*Proceedings of the Reactor Physics Asia 2015 (RPHA15) Conference*
*Jeju, Korea, Sept. 16-18, 2015*

# A Comparative Physics Study of New Ultra-Long-Life Sodium-Cooled Cores using Uranium and Thorium Fuels

HaeLee Hyun and Ser Gi Hong*
[a]*Department of Nuclear Engineering, Kyung Hee University:1732 Deogyeong-daero,Giheung-gu, Yongin-si, Gyeonggi-do, 446-701, Korea*
*Corresponding author: sergihong@khu.ac.kr

## 1. Introduction

One of the distinct features of fast spectrum reactors is the fact that a significant breeding of fissile fuel materials can be achieved using fast neutrons, which can be effectively utilized to design ultra-long-life cores having operational cycle length of several tens of years without refueling [1,2,3]. Recently, we have suggested a new burning strategy using axial blanket-driver-blanket core configuration [4], which allows the ultra-long-life operation with satisfaction of self-controllability under unprotected accidents. In the new burning strategy, the power distribution was controlled by using partial use of dummy rods in some core regions in order to reduce sodium void reactivity worth. The objective of this work is to assess the neutronic feasibility of the ultra-long-life cores using uranium-thorium mixed fuel and to compare their neutronic characteristics with those of the uranium fueled core.

## 2. Description of the Actual Work

### *2.1 Computational Methodology*

The depletion analysis of the long-life cores was done with twenty five group cross sections and REBUS-3[5] non-equilibrium model. These twenty five group cross section were prepared by condensing the 150 group cross section library of ISOTXS format with the core region-wise neutron spectra and the TRANSX code [6]. The 150 group cross section library of MATXS format was generated at KAERI based on ENDF/B-VII.r0. The core region-wise neutron spectra were generated by using the DIF3D [7] R-Z diffusion calculation and the 150 group cross section. On the other hand, the core physics parameters such as temperature reactivity coefficients and sodium void reactivity worth were evaluated by using the VARIANT $SP_N$ option in DIF10.0 [7] and 80 group cross sections.

### *2.2 Core Design and Performance Analysis*

The binary metallic fuel of U-10Zr was considered both for the driver and blanket fuels due to its high heavy metal density. Also, the use of metallic fuel is useful to achieve hard neutron spectra which are useful in maximizing the breeding. The core rates 990 MWt power. The fuel rods are closely packed in a triangular pitch with wire-wrap inside a hexagonal duct. The number of fuel rods in an assembly is 127. The radial core configuration is given in Fig. 1. As shown in Fig. 1, the core consists of inner and outer core regions. Fig. 2 shows the axial cut view of the ultra-long-life core using uranium metallic fuel. As shown in Fig. 2, the inner core region fuel assembly is comprised of three axial regions : 1) lower blanket fuel, 2) central driver fuel, and 3) upper blanket fuel. The outer core region fuel assemblies are also comprised of lower blanket, central driver, and upper blanket regions. However, it should be noted that the driver region of the outer core region is taller than the one of the inner core region and that the driver region of the outer core is subdivided into two sub-regions. In the reference core using uranium fuels, 24 dummy rods per assembly in the lower blanket of inner core were used to shift the power distribution toward the upper sodium plenum at high burnup, which increases neutron leakage under sodium coolant voiding in order to reduce sodium void reactivity worth and the 6 dummy rods per assembly in the lower driver of outer core were used to reduce the burnup reactivity swing. In this work, the dummy rod is just the rod that is 100% structural material without fuel and its outer diameter is the same as that of the fuel rod.

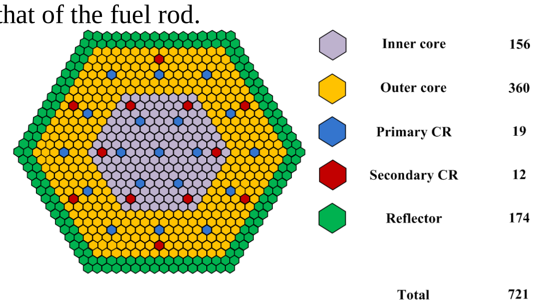



Fig. 1. Radial core configurations

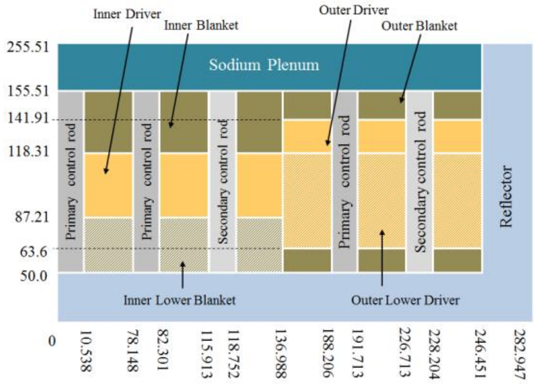



Fig. 2. Axial cut view of the reference uranium fueled ultra-long-life core

The main design parameters are summarized in Table I.

As shown in Table I, we employed thick fuel rods of 1.5 cm outer radius and lower power density to achieve ultra-long-life. Specifically, the lower power density is required to limit the peak linear power density within the typical limiting value (~500 W/cm) in sodium-cooled fast reactors. The heights of the driver fuels are optimized to achieve small burnup reactivity swing with ultra-long-life. The heights of driver fuels in the inner and outer cores were determined to be 30 cm and 75 cm, respectively. The use of thicker blanket fuel in the inner core than in the outer one is important to increase the core life by effectively using the leaking neutrons through outer driver fuel in the blanket fuels of the inner core where the radial neutron leakage is small.

Table I. Main design parameters

| Design parameter | Case A | Case B |
|---|---|---|
| Fuel type | | |
| - Driver | U-10Zr | U-10Zr |
| - Upper Blanket | U-10Zr | U-10Zr |
| - Lower Blanket | U-10Zr | 50Th-U-10Zr |
| Power (MWe/MWt) | 390/990 | 390/990 |
| Average linear power density (W/cm) | 157.4 | 151.1 |
| Driver height (cm, Inner/Outer Driver) | 30/75 | 30/75 |
| Number of rods per FA (fuel/dummy) | | |
| - Inner Blanket, Outer Blanket | 127/NA | 127/NA |
| - Inner Driver | 127/NA | 127/NA |
| - Inner Lower Blanket | 103/24 | 127/NA |
| - Outer Driver | 127/NA | 127/NA |
| - Outer Lower Driver | 121/6 | 127/NA |
| Duct wall thickness (mm) | 3.5 | 3.5 |
| Assembly pitch (cm) | 20.07 | 20.07 |
| Rod outer diameter (mm) | 15.0 | 15.0 |
| Wire wrap diameter (mm) | 1.4 | 1.4 |
| Clad thickness (mm) | 0.55 | 0.55 |

In this work, metallic thorium fuel is considered to check the possibility of thorium fuel to improve the performances of the ultra-long-life core. Actually, we tried to use the thorium fuel in the driver fuels but we failed to achieve ultra-long-life core. As the result, we determined to use metallic uranium-thorium mixed fuel (i.e., only Th-U-10Zr) in the lower blankets both of the inner and outer cores in order to further reduce sodium void worth by further shift the power distribution toward the upper sodium plenum region without loss of core life. The radial configuration of the ultra-long-life core using uranium-thorium mixed fuel in lower blankets is the same as that of the uranium fueled core but we could simplify the axial design of the fuel with uranium-thorium mixed fuel in lower blankets. The axial cut view of this new core is given in Fig. 3 and its main design parameters are compared in Table I. As shown in Table I, we completely removed the dummy rods in the new core using uranium-thorium mixed fuel in lower blankets while the other design parameters are all the same as those of the reference uranium fueled core except for the fact that the average linear power density is reduced from 157.4 to 151.1 W/cm due to the removal of the dummy rods. Fig. 4 compares the evolutions of k-effective versus time of the new cores with that of the reference uranium fueled core. In particular, Fig. 4 shows the effect of the thorium content in the lower blanket. As expected, as thorium content increases in lower blanket, the core life gets shorter. Basically, the removal of the dummy rods increases maximum value of k-eff (so, increases burnup reactivity swing) in comparison with the reference core. Even if thorium content increases up to 60 wt%, the initial peak value of k-eff is higher than that of the reference core. The second peaks of the k-eff curve are reduced with higher thorium content because the second peaks are generated by the breeding in the blanket and the thorium blanket has lower breeding capability.

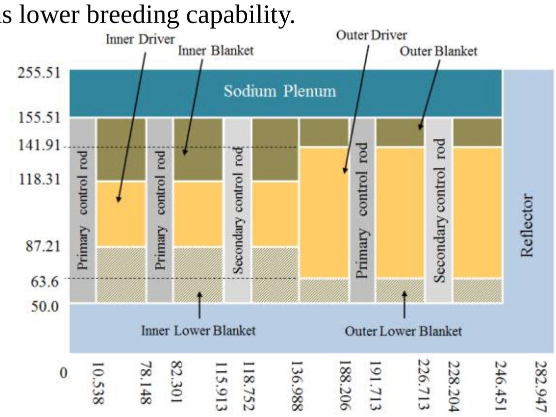



Fig. 3. Axial cut view of the ultra-long-life core using thorium fuel in lower blankets

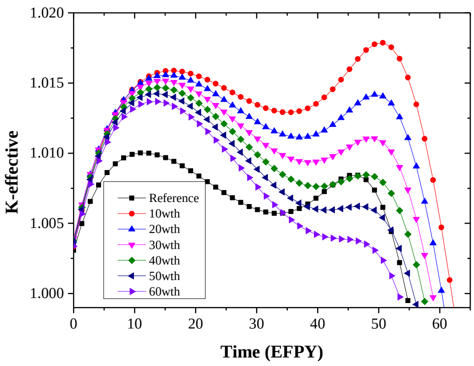



Fig. 4. Comparison of the evolutions of $k_{eff}$ over time

Fig. 5 shows the effects of thorium content in the lower blanket fuel on the core average discharge burnup and sodium void worth at EOL versus thorium contents. As shown in Fig. 5, the sodium void worth and the core average discharge burnup monotonically decrease as thorium content in the lower blanket increases. The sodium void worth given in this figure was evaluated by assuming the 50% voiding of sodium coolants both in the active core regions and upper sodium plenum because some numerical instabilities were occurred with DIF3D calculation for the full sodium voiding. In the future, we will consider the sodium void worth evaluation using MCNP transport calculation. With consideration of the discharge burnup, peak power density at EOL (End of Life), sodium void worth at EOL, and burnup reactivity swing, we determined the case using 50 wt% thorium in lower blanket as the best candidate core. The performances of the new cores having different thorium contents are summarized in Table II.

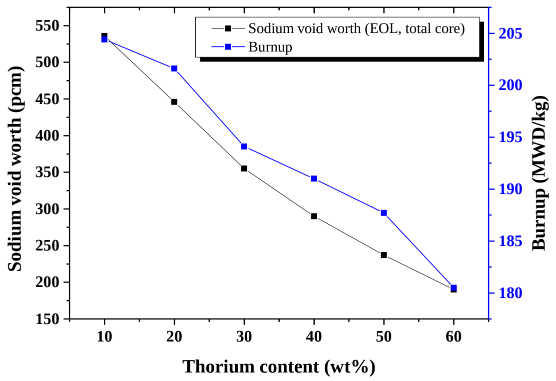



Fig. 5. Comparison of the sodium void worth and burnup

As shown in Table II, the new core using 50 wt% thorium contents in lower blanket has comparable core life of 54.7 EFPYs (Effective Full Power Years) to the reference core. This core using 50 wt% thorium in lower blanket has a larger burnup reactivity swing by 269 pcm and slightly higher core average discharge burnup of 188 MWD/kg than the reference core. The initial uranium enrichments of the driver fuel of these cores are nearly the same each other (~13.3 wt%). The new core using 50 wt% thorium in lower blanket has a smaller peak power density of 462 W/cm than the reference core. Table III summarizes the temperature reactivity coefficients, sodium void worth both at BOL (Beginning of Life) and EOL, and control rod worth of the reference core and new core having 50 wt% thorium content in lower blanket. The new core has slightly less negative Doppler coefficients, similar values of reactivity coefficients by fuel and core radial expansions, and less negative reactivity coefficients by coolant expansion at EOL than the reference core. It should be noted that the both cores have large negative sodium void worth for full sodium coolant voiding at BOL. These negative sodium void worths at BOL are typical trend for the general ultra-long-life cores using blanket. On the other hand, it has been known that the ultra-long-life cores have typically large positive values of sodium void worth at EOL because these cores have high TRU contents at EOL due to their high discharge burnup and core shapes having small neutron leakage. On the other hand, Table III shows that our ultra-long-life cores have relatively small sodium void worths at EOL and the new core has smaller sodium void worth than the reference core at EOL.

Table II. Performances of long-life cores

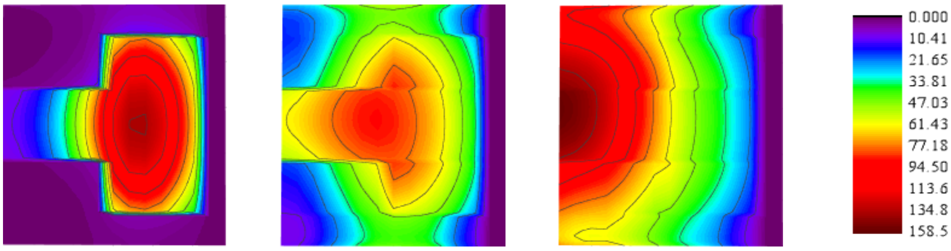
| Parameter | Reference | 10wt% Th case | 30wt% Th case | 50wt% Th case |
|---|---|---|---|---|
| Cycle Length, effective full power(EFPY) | 53.4 | 61.6 | 57.5 | 54.7 |
| Reactivity swing (pcm) | 991 | 1755 | 1494 | 1260 |
| Average discharge burnup (MWD/kg) | | | | |
| - Total Core | 181.8 | 204.3 | 194.1 | 187.7 |
| - Driver | 227.9 | 247.6 | 237.3 | 230.2 |
| - Blanket | 109.0 | 135.7 | 122.5 | 114.5 |
| Average conversion ratio over life | 0.95 | 0.94 | 0.95 | 0.95 |
| TRU weight fraction (wt%, EOL) | | | | |
| - Driver/Blanket | 10.9/10.5 | 11.4/10.6 | 11.1/9.1 | 10.9/8.0 |
| Volumetric power density (W/cm$^3$) | 51.7 | 51.7 | 51.7 | 51.7 |
| Heavy metal inventories (t) | 103.5 | 106.5 | 104.6 | 103.1 |
| Initial uranium enrichment (driver/blanket, %) | 13.35/0.2 | 13.03/0.2 | 13.08/0.2 | 13.25/0.2 |
| Peak linear power density (W/cm, BOL/EOL) | 409/496 | 403/494 | 406/466 | 407/462 |
| Fast neutron fluence (n/cm$^2$) | 1.32x10$^{24}$ | 1.48x10$^{24}$ | 1.41 x10$^{24}$ | 1.37x10$^{24}$ |

Table III. Reactivity coefficients and results of quasi-static reactivity balance analysis (BOL/EOL)

| Parameter | Reference | 50wt% Th case |
|---|---|---|
| Fuel Doppler coefficient (pcm/K, 890K) | -0.537/-0.510 | -0.511/-0.464 |
| Radial expansion coefficient (pcm/K) | -0.771/-0.745 | -0.777/-0.729 |
| Fuel axial expansion coefficient (pcm/K) | | |
| - Fuel only | -0.598/-0.523 | -0.601/-0.514 |
| Coolant expansion coefficient (pcm/K) | -0.065/0.489 | -0.058/0.439 |
| Sodium void worth (pcm, BOL/EOL) | | |
| - Total core | -1471 $^a$/368 $^b$ | -1406 $^a$/248 $^b$ |
| - Inner core | -197 $^a$/280 $^b$ | -204 $^a$/191 $^b$ |
| - Outer core | -1394 $^a$/42 $^b$ | -1321 $^a$/3 $^b$ |
| Control rod worth (pcm, BOL/EOL) | | |
| - Primary | 5677/6488 | 5506/6563 |
| - Secondary | 2029/2248 | 1888/2428 |
| Results of quasi-static reactivity balance analysis | | |
| A (pcm, <0) | -196/-155 | -167/-147 |
| B (pcm, <0) | -226/-158 | -211/-155 |
| C (pcm/℃,< 0) | -2.144/-1.267 | -1.947/-1.268 |
| A / B (≤ 1 ) | 0.868/0.983 | 0.790/0.948 |
| (≤1)/B(≤2) | 1.471/1.267 | 1.430/1.270 |
| $\rho_{ex}$ /\|B\| (≤1 ) | 0.319/NA | 0.433/NA |

$^a$ 100% sodium voiding $^b$ 50% sodium voiding

Table III also compares the results of the quasi-static reactivity balance analyses for the two cores to show the inherent safety features in terms of the self-controllability under unprotected accidents such as ULOF (Unprotected Loss Of Flow), ULOHS (Unprotected Loss Of Heat Sink), and UTOP (Unprotected Transient Over Power). This quasi-static reactivity balance analysis method was originally proposed by Wade and Hill [8] to determine the self-controllability of the sodium cooled metallic fueled cores under the unprotected accidents. The self-controllability means that the reactor leads to a passively safe shutdown state only through reactivity feedback effects without exceeding the limits ensuring core integrity. In Table III, the quantities A, B, and C are functions of the reactivity coefficients given in Table II. These quantities should be negative for the self-controllability. Also, three additional conditions for the self-controllability are given in Table III. As shown in Table III, the both cores satisfy all the conditions for the self-controllability both at BOL and EOL. Our study showed that the most difficult condition to be satisfied is the fourth condition (i.e., $A/B \le 1$) at EOL due to the positive values of the reactivity coefficient by coolant expansion. Finally, Fig. 6 compares the R-Z power distributions of the new core using 50 wt% thorium in lower blanket at 0, 34 EFPY, and 54.7 EFPY (i.e., EOL). As shown in Fig. 6, the power is mainly generated from outer driver at BOL and then it propagates toward the inner region and upper region. At EOL, the power distribution is shifted toward upper sodium plenum region due to the use of uranium-thorium mixed fuel in lower blankets as expected.




(a) 0 year (b) 34year (c) 54.7year

Fig. 6. R-Z power distribution of the cores at several burnup time points (Unit: W/cm$^3$)

## 3. Summary and Conclusions

In this work, we designed new sodium-cooled ultra-long-life core using uranium-thorium mixed fuel. In particular, the uranium-thorium mixed fuel are used in the lower blanket fuels of our ultra-long-life core which led to the simplification of the axial core configuration through the complete removal of the dummy rods in the original design. The neutronic analyses showed that the new core using 50 wt% thorium content in lower blanket has ultra-long-life of 54.7 EFPYs, high core average discharge burnup of 188 MWD/kg, and small burnup reactivity swing of 1260 pcm. In particular, the new core has smaller sodium void worth and smaller reactivity coefficient by coolant expansion at EOL than the original uranium fueled core, which led to the satisfaction of all the conditions for self-controllability under the unprotected accidents.

## Acknowledgements

This work was supported from National Research Foundation (NRF) of Korea through project No. NRF-2014M2B2A9030201.

## References

1. Greenspan E, "A Phased Development of Breed-and-Burn Reactors for Enhanced Nuclear Energy Sustainability," *Sustainability*, **4**, 2745-2764; doi:10.3390/su4102745 (2012).
2. Sekimoto H, et al., "CANDLE: The New Burnup Strategy," *Nucl. Sci. and Eng.*, **139**, 306 (2001).
3. Tak T et al., "Optimization Study of Ultra-Long Cycle Fast Reactor Core Concept," *Ann. Nucl. Energy*, **73**, 145 (2014).
4. Hyun H, Hong SG, "Physics Study of Sodium Cooled Ultra-Long-Life Core using New Driver-Blanket Burnup Strategy," submitted to the 2015 ANS Winter Meeting.
5. Toppel BJ, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, ANL, IL, USA (1983).
6. MacFarlane RE, "TRANSX 2: A Code for Interfacing MATXS Cross-Section Libraries to Nuclear Transport codes," LA-12312-MS,LANL,NM,USA. (1992).
7. Derstine KD, "DIF3D 10.0: Code System Using Variational Nodal Methods and Finite Difference Methods to Solve Neutron Diffusion and Transport Theory Problems," CCC-784, ANL, IL, USA (2011).
8. Wade DC, Chang YI, "The Integral Fast Reactor Concept : Physics of Operation and Safety," *Nucl. Sci. and Eng.*, **100**, 507 (1988).