A Neutronics Analysis for 6 MWth Small-Size Molten Chloride Salt Reactor Core

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1. Introduction

In the recent years, there has been a growing interest in the Molten Salt Reactor (MSR), which is originally suggested by the Oak Ridge National Laboratory (ORNL) for a military aircraft nuclear propulsion project, owing to its unique characteristic in liquid fuel cycle management [1-3].

Owing to the characteristics of liquid fuel, it is believed that MSR can be a feasible candidate for small-size reactor. One remarkable advantage of the MSR is a potential in minimizing reactor core size due to 1) sufficient margin in fuel and structure temperature, and 2) inherent strong negative reactivity coefficients of liquid fuel.

In this study, based on the molten chloride salt fast reactor design suggested by the Ref. [4], feasibility of small-size core concept with TRU fuel was studied with the MCNP6 and ENDF/B-VII.1 library [5].

2. A Reference Model and Neutronic Characteristics of Reflector Materials

The spherical core model with molten chloride fuel, reactor vessel, outer reflector, and shield is considered as a reference core model with 650 ℃ operating temperature. The detailed geometry and materials are shown in Table I [6] and detailed fuel compositions are based on the KAERI's internal research results [7]. The criticality of the reference core is 1.00294 (1σ = 0.00057).

Table I: Materials and geometry of the reference core

Component	Materials	Outer radius, cm
Fuel	$NaCl\text{-}MgCl_2\text{-}TRUCl_3$	58.5
Reactor vessel	Hastelloy-N	63.5
Outer reflector	SS304	103.5
Shield	B_4C	123.5

2.1 Effect of Various Outer Reflector Materials

To reduce core size preserving criticality, various outer reflector materials as shown in table II are considered as outer reflectors to enhance neutronic economy.

Table II: Considered outer reflector materials

Materials	Densities, g/cm^3
SS304	8.00
Beryllium metal	1.85
Beryllium pebble	1.17
MgO	3.60
BeO	3.01
$Be_{12}Ti$	2.26
Graphite	1.78

Criticalites with various outer reflector materials are shown in Fig.1.

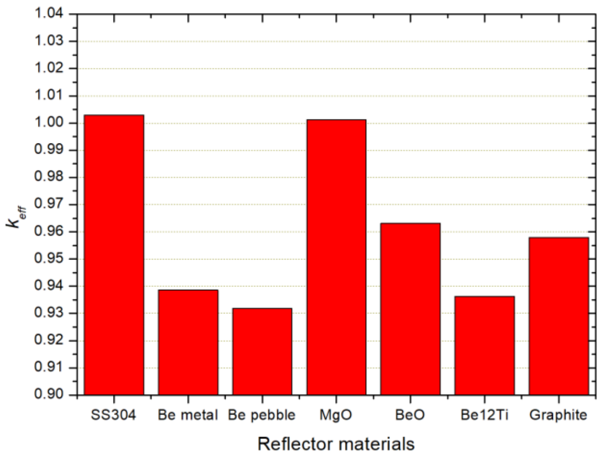
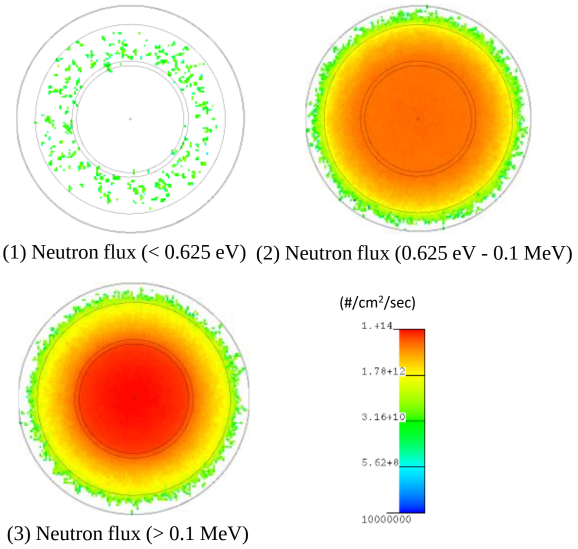


Fig. 1. Criticalities with various outer reflector materials

In spite of good moderating characteristics of candidate outer reflector materials, lower criticalities are resulted due to the significant thermal neutron absorption at reactor vessel materials as shown in Figs. 2 and 3.

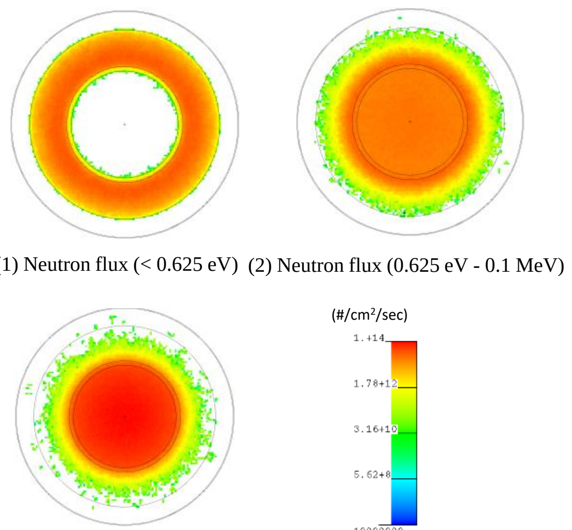
More detailed results of thermal neutron absorption characteristics of reactor vessel materials will be discussed at section 2.2.



(1) Neutron flux (< 0.625 eV) (2) Neutron flux (0.625 eV - 0.1 MeV)

(3) Neutron flux (> 0.1 MeV)

Fig. 2. Neutron flux distributions with SS304 outer reflector



(1) Neutron flux (< 0.625 eV) (2) Neutron flux (0.625 eV - 0.1 MeV)

(3) Neutron flux (> 0.1 MeV)

Fig. 3. Neutron flux distributions with beryllium metal outer reflector

2.2 Neutron absorption characteristic of reactor vessel materials

Figs. 4 and 5 show microscopic and macroscopic neutron absorption reaction rates of candidate reactor vessel materials, Hastelloy-N and SS316. Reaction rates are calculated using the reference core model.

At a view point of microscopic reaction rate, W and Mo are the most significant neutron absorption elements. Mn and Ni are the next significant neutron absorption elements.

In case of Hastelloy-N, due to the relatively large Ni (70.6%) and Mo (16.8%) fractions, more significant macroscopic neutron absorption reaction rates were resulted than SS316 material.

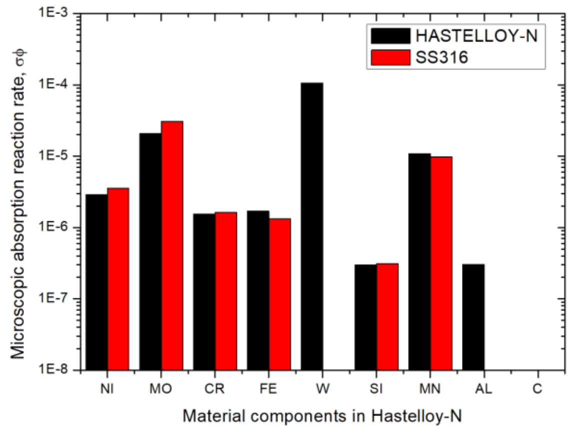


Fig. 4. Microscopic neutron absorption reaction rates

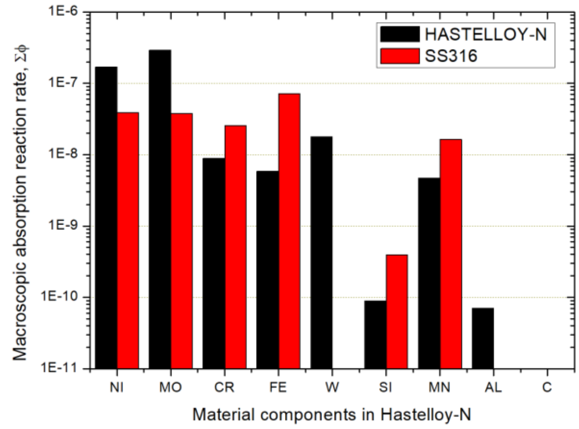


Fig. 5. Macroscopic neutron absorption reaction rates

Hence SS316 is a better reactor vessel material than Hastelloy-N at a view point of neutronics. However, since absorption reaction rates of SS316 is not small enough, the improvement of criticalities are limited as shown in Fig. 6.

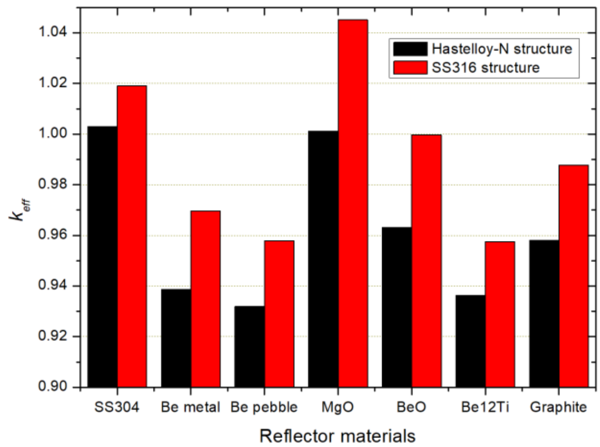


Fig. 6. Criticalities with various outer reflector materials for Hastelloy-N and SS316 reactor vessel materials

In Fig. 6, MgO showed the best results due to appropriate moderation characteristics. Beryllium materials showed relatively lower criticalities due to over-moderation and thermal neutron absorption at SS316 material.

2.3 Effect of Various Inner Reflector Materials

To decrease neutron absorption by structure material, inner reflector concept is adopted as shown in fig. 7. To study effect of inner reflector materials, 40 cm thickness total reflector size and SS304 outer reflector material are assumed.

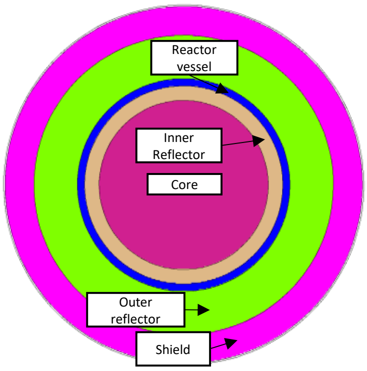


Fig. 7. An example of core model with inner reflector

Criticalites with various inner reflector materials are shown in Fig.8, in which BeO showed the best results among considered inner reflector materials due to significant moderation by (n,2n) reactions.

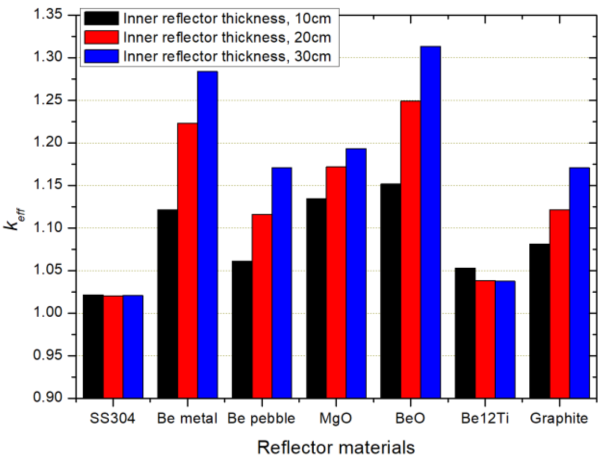


Fig. 8. Criticalities with various inner reflector materials

Effective neutron dose rates at 1 cm distance from surface of shield are shown in Fig. 9. Although SS304 outer reflector material was assumed, neutron moderations by inner reflector showed significantly lower neutron dose rates.

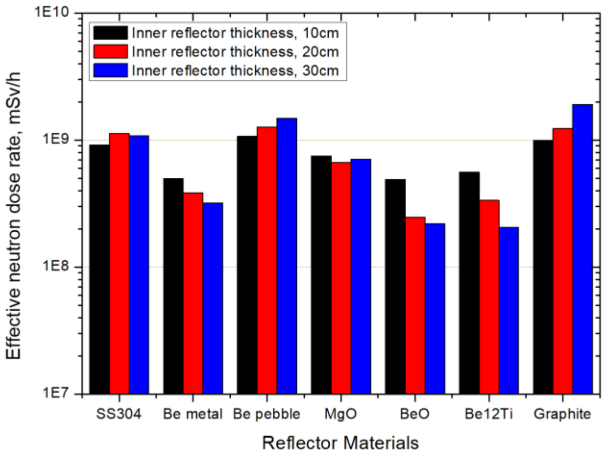


Fig. 9. Effective neutron dose rates with various inner reflector materials

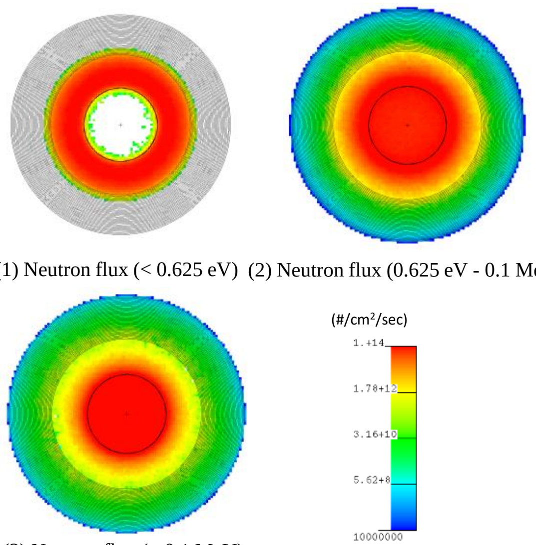
3. A Small-Size Reactor Core Model

Based on the reflector material characteristics studied so far, 30 cm-thickness BeO inner reflector with inner reflector case, reported at Ref. [8], are considered. The final reactor core dimension for criticality is listed at Table III. In which, SS316 is selected as a reactor vessel material.

Table III: Core dimension for a small-size reactor

Component	Outer radius, cm
Core	33
Inner reflector case	33.4
Inner reflector	63.4
Reactor vessel	68.4

Neutron flux distributions of a small-size core with 30 cm-thickness B_4C shield is shown in Fig. 10 and those with 45 cm-thickness $Mg(BH_4)_2$ and 5 cm-thickness B_4C shield is shown in Fig. 11 [8].



(1) Neutron flux (< 0.625 eV) (2) Neutron flux (0.625 eV - 0.1 MeV)

(3) Neutron flux (> 0.1 MeV)

Fig. 10. Neutron flux distributions for 30 cm-thickness B_4C shield case

Effective neutron dose rates at 1 cm distance from surface of shield are 3.68430E+08 mSv/h and 1.35815E+05 mSv/h, respectively. Standard deviation of effective neutron dose rates is less than 5 %. Neutron flux at 1 cm distance from surface of shield are 2.45868E+08 #/cm^2/sec and 4.39145E+04 #/cm^2/sec, respectively. For usual worker dose rate limit, 5 μSv/hr, resulted dose rates are still significant [9]. However, considering SDDR (ShutDown Dose Rate) limit of ITER, 6.0E+4 #/cm^2/sec, 45 cm-thickness $Mg(BH_4)_2$ and 5 cm-thickness B_4C shield case satisfied the SDDR limitation [10].

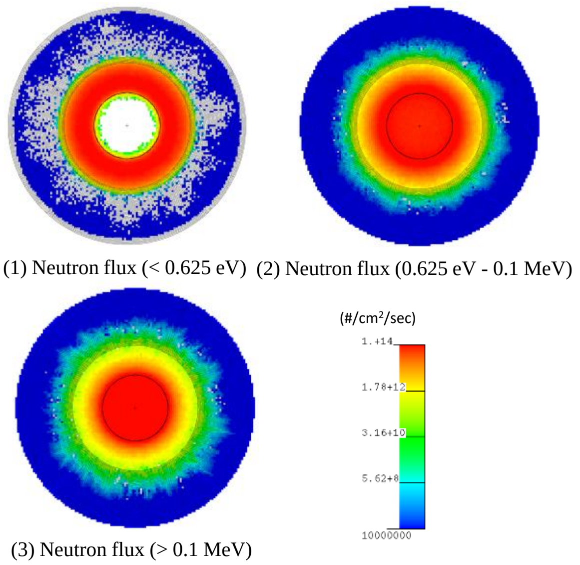


Fig. 11. Neutron flux distributions for 45 cm-thickness $Mg(BH_4)_2$ and 5 cm-thickness B_4C shield case

Since thermal neutron is mostly absorbed by few cm-thickness B_4C shield, neutron dose rate at 1 cm distance from surface of shield is originated by fast neutron as shown in Fig. 12. For fast neutron shield, as shown in Figs. 10 and 11, high-moderating power material such as $Mg(BH_4)_2$ is better than conventional neutron absorbing material such as boron.

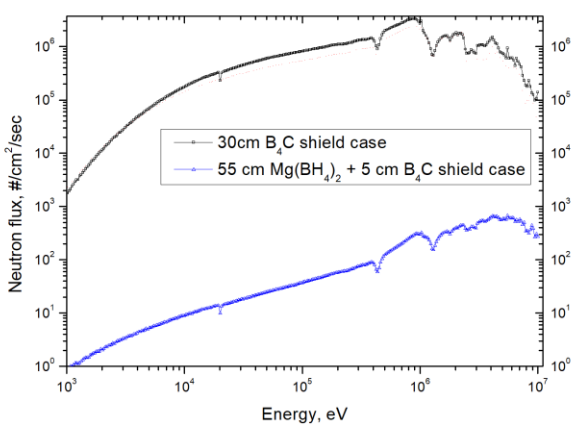


Fig. 12. Neutron spectrum of a small-size core at 1 cm distance from surface of shield

3. Conclusions

In this paper, a 33 cm-radius small-size reactor core with molten chloride TRU fuel is suggested with 30 cm-thickness inner reflector, 45 cm-thickness $Mg(BH_4)_2$ and 5 cm-thickness B_4C shield concept. The outer diameter of reactor with neutron shield is expected to the 236.8 cm with 0.4 cm-thickness inner reflector case.

However, the size of core may be increased when appropriate control, I&C, cooling systems are considered together. In addition, usage of HALEU (High-Assay Low-Enriched Uranium) fuel with identical molten chloride composition will also increase core size.

Hence, appropriate control system design, neutron shield design with high-moderating power material considering cooling, and usage of HALEU fuel instead of TRU fuel are planned as a future study.

4. Acknowledgments

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