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The effect on radiation damage of structural material in a hybrid system by using a Monte Carlo radiation transport code

Mehtap Günay [a,*], Başar Şarer [b], Hızır Kasap [a]

[a] İnönü University, Science and Art Faculty, Physics Department, Malatya, Turkey
[b] Gazi University, Science Faculty, Physics Department, Ankara, Turkey

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ABSTRACT

In this study, the molten salt-heavy metal mixtures 99–95% Li20Sn80-1-5% SFG-Pu, 99–95% Li20Sn80-1-5% SFG-PuF4, 99-95% Li20Sn80-1-5% SFG-PuO2 were used as fluids. The fluids were used in the liquid first-wall, blanket and shield zones of the designed hybrid reactor system. 9Cr2WVTa ferritic steel with the width of 4 cm was used as the structural material. The parameters of radiation damage are proton, deuterium, tritium, He-3 and He-4 gas production rates. In this study, the effects of the selected fluid on the radiation damage, in terms of individual as well as total isotopes in the structural material, were investigated for 30 full power years (FPYs). Three-dimensional analyses were performed using the most recent version of the MCNPX-2.7.0 Monte Carlo radiation transport code and the ENDF/B-VII.0 nuclear data library.

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1. Introduction

The 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 a hybrid reactor system 14.1 MeV fusion neutrons and 3.5 MeV alpha particles are released when D-T fuel enters into the fusion reaction. The energy produced in a hybrid reactor should be as high as possible compared to the energy produced by plasma. The plasma is surrounded by a wall of fertile material. Thus, the high-energy 14.1 MeV fusion neutrons that are emitted from the plasma react with the fertile materials, resulting in fissile materials (Şahin and Übeyli, 2005; Şahin, 2007; Şarer et al., 2007; Günay et al., 2011, 2013).

APEX (Advanced Power Extraction) was developed in the USA in early 1998 to investigate fusion energy technology. In APEX, the traditional solid first-wall that surrounds the plasma is replaced by a flowing liquid-wall layer. The flowing liquid wall is used in APEX 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 and The APEX Team, 1999; Abdou et al., 1999, 2001, 2005; Abdou, 2001, 2004; Ying et al., 1999; Youssef and Abdou, 2000; Youssef et al., 2002).

Structural materials are exposed to high neutron flux, γ-rays, and energetic particles that continuously eject atoms from their lattice sites. This leads to various types of damage in the material due to neutron attenuation such as hardening, swelling and embrittlement, which directly influence its function.

Radiation damage refers to the localized disruption of the crystal lattice of a solid by high-energy radiation passing through it. The abilities to reduce the activation and radiation damage to structural materials are among the most important advantages of liquid walls, particularly the "thick" liquid wall concepts in the fusion reactor. The important damage for structural materials is a displacement of atoms in the lattice as a result of collisions with very fast neutrons. The serious damage mechanism is gas production in the metallic lattice resulting from nuclear reactions. Proton, deuterium, tritium, He-3 and He-4 gas production are radiation damage parameters. Radiation effects are the consequence of radiation damage on the mechanical and physical properties of the solid. Whereas the hydrogen atom and its isotopes produced by (n, p), (n, d), and (n, t) reactions can diffuse out of the first wall material with high temperatures, the helium atoms produced by (n, α) and (n, He-3) reactions will accumulate in the first wall of the hybrid reactor and produce helium gas bubbles. These reactions will limit the operational lifetime of the structural material of the reactor. The lifetime of the structural material, which consists of ferritic steel behind the liquid second-wall, is assumed to be the life of

* Corresponding author. Tel./fax: +90 422 377 38 78.
E-mail address: mehtap.gunay@inonu.edu.tr (M. Günay).

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the plant, which is 30 years (Duderstadt and Moses, 1982; Blink et al., 1985; Perlado et al., 1995; Ünalan, 1998).

In this study, a hybrid reactor system was designed using APEX fusion technology. 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 spent fuel-grade plutonium is 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 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 the hybrid reactor system designed in this study to reduce the amount of spent fuel-grade plutonium.

In this study, $Li_{20}Sn_{80}$ was chosen for the molten salt because it has low melting temperatures and low vapor pressure. Therefore, sufficient tritium can be generated and nuclear heat can be transferred out of the blanket. 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 value is 0.5 barns (Piera et al., 2010). This reaction, with a fast neutron spectrum, contributes to neutronic measurements, such as neutron flux, energy multiplication, and heating. Therefore, in this study, a Be zone with a thickness of 3 cm was used between the liquid first-wall and the blanket. In this study, 9Cr2WVTa ferritic steel was chosen as a structural material because it has low activation, which can help to extend the lifetime of the reactor system.

In this study was used ENDF/B-VII.0 as nuclear data library, 9Cr2WVTa ferritic steel as structural material, 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 the molten salt-heavy metal mixture as the fluids in the liquid first-wall, blanket and shield zones of the designed hybrid reactor system. The fluids were used to decrease the amount of spent fuel-grade plutonium in the hybrid reactor system. The radiation damage according to each isotopes of structural material and total in structural material was calculated using the most recent version MCNPX-2.7.0 Monte Carlo code for a full power operation period of 30 years (FPYs) in structural material of the designed system. The main objective of this study is to investigate the effect of the selected fluid on the radiation damage according to each isotopes of structural material and total in the structural material for 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 used in the study is in the shape of torus. The radius of the torus is 552 cm. The fast-flowing liquid first-wall is 2 cm thick, and the slow-flowing layer (blanket) is 50 cm thick. A Be zone with a thickness of 3 cm 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 (Şarer et al., 2009; Günay, 2013).

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).

The Monte Carlo method is generally preferred due to its success with three-dimensional complex geometry configurations of materials and physics problems using deterministic methods. Therefore, the transport equation is currently handled only by the Monte Carlo method. The MCNPX-2.7.0 transport code uses standard cross-section libraries compiled from ENDF/B-VII.0 for neutron, proton and photonuclear interactions for energies below 20 MeV. For certain materials, cross-sections are available up to 150 MeV; otherwise, nuclear models are used to calculate the missing cross-sections for this energy range. Physics models can also be used for any energy below the upper limit of the tables. Alternative data tables are available, although physics models are often poor below these limits (Şarer et al., 2012).

It is important for neutronic calculations to know the flux distribution of the neutrons in a nuclear fission reactor, fusion reactor, and hybrid reactor. The Boltzmann equation is a common way to calculate neutron flux in a reactor, which is given in below:

$$\frac{1}{v}\frac{\partial}{\partial t}\phi(r,\Omega,E,t)+\Omega.\nabla\phi(r,\Omega,E,t)+\sum_{t}(r,E,t)\phi(r,\Omega,E,t) = q(r,\Omega,E,t) \quad (1)$$