Roadmap Design for Thorium-Uranium Breeding Recycle in PWR

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Abstract. The paper was focused on designing a roadmap to finally approach sustainable Thorium-Uranium (^{232}Th-^{233}U) Breeding Recycle in current PWR, without any other change to the fuel lattice and the core internals, but substituting the UOX pellet with Thorium-based pellet. At first, the paper presented some insights to the inherence of Thorium-Uranium fuel conversion or breeding in PWR based on the neutronics theory and revealed the prerequisites for Thorium-Uranium fuel in PWR to achieve sustainable Breeding Recycle; And then, various Thorium-based fuels were designed and examined, and the calculation results further validated the above theoretical deductions; Based on the above theoretical analysis and calculation results, a roadmap for sustainable Thorium-Uranium breeding recycle in PWR was outlined finally.

1. INTRODUCTION

Thorium is about three times more abundant in nature compared to uranium and exists mainly as fertile isotope ^{232}Th. It is well-known that ^{232}Th can be converted to man-made fissile isotope ^{233}U with higher conversion ratio or even breeding in thermal reactor while the fuel lattice or the core is well designed.

Nevertheless, if the conversion of ^{232}Th to ^{233}U is only driven by other fissile isotopes (so-called "seeds" , such as ^{235}U or ^{239}Pu), it will be difficult to maintain a sustainable fuel recycle; and if there is any redesign on the existing fuel lattice or the core structure, the plan shall become quite difficult for implementation. Therefore, the most attractive case is to achieve sustainable Thorium-Uranium (^{232}Th-^{233}U) Breeding Recycle in existing thermal reactor without any change to the design of fuel lattice or core internals.

The paper is focusing on designing a roadmap to finally approach sustainable Thorium-Uranium (^{232}Th-^{233}U) Breeding Recycle in current PWR, without any other change to the fuel lattice and the core internals, but substituting the UOX pellet with thorium-based pellet.

At first, the paper will present some insights to the inherence of Thorium-Uranium fuel conversion or breeding in PWR from the view of neutronics theory in Section 2; and then a series of calculations are performed and summarized in Section 3 for Thorium-based fuels blended with various fissile seeds, such as Medium Enriched Uranium (MEU), reactor-grade plutonium and reactor-grade ^{233}U. The calculation results will demonstrate that it is possible to achieve sustainable Thorium-Uranium (^{232}Th-^{233}U) Breeding Recycle in current PWR. Based on the above theoretical analysis and calculation results, Section 4 will present a roadmap to finally approach sustainable Thorium-Uranium breeding recycle in PWR.

2. INSIGHTS TO THE INHERENCE OF THORIUM-BASED FUEL IN PWR

The chemical performance of natural Thorium is quite stable; Thorium dioxide is the main compound of the natural Thorium. The half lives of other Thorium sisters are quite short, except for ^{232}Th, whose half life is about 3 times of ^{238}U, therefore, the natural Thorium is almost pure ^{232}Th. The threshold energy of ^{232}Th fission is quite high (>1.4Mev) and the fission cross section is smaller, therefore, Thorium in reactor mainly works as fertile isotope rather than fissile material. ^{232}Th converts into ^{233}Th with Gamma ray emitted after one neutron absorbed, ^{233}Th normally emits an electron by β^- decay to

become protactinium-233 (^{233}Pa), and then, ^{233}Pa emits another electron by a second β^- decay to become ^{233}U, which is nice fissile material and can provide around 2.3 fission neutrons for next round fuel transmutation. Therefore, the performance of the Thorium-based fuel in reactor core is strongly dependent on the synthetical performance of each isotope, especially ^{232}Th and ^{233}U, in this transmutation process, which is illustrated in Fig 1.

$$^{232}_{90}Th + ^{1}_{0}n \rightarrow ^{233}_{90}Th + \gamma \downarrow$$
$$^{233}_{90}Th \xrightarrow[22\,\mathrm{min}]{\beta^-} ^{233}_{91}Pa \xrightarrow[27day]{\beta^-} ^{233}_{92}U + ^{1}_{0}n \rightarrow FPs + \sim 2.30\,^{1}_{0}n$$

Fig. 1. Isotopes transmutation process for Thorium-based fuel

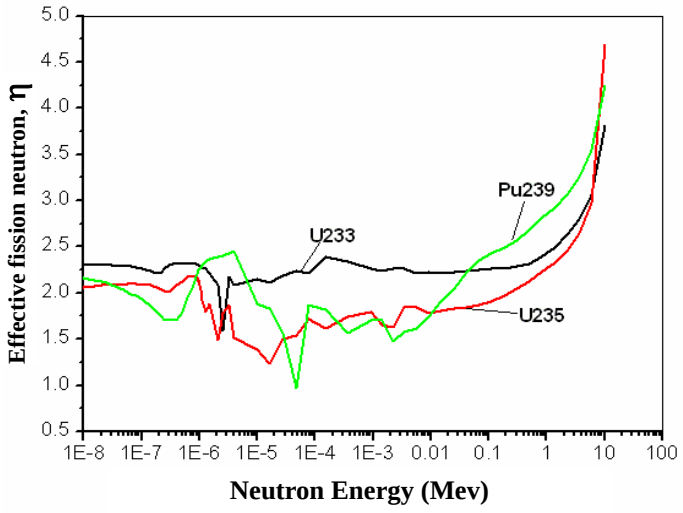

Fig. 2. Effective fission neutrons vs. neutron energy for ^{233}U, ^{235}U and ^{239}Pu

The fission performance of ^{233}U is quite excellent [1, 2]. FIG.2 compared the effective fission neutron number (η) of ^{233}U vs. neutron energy with ^{235}U and ^{239}Pu, which indicates that the η of ^{233}U is bigger than the η of ^{235}U and ^{239}Pu within a quite wide energy span from thermal to epithermal energy region. This is also the well-known evidence that Thorium-based fuel can achieve higher conversion ration or even breeding in thermal reactor. Nevertheless, it is emphasized here that the excellent fission performance of ^{233}U does not only exist in thermal energy group but also epithermal or even beyond epithermal energy group (vs. ^{235}U).

Theoretically [3], the conversion or breeding ratio (CBR) of nuclear fuel is defined as follows:

$$CBR(t) = \frac{capture \cdot rate \cdot of \cdot fertiles}{absorption \cdot rate \cdot of \cdot fissiles} \qquad (1)$$

As for the Thorium-based fuel, the CBR can be expressed as following form:

$$CBR(t) = \frac{\int N^{Th232}(t)\sigma_c^{Th232}(E)\phi(E,t)dE}{\int N^{U233}(t)\sigma_a^{U233}(E)\phi(E,t)dE} = \frac{N^{Th232}(t)\int \sigma_c^{Th232}(E)\phi(E,t)dE}{N^{U233}(t)\int \sigma_a^{U233}(E)\phi(E,t)dE} \qquad (2)$$

Formula (2) states that the conversion or breeding ratio is dependent on the number densities of ^{232}Th and ^{233}U, the neutron spectra in the fuel lattice, the ^{232}Th capture cross section and ^{233}U absorption cross section vs. neutron energy.

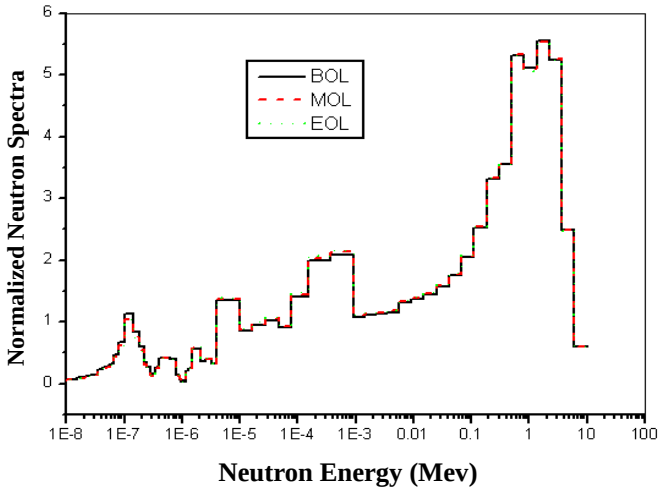

Fig. 3. Neutron Spectra in a typical PWR fuel lattice

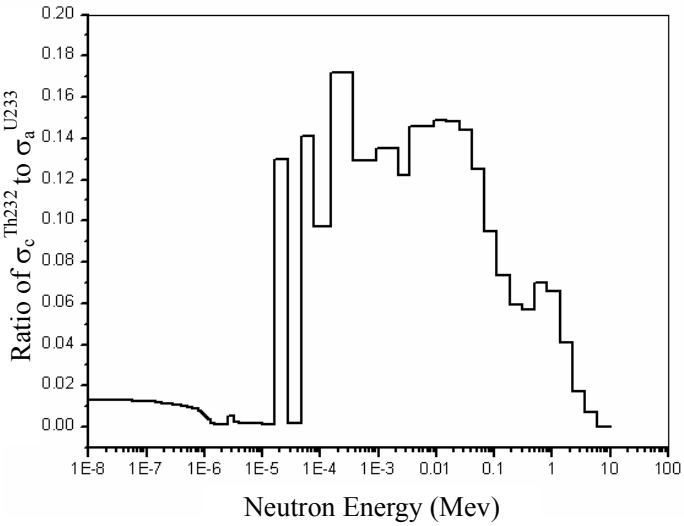

Fig. 4. The ratio of ^{232}Th capture cross section to ^{233}U absorption cross section vs. neutron energy

The neutron spectra in fuel lattice are strongly dependent on the core type, such as thermal reactor or fast reactor; but, for a typical PWR core, the neutron spectra is almost constant, see FIG. 3, which illustrated the spectra of a PWR fuel lattice at BOL, MOL and EOL respectively. We can see that the variation of the spectra is almost neglectable.

It can be seen from formula (2) that CBR(t) has an implicit relation with the ratio of ^{232}Th capture cross section to ^{233}U absorption cross section, which is only dependent on the inherence of these two nuclides and independent to any core type. FIG. 4 illustrated the ratio of σ_c^{Th232} to σ_a^{U233} vs. neutron energy. It can be seen that the ratio of σ_c^{Th232} to σ_a^{U233} around thermal group is far smaller than the ratio around epithermal group. Since this distribution is almost constant, combined FIG. 4 and formula (2), we can deduce that:

- The conversion of ^{232}Th to ^{233}U in Thorium-based fuel is mainly dominated by epithermal neutrons;
- The hardener neutron spectra in thermal reactor is beneficial to improve the conversion or breeding ratio in Thorium-based fuel;
- The design of pure Thorium fuel rod in moderator region is not a good choice from the

viewpoint of isotopes conversion in Thorium-based fuel; the design of thorium blended with other fissile seeds (such as ^{233}U, ^{235}U or ^{239}Pu) is beneficial to enhance the conversion or breeding ratio in Thorium-based fuel.

Since the neutron spectra in a given fuel lattice is almost changeless and the Ratio of σ_c^{Th232} to σ_a^{U233} vs. neutron energy is also a constant distribution, for a given PWR fuel lattice, now we can set the ratio of the two integrations in formula (2) as a constant, which is the Conversion or Breeding Performance Index (CBPI) for a given fuel lattice, i.e.

$$CBR(t) = CBPI \cdot \frac{N^{Th232}(t)}{N^{U233}(t)}, \qquad CBPI = \frac{\int \sigma_c^{Th232}(E)\phi(E,t)dE}{\int \sigma_a^{U233}(E)\phi(E,t)dE} \qquad (3)$$

As we know, CBR(t) > 1 means the fuel has achieved breeding, while CBR(t) =1 means the inventory of fissile isotope is in a state of quasi-equivalence, where the production rate of fissile isotopes is equal to the loss rate. Now, let set CBR(t) =1, then formula (3) becomes,

$$1 = CBPI \cdot \frac{N^{Th232}(t)}{N^{U233}(t)}, \quad or \quad \frac{N^{U233}(t)}{N^{Th232}(t)} = CBPI \qquad (4)$$

Formula (4) indicates that the prerequisite to achieve quasi-equilibrium state for Thorium-based fuel is that the ratio of ^{233}U inventory to ^{232}Th inventory must be equal to the CBPI of the fuel lattice. For the typical PWR fuel lattice, calculation result shows that the CBPI is around 0.02. Combined formula (3) and formula (4), now we can deduce that:

- The prerequisite for Thorium-Uranium fuel breeding cycle is that the ratio of ^{233}U inventory to ^{232}Th inventory must be less CBPI (~0.02), otherwise, the CBR(t) should be less than 1, it is impossible to maintain sustainable fuel cycle;
- For the Thorium-based fuel using ^{233}U as seeds, if the ratio of ^{233}U inventory to ^{232}Th inventory is greater than CBPI (typically 0.02, most of other authors used 0.04~0.05 for PWR), the fuel system could not achieve breeding, the extra ^{233}U is mainly contributing fission energy;
- For the Thorium-based fuel using other fissile isotopes (such as ^{235}U or ^{239}Pu) as seeds, since the initial ^{233}U inventory is almost zero, the CBR(t) will be much bigger than 1 at early stage, then CBR(t) will approach 1 along with the burnup accumulated, and the maximum ^{233}U inventory will be the product of CBPI and instant ^{232}Th inventory;
- As the purity of the seeds is varied evidently, e.g. most of the composition in MEU is ^{238}U or there are some non-fissile isotopes in reactor-grade Plutonium, for a given fuel lattice, the initial ^{232}Th inventory shall be varied a lot, and then the final ^{233}U inventory will be also quite different.

Now, we can find that CBPI is actually a threshold value for the ratio of ^{233}U inventory to ^{232}Th inventory, which identifies the conversion or breeding capability in a given fuel lattice.

All above analysis assumed that ^{232}Th is converted into ^{233}U directly after neutron capture. Actually, the direct daughter of ^{232}Th neutron capture is ^{233}Th, which has another branch to capture neutron and become ^{234}Th (and then ^{234}U). Therefore, the real CBPI in Thorium-based fuel shall be a little bit less than the estimated value (typically 0.02). Nevertheless, this discrepancy will not affect all above deductions, and the calculation results in Section 3 will further validate these deductions.

3. CALCULATION RESULTS EVALUATION FOR THORIUM-BASED PWR FUEL ASSEMBLY

Since ^{233}U does not exist in nature, it is not like Uranium-Plutonium fuel cycle, in which the naturally existing ^{235}U works as seeds to drive the fuel conversion, Thorium-based fuel need extra seeds to start the fuel cycle initially. Generally, Thorium-based fuel can have two forms in reactor core: Discrete and Integral.

The discrete can be discrete fuel rod or discrete fuel assembly, which means that the Thorium-based fuel is laid in moderator region. Based on the previous analysis, the discrete design is not beneficial to the fuel conversion and breeding; on the other hand, since the reactivity of the discrete fuel is almost zero at the beginning, it will also be a bigger challenge to controlling the core power distribution. So, our calculation will forget this kind of design.

The integral means Thorium is blended with other fissile isotopes, such as ^{233}U, ^{235}U or ^{239}Pu, based on the previous deductions, it is beneficial to enhance the conversion or breeding from ^{232}Th to ^{233}U especially in the early stage of the fuel lifetime, and also makes the core power distribution easily controlled as it already has reactivity at the beginning. So, the integral design will be our choice to do further calculation and analysis in this paper.

Table 1 listed the main lattice parameters of a typical 17x17 PWR fuel assembly, which is the basic database in our calculation. The lattice code used in our calculation is DRAGON 3.06 [4], which is an open source code developed by École Polytechnique de Montréal; the library to DRAGON here is IAEA version of WLUP format microscopic cross-section library [http://www-nds.iaea.org/wimsd].

Table 1. Lattice parameters for a typical 17x17 PWR fuel assembly

Item	Parameter
System Pressure, MPa	15.5
Moderator temperature, ^{o}C	302.0
Soluble Boron, ppm	500.0
Specific Power, Kw/Kg	38.0
Assembly Pitch, cm	21.5
Active height, cm	364.0
Fuel rod pitch, cm	1.26
Fuel rod diameter, cm	0.95
Cladding thickness, cm	0.057
Pellet diameter, cm	0.8192

Table 2. Pellet designs for Thorium-based fuel with various seeds

Cases	Pellet Type	Ingredient	
Reference case	UO_2,$^{235}U/U$=4.9w/o	UO_2,w/o	100.00
Case A	$(U+Th)O_2$, $^{235}U/U$=10.0w/o	UO_2,w/o	45.00
		ThO_2,w/o	55.00
Case B	$(U+Th)O_2$, $^{235}U/U$=20.0w/o	UO_2,w/o	23.00
		ThO_2,w/o	77.00
Case C	$(Pu+Th)O_2$,(^{239}Pu+^{241}Pu)/Pu=66.1w/o	PuO_2,w/o	7.90
		ThO_2,w/o	92.10
Case D	$(U+Th)O_2$,(^{233}U+^{235}U)/U=89.5w/o	UO_2,w/o	4.62
		ThO_2,w/o	95.38
Case E	$(U+Th)O_2$,(^{233}U+^{235}U)/U=89.5w/o	UO_2,w/o	2.50
		ThO_2,w/o	97.50
Case F	$(U+Th)O_2$,(^{233}U+^{235}U)/U=89.5w/o	UO_2,w/o	2.00
		ThO_2,w/o	98.00
Case G	$(U+Th)O_2$,(^{233}U+^{235}U)/U=89.5w/o	UO_2,w/o	1.50
		ThO_2,w/o	98.50

Table 2 presented the pellet designs for Thorium-based fuel with various seeds, which will be

examined in this paper. The Reference case in Table 2 is UOX fuel (enriched ^{235}U=4.9w/o), whose purposes here are to provide a target to be compared with other Thorium-based fuel and the ingredient of the reactor-grade Plutonium. Case A and Case B are the Thorium-based fuel using MEU (10w/o and 20w/o enriched ^{235}U respectively) as seeds; Case C uses reactor-grade Plutonium as seeds, the ingredient of reactor-grade Plutonium is obtained from Reference case at 60GWD/tHM; Case D is the Thorium-based fuel using reactor-grade ^{233}U as seeds, the ingredient of reactor-grade ^{233}U is obtained from Case C at 60GWD/tHM. The fractions for the oxides of Thorium and the seeds in Case A, B, C and D are designed so that the reactivity of each case at 60GWD/tHM is equivalent with the Reference case (see FIG. 5). Case E, F and G are 3 cases using reactor-grade ^{233}U as seeds, the fractions of the Uranium dioxide are 2.5w/o, 2.0w/o and 1.5w/o respectively, where 2.0w/o is the estimated threshold value (CBPI) for conversion and breeding.

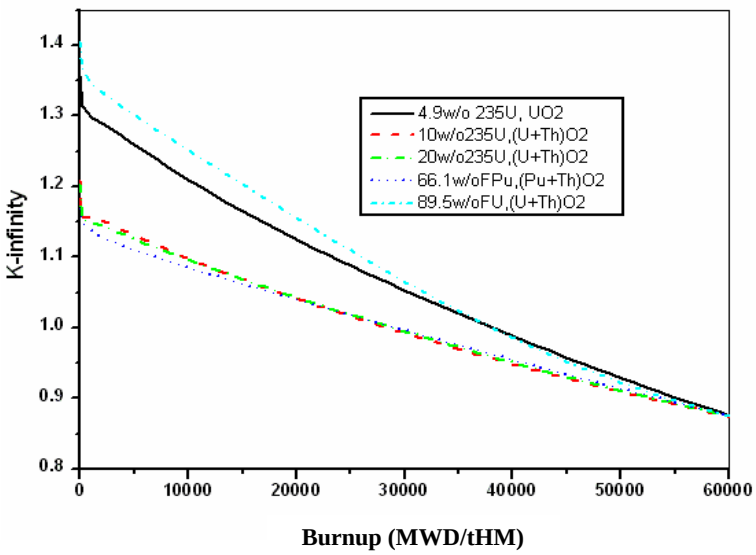

Fig. 5. Reactivity vs. burnup for Reference case and Case A~D

The Table 3 summarized calculation results of actinides inventories at initial (0 MWD/tHM) and final (60GWD/tHM) state for Reference case and Case A~D, where the mass density is averaged over assembly volume. It can be seen from table 3 that the final composition of actinides in each Thorium-based fuel have significant difference.

Firstly, as for the Thorium-based fuels using MEU (10w/o and 20w/o enriched ^{235}U in Case A and B respectively) as seeds, since the initial enrichment is different, the weight percent of Uranium dioxide and Thorium dioxide have significant variation, then the mass density of ^{233}U in the discharged fuels is varied from 0.027 g/cm^3 (1.112×0.0245 for 10w/o enriched ^{235}U) to 0.035 g/cm^3 (0.540×0.0644 for 20w/o enriched ^{235}U). This result validated the deduction that the final inventory of ^{233}U is proportional to the ^{232}Th inventory. And also, since the main composition in MEU is ^{238}U, which is not easily fissile and remained mostly in the discharged fuel, the main nuclide in the recycled Uranium from the discharged fuels is ^{238}U, so that the weight percent of ^{233}U in the recycled Uranium is much lower, in other words, the weight percents of all fissile Uranium are much less than the initial enrichments of ^{235}U in the MEU. Therefore, it is impossible to utilize this recycled Uranium as seeds to drive next round Thorium fuel cycle if we only extract the remained Uranium from the discharged fuel with chemical process. And if we want to utilize this recycled Uranium in next round Thorium fuel cycle, the enrichment process might get to be adopted, which will have much adverse influence on the complexity of fuel processes and also the fuel cost. As a result, the MEU is not a good seeds to mix with Thorium and difficult to support sustainable Thorium fuel cycle.

Secondly, in the discharged Thorium-based fuel using reactor-grade Plutonium as seeds (Case C), the purity of fissile Uranium (^{233}U and ^{235}U) is the highest, nearly 90w/o and mainly in ^{233}U. After the

Uranium is extracted from the discharged fuel through chemical process, it can be directly used as seeds to mix with Thorium and enter next round fuel recycle. And for the discharged Thorium-based fuel using above-mentioned recycled Uranium (mainly in ^{233}U) as seeds (Case D), the purity of fissile Uranium (^{233}U and ^{235}U) is still higher, nearly 68w/o; the

Table 3. Summarized calculation results of initial and final actinides inventories for Reference case and Case A~D

		Reference Case UO_2 $^{235}U/U=4.9w/o$	Case A $(U+Th)O_2$ $^{235}U/U=10.0w/o$	Case B $(U+Th)O_2$ $^{235}U/U=20.0w/o$	Case C $(Pu+Th)O_2$ FPu/Pu=66.1w/o	Case D* $(U+Th)O_2$ FU/U=89.5w/o
Initial, 0 GWD/tHM	Th		1.504 g/cm^3 $^{232}Th=100w/o$	2.106 g/cm^3 $^{232}Th=100w/o$	2.285 g/cm^3 $^{232}Th=100w/o$	2.346 g/cm^3 $^{232}Th=100w/o$
	U	2.743 g/cm^3 $^{235}U=4.9w/o$ $^{238}U=95.1w/o$	1.234 g/cm^3 $^{235}U=10w/o$ $^{238}U=90w/o$	0.630 g/cm^3 $^{235}U=20w/o$ $^{238}U=80w/o$		0.114 g/cm^3 $^{232}U=0.45w/o$ $^{233}U=87.32w/o$ $^{234}U=9.92w/o$ $^{235}U=2.16w/o$ $^{236}U=0.15w/o$
	Pu				0.226 g/cm^3 $^{238}Pu=3.54w/o$ $^{239}Pu=50.94w/o$ $^{240}Pu=22.99w/o$ $^{241}Pu=15.15w/o$ $^{242}Pu=7.38w/o$	
Final, 60GWD/tHM	Th		1.430 g/cm^3 $^{232}Th=100w/o$	2.008 g/cm^3 $^{232}Th=100w/o$	2.208 g/cm^3 $^{232}Th=100w/o$	2.236 g/cm^3 $^{232}Th=100w/o$
	Pa		0.002 g/cm^3 $^{231}Pa=6.77w/o$ $^{233}Pa=93.23w/o$	0.003 g/cm^3 $^{231}Pa=6.69w/o$ $^{233}Pa=93.31w/o$	0.002 g/cm^3 $^{231}Pa=11.48w/o$ $^{233}Pa=88.52w/o$	0.003 g/cm^3 $^{231}Pa=4.73w/o$ $^{233}Pa=95.27w/o$
	U	2.532 g/cm^3 $^{234}U=0.02w/o$ $^{235}U=0.88w/o$ $^{236}U=0.71w/o$ $^{237}U=0.00w/o$ $^{238}U=98.39w/o$	1.112 g/cm^3 $^{232}U=0.01w/o$ $^{233}U=2.45w/o$ $^{234}U=0.48w/o$ $^{235}U=1.82w/o$ $^{236}U=1.50w/o$ $^{237}U=0.00w/o$ $^{238}U=93.74w/o$	0.540 g/cm^3 $^{232}U=0.04w/o$ $^{233}U=6.44w/o$ $^{234}U=1.28w/o$ $^{235}U=3.53w/o$ $^{236}U=3.19w/o$ $^{237}U=0.01w/o$ $^{238}U=85.52w/o$	0.045 g/cm^3 $^{232}U=0.45w/o$ $^{233}U=87.32w/o$ $^{234}U=9.92w/o$ $^{235}U=2.16w/o$ $^{236}U=0.15w/o$ $^{237}U=0.00w/o$ $^{238}U=0.00w/o$	0.067 g/cm^3 $^{232}U=0.32w/o$ $^{233}U=61.30w/o$ $^{234}U=28.86w/o$ $^{235}U=7.12w/o$ $^{236}U=2.39w/o$ $^{237}U=0.01w/o$ $^{238}U=0.01w/o$
	Np	0.003 g/cm^3 $^{237}Np=90.41w/o$ $^{239}Np=9.59w/o$	0.002 g/cm^3 $^{237}Np=93.84w/o$ $^{239}Np=6.16w/o$	0.002 g/cm^3 $^{237}Np=95.97w/o$ $^{239}Np=4.03w/o$	0.00001 g/cm^3 $^{237}Np=99.99w/o$ $^{239}Np=0.00w/o$	0.00018 g/cm^3 $^{237}Np=99.99w/o$ $^{239}Np=0.00w/o$
	Pu	0.038 g/cm^3 $^{238}Pu=3.54w/o$ $^{239}Pu=50.94w/o$ $^{240}Pu=22.99w/o$ $^{241}Pu=15.15w/o$ $^{242}Pu=7.38w/o$	0.021 g/cm^3 $^{238}Pu=5.31w/o$ $^{239}Pu=49.70w/o$ $^{240}Pu=19.58w/o$ $^{241}Pu=16.57w/o$ $^{242}Pu=8.85w/o$	0.013 g/cm^3 $^{238}Pu=8.53w/o$ $^{239}Pu=47.03w/o$ $^{240}Pu=17.07w/o$ $^{241}Pu=17.06w/o$ $^{242}Pu=10.31w/o$	0.084 g/cm^3 $^{238}Pu=7.11w/o$ $^{239}Pu=12.96w/o$ $^{240}Pu=33.68w/o$ $^{241}Pu=24.66w/o$ $^{242}Pu=21.59w/o$	0.0009 g/cm^3 $^{238}Pu=81.28w/o$ $^{239}Pu=12.06w/o$ $^{240}Pu=3.07w/o$ $^{241}Pu=2.61w/o$ $^{242}Pu=0.98w/o$
	Am	0.001 g/cm^3 $^{241}Am=21.56w/o$ $^{242}Am=0.4w/o$ $^{243}Am=78.02w/o$	0.0007 g/cm^3 $^{241}Am=20.01w/o$ $^{242}Am=0.37w/o$ $^{243}Am=79.61w/o$	0.0005 g/cm^3 $^{241}Am=17.27w/o$ $^{242}Am=0.32w/o$ $^{243}Am=82.41w/o$	0.010 g/cm^3 $^{241}Am=20.13w/o$ $^{242}Am=0.47w/o$ $^{243}Am=79.40w/o$	2.e-7 g/cm^3 $^{241}Am=22.91w/o$ $^{242}Am=0.38w/o$ $^{243}Am=76.71w/o$

	Cm	0.0047 g/cm^3 ^{242}Cm=15.55w/o ^{243}Cm=0.48w/o ^{244}Cm=83.97w/o	0.0032 g/cm^3 ^{242}Cm=14.71w/o ^{243}Cm=0.47w/o ^{244}Cm=88.82w/o	0.0023 g/cm^3 ^{242}Cm=13.61w/o ^{243}Cm=0.44w/o ^{244}Cm=85.95w/o	0.0063 g/cm^3 ^{242}Cm=8.93w/o ^{243}Cm=0.39w/o ^{244}Cm=90.68w/o	6.e-8 g/cm^3 ^{242}Cm=25.11w/o ^{243}Cm=0.48w/o ^{244}Cm=74.41w/o

* The contribution of ^{233}Pa decay to ^{233}U during cooling is ignored conservatively.

later calculation results will demonstrate that the recycled Uranium with 68w/o of fissile nuclides also has the capability to work as seeds to drive Thorium-based fuel recycle and will maintain this purity of fissile Uranium in next round recycles without significant degradation.

Thirdly, it can be seen from Table 3 that the content of transuranics in discharged fuel are obviously diverse from each case. Thereinto, the inventories of Plutonium and MA (Minor Actinides, such as Np, Am, Cm) in the discharged fuels blended with MEU (Case A and B) are generally equivalent to the Reference case; while the inventories of remained Plutonium in Case C are still much bigger than Reference case and other Thorium-based cases, even though most of fissile Plutonium have been burnt out, and the MA inventories in Case C are also the highest among all cases; whereas, for the Case D using ^{233}U as seeds, because there is a far way for ^{232}Th and ^{233}U to become Plutonium and MA, the inventories of Plutonium and MA (Np, Am, Cm) in the discharged fuel are far lower than other cases, only with the magnitude 10^{-3}~10^{-5} of the Reference case. It is well-known that Plutonium is the crucial material controlled under nonproliferation and Minor Actinides are the main contributors to the high radiotoxicity and decay heat, therefore, the Thorium-based fuel mixed with ^{233}U seeds naturally satisfies some important items for Gen-IV reactors.

The above analysis indicates that Thorium-based fuel mixed with ^{233}U has excellent performance from the standpoint of fuel cycle and Gen-IV reactors. Unfortunately, there is no ^{233}U existing in nature for us to start Thorium-Uranium fuel cycle right now; and even though we can produce some man-made ^{233}U via Thorium-based fuel mixed with MEU or reactor-grade Plutonium, it is still impossible to utilize the above excellent performance if we could not approach sustainable Thorium-Uranium breeding recycle. Just as we discussed above, it is difficult for MEU to produce ^{233}U and then feed Thorium-Uranium fuel cycle, and actually, it is still a bigger challenge to obtain industrialized MEU currently; reactor-grade Plutonium is a good driver to generate ^{233}U with high purity, but it will become big issues on the high radiotoxicity management and nonproliferation if we expect it as a long term ^{233}U supplier, and also, reactor-grade Plutonium is the fuel of Fast Breeder reactor (FBR), which is another path to pursue long term supply of nuclear energy. There are also a lot of conceptual designs for innovative reactors to utilize Thorium resource, but it is a long term version before these innovative reactors get into industrialization. Therefore, the most attractive case is to finally approach sustainable Thorium-Uranium Breeding Recycle in currently existing reactors, without any other change to the fuel lattice and the core internals, but substituting the UOX pellet with Thorium-based pellet.

The theoretical analysis in Section 2 has revealed that it is possible to approach sustainable Thorium-Uranium Breeding Recycle in current PWR if and only if the ratio of ^{233}U inventory to ^{232}Th is less than a threshold value (CBPI), which is an inherent index of a given PWR fuel lattice. Case E, F and G just modeled the 3 Thorium-based fuels with ^{233}U seeds where the weight percent of Uranium dioxide are 2.5w/o, 2.0w/o and 1.5w/o respectively and 2.0w/o is nearly the threshold value of the fuel lattice used in this paper. FIG. 6 presented the reactivity and CBR vs. burnup for Case E, F and G. Just as we predicted in previous section, it can be seen from FIG. 6 that the Case E (1.5w/o UO_2) has achieved breeding, whose CBR is always greater than 1 and the reactivity maintains very little variation during whole lifetime, Case G (2.5w/o UO_2) never achieved breeding and Case F (2.0w/o UO_2) achieved breeding at EOL. Table 4 summarized the calculation results of Uranium inventories at initial (0GWD/tHM) and final (40GWD/tHM) burnup for Case E, F and G respectively, which is also a validation to the deduction in Section 2.

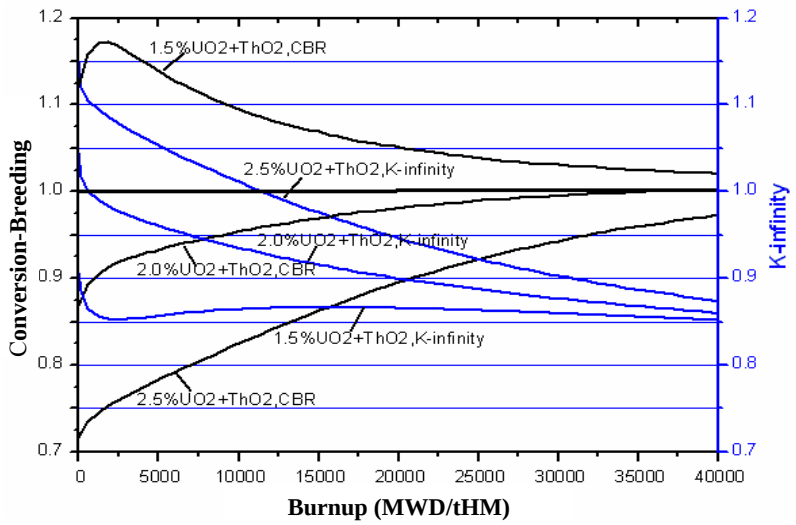

Fig. 6. Reactivity and CBR vs. burnup for Case E, F and G

Table 4. Summarized calculation results of Thorium and Uranium inventories at initial and final state for Case E, F and G

		Case E,$(U+Th)O_2$ * 2.5%UO_2,FU/U=89.5w/o	Case F,$(U+Th)O_2$ * 2.0%UO_2,FU/U=89.5w/o	Case G,$(U+Th)O_2$ * 1.5%UO_2,FU/U=89.5w/o
Initial, 0 GWD/tHM	Th	^{232}Th=2.399 g/cm^3 (100w/o)	^{232}Th=2.411 g/cm^3 (100w/o)	^{232}Th=2.423 g/cm^3 (100w/o)
	U	^{232}U=0.0000 g/cm^3 (0.45w/o) ^{233}U=0.0542 g/cm^3 (87.32w/o) ^{234}U=0.0062 g/cm^3 (9.92w/o) ^{235}U=0.0013 g/cm^3 (2.16w/o) ^{236}U=0.0001 g/cm^3 (0.15w/o) ^{237}U=0.0000 g/cm^3 (0.00w/o) ^{238}U=0.0000 g/cm^3 (0.00w/o) 0.0618 g/cm^3	^{232}U=0.0000 g/cm^3 (0.45w/o) ^{233}U=0.0434 g/cm^3 (87.32w/o) ^{234}U=0.0049 g/cm^3 (9.92w/o) ^{235}U=0.0011 g/cm^3 (2.16w/o) ^{236}U=0.0001 g/cm^3 (0.15w/o) ^{237}U=0.0000 g/cm^3 (0.00w/o) ^{238}U=0.0000 g/cm^3 (0.00w/o) 0.0494 g/cm^3	^{232}U=0.0000 g/cm^3 (0.45w/o) ^{233}U=0.0325 g/cm^3 (87.32w/o) ^{234}U=0.0037 g/cm^3 (9.92w/o) ^{235}U=0.0008 g/cm^3 (2.16w/o) ^{236}U=0.0001 g/cm^3 (0.15w/o) ^{237}U=0.0000 g/cm^3 (0.00w/o) ^{238}U=0.0000 g/cm^3 (0.00w/o) 0.0371 g/cm^3
Final, 40GWD/tHM	Th	^{232}Th=2.302 g/cm3 (100w/o)	^{232}Th=2.304 g/cm3 (100w/o)	^{232}Th=2.306 g/cm3 (100w/o)
	U	^{232}U=0.0002 g/cm^3 (0.28 w/o) ^{233}U=0.0378 g/cm^3 (69.86w/o) ^{234}U=0.0124 g/cm^3 (22.97w/o) ^{235}U=0.0029 g/cm^3 (5.35w/o) ^{236}U=0.0008 g/cm^3 (1.55w/o) ^{237}U=0.0000 g/cm^3	^{232}U=0.0002 g/cm^3 (0.31 w/o) ^{233}U=0.0365 g/cm^3 (70.72w/o) ^{234}U=0.0115 g/cm^3 (22.32w/o) ^{235}U=0.0026 g/cm^3 (5.12 w/o) ^{236}U=0.0008 g/cm^3 (1.53w/o) ^{237}U=0.0000 g/cm^3	^{232}U=0.0002 g/cm^3 (0.33 w/o) ^{233}U=0.0357 g/cm^3 (71.72w/o) ^{234}U=0.0108 g/cm^3 (21.61w/o) ^{235}U=0.0024 g/cm^3 (4.87w/o) ^{236}U=0.0007 g/cm^3 (1.46w/o) ^{237}U=0.0000 g/cm^3

		(0.00w/o) ^{238}U=0.0000 g/cm^3 (0.00w/o) 0.0541 g/cm^3	(0.00w/o) ^{238}U=0.0000 g/cm^3 (0.00w/o) 0.0516 g/cm^3	(0.00w/o) ^{238}U=0.0000 g/cm^3 (0.00w/o) 0.0494 g/cm^3

* The contribution of ^{233}Pa decay to ^{233}U during cooling is ignored conservatively.

FIG. 7 and Table 5 summarized the calculation results for multiple-round Thorium-Uranium fuel recycle, where R1, R2 and R3 represent round 1, 2 and 3 respectively and R1 is connected to Case G in Table 4. It can be seen from FIG. 7 that the recycled Uranium in sustainable Thorium-Uranium fuel cycle can maintain similar breeding capability after the ingredient is well-designed; and also, we can find from Table 5 that the weight percent change (degradation) of fissile Uranium in recycled Uranium is also smaller.

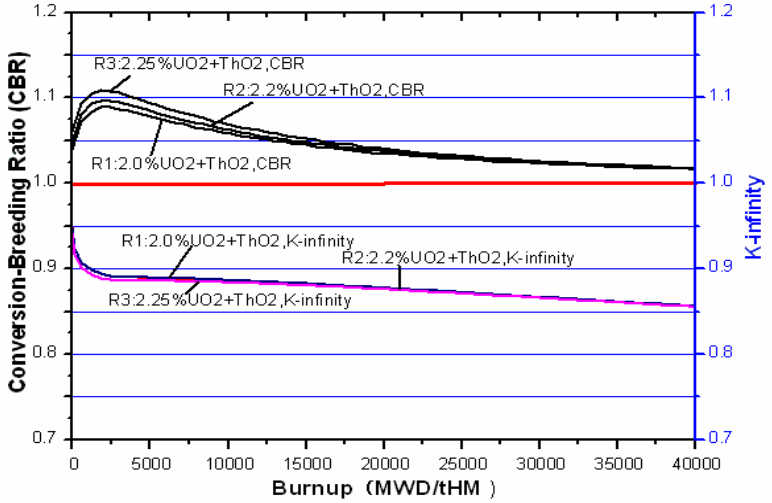

Fig. 7. Conversion-Breeding Ratio and K-infinity vs. Burnup for Multiple-Round Thorium-Uranium Fuel Recycle

Table 5. Summarized calculation results of Thorium and Uranium inventories
For multiple-round Thorium-Uranium fuel recycle

		R1:(U+Th)O_2 * 2.0%UO_2,FU/U=76.6w/o	R2:(U+Th)O_2 * 2.2 %UO_2,FU/U=71.0w/o	R3:(U+Th)O_2 * 2.25%UO_2,FU/U=70.1w/o
Initial, 0GWD/tHM	U	^{232}U=0.33 w/o ^{233}U=71.72w/o ^{234}U=21.61w/o ^{235}U=4.87w/o ^{236}U=1.46w/o ^{237}U=0.00w/o ^{238}U=0.00w/o 0.04938 g/cm^3	^{232}U=0.29 w/o ^{233}U=64.75w/o ^{234}U=25.58w/o ^{235}U=6.27w/o ^{236}U=3.10w/o ^{237}U=0.01w/o ^{238}U=0.01w/o 0.05431 g/cm^3	^{232}U=0.28 w/o ^{233}U=61.30w/o ^{234}U=27.06 w/o ^{235}U=6.79 w/o ^{236}U=4.56w/o ^{237}U=0.01w/o ^{238}U=0.01w/o 0.05554 g/cm^3

Final, 40GWD/tHM	U	^{232}U=0.29 w/o ^{233}U=64.75w/o ^{234}U=25.58w/o ^{235}U=6.27w/o ^{236}U=3.10w/o ^{237}U=0.01w/o ^{238}U=0.01w/o 0.05563 g/cm^3	^{232}U=0.28 w/o ^{233}U=61.30w/o ^{234}U=27.06 w/o ^{235}U=6.79 w/o ^{236}U=4.56w/o ^{237}U=0.01w/o ^{238}U=0.01w/o 0.05884 g/cm^3	^{232}U=0.27 w/o ^{233}U=60.22w/o ^{234}U=27.13w/o ^{235}U=6.84w/o ^{236}U=5.52w/o ^{237}U=0.01w/o ^{238}U=0.02w/o 0.05986 g/cm^3

* The contribution of ^{233}Pa decay to ^{233}U during cooling is ignored conservatively.

All above calculation results demonstrated that it is possible to approach sustainable Thorium-Uranium Breeding Recycle in current PWR fuel lattice so long as the ingredient of fuel pellet is well designed. It is a pity that the K-infinite of the Thorium-based fuel at this condition is always smaller than 1 and impossible to maintain the critical core if the core is fully loaded with Thorium-based fuel. Fortunately, after comparing the reactivity of this Thorium-based fuel with those fuel assemblies loaded on the periphery of Low-leakage Long cycle PWR core, which is very popular reload core design for PWR now, we found that the reactivity is equivalent. That means we can substitute those highly burnt UOX fuel assemblies with well-designed Thorium-based fuel assemblies on the periphery of current PWR core and get a mixed core design, which has following advantages:

- Just like that in Fast Breeder reactor, a blanket region is established, where extra ^{233}U may be bred and sustainable Thorium-Uranium Breeding Recycle may be maintained;
- Approximately 30% UOX fuel assemblies may be saved;
- Low-leakage core is naturally composed, as the reactivity of Thorium-based fuel is always lower;
- Longer cycle lifetime may be achieved, as the reactivity change of Thorium-based fuel vs. burnup is very small;
- The core characteristics, especially the dynamics, are still dominated by UOX fuel, because the most reactive region of the core is still occupied by UOX fuels.

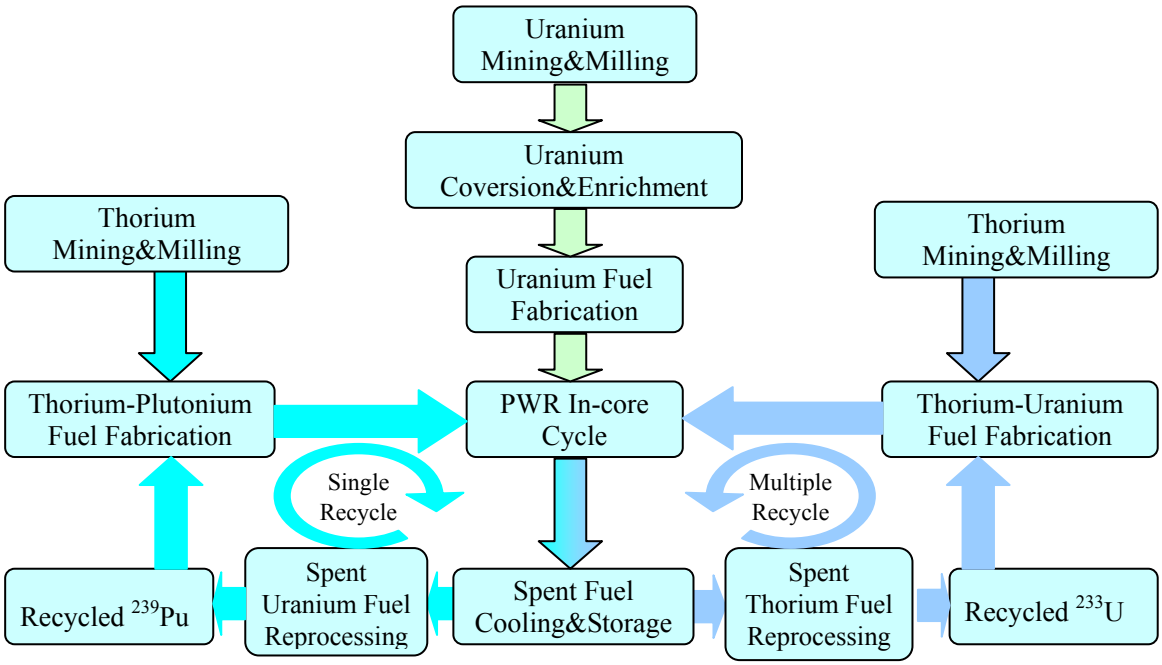

Fig. 8. Roadmap for Thorium-Uranium Recycle in PWR

4. ROADMAP TO APPROACH THORIUM-URANIUM BREEDING RECYCLE IN PWR

Based on the above theoretical analysis and calculation results, a roadmap to approach sustainable Thorium-Uranium (^{232}Th-^{233}U) Breeding Recycle in current PWR is on the surface now, as shown in FIG. 8. The middle axis in FIG.8 is a concise path of conventional Once-through fuel cycle, on this base, the roadmap to approach Thorium-Uranium Breeding Recycle in PWR includes 2 stages as follows.

In the first stage, the recycled reactor-grade plutonium from current PWR spent fuel is used as seeds to mix with thorium; then the thorium-plutonium fuels are loaded into PWR core and will produce higher purity reactor-grade ^{233}U;

In the second stage, after the thorium-plutonium fuels are discharged from PWR core and stay in spent fuel pool for proper time, the fuels will be reprocessed, the reactor-grade ^{233}U will be extracted and used as seeds to mix with thorium; then thorium-uranium fuels shall be loaded into the periphery of PWR core to compose so-called “blanket” for Low-Leakage and Long-Cycle reload core design, in which Thorium-Uranium (^{232}Th-^{233}U) Breeding Recycle will be achieved. After the thorium-uranium fuels are discharged from PWR core and cooled for enough time, the fuel will be reprocessed and the bred reactor-grade ^{233}U will be extracted, and then, the next round recycle could be started with extra reactor-grade ^{233}U accumulated.

5. CONCLUSION

The paper revealed and demonstrated the inherent prerequisites to approach sustainable Thorium-Uranium breeding recycle in current PWR, and then, combined with the features of Low-leakage Long-cycle reload PWR core design, a roadmap to approach sustainable Thorium-Uranium breeding recycle in current PWR is presented.

The further PWR core design and evaluation with Thorium-Uranium (^{232}Th-^{233}U) blanket will be presented in future reports.

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REFERENCE

[1] Dekoussar, V. etc. *Thorium fuel cycle — Potential benefits and challenges*, IAEA-TECDOC-1450, 2005.5

[2] A. PUILL, *Thorium Utilization in PWRs. Neutronics Studies*, *Thorium fuel utilization:Options and trends*, IAEA-TECDOC-1319,2002.11

[3] Xie Zhongsheng etc., *Nuclear Reactor Physics Analysis*, Xi’an Jiaotong Univerity Press, 2004.

[4] G. Marleau, A. Hébert and R. Roy, *A User Guide for DRAGON 3.06*, Report IGE-174 Rev.7, Institut de génie nucléaire, École Polytechnique de Montréal, Montréal, Québec (2008).