

Urania

Jurnal Ilmiah Daur Bahan Bakar Nuklir

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NEUTRONIC CALCULATION FOR PWR MOX FUEL PIN CELLS WITH WIMSD-5B CODE

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(Naskah diterima: 18–01–2022, Naskah direvisi: 25–01–2022, Naskah disetujui: 11–02–2022)

ABSTRACT

NEUTRONIC CALCULATION FOR PWR MOX FUEL PIN CELLS WITH WIMSD-5B CODE. The WIMSD-5B thermal reactor lattice cell code is used in many laboratories for research reactor calculations and power reactors. The program uses the Wigner-Seitz approximation for cell pin calculations. The approximation has been widely applied to the pin of UO₂ cells and has shown good results in previous studies but can produce incorrect results when used for pin cells in MOX fuels. This paper investigates the use of the WIMS-5B code to calculate the neutron multiplication factor and depletion for MOX fuel pin cells. Calculations were performed using the WIMSD-5B code updated with the ENDF-BVIII.0 library. The outer scattering boundary condition was used to overcome the effect of the Wigner-Seitz approach on the lack of MOX fuel. Results of this study indicates that most of the results obtained using ENDF-BVIII.0 are better than ENDF-BVII.1, and this is in line with expectations. The difference in the maximum k-inf value obtained from this library occurs in the fuel that has the greatest enrichment. On the other hand, the addition of the outer scattering limit improves the results obtained using ENDF-BVIII.0, causing a slight improvement for other libraries. This shows that by using appropriate libraries and the addition of the scattering outer limit, the Wigner Seitz approximation for the MOX pin cell pins in WIMS-D5 can yield quite accurate results.

Keywords: Wigner-Seitz approximation, WIMS-D5 code, MOX fuel, Doppler reactivity.

ABSTRAK

PERHITUNGAN NEUTRONIK PIN CELL BAHAN BAKAR MOX PWR DENGAN PROGRAM WIMSD-5B. Program komputer sel kisi reaktor termal WIMSD-5B digunakan di banyak laboratorium untuk penelitian perhitungan reaktor riset dan reaktor daya. Program menggunakan pendekatan Wigner-Seitz untuk perhitungan pin sel. Aproksimasi telah diterapkan secara luas pada pin sel UO_2 telah ditunjukkan hasil yang baik dalam penelitian sebelumnya namun dapat menghasilkan hal yang salah jika digunakan untuk pin sel pada bahan bakar MOX. Makalah ini menyelidiki penggunaan kode WIMSD-5B untuk menghitung faktor perkalian neutron dan koefisien Doppler untuk sel pin bahan bakar MOX. Perhitungan dilakukan menggunakan kode WIMSD-5B yang diperbarui dengan pustaka ENDF-BVIII.0. Kondisi batas hamburan luar digunakan untuk mengatasi efek dari pendekatan Wigner-Seitz yang menjadi kekurangan bahan bakar MOX. Hasil dari penelitian ini menunjukkan bahwa sebagian besar hasil yang diperoleh dengan menggunakan ENDF-BVIII.0 lebih baik dibandingkan dengan ENDF-BVII.1 dan hal ini sudah sesuai dengan harapan. Perbedaan nilai k_{inf} maksimum yang diperoleh dari pustaka ini terjadi pada fuel yang mempunyai pengkayaan terbesar. Pada sisi lain, penambahan batas hamburan luar meningkatkan hasil yang diperoleh dengan menggunakan ENDF-BVIII.0 menyebabkan sedikit perbaikan untuk pustaka lain. Hal ini menunjukkan bahwa dengan menggunakan pustaka yang sesuai dan penambahan batas luar hamburan, pendekatan Wigner Seitz untuk pin sel pin MOX di WIMSD-5B dapat menghasilkan yang cukup akurat.

Kata kunci: Aproksimasi Wigner-Seitz, WIMS-D5, bahan bakar MOX, reaktivitas Doppler.

INTRODUCTION

The WIMS computer program is a tool used to calculate neutronic parameters such as the effective neutron multiplication constant, cross-section, and isotope depletion in nuclear materials [1]. At this time, the WIMS program uses a new nuclear data library ENDFB-VIII.0 so it is necessary to calculate the neutronic parameters of MOX fuel pin cells for PWR cores. Because MOX fuel is currently being developed for use in PWR cores. MOX PWR fuel has several advantages compared to UO_2 fuel, namely, it can be produced from recycling waste UO_2 fuel [2]. The research is principally intended to be used in generating a macroscopic cross-section of MOX fuel and waste characteristics which will form the basis for the study and design of fuel reprocessing plants, spent fuel delivery canisters, sewage treatment and disposal facilities, and canisters. waste delivery. The fuel operating cycle is being checked in general terms, and as such it is expected to accommodate a wide range of fuel characteristics, it is only necessary that the WIMS results be representative of this range. Satisfactory results were obtained by adjusting only the resonance integrals of the main fissile and fertile species to obtain agreement with the composition of spent fuel from exogenous sources. Generic fuel cycle studies, many of which are environmental impact studies, require relatively precise calculations of Pu isotopes [3,4]. This paper discusses the calculation of the neutronic fuel MOX or uranium-plutonium oxide with the PWR reactor model. The PWR model is based on the Westinghouse design[5] using a cross-section generated from the new compilation of nuclear data, namely ENDFB.VIII.0 [6]. The creation of the information required for this reactor model begins with the collection and initial processing of the cross-sectional data contained in the 69-neutron energy group library. Those nuclides whose presence in the reactor will have the greatest effect on the neutron spectrum and depletion characteristics and a larger library containing many of the nuclides of interest in the WIMSD code but having little effect on the spectrum and fuel depletion. The purpose of this study is to support the government's program in assessing the safety of nuclear power plants, especially PWR in terms of calculating the depletion of PWR core cell fuel with UO_2 and PuO_2 fuel.

a. PWR Core Description

A pressurized water reactor (PWR) is a type of nuclear reactor used to generate electricity and propel nuclear submarines and naval vessels [7]. PWR uses light water (H_2O) as a coolant and a neutron moderator. The PWR has quickly grown to become the most widely used reactor for nuclear power generation; with 297 operating worldwide by 2020 [8]. PWR makes it the most dominant reactor in use in the world. Its use on naval ships and nuclear ships is of great importance to various militaries around the world. Nuclear power has a major advantage over fuels such as gasoline or diesel in that it allows ships to run for very long periods of time without needing to be refueled. PWR is a good reactor for this ship because it has a high specific power due to the use of high pressure. This allows the reactor to be quite compact, especially with the use of enriched uranium. The PWR reactor must use enriched uranium as its nuclear fuel, due to the use of light water as a coolant and moderator. This is because light water will absorb too many neutrons if natural uranium is used, so the content of fissile uranium-235 fuel must be increased. This is done through a uranium enrichment process, in which the concentration of Uranium-235 is increased from 0.7% to about 4% [9]. The enriched uranium is packed into a fuel rod which is assembled into a fuel device, as shown in Figure 1. One device consists of 264 fuel rods for PWR [9]. The fuel device is arranged vertically in the fuel tube in the core. As the fuel is fired by thermal neutrons inside the reactor core, its density increases gradually, creating small cavities in the fuel tube. This empty space can cause problems due to the high pressure on the tube and will increase the chance of rupture. To avoid this problem, the tube is pressurized with helium of about 3.4 MPa. As the fission gas products accumulate over the lifetime of the fuel, the pressure gradually equals the high pressure of the reactor core.

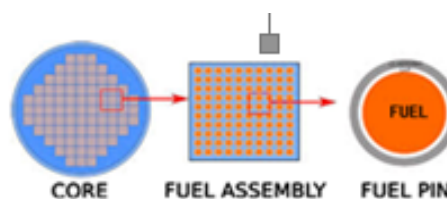


Figure 1. PWR reactor core and fuel assembly [10]

METHODOLOGY

Winfrith Improved Multigroup Scheme (WIMS) is a computer program for calculating reactor lattice cells in various reactor systems [13]. This program can calculate the geometry of the fuel rod (fuel rod) in a regular arrangement in groups and neutron energy structures selected mainly for thermal calculations. The basic library of 60 neutron energy groups has been divided into energy ranges of 14 groups for fast neutron energies, 13 groups for resonant neutron energies, and 42 groups for thermal neutron energies [14], but users can also choose an accurate solution in multiple groups or quick calculations in multiple groups. A temperature-dependent thermal scattering matrix for scattering law is included in the library used in WIMS especially collisions with light atomic coolant/moderator light water (H₂O). WIMSD-5B is the latest version of the WIMS program that uses ENDF-BVIII.0 nuclear data [15]. The resonance treatment is based on the use of the equivalence theorem with a resonance integral library which is accurately evaluated for homogeneous systems at various temperatures. Using collision theory in the calculation of an accurate spectrum of 69 neutron energy groups for the main region of the fuel lattice is a simplified geometric representation of the lattice cell modeled as a pin cell. The calculated spectrum is then used for cross-sectional condensation to the selected number of neutron energy groups in this case 69 groups for the solution of the transport equation in the lattice cell geometry. The transport equation is solved by using the Carlson DSN method. The output of the WIMSD-5B program provides eigenvalues (k-inf), macroscopic constants, and fuel isotope inventories after the reactor is operated with a set fuel fraction level of 27.0 GWd/tHM. Figure 2 is a PWR fuel cell pin model with 3 categories of MOX fuels, namely Fuel A, B, and C.

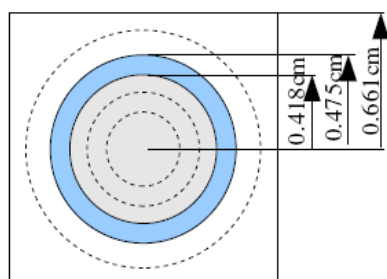


Figure 2. WIMSD-5B input model [17]

Density of the MOX fuel A is 4.3%, B = 7.0%, and C = 8.7% [16]. The calculation of the atomic density of fuel (UO₂-PuO₂) A, B, and C can be seen in Table 1, while the atomic density for the cladding and coolant can be seen in Table 2. In addition to the atomic densities of fuels A and B also the WIMSD-5B program ID for each nuclide is provided in Table 3. In this calculation, the temperature for the fuel is 841 K and the cladding is set to 620 K and the coolant is 558 K [17]. This temperature value is adjusted to the temperature of the MOX fuel and the PWR reactor cladding as operation.

Table 1. The atomic density of MOX fuel

Nuclide	Fuel		
	Density A (/barn cm)	Density B (/barn cm)	Density C (/barn cm)
²³⁵ U	5.00E-05	5.00E-05	5.00E-05
²³⁸ U	2.21E-02	2.21E-02	2.21E-02
²³⁸ Pu	1.50E-05	2.24E-05	3.00E-05
²³⁹ Pu	5.80E-04	9.30E-04	1.16E-03
²⁴⁰ Pu	2.40E-04	3.90E-04	4.90E-04
²⁴¹ Pu	9.80E-05	1.52E-04	1.90E-04
²⁴² Pu	5.40E-05	8.40E-05	1.05E-04
²⁴¹ Am	1.30E-05	2.00E-05	2.50E-05
¹⁶ O	4.63E-02	4.63E-02	4.63E-02

Table 2. The atomic density of cladding

Nuclide	WIMS ID	Atomic Density (/barn cm)	Temperature (K)
Zr	91	4,30E-02	620
H	3001	4,77E-02	
O	6016	2,38E-02	558
B	1011	2.78E-05	

RESULTS AND DISCUSSION

After performing calculations using the WIMSD-5B program, the results can be seen in Figure 3, namely the k-inf values for fuels A, B and C. The initial k-inf values for the 69 neutron energy groups are 1.17341 for fuels A, 1.18828 for fuel B, and 1.19583 for fuel C. This is because the amount of U-235 in fuel C is greater than fuel A and B. The figure shows the difference in the k-inf value profile for fuel A, B and C along with the deflection of the fuel up to a fuel fraction of 27.0 MWd/Te. The atomic densities of both fuels A, B and C are solid, but at the end of the cycle, with the same fuel burn up, the k-inf values are almost the same. When the reactor operates, uranium and plutonium are burned as fuel, with decreasing amounts of uranium and plutonium, the k-inf value also decreases.

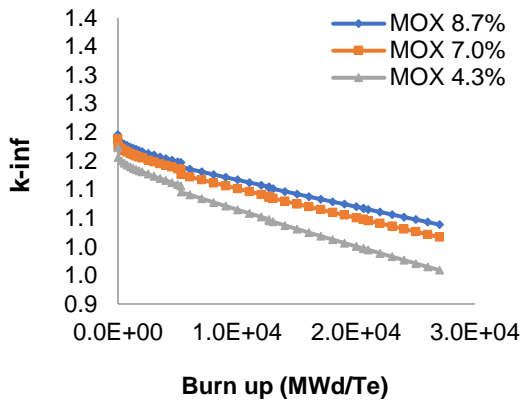


Figure 3. k-inf with burn up for MOX fuel

In Table 3, the results of the calculation of the k-inf value for the three MOX fuels at the beginning of the cycle (BOC) are obtained. In fuel A, the smallest k-inf value is obtained and in fuel C the largest value is in accordance with the enrichment value. However, when compared to the k-inf values for the ENDFB-VII.1 and ENDFB-VIII.0 libraries, the best value is obtained in the latest ENDFB-VIII.0 and has a more complete isotope. The greater the value of enrichment, the difference in the use of libraries is more significant.

Table 3. K-inf at BOC for MOX fuel

Fuel	ENDFB-VII.0	ENDFB-VIII.0	Different
A	1.17341	1.17397	0.00056
B	1.18828	1.18951	0.00113
C	1.19583	1.19732	0.00149

Table 4 also shows the results of calculating the k-inf value for the three MOX fuels at the end of the cycle (EOC). In fuel A, the smallest value of k-inf at the end of the cycle is obtained and in fuel C the largest value is in accordance with the enrichment value. However, when compared to the k-inf value for the end of the library cycle ENDFB-VII.1 and ENDFB-VIII.0, the best value was obtained at ENDFB-VIII.0. The greater the enrichment value, the smaller the difference in the use of the library.

Table 4. K-inf at EOC for MOX fuel

Fuel	ENDFB-VII.0	ENDFB-VIII.0	Different
A	0.95904	0.96537	0.00633
B	1.01677	1.02187	0.00510
C	1.03888	1.04339	0.00451

The value of the atomic density for fuels A, B and C can be seen in Figure 4. When the reactor is operating, the fuel continues to be

burned by firing thermal neutrons at uranium and plutonium nuclides so that over time the amount of fuel fuel decreases but the value of the fuel burn up increases, this is often referred to as fuel utilization. The fuel burn up in nuclear power plants (Nuclear Power Plants) often uses the GWd/tHM unit. This indicates the amount of energy produced during the normal operation of the PWR reactor. The amount of energy generated by the PWR nuclear power plant is getting better if the amount of energy is greater. In this study, the fuel fraction was calculated using the WIMSD-5B program up to 27.0 GWd/tHM with a time of 1831 days.

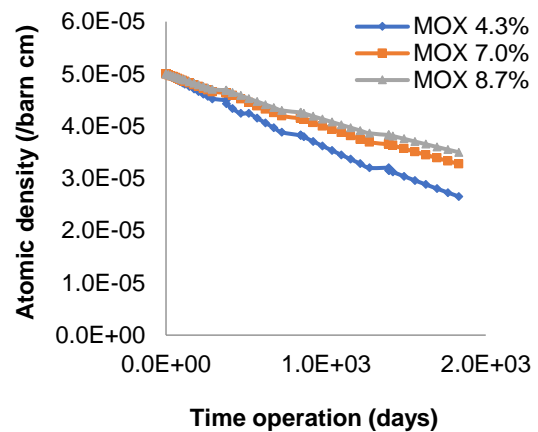


Figure 4. Atomic density with time operation

Figure 5 is the result of the calculation of the WIMSD-5B program for the total neutron flux values for fuel A, B and C. The total neutron flux in the fuel is a combination of all neutron fluxes from fast energy to thermal energy. The total neutron flux depends on the amount of fuel (uranium and plutonium) in the fuel, so the longer it is burned, the higher the total neutron flux, but in nuclear power plants, the neutron flux is not used for research. It is clear that if the amount of uranium decreases due to fuel deflection, the total neutron flux increases with operating time. The resulting total neutron flux is in the order of 10^{14} . The results of this calculation are in accordance with reference [18]. The total neutron flux also depend on level of pin power. In this case there are 4 level of pin power namely 17.24 MW/te, 19.43 MW/te, 17.04 MW/te, and 14.57 MW/te. It can be seen that from level of pin power 17.24 MW/te to 19.43 MW/te, the total flux neutron is increasing and then decreasing because the pin power lowering to 17.04 MW/te and also decreasing for pin power 17.04 MW/te.

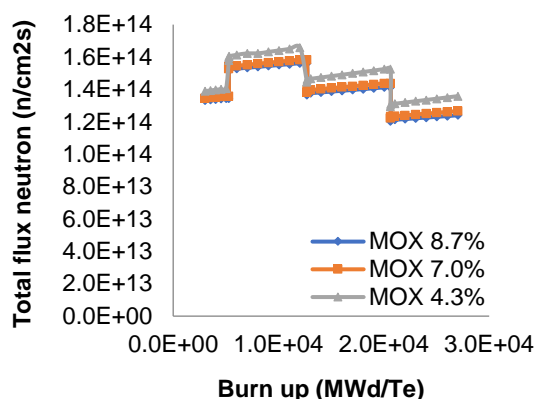


Figure 5. Total neutron flux with burn up

Figures 6, 7 and 8 showed the results of the calculation of the Pu nuclide inventory in fuel A, B and C. The calculation results show that the total density of the 5 Pu atoms (238-242Pu) in fuel A. All of the Pu atomic densities decrease with increasing fuel burn up. The largest Pu-239 decreased by 36.8% compared to the other Pu because the absorption cross-section of Pu-239 was greater than the others Pu.

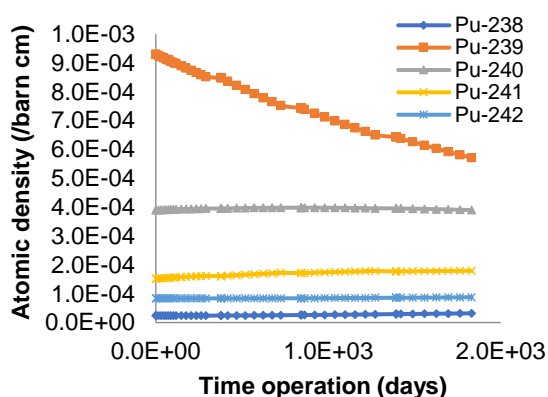


Figure 6. Atomic density with time operation in fuel A

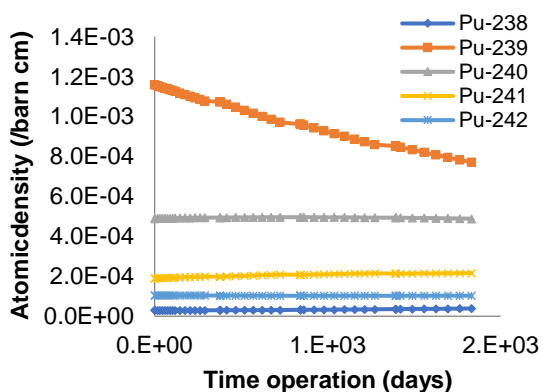


Figure 7. Atomic density with time operation in fuel B

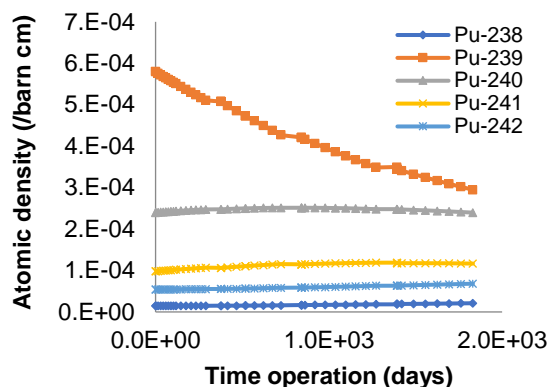


Figure 8. Atomic density with time operation in fuel C.

The isotope profile of the Pu atom inventory of fuel A, B and C can be seen in Figures 6, 7 and 8 they have the same trend, the largest being Pu-239 which is burned at around 57.7% so that the graph is steeper than fuel A.

CONCLUSIONS

A study of the fuel cell burnup parameters of light water-cooled PWR reactor has been carried out. Mix uranium oxide and plutonium oxide fuel cells with neutron energy groups 69 have a significant difference, namely producing k-inf values of 1.17341 for fuels A, 1.18828 for fuel B, and 1.19583, respectively. The burnup value is in accordance with the reality, namely 27.12 GWD/TU and results in 3 years of refueling. The results of the flux distribution analysis are highly dependent on power and are relatively in accordance with the neutron spectrum of the 1000 MWe PWR nuclear power plant core in the fuel area. To increase the accuracy of the analysis, in future research, the model will be improved and compared to the more efficient MCNP calculation because the program execution time is still quite long.

ACKNOWLEDGEMENT

The author would like to thank the head of the PTKRN Center who has provided the opportunity to conduct final research at PTKRN.

AUTHOR CONTRIBUTIONS

Santo Paulus Rajagukguk carried out cell calculations using WIMSD-5B. Tukiran Surbakti participated as a reviewer and did data analysis. Santo Paulus Rajagukguk and Tukiran Surbakti as main contribution of this paper. All authors read and approved the final version of the manuscript.

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