INVESTIGATION ON PWR-to-PWR FUEL RECYCLE BY DUPIC PROCESS

Masood Iqbal[1] and Chang Joon Jeong
Korea Atomic Energy Research Institute
P.O. Box. 105, Yuseong, Daejeon, 305-600 Korea
masiqbal@hotmail.com ; cjjeong@kaeri.re.kr

ABSTRACT

PWR-to-PWR fuel cycle model has been developed to recycle the spent fuel using the dry fabrication process. Two types of fuels were considered; first fuel was based on low initial enrichment with low discharge burnup and second one was based on more initial enrichment with high discharge burnup in PWR. For recycling calculations, the HELIOS code was used, in which all of the available fission products were considered. The decay of 10 years was applied for reuse of the spent fuel. Sensitivity analysis for the fresh feed material enrichment has also been carried out. With the increase of mixing material enrichment, saving of uranium reserves would be decreased. The uranium saving of low burned fuel increased from 4.2% to 7.4% in fifth recycling step for 5 wt% to 19.99 wt% mixing material enrichment. While for high burned fuel, there was no uranium saving, which implies that higher uranium enrichment required than 5 wt%. The required fresh fuel mixing with 15 wt% is about 20.0% and 37.0% of total fuel volume for low and high burned fuel, respectively. With multiple recycling, reductions in waste for low and high burned fuel became 80% and 63%, for first recycling, respectively. In this way, waste can be reduced more and the cost of the waste disposal reduction can provide the economic balance.

1. INTRODUCTION

The spent fuel from nuclear reactor is main concern in the world due to its radioactive hazards, and many studies have been carried out to reduce the spent fuel. For that purpose, recycling of the spent fuel was came into exist. Recycling through reprocessing does not provide nuclear proliferation resistance. Recently some studies have been performed for the use of PWR spent fuel into the CANDU reactors directly through Oxidation Reduction of oxide fuel (OREOX) process which is known as DUPIC fuel. The OREOX process is a proliferation resistance process because this is a dry fabrication process and there is no extraction of any sensitive material from the spent fuel. Only gaseous fission products and some percentage of other actinides will go away. As spent PWR fuel contains about 0.9 wt% of U^{235}, 0.56 wt% of Pu^{239}, and 0.08 wt% of Pu^{241}, resulting in a total fissile content of 1.5 wt% [1]. Economic analysis for DUPIC fuel handling, fabrication, cycle and disposal has also proved it to be feasible. With DUPIC fuel cycle it has been found that it can save uranium resources by 20 to 23% and also reduce the spent fuel arising by 65 to 67% [2-5].

In this study recycling of the PWR fuel into PWR fuel has been carried out using only dry fabrication process. During recycling some amount of fresh fuel was mixed to compensate the negative reactivity of fission products and to increase the fissile contents to achieve the desired burnup. First model was based on 3.5 wt% initial enrichment with burnup of 35000 MWd/T and second was based on 5.0 wt%

[1] Permanent address: Pakistan Institute of Nuclear Science and Technology
P.O. Nilore, Islamabad, Pakistan.

initial enrichment with 60000 MWd/T burnup in PWR. The low and high burned spent fuels were reused in PWR reactor with multiple recycling schemes. HELIOS computer code was used for calculations, and the available fission products in HELIOS library were used. Also, the decay of 10 years was applied for reuse of the spent fuel.

2. REACTOR CALCULATIONAL MODELING

2.1 PWR LATTICE MODEL

The reference PWR fuel assembly for spent fuel employed in this study is a typical 17 x17 fuel assembly of 950 MW (electric) PWR of Yonggwang power plant[6]. The initial uranium enrichment for low and high discharge burnup were 3.5 and 5.0 wt%, respectively. The design parameters are shown in Table I. To obtain the spent fuel composition pin cell calculations were performed. The geometry is illustrated in Fig. 1. The cell pitch was adjusted according to the fuel to moderator ratio. The gap between fuel and clad was treated separately. The specular reflective boundary condition was used to all the external surfaces of the cell. The normal operating temperature for fuel, clad and coolant/moderator were taken as 1000, 585, and 580 ^oK, respectively. No burnable poison was considered throughout PWR pin cell calculations.

Table I. Design parameters of typical PWR.

Parameters	Value
Rated Power ($MW_{thermal}$)	2775
Number of assemblies/channels	157
Active core height (cm)	365.76
Type	17 X 17
Cladding material	Zr-4
Fuel temperature (^oK)	1000
Clad temperature (^oK)	585
Moderator/coolant temperature (^oK)	580
Pin radius (cm)	0.4025
Clad inner radius (cm)	0.411
Clad outer radius (cm)	0.475
Lattice pitch (cm)	1.26
Power density (W/g)	41.73
H_2O / U molecular ratio, Lattice	2.8

2.2 LINEAR REACTIVITY MODEL

To evaluate the discharge burnup in PWR, linear reactivity model was used. In single batch refueling scheme the discharge burnup can be calculated directly from the burnup versus system reactivity. Fig. 2 shows the behavior of system reactivity with burnup for 17 X 17 PWR fuel assembly, for which the discharge burnup could be calculated on the basis of the excess reactivity of 0.045 for leakage. This value is a typical for an out-in fueling pattern. [7]

Generally multi-batch refueling scheme is used for PWR system. To calculate the discharge burnup in multi-batch refueling scheme, using the linear reactivity model with equal power sharing of assemblies, following formula has been used. [8]

$$B_s = \frac{2(m+h)^2}{(m+1)(m+2h)} B_1 \tag{1}$$

where B_1 is the single batch discharge burnup predicted by the lattice code and m is an integer and h is a fraction $0<h<1$. The m and h can be calculated by this expression:

$$S=m+h \tag{2}$$

where in the reload batch fraction $1/S$ be equal to the number of fresh fuel assemblies refueled at each cycle divided by the total number of assemblies in the core. Using this methodology, the discharge burnup with different batch loading was calculated. In our case we considered 1/3 batch size that is 52 fresh assemblies out of 157 total are replaced during the reloading.

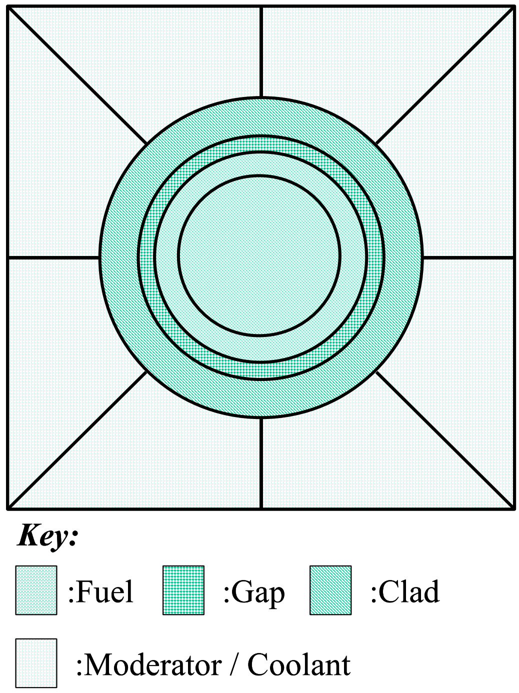

Figure 1. PWR pin cell geometry used in HELIOS calculations (Not on scale).

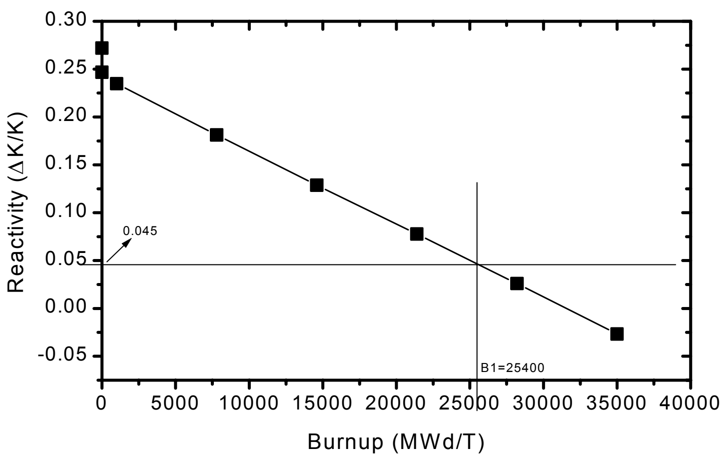

Figure 2. Reactivity with burnup for low burned fuel.

3. ANALYSIS

The low and high burned PWR spent fuel was direct utilized into PWR reactor after OREOX processing and with mixing of enriched fresh fuel.

3.1 MULTIPLE RECYCLING

For multiple recycling, enrichments for mixing of fresh fuel were taken from 5 wt% to 19.99 wt%. For the 5 wt% fresh fuel mixing, the desired discharge burnup (35000 MWD/T) was achieved after replacing 65% of spent fuel for low burned spent fuel while for high burnup fuel 100% has to be replaced. For 10 wt% fresh fuel mixing, the discharge burnup of 35000 MWd/T was obtained after replacing 31% of spent fuel for low burned fuel and for high burned fuel the replacement was 55%. In 15.0 wt% fresh fuel mixing, 20% and 37% of spent fuel was replaced for low and high burned fuels, respectively. For 19.99 wt% fresh fuel mixing, the spent fuel was replaced 16% and 29% for low and high burned fuel, respectively.
The system reactivity versus fuel burnup for low and high burned fuel due to the different fresh fuel enrichment are shown in Figs. 3-4. As the fresh fuel enrichment increases the system reactivity becomes lower for same discharge burnup. It is because of the presence of Pu loading.

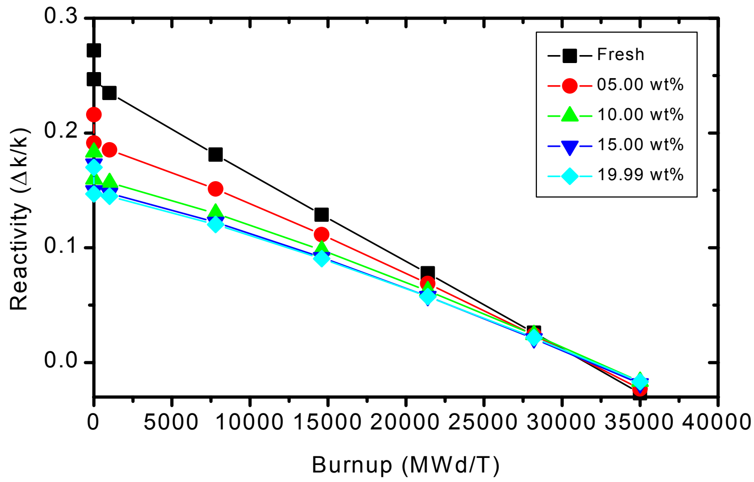

Figure 3. Reactivity versus burnup for different recycling steps with different fresh fuel enrichment in low discharge burnup spent PWR.

For high enriched fresh fuel mixing, the presence of Pu is more. The weight of important heavy elements at fresh and discharge stage was also calculated as shown in Tables II to III. It is well clear that the Pu contents increases as fresh fuel enrichment increases. The ^{239}Pu contents for low burned fuel are 0.18387, 0.36829, 0.42923, 0.25714 wt% with mixing of 5, 10, 15 and 19.99 wt% of fresh fuel, respectively. The ^{239}Pu contents for high burned fuel are 0.28093, 0.39687, and 0.44908 wt% with mixing of 10, 15 and 19.99 wt% of fresh fuel, respectively. The fissile contents at fresh and discharge stages were also calculated shown in above-mentioned tables. The fissile contents at fresh stage, for fresh fuel are 3.5 wt% in low burned fuel. This value increases as we increase the mixing fuel enrichment. For 5, 10, 15 and 19.99 wt% mixing fuels, the fissile contents are 3.8306, 4.25791, 4.34552, and 4.43127 wt%, respectively. In high burned fuel, the fissile contents at fresh stage are 5.0, 6.30558, 6.68934, and 7.09485 wt% for fresh and mixing of 10, 15, 19.99 wt% fuels, respectively.

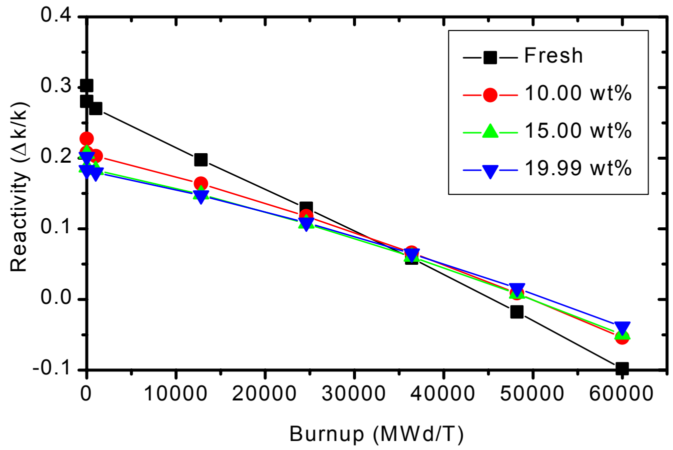

Figure 4. Reactivity versus burnup for different recycling steps with different fresh fuel enrichment in high discharge burnup spent PWR.

Table II. Composition of low burned spent PWR fuel.

Actinide	PWR Fresh Fuel		5 wt% Mixing		10 wt% Mixing		15 wt% mixing		19.99 wt% mixing	
	Fresh	Disch.*	Fresh	Disch.	Fresh	Disch.	Fresh	Disch.	Fresh	Disch.
U^{235}	3.50000	0.92992	3.61841	1.16728	3.83290	1.46086	3.85018	1.51941	3.90372	1.57740
U^{236}	0.00000	0.44581	0.15134	0.57807	0.30313	0.72072	0.35329	0.76361	0.37626	0.78722
U^{238}	96.5000	97.6021	95.8843	97.0461	95.1710	96.3966	94.9889	96.2211	94.8598	96.1038
Np^{237}	0.00000	0.04246	0.01480	0.06200	0.02964	0.08176	0.03454	0.08813	0.03679	0.09111
Pu^{238}	0.00000	0.01344	0.00464	0.03306	0.00928	0.05255	0.01082	0.05894	0.01152	0.06177
Pu^{239}	0.00000	0.53143	0.18387	0.58740	0.36829	0.65525	0.42923	0.67634	0.45714	0.68874
Pu^{240}	0.00000	0.22090	0.07491	0.24012	0.15005	0.26994	0.17488	0.28315	0.18625	0.28890
Pu^{241}	0.00000	0.13518	0.02832	0.15724	0.05672	0.18805	0.06611	0.19947	0.07041	0.20509
Pu^{242}	0.00000	0.04967	0.01686	0.07326	0.03377	0.09496	0.03936	0.10258	0.04192	0.10541
Am^{241}	0.00000	0.00325	0.01850	0.00636	0.03707	0.01117	0.04320	0.01296	0.04601	0.01396
Am^{243}	0.00000	0.00991	0.00336	0.02156	0.00673	0.03098	0.00785	0.03401	0.00830	0.03517
Fissile Contents	3.50000	1.59653	3.83060	1.91192	4.25791	2.30416	4.34552	2.39522	4.43127	2.47123

* Discharge burnup condition.

3.2 MASS FLOW CALCULATIONS

To calculate the mass flow during the recycling steps, typical PWR with power of 950 MWe, 34.23% efficiency and 0.8 capacity factor was used. The discharge burnup for PWR was considered as 35000 MWd/T and 60000 MWd/T for low and high burnd fuel. For material flow calculations, the tail assay in the enrichment facility is 0.25 wt%.
To calculate the uranium requirement for different uranium enrichment following relation was used

$$M_f = M_p \frac{(e_p - e_t)}{(e_f - e_t)} \qquad (3)$$

where e_p = Fresh feed material enrichment
e_f = Feed material enrichment for natural uranium (0.711 wt%)
e_t = Tail assay (0.25 wt%)

M_p = Mass of uranium to be charged in the DUPIC facility
M_f = Mass of uranium feed in enrichment plant

Table III. Composition of high burned spent PWR fuel.

Actinide	PWR Fresh Fuel		10 wt% Mixing		15 wt% mixing		19.99 wt% mixing	
	Fresh	Disch.*	Fresh	Disch.	Fresh	Disch.	Fresh	Disch.
U^{235}	5.00000	0.79362	5.97060	1.60857	6.21612	1.88177	6.55937	2.16109
U^{236}	0.00000	0.73669	0.32246	1.12890	0.45555	1.26970	0.51547	1.35894
U^{238}	95.0000	97.0318	93.0819	95.4330	92.4453	94.8381	91.9260	94.3777
Np^{237}	0.00000	0.09068	0.04043	0.13886	0.05711	0.15804	0.06462	0.16744
Pu^{238}	0.00000	0.04498	0.01952	0.10020	0.02758	0.12436	0.03121	0.13503
Pu^{239}	0.00000	0.63133	0.28093	0.77249	0.39687	0.82923	0.44908	0.86820
Pu^{240}	0.00000	0.29592	0.12939	0.31855	0.18279	0.33664	0.20684	0.34365
Pu^{241}	0.00000	0.20009	0.05405	0.24042	0.07635	0.26181	0.08640	0.27370
Pu^{242}	0.00000	0.10509	0.04600	0.13026	0.06499	0.14652	0.07354	0.15116
Am^{241}	0.00000	0.00659	0.03609	0.01248	0.05098	0.01601	0.05769	0.01846
Am^{243}	0.00000	0.03080	0.01347	0.05028	0.01903	0.05914	0.02153	0.06202
Fissile Contents	5.00000	1.62504	6.30558	2.62148	6.68934	2.97281	7.09485	3.30299

* Discharge burnup condition

The calculations were performed to get the loading of uranium per year in PWR and for discharge burnup of 35000 and 60000 MWd/T for low and high burned fuel respectively. The spent fuel as waste was also calculated for once through cycle and multiple cycles. The flows of the feed material (5 wt% case) and spent fuel for once through and multiple recycle models are shown in Figs. 5 – 6 for low and high burned fuels, respectively. The loading and disposal of uranium for feed material of different enrichments are given in Table IV. In the mixing of 5, 10, 15 and 19.99 wt% fresh fuel, the uranium loadings are 411.11, 402.46, 412.44 and 394.27 klb for low burned fuel. The uranium loadings for high burned fuel are 414.73, 422.08 and 442.74 klb for mixing with 10, 15 and 19.99 wt% fresh fuel. In low burned fuel, the wastes are 15.12, 7.12, 4.65, and 3.49 THM with mixing of 5, 10, 15 and 19.99 wt% fresh fuel. The wastes for high burned fuel are 7.43, 4.99, and 3.92 THM after mixing with 10, 15 and 19.99 wt% fresh fuel. The feed material loading as a result of fresh fuel mixing is depicted in Fig. 7. With mixing of 10 wt% fresh fuel, the uranium loading will be reduced to 31% and 55% for low and high burned fuel, respectively. The uranium saving for different enrichments in multiple recycling is shown in Fig. 8 for low burned PWR fuel. For first recycling step, the uranium saving would be 2.5, 3.5, 4.6, and 4.5% for mixing with 5, 10, 15, and 19.99 wt% fuel, respectively. The reduction in disposal of spent fuel in multiple recycling was also calculated for low and high burned PWR fuel. Sensitivity of disposal reduction due to feed material enrichment is shown in Fig. 9. The waste reductions for low and high burned fuel are 69 and 45%, respectively, after mixing with 10 wt% fresh fuel.

SUMMARY AND CONCLUSION

Recycling of spent PWR fuel in the PWR has been studied, for which dry fabrication process was considered. During the dry process, different enrichment of U^{235} was used for mixing. Two types of fuel cycle model for PWR were considered.

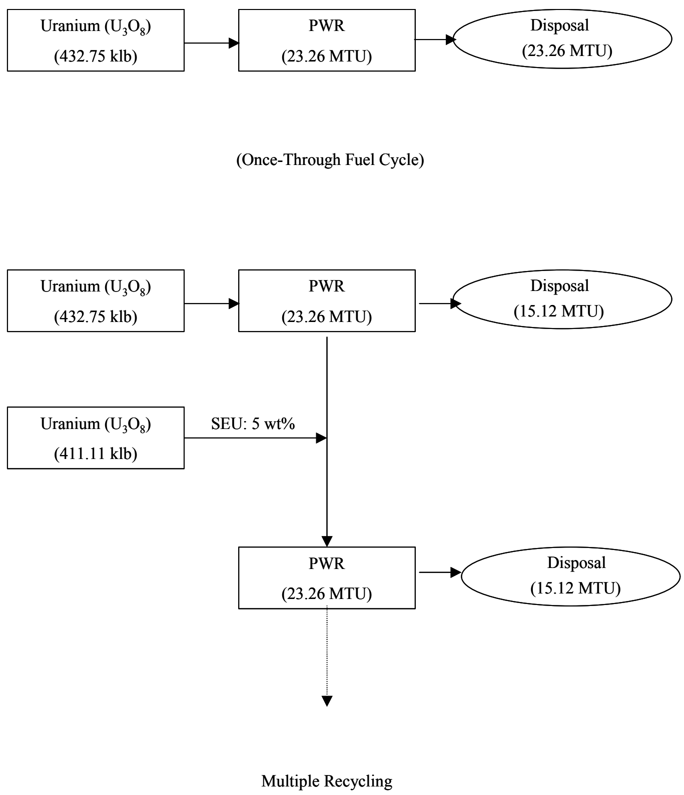

Figure 5 Recycling scheme and mass flow for low burned spent PWR fuel with 5 wt% uranium mixing.

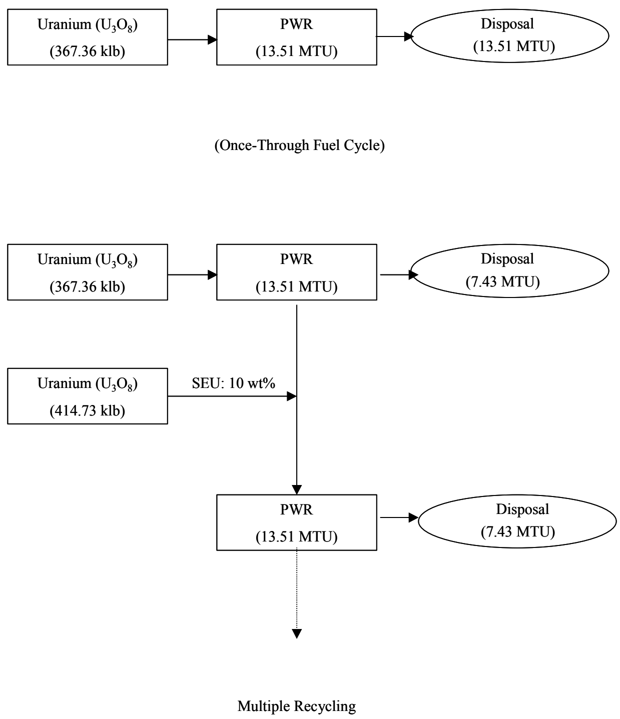

Figure 6 Recycling scheme and mass flow for high burned spent PWR fuel case with 5 wt% uranium mixing.

Table IV. Uranium loading and waste disposal during multiple recycling for mixing of different fresh uranium enrichments.

	Case	5 wt%	10 wt%	15 wt%	19.99 wt%
Loading (klb U_3O_8)	Low	411.11	402.46	412.44	394.27
	High	--	414.73	422.08	442.74
Disposal (THM)	Low	15.12	7.21	4.65	3.49
	High	--	7.43	4.99	3.92

Low - PWR spent fuel from low burnup
High - PWR spent fuel from high burnup
THM - Tons of Heavy metal

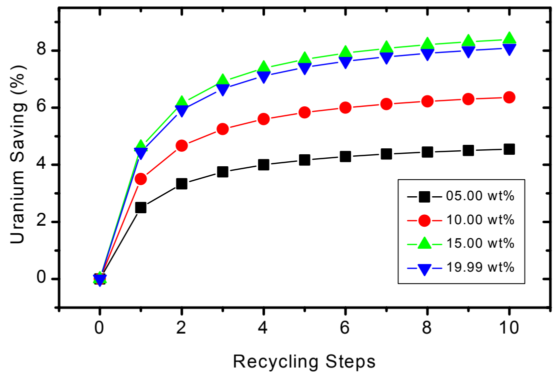

Figure 7 Change in feed material loading with fresh fuel enrichment.

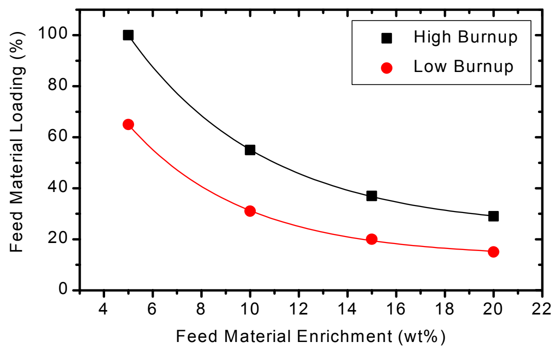

Figure 8 Uranium saving of multiple recycling with different fresh fuel enrichment for low burned spent PWR fuel.

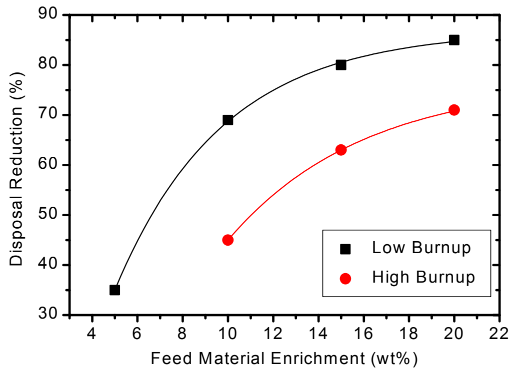

Figure 9. Change in disposal reduction with feed material enrichment.

First model was based on 3.5 wt% initial enrichment with burnup of 35000 MWd/T and second was based on 5.0 wt% initial enrichment with 60000 MWd/T burnup in PWR. Recycling calculations were performed using the HELIOS code, in which all of the available fission products were considered. The decay of 10 years was applied for reuse the spent fuel.

For 5, 10, 15 and 19.99 wt% fresh fuel mixing, the fissile contents are 3.8306, 4.25791, 4.34552, and 4.43127 wt%, respectively. In high burned fuel, the fissile contents are 6.30558, 6.68934, and 7.09485 wt% for mixing of 10, 15, 19.99 wt% fresh fuels, respectively.

In mass flow analysis, uranium saving/loss and waste reduction were calculated. In mixing of 5, 10, 15 and 19.99 wt% fresh fuel, the uranium loadings are 411, 403, 412 and 394 klb for low burned fuel. The uranium loadings for high burned fuel are 415, 422 and 443 klb for mixing with 10, 15 and 19.99 wt% fresh fuel, respectively. In low burned fuel, the wastes are 15.12, 7.21, 4.65, and 3.49 THM with mixing of 5, 10, 15 and 19.99 wt% fresh fuel, respectively. The wastes for high burned fuel are 7.43, 4.99, and 3.92 THM after mixing with 10, 15 and 19.99 wt% fresh fuel, respectively. With mixing of 10 wt% the uranium loading will be reduced to 31% and 55% for low and high burned fuel, respectively.

For first recycling step of low burned fuel, the uranium saving would be 2.5, 3.5, 4.6, and 4.5% for mixing with 5, 10, 15, and 19.99 wt% fresh fuel, respectively. The waste reductions for low and high burned fuel are 69 and 45%, respectively, after mixing with 10 wt% fuel. Although with high enrichment we have decrease in waste disposal. The uranium saving is also one of the parameter involved in multiple recycling. If enrichment of the mixing material increased the saving of uranium reserves would decreased.

From this study, it could be inferred that multiple recycling is possible in PWR using dry fabrication process. As for as mixing material enrichment is concerned, 15 wt% fresh fuel provides better results in uranium saving and disposal reduction as well as for low burned fuel. In high burned fuel, uranium saving is not expected, but waste disposal can be reduced. For mixing of 15 wt% fresh fuel, the required mixing is about 20.0 and 37.0% of fuel volume for low and high burned fuel, respectively. With multiple recycling, reductions in waste disposal for low and high burned fuel became 80 and 63%, respectively, for first recycling. The uranium saving for multiple recycling is 4.6% for low burned during fuel first step. Although mixing of fresh fuel is required, the cost of the waste disposal reduction can provide the economic balance. It is recommended that the economic analysis should be performed for multiple recycling in PWR.

ACKNOWLEDGEMENTS

This work has been carried out under the Nuclear Research and Development Program of the Korea Ministry of Science and Technology.

REFERENCES

1 H. B. CHOI, B. W. RHEE and H. PARK, "Physics study on direct use of spent pressurized water reactor fuel in CANDU (DUPIC)," *Nucl. Sci. Eng.*, **126**, pp.80-93 (1997).

2 H. B. CHOI, W. I. KO and M. S. YANG, "Economic analysis on direct use of spent pressurized water reactor fuel in CANDU reactors – I: DUPIC fuel fabrication cost," *Nucl. Tech.* **134**, pp.110-129 (2001).

3 H. B. CHOI, W. I. KO, M. S. YANG, I. HAMGUNG and B. G. NA, "Economic analysis on direct use of spent pressurized water reactor fuel in CANDU reactors – II: DUPIC fuel-handling cost," *Nucl. Tech.* **134**, pp. 130-148 (2001).

4 W. I. KO, H. B. CHOI, G. ROH and M. S. YANG, "Economic analysis on direct use of spent pressurized water reactor fuel in CANDU reactors – III: Spent DUPIC fuel disposal cost," *Nucl. Tech.* **134**, pp. 149-166 (2001).

5 W. I. KO, H. B. CHOI and M. S. YANG, "Economic analysis on direct use of spent pressurized water reactor fuel in CANDU reactors – IV: DUPIC fuel cycle cost," *Nucl. Tech.* **134**, pp. 167-186 (2001).

6 J. KANG et. al, "*Nuclear design report: Yonggwang 1 cycle 5*,"KWU B324/90/e204, KAERI & SIEMENS AG/UB KWU (1990).

7 M.J.DRISCOLL, T. J. DOWNAR and E. E. PILAT, "*The linear reactivity model for nuclear fuel management*," American Nuclear Society, La Grange Park, Illinois, USA (1990).

8 D. ROZON and W. SHEN, "A parametric study of the DUPIC fuel cycle to reflect pressurized water fuel management strategy," *Nucl. Sci. Eng.*, **138**, pp.1-25 (2001).