TR0700351

NEUTRONICS OF A LIQUID SALT COOLED - VERY HIGH TEMPERATURE REACTOR

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ABSTRACT

Background

During last few years, the interest in the innovative, Liquid Salt cooled - Very High Temperature Reactor (LS-VHTR), has been growing. The preconceptual design of the LS-VHTR was suggested in Oak Ridge National Laboratory (ORNL) [1] and nowadays, several research institutions contribute to the development of this concept. The LS-VHTR design utilises a prismatic, High Temperature Reactor (HTR) fuel [2] in combination with liquid salt as a coolant. This connection of high-performance fuel and a coolant with enhanced heat transfer abilities enables efficient and economical operation. Main objective of the LS-VHTR operation may be either an efficient electricity production or a heat supply for a production of hydrogen or, combination of both. The LS-VHTR is moderated by graphite. The graphite matrix of the fuel blocks, as well as the inner and outer core reflectors serve as a thermal buffer in case of an accident, and they provide a strong thermal feedback during normal reactor operation. The high inherent safety of the LS-VHTR meets the strict requirements on future reactor systems, as defined by the Gen IV project.

This work, purpose, scope, contribution to the state-of-art

The design, used in the present work is based on the first ORNL suggestion [1]. Recent study is focused on comparison of the neutronic performance of two types of fuel in the LS-VHTR core, whereas, in all previous works, only uranium fuel has been investigated. The first type of fuel, which has been employed in the present analysis, is based on the spent Light Water Reactor (LWR) fuel, whereas the second one consists of enriched uranium oxide. The results of such a comparison bring a valuable knowledge about limits and possibilities of the LS-VHTR concept, when employed as a spent fuel burner.

Method

Figure 1 shows a 3-D drawing of the LS-VHTR core, which contains 324x10 hexagonal fuel blocks. Each fuel block contains 216x10 fuel pins, which consists of TRISO particles incorporated into a graphite matrix. The external radius of a TRISO particle has been set to 410μm, the radius of the fuel kernel to 150 μm, in case of plutonium fueled core, and 215 μm in case of uranium fueled core. The core was modelled in stochastic, three-dimensional code MCNP, version 4c3, in the finest detail. First, an undermoderated core setup was found for both types of fuel by modifying the fuel to moderator ratio; then, the void and the thermal coefficients of reactivity were investigated. Few single - component molten salts were involved in the study of the void effect, in order to estimate worth of these components; NaF, BeF_2, LiF, ZrF_4. As a reference multi-component salt, Li_2Be_4F, referred to as FLiBe, was investigated.

Results

Table 1 summarizes the total void worth achieved with the individual liquid salt components and FLiBe. It can be seen, that removing BeF_2 from the core brings a negative reactivity contribution, while other three components, NaF, LiF and ZrF_4 would in a mixture contribute to the reactivity positively. Voiding FLiBe, which is a mixture of 66% of LiF and 34% BeF_2, is equivalent to a negative reactivity insertion. Both the moderator and the fuel temperature coefficients of reactivity are large and negative for both plutonium and uranium fueled core. In the operational temperature interval (1200 K for graphite and 1500 K for fuel), the total temperature feedback is -7.82 pcm/K for the plutonium fueled core and -2.47 pcm/K for the uranium fueled core. This results show, that the LS-VHTR core has a potential to meet the basic safety requirements as both uranium, and spent LWR fuel burner.

Figures and Tables

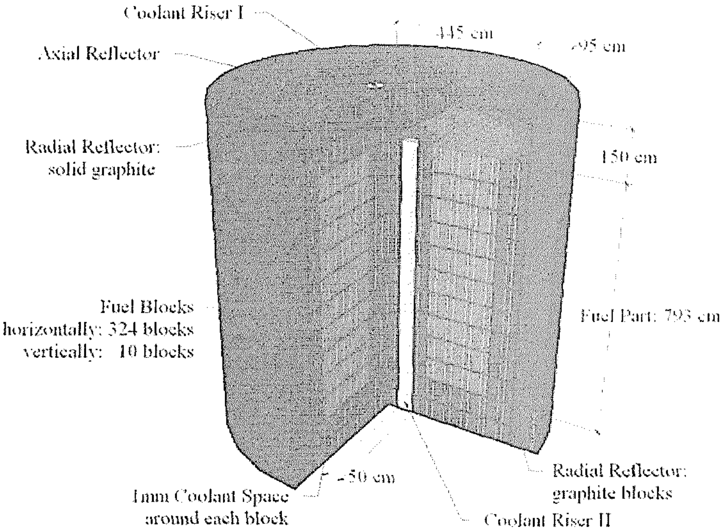

Figure 1: Core of LS-VHTR

Table 1 : Total void worths for selected liquid salts and core configurations

Liquid Salt	**Plutonium Fuel, kernel 150 µm, pf 14.45%**		**Uranium Fuel; kernel 215 µm, pf 24.32%, enrichment 15%**	
	[$]	**[pcm]**	**[$]**	**[pcm]**
Li_2BeF_4	-2.40	-641	-0.42	-308
BeF_4	-5.26	-1405	-1.61	-1194
Li_2	+0.31	+83	+0.61	+449
NaF	+9.48	+2530	+7.50	+5545
ZrF_4	+4.61	+1230	+2.09	+1551

References

[1] D. T. Ingersoll, L. J. Ott, J. P. Renier, S. J. Ball, W. R. Corwin, C. W. Forsberg, D. F. Williams, D. F. Wilson, L. Reid, G. D. Del Cul, P. F. Peterson, H. Zhao, P. S. Pickard, E. J. Parma. *Status of Preconceptual Design of the Advanced High-Temperature Reactor.* (ORNL, The United States of America, Tennessee 2004).

[2] A. Talamo, W. Gudowski, F. Venneri, *Annals of Nuclear Energy*, **31**, 173-196 (2004), The burnup capabilities of the Deep Burn Modular Helium Reactor analyzed by the Monte Carlo Continuous Energy Code MCB.