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# DUPIC 핵연료주기의 환경친화성 분석

# Analysis of Environmental Friendliness of DUPIC Fuel Cycle



2001. 7

Korea Atomic Energy Research Institute

# 제 출 문

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본 보고서를 “경중수로 연계 핵연료주기 기술개발” 과제 (세부과제 "DUPIC 핵물질 안전조치 기술개발")의 기술보고서로 제출합니다.

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# Abstract

Some properties of irradiated DUPIC(Direct Use of Spent PWR Fuel In CANDU) fuels are compared with those of other fuel cycles. The properties include the radio-toxicity, decay heat, activity and actinide content embedded in various spent fuels or high level wastes which could be measures for the effectiveness of waste management. The other fuel cycles considered in this study are PWR of the 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 once-through mode with reactor grid equivalent to DUPIC fuel cycle (PWR-CANDU-OT). It was indicated from radio-toxicity analysis that the toxicity of the DUPIC option based on 1 GWe-yr is much smaller than those of other fuel cycle options, and is just about half the order of magnitude of other fuel cycles until decayed to a level below toxicity of initial ore. It means that the DUPIC option could have an indirect benefit on the safety of long term spent fuel disposal. It was indicated from the decay heat analysis that for the first 300 years the time PWR-MOX option has 1.7~2.0 times higher heat output and PWR-CANDU-OT option has 1.4~1.7 times higher heat output, compared to DUPIC option case. From total activity analysis of various fuel cycle options, the activity per metric ton heavy metal of spent fuel is the lowest in natural uranium CANDU fuel, but, in the case of activity based on 1 GWe-yr, the DUPIC option has the smallest activity. In the meanwhile, from the activity analysis of $^{99}$Tc and $^{237}$Np, which are important to the long-term transport of fission products stored in geologic media, the DUPIC option, was being contained only about half of those other options. It was found from the actinide content estimation that the MOX option has the lowest plutonium arising based on 1 GWe-year and followed by the DUPIC option. However, fissile Pu content generated in the DUPIC fuel was the lowest among the fuel cycle options. On the whole, the CANDU-OT option has the largest fissile plutonium as well as gross plutonium based on 1 GWe-year, which means negative points in nuclear proliferation resistance aspects. In conclusions, the irradiated DUPIC fuels would have good properties on waste management and proliferation resistance, compared to other fuel cycle cases.

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# I. Introduction

The commercial nuclear fuel cycles in operation in the world include the once-through light water reactor (LWR) fuel cycle (in the U.S. and Sweden), once-through heavy water reactor (HWR) fuel cycle (in Canada), the LWR fuel cycle with the recycled MOX(Mixed Oxide) fuel (in Japan, France and Russia etc.). The present civil use of recycled uranium and plutonium in LWR involves the development and the utilization of large scale reprocessing plants and MOX fuel fabrication facilities. There are some other alternative fuel cycles, which are currently used on a limited scale or are under development. They include thorium fuel cycle (in India), dry recycle and DUPIC(Direct Use of Spent PWR Fuel in CANDU) fuel cycle [1-4] (under development in Korea).

This study examines whether the DUPIC fuel cycle will make radioactive waste management more effective, compared with other fuel cycles such as PWR(Pressurized Waster Reactor) once through cycle, CANDU(Canadian Deuterium Uranium) once-through cycle and thermal recycling option to use existing PWR with MOX fuel. This study is focused on the decay heat, activity, and radio-toxicity which could be a measure for the effectiveness of waste management for various nuclear fuel cycles. Actinide isotope content, which could be a measure for proliferation resistance, is also examined and compared each other. Those properties in conventional fuel cycles have been well established and compared each other[5-7], but the characteristics of the DUPIC fuel cycle, in which PWRs are linked to a CANDU, has not been established systematically. Of those properties, ingestion toxicity, which is the total volume of water to dilute the wastes to public drinking water standards, is a crude measure of the potential danger of radioactive material. The decay heat and activity properties could be used for the design of transportation cask, interim storage, final disposal facility and their treatment systems.

For reasonable comparison, all amounts calculated in this study are expressed on the basis of one gigawatt(GWe) year of reactor operation as well as metric ton heavy metal(MTHM) of spent fuels or High level wastes. For this, fuel cycle scenarios first are set up and reactor parameters and their fuel characteristics are assumed appropriately. Using those characteristics, fuel material flows are estimated based on 1 GWe-yr. And then the various fuel properties are assessed by use of ORIGEN computer code[8] and compared each other.

Reference fuel cycle models and approaches to estimate the properties of irradiated fuels are given in Section II. The material flow of each fuel cycle is described in Section III. The transuranium isotope content, decay heat, toxicity index and activity are described in section IV through section VII, respectively.

# II. Reference Fuel Cycle Model

## II.A. Fuel Cycle Model

Fig. 1 shows fuel cycle options considered in this study and steps or components consisting of fuel cycles. The first cycle 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 CANDU reactors 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 CANDU reactors. 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.

## II.B. Reference Reactors and Fuels

For material flow of each fuel cycle, reference PWR and CANDU reactors 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 reactor, 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

$$\text{Fuel loading per core} = \frac{P \times 100}{\varepsilon \times SH} \qquad (1)$$

where *P*, *SH* and $\varepsilon$ are the electric power (MWe) of a CANDU reactor, the specific heat (MWt/MTHM) and efficiency (%), 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% $U^{235}$ was chosen as a reference PWR fuel in DUPIC fuel cycle development in Korea [2, 4].

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 [8]. It found that content 0.82wt% of $U^{235}$ and 0.89wt% 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 35,000MWD/MTU can be used in the PWR core. In this situation, PWR core with MOX fuel consists of 10.22 MTHM MOX fuel and 59.28 MTU LEU fuel per reactor core.

In a CANDU reactor, the discharge burnup of natural CANDU fuel is assumed to be 7500 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 [2, 4].

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

$$\text{Annual requirement} = \frac{P \times 365 \times C}{\varepsilon \times BU} \qquad (2)$$

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 the table.

Table 1 Characteristics of Reference Reactors

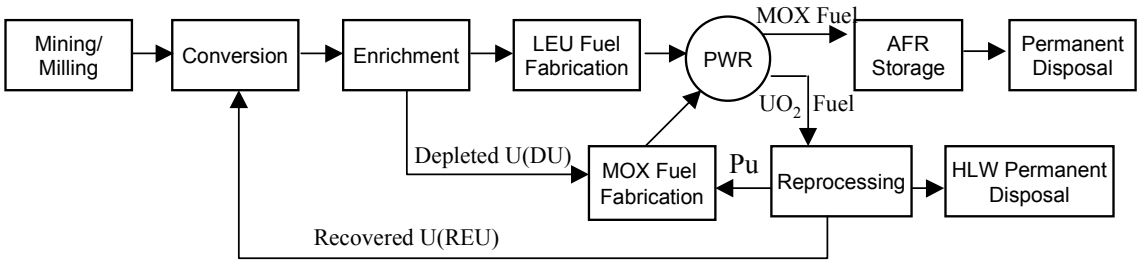
| Reactor parameters | PWR | CANDU |
|---|---|---|
| - Electric power (MWe) | 950 | 713 |
| - Thermal efficiency (%) | 34 | 33 |
| - Thermal power (MWt) | 2,794 | 2,161 |
| - Specific power (MWt/ton U) | 40.2 | 25.5 |
| - Load factor | 0.8 | 0.9 |
| - Cycle length (Full Power Day) | 290 | - |
| - No. of fuel assemblies or bundles per core | 157 | 4,560 |
| - No. of batches for PWR | 3 | - |
| - Loading per core (MTU) | 69.5 | 84.7 |

Table 2 Characteristics of Reference Reactors and Fuels

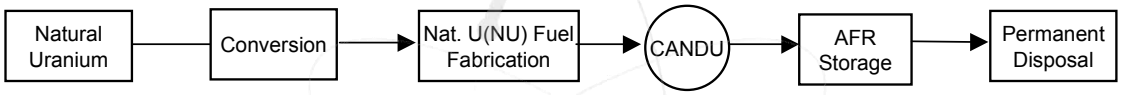
| Item | Characteristic Parameters | | | |
|---|---|---|---|---|
| | PWR with LEU fuel | PWR with LEU and MOX fuel* | CANDU with NU fuel | CANDU with DUPIC fuel |
| **Reactor** | | | | |
| - Loading per core (MTU) | 69.5 | 69.5<br>(10.22 MOX)<br>(59.28 LEU) | 84.7 | 84.7 |
| - Annual fuel requirement (MTU) | 23.31 | 23.31<br>(3.43 MOX)<br>(19.88 LEU) | 94.63 | 46.09 |
| **Fuel** | | | | |
| - Initial enrichment | 3.5% | 5% $Pu_f$ MOX<br>3.5% LEU | Nat. U | PWR S/F |
| - No. 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<br>(3.61 MOX)<br>(20.93 LEU) | 132.73 | 64.64 |

*14.7% of the fuel in MOX and 85.3% of the fuel in LEU(Low Enriched Uranium) – an equilibrium state is reached when all spent PWR fuels are reprocessed to make needed MOX fuels.

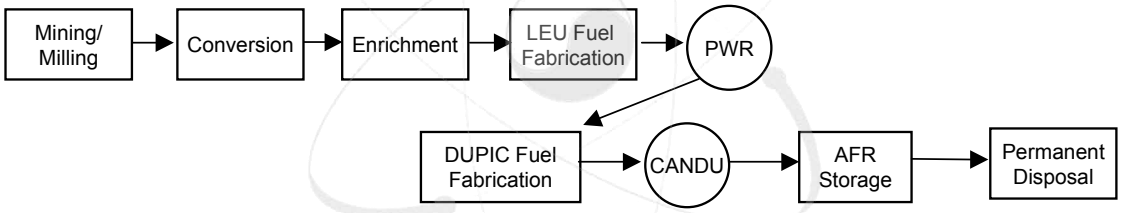

**(PWR-OT)**



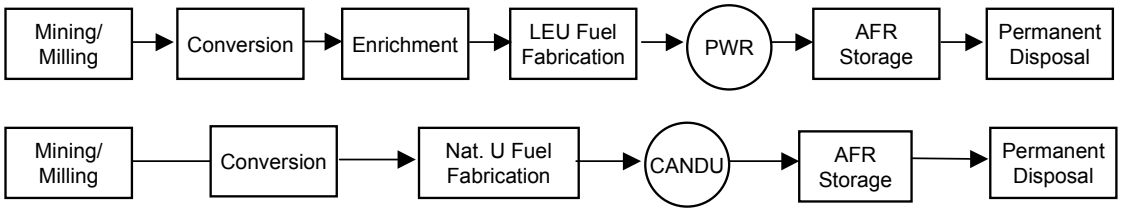

**(PWR-MOX)**




**(CANDU-OT)**




**(DUPIC)**




**(PWR-CANDU-OT)**

Fig. 1 Fuel Cycle Options

## III. Material Flow for Fuel Cycle Options

For PWR-CANDU-OT and DUPIC fuel cycle, the equilibrium core ratio between PWRs and CANDU reactors 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 reactor can be calculated as follows;

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

Where $M_{DUPIC}$, $M_{PWR}$, and $L_{DUPIC}$ are annual requirement of DUPIC, annual requirement of PWR and 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 followings;

$$\text{Electricity generation portion of PWR} = \frac{P_{PWR} \times R_C}{P_{PWR} \times R_C + P_{CANDU}} \quad (4)$$

Where $P_{PWR}$ and $P_{CANDU}$ are electricity powers of PWR and CANDU, respectively. So the portion 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. Enrichment amount in unit of Separative Work Unit (SWU) is calculated as follows:

$$\text{SWU} = M_p V_p + M_t V_t - M_f V_f \quad (5)$$

Where $M_p$ = mass of uranium to be charged in the fuel fabrication facility,

$M_f$ = mass of uranium feed in enrichment plant (and output of conversion plant), and

$M_t$ = mass of uranium discharged from the enrichment plant(So called depleted uranium).

$$V_x = (2e_x - 1)\ln\frac{e_x}{(1-e_x)} \quad (6)$$

and $x$ is subscript for $f$, $p$ or $t$,

where $e_p$ = fraction of $^{235}U$ in the uranium feed (3.5 wt% in this study),

$e_t$ = fraction of $^{235}U$ in the tails (0.25 wt% in this study), and

$e_f$ = fraction of $^{235}U$ of uranium to be charged in enrichment plant (0.711 wt% in this study).

Then, $$M_f = M_p \frac{(e_p - e_t)}{(e_f - e_t)} \quad (7)$$

and $$M_t = M_f - M_p \quad (8)$$

From the above equations, if $M_p$ and three fractions of the $^{235}U$ in enrichment plant are known, the SWU as well as $M_f$ and $M_t$ (depleted uranium) can be calculated.

The requirement of natural uranium resources are converted to that of uranium ($U_3O_8$) by the following formulation:

$$M_n = M_R \times \frac{e_p - e_t}{e_f - e_t} \times (1 + l_1) \times \frac{W_{U_3O_8}}{W_{U_3}} \times (1 + l_2) \quad (9)$$

where $M_n$ is the mass of uranium ($U_3O_8$) to feed, $M_R$ is the mass of uranium charged to the reactor, and $W_{U_3O_8}/W_{U_3}$ is the weight fraction of uranium in uranium ($U_3O_8$), and $l_1$ *and* $l_2$ are process loss rate of conversion and fuel fabrication, respectively.

The results of the material balance analyses, which were calculated by equation 1 through equation 9 with reference reactors parameters (shown in Table 1) and their fuel characteristics(shown in Table 2), are shown in Fig. 2. All values were expressed on basis of 1 GWe-yr for all fuel cycle options.

From the material flow of the Fig. 2, we can find interesting values on natural uranium resources and spent fuel arisings for each fuel cycle. It indicated that the DUPIC fuel cycle with PWR and CANDU reactor requires only ~341 $lbU_3O_8$ of natural uranium which is just for PWR fuel with enrichment of 3.5 wt% $^{235}U$. On the other hand, ~341 $lbU_3O_8$ of natural uranium for PWR fuel and ~96 $lbU_3O_8$ of natural uranium for CANDU fuel are required for the PWR-CANDU once-through cycle. It means that the DUPIC option has ~22% uranium resources saving based on weight(ton), compared with PWR-CANDU-OT fuel cycle. We found that the natural uranium resource of the DUPIC fuel cycle is a little smaller than that of reprocessing cycle. In addition, the amount of spent fuel annually discharged from the DUPIC fuel cycle generates only ~18 MTHM/GWe-yr while once-through fuel cycle(PWR-CANDU-OT) is ~54 MTHM/GWe-yr. The DUPIC fuel cycle generates ~67% less

spent fuels than that of PWR-CANDU-OT cycle. Relative amount of natural resources saving and spent fuel arisings reduction to maximum values are summarized in Table 3. Compared between PWR-OT and CANDU-OT, it is indicated that PWR-OT requires the largest natural uranium resources(~466 $lbU_3O_8$/GWe-yr) and CANDU-OT generates the largest spent fuels(~133 MTHM/GWe-yr).

Table 3 Summary of the Natural Uranium Resources and Spent Fuel Arisings Based on Metric Ton

| | Nuclear Fuel Cycles | | | | |
|---|---|---|---|---|---|
| | PWR-OT | PWR-MOX | CANDU-OT | DUPIC | PWR-CANDU |
| Natural Uranium Saving Rate | 0.00% | 18.68% | 24.8% | 26.73% | 6.18% |
| Disposal Waste (SF/HLW) Reduction Rate | 81.51% | 81.51% | 0.00% | 86.69% | 59.04% |

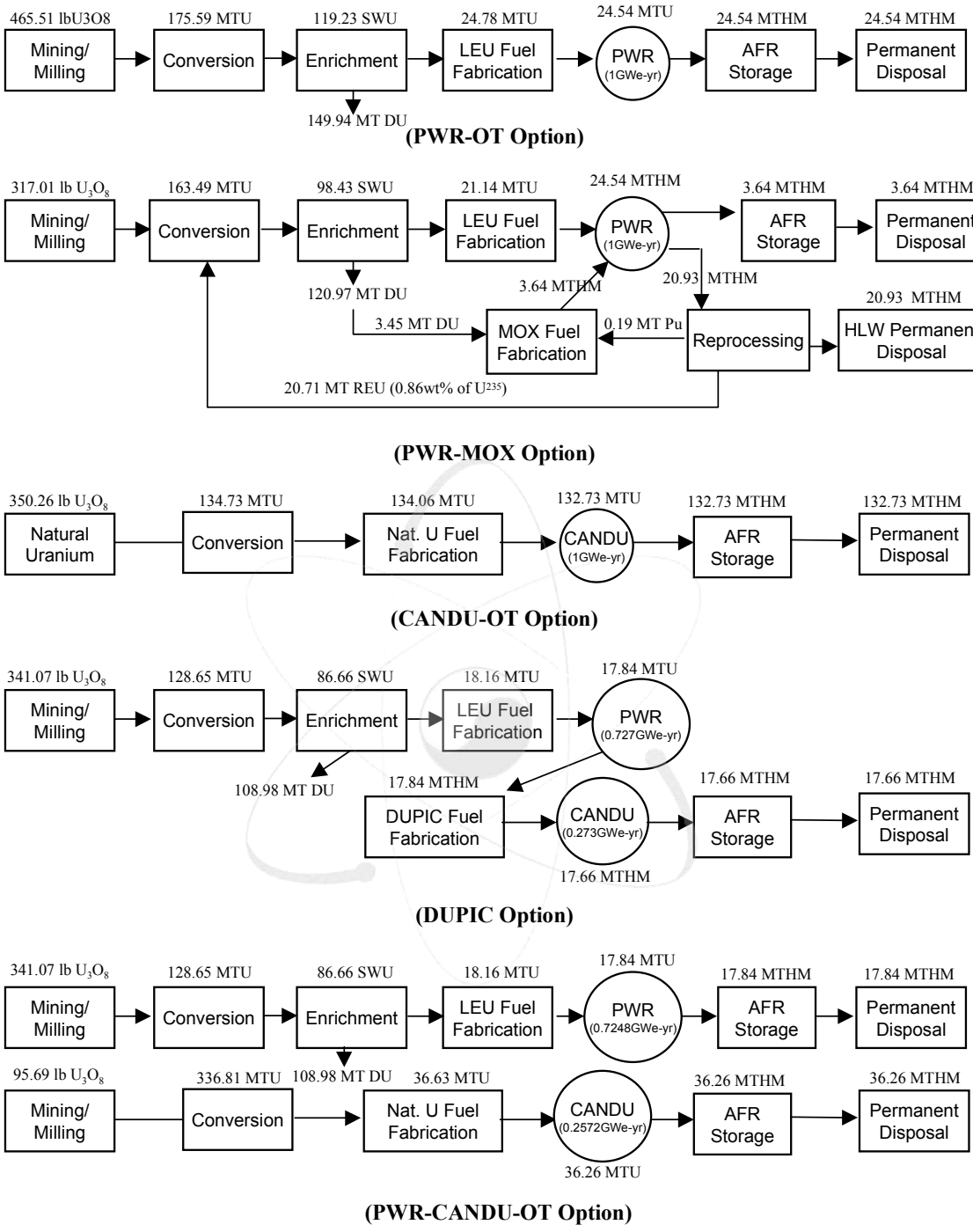



Fig. 2 Material Flows of Fuel Cycle Options on the Basis of one GWe-year

# IV. Major Actinides Content

In this chapter, we will evaluate transuranium isotope composition including plutonium contained in spent fuels generated in the five alternative fuel cycles. The compositions are assessed with metric ton basis and then those values are translated into 1 GWe-yr basis. Especially the plutonium and its content contained in spent fuels are important because it could be a measure of proliferation resistance.

In order to calculate how much plutonium there is in spent fuels, we have used a burnup simulation code, ORIGEN 2 computer code[8]. 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 reactor, isotope contents of PWR spent fuel, which are also calculated by the ORIGEN code, has been used. All actinides, transuranic isotopes and 140 fission products contained in PWR spent fuels were inputted in the code. It excludes 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 1700 $^{o}$C [9]. The removed fission products referred to the KAERI(Korea Atomic Research Institute) report[9], are described in Table 4.

Table 5 and Fig. 3 show the comparison of major actinides contained in four different spent fuels with 10 years of cooling time. 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 0.84 wt% $^{235}$U and 0.88 wt% Pu, and about 68.4% of the Pu is fissile isotopes($^{239}$Pu and $^{241}$Pu). In the meanwhile, MOX spent fuel is containing only 0.11 % wt% $^{235}$U but is containing as much as 0.88 wt% Pu. Both DUPIC spent fuels and CANDU spent fuels are containing about only 0.22 wt% U235, but the DUPIC spent fuel is containing about two times more plutonium contents, 0.84 wt%, than the CANDU spent fuel case.

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 the 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 (~0.33wt%) is much lower than that of MOX spent fuel(~0.82wt%). On the other hand, $^{236}$U produced by (n, $\gamma$) 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 0.22wt% $^{236}$U but the MOX spent fuel is containing only

0.11wt% $^{236}U$. Therefore, it is concluded that MOX spent fuel is more effective than the DUIC spent fuel for multiple recycling.

Figure 3 shows that the MOX spent fuels have much more miner actinide such as Cm and Am 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 Fig. 2, total plutonium embedded in spent fuels are calculated on the basis of 1 GWe-year and then compared in Fig.4. Total plutonium generated during 1 GWe-year is showing to be the biggest (~535g-Pu/GWe-yr) in CANDU-OT option and the least (~88g-Pu/1GWe-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 ~141g-Pu/GWe-yr which is a little higher than the PWR-MOX case, but the DUPIC option has the lowest fissile Pu 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 in nuclear proliferation resistance aspects.

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 |

Table 5 Major Actinides Content Contained in Various Spent Fuels

| isotopes | PWR | | MOX | | DUPIC | | CANDU | |
|---|---|---|---|---|---|---|---|---|
| | g/MTU* | Wt%** | g/MTU | Wt% | g/MTHM | Wt% | g/MTU | Wt% |
| U234 | 200.4 | 0.0208 | 92.8 | 0.0096 | 187.3 | 0.0198 | 44.5 | 0.0045 |
| U235 | 8190.0 | 0.8497 | 1074.0 | 0.1113 | 2098.0 | 0.2212 | 2194.0 | 0.2211 |
| U236 | 4360.0 | 0.4523 | 235.4 | 0.0244 | 5073.0 | 0.5348 | 741.4 | 0.0747 |
| Np237 | 522.9 | 0.0542 | 92.6 | 0.0096 | 542.5 | 0.0572 | 28.0 | 0.0028 |
| Pu238 | 153.7 | 0.0160 | 930.1 | 0.0964 | 387.9 | 0.0409 | 3.4 | 0.0003 |
| Pu239 | 5329.0 | 0.5529 | 7867.0 | 0.8156 | 3157.0 | 0.3328 | 2796.0 | 0.2818 |
| Pu240 | 2203.0 | 0.2286 | 6459.0 | 0.6696 | 2792.0 | 0.2944 | 1046.0 | 0.1054 |
| Pu241 | 751.5 | 0.0780 | 3683.0 | 0.3818 | 523.5 | 0.0552 | 130.7 | 0.0132 |
| Pu242 | 456.8 | 0.0474 | 4811.0 | 0.4988 | 1099.0 | 0.1159 | 52.8 | 0.0053 |
| Am241 | 497.3 | 0.0516 | 2834.0 | 0.2938 | 426.5 | 0.0450 | 82.7 | 0.0083 |
| Am243 | 91.3 | 0.0095 | 1251.0 | 0.1297 | 192.2 | 0.0203 | 1.9 | 0.0002 |
| Cm244 | 18.5 | 0.0020 | 269.3 | 0.0279 | 54.9 | 0.0058 | 0.1 | 0.0000 |
| Cm245 | 1.0 | 0.0001 | 13.7 | 0.0014 | 0.9 | 0.0001 | 0.0 | 0.0000 |

*MTU means the initial metric ton uranium used in ORIGEN code input.

**wt% means the weight percent of heavy metal came from the ORIGEN output.

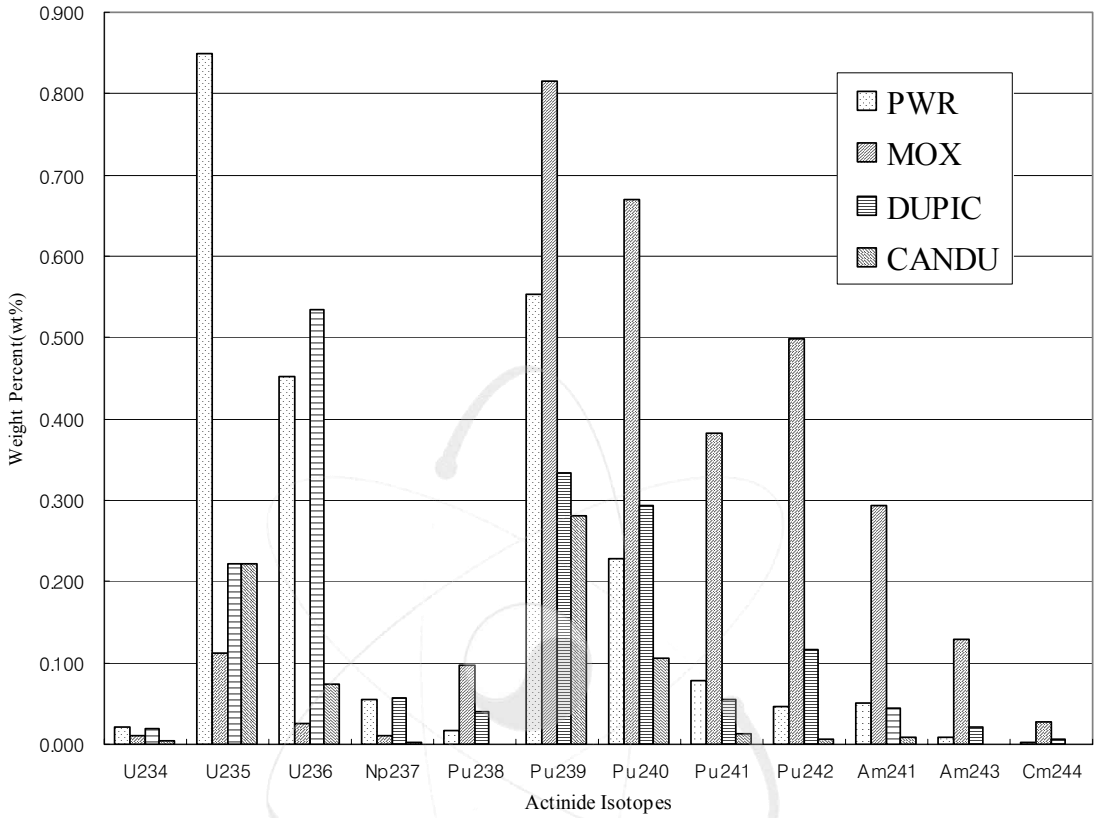



Fig. 3 Weight Percent of Major Actinides embedded in Various Spent Fuels

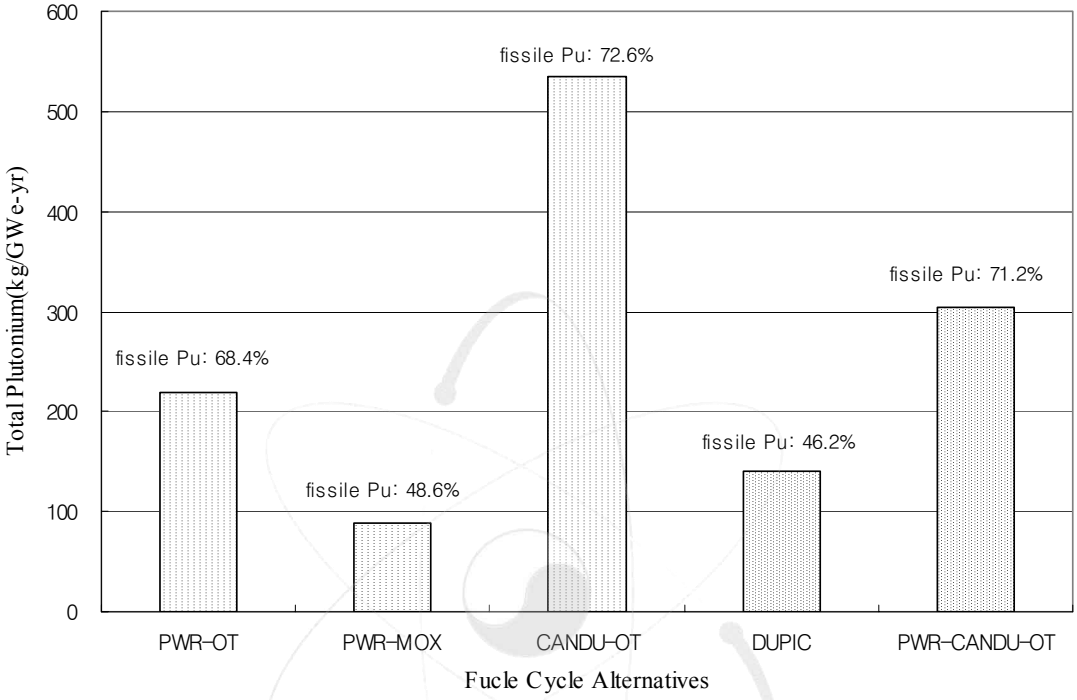



Fig. 4 Total Plutonium of Fuel Cycle Alternatives Based on 1 GWe-year

# V. Radiation Barrier of Various Spent Fuels

In this chapter, we examines quantitatively radiation barrier, which is one of nuclear proliferation barriers, and the radiation barrier performances are compared with each fuel cycle options.

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 away a significantly lager mass than just the stolen object.

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 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 fields 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 described in Sources, Effects and Risk of Ionizing Radiation[11], 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 the 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/hr (roughly the field 1 m from a commercial spent fuel assembly), the thief receives 1 Sv every 6 minutes. A successful overt theft is estimated to take only ten or twenty minutes so even if the thief is exposed to the full field during a twenty minutes 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 minutes theft is 3 Sv/hr. If the thief is willing to accept a dose of 0.25 Sv, which is a dose limit for planned special exposure for adult[12], the dose rate required during 20 minutes theft is ~0.75 Sv/hr. 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[12].

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

In order to see the performance of radiation barrier in various 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[13] is used. Geometry of the assembly for shielding calculation is shown in Fig. 5. 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/cm3 and 5.5 g/cm3, 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 section were used. It is indicated from the table that PWR spent fuel assembly with dose rate of 13.67 Sv/hr and MOX spent fuel assembly with dose rate of 11.55 Sv/hr have a good radiation barriers for theft. The DUPIC spent fuel bundle with dose rate of 7.12 Sv/hr has a moderate radiation barrier for theft. On the other hand, the CANDU spent fuel bundle with dose rate of 0.37 Sv/hr 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/hr 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.

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 fuel 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 [14]. 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 [15].

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[16]. 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.

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 hr | 1-2 hr | 0.5-1 hr | 0.5 hr | minutes |
| Critical period | | 2-6 wk | 2-6 wk | 2-6 wk | 3-14 d | 1-48 hr |
| Incidence of death | | 0-10% | 0-90% | 0-90% | 90-100% | 100% |
| Death occurs in | | months | weeks | weeks | 2 weeks | 1-48 hr |
| Leading system | | Blood-forming | | | Gastrointestinal | Nervous |

Table 7 Dose Rate form Various Nuclear Fuels

| Fuels | Dose Rate(Sv/hr) per assembly or bundle | Dose Rate(Sv/hr) for obtaining 1 MTHM at a time | Dose Rate(Sv/hr) for Obtaining 1 SQ Pu at a time |
|---|---|---|---|
| PWR SF | 13.67 | 31.27 | 27.52 |
| MOX SF | 11.55 | 26.25 | 11.55 |
| DUPIC SF | 7.12 | 352.48 | 370.24 |
| CANDU SF | 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.

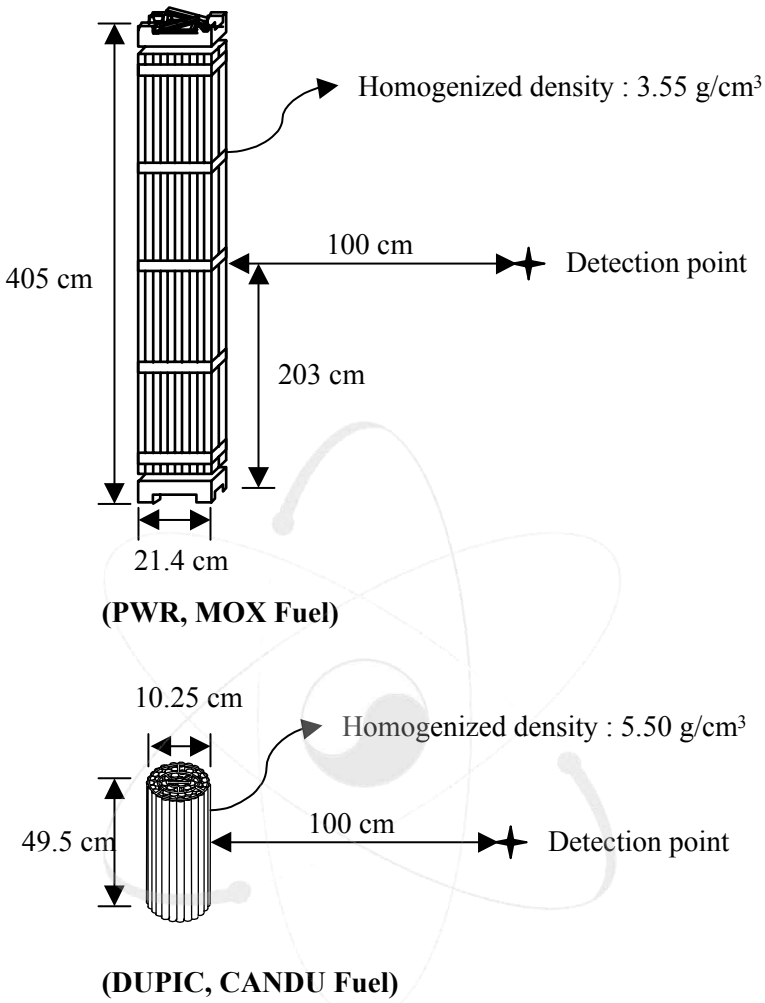



Fig. 5 Geometry for Calculation of Radiation Shielding

## V. Decay Heat Generation

One way to quantify the ease or difficulty of waste management such as storage and disposal of spent fuels is to measure the decay heat generated in wastes in a given nuclear fuel cycle. Generally, for low level wastes, the key cost driver is the waste volume but the key cost driver in spent fuels and high level wastes(HLW) is the decay heat. The decay heat generally affects dry storage or disposal waste spacing. This spacing is important in disposal, because they affect the number spent fuel canisters that can be placed in the repository of a given size and thus the disposal cost.

In order to forecast the decay heat of spent fuels and HLWs, the ORIGEN 2 code[9] was used. Fig. 6 shows the results of the decay heat analysis of the four different spent fuels on basis of one metric ton heavy metal. On the whole, it is shown that the MOX spent fuel has the largest decay heat. The decay heat of the MOX spent fuels is higher than that of the PWR spent fuel even though the burnup of the MOX fuels is the same as the PWR fuel. The decay heat of the DUPIC spent fuel is similar to that of the PWR spent fuels but in short term until 100 years the DUPIC case is a little the lower. The decay heat of the CANDU spent fuel is the lowest as expected.

It is more meaningful to compare all decay heat of all wastes generated in fuel cycles. In fact, some high level wastes are generated in reprocessing process and DUPIC fuel fabrication process. So we have calculated decay heats generated in HLWs which come from both reprocessing and DUPIC fuel fabrication. It is assumed that the reprocessing wastes include all fission products, miner actinide and process loss in reprocess and MOX fabrication process. In this study, the process losses are assumed to be 1% of U and Pu to be reprocessed. In order to calculate the decay heat of DUPIC HLW, dirty scrap and fission products(Kr, Cs, Ru) with high decay heat are considered. Dirty scrap is assumed to be 1% of spent fuel to be treated[9]. Figure 7 compares the decay heat of the two HLWs. It is seen from the figure that decay heat of the DUPIC HLW is only about 2% ~ 46% of the decay heat of reprocessing HLW for 10~300 years of cooling time.

Fig. 8 shows total decay heat based on 1 GWe-yr obtained by multiplying decay heat per metric tone by fuel cycle material flow of Fig. 2. The decay heat generated from HLWs are added to both DUPIC option and PWR-MOX option. On the whole, the decay heat of the DUPIC option is the lowest until about 2000 years of cooling time. On the other hand, the decay heat of the PWR-MOX option is the smallest between about 3000 years and 100,000 years of cooling time. It is mainly due to the decrease of plutonium, which is a main source of decay

heat during that period. From the figure, we can see some interesting results for the CANDU option. Even though the decay heat per metric ton of the CANDU spent fuel is the lowest, the decay heat per GWe-yr of the CANDU-OT is rather higher than any others. It is due to the increase of the spent fuel amount generated for one GWe-yr, compared to other options.

In fact, a very long-term decay heat is not important for repository. The heat generated by radioactive decay in a sealed repository will raise rock temperatures to a maximum from a few decades to about 300 years after repository closure and then gradually subside[10, 17]. This maximum decay heat affects disposal waste spacing and then disposal cost. It depends on the spacing of canisters in the repository, and the thermal conductivity of the host rock etc.

In order to compare the decay heat during that period, the total decay heat from 30 years to 300 years of cooling time is described in Fig. 8. The decay heat of the DUPIC option clearly is the lowest during that period. As opposed to the other period, the PWR-MOX option has the higher heat output during the first 300 years than other cases.

In order to make the differences among fuel cycle options more apparent, Fig. 9 shows the ratio of the heat output of fuel cycle options compared with that of DUPIC option. Whenever the ratio is greater than 1.0, decay heat of the fuel cycle is greater than that of DUIPIC option. From the Fig. 10, one can see that the PWR-MOX option clearly has the higher heat output which is about 1.7~2.0 times higher heat output. And PWR-CANDU-OT option has 1.4~1.7 times higher heat output compared to DUPIC option. It means that PWR-MOX option would take up 1.7~2.0 times more space than the DUPIC case.

As previously, in order to determine why the decay heat of Fig. 8 and 9 change the way it does, we have extracted from ORIGEN results the key elements that are generating the heat. Fig. 11 through Fig. 15 show the contribution of each element to the decay heat. On the whole, it is indicated that total decay heat generated in fuel cycles governed by fission products during the first 100 years through 200 years. From ~200 year to ~ 1000 years, the decay heat governed by americium, and then plutonium after that period.

Table 8 shows the comparison of the isotope decay heat output at 50 years and 300 years of cooling time compared with DUPIC option. The first column lists the heat generated by seven isotopes that are producing almost all of heat in the fuel cycles. On the whole, the decay heat of MOX spent fuel is the highest at both 50 years and 300 years, and the decay heat of CANDU spent fuel is the lowest at both 50 years and 300 years as expected.

As can be seen from the table, at 50 years of cooling time, the decay heat of the DUPIC spent fuel governed by $^{238}$Pu. However, the decay heat of the PWR and CANDU spent fuels

governed by $^{90}Sr$ and $^{137}Cs$. On the other hand, the decay heat of the MOX spent fuel governed by $^{241}Am$. The decay heat of DUPIC spent fuel is 80% ~ 93% of the decay heat of the PWR spent fuel. The main difference between the two spent fuel are the decrease of fission products($^{90}Sr$ and $^{137}Cs$) and buildup of $^{238}Pu$ in DUPIC spent fuels. The decrease of by $^{241}Am$ in DUPIC spent fuel compared with PWR spent fuel can explain with that $^{241}Am$ could be annihilated in a reactor.

Table 8 Comparison of the Isotope Decay Heat Output of Spent Fuels

(Unit : W/MTHM)

| isotopes | DUPIC Spent Fuel | | PWR Spent Fuel | | MOX Spent Fuel | | CANDU Spent Fuel | |
|---|---|---|---|---|---|---|---|---|
| | 50 years | 300 years | 50 years | 300 years | 50 years | 300 years | 50 years | 300 years |
| $^{90}$Sr* | 52 | 0 | 160 | 0 | 74 | 0 | 34 | 0 |
| $^{137}$Cs* | 73 | 0 | 166 | 1 | 168 | 1 | 37 | 0 |
| $^{238}$Pu | 161 | 22 | 64 | 9 | 387 | 56 | 1 | 0 |
| $^{239}$Pu | 6 | 6 | 10 | 10 | 15 | 15 | 5 | 5 |
| $^{240}$Pu | 20 | 20 | 16 | 15 | 47 | 46 | 7 | 7 |
| $^{241}$Am | 95 | 69 | 123 | 91 | 647 | 476 | 21 | 16 |
| $^{244}$Cm | 34 | 0 | 11 | 0 | 165 | 0 | 0 | 0 |
| all others | 4 | 2 | 4 | 1 | 23 | 12 | 0 | 0 |
| total | 445 | 119 | 555 | 128 | 1526 | 606 | 106 | 29 |

* Included daughter products

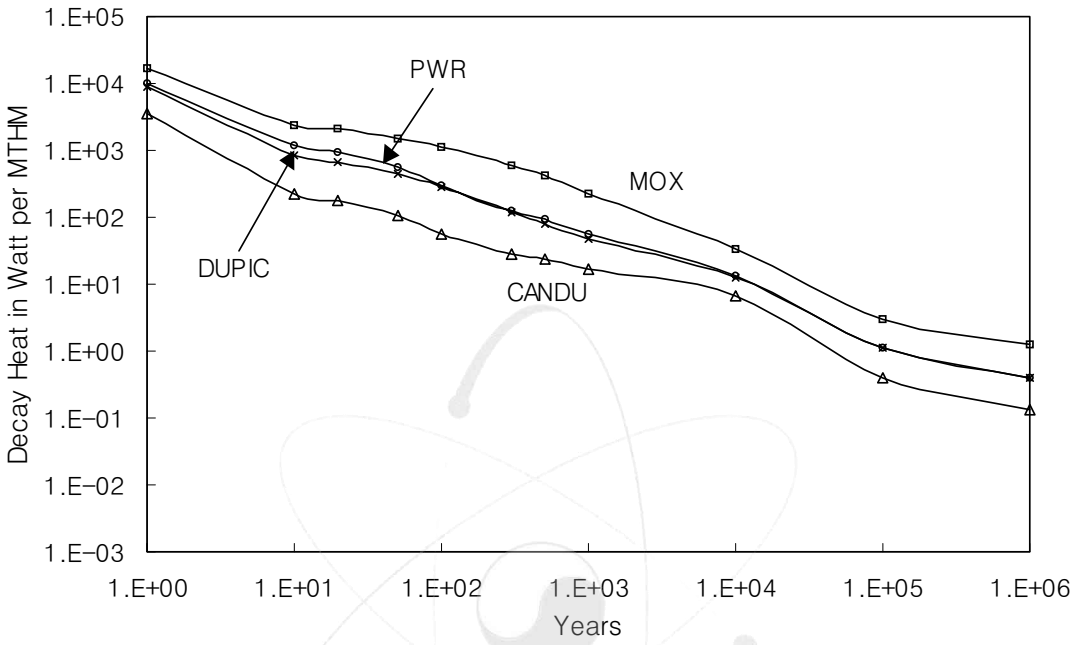



Fig. 6 Decay Heat Generated from Various Spent Fuels

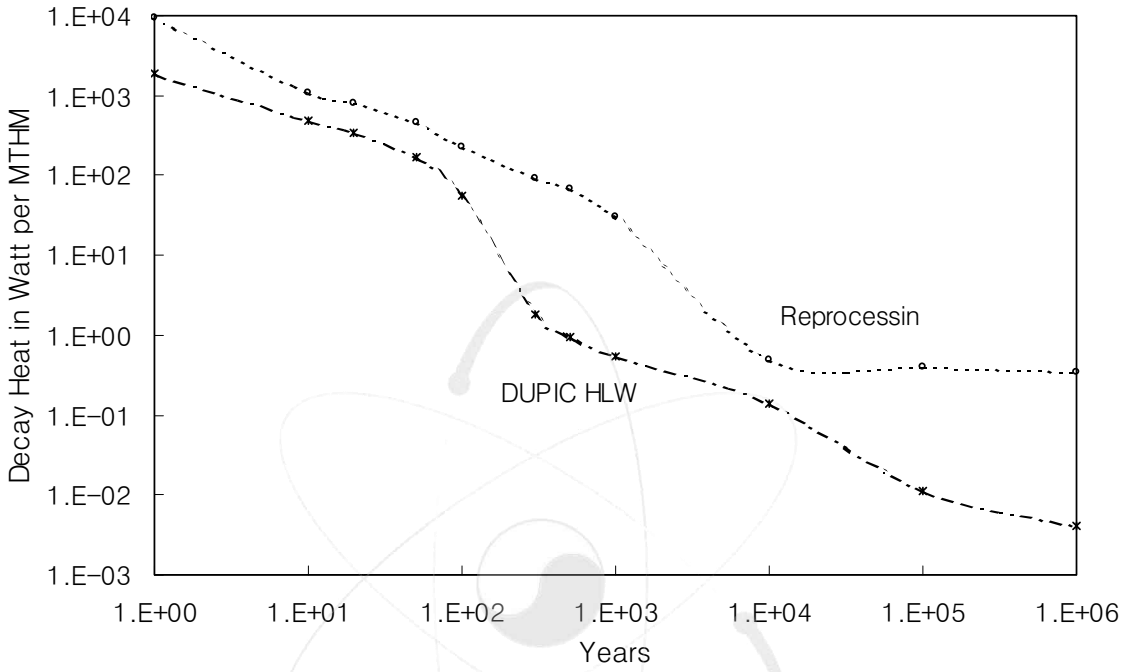



Fig. 7 Decay Heat of HLWs Based on Initial Heavy Metal to be Treated

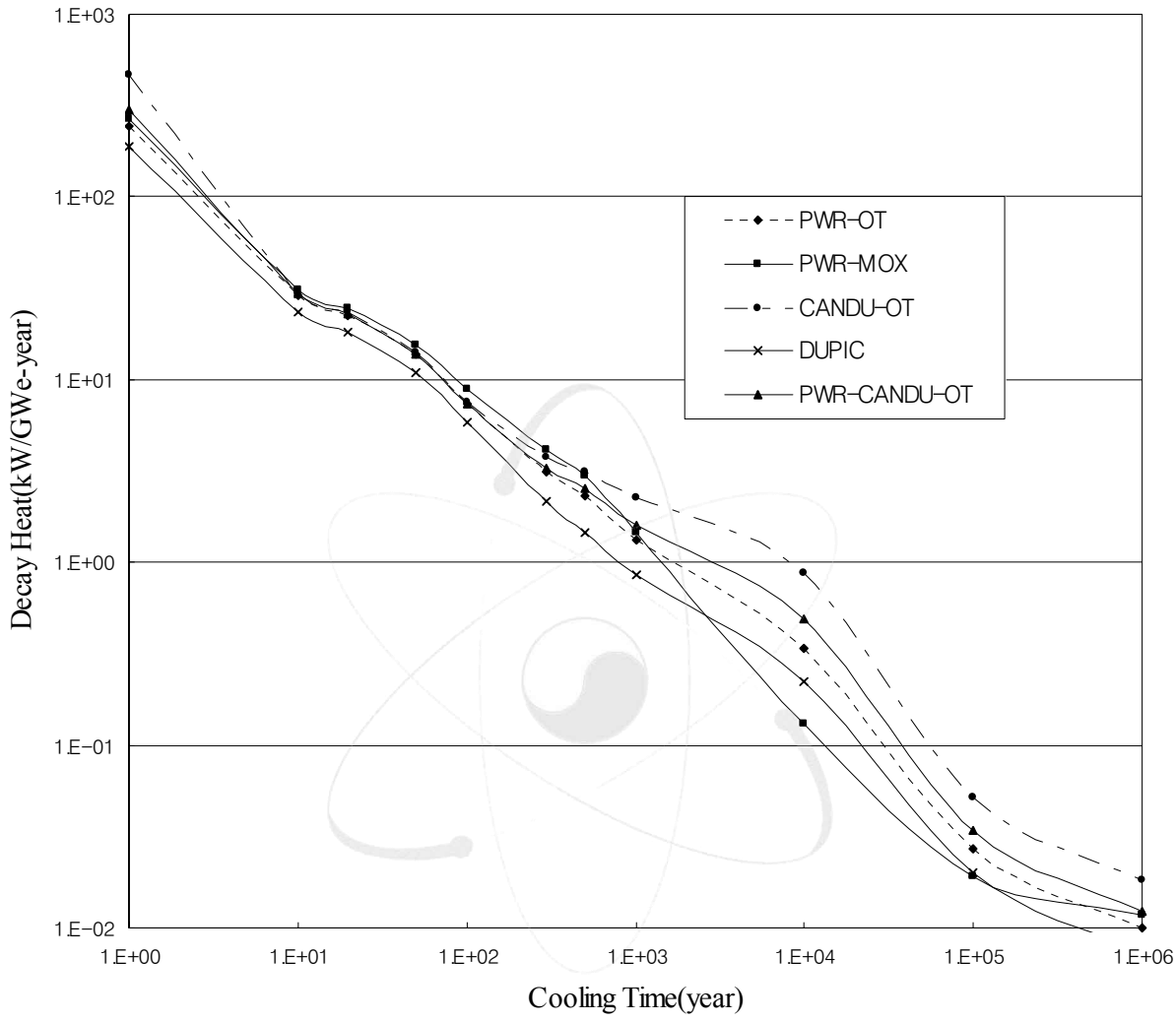



Fig. 8 Decay Heat Generated from Wastes in Nuclear Fuel Cycles(Based on 1 GWe-yr)

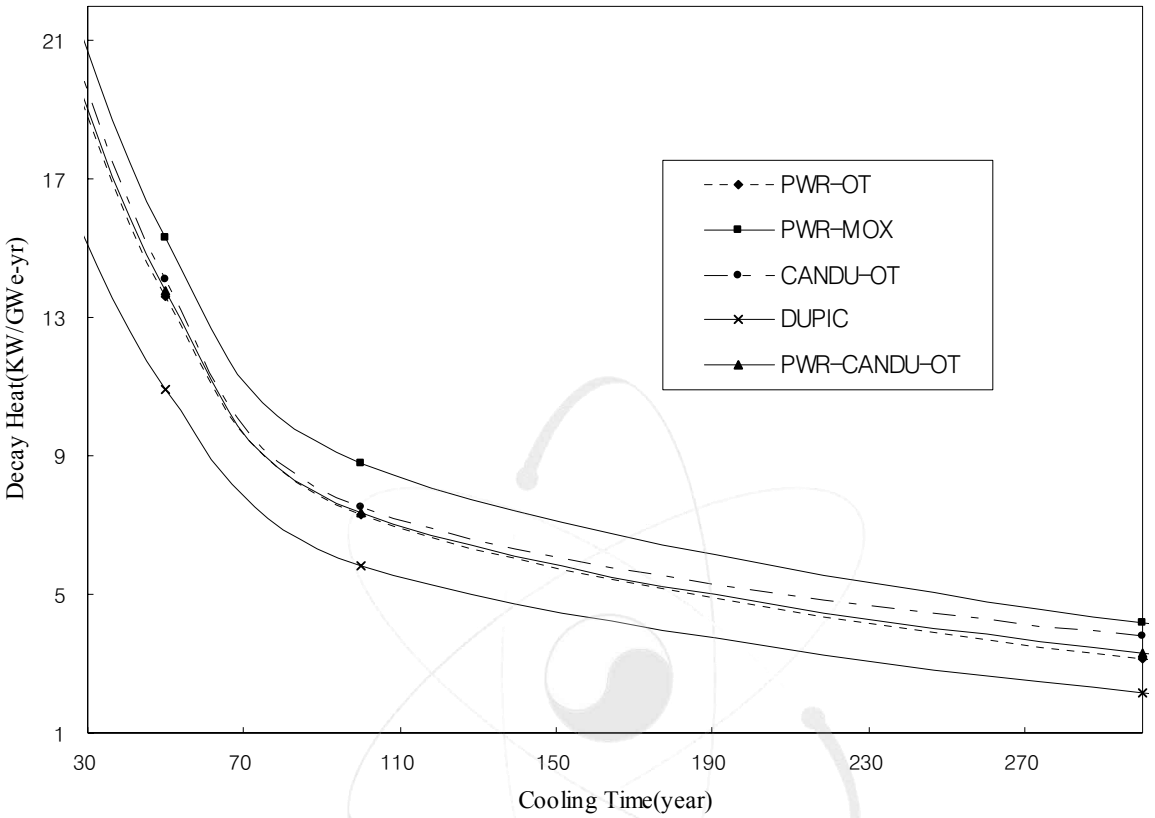



Fig. 9 Decay Heat of Wastes generated in Nuclear Fuel Cycles

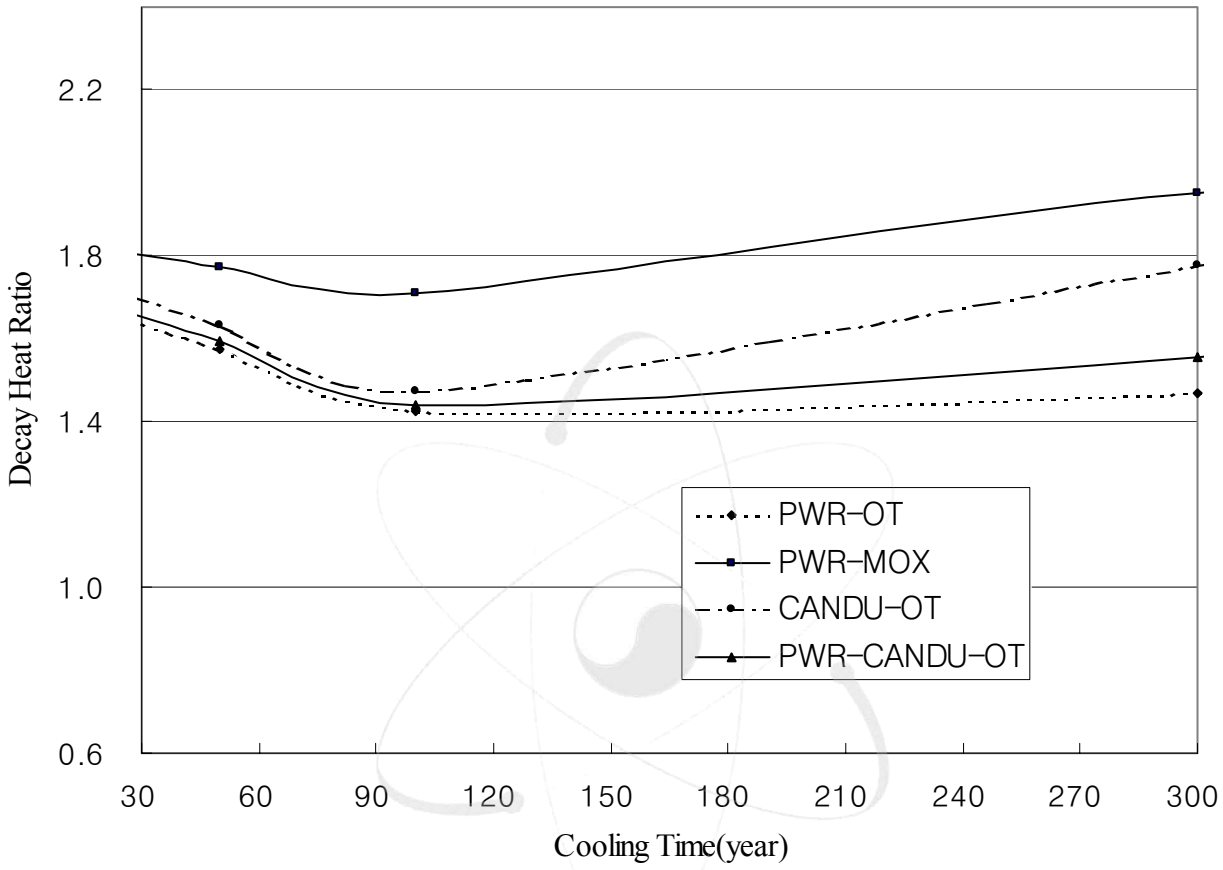



Fig. 10 The Ratio of the Heat Output of Fuel Cycle Options Compared with that of DUPIC Option

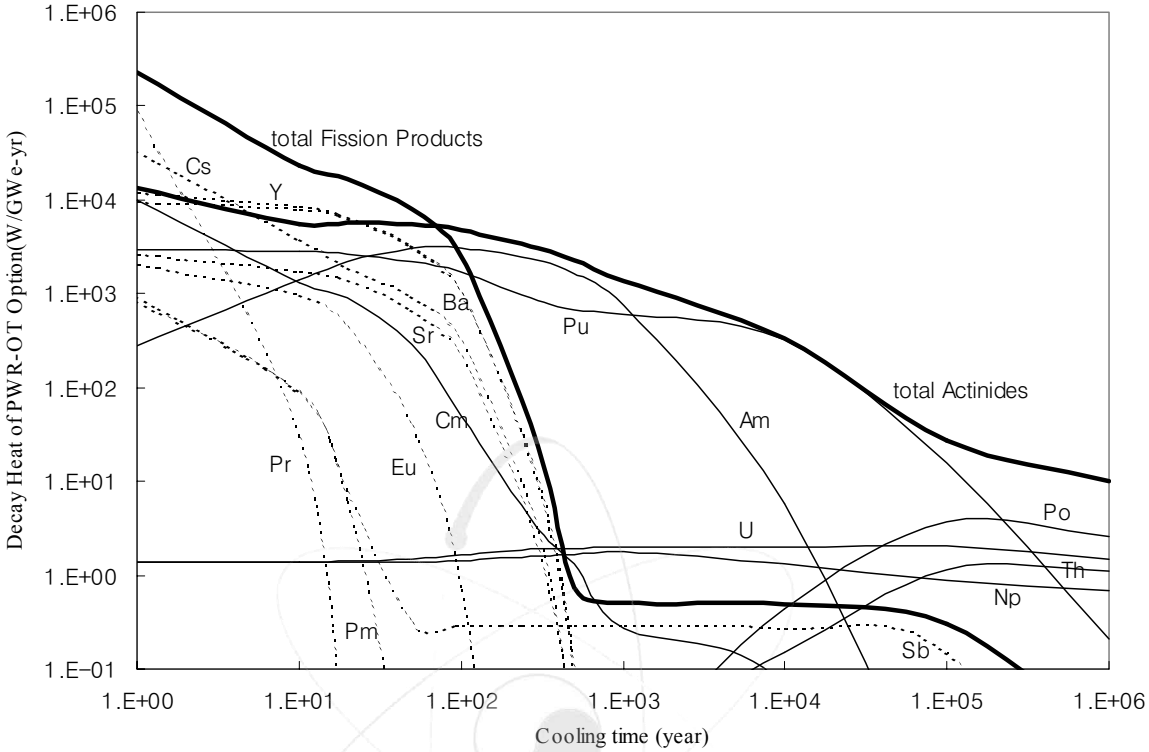



Fig. 11 Isotopes Contribution to the Decay Heat of PWR-OT Option

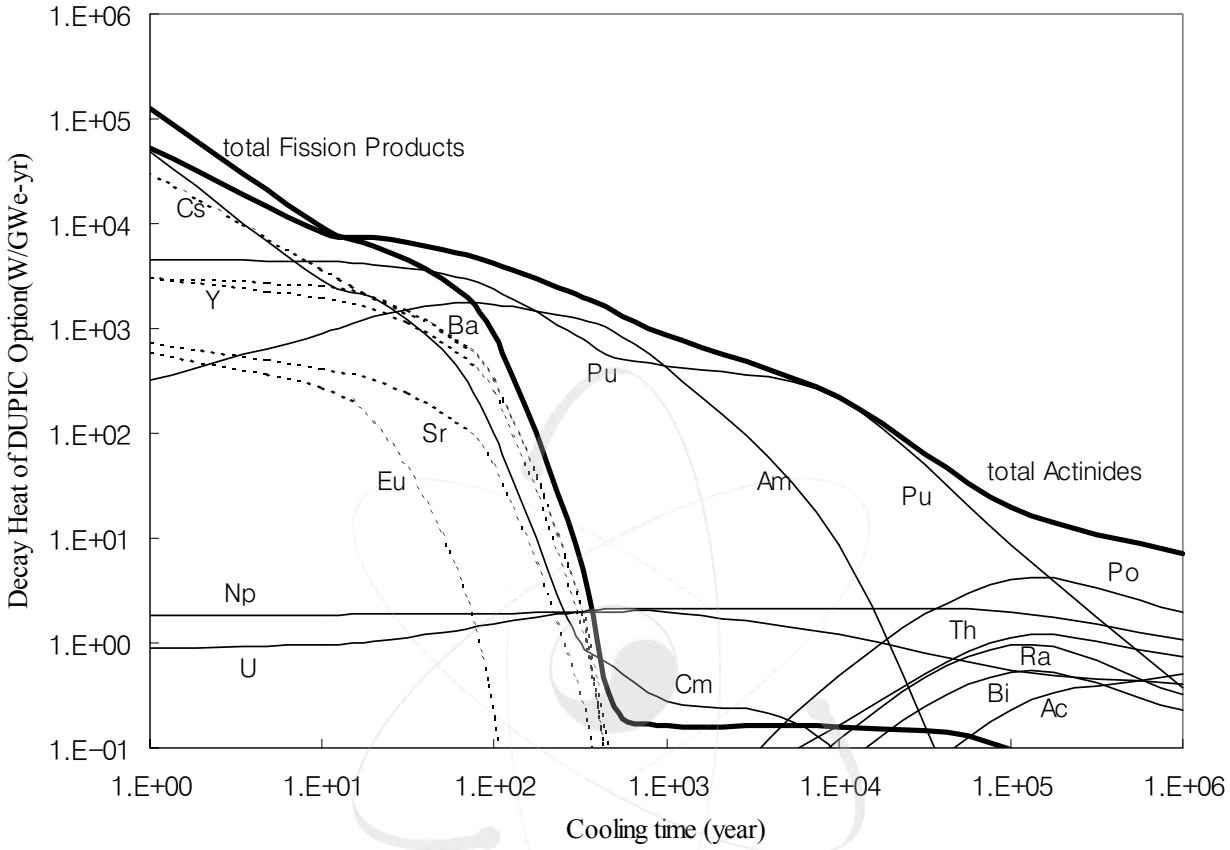



Fig. 12 Isotopes Contribution to the Decay Heat of DUPIC Option

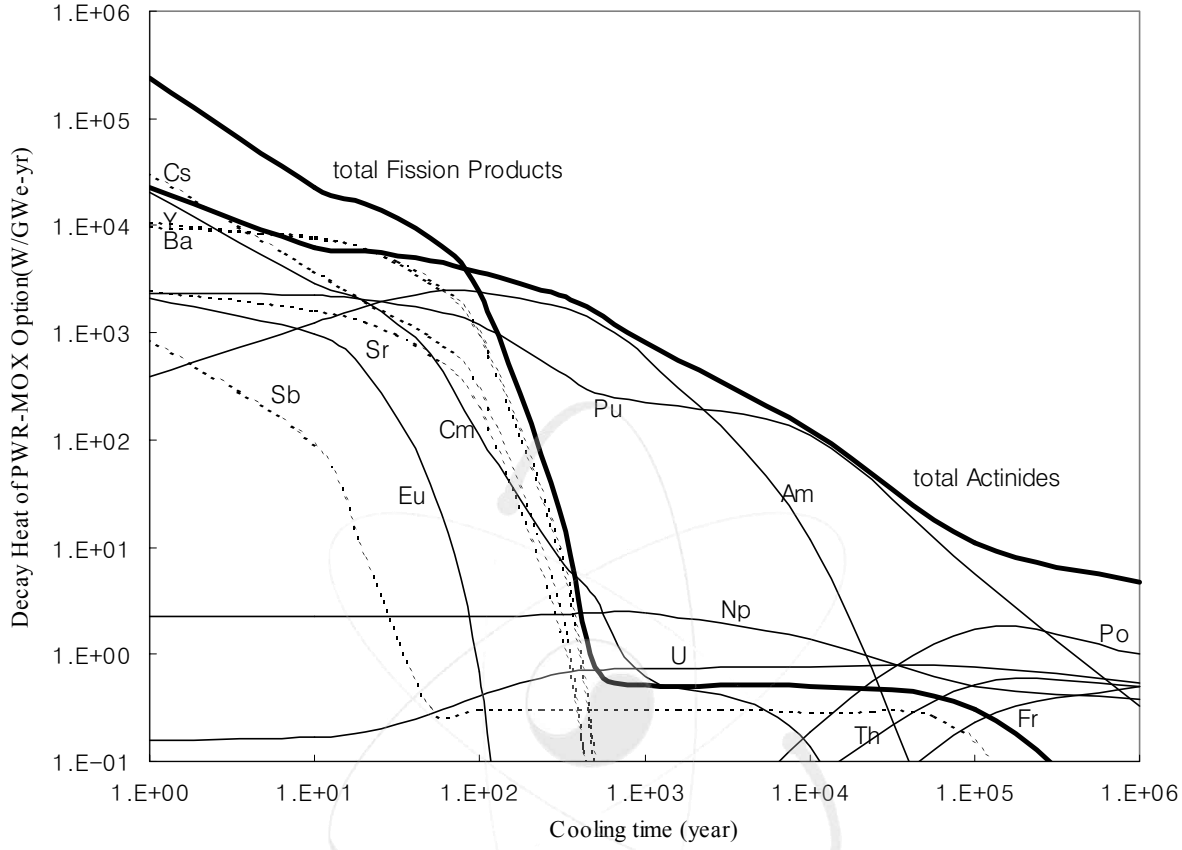



Fig. 13 Isotopes Contribution to the Decay Heat of PWR-MOX Option

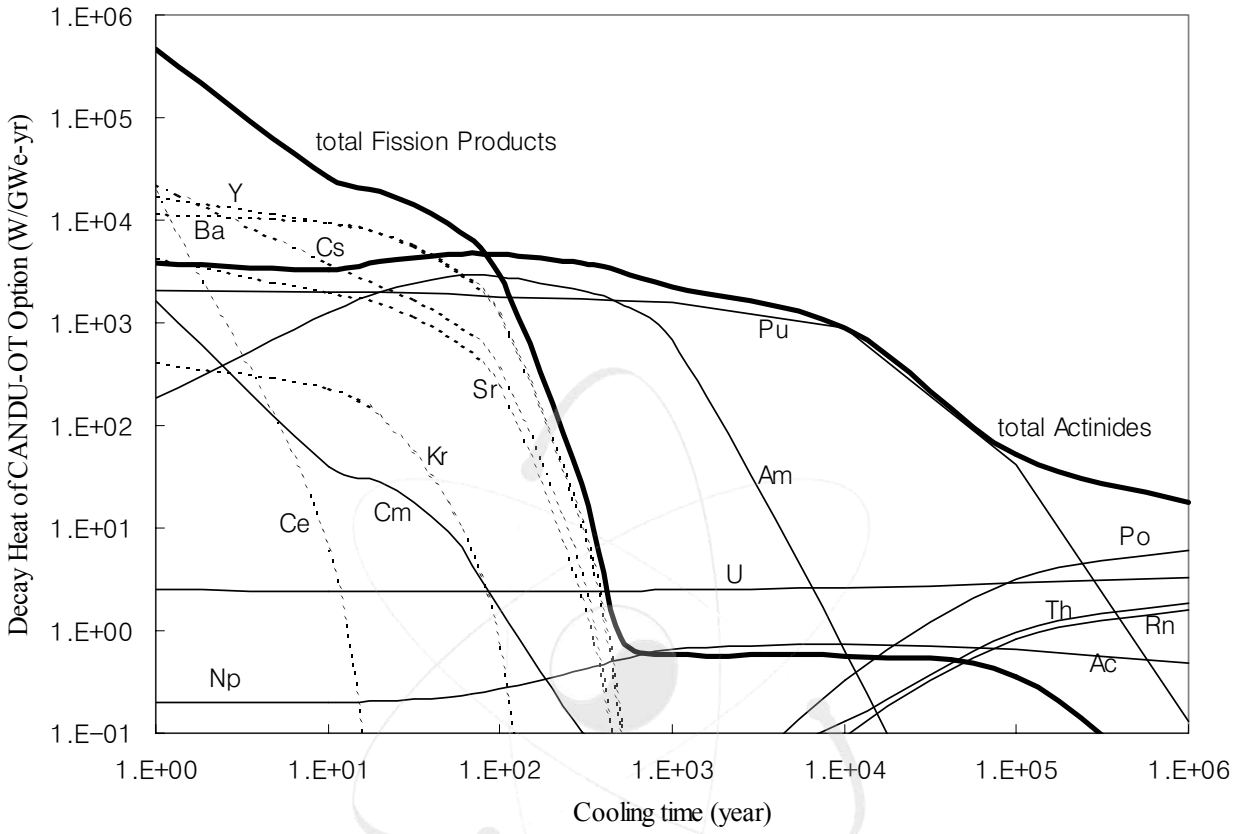



Fig. 14 Isotopes Contribution to the Decay Heat of CANDU-OT Option

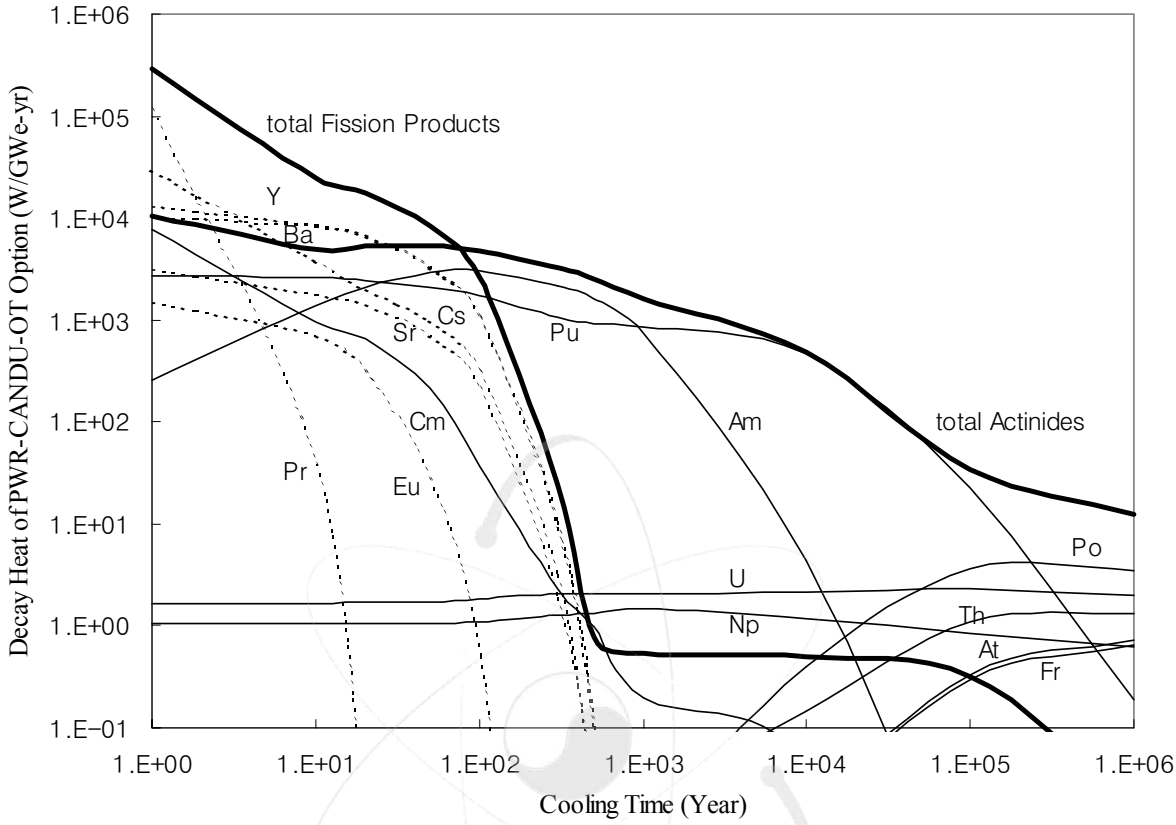



Fig. 15 Isotopes Contribution to the Decay Heat of PWR-CANDU-OT Option

# VI. Activities

The rate of radioactivity disintegration, activity, could be a measure of the importance of waste management in irradiated fuel and in radioactive wastes. That is because the activities of spent fuels could be important for the design of transportation cask, interim storage, final disposal facility and their treatment systems.

In general, $^{137}Cs$ and $^{90}Sr$ are important in determining the radioactivity of fission products after long decay period. However, $^{99}Tc$ contributes to the very long-term radioactivity of stored fission product wastes because the $^{99}Tc$ may be important to the long-term transport of fission products stored in geologic media. For actinide, $^{237}Np$ is an important long-term constituent of radioactive wastes, particularly because its transport through some geologic media is not as delayed as that of other actinides, and because of the toxicity of radionuclide in its decay chain, especially, $^{225}Ra$. Total activities of fuel cycle options are examined and then two isotopes mentioned above are addressed in this study.

Figure 16 shows fission products and actinides activity of various spent fuels as a function of cooling time. On the whole, it is indicated that total activities in spent fuels governed by fission products during first about 100 years and actinides after that period. The CANDU spent fuel has the smallest activity per metric tone for both fission products and actinides. On the order hand, the MOX spent fuel has the largest activity per metric tone for both fission products and actinides. It is shown that activity of the DUPIC fuel is similar to that of PWR spent fuel for actinides, but fission product activity of the DUPIC fuel is a little smaller than to that of PWR spent fuel. The fission products activity of the MOX fuel is similar to that of PWR spent fuel, but actinide activity of the MOX fuel is much higher than to that of PWR spent fuel.

Figure 17 shows the total activity of various fuel cycle options based on 1 GWe-yr. Unlike the Fig. 16, the DUPIC option has the smallest activity per 1 GWe-yr. It is due to the decrease of total weight of spent fuel needed for 1 GWe-yr.

Figure 18 shows activities of two isotopes, $^{99}Tc$ and $^{237}Np$, as a function of cooling time. For $^{99}Tc$, the DUPIC option is containing only about half $^{99}Tc$ of other options. The $^{237}Np$ content of the DUPIC option is also the smallest but the difference is not as big as $^{99}Tc$ case. In conclusion, it seems that the DUPIC option has indirect benefit about safety of long term spent fuel disposal.

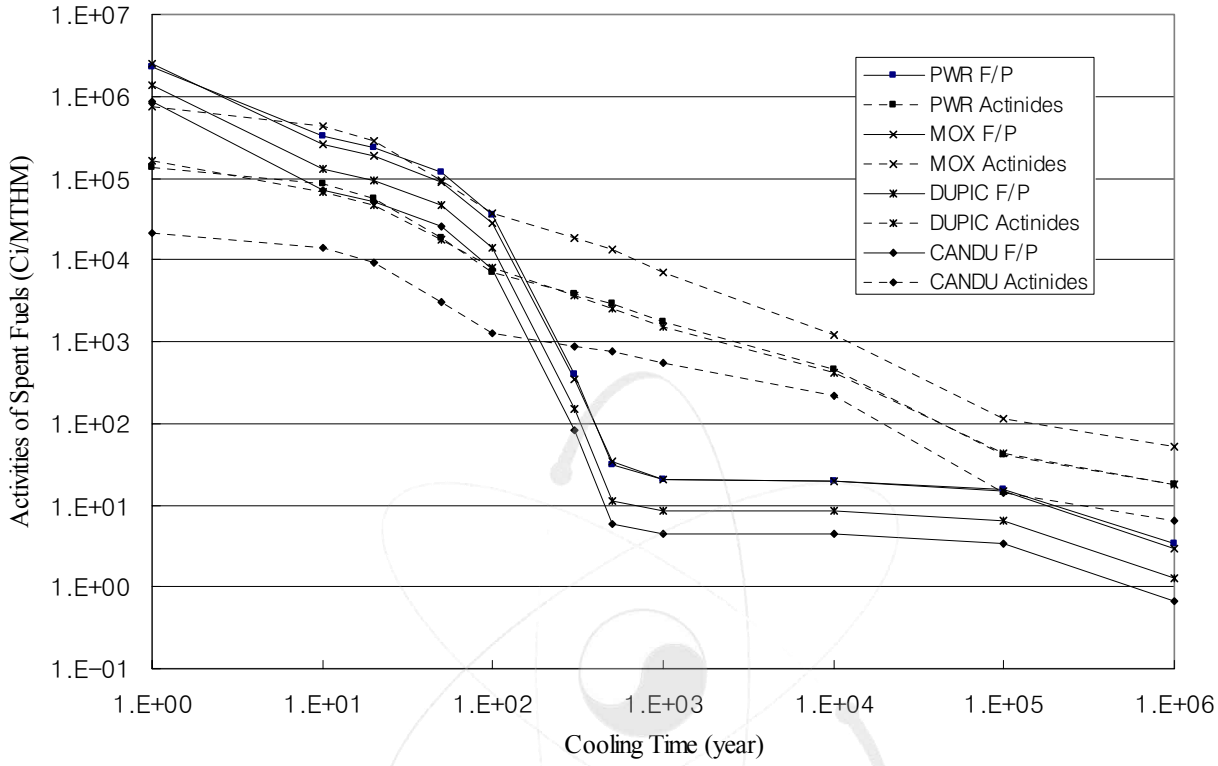



Fig. 16 Activities of Various Spent Fuels as a Function of Cooling Time

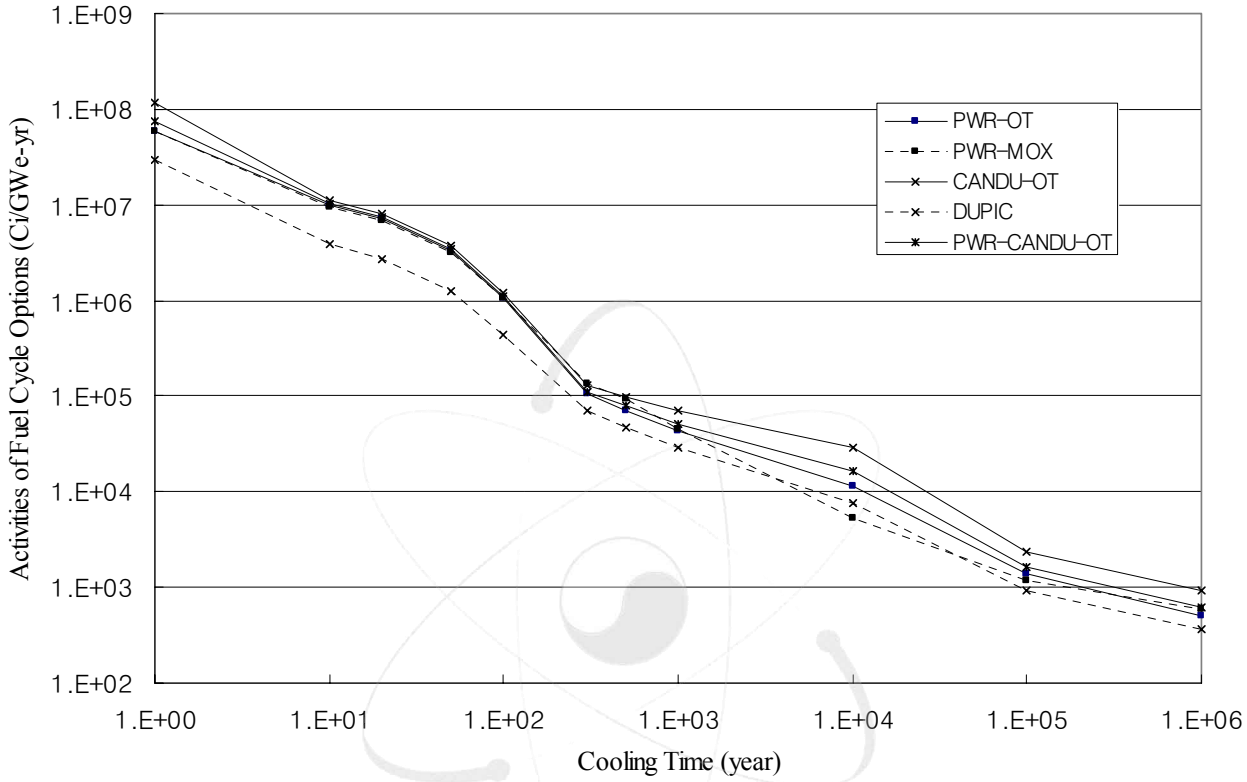



Fig. 17 Activities of Various Fuel Cycle Options (Based on 1GWe-yr)

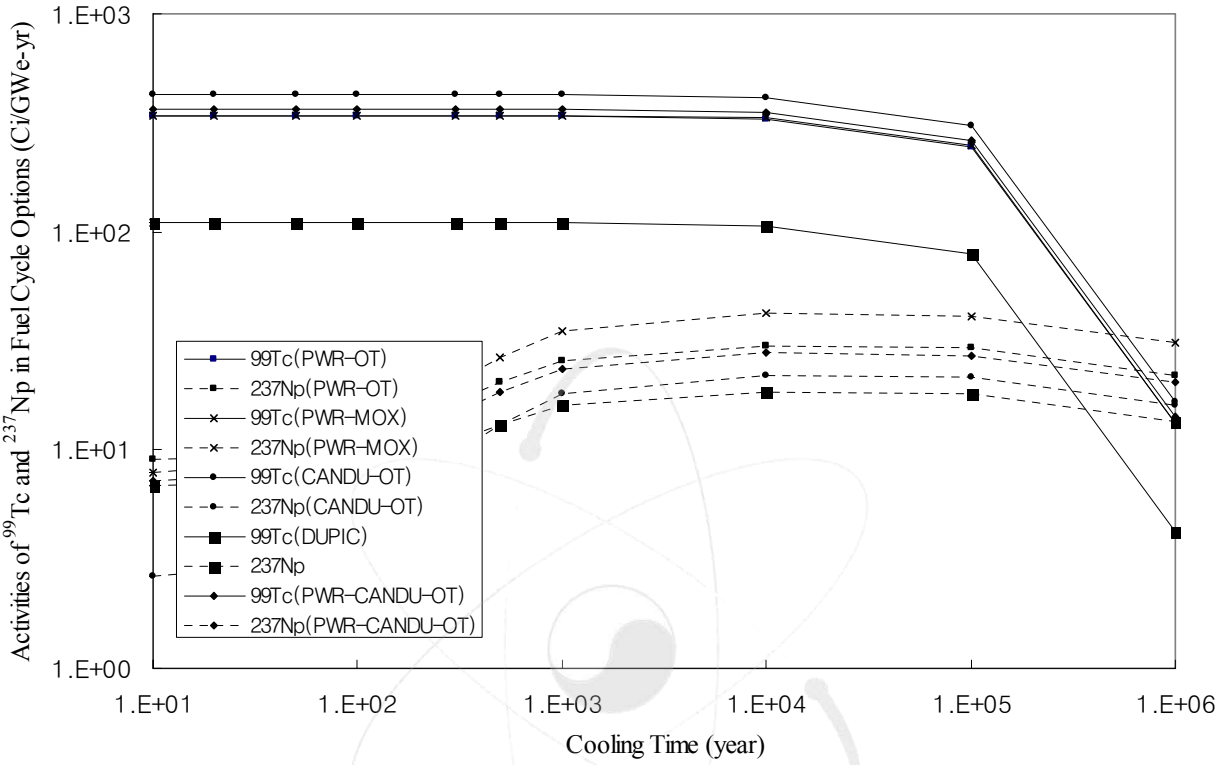



Fig. 18 Activities of $^{99}$Tc and $^{237}$Np in Various Fuel Options

## VII. Radioactive Toxicity

In fact, the activity is only a crude measure of the importance of waste management in irradiated fuel and in radioactive wastes. A more meaningful measure of potential biological hazard must also include the sensitivity of humans to inhalation or ingestion of these radionuclides. For this purpose, Toxicity index has been used as follows[18];

$$\text{Toxicity Index} = \sum_i \frac{\lambda_i N_i}{C_{ik}} \qquad (10)$$

Where $\lambda_i$ = radioactive decay constant for nuclide $i$

$N_i$ = number of atoms of nuclide $i$

$C_{ik}$ = maximum permissible concentration limit for nuclide $i$ in medium $k$ (ie. air or water)

The toxicity index is the volume of air or water with which the mixture of radionuclides must be diluted so that breathing the air or drinking the water will result in the accumulation of radiation dose at a rate no greater than the dose limit. However, the toxicity index still does not measure ultimate hazards and risk, because it does not take into account the mechanisms by which the radionuclides could be released to air or water and transported to humans.

The ingestion toxicity indices are more important than the inhalation indices. Because the actinide and most fission products are nonvolatile and because the wastes are expected to be geologically isolated, ingestion toxicity is probably a more important measure than inhalation toxicity. So only the ingestion toxicity indices are examined in this chapter.

Ingestion toxicity indices for fission products and actinides of PWR spent fuel and DUPIC spent fuel are compared in Fig. 19 and Fig. 20, respectively, as a function of storage time. It is apparent that the relative high toxicity, low C, of bone-seeking Sr-90 makes this nuclide more important than any other fission product in terms of potential ingestion toxicity during the first few hundred years after discharge from the reactor. Thereafter, the long-lived thyroid-seeking $^{129}$I is potentially the importance of the fission products. During the first 300 years total toxicity index are governed by the fission products, mainly $^{90}$Sr. It is controlled by $^{241}$Am and $^{243}$Am from about 300 years to about 2000 years, followed successively by $^{239}$Pu and $^{240}$Pu from 2000 years to 80,000 years. Subsequently, the most important radionuclide is $^{226}$Ra, which is formed from the decay of $^{234}$U, $^{238}$Pu, $^{242m}$Am, $^{242}$Cm and $^{238}$U. Compared ingestion toxicity of the DUPIC spent fuel with the PWR case, the DUPIC case is a little higher than the PWR case for

actinide toxicity but is a little lower than PWR case for fission product toxicity. Therefore, for short term(about 200 years) when is governed by fission products, the DUPIC case is a little lower than PWR case. On the other hand, for long term when is governed by actinides, the PWR case is a little lower than the DUPIC PWR case.

Fig. 21 compares the ingestion toxicity based on metric tone heavy metal of spent fuel and HLWs. As opposed to the activity and decay heat, the ingestion index of the MOX fuel is lower than PWR spent fuel case during the period governed by fission products, but much higher during the period governed by actinides. It seems that recycling plutonium increases the production of americium and curium by about half order of magnitude during the period governed by actinides and $^{226}$Ra. In the meanwhile, the ingestion index of the DUPIC spent fuel is lower than PWR spent fuel case during the period governed by fission products, but a little higher or similar during the period governed by actinides.

The ingestion toxicity index for five fuel cycle options is compared based on 1 GWe-yr in Fig. 22. Although the true hazards of radioactive wastes are not measured by these toxicity indices, some prospective can be obtained by comparing the total ingestion toxicity index to the similar toxicity index for the ore used to fuel the reactor to generate these wastes. The ore toxicity is due mainly to the $^{226}$Ra, which is in secular equilibrium. The ingestion index in Fig. 20 are also compared with the toxicity of uranium ore mined for one GWe-year of reactor operation. Since the ore toxicity is due mainly to $^{226}$Ra, which is in secular equilibrium in the $^{238}$U decay chain, we can calculate the ore toxicity. At secular equilibrium, the activity of $^{226}$Ra and $^{238}$U are the same. So the activity is

$$\frac{(0.9927^{238}U/U)(10^6 g/MT)(6.0225\times10^{23}/g-atom)(0.693)}{(238gU/g-atom)(4.51\times10^9 yr)(3.154\times10^7 s/yr)[3.7\times10^{10}/(Ci.s)} = 0.33Ci/MTU.$$

The maximum permissible concentration limit($C_{ik}$) for $^{226}$Ra is $3\times10^{-8}\mu Ci/cm^3$ [19]. From the material flow of Fig. 2, the uranium ore of 2.465 lb$U_3O_8$ mined for PWR fuel with one GWe-year can be converted to about 1210 MTU. Therefore, the toxicity of uranium ore mined for one GWe-year of PWR reactor operation by the equation 10 becomes

$$\frac{(1210MTU)(0.33\times10^6\mu Ci/MTU)}{(3\times10^{-8}\mu Ci/cm^3)(1\times10^6 cm^3/m^3)} = 1.33\times10^{10} m^3.$$

The toxicity of uranium ore for all fuel cycle options is compared in Table 9. From Fig. 22, the ingestion toxicity of each fuel cycle option decays to a level below that of the initial ore after period of about 600 ~ 3000 years. It ultimately decays to toxicity that is a fraction of a

percent of the toxicity of the original ore consumed to generate these wastes. From Fig. 22, it is indicated that the DUPIC option and the PWR-MOX option decay to a level below that of the initial ore after about 600 years and 3000 years, respectively.

It is likely that the long-term hazards from geologically isolated high-level wastes will be less than those already experienced due to the naturally occurring uranium mineral s. The period of greatest importance in high level waste management is probably the earlier, 600 ~ 3000 year. In conclusions, up to that period, the toxicity of the DUPIC option is much smaller than other fuel cycle option.

Table 9 Toxicity of Uranium Ore for Fuel Cycle Options

| Fuel Cycle Options | Ore Toxicity for 1GWe-year ($m^3$ water) |
|---|---|
| PWR-OT | 1.33E+10 |
| PWR-MOX | 9.07E+09 |
| CANDU-OT | 1.00E+10 |
| DUPIC | 9.75E+09 |
| PWR-CANDU-OT | 1.25E+10 |

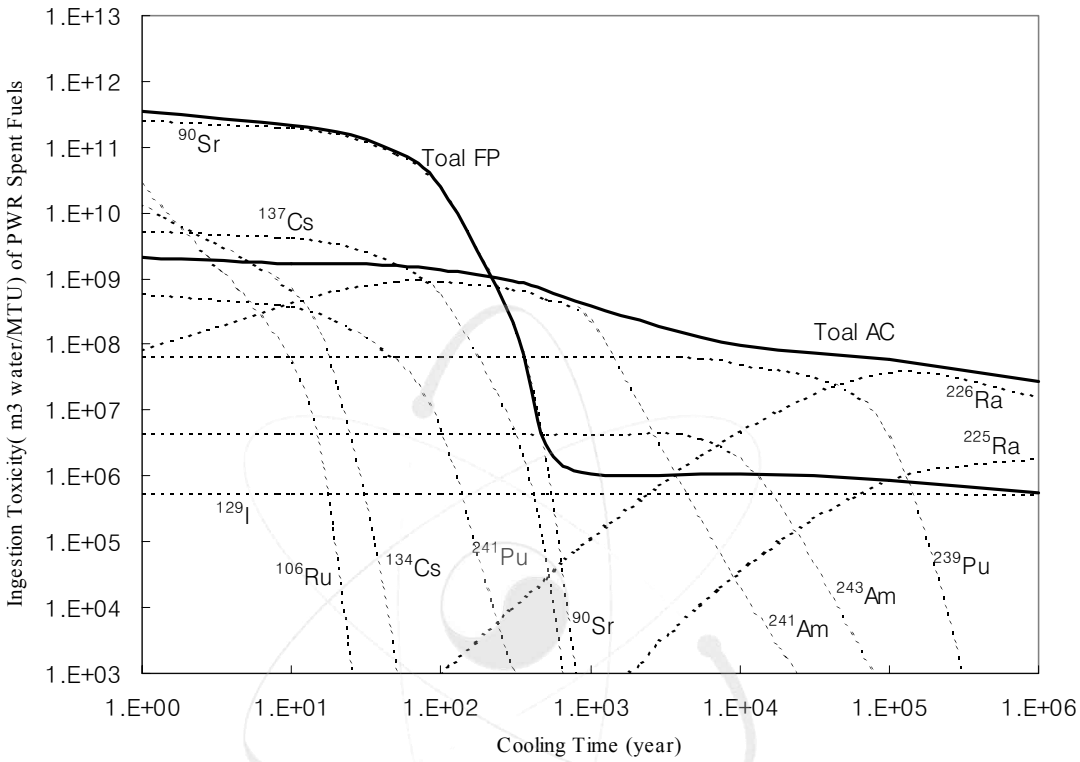



Fig. 19 Isotopes Contribution to the Long Term Ingestion Hazard Index of PWR Spent Fuels

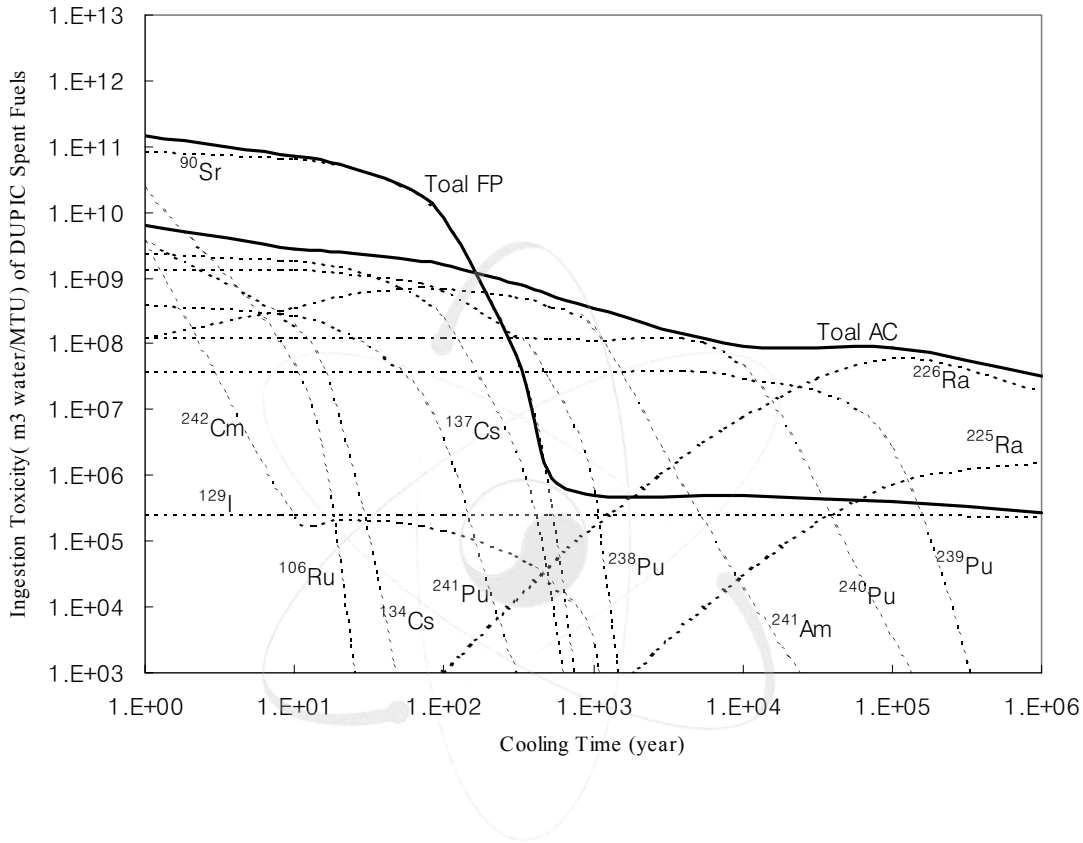



Fig. 20 Isotope Contribution to the Long Term Ingestion Hazard Index of DUPIC Spent Fuels

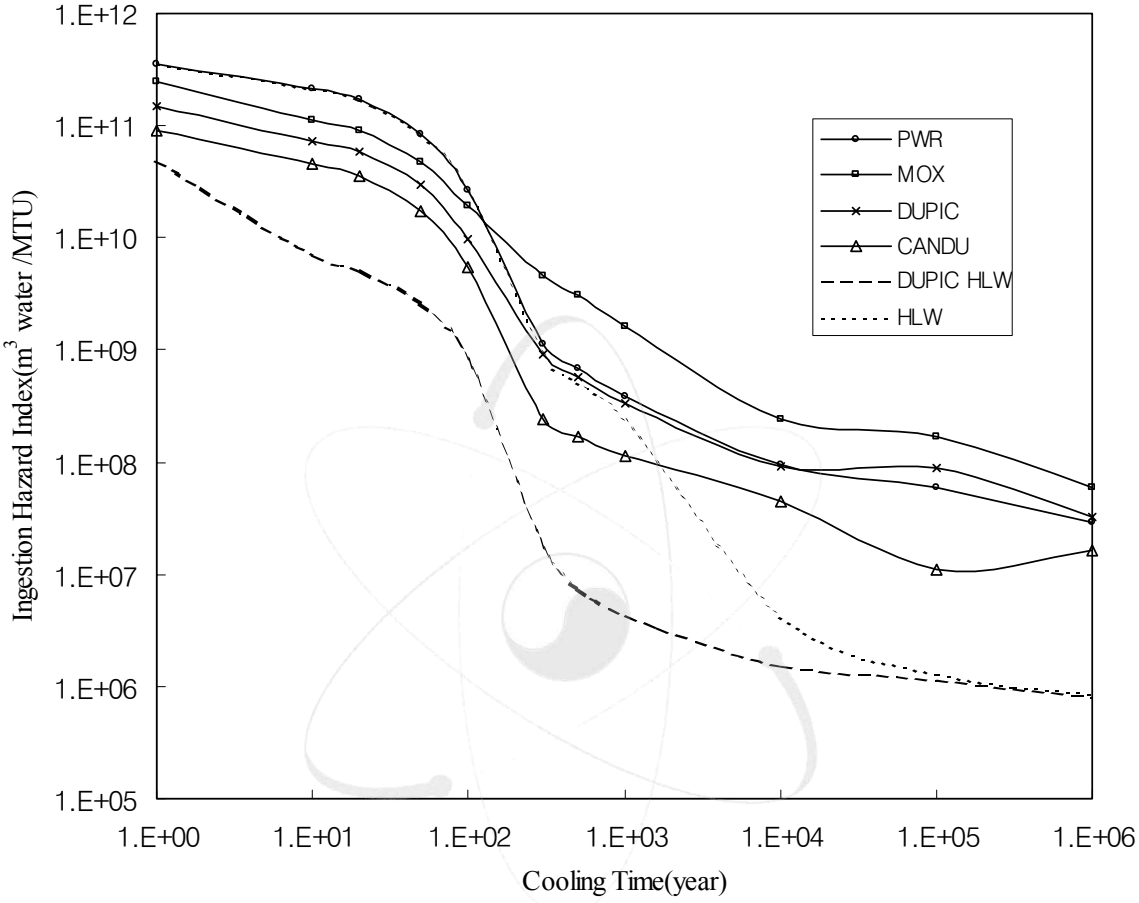



Fig. 21 Ingestion Hazard Index for Various Spent Fuels and HLWs

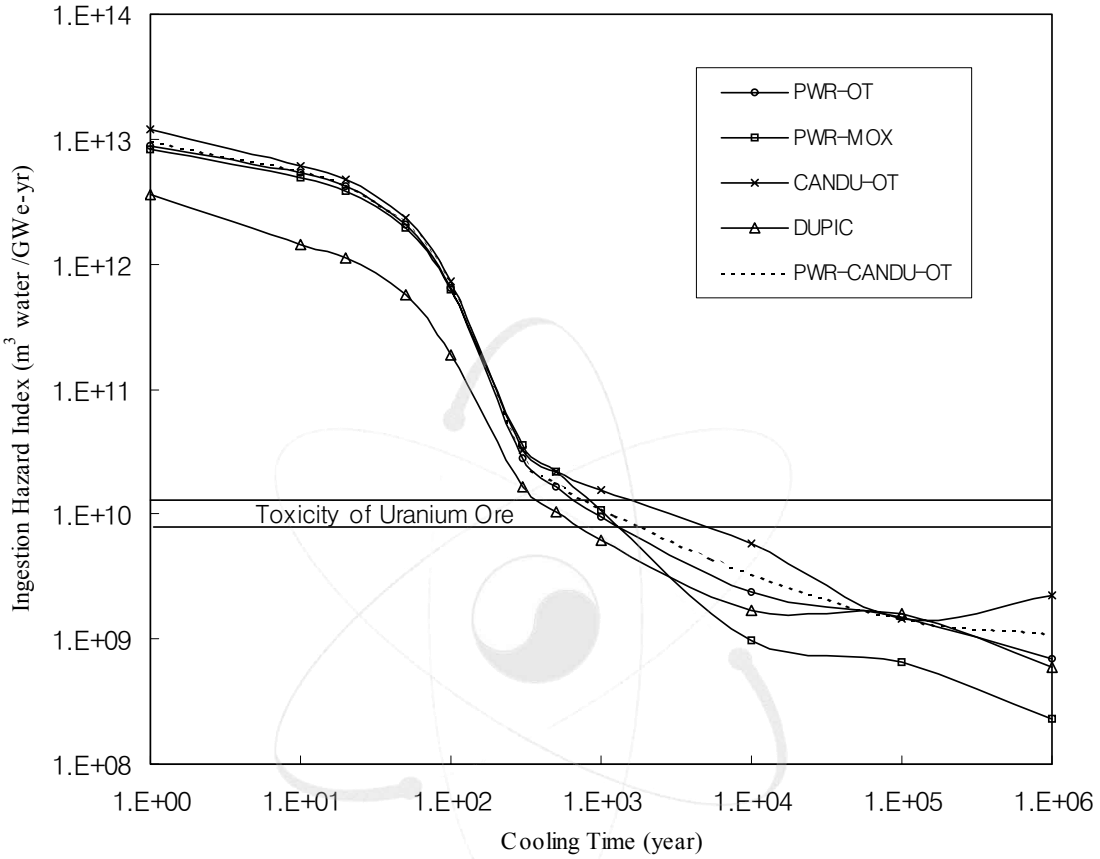



Fig. 22 Ingestion Hazard Index for Various Fuel Cycle Alternatives (Based on 1 GWe-yr)

# VIII. Conclusion

This study compares some properties of irradiated fuel generated from alternative fuel cycles(DUPIC, CANDU-OT, PWR-OT, PWR-MOX and PWR-CANDU-OT) that generated the same amount of electricity.

It was found from the actinides content estimation that the MOX option has the lowest plutonium arising based on 1 GWe-year and followed by the DUPIC option. However, the fissile Pu content generated in the DUPIC fuel is shown to be the lowest among the fuel cycle options. On the whole, the CANDU-OT option has the largest fissile plutonium as well as gross plutonium, which means negative points in nuclear proliferation resistance aspects.

It was indicated from the radio-toxicity analysis that the toxicity of the DUPIC option based on 1 GWe-yr is much smaller than those of other fuel cycle options, and is just about half order of magnitude of other fuel cycles until decayed to a level below toxicity of initial ore. It means that the DUPIC option could have indirect benefit on safety of long term spent fuel disposal.

It was indicated from the decay heat analysis that for the first 300 years the time PWR-MOX option has 1.7~2.0 times higher heat output and PWR-CANDU-OT option has 1.4~1.7 times higher heat output, compared to DUPIC option case.

From total activity analysis of various fuel cycle options, the activity per metric ton heavy metal of spent fuel is the lowest in natural uranium CANDU fuel, but, in case of activity based on 1 GWe-yr, the DUPIC option has the smallest activity. It seems to be due to the decrease in the total weight of spent fuel needed for 1 GWe-yr. In the meanwhile, from the activity analysis of $^{99}Tc$ and $^{237}Np$, which are important to the long-term transport of fission products stored in geologic media, the DUPIC option, was being contained only about half of those other options.

It was found from the actinide content estimation that the MOX option has the lowest plutonium arising based on 1 GWe-year and followed by the DUPIC option. However, fissile Pu content generated in the DUPIC fuel was the lowest among the fuel cycle options. On the whole, the CANDU-OT option has the largest fissile plutonium as well as gross plutonium based on 1 GWe-year, which means negative points in nuclear proliferation resistance aspects. In conclusions, the irradiated DUPIC fuels would have good properties on waste management and proliferation resistance, compared to other fuel cycle cases.

In conclusions, the irradiated DUPIC fuels would have good properties on waste management and proliferation resistance, compared to other fuel cycle cases.

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Abstract (15-20 Lines)

Some properties of irradiated DUPIC fuels are compared with those of other fuel cycles. It was indicated that the toxicity of the DUPIC option based on 1 GWe-yr is much smaller than those of other fuel cycle options, and is just about half the order of magnitude of other fuel cycles. From the activity analysis of $^{99}Tc$ and $^{237}Np$, which are important to the long-term transport of fission products stored in geologic media, the DUPIC option, was being contained only about half of those other options. It was found from the actinide content estimation that the MOX option has the lowest plutonium arising based on 1 GWe-year and followed by the DUPIC option. However, fissile Pu content generated in the DUPIC fuel was the lowest among the fuel cycle options. From the analysis of radiation barrier in proliferation resistance aspect, the fresh DUPIC fuel can play a radiation barrier part, better than CANDU spent fuels as well as fresh MOX fuel. It is indicated that the DUPIC fuel cycle has the excellent resistance to proliferation, compared with an existing reprocessing option and CANDU once-through option. In conclusions, DUPIC fuel cycle would have good properties on environmental effect and proliferation resistance, compared to other fuel cycle cases.

| Subject Keywords (About 10 words) | DUPIC Fuel Cycle, Proliferation Resistance, Environmental Effect, Spent Fuel Radio-toxicity, Spent Fuel Activity |
|---|---|

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| 연구위탁기관 | | 계약 번호 | |

초록 (15-20줄내외)

DUPIC 핵연료주기의 환경영향을 평가하기 위하여 DUPIC 사용후핵연료에 대한 여러 특성을 다른 핵연료주기의 핵연료들과 비교 평가하였다. 방사성 독성도 평가에서 DUPIC 사용후핵연료는 다른 핵연료주기에 비해 단위전력 생산량당 독성도가 가장 작은 것으로 나타났다. 지하 처분장에서 환경에 영향을 미치는 주요 동위원소인 $^{99}$Tc 과 $^{237}$Np의 방사능 정도를 단위전력 생산량당 평가한 결과 DUPIC 주기는 다른 주기의 1/2 정도에 불과한 것으로 평가되었다. 한편 단위전력 생산량당 플루토늄 생성량은 MOX 주기가 가장 작은 것으로 평가되었으며, DUPIC 주기는 MOX 보다는 약간 많지만 다른 직접처분 주기들 보다는 매우 작게 생산되는 것으로 나타났다. 핵확산 저항성 관점에서 핵연료의 방사선 장벽 정도를 평가한 결과 신DUPIC 핵연료는 사용후 CANDU 핵연료나 신 MOX 핵연료 보다도 높은 방사선량을 갖는 것으로 나타났으며, 이 양은 핵확산저항성 관점에서 방사선 장벽으로서의 역할을 충분히 할 것으로 평가되었다. 결론적으로 DUPIC 핵연료주기는 환경영향 측면뿐만 아니라 핵확산저항성 관점에서 타핵연료주기에 비해 우수한 것으로 나타났다.

주제명키워드 (10단어내외) 듀픽 핵연료주기, 핵확산 저항성, 핵연료주기의 환경 영향, 사용후핵연료의 붕괴열, 사용후핵연료의 방사능, 사용후핵연료의 선량, 독성도