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Neutronic investigation of the application of certain plutonium-mixed fluids in a fusion–fission hybrid reactor

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ABSTRACT

In this study, the fluids that were investigated were contained increased mole fractions of the mixtures molten salt 99–95% $Li_{20}Sn_{80}$ and, the heavy metals 1–5% SFG-Pu, SFG-PuF_4, SFG-PuO_2. The fluids were used in the liquid first wall, blanket and shield zones of the designed hybrid reactor system. Beryllium (Be) zone with the width of 3 cm was used for the neutron multiplicity between liquid first wall and blanket. Four centimeter thick 9Cr2WVTa ferritic steel was used as the structural material.

The nuclear parameters of a fusion–fission hybrid reactor such as neutron flux, heating, fission reaction rate were investigated according to the mixture components, radial energy spectrum in the designed system. In this study, the effect of spent fuel-grade Pu content in the designed system on these nuclear parameters were calculated by using the three-dimensional Monte Carlo code MCNPX-2.7.0 and nuclear data library ENDF/B-VII.0.

1. Introduction

During their over 50 years of operation nuclear installations have been producing increasing amounts of highly radioactive waste. The waste management required as a result of fuel burning to generate energy from traditional nuclear reactors is one of the most important problems today. Transformation of wastes into stable and short-lived isotopes through nuclear reactions is a radical solution. For this reason, a hybrid reactor system, in which nucleus fusion and fission events can be operated simultaneously, was developed to obtain more energy and nuclear fuel, decrease the waste amounts in storage areas, recover transuranic elements in used fuels through reprocessing, and render fission products harmless. An important advantage of the hybrid reactor system is that it has a subcritical reactor system in which the reactor can operate securely.

In hybrid reactors, about 80% of the fusion power (D,T), 14.1 MeV, is carried with neutrons that penetrate the first wall and blanket and dissipate energy through exothermic nuclear reactions. The hybrid reactor produces 30 times more nuclear fuel per nuclear energy quantity compared with fast reactors (Şahin and Übeyli, 2005; Şahin, 2007; Günay et al., 2011; Günay, 2013a,b).

The first wall surrounding the plasma is exposed to high energy fusion neutrons, gamma ray and charged particle flux. In order to reduce these negativities, the idea that first wall surrounding the plasma should be liquid instead of solid was initially suggested by Christofilos (Christofilos, 1989; Moir, 1997). The traditional solid first-wall that surrounds the plasma is replaced by a flowing liquid wall layer. The flowing liquid wall is used as both a liquid first-wall and as a liquid second-wall. Both liquid walls confine charged particles, thus significantly reducing radiation damage in the structural materials, and conserve the energy of the neutrons by converting it into heat (Abdou et al., 2005; Abdou, 2001, 2004; Ying et al., 1999; Youssef and Abdou, 2000; Youssef et al., 2002).

In this study, a hybrid reactor was designed using fusion technology. The hybrid reactor system in this study could generate secure energy in large quantities with D–T fuel usage and subcritical study. Furthermore, it would be possible to enable the production of a self sufficient fuel for the reactor through the reaction of the neutrons released by the plasma. A substantial amount of spent fuel-grade (SFG) plutonium (Pu) from the current nuclear reactors has been stored for future use. The isotopic distribution of SFG-Pu is 2.4% ^{238}Pu, 58.5% ^{239}Pu, 24% ^{240}Pu, 11.2% ^{241}Pu, and 3.9% ^{242}Pu (IAEA, 2003; Şahin et al., 2006, 2010). SFG-Pu is valuable for its use as a fissile material, but it is dangerous when misused. Therefore, extreme care must be taken when working with spent fuel-grade plutonium methods. The transmutation of these spent fuel-grade plutonium resources, which contain a huge amount of energy, is very critical. For this reason, the radiation from this material to the environment should be prevented and reduced. With this purpose, radioactive materials with a plutonium additive were used in designed hybrid reactor system in this study to reduce the amount of spent

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fuel-grade plutonium from the current nuclear reactors, which is very dangerous in the case of unintentional misuse.

The hybrid reactor has to breed the tritium it requires. Therefore, the working liquid in the hybrid reactor system must contain a lithium compound medium to breed adequate tritium amount (Abdou et al., 2005; Jung and Abdou, 1983; Sawan and Abdou, 2006; Şahin, 2007). The main tritium breeders are natural lithium, $Li_{17}Pb_{83}$, $Li_{20}Sn_{80}$, Flibe and Flinabe. The natural lithium, an alkali metal, is one of the primary candidates due to its high tritium breeding ratio, but is very chemically active and reacts violently with air and water (Moriyama et al., 1998). $Li_{17}Pb_{83}$ and Flibe are very low tritium breeding according to natural lithium (Moriyama et al., 1995). Flinabe is an attractive tritium breeder due to its relatively low melting temperature and low vapor pressure. However, it has poor tritium breeding potential and activation of sodium in the fusion environment is a concern (Youssef et al., 2002). $Li_{20}Sn_{80}$ is recently investigated as a new candidate tritium breeders to be used in fusion reactors. $Li_{20}Sn_{80}$ has low melting temperatures and low vapor pressure. $Li_{20}Sn_{80}$ is high tritium breeding according to natural lithium (Sze et al., 1999). Although Flinabe has slightly higher Li atomic density than $Li_{20}Sn_{80}$, it contains F deteriorating neutron economy due to its high neutron absorption cross-section. Therefore, Flinabe has lower tritium breeders than that with $Li_{20}Sn_{80}$ (Yalçın et al., 2005).

In this study, $Li_{20}Sn_{80}$ was chosen as the molten salt due to sufficient tritium can be generated and nuclear heat can be transferred out of the blanket. In this study, hybrid reactor system was designed using 9Cr2WVTa ferritic steel structural material, ENDF/B-VII.0 nuclear data library and 99–95% $Li_{20}Sn_{80}$-1–5% SFG-Pu, 99–95% $Li_{20}Sn_{80}$-1–5% SFG-PuF_4, 99–95% $Li_{20}Sn_{80}$-1–5% SFG-PuO_2 as the fluids. The fluids were used to decrease the amount of spent fuel-grade Pu in the liquid first wall, blanket and shield zones of the hybrid reactor system.

Three-dimensional neutronic measurements of average neutron flux, heating, fission reaction rate was calculated according to the mixture components, radial, energy spectrum in the designed system. MCNPX-2.7.0 Monte Carlo code was used for three-dimensional neutronic measurements. The main objective of this study is to investigate the effect on the neutronic measurements of spent fuel-grade plutonium content in the liquid first wall, blanket and shield zones of the designed hybrid system.

2. Method

2.1. Geometry description

The radial structure of the hybrid reactor system is shown in Table 1. The hybrid reactor system is toroidal. The radius of the toroidal is 552 cm. The fast-flowing liquid first-wall is 2 cm thick, and the slow–flowing layer (blanket) is 50 cm thick. The beryllium (Be) is a neutron multiplier that increases the neutronic measurements by (n, 2n) reactions such as the energy multiplication, heating. Therefore, in this study, a Be zone with a thickness of 3 cm, contributing to the neutron multiplication effect by (n, 2n) reactions, was used between the liquid first wall and the blanket. A backing solid wall of 4 cm thickness and made of 9Cr2WVTa ferritic steel, which has a low activation as a structural material, follows the blanket zone. A shielding zone of 50 cm thickness (outboard) and 49 cm thickness (inboard) is located behind the backing solid wall for the outboard and inboard builds, respectively, and is assumed to have a structure-to-breeder (coolant) volume ratio of 60:40. The vacuum vessel wall is 2 cm thick and made of SS316LN stainless steel. The interior is 16 cm thick (inboard) and 26 cm thick (outboard) with the SS316LN stainless steel cooled with water by a structure-to-water ratio of 80:20 (Ying et al., 1999).

Table 1
The radial build of the hybrid reactor system design.

Inboard side		Outboard side	
Zone	r (cm)	Zone	r (cm)
SS316LN	276	Plasma	667
Vacuum vessel[a]	278	SOL	695
SS316LN	294	Liquid First Wall[d]	697
GAP	296	Be[e]	700
Shield[b]	301	Blanket[d]	750
Ferritic Steel[c]	350	Ferritic Steel[c]	754
Blanket[d]	354	Shield[b]	804
Be[e]	404	GAP	838
Liquid First Wall[d]	407	SS316LN	840
SOL	409	Vacuum vessel[a]	866
Plasma	437	SS316LN	868

[a] 80% SS316LN, 20% H_2O.
[b] 60% 9Cr2WVTa, 40% (99–95% $Li_{20}Sn_{80}$-1–5% SFG-Pu, 99–95% $Li_{20}Sn_{80}$-1–5% SFG-PuF_4, 99–95% $Li_{20}Sn_{80}$-1–5% SFG-PuO_2).
[c] 100% 9Cr2WVTa.
[d] 99–95% $Li_{20}Sn_{80}$-1–5% SFG-Pu, 99–95% $Li_{20}Sn_{80}$-1–5% SFG-PuF_4, 99–95% $Li_{20}Sn_{80}$-1–5% SFG-PuO_2.
[e] 100% Be.

2.2. Numerical calculations

The calculation of all the parameters of fission and fusion reactors, accelerator-driven systems and other areas of nuclear technology depend on cross-section data. The experimental data are limited for neutron-produced reactions in certain energy intervals. Certain cross-sections, such as fission reactions, are insufficient. There are significant differences between the cross-section and the nuclear data libraries in all energy intervals from thermal to high energies. These differences will affect all of the parameter calculations used in areas of nuclear technology.

Nuclear reaction cross-sections can be obtained in three different ways: experimental measurement, theoretical calculation and Evaluated Nuclear Data Files (ENDFs). For wide ranges of energy, measuring the cross-sections for all of the isotopes in the periodic table is infeasible both physically and economically. Therefore, model calculations play an important role in the evaluation of nuclear data (Günay et al., 2013; Günay, 2013a,b).

The evaluated nuclear data file ENDF/B was first developed in the USA in 1968. New versions were published periodically following large-scale investigations and additional research. ENDF/B-VII includes data from 10^{-11} MeV to 20 MeV for all isotopes and up to 150 MeV for certain isotopes (Chadwick et al., 2006; Pelowitz, 2011).

2.2.1. Neutron flux

It is important for neutronic calculations to know the flux and energy distribution of the neutrons in a nuclear fission reactor, fusion reactor, and hybrid reactor. The neutron flux is defined as the total path of thermal neutrons in a unit volume around point *r* in a second. The neutron flux expresses a scalar value of dimensions $m^{-2}s^{-1}$. In the nuclear reactors, fast neutrons are nearly exclusively produced in the fission reactions. In the nuclear reactors, the fast neutrons gradually slow due to collisions with the atoms of the moderator and the various nuclear processes taking place in the reactor (Duderstadt and Hamilton, 1976). As these formations are the functions of neutron energy, the neutron flux distribution should also be expressed as a function of neutron energy.

The uncertainties associated with the neutron flux distribution are neutron cross-sections (the inelastic, elastic, and capture group cross sections), neutron source distribution (source energy distributions, spatial source distribution), geometrical dimensions and material densities. These uncertainties can change according to