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Assessment of Reduced Moderation Water Reactor Fuel Cycle

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Abstract

An assessment of the fuel cycle performance of the Reduced-Moderation Water Reactor (RMWR) concept has been performed. The transuranics consumption rate, key safety coefficients, fuel handling issues, natural uranium requirements, and radiotoxicity of the waste passed to the repository or interim storage were evaluated and compared to other reactor-based transmutation options.

The decay heat and dose rate of the RMWR fuel are about 2 to 3 times higher than the corresponding values of the multirecycling LWR fuel (CORAIL concept) at the fabrication and charge stages because of the high plutonium content of the RMWR fuel. The radiotoxicity per Giga-Watt-day electric of the RMWR waste is smaller than that of the CORAIL waste 10 years after discharge, mainly due to the smaller fission products content arising from a lower discharge burnup. However, the radiotoxicity at a few hundred years is higher than those of the other reactor-based transmutation cases because of the higher amount of Am-241 produced in the RMWR case.

By recycling both the plutonium and uranium, the RMWR concept has excellent features in terms of the uranium utilization and waste reduction. However, the tradeoff for this benefit is an increase in the amount of heavy metal to be separated and reprocessed per unit electricity production.

KEYWORDS: Reduced Moderation Water Reactor, TRU transmutation, Fuel handling issues, Pu multirecycling.

1. Introduction

The Reduced-Moderation Water Reactor (RMWR) concept is being developed by the Japan Atomic Energy Research Institute (JAERI) in order to utilize uranium resources effectively and reduce spent fuel production. [1-5] In the RMWR design, a high conversion ratio (over 1.0) is attained by using a tight-lattice fuel rod arrangement and a high void fraction. The moderator-to-fuel volume ratio is less than 0.5 and the spacing between fuel rods is significantly narrower than those of typical light water reactors. The resulting neutron spectrum of the RMWR core is similar to that of a fast reactor rather than a typical LWR. In addition, the RMWR has a pancake shaped core to reduce the void reactivity coefficient by enhancing the axial leakage. Axially, two sub-cores are sandwiched between three depleted uranium layers, forming alternating zones of blanket and core.

Small production or stabilization of the transuranics (TRU) nuclides in a nuclear fuel cycle would provide the benefit of reducing the radiotoxicity of waste to be sent to a repository and would support nuclear power as a viable option in the future energy production. The hard spectrum of the RMWR core has a potential to mitigate the generation of higher actinides; however, the high conversion ratio may increase the plutonium (Pu) production. Thus, a fuel cycle analysis of the RMWR design has been performed to evaluate the transuranics consumption rate and the key safety coefficients. In this paper, the neutronics properties, mass flow rate, radiation hazard, and fuel handing issues of the RMWR fuel cycle are evaluated and compared to other reactor-based transmutation options such as the Advanced Light Water Reactor (ALWR) and the CORAIL concepts. [6-9]

2. Reduced-Moderation Water Reactor

JAERI has proposed several RMWR configurations based on traditional BWR and PWR designs [1-4]. Table 1 compares the design parameters of several RMWR cores with those of typical light water reactors. The operating conditions of the RMWR core are similar to those of a typical BWR, but the RMWR core designs have a tight triangular lattice and high void fraction. The high fissile content of the RMWR is due to the hard-spectrum of the design. Among the several candidate RMWR concepts, the small-scale RMWR core design is the basis of the fuel cycle analyses that have been performed in this work.

Table 1. Comparison of Design Parameters of RMWR Concepts

	PWR	BWR	High conversion RMWR	Long cycle RMWR		Large -scaled RMWR	Small -scaled RMWR
				Non-void	Void		
Thermal power, MW	3411	3579	3188	3926		3926	990
System pressure, MPa	15.5	7.2	7.2	7.2		7.2	7.2
Number of fuel assembly	193	748	924	252	61	900	283
Active height, cm	366.00	376.00	68.00	160.00		69.50	76.00
Effective radius of core, cm	168.50	183.00	360.00	290.00		380.00	197.52
Power density, MW/m^3	104.48	90.47	115.15	92.87		124.52	[a] 106.28
Average burnup, MWD/t	33.00	27.50	45.00	60.00		60.00	[b] 60.00
Average linear heat rate, W/cm	182.91	201.99	300.23	207.61		289.24	[a] 212.87
Fuel Form	UO_2	UO_2	MOX	MOX	MOX	MOX	MOX
Fissile enrichment, %	4	3	18	18	9.8	18	18
Fuel radius, cm	0.4095	0.5020	0.6440	0.5555	0.5555	0.6050	0.5670
Fuel pitch, cm	1.2500	1.6200	1.5800	1.3210	1.3210	1.5000	1.4300
Lattice shape	square	square	triangular	triangular	triangular	triangular	triangular
Assembly pitch, cm	21.50	15.50	22.00	30.42	30.42	22.80	22.36
Core average void fraction	0	0.4	0.7	0.6	0.6	0.7	0.7

a) Including core and blanket.
b) Average value of core only.

The assembly design of the small-scaled RMWR contains 217 MOX pins in a hexagonal duct; a bottom-entry, Y-shaped control blade is positioned between assemblies. To reduce the peak pin power, which results from the presence of the water gap between assemblies, the assembly consists of pins with four different fissile contents (14 to 19% Pu-fissile). The Y-shaped control rods contain B_4C with highly enriched B-10. A rod follower made of graphite is in the upper part to remove the water between fuel assemblies. The control rods

with the follower are inserted from the bottom of the core as in BWRs. In this study, however, the fuel cycle analyses were performed under the assumption that the control rods are fully out of the core. The core consists of 283 hexagonal fuel assemblies. The RMWR utilizes a pancake shaped core to enhance the axial leakage and make the void reactivity coefficient more negative. Axially, two short sub-cores are sandwiched between two depleted uranium layers and an internal blanket zone (between the cores). The average void fraction of the small-scale RMWR is reported as 69% in Ref. 5, but the distribution was not specified; An axial distribution for the void fraction was assumed in this study.

For Pu multirecycling in the RMWR concept, the fuel pins composing the reflector (i.e., upper and lower blankets), inner blanket and core zones are fabricated from external depleted uranium feed and the Pu and uranium extracted from the discharge of the previous cycle. A lead-time of two years is assumed from the assembly fabrication to its loading into the reactor. After the assembly is discharged from the reactor, a five-year post-irradiation cooling period is allowed before separation of the discharged fuel. Following the separation, the Pu and uranium are recycled to make the MOX fuel pins, while all fission products and minor actinides are passed to the repository.

3. Calculation Methods

The neutronic environment of the RMWR core is significantly different from that of typical light water reactors due to its tight lattice, high void fraction, and high Pu content (see Figure 1). An initial investigation of the self-shielding effect of the RMWR fuel was performed by comparing the eigenvalues of the homogeneous and heterogeneous fuel cells. The differences in the eigenvalues of the heterogeneous and homogeneous calculations are 36% and 0.9% for the traditional LWR UO_2 fuel and RMWR fuel, respectively. This result indicates that the RMWR core is closer to a fast system rather than a thermal system.

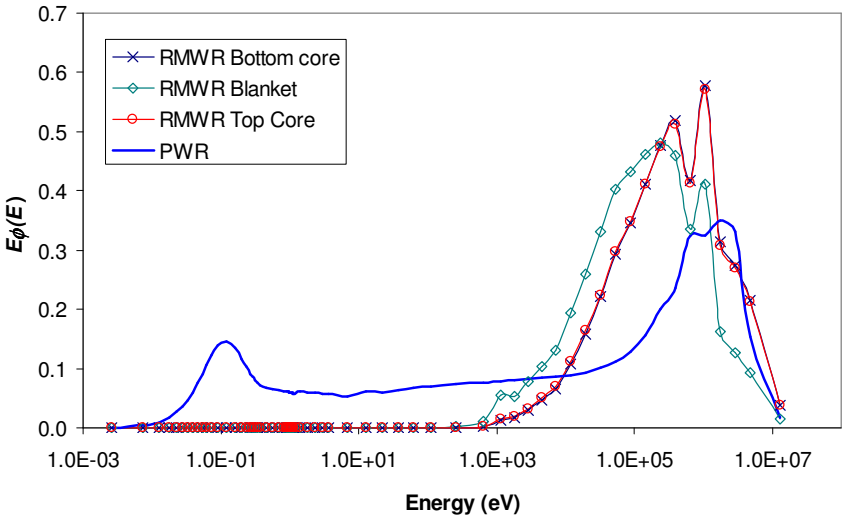

Figure 1. Spectrum Comparison between RMWR and a typical PWR

Numerical benchmark problems defined for the RMWR fuel pin have also been evaluated using traditional LWR lattice codes (WIMS8, DRAGON) and a fast reactor lattice code, MC^2-2[10]. The reference solutions for comparison were obtained with the continuous-energy Monte Carlo code, MCNP4C. The eigenvalue calculated by MC^2-2 is closer to the result of

the MCNP4C calculation than those of other codes, mainly due to the fast spectrum of the RMWR core. Similar trends were also found in the results for the unit assembly benchmark.

Based on these findings, MC^2-2 was selected as the lattice code for the RMWR fuel cycle analysis. The code is used to generate 230-group cross sections that are then additionally collapsed with region-dependent spectrum into 33-group cross sections utilized in the flux calculations for the fuel cycle study. The REBUS-3 code is used for the detailed equilibrium cycle calculation. A three-batch fuel management scheme was modeled with the code, assuming an out-in in-core fuel movement and a cycle length of 24 months.

The ORIGEN2 code was used for evaluating the radioactivity properties of the RMWR fuel at the charge and discharge stages and the spent fuel to be sent to the repository. In these calculations, the masses of heavy-metal isotopes and their cycle-averaged one-group cross sections obtained from the REBUS-3 calculation are used as input to the ORIGEN2 calculations.

The MCNP4C code was used for calculating the neutron and gamma dose rates. These calculations were done for fuel pellet and fuel pin. For these calculations, the neutron and gamma sources of the equilibrium core were obtained from the ORIGEN2 calculations. With the given masses and source distributions, the MCNP4C code calculates the gamma or neutron flux at the spatial zone boundary of interest. The doses are obtained by multiplying these fluxes with the dose conversion factors specified in the MCNP4C users' manual.[11] The MCNP4C models used for dose calculations take into account the spectral differences of the spontaneous fission neutrons and neutrons from (α,n) interactions. They also utilize the gamma source spectra obtained from ORIGEN2 calculations. For the MCNP4C calculations, the gammas or neutrons are distributed uniformly over the volume of the fuel pellet or pin. A vacuum boundary condition is used for all the cases. In the fuel pin case, the pin is assumed clad with Zircalloy, and the full length (160 cm height) of the pin is modeled.

4. RMWR Equilibrium Core with Pu Multirecycling

4.1. Mass flow

The mass flow of the RMWR equilibrium cycle core has been evaluated. The equilibrium core was obtained using REBUS-3 RZ calculations with a three-batch, out-in in-core fuel management scheme. The cycle length of the equilibrium core was assumed 24 months with 90% capacity factor. The initial isotopic vector of the fresh MOX fuel was obtained from a LWR spent nuclear fuel with a burnup of 45,000 MWd/t (3.8% enriched UO_2 and 7-year cooling).

Table 2 is a summary of four Pu recycle stages in the RMWR core, compared with the results of Pu multirecycling in the CORAIL concept (simply, CORAIL-Pu). Note that the CORAIL concept denotes a full-core loading of the so-called CORAIL assembly, which employs a standard 17x17 PWR fuel assembly containing 180 UO_2 pins in the interior and 84 MOX pins in the peripheral region. [6]

The Pu content of the MOX fuel pin is quickly converged with multirecycling in the RMWR core mainly because the high conversion ratio maintains the fissile content of plutonium; the average fissile content converges to about 18% (this value is similar to that obtained by JAERI [5]). Also, the fast spectrum of the RMWR core mitigates the generation of minor actinides (MA) with multirecycling. However, in the CORAIL-Pu case, the generation rate of MA increases with Pu multirecycling. Note that the capture-to-fission cross section ratio of actinides is smaller in fast spectrum systems than in typical LWRs

In the current assessment, the core dimensions specified in the JAERI paper have been utilized; No attempt has been made to optimize the core dimensions to get identically zero Pu production rate. Since the conversion ratio is greater than 1.0, the total net TRU mass balances become positive in the RMWR core; about 50 kg of TRU is generated every year from a 990 MWt RMWR core. For comparison purposes, the net mass balances are normalized to one Giga-Watt day electricity production and compared to other thermal transmutation options in Table 3.

Table 2. Summary of Masses and Isotopic Vectors of RMWR Multirecycling Scheme

		RMWR				CORAIL-Pu	
Recycling Cycle		1st	2nd	3rd	4th	1st cycle	7th cycle
U enrichment in MOX fuel, %		0.20	0.20	0.20	0.20	4.15	4.57
Pu content in MOX fuel, %		30.6	31.0	31.0	30.8	6.50	8.18
Fissile content in MOX fuel, %		17.54	17.79	17.89	17.96	4.12	3.84
Breeding ratios (fuel / blanket)		1.03	1.05	1.06	1.06 (0.42 / 0.64)	N/A	N/A
Isotopic vector	Pu-238	2.68	2.34	1.91	1.56	2.7	3.9
	Pu-239	47.75	51.92	53.70	54.70	56.0	36.1
	Pu-240	30.33	31.28	31.96	32.43	25.9	27.0
	Pu-241	9.65	5.45	4.09	3.67	7.4	10.8
	Pu-242	8.58	8.45	7.93	7.26	7.3	21.1
	Am-241	1.01	0.55	0.41	0.37	0.7	1.1
Net mass balance of core, kg/year	Heavy metal	-330	-330	-330	-330	-1287	-1287
	U	-375	-376	-378	-379	-1379	-1390
	Pu	15	22	27	29	51	41
	MA	30	24	22	20	41	62

Table 3. Comparison of Fuel Cycle Performance Data

		RMWR	CORAIL		ALWR [c]
			Pu [a]	TRU [b]	
Thermal power, MWt		990	3800	3800	3000
Cycle length, month		24	15	15	18
Capacity factor		0.9	0.9	0.9	0.9
Thermal efficiency		0.333	0.333	0.333	0.333
Total HM mass in core, ton		72.411	103.641	103.641	86.85
Number of assembly		283	193	193	193
Electricity production per one metric ton initial heavy metal, GWe-d/MTIHM		2.99	5.01	5.01	5.67
Cycle length, GWd/t		8.97	15	15	16.67
Discharge burnup (core/blanket in RMWR), GWd/t		58 / 11	45	45	50
Net mass balance in kg/year	HM total	-330	-1287	-1287	-1072
	U	-379	-1390	-1359	-1351
	Pu	29	41	77	257
	MA	20	62	-5	21
	TRU	49	103	72	279
Normalized net mass balance in kg/GWe-d	HM total	-3.04	-3.10	-3.10	-3.27
	U	-3.50	-3.34	-3.27	-4.12
	Pu	0.26	0.10	0.19	0.78
	MA	0.19	0.15	-0.01	0.07
	TRU	0.45	0.25	0.17	0.85

a) Pu multirecycling in CORAIL concept. Net mass balance per assembly is HM/U/Pu/MA/TRU = -25/ -27 /0.8/1.2/2 kg

b) TRU multirecycling in CORAIL concept. Net mass balance per assembly is HM/U/Pu/MA/TRU = -25/ -26.4/1.5/-0.1/1.4 kg

c) ALWR stands for Advanced Light Water Reactor. Net mass balance per assembly is HM/U/Pu/MA /TRU = -25/-31.5/6.0/0.5/6.5 kg

In this Table, the reactor capacity factor and thermal efficiency are fixed for all cases and the net mass balances of the other thermal transmutation options reported in Refs. 7 through 9 were used.

The TRU generation rate of the RMWR core is smaller than those of other thermal transmutation options; about half of the CORAIL-Pu option and one sixth of the ALWR option. The normalized net TRU mass balance (0.45kg/GWe-d) is, however, larger than those of the CORAIL options (0.25 or 0.17 kg/GWe-d for CORAIL-Pu and –TRU, respectively) due to the smaller thermal power of the RMWR core. On the basis of energy produced, the MA content is about 3 times higher for the RMWR fuel compared to the ALWR discharge fuel; this high MA content in the RMWR is due to the high Pu-content.

4.2. Void Reactivity Coefficient

The void reactivity coefficient is a primary concern in the design of the RMWR core due to its fast spectrum as well as moderating coolant. Table 4 provides a comparison of the void coefficients for the RMWR startup core at zero burnup and discharge burnup. In spite of the different core geometries, number of neutron groups and equation solvers, the void reactivity coefficients are comparable to each other at the beginning and end of cycle. The void reactivity coefficient is negative at zero burnup and the absolute value is comparable to that reported by JAERI (see Ref. 5).

The value of this reactivity coefficient is however positive at the end of cycle because higher-actinides build up in the blanket zones reduces core leakage and the buildup of fission products hardens the spectrum. The void coefficients of the equilibrium cycle core were also calculated. They were found to be 698 pcm/100%-void and 1095 pcm/100%-void at beginning of cycle and end of cycle, respectively. Note that a value of -500 pcm/100%-void is reported in JAERI paper for a small-scale RMWR core but that the corresponding burnup point was unspecified. Additionally, an axial distribution for the void fraction was assumed in this study.

Table 4. Comparison of Void Reactivity Coefficients of RMWR Startup Core

Burnup		0 EFPD		1314 EFPD	
Solver		TWODANT	DIF3D	TWODANT	DIF3D
neutron group		230	33	230	33
Geometry		R-Z	HEX-Z	R-Z	HEX-Z
k_∞	Normal	1.10498	1.08348	1.02215	1.01168
	100% void	1.09858	1.07869	1.03496	1.02628
Void coefficient (pcm/100% void)		-527	-410	1211	1406

5. Radiotoxicity and Fuel Handling Issues of RMWR Fuel Cycle

5.1. Radiotoxicity

A variety of measures are available to quantify the *radiotoxicity* of spent nuclear fuel, e.g., dose equivalent derived from the ICRP database and water dilution volume from *10CFR20*, or cancer dose measure. [12-14] For consistency with the previous studies, [7-9] the cancer dose measure was selected in this work.

The radiotoxicity of the waste sent to a geologic repository from the Pu recycling in the RMWR core was estimated by evaluating the cancer dose up to 10 million years after disposal. The radiotoxicity values are normalized to the energy production of one Giga-Watt day electric (GWe-d). Note that in the RMWR concepts, the waste passed to the repository is

limited to all fission products and minor actinides; the Pu is recycled and the discharged uranium is used as make up feed.

The normalized radiotoxicity sent to the repository from the RMWR core is provided in Figure 2, with a comparison made to other thermal transmutation options. Note that 0.1% loss of Pu at the separation stage was assumed in the CORAIL Pu multirecycling case but ignored in the RWMR case. For all cases but the reference UO_2, the discharged material is from the equilibrium cycle of a multirecycling scheme (the reference UO_2 denotes a once- through cycle of a typical PWR). The CORAIL results were previously reported in Refs. 7 through 9.

The normalized radiotoxicity levels of the spent nuclear fuel sent to the repository from the RMWR core is initially comparable to the waste from the reference UO_2 or CORAIL-Pu options. After a few hundred years, however, the normalized radiotoxicity of the RMWR spent nuclear fuel is higher than those of the reference UO_2 or CORAIL-Pu options, mainly due to the high content of Am-241 ($T_{1/2}$= 432.2 year) in the spent nuclear fuel. Note that the high Pu content of the RMWR core increases the Am-241 content of the discharge fuel; the Am-241 mass at discharge stage is about a factor of 2.8 higher than that of the charge stage.

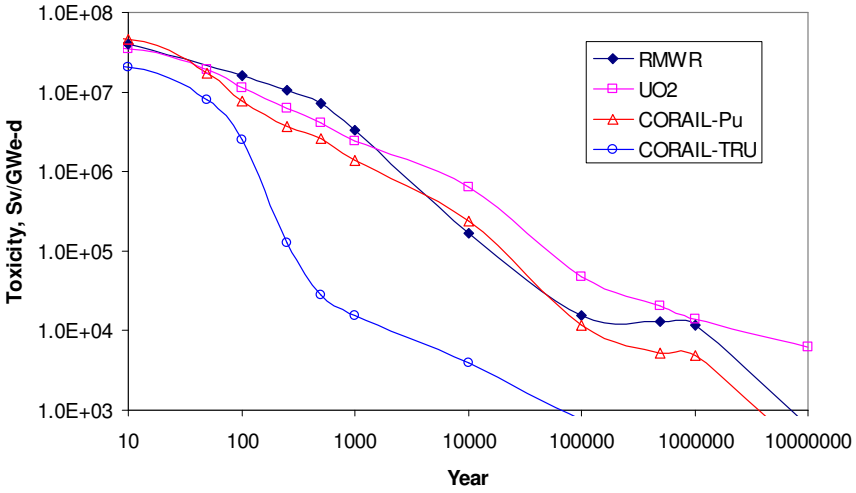

Figure 2. Comparison of Radiotoxicity in Terms of Cancer Dose

5.2. Fuel Handling Issues

To understand the impact of Pu recycling in the RMWR concept on fuel handling and processing, three key parameters were identified and evaluated: total decay heat, gamma energy emission rate, and neutron emission rate. The values of these parameters for the RMWR with Pu recycling are provided in Table 5 (similar data for the UO_2, MOX, CORAIL-Pu, and CORAIL-TRU fuel cycle options are also given). In these calculations, all property values were evaluated for one metric ton of charged heavy metal and then normalized to those of the MOX fuel at fabrication.

The radioactivity properties of the RMWR fuel are factors of 2 to 3 larger than the corresponding values for the CORAIL-Pu case at the fabrication and charge stages because of high Pu-content of the RMWR fuel. But they are quite similar to the values for the MOX case at the charge stage, mainly due to the similar Pu-content in the total heavy metal in both cases. This implies that the decay heat loads of the charged RMWR fuel are not necessarily problematic; Note that the fabrication and charge of MOX fuel are currently done in France and Japan.

At discharge, the total decay and gamma heat loads are expected to be proportional to the discharge burnup because they are dominated by fission products. Thus, all options except the RMWR case have similar values at discharge because of similar discharge burnup. The values for the RMWR fuel are however smaller than those for the other fuel cycles at discharge because of its lower average (core and blanket) discharge burnup (see Table 3).

Table 6 provides the dose rate of the MOX pellet and fuel pins, for the different concepts. Relative to the CORAIL-Pu case, the MA of the CORAIL-TRU option increases the dose rate at the pellet surface significantly; similarly, the higher content of the Pu in MOX pin of the RMWR case increases the dose rate. Note that the Pu and TRU contents are 8.2% and 12.2% in the CORAIL-Pu and CORAIL-TRU cases, respectively.

Table 5. Comparison of Normalized Radioactive Properties for Several Fuel Cycles

		UOX	MOX [a]	CORAIL-Pu	CORAIL-TRU	RMWR [b]
Decay heat (Watt)	Fabrication	0	1 (1983)[c]	1	19	2
	Charge	0	1	0.3	6	1
	Discharge	1038	1001	1010	1009	375
	After cooling	1	3	2	7	1
Neutrons [d] (x1.0E6/sec)	Fabrication	0	1(95)	1	80662	3
	Charge	0	1	0	15922	1
	Discharge	13	140	101	95740	33
	After cooling	1	68	67	26466	16
Gamma (Watt)	Fabrication	0	1(0.56)	1	42	3
	Charge	0	2	1	13	1
	Discharge	1012527	933562	978080	953389	356385
	After cooling	1904	1601	1840	1747	603

a) Pu content is 9.4%.
b) Radiation properties were evaluated with one metric ton heavy metal of MOX fuel at fabrication stage, but total fuel pin (MOX fuel, blanket and axial reflectors) for other stages.
c) Actual value per one metric ton initial heavy metal.
d) Neutron emission rate due to spontaneous fission and (α,n) reactions.

Table 6. Comparison of Dose Rate of Equilibrium Fuel at Charge Stage (rem/hr)

		CORAIL-Pu	CORAIL-TRU	RMWR (peak /average) [a]
Surface of MOX fuel pellet	SF [b]	0.014	1039	0.038
	(α,n)	0.016	0.34	0.039
	photon	4.893	110.6	8.761
	Total	4.923	1149.94	8.838
At cladding Surface of total fuel pin	SF	0.019	1436	0.153 / 0.055
	(α,n)	0.021	0.43	0.139 / 0.050
	photon	0.094	18.46	0.273 / 0.102
	Total	0.134	1454.89	0.565 / 0.207
One meter away from cladding surface of total fuel pin	SF	6.13E-05	4.61	1.36E-04 / 1.29E-04
	(α,n)	6.22E-05	1.29E-03	1.10E-04 / 1.10E-04
	photon	2.96E-04	0.06	3.29E-04 / 3.10E-04
	Total	0.0004	4.671	5.75E-04 / 5.49E-04

a) Value in MOX fuel zone and average value of total fuel pin (i.e., MOX fuel, blanket and axial reflectors).
b) Spontaneous fission neutron
Note : 1. Flux-to-dose conversion factors for neutrons and photons are from ANSI/ANS-6.1.1-1977.
2. Spectrum for the spontaneous fission neutrons is from MCNP4C and that for photons comes from ORIGEN2 calculations.
3. Spectrum for (α,n) neutrons is based on generic data for MOX fuel (see Ref. 9)

The radiation dose rate from the MOX fuels can be reduced significantly by the use of Zr clad. This is because relatively low energy gammas (photons) are dominant contributor to the dose rate and are shielded effectively by Zr clad. The results also confirm the greatly reduced values at a distance of one meter away from the fuel pin.

6. Fuel Cycle Mass Flows

Several fuel cycle options were analyzed with a simple lumped model. Table 7 contains input data for the fuel cycle analyses; for simplicity, the same fuel cycle option was assumed for each park and the thermal power was fixed as 300 GWt, which is similar to the current nuclear capacity in the U.S. Additionally, 90% capacity factor and 50 GWd/MTIHM discharge burnup were assumed for the UO_2 case.

Table 7. Input Data for Fuel Cycle Options

Fuel Cycle	ALWR	CORAIL-Pu	RMWR
Capacity (GWt) [a]	300	300	300
Discharge burnup (GWd/t)	50	45	27
Capacity factor (%)	90	90	90
U enrochement (UOX/MOX), %			
UOX fuel	4.2	4.57	0.2
MOX fuel	N/A	0.2	0.2
MOX fuel			
Volume fraction in core, %	N/A	31.8	33.8
Pu content in MOX, %	N/A	8.18	30.77
Discharge vector, %			
Uranium	93.4	92.41	86.4
Pu	1.27	2.73	10.61
MA	0.15	0.25	0.21
Fission products	5.18	4.61	2.78

a) The capacity 300GWt is similar to the current nuclear capacity in the U.S.

The enrichments of the natural and depleted uranium materials were assumed to be 0.71 % and 0.2%, respectively. Losses at the conversion, fabrication and reprocessing stages were ignored in this work. However, 0.1% loss was assumed at the reprocessing stage in the CORAIL-Pu fuel cycle. Figures 3 through 5 show the simplified representation of the fuel cycles for 300 GWt ALWR, CORAIL-Pu and RMWR nuclear parks. All mass flows were evaluated based on the data in Table 8.

In the once-through ALWR park (see Figure 3), the gross amount of the spent nuclear fuel is the same as the heavy metal mass loaded at the charge stage; the required capacity of the repository or interim storage for the spent nuclear fuel is 1,971 metric ton per year. By recycling the Pu in the CORAIL or RMWR fuel cycles, however, the waste passed to the repository or interim storage could be limited to less than 110 metric ton per year.

For Pu multirecycling in the RMWR core, it is important to note that natural uranium feed is unnecessary because both the Pu and uranium are recycled (depleted uranium feed is used). However, 15,459 and 12,795 metric ton per year of natural uranium are required to supply the enrichment needs for the once-through ALWR and the CORAIL-Pu multirecycling cases, respectively. Note that effective utilization of uranium resources is one of the primary goals of the RMWR core. However, the tradeoff for this benefit is an increase in the amount of heavy metal at the separation and reprocessing stages. The amount of the heavy metal mass to be reprocessed in the RMWR fuel cycle (3541 metric ton per year) is a factor of 5 larger than that of the CORAIL fuel cycle (689 metric ton per year); definitely, this negatively impacts the fuel cycle cost. A large fraction of the heavy metal mass (65%) is from the processing of

the blanket in the RMWR concept; thus, significant reduction could readily be achieved if an external TRU feed is available (conversion ratio <1).

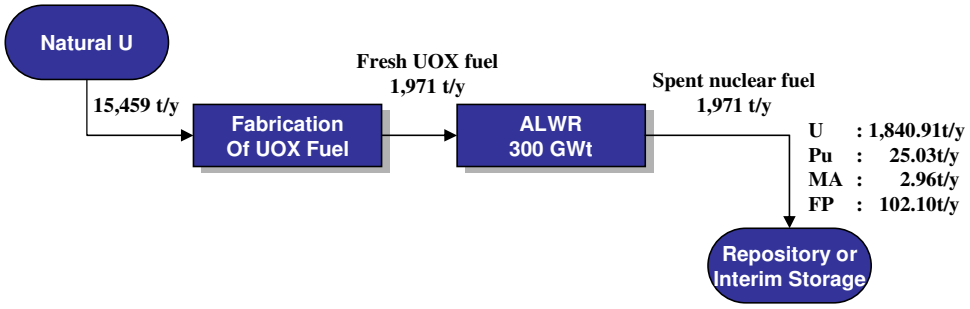

Figure 3. Once-through Fuel Cycle of ALWR

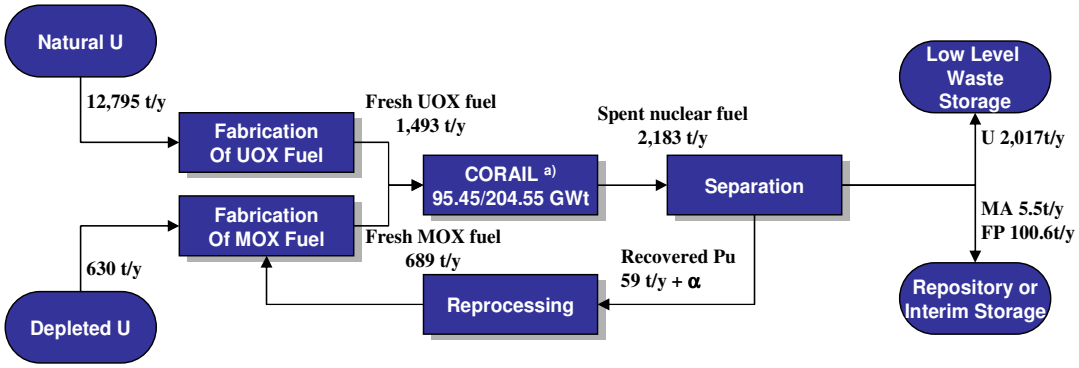

Figure 4. Pu multirecycling in CORAIL concept

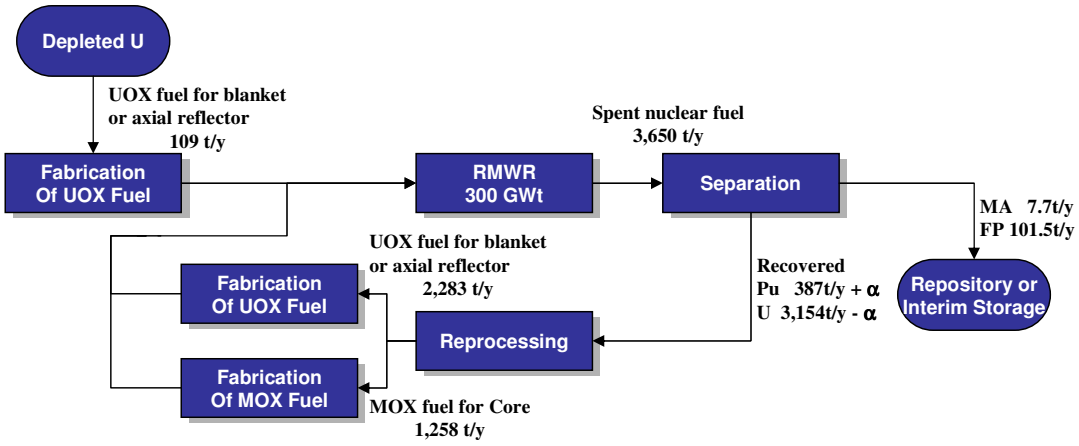

Figure 5. Pu Multirecycling in RMWR Concept

7. Conclusions

The fuel cycle performance of the RMWR concept was evaluated in terms of normalized net mass balance, reactivity coefficients, radiotoxicity, decay heat, required natural uranium and waste mass passed to the repository or interim storage.

It is found that the RMWR fuel cycle has excellent features for improving the utilization of uranium resources and reducing the radioactive waste passed to the repository or interim storage by recycling both Pu and uranium with a high conversion ratio. The conversion ratios of the fuel and blanket zones are 0.5 and 3.1, respectively, and the core average value is slightly greater than 1.0. Thus, the required natural uranium is zero and the wastes passed to the repository are limited mainly to the fission products and minor actinides (and fractional losses of the Pu and uranium in the separation stage). Due to the high Pu content of the RMWR fuel, the decay heat or dose rate is about 2 ~ 3 times higher than the corresponding values for the CORAIL-Pu case at the fabrication and charge stages; but similar to the corresponding values for the MOX case at the charge stage. The radiotoxicity of the RMWR waste at the repository is smaller than that for the CORAIL-Pu waste 10 years after discharge, mainly due to the smaller fission products content arising from a lower discharge burnup. But, the radiotoxicity at few hundred years is higher than those for the other transmutation cases because of the higher amount of Am-241 produced in the RMWR case.

By recycling both the Pu and uranium, the RMWR concept has excellent features in terms of uranium utilization and waste reduction. However, the tradeoff for this benefit is an increase in the amount of heavy metal at the separation and reprocessing stages. The amount of the heavy metal mass to be reprocessed is a factor of 5 larger in the RMWR fuel cycle, compared to the CORAIL-Pu fuel cycle; this impacts negatively on the fuel cycle cost. A large fraction of the heavy metal mass (65%) is from the processing of the blanket in the RMWR concept. A primary difference between the CORAIL-Pu and RMWR concepts is that uranium enrichment is used to support the Pu multirecycling in the former case. This is quite important for the overall uranium utilization since no new uranium mining/enriching is required for the RMWR.

A potential safety-related issue is that the RMWR core could have a positive void reactivity coefficient. It is negative at the beginning of the startup core but positive at the end of cycle and throughout the equilibrium cycle. Even though a negative void reactivity coefficient was previously reported (Ref. 5), the void coefficients of the equilibrium RMWR core were found to be positive in this work (this difference could be due to the assumed axial void fraction distribution). Another issue is the axial power profile that can be skewed asymmetrically to the bottom core. This design could result in power oscillations. Additional efforts are therefore required to investigate these safety-related issues.

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