

**Neutronic Calculation of Radiation Damage in First Wall of a FusionFission Reactor**

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**Abstract**

A hybrid reactor is a combination of the fusion and fission processes. The fusion plasma is surrounded with a multi-layered cylindrical blanket made of the fertile materials and coolant zone made of LiPb. In this concept, a line neutron source in a cylindrical cavity simulates the fusion plasma chamber. The latter is surrounded by a FW which various materials will be used for investigation. The (D, T) fusion neutron driver of MFE for the hybrid reactor has been carried out. The temporal neutronic performance of the hybrid blanket have been evaluated for the NWL of 2.25 MW/m2 by full reactor power (plant factor PF=100%). This corresponds to the fusion neutron flux of 1014 (14.1 MeV) n/cm2s at First Wall (FW) for conventional (D,T) driven hybrid reactor. The fissile zone is composed of typical LWRs spend fuel which contain natural uranium dioxide (UO2) in hexagonal geometry as 10 rows having pitch length = 1.25 cm in the radical direction. In this study, the nuclear heat deposition in the first wall (FW), tritium breeding ratio (TBR) in the blanket and radiation damage such as displacement per atom (DPA), He-production (n, α), H-production (n, p); (n, d); (n, t) for composed of FW made of different materials, namely, ferritic/martensitic steels, vanadium alloy, silicon carbide, copper alloy, and stainless steel as a lifetime of one full power year (FPY) have been calculated based on different data libraries. Neutronic calculations were performed by Monte Carlo Neutron-Particle Transport code MCNP5 version 1.40 in three-dimensions using three different data libraries; ENDF/B-V, ENDF/B-VI and CLAW-IV for comparing neutronic parameters.

**Keywords:** fusion, dielectrics, inertial electrostatic confinement

**INTRODUCTION**

A fusion–fission (hybrid) is a combination of the fusion and fission processes. The fusion plasma is surrounded with a multi-layered cylindrical blanket made of the fertile materials (238U or 232Th) to convert them into fissile materials (239Pu or 233U) by transmutation through the capture of the high-yield fusion neutrons and neutron multiplier and coolant zone made of LiPb. A hybrid reactor is based on either magnetic fusion energy (MFE) or inertial fusion energy (IFE). The neutron source is volumetric in MFE systems, whereas the target represents a point neutron source in IFE plants [1]. In a hybrid reactor system, a fusion breeder can produce up to 30 times more fissile fuel than a fission breeder per unit of energy [2].

In this study, the different enrichments and natural lithium were used as a moderator in order to contribute tritium breeding ratio (TBR) and the neutronic parameters were eterminate as: neutron spectrum and nuclear heat deposition in the FW, tritium breeding capability in the blanket and radiation damage such as DPA, He-production (n,a), H-production (n, p); (n, d); (n, t) for different FW materials, namely, ferritic/martensitic steels, vanadium alloy, silicon carbide, copper alloy, and stainless steel as a lifetime of 1 full power year (FPY) based on different data libraries. Neutronic analyses were performed using MCNP5 version 1.40 [15] and three different data libraries; ENDF/B-V [16], ENDF/B-VI and CLAW-IV [20].

**MODELING**

The (D, T) fusion neutron driver of MFE for the hybrid reactor were carried out. The temporal neutronic performance of the hybrid blanket have been evaluated for the NWL of 2.25 MW/m2 by full reactor power (plant factor PF=100%). Hence, this corresponds to the fusion neutron flux of 1014 (14.1 MeV) n/cm2s at FW for conventional (D,T) driven hybrid reactor. Neutronic calculations were performed by Monte Carlo Neutron-Particle Transport code MCNP5 version 1.40 in 3-dimensions using three different data libraries; ENDF/B-V, ENDF/B-VI and CLAW-IV for comparing neutronic parameters. Fig. 1 shows the basic structure of the hybrid blanket adapted from previous studies [21] to this work.

The fissile zone is composed of typical LWRs spend fuel which contain natural uranium dioxide (UO2) in hexagonal geometry as 10 rows having pitch length = 1.25 cm in the radical direction. The fuel zone is considered to be cooled with natural lithium and different enrichments lithium were used as a moderator which contributes to tritium breeding ratio, at the same time, as a working fluid for the nuclear heat transfer out of the fuel zone. The coolant to fuel volume fraction is Vc/Vf = 2. Under consideration of the volume for the fuel cladding material also, the coolant occupies a volume fraction of 62.6% in the fuel zone. The radial reflector is made of Li2O for production of tritium (T) and graphite in a sandwich structure.

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FIG. 1. Cross sectional view of the investigated blanket.

**NUMERICAL RESULTS**

**Tritium breeding ratio:** The tritium self-sufficiency is necessary for selfsustaining fusion plants operating on the DT fuel cycle. There is no practical external source of tritium for fusion reactor. All subsequent DT experimental devices and power plants have to breed artificially their own tritium. To ensure tritium self-sufficiency, the calculated achievable tritium breeding ratio (TBR) must be equal to or greater than the minimum required TBR [27]. For a self-sustaining fusion reactor, the TBR > 1.05 will be required during the operation period [22-26]. It also depends on the type of coolant containing 6Li enrichment, first-wall structure and fuel cladding materials, and neutron multiplier, as well as the dimensions and structure of the blanket used. Table I shows TBR values in the natural lithium coolant and the sandwich structure made of Li2O for different FW structure materials with ENDF/B-V, ENDF/B-VI and CLAW-IV data libraries using this study. One can see this table that TBR values are greater than the minimum requirement (TBR > 1.05) for all investigated materials.

**Radiation damage:** In a fusion-fission (hybrid) reactor, the FW will be exposed to plasma particles and lectromagnetic radiation. Moreover, the FW will be suffered from irradiation by 14 MeV neutrons for DT driven systems. The high energy neutrons will cause atomic displacement via displacement cascades and gas productions various nuclear reactions within structural materials. The radiation damage will limit life time of the FW material. Design concepts for fusion energy reactors indicate a life time of FPY for the FW structure. This means that every year first wall material must be replaced [28, 29]. Table II shows the DPA, helium and hydrogen production values at the FW for all investigated materials with NWL=2.25 MW/m2 of a convensional (D,T) driven hybrid reactor.

Table I: TBR values in the hybrid reactor blanket with differentStructural materials.

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Table II: DPA, He and H-production values in the FW for different

Structure materials.

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**Nuclear heating in the first wall:** Table III shows delta \_T (oC), difference between the aximum temperature surface (the maximum practical operating temperature limit) of the FW and the coolant (moderator) temperature for the FW material used in this study. The \_T was calculated by Fourier law for conducting wall. When the lowest \_T value was obtained 1.2oC for SiC-SiC composite due to low conductivity, the highest \_T value was calculated 160.3oC for copper alloy (Cu0.5Cr0.3Zr) due to high conductivity. In this respect, all investigated materials can be operated with 2.25 MW/m2 of NWL for a relevant temperature range.

Table III: Heating release and maximum operation temperature in the FW.

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**CONCLUSIONS**

The following conclusions can be summarized as follows:

1. In the FW, the highest DPA/FPY value 31.2 was found for V4Cr4Ti while the lowest value 27.8 was for both ODS and Cu0.5Cr0.3Zr after 1 year operation period. When a conservative limit of 100 DPA was selected for the radiation damage, the replacement of the FW structure would be required every ~ 3.2-3.6 years for all investigated materials.

2. The highest He-production value was found 679.5 appm/FPW for SiC/SiC composite, whereas low values were found 52.9 and 80.8 appm/FPW for SS316 and V4Cr4Ti, respectively. Therefore, the FW replacement every 9 mounts, 9.5 and 6.2 years will be needed for SiC/SiC composite, SS316 and V4Cr4 Ti, respectively. Furthermore, 3.8, 2.6, 2.5 and 1.9 years will be for Cu0.5Cr0.3Zr, ferritic steels, ODS and SS304, respectively. Although the DPA values were very similar, the He-production values were quite different for each other, except ferritic steels and ODS. He-productions decrease rather quietly with increasing enrichment of 6Li.

3. The FW replacement every 9 mounts and 3.5 years must be needed for SiC/SiC composite and SS316. Hence, SS316 showed the best performance with respect to radiation damage limit for all investigated materials.

4. TBR>1.05 will be required during the operating period, all calculated TBR values for investigated materials were greater than the minimum requirement (TBR>1.05). The best TBR performance was achieved with vanadium alloy (V4Cr4Ti) for structure and fuel cladding materials. While the highest TBR value was 1.177 for vanadium alloy, the lowest TBR value was 1.078 for SiC/SiC composit in this study. TBR values increase with enrichment of 6Li.

5. The highest surface heat flux 123340.73 W/m2 was obtained in the SiC/SiC composite structure, on the other hand, the lowest surface heat flux 129721.45 W/m2 was in the copper alloy (Cu0.5Cr0.3Zr) at the FW surface. Using Fourier law, while the lowest \_T value 1.2oC for SiC-SiC composite was calculated due to low conductivity, the highest \_T value 160.3oC for copper alloy (Cu0.5Cr0.3Zr) with high conductivity was obtained.

6. H-production values could not take into consideration due to all hydrogen isotopes produced by (n,p), (n.d), (n,t) reactions which will diffuse out of the metallic lattice or form metal hybrids.

**Acknowledgment:** This paper is the summarized version of my previous papers (Annals of Nuclear Energy 34 (2007) 861–870 and Energy Conversion and Management Vol. 49, pp.1960–1965, 2008).

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