn-up levels, conversion ratio, and the density of nuclides, on SFR design with different fuel types (uranium-plutonium zirconium ([U-Pu]Zr), uranium-plutonium nitride ([U-Pu]N), and mixed oxide (MOX) are investigated. MOX and (U-Pu)Zr have been studied as SFR fuels,[18] and (U-Pu)N is deemed the most promising fuel previously used in the simulation of a modified CANDLE reactor.[9,19] Such analysis with varying the fuel types with the modified CANDLE burn-up shuffling strategy in the radial direction is expected to help identify the optimal type of fuel for the SFR.

2 | DESIGN CONCEPT AND METHODOLOGY

Reactor designs using MOX[20] and (U-Pu)N[21] fuels have been studied previously, including calculations at the level of nuclear fuel cells.[22] In addition, metallic uranium-based (U-Zr) fuels have been previously considered for use in carbon dioxide gas-cooled fast reactors (GFR).[23] The initial parameters of the SFR design for this study are listed in Table 1.

The designed reactor core was divided into 10 regions that have the same volume of nuclear fuel in the radial direction. At the beginning of reactor operation, fresh fuel was loaded in the first region of the reactor core. The results of the burn-up in the first region were shuffled into the second region after 10 years of simulated burn-up. The results of burn-up in the second region were shuffled into the third region, and so on. The fuel movement continued until the results of the burn-up in the ninth region were shuffled into the 10th. Furthermore, the results of the burn-up process in the 10th region were

TABLE 1 Initial parameters of the sodium-cooled fast reactor (SFR) core design

Parameters	Specification
Cell geometry	Cylinder
Core geometry	2-D cylinder
Number of regions in radial	10 regions
Refueling period	10 years
Reactor lifetime	100 years
Output power	500 MWth
Fuel	MOX, (U-Pu)N, (U-Pu)Zr
Cladding	SS316
Coolant	Na
Configuration of coolant	Pool type
Fuel fraction	65%
Cladding fraction	10%
Coolant fraction	25%
Pin pitch	1.4 cm
Active core height	210 cm
Active core diameter	220 cm
Reflector width	50 cm

removed from the reactor core so that the first region could be filled with fresh fuel for up to 100 years of reactor operation. In other words, when fresh fuel was loaded into the first region, the fuel in the 10th region from the previous burn-up period could be removed from the core.[9,10] The same mechanism was used for MOX, (U-Pu)Zr, and (U-Pu)N fuels. Furthermore, global parameters such as the effective and infinite multiplication factors (k_{eff} and k_{inf}) and burn-up process were investigated. The mechanism of the burn-up process using the radial shuffling strategy is shown in Figure 1. The characteristics of the fuel density and smear density of each fuel type are shown in Table 2. The composition of the fresh fuels of interest in this study is shown in Table 3.

The SFR design was completed using the SRAC system.[17] The library data used in the study were from the fourth version of the Japanese Evaluated Nuclear Data Library (JENDL-4.0) with a two-dimensional RZ shuffling strategy of the cylinder core for different nuclear fuels to an output power of 500 MWth. The modified CANDLE burn-up scheme was proposed to fulfill the requirement that the reactor core maintain its criticality for 10 years of burn-up, allowing optimal neutronic calculation of the reactor core.[10] In the SRAC system, the group constant and burn-up in the nuclear fuel cell calculation was used in the collision probability calculation (PIJ) and diffusion calculation (CITATION) modules for

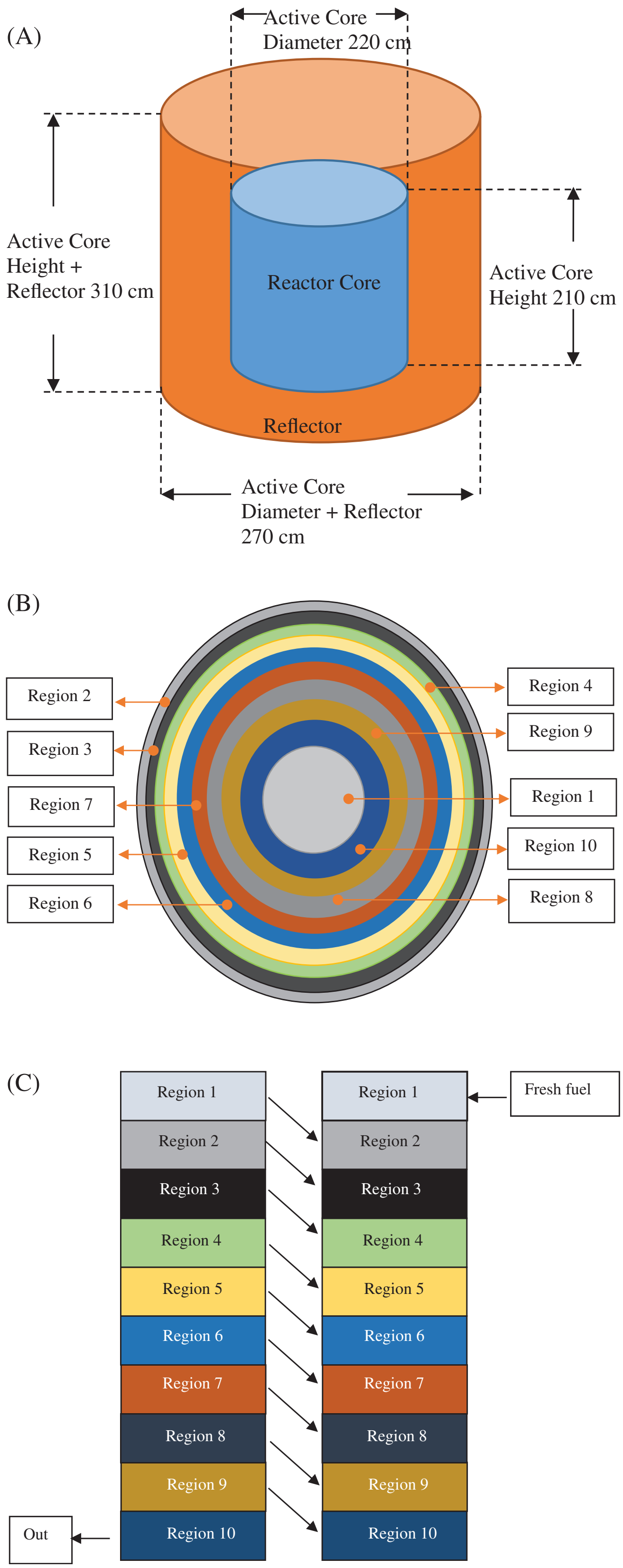

FIGURE 1 (A) Geometry and core size; (B) cross section of the sodium-cooled fast reactor (SFR) core region in the radial direction; (C) shuffling strategy in the radial direction [Colour figure can be viewed at wileyonlinelibrary.com]

TABLE 2 Density of fuel types with smear density

Fuel Types	Density (g/cm^3)	Smear Density (%)[18]
MOX	10.5[24,25]	85
(U-Pu)N	14.32[26]	85
(U-Pu)Zr	15.8[27]	85

TABLE 3 The composition of each design fuel type

Fuel types	Fuel content	Composition (%)
MOX	^{235}U	0.71
	^{238}U	99.29
	^{239}Pu	0[a]
	^{16}O	100.00
(U-Pu)N	^{235}U	0.71
	^{238}U	99.29
	^{239}Pu	0[a]
	^{15}N	100.00
(U-Pu)Zr	^{235}U	0.71
	^{238}U	99.29
	^{239}Pu	0[a]
	^{40}Zr	10.00

[a]At the beginning of the burn-up period, ^{239}Pu is not yet formed as a result of natural uranium utilization.

multi-group diffusion calculation and burn-up analysis in the reactor core. The PIJ and CITATION modules were selected for use in this study because the fuel used in the reactor core had a homogeneous distribution in each region. As the burn-up process used a modified CANDLE scheme, the reactor core was in a quasi-equilibrium state. The first neutronic calculation required to determine the distribution of power density in each region of the core reactor was the burn-up calculation. After obtaining the latest burn-up analysis, multi-group diffusion calculations were carried out to obtain the new power density distribution, which was then used to update the burn-up calculation.[10] The detailed procedure is as follows. First, cell calculation and burn-up processes such as the infinite multiplication factor (k_{inf}), burn-up level, integral conversion ratio, and fuel nuclide density of each fuel were executed by the PIJ module. In this first step, the required input data were the initial guess of the power level; diameter and geometry of the cell; type of fuel cell; fraction of fuel; structure; and coolant (see Table 1). Second, the results of cell calculations in the PIJ module were homogenized and collapsed into eight energy groups of macroscopic cross-section data. These group constants were used in the two-dimensional RZ geometry of multi-group diffusion calculations in the reactor core. These calculation processes were repeated according to the desired burn-up processes and fuel involved, and the results were saved in the user library. Data from the user library were used as input for the CITATION module to calculate the effective multiplication factor (k_{eff}) and burn-up levels in the reactor core. In this step, discretization was carried out by dividing the radius of the reactor core into 20 meshes of 10 regions. From the CITATION data, the new burn-up level values were obtained, which were later used in the PIJ module. The processes were repeated until a homogeneous power level was obtained. Homogeneous power is reached when the difference in power level in each region is less than 10^{-6}.

3 | RESULTS AND DISCUSSION

3.1 | Criticality and neutronics reactor parameters for different fuels

The neutronic analysis of the SFR design was considered for three types of nuclear fuel (metallic fuel [(U-Pu)Zr], nitride fuel [(U-Pu)N], and oxide fuel [MOX]) with an output power of 500 MWth. Figure 2 shows the results of the effective multiplication factor (k_{eff}) calculation of the first burn-up period of the reactor for each fuel. The k_{eff} value was obtained from the CITATION module.

In order to calculate the expected burn-up over a 10-year period, the calculation was repeated five times, once every 2 years. The data were calculated for an output power of 500 MWth with an active core radius of 110 cm and an active core height of 210 cm. The (U-Pu)N fuel provided the highest multiplication factor among all the fuels and was in a critical condition for the full burn-up time. The (U-Pu)Zr fuel was subcritical at the beginning of life (BOL) and reached criticality at the end of life (EOL). On the other hand, the MOX fuel was subcritical throughout the burn-up time. One of the factors that influenced reactor criticality was the atomic density of the constituent core materials. Table 4 shows that the (U-Pu)N fuel had the highest atomic density compared to the other types of fuel. Higher fuel density and a higher number of neutrons produced from fission reactions result in a higher reactor criticality level. Continuation of fission in the reactor core requires a higher fuel fraction of more extensive fissile materials. This is because, in the fast reactor, the fission reaction rate pertains only to the fuel region.[7] Figure 2 also shows that the k_{eff} of the (U-Pu)Zr fuel indicates that it is the best option for the SFR core design using the modified CANDLE burn-up strategy. However, in general, modified CANDLE reactor

FIGURE 2 Effective multiplication factor k_{eff} of first burn-up period for each fuel [Colour figure can be viewed at wileyonlinelibrary.com]

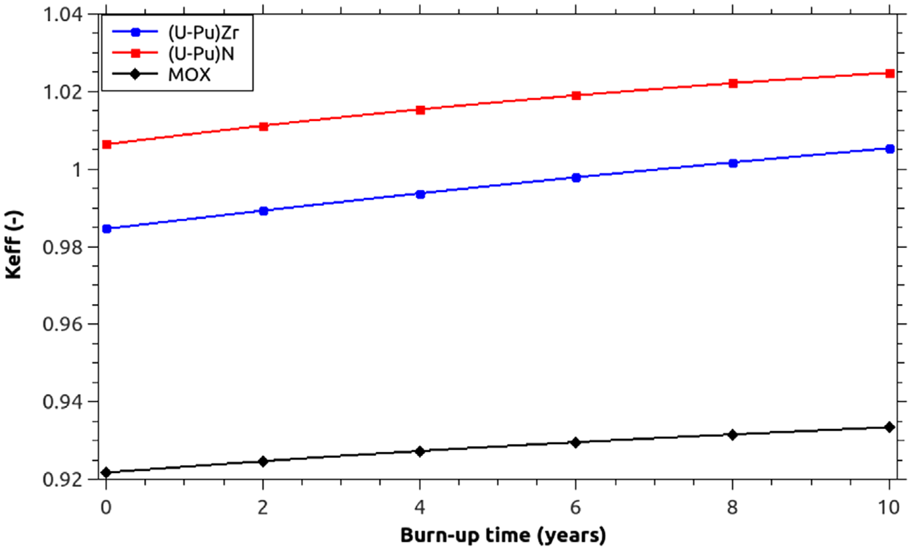

TABLE 4 Atomic density of the fuel types at the beginning of life (BOL)

Fuel types	Fuel content	Atomic density (g/cm^3) 85%^{TD}
MOX	^{235}U	1.4317E−04
	^{238}U	1.9760E−02
	^{239}Pu	0
	^{16}O	1.9904E−02
(U-Pu)N	^{235}U	2.0919E−04
	^{238}U	2.8872E−02
	^{239}Pu	0
	^{15}N	2.9083E−02
(U-Pu)Zr	^{235}U	1.7668E−04
	^{238}U	2.4385E−02
	^{239}Pu	0
	^{40}Zr	2.4563E−03

studies use (U-Pu)N as fuel in core designs.[28-31] Even though criticality was not achieved for the (U-Pu)Zr and MOX fuels, they can be used for the SFR core design with the modified CANDLE burn-up strategy. In Sections 3.2 and 3.3 of this paper, optimization is carried out for these fuels to show that the reactor can reach criticality and is expected to operate successfully for 10 years without refueling, with only natural uranium as the fuel cycle input.

Figure 3 shows the k_{inf} values of the three types of fuel. The value of k_{inf} is a ratio of the number of neutrons in one generation with the number in the previous generation without considering neutron leakage. The k_{inf} value of the (U-Pu)N fuel rapidly increased to the supercritical condition peak after 50 years. After that, it slowly decreased to the critical condition at 72 years and remained there until the EOL. The (U-Pu)Zr fuel reached the critical condition after 20 years of burn-up time and continued to increase until it began to decrease at 60 years (which was a little bit earlier than the (U-Pu)N fuel), continuing to be critical until the EOL. The MOX fuel had roughly the same pattern as the (U-Pu)Zr fuel; after reaching a critical state at 20 years, its k_{inf} continued to increase until beginning to decrease at 52 years and becoming subcritical at the EOL.

The k_{inf} value of all fuel types increased rapidly at the beginning of the first fuel burn-up period. This was caused by the presence of natural uranium in the fresh fuel.[9] At the beginning of the burn-up period, the amount of fissile material inside the reactor core is still small; however, it increases rapidly, resulting in the increase in the burn-up rate. The fertile material uranium-238 (^{238}U) produces the fissile material plutonium-239 (^{239}Pu) and therefore leads to an increased neutron population until the peak critical condition is reached. This process is caused by the reduced amount of fissile material in the reactor. As a result, the neutron population in the reactor core also decreases.

Figure 4 shows the burn-up trend for the studied fuel types as a function of operation time. The output power was set to 500 MWth. The burn-up process for all the fuel types followed a similar pattern during the reactor operating life. The burn-up level increased slowly during the early burn-up period and then rapidly in the subsequent burn-up period.[10] In the first 10 years of reactor operation or the first fuel burn-up period, the reactor was filled with fresh fuel, and the natural uranium fraction retarded the burning process. With the addition of fresh fuel every 10 years, the burn-up rate increased over the prior fuel load for the next 40 to 50 years.[9] All fuel materials in a nuclear reactor will continue to undergo reactions while the reactor operates.

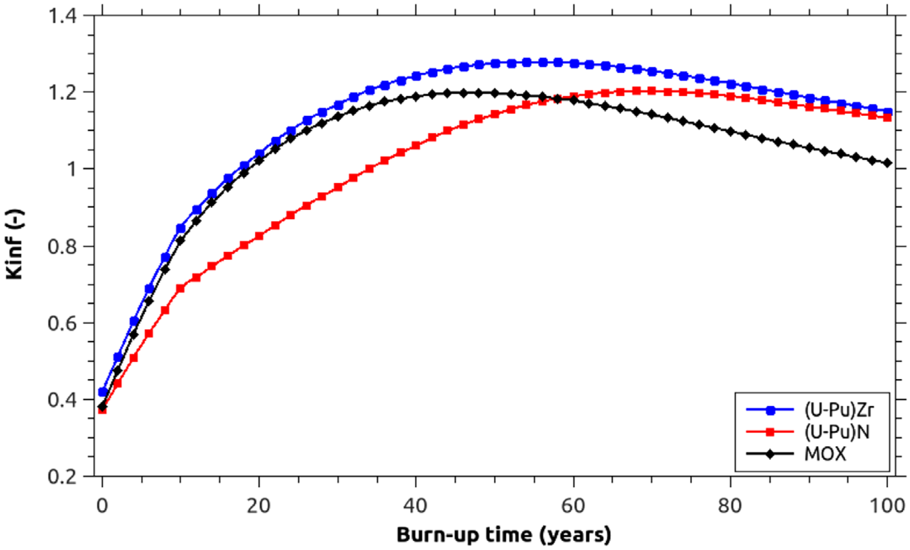

FIGURE 3 Infinite multiplication factor k_{inf} of the fuels during the reactor life [Colour figure can be viewed at wileyonlinelibrary.com]

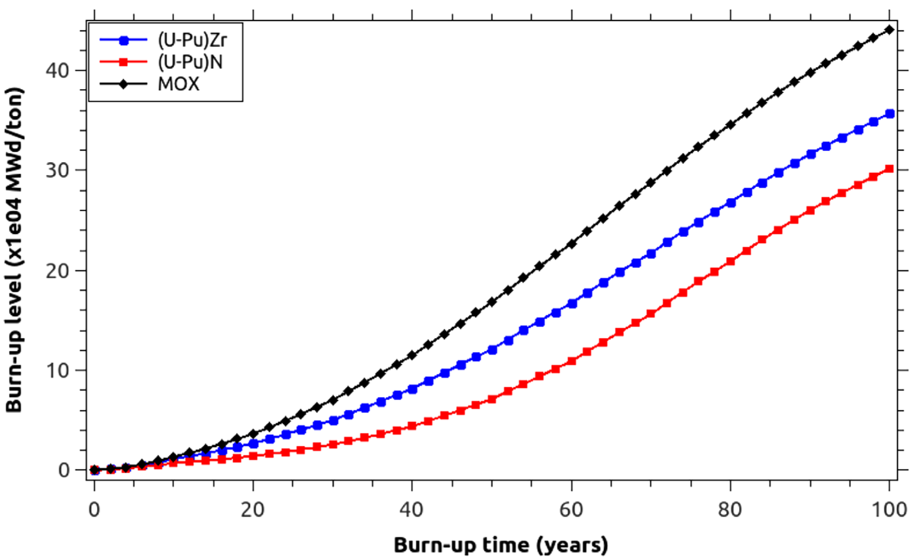

FIGURE 4 Burn-up trend of fuels during reactor life [Colour figure can be viewed at wileyonlinelibrary.com]

Figure 4 shows that the MOX fuel has the highest burn-up level. Table 2 indicates that the MOX fuel has the smallest density, and in order to achieve 500 MWth of power, more fuel must be burned to reach a critical condition. However, Figure 2 shows that, despite the high burn-up rate of the MOX fuel, it was not able to reach critical conditions during the burn-up time. The (U-Pu)N fuel had the lowest burn-up level; however, Figure 2 indicates that the amount of fuel burned in the reactor core was sufficient to maintain the reactor in a critical condition during the burn-up period. The maximum burn-up levels of the three types of fuel ([U-Pu]N: 30.15% heavy metal [HM], [U-Pu]Zr: 35.71% HM, and MOX: 44.07% HM) are relatively high given the assumed cladding material resistance. Currently, the cladding material SS316, which was used in SFR core designs, is able to withstand irradiation up to 150 displacements per atom.[32] Nagata et al[33] and Sekimoto et al[34] provide a solution for this problem, which is called a recladding strategy. In this strategy, when the cladding material has reached its maximum irradiation, it is replaced with a new cladding.

The conversion ratio compares the amount of fertile material (^{238}U) that turns into fissile material (^{239}Pu). Figure 5 shows the conversion ratio as a function of the burn-up period for the three fuel types, assuming 500 MWth output power of the reactor. The conversion ratio of all fuel types decreased from 15 for all fuel types to approximately 1 during the burn-up process. There was no significant difference in the pattern by fuel type. At the beginning of fuel burn-up when the fuel is dominated by natural uranium, the conversion ratio of the (U-Pu)Zr fuel was higher than that of MOX and (U-Pu)N fuel. Consequently, the proportion of atomic density of fertile to fissile material was substantial. The conversion

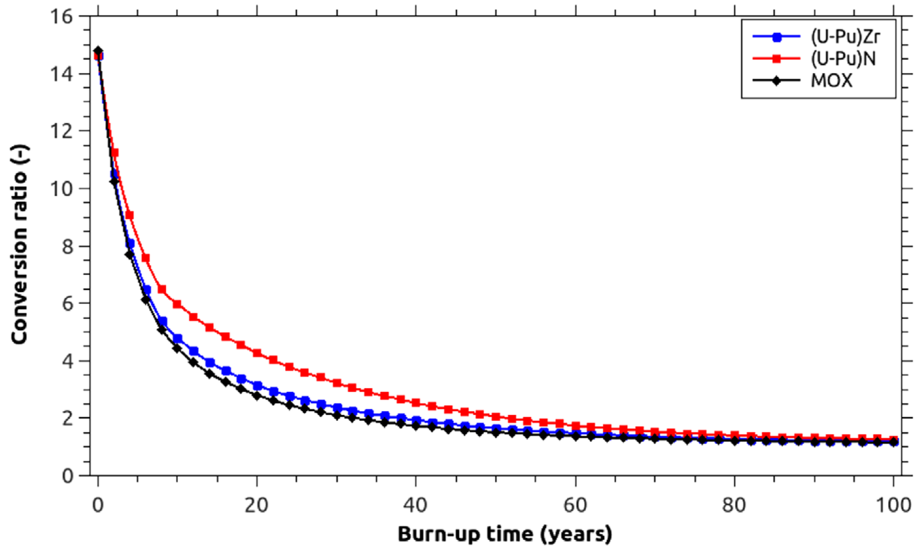

FIGURE 5 Conversion ratio of each fuel type during operation [Colour figure can be viewed at wileyonlinelibrary.com]

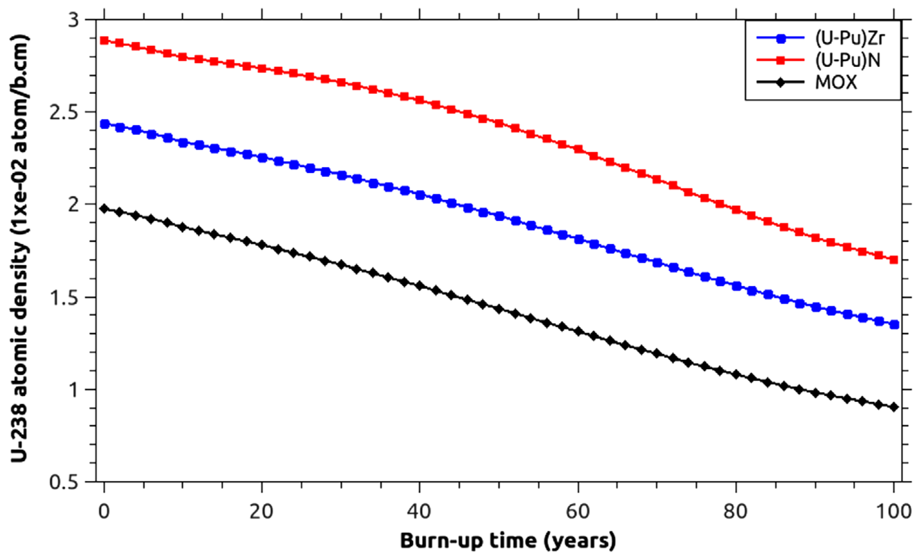

FIGURE 6 Uranium-238 (^{238}U) atomic density change during operation for each fuel [Colour figure can be viewed at wileyonlinelibrary.com]

ratio of all fuel types decreased rapidly in the first burn-up period, decreased at a slower rate through the sixth fuel burn-up period, and yet more slowly until the end of the reactor life. As the burn-up process occurs, the ^{238}U isotope density decreases, the amount of plutonium formed increases, and the atomic ratio of the fertile-to-fissile material is reduced. This causes the conversion ratio to decrease.[9] Figure 5 shows that the (U-Pu)Zr fuel for the SFR design experienced a greater change of fertile material into fissile material in the operation of the reactor than (U-Pu)N. Based on Figure 4, (U-Pu)Zr has a higher amount of fissile material in the core than (U-Pu)N; this was proportional to the density of each fuel. If the fuel density is high, the change in ^{238}U to ^{239}Pu is also considerable.

Figure 6 shows the change in the atomic density of the fertile material ^{238}U during the operation time of the reactor. The atomic density of ^{238}U for all types of fuel decreased steadily throughout reactor operation due to the fission process in the reactor core.

Figure 7 shows the change in the atomic density of the fissile material ^{239}Pu during operation of the reactor. At the beginning of reactor operation, the atomic density of ^{239}Pu for each fuel was zero because it had not yet been produced. After 40 years of reactor operation time, the ^{239}Pu atomic density had increased rapidly due to the conversion of fertile into fissile material. Larger amounts of ^{239}Pu increase the criticality of the reactor.

Based on Figures 6 and 7, the (U-Pu)N fuel had the highest atomic density of both ^{238}U and ^{239}Pu, followed by (U-Pu)Zr, and the lowest densities belonged to MOX during the burn-up time. The atomic densities of ^{238}U and ^{239}Pu were approximately proportional to the atomic density of each fuel, as shown in Table 3.

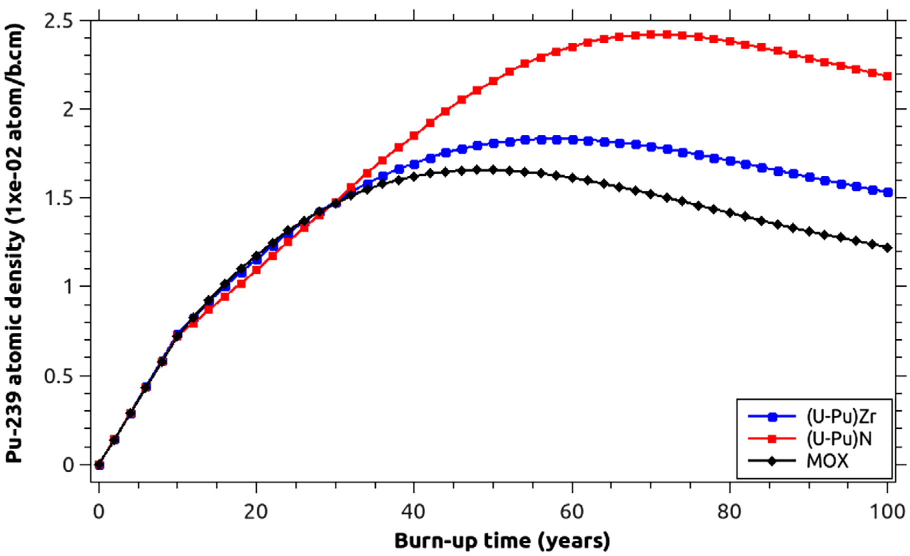

FIGURE 7 Plutonium-239 (^{239}Pu) atomic density change during reactor operation for each fuel [Colour figure can be viewed at wileyonlinelibrary.com]

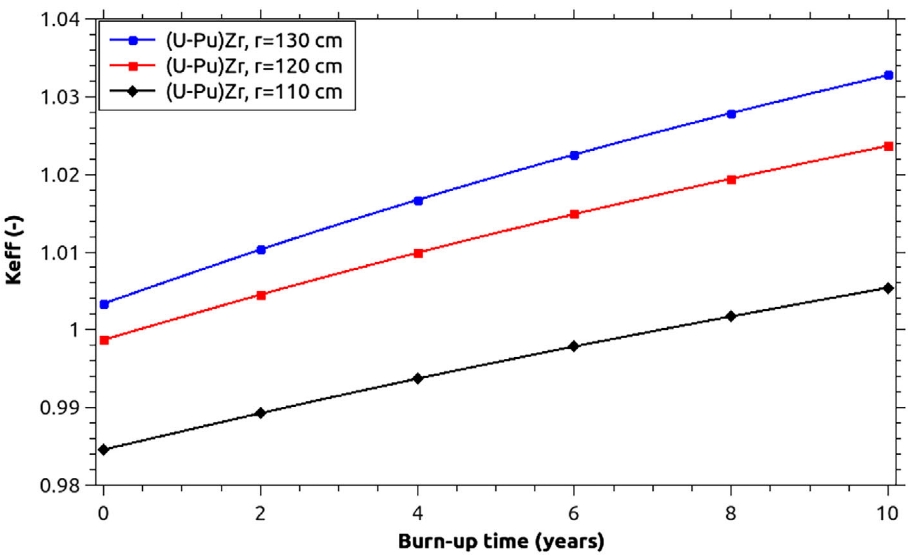

FIGURE 8 The effective multiplication factor k_{eff} of the first burn-up period of uranium-plutonium zirconium ((U-Pu)Zr) fuel for $r = 110$, 120, and 130 cm [Colour figure can be viewed at wileyonlinelibrary.com]

3.2 | Optimization of (U-Pu)Zr fuel in SFR design using modified CANDLE strategy

Initially, the SFR with the (U-Pu)Zr fuel was designed with a 110 cm active core radius and 210 cm active core height for 500 MWth output power. The data showed that the reactor was in the subcritical condition at the BOL, but it reached critically by the EOL. Hence, the SFR design was adjusted to allow the reactor to reach criticality more efficiently along the burn-up time. The optimization of the SFR was carried out by increasing the value of the active core radius in 10 cm increments, while maintaining a constant active core height of 210 cm. The results are shown in Figures 8 to 10. Figure 8 shows that the SFR with (U-Pu)Zr fuel reached a critical condition quicker than the other fuels at 130 cm of active core radius. At an active core radius of 120 cm, the reactor was still in a subcritical condition at BOL, though it reached criticality during the following year of the burn-up period. Figures 9 and 10 show the integral conversion ratio and atomic density of ^{239}Pu, respectively, which both increased as the active core radius was increased.

3.3 | Optimization of MOX fuel in SFR design using modified CANDLE strategy

MOX is a great fuel candidate for SFR because it has the advantages of simplicity and fewer process steps in fabrication, a high melting point, and good chemical stability.[18] However, in reactors using the modified CANDLE strategy with only natural uranium as the fuel cycle input, optimization is needed to allow the reactor to

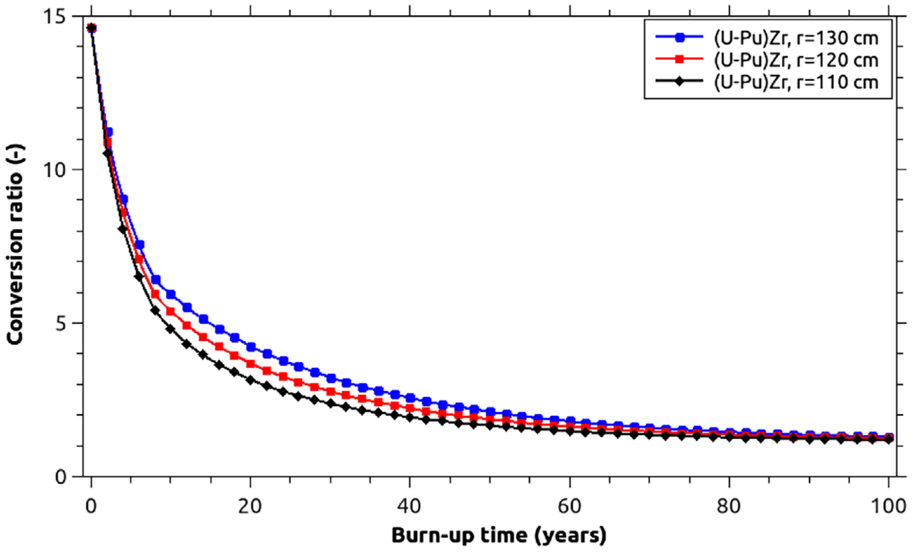

FIGURE 9 Conversion ratio of uranium-plutonium zirconium ((U-Pu)Zr) fuel for r = 110, 120, and 130 cm [Colour figure can be viewed at wileyonlinelibrary.com]

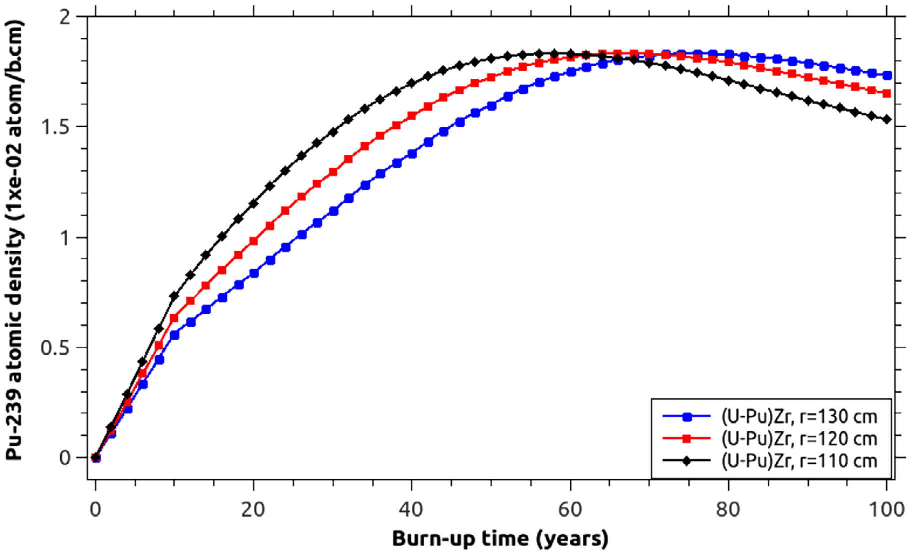

FIGURE 10 Plutonium-239 (^{239}Pu) atomic density change during operation for uranium-plutonium zirconium ((U-Pu)Zr) fuel for r = 110, 120, and 130 cm [Colour figure can be viewed at wileyonlinelibrary.com]

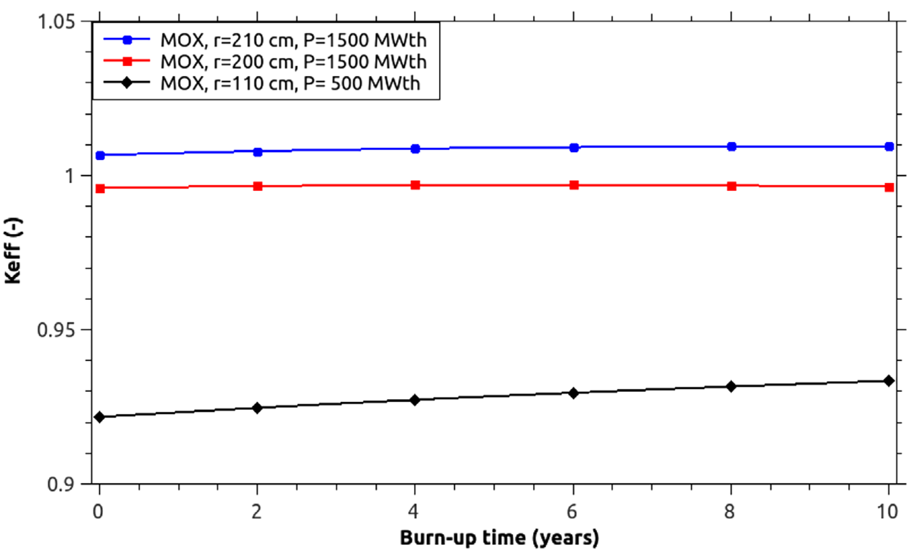

FIGURE 11 The effective multiplication factor k_{eff} of the first burn-up period of mixed oxide (MOX) fuel for active core radius 110, 200, and 210 cm [Colour figure can be viewed at wileyonlinelibrary.com]

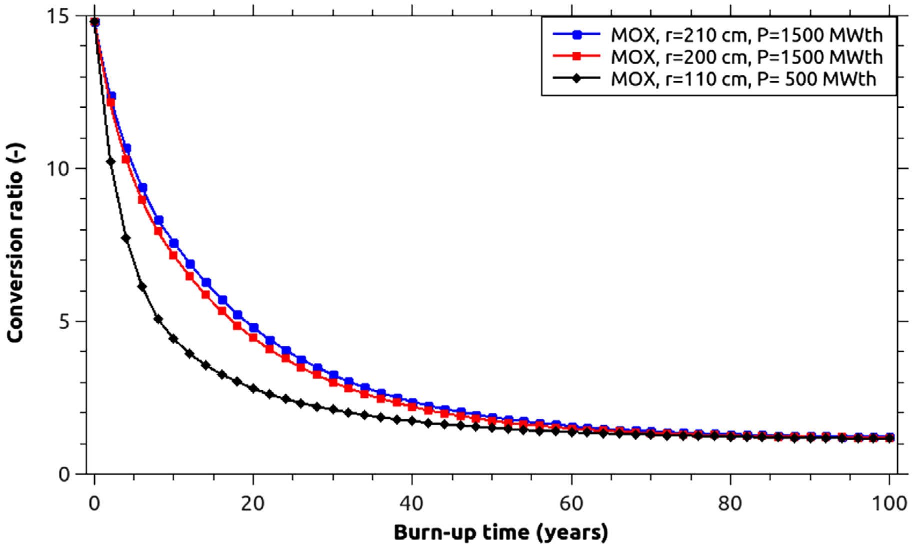

FIGURE 12 Conversion ratio of mixed oxide (MOX fuel) for active core radius 110, 200, and 210 cm [Colour figure can be viewed at wileyonlinelibrary.com]

reach critical conditions during the burn-up period. Initially, the reactor core was designed with a radius of 110 cm and height of 210 cm for a 500 MWth output power. In the previous section, optimization of the (U-Pu)Zr fuel was successfully carried out by increasing the core radius by 20 cm, but this was not the case for MOX fuel. Criticality in the core was not obtained even when the radius of the core was increased to 200 and 210 cm.

As a result, the reactor output power was increased to 1500 MWth in addition to the increased active core radius, and the results are shown in Figure 11. Figure 11 shows that the reactor reached a critical condition when the core radius was 210 cm, the height was 210 cm, and the output power was 1500 MWth. Figure 12 shows the integral conversion ratio, which increased significantly as the radius and output power of the reactor were increased.

4 | CONCLUSIONS

The modified CANDLE radial shuffling strategy was successfully used to design an SFR to manage fuel burn-up with natural circulation in a long-life reactor. The reactor core design utilized natural uranium in the form of two-dimensional cylinders with three fuels: MOX, (U-Pu)N, and (U-Pu)Zr. The three types of fuel were modeled using a shuffling burn-up strategy that was optimized for a burn-up period of 10 years. The data obtained from the models demonstrate that the reactor will work well using this strategy for 100 years of reactor operation. Global neutronic parameters, such as the effective (k_{eff}) and infinite (k_{inf}) multiplication factor, and burn-up analyses, such as the burn-up level, conversion ratio, and atomic density of ^{238}U and ^{239}Pu, were observed and optimized. Overall, for an output power of 500 MWth, 110 cm of active core radius, and 210 cm of active core height, this study indicates that (U-Pu)N fuel is the optimal SFR fuel and has a higher chance of being utilized for SFR design with a modified CANDLE burn-up strategy. The (U-Pu)Zr fuel with an optimized core radius of 130 cm would also be effective if utilized in an SFR. Finally, after optimization of the active core radius to 210 cm and increasing the power to 1500 MWth, the MOX fuel could also be a good candidate fuel in an SFR with a modified CANDLE burn-up strategy.

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DATA AVAILABILITY STATEMENT

The data that support the findings of this study are openly available in [repository name e.g., "figshare"] at http://doi.org/[doi], reference number [reference number].

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REFERENCES

1. Zhang K. The multiscale thermal-hydraulic simulation for nuclear reactors: a classification of the coupling approaches and a review of the coupled codes. *Int J Energy Res*. 2020;44(5):3295-3315. https://doi.org/10.1002/er.5111.

2. Agency, OECD Nuclear Energy SC for N. Benchmark for neutronic analysis of sodium-cooled fast reactor cores with various fuel types and core sizes. Nuclear Energy Agency; February 2016:87.
3. Ichimiya M. The status of generation IV sodium-cooled fast reactor technology development and its future project. *Energy Procedia*. 2011;7:79-87. https://doi.org/10.1016/j.egypro.2011.06.011.
4. Korkmaz ME, Agar O. The investigation of burnup characteristics using the serpent Monte Carlo code for a sodium cooled fast reactor. *Nucl Eng Technol*. 2014;46(3):407-412. https://doi.org/10.5516/NET.07.2013.050.
5. You W, Hong SG. An advanced sodium-cooled fast reactor core concept using uranium-free metallic fuels for maximizing TRU burning rate. *Sustainability*. 2017;9(12):1-21. https://doi.org/10.3390/su9122225.
6. Takeda T. Improvement of neutronics calculation methods for fast reactors. *Prog Nucl Sci Technol*. 2011;2:289-293. https://doi.org/10.15669/pnst.2.289.
7. Shafii MA. Analysis of neutron fission reaction rate in the nuclear fuel cell using collision probability method with non flat flux approach. *J Phys Conf Ser*. 2017;877(1):012013. https://doi.org/10.1088/1742-6596/877/1/012013.
8. Shafii MA, Usman J, Tongkukut SHJ, Abdullah AG. The Pij matrix and flux calculation of one-dimensional neutron transport in the slab geometry of nuclear fuel cell using collision probability method. *MATEC Web Conf*. 2018;197:10-13. https://doi.org/10.1051/matecconf/201819702006.
9. Su'ud Z, Miftasani F, Ilham M, Sarah A, Ariani M, Sekimoto H, Waris A, Permana S. Design study of small modified CANDLE based long life gas cooled fast reactors. *Energy Procedia*. 2017;131:6-14. https://doi.org/10.1016/j.egypro.2017.09.439.
10. Su'ud Z, Ilham M, Widiawati N, Sekimoto H. Modified CANDLE burnup calculation system, its evolution, and future development. *J Phys Conf Ser*. 2018;1090(1):1-11. https://doi.org/10.1088/1742-6596/1090/1/012006.
11. Şahin S, Şahin HM, Al-Kusayer TA, Sefidvash F. An innovative nuclear reactor for electricity and desalination. *Int J Energy Res*. 2011;35(2):96-102. https://doi.org/10.1002/er.1772.
12. Sekimoto H. Basic theory of CANDLE burn-up. International Conference on the Physics of Reactors 2008, PHYSOR 08. 2008; 2 (September 2008):1533-1540.
13. Zhang D, Zheng M, An H, et al. Numerical studies of stepwise radial fuel shuffling in a traveling wave reactor. *Sci China Technol Sci*. 2014;57(6):1229-1237. https://doi.org/10.1007/s11431-014-5514-4.
14. Chen XN, Zhang D, Maschek W. Fundamental burn-up modes of radial fuel shuffling. *Int Conf Math Comput Methods Appl Nucl Sci Eng*. 2011;1-11.
15. Chen XN, Kiefhaber E. Asymptotic solutions of serial radial fuel shuffling. *Energies*. 2015;8(12):13829-13845. https://doi.org/10.3390/en81212398.
16. Zhang D, Chen XN, Flad M, Rineiski A, Maschek W. Theoretical and numerical studies of TWR based on ESFR core design. *Energy Convers Manage*. 2013;72:12-18. https://doi.org/10.1016/j.enconman.2013.01.037.
17. Okumura K, Kugo T, Kaneko K, Tsuchihashi K. SRAC 2006: A comprehensive neutronics calculation code system, SRAC_JAEA-Data-Code-2007-004.pdf.
18. International Atomic Energy Agency. Status and trends of nuclear fuels technology for sodium cooled fast reactors. IAEA Nuclear Energy Series; 2011;(NF-T-4.1): IAEA Nuclear Energy Series No. NF-T-4.1.
19. Description V. International conference on advances in nuclear science and engineering 2015. *J Phys Conf Ser*. 2017;799:011001. https://doi.org/10.1088/1742-6596/799/1/011001.
20. Rochman D, Koning AJ, da Cruz DF, Sjöstrand H. Uncertainties on reactivity for UO_2 and MOX systems due to nuclear data; January 2014:22.
21. Novalianda S, Ariani M, Monado F, Su'ud Z. Neutronic design of uranium-plutonium nitride fuel-based gas-cooled fast reactor (GFR). *J Pendidik Fis Indones*. 2018;14(2):92-98. https://doi.org/10.15294/jpfi.v14i2.6602.
22. Shafii MA, Su'ud Z, Waris A, Kurniasih N. Nuclear fuel cell calculation using collision probability method with linear non flat flux approach. *World J Nucl Sci Technol*. 2012;2(2):49-53. https://doi.org/10.4236/wjnst.2012.22008.
23. Susanty E, Ariani M, Su'Ud Z, Monado F. Calculation of burnup fuel cell uranium metallic with carbon dioxide cooled. *J Phys Conf Ser*. 2019;1282(1):012028. https://doi.org/10.1088/1742-6596/1282/1/012028.
24. Şahin S, Şahin HM, Acir A. Criticality and burn up evolutions of the fixed bed nuclear reactor with alternative fuels. *Energy Convers Manage*. 2010;51(9):1781-1787. https://doi.org/10.1016/j.enconman.2009.12.044.
25. Helgesson P, Rochman D, Sjöstrand H, Alhassan E, Koning A. UO_2 versus MOX: propagated nuclear data uncertainty for k_{eff}, with burnup. *Nucl Sci Eng*. 2014;177(3):321-336. https://doi.org/10.13182/NSE13-48.
26. Rogozkin BD, Stepennova NV, Fedorov YE, et al. (2003) Mononitride U–Pu mixed fuel and its electrochemical reprocessing in molten salts. IAEA-TECDOC-1348.
27. Kim KH, Kim JH, Oh SJ, Lee JW, Lee HJ, Lee CB. Fabrication of U-10 wt.% Zr metallic fuel rodlets for irradiation test in BOR-60 fast reactor. *Sci Technol Nucl Install*. 2016;2016:1-7. https://doi.org/10.1155/2016/4385925.
28. Widiawati N, Su'ud Z, Irwanto D, Permana S, Takaki N, Sekimoto H. Enhancing the performance of a long-life modified CANDLE fast reactor by using an enriched 208Pb as coolant. *Nucl Eng Technol*. 2021;53:423-429. https://doi.org/10.1016/j.net.2020.07.008.
29. Ariani M, Su'ud Z, Waris A, Asiah N, Shafii MA. Effect of fuel fraction on small modified CANDLE burn-up based gas cooled fast reactors. *AIP Conf Proc*. 2010;1325:249-252. https://doi.org/10.1063/1.3537909.
30. Su'ud Z, Sekimoto H. The prospect of gas cooled fast reactors for long life reactors with natural uranium as fuel cycle input. *Ann Nucl Energy*. 2013;54:58-66. https://doi.org/10.1016/j.anucene.2012.09.014.
31. Irka FH, Aryani M, Suud Z, Sekimoto H. Design study of gas cooled fast reactor with natural uranium as fuel cycle input using radial shuffling strategy. *AIP Conf Proc*. 2012;1448:74-81. https://doi.org/10.1063/1.4725440.

32. IAEA. IAEA nuclear energy series structural materials for liquid metal cooled fast reactor fuel assemblies—operational behaviour. *IAEA Nucl Energy Ser*. 2012;NF-T-4.3:103.
33. Nagata A, Takaki N, Sekimoto H. A feasible core design of lead bismuth eutectic cooled CANDLE fast reactor. *Ann Nucl Energy*. 2009;36:562-566.
34. Sekimoto H, Nagata A. Performance optimization of the CANDLE reactor for nuclear energy sustainability. *Energy Convers Manage*. 2010;51:1788-1791.

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