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Optimization Study of Ultra-long Cycle Fast Reactor Core

Taewoo Tak[a], Deokjung Lee[a*]
[a]*Ulsan National Institute of Science and Technology, UNIST-gil 50, Ulsan, 689-798, Korea*
[*]*Corresponding author: deokjung@unist.ac.kr*

1. Introduction

Ultra-long Cycle Fast Reactor (UCFR) had been developed with the purpose of 60-year operation. It utilizes a breed-and-burn strategy by using low enrichment uranium (LEU) as an igniter and natural uranium (NU) as a blanket material. Since it was reported only neutronics point of view, it is expected to optimize UCFR to perform a feedback from thermal-hydraulic (TH) analysis and mitigate the power peaking issue. Also, further feasibility study on using spent fuel (SF) for the blanket material was expected, which contributes to nonproliferation issues as well as uranium utilization [1].

In this paper, the optimization of UCFR-1000 has been performed. First, the overall core shape and size have been changed to satisfy the safety limit from the TH feedback of the previous UCFR-1000 [2]. Second, PWR SF has been tried as the blanket material as well as NU for the fuel composition optimization, which could be one way of solving the SF storage issue in Korea. Core performance and depletion tendency have been analyzed. Last, a small-size UCFR with a power rating of 100 MWe (UCFR-100) has also been developed regarding the neutron flux and fast neutron fluence. It was analyzed with both NU and PWR SF as it has been performed for UCFR-1000. The calculation and evaluation have been performed by McCARD Monte Carlo code [3].

2. Design Optimization of UCFR-1000

The core design parameters have been changed from those of the old model [1] in the process of optimizing the core size and core shape for satisfying the safety limit from the TH feedback. Thermal-hydraulic safety analysis has been performed using Multi-dimensional Analysis for Reactor Safety - Liquid Metal Reactor (MATRA-LMR) code developed by Korea Atomic Energy Research Institute (KAERI) [4] and Steady-state LMR core Thermal-Hydraulic analysis code based on ENergy model (SLTHEN) [5]. For UCFR, MATRA-LMR performs a single assembly analysis and seeks a peak temperature of center fuel pin and SLTHEN performs whole assembly analysis with the radial peaking factors. The performance result of the two codes is compared the sodium cooled fast reactor thermal safety margin and temperature limit of Design Basis Event (DBE) that are the maximum fuel rod temperature of 955 ℃, the maximum cladding temperature of 650 ℃, the coolant exit temperature of 560 ℃, and the limit of coolant temperature of 1055 ℃. The TH simulation of the old model has 1478 ℃, 876 ℃, 867 ℃, and 876 ℃ for each so the purpose of geometric optimization is to decrease the temperatures by 1/3 so that they are lower than each limit. PWR SF loaded UCFR-1000 has also been analyzed concurrently.

2.1. Core Design Optimization

As a geometric optimization way of decreasing the temperatures in TH simulation feedback by 1/3, the axial peak factor has decreased by 1/3 through shortening the axial pin length, which leads decrease of the coolant flow and maximum neutron flux. The axial pin length could be shortened as the core was enlarged in radial and the axial movement of the active core got slower, which makes it possible to operate the cycle length with shortened fuel pin.

Table I shows the design parameters of the optimized UCFR-1000 loading NU model and SF model. The design objective is to analyze a core which operates 60 years once through without refueling with the power rating of 1000 MWe and maintains criticality with non-enriched fuel such as natural uranium or PWR spent fuel. The simulation result of the geometrically optimized UCFR-1000, it has the maximum fuel rod temperature of 767 ℃, the maximum cladding temperature of 566 ℃, and the maximum coolant exit temperature of 541 ℃ so the optimization succeeded in satisfying the TH safety limit [6].

Table I: Core design parameter of UCFR-1000

Parameters	Old model	UCFR-1000 (NU)	UCFR-1000 (SF)
Thermal power (MWth / MWe)	2600 / 1000	2600 / 1000	2600 / 1000
Cycle length (effective full power years)	60 (Once through)	60 (Once through)	60 (Once through)
Initial heavy metal loading (t)	201	238	236
Core volume (kL)	32.1	42.4	42.4
Equivalent core diameter (m)	4.8	6.4	6.4
Active core height (cm)	360	240	240
Average specific power density (MW/t)	12.9	10.9	11.0

Average volumetric power density (W/cc)	81	61.3	61.3
Average linear power (W/cm)	210.0	158.7	158.7
Axial blanket form	NU-10Zr	NU-10Zr	SF-7Zr
Uranium enrichment of bottom-driver (%)	12.3	11.9	11.9

Figure 1 shows the core layout of UCFR-1000 which has a total of 1093 assemblies in 20 radial rings. 324 reflector assemblies surround the fuel region in 3 layers and the thickness of the reflector region is about 50 cm.

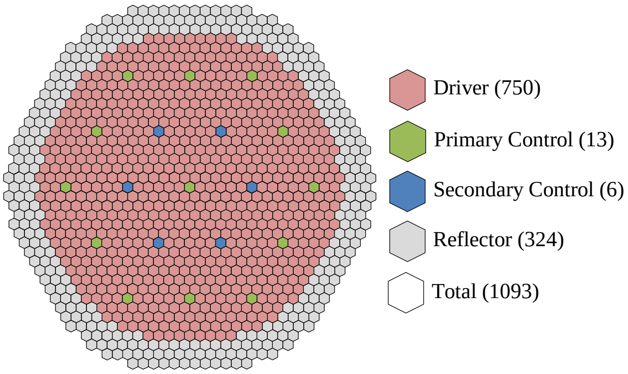

Fig. 1. Core layout of UCFR-1000.

2.2. Performance Evaluation

The Figure 2 shows the multiplication factor behavior over the 60-year depletion. All the calculations for this paper were performed with the following conditions: (a) all control rods out state, (b) the depletion calculation step size is 1 year, (c) the number of neutron histories is determined so that the average standard deviation of all the steps is no more than 10 pcm, (d) the average fuel temperature was assumed to be 900K and 800K for the other materials. It is noted that the core can maintain criticality for 60 years with full power rating with both kinds of fuel. Because the spent fuel has less amount of ^{238}U than natural uranium, the fuel of SF-10Zr form is not enough to operate the 60-year. For the criticality over 60 year, the zirconium fraction has decreased to SF-7Zr.

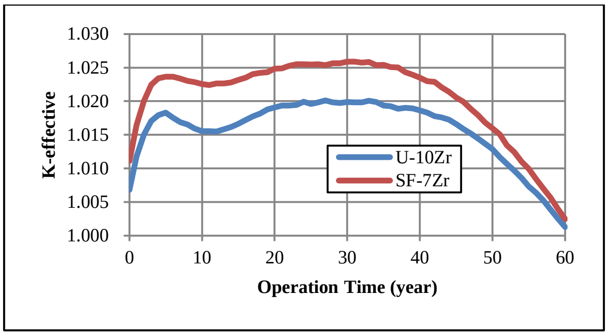

Fig. 2. K-effective vs. time of UCFR-1000.

The Figure 3 presents normalized core axial power distributions at the beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) of 60-year operation of both UCFR-1000 models. In this figure, the active core is moving from the bottom to the top of the core as the core burning continues. The axial movement speed of the active core is measured to be 3 cm/year which is decreased value from that of old model of 5 cm/year. The axial peak factors of NU model are 3.69 at BOC and 2.18 at MOC and these are decreased values from those of old model of that 6.59 at BOC and 3.33 at MOC. It has been confirmed that the geometric optimization is successful to decrease the axial movement speed of the active core and the axial peak factors by more than 1/3, which leads the decrease of fuel pin length by 1/3.

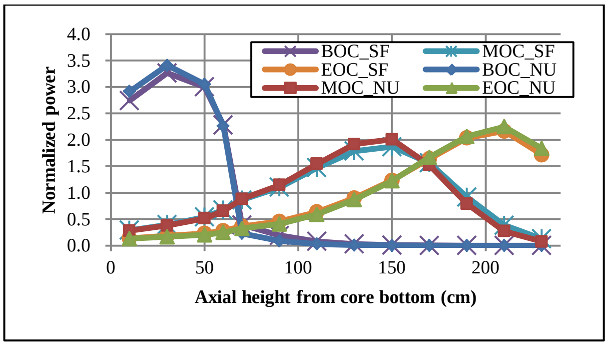

Fig. 3. Normalized axial power distribution of UCFR-1000.

In Table II, maximum neutron flux, fast fluence, coolant velocity and discharge burnup have decreased from those of the old model as the core volume increased. It is noted that the maximum neutron flux and average coolant velocity have decreased by 1/3 from those of the old model

Table II: Core parameters of UCFR-1000

Parameters	UCFR-1000 (NU)	UCFR-1000 (SF)
Maximum neutron flux (#×10^{15}/cm^{2}sec)	4.7	4.5
Fast neutron fluence (#×10^{24}/cm^2)	2.18	2.14
Average coolant velocity (m/s)	2.8	2.8
Average discharge burnup (GWD/t / %)	239.4 / 25.2	241.6 / 25.4

The core breeding ratio was measured and it is summarized in Table III. The breeding ratio is the amount of produced fissile to that of consumed fissile and the calculation was done with the assumption that all the ^{238}U is converted to ^{239}Pu. At BOC, the active core is in the enriched uranium region and the amount of consumed ^{235}U is relatively large while the amount of consumed ^{238}U is not enough to make the breeding ratio lower than 1. In contrast, once the core moves upward to the blanket region where the amount of ^{235}U is less than 1 % and the amount of ^{238}U is dominant, ^{239}Pu becomes a main fuel of the active core from the capture reaction of ^{238}U so that the breeding ratio is greater than 1. The SF loaded model has ^{239}Pu in the initial core but the amount of it has little effect on the breeding ratio values, so the breeding ratio change progress is no different from the NU loaded model

Table III: Fissile number and breeding ratio of UCFR-1000 (NU/SF)

	Initial	BOC	MOC	EOC
^{235}U	2.43E+04 2.52E+04	2.19E+04 2.28E+04	4.18E+03 4.82E+03	1.72E+03 1.97E+03
^{238}U	5.78E+05 5.63E+05	5.75E+05 5.61E+05	4.87E+05 4.75E+05	4.01E+05 3.92E+05
^{239}Pu	- 2.78E+03	2.28E+03 5.01E+03	2.83E+04 2.99E+04	3.85E+04 3.87E+04
Total	6.02E+05 5.91E+05	5.99E+05 5.88E+05	5.20E+05 5.10E+05	4.41E+05 4.33E+05
Breeding ratio	- -	0.96 0.96	1.14 1.12	1.05 1.03

3. Design Optimization of UCFR-100

In this section, for more feasible and deployable model development, UCFR-100 has been improved with the purpose that the maximum neutron flux decreases to 14th order per square centimeter per second and the fast neutron fluence decreases by half from 2.57×10^{24}/cm^{2} of previous UCFR-1000. In this section, The core design optimizations of UCFR-100, according to the analysis of improved UCFR-1000, are presented with both NU and PWR SF.

3.1. Core Design Optimization

Table IV shows the core design parameters of UCFR-100. The power rating has been lowered to one tenth from that of UCFR-1000 for the decrease of the maximum neutron flux and fast neutron fluence. The equivalent core diameter and the active core height decreased, which causes both the initial heavy metal loading and core volume decrease to one fifth. This leads to decreases in power density, linear power, maximum neutron flux, fast neutron fluence, average coolant velocity, and average discharge burnup. This improved UCFR-100 has a maximum fuel rod temperature of 529 ℃, maximum cladding temperature of 499 ℃, and maximum coolant exit temperature of 486 ℃, which also satisfies the thermal safety limit like improved UCFR-1000.

Table IV: Core design parameter of UCFR-100

Parameters	UCFR-100 (NU)	UCFR-100 (SF)
Thermal power (MWth / MWe)	260 / 100	260 / 100
Cycle Length (effective full power years)	60 (Once through)	54 (Once through)
Initial heavy metal loading (t)	53	50
Core Volume (kL)	8.2	8.2
Equivalent Core Diameter (m)	3.8	3.8
Active core height (cm)	100	100
Average Specific power density (MW/t)	4.9	5.2
Average Volumetric power density (W/cc)	31.7	31.7
Average Linear Power (W/cm)	82.1	82.1
Axial blanket form	NU-5Zr	SF-5Zr
Uranium enrichment of bottom-driver (%)	12.1	12.1

The number of total assemblies and that of fuel assemblies are halved from UCFR-1000. The number of control assemblies also decreased from 19 to 13. Figure 4 shows the core layout of UCFR-100. 234 reflector assemblies in 3 layers surround the fuel region, and the thickness of the radial reflector region is 50 cm.

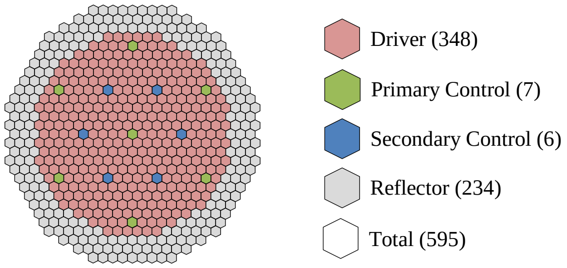

Fig. 4. Core layout of UCFR-100.

3.2. Performance Evaluation

The depletion calculations for the two UCFR-100 models were performed under identical condition as for UCFR-1000. Figure 5 shows the multiplication factor behavior over the operation time and the SF model ends operation at 54 years. SF has ^{235}U of 0.91 %, which is greater than that of NU, and this makes the graph of SF have higher values in the first 40 years. SF, however, has ^{238}U of 93.3 % which is less than that of NU and this makes the graph of SF have lower values in the latter 20 years. In this evaluation for UCFR-100 with SF, the EOC of the SF model is 54-year.

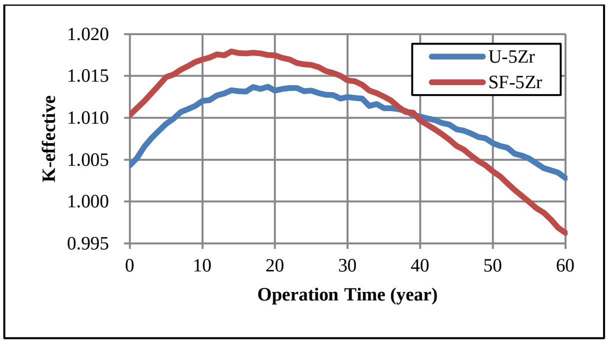

Fig. 5. K-effective vs. time of UCFR-100.

Figure 6 shows normalized core axial power distributions of UCFR-100 at BOC, MOC, and EOC. As UCFR-1000, the active core moves from the bottom to the top of the core as the core burning continues. The axial peaking has decreased by half from that of UCFR-1000, and the speed of the core movement is 0.67 cm/year, which is about 5 times slower than that of UCFR-1000 because of the lower power density.

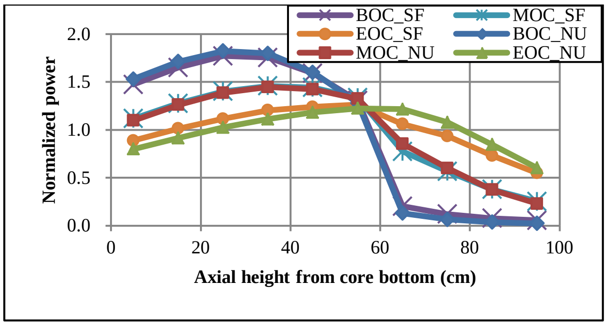

Fig. 6. Normalized axial power distribution of UCFR-100.

In Table V, the maximum neutron flux and average coolant velocity decreased to one fifth that of UCFR-1000, but fast neutron fluence decreased only by half because the maximum neutron flux is not an average value. The fast neutron fluence is still larger than the HT-9 irradiation experience. Further study of power flattening and the shortening of the refueling period could help reduce the fast fluence.

Table V: Core parameters of UCFR-100

Parameters	UCFR-100 (NU)	UCFR-100 (SF)
Maximum Neutron Flux ($\#\times10^{15}/cm^2sec$)	0.94	0.92
Fast Neutron Fluence ($\#\times10^{24}/cm^2$)	1.14	1.11
Average Coolant Velocity (m/s)	0.5	0.5
Average Discharge Burnup (GWD/t / %)	107.9 / 11.4	114.7 / 12.1

4. Conclusion

The optimization of UCFR core design has been performed for the aspects of core geometry, fuel composition, and power rating. The UCFR-1000 has been optimized geometrically in the aspects of core size and core shape with the consideration of TH feedback. Enlarging the core in radial direction is followed by slowed axial movement of the active core, shortening the fuel pin length, decreased axial peak factor, and decreased maximum neutron flux by 1/3. A 60-year full power operation without refueling has been achieved with both NU and PWR SF as blanket materials. The core parameters such as multiplication factors and power peaking factors show that the breed-and-burn concept has been successfully implemented for the UCFR, which causes that the total number of uranium atoms decreases and that of plutonium atoms increases in the core. A small-size UCFR, UCFR-100, has been also optimized in the aspects of maximum neutron flux and high energy neutron. The UCFR-100 core successfully lowered the maximum neutron flux to 14th order and fast neutron fluence by half from that of previous UCFR-1000. It was confirmed for both UCFR-1000 and UCFR-100 that the ultra-long cycle operation satisfying safety limits is feasible with respect to nuclear isotopics and criticality, and the SF loading as a blanket material is also feasible without performance degradation.

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