*Article*

# Core Design and Deployment Strategy of Heavy Water Cooled Sustainable Thorium Reactor

**Naoyuki Takaki [1],\* and Deby Mardiansah [2]**

[1]  Department of Nuclear Safety Engineering, Tokyo City University, Tokyo, Japan
[2]  Department of Nuclear Engineering, Tokai University Tokyo, Japan;
    E-Mail: 9aarm010@mail.tokai-u.jp

\*  Author to whom correspondence should be addressed; E-Mail: ntakaki@tcu.ac.jp;
    Tel.: +81-3-5707-0104 (ext. 3805); Fax: +81-3-5707-1180.

**Abstract:** Our previous studies on water cooled thorium breeder reactor based on matured pressurized water reactor (PWR) plant technology concluded that reduced moderated core by arranging fuel pins in a triangular tight lattice array and using heavy water as coolant is appropriate for achieving better breeding performance and higher burn-up simultaneously [1–6]. One optimum core that produces 3.5 GW thermal energy using Th-$^{233}$U oxide fuel shows a breeding ratio of 1.07 and averaged burn-up of about 80 GWd/t with long cycle length of 1300 days. The moderator to fuel volume ratio is 0.6 and required enrichment of $^{233}$U for the fresh fuel is about 7%. The coolant reactivity coefficient is negative during all cycles despite it being a large scale breeder reactor. In order to introduce this sustainable thorium reactor, three-step deployment scenario, with intermediate transition phase between current light water reactor (LWR) phase and future sustainer phase, is proposed. Both in transition phase and sustainer phase, almost the same core design can be applicable only by changing fissile materials mixed with thorium from plutonium to $^{233}$U with slight modification in the fuel assembly design. Assuming total capacity of 60 GWe in current LWR phase and reprocessing capacity of 800 ton/y with further extensions to 1600 ton/y, all LWRs will be replaced by heavy water cooled thorium reactors within about one century then thorium reactors will be kept operational owing to its potential to sustain fissile fuels while reprocessing all spent fuels until exhaustion of massive thorium resource.

**Keywords:** thorium; heavy water; deployment scenario; transition; sustainer; $^{233}$U

## 1. Introduction

The price of uranium had shot up over $US 100 a pound in 2007 and recently fluctuated around $50. There is, however, no denying that it is interesting to rethink alternate promising fertile material that is applicable for sustainable energy supply. Additionally, in the rare-earths sector, thorium-bearing waste is continuously generated by the extraction from monazite sands, separation and refining operations. One option for the safe management of radioactive thorium waste instead of storing is utilization as nuclear fuel which is currently a request of the mining and nuclear industries.

The water-cooled plant technology is in common use throughout the world and LWRs are already well-established as commercial plants through extensive operation and maintenance experiences. The transparent coolant is desirable especially from the aspect of maintenance. If a breeder reactor is designed based on this common plant technology, there could be less requirement for investment and time for new technology development, education for operators and so on.

Several core design studies have been conducted to clarify the breeding potential of water cooled, oxide fuel thorium reactor based on current LWR technologies in our previous works. It was concluded that reduced moderated core by arranging fuel pins in triangular tight lattice array and using heavy water as coolant is appropriate for achieving better breeding performance and higher burn-up simultaneously. It is often written in textbooks that thorium fueled reactor can be designed as a breeder with thermalized spectrum because $^{233}U$ shows superior eta-value even in thermal energy regions. Our studies, however, indicated that optimum neutron spectrum for thorium breeder reactor is not in the thermal region but in the intermediate or sub-fast energy region. Based on such findings, core designs of thorium reactors in transition phase and sustainer phase are examined and the deployment strategy to mesh them with the current LWR cycle is studied here.
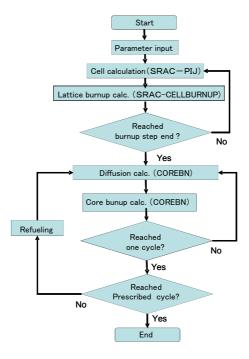
The objectives of this study are to: (1) design heavy water cooled reactor (HWR) cores used for the transition phase and sustainer phase; and (2) depict deployment scenario of thorium reactors by using plutonium generated in current LWRs.

## 2. Methods and Models

### 2.1. Neutronic Analysis

The nuclear data JENDL-3.3 have been employed in a series of analyses to generate effective cross sections. The calculation code used is the standard thermal reactor analysis code (SRAC) system [7]. For the SRAC analyses, cell calculations using the collision probability method are performed to collapse detailed 107 group microscopic cross sections into 10 effective group macro cross sections.

Core calculations are performed with the COREBN module in the SRAC System using collapsed 10 group cross section. COREBN is an auxiliary code of the SRAC system for multi-dimensional core burn-up calculation based on the diffusion theory and interpolation of macroscopic cross-section tabulated to local parameters such as burn-up, moderator temperature and so on. The numbers of nuclides treated in burn-up calculation is 29 for heavy nuclides and 66 for fission products, including one pseudo nuclide that represents other remaining fission products (FPs). The whole process of calculation is shown in Figure 1.

**Figure 1.** Flow of calculation.



The definition of the instantaneous conversion ratio used here is defined as the ratio of neutron capture rate by fertile to neutron absorption rate by fissile, as indicated in Equations (1,3). The neutron capture reactions by $^{232}$Pa impede generation of fissile $^{233}$U, so that the capture rate by $^{232}$Pa is subtracted in the numerator in Equation (1). This instantaneous quantity is referred to as breeding ratio (BR) if it is more than unity.

$$CR = \frac{\int\limits_{Core} \sum\limits_{g} \Sigma_{cg}^{fert}(r)\varphi_g(r)dV}{\int\limits_{Core} \sum\limits_{g} \Sigma_{ag}^{fiss}(r)\varphi_g(r)dV} \tag{1}$$

Here,

$$\Sigma_{cg}^{fert}(r) = \Sigma_{cg}(r)\left(^{232}Th + ^{234}U + ^{238}U + ^{240}Pu - ^{233}Pa\right) \tag{2}$$

$$\Sigma_{ag}^{fiss}(r) = \Sigma_{ag}(r)\left(^{233}U + ^{235}U + ^{239}Pu + ^{241}Pu\right) \tag{3}$$

Another index, fissile inventory ratio (FIR) that indicates fissile breeding gain through the residence period of fuels in a core is also introduced for the scenario analysis described in Section 4.
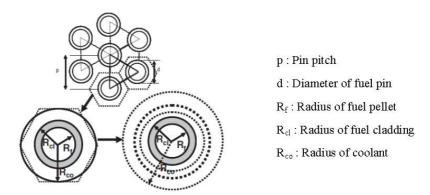
$$FIR = \left(\frac{Discharge \ \ Fissile \ \ Inventory}{Initial \ \ \ \ Fissile \ \ Inventory}\right) \tag{4}$$
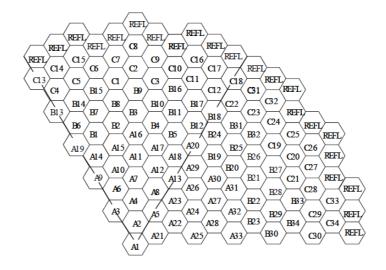
*2.2. Core Model*

As for thorium fueled reactor, intermediate or sub-fast neutron spectra are suitable for achieving high conversion or breeding ratios. This requires the core to have a low moderation, *i.e.*, narrow gaps between fuel pins. Therefore, a fuel pin arrangement in tight triangular lattice has been employed.

The cell model was a one-dimensional cylindrical model that is divided into radius of fuel, cladding and coolant, as shown in Figure 2. The radius of fuel and cladding were fixed and the radius of the coolant was changed to obtain different moderator to fuel volume ratio (MFR). Three dimensional HEX-Z model shown in Figure 3 is used for the hexagonal type of fuel assemblies.

**Figure 2.** Triangular lattice of fuel pins.



p : Pin pitch

d : Diameter of fuel pin

$R_f$ : Radius of fuel pellet

$R_{cl}$ : Radius of fuel cladding

$R_{co}$ : Radius of coolant

**Figure 3.** Core model with hexagonal fuel assemblies.



*2.3. Deployment Scenario Model*

In order to start up a thorium reactor, fissile materials have to be externally supplied and mixed with thorium in the fuel preparation process. Although enriched uranium is a promising material for this purpose, using enriched uranium spoils the unique performance of thorium core because of the prominent existence of $^{238}$U in the core. Nowadays the global stock pile of separated or non-separated civilian plutonium keeps growing through commercial operation of LWRs. Therefore it is anticipated to use plutonium as a form of Th-Pu oxide in reactors in the intermediate transition phase that bridges between the current LWRs phase and future thorium sustainer phase, as illustrated in Figure 4.

- In current phase, plutonium produced in LWRs is sent to fuel fabrication facility in transition phase after mixing with fertile thorium to exclude possible existence of plutonium in sole substance form.

- In transition phase, fissile $^{233}$U is generated from main substance of thorium. Recovered $^{233}$U by reprocessing process is sent to fuel fabrication facility in the next sustainer phase in a mixed form with thorium. Recovered plutonium is recycled in the closed cycle by mixing with makeup plutonium sent from current phase.
- In sustainer phase, Th-$^{233}$U fueled reactor is introduced by using $^{233}$U supplied from the transition phase cores. Once a Th-$^{233}$U reactor started up, it can be sustainably operational by recycling self-produced $^{233}$U.
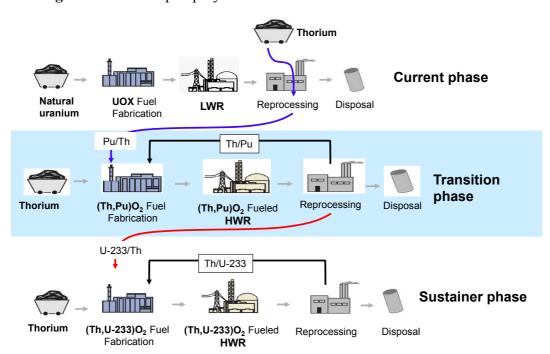
**Figure 4.** Three step deployment scenario of sustainable thorium reactor.



## 3. Core Designs

As there is no naturally occurring $^{233}$U, plutonium is used as igniter to produce $^{233}$U in the transition phase. Our previous study showed that production performance of $^{233}$U is superior for heavy water cooled reactor than for light water reactor. Therefore the authors tried to propose a heavy water cooled core that can be applicable both for transition phase and sustainer phase reactor. Key design specifications are listed in Table 1. The plutonium composition is that of reprocessed spent $UO_2$ fuel discharged from PWR with an average burn-up of about 45GWd/t. In this section, design parameters of heavy water core are adjusted for each phase core.

### 3.1. Core for Transition Phase

Before studying heavy water cooled reactor for transition phase, the authors examined the potential of light water reactor (PWR) fueled with Th-Pu oxides in terms of $^{233}$U breeding. Replacing uranium oxide (UOX) fuel by Th-Pu fuel in PWR core improves the internal conversion ratio and reduces burn-up reactivity loss [8]. This also results in increasing burn-up while keeping negative coolant coefficient. The evaluated production rate is, however, about 300 kg/year at most. The sustainer core in the next phase requires about 4 tons of $^{233}$U per 1GWe in output; therefore it should be concluded that

light water cooled PWR does not have sufficient potential to produce [233]U for smooth replacement of UOX LWRs by Th-[233]U fueled sustainer reactors. Therefore the following discussion on the core in transition phase focuses only on heavy water cooled reactor.

**Table 1.** Design specification of cores for transition phase and sustainer phase.

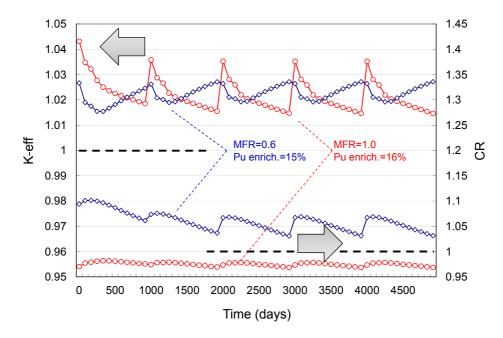| Parameters | Values |
|---|---|
| Thermal power output (GWt) | 3.0 for transition phase core |
| | 2.5–3.5 for sustainer phase core |
| Fuel pellet diameter (inner) (cm) | 0.82–1.31 |
| Fuel pin diameter (outer) (cm) | 1.24–1.41 |
| Moderator to fuel volume ratio (MFR) (-) | 0.5–1.0 |
| Fuel pin arrangement | Triangular lattice |
| Core diameter (cm) | 368–405 (function of MFR) |
| Core height (cm) | 370 |
| Coolant | Heavy water |
| Burn-up (GWd/t) | >70 |
| Cycle length (days) | 700–1300 |
| Batch number | 3 |
| Fuel material | Th-Pu oxide for transition phase core |
| | Th-[233]U oxide for sustainer phase core |
| Cladding | Zircaloy-4 |
| Pu vector (wt%) | [238]Pu: 2.70, [239]Pu: 50.40, [240]Pu: 24.10, [241]Pu: 15.20, [242]Pu: 7.10 |

One important design parameter moderator to fuel volume ratio (MFR) is surveyed in the range of 0.5 to 1.0 to achieve better conversion capability and negative void reactivity coefficient. The fuel pin diameter is kept constant and only fuel pin pitch is changed here.

Figure 5 shows the variation of the effective multiplication factor (k-eff) and conversion ratio for 5 cycle burn-up as a function of MFR. The cycle length is set as 1000 days. When MFR = 0.6, the k-eff is smaller than that of MFR = 1.0 at beginning of cycle (BOC), but it increases at the middle of each cycle until end of cycle (EOC) because the conversion ratio for MFR = 0.6 exceeds unity.

The obtained void reactivity coefficients at the 5th cycle are shown in Table 2. These coefficients are calculated by voiding 95% of water from flooded condition. The coefficients for smaller MFR are evaluated as positive, whereas negative values are achieved for higher MFR of 1.0.

The transition phase cores annually produce more than 800 kg of [233]U as summarized in Table 3. Therefore, it takes about 16 years to yield enough fissile material to start-up one 3,5 GWt rating sustainer phase core with [233]U fissile inventory of 13.5 tons. The core having MFR = 1.0 should be chosen from a view point of negative void coefficient, whereas better [233]U production capability is derived for less moderated core.

**Figure 5.** k-eff and conversion ratio for two different MFRs (Th-Pu fueled core for transition phase).



**Table 2.** Void reactivity coefficients (5th cycle) (Th-Pu fueled core for transition phase).

| MFR | BOC (% dk/kk'/%void) | EOC (% dk/kk'/%void) |
|---|---|---|
| 0.6 | +1.68e−4 | +1.60e−4 |
| 0.8 | +5.23e−5 | +6.55e−5 |
| 1.0 | −3.19e−3 | −3.85e−3 |

**Table 3.** Production rate of $^{233}$U (Th-Pu fueled core for transition phase).

| MFR | | 0.6 | 1.0 |
|---|---|---|---|
| Production rate of $^{233}$U (kg/cycle) | 1st cycle | 425 | 431 |
| | 2nd cycle | 1353 | 1267 |
| | 3rd cycle | 2489 | 2260 |
| | 4th cycle | 2439 | 2295 |
| | 5th cycle | 2431 | 2308 |
| Annual average rate (kg/year) | | 890 | 820 |

*3.2. Core for Sustainer Phase*

Through the preliminary survey, the enrichment of $^{233}$U in $^{232}$Th oxide fuel is determined as 8 wt% and cycle length is set to 700 days. Results obtained by two dimensional diffusion calculations are shown in Figure 6. Here the cooling and recycling time between cycles are ignored because of the small neutronic impact and for simplification of analysis.
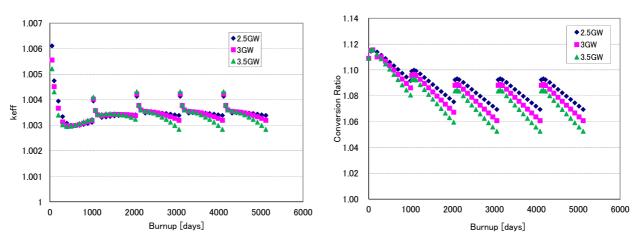
It is observed that for an MFR of 1.0, there is a sharp fall in the reactivity as a function of burn-up whereas this slope becomes more level for smaller MFR. In the case of MFR equaling 0.5, the core cannot remain critical. The curve of keff is almost level with 0.2% dk/k reactivity swing in an equilibrium cycle for an MFR of 0.6.The instantaneous conversion ratio for MFR of 0.6 is around 1.1 at the beginning of each cycle and remains above 1.07 at the end. The batch number is set to 3 and the refueling scheme employs the out-in method. The radial power peaking factor of less than 1.3 and negative void coefficient is derived. It is thus concluded that the desired MFR for high breeding potential and minimum reactivity swing is 0.6.

**Figure 6.** k-eff and conversion ratio for different MFR (Th-$^{233}$U fueled core for sustainer phase).
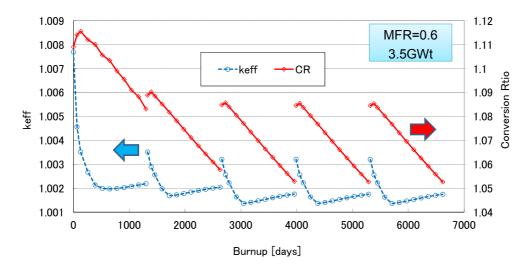


To understand the effect of thermal output on reactivity, it is changed in the range of 2.5 GWt to 3.5 GWt for the core of MFR = 1.0. The cycle length was also extended from 700 days to 1000 days for this survey and the result is illustrated in Figure 7. The values of k-eff at EOC for later cycle decreases by increasing power output, however criticality is maintained. The conversion ratio is also worse with power increase but is more than unity even in the highest case of 3.5 GWt. This behavior is caused by boosted neutron capture reactions of $^{233}$Pa and impeded productions of $^{233}$U for higher neutron flux that is proportional to power output. Considering these results, the thermal power of the thorium breeder is fixed at 3.5 GWt in the following discussion.

**Figure 7.** k-eff and conversion ratio for different power output (Th-$^{233}$U fueled core for sustainer phase, MFR = 1.0).

The flat reactivity for MFR = 0.6 shown in Figure 6 implies that there is room for enhancing burn-up which is a key performance affecting electricity generation cost. Hence similar evaluation on k-eff and conversion ratio for MFR = 0.6 core is also performed. To maintain conversion ratio more than 1.05, the cycle length is determined as 1300 days for the enhanced thermal power core. The core has superior neutronic performance, so that k-eff recovers with burn-up after rapid drop in the beginning of each cycle, as can be seen in Figure 8.

**Figure 8.** k-eff and conversion ratio of 3.5 GWt core with 1300 days cycle length (Th-$^{233}$U fueled core for sustainer phase, MFR = 0.6).



The void coefficient at the localized position in the core is calculated and summarized in Table 4. To calculate the coefficient, the coolant in a particular zone is voided by 95% then the inserted reactivity is divided by the voided fraction. It is observed that coefficient is negative in the whole region of the core.
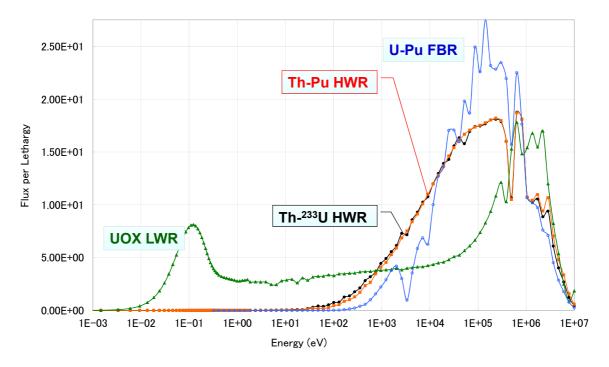
**Table 4.** Zone wised void reactivity coefficients (Th-$^{233}$U fueled core for sustainer phase).

|  | Inner Core Zone (% dk/kk'/%void) | Middle Core Zone (% dk/kk'/%void) | Outer Core Zone (% dk/kk'/%void) |
|---|---|---|---|
| Upper Zone (250~370 cm) | −1.68E−03 | −1.04E−03 | −4.64E−04 |
| Middle zone (170~250 cm) | −4.02E−03 | −2.52E−03 | −1.14E−03 |
| Lower zone (0~120 cm) | −1.51E−03 | −9.26E−04 | −4.10E−04 |

Figure 9 illustrates the neutron spectrum for Th-Pu core for transition phase and Th-$^{233}$U core for sustainer phase, in comparison with conventional UOX-LWR and sodium cooled, U-Pu oxides fuel fast breeder reactor. The spectra for two heavy water cooled cores almost correspond with each other. There are more neutrons in the energy range below 1keV in Th-$^{233}$U core than in Th-Pu core. Both cores have softer spectrum compared to fast breeder reactor (FBR), whereas they show much harder spectrum than LWR. Although the reproduction factor of neutrons (eta-value) for $^{233}$U depends

less on neutron energy than other fissile materials, harder spectra obtained in HWR enables better neutron economy because of less neutron consumption by structure materials and higher eta-value.

**Figure 9.** Neutron spectra for heavy water cooled thorium reactors in transition and sustainer phase in comparison with LWR and FBR.
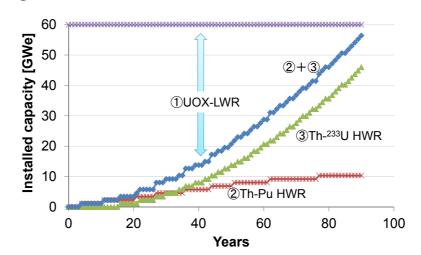


## 4. Deployment Scenario

In the deployment scenario study, introduced numbers of current LWRs, Th-Pu fueled HWR in transition phase and Th-$^{233}$U fueled HWR in sustainer phase are evaluated considering mass balance of required fissile materials and reprocessing capacity under the following assumptions [9].
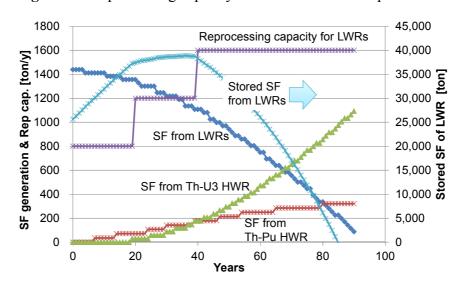
- Total nuclear capacity is kept at 60 GWe.
- At the beginning of the scenario, LWR fueled with low enriched uranium is a sole reactor type.
- At the beginning of the scenario, there is 25,000 tons of spent fuels stored.
- Reprocessing capacity for LWR spent fuel is 800 tHM/y in the early stage and can be flexibly increased by 400 tHM/y when it is necessary.
- In the beginning of the transition phase, current LWRs are gradually replaced by Th-Pu oxide fuel HWR if enough plutonium is accumulated for its start-up by reprocessing LWR spent fuels.
- Th-Pu HWR produces $^{233}$U and the reprocessing capacity for spent fuels discharged from Th-Pu HWR is assumed to be adjustable to the amount of spent fuels generated.
- In the beginning of the sustainer phase, remaining LWRs are gradually replaced by Th-$^{233}$U fueled HWR if enough $^{233}$U is accumulated for its start-up by reprocessing spent fuels of Th-Pu HWR.
- Time periods required for cooling after discharge, reprocessing and fabrication are assumed to be 5 years, 1year and 1year, respectively.

Figure 10 indicates the transition of reactor types in which current LWRs are gradually replaced by HWRs that utilize Th-Pu or Th-$^{233}$U as their fuel. By using plutonium obtained in LWR reprocessing plant, Th-Pu HWRs are deployed by replacing UOX-LWRs.

**Figure 10.** Transition from UOX-LWR to thorium fueled HWR.



The production of LWR spent fuel slows down with decreasing operational LWRs. However the stored LWR spent fuel accumulates as shown in Figure 11. To avoid huge accumulation of more than 40,000 tons of spent fuel, the reprocessing capacity is increased by 400 ton/y at 20 years elapsed from the beginning. This contributes to produce more plutonium and accelerate the introduction of Th-Pu HWR. Although the reprocessing capacity is increased by 400 ton/y again at 40 years, introduction pace of Th-Pu HWR is gradually decreased because the conversion ratio of Th-Pu HWR is 0.95 at the most. In contrast, Th-$^{233}$U HWRs increase exponentially and overtake Th-Pu HWR before 40 years has elapsed. This scenario analysis showed that 60 GWe UOX-LWRs can be replaced by thorium HWRs within a century.

**Figure 11.** Reprocessing capacity and mass balance of spent fuels.

## 5. Conclusions

Nowadays thorium is recognized as an obstacle to mining activities in the rare-earths sector. It is a by-product continuously generated by extraction from monazite sands, separation and refining operations. One option for the safe management of radioactive thorium waste is utilization as nuclear fuel.

It is feasible to design a sustainable thorium reactor, based on matured current LWR plant technology. The reactor core is thorium oxide fueled, heavy water cooled, PWR type, loaded with fuel pins in triangular tight lattice array. The optimum moderator to fuel volume ratio is in the range of 0.6 to 1.0.

To make thorium reactor sustainable energy source, fissile material mixed with fertile thorium must be $^{233}$U which is not a naturally occurring nuclide. Therefore enough $^{233}$U must be generated by transmuting $^{232}$Th in other preceding reactors in the intermediate transition phase that bridges between the current LWRs phase and future thorium sustainer phase, by using presently available fissile. The fissile material assumed in this study is civilian plutonium contained in stock piles of LWR spent fuels.

The design parameters for each thorium fueled core in the transition and sustainer phases are determined and the deployment scenario of thorium reactors is depicted. A Th-Pu fueled heavy water reactor core, designed for the transition phase, produces 3 GWt thermal energy and about 300 kg of $^{233}$U annually. The MFR of the core is set to 1.0 to satisfy both high conversion ratio of 0.95 and negative coolant reactivity.

A core designed for sustainer phase that produces 3.5 GW thermal energy using Th-$^{233}$U oxide fuel shows breeding ratio of 1.07 and average burn-up of about 80 GWd/t with long cycle length of 1300 days. The MFR is 0.6 and required enrichment of $^{233}$U for the fresh fuel is about 7%. The coolant reactivity coefficient is negative during all cycles despite it being a large scale breeder reactor.

In the deployment scenario study, introduced numbers of current LWR, Th-Pu fueled HWR in transition phase and Th-$^{233}$U fueled HWR in sustainer phase are evaluated considering mass balance of required fissile materials. The total reactor capacity is kept constant at 60 GWe. It is concluded that all LWRs will be replaced by thorium fueled reactors within a century. Although this time period requirement is longer than for sodium cooled fast breeder reactors, using thorium in a water cooled reactor requires less effort in developing commercial breeder plants and contributes to diversify fission resources.

## Conflict of Interest

The authors declare no conflict of interest.

## References

1. Sidik, P.; Takaki, N.; Sekimoto, H. Impact of different moderator to fuel ratios with light and heavy water cooled reactors in equilibrium states. *Ann. Nucl. Energy* **2006**, *33*, 561–572.
2. Takaki, N.; Sidik, P.; Sekimoto, H. *Feasibility of Water Cooled Thorium Breeder Reactor Based on LWR Technology*; American Nuclear Society: La Grange Park, IL, USA, 2007; pp. 1733–1738.

3.    Sidik, P.; Takaki, N.; Sekimoto, H. Feasible region of design parameters for water cooled thorium breeder reactor. *J. Nucl. Sci. Technol.* **2007**, *44*, 946–957.

4.    Sidik, P.; Takaki, N.; Sekimoto, H. Breeding capability and void reactivity analysis of heavy-water-cooled thorium reactor. *J. Nucl. Sci. Technol.* **2008**, *45*, 589–600.

5.    Sidik, P.; Takaki, N.; Sekimoto, H. Breeding and void reactivity analysis on heavy metal closed-cycle water cooled thorium reactor. *Ann. Nucl. Energy* **2011**, *38*, 337–347.

6.    Takaki, N.; Srivastava, A.; Takeda, A. Core Design of Heavy Water Cooled Thorium Breeder Reactor with Negative Void Reactivity and Improved Breeding Performance. In *Proceedings of ICAPP 2011*; Curran Associates, Inc.: Nice, France, 2011.

7.    Okumura, K. COREBN: A Core Burn-Up Calculation Module for SRAC2006. In *JAERI-Data/ Code 2007-003*; Japan Atomic Energy Research Institute (JAERI): Ibaraki-ken, Japan, 2007.

8.    Björk, K.I.; Fhager, V. Comparison of Thorium-Plutonium fuel and MOX Fuel for PWRs. In *Proceeding of the Global 2009*, Paris, France, 6–11 September 2009.

9.    Mardiansah, D. Deployment Scenario of Water Cooled Thorium Breeder Reactor. Master Thesis, Tokai University, Tokyo, Japan, 2011.