Neutronic Analysis of an Ultra Long Cycle VSMLFR (Very Small Modular Lead-cooled Fast Reactor)

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1. Introduction

Recently, there have lots of interest in designing small modular reactor (SMR)s even including micro reactors because they have many desirable features such as low investment risk, factory-fabrication including on-site installation, and improved passive safety. In particular, they can provide electricity and heat efficiently in remote area and is transportable by truck, shipping vessel, airplane or railcar. Also, there have been several different types of SMRs or Micro-reactors using different coolants (e.g., water-coolant, liquid-metal coolant, and gas-coolant). Actually, the water cooled SMRs has advantages that they use many proven technologies accumulated in the commercial PWRs and use of LEU (Low Enriched Uranium) fuels while their primary coolant system should be pressurized and low plant efficiency. The gas cooled SMRs can provide very high temperature heat which can be used to produce hydrogen or electricity with high efficiency while they require lots of graphite moderator for thermal spectrum but it can become lots of activated wastes. Also, the typical gas cooled SMRs uses very low power density.

On the other hand, the liquid metal cooled SMRs such as sodium or lead cooled ones can be designed to have an ultra-long operation cycle due to the breeding and to provide high temperature heat while they usually require high fissile content fuel due to large neutron leakage. In particular, the lead-cooled reactors have several desirable features such as advantages coming from fast spectrum, no chemical reaction issue of coolant with air and water, no requirement of solid reflector, much less concern of coolant voiding than sodium, and no pressurization need of the coolant system. Also, the autonomous load following can be achieved due to high negative feedback coefficients and very small Xenon poisoning effect.

The objective of this work is to design a very small modular lead-cooled fast reactor (VSMLFR) of 60MWt (~20 MWe) enabling to make breeding performance with low enrichment uranium (LEU) for operation over 20~30 years without refueling. In particular, this reactor concept does not use the blanket fuels to achieve breeding, which leads to the high proliferation resistance.

2. Computational methods and core design

2.1 Computational methods

The depletion analysis of the core was done using the Serpent2 Monte Carlo reactor physics burnup calculation code which was developed by VTT [1]. Serpent2 code has been widely used for modeling small research reactors and other closely-coupled systems. The ENDF/B-VII.r0 point-wise cross section library was used for all the depletion calculations and core physics parameters. Full-core 3-D analysis was performed with preserving fuel pin level heterogeneities and Chebyshev Rational Approximation Method (CRAM) option is used for burnup depletion modeling. We used 100 inactive and 600 active cycles with 50000 histories each both for depletion calculation giving ~10 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 considered a VSMLFR of which thermal power is 60MWt. The fuel is U-10Zr metallic fuel. To enable to use LEU fuel, 5490 fuel rods of a single uniform uranium enrichment are arranged with a triangular lattice structure without the assembly duct and these fuel rods are hold through grid spacers, which eliminates potential flow blockage by allowing crossflow paths. A 75% smear density was used to consider swelling of the metallic fuels. The active fuel length is 150 cm and the fuel outer diameter is 1.44 cm to achieve breeding ratio slightly larger than 1.0 and a 150cm fission gas plenum above fuel is considered to reduce fission gas pressure. The reactivity controls are achieved using 18 outer peripheral control assemblies and one central control assembly. At this stage, the detailed design of the control assemblies is not performed, which will be refined in the near future. As the preliminary concept, the combination of B_4C and W is considered by 40 vol% B_4C +60 vol% W for control material. Tungsten is considered to make the density of the control elements exceed the density of the heavy lead coolant, to help scramming by gravity. The boron is enriched to 92% ^{10}B [2]. We are considering a grouping of the peripheral control assemblies in which the first group controls the reactivity for compensating the small burnup reactivity and power change while the second group shutdown the core. This core does not use solid reflector but the liquid lead surrounding active core plays a role as reflector. Fig. 1. and Fig. 2 show the radial and axial core layouts, respectively. Table I summarizes the main design parameters.

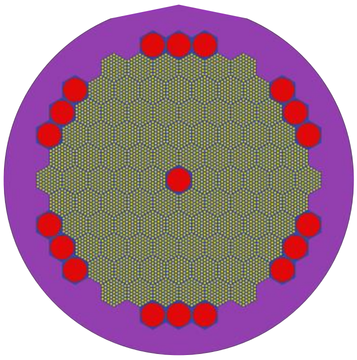

Fig. 1. Radial Core Layout

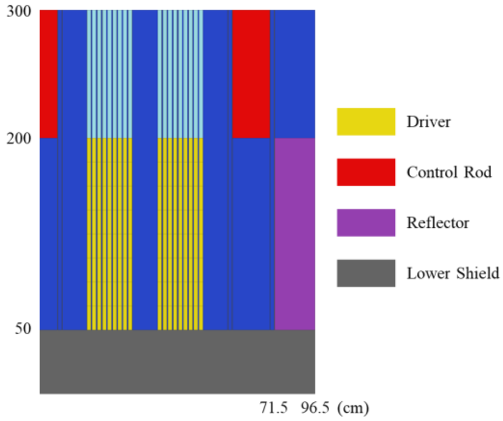

Fig. 2. Axial Core Layout

Table I. Main design parameters of the core

Design parameter	***Value***
Power (MWt)	60
Active core height (cm)	150
Average LPD (W/cm)	72.86
Fuel type	U-10Zr
Number of rods per core	5490
Smear density of fuel (%)	75
Fuel pin outer diameter (cm)	1.44
Cladding thickness (mm)	0.55
Control assembly duct wall thickness (mm)	3.5
Number of Control Rods	19
Control Rod type	40 vol% B_4C + 60 vol% W

The core dimensions (diameter and height), initial uranium enrichment, and the liquid reflector thickness are determined through a parametric study such that the core reactivity initially increases to have ultra-long cycle length of several tens of years within the small burnup reactivity swing less than 1$ for a set of fuel rod dimension and P/D ratio. A large P/D ratio is helpful to achieve enhanced safety against flow blockage and to improve heat removal through higher natural circulation by reducing pressure drop [3]. However, it harms breeding performance by increasing neutron leakage. So, it is important to find minimum P/D ratio that can makes breeding. The results of the parametric study are described in the next section.

With the fuel outer diameter of 1.44cm, we considered 1.20 (Case 1), 1.25 (Case 2), and 1.30 (Case 3) for P/D ratio to show the sensitivity of P/D ratio on the evolution of k_{eff} and coolant velocity. From the thermal-hydraulic safety criteria adopted in the design of internationally representative lead-cooled fast reactors, the coolant velocity must be limited to about 2 m/s to protect corrosion of the structural material. The coolant velocity can be calculated using the following energy conservation:

$$q = \rho V_c A_{flow} C_p \Delta T \quad (1)$$

In this equation, q, ρ, V_c, A_{flow}, ΔT and C_p represent reactor thermal power, coolant density, average coolant velocity, coolant flow area, coolant temperature rise through core, and specific heat, respectively. The inlet and outlet temperatures are, respectively, 400 and 480℃. Lead has high melting point (327.5℃), therefore the inlet coolant temperature must be high enough to ensure that the solidification of lead does not happen in the reactor [4]. Also, it needs to increase inlet temperature for high thermal efficiency but high coolant temperature incurs higher cladding temperature, which give rise to material issues. Also, molten lead interacts with structural materials, mainly with the mechanisms of corrosion at high temperature and erosion. With this consideration, 400°C and 480°C are selected as the inlet and outlet coolant temperatures, respectively. The lead density and the specific heat are calculated by each function of temperature as [5]

$$\rho[\text{kg}\cdot\text{m}^{-3}] = 11367 - 1.1944 \times T \quad (600K < T < 1500K) \quad (2)$$

$$C_p[\text{J}\cdot\text{kg}^{-1}\cdot\text{K}^{-1}] = 175.1 - 2.961 \times 10^{-2}T + 1.985 \times 10^{-5}T^2 - 2.099 \times 10^{-9}T^3 - 1.524 \times 10^{6}T^2 \quad (3)$$

Table II shows that differences of the coolant velocity for three P/D ratio cases. A high P/D ratio means the core has large coolant flow area, giving slow coolant flow. As shown in Table II, the Case 1 with the smallest P/D ratio has 0.77 m/s, which is far below the limiting value of 2 m/sec.

3. Results

In this section, the results of the parametric study are described on the P/D ratio and reflector thickness. Additionally, the reactivity worth of the control assemblies is analyzed. The initial uranium enrichment of the core was determined to give the initial k_{eff} of 1.004 and the cycle length is considered as the time interval over which k_{eff} is maintained over 1.0025 with a margin of reactivity of 250 pcm.

3.1 P/D ratio

The ^{235}U contents are estimated to be 11.70, 12.09, and 12.41wt% for the Cases 1, 2 and 3, respectively, due to their different fuel volume fractions. The evolutions of k_{eff} as depletion time for the different P/D ratio cases are compared in Fig. 3. As shown in Fig. 3, the k_{eff} value evolution for the Case 3 having P/D=1.30 gives only very small change over ~17 years. On the other hand, the Cases 1 and 2 having P/D=1.20 and 1.25, respectively, show higher breeding performances leading to higher cycle lengths of 43 and 29 EFPYs, with burnup reactivity swings of 864 pcm and 330 pcm, respectively. Table III summarizes the main performance parameters of the three different P/D ratio cases. Effective delayed neutron fractions (β_{eff}) for the Cases 1 and 2 are 735 pcm and 737 pcm, respectively, which are comparable with the burnup reactivity swings. The core average burnups of Cases 1, 2, and 3 are 77.6, 52.3, 30.7 MWd/kg, respectively. In particular, it is noted that the Cases 2 and 3 have small burnup reactivity swing less than 1$.

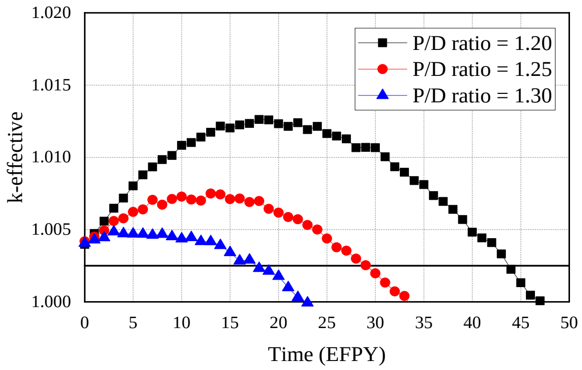

Fig. 3. Comparison of the k_{eff} evolutions for three different P/D ratio cases

3.2 Reflector thickness

For a small fast reactor core, neutron leakage through core boundary is a key factor on the initial reactivity due to large mean free path of the fast neutrons. In addition, the large mean free path of fast neutrons makes it possible to easily control the core reactivity using reflector or absorbers outside the core [6]. Also, it is known that the pure lead and lead-based reflectors have higher performance than the HT9 reflector in sodium cooled fast reactors [7]. So, at present, we considered the pure lead coolant as the reflector. From the results of Sec. 3.1, we fixed the P/D ratio to be 1.25 for three different reflector thickness cases (20, 25, and 30 cm) considered in this section. The initial enrichments of uranium for the Cases 4, 5, and 6 are estimated to be 12.26, 12.09, 11.95 wt%, respectively. The evolutions of k_{eff} as depletion time for the three different reflector thickness are compared in Fig. 4. As seen Fig. 4, the Case 6, with large reflector thickness, has the highest breeding performance with longest cycle length due to the smallest neutron leakage. The cases 5 and 6 have reactivity swing by 330 pcm and 548 pcm respectively, which are lower than their β_{eff} values of 737 pcm and 734 pcm. Table IV summarizes the main performance parameters of the three different reflector thickness cases. The Case 4, 5 and 6 achieve a lifetime of 20, 29, and 35 EFPYs, corresponding to the average fuel burnups of 36.1, 52.3, and 63.1 MWd/kg, respectively.

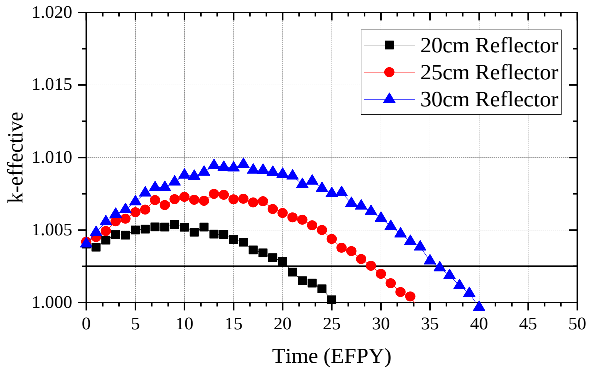

Fig.4. Comparison of the k_{eff} evolutions for three different reflector thickness cases

3.3 Control rod worth

As mentioned in Sec. 2, the core has 1 central control rod and 18 peripheral control rods. We analyzed the reactivity worth with moving all control rods together. In this calculation we chose the core design which has P/D ratio of 1.25 and reflector thickness of 25cm. The reactivity worth curve of the control rods is shown in Fig. 5. The control rod worth is calculated with 20cm-wise insertion. The reactivity worth of the full insertion of all the control assemblies at BOC and EOC are estimated to be 9111 pcm and 9628 pcm, respectively. The curve for the control reactivity worth shows the typical S-shape. Even if we did not perform the detailed shutdown margin of the control rods, it can be considered that the reactivity control assemblies have sufficiently large shutdown margin due to the very small burnup reactivity swing of the core.

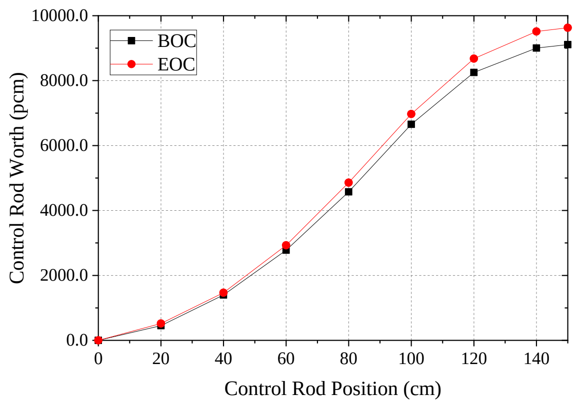

Fig.5. Reactivity worth curve of the control rods

4. Conclusions

In this work, we showed that it is possible to design a very small lead-cooled fast reactor core having ultra-long cycle (20~30 years) with low enriched uranium metal fuel through the depletion analysis with the Monte Carlo code Serpent2. In particular, the analysis was performed with the change of P/D ratio and reflector thickness to find the feasible design candidate core such that it has small burnup reactivity swing less than 1$ to remove prompt critical and to reduce the reactivity control requirement. From the analysis, it was shown that such cores can be designed having P/D=1.25~1.30, initial uranium enrichments less than 12.5wt% and high average fuel burnup of 63 MWD/kg. Also, we showed that the reactivity control assemblies can have sufficiently large reactivity worth. In the future, we will refine the core design considering more realistic configurations and optimize the cycle length including fuel burnup and core size.

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Table II. Velocity analysis for three different P/D ratio cases

Parameters	***Case 1***	***Case 2***	***Case 3***
P/D ratio	1.20	1.25	1.30
Pin pitch	1.73	1.80	1.87
Volume fraction (%) (fuel/coolant/structure)	52.1/38.9/9.0	48.1/43.7/8.3	44.4/47.9/7.7
Flow Area (cm^2)	5687.83	6932.29	8227.56
Velocity (m/s)	0.77	0.63	0.53

Table III. Summary of performance parameters for three different P/D ratio cases

Parameters	***Case 1***	***Case 2***	***Case 3***
P/D ratio	1.20	1.25	1.30
Cycle length (EFPY)	43	29	17
Burnup reactivity swing (pcm)	864	330	79
Effective delayed neutron fractions (pcm)	735	737	736
Uranium enrichment (%)	11.70	12.09	12.41
Burnup (MWd/kg)	77.6	52.3	30.7

Table IV. Summary of performance parameters for three different reflector thickness cases

Parameters	***Case 4***	***Case 5***	***Case 6***
Reflector thickness (cm)	20	25	30
Cycle length (EFPY)	20	29	35
Burnup reactivity swing (pcm)	117	330	548
Effective delayed neutron fractions (pcm)	736	737	734
Uranium enrichment (%)	12.26	12.09	11.95
Burnup (MWd/kg)	36.1	52.3	63.1