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Analysis of the Reactivity Coefficient of the PWR Thorium Fuel

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A B S T R A C T

In design, control, and safety, especially in Pressurized Water Reactors, the Reactivity Coefficient parameter is crucial. The validation of every new library for an accurate parameter prediction is then crucial. The purpose of this work is to determine the value of the reactivity coefficient at the Beginning of the Cycle (BOC) and End of the Cycle (EOC) using the WIMSD code based on ENDF/B-VIII.0 nuclear data files. The PWR-1175 MWe experiment critical reactors, which use Th-UO_2 fuel pellets, are a set of light water-moderated lattice experiments that were used for this purpose. The study applied the new cross-section libraries for WIMSD-5B with ENDF/B-VIII.0 lattice code. The results showed that the fuel temperature reactivity coefficients for the PWR reactor at BOC and EOC using new libraries are –4.07 pcm/K and –2.72 pcm/K, respectively. Moderator Temperature Reactivity Coefficient at BOC and EOC are -1.8E-03 pcm/K and 3.73 pcm/K, respectively. Compared to the experimental data of the reactor core, the difference is in the range of 5.0 %. It can be concluded that for the PWR using thorium fuel as a model, all reactivity coefficients are negative and it is a good design for the safety of operation.

1. INTRODUCTION

The Nuclear Power Plant (NPP), which is appropriate for usage in Indonesia, is still being studied by BRIN. Various studies examined the safety, culture, and economy of nuclear power facilities that have been in operation worldwide. However, because the study of a single kind of nuclear power plant involves a wide range of topics, from the parameters of the fuel cell's performance to the reactor's overall performance, the research is done gradually and continuously. Studies on the safety of particular kinds of nuclear power plants must be conducted to bolster this research. Analyzing pin cell parameters is typically the first step in any investigation into a nuclear reactor's safety.

The first step in building a reactor as a power plant is determining how the performance of the reactor core relates to fuel cell characteristics that satisfy specific needs. The typical Pressurized Water Reactor (PWR) was employed for the research of the core of a PWR nuclear power station. Every cycle, the reactor's operating condition causes the core's reactivity to fluctuate. This shift is brought about by alterations in the fuel's composition, the production of xenon poison, and the physical characteristics of

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the fuel's fundamental building blocks. The term "reactivity coefficient" in reactor control refers to the variation in reactivity brought by the variations in fuel temperature, moderator, and moderator density. This coefficient is intended to be negative.

Because of its intrinsic safety, the reactor can be safely regulated even in the event of an increase in fuel or moderator temperature brought on by an increase in reactor power [1]. This is achieved by reducing the core reactivity. In particular, the fuel temperature reactivity coefficient (FTRC) plays a significant role in the design, operation, and safety of PWR reactors. Therefore, it is thought to be crucial to use the novel cross-section procedure to do precise calculations for the FTRC parameters. Using the new WIMSD library and the nuclear data file ENDF/B-VIII.0 [2], the calculation aims to find the value of FTRC. Because of this, fuel calculations for cells in a typical PWR reactor employing thorium-uranium oxide (Th-UO_2) fuel and light water as moderators were carried out [3]. The Sn approach was used to calculate the one-dimensional transport to conduct the analysis.

Through the use of unit cell geometry, or a model of the reactor core's fuel lattice, the fuel and moderator temperature reactivity coefficient parameter is determined. Two fuel units and a moderator make up a single-unit cell. The annulus, a cell computation software, requires cell dimension data, which can be produced from a single equivalent unit cell. With the new nuclear data library (ENDF / B-VIII.0) available, the calculations were performed using the WIMSD-5B program. The variance of fuel temperature was increased to do the reactivity coefficient parameter analysis. Next, an analysis is conducted using the reference results and the calculation results.

2. THEORY

2.1. Reactivity Coefficient

Reactivity is the term used to describe changes brought by reactor conditions to the effective multiplication factor of the reactor core. If the control rod's working circumstances are altered, the reflector or core configuration is changed, or a neutron source or absorber is inserted into the core, the reactivity of the core will change [4]. Reactivity can be mathematically represented by the following equation.

$$\rho = k_{eff} - 1/k_{eff} \quad (1)$$

with,
ρ = reactivity
k_{eff} = effective multiplication factor

The change in the neutron population in one cycle per population at the end of the cycle is another definition of reactivity. Even though the reactor is meant to run at constant power, it contains built-in elements that can alter the reactivity. Increased xenon concentrations in the fuel in the reactor core, and voids in the moderator or coolant are the intrinsic factors that most significantly affect the change in reactivity. The magnitude of the reactivity coefficient (α) represents the change in reactivity brought on by the aforementioned components.

2.2. Temperature Reactivity Coefficient

The partial derivative of the reactivity to temperature changes is known as the temperature reactivity coefficient (α_T) [5].

$$\alpha_T = \delta\rho/\delta T \quad (2)$$

with,
$\delta\rho$ = reactivity changed
δT = temperature changed

The stability of the nuclear reaction in the reactor will be determined by the temperature reactivity coefficient. If the temperature rises and the temperature reactivity coefficient is positive, this will lead to an increase in reactivity and an increase in reactor power. On the other hand, if the temperature reactivity coefficient is negative, rising temperatures will result in decreased reactivity, which will subsequently lead to a reduction in reactor power, ultimately maintaining the reactor in a safe state.

The Doppler effect is one of the most prevalent effects in nuclear reactors [6]. The phenomenon is an expansion of the resonant neutron area in the neutron energy cross-section as the fuel's temperature rises. The phenomena of absorption of resonant neutrons—neutrons with a steep energy curve—is significantly impacted by the broadening of the resonant area. It is widely known that there is significant absorption in the resonant neutron energy group in the macroscopic cross-section of ^{238}U. The fertile element's resonant neutron absorption rate increases as a result. The core reactivity decreases as a result of the fuel element's temperature increase, which also increases the resonance neutron absorption rate at ^{238}U. The change in reactivity of the unity of the moderator temperature change is the expression of the reactivity coefficient of the core moderator temperature.

$$\alpha_{Tm} = \delta\rho/\delta T_m \quad (3)$$

The reactivity coefficient of the α_{Tm} can be calculated by approximating it,

$$\alpha_{Tm} = \Delta\rho/\Delta T_m \quad (4)$$

The type and fuel of the reactor have an impact on the coefficient reactivity value as well. The reactor's safety quality is supported by its negative temperature reactivity coefficient, which indicates that as the temperature rises, reactor power will drop.

2.3. Doppler effect

The widening of the peak energy of resonant neutrons, or neutrons with a distinct energy curve consisting of peaks and valleys that are easily discernible on the small absorption cross-sectional curve of ^{238}U in Fig. 1, is known as the Doppler effect. This widening results from the reactor core's temperature increase as the fission reaction takes place. As the resonant neutron has an energy value that matches the excitation energy value of the ^{238}U, it is known that it has a fairly high cross-sectional view of the reaction towards ^{238}U. This means that as the resonant neutron peak widens, the absorption of neutrons by ^{238}U will increase, resulting in a reduction in the number of thermal neutrons absorbed by ^{235}U so that the k_{eff} is reduced.

Because of the target nucleus's thermal motion, which raises the likelihood of neutron absorption, rising temperatures have an impact on the resonant neutron peaks' broadening. A rise in temperature causes the target core to oscillate in the direction of its normal location. Consequently, not only do specific energy neutrons get absorbed, but also other neutrons with energy inside the neutron energy interval that would have had a high absorption probability in the past. This is because when the target nucleus shifts in the direction of the incoming neutrons, it will absorb the neutrons with less energy than it should. Otherwise, the opposite will occur. For the peak resonant neutron energy of the ^{238}U atom, which is 6.67 eV, to be wider at high temperatures. The target core's thermal energy rises with increasing reactor core temperature, making it easier to absorb neutrons with energies lower and higher than the target core's excitation energy value.

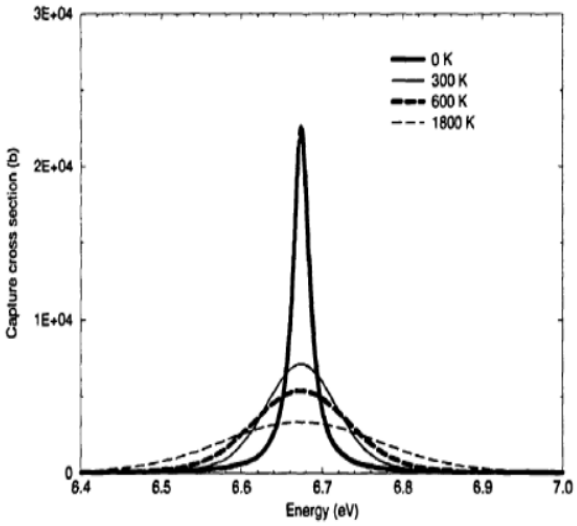

Fig. 1. Doppler effect[7]

Fuel reactivity will change as a result of Doppler broadening, or the expansion of the resonant peak. It is well known that high-energy neutrons are produced during the fission process. These neutrons are then moderated by collisions with moderating particles, causing a slow decrease in energy. The likelihood of neutrons being absorbed by the ^{238}U nucleus will be quite high when they approach the resonance energy interval value, whereas for ^{235}U the converse will occur. Reactor reactivity decreases as a result of this occurrence. Cross-section of ^{238}U absorption in the resonant zone decreases with increasing temperature, nevertheless, the neutron flux in the resonant region is getting bigger, such that it directly impacts the thermal neutron absorption by ^{235}U. Because neutrons travel randomly and repeatedly collide with target and moderator nuclei (H_2O), the movement of neutrons in the reactor core is extremely complex. Consequently, neutrons that were in one area of the reactor and were going in a certain direction and with a particular energy at a different time will now arrive in another area of the reactor with a different energy and direction of movement. From the first space and energy region to the second space and energy zone in this instance, the neutron is said to have been transferred. The term "transport theory" is frequently used to describe the study of this phenomenon [8]. The Sn method (discrete ordinate), which is employed for lattice calculation, was utilized to solve the transport equation contained in the WIMS program. The neutron power distribution and the infinite multiplication factor are calculated using lattice methods. Nuclear isotope data with numerous energy groups and reactor descriptions, expressed by unit cells or macrocells, are inputs used in lattice calculations. The buckling correction approach can

be used to calculate the leakage factor, which is used to calculate the value of k_{eff}.

3. BRIEF DESCRIPTION OF PWR CORE

Westinghouse developed the PWR. The core configuration of a PWR reactor is depicted in Fig. 2. Three levels of enrichment that make up the PWR core at the BOC are 2.4%, 3.1%, and 3.9%, and they are constructed into 49 fuel assemblies, 48 fuel assemblies, and 48 fuel assemblies, respectively [9]. A total of 145 fuel assemblies comprise the PWR reactor core. Table 1 summarizes the fuel, reflector devices, and PWR active core dimensions. PWR reactors use Zircalloy-4 as the fuel cladding, with UO_2 fuel pellet [10].

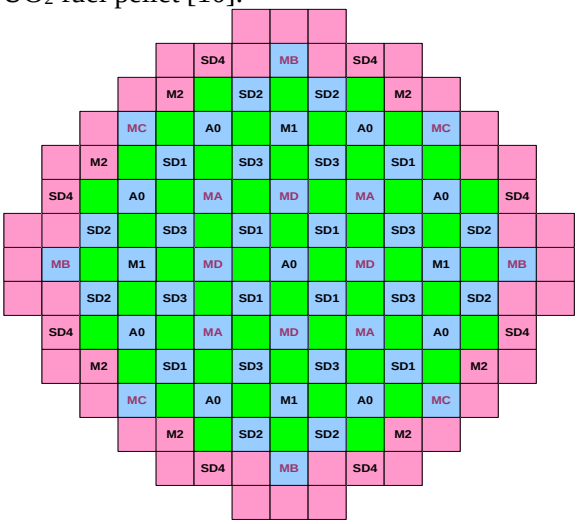

Fig. 2. PWR core configuration [11]

There is a compartment used to hold gaseous fission products at the top or bottom of the fuel cladding. A total of 289 (17×17) elements, comprising 264 fuel element pieces, 24 guide thimble elements, and one instrumentation tube, are included in each set of fuel assembly. Additionally, the fuel has 24 control devices overall on a single device, which is commonly referred to as a rod cluster control assemblie (RCCA). Ag-In-Cd is the primary component used to construct the control element, while zircaloy is used for the cladding. Axial power distribution and reactivity variations are managed by the control element device RCCA.

In addition to RCCA, the PWR core is equipped with GRCA control elements, or gray rod cluster assemblies, which are used to adjust the core's reactivity in response to load variations. When combined with the soluble boron, light water serves as a moderator and cooler as well as a neutron absorber [12]. Because of variations in the burnup percentage within the core, the concentration of soluble boron varies in proportion to changes in its reactivity.

Table 1. Nuclear data of fuel assembly a typical PWR [13]

Plant data	Values
Net electrical output, MW	1175
Fuel assembly	17×17
Number of fuel rods	264
Number of guide tubes	24
In-core instrumentation	1
Full length (without control spider), mm	4058
Width, mm	214
Rod pitch, mm	12.6
Mass of UO_2, Kg	460
Initial internal pressure (He), bar	24.1
The outside diameter of the fuel rod, mm	9.5
Fuel pellet material	UO_2
Density, g/cc	10.4
Fuel temperature,°C	811
Coolant temperature,°C	570
Cladding temperature, °C	620
Cladding material	Zircaloy-4
Cladding thickness, mm	0.57

3. METHODOLOGY

Cell Calculation

Using transport theory, the WIMS code determines the neutron flow in a one-dimensional cell as a function of energy and space. The transport equation is solved by applying the discrete ordinates method. This program package is then used for the fuel cell calculation step. This program's objective is to process the chancellor's core input and produce an output representing the material's macroscopic cross-sectional constant, which comprises the reactor core. In this program, the reactor core element's fuel assembly is represented as a collection of annuli consisting of moderator, meat, cladding, and extra region.

Reactor fuel with an elemental composition and variations in the fuel element temperature value make up the input prepared for the WIMS program code. The data used for the lattices have been done with hot zero power (HZP) condition Tf and T_m = 621 K. Meanwhile for hot full power (HFP) condition T_f = 841 K, Tc = 621 K, and T_m = 583 K [14].

The quantity of power is condensed into just 2 groups (few groups) in the first section using the neutron spectrum in a specific geometry and groups that are computed based on the software library (69 groups)[15]:

-Fast neutrons, groups 1 through 45, with energies between 5.531eV and 10 MeV.

- Thermal neutrons with energy less than 0.625 eV, groups 46–69.

The atomic density of the isotope provided in the program input and the microscopic cross-section of the program library are used to immediately derive the macroscopic cross-section of the neutron energy, which is needed as the coefficient of the multi-group equation.

In the second section, several groupings were computed. The fuel meat region is indexed 1, cladding is indexed 2, moderators are indexed 3, and the additional region is indexed 4. The cell is made up of these four regions. Each region's size and makeup are determined by program inputs. The multi-group constant was divided into four groups once the multi-group spectrum for each of the four regions was obtained.

The PWR core with a 17 × 17 fuel assembly uses the same UO_2 fuel cell pin. In one cycle, two years of full power operation equals one full power day, a fuel burn-up of 70 GWd/t results in the 3.1% enriched fuel.

The unit cell of the PWR pin geometry is shown in Fig. 3.

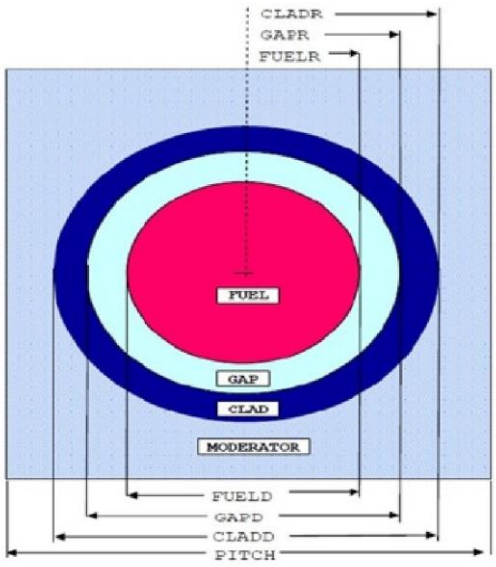

Fig. 3. Unit cell of PWR pin [16]

Since the WIMS algorithm is limited to computing one-dimensional neutron transport, core cell modeling is required. Calculating the formation of group constants in four energy groups is done by cell modeling. Computation of cells using the WIMS software package from the LWR cell unit, which is composed of squarely arranged fuel clusters, as seen in Fig. 3. After that, the dimensions of each cell unit are determined. A fuel and a moderator make up a single unit cell. As seen in Fig. 3, the cell dimension data is retrieved from the equivalent unit cell and used as input data for the WIMS program known as the annulus. Table 1 displays the atomic density that forms the fuel pin. The goal of group constant generation is to homogenize a cell to determine the average value of the group constants within it. To get group constant values that match the core circumstances, the experiment's core buckling value (Bz^2) was determined, as shown in Table 1. Under the aforementioned circumstances, group constant calculations were performed for the primary constituent materials. By enriching the fuel for each reactor as shown in Table 1, the core k_{inf} and the reaction rate were computed.

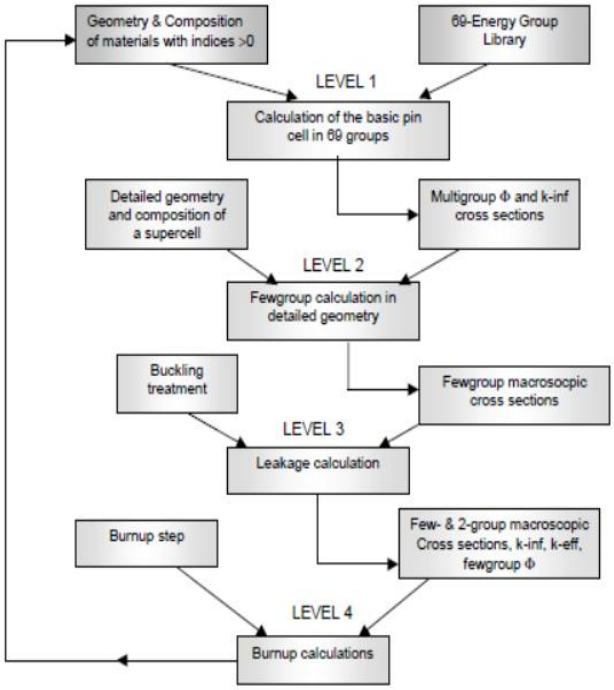

Fig. 4. Flowchart of WIMS code [17]

The root of the area above divided by 3.14 yields the moderator's radius around the fuel element, which is 0.53 cm. As seen in Fig. 3, the cell dimension data is retrieved from the equivalent unit cell and used as input data for the WIMS program known as the annulus. Table 1 displays the atomic density that forms the fuel pin. The goal of group constant generation is to homogenize a cell to determine the average value of the group constants within it. The experiment's core buckling value (Bz^2) was determined to yield group constant values that match the core conditions, as shown in Table 1.

The calculation of group constants was carried out for the core constituent materials under the conditions above. The k_{inf} and the reaction rate were calculated by enriching the fuel for each reactor as in

Table 1. Table 1 displays the pin cell shape and temperature settings. To carry out a coefficient analysis The following is how the moderator temperature reacts: The leakage factor (buckling) from the experimental data is used to calculate cells at room temperature (T0). Next, compute the cells under high temperatures (temperature T1). Here, there is a change in the density, cladding, moderator, and fuel temperature. Equation 2 is then used to determine the coefficient of reactivity, and Fig. 4 shows the computation flow chart.

4. RESULTS AND DISCUSSION

The values of k_{inf} PWR pin cell and fuel burn up utilizing ENDF/B-VIII.0 are shown in Table 2 based on the WIMSD-5B calculation results using the most recent library. The longer the reactor operates, the greater the amount of uranium that is burned and converted into energy. This is indicated by the k_{inf} value which is decreasing and the energy produced is indicated by the higher burn-up value. After the reactor has operated for 813 days and the burn-up value is 31,000 MWD/kg, the k_{inf} value is below 1.0, meaning that a certain amount of fresh fuel is needed for the reactor to stay critical.

Table 2. The burn-up calculation for typical PWR using Th-UO_2 fuel

Burn-up (MWd/kg)	k_{inf}	Time operation (days)
0.000	1.24578	0
0.114	1.20786	30
5.835	1.15579	153
10.411	1.12963	273
19.563	1.08253	513
31.004	1.02972	813
40.156	0.990771	1053
49.308	0.955111	1293
51.596	0.946666	1353
60.749	0.915423	1593
72.189	0.882305	1893

Table 3. Doppler Coefficient of Reactivity for typical PWR using Th-UO2 fuel

Fuel Temperature (K)	k_{inf} BOC (%Δk/k)	k_{inf} EOC (%Δk/k)	ρ-BOC (%)	ρ-EOC (%)	α_fBOC (pcm/K)	α_fEOC (pcm/K)
600	1.265646	1.037573	0.209889	0.036212	-	-
700	1.258518	1.034341	0.205415	0.033201	-4.474	-3.011
800	1.251930	1.031378	0.201233	0.030423	-4.182	-2.778
900	1.245780	1.028584	0.197290	0.027789	-3.943	-2.634
1000	1.240107	1.025983	0.193618	0.025325	-3.672	-2.464

From the calculation result of the WIMSD code, the Doppler Coefficient of Reactivity using a Th-UO2 fuel pin can be seen in Table 3. From the Table, the k_{inf} value at the temperature of the fuel was 600 K and the cladding was 583 K, the k_{inf} value at BOC was 1.265646, and the highest fuel temperature at 1000 K was 1.240107. For the fuel, the temperature is increased from 600 K to 1000 K and the moderator temperature is consistent at 621 K, the k_{inf} value decreases so that it can be said that the fuel reactivity coefficient is negative. This is appropriated by the typical PWR design [12]. The result at the EOC also shows that k_{inf} values decreased with the higher fuel temperature. The value of the fuel reactivity coefficient is larger the higher the temperature. The average value is around -4.07 pcm/K at BOC and -2.72 pcm/K at EOC. The parameter of fuel temperature coefficient reactivity at BOC and EOC from calculation results compared to experimental data for ENDF/B-VIII.0, the difference is around 4.16% [12],

Table 4. Coefficient reactivity of moderator for typical PWR using Th-UO2 fuel

Fuel Temperature (K)	k_{inf} BOC (%Δk/k)	k_{inf} EOC (%Δk/k)	ρ BOC (%)	ρ EOC (%)	$\alpha_{f\ BOC}$ (pcm/K)	$\alpha_{f\ EOC}$ (pcm/K)
550	1.249362	1.030217	0.1995914	0.0293307	-	-
600	1.249361	1.030214	0.1995908	0.0293279	-1.2E-03	-5.6E-03
650	1.249359	1.030213	0.1995895	0.0293269	-2.1E-03	-2.0E-03
700	1.249357	1.030211	0.1995883	0.0293251	-2.1E-03	-3.6E-03

The result of the calculation for the typical PWR using thorium fuel as a model for moderator temperature coefficient reactivity is presented in Table 4. From the table, it can be seen that the higher the moderator temperature, the smaller the k_{inf} values. In general, this causes the k_{iff} value in the core to decrease. The results of the calculation for moderator temperature coefficient reactivity using

ENDF/B-VIII.0 are almost the same as the reference. The differences are just 3.1 wt% enrichments and they are not significant for the fuel temperature coefficient reactivity parameter because small difference. The result of the calculation using ENDF/B-VIII.0 of the typical PWR core for fuel temperature coefficient reactivity is presented in Table 4.

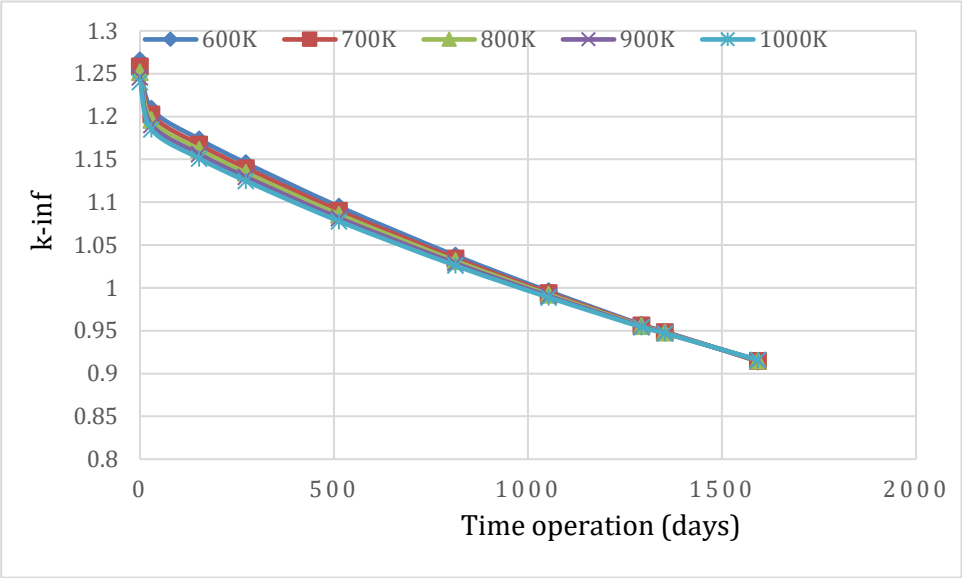

Fig. 5. K_{inf} values as a function of fuel temperature

From Fig. 4, it can be seen that in general, the k_{inf} value decreases with higher fuel temperature. The results of the calculation for fuel temperature coefficient reactivity using ENDF/B-VIII.0 are different at the BOC but at the EOC is almost the same. The differences are 3.9 % at the BOC, 2.9 % at the MOC, and 2.1 % at the EOC. The moderator coefficient of reactivity is negative and they are significant for the operation safety. Fuel and moderator temperature coefficient reactivity parameters have fulfilled the criteria of operation safety.

WIMS calculations for moderator and fuel temperature reactivity coefficient parameters using the new nuclear data ENDF/B-VIII.0 give consistent results to the design using thorium oxide fuel.

5. CONCLUSION

The reactivity coefficient of the fuel and moderator for the PWR core with uranium-thorium fuel pin cell was found to be negative. The value of the reactivity coefficient is very dependent on the temperature of the fuel and moderator, with each increase in fuel temperature and moderator the reactivity coefficient value is not the same. However, with every increase in temperature, both fuel and moderator, the value is always negative. This is by design and operation safety criteria. The average fuel reactivity coefficient value is greater than the average moderator reactivity coefficient value.

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AUTHOR CONTRIBUTION

Santo Paulus Rajagukguk carried out a model of pin cell input code, and Purwadi carried out cell calculation using WIMSD-5B. Syaiful Bahkri participated as a reviewer and did data analysis. Santo Paulus Rajagukguk and Purwadi as the main contributions of this paper. All authors read and approved the final version of the manuscript.

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