Neutronic Analysis of a Small Breed-and-Burn Fast Reactor Cores using Recycling Options

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I. Introduction

Recently, there have been a lot of researches [1][2][3] and interests on fast reactors as the way to improve utilization of uranium resources and to mitigate nuclear waste problems. In particular, spent fuel of LWRs can be reused as nuclear fuel in fast reactors through reprocessing, and such a coupling cycle of PWRs and fast reactors can resolve much of the back-end fuel cycle issues.

In this work, a small ultra-long-life sodium cooled fast reactor core loaded with TRU from PWR spent fuels is neutronically designed and analyzed. In addition, we considered multi-cycling and partial recycling strategy with dry processing such as AIROX [4] (Atomics International Reduction OXidation) and pyro-processing which have good proliferation resistance. By applying the multi-recycling with proliferation resistant processing, we intended to achieve high discharge burnup and to mitigate the fuel integrity issues under ultra-long-life operation.

II. Computational Methods and Models

We performed the core neutronic calculations including depletion analysis with the Monte Carlo reactor physics code 'SERPENT' [5] developed by VTT and modeled full heterogeneities of the core down to fuel pellets and claddings. In particular, the accurate consideration of neutron leakage through the small size fast reactor core is quite important to achieve a high level of accuracy in the core design and analysis. In all the calculations, we used the point-wise ENDF/B-VII.r0 cross section libraries provided by SERPENT. For the depletion calculations, we considered five axial zones in active core region and each fuel assembly was assigned to single radial depletion zone. Therefore, a single depletion zone is assigned to each axial zone of all the fuel rods in a fuel assembly. We used 400 active cycles and 100 inactive cycles with 10000 particles for each cycle, which led to a statistical standard deviation of ~30 pcm during depletion calculations and one depletion step is set to one year. To automatically apply recycling and shuffling process to SERPENT input file, a small in-house program was made to reduce the mistakes in preparing the input files. In this study, pyro-process and AIROX processes are used for reprocessing the discharged fuels. The AIROX process is a dry oxidation-reduction process for the oxide fuel that uses only gaseous and solid materials. In this work, it is assumed that AIROX process is applicable to metallic fuel with some additional processes. During oxidation and reduction processing of AIROX, some volatile and semi volatile fission products are removed. In case of pyro-process, we assumed that all fission products are removed and then the depleted uranium is supplied to make up the heavy metal consumption. During AIROX process, it is assumed that volatile fission products ^{3}H, ^{14}C, Kr, and Xe isotopes are completely removed while 90% of the semi-volatile fission products Ru and Cs isotopes, and 75% of the semi-volatile ones Cd, In, and Te are removed. The removal fractions for each volatile and semi-volatile fission products during AIROX are shown in Table I. However, the makeups for removal of the volatile fission products are not considered.

Table I. Removal rate of volatile nuclides (AIROX)

Removal rates	Volatile FP
100%	H^3, C^{14}, Kr, Xe
90%	Ru, Cs
75%	Cd, In, Te

III. Core Design and Performance Analysis

III.A. Description of Core Design

We considered a small SFR core having 330MW thermal output. A ternary metallic fuel of TRU-U-22Zr are employed for all fuel regions. The configuration of the core is shown in Fig. 1. The fuel regions are divided into four concentric annular regions (Regions A, B, C, and D). For the first cycle, a single fuel composition of 14wt% TRU-U-22Zr are loaded in all the regions at BOC. Two rings of the lead (Pb) reflector assemblies surround the active core regions and they are surrounded by the radial shield assemblies. The reactivity of the core is controlled using 13 control assemblies which are not yet optimized at present.

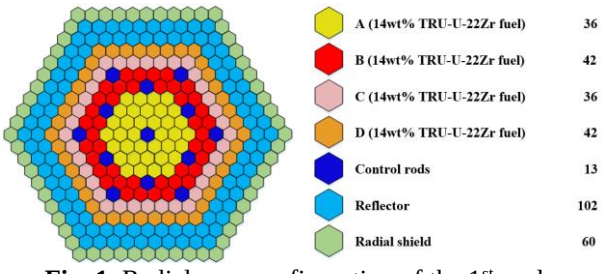

Fig. 1. Radial core configuration of the 1st cycle.

Table II summarizes the main design parameters of the reference core. The active fuel length is 100 cm and fat fuel rods of 1.5 cm outer diameter are adopted for achieving high breeding ratio. Each fuel assembly is comprised of 169 fuel rods within a 3.5 mm thick outer hexagonal duct. The average linear heat generation rate is 125 W/cm and average volumetric power density is 48.8 W/cm^3. As shown in Table I, the fuel volume fraction for the fuel assemblies is very high (i.e., 64.1%) for achieving ultra-long-cycle length. This fuel volume fraction includes the sodium bond region between fuel slug and cladding. In addition, the composition of the TRU nuclides in the initial loading was assumed to be that of the PWR spent fuel with 50 MWD/kg burnup and 10-year cooling. In this work, we considered three different schemes for recycling and shuffling of the spent fuels from the core and they are described in Figs. 2 and 3.

Table II. Main design parameters

Parameters	**Values**
Power (MWe/MWt)	130/330
Average linear heat generation (W/cm)	125
Average volumetric power density (W/cc)	48.8
Active core height (cm)	100
Number of rods for each fuel assembly	169
Fuel rod outer diameter (cm)	1.5
Cladding thickness (mm)	0.55
Fuel smear density (% of theoretical density)	75
Fuel rod pitch (cm)	1.55
Pitch-to-diameter (P/D) ratio	1.03
*Duct thickness (*mm)	3.5
Fuel assembly pitch (cm)	21.493
Volume fractions for fuel assemblies (%)	
Fuel/structure/coolant	64.1/22.2/13.7
Reflector composition (volume fractions (%))	
Pb/coolant/structure	93.1/3.7/3.2
Control rod assembly composition (volume fractions (%))	
[a]B_4C/coolant/structure	45.9/44.5/9.6

[a]B_4C : B-10 enrichment: 60 wt%.

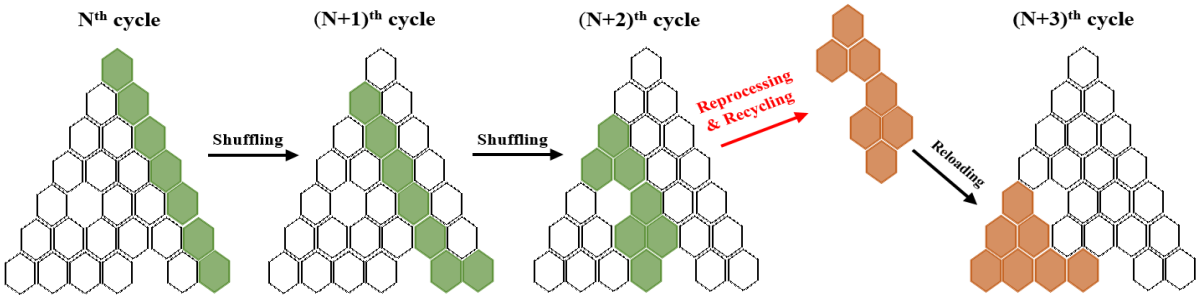

Fig. 2. Shuffling strategy for Schemes I & II (1/6 core)

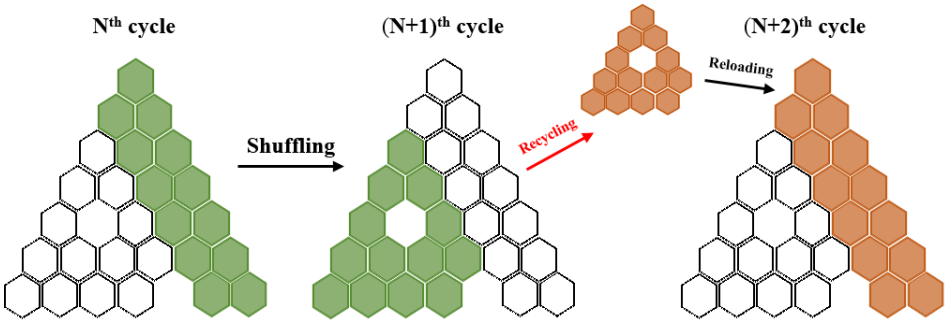

Fig. 3. Shuffling strategy for Scheme III (1/6 core)

Fig. 2 explains the shuffling and partial recycling schemes used in the first (Scheme I) and the second (Scheme II) schemes. These two schemes use four batch fuels. The fresh batch fuel assemblies are loaded in the outermost region of the core and they are burned for three subsequent cycles with movement to the inner region at the end of each cycle. Finally, three times burnt batch fuel assemblies are discharged and reloaded into the central region after reprocessing. The reloaded batch fuel assemblies are finally discharged after one additional cycle depletion. The difference between Schemes I and II is the fact that the AIROX process is applied to the first scheme while the pyro-process to the second one. In these two schemes, it should be noted that they are not closed cycle because the discharged fuels after additional one cycle are not re-used. Unlike the previous two schemes, as shown in Fig. 3, the last scheme (Scheme III) is a closed fuel cycle consisting of two fuel batches. For this scheme, after each cycle, the fuel batch placed in the outer region is moved into the inner fuel region without processing while the one placed in the inner region is reprocessed with pro-processing and then reloaded to the outer fuel batch. This shuffling including pyro-processing is continued by supplying only depleted uranium after the 1st cycle.

The main performance parameters for the core using these schemes are summarized in Table III. The evolutions of k_{eff} for the first cycle and the equilibrium cycles for three different schemes. The first cycle which is common for three different schemes has desirable shape of the excess reactivity change having small burnup reactivity swing of 1335 pcm over 24 EFPYs. The equilibrium cycle of the first scheme has the smallest cycle length of 11 EFPYs and its k_{eff} monotonically decreases as time. Its shortest cycle length is due to the fact that AIROX process is applied without makeup of depleted uranium. On the other hand, the equilibrium cycle for the second scheme has much longer cycle length of 19 EFPYs than that of the first scheme. Also, it is noted that k_{eff} initially increases due to higher breeding resulted from the makeup of depleted uranium. This equilibrium cycle of the second scheme has 2456 pcm burnup reactivity swing. The equilibrium cycle of the last scheme using two batch fuels has the longest cycle length of 22 EFPYs comparable to the first cycle but it has the largest burnup reactivity swing of 3909 pcm. The large initial reactivity and long cycle length of this cycle are due to the fact that half of the fuels are reprocessed with pyro-processing and so large amount of depleted uranium makeup is required. In Table III, the cycle average burnups for each fuel region are analyzed. The first cycle has high average burnup of 91.9 MWD/kg. The equilibrium cycle of the first scheme has small cycle average core burnup of 45.5 MWD/kg but it has high discharge burnup of 104 MWD/kg before reprocessing and the one of the second scheme has much higher discharge burnup of 169 MWD/kg due to much longer cycle length than the first scheme.

Table III Comparison of the core performances

Parameters	1st cycle	Scheme I (10th cycle)	Scheme II (10th cycle)	Scheme III (7th cycle)
Cycle length (EFPY)	24	11	19	22
Burnup Reactivity swing (pcm)	1335	1451	2456	3909
Cycle Average burnup (MWD/kg)				
Total core	91.9	45.5	76.0	86.8
Region A	145.8	75.9	128.3	124.9
Region B	104.0	51.7	88.2	99.7
Region C	72.4	34.2	55.9	72.9
Region D	50	23.0	36.3	53.0
Discharge burnup (MWD/kg)	N/A	104	169	168

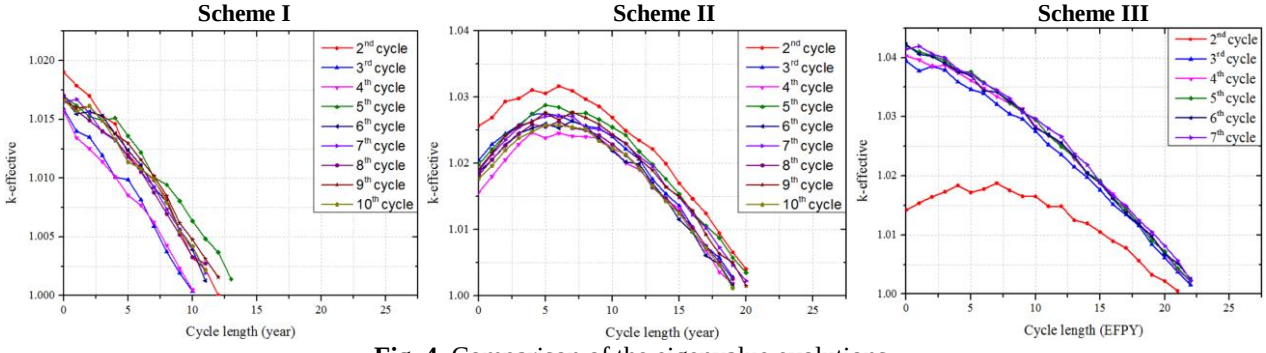

Fig. 4. Comparison of the eigenvalue evolutions.

Table IV. Comparison of the reactivity coefficients

Parameters	Scheme I (10^{th} cycle)	Scheme II (10^{th} cycle)	Scheme III (7^{th} cycle)
Fuel axial expansion (pcm/K)	[A]-0.3105/ [B]-0.2515	-0.3108/-0.2586	-0.2766/-0.2457
Radial expansion (pcm/K)	-0.8816/-0.9123	-0.8641/-0.9049	-0.8472/-0.9027
Sodium coolant expansion (pcm/K)	0.4067/0.4584	0.3895/0.4181	0.4319/0.4255
Fuel Doppler coefficient (pcm/K, 900K)	-0.2226/-0.2350	-0.2801/-0.1606	-0.3045/-0.3498
Sodium void reactivity worth (pcm)	1230/1330	1176/1270	1112/1261
Effective delayed neutron fraction	0.00343/0.00336	0.00347/0.00333	0.00346/0.00337

[A]Values at BOC.
[B]Values at EOC.

Table IV summarizes the reactivity coefficients including sodium void reactivity worth. As shown in Table IV, all the reactivity coefficients except for those by sodium coolant expansion are negative for all the cases. All cases have positive sodium void reactivity worth of about 1200 pcm at both BOC and EOC, but these positive sodium void worth are less than 4$.

For Schemes I and II, the finally discharged fuel batch is disposed to repository while the only fission products contained in the discharged fuel batch for Scheme III are disposed to repository after reprocessing. Table V summarizes the inventory of the discharged batch fuel for Scheme III, which shows the amount of the disposed fission products is ~2744.3 kg.

Table V. Nuclide-wise inventory of discharged fuel batch for Scheme III

Nuclide	Mass (kg)
U-234	29.3
U-235	6.5
U-236	11.0
U-238	11052.6
Np-237	18.8
Pu-238	43.5
Pu-239	1297.4
Pu-240	560.8
Pu-241	45.2
Pu-242	65.9
Am-241	55.6
Am-243	20.3
Cm-244	9.3
FP	2744.3
Total	20398.6

IV. Summary and Conclusions

In this study, the feasibility of multi-recycling on a small ultra-long-life SFR core is neutronically analyzed. In particular, three different shuffling schemes having AIROX or pyro-processing are considered and the reload core analysis from initial to equilibrium cycles is performed. From the study, it is concluded that the multi-recycling on the small ultra-long-life SFR core is neutronically feasible and equilibrium cycles can be achieved with long cycle lengths and negative reactivity coefficient except for small positive sodium void reactivity. Of them, a simple two batch scheme using pyro-processing has desirable features such as ultra-long-cycle length, high fuel burnup, and a small amount of actinide waste stream going to a final disposal repository due to its close fuel cycle.

Acknowledgement

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