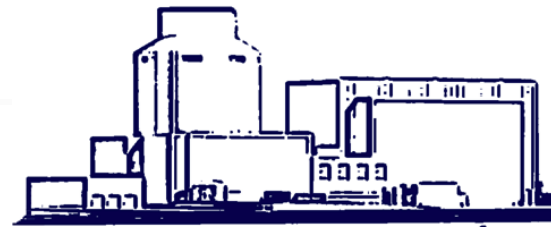


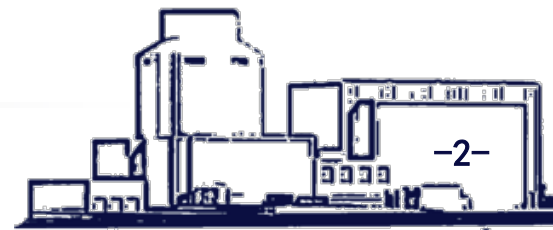
# Roadmap Design for Thorium-Uranium Breeding Recycle in PWR

Si Shengyi

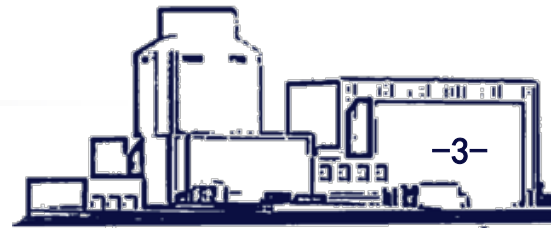
Shanghai Nuclear Engineering Research and Design Institute  
Shanghai, China  
2009.10



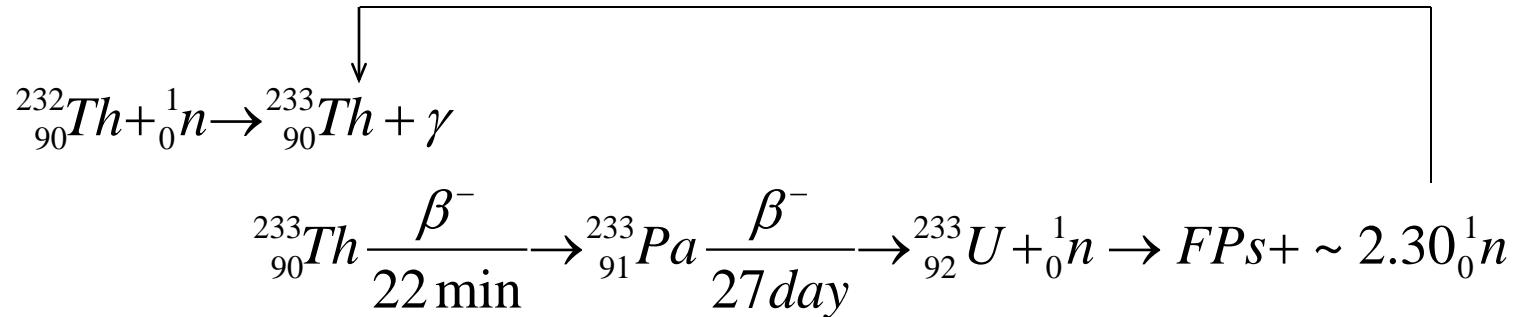
- 1. Introduction**
- 2. Insights to the Inherence of Thorium-based Fuel in PWR**
- 3. Calculation Results Evaluation for Thorium-based PWR Fuel Assembly**
- 4. Roadmap to Approach Thorium-Uranium Breeding Recycle in PWR**



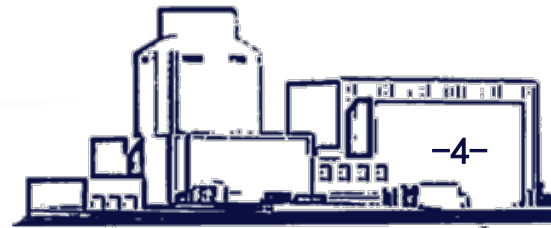
- ◆ The paper is focusing on designing a roadmap to finally approach sustainable Thorium-Uranium ( $^{232}\text{Th}$ - $^{233}\text{U}$ ) Breeding Recycle in current PWR,
- ◆ without any other change to the fuel lattice and the core internals,
- ◆ but substituting the UOX pellet with thorium-based pellet.



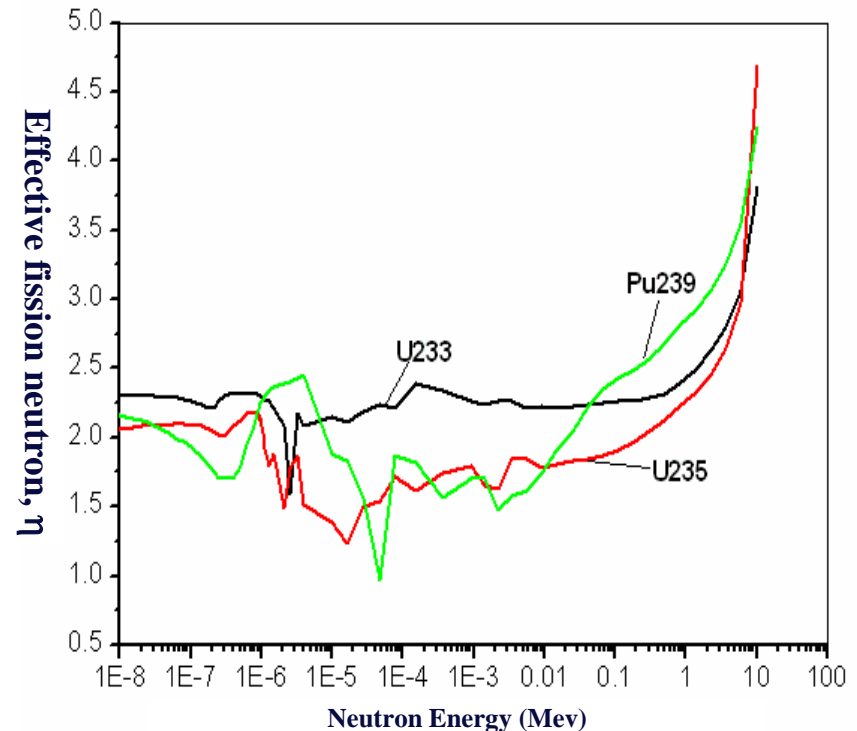
## ◆ Nuclear reaction for Thorium-Uranium ( $^{232}\text{Th}$ - $^{233}\text{U}$ ) cycle



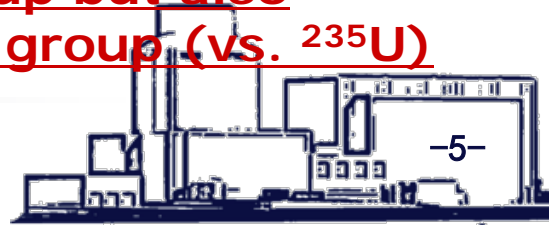
- ◆ the performance of the Thorium-based fuel in reactor core is strongly dependent on the synthetical performance of each isotope, in this transmutation process , especially  $^{232}\text{Th}$  and  $^{233}\text{U}$



- ◆ The fission performance of  $^{233}\text{U}$  is quite excellent ;
- ◆ the  $\eta$  of  $^{233}\text{U}$  is bigger than the  $\eta$  of  $^{235}\text{U}$  and  $^{239}\text{Pu}$  within a quite wide energy span from thermal to epithermal energy region ;
- ◆ This is the well-known evidence that Thorium-based fuel can achieve higher conversion ratio or even breeding in thermal reactor



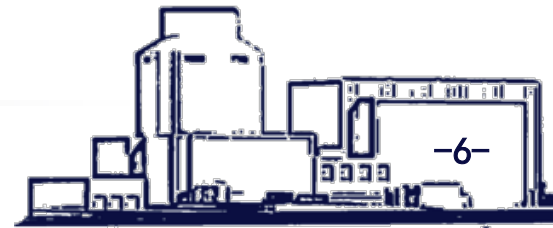
it is emphasized here that the excellent fission performance of  $^{233}\text{U}$  does not only exist in thermal energy group but also epithermal or even beyond epithermal energy group, (vs.  $^{235}\text{U}$ )



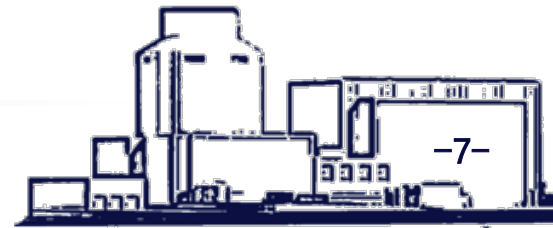
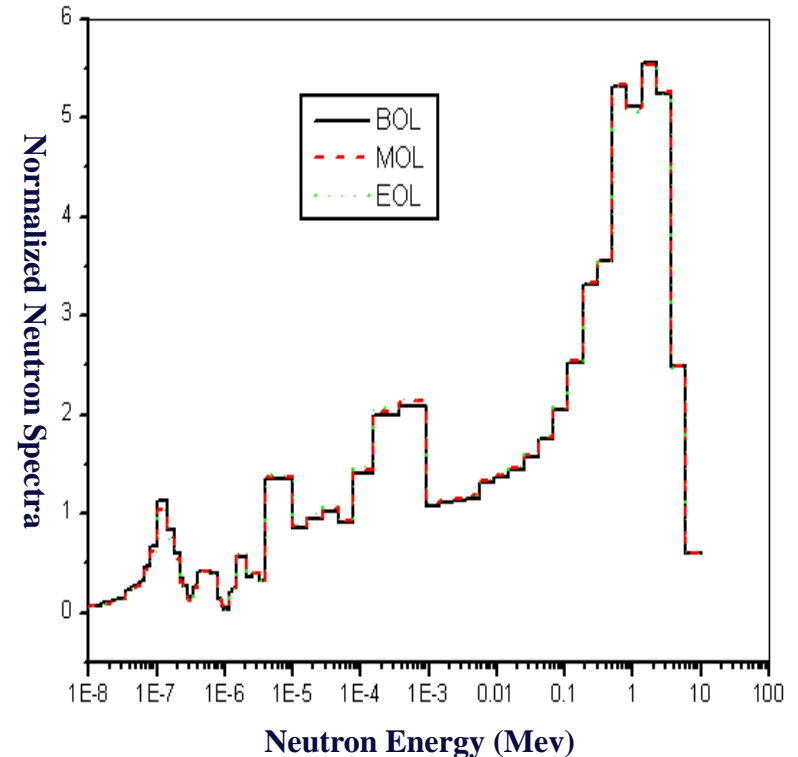
- ◆ conversion or breeding ratio (CBR) for Thorium-Uranium ( $^{232}\text{Th}$ - $^{233}\text{U}$ ) cycle can be expressed as follows:

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

- ◆ The above formula indicates that the conversion or breeding ratio is dependent on the number densities of  $^{232}\text{Th}$  and  $^{233}\text{U}$ , the neutron spectra in the fuel lattice, the  $^{232}\text{Th}$  capture cross section and  $^{233}\text{U}$  absorption cross section vs. neutron energy

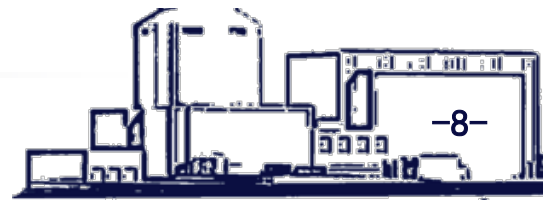
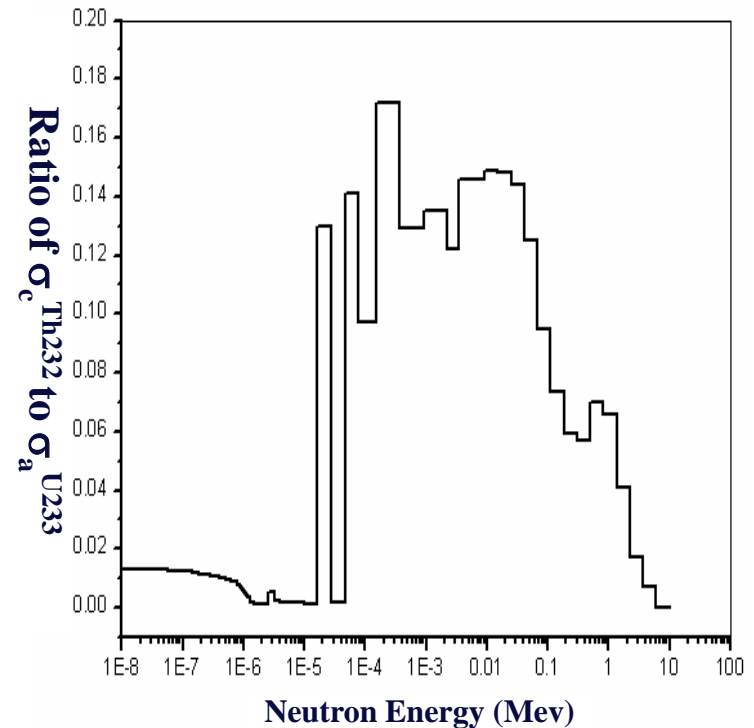


- ◆ The neutron spectra in fuel lattice are strongly dependent on the core type, such as thermal reactor or fast reactor;
- ◆ But, for a typical PWR core, the neutron spectra is almost constant
- ◆ Right Figure illustrated the spectra of a PWR fuel lattice at BOL, MOL and EOL respectively.
- ◆ We can see that the variation of the spectra during lifetime is almost neglectable



- ◆ CBR(t) has an implicit relation with the ratio of  $^{232}\text{Th}$  capture cross section to  $^{233}\text{U}$  absorption cross section ;
- ◆ The ratio is only dependent on the inherence of these two nuclides and independent to any core type ;
- ◆ Right Figure illustrated the ratio of  $\sigma_c^{Th232}$  to  $\sigma_a^{U233}$  vs. neutron energy;
- ◆ It can be seen that the ratio of  $\sigma_c^{Th232}$  to  $\sigma_a^{U233}$  around thermal group is far smaller than the ratio around epithermal group .

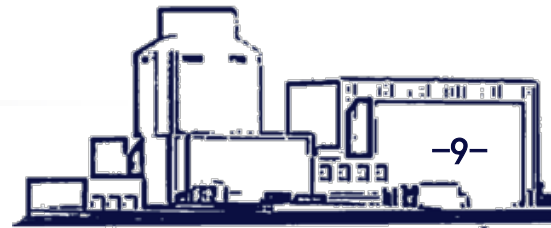
$$CBR(t) = \frac{N^{Th232}(t) \int \sigma_c^{Th232}(E) \phi(E, t) dE}{N^{U233}(t) \int \sigma_a^{U233}(E) \phi(E, t) dE}$$





## Deductions from the above insights

- ◆ The conversion of  $^{232}\text{Th}$  to  $^{233}\text{U}$  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  $^{233}\text{U}$ ,  $^{235}\text{U}$  or  $^{239}\text{Pu}$ ) is beneficial to enhance the conversion or breeding ratio in Thorium-based fuel.



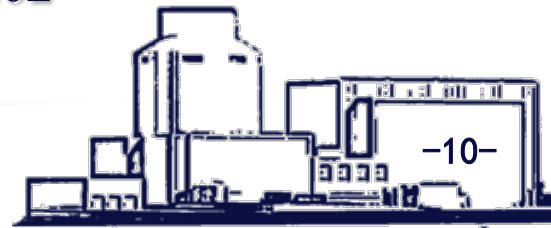
- ◆ Conversion or Breeding Performance Index (CBPI) for a given fuel lattice is a constant

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

- ◆  $CBR(t) = 1$  means the inventory of fissile isotope is in a state of quasi-equivalence, i.e.

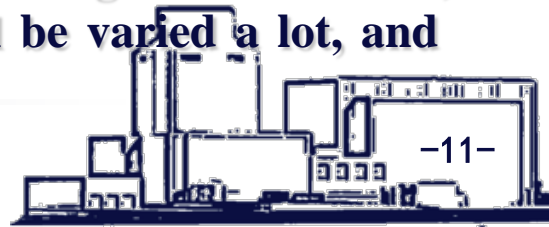
$$1 = CBPI \cdot \frac{N^{Th232}(t)}{N^{U233}(t)}, \quad \text{or} \quad \frac{N^{U233}(t)}{N^{Th232}(t)} = CBPI$$

- ◆ Prerequisite to achieve quasi-equilibrium state for Thorium-based fuel is that the ratio of  $^{233}\text{U}$  inventory to  $^{232}\text{Th}$  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



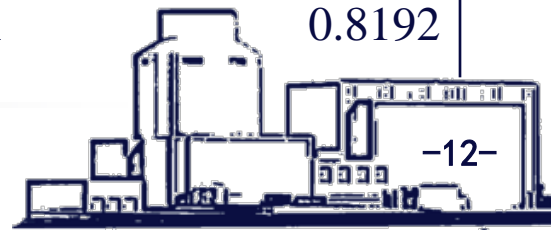
## Deductions from the above insights

- ◆ The prerequisite for Thorium-Uranium fuel breeding cycle is that the ratio of  $^{233}\text{U}$  inventory to  $^{232}\text{Th}$  inventory must be less than CBPI ( $\sim 0.02$ ), otherwise, the CBR(t) should be less than 1, it is impossible to maintain sustainable fuel cycle;
- ◆ For the Thorium-based fuel using  $^{233}\text{U}$  as seeds, if the ratio of  $^{233}\text{U}$  inventory to  $^{232}\text{Th}$  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  $^{233}\text{U}$  is mainly contributing fission energy;
- ◆ For the Thorium-based fuel using other fissile isotopes (such as  $^{235}\text{U}$  or  $^{239}\text{Pu}$ ) as seeds, since the initial  $^{233}\text{U}$  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  $^{233}\text{U}$  inventory will be the product of CBPI and instant  $^{232}\text{Th}$  inventory;
- ◆ As the purity of the seeds is varied evidently, e.g. most of the composition in MEU is  $^{238}\text{U}$  or there are some non-fissile isotopes in reactor-grade Plutonium, for a given fuel lattice, the initial  $^{232}\text{Th}$  inventory shall be varied a lot, and then the final  $^{233}\text{U}$  inventory will be also quite different.

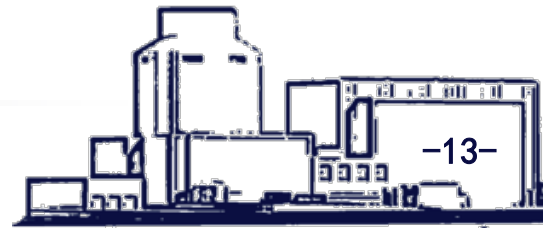
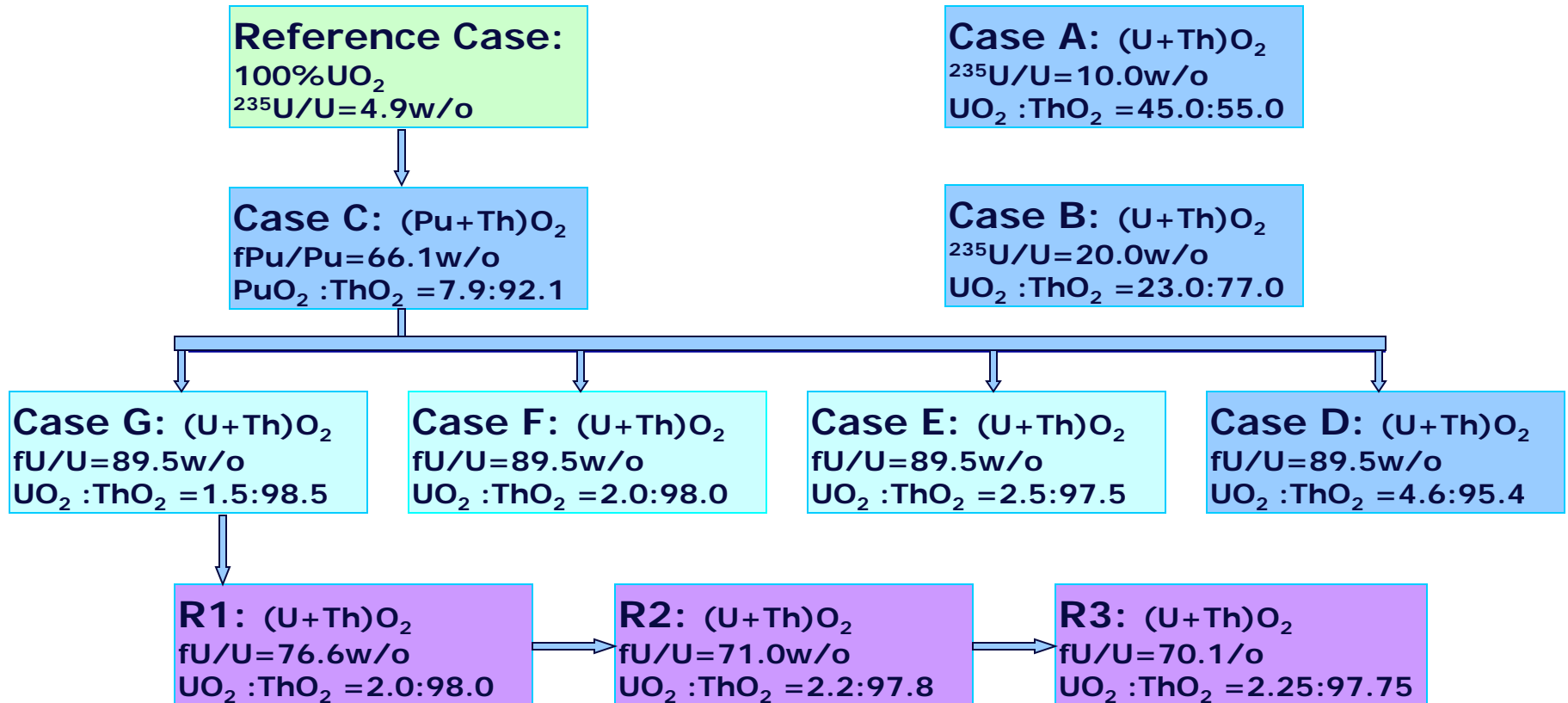


- ◆ **Discrete and Integral Thorium-based fuel;**
- ◆ **The calculation in this paper is focusing on Integral Thorium-based fuel;**
- ◆ **Right table listed the main lattice parameters of a typical 17x17 PWR fuel assembly, which is the basic database in our calculation.**
- ◆ **The lattice code used in our calculation is DRAGON 3.06**
- ◆ **The library to DRAGON here is IAEA version of WLUP format microscopic cross-section library**

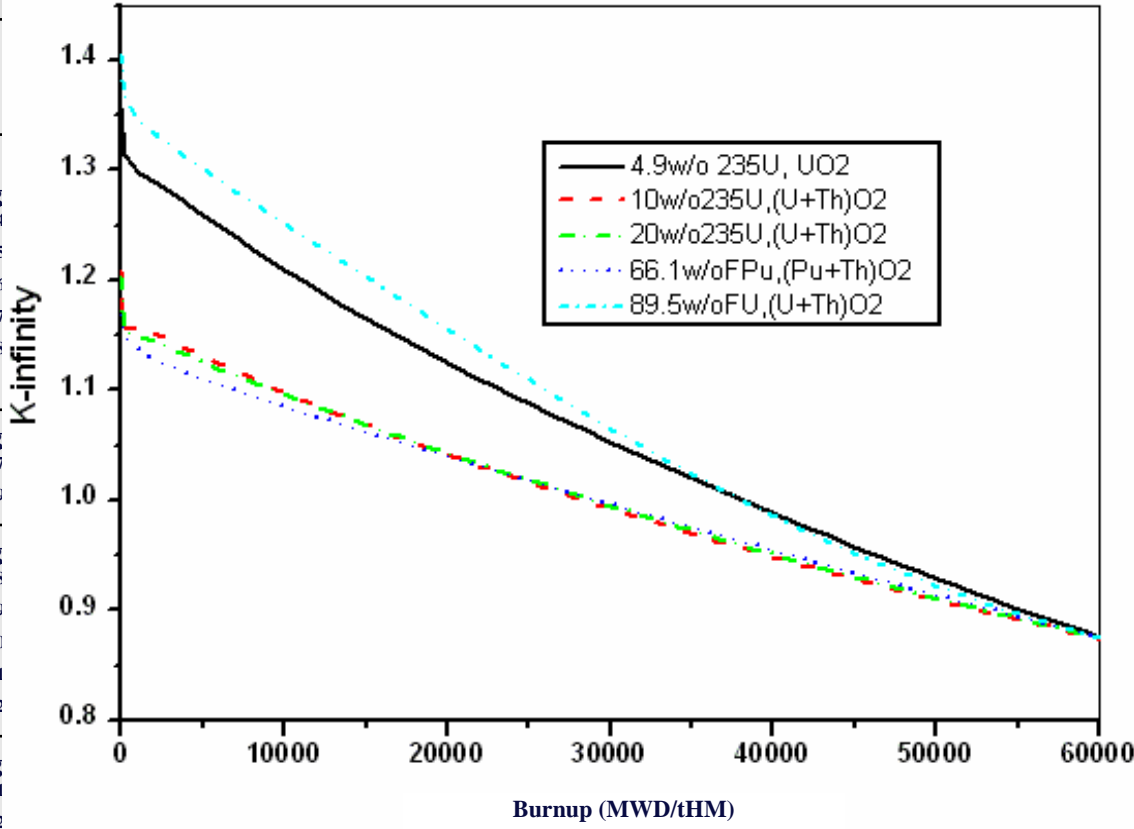
Item	Parameter
System Pressure, MPa	15.5
Moderator temperature, °C	302.0
Soluble Boron, ppm	500.0
Specific Power, Kw/Kg	38.0
Assembly Pitch, cm	21.5
Active height, cm	364.0
Fuel rod pitch, cm	1.26
Fuel rod diameter, cm	0.95
Cladding thickness, cm	0.057
Pellet diameter, cm	0.8192



Examined cases in this paper



		Reference Case UO <sub>2</sub> <sup>235</sup> U/U=4.9w/o	Case A (U+Th)O <sub>2</sub> <sup>235</sup> U/U=10.0w/o	Case B (U+Th)O <sub>2</sub> <sup>235</sup> U/U=20.0w/o	Case C (Pu+Th)O <sub>2</sub> FPu/Pu=66.1w/o	Case D* (U+Th)O <sub>2</sub> FU/U=89.5w/o	
Final, 60GWD/tHM	Th		<u>1.430 g/cm<sup>3</sup></u>	<u>2.008 g/cm<sup>3</sup></u>	<u>2.208 g/cm<sup>3</sup></u>	<u>2.236 g/cm<sup>3</sup></u> <sup>232</sup> Th=100w/o	
	Pa					<u>103 g/cm<sup>3</sup></u> <sup>231</sup> Pa=4.73w/o <sup>233</sup> Pa=95.27w/o	
	U	<u>2.532 g</u> <sup>234</sup> <sup>235</sup> <sup>236</sup> <sup>237</sup> <sup>238</sup>					<u>167 g/cm<sup>3</sup></u> <sup>232</sup> U=0.32w/o <sup>233</sup> U=61.30w/o <sup>234</sup> U=28.86w/o <sup>235</sup> U=7.12w/o <sup>236</sup> U=2.39w/o <sup>237</sup> U=0.01w/o <sup>238</sup> U=0.01w/o
	Np	<u>0.003 g</u> <sup>237</sup> <sup>235</sup>					<u>10018 g/cm<sup>3</sup></u> <sup>237</sup> Np=99.99w/o <sup>239</sup> Np=0.00w/o
	Pu	<u>0.038 g</u> <sup>238</sup> <sup>239</sup> <sup>240</sup> <sup>241</sup> <sup>242</sup>					<u>1009 g/cm<sup>3</sup></u> <sup>238</sup> Pu=81.28w/o <sup>239</sup> Pu=12.06w/o <sup>240</sup> Pu=3.07w/o <sup>241</sup> Pu=2.61w/o <sup>242</sup> Pu=0.98w/o
	Am	<u>0.001 g</u> <sup>241</sup> <sup>242</sup>					<u>0.7 g/cm<sup>3</sup></u> <sup>241</sup> Am=22.91w/o <sup>242</sup> Am=0.38w/o <sup>243</sup> Am=76.71w/o
			<sup>243</sup> Am=78.02w/o	<sup>243</sup> Am=79.61w/o	<sup>243</sup> Am=82.71w/o	<sup>243</sup> Am=79.40w/o	
	Cm	<u>0.0047 g/cm<sup>3</sup></u> <sup>242</sup> Cm=15.55w/o <sup>243</sup> Cm=0.48w/o <sup>244</sup> Cm=83.97w/o	<u>0.0032 g/cm<sup>3</sup></u> <sup>242</sup> Cm=14.71w/o <sup>243</sup> Cm=0.47w/o <sup>244</sup> Cm=88.82w/o	<u>0.0023 g/cm<sup>3</sup></u> <sup>242</sup> Cm=13.61w/o <sup>243</sup> Cm=0.44w/o <sup>244</sup> Cm=85.95w/o	<u>0.0063 g/cm<sup>3</sup></u> <sup>242</sup> Cm=8.93w/o <sup>243</sup> Cm=0.39w/o <sup>244</sup> Cm=90.68w/o	<u>6.e-8 g/cm<sup>3</sup></u> <sup>242</sup> Cm=25.11w/o <sup>243</sup> Cm=0.48w/o <sup>244</sup> Cm=74.41w/o	

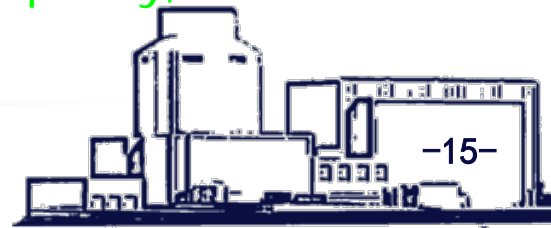


# Calculation Results Evaluation for Thorium-based PWR Fuel Assembly

Extraction from the above table

	Ref. Case	Case A	Case B	Case C	Case D
$^{233}\text{U}$ mass density, g/cm <sup>3</sup>		<b>0.027</b>	<b>0.035</b>	<b>0.039</b>	<b>0.041</b>
$^{233}\text{U}/\text{U}$ , w/o		<b>2.45</b>	<b>6.44</b>	<b>87.32</b>	<b>61.30</b>
Pu mass density, g/cm <sup>3</sup>	<b>0.038</b>	<b>0.021</b>	<b>0.013</b>	<b>0.084</b>	<b>0.001</b>
MA mass density, g/cm <sup>3</sup>	<b>~0.009</b>	<b>~0.006</b>	<b>~0.005</b>	<b>~0.016</b>	<b>~0.0002</b>

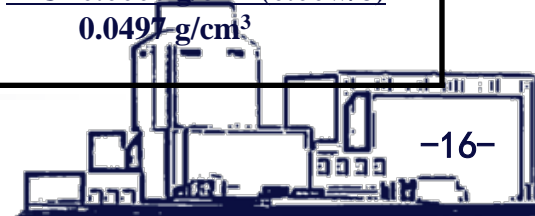
- MEU(Case A and B): **less yield, quite low enrichment, Pu and MA equivalent to Ref. case**, not a good seed;
- Reactor-grade Plutonium (Case C): **higher yield, highest purity  $^{233}\text{U}$ , highest Pu and MA yield**;
- Reactor-grade  $^{233}\text{U}$  (Case D): **appropriate yield and purity, lowest Pu and MA**





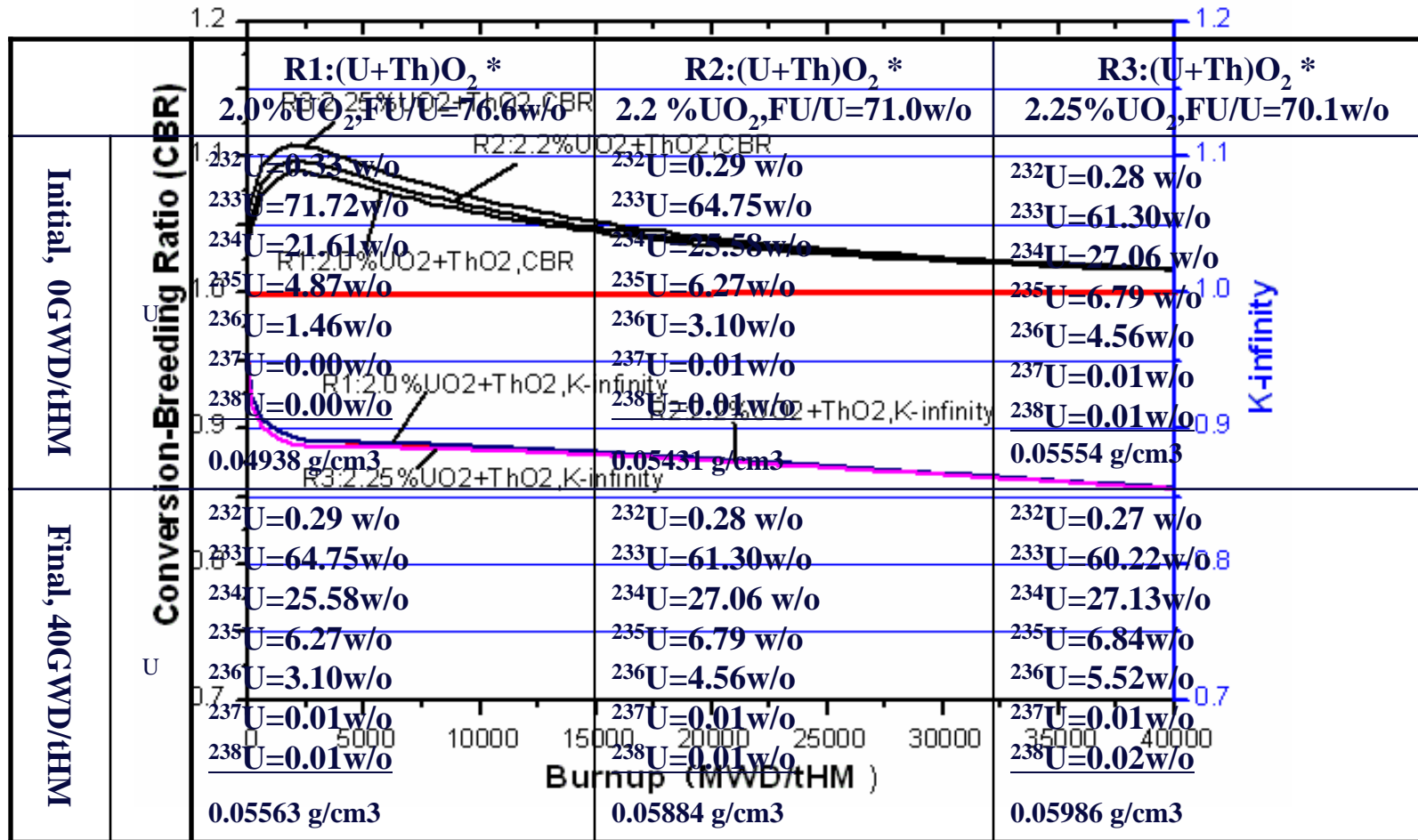
# Calculation Results Evaluation for Thorium-based PWR Fuel Assembly

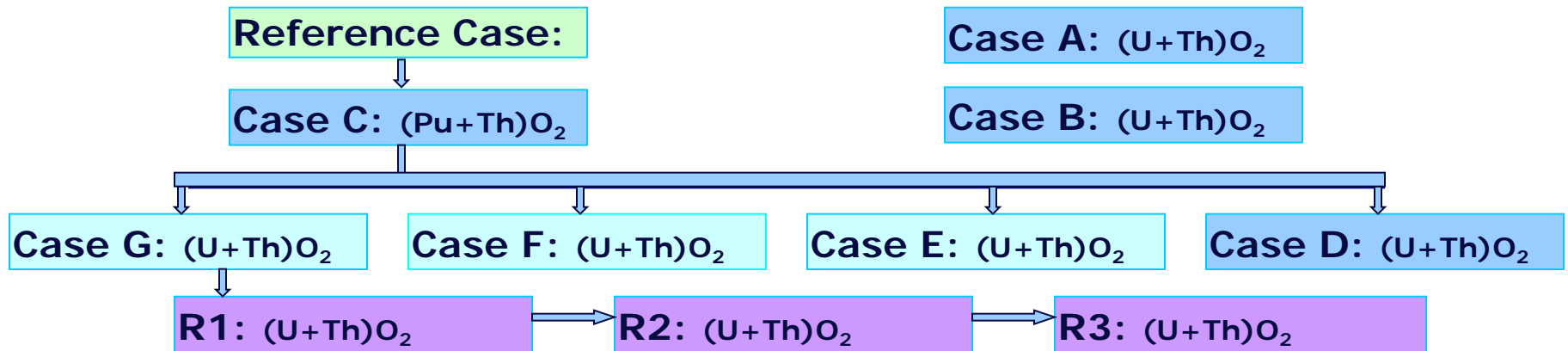
		Case E, (U+Th)O <sub>2</sub> * 2.5%UO <sub>2</sub> , FU/U=89.5w/o	Case F, (U+Th)O <sub>2</sub> * 2.0%UO <sub>2</sub> , FU/U=89.5w/o	Case G, (U+Th)O <sub>2</sub> * 1.5%UO <sub>2</sub> , FU/U=89.5w/o
Initial, 0 GWD/tHM	Th	<sup>232</sup> Th=2.399 g/cm <sup>3</sup> (100w/o)	<sup>232</sup> Th=2.411 g/cm <sup>3</sup> (100w/o)	<sup>232</sup> Th=2.423 g/cm <sup>3</sup> (100w/o)
	U	<sup>232</sup> U=0.0000 g/cm <sup>3</sup> (0.45w/o) <sup>233</sup> U=0.0542 g/cm <sup>3</sup> (87.32w/o) <sup>234</sup> U=0.0062 g/cm <sup>3</sup> (9.92w/o) <sup>235</sup> U=0.0013 g/cm <sup>3</sup> (2.16w/o) <sup>236</sup> U=0.0001 g/cm <sup>3</sup> (0.15w/o) <sup>237</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) <sup>238</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) 0.0618 g/cm <sup>3</sup>	<sup>232</sup> U=0.0000 g/cm <sup>3</sup> (0.45w/o) <sup>233</sup> U=0.0434 g/cm <sup>3</sup> (87.32w/o) <sup>234</sup> U=0.0049 g/cm <sup>3</sup> (9.92w/o) <sup>235</sup> U=0.0011 g/cm <sup>3</sup> (2.16w/o) <sup>236</sup> U=0.0001 g/cm <sup>3</sup> (0.15w/o) <sup>237</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) <sup>238</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) 0.0494 g/cm <sup>3</sup>	<sup>232</sup> U=0.0000 g/cm <sup>3</sup> (0.45w/o) <sup>233</sup> U=0.0325 g/cm <sup>3</sup> (87.32w/o) <sup>234</sup> U=0.0037 g/cm <sup>3</sup> (9.92w/o) <sup>235</sup> U=0.0008 g/cm <sup>3</sup> (2.16w/o) <sup>236</sup> U=0.0001 g/cm <sup>3</sup> (0.15w/o) <sup>237</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) <sup>238</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) 0.0371 g/cm <sup>3</sup>
Final, 40GWD/tHM	Th	<sup>232</sup> Th=2.302 g/cm <sup>3</sup> (100w/o)	<sup>232</sup> Th=2.304 g/cm <sup>3</sup> (100w/o)	<sup>232</sup> Th=2.306 g/cm <sup>3</sup> (100w/o)
	U	<sup>232</sup> U=0.0002 g/cm <sup>3</sup> (0.28 w/o) <sup>233</sup> U=0.0378 g/cm <sup>3</sup> (69.86w/o) <sup>234</sup> U=0.0124 g/cm <sup>3</sup> (22.97w/o) <sup>235</sup> U=0.0029 g/cm <sup>3</sup> (5.35w/o) <sup>236</sup> U=0.0008 g/cm <sup>3</sup> (1.55w/o) <sup>237</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) <sup>238</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) 0.0541 g/cm <sup>3</sup>	<sup>232</sup> U=0.0002 g/cm <sup>3</sup> (0.31 w/o) <sup>233</sup> U=0.0365 g/cm <sup>3</sup> (70.72w/o) <sup>234</sup> U=0.0115 g/cm <sup>3</sup> (22.32w/o) <sup>235</sup> U=0.0026 g/cm <sup>3</sup> (5.12 w/o) <sup>236</sup> U=0.0008 g/cm <sup>3</sup> (1.53w/o) <sup>237</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) <sup>238</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) 0.0516 g/cm <sup>3</sup>	<sup>232</sup> U=0.0002 g/cm <sup>3</sup> (0.33 w/o) <sup>233</sup> U=0.0357 g/cm <sup>3</sup> (71.72w/o) <sup>234</sup> U=0.0108 g/cm <sup>3</sup> (21.61w/o) <sup>235</sup> U=0.0024 g/cm <sup>3</sup> (4.87w/o) <sup>236</sup> U=0.0007 g/cm <sup>3</sup> (1.46w/o) <sup>237</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) <sup>238</sup> U=0.0000 g/cm <sup>3</sup> (0.00w/o) 0.0497 g/cm <sup>3</sup>



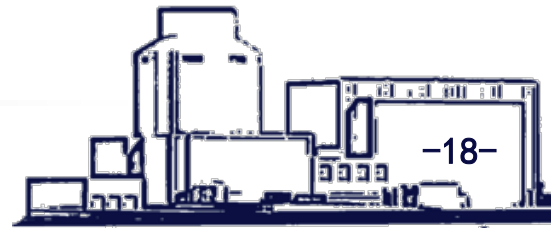


# Calculation Results Evaluation for Thorium-based PWR Fuel Assembly

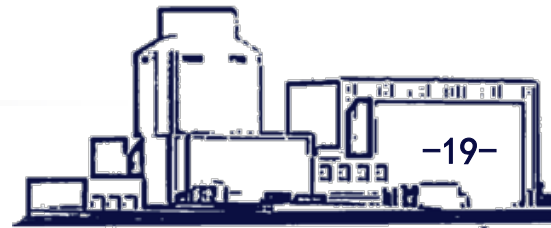




- ◆ Case A, B, C, D indicate that reactor-grade  $^{233}\text{U}$  is the best seeds for Thorium-Uranium fuel cycle
- ◆ Case E, F, G indicate that Thorium-Uranium fuel cycle using reactor-grade  $^{233}\text{U}$  can achieve breeding;
- ◆ Case R1, R2, R3 indicate that the sustainable Thorium-Uranium Breeding Recycle is approachable

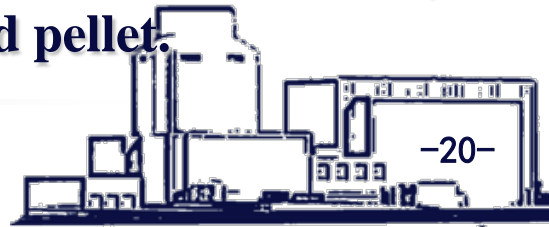


- ◆ All above calculation results demonstrated that it is possible to approach sustainable Thorium-Uranium Breeding Recycle in current PWR fuel lattice so long as the ingredient of fuel pellet is well designed.
- ◆ It is a pity that the  $K$ -infinite of the Thorium-based fuel at this condition is always 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, 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

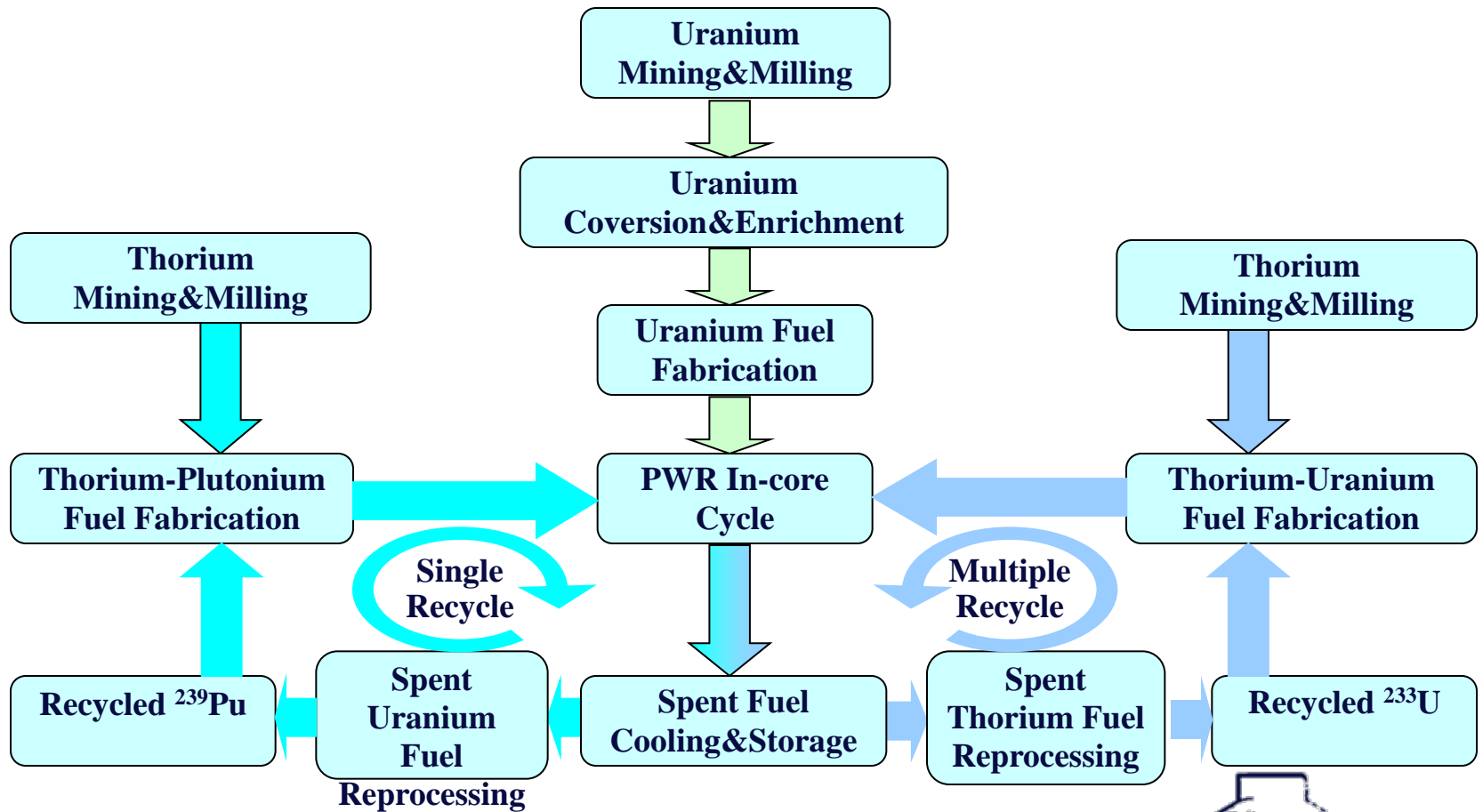


The Mixed core has following advantages:

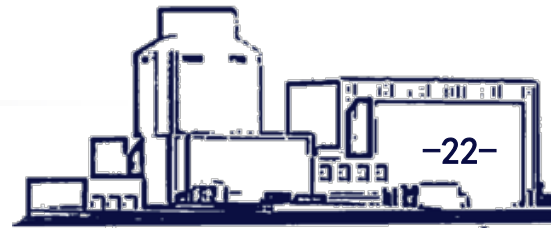
- ◆ Just like that in Fast Breeder reactor, a blanket region is established, where extra  $^{233}\text{U}$  may be bred and sustainable Thorium-Uranium Breeding Recycle may be maintained;
- ◆ Approximately 30% UOX fuel assemblies may be saved;
- ◆ Low-leakage core is naturally composed, as the reactivity 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
- ◆ Without any other change to the fuel lattice and the core internals, but substituting the UOX pellet with thorium-based pellet.



## Roadmap for Thorium-Uranium Recycle in PWR



- ◆ The lattice code used in this paper is **DRAGON 3.06**,
- ◆ The library used for **DRAGON** is IAEA version of **WLUP** format microscopic cross-section library.
- ◆ The author is highly appreciating the originators of above products.



THANKS FOR  
YOUR  
ATTENTION

