IRRADIATION CHARACTERISTIC OF NATURAL UO_2 PIN PHWR TARGET AT PRTF FACILITIES OF RSG – GAS CORE

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ABSTRACT

IRRADIATION CHARACTERISTIC OF NATURAL UO_2 PIN PHWR TARGET AT PRTF OF RSG – GAS CORE. The RSG-GAS reactor has a facility for irradiation of the fuel pin of nuclear power reactor, namely Power Ramp Test Facility (PRTF). The in-house fabrication PWR fuel pin has prepared for irradiations in the PRTF facility, currently, while the various enrichments of uranium are analyzed using the analytical tool. In the next step, it is planned to perform an irradiation of PHWR fuel pin sample of natural UO_2 in the facility. Before irradiation in the core, it should be analyzed by using the analytical tool. The objectives of this paper are to optimize irradiation time based on the burn-up, the generated linear power and the neutron flux level at the target. The 3-dimension calculations have been carried out by using the CITATION code in the SRAC2006 code system. Since the coolant of the reactor is H_2O, the effect of moderators in the pressurized tube, H_2O and D_2O, were analyzed, as well as pellet radius and moderator densities. The calculation results show that the higher linear power as irradiation time longer is occurred preferably in the D_2O moderator than in H_2O. For the D_2O moderator, the higher pressure affects the lower density and longer irradiation time. The maximum irradiation time for natural UO_2 fuel pin with the pressurized D_2O moderator is about 9.5×10^4 h, with the linear power of 700 W/cm. During irradiation, neutronic parameters of the core such as excess reactivity and ppf show a very small change, still far below design value.

Keywords: PHWR, Neutron Flux, Thermal Power, PRTF, RSG-GAS

ABSTRAK

KARAKTERISTIK IRADIASI TARGET PIN PHWR UO_2 ALAM PADA PRTF TERAS RSG – GAS. *Teras RSG-GAS dilengkapi dengan fasilitas untuk uji iradiasi bahan bakar nuklir atau disebut dengan Power Ramp Test Fasility (PRTF). Saat ini sedang dilpersiapkan untuk dilakukan uji sample pin bahan bakar PWR pada fasilitas PRTF. Analisis terhadap uji iradiasi sample pellet UO_2 dengan berbagai pengkayaan telah dilakukan menggunakan paket program komputer. Dimasa yang akan datang, uji iradiasi pin bahan bakar PHWR UO_2 alam juga sedang dalam perencanaan. Sebelum diiradiasi di dalam teras, maka terlebih dahulu harus dilakukan analisis dengan menggunakan paket program komputer. Tujuan dari penelitian ini adalah optimasi uji iradiasi pin bahan bakar UO_2 alam sebagai fungsi waktu iradiasi berdasarkan burn-up, daya linier dan fluks neutron. Perhitungan teras RSG-GAS dilakukan dengan paket program SRAC2006 modul CITATION dalam bentuk geometri 3 dimensi. Analisis dilakukan terhadap pengaruh penggunaan jenis moderator pada tabung tekan iradiasi (H_2O dan D_2O), perubahan ukuran pelllet UO_2 dan perubahan besarnya densitas moderator D_2O. Dari analisis hasil perhitungan diketahui bahwa semakin lama waktu iradiasi akan menghasilkan daya termal yang semakin besar jika menggunakan moderator D_2O dibandingkan H_2O. Semakin tinggi tekanan atau semakin kecil densitas moderator, maka akan menghasilkan daya termal yang semakin besar seiring bertambah lamanya waktu iradiasi. Batas maksimal waktu iradiasi untuk pin bahan bakar UO_2 alam dengan moderator D_2O bertekanan adalah sekitar $9,5\times10^4$ jam, dengan batasan daya linier desain kemampuan peralatan, 700 W/cm. Selama iradiasi, nilai parameter neutronik teras reaktor seperti reaktivitas lebih dan ppf hanya menunjukkan perubahan yang sangat kecil, masih jauh dibawah batas yang ditetapkan dalam desain.*

*Kata kunci**:** PHWR, Fluks Neutron, Daya Termal, PRTF, RSG-GAS*

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INTRODUCTION

Power Ramp Test Facility (PRTF) [1] is one of the irradiation test facility that is installed in the RSG-GAS core. PRTF is used to test power ramp for the fuel of nuclear power plants. This facility is developed to support the R&D on the nuclear fuel fabrication, especially Ligth Water Reactor (LWR) and Pressurized Heavy Water Reactor (PHWR). This facility has a cooling system to obtain a real pressure as in the power reactor. Some safety instrumentations are installed in the facility, especially the fission gas release detection system. Not all research reactors have this facility due to the complexity of the system. Some studies and experiments using such kind of this facility have been carried out in order to obtain a better performance that are more economical, optimal and reliable [2-4].

The previous studies are focused on the irradiation of Pressurized Water Reactor (PWR) fuel pin using PRTF facility [5-8]. Some important parameters, such as neutron flux and thermal power in the fuel pin were analyzed with UO_2 enrichments (0.7 – 4.8 %). From the studies, it is known that linear thermal heat generated by the UO_2 pellets does not exceed the limits of the irradiation facility PRTF (< 700 W/cm) [9]. Based on the calculated results, the irradiation of the fuel pin will be carried out by using the PRTF facility. For the next step, the fuel pin of PHWR that is in accordance with natural uranium is also planned to be irradiated since the natural uranium pellets have been fabricated.

In the real condition, the enriched UO_2 fuel in the PWR core using high pressurized water as a moderator, while the natural UO_2 fuel in the PHWR is using low pressurized D_2O as a moderator. If the fuel pellet of PHWR is irradiated at PRTF facility, some technical aspects should be considered, especially moderator. Currently, the facility is using H_2O as moderator, thus the natural uranium fuel pellet is irradiated by using H_2O moderator. The irradiation results should be corrected with moderator effect to obtain the evaluated fuel performance. Since, the absorption macroscopic cross section (Σ_a) of D_2O moderator is smaller than the H_2O, therefore it will affect the characteristics of UO_2 fuel pellets during irradiation test. By using previous studies of the high burn-up PHWR fuel, the evaluation of the irradiated natural uranium pellet could be carried out [10].

The objectives of this research work are to determine the optimum irradiation time and to obtain the high burn-up without exceeding the linear power limit of fuel pellet in the PRTF facility. The calculations were carried out for H_2O and D_2O moderators. Using the calculation results, the evaluation of fuel characteristic can be carried out. The core, target and the PRTF facility are modeled in 3-dimension. The core calculations were performed by using the CITATION module in the SRAC2006 code. The homogenization of fuel pin PHWR macroscopic crossection, power and size dimensional of reactor data are needed as input data. The output data are thermal power and flux neutron at the PRTF facility, power distribution of the fuel, and reactivity of the core. The code has been validated with some research reactors, Material Testing Reactors (MTRs) type, and PWRs. The results have a very good agreement [11, 12]. The analysis were performed to see the effect of irradiation time on thermal power produced and neutron flux at PRTF in various radius of pellets and the density of moderator. From the analysis, the neutron flux, linear power and fission product as a function of irradiation time are optimized.

METHODOLOGY

Modelling of Target Irradiation

The radial model of natural uranium fuel pin of PHWR is shown in Figure 1. The fuel pin lattice is composed by fuel pellets, cladding and moderator with a diameter of 0.905 cm, 1.07 cm and 2.6 cm, respectively. There is He gas gap in between the pellets and the cladding. The calculations are carried out for some densities of D2O moderator, in the range of 70% - 100% of 1.105 g/cm3, for given pressure. The calculated of 100% density of D2O were compared with the 100% density of H2O.

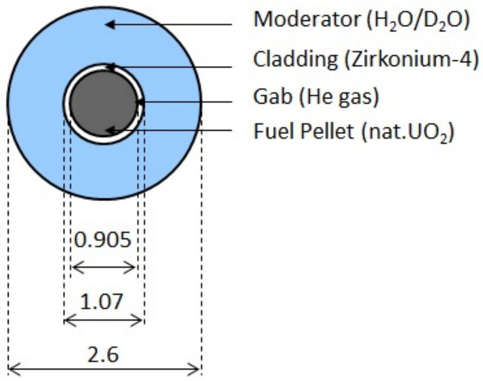

Figure 1. Radial model of natural UO_2 fuel pin of PHWR target irradiation (unit in cm).

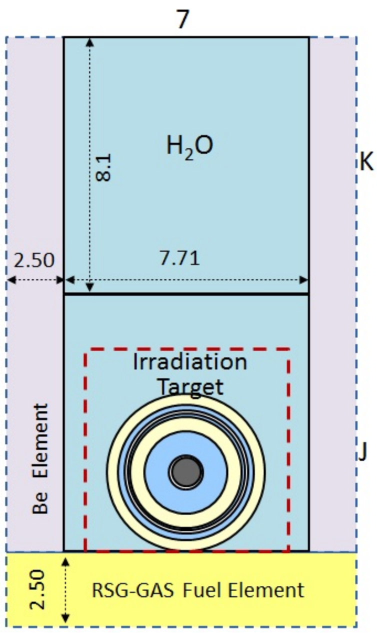

Figure 2. Radial model of PRTF with fuel pellet in the RSG-GAS core (unit in cm).

Figure 2 shows the radial model of the PRTF facility with the fuel pellet in the core. The target can be moved from J-7 to K-7 core grid position to make a transient load in the fuel pellet. The closest distance of the PRTF to the periphery is 2.5 cm.

In the cell calculation, the fuel pin, pressure tube, secondary water vessel and outer capsule are modeled in one lattice cell. The fuel pin is inserted in the $AlMg_3$ pressure tube with inlet and outlet diameter of 2.6 and 3.5 cm, respectively. The pressure tube is installed in the secondary water vessel with the inner and outer diameter of 3.7 cm and 3.9 cm, respectively. The outer capsule, with an inner diameter of 4.3 cm and an outer diameter of 5 cm, covers the secondary water vessel. In the pressure tube, the pressure can be adjusted while the H_2O is inserted in secondary water vessel and outer capsule. For the PHWR fuel pin, the pressure of D_2O moderator is low. All cell calculations were carried out by using neutron transport method, PIJ code in the SRAC2006 system.

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Optimization Method

Figure 3 shows the calculation flowchart for thermal power, neutron flux and nuclides Pu-239 in the natural UO_2 fuel pin of PHWR. The neutron diffusion constant were generated for several irradiation times and moderator densities. The neutron energy group boundaries are listed in Table 1. The irradiation time steps are 0.1, 1, 2, 3 and 4 (in units of 10^4 hours). Note, those maximum irradiation times based on the time for the fuel exists in the real nuclear power plant (NPP), about 38800 hours. From the calculations, the fission product density can be determined. The 3-dimension geometry of Multi Purpose Reactor G.A. Siwabessy (RSG-GAS) core was used in the core calculations. The RSG-GAS core is fueled by U_3Si_2-Al 2.96 gU/cm^3, in the equilibrium condition with thermal power about 15 MW. And we assumed that the target is in the closed position (J-7).

Table 1. Neutron energy grouping on the irradiation target calculations.

Group	Energy (eV)		Category
	Upper	Lower	
1	1.0000E+7	6.738E+4	Fast Neutron
2	6.738E+4	0.68256	Epithermal Neutron
3	3.9279E+0	4.5785E-2	Thermal Neutron

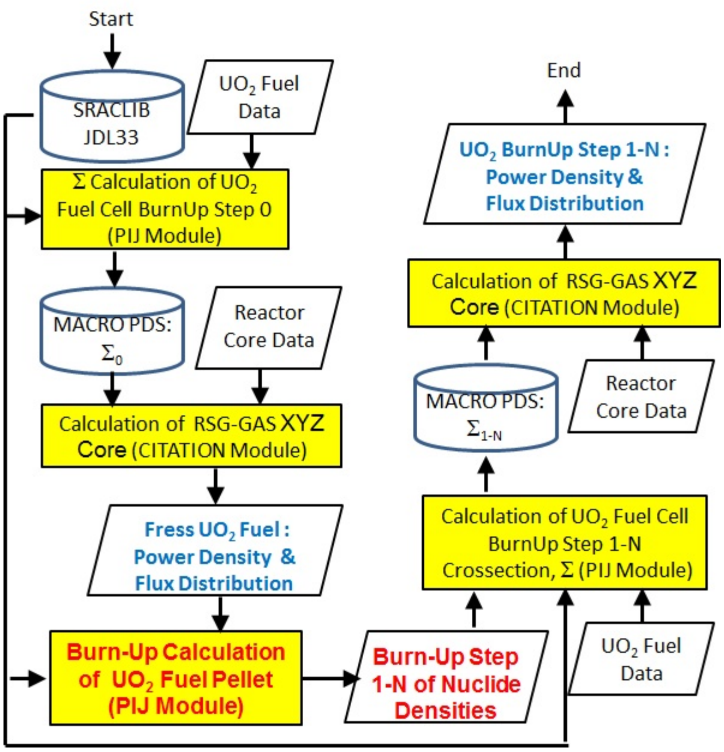

Figure 3. Calculation flowchart for the PHWR fuel pin target in the PRTF facility of RSG-GAS.

RESULTS AND DISCUSSIONS

Effect of Moderator (H_2O / D_2O)

Figure 4 shows the change of thermal power as a function of irradiation time for H_2O and D_2O moderator. From the figure, it can be seen that the selection of the type of moderator greatly affect the generated thermal power. If using a H_2O moderator, at the beginning of irradiation time (t = 1000 hours), the thermal power of UO_2 fuel pin shows a slight rise and then decline before tend to constant/fixed. A slightly increase of the thermal power is caused by production in small quantity of fissile material, namely Pu-239 nuclide. Then, due to the content of U-235 in the UO_2 pellets decreases and the amount of Pu-239 produced approaching constant, then thermal power generated from irradiation target will be reduced with increasing irradiation time. Moreover, the small amount of Pu-239 formed above is due to the small number of neutrons absorbed by the fuel. As it is known that H_2O has a large neutron absorption cross section (Σ_a=0.0222 cm^{-1}), therefore large number of thermal neutrons will be absorbed, and contrary small number of Pu-239 produced.

Meanwhile, if using D_2O moderator, the generated thermal power will be greater with increasing of irradiation time. That is caused by the D_2O moderator which has a small neutron absorption cross section (Σ_a=4.42×10^{-5} cm^{-1}). Therefore, it will affect the number of neutrons absorbed by the fuel U-238 and Pu-239. Pu-239 is a nuclide fissile and when it reacts with thermal neutrons, it will produce greater energy compared with U-235.

According to Figure 4, if X is irradiation time, and Y is thermal power generated by UO_2 fuel pin using D_2O moderator, then the correlation between them can be approached by the equation Y = 55.917X+168.84. Therefore, from the equipment design, maximum thermal power generated by pin UO_2 is limited to 700 watt / cm (Y), and the maximum irradiation time value (X) is equal to 9.5E4 hours. As a note that the amount of thermal power and linear power density have almost no difference, because the high fuel UO_2 pellets is about 0.98 cm (≈1 cm).

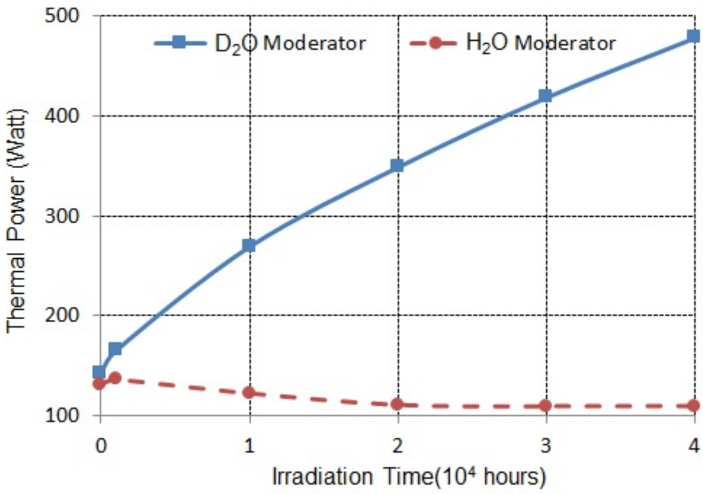

Figure 4. Effect of irradiation time on thermal power generated by UO_2 pin with variation in moderator type.

The effect of irradiation time on the nuclides weight produced by fuel pin PHWR of natural UO_2 is shown in Figure 5. From the figure, it can be seen that the use of D_2O as a moderator increases the production of Pu-239 as time increases. Number of produced Pu-239 using D_2O modeator is much greater than using H_2O moderator. As mentioned previously, it was due to the nature of the D_2O moderator that has smaller macroscopic absorption cross section than H_2O. Therefore, the absorption of thermal neutrons by U-238 fuel is more effective to produce Pu-239. Moreover, this figure also shows that the effects of Pu-239 fission reaction would slow consumption of U-235. It is seen as the gradient decrease for U-235. That is caused by the

P-ISSN: 1411–240X E-ISSN: 2527-9963
Nomor: 632/AU3/P2MI-LIPI/03/2015

microscopic cross section of the thermal neutron fission nuclides Pu-239 (742.5 barns) which is greater than the U-235 (582.2 barns).

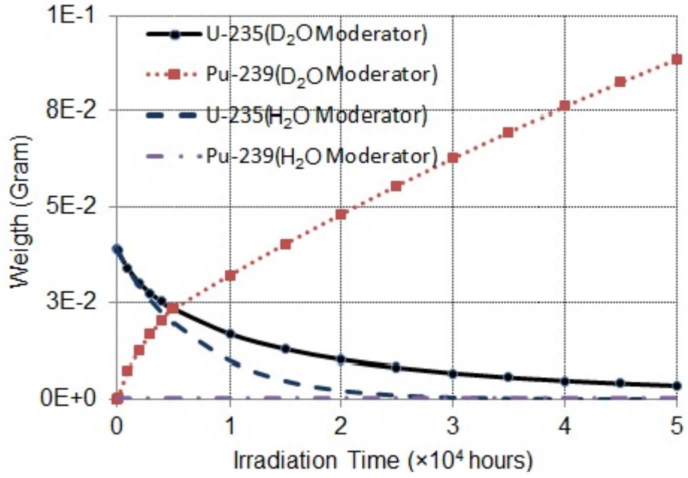

Figure 5. Effect of irradiation time on the nuclide weigth with variation in moderator type

Figure 6 and 7 show the difference of the fast and thermal neutron flux as a function of irradiation time on fuel cell lattice irradiation targets effect type of moderator used (D_2O and H_2O), respectively. The number of fast and thermal neutron flux is a mixture of flux generated by the target irradiated UO_2 pellets and from the fuel of reactor core RSG-GAS.

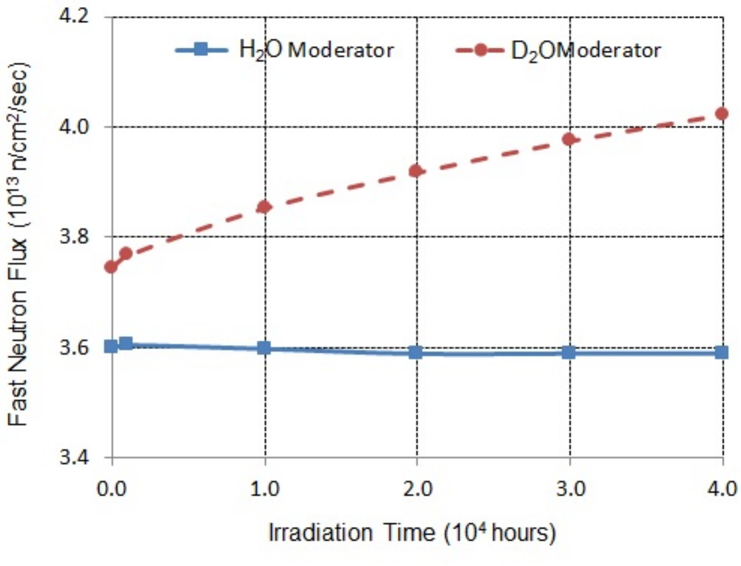

Figure 6. Effect irradiation time to fast neutron flux at the target irradiation with variation in moderator type

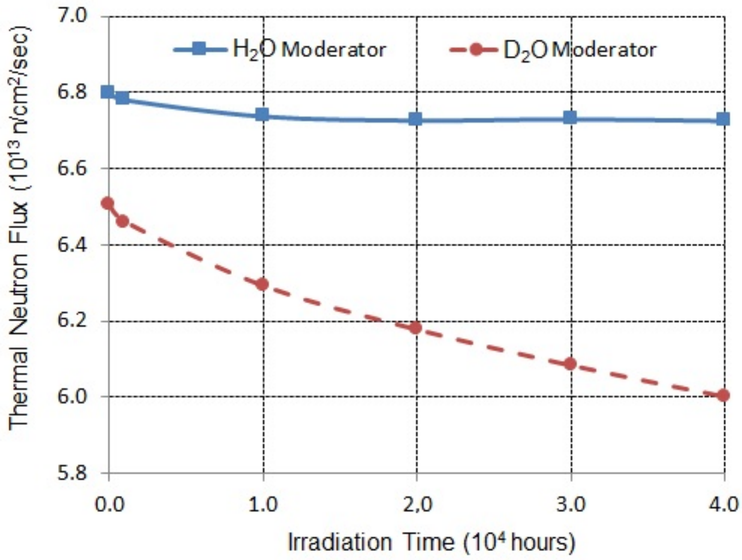

Figure 7. Effect of irradiation time on thermal neutron flux at the target irradiation with variation in moderator type

Figure 6 shows that the fast neutron flux on the fuel lattice cell target irradiation using D_2O moderator is greater than H_2O moderator. It was caused by the Pu-239 and the D_2O properties that have less thermal neutrons absorption. So that thermal neutrons are absorbed by fuel and fission reaction with Pu-239 occurs, then produce greater number of fast neutrons. As it is known that Pu-239 is more reactive than U-235, thus the Pu-239 would produce more fast neutrons than U-235.

Otherwise, effect of irradiation time on thermal neutron flux at target irradiation is showed at Figure 7. The figure shows that the irradiation target moderated by D_2O has a smaller thermal neutron flux compared with H_2O. It is due to fuel lattice cell that using D_2O moderator produces more Pu-239 than H_2O moderator. Pu-239 has about **750 barns** of microsscopic crossection for fission reaction with neutron thermal (0.025 eV), while fission microsscopic crossection of U-235 is about 584.994 barns. This result in more thermal neutrons will be absorbed by Pu-239 than U-235.

Table 2. Effect of irradiation time on reactivity change and PPF of RSG-GAS with variation in moderator type of target irradiation

Irradiation Time (10^4 hours)	Moderator H_2O		Moderator D_2O	
	$\Delta\rho$(%Δk/k)	PPF	$\Delta\rho$(%Δk/k)	PPF
0 (without target)	0.0000	1.1654	0.0000	1.1654
0	0.0317	1.1682	0.0519	1.1700
0.1	0.0319	1.1686	0.0532	1.1703
1	0.0299	1.1680	0.0581	1.1702
2	0.0279	1.1684	0.0610	1.1709
3	0.0274	1.1683	0.0645	1.1706
4	0.0273	1.1683	0.0672	1.1709

Effect of irradiation time on reactivity change and power peaking factor (ppf) of RSG-GAS during irradiation of PHWR fuel pin is showed in the Table 2. It can be seen that increasing of excess reactivity and ppf gives insignificant change. It is caused by the small amount of loaded uranium-235. In another hand, the position of the PRTF target irradiation facility is not in the active

P-ISSN: 1411–240X E-ISSN: 2527-9963
Nomor: 632/AU3/P2MI-LIPI/03/2015

core. The active core is composed from 48 fuel, 4 Irradiation Position (IP), 4 Central Irradiation Position (CIP) and 8 Beryllium Element in the 8×8 arrangement. Meanwhile, the PRTF irradiation target facility is located outside the active core. Those position has much more smaller thermal neutron flux than at the inner of active core such as IP or CIP. The decreasing of water volume in PRTF facility is caused by irradiation target loading that gives effect to decreasing number of moderation reaction from fast energy to thermal energy neutron.

In case of fuel lattice cell target irradiation using D_2O moderator, core excess reactivity change is slightly greater then using H_2O moderator. It is caused by the characteristic of D_2O moderator and amount of produced Pu-239 nuclide. The D_2O moderator has smaller neutron absorption cross section than H_2O. The value of neutronic parameter of the core shows a very small change, far below of design value (reserved excess reactivity for experiment ≤ 2 %Δk/k and ppf ≤ 1.4).

Effect of UO_2 Pellet Radius

Figure 8 shows the effect of irradiation time on thermal power generated by natural UO2 pin with variation in radius of pellets. From the figure, it can be seen as irradiation time increases thus thermal power increases for all of pellet radius. This penomenon is caused by the increasing content of U-235 and Pu-239 nuclides as a function of pellet radius. In addition to that, the largest thermal power is still below the accepted standards in designing the equipment, which is equal to 700 W/cm for the maximum irradiation time of 40,000 hours (≈40 GWD/t) with height of fuel pellet is about 0.98 cm.

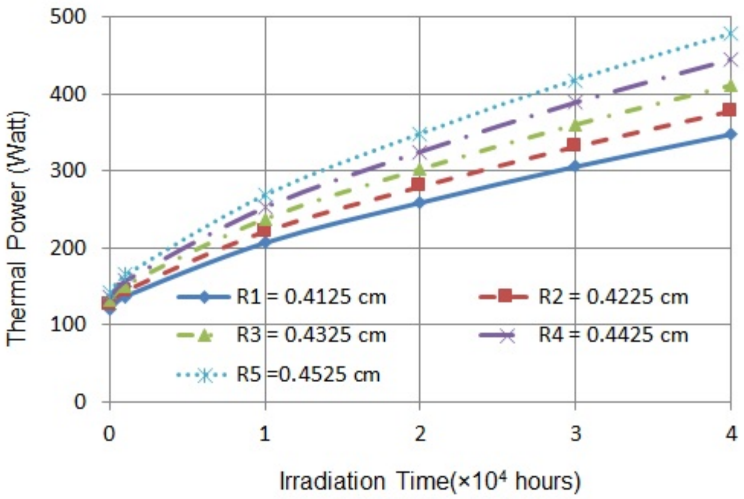

Figure 8. Effect of irradiation time on thermal power generated by UO_2 pin with variation in radius of pellet

Effect of irradiation time on fast and thermal neutron flux at the target irradiation with variation in radius of pellet is showed at the Figure 9 and 10, respectively. Figure 9 indicates that there is dependency of pellet radius with fast neutron flux. As mentioned previously, it is caused by the acumulation of Pu-239 nuclide. Pu-239 produces fast neutrons through fission reaction. From Figure 10, the larger radius of pellet will result in smaller thermal neutron flux. It is caused by absorbtion reaction between Pu-239 nuclides and thermal neutron.

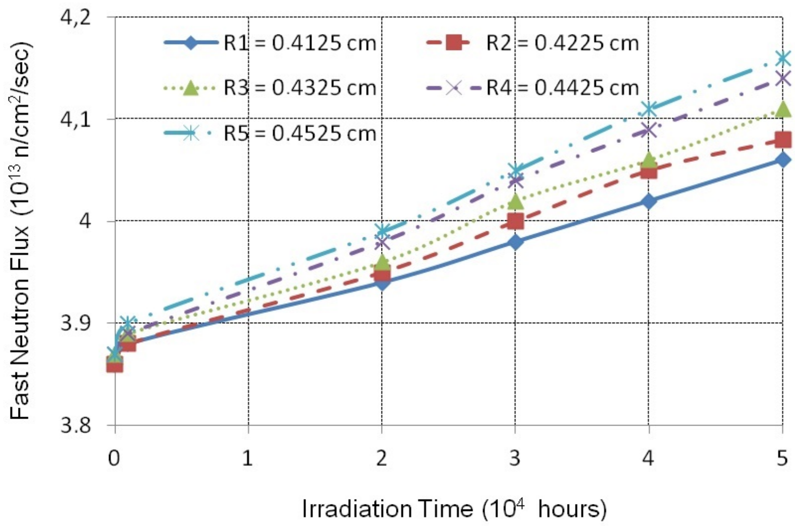

Figure 9. Effect of irradiation time on fast neutron flux at the target irradiation with variation in radius of pellet

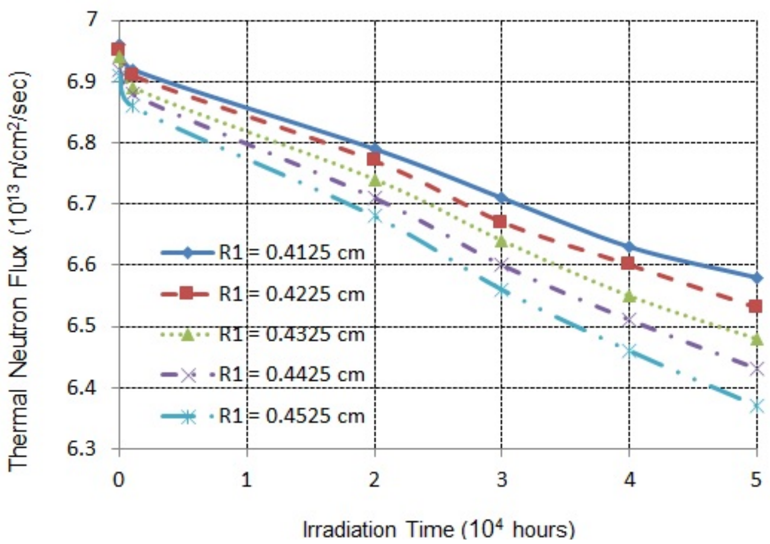

Figure 10. Effect of irradiation time on thermal neutron flux at the target irradiation with variation in radius of pellet

Effect of D_2O Moderator Densities

Figure 11 shows effect of irradiation time on power generated by UO_2 pin with variation in D_2O moderator density. The D_2O moderator density changes are assumed that there is an effect of the change in pressure inside the pressurized tube. The increasing of pressure (decreasing density) of the D_2O moderator causes thermal power produced by fuel pin of natural UO_2 increase. This phenomenon due to the smaller density of the moderator, then the number of atoms D will be smaller. Therefore, it will also lead to fewer thermal neutrons absorbtion by the moderator. From these figure is also showed that the longer irradiation time will also result greater thermal power. As well as mentioned above, it is also caused by the accumulation of Pu-239 nuclides.

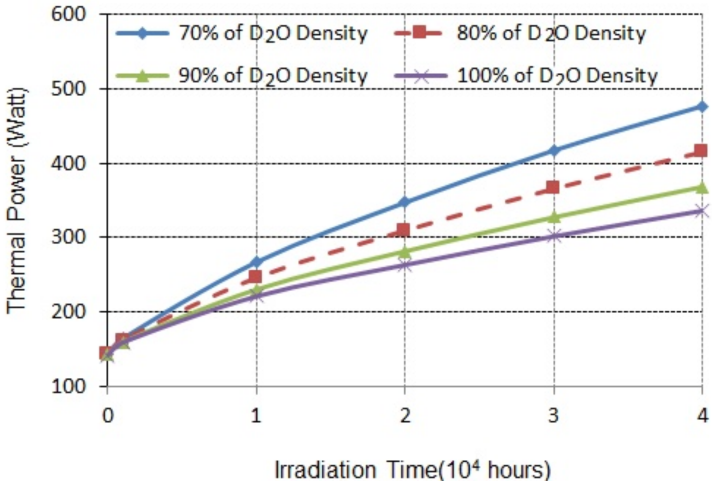

Figure 11. Effect of irradiation time on power generated by UO_2 pin with variation in D_2O moderator density

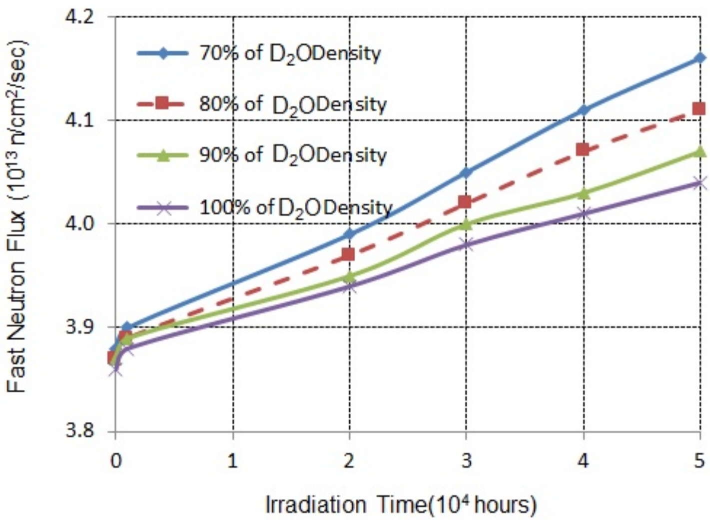

Figure 12. Effect of irradiation time on fast neutron flux at the target irradiation with variation in D_2O moderator density

Effect of irradiation time on thermal and fast neutron flux at the target irradiation with variation in D_2O moderator density is showed in Figure 12 and 13, respectively. In the beginning time of irradiation, the change of the moderator density showed only slightly change in the thermal neutron flux and almost same in the fast neutron flux. But with increasing of irradiation time, then the difference of fast and thermal neutron flux will be even greater due to accumulation of the Pu-239 produced.

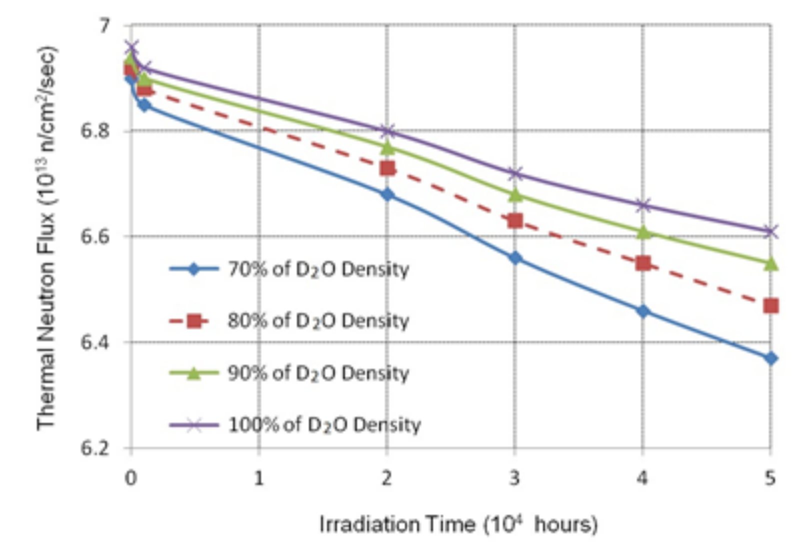

Figure 13. Effect irradiation time to thermal neutron flux at the target irradiation with variation in D_2O moderator density

CONCLUSION

The results of calculations show that using D_2O moderator in PHWR fuel pin with natural UO_2 is more effective than H_2O. The longer irradiation time will generate a higher thermal power. The higher pressure, or the smaller moderator density, will generate higher thermal power due to increasing length of irradiation time. The maximum limit of irradiation time to fuel pin PHWR natural UO_2 with high pressure D_2O moderator is approximately 9.5E4 hours, with restrictions linear power design capabilities of the equipment about 700 W/cm. During irradiation, neutronic parameters of the core such as excess reactivity and ppf have very small change, still far below design value.

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