INFLUENCE OF MODERATOR-TO-FUEL VOLUME RATIO ON Pu AND MA RECYCLING IN EQUILIBRIUM FUEL CYCLES OF PWR

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ABSTRACT

An influence of moderator-to-fuel volume ratio (MFR) changes by changing the pin-pitch of fuel cell on characteristics of several equilibrium fuel cycles for PWR has been studied. Special emphasis was given on the required uranium enrichment as well as the amount of natural uranium for criticality of PWR and the toxicity of discharged heavy metal (HM) in spent fuel. A conversion ratio (CR) increases with increasing number of recycled plutonium and/or minor actinides (MA) in the reactor. If plutonium and MA are recycled with MFR = 0.4, the CR becomes more than unity, and the system changes to a breeder reactor. Moreover the depleted uranium is more than enough for loaded fuel. Considering the MFR values of 1.0 and above, the required enrichment becomes minimum at MFR = 2.0, which is the typical MFR value of the present PWR.

1. INTRODUCTION

The delay of commercialization of fast reactor argues for another solution to the plutonium produced in light water reactor (LWR). As one option, recently, the plutonium recycling in LWR becomes an important consideration[1]. We have reported the study on plutonium and minor actinides (MA) recycling in equilibrium fuel cycles for pressurized water reactors (PWR). The results showed that plutonium and MA recycling can reduce the required uranium enrichment and the required natural uranium supply by small amount[2].

In the present study, the influence of moderator-to-fuel volume ratio (MFR) changes on characteristics of several equilibrium fuel cycles for PWR are investigated by changing the pin-pitch of fuel cell, as shown in Table I. Special emphasis is given on the required uranium enrichment as well as the amount of natural uranium for criticality of PWR and the toxicity of discharged heavy metal (HM) in spent fuel. Since the emphasis of the present study lie on the systematic understanding rather than applicability, we have chosen the broad range of the investigated MFR values from 0.1 to 4.0.

The following fuel cycles of 3 GWt PWR are investigated, where all fission products (FP) and final

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products of HM natural decay chain (tellurium – francium) are discharged from the reactor at a standard rate (33%/year).[2,3]

Case 1: All HMs are discharged from the reactor with the standard rate.

Case 2: All HMs except Pu are discharged from the reactor with the standard rate. Plutonium is discharged at the rate of one-half of the standard rate.

Case 3: All HMs except Pu are discharged from the reactor with the standard rate. Plutonium is confined in the reactor.

Case 4: All HMs except uranium are confined in the reactor. Uranium is discharged from the reactor with the standard rate.

Table I Basic physics characteristics of studied PWR

Thermal power Output	3000 MW
Fuel pellet average power density	280 Wcm^{-3}
Fuel pellet diameter (= inner pin diameter)	0.80 cm
Fuel pin outer diameter	0.96 cm
Pin pitch	0.88 – 1.65 cm
Moderator-to-fuel volume ratio (MFR)	0.1 – 4.0
Materials	
Fuel type	Oxide
Cladding	Zircaloy-4
Coolant	Light water

2. NEUTRON SPECTRA AND CONVERSION RATIO

In general, the changes of MFR result in change of a neutron spectrum as well as one-group microscopic cross-sections. Figure 1 shows the neutron spectra for all MFR values of the investigated Case 1. As can be expected, the neutron spectra become harder with decreasing MFR. For MFR = 0.1 the neutron spectrum even closes to that of fast reactor. The investigated Case 2 through 4 results in the similar trends of the neutron spectra. Even though the neutron spectra for Cases 2 through 4 are not shown here, the neutron spectrum becomes harder with increasing number of confined HM in the core. This fact was clearly confirmed from the results in our previous study[2].

One of the important parameters for nuclear reactor systems is their conversion ratio (CR). Figure 2 shows the conversion ratio of all investigated Cases as a function of the MFR values. The conversion ratio here was calculated from the following equation[4]:

$$CR = \frac{Capture\_rate\_of\_(^{238}U + ^{238}Pu + ^{240}Pu)}{Absorption\_rate\_of\_(^{235}U + ^{239}Pu + ^{241}Pu) + Decay\_rate\_^{241}Pu} \quad (1)$$

The decay rate of ^{241}Pu was considered in calculating CR since the half-life of this nuclide is only 14.4 years. As expected, the CR increases with decreasing of the MFR values, though there is a small negligible local optimum

at MFR = 1.0 for Case 1. Simultaneously, the CR also increases with increasing number of confined HM in the reactor. For Case 4 with MFR = 0.4, the CR becomes more than unity, and the system changes to become a breeder reactor.

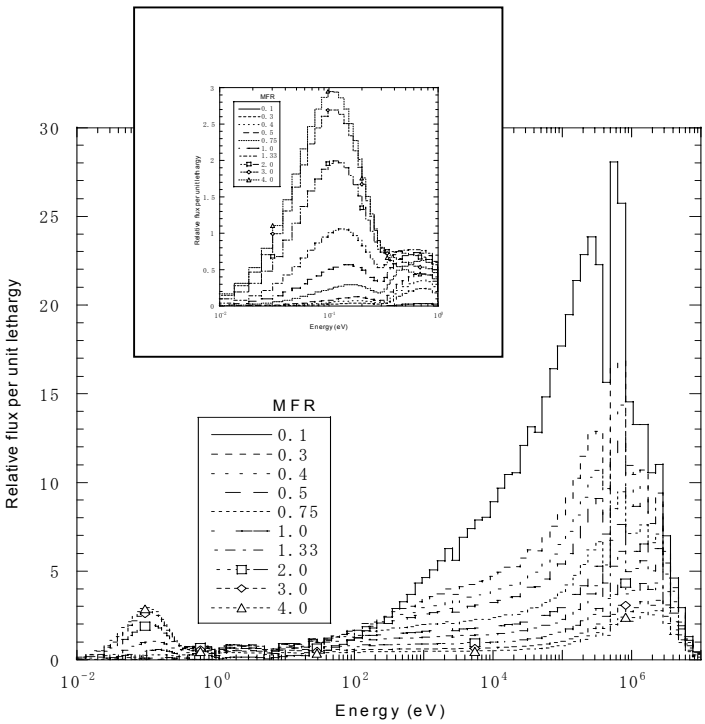

Figure 1. Neutron spectrum for Case 1

Safety aspect was not considered yet in this study. However, for void coefficient problem, it can be analyzed from Figure 2. The MFR value of the present standard PWR of about 2.0 was chosen from the safety aspect point of view. When we consider MFR > 2.0, the void coefficient would be negative since the moderation increases. If we consider 1.0 <MFR<2.0 the void coefficient would be lower than that of MFR = 2.0 (still negative). The positive void coefficient will occur for MFR<1.0 of Case3 and Case 4. It should be noted that in this study the cell calculation with infinite cylindrical geometry was employed. The void coefficient problem for MFR<1.0 of Case 3 and Case 4 can be overcome by increasing leakage.

3. REQUIRED URANIUM ENRICHMENT AND NATURAL URANIUM SUPPLY

The burnup equation of the equilibrium fuel cycle can be expressed in a matrix form as follows[2,3]:

$$\mathbf{Mn} = \mathbf{s} \quad , \tag{2}$$

where the elements of matrix **M** comprise all the transmutation parameters of all nuclides such as natural decay constants and microscopic transmutation cross-section. The **n** and **s** are the vectors of the number density of nuclides in the reactor core and the supply rate of fuel nuclides (uranium), respectively. The one-group

microscopic cross-sections for Eq. (1) are calculated by using SRAC95[5] code with nuclear data prepared from JENDL-3.2[6].

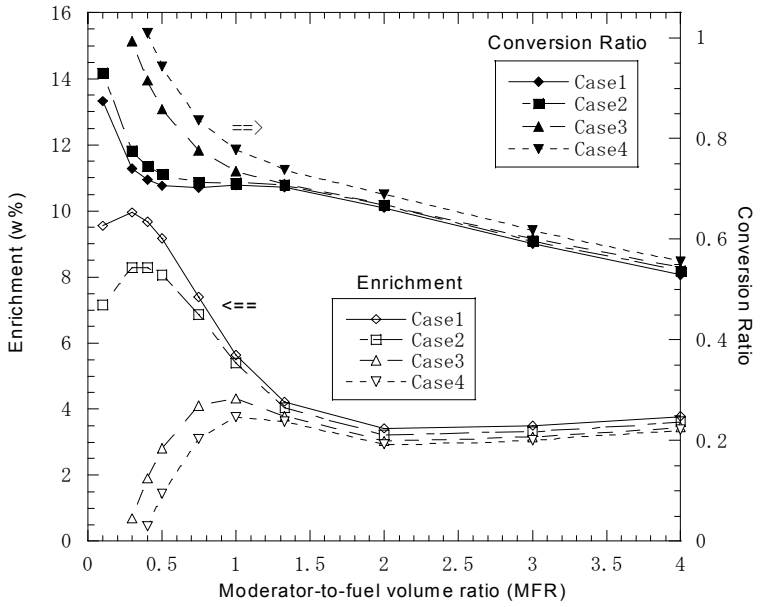

Figure 2. Conversion ratio and required uranium enrichment

In order to evaluate the performance of the investigated fuel cycles, we used the nuclide importance values. The nuclide importance vectors **f** and **a** can be calculated from the following adjoint equations:

$$\mathbf{M}^t\mathbf{f} = \phi\nu\boldsymbol{\sigma}_\mathbf{f}, \qquad \mathbf{M}^t\mathbf{a} = \phi\boldsymbol{\sigma}_\mathbf{a}, \tag{3}$$

where $\mathbf{M}^t$ is the adjoint matrix of **M**. ϕ is neutron flux. $\boldsymbol{\sigma}_\mathbf{f}$ and $\boldsymbol{\sigma}_\mathbf{a}$ are the vectors of microscopic fission cross-sections and microscopic absorption cross-sections, correspondingly. The ν represents the number of neutrons produced in each fission reaction. We have called **f** and **a** as fission neutron importance and absorbed neutron importance, respectively. The fission neutron importance represents the number of neutrons produced from fission of one nucleus of the studied nuclide and its family members (reaction products) during its existence in the reactor. While the absorbed neutron importance represents the number of neutrons absorbed by one nucleus of the studied nuclide and its family members during its presence in the reactor[2].

By using the nuclide importance vectors, the infinite multiplication factor, k, can be expressed as the following equation.

$$k = \frac{(\mathbf{v\sigma_f}, \mathbf{n})}{\alpha(\mathbf{\sigma_a}, \mathbf{n})} = \frac{(\mathbf{f}, \mathbf{s})}{\alpha(\mathbf{a}, \mathbf{s})}, \tag{4}$$

where α is a correction parameter for estimating neutron absorption of non-fuel nuclides such as coolant and structural materials. The actual calculation for k is performed by SRAC95 code.

In order to judge the criticality of the system, the neutron leakage from the system should be evaluated. For current PWR, the neutron leakage is estimated about 2% of produced neutrons. Then the following condition is employed for the criticality condition in the present study.

$$k = 1.02 \equiv k_c \tag{5}$$

The uranium enrichment to satisfy the criticality condition for each case is calculated as follows. The equilibrium burnup calculation is performed to determine the flux level and the number density of each nuclide in the fuel pellet. This calculation is coupled with SRAC95 cell calculation code in order to get the neutron spectrum and the one-group microscopic cross-sections of each investigated case. From this coupling calculation procedure we evaluate the value of the infinite multiplication factor, k. If k equal to k_c, then we choose the initial enrichment input as the required uranium enrichment for the criticality of the investigated case. In case k differs from k_c, the uranium enrichment is determined by using Eq. (4) and the following ones.

$$s_{24} + s_{25} + s_{28} = 100 \ , \tag{6}$$

$$100 s_{24} - 0.9937 s_{25} = -0.1925 \ , \tag{7}$$

where s_x is an atomic percent of uranium isotopes (^{234}U, ^{235}U and ^{238}U) in the supplied fuel. The Eq. (7) is given by enrichment condition[2].

The required amount of natural uranium, S_0, which given by the following Eq. (7) is calculated by means of two different enrichment processes where the concentration of ^{235}U in the tail was chosen to be 0.3 $^{w}/_{0}$ and 0.1 $^{w}/_{0}$, respectively[3].

$$S_0 = \frac{(e_1 - e_2)}{(e_0 - e_2)} S_1 \ , \tag{8}$$

where S_1 is the amount of required enriched uranium. Here e_0, e_1, and e_2 are the ^{235}U abundance in natural uranium (0.711 $^{w}/_{0}$), the concentration of ^{235}U in the produced enriched uranium and the concentration of ^{235}U in the tail of enrichment plant, correspondingly.

Figure 2 also shows the required uranium enrichment for criticality of all investigated Cases. Considering the MFR values of 1.0 and above, the required enrichment becomes minimum at MFR =2.0 for all investigated fuel cycles. In dealing with the tight lattices, i.e., the MFR values of less than 1.0, the investigated fuel cycles can be separated into the plutonium discharging case (Cases 1 and 2) and the plutonium recycling case (Case 3 and 4). For the plutonium discharging case, in general, the required uranium enrichment increases with decreasing of the MFR values. On the other hand, for the plutonium recycling case, the required uranium enrichment decreases significantly with decreasing of the MFR values. For Case 4 with MFR = 0.4 (CR = 1.009) the depleted uranium is more than enough for loaded fuel. This is the reason why we did not continue to evaluate for the MFR < 0.3 and the MFR < 0.4 of Cases 3 and 4, respectively. It can be expected that the tail

uranium (0.3w% ^{235}U) is enough for the loaded fuel in Case 4 with the MFR of less than 0.4.

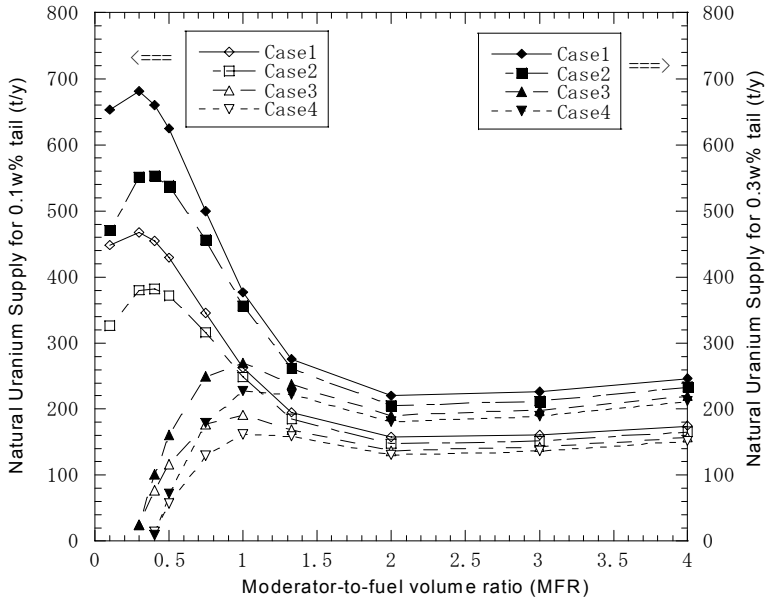

Figure 3. Required natural uranium supply for all Cases

The annual amount of required natural uranium supply has the same pattern likes the required uranium enrichment as shown in Figure 3. Both for 0.3 w% and 0.1 w% tails of the enrichment processes are illustrated in this figure. Again, when we consider the MFR values of 1.0 and above, the required natural uranium supply becomes minimum at the MFR =2.0 for all investigated fuel cycles. And if we have a look at the MFR values of less than 1.0, for the plutonium discharging case, in general, the required natural uranium supply increases with decreasing of the MFR values. On the other hand, for the plutonium recycling case, the required natural uranium supply decreases considerably with decreasing of the MFR values.

The MFR = 2.0 is the moderator-to-fuel volume ratio of the present standard PWR[7]. These results confirmed that the present-day PWR is the optimum PWR design from the point of views of the required uranium enrichment and the required amount of natural uranium supply.

The burnup and the neutron flux for all fuel cycle cases are listed in Table II**.** The burnup is defined as the produced energy (GWd) per ton loaded fuel. According to Table II, the burnup for Case 1 was similar for all the MFR values since the annual amount of loaded enriched uranium fuel was same. The burnup increases with increasing number of confined HM as also mentioned in reference[2]. The neutron flux increases with decreasing of the MFR values. The latter parameter decreases with increasing number of confined HM in the reactor core. In thermal energy region microscopic absorption cross-section of fissile plutonium and trans-plutonium nuclides

with odd number of neutrons like ^{242}Am, ^{242m}Am, ^{244}Am, ^{243}Cm, and ^{245}Cm have the absorption cross-sections larger than ^{235}U[8], therefore by recycling Pu and trans-plutonium in PWR the flux depression occurs, and the neutron spectrum becomes harder.

Table II Loaded fuel, burnup, and neutron flux for all Cases

Physics Parameters	Case	Moderator-to-fuel volume ratio (MFR)									
		0.1	0.3	0.4	0.5	0.75	1.0	1.33	2.0	3.0	4.0
Burnup (GWd/t fuel)	1	37.80	37.81	37.81	37.81	37.81	37.81	37.81	37.81	37.79	37.81
	2	38.80	38.58	38.51	38.45	38.31	38.20	38.10	38.01	37.97	37.95
	3		42.99	42.33	41.75	40.59	39.74	39.01	38.48	38.34	38.30
	4			43.18	42.66	41.56	40.65	39.76	39.01	38.81	38.77
Neutron Flux (x 10^{14}/ cm2.s)	1	13.3	8.33	7.20	6.40	5.16	4.44	3.83	3.09	2.48	2.11
	2	13.7	8.48	7.30	6.47	5.18	4.42	3.80	3.06	2.46	2.09
	3		9.10	7.72	6.76	5.26	4.40	3.73	2.98	2.40	2.04
	4			7.84	6.85	5.30	4.41	3.71	2.95	2.36	2.01

The change of MFR results in the change of the plutonium isotopes composition, i.e., a plutonium vector as well as a plutonium quality that is the percentage of the amount of fissile plutonium in the total plutonium. The plutonium quality for all investigated fuel cycles are summarized in Table III. From results in this table, for the plutonium discharging cases (Cases 1 & 2) the plutonium quality decreases monotonically with increasing of the MFR values. In case of the plutonium recycling cases (Cases 3 & 4) the plutonium quality reaches the maximum values at the MFR = 0.75. The plutonium quality is inversely proportional to the increasing number of confined HM in reactor.

Table III Plutonium quality (% fissile Pu) for all investigated Cases

Case	Moderator-to-fuel volume ratio (MFR)									
	0.1	0.3	0.4	0.5	0.75	1.0	1.33	2.0	3.0	4.0
1	87.60	86.85	86.45	85.78	83.00	79.06	73.45	65.32	59.91	57.68
2	81.65	82.23	82.45	82.24	80.13	76.32	70.20	60.60	54.03	51.17
3		62.07	64.88	67.11	69.42	67.79	62.05	49.53	40.10	35.44
4			62.60	64.29	65.73	64.05	59.25	47.97	38.54	33.74

Figure 4 shows the variation of the number of neutrons produced per neutron absorbed in the isotope (η value) with energy for ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu.[9] The average value (on the neutron spectrum) of η of ^{239}Pu is higher than that of ^{235}U, and the η value of ^{235}U is less than 2.0 at fast energy region up to 200 keV. Therefore, ^{239}Pu is a better fissile isotope than ^{235}U for tight-lattice PWR systems. This fact together with the change of number density (not shown here) may explain the pattern of the required uranium enrichment as a function of the MFR and the fuel cycle in the low MFR value region (MFR < 1.0). For Cases 1 & 2, the number density of produced ^{239}Pu still smaller than that of ^{235}U, so the better performance of ^{239}Pu could not play a dominant role.

The number density of the produced ^{239}Pu in Case 2 is larger than in Case 1, so that the required enrichment as well as the required amount of natural uranium for Case 2 is smaller than that for Case 1. For Cases 3 & 4, the number density of produced ^{239}Pu really larger than that of ^{235}U, so the better performance of ^{239}Pu could dominate the whole fissile in the core. Therefore, the required enrichment as well as the required amount of natural uranium decreases with decreasing of the MFR value for the low MFR region.

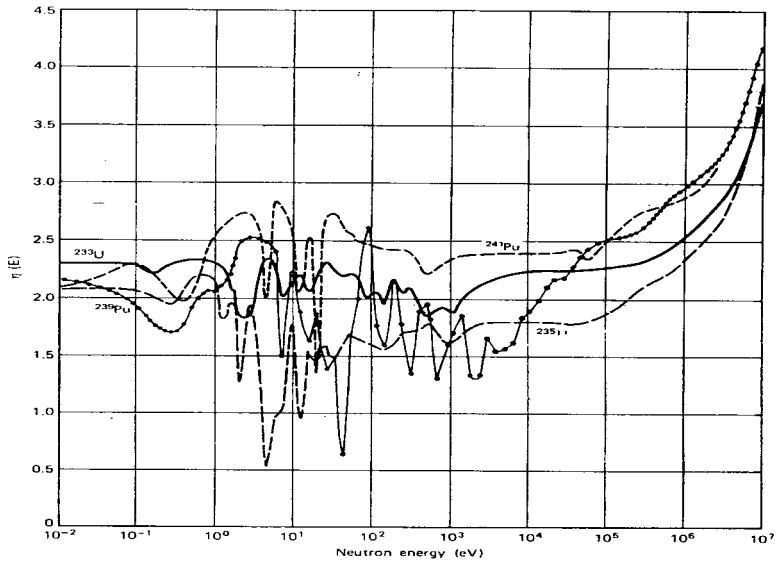

Figure 4. η value for fissile as a function of energy

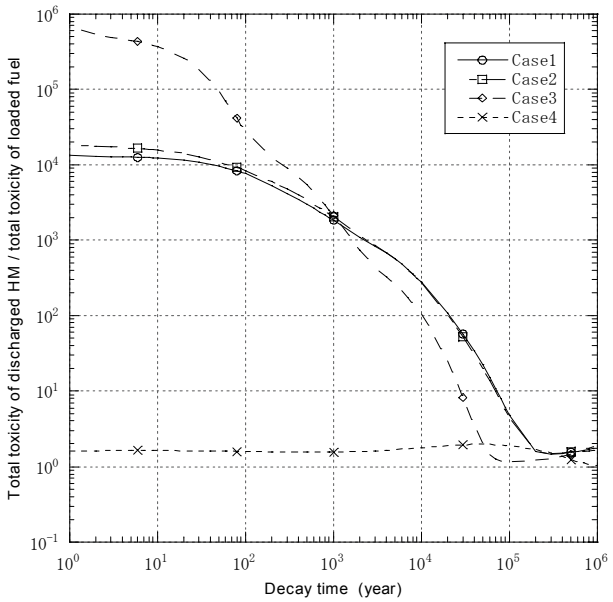

Figure 5 Toxicity ratio for MFR = 0.4

4. NUCLIDE IMPORTANCE AND TOXICITY OF HEAVY METALS

Table IV shows the fission neutron importance and the absorbed neutron importance of selected actinides for all considered Cases. For Cases 1 & 2, the fission neutron importance of fissile increases with increasing number of confined HM and simultaneously with increasing of the MFR values. The fission neutron importance of transuranium become close to the ν-value, the number of neutrons produced in one fission reaction, when they are confined in the reactor. The absorbed neutron importance shows the same trend as that of the fission neutron importance. The fission neutron importance and the absorbed neutron importance of ^{243}Am and ^{244}Cm are not so sensitive to the change of the MFR value, when they are recycled in the core with the MFR value less than 2.0.

Figure 5 and Figure 6 show the ratio of the total toxicity of discharged HM from the reactor to the total toxicity of loaded fuel into the reactor (toxicity ratio of HM) for the MFR = 0.4 and the MFR = 4.0, respectively. Two extreme values of the MFR were selected to demonstrate that for plutonium recycling cases (Case 3 & 4) the toxicity ratio decreases with increasing of the MFR value. As well known, the main toxicity contributor in the spent fuel is plutonium.[2] The production of plutonium (especially ^{239}Pu, ^{240}Pu, and ^{241}Pu) is inversely proportional to the MFR value for plutonium recycling cases.

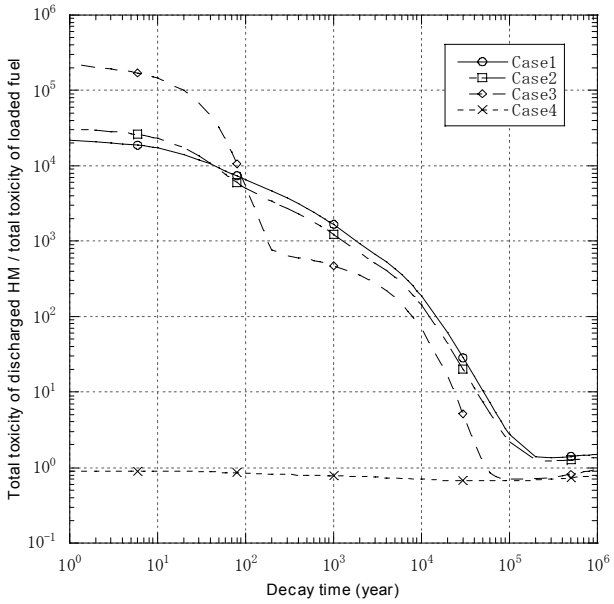

Figure 6 Toxicity ratio for MFR = 4.0

5. CONCLUSIONS

The influence of MFR changes on characteristics of several equilibrium fuel cycles for PWR has been

studied. In this study, MFR change of 0.1 to 4.0 was achieved by changing the pin-pitch of the fuel cell.

The conversion ratio is inversely proportional to the MFR and it is directly proportional to the number of confined HM in the reactor.

Considering the MFR values of 1.0 and above, the required enrichment and the required amount of natural uranium supply become minimum at the MFR =2.0 for all investigated fuel cycles. These results confirmed that from the required uranium enrichment and the required amount of natural uranium supply point of views, the present PWR design is considered as the optimum.

For the tight lattices (MFR < 1.0), the investigated fuel cycles can be separated into the plutonium discharging case (Cases 1 & 2) and the plutonium recycling case (Cases 3 & 4). For the plutonium discharging case, in general, the required uranium enrichment as well as the required amount of natural uranium supply increases with decreasing of the MFR values. On the other hand, for the plutonium recycling case, these two characteristics decreases significantly with decreasing of the MFR values. For Case 4 with the MFR = 0.4 the depleted uranium is enough for loaded fuel. Therefore, the high conversion tight lattice PWR can improve the fuel utilization only when we consider the plutonium recycling or plutonium and MA recycling. Once through cycle of high conversion tight lattice PWR with UOX or MOX fuel would exhaust the uranium very quickly.

The production of plutonium and MA is inversely proportional to the MFR so that the toxicity ratio of HM is also inversely proportional to the MFR value.

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Table IV Nuclide importance values of selected actinides

Case	Importance	MFR	^{234}U	^{235}U	^{238}U	^{237}Np	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{243}Am	^{244}Cm
1	Fission	0.3	0.225	0.609	0.048	0.240	0.424	0.850	0.555	1.039	0.473	0.550
		0.4	0.236	0.620	0.049	0.237	0.416	0.938	0.654	1.086	0.510	0.604
		0.5	0.248	0.639	0.051	0.239	0.420	1.036	0.743	1.141	0.543	0.651
		0.75	0.291	0.729	0.056	0.266	0.477	1.301	0.947	1.308	0.619	0.759
		1.0	0.351	0.870	0.061	0.328	0.600	1.558	1.137	1.497	0.687	0.857
		1.33	0.437	1.071	0.065	0.440	0.819	1.804	1.326	1.697	0.736	0.936
		2.0	0.536	1.318	0.064	0.604	1.135	2.016	1.481	1.872	0.709	0.938
		3.0	0.566	1.435	0.058	0.673	1.296	2.088	1.506	1.926	0.606	0.845
		4.0	0.553	1.463	0.052	0.668	1.328	2.092	1.478	1.929	0.517	0.756
	Absorption	0.3	0.389	0.403	0.066	0.532	0.359	0.592	0.809	0.664	1.241	0.499
		0.4	0.426	0.413	0.066	0.565	0.377	0.678	0.972	0.713	1.267	0.551
		0.5	0.456	0.428	0.067	0.596	0.402	0.769	1.102	0.763	1.286	0.591
		0.75	0.522	0.485	0.071	0.687	0.506	1.000	1.347	0.899	1.321	0.661
		1.0	0.584	0.569	0.074	0.804	0.681	1.214	1.529	1.039	1.341	0.708
		1.33	0.652	0.686	0.077	0.975	0.962	1.410	1.686	1.175	1.339	0.734
		2.0	0.719	0.824	0.074	1.197	1.346	1.563	1.789	1.274	1.270	0.708
		3.0	0.725	0.878	0.067	1.278	1.529	1.595	1.773	1.277	1.152	0.636
		4.0	0.698	0.881	0.059	1.258	1.562	1.577	1.718	1.250	1.056	0.573
2	Fission	0.3	0.229	0.610	0.064	0.362	0.768	1.229	0.861	1.395	0.473	0.550
		0.4	0.234	0.611	0.063	0.361	0.754	1.304	0.964	1.424	0.501	0.593
		0.5	0.242	0.621	0.063	0.365	0.754	1.390	1.055	1.460	0.526	0.631
		0.75	0.271	0.684	0.066	0.401	0.814	1.624	1.254	1.581	0.586	0.721
		1.0	0.319	0.801	0.069	0.476	0.954	1.851	1.434	1.728	0.647	0.811
		1.33	0.399	0.998	0.072	0.612	1.211	2.073	1.621	1.897	0.705	0.901
		2.0	0.509	1.275	0.070	0.807	1.575	2.260	1.777	2.043	0.699	0.930
		3.0	0.548	1.411	0.063	0.882	1.743	2.318	1.803	2.078	0.607	0.851
		4.0	0.538	1.446	0.056	0.874	1.778	2.322	1.781	2.075	0.523	0.769
	Absorption	0.3	0.395	0.405	0.077	0.621	0.631	0.857	1.071	0.919	1.238	0.496
		0.4	0.425	0.410	0.077	0.655	0.656	0.945	1.244	0.963	1.258	0.543
		0.5	0.450	0.418	0.077	0.690	0.688	1.034	1.384	1.006	1.273	0.578
		0.75	0.502	0.458	0.079	0.792	0.816	1.254	1.643	1.120	1.301	0.644
		1.0	0.554	0.528	0.081	0.927	1.023	1.454	1.833	1.240	1.320	0.692
		1.33	0.618	0.642	0.082	1.132	1.362	1.640	2.002	1.363	1.325	0.727
		2.0	0.693	0.797	0.079	1.400	1.814	1.780	2.117	1.447	1.269	0.716
		3.0	0.706	0.863	0.071	1.492	2.009	1.802	2.104	1.436	1.158	0.651
		4.0	0.683	0.871	0.063	1.471	2.043	1.783	2.055	1.402	1.064	0.593
3	Fission	0.3	0.257	0.648	0.129	1.016	2.684	2.736	2.383	2.279	0.544	0.633
		0.4	0.249	0.623	0.120	1.013	2.653	2.711	2.334	2.255	0.532	0.632
		0.5	0.243	0.608	0.112	1.013	2.628	2.692	2.301	2.241	0.526	0.635
		0.75	0.240	0.609	0.100	1.035	2.588	2.666	2.264	2.234	0.531	0.660
		1.0	0.258	0.666	0.092	1.079	2.578	2.660	2.270	2.253	0.561	0.713
		1.33	0.316	0.827	0.086	1.157	2.602	2.669	2.303	2.294	0.620	0.805
		2.0	0.446	1.171	0.078	1.261	2.648	2.684	2.341	2.336	0.663	0.893
		3.0	0.505	1.354	0.069	1.279	2.660	2.684	2.333	2.330	0.595	0.845
		4.0	0.504	1.404	0.062	1.251	2.657	2.680	2.317	2.314	0.522	0.777
	Absorption	0.3	0.438	0.444	0.127	1.107	2.103	1.959	2.369	1.593	1.289	0.554
		0.4	0.449	0.429	0.120	1.144	2.163	1.999	2.445	1.611	1.278	0.572
		0.5	0.454	0.419	0.114	1.180	2.215	2.031	2.505	1.629	1.270	0.587
		0.75	0.467	0.416	0.105	1.285	2.345	2.090	2.616	1.681	1.262	0.622
		1.0	0.490	0.448	0.099	1.419	2.502	2.135	2.699	1.737	1.268	0.659
		1.33	0.540	0.540	0.094	1.616	2.726	2.175	2.774	1.797	1.281	0.703
		2.0	0.634	0.733	0.086	1.853	2.959	2.194	2.815	1.828	1.255	0.724
		3.0	0.664	0.827	0.077	1.905	3.014	2.177	2.795	1.804	1.159	0.677
		4.0	0.648	0.844	0.068	1.868	3.014	2.157	2.767	1.774	1.073	0.629
4	Fission	0.4	0.266	0.671	0.133	2.811	2.810	2.968	3.068	3.072	3.461	3.461
		0.5	0.257	0.649	0.125	2.799	2.797	2.965	3.067	3.070	3.461	3.461
		0.75	0.244	0.630	0.110	2.778	2.775	2.957	3.063	3.064	3.459	3.459
		1.0	0.251	0.663	0.101	2.778	2.775	2.950	3.061	3.062	3.459	3.458
		1.33	0.294	0.791	0.093	2.809	2.807	2.946	3.062	3.063	3.459	3.458
		2.0	0.420	1.136	0.084	2.871	2.871	2.942	3.063	3.063	3.451	3.451
		3.0	0.486	1.339	0.074	2.891	2.892	2.939	3.057	3.058	3.431	3.430
		4.0	0.488	1.394	0.066	2.894	2.895	2.936	3.052	3.052	3.410	3.410
	Absorption	0.4	0.470	0.480	0.130	3.105	2.270	2.188	2.983	2.225	3.412	2.429
		0.5	0.471	0.465	0.124	3.174	2.326	2.233	3.081	2.268	3.449	2.466
		0.75	0.471	0.447	0.113	3.332	2.461	2.313	3.247	2.352	3.511	2.529
		1.0	0.482	0.463	0.106	3.505	2.616	2.362	3.346	2.409	3.548	2.566
		1.33	0.519	0.536	0.099	3.752	2.847	2.399	3.412	2.449	3.579	2.595
		2.0	0.612	0.732	0.091	4.058	3.131	2.412	3.440	2.458	3.620	2.635
		3.0	0.648	0.835	0.080	4.157	3.212	2.400	3.442	2.454	3.665	2.679
		4.0	0.633	0.852	0.072	4.178	3.225	2.388	3.441	2.452	3.701	2.715