Proceedings of the Korean Nuclear Society Autumn Meeting
Yongpyong, Korea, 2003

A New Design Concept for Single Fuel Enrichment in Self-Sustaining Lead-Cooled Reactor

Y. Kim, C. H. Cho, S. J. Kim, and T. Y. Song
Korea Atomic Energy Research Institute, Korea

Abstract

New design measures have been studied to achieve single fuel enrichment (SFE) in a 900 MWth lead-cooled breakeven reactor with a burnup reactivity swing smaller than the β_{eff} value. The conversion ratio of the core is almost unity and depleted uranium is only used as the feeding material. For the SFE, a new fuel assembly design has been introduced, in which a combination of B_4C burnable absorber (BA) rods and neutron streaming tubes are utilized to control the power distribution. The BA rods are designed to have top and bottom cutback zones to reduce the peak fast fluence, which is a limiting factor in a fast reactor loaded with a metallic alloy fuel. An 18-month cycle core has been designed and its various characteristics are analyzed. Additionally, a subchannel thermal-hydraulic analysis has been performed for the peak power assembly to characterize its thermal-hydraulic properties.

I. Introduction

For the sustainable nuclear energy development, it is well perceived that fast spectrum reactors could play an important role. Recently, lead- or lead-alloy-cooled fast reactors have been actively studied worldwide due to its potential advantages over the conventional sodium coolant. Nowadays, fast reactors are not required to breed fissile materials such as Pu-239, instead, related studies are mainly focused on the breakeven self-sustaining reactor or transmutation of radioactive nuclides. This paper is concerned with core physics studies of a lead-cooled breakeven reactor.

In the conventional fast reactor design, the reactor core is divided into several zones and the fuel enrichment is adjusted in each zone to control the power distribution. The zone-wise different enrichment leads to a zone-wise different conversion ratio. Consequently, the fissile elements generally need to be separated from the discharged fuel and this makes the related fuel cycle costly and susceptible to the proliferation concern. Also, in this fuel cycle, fabrication and management of fuel are quite complicated and costly, hampering the competitiveness of fast reactors.

In the Russian BREST[1] breakeven reactor using a nitride fuel, a single fuel enrichment (SFE) concept was adopted in order to mitigate the above-mentioned problems resulting from a zone-dependent fuel enrichment. For the SFE, zone-dependent fuel rod diameters were in the BREST design: the core was partitioned into 3 zones and a different rod diameter with a different pitch-to-diameter ratio was utilized in each zone. Although the core provides a good neutronic performance, the design is subject to several potential drawbacks. First, the coolant flow patter would be very complicated in the core and detailed thermal-hydraulic experiment for the complicated geometry should be performed for realization of the core design. Secondly, in the design, three types of fuel rods are required in spite of the SFE concept. Taking into consideration the fabrication and quality control of the fuel, an identical rod type would be preferable to several types of fuel rods.

In this work, new design measures have been studied to achieve SFE with an identical rod type in a lead-cooled breakeven reactor with a burnup reactivity swing smaller than the β_{eff} value. Since the core produces self-sufficient fissile elements, depleted uranium is only fed to the core. In the closed fuel cycle of the core, the transuranics (TRU) are not separated from the spent fuel. The fission products are only removed from the spent fuel and all the remaining actinides are recycled into the core in a single stream.

All the neutronic analyses have been done with the REBUS-3[2]/DIF3D[3] code system and the ENDF-B/VI cross section data. In the DIF3D calculations, a diffusion nodal option is used with a 9-group cross sections. The multi-group cross sections are obtained with the TRANSX[4]/ TWODANT[5] codes.

II. Design Concept for Single Fuel Enrichment

In this paper, a 900 MWth lead-cooled reactor is considered. The reactor core is divided into 3 zones and comprises 198 ductless hexagonal fuel assemblies containing 204 fuel rods and 13 tie rods (TRs) for grid spacers, as shown in Fig. 1. Fuel is the conventional metallic alloy of U-TRU-10Zr with a lead bonding. All the control and shutdown systems are placed in the reflector zone since the core is designed to have a very small reactivity swing. For additional shutdown margin, the core is also equipped with a central safety assembly. In the safety and shutdown systems, an 80%-enriched B_4C is used as absorber.

Accounting for the relatively small void reactivity and the high boiling temperature of the lead coolant, the height of the active core is set to 120 cm. In general, the maximum coolant speed is known to be about 2 m/sec due to the corrosion and erosion effects of lead. Consequently, a relatively open lattice is used in the core: the pitch-to-diameter ratio of fuel rods is 1.41 in the fuel assembly. The coolant inlet temperature is set to 420 °C since the melting temperature of lead is high (327 °C), and the coolant outlet temperature is 540 °C, resulting in ~1.65 m/sec average coolant speed.

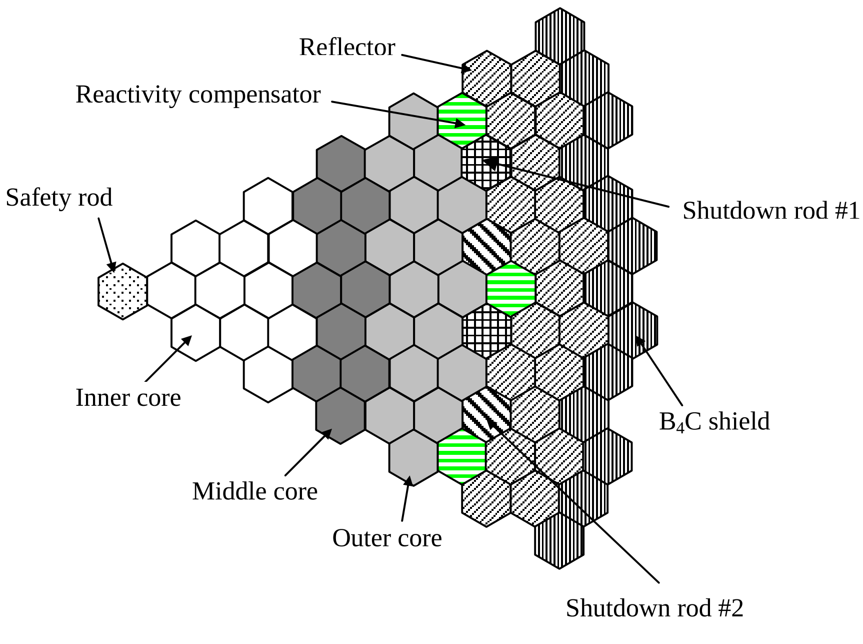

Fig. 1. Schematic onfiguration of core with a 1/6 symmetry

Two methods have been considered for the SFE design with an identical fuel rod: a B_4C burnable absorber loaded inside TRs and replacing some fuel rods by a neutron streaming tube (NST). It is easy to flatten the power distribution with the absorber rod. However, the B_4C option provides a relatively short cycle length due to a poor neutron economy and also it makes coolant void reactivity less favorable. The less favorable void reactivity is mainly because the capture cross section of B-10 is reduced when the neutron spectrum becomes harder. It is worthwhile to note that other absorbers but B-10 can hardly used in a breakeven core since the neutron economy is very poor. In the NST scheme, the number of NSTs per assembly is adjusted zone-wise to control the power distribution. This method has little impacts on the neutron economy. Generally, the NSTs enhance the neutron streaming effect and thus reduced the void reactivity a little. However, it has a drawback that the peak linear power may increase substantially if a large number of NSTs is used in an assembly, which is the usual case. Also, the peak fast fluence is generally a limiting factor when the NST scheme is applied in a metal-fueled core.

To make best use of the two concepts, they have been combined in this study: 24 NSTs and B_4C in the inner core, 18 NSTs and no B_4C in the middle core, and no NSTs and no B_4C in the outer core. Figure 2 shows the schematic configuration of the fuel assemblies in the inner and middle cores. In fast reactors, the fast fluence generally determines the lifetime of a fuel assembly. Thus, for a long-cycle, high-burnup core, the peak fast fluence needs to be significantly reduced. The peak fast fluence usually occurs in the vicinity of the core midplane. In order to reduce the fast fluence, we introduced top and bottom cutback zones in the BA rods. In the cutback zone, B_4C absorber is not loaded, thus this suppresses the midplane power. It is worthwhile to note that top/bottom cutbacks enhances B-10 depletion rate since the absorber is loaded in a high-flux region.

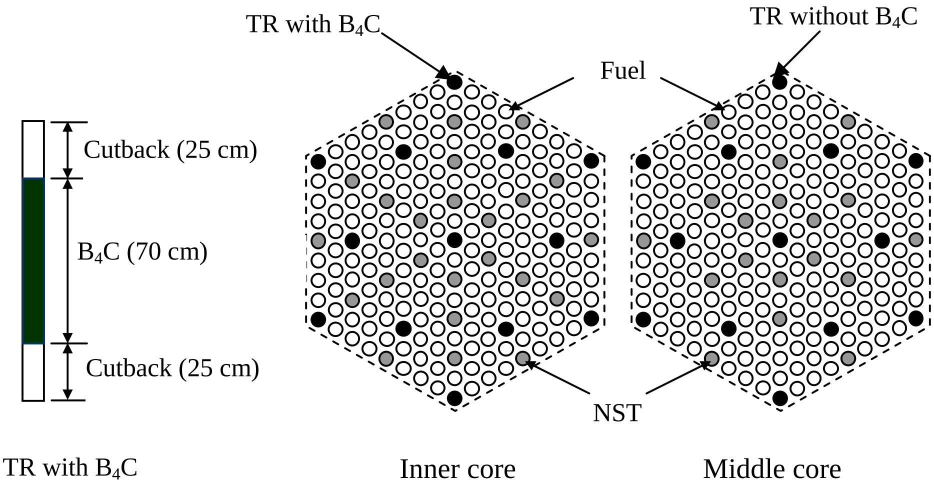

Fig. 2. Configurations of fuel assemblies in inner and middle cores

III. Core Characteristics

With the aforementioned design concepts, an 18-month-cycle core has been designed to have 480 effective full power days. In the design, a 4-batch fuel management is used in each zone. A 67%-enriched B_4C is loaded inside TRs in the inner core. In the core analysis, it is assumed that 99.9% of the actinides is recovered in the reprocessing and 5% of the rare earth elements is recycled into the core.

Table I summarizes the characteristics of the breakeven core in an equilibrium cycle. It is clear that the burnup reactivity change over a cycle is significantly smaller than β_{eff}. Consequently, it is very unlikely that a malfunction of the control system will lead to a power excursion accident in the core. Table I shows that the fuel composition is very similar to the conventional ternary fuel for fast reactors. Thus, the current metal fuel technologies could be applied to the core. The peak fast neutron fluence satisfies the generic design limit of 4.0×10^{23} n/cm^2 for the HT-9 steel. It can be reduced further if more B_4C absorber rods are used, sacrificing neutron economy and void reactivity. It should be mentioned that the peak fast neutron fluence is far beyond the limit value, if the conventional core design method or the NST-only approach is used for the breakeven core considered in this paper. In Table I, it is observed that the discharge burnup of B-10 is fairly high. This is largely attributed to the fact that B4C absorber is loaded in the inner zone with a high neutron flux.

Figure 3 shows the power distributions of the core. One can see that the radial power distribution is well controlled. Also, it is clearly observed that the BA rods with cutbacks significantly flatten the axial power distribution in the inner core, reducing the peak fast fluence.

Table I. Equilibrium cycle characteristics of the breakeven core

Parameter		Value
Fuel composition, w/o		77.3U-12.7TRU-10Zr
Reactivity swing, pcm		250
Delayed neutron fraction (β_{eff})		0.00340
Neutron generation time, μsec		0.515
Core power density, W/cc		130
3-D power peaking (BOC, EOC)		(1.42, 1.41)
Linear power (avg., peak), W/cm		(197,259)
Fuel discharge burnup (avg., peak), a/o		(8.1,11.0)
Peak fast fluence, n/cm^2		3.94x10^{23}
Initial B-10 inventory, kg		9.4 (59.5* a/o)
Uranium consumption, kg/cycle		445
Equilibrium loading, kg/cycle	Feed uranium	445
	Recycled actinide	5,038
	Total actinide	5,483
Heavy metal inventory (BOC, EOC), kg		(21,335, 20,879)

* B-10 discharge burnup

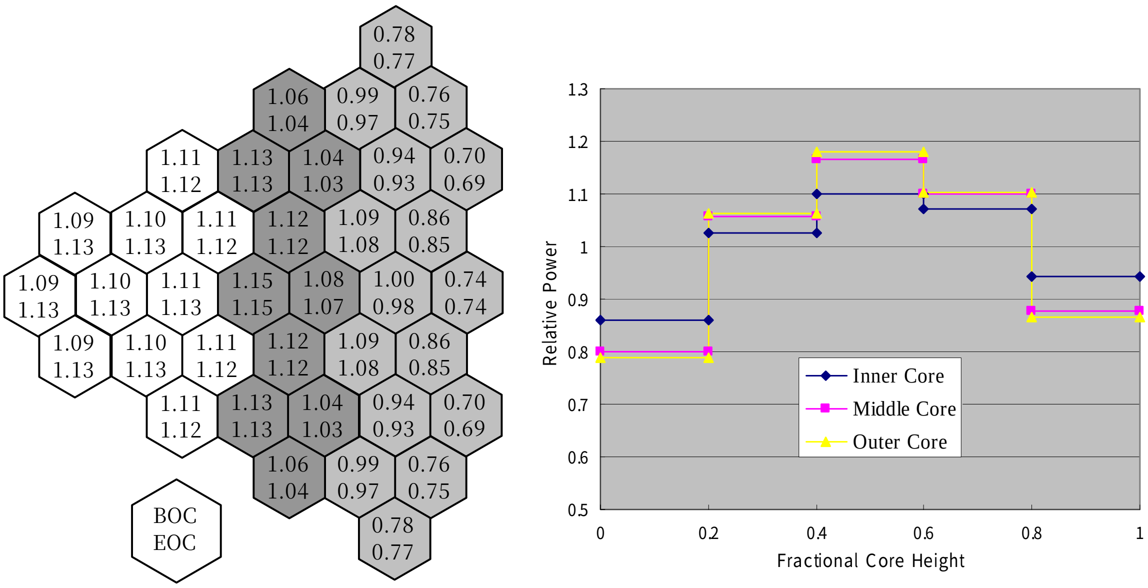

Fig. 3. Planar and axial power distributions in an equilibrium cycle

Basic requirement for a breakeven core is that the fissile inventory should be constant over an irradiation cycle. The conversion ratio of the core is about 0.995 for the core design in Table I. In Fig. 4, the evolution of fissile inventory is given for an equilibrium cycle of the breakeven core.

One can see that the fissile inventory in each core is kept almost constant over a cycle in each zone. This is because the conversion ratios in the three zones are very similar.

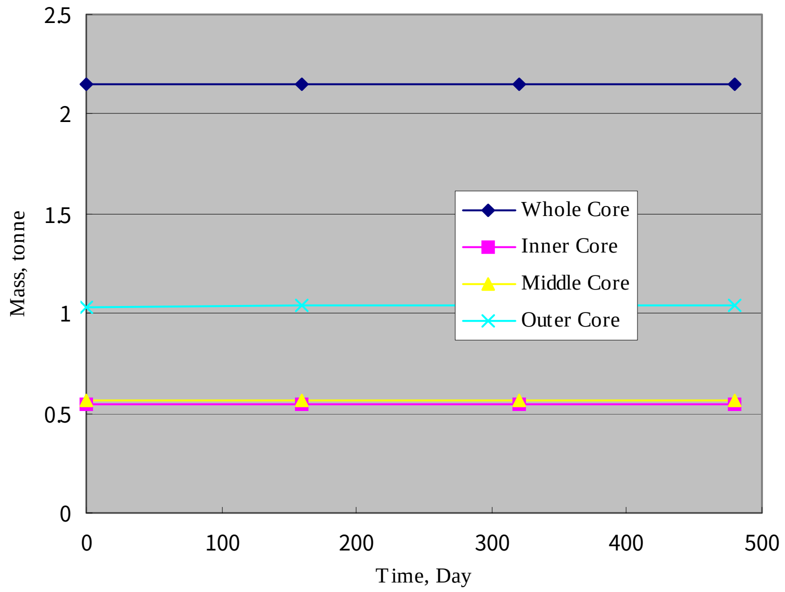

Fig. 4. Evolution of the fissile nuclides in the breakeven core

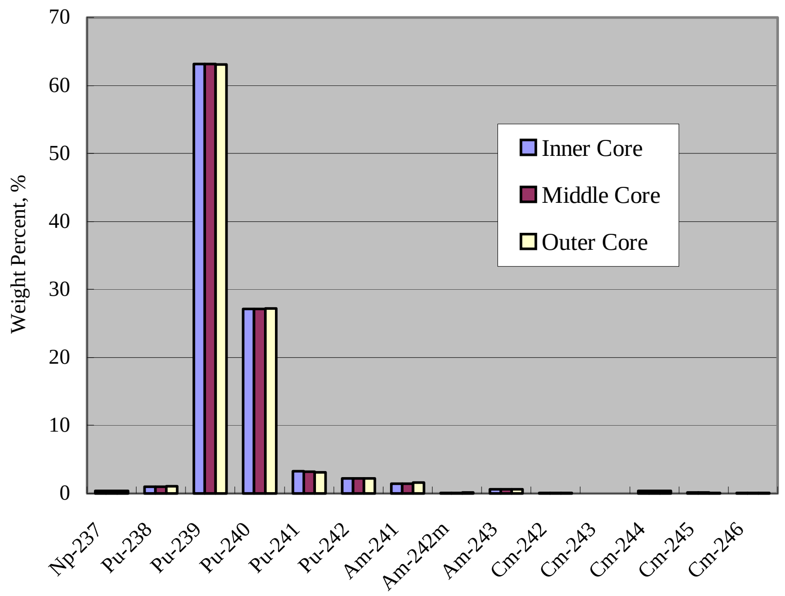

Fig. 5. TRU compositions of discharge fuels

Figure 5 shows the TRU composition of discharged fuel in each zone. It is clear that the TRU vectors in the three zones are very similar. It is worthwhile to note that in the conventional core design each zone has a significantly different TRU composition due to the differences in the zone-wise conversion ratios. These unique features of the SFE breakeven core may lead to the proliferation-resistant reprocessing technology in which only fission products are removed from the spent fuel and the depleted uranium is added for fabrication of a new fuel.

In Fig. 6, the core reactivity change is provided as a function of the coolant level. In the case of the coolant loss in the whole core, the reactivity decreases almost monotonically as the coolant level decreases. Meanwhile, when the coolant level is lowered only in the active core, the core can have a small positive reactivity (smaller than β_{eff})). However, this is very unlikely to happen in the core.

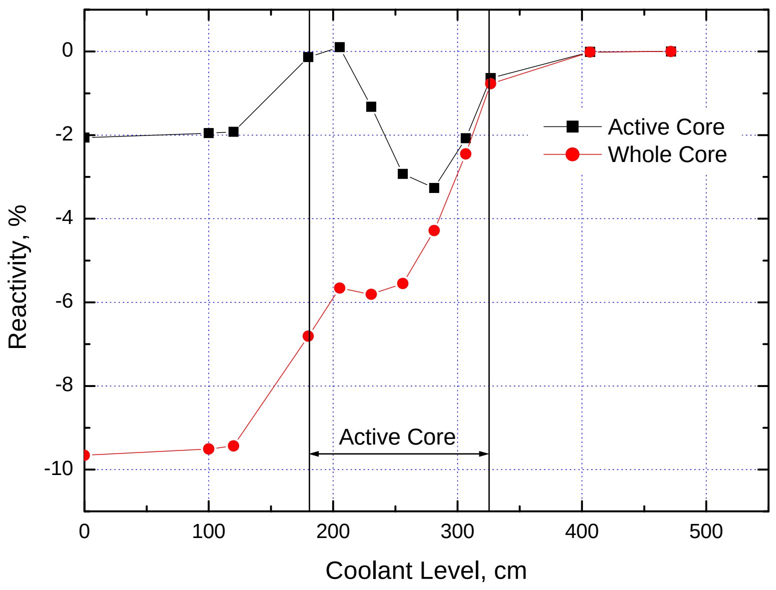

Fig. 6. Reactivity change due to coolant level change

A subchannel thermal-hydraulic analysis was performed with the MATRA[6] code for the peak power assembly in each zone and the results are shown in Table II. In the current TH analysis, it was assumed that the coolant flow rate was uniformly distributed over the all assemblies. It is observed that the peak temperature of cladding outer surface could be maintained below 600 C in the new core design. It is expected that the peak temperatures could be further reduced if the core design is optimized. In the actual design, the coolant flow could be adjusted such that the flow rate is a little larger in the inner core with more NSTs than in outer core, leading to a reduced cladding and coolant temperature in the inner core.

Table II. Subchannel analysis for peak power assembly in each zone

Parameter	Inner core	Middle Core	Outer Core
Peak power, MW	5.10	5.20	4.96
Coolant speed (avg./max), m/sec	1.64/1.78	1.63/1.77	1.61/1.73
Coolant exit temperature (avg./max.), C	550/571	552/572	548/554
Peak cladding outer surface temperature, C	593	590	573
Peak fuel temperature, C	695	707	674
Pressure loss, Mpa	0.187	0.187	0.184

IV. Conclusions

A combination of B_4C absorbers and neutron streaming tubes is an effective way to design a long-cycle, high-burnup lead-cooled core with single fuel enrichment and an identical rod type. With the proposed design approach, the fuel reprocessing/fabrication processes could be significantly simplified. A B_4C absorber rod with top/bottom cutback zones could be utilized to mitigate the neutron fluence issue in fast reactors. Introduction of the B_4C absorber in the core slightly increases the coolant void reactivity a little. However, we think that the increased void reactivity is not a big concern in a lead-cooled reactor regarding the high boiling temperature of the coolant.

For a more environment-friendly fuel cycle, the long-lived fission product produced in the breakeven core needs to be transmuted in the core. A related work is underway for the transmutation of self-generated Tc-99 and I-129.

References

1. Adamov, A. et al., “Conceptual Design of BREST-300 Lead-Cooled Fast Reactor,” RDIPE, Moscow, Russia (1994).
2. B. J. Toppel, “A User’s Guide to the REBUS-3 Fuel Cycle Analysis Capability,” ANL-83-2, Argonne National Laboratory (1983).
3. K. L. Derstine, “DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite Difference Diffusion Theory Problems,” ANL-82-64, Argonne National Laboratory (1984).
4. R.E. MacFarlane, “TRANSX2: A Code for Interfacing MATXS Cross Section Libraries to Nuclear Transport Codes,” LA-12312-MS, Los Alamos National Laboratory (1992).
5. R.E. Alcouffe et al., “User’s Guide for TWODANT: A Code Package for Two-Dimensional, Diffusion-Accelerated Neutral Particle Transport,” LA-10049-M, Los Alamos National Laboratory (1990).
6. Y. J. Yoo et al., “Development of a Subchannel Analysis Code MATRA Applicable to PWRs and ALWRs,” J. of the Korean Nuclear Society, **31**, No. 3, 314 (1999).