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Analysis of Nuclear Proliferation Resistance of DUPIC Fuel Cycle

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This study compares the proliferation resistance of DUPIC (Direct Use of Spent PWR Fuel in CANDU) fuel cycle with other fuel cycle cases. The other fuel cycles considered in this study are PWR of once-through mode (PWR-OT), PWR of reprocessing mode (PWR-MOX), in which spent PWR fuel is reprocessed and recovered plutonium is used for making MOX (Mixed Oxide), CANDU with once-through mode (CANDU-OT), PWR fuel and CANDU fuel in a once-through mode with reactor grid equivalent to DUPIC fuel cycle (PWR-CANDU-OT). This study is focused on intrinsic barriers, especially, radiation field of the diverted material, which could be a significant accessibility barrier, amount of special nuclear material based on 1 GWe-yr that has to be diverted and the quality of the separated fissile material. It is indicated from plutonium analysis of each fuel cycle that the MOX spent fuel is containing the largest plutonium per MTHM but PWR-MOX option based on 1 GWe-yr has the best benefit in total plutonium consumption aspects. The DUPIC option is containing a little higher total plutonium based on 1 GWe-yr than the PWR-MOX case, but the DUPIC option has the lowest fissile plutonium content which could be another measure for proliferation resistance. On the whole, the CANDU-OT option has the largest fissile plutonium as well as total plutonium per GWe-yr, which means negative points in nuclear proliferation resistance aspects. It is indicated from the radiation field analysis that fresh DUPIC fuel could play an important radiation barrier role, more than even CANDU spent fuels. In conclusion, due to those inherent features, the DUPIC fuel cycle could include technical characteristics that comply naturally with the Spent Fuel Standard, at all steps along the DUPIC linkage between PWR and CANDU.

KEYWORDS: DUPIC (direct use of spent PWR fuel in CANDU), (DUPIC) fuel cycle, nuclear fuel cycle analysis, nuclear proliferaion resistance, proliferation resistance barrier, safeguards, plutonium analysis, candu type reactors, spent fuels, fuel cycles

I. Introduction

The question of "how to manage the spent fuel discharged from reactors" has been a key factor to be considered, as part of the sustainable supply of nuclear energy in the future. The proliferation risk could be one of the important criteria for determining a specific policy of the spent fuel management. In this respect, the proliferation risk in backend fuel cycle has long been a topic of polemics as was culminated by the International Nuclear Fuel Cycle Evaluation (INFCE)[1] under the auspice of IAEA and Nonproliferation Alternative System Assessment Program (NASAP)[2] under the auspice of the U.S. Government. Recently, consideration of proliferation resistance or vulnerability has been a topic of renewed interest in the content of the Fissile Materials Disposition Program for disposing of surplus U.S. weapon plutonium.[3,4]

The INFCE and NASAP were addressed to systematic identification of diversion problems and to recommendation of possible measures to mitigate them, suggesting some technical means to enhance deterrence to access to special nuclear materials in the conventional recycle stream, such as: partial decontamination, spiking with fission products, and denaturing. Looking back to the INFCE era, however, there has not been significant enhancement for proliferation resistance in backend fuel cycle technologies. In addition, there was no consensus reached on non-proliferation standards and there has been essentially no follow up on those proposals.

A new fuel cycle concept, so-called DUPIC (Direct Use of Spent PWR Fuel in CANDU) has been developed since the early nineties.[5–8] The basic idea of this fuel cycle alternative is that the spent fuel from LWR contains enough fissile remnant to be burnt again in CANDUs (Canadian Deuterium Uranium, Reactors) by direct refabrication, without separating the residual fissile materials, before eventual disposal of the spent fuel. The DUPIC fuel cycle may offer a good alternative to other conventional options, considering the high proliferation resistance. In this regard, this paper examines quantitatively some nuclear proliferation barriers of the DUPIC fuel cycle, and compared with other fuel cycle options.

There are several proliferation barriers that make it difficult to recover weapon-usable plutonium from material that has been successfully removed from within a fuel cycle system. The proliferation resistance barrier generally consists of an intrinsic barrier such as radiation field and institutional barrier such as material accounting and physical protection. Many studies[9–14] have focused on the intrinsic barriers of the proliferation resistance. Generally, the intrinsic barriers include time required to recover a significant quantity of weapon-usable plutonium, minimum cost of weapons' construction from source material available, radiation field of the diverted material and amount of material that has to be diverted, material form, the quality of the separated fissile material *etc*. This study is focused on intrinsic barriers, especially, radiation field of the diverted material, the quality of the separated fissile material and amount of material that has to be diverted.

A radiation field could be apparently a significant accessibility barrier if the field is high enough to force a theft to shield the object during a theft. The shielding material, being heavy and cumbersome, and/or remote handling would force the theft to use lifting equipment during the thief and to haul

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away a significantly lager mass than just the stolen object.

However, the self-protecting field of penetrating radiation emitted by spent fuel decreases steadily with time, and effectively disappears after several hundred years of cooling. After this period, the spent fuel could be processed in a contact-handled glove box facility, rather than in a shielded, remotely-operated facility. It has been suggested that this would provide a strong incentive for a proliferant to mine long-cooled spent fuel from a repository.[15,16] They have insisted that the repository is just a plutonium mine and it is, therefore, more important to achieve the near-total "destruction" of plutonium.[17] In this regard, the quality of the separated fissile material and total amount of plutonium, based on 1 GWe-yr, which could be an another measure for proliferation resistance, were estimated in this study.

The DUPIC fuel cycle concept and other fuel cycle models to estimate the proliferation barriers are given in Chap. II. The plutonium burnout and fissile plutonium content of fuel cycles are described in Chap. III. The radiation field analysis is described in Chap. IV.

II. Reference Fuel Cycle Models

1. Fuel Cycle Model

In general, the conventional backend fuel cycle has evolved into two different directions, depending upon national policies: either direct disposal in deep geological formations, or the reprocessing of spent LWR fuel for MOX (Mixed Oxide) fuel recycle in LWR (or FBR). The predominant nuclear reactor types in the current world market are LWR and HWR, and many experts are forecasting that this trend will likely continue for the foreseeable future.[18] In the light of this forecasting, various fuel cycle options, which could be considered in U–Pu cycle especially with LWRs and/or HWRs, are described in **Fig. 1**.

The first cycle considered in this study is low-enriched uranium in PWR of once-through mode (hereafter called "PWR-OT"). The second cycle is mixed oxide fuel in PWR of reprocessing mode (hereafter called "PWR-MOX"), in which spent PWR fuel is reprocessed and recovered plutonium is used for making MOX fuel (5% of plutonium content) and recovered uranium is inputted into a conversion plant. The MOX spent fuel will be disposed of without further plutonium or uranium recovery. Some depleted uranium generated in the enrichment plant will be used for making MOX fuel. The third cycle is natural uranium in CANDU with once-through mode (hereafter called "CANDU-OT"). The forth cycle is the DUPIC fuel cycle in which PWRs are linked to a CANDU (hereafter called "DUPIC"). The fifth cycle is PWR fuel and CANDU fuel in once-through mode with reactor grid equivalent to DUPIC fuel cycle (hereafter called "PWR-CANDU-OT").

In the DUPIC fuel cycle, spent PWR fuel is directly refabricated into CANDU fuel to be burnt again in CANDUs before being disposed of permanently. On the other hand, the once-through fuel cycle (PWR-CANDU-OT) is to dispose of all spent fuel generated from both PWR and CANDUs. As shown in Fig. 1, the front-end fuel cycle components for a PWR were established to be the same for both fuel cycles. For the DUPIC fuel cycle, however, several services such as DUPIC fuel fabrication included but the front-end fuel cycle components for CANDU is not needed.

The DUPIC fabrication process involve the direct refabrication of PWR spent fuel in the CANDU fuel. The spent fuel materials is recovered from the PWR spent fuel by disassembling and decladding using only thermal and mechanical processes. The powder preparation process called OREOX (Oxidation REduction of OXide fuel) is considered the most critical process for producing resinterable powder feedstock. Once the resinterable powder is prepared, the pellet and rod manufacturing processes are almost same as the conventional powder/pellet route in fuel fabrication. All these works are performed in a hot cell with remote technologies because fuel materials have still high radioactivity generated especially by fission products.

The fuel materials would flow along with the bulk stream through the powder preparation and scrap recovery, except for a small amount of irrecoverable discards. The waste stream from the DUPIC fuel fabrication processes would mainly consist of the metallic components from spent LWR fuel, and the gases and semi-volatile fission products released from the bulk fuel material treatment, in addition to the measurable discards and losses. There is no liquid waste arising from the DUPIC fuel fabrication processes which depend entirely on dry method, in contrast to wet processes from which liquid waste as effluent arise.

2. Reference Reactors and Fuels

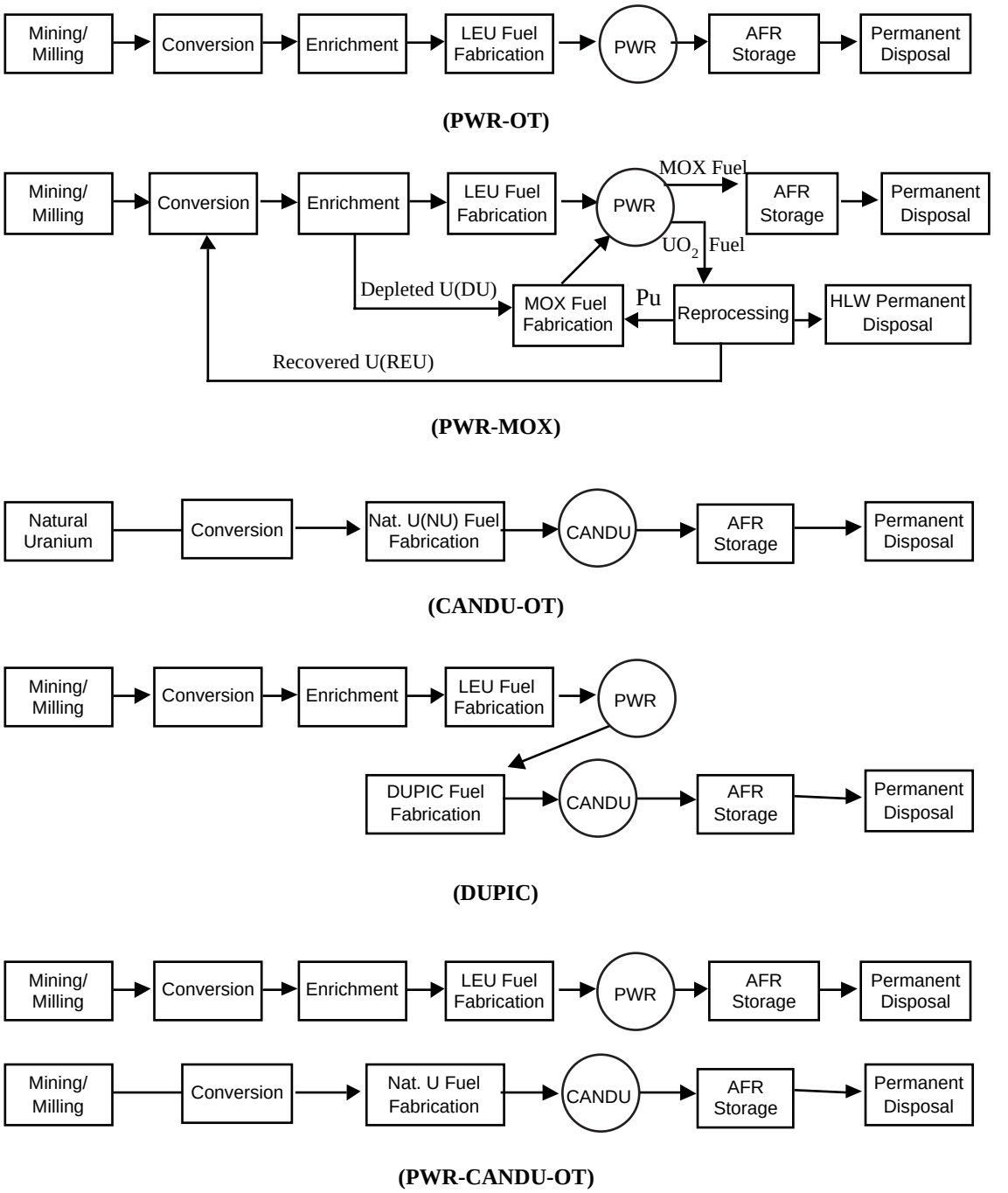
For material flow of each fuel cycle, reference PWR and CANDU have to be chosen first, and their fuel characteristics (*e.g.*, initial enrichment and discharge burnup) need to be defined reasonably. For a practical analysis, a 950 MWe PWR and a 713 MWe CANDU, which are now operating in Korea, were taken as reference reactor systems. The characteristic parameters of the reference reactor systems are summarized in **Table 1**, which will be used as input data for determining the fuel material balance. In the table, the amount of fuel loaded per reactor is estimated based on the reactor parameters such as

$$\text{Fuel loading per core} = \frac{P \times 100}{\varepsilon \times SH}, \tag{1}$$

where P, SH and ε are the electric power (MWe) of a CANDU, the specific heat (MWt/MTHM) and efficiency (%),

Table 1 Characteristics of reference reactors

Reactor parameters	PWR	CANDU
Electric power (MWe)	950	713
Thermal efficiency (%)	34	33
Thermal power (MWt)	2,794	2,161
Specific power (MWt/t U)	40.2	25.5
Load factor	0.8	0.9
Cycle length (Full Power Day)	290	—
Number of fuel assemblies or bundles per core	157	4,560
Number of batches for PWR	3	—
Loading per core (MTU)	69.5	84.7



LEU: Low Enriched Uranium, AFR: Away From Reactor, DU: Depleted Uranium, HLW: High Level Waste

Fig. 1 Fuel cycle options and their steps

respectively.

Table 2 shows the reference fuels of each fuel cycle. It is assumed that LEU (Low Enriched Uranium) PWR fuels and MOX fuels are burnt up to 35,000 MWd/MTU although recent PWR fuels have been mostly over 40,000 MWd/MTU fuel. The reason is that 35,000 MWd/MTU with initial enrichment of 3.5% ^{235}U was chosen as a reference PWR fuel in DUPIC fuel cycle development in Korea.[6,8)]

In PWR-MOX fuel cycle, the plutonium recovered from reprocessing of LEU PWR fuel is made into MOX fuel, which is burned in PWR, and then discharged MOX spent fuel is disposed of. In order to calculate how much plutonium is in PWR spent fuel burnt with 35,000 MWd/MTU, we have used ORIGEN 2 computer code.[19)] It found that content 0.82 wt% of ^{235}U and 0.89 wt% of Pu were still included in the spent fuels. If the MOX fuel is made from depleted uranium and 5% plutonium content, an equilibrium state could be reached when the MOX burning reactor uses a core which is 14.7% of the fuel in MOX and 85.3% of the fuel in LEU. It means that all reprocessed plutonium from LEU PWR spent fuel with 35,000 MWd/MTU can be used in the PWR core. In this situation, PWR core with MOX fuel consists of 10.22 MTHM

Table 2 Characteristics of reference reactors and fuels

Item	Characteristic parameters			
	PWR with LEU fuel	PWR with LEU and MOX fuel[a)]	CANDU with NU fuel	CANDU with DUPIC fuel
Reactor				
-Loading per core (MTU)	69.5	69.5 (10.22 MOX) (59.28 LEU)	84.7	84.7
-Annual fuel requirement (MTU)	23.31	23.31 (3.43 MOX) (19.88 LEU)	94.63	46.09
Fuel				
-Initial enrichment	3.5%	5% Pu_f MOX 3.5% LEU	Nat. U	PWR S/F
-Number of fuel rods per assembly	264	264	37	43
-Discharge burnup (MWd/kgHM)	35	35	7.5	15.4
Normalization of fuel				
-Required fuel amount for 1 GWe-yr (MTU or MTHM)	24.54	24.54 (3.61 MOX) (20.93 LEU)	132.73	64.64

[a)] 14.7% of the fuel in MOX and 85.3% of the fuel in LEU —an equilibrium state is reached when all spent PWR fuels are reprocessed to make needed MOX fuels.

MOX fuel and 59.28 MTU LEU fuel per reactor core.

In a CANDU, the discharge burnup of natural CANDU fuel is assumed to be 7,500 MWd/MTHM, and the discharge burnup of DUPIC fuel is assumed to be 15,400 MWd/MTHM which is a reference fuel in DUPIC fuel development.[8)]

The annual requirement of nuclear fuels is calculated based on fuel burnup and other parameters such as

$$\text{Annual requirement} = \frac{P \times 365 \times C}{\varepsilon \times BU}, \tag{2}$$

where C and BU are the capacity factor (%) and burnup (MWd/MTHM), respectively. The annual requirements per unit are translated into annual requirement based on 1 GWe-yr as shown in the last row of Table 2.

3. Material Flow for Fuel Cycle Options

For PWR-CANDU-OT and DUPIC fuel cycle, the equilibrium core ratio between PWRs and CANDUs have to be known so that all PWR spent fuels can make DUPIC fuels. It is possible to calculate with the annual requirement of PWR and CANDU with DUPIC fuel. The equilibrium core ratio between PWRs and a CANDU can be calculated as follows;

$$\text{Equilibrium core ratio } (R_C) = \frac{M_{\text{DUPIC}} \times (1 + L_{\text{DUPIC}})}{M_{\text{PWR}}}, \tag{3}$$

where M_{DUPIC}, M_{PWR}, and L_{DUPIC} are the annual requirement of DUPIC, the annual requirement of PWR and the loss rate in DUPIC fabrication plant, respectively. In this study, loss rate in DUPIC fabrication plant is assumed to be 1%. Since M_{DUPIC}, and M_{PWR} are 46.09MTHM and 23.31MTU, respectively, the equilibrium core ratio is 1.997.

In the mean while, portion of electricity generation between PWR and CANDU for 1 GWe-yr can be calculated as follows;

$$\text{Electricity generation portion of PWR} = \frac{P_{\text{PWR}} \times R_C}{P_{PWR} \times R_C + P_{\text{CANDU}}}, \tag{4}$$

where P_{PWR} and P_{CANDU} are the electricity powers of PWR and CANDU, respectively. So the portions of PWR and CANDU generation will be 72.68% and 27.32%, respectively. The portions of electricity generation will be applied to both PWR-CANDU-OT and DUPIC fuel cycle.

In this study, it is assumed that the loss factors are 0.5% for conversion and for CANDU fuel fabrication, 1% for PWR, DUPIC and MOX fuel fabrication and for reprocessing plant.

The results of the material balance analyses, which were calculated by Eqs. (1) through (4) with reference reactors parameters (shown in Table 1) and their fuel characteristics (shown in Table 2), are shown in **Table 3**. All values were expressed on basis of 1 GWe-yr for all fuel cycle options. It was found that the amount of fuel in the DUPIC fuel cycle needs only ~18 MTHM/GWe-yr while once-through fuel cycle (PWR-CANDU-OT) needs ~54 MTHM/GWe-yr. In the meanwhile, it is indicated that PWR-OT and CANDU-OT need 24.54 MTHM and 132,74 MTHM for 1 GWe-year, respectively.

III. Plutonium Burnout Analysis

As a commercial nuclear fuel burns, plutonium is produced by neutron captures in ^{238}U. The plutonium content of the fuel increases with burnup. The plutonium burnout or dilution may perhaps be an important one in the long-term perspective of non-proliferation in that some regard the spent fuel repository as a potential "plutonium mine." The low concen-

Table 3 Required fuels for five fuel cycle options

(Based on 1 GWe-yr)

Fuel types	Nuclear fuel cycles (MTHM)				
	PWR-OT	PWR-MOX	CANDU-OT	PWR-CANDU-OT	DUPIC
PWR	24.54	20.93		17.84	—
CANDU	—	—	132.73	36.26	—
MOX	—	3.61	—	—	—
DUPIC	—	—	—	—	17.66

tration of plutonium in spent fuel simply makes the mass of fuel material that must be removed to obtain 8 kg of plutonium (Significant Quantity) during a theft large. In addition, the high concentration of fissile plutonium in spent fuel makes the manufacture of nuclear weapon difficult because of large heat and spontaneous fission neutron generation.

In this chapter, we will evaluate actinide isotope composition including plutonium contained in spent fuels generated in the five alternative fuel cycles. The compositions are assessed on the basis of metric ton heavy metal and then those values are translated into 1 GWe-yr basis.

In order to calculate how much plutonium is in spent fuels, we have used a burnup simulation code, ORIGEN 2 computer code.[19] The ORIGEN 2 code users must supply the input characteristics to the program. Our MOX fuel is made from depleted uranium and we use a 5% plutonium content as described in previous chapter.

For reactor simulation of the DUPIC fuel in CANDU, isotope contents of PWR spent fuel, which are also calculated by the ORIGEN code, has been used. All actinides and 140 fission products contained in PWR spent fuels were inputted in the code for the DUPIC fuel. It has excluded some fission products removed during DUPIC fuel fabrication process. It is assumed that volatile isotopes during oxidation and reduction process are removed and semi-volatile isotope such as cesium and ruthenium are removed during sintering process working at 1,700°C. The removed fission products referred to the KAERI (Korea Atomic Research Institute) report,[20] are given in **Table 4**.

Table 5 and **Fig. 2** show the comparison of major actinides contained in four different spent fuels with 10 years of cooling time, which are obtained from the ORIGEN 2 code. The weight percent per heavy metal as well as mass per initial uranium or heavy metal are shown in the table. PWR spent fuels are still containing about 0.85 wt% ^{235}U and 0.92 wt% Pu, and about 68.4% of Pu is fissile isotopes (^{239}Pu and ^{241}Pu). On the other hand, MOX spent fuel is containing only 0.11% wt% ^{235}U but it is containing as much as 2.46 wt% Pu. Both DUPIC spent fuels and CANDU spent fuels are containing about only 0.22 wt% ^{235}U, but the DUPIC spent fuel is containing about two times more plutonium contents, 0.84 wt%, than the CANDU spent fuel case, 0.42 wt%.

From the plutonium weight percent of spent fuel, one can calculate the total mass for recovering one Significant Quantity (SQ: 8 kg for Pu). The resources required for the recovery of one SQ from spent fuel would be ~0.90 MTHM for PWR spent fuel, ~0.34 MTHM for MOX spent fuel, ~1 MTHM for DUPIC spent fuel and ~1.99 MTHM for CANDU spent fuel. Therefore, it is indicated that the amount for obtaining 1 SQ plutonium will be 2 assemblies for PWR, 1 assembly for MOX, ~52 bundles for DUPIC and ~105 bundles for CANDU spent fuel.

Table 4 Release rate during DUPIC fuel fabrication process

Isotopes	Release rate (%)	Isotopes	Release rate (%)
H	100	C	100
Kr	100	Ru	100
Cd	75	Te	75
Ir	75	I	100
Xe	100	Cs	100

In case of multiple recycling of the MOX and DUPIC spent fuels, it is interesting to see which one is more effective. As shown in Table 5, ^{235}U enrichment of DUPIC spent fuel is a little higher than that of MOX spent fuel, but ^{239}Pu content of DUPIC spent fuel (~0.33 wt%) is much lower than that of MOX spent fuel (~0.82 wt%). On the other hand, ^{236}U produced by (n, γ) reactions in ^{235}U is important because of its neutron absorption. If the uranium containing ^{236}U is recycled, a slightly greater fissile concentration in the fresh fuel to the reactor is required. DUPIC spent fuel is containing 0.22 wt% ^{236}U but the MOX spent fuel is containing only 0.11 wt% ^{236}U. Therefore, it is inferred that MOX spent fuel is more effective in fissile material utilization aspect than the DUPIC spent fuel for multiple recycling.

Figure 2 shows that the MOX spent fuels have much more minor actinide such as curium and americium than other cases, as expected. The miner actinide of DUPIC spent fuels is a little more than the PWR spent fuel case.

Using the material flow of Table 3, total plutonium embedded in spent fuels are calculated on the basis of 1 GWe-yr. The results are shown in **Fig. 3**. It is indicated that total plutonium generated during 1 GWe-yr is the biggest (~535 g-Pu/GWe-yr) in CANDU-OT option and the least (~88 g-Pu/1 GWe-yr) in PWR-MOX option. It means that the PWR-MOX option has some benefits in plutonium consumption aspects. In the meanwhile, the DUPIC option is containing ~141 g-Pu/GWe-yr which is a little higher than the PWR-MOX case, but the DUPIC option has the lowest fissile plutonium content which could be another measure for proliferation resistance. On the whole, the CANDU-OT option has the largest fissile plutonium as well as gross plutonium, which means negative points

Table 5 Major actinides content contained in various spent fuels

Isotopes	PWR		MOX		DUPIC		CANDU	
	g/MTU[a)]	wt%[b)]	g/MTU	wt%	g/MTHM	wt%	g/MTU	wt%
^{234}U	200.4	0.0208	92.8	0.0096	187.3	0.0198	44.5	0.0045
^{235}U	8,190.0	0.8497	1,074.0	0.1113	2,098.0	0.2212	2,194.0	0.2211
^{236}U	4,360.0	0.4523	235.4	0.0244	5,073.0	0.5348	741.4	0.0747
^{237}Np	522.9	0.0542	92.6	0.0096	542.5	0.0572	28.0	0.0028
^{238}Pu	153.7	0.0160	930.1	0.0964	387.9	0.0409	3.4	0.0003
^{239}Pu	5,329.0	0.5529	7,867.0	0.8156	3,157.0	0.3328	2,796.0	0.2818
^{240}Pu	2,203.0	0.2286	6,459.0	0.6696	2,792.0	0.2944	1,046.0	0.1054
^{241}Pu	751.5	0.0780	3,683.0	0.3818	523.5	0.0552	130.7	0.0132
^{242}Pu	456.8	0.0474	4,811.0	0.4988	1,099.0	0.1159	52.8	0.0053
^{241}Am	497.3	0.0516	2,834.0	0.2938	426.5	0.0450	82.7	0.0083
^{243}Am	91.3	0.0095	1,251.0	0.1297	192.2	0.0203	1.9	0.0002
^{244}Cm	18.5	0.0020	269.3	0.0279	54.9	0.0058	0.1	0.0000
^{245}Cm	1.0	0.0001	13.7	0.0014	0.9	0.0001	0.0	0.0000

[a)]MTU means the initial metric ton uranium used in ORIGEN code input.
[b)]wt% means the weight percent of heavy metal came from the ORIGEN output.

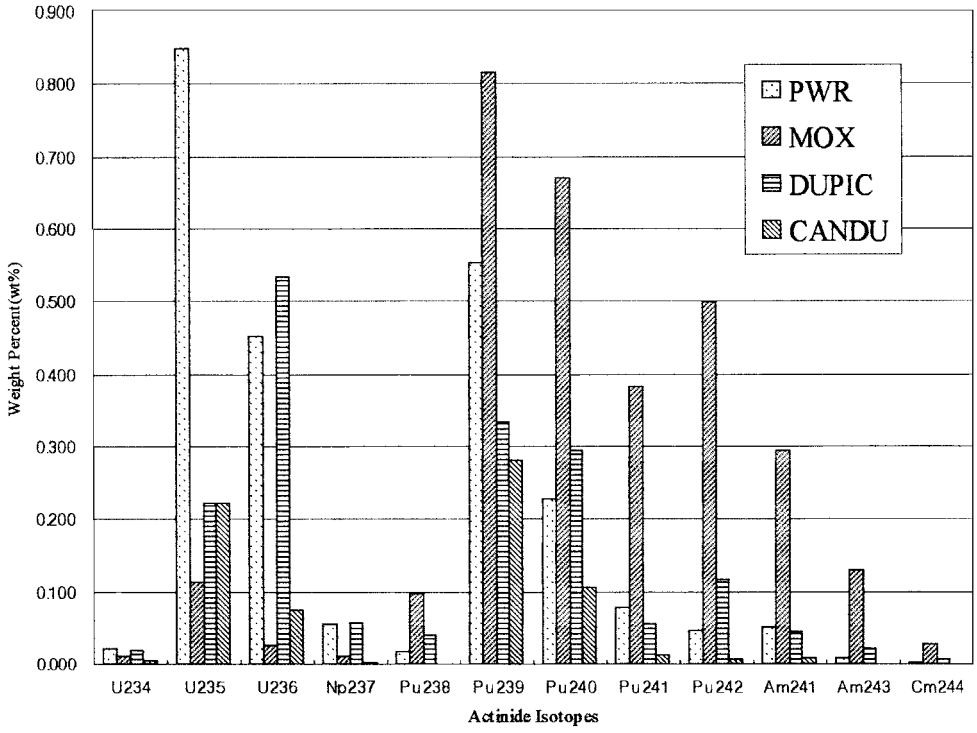


Fig. 2 Weight percent of major actinides embedded in various spent fuels

in nuclear proliferation resistance aspects.

IV. Radiation Barrier Analysis

The proliferation resistance of spent fuel is commonly attributed to its radiation field. We considered a radiation field to be a significant accessibility barrier if the field is high enough to force a thief to shield the object during a theft. The shielding material, being heavy and cumbersome, and/or remote handling would force the thief to use lifting equipment during the thief and to haul away a significantly lager mass than just the stolen object.

The magnitude of the radiation field near a spent fuel assembly depends on a number of factors, including design of the assembly, burnup of the fuel, and decay time since irradiation. Fresh DUPIC fuel still containing fission products and four spent fuel assemblies generated in five fuel cycle alternatives given in Chap. III were considered in this study. For our purpose, the decay time will be fixed at 10 years post irradiation.

It is important to note how much radiation field is enough to force a thief to shield the spent fuel during a theft. The effects of acute doses of radiation on human beings are de-

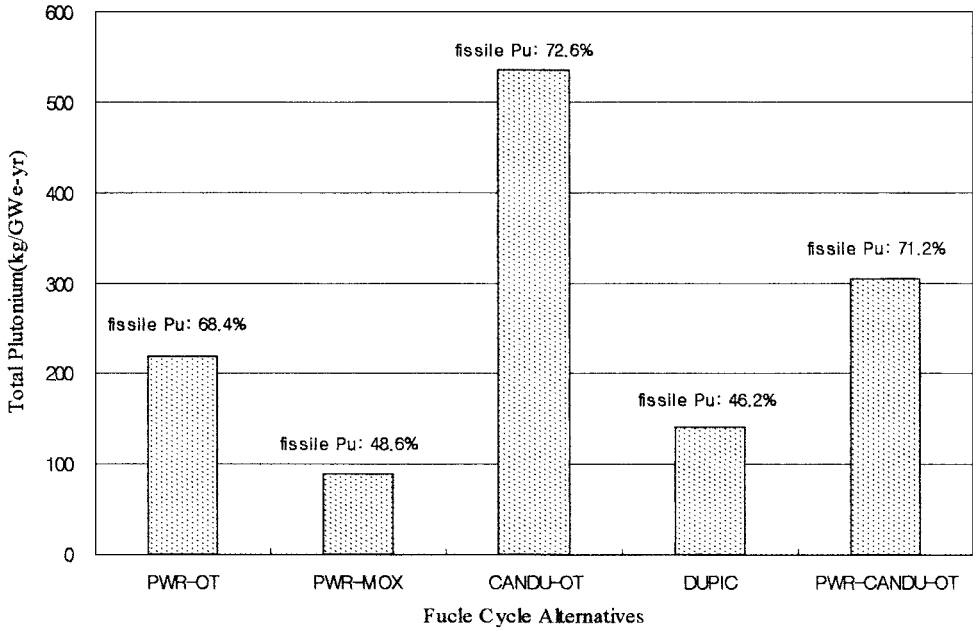


Fig. 3 Total plutonium of fuel cycle alternatives based on 1 GWe-yr

Table 6 Effects of acute ionizing radiation doses

	Acute dose range (Sv)				
	1–2	2–5	5–10	10–50	>50
Initial symptoms					
Incidence	0–50%	50–90%	100%	100%	100%
Latency	>3 h	1–2 h	0.5–1 h	0.5 h	minutes
Critical period	2–6 wk	2–6 wk	2–6 wk	3–14 d	1–48 h
Incidence of death	0–10%	0–90%	0–90%	90–100%	100%
Death occurs in	months	weeks	weeks	2 wk	1–48 h
Leading system	Blood-forming			Gastrointestinal	Nervous

scribed in Sources, Effects and Risk of Ionizing Radiation,[21] and is summarized in **Table 6**. The effects of an acute radiation exposure can be divided into three phases. During the initial phase, the symptoms of radiation sickness appear. Latent period follows in which the symptoms largely disappear and it is possible for the exposed individual to perform useful tasks. The final phase follows in which the symptoms of radiation sickness recur and may include skin hemorrhages, diarrhea, and hair loss. The final phase persists through the recovery or death of the individual. The time to onset and duration of the phases and the severity of the symptoms depends on the dose received and vary from individual to individual. The doses listed in Table 6 are whole-body doses on human beings. For doses in the range of 0.25 to 1 Sv, significant changes in the blood can occur but few, if any, outward signs of radiation injury are apparent. For doses in the range of 1 to 2 Sv, the symptoms of radiation sickness are mild and do not occur until several hours after the exposure. For 10 Sv, survival is unlikely.

The doses taken by thief are the time integral of the dose rate at the midline of the thief. For a 10 Sv/h (roughly the field 1 m from a commercial spent fuel assembly), the thief receives 1 Sv every 6 min. A successful overt theft is estimated to take only 10 or 20 min so even if the thief is exposed to the full field during a 20 min theft, the dose accumulated will be about 3 Sv. Such a dose, while it will eventually cause the symptoms of radiation sickness to appear, is unlikely to produce any symptoms during the course of the theft and is unlikely to result in death. If the thief is willing to accept a dose of 1 Sv, the level at which no outward signs of radiation sickness occur, the dose rate required during 20 min theft is 3 Sv/h. If the thief is willing to accept a dose of 0.25 Sv, which is a dose limit for planned special exposure for adult,[22] the dose rate required during 20 min theft is ~0.75 Sv/h. The planned special exposure in a nuclear related facilities is permitted only in an exceptional situation when alternatives that might avoid the dose estimated to result from the planned special exposure are unavailable or impractical.[22]

As a result, we would say that spent fuel with above 10 Sv/h has a good radiation barrier for theft. The spent fuel with between 3 Sv/h and 10 Sv/h is a moderate radiation barrier for theft. Below 0.75 Sv/h, however, can no longer serve as a radiation barrier.

In order to see the performance of radiation barrier in var-

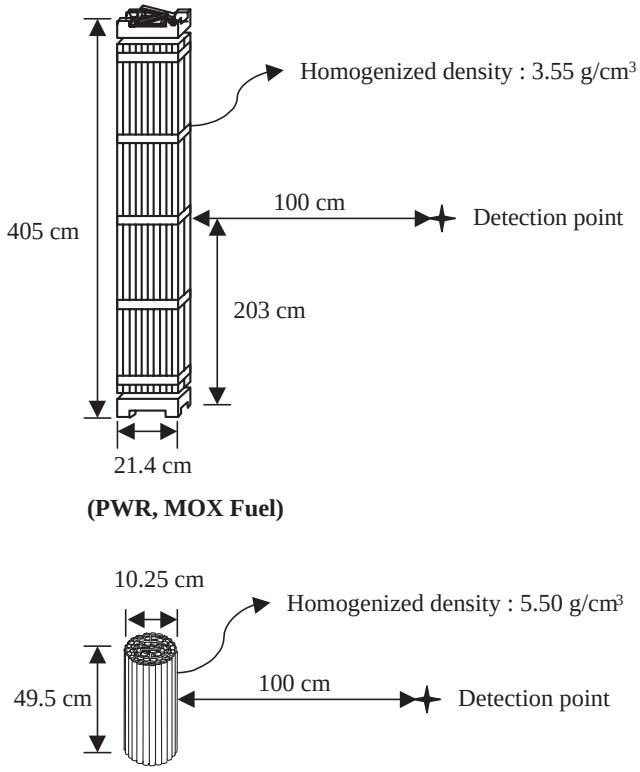


Fig. 4 Geometry for calculation of radiation shielding

ious fuels, we have calculated the radiation dose at 1 m from the surface of the fuel assembly or bundles at the mid-plane. For this, photon spectrums obtained from ORIGEN code are used. In order to calculate the radiation field, gamma shielding computer code, Microshield[23] is used. Geometry of the assembly for shielding calculation is shown in **Fig. 4**. Homogenized density concept for shielding calculation was used in this study. It was indicated that the homogenized density of PWR/MOX assembly and DUPIC/CANDU bundle, which are calculated from the geometry in Fig. 4, are 3.55 g/cm^3 and 5.5 g/cm^3, respectively.

Table 7 shows the radiation doses at 1 m from the surface of the fuel assembly or bundles at the mid-plane. The second and third columns of the table are the dose rate for obtaining 1 MTHM at a time and the dose rate for obtaining 1 SQ Pu at a time, respectively. For the dose rate for obtaining 1 SQ Pu, the plutonium contents of spent fuels obtained in previous chapter were used. It is indicated from the table that PWR spent fuel assembly with dose rate of 13.67 Sv/h and MOX spent fuel assembly with dose rate of 11.55 Sv/h have a good radiation barriers for theft. The DUPIC spent fuel bundle with dose rate of 7.12 Sv/h has a moderate radiation barrier for theft. On the other hand, the CANDU spent fuel bundle with dose rate of 0.37 Sv/h can no longer serve as a radiation barrier for theft. In the mean while, the fresh DUPIC fuel with dose rate of 11.8 Sv/h has a moderate radiation barrier for theft.

For the dose rate for obtaining 1 SQ Pu, it is indicated that all spent fuel has a good radiation barrier for theft but the MOX spent fuel with high plutonium content could be the worst one and the DUPIC spent fuel is the best one.

Table 7 Dose rate form various nuclear fuels

Fuels	Dose rate (Sv/h) per assembly or bundle	Dose rate (Sv/h) for obtaining 1 MTHM at a time	Dose rate (Sv/h) for obtaining 1 SQ Pu at a time
PWR spent fuel	13.67	31.27	27.52
MOX spent fuel	11.55	26.25	11.55
DUPIC spent fuel	7.12	352.48	370.24
CANDU spent fuel	0.37	18.32	38.85
Fresh DUPIC fuel	11.8	58.41	50.79

All spent fuels are assumed to cool for 10 years after irradiation.
Fresh DUPIC fuel is made form PWR spent fuel with cooling time of 10 years.
Radiation dose at 1 m from the surface of the fuel assembly or bundles at the mid-plane.

It is important to note that the fresh DUPIC fuel can play a radiation barrier part, better than CANDU spent fuels as well as fresh MOX fuel. Therefore, we would say that the DUPIC fuel cycle has the excellent resistance (radiation barrier) to proliferation, compared with an existing reprocessing option and CANDU once-through option. In addition, no fissile material is separated in the DUPIC fuel fabrication process.

Moreover, DUPIC is refabricated directly from highly radioactive spent PWR fuel in heavily shielded enclosure, and therefore, access to the sensitive materials is extremely difficult because of the high radiation field. The DUPIC processing is self-contained, and there is no transport of intermediate materials outside of the facility: spent LWR fuel enters the facility, and fresh CANDU-DUPIC fuel leaves. This feature of the DUPIC technology may be concordant with the PIPEX concept as was proposed during the INFCE.[1] The PIPEX approach to reducing access to nuclear materials at the reprocessing and conversion stage would be to make use of the heavy concrete shielding that provides protection against radiation in reprocessing plants to give a physical barrier against diversion.[24]

Due to those inherent features, it is inferred that the DUPIC fuel cycle could be a new fuel cycle alternative with high proliferation resistance close to “Spent Fuel Standard (SFS)” concept that was recently chartered by National Academy of Science in USA in the review on disposition alternatives of weapon plutonium.[3] The key idea behind the SFS is to utilize the hostile conditions of spent nuclear fuel, as an inherent barrier to any clandestine access to the nuclear material contained therein.

V. Conclusions

This study is focused on intrinsic barriers, especially, radiation field of the diverted material, amount of special nuclear material based on 1 GWe-yr and the quality of the separated fissile material. It is concluded from plutonium analysis of each fuel cycle that the DUPIC (Direct Use of Spent Fuel in CANDU) option has some benefits in plutonium consumption

and its quality aspects. From the radiation field analysis, it is also concluded that fresh DUPIC fuel could play an important radiation barrier role, more than even CANDU spent fuels.

In conclusions, due to those inherent features, the DUPIC fuel cycle could include technical characteristics that comply naturally with the Spent Fuel Standard, at all steps along the DUPIC linkage between PWR and CANDU. Therefore, in consideration of the huge amount of spent LWR fuel stockpile to be managed in the future, the DUPIC option can be a synergistic alternative for exhaustive burnout of the fissile residuals, even without separating them, taking advantage of the high neutron efficiency of heavy water.

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