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Pre-Conceptual Core Design of a LBE-Cooled Fast Reactor (BLESS)

Ziguan Wang, Luyu Zhang, Eing Yee Yeoh, Linsen Li, Feng Shen

State Power Investment Corporation Research Institute, Beijing 102209, P. R. China.
**Email: wangziguan@snptc.com.cn*

Abstract - *A Generation 4th reactor development project named BLESS is proposed and designed to meet the public demands of a safer, more economical and more environmental-friendly nuclear system. In this project, the first reactor is called BLESS-D (Breeding Lead-base Economical Safe System - Demonstration). In the roadmap among several proposed BLESS reactor, BLESS-D is a pool-type reactor cooled by Lead-bismuth eutectic(LBE). The thermal power is 300 MW while the electric power is set at about 100 MW. UO_2 fuel rod is currently chosen as fuel in order to take the advantages of mature fuel-fabrication industry. BLESS-D is devoted to demonstrate the technology of China LBE-cooled fast reactor. It is expected that the design of BLESS-D can validate and demonstrate crucial technical problem solutions and be expended to an industrial scale (about 1000MWe) or be converted to modular design in order to meet different requirements. In this paper, a preliminary configuration of the 300 MW Pb-Bi cooled fast reactor BLESS has been established. Neutronics analysis was performed by a Monte-Carlo code RMC (Reactor Monte Carlo) developed by Tsinghua university, China. Different schemes parameters of core design including fuel assembly scale, fuel pitch, fuel enrichment, control-assembly arrangement and reactor size was analyzed and compared to get an optimized solution. The calculation results showed that the reactivity control performance meets the safety criteria, giving a coolant reactivity of -1.53 pcm/K, coolant voiding coefficient -31.5 pcm/%, and fuel Doppler coefficient -0.37pcm/K. The power distribution was not flattened enough with a radical power peak factor 1.364 at BOL, and 1.345 at EOL. Fuel pin level power distribution is also calculated to support thermal-hydraulics analysis.*

I. INTRODUCTION

Lead-cooled Fast Reactor (LFR), as one of the six nuclear reactor technologies selected by the Generation IV International Forum (GIF)[1], has become one of the most promising concepts and attracted more attention from the industry.

In recent years, many types of design of Lead-cooled fast reactor are proposed by research organizations, for example, SVBR-100 (Ref. 2) and BREST-300 (Ref. 3) in Russia, ALFRED and ELSY (Ref. 4) in Europe, and SSTAR (Ref. 5) in the USA,

A project of a Lead-Bismuth Eutectic (LBE) cooled Fast Reactor (Breeding Lead-based Economical and Safe System, BLESS) named BLESS has been proposed by China State Power Investment Corporation Research Institute and designed to meet the public demands of a safer, more economical and more environmental-friendly nuclear system.

In this project, the first reactor is called BLESS-D (Breeding Lead-base Economical Safe System - Demonstration). In the roadmap among several proposed BLESS reactor, BLESS-D is a pool-type reactor cooled by LBE. The thermal power is 300MW while the electric power is set at about 100 MW. UO_2 fuel rod is currently chosen as fuel in order to take the advantages of mature fuel-fabrication industry. BLESS-D is devoted to demonstrate the technology of China LBE-cooled fast reactor.

It is expected that the design of BLESS-D can validate and demonstrate crucial technical problem solutions and be expended to an industrial scale (about 1000 MWe) or be converted to modular design in order to meet different requirements. The definition of the core and its neutronic characterization are presented in this paper.

II. CORE DESIGN

BLESS-D is a LBE-cooled fast neutron reactor with UO_2 fuel enriched to 12%, 16% and 19.75% (^{235}U enrichment) in 3 fuel region. The parameters for pre-conceptual design of BLESS-D are shown in TABLE I.

TABLE I. Bless-D Preliminary Conceptual Design Parameters

Parameters	Value
Thermal Power	300 MW
Coolant	LBE
Fuel	UO_2
^{235}U Enrichment (region 1/2/3)	12% /16%/ 19.75%
Average linear power density	185 W/cm
Core diameter	2422 mm
Core height	700 mm
Fuel pin pitch	10.5 mm
Fuel rod diameter	9.29 mm
Number of FA (region 1/2/3)	30/84/138
Number of Pin per FA	127
Reactor control system (CS) assembly	7
Reactor safety system (SS)	6

assembly	

A. Fuel Pin and Assembly Design

Each fuel assembly of BLESS-D includes 127 fuel pins, with UO_2 pellets as fuel and austenitic steel as the cladding. The fuel pin consists of austenitic steel cladding tube, 10 cm of upper reflector, 30 cm of lower reflector and 60 cm of gas plenum. The active region height is 70 cm. The fuel pins are in triangular arrangement with the pitch of 10.5 mm. The fuel assembly of BLESS-D are hexagonal and all assemblies have the same configuration as shown in Fig.1.

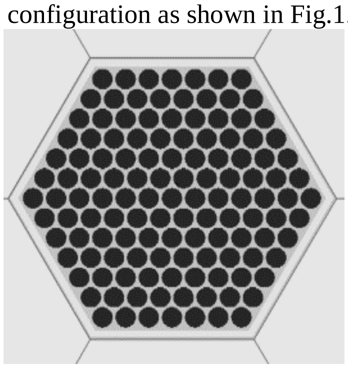

Fig. 1. Fuel assembly layout

B. Control rod assembly

The BLESS-D control and safety system contains 13 control assemblies arranged in two independent systems: reactor control system (CS) and reactor safety system (SS). The CS with 7 control assemblies serves as reactor startup, regulation of reactor power and reactivity during the fuel cycle, and reactor shut-down. The SS, with 6 control assemblies, is used only for reactor SCRAM case. All control assemblies have the same configuration as shown in Fig. 1. The control assembly is made of a cylindrical absorber bundle with 7 absorber pins cooled by primary LBE coolant (Fig. 2). The absorber material is B_4C with 90 a.t.% ^{10}B. All the control rod assembly are enclosed in a cylindrical guiding tube positioned in a particular place in the core map, as shown in Fig. 3.

For the CS, the total control rod length has been set to 90 cm. When the whole control rod assembly being inserted into the core, the bottom of the control rod assembly is 10 cm lower than the bottom of the core active area to get a larger control rod worth.

For the SS, the total control rod length has also been set to 90 cm, but all the 6 control rod assembly are extracted 10 cm higher than the top of core during the normal operation. During the emergency shutdown case, by unlocking an electromagnet device, the SCRAM action happens to shut-down the reactor in a short time. A tungsten ballast has been applied at the top of SS to guarantee the successful insertion of the control rod assembly.

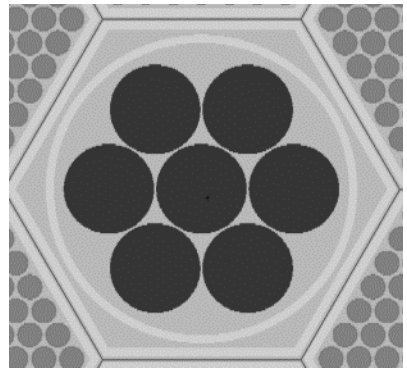

Fig. 2. Control rod assembly layout

C. Core Arrangement

According to the analysis results of fuel pellet and fuel rod performance, a reference maximum liner power density has been fixed to 185 W/cm. Considering the thermal power target is 300 MW, BLESS-D core consists of 252 fuel assemblies. The fuel assemblies have been arranged to approximate a right cylinder. The fuel assemblies of the core are arranged in 3 regions, of which the ^{235}U enrichments are 12 % in innermost region, and 16% in middle region, and 19.75% in the outermost region. ^{235}U enrichment are chosen by taking consideration of both the conversion ratio influence, shutdown margin, and flattening of the power distribution. In this case, the core radical power peak factor was confined within 1.37. The 252 fuel assemblies are surrounded by two rings of stainless shielding assemblies, which has been filled with stainless steel reflector block. The core arrangement of BLESS-D is shown in Fig. 3

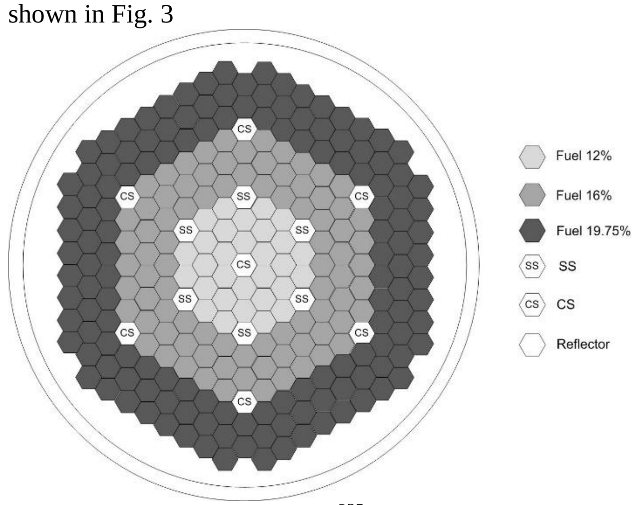

Fig. 3. Core Arrangement and ^{235}U enrichment of BLESS-D.

III. NEUTRONICS ANALYSIS

The neutronic parameters are calculated to study the core characteristics of BLESS-D.

A. Calculation Tools

The neutronics analysis was performed using the Monte Carlo Code RMC (Reactor Monte Carlo)[6]. The RMC code is a new Monte Carlo neutron transport code developed by Department of Engineering Physics in Tsinghua University, Beijing, China. The existing research shows that the RMC code gives a highly accurate results in the reactor neutronic calculation and burn-up calculation. Besides, by the application of brand new MPI parallel computation technique, high speed calculation has become the most attractive character of the RMC code, which is much more helpful in big-scale reactor and high accuracy calculation.[7] The nuclear data used in this calculation are all from ENDF/B VII.1 evaluated nuclear data library.

B Results and Analysis

B. 1. Burn-up performance

First, burn-up performance is analyzed. For now, a once-through refueling strategy is used. The calculation was done with DEPTH module from RMC code. The DEPTH module coupled RMC critical calculation module and point-burnup module, and was capable to deal with more than 1500 nuclides in the burnup chain. The burnup libraries come from both Origin-S and Origin-2. In the burnup calculation of BLESS-D, typical actinides and fission products are considered in the burnup calculation. For the sake of simplicity, the whole core was treated as one burnup calculation region at axial direction, and 3 regions at radical direction according to 3 different ^{235}U enrichment region. The calculation time step was set to 20 days and 40 days at the first to time steps, and 80 days for the rest.

As a power plant demonstration reactor, BLESS-D aims at achieving a high power performance but not a rather long life cycle and high breeding ratio. In this scenario, the results show that the fuel cycle lifetime is about 1200 EFPDs, or 3.28 EFPYs. And the average discharge burn-up is 35 GWd/tU. Fig. 4 shows the k_{eff} variation against burn-up.

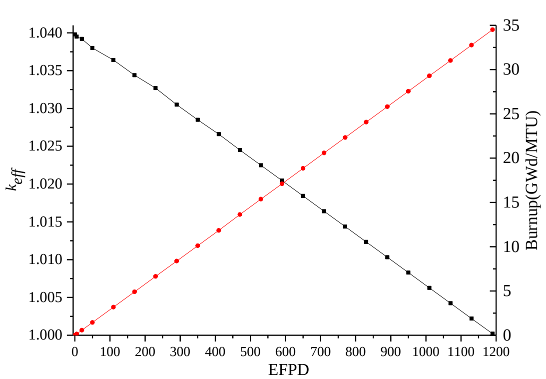

Fig. 4. k_{eff} Variations with the Burnup.

B. 2. Reactivity feedback and control parameters

As discussed above, the reactor control system (CS) is supposed to compensate the following reactivity variation:

- CZP (Cold Zero Power) to HFP (Hot Full Power) transition.
- Excess reactivity at BOL.
- Shutdown of the reactor with the most valuable control rod assembly stuck outside the core.

The total/specific reactivity requirements and reactivity worth of the CS are listed in TABLE II. The CS total requirements are the sum of the italic items listed below

TABLE II. Total/Specific Reactivity Requirements and Reactivity Worth of CS

Reactivity parameters	**Results**
Reactivity worth of CS	8689 pcm
CS total requirements	7138 pcm
CZP to HFP transition	*1429 pcm*
Excess reactivity at BOL	*3867 pcm*
Shutdown margin	*1000 pcm*
Most valuable CS assembly worth	*842 pcm*

On the other hand, the SS has to guarantee the successful SCRAM action and to ensure the enough shutdown margin. The HFP to CZP transition should also be taken into account with the most effective control rod stuck outside the core.

The total/specific reactivity requirements and reactivity worth of the SS are listed in TABLE III. The SS total requirements are the sum of the italic items listed below

TABLE III. Total/Specific Reactivity Requirements and Reactivity Worth of SS

Reactivity parameters	**Results**
Reactivity worth of SS	4593 pcm
SS total requirements	3185 pcm
Shutdown margin	*1000 pcm*
CZP to HFP transition	*1429 pcm*
Most valuable SS assembly worth	*756 pcm*

The refueling shutdown margin (k_{eff} = 0.95) is supposed to be achieved by the CS and the SS together, as shown in TABLE IV. The total requirements of refueling are the sum of the italic items listed below

TABLE IV. Refueling Shutdown Reactivity Requirements

Reactivity parameters	**Results**
Total worth of CS + SS	14470 pcm
Total requirements of refueling	12894 pcm
HFP to CZP transition	*1429 pcm*
Excess reactivity at BOL	*3867 pcm*
Refueling Shutdown requirement k_{eff}	*5000 pcm*
Shutdown margin	*1000 pcm*
Most valuable CS assembly worth	*842 pcm*
Most valuable SS assembly worth	*756 pcm*

It could be seen that the worth of every control system in the above 3 cases is all 10 % larger than the reactivity control requirements, thus it can be concluded that the reactivity control system design meets the safety criteria.

B. 3. Reactivity Coefficient.

In the preliminary calculation, reactivity coefficient at BOL has been calculated including fuel doppler coefficient, coolant temperature coefficient, and coolant voiding coefficient. A perturbation method is used in RMC code. Effective delayed neutron fraction is also calculated to support both neutronics design and safety analysis. Table V gives the reactivity coefficients calculation results. From the results, it can be concluded that Fuel Doppler coefficient, coolant temperature coefficient and coolant voiding coefficient are all negative, which meets the design criteria. For now, the reactivity coefficient induced by mechanical expansion are not calculated here because of the difficulty of calculation by RMC code.

TABLE V. Reactivity Coefficient Parameters

Reactivity Coefficient	Results
Effective delayed neutron fraction	327 pcm
Fuel Doppler coefficient	-0.37 (pcm/K)
Coolant temperature coefficient	-1.53 (pcm/K)
Coolant voiding coefficient	-31.5 (pcm/%)

III. B. 4. Power Distribution

Fig.5 and Fig.6 show the relative power distribution of a 1/6 core at the beginning of life (BOL) and end of life (EOL). The control rods of CS system are inserted at critical position, which makes k_{eff} always equals to unity, at both BOL and EOL. The control rods of SS system are extracted from the core at all time. When performing the calculation, every single fuel assembly are tallied using energy deposition tally averaged over a cell. In order to flatten the power distribution curve, the outermost fuel enrichment region is of the highest enrichment. It can be noticed that at BOL, the power peak factor is 1.364, which locates at the outermost fuel region. Whereas at EOL, the power peak factor 1.345 at the same position. It could be seen that at EOL, as the burning of high enrichment fuel, the power distribution tends to be more flattened than that of BOL. However, the peak factor is still larger than expected and could be a problem for the thermal-hydraulic design. It is necessary that the fuel assemblies' arrangement and control rod position should be further optimized in order to get lower peak factor.

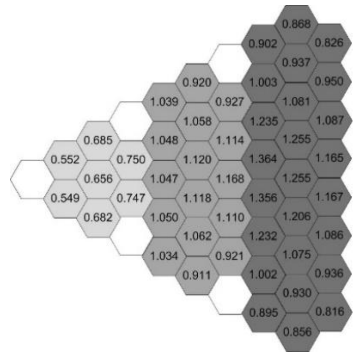

Fig. 5. Relative Power Distribution of 1/6 Core at BOL, the peak factor is 1.364

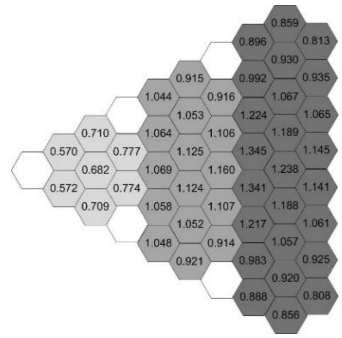

Fig. 6. Relative Power Distribution of 1/6 Core at EOL, the peak factor is 1.345

B. 5. Power distribution inside a fuel assembly

The fuel pin power peak factor inside a fuel assembly is a significant input parameter for the sub-channel analysis in thermal-hydraulics calculation. Basically, in BLESS-D reactor core, inside the fuel assembly closed to the control rod assembly of the CS has the most unflatten power distribution. In order to get the worst thermal condition, this fuel assembly are chosen to perform power distribution calculation. Fig. 7 gives the fuel pin level power peak factor inside a fuel assembly (19.75% enrichment) next to a control rod assembly. It can be seen that the peak factor is 1.142. This value is small enough for thermal-hydraulics design in the sub-channel calculation.

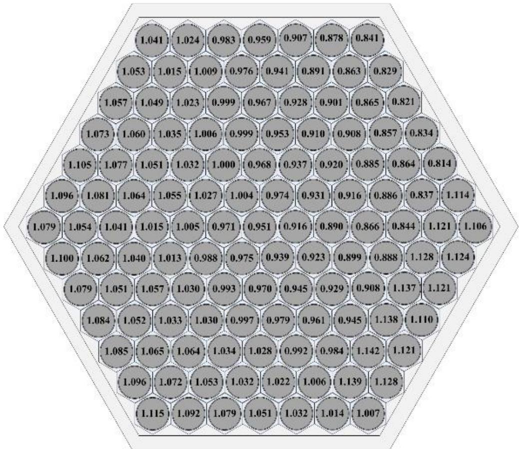

Fig. 7. Fuel pin level power peak factor inside a fuel assembly

IV. CONCLUSION

A preliminary configuration of the 300 MW Pb-Bi cooled fast reactor BLESS-D has been established. UO_2 was used as fuel with 3 compositions of ^{235}U enrichment 12%, 16%, and 19.75%. Neutronics analysis shows that fuel cycle lifetime reaches 3.28 EFPYs and the average discharge burn-up is 32.5 GWd/tU. The reactivity control performance meets the safety criteria, giving a coolant reactivity of -1.53 pcm/K, coolant voiding coefficient -31.5 pcm/%, and fuel Doppler coefficient -0.37pcm/K. The power distribution was not flattened enough with a radical power peak factor 1.364 at BOL, and 1.345 at EOL. Fuel pin level power distribution is also calculated to support thermal-hydraulics analysis. Considering the results above, further design optimizations including a refined core configuration and thermo-hydraulic analysis are deemed necessary to achieve a better performance

REFERENCES

1. Abram T, Ion S. Generation-IV nuclear power: A review of the state of the science[J]. Energy Policy, 2008, 36(12): 4323-4330.
2. Zrodnikov A V, Toshinsky G I, Komlev O G, et al. SVBR-100 module-type fast reactor of the IV generation for regional power industry[J]. Journal of Nuclear Materials, 2011, 415(3): 237-244.
3. Adamov E O, Orlov V V, Filin A I, et al. Conceptual design of BREST-300 lead-cooled fast reactor[C] //Proceedings, International Topical Meeting on Advanced Reactor Safety, ARS ‘94, Pittsburgh, USA. 1994: 509-516.
4. Alemberti A, Frogheri M, Mansani L. The lead fast reactor: demonstrator (ALFRED) and ELFR design[J]. 2013.
5. Smith C F, Halsey W G, Brown N W, et al. SSTAR: The US lead-cooled fast reactor (LFR)[J]. Journal of Nuclear Materials, 2008, 376(3): 255-259.
6. ANG K., LI Z. G., SHE D, etc, “Progress on RMC – a Monte Carlo Neutron Transport Code for Reactor Analysis”, *Proceeding of M&C 2011*, Rio de Janeiro, RJ, Brazil, May 8-12 2011.
7. YANG F, YU G, WANG K. Hybrid Shared Memory/Message Passing Parallel Algorithm in Reactor Monte Carlo Code RMC[C]. Proceedings of the Reactor Physics Asia 2015 (RPHA15) Conference Jeju, Korea, Sept. 16-18, 2015