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Three-dimensional Monte Carlo calculation of gas production in structural material of APEX reactor for some evaluated data files

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ABSTRACT

Proton and He-4 gas production rates in the structural material of a fusion-fission hybrid reactor were calculated with the three-dimensional Monte Carlo method by the MCNPX-2.5.0 code. We examined these reaction rates with five nuclear data libraries: ENDF/B-VII.0 $T = 300$ K, JEFF-3.1 $T = 300$ K, JENDL-4.0 $T = 300$ K, ROSFOND $T = 300$ K and CENDL-3.1 $T = 300$ K. The production from each isotope of structural material made of ferritic steel was calculated. The neutron flux load is assumed to be 10 MW/m^2.

1. Introduction

Generating energy from traditional nuclear reactors, waste management as a result of fuel burning is one of the most important problems today. Transformation of waste to stable and short-lived isotopes with nuclear reactions is a radical solution. With this aim, a hybrid reactor system was developed where fusion and fission occur at the same time.

The fuel used in hybrid reactors is generally D–T fuel. When D–T fuel enters into fusion reaction, 14.1 MeV fusion neutrons and 3.5 MeV alpha particles are liberated (Nygren et al., 2004). In hybrid reactors fissile fuel breeding is just as important as energy production. 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 ^{238}U or ^{232}Th resulting in fissile materials ^{239}Pu or ^{233}U (Şahin and Übeyli, 2005; Şahin, 2007; Şarer et al., 2007).

$$^{238}\mathrm{U}(n,\gamma)^{239}\mathrm{U} \xrightarrow{\beta^-} {}^{239}\mathrm{Np} \xrightarrow{\beta^-} {}^{239}\mathrm{Pu}$$

$$^{232}\mathrm{Th}(n,\gamma)^{233}\mathrm{Th} \xrightarrow{\beta^-} {}^{233}\mathrm{Pa} \xrightarrow{\beta^-} {}^{233}\mathrm{U}$$

The hybrid reactor has to breed the tritium it requires (Abdou et al., 2005; Jung and Abdou, 1983; Sawan and Abdou, 2006; Şahin, 2007). Tritium is bred through the reaction of lithium with neutrons.

In fusion-fission hybrid reactors, about 80% of the energy produced by plasma, 14.1 MeV, is carried off by neutrons. Structural materials are exposed to high neutron flux, which continuously kicks atoms out of their lattice sites. This leads to various types of damage in the material, such as hardening, swelling and embrittlement, which directly influence its function.

Gas production in the metallic lattice, resulting from nuclear reactions, is one of the important damage mechanisms. While the hydrogen atom and its isotopes can diffuse out of the first wall material with high temperatures, the helium atoms will cumulate in the first wall of the hybrid reactor. All this damage will affect the strain of the first wall and restrict the operation lifetime of the reactor (Abdou et al., 2001; Blink et al., 1985; Duderstadt and Moses, 1982; Perlado et al., 1995; Ünalan, 1998; Youssef et al., 1998).

The liquid wall concept was first proposed by Christofilos in 1971 (Christofilos, 1989; Moir, 1997). The liquid wall concept proposed by Christofilos was used in APEX (Advanced Power Extraction). APEX was developed in the USA in early 1998 to investigate fusion energy technology (Abdou and The APEX Team, 1999; Abdou et al., 1999, 2001, 2005; Abdou, 2004; Şarer et al., 2007). In APEX the traditional solid first wall that surrounds the plasma is replaced by a flowing liquid layer. The flowing liquid wall is used in APEX in two ways; as a liquid first wall and as a liquid second wall. The liquid first wall is located beyond the plasma and the liquid second wall is located beyond the liquid first wall. The liquid first wall is 2 cm thick and flows at 20 m/s. The liquid second wall is 40 cm thick and flows at the slightly slower speed

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of 8 m/s. Both liquid walls confine charged particles, thus significantly reducing radiation damage in structural materials, and confine the energy of the neutrons, 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).

The aim of the APEX reactor is to enable generation of high energy with D–T fuel, provide ^{239}Pu or ^{233}U fissile materials through the reaction of ^{238}U or ^{232}Th fertile materials with high-energy neutrons, and enable tritium generation through the reaction of released neutrons with the lithium-containing wall (Günay et al., 2011).

In this study, the thickness of the blanket layer of the designed APEX was 50 cm and a liquid composed of 10% of UF_4 heavy metal and 90% of Flibe molten salt mixture was used in the liquid first wall, blanket, and shield parts. In this study, we calculated proton and He-4 gas production rates because the high-energy neutrons liberated from the plasma and from the other nuclear reactions will produce radiation damage such as proton and He-4 gas production via nuclear reactions within the structural material. The main objective of this study is to investigate the effect of ENDF libraries on proton and He-4 gas production rates in the structural material of the APEX design. In this study, we calculated these gas production rates in structural material with five nuclear data libraries currently the most recent: ENDF/B-VII.0 T = 300 K, JEFF-3.1 T = 300 K, JENDL-4.0 T = 300 K, ROSFOND T = 300 K and CENDL-3.1 T = 300 K with the help of Monte Carlo method the MCNPX-2.5.0 in three-dimensional. Analysis was performed for neutron wall loading 10 MW/m^2 and fusion power 4000 MW.

2. Numerical calculations

2.1. Geometry description

The dimensions and materials used for the APEX design in this study are given in Fig. 1. APEX 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, the blanket, is 50 cm thick. A backing solid wall of 4 cm thickness, ferritic steel, follows the liquid first wall/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 Type 316LN stainless steel, and the interior is 16 cm thick (inboard) and 26 cm thick (outboard) with Type 316LN stainless steel cooled with water by a structure-to-water ratio of 80:20 (Ying et al., 1999). Atomic fractions and atomic densities of the structural material (ferritic steel) for selected libraries are given in Table 1.

2.2. Evaluated nuclear data files and codes

The neutron transport calculations have been carried out with Monte Carlo methods. MCNP (Monte Carlo Neutral Particle) is a three-dimensional particle transport code. The MCNP code uses evaluations from the ENDF system (Chadwick et al., 2006; Şarer et al., 2009). In this study, three-dimensional analysis has been made using the NJOY99 (Oak Ridge National Laboratory, 1998) and MCNPX-2.5.0 (Hendricks et al., 2005; Pelowitz, 2005), which is the Monte Carlo code, and ENDF/B-VII.0 (Chadwick et al., 2006), JEFF-3.1 (Chadwick et al., 2006; Koning et al., 2004), JENDL-4.0 (Nakagawa et al., 1995; Shibata et al., 2002, 2011), ROSFOND (IAEA, 2010; IAEA Technical, 2010), CENDL-3.1 (Ge et al., 2011; 11th IAEA Consultants' Meeting of the Nuclear Reaction Data Centers, 1991) nuclear data libraries.

2.3. Proton and helium-4 production

Structural materials are exposed to high neutron flux, which continuously kicks atoms out of their lattice sites. This leads to various types of damage in the material, 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. Proton and He-4 gas production amounts are the parameters of radiation damage. In structural material one desires a minimum production of proton and He-4 gas, which are radiation damage parameters.

In the plasma zone, fusion source neutrons were sampled from an isotropic neutron distribution having a step function. In this study, the average 14.1 MeV fusion neutron flux at the inboard and outboard blanket surfaces is 3.99×10^{15} n cm^2/s and 4.3×10^{15} n cm^2/s, respectively, which corresponds to an average neutron wall loading of 10 MW/m^2 for ENDF/B-VII.0. The lifetime of the structural material, which consists of ferritic steel behind the blanket, is assumed to be the life of the plant, that is 30 years.

To see the impact of the liquid first wall and liquid blanket on proton and helium-4 production in the ferritic steel first wall, we compared the case no first wall and no liquid blanket with the cases with a first liquid wall and with a liquid blanket using the Monte Carlo code MCNP with ENDF/B-VII.0 evaluated data file. For the case without the first liquid wall and the liquid blanket, the proton and helium-4 production at the ferritic steel first wall

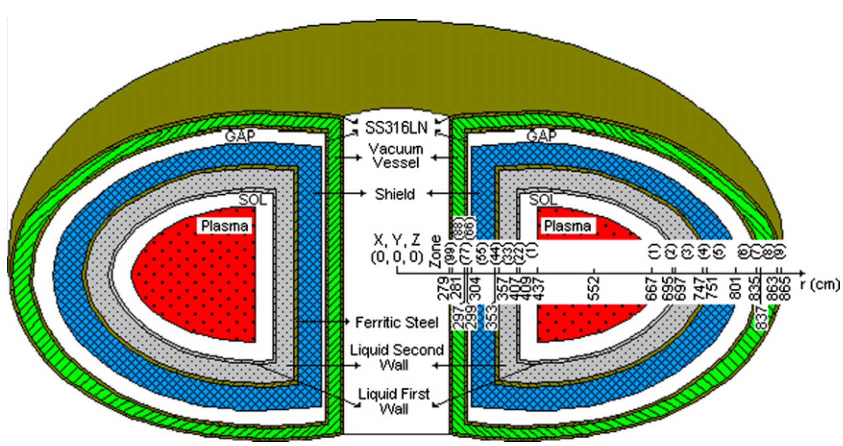

Fig. 1. The radial build of the liquid first wall/blanket concept with shielding.