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Short Communication

Producing Hydrogen-3 by irradiating lithium orthosilicate targets in a fission research reactor

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ABSTRACT

Neutron flux in the eight irradiation boxes of the 5 MW Tehran low power research reactor computes by MCNP Monte Carlo N-Particle transport code. The lithium orthosilicate, an interesting material in fusion engineering, encapsulated by quartz cylindrical containers positioned in the irradiation boxes to produce tritium. According to the weight of the tritium produced in the Tehran research reactor in comparison with reference reactor, this type of low power reactor can be used as a source of external tritium supplier in the fusion research facility.

Introduction

Hydrogen-3 (Tritium) is the most important fuel for future power plant, fusion reactors. It is not available in nature and so must be produced through transmutation of lithium. Currently, lithium ceramics used as breeder material in fusion reactor blanket designs that surround the fusion plasma [1]. For example, lithium orthosilicate (Li_4SiO_4) pebbles selected as a breeder material for the European helium cooled pebble bed blanket [2]. Lithium orthosilicate has the second highest lithium atom density after lithium oxide. Moreover, it does not appear to have a severe problem with water and CO_2 reactions [3]. The attainable tritium breeding ratio in the fusion reactor blanket system must be larger than the required breeding ratio. Construction of an outside tritium supply is one of the possible ways to compensate the lack of tritium breeding ration [4].

In 2008, out of pile tritium release experiments for the required amount of fusion research studies were performed using slightly over-stoichiometric lithium orthosilicate pebbles which irradiated in the 5 MW light water tank-type Kyoto University research reactor [1].

It is reported recently that high temperature gas cooled reactor with a total thermal output power of 3 GW may become a candidate to produce 5–8 kg of tritium maximum in a year [5–8]. In the literature, MCNP5 based McDeLicious code and MVP-BURN Monte Carlo transport code can be used to calculate the inventory of the tritium.

In this study, a 5 MW research reactor having an orthosilicate tritium breeder targets modeled by MCNP code to calculate the number of generating neutrons

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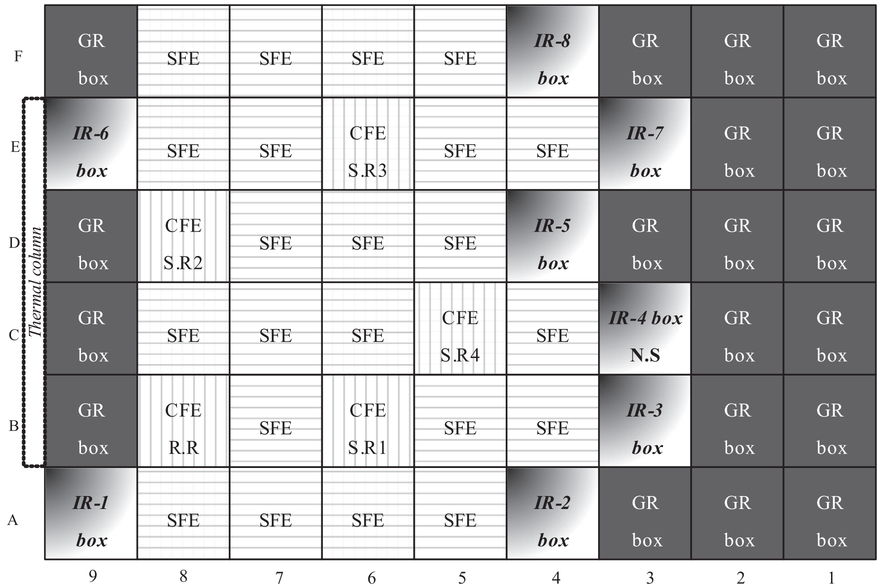

Fig. 1 – Configuration of Tehran research reactor core. SFE = standard fuel element; CFE = control fuel elements; R. R = regulating rod; GR box = graphite box; IR-x = irradiation box number; N. S = neutron source.

(neutron flux) in different irradiation sites. However, golden foil activation experiment can be used to obtain an exact shape of the neutron flux distribution in the core at different points [9]. Using the shape of the neutron flux, the amount of the produced tritium can be calculated analytically.

Methods and materials

MCNP code had been verified for tritium production evaluation procedure in fusion engineering [10]. Therefore, Tehran research reactor (TRR), a fission research reactor, modeled using the MCNP code which the lithium orthosilicate targets positioned in the irradiation boxes. The number of neutrons calculated in the empty irradiation boxes to find the irradiation box with highest neutron flux. Moreover, the number of neutrons calculated in the irradiation boxes which containing quartz tubes. Finally, the mass and density of the hydrogen-3 can be calculated analytically in the irradiation boxes.

TRR simulation by MCNP code

The Tehran 5 MW research reactor is loaded with U_3O_8–Al fuel element, with an enrichment of 19.7% of uranium, Known as low enriched. Maximum neutron flux in the reactor core is about 3×10^{13} neutrons/cm^2/s. Light water acts as coolant, moderator, reflector and biological shield. The core contains two types of the fuel elements, standard fuel elements (SFE) with 19 fuel plates and control fuel elements (CFE) with 14 fuel plates. However, the CFE conjugated with four Ag–In–Cd shim safety rods and a 316L-SS fine regulating rod. The reactor core configuration which used in this study is shown in Fig. 1 [11].

All components of the reactor core were simulated by MCNP-4C code developed by the Los Alamos National Laboratory [12]. Fig. 2 shows the cross view of the modeled core in the MCNP code.

Center of the some fuel boxes considered as fission points. Thus KSRC source point card applied to KCODE calculation. To compute the neutron flux in irradiation boxes, F4 tallies were used in a range of energy with the proper E4 card. The E4 card considered between 10^{-5} and 10^5 MeV (E4 1e-5 1e-4 1e-3 1e-2 1e-1 1 1e1 1e2 1e3 1e4 1e5).

Results and discussion

Neutron flux in empty irradiation boxes

Since the MCNP code computes the number of generating neutrons per collided particle in the computational cells. So,

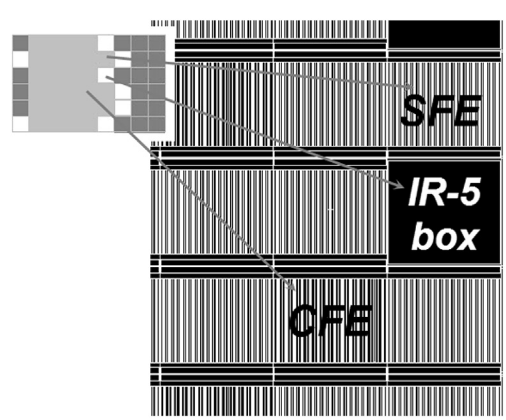

Fig. 2 – A cross view of the TRR core in MCNP code.