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Monte Carlo calculations of the nuclear effects of certain fluids in a hybrid reactor

Mehtap Günay*

İnönü University, Science and Art Faculty, Physics Department, Malatya, Turkey

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ABSTRACT

This study analyzes the effects of certain heavy-metal-salt fluids on nuclear parameters in a fusion–fission hybrid reactor. Calculated parameters include the tritium breeding ratio (*TBR*), energy multiplication factor (*M*), heat deposition rate, fission reaction rate, and fissile fuel breeding in the reactor's liquid first wall, blanket, and shield zones; gas production rates in the structural material of the reactor were calculated, as well. The fluid mixtures consisted of 93–85% $Li_{20}Sn_{80}$ + 5% SFG-PuO_2 and 2–10% UO_2, 93–85% $Li_{20}Sn_{80}$ + 5% SFG-PuO_2 and 2–10% NpO_2, and 93–85% $Li_{20}Sn_{80}$ + 5% SFG-PuO_2 and 2–10% UCO. The fluids were used in the liquid first wall, blanket, and shield zones of a fusion–fission hybrid reactor system. A 3 cm wide beryllium (Be) zone was used for neutron multiplier between the liquid first wall and the blanket. The structural material used was 9Cr2WVTa ferritic steel, measuring 4 cm in width. Three-dimensional analyses were performed using the Monte Carlo code MCNPX-2.7.0 and the ENDF/B-VII.0 nuclear data library.

1. Introduction

In traditional nuclear reactor, harmful long-lived fission products and minor actinides are produced by fission and other nuclear reactions. Therefore, the waste management required as a result of fuel burning to generate energy from a nuclear reactors is an important problem. One possible solution is the transmutation of nuclear waste into stable and short-lived isotopes through nuclear reactions in order to reduce the amount of the long-lived fission products and minor actinides. Therefore, a hybrid reactor system has been developed in which nuclear fusion and fission events occur simultaneously to obtain both more energy and nuclear fuel, decrease the waste in storage areas, recover transuranic elements in used fuels through reprocessing.

The fuels used in hybrid reactor systems are generally deuterium–tritium (D–T) or deuterium–deuterium (D–D) fuels. When D–T fuel undergoes the fusion reaction, 14.1 MeV fusion neutrons and 3.5 MeV alpha particles are released. The plasma is surrounded by a wall of fertile material (^{238}U, ^{232}Th), which is not itself capable of undergoing fission with thermal neutrons but can undergo conversion by high energy neutrons, such as the 14.1 MeV neutrons produced by the fusion reaction. Thus, the high energy fusion neutrons convert fertile materials to fissile fuel (^{239}Pu, ^{233}U) and fission neutrons and produce energy. A hybrid reactor (HR) produces 30 times more fissile fuel per nuclear energy (E) quantity than a fast breeder (FB) (Şahin and Übeyli, 2005; Şahin, 2007; Şarer et al., 2007; Günay et al., 2011, 2013, 2014; François et al., 2013). The fissile breeding ratio (BR) value can be obtained from the following equation (Teller, 1981; Şahin et al., 2001).

$$\frac{\left(\frac{BR-1}{E}\right)_{HR}}{\left(\frac{BR-1}{E}\right)_{FB}} = \frac{\left(\frac{1.8-1}{27}\right)}{\left(\frac{1.2-1}{200}\right)} = 30$$

To increase the lifetime of the reactor, Christofilos suggested that the first wall surrounding the plasma should be liquid instead of solid (Christofilos, 1989; Moir, 1997). The liquid first wall in a hybrid system is located beyond the plasma and a liquid second wall is located beyond the liquid first wall. Both liquid walls confine charged particles, significantly reducing radiation damage to the structural materials, and absorb the energy of the neutrons by converting it into heat (Abdou et al., 1999, 2001; Abdou, 2001; Abdou, 2004; Abdou et al., 2005; Ying et al., 1999; Youssef and Abdou, 2000; Youssef et al., 2002).

* Corresponding author. Tel.: +90 4223773878.
E-mail address: mehtap.gunay@inonu.edu.tr.

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The hybrid reactor system can generate large quantities of energy securely using D–T fuel in a subcritical system ($k_{eff} < 1$, k_{eff}: effective multiplication constant) to avoid a critical accident (Wu et al., 2013). Furthermore, it self-sufficiently enables the production of its own fuel through the reactions of the neutrons released by the plasma. A substantial amount of spent fuel-grade (SFG) plutonium (Pu) from current nuclear reactors has been stored for future use. The isotopic distribution of SFG-Pu has the composition: 2.4% ^{238}Pu, 58.5% ^{239}Pu, 24% ^{240}Pu, 11.2% ^{241}Pu, and 3.9% ^{242}Pu (IAEA, 2003). SFG-Pu is valuable, but the emission of the radiation from this material into the environment should be prevented for human health and safety because SFG-Pu is a toxic substance. Therefore, SFG-Pu must be kept under strong security or it must be reduced by nuclear transmutation in suitable reactors to maintain environment safety. Nuclear transmutation is the conversion into another of one isotope by nuclear reactions or radioactive decay. This process can occur through natural, or it may be artificially induced by human intervention (Liu et al., 2015). For this reason, to reduce the amount of SFG-Pu needed through transmutation, radioactive materials with a plutonium additive were used in this study in the hybrid reactor system.

This study used UO_2, NpO_2, and UCO as fluids at 2–10% rates to produce fissile fuel, fission neutrons, and energy from the high energy fusion neutrons and to increase neutron performance. To ensure sufficient tritium breeding, $Li_{20}Sn_{80}$ molten salt was used, which has a low melting temperature and low vapor pressure. $Li_{20}Sn_{80}$ has been recently investigated as a new candidate tritium breeder to be used in fusion reactors. Beryllium (Be) is used as a neutron multiplier. The Be (n,2n) reaction has an effective threshold of 2.5 MeV, above which the cross-section is 0.5 b (Kowash, 2002; Piera et al., 2010). This reaction, with its fast-neutron spectrum, effects on nuclear parameters such as tritium breeding, energy multiplication and heat deposition in blanket (Morley et al., 1999; Kondo et al., 2011). Therefore, in this study, a Be zone with a thickness of 3 cm was used between the liquid first-wall and the blanket. 9Cr2WVTa ferritic steel was chosen as a structural material because it has a low activation cross-section, which can help to extend the reactor system life.

In this study, the MCNPX-2.7.0 Monte Carlo code and the ENDF/B-VII.0 nuclear data library were used for the three-dimensional neutron measurements.

Calculations were made of the contributions of the fluids to the three-dimensional neutron measurements, such as the tritium breeding ratio (*TBR*), energy multiplication factor (*M*), heat deposition rate, fission reaction rate, fissile fuel breeding in liquid first wall, blanket, and shield zones and the gas production rates in structural material. The primary objective of this study is to investigate the effect of the selected fluid on the neutron-related quantities for the hybrid system being designed.

2. Calculation of nuclear quantities for the hybrid reactor

2.1. Hybrid reactor

Fig. 1 shows radial cross-sectional of the designed hybrid reactor system (a) and zone geometry from top (b). The hybrid reactor system was toroidal; and the radius of the torus is 552 cm. The fast-flowing liquid first wall was 2 cm thick, and the slow-flowing layer (blanket) was 50 cm thick. The fast and slow-flowing layers flowed around the torus because of the effect of the magnetic field on the liquid. Beryllium (Be) was used as a neutron multiplier. Therefore, in this study a Be zone was used between the liquid first wall and blanket. A solid backing wall 4 cm thick made of 9Cr2WVTa ferritic steel followed the blanket zone; the steel had a low activation cross-section as a structural material. A shielding zone 50 cm thick (outboard) and 49 cm thick (inboard) was located behind the backing solid wall for the outboard and inboard builds, respectively, and was assumed to have a structure-to-breeder (coolant) volume ratio of 60:40. The vacuum vessel wall was SS316LN stainless steel 2 cm thick. The interior was SS316LN stainless steel, 16 cm thick (inboard) and 26 cm thick (outboard), cooled with water using a structure-to-water ratio of 80:20 (Ying et al., 1999).

2.2. Numerical calculations

The evaluated nuclear data file ENDF/B was first developed in the USA in 1968 (Oblozinsky, 2006). New versions are published periodically following large-scale investigations and additional research. ENDF/B-VII libraries, which are used for theoretical calculations in MCNPX, include nuclear data from 10^{-11}–20 MeV for certain isotopes and up to 150 MeV for other isotopes (Chadwick et al., 2011; Kahler et al., 2011; Pelowitz, 2011; Pritychenko et al., 2010; Smith, 2011). The data of the ENDF/B-VII libraries are very important for theoretical calculations.

The Monte Carlo method is generally used because of the complex three-dimensional configurations of the materials and the very complicate problems as optimization, numerical integration couldn't solve of deterministic transport methods, which it was used before the Monte Carlo method was developed (Dufek, 2009). But, Monte Carlo method does not solve an explicit equation, it can solve by simulating particles and recording some aspects (tallies) of particles average behavior. Therefore, the Monte Carlo method solves transport problem by simulating particle histories (Briesmeister, 2000). The MCNPX-2.7.0 transport code uses standard cross-section libraries compiled from ENDF/B-VII.0 for neutron, proton and photonuclear interactions. Different intranuclear, preequilibrium and evaporation-fission models have been implemented into MCNPX-2.7.0 version, which offers seven different options based on four physics packages: Bertini (Bertini, 1963, 1969) and ISABEL (Yariv and Fraenkel, 1979, 1981), INCL4 (Boudard et al., 2002; Cugnon, 1987; Cugnon et al., 1997), the CEM2k (Mashnik and Toneev, 1974) package and two evaporation-fission models Dresner (Dresner, 1962), ABLA (Junghans et al., 1998). Bertini, ISABEL, and INCL4 are INC models, which can be coupled with ABLA and Dresner evaporation–fission codes. CEM2k is a cascade-preequilibrium-evaporation model (Şarer et al., 2012; Günay and Kasap, 2014). In this study, the neutron, proton and photonuclear cross section libraries LA150n, LA150u and LA150p include cross section data for four tungsten isotopes in structural material and shield zones: 182W, 183W, 184W and 186W. The libraries have no data for 180W. Hence the isotopic fraction of this isotope (0.13%) was merged in the 182W.

In this study, the fluids were composed of molten salt (93–85% $Li_{20}Sn_{80}$ + 5% SFG-PuO_2) as the main constituent, with increasing mole fractions of 2–10% rates of the heavy metals UO_2, NpO_2, and UCO. The fluids were used to increase neutron performance and to decrease the amount of SFG-PuO_2 in the liquid first wall, blanket, and shield zones of the hybrid reactor system. In this study, neutron transport properties such as the tritium breeding ratio (*TBR*), energy multiplication factor (*M*), heat deposition rate, fission reaction rate, fissile fuel breeding, and parameters of radiation damage were investigated using the MCNPX-2.7.0 Monte Carlo code and the ENDF/B-VII.0 nuclear data library. The high neutron wall loading is important for efficient thermodynamic working cycles, capturing and removing the fusion power in hybrid reactor technology (Teller, 1981; Ehrlich et al., 2001). The first wall material should have high power density and high power conversion efficiency for high neutron wall loading in hybrid reactor. The high power density