Neutronic Analysis of the Moderator Effect for an Ultra Long Cycle SMSFR (Small Modular Sodium-cooled Fast Reactor)

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1. Introduction

Recently, there have lots of interest in ultra-long cycle fast spectrum cores which can be operated over several tens of years without refueling.[1] These type reactors have several good features over the typical 1~2 years cycle operation such as high fuel economy and small amount of spent fuel generation per energy. Recently, we also have designed SMSFR (Small Modular Sodium-cooled Fast Reactor) which has the concept of long-life reactor without refueling. It can be operated for more than 30 years. This core rates 330MWt and its uses U-TRU-10Zr driver and Th-10Zr blanket metallic fuels. At present, the core is not fully optimized.

The main objective of this work is to analyze the effects of moderator pins in fuel assemblies on the core physics characteristics.[2] For this purpose, we designed three different cases having different number of the ZrH_2 moderator pins. We analyzed the effects of the ZrH_2 moderator pins on the reactivity change over cycle, cycle length, burnup, power distribution, sodium void worth, and neutron spectra. In particular, we analyzed the sodium void worth using nuclide-wise reactivity decomposition based on the neutron balance obtained with the Serpent Monte Carlo code.

2. Computational methods and core design

2.1 Computational methods

The depletion analysis of the core was done using the Serpent2 Monte Carlo code which was developed by VTT.[3] In particular, the geometric modeling using Serpent was performed with explicit heterogeneous representation down to the fuel or moderator rods to take into account the heterogeneity effect caused by moderator pins. The ENDF/B-VII.r0 point-wise cross section library was used for all the depletion calculations and core physics parameters. We used 100 inactive and 1000 active cycles with 70000 histories each both for depletion calculation giving ~8 pcm statistical errors. Each assembly was treated as depletion zone and the active core was divided into eight axial depletion zones. The depletion time step size is one year.

2.2 Core design model

We designed SMSFR (Small Modular Sodium-cooled Fast Reactor) of which thermal power is 330MWt. This core uses both the driver and thorium blanket fuels. The driver fuel is metallic fuels of U-TRU-10Zr where uranium is depleted uranium and TRU is from PWR spent fuel of 50 MWd/kg burnup and 10 years cooling time. The thorium blanket is Th-10 Zr binary metallic fuel. 169 fuel rods are arranged with a triangular lattice structure within a 3.5 mm thick HT9 duct. 75% smear density was used to consider swelling of the metallic fuels. The active fuel length is 100 cm and the fuel outer diameter is 1.37 cm. The main design parameters are summarized in Table I. The core uses a tight triangular lattice of P/D ratio of 1.13 to achieve an ultra-long cycle. The average linear heat rate and volumetric power density are 112.2 W/cm and 44.7 W/cm^3, respectively, which are chosen to achieve ultra-long cycle.

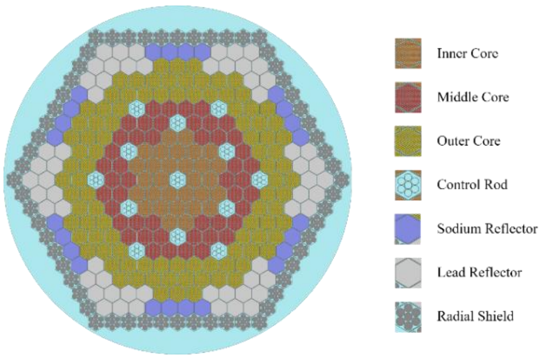

Fig. 1. Radial core layout

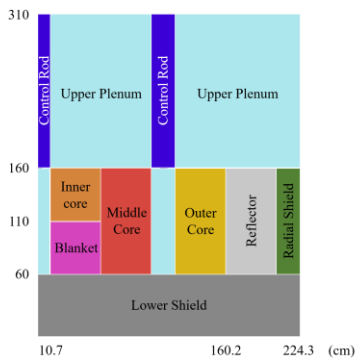

Fig. 2. Axial core layout

As shown in Fig. 1 and 2, the active core is divided into three regions. The innermost region (inner core) is loaded with axially two-region fuel rods comprised of lower thorium blanket (i.e., Th-10Zr) and upper TRU-U-10Zr driver fuels. This innermost region is surrounded by two successive driver regions (i.e., middle and outer cores). The active cores are surrounded by lead reflectors and sodium ducts. The sodium ducts are designated as sodium reflector in Fig. 1 and they are considered to reduce sodium void worth. For the driver fuels in the inner core, the initial TRU content is fixed to 16wt% TRU while the TRU contents of the driver fuels in the middle and outer cores are adjusted to achieve initial effective multiplication factor (k_{eff}) of 1.005. In the fast reactor society, there have been several studies that consider ZrH_2 or BeO moderator rods to reduce sodium void worth in TRU or Pu burner or breakeven cores. However, to our knowledge, there have no studies to use moderator rods in ultra-long-cycle cores. In this work, we considered three (Case 2) and seven (Case 3) ZrH_2 moderator rods per each fuel assembly only in the middle core as shown in Fig. 2. In addition, we also considered the core having no moderators (Case 1).

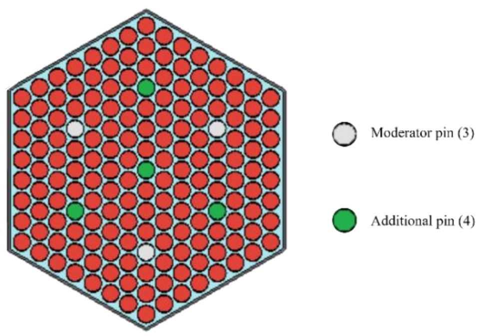

Fig. 3. Configuration of the fuel assembly having moderator pins

Table I. Main design parameters of the core

Design parameter	*Value*
Power (MWe/MWt)	130/330
Active core height (cm)	100
Active core radius (cm)	153.38
Average LPD (W/cm)	112.22
Average PD (W/cc)	44.65
Fuel type	U-TRU-10Zr
Blanket type	Th-10Zr
Number of rods per FA	169
Smear density of fuel (%)	75
Fuel pin outer diameter (cm)	1.37
Cladding thickness (mm)	0.55
Fuel pin pitch (cm)	1.55
P/D ratio	1.13
Duck wall thickness (mm)	3.5
Assembly pitch (cm)	21.363
Volume fraction (%) (fuel/coolant/structure)	53.3/33.8/12.9
Moderator type	ZrH_2

3. Results

The TRU contents in the middle and outer cores giving initial k_{eff} of 1.005 are estimated to be 13.64 and 13.83wt% for Cases 1 and 2, respectively, which means the core having moderator rods requires higher fissile content due to high capture resulted from soften neutron spectra. The evolutions of k_{eff} as depletion time are compared in Fig. 4. As shown in Fig. 4, the k_{eff} value monotonically decreases as time for the Case 3 core having 7 ZrH_2 rods for each assembly, which means the long cycle operation is impossible for this core due to the low breeding resulted from too soft neutron spectrum. On the other hand, the Cases 1 and 2 cores have cycle lengths of 41 and 31 EFPYs, respectively. Cycle length is ended when k_{eff} reached the initial k_{eff}(1.005). Table II summarizes the main performance parameters of the Cases 1 and 2 cores. The Case 2 core having 3 ZrH_2 rods per assembly has small burnup reactivity swing of 670 pcm in spite of its smaller cycle length than the Case 1 core. The Case 2 core has a significantly reduced average burnup of 97.8 MWD/kg in comparison with 129.4 MWD/kg for Case 1. In particular, the Cases 1 and 2 cores have considerably high burnup of 95.8 and 55.5 MWD/kg, respectively in the thorium blankets. Also, it is noted that the Case 2 core has higher radial power peaking factor than the Case 1 core due to the fact that high powers occur near the moderator rods (as shown in Fig. 6). The Case 2 core has much more negative Doppler coefficient at both BOC and EOC than the Case 1 core due to its softer neutron spectrum. Fig. 5 compares the neutron spectra of the considered three cases. As shown in Fig. 5, the use of ZrH_2 rods soften the core neutron spectra.

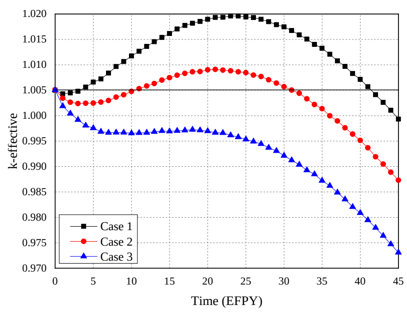

Fig.4. Comparison of the k_{eff} evolutions for three different moderator cases

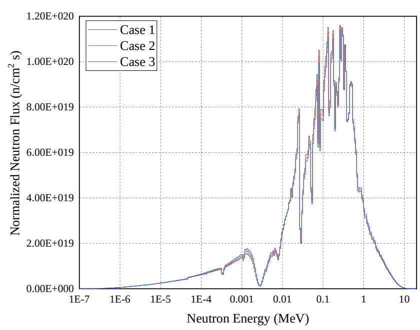

Fig.5. the neutron spectra of the three cases

In Table II, the sodium void worth is given at BOC and 31 years. We choose 31 year rather than EOC of Case 1 for Case 1 core to coincide the burnup. The Case 2 has significantly lower sodium void worth by 534pcm and by 182 pcm, respectively at BOC and 31 years than the Case 1 core for sodium voiding both in active core and upper gas plenum regions. For additional sodium voiding in the sodium reflectors, the Case 1 core has lower sodium void worth by 549 pcm and 244 pcm at BOC and 31 years, respectively. Fig. 6 compares the pin-wise power distributions at BOC, MOC, and EOC. Fig. 6 shows that the Case 2 core has higher pin powers in the middle core having moderator rods than Case 1 core and that there are power peakings near the moderator rods. However, it is considered that these levels of power peakings are not problematic.

The decomposition of sodium void worth which was performed using the reaction rates provided from Serpent at BOC. We used the normalization of the reaction rates and leakage rate to the fission production rate.[4] The results of the decomposition are given in Table III which shows that the Case 2 core has slightly less contribution by leakage to the sodium void worth than the Case 1 core due to its softer spectrum. The main difference in the contribution to sodium void worth is from the capture. That is to say, the Case 2 has smaller contribution by 294 pcm from capture, which means that the capture rate decreases by a smaller amount from sodium voiding due to softer spectrum in the Case 2 core than in the Case 1 core.

Table IV summarizes the nuclide-wise contributions to the sodium void worth at BOC. The major positive contribution comes from the capture reduction of ^{238}U which is most abundant resonance nuclide, under sodium voiding and the TRU nuclides such as ^{237}Np, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{241}Am give considerable positive contributions of 130~1013 pcm. In particular, it is noted that ^{239}Pu and ^{241}Pu give positive contributions both in fission and capture and their positive contributions are considerable. The smaller sodium void worth for Case 2 is resulted from the reduction of ^{238}U capture contribution by spectrum softening than Case 1.

4. Conclusions

In this work, a small modular SFR core was designed to have ultra-long-life and the effect of moderator rods on this core is analyzed using the Serpent code. From the results of the analysis, it was shown that the effect of ZrH_2 moderator rods is quite large on the ‘ capability and so use of the 7 ZrH_2 rods per assembly even in the middle core led to the failure to achieve ultra-long-cycle. On the other hand, the 3 ZrH_2 rods per assembly in the middle core was feasible even if the cycle length was significantly reduced. Also, the use of 3 ZrH_2 rods per assembly leads to considerable reduction of sodium void worth. From a decomposition analysis based on neutron balance, it was shown that the reduction of sodium void is mainly resulted from the reduction of the contribution from capture due to the softer neutron spectrum. However, the thermal stability of ZrH-based moderator is not assured up to the temperature of the sodium voiding. For further application in this core, using YH-based moderator could solve the problem of thermal stability.[5]

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Table II. Summary of performance and safety parameters

Parameters	***Case1***	***Case2***
Cycle length (EFPY)	41	31
Burnup reactivity swing (pcm)	1531	670
TRU contents in middle and outer core (%)	13.64	13.83
Burnup (MWd/kg, EOC)		
Total core	129.4	97.8
Inner core (blanket/driver)	95.8/186.7	55.5/140.4
Middle core	159.4	121.8
Outer core	109.0	83.9
Maximum LPD (W/cm, BOC/EOC)	147.7/149.0	166.1/165.4
Radial power peaking factor (BOC/EOC)	1.316/1.328	1.480/1.474
Doppler coefficient (pcm/K, BOC/EOC)	-0.3466/-0.1780	-0.5249/-0.3664
Voided case (pcm, BOC/EOC)		

With upper plenum	1639.8/2121.0	1106.1/1939.5
And with sodium reflector	1305.1/1904.3	756.4/1660.0

Table III. Decomposition analysis of sodium void worth at BOC

	Case 1	***Case 2***
Sodium void worth decomposition (pcm)		
1Leakage	0.0604[1)]/-5107.5[2)]	0.0601/-4995.8
Capture	0.5963/6338.9	0.5965/5673.6
Fission	0.415/40.9	0.3417/45.7
(n,2n)	0.0033/32.8	0.0033/32.9
Total	0.9949/1305.1	0.9949/756.4

1) Normalized reaction rate to fission production rate at nominal state
2) Contributions to sodium void worth (pcm)

Table IV. Nuclide-wise contributions to sodium void worth at BOC

Nuclide	***Case 1***				***Case 2***			
	Fission	**Capture**	**(n,2n)**	**Total**	**Fission**	**Capture**	**(n,2n)**	**Total**
^{232}Th	-5.9	42.7	1.7	38.5	-5.8	23.7	1.6	19.5
^{235}U	32.0	17.8	0.1	49.9	30.6	15.9	0.0	46.6
^{238}U	-272.9	3473.8	25.9	3226.7	-311.2	2952.2	27.9	2668.8
^{237}Np	-53.9	256.9	0.1	203.0	-52.6	250.3	0.1	197.8
^{238}Pu	-20.0	47.3	0.1	27.4	-20.6	47.8	0.0	27.3
^{239}Pu	347.2	637.8	0.4	985.4	379.3	633.4	0.5	1013.2
^{240}Pu	-161.4	292.2	0.1	130.9	-149.3	295.4	0.2	146.3
^{241}Pu	259.6	90.8	0.4	350.9	257.5	88.4	0.4	346.4
^{242}Pu	-44.5	85.6	0.2	41.4	-44.0	84.2	0.1	40.3
^{241}Am	-28.2	242.1	0.0	213.9	-26.9	234.7	0.0	207.8
^{242}Am	0.6	0.1	0.0	0.7	0.4	0.2	0.0	0.6
^{243}Am	-6.5	82.9	0.0	76.4	-7.1	85.0	0.0	78.0
^{243}Cm	0.1	0.0	0.0	0.1	0.2	0.1	0.0	0.4
^{244}Cm	-6.4	11.6	0.0	5.2	-6.4	11.6	0.0	5.2
^{245}Cm	1.1	0.5	0.0	1.7	1.4	0.5	0.0	2.0
Total	40.9	5282.4	29.0	5352.3	45.7	4723.5	31.0	4800.2

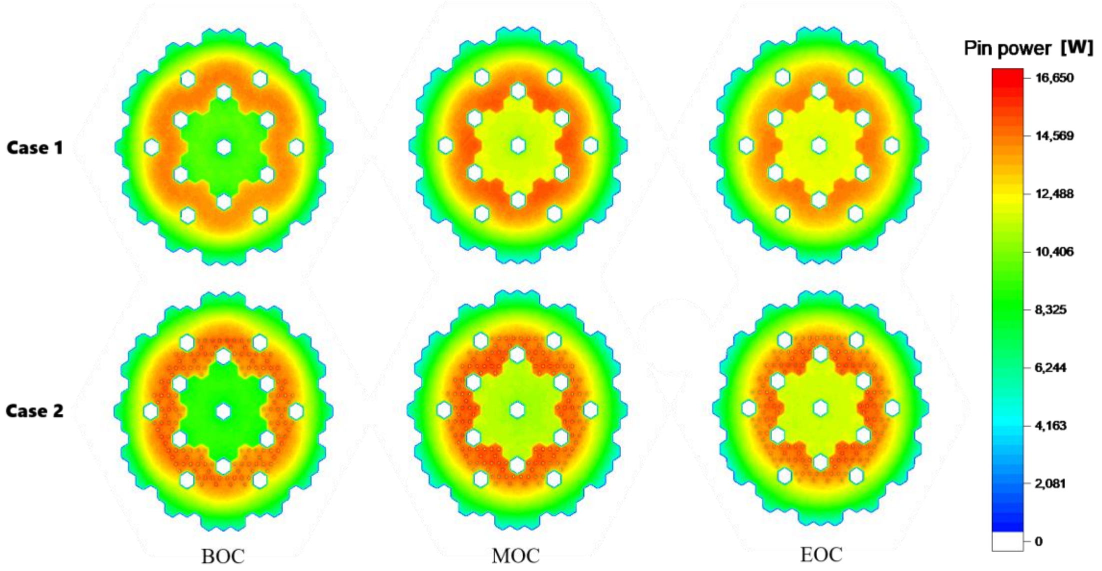

Fig.6. Configurations of power distribution