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Investigation of the Use of Alternative Thorium-based Fuel in Generation-IV Fast Reactor

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Abstract

The future demand on energy sources is expected to increase drastically. Of the available energy sources, nuclear energy is considered a promising candidate for satisfying this demand with minimum environmental impacts. However, the sustainability of this source relies on the availability of nuclear fuel. The main aim of this work is to analyze the use of thorium as alternative fuel in one of generation IV fast reactors. The selected design for this work is the Gas cooled Fast Reactor. MCNPX transport code was used to design a representative fuel assembly of this advanced concept to simulate the neutronic performance of thorium-based fuel compared to the reference uranium-based fuel. The results of fuel burnup have showed that the use of thorium-based fuel can lead to a smaller reactivity swing during the fuel cycle which allow for a longer fuel burnup duration compared to the reference fuel.

Keywords: *Thorium; Fast Reactors; GFR2400; MCNPX; GEN-IV; Safety Parameters*

1. Introduction

Currently, a number of nations have taken the lead in developing what are known as generation IV (GEN-IV) reactors. Ten of these nations chose six cutting-edge systems for study and development in 2002. By 2030, these reactors should be at a commercially viable stage and prepared for deployment (GENIV, 2017). The GEN-IV concepts promote to be more advanced in terms of security, sustainability, economics, and proliferation resistance than the preceding generations. In terms of the sustainability objective, these cutting-edge systems will enable a better and more effective use of the available natural resources, which will allow for extending the life of nuclear energy. However, for sustainability goal to be fully fulfilled by GEN-IV concepts, it is necessary to search for alternative fuels.

The future demand on energy sources is expected to increase with high rate. Of the available energy sources, nuclear energy is thought of as an excellent candidate for satisfying this demand with minimum environmental impact (IAEA, 2016). These days, mixed oxide (MOX) and uranium dioxide (UO_2) are used in the majority of nuclear fuel cycles. However, over the crust of the Earth, thorium ore is thought to be 3–4 times more common than uranium ore (IAEA, 2005). As a result, it serves as an excellent alternate fuel for both fast and thermal reactors. The use of thorium alone, however, cannot start a fission chain reaction since it is not a fissile isotope. Nevertheless, neutron capture can transform it into the fissile isotope ^{233}U. Fast spectrum reactors are a suitable choice for carrying out this procedure since the high neutron flux increases the likelihood of the breeding process.

To investigate the feasibility of using alternative thorium-based fuel in GEN-IV fast reactor, the Gas-cooled Fast Reactor (GFR2400) was selected as the design for this investigation. Using the MCNPX Monte Carlo transport code, a representative fuel assembly for this cutting-edge concept was constructed. The designed model is employed to perform a comparative neutronic analysis

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between the performance of the thorium-based fuel and the reference uranium-based fuel. Recent research on thorium has demonstrated its potential for use in GEN-IV advanced reactors. Related to Gas cooled Fast Reactors (GFRs), (Ibrahim, A. et al. 2018) investigated the feasibility of the use of thorium as a fuel in the gas cooled fast reactor GFR2400 under open and closed fuel cycles scheme. Also, (Y. Lima-Reinaldo et al. 2019) studied the use of thorium under a single batch irradiation scheme.in heterogeneous GFR2400 core model. (György,H,. and Czifrus, S., 2016) used infinite lattice models of different GEN-IV fuel assemblies to analyze the feasibility of heterogeneous use of thorium in different configurations. Although, a number of GEN-IV concepts have been investigated and analyzed in the previous literature, an analysis of neutronic and safety parameters using different transport code like MCNPX can support and give better understanding of the results previously obtained for the investigated concept.

1.1. Gas-cooled Fast Reactor

The investigated Gas-cooled Fast Reactor is the GFR2400. The design of this concept has been proposed by the French Atomic Energy Commission (Richard, P., et al., 2010). It is a power scale reactor of thermal power 2400 MWth with ceramic fuel pin element and employ helium gas as a coolant (Zabiego, M., et al., 2013; Perko, Z., et al., 2015). Helium gas has been chosen as coolant for its transparency and very small absorption cross section; besides it is chemically inactive and thus there is no interaction with structural materials.

The GFR2400 core consist of 516 hexagonal FAs (Ibrahim, A., et al., 2022). These assemblies are divided into two core zones: the inner core contains 264 FA while the outer one contains 252 FA. The fuel in these assemblies is made of (U, Pu)C pellets of active length of 165 cm. However, each core zone has different volume fraction of fissile plutonium namely, 17.65% for outer and 14.12% for inner core zone. The reason for the use of different enrichments is to have a flat power distribution across the core by end of cycle (EOC). The fuel assembly contains a triangular arrangement of 217 fuel pins within a hexagonal wrapper made of ceramic SiC which is also used as the cladding material for the fuel pins. And, to strength the clad a thin rhenium layer is used as an inner coating of the clad and a gap is left between the clad and the fuel pellets to prevent the interaction between them (Perko, Z., et al., 2015). The modelled design of the fuel assembly and fuel pin are shown in Fig. 1 while Table 1 depicts the dimensions of GFR2400 fuel assembly.

According to the reference design, natural uranium is used to form the (U, Pu)C fuel while a twice recycled mixed oxide (MOX) fuel is used as fissile material (Table 2). In this analysis, the PuC content in the fuel is 15.81 w% which represent the average value of core enrichments. The fuel average density of the mixed oxide fuel (MOX) is 10.90 g.cm^{-3} which represents 80.0% of the TD accounting for the porosity of the ceramic fuel pellets.

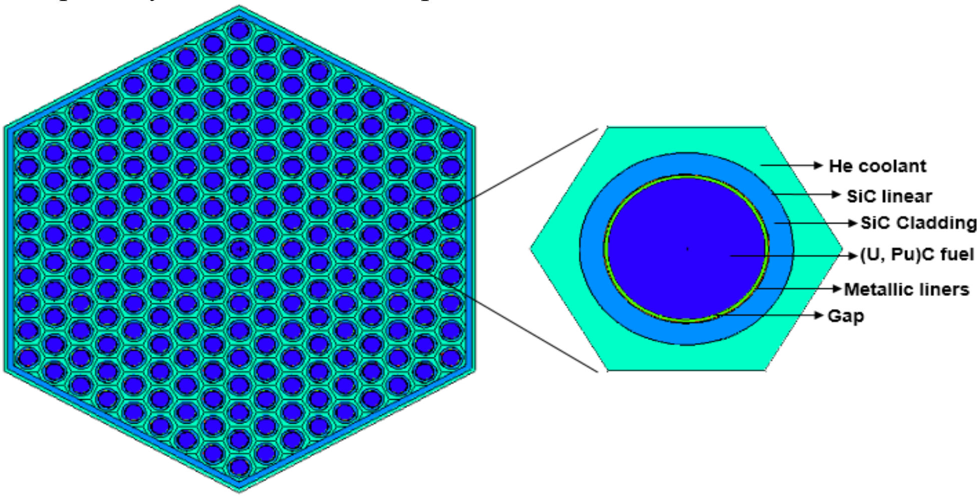

Fig. 1: configuration of the GFR2400 fuel assembly at left and fuel pin lattice at right.

Table 1: Design parameters of the GFR-2400 fuel assembly (Ibrahim et al., 2022).

Parameter	Value	Parameter	Value
Fuel	MOX	Assembly average power [MWth]	4.651
Fuel pellet radius [mm]	3.355	No. of pins in FA	217
Gap thickness [mm]	0.145	Fuel pin lattice pitch [mm]	11.57
W14Re liner thickness [mm]	0.04	Wrapper pitch (inside) [mm]	172.6
Re liner thickness [mm]	0.01	Wrapper pitch(outside) [mm]	175.29
Clad thickness [mm]	1.0	Coolant pitch (outside FA) [mm]	178.29
SiC liner thickness [mm]	0.03	Active fuel height [cm]	165

Table 2: GFR2400 plutonium vector composition.

Isotope	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
Concentration (w%)	2.70	56.0	25.9	7.40	7.30	0.70

2. The Performed Neutronic Calculations

MCNPX code version 2.7 (Pelowitz, D.B. 2011) with ENDF-VII cross section library was used to perform calculations on the use of thorium as alternative fuel in the Gas-cooled fast reactor (GFR) by modeling a single representative fuel assembly of this concept (see Figs. 1). By using reflective boundary conditions, the fuel assembly was modeled as infinite lattice model to be representative to the GFR2400 core. Thorium was applied in a similar manner as uranium i.e. uranium carbide (UC) was replaced by thorium carbide (ThC) while, the reference fissile material (PuC) was not changed in thorium model. Nevertheless, the average enrichment of fissile material was increased to achieve the same initial reactivity as the reference uranium-based fuel namely, the enrichment was increased from 15.76 to 21.1%. Also, a higher smeared density was considered i.e. the density of thorium based fuels was assumed to be 95 % of the TD instead of 80%. For simplicity from this point and forward the reference uranium fuel will be referred to as U-Pu while, the thorium-based fuel will be referred to as Th-Pu fuel. For each assembly modeled the length of operation cycle was equal to 1440 Effective Full Power Day (EFPD). The total number of cycles was 150 per each time step with 20 cycles to be skipped and the number of starting neutrons was 10000 per cycle. The standard deviation in k_{inf} values was around 0.00032.

3. Results and Discussion

3.1. Analysis of Neutronic Parameters and Fuel Burnup Results

Figure 2 shows the variations in the infinite neutron multiplication factor (k_{inf}). The figure compares the change in k_{inf} values with burnup time for the reference uranium and thorium-based fuels. For thorium-based fuel the following can be noticed: during the first 150 days, k_{inf} value continue to decrease; this is because the buildup of ^{233}Pa which has high capture cross section and acts as a neutron poison, compared to the fertile isotope ^{232}Th. The ^{233}Pa takes about 150 EFPD to reach its highest concentration in the fuel, see Fig. 3. After 150 EFPDs, the k_{inf} value starts to increase again driven by the buildup of ^{233}U in the fuel (see Fig. 3). A similar behavior was reported in a previous study by (György, H., Czifrus, S., 2016). For thorium fueled GFR2400 the change in reactivity at EOC is -720 pcm compared to -3240 for uranium-based fuel. Although, for thorium fueled model the k_{inf} remains below the initial value, but it shows much smaller reactivity loss compared to the uranium case. It is clear that the evolution of k_{inf} for the thorium-based fuel has a trend that is different from that of the reference uranium fuel, with very low reactivity loss. In Fig. 3 it can be seen that the production of ^{233}U through breeding from ^{232}Th is notable for the Th-Pu fuel. This production is driven by the fission of ^{239}Pu. The production of ^{233}U increases the reactivity allowing for a longer fuel cycle compared to the reference uranium fuel cycle. By EOC the thorium fueled model will contain 4.47E+03 kg of ^{233}U which represent 30% of the EOC fissile content. Meanwhile, the concentration of ^{239}Pu in plutonium vector is degraded from 56.0% at BOC to 48.01% at EOC for thorium fueled model. Also, as shown in Fig. 4 the total mass of the loaded plutonium is decreased significantly in case of the thorium-based fuel while it is slightly increased in case of reference uranium-based fuel. Therefore, thorium-based fuel can help in getting red off the present stock piles of recycled plutonium which are retrieved from nuclear waste. Finally, the fuel burnup has reached 51.1 GWd/ tHM vs. 55.7 GWd/tHM for thorium-based fuel. This difference arises from the difference in density (or heavy metal content) between the two models as uranium has relatively larger density and hence lower specific power compared to thorium based-fuel.

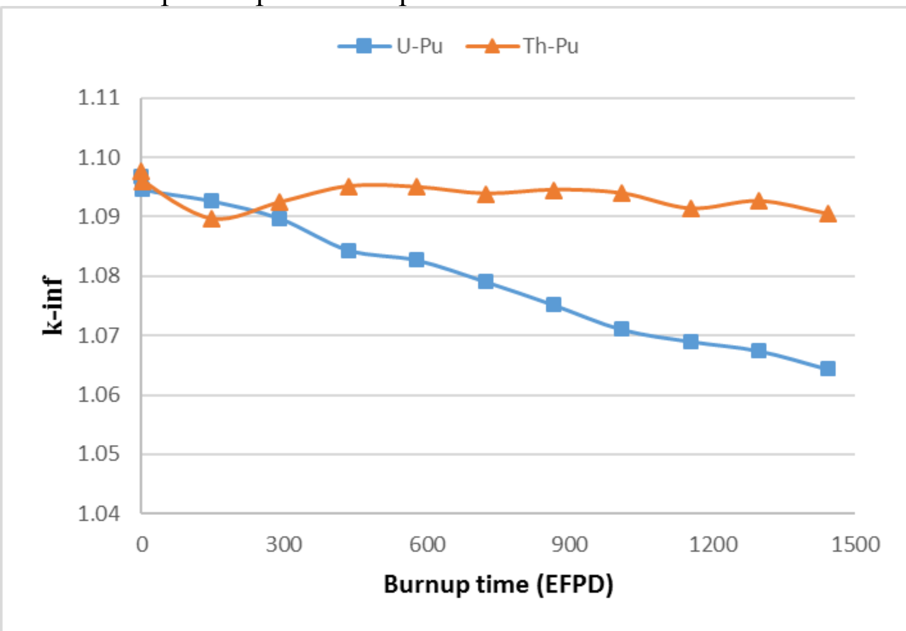

Fig. 2: k_{inf} evolution of GFR2400 for uranium and thorium based fuels.

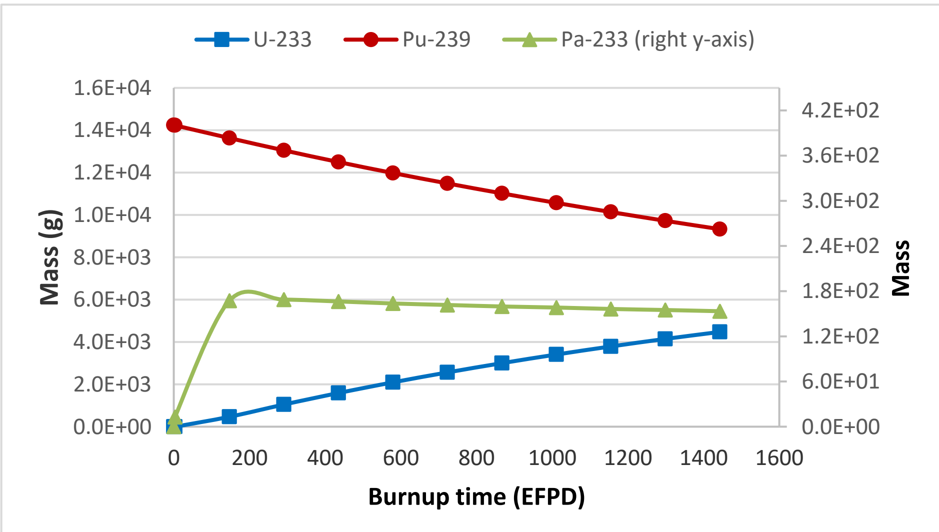

Fig. 3: Evolution of ^{233}U, ^{239}Pu and ^{233}Pa in thorium fueled GFR2400.

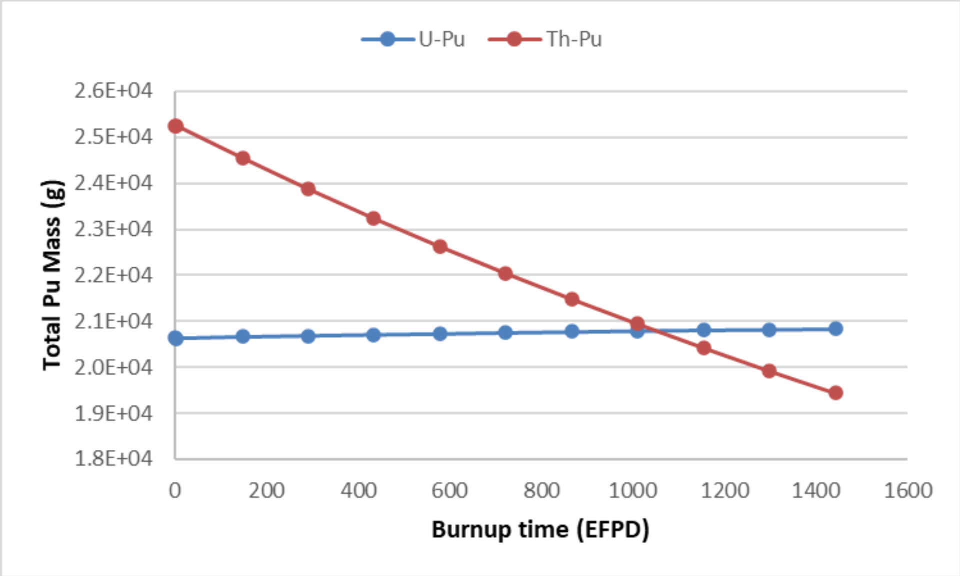

Fig. 4: Evolution of total plutonium mass for uranium and thorium-based fuels.

Table 3 presents a number of important neutronic parameters measured at BOC for both uranium (U-Pu) and thorium (Th-Pu) based fuels. As results indicate, the contribution of thermal (<0.625 eV) neutrons to fission is negligible and fission is mainly caused by intermediate (<0.1 MeV), and fast (> 0.1 MeV) neutrons with the percentage of the fast neutrons being higher than of intermediate neutrons. Also, Table 3 indicates that the average neutron energy causing fission is shifted towards fast energy region (>0.1 MeV) as the low moderation of coolant has a determinant influence on the energy distribution of the neutrons. On the other hand, the average number of neutrons produced per fission ($\bar{\upsilon}$) at BOC depends on the type of fissile material used at the beginning of fuel irradiation. Therefore, $\bar{\upsilon}$ values for the two concepts at BOC are comparable for thorium and uranium based fuels as the fissile material (239Pu: $\bar{\upsilon}$ =3.00±0.14) is used in the investigated concepts and it is responsible for most of fission reactions at BOC for both fuels.

Table 3: BOC neutronic parameters of the investigated concepts for both uranium and thorium based fuels.

Parameter	GFR2400	
	U-Pu	Th-Pu
Neutrons contributions to fission: thermal - intermediate - fast neutrons (%)	0.00% - 43.44% - 56.56%	0.00% - 43.83% - 56.17%
The average neutron energy causing fission (Mev)	0.89	0.67
The average number of neutrons produced per fission	2.912	2.950

The neutron spectrum of the GFR2400 for uranium and thorium-based fuels can be seen in Fig. 5. Neutron flux spectrum is obtained by using F4 tally in MCNP along with the energy card E0 to divide the total flux into energy groups (Pelowitz, D.B. 2011). From the figures, the neutron flux is negligible in the energy range below 10^{-4} MeV. While, a higher neutron flux is observed in the energy range below 1 MeV for both reference and thorium models. There is a peak at higher energy range near 0.1 MeV, which implies that there is less moderation as expected and desired for a fast spectrum concepts. The shape of the spectrum shows good agreement with the results presented in Table 3 for the contributions of neutrons belonging to different energy groups.

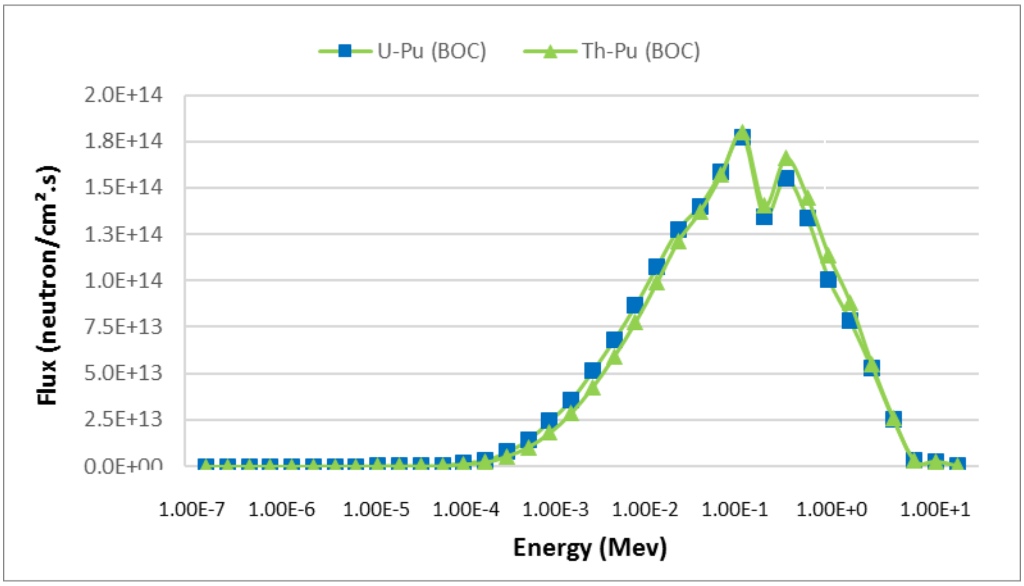

Fig. 5: Neutron energy flux spectrum of GFR2400 at BOC.

3.2 Safety Parameters Analysis

A main concern when considering the use of a different fuel in a nuclear reactor is its impact on the safety features, which may change from the initial load through the transition to the end of cycle. In this subsection, the U–Pu, and Th–Pu are compared employing the safety parameters summarized in Table 4. These parameters were calculated as in (Ibrahim,, A. 2023).

The generation time of prompt neutrons (Λ) is often used to define the dynamic response of a nuclear reactor (Hetrick, D.L., 1971). As results indicate the thorium fueled model is characterized by relatively lower Λ compared to the uranium fueled counterpart. This is due to their relatively higher fissile enrichment of plutonium as the prompt neutron generation time is inversely proportional to the fissile material content.

The delayed neutrons play a crucial role in nuclear reactors for controlling the rate of the increase in the generated power. The contribution of delayed neutrons to reactivity is estimated by calculating the effective delayed neutron fraction (β_{eff}). The obtained results indicate that the thorium fueled models are characterized by a lower β_{eff} compared to the uranium models. The reason for this difference is ^{238}U which has a relatively high delayed neutron yield and significant fission contribution. ^{232}Th has even higher yield of delayed neutron compared to ^{238}U but much smaller fission rate contribution. Furthermore, thorium fueled model is characterized by a lower fertile fraction and a higher plutonium content.

During burnup the increases in fuel temperature is followed by a reactivity change due to the broadening resonances of fuel nuclides with the increase in temperature this effect is referred to as Doppler effect. The strength of this effect is determined by estimating factor called Doppler constant (DC). Results of DC depicted in Table 4 indicate that the obtained values of DC for the thorium fueled models are lower than their uranium fueled counterpart. This is mainly due to the low fertile fraction of ^{232}Th and higher plutonium content of this model compared to the uranium one.

Void effect is defined as the change in reactivity corresponding to a loss of coolant. For a fast reactor these accidents are definitely the most challenging situations due to the fast spectrum of the core. The calculated value of void coefficient corresponds to the most serious of such accidents, i.e. when the model suffers total void i.e. loss of helium gas in case GFR2400. This situation is accounted for in MCNPX input file by decreasing the density of coolant (He) by 99%. The obtained value of the void coefficient for thorium fueled model show improvement compared to uranium fueled models. This is mainly attributed to the high fission threshold of ^{232}Th and its relatively low sensitivity to spectrum hardening compared to ^{238}U.

Table 4: safety parameter of studied concepts for both uranium and thorium based fuels at BOC.

Parameter	GFR2400	
	U-Pu	Th-Pu
Prompt neutron generation time (μs)	2.21	2.14
B_{eff} (pcm)	408.93	263.41
Dopler constant (pcm)	1761.65	1301.37
Void coefficient (pcm)	308.82	275.22

4. Conclusions

The main aim of this work is to investigate the use of alternative thorium-based fuel in generation-IV Gas-cooled Fast Reactor (GFR2400). The results of the neutronic analysis indicate that the use of thorium has a positive influence on the trend of the multiplication factor evolution. Whereas, a fuel burnup cycle of a longer length can be achieved or a small reactivity swing is attainable during a target cycle length. On the other hand, the total mass of the loaded plutonium is decreased significantly in case of the thorium-based fuel while it is slightly increased in case of reference uranium-based fuel. Therefore, thorium-based fuel can help in getting red off the present stock piles of recycled plutonium which are retrieved from nuclear waste and at the same time produce fissile ^{233}U which is preferred over ^{239}Pu for non-proliferation reasons. Regarding safety parameters, for thorium fueled model the effective delayed neutron fraction and Doppler constant are deteriorated at BOC state. However, the void reactivity coefficient in case of thorium-based fuel shows a significant improvement which have the potential for enhancing overall safety of the model.

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