Pre-conceptual Design of a 300MWth Pebble-bed Type HTGR: Optimization of Fuel and Fuel Management

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1. Introduction

The pre-conceptual design for a 300MWth pebble-bed type HTGR core was performed using the VSOP94 code system.[1] The fuel and core specifications were directly extended from those of the 150MWth core investigated elsewhere.[2] The core characteristic parameters including the power and temperature distributions, the reactivity temperature coefficients, and the neutron fluences at the various fuel/reflector interfaces were evaluated for both the initial core and the equilibrium core conditions.

The optimization of some design parameters of the fuel pebble was tried by investigating the sensitivity of the fuel neutronic performance with respect to the fuel kernel size, the particle packing fraction, and the fuel zone size. In addition, a clue was pursued to find the reasonable number of pebble passes through the core from the first feed to the final discharge.

2. Pre-conceptual Design

2.1 Core Specification and Characteristics

The main design parameters of the HTGR core explored in this study are shown in Table. I. The design specifications of the fuel particle and the fuel pebble that are not listed here are taken to be almost same to those of the typical German design.

Not in practice but for simplicity, the initial core is assumed to consist of only the low enriched fuel pebbles without any moderator pebbles. It is regarded for the core to fully reach the equilibrium condition after it depletes three times longer than a pebble lifetime in the core.

This core was found to be able to achieve the helium outlet temperature of 1,000 °C while maintaining the maximum fuel temperature below the normal operation limit of 1,200 °C. This may be a significant advantage of the pebble-bed type HTGR against the prismatic type HTGR. Note that the prismatic type HTGR cannot easily achieve the 1,000 °C outlet temperature due to its much higher fuel temperature. The average discharge burnup was evaluated to be about 115,000 MWD/MTU with 9.76 % enriched fuel pebbles. Some other important core characteristic parameters calculated in this study can be found in Ref. [3].

The passive cooling ability of this core under various accident scenarios should be assessed. If it fails in showing this ability, the rated thermal power may be reduced.

Table I. Core Specifications and Characteristics

Parameter	Value
Thermal power (MW)	300
Inlet/Outlet helium temperature (°C)	490/1,000
Active core radius (cm)	175
Thickness of outer reflector (cm)	100
Active core height (cm)	890
Thickness of top/bottom reflector (cm)	100/270
Top cavity height (cm)	55
Average power density (W/cc)	3.5
Uranium loading in core (Kg)	3,060
Average pebble packing fraction	0.61
Number of pebbles in core	433,000
Average number of pebble passes	10
Average residence time of pebbles (day)	1,200
Total/Fresh pebble feed rate (/day)	3,600/360
U^{235} enrichment of initial/equilibrium core (%)	2.65/9.76
Average discharge burnup (MWD/MTU)	115,000

2.2 Fuel Optimization

Encouraged from the conclusion of Ref. [4] pointing out the great possibility that the current design of the typical German fuel pebble can be improved, an effort to optimize the design parameters of the fuel pebble is made. The design parameters considered in the optimization are the kernel radius of the particle, the fuel zone radius in the pebble, and the particle packing fraction in the fuel zone.

Fig. 1 shows the core effective multiplication factors at various optimization parameters listed above. All the curves but the red ones are calculated at the fixed uranium loading per pebble. The red curves in this figure are obtained with varying the fuel zone radius at the fixed packing fraction of same size particles. In this case, the uranium loading varies with the radius. This red curves show that the more uranium loading results in the higher effective multiplication factor. This is quite contradictory to the result of Ref. [4] where a peak of the multiplication factor can be seen near the optimum fuel zone radius. The other curves in this figure show that the effective multiplication factors are almost independent of the corresponding optimization parameters at the least equilibrium core condition although they show some dependencies at the initial core condition. Considering that the pebble-bed core is operated at the equilibrium condition over 90% of its lifetime, this leads to the conclusion that there is not much need to optimize the design parameters investigated at the least here.

It can be judged that the conflict between our results and those in Ref. [4] is ascribed to the different code systems used. This leaves room to promote an international collaboration in benchmarking the code systems.

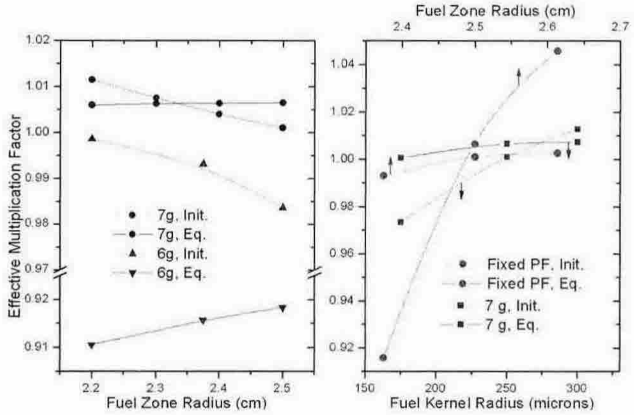

Fig. 1 Optimization of Fuel Design Parameter

2.3 Fuel Management Optimization

Controlling the average number of pebble passes through the core is one of not-many possible means to manage the fuel in a pebble-bed reactor. The higher value we have the better performance we can expect in a viewpoint of fuel utilization, because the perfect online refueling results theoretically in the best fuel utilization. However, the high number of pebble passes through the core has adverse effects such as the graphite dust problem in the Helium coolant system and the heavy burden to the fuel handling system. Note that the dust is not mainly produced in the core but in the fuel handling system.

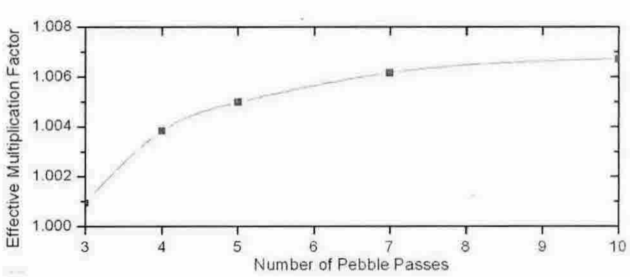

Fig. 2. Optimization of Number of Pebble Passes

A clue is pursued to determine the pass number that is as small as possible but does not inflict a serious loss in fuel utilization. Fig. 2 shows the effective multiplication factors vs. the pass numbers at the equilibrium core condition. The smaller pass number in this figure means the longer core stay per pebble pass, because the discharge burnup for all the cases is fixed to 115,000 MWD/MTU. It can be concluded from this figure that a number near seven that is taken by the South African PBMR is acceptable.

3. Conclusion

The pre-conceptual design for a 300MWth pebble-bed type HTGR core performed in this work shows that this core can achieve the outlet temperature of 1,000 °C without violating the limit of the maximum fuel temperature and that it can reach the average discharge burnup of about 115,000 MWD/MTU with the 9.76 % enriched fuel.

The results of the optimization of the fuel pebble shows that there is not much need to optimize the fuel design parameters as far as the equilibrium core is considered.

Furthermore, a number near seven as the average number of pebble passes through the core is found to be acceptable in the sense that this number causes neither the big dust problem in the coolant system nor a serious loss in fuel utilization.

The passive cooling ability of this core during various accidents should be assessed in the near future. In order to resolve some conflicts between our results and others, an international collaboration in benchmarking the computer code systems may be promoted.

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REFERENCES

[1] E. Teuchert et al., "VSOP('94) Computer Code System for Reactor Physics and Fuel Cycle Simulation," FZJ Internal Report, *Jul-2897*, 1994.

[2] J.M. Noh et al., "A Preliminary Conceptual Design for a 150 MWth Pebble Bed Reactor Core Using the VSOP94 Code Package," Proc. KNS Spring Meeting, Gyeong-ju, Korea, May 29-30, 2003, CD-ROM, Kor. Nucl. Soc., 2003. (in Korean)

[3] Basic Study on High Temperature Gas Cooled Reactor Technology for Hydrogen Production, KAERI Internal Report, *KAERI/RR-2453/2003*, 2003. (in Korean)

[4] A. M. Augouag et al., "Optimal Moderation in the Pebble-Bed Reactor for Enhanced Passive Safety and Improved Fuel Utilization," Proc. PHYSOR 2004, Chicago, Illinois, April 25-29, 2004, CD-ROM, Am. Nucl. Soc., 2004.