Research Article

Neutronic Performance of the VVER-1000 Reactor Using Thorium Fuel with ENDF Library

Sonia M. Reda ¹, Ibrahim M. Gomaa,² Ibrahim I. Bashter,¹ and Esmat A. Amin³

¹Physics Department, Faculty of Science, Zagazig University, Zagazig, Egypt

²Zagazig Higher Institute for Engineering and Technology, Zagazig, Egypt

³Nuclear and Radiological Regulatory Authority, Cairo, Egypt

Correspondence should be addressed to Sonia M. Reda; smali@ju.edu.sa

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In this paper, neutronic calculations and the core analysis of the VVER-1000 reactor were performed using MCNP6 code together with both ENDF/B-VII.1 and ENDF/B-VIII libraries. The effect of thorium introduction on the neutronic parameters of the VVER-1000 reactor was discussed. The reference core was initially filled with enriched uranium oxide fuel and then fueled with uranium-thorium fuel. The calculations determine the delayed neutron fraction β_{eff} , the temperature reactivity coefficients, the fuel consumption, and the production of the transuranic elements during reactor operation. β_{eff} and the Doppler coefficient (DC) are found to be in agreement with the design values. It is found that the core loaded with uranium and thorium has lower delayed neutron fraction than the uranium oxide core. The moderator temperature coefficients of the uranium-thorium core are found to be higher than those of the uranium core. Results indicated that thorium has lower production of minor actinides (MAs) and transuranic elements (mainly plutonium isotopes) compared with the relatively large amounts produced from the uranium-based fuel UO_2 .

1. Introduction

Countries are building nuclear power plants to meet their energy needs; the site of nuclear power plant in Egypt is El Dabaa. The VVER-1200 is the predecessor of the VVER-1000 reactor. The VVER-1000 is a 3000 MW thermal power nuclear power plant which is cooled and moderated by light water. The core is filled with 163 enriched uranium (UO_2) fuel assemblies (FAs). The reactor includes international safety standards with evolutionary design improvement in the areas of fuel technology, modularized construction, safety systems, and standardized designs [1, 2].

In fact, thorium is three times more abundant than uranium, and the feasibility of loading the core with thorium uranium fuel was carried out. Thorium is made up mainly of “fertile” isotope (^{232}Th). Experiments have been made on power reactors that were successfully operated using ThO_2 - UO_2 fuel in light water reactors (LWRs). ^{233}U and ^{232}Th are the best “fissile” and “fertile” materials, respectively, for thermal neutron reactors [3].

The neutronic parameter effects due to loading thorium as a part of VVER-1200 fuel under normal operation were studied by Dwiddar et al. [4]. The investigation used two different configurations, mixed uranium fuel with thorium and seed-blanket fuel. The amount of thorium inserted and the location of the thorium assemblies in the core of the reactor presented the master factor in determining the value of k_{eff} and the core cycle length. The results concluded that the safest position of thorium is in the periphery of the core, and it is recommended to provide a blanket of mixed thorium uranium fuel for the entire core as the peripheral layer with the highest value of uranium enrichment. Mixing thorium with plutonium instead of uranium for the uranium enrichment, limited to 5%, was suggested.

Minimizing the fission products (FPs) using thorium as a fuel in place of the traditional UO_2 fuel has been studied by Galahom [5]. This paper discussed the VVER-1200 neutronic features for one assembly. The assembly fuel is a mix of thorium (fertile) with uranium (fissile). The neutronic features were calculated and compared with the traditional

UO₂ fuel by using MCNPX code. The purpose of this study is to minimize the production of long-lived actinides and improve the fuel cycle length. The moderator temperature coefficient and the percentage of fissile inventory have been measured at different burn steps. The results showed that thorium fuels were better than UO₂, and by replacing ²³⁸U with ²³²Th, we need much ²³⁵U as a fissile material to sustain the same burnup level. For ²³²Th+²³⁵U fuel, the Doppler coefficient (DC) was more negative [5]. Production of long-lived actinides using thorium, fertile material, was reduced.

The possible advantages of the seed-blanket (SB) assembly used in the VVER-1200 core instead of the homogeneous assembly were investigated using MCNPX 2.7 code and the ENDF/B-VII library [6]. The blanket region fertile material was ²³²Th, while in the seed region, four different fissile materials have been investigated. The work concluded that in the homogeneous assembly, the power distribution is flatter than that of the heterogeneous assembly. The suggested fuels achieved a longer fuel cycle and a higher conversion ratio in the SB assembly than the homogeneous assembly. Moreover, using ²³²Th instead of ²³⁸U resulted in reducing the plutonium and the production of transuranic atoms.

Heterogeneous and homogeneous seed-blanket concepts were used to study the thermal hydraulics feasibility and the neutronics to change the AP1000 PWR reactor from UO₂ to (U-Th)O₂ [7]. The burnable poison geometry and materials in the core were not varied and only the fuel pin material was varied keeping the low enriched uranium (LEU) for ²³⁵U. The study used three different fuel types, the first being 68% ThO₂-32% UO₂; the second being 76% ThO₂-24% UO₂; and the third being 80% ThO₂-20% UO₂. The work goals were to increase ²³³U and minimize production of plutonium. The results showed that the homogeneous concept with three distinct types of fuel meets the optimization requirements. The results revealed that (U-Th)O₂ has many benefits, including a lower power density, retaining the same 18 months cycle, and a lower concentration of B-10 in the soluble poison and the elimination of B-10 in the coated integral boron poison [7].

In this paper, the neutronic performance and core analysis of the VVER-1000 one-six core were performed from the beginning of life (BOL) at cold zero power state (CZP) to the end of life (EOL) at hot full power state (HFP). The VVER-1000 reference core was initially filled with enriched uranium oxide fuel and then fueled with uranium-thorium fuel. Since thorium-based fuel is more favorable from the safeguard viewpoint, it is investigated in the present work as a fuel option to increase the fuel cycle length and reduce the plutonium amount produced. The neutronics calculations were made by MCNP6 code with two cross section libraries ENDF/B-VII.1 and ENDF/B-VIII.

2. The VVER-1000 Reactor Core Description

The VVER-1000 reactor type is a four-loop Russian version of the pressurized water reactor (PWR) producing about 1000 MW electric power. Figure 1 shows the core loading pattern for the first cycle of operation at the beginning of life

(BOL) [8]. The reactor fresh core consists of FAs, which differ from one another by the enrichment of the fuel. The brief details of the core FAs were illustrated by Aghaie et al. [9]. The reactor core, FA, and fuel rod characteristics are shown in Table 1 [10].

In order to provide lower center temperatures and a free volume to allow any released fission gas to expand and thus decrease internal pressure, the fuel contains a central void hole in its fuel pellet [11]. The form of FAs, numbers, and average enrichment loaded for the first cycle of operation are shown in Table 2 [12].

The core configuration for the mixed thorium-uranium fuel used in this simulation is shown in Figure 2, where the 3rd fuel batch with 4.95% enrichment is composed of 50% UO₂ and 50% ThO₂.

3. Methods of Calculation and Validation

MCNP6 is the result of merging MCNP5 and MCNPX codes with new available options capabilities and features. MCNP6 includes more capabilities such as Shannon entropy, more options for tally treatments and tagging, variance reduction control methods, and more options for geometry and particles. This code is high fidelity due to its accuracy in simulating geometries and materials with continuous energy processing of nuclear parameters [13].

The largest update to the ENDF library is represented in ENDF/B-VIII. These neutron sublibraries have been expanded by 32% to include 557 evaluations. ENDF/B-VIII and ENDF/B-VII.1 were the libraries used in this study [14, 15].

In this study, the VVER-1000 one-six core shown in Figure 3 was simulated. It was fueled firstly with UO₂ and then fueled with ThO₂+UO₂. The neutronic parameters are presented for the two fuels. The simulation performed used 100000 neutron per cycle, 150 skipped cycles, and 250 active cycles.

ENDF/B-VIII library has the maximum neutron reactions changes on nuclides including actinides which affect nuclear criticality simulations. Two of the most important re-evaluated isotopes are uranium-235 and uranium-238. The re-evaluated parameters include the following.

- (i) *The capture and fission cross section values*: the cross sections for capture reaction have been reduced as compared with ENDF/B-VII.1 in the 0.5–2 keV region, but increased in energies up to 80 keV. The fission cross section evaluation agrees with the ENDF/B-VII.1 uncertainties around 0.4% higher for incident energies less than 15 MeV [14, 16].
- (ii) *The neutron multiplicity*: the neutron multiplicity ν was re-evaluated and expanded below 30 eV to include fluctuations. Unlike ENDF/B-VII.1 in which ν is constant over a wide range of incident neutron energy, there is no constant ν in ENDF/B-VIII, but its value varied as the incident neutron energy varied [14, 16].
- (iii) *The thermal neutron constants (TNCs)*: TNC new evaluations for uranium-235 show a neutron multiplicity reduction and a fission cross section

FIGURE 1: Core loading pattern for the first cycle at beginning of cycle (BOC) [8].

increment as compared with ENDF/B-VII.1 evaluations at thermal energy. The comparison between the evaluated thermal neutron constants for ENDF/B-VII.1, ENDF/B-VIII, and the standard 2017 values [14, 16] is illustrated in Table 3.

- (iv) (n,f) Prompt fission neutron spectrum (*PFNS*): the ENDF/B-VIII evaluation for the *PFNS* mean energy is clearly softer than that of ENDF/B-VII.1, but it fits well with experimental data. The new average released neutron energy becomes 2.00 ± 0.01 MeV; on the other hand, it was 2.03 MeV in thermal range [16].

- (v) *Cross sections of (n, n') and (n, xn)* : the ENDF/B-VIII evaluation for the total inelastic scattering cross section (n, n') is slightly decreased than the last ENDF/B-VII.1 evaluation. ENDF/B-VIII valuation for the (n, xn) secondary neutron was not changed but showed a difference above 14 MeV.

- (vi) *Nubar*: evaluators used the parameter *nubar* to study criticality problems since criticality is highly sensitive to *nubar*. Several simulations have shown that the use of the new *PFNS* and *TNCs* of ENDF/B-VIII produces marginally higher k_{eff} values than those of ENDF/B-VII.1 [17, 18].

TABLE 1: Reactor core and fuel assembly design data [10].

Reactor core	Value
Fuel assembly form	Hexagonal
Equivalent diameter of the core (cm)	415
Active height of the fuel (cm)	353
Fuel assemblies number	163
Thermal power (MW)	3000
Coolant/moderator	H ₂ O + H ₃ BO ₃
Rod pitch (cm)	1.275
Fuel Assembly (FA)	Value
Assembly pitch (cm)	23.6
Number of pins in the FA	331
Fuel rods number in the FA	311
Fuel pin pitch (cm)	1.275
Length of fuel pin (cm)	353
Diameter of fuel pellet central hole (cm)	0.15
Outer/inner diameter of fuel pin (cm)	0.757/0.15
Outer/inner diameter of cladding (cm)	0.91/0.773
Outer/inner diameter of guide tube (cm)	0.126/0.106
Outer/inner diameter of central tube (cm)	0.13/0.11
Fuel pin material	UO ₂
Density of fuel pin material (g/cm ³)	10.5
Density of clad material (g/cm ³)	6.4516
Composition of clad material (%)	99% Zr, 1% Nb
Density of guide tube material (g/cm ³)	6.4516
Absorber material	Boron

The above 6 factors explain the marginal increment of ENDF/B-VIII than ENDF/B-VII.1 in the criticality calculations as will be shown in results.

4. Results and Discussion

The neutronic calculations of the VVER-1000 reactor are presented starting from the beginning of life (BOL) to the end of life (EOL). At BOL, the reactivity coefficients were estimated. The reactivity coefficients together with the delayed neutron fraction β_{eff} are the key parameters in the reactor dynamics response. The reactivity coefficients include the moderator temperature coefficient (MTC) and Doppler coefficient (DC). The end of life calculations include the burnup results (fuel consumptions and actinide productions). The neutronic parameters were calculated by MCNP6 with ENDF/B-VII.1 and ENDF/B-VIII and compared with the reactor design parameters [19] and published results [9, 20]. It is necessary for any reactor core to have delayed neutrons since they can control the increase in the reactor power [21]. β_{eff} is calculated by using the following equation:

$$\beta_{\text{eff}} = \frac{k_{\text{eff}} - k_{\text{prompt}}}{k_{\text{eff}}}, \quad (1)$$

where k_{prompt} is the effective multiplication factor using only prompt neutrons, while k_{eff} is the effective multiplication factor using both prompt and delayed neutrons. The fuel temperature coefficient of reactivity (DC) can be calculated by using the following equation:

$$\text{DC} = \frac{k_2 - k_1}{k_2} \frac{1}{\Delta T}. \quad (2)$$

K_1 is calculated in case moderator and fuel are at 300 K and k_2 is calculated in case moderator is at 300 K while fuel is at 600 K [22]. MTC was determined by the following equation:

$$\text{MTC} = \frac{k_3 - k_2}{k_3} \frac{1}{\Delta T}. \quad (3)$$

K_3 is calculated in case fuel and moderator are at 600 K.

4.1. The Beginning of Life (BOL) Results. The comparison between the calculated parameters with the published results is shown in Table 4. This comparison allows assessing the validity of the current VVER-1000 simulation. A general overall good agreement can be observed. In case of fueling the core with UO₂, β_{eff} values were 0.0078 and 0.00688 with ENDF/B-VII.1 and ENDF/B-VIII, respectively, while the expected value from the ²³⁵U fission is 0.00650 according to Lamarsh and Baratta [23], so it was realistic to see the core has a delayed neutron fraction that is greater than this expected value. According to RFANE [19], the delayed neutron fraction is 0.00740 and 0.007110 according to Gholamzadeh et al. [20] which is in agreement with the obtained results. The fraction of the delayed neutrons differs from fuel material to another. β_{eff} was 0.0031 and 0.0069 for ²³³U and ²³⁵U materials, respectively, according to IAEA [3] which means that the ThO₂+UO₂ core is expected to have a low delayed neutron fraction and this is clear in Table 4. β_{eff} values are 0.00628 and 0.00545 with ENDF/B-VII.1 and ENDF/B-VIII falling in the expected range for ThO₂+UO₂ core.

4.2. Temperature Coefficients of Reactivity. This section discusses the reactor operation safety parameters which are the variation of the reactivity with temperatures (the temperature coefficients of reactivity). Table 4 presents the main operating safety parameters (DC and MTC). As the temperature of the moderator increases, its density decreases and lower fraction of neutrons is slowed down, resulting in negative change of reactivity. On the other hand, the neutron spectrum hardens due to the decrease of moderator fraction, which will decrease the core reactivity. The above two factors compete with each other and might lead to a negative MTC.

The DC is the most important safety parameter because it measures the reactor operation stability. The higher temperature of fuel permits the fertile material to absorb many neutrons away from the fission. ²³²Th absorbs larger neutrons than ²³⁸U in higher fuel temperatures. Therefore, thorium-based fuel has more negative DC than the uranium fuels; this is clearly shown in Table 4. The MTC is the change in reactivity by changing the moderator temperature. The reactor is designed to have negative MTC value to provide negative reactivity feedback, which indicates that the more the temperature increases, the more the reactivity decreases. The UO₂ MTC is more negative than ThO₂+UO₂; this is because of the larger fast fission cross section of U-238 than that of Th-232, and this agrees with the results in Table 3. The higher temperature of fuel permits the fertile material to

TABLE 2: FA types, enrichment, and number loaded in the core for the first operational cycle [12].

FA type	Average enrichment %	No. of UO ₂ rods/enrichment %	No. of burnable absorber rods (BA)
16	1.6	311/1.6	—
24	2.4	311/2.4	—
36	3.62	245/3.7 66/3.3	—
20 B 24	2.4	311/2.4	18
36 B 24	2.4	311/2.4	18
36 B 36	3.62	245/3.7 66/3.3	18

FIGURE 2: Mixed fuel with uranium and thorium in the 4.95% enrichment pattern core configuration.

absorb many neutrons away from the fission. Th-232 absorbs larger neutrons than U-238 in higher fuel temperatures [5].

4.3. The Middle of Life (MOL) Results. The behavior of k_{eff} over time for the VVER-1000 core can be seen in Figure 4. The calculations were performed along with 5 burnup steps of 100 days. The standard deviations for k_{eff} values ranged from 0.00036 to 0.0004 for UO₂ fuel and from 0.00036 to 0.00041 for ThO₂+UO₂ fuel. k_{eff} values were 1.19911 and 1.23126 for UO₂ and ThO₂+UO₂, respectively, at BOL. It then decreased with the depletion of the fuel. ThO₂+UO₂ core can keep criticality for more than 500 effective full power days while the UO₂ core can keep criticality nearly up to 450 effective full power days. In the following figures, UO₂-7 represents current results for UO₂ core by ENDF/B-VII.1; UO₂-8 represents current results for UO₂ core by ENDF/B-VIII; UTh-7 represents current results

FIGURE 3: The VVER-1000 one-six core.

TABLE 3: TNCs values, for uranium-235, for ENDF/B-VII.1, B-VIII, and standards 2017.

	B-VII.1	B-VIII.0	Standards 2017
$\sigma_f(b)$	584.99	586.80	587.20 (1.4)
$\sigma_v(b)$	98.69	99.40	99.30 (2.0)
$\sigma_{el}(b)$	15.11	14.11	14.09 (22)
ν_{tot}	2.4367	2.4298	2.4250 (50)
α	0.1687	0.1694	0.1690

for ThO₂+UO₂ core by ENDF/B-VII.1; UTh-8 represents current results for ThO₂+UO₂ core by ENDF/B-VIII.

As mentioned before, this simulation uses the ThO₂+UO₂ core configuration where the 3rd fuel batch with 4.95% enrichment is composed of 50% UO₂ and 50% ThO₂. Not only the thorium amount inserted but also the thorium assemblies location in the reactor play the principal role in k_{eff} value determination and the core cycle length. The thorium location in the periphery results in higher value of k_{eff} and longer length of cycle [3].

TABLE 4: Comparison between current calculated parameters with references [9, 19, 20].

Data source	DC (1/°C)	MTC (1/°C)	β_{eff}
UO ₂ -7 (current results for UO ₂ by ENDF/B-VII)	-2.88×10^{-5}	-1.39×10^{-4}	0.0078
UO ₂ -8 (current results for UO ₂ by ENDF/B-VIII)	-2.61×10^{-5}	-1.13×10^{-4}	0.00688
MCNXP 2.6, ENDF/B-VI [20]	—	—	0.00711
WIMSD4 [9]	-2.04×10^{-5}	-1.15×10^{-4}	—
Reactor design [19]	-2.11×10^{-5}	-1.30×10^{-4}	0.00740
UTh-7 (current results for UO ₂ +ThO ₂ by ENDF/B-VII)	-2.54×10^{-5}	-0.339×10^{-4}	0.00628
UTh-8 (current results for UO ₂ +ThO ₂ by ENDF/B-VIII)	-3.055×10^{-5}	-0.416×10^{-4}	0.00545

FIGURE 4: k_{eff} variation with time.

After 500 effective full power days, the ThO₂+UO₂ core remains supercritical and this can be explained by the fact that since only half of the UO₂ fuel is replaced with ThO₂ which is located in the low neutron flux periphery assemblies. Moreover, insertion of thorium increases the first core cycle length compared with UO₂ core fuel cycle, and increasing the UO₂ enrichment may help this. For this reason, in order to obtain more optimum advantage of thorium, it has to be rearranged in the middle of the core. This strategy can be realized by locating ThO₂ in the core periphery for the first operational cycle and then shuffling them into the core interior in the subsequent cycles.

Thorium was mixed with ²³⁵U to provide power till building enough ²³³U amounts. Figure 5 illustrates the change in the burnup rate over time. As time passes, the burnup rate increases till it reaches 20.13 and 21.3 GWD/MTU for the UO₂ and ThO₂+UO₂ one-six core, respectively, at end of life (EOL). Figure 5 shows that the core burnup results of ENDF/B-VII.1 library are typically the results of the ENDF/B-VIII library, and the changes appear only in the criticality calculations.

4.4. The End of Life (EOL) Results. The ²³⁵U is the core fissionable material in that it steadily decreases as it burns over the lifetime of the core, in addition to increasing the output of ²³⁹Pu and ²⁴⁰Pu levels towards the EOL. The fissile component ²³⁵U is depleted from the startup until EOL, and other fissile components are created when ²³⁸U is

FIGURE 5: The change in the burnup over time for one-six core.

transmuted to higher actinides, especially ²³⁹Pu and ²⁴⁰Pu. The quantity of total fission products (FPs) in the core grows up as the burnup level increases. FPs are neutron absorbers which have a strong adverse effect on the neutron economy of the core as time passes. Figures 6 and 7 show the fuel consumption and the production of actinides that build up in the UO₂ and ThO₂+UO₂ cores.

The UO₂ core contains ²³⁵U+²³⁸U; the two isotopes were re-evaluated in the ENDF/B-VIII library which means marginal differences in results, and this appears in Figure 6(a) especially for the consumption of ²³⁵U. On the other hand, ENDF/B-VIII did not re-evaluate thorium. The

FIGURE 6: The fuel consumption.

FIGURE 7: Production of actinides.

FIGURE 8: Neptunium production.

FIGURE 9: The thorium consumptions.

thorium core contains $^{235}\text{U} + ^{238}\text{U} + ^{232}\text{Th}$, hence the slight difference in the UTh-7 and UTh-8 mainly from $^{235}\text{U} + ^{238}\text{U}$ cross section difference. The principal benefit of using thorium as a fuel option is to get as much fissile isotope ^{233}U as possible and eliminate the plutonium production, and this is clearly demonstrated in Figure 7.

Figure 8 shows the neptunium isotope masses produced during the reactor operational cycle. Figure 9 presents the consumption of thorium in the $\text{ThO}_2 + \text{UO}_2$ core. There was a thorium consumption difference between ENDF/B-VII.1 and ENDF/B-VIII at 200 days, but we do not have any explanation for this difference.

5. Conclusion

From the beginning of life to the end of life, the VVER-1000 neutronic parameters were calculated using MCNP6 code. These parameters were determined for a UO_2 and a mixed $\text{ThO}_2 + \text{UO}_2$ fuel. The obtained results fall within an acceptable range with respect to the reactor design parameters and the published data. The delayed neutron fraction β_{eff} values were found to be 0.0078 and 0.00688 for UO_2 fuel, with ENDF/B-VII.1 and ENDF/B-VIII, respectively, which are in an agreement with the reference value of 0.00740. On

the other hand, $\text{ThO}_2 + \text{UO}_2$ has lower delayed neutron fraction β_{eff} values (0.00628 and 0.00545) than UO_2 fuel. For the UO_2 fuel, its MTC value was -1.39×10^{-4} ($^{\circ}\text{C}$) while for the $\text{ThO}_2 + \text{UO}_2$ fuel, it was -0.339×10^{-4} ($^{\circ}\text{C}$) with ENDF/B-VII.1. All fall in the reactor design operational limits. It has been shown that the MTC of the UO_2 core is more negative than that of the $\text{ThO}_2 + \text{UO}_2$ one; this is because of the larger fast fission cross section of ^{238}U than that of ^{232}Th . Thorium-based fuel has more negative DC than the uranium fuels. The behavior of k_{eff} over time for both cores explained the slight increment of ENDF/B-VIII than ENDF/B-VII.1 in the criticality calculations for the UO_2 fuel. k_{eff} decreased with the depletion of the fuel indicating that the UO_2 core has to be refueled earlier than the $\text{ThO}_2 + \text{UO}_2$ core for the first cycle of operation. The plutonium isotopes produced due to the $\text{ThO}_2 + \text{UO}_2$ core were lower than those of the UO_2 core.

Data Availability

The data used to support the findings of this study are included within the article.

Conflicts of Interest

The authors declare that they have no conflicts of interest.

References

- [1] G. P. Nyalunga, "Developing a Fresh core neutronic model at 300K for a VVER-1000 reactor type using MCNP6," Master's Thesis, Potchefstroom Campus of the North-West University, Potchefstroom, South Africa, 2016.
- [2] T. Lotsch, V. Khalimonchuk, and A. Kuchin, *Proposal of a Benchmark for Core Burnup Calculations for a VVER-1000 Reactor Core*, International Atomic Energy Agency, Vienna, Austria, 2009, http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/41/035/41035568.pdf.
- [3] International Atomic Energy Agency, *Thorium Fuel Cycle - Potential Benefits and Challenges*, IAEA-TECDOC-1450, Vienna, Austria, 2005.
- [4] M. S. Dwidar, A. A. Badawi, H. H. Abou-Gabal, and I. A. El-Osery, "Investigation of different scenarios of thorium-uranium fuel distribution in the VVER-1200 first core," *Annals of Nuclear Energy*, vol. 85, pp. 605–612, 2015.
- [5] A. A. Galahom, "Minimization of the fission product waste by using thorium based fuel instead of uranium dioxide," *Nuclear Engineering and Design*, vol. 314, pp. 165–172, 2017.
- [6] A. A. Galahom, "Improvement of the VVER-1200 fuel cycle by introducing thorium with different fissile material in blanket-seed assembly," *Nuclear Science and Engineering*, vol. 193, no. 6, pp. 638–651, 2019.
- [7] J. R. Maiorino, G. L. Stefani, J. M. L. Moreira, P. C. R. Rossi, and T. A. Santos, "Feasibility to convert an advanced PWR from UO_2 to a mixed U/ThO_2 core-Part I: parametric studies," *Annals of Nuclear Energy*, vol. 102, pp. 47–55, 2017.
- [8] A. Pirouzmand and M. Kazem Dehdashti, "Estimation of relative power distribution and power peaking factor in a VVER-1000 reactor core using artificial neural networks," *Progress in Nuclear Energy*, vol. 85, pp. 17–27, 2015.
- [9] M. Aghaie, A. Zolfaghari, and A. Minuchehr, "Coupled neutronic thermal-hydraulic transient analysis of accidents in PWRs," *Annals of Nuclear Energy*, vol. 50, pp. 158–166, 2012.

- [10] M. Jabbari, K. Hadad, G. R. Ansarifar, Z. Tabadar, and M. Hashemi-Tilehnoee, "Power calculation of VVER-1000 reactor using a thermal method, applied to primary-secondary circuits," *Annals of Nuclear Energy*, vol. 77, pp. 129–132, 2015.
- [11] T. Lotsch, V. Khalimonchuk, and A. Kuchin, *Solutions for the Task 1 and Task 2 of the Benchmark for Core Burnup Calculations for a VVER-1000 Reactor*, International Atomic Energy Agency, Vienna, Austria, 2011.
- [12] M. R. Karahroudiand and S. A. M. Shirazi, "Study of power distribution in the CZP, HFP and normal operation states of VVER-1000 (Bushehr) nuclear reactor core by coupling nuclear codes," *Annals of Nuclear Energy*, vol. 75, pp. 38–43, 2015.
- [13] D. B. Pelowitz, *MCNP6™ User's Manual, Version 1.0*, Elsevier, Amsterdam, Netherlands, (LA-CP-13-00634), 2013.
- [14] R. Capote and A. Trkov, "IAEA CIELO evaluation of neutron-induced reactions on ^{235}U and ^{238}U targets for incident energies up to 30 MeV," *Nuclear Data Sheets*, vol. 148, pp. 148–254, 2018.
- [15] J. L. Conlin, W. Haeck, D. Neudecker, D. K. Parsons, and M. C. White, *Release of ENDF/B-VIII.0-Based ACE Data Files, XCP-5*, Los Alamos National Laboratory, Los Alamos, NM, USA, (LA-UR-18-24034), 2018.
- [16] D. A. Brown, "ENDF/B-VIII.0: the 8th major release of the nuclear reaction data library with CIELO-project cross sections, new standards and thermal scattering data," *Nuclear Data Sheets*, vol. 148, 2018.
- [17] M. B. Chadwick, R. Capote et al., "CIELO collaboration summary results: international evaluations of neutron reactions on uranium, plutonium, iron, oxygen and hydrogen," *Nuclear Data Sheets*, vol. 148, 2018.
- [18] S. M. Reda, I. M. Goma, I. I. Bashter, and E. A. Amin, "Effect of MOX fuel and the ENDF/B-viii on the AP1000 Neutronic parameters calculations by using MCNP6," *Nuclear Technology & Radiation Protection*, vol. 34, no. 4, pp. 325–335, 2019.
- [19] Russia Federal Agency on Nuclear Energy (FRANE), "Final safety assessment report (FSAR) for BNPP," *Accident Analysis*, Vol. 4, Russia Federal Agency on Nuclear Energy, Moscow, Russia, 2005.
- [20] Z. Gholamzadeh, S. A. Hossein Feghhi, L. Soltani, M. Rezazadeh, C. Tenreiro, and M. Joharifard, "Burn-up calculation of different thorium-based fuel matrixes in a thermal research reactor using MCNPX 2.6 code," *Nukleonika*, vol. 59, no. 4, pp. 129–136, 2014.
- [21] G. Farkas, S. Michálek, and J. Haščík, "MCNP5 delayed neutron fraction (β_{eff}) calculation in training reactor VR-1," *Journal of Electrical Engineering*, vol. 59, no. 4, pp. 221–224, 2008.
- [22] W. Compton, "Impact of configuration variations on small modular reactor core performance," M.Sc. thesis, Missouri University of Science and Technology, Rolla, Missouri, 2015.
- [23] J. R. Lamarsh and A. J. Baratta, *Introduction to Nuclear Engineering*, Prentice Hall Inc., Hoboken, NJ, USA, Third edition, 2001.