Investigation of Hydrogen Isotopes Interaction with ITER and DEMO Reactors Structural Materials in the Republic of Kazakhstan

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Abstract. Study of retention and permeability of hydrogen isotopes in structural materials is one of the key problems for design and operation of thermonuclear reactors. Given article is denoted to investigations conducted within the framework of the ITER Project in Kazakhstan and to support of studies of characteristics of structural materials of DEMO reactor. The basic direction of the work is a verification of fusion reactor safety. The basic investigated materials are stainless steel, vanadium alloys (V_4Cr_4Ti, V_6Cr_5Ti), low-activated steels such as MANET or F82H, beryllium, $Li_{17}Pb_{83}$ eutectics and ceramics Li_2TiO_3 which are considered to be structural materials of thermonuclear reactors. The basic parameters of hydrogen isotope interaction with the specified structural materials have been determined by permeation and thermo-stimulated gas release techniques in the process of reactor irradiation. As a irradiation source the reactor IVG.1M of National Nuclear Center of the Republic of Kazakhstan is used. For carrying out of research work in the support of ITER Project and for personal training in Republic Kazakhstan the creation of Kazakhstan tokamak (KTM) for material studies is undergoing. At present time are elaborated the detailed design of installation, the program of research studies.

1. Introduction

Given article is denoted to investigations conducted within the framework of the ITER Project in Kazakhstan and to support of studies of characteristics of structural materials of DEMO reactor. The object of this work is to investigate hydrogen isotopes permeation and inventory in structural materials of the fist wall, divertor plates and the most prominent materials in application to tritium breeder for the further fusion devices. Tokamak KTM will allow to test the models and units in regimes, required for revision of ITER reactor parameters and future fusion reactors development.

2. The main areas of activity and results

2.1. Hydrogen isotopes permeation through stainless steel, vanadium alloys (V4Cr4Ti), low-activated steels MANET and F82H, included different type of coatings.

The experimental results on hydrogen and deuterium permeation at the temperature range 1073 – 573 K through structural materials during in-pile irradiation were represented. The experiments showed that in different materials reactor irradiation has different effect on hydrogen transportation processes. The experiments on in-pile loading of vanadium alloy specimens at the neutron flux density 10^{14} n/cm^2, different temperatures and pressure were carried out using the IVG.1M reactor of Kazakhstan National Nuclear Center. Hydrogen contents calculated from TDS spectra for both in-pile loaded and control samples are presented in [1]. The results of this experiment demonstrate that equilibrium saturation with hydrogen was not achieved. In case martensitic steel F82H, permeation of deuterium through F82H was investigated using two tips of experiment with different surface conditions: permeation rate measurements and accumulation measurements.

2.2. Tritium permeation though martensitic steel MANET from $Li_{17}Pb_{83}$ in-pile reactor experiments included different type of Al_2O_3 protected coatings.

Reactor experiments and control out of pile experiments have been carried out.

Control out of pile experiments were carried out with tubular samples of MANET steel (diameter 13 mm, wall thickness1.5 mm, effective length 100 mm), which then were used in the reactor experiments. Three types of MANET steel samples were examined: without coating, with CEA(France) coating, with FZK(Germany) coating.

Control experiments were carried out by differential mode of hydrogen permeation technique in the temperature range 673- 973 K and deuterium pressure at the upstream hydrogen pressure 100-1000 Pa. The results of experiments showed that deuterium flows through investigated FZK coated MANET steel samples are 50-60 time less than trough uncoated samples. CEA coating reduces deuterium permeation constant trough the sample about 2,5 times.

Experiments on tritium permeation from $Li_{17}Pb_{83}$ eutectics through MANET steel samples both coated and uncoated were carried out simultaneously – two tube samples were immersed into one irradiated liquid alloy of $Li_{17}Pb_{83}$ eutectics. Temperature of the samples during the experiments was 773 –973 K. Tritium release rate from eutectics (out of free surface) and tritium flows through investigated steel tube samples both bare and Al_2O_3 protected were measured. Permeation through the tube and release through open surface are strongly influenced by radiation-induced effects. Therefore, the real permeation rate and the thermally activated rate of release from eutectics are much less than rates observed experimentally.

The in-pile experiments showed that:

- Part of tritium, passed through the sample of MANET steel without coating in respect to total amount of tritium generated in the eutectic volume, is about 2%
- Tritium flow through the MANET steel sample with FZK coating was not registered, because it was lower that the sensitivity of measuring device (less than 10^{-13} mole/s).
- CEA coating reduces tritium penetration through the eutectic about 1.5-2 times.

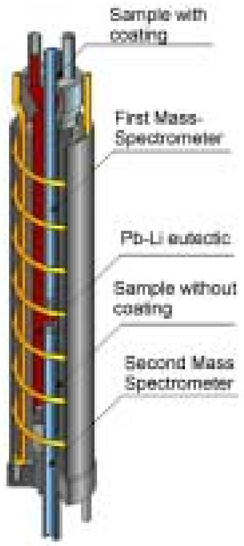

Fig.1. Layout of diffusion cell with two tubular samples

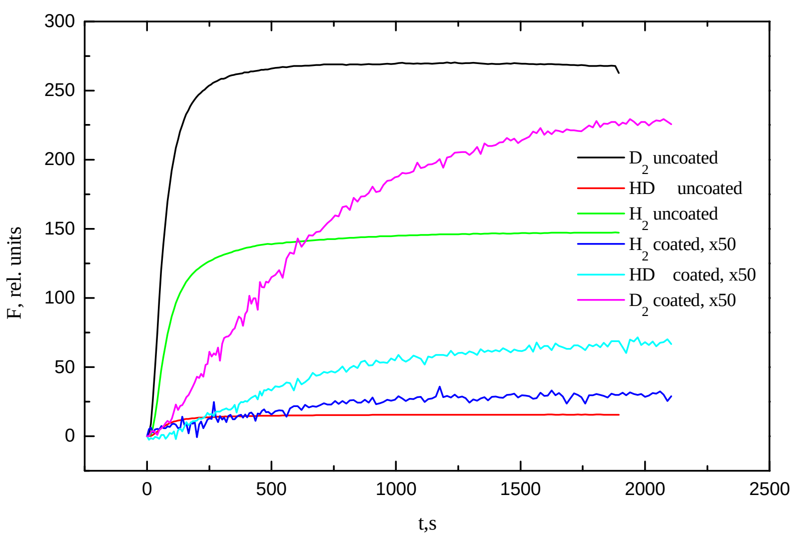

Fig.2. Typical kinetic curves of hydrogen permeation for 3 masses (T=773K) during control experiments with and without FZK coaling.

No degradation of the protection layer was observed during a few in-pile experiments. This work were in the frames of ISTC project #K-039. The diffusion cell with PbLi eutectic and results of experiments are show in Fig. 1 and 2.

Also we carrying out the in-pile experiments on tritium permeation through Cr18Ni10Ti stainless steel tube and martensitic steel F82H from $Li_{17}Pb_{83}$ eutectics at temperature range 753-993 K. Tritium release rate and tritium flow through tubular samples of these steels are determined.

2.3. In-pile study experiments on lithium containing ceramics.

The goal of the experiment - to carry out radiation testing of 95% enriched with ^6Li isotope lithium ceramics(Li_2TiO_3) up to the burn-up level 15-20%, at the IVG.1M and WWR-K reactors with in situ registration of released tritium and helium. At present time the $^6Li(n\alpha)^3T$ reaction rates have been calculated for suggested versions of irradiation ampoule designs. Two types of ampoules have been considered:

- active ampoules, to be used for irradiation of two grams of the 1-mm diameter ceramic pebbles of two compositions (Li_2TiO_3 and Li_2TiO_3+TiO_2), enriched by the Li^6 isotope up to ~95%, with continuous in situ measurement of the generated tritium;
- passive ampoules, to be used for allocation of the same two grams of pebbles and 10 pellets of the same ceramic composition, having the 8-mm diameter and 1-mm thickness; without tritium monitoring.

As a result, the following outcomes have been obtained.

- Heat generation in lithium ceramics comprises: (115±11) $W \cdot g^{-1}$ – for pebbles, and (95±10) $W \cdot g^{-1}$ - for pellets.
- Average lithium burn up in the ceramics for 200 days of reactor operation at nominal power, for the computational model applied comprises: from 22.7% to 24.9% - for pebbles; from 18.1% to 20.1% – for pellets.

The experiments were carried at IVG.1M reactor on lithium ceramics energy-release (2mm pebbles with ^6Li of natural enrichment) and verification of tritium registration system.

The coolant was supplied to ampoule before the experiment beginning (the temperature of ampoule device surface never exceed 300 K). Four stages of reactor power were passed sequentially - 1, 2, 3 and 6 MW. At each stage the temperature of container was measured and mass-spectrum registration of ampoule gases was provided. After the reactor had reached needed power lever we measured the temperature of container and tritium release from container with ceramics till they reached stable values. Then the reactor power was lowered to minimal level. Next power stage was reached after container was cooled down below 400 K. When, in such a way, the reactor had reached the final power level of 6 MW, it stayed at that level during 5 hours till the shut down. Total dynamics of temperature changes and tritium release is shown in Fig. 3. During the experiment we did not detected any helium flows from ceramics. Effect of sharp increase of tritium flow was registered while cooling down containers at temperature 1123 K. It is shown in Fig. 4.

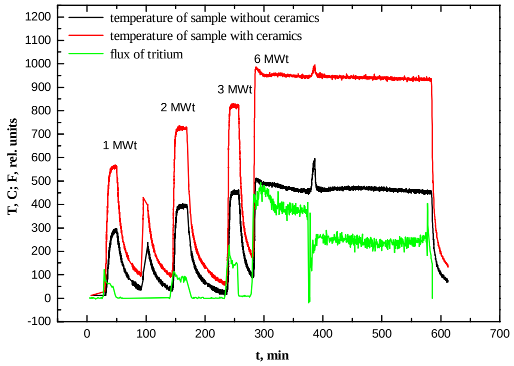

Fig. 3. Dynamics of tritium release and temperatures in a system during the whole experiment.

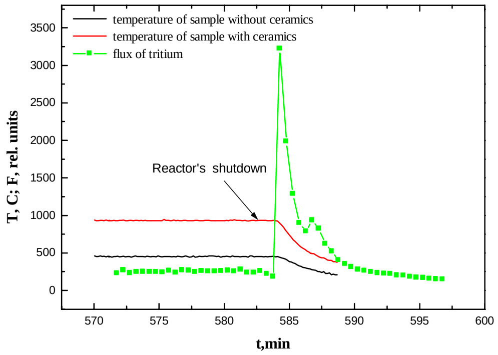

Fig. 4. Effect of tritium flux increase at reactor shutdown from 6MW power level. (Ceramics temperature at that moment was 850℃)

2.4. Retention of hydrogen isotopes in re-sputtered layers of candidate material for fusion.

Presented results of experiments on the determination of gas release from samples of sputtered beryllium, graphite, tungsten, graphite and tungsten sputtered simultaneously, produced by different methods (plasma booster "OSPA", TRINITI and installation "Argamak", INP NNC RK). The curves of gas release under different heating rates in interval of temperatures from 300 to 1200 K are registered and parameters of gas release and retention of hydrogen isotopes in given materials are determined. Modeling of hydrogen isotopes release from the samples of co-deposited layers of these materials allowed to determine hydrogen (deuterium) diffusion coefficients and energy of release from traps. Results of experiments are shown in Table 1.

TABLE 1. PARAMETERS OF HYDROGEN ISOTOPE RETENTION AND DIFFUSION IN CODEPOSITED LAYERS

Material	Hydrogen isotope retention	Diffusion coefficient, m^2/s	Energy of release from traps, *kJ/mole*
Beryllium	$H_2 – 3.8\times10^3$	$7\times10^{-9}exp(-30/RT)$	270
Beryllium oxide		$1\times10^{-9}exp(-165/RT)$	
Graphite	$H_2 – 2.5\times10^4$ $D_2 – 1.9\times10^4$	$2\times10^{-4}exp(-250/RT)$	350
Tungsten	$H_2 – 2.3\times10^4$ $D_2 – 2.9\times10^4$	$6\times10^{-3}exp(-107/RT)$	272
Codeposited graphite and tungsten	$H_2 – 4.2\times10^4$ $D_2 – 6.6\times10^4$		

2.5. Kazakhstan tokamak (KTM) for material studies

For carrying out of research work in the support of ITER Project and for personal training in Republic Kazakhstan the creation of Kazakhstan tokamak (KTM) for material studies is undergoing. The main purpose of the KTM tokamak is: scientific and technological support of the ITER project and future fusion reactors development, experimental studies of the first wall and divertor materials under the ITER-like load conditions, tests of the invessel components and divertor elements, studies of various methods of reducing heat loads on the divertor plates.

The main tasks of the KTM tokamak for divertor studying: developed a new candidate materials for the divertor plates, erosion, sputtering and arching processes the divertor plates, dynamics of the surface and volume temperature divertor plate distribution, methods of reduction of local heat loads on divertor plates. Investigation for first wall and in-vessel elements protection: development of the first wall materials (recycling, sputtering, erosion, thermal stability), influence of the protection materials on $Z_{eff,}$ porous materials for reducing heat loads on the walls and decreasing the main plasma contamination.

Basic ideology of the KTM tokamak chosen on aspect ratio A=2 and elongation K=1.7 will allow: to provide 4-5 s nominal current duration 0.75 MA by central solenoid, to obtain stable divertor configuration with minimum requirements to passive and active stabilization systems, to use compact vacuum chamber and electromagnetic system, to reach the necessary power of plasma flux into the divertor at moderate additional RF heating power.

Unique to this facility will be the presence of sluice-ways for ports in limiter and divertor areas necessary for input of targets and diagnostics. Appearance 3D-view of the KTM, is shown in Fig. 5

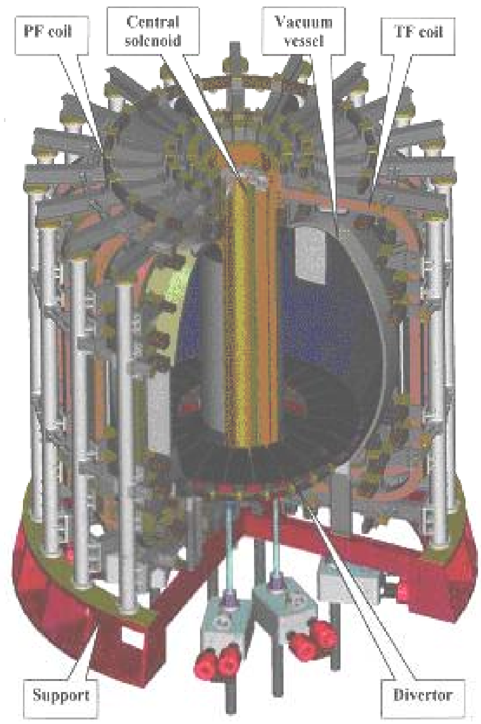

Fig 5. 3D view of the KTM tokamak

Design of KTM is conducted by common efforts of Russian and Kazakhstan scientists and specialists. At present time are elaborated the detailed design of installation, the program of research studies. Activity on development of technological processes for fabrication and prototyping of responsible elements and details is initiated and technical documentation is issued.

The main results of our investigation are represented in [2-4].

At this moment the talks are conducted about the further participation of Republic Kazakhstan in the ITER Project in partnership with Russia and about co-ordinations of research activity. There are plans of Kazakhstan participation in fabrication of high-tech elements of thermal protection of beryllium and components of superconductors for magnetic system, as well as scientifically- technical support of these work.

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