Investigations in the Area of Thermonuclear Structural Material in the Republic of Kazakhstan

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Abstract

The investigations in the area of structural materials for fusion program initiated within the framework of ITER project in the Republic of Kazakhstan are devoted basically in the following direction: to studying the behaviour of hydrogen isotopes in structural elements of the first wall and the divertor in conditions simulating real conditions of material operation, accident situations arising during steam interaction with the beryllium armour of the first wall during accidental coolant loss, to establish an experimental facility for study aspects of tritium safety of thermonuclear installations, for example, levels of tritium accumulation and release; efficiency of barrier layers and protective coating; influence of brazing and welding zones on tritium permeation. The work on determination of tritium release from lead/lithium eutectic alloy by mass- spectrometry method and the development of permeation barriers has begun. At present, work has begun to create Kazakhstan's own tokamak type reactor for investigation of the behaviour of various first wall materials and divertor plates during normal and accident conditions. The concept of spherical tokamak will be used in the construction of KTM reactor.

1. Introduction

The investigations in the area of structural materials for fusion program initiated within the framework of ITER project in the Republic of Kazakhstan are devoted basically to studying the behavior of hydrogen isotopes in structural elements of the first wall and the divertor in conditions simulating real conditions of material operation and also accident situations arising during steam interaction with the beryllium armour of the first wall during accidental coolant loss. The basic content of the work is a verification of ITER reactor safety

2. The following works are carried out and results are obtained

The basic investigated materials are beryllium, stainless steel, graphite, tungsten, duplex and three-layered structures Be/Cu and Be/Cu/stainless steel, vanadium alloys (V4Cr4Ti, V6Cr5Ti, etc.), low-activated steels such as MANET or F82H, 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. The main results of our investigation are represented in /1 – 5/.

1. The technology is tested and the samples of duplex structure (Be-Cu) and triplex structure (Be-Cu-SS) are manufactured for ITER program.

2. The works on investigation of hydrogen isotopes permeation through structural materials are carried out. Investigations of hydrogen permeation were carried out for the samples of Cr18Ni10Ti steel, nickel, vanadium alloy V-4Cr-4Ti, copper alloy CuCRZr01 in IVG.1M reactor of the Institute of Atomic Energy (National Nuclear Center of Kazakhstan) at the fast neutron flux 10^{13} n/cm^{2}s, thermal neutron flux 10^{14} n/cm^{2}s, in the temperature range 523-1023K and input hydrogen pressures from 10^{-2} - 10^{6} Pa. It is shown that reactor

irradiation stimulates the process of hydrogen permeation through the materials, specified above. In case of stainless steel and nickel the influence of reactor irradiation results in the increase of apparent diffusion coefficients and, less significantly, of permeation constant. In case of copper alloy we observe inverse effect – permeation constant increases, but diffusion coefficient is not changed. For the vanadium alloy reactor irradiation decrease of permeation constants and apparent diffusion coefficients. For all the cases the effect of irradiation influence increases as the temperature decreases, hydrogen input pressure and intensity of irradiation increase. Postirradiation effects are observed – the values of diffusion parameters do not return to the initial values after stopping of irradiation.

3. Using the technique of thermodesorption in a linear temperature ramp rate mode the investigations of hydrogen isotopes retention and release from the samples of beryllium of different grades (EHP-56, HP-56, IP-56), RG-Ti graphite and vanadium alloy V-4Cr-4Ti, irradiated in hydrogen and deuterium atmosphere in IVG.1M and RA reactors up to the fluence $5*10^{18}$ n/cm^2 at different temperatures and pressures are carried out. Comparative analysis of hydrogen isotopes release from the samples of irradiated beryllium of different grades without preliminary treatment revealed that hydrogen isotopes retention depends on the parameters of irradiation, production technique and grain size. Minimal gas release was observed for the samples of hot pressed beryllium HP-56 with lowest content of oxygen. Maximal gas release was observed for the samples of extruded hot pressed beryllium, that may be explained by high level of deformation and high oxygen content. It is shown that hydrogen inventory for the samples, irradiated in RA reactor is one order of magnitude less than for the samples, irradiated in IVG.1M reactor, where the flux of thermal neutrons is two times higher. Deuterium gas release is the same for the samples of all the investigated beryllium grades, preliminary annealed at elevated temperatures.
   It is shown that the increase of the rate of thermally stimulated hydrogen release from irradiated samples of graphite, beryllium and vanadium alloy is observed only for the samples, irradiated in the process of irradiation, and this effect increases with the increase of reactor irradiation intensity and temperature. The effect of influence of low fluence reveals in the increasing of catalytic activity of the surface of graphite, beryllium and vanadium alloy, decay of hydrogen complexes and compositions, increasing of hydrogen capture and release rates from different types of defects at the surface and near-surface layer.

4. The works on investigation of beryllium for first wall armour and steam interaction are carrying out to verify computer codes for simulation of Loss Of Coolant Accident in ITER reactor.

During a loss of coolant accident it is necessary to meet the requirements of restriction of intensity of steam interaction with first wall beryllium armour facing the plasma and thus restricts the amount of hydrogen formed during an exothermal reaction of beryllium and steam. Therefore, the work on verification of computer codes for accident simulation of coolant loss in the ITER reactor were conducted. Experiments for measuring beryllium emissivity in a wide temperature range have been performed. The experiments to obtain data for verification of calculation codes describing accident situations involving water coolant release into the vacuum chamber of ITER reactor were performed at Institute of Atomic Energy of the Kazakhstan National Nuclear Center. Samples of beryllium were manufactured at "Ulba" beryllium plant (Kazakhstan). Experimental samples of the beryllium DB-56 with a purity of 98.79% were used.

Experiments conditions were as follows:

Atmosphere in the chamber ....steam;
Pressure in the chamber ....up to 30 torr;
Chamber wall temperature... about 470 K.
Initial temperature of beryllium surface .... 670K …1370 K;
Pressure of steam in the chamber ....0.1- 0.2 MPa;
Steam temperature..about 470 K;
Steam supply time..about20 - 60 s.

The following results are obtained:

- The changes in the emissivity of beryllium sample surface in the result of short term exposure (10 min) at the temperature 1280 K and air pressure 10^3 Pa are not found
- Self-sustaining steam/beryllium reaction in the investigated range of the parameters is not registered,
- The coincidence of emissivity dependencies on temperature for the samples with central opening, oxidized in steam atmosphere at different pressures and the same temperature.

5. Within the framework of the international cooperation project with Russian, American and Italian partners financed by ISTC, there is a plan to develop and to establish an experimental facility for material science investigations. This facility would be located on the territory of the former test site at the National Nuclear Center in order to study aspects of tritium safety of thermonuclear installations. In particular:
   - levels of tritium accumulation and release;
   - efficiency of barrier layers and protective coating;
   - influence of brazing and welding zones on tritium permeation;
   - effects connected with a combined irradiation by fluxes of alpha-particles, gamma-photons and neutrons.

The project of the DEMO thermonuclear reactor foresees the use of a liquid metal Pb+17%Li blanket for tritium production. The main obstacle facing the designers of the blanket, is the prevention of tritium release into the coolant through the blanket wall. One of materials suggested as a blanket wall structural material is martensite steel.

Recently, work on determination of tritium release from lead/lithium eutectic alloy and the development of permeation barriers has begun**.** In the first stage, the design of a diffusion cell was developed and tested. The test results have proven the validity of the design, and the mass-spectrometer measurements have measured tritium and HT permeation in the outlet chamber of the diffusion cell. Simultaneously, the pressure above the eutectic alloy has been measured. We are preparing for the testing of diffusion cell in which it is possible to measure tritium permeation through the materials both with a coating and without it. Assessment and experimental testing of produced tritium in Pb-Li eutectic during irradiation in IVG.1M reactor is carried out. Tritium permeation from PbLi eutectic through the tubular stainless steel sample is measured and tritium permeation constants are calculated. It is obtained that tritium flux at T=773 K is equal to $2.25 \cdot 10^{-11}$ mole/m^{2}s, at T=1373 K is equal to $1.35 \cdot 10^{-8}$ mole/m^{2}s. Fig 1. shows the temperature change along the diffusion cell, quantity of tritium and hydrogen inside the input chamber, and also the gas pressure over eutectic during and after in-pile experiment. Fig 2 shows the kinetics of partial pressures H_2, HD, HT and T_2 in output chamber in the course of in-pile experiment. Calculated fluxes of tritium through stainless steel are comparable with the fluxes from gas phase at input tritium pressure 10^{-4} Pa. Activation energy of tritium permeation is estimated from the results of after in-pile experiments and is equal to 0.58 eV.

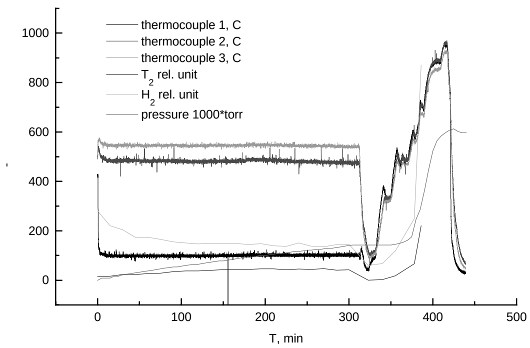

FIG. 1. The temperature change along the diffusion cell, quantity of tritium and hydrogen inside the input chamber, and also the gas pressure over eutectic during and after in-pile experiment.

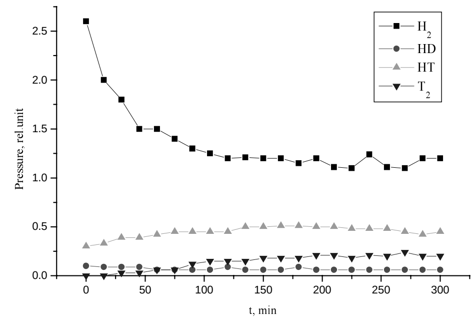

FIG. 2. The kinetics of partial pressures H_2, HD, HT and T_2 in output chamber in the course of in-pile experiment.

During project implementation, models of tritium transport in the system: structural material/liquid metal coolant (Pb17Li) and principles of gamma-ray nuclear-physical diagnostics of thermonuclear plasma will be developed.

The study of tritium behaviour during irradiation is important not only for the analysis of radiation safety but also for an economic evaluation of simulations and subsequent live tests taking into consideration the high cost of tritium. Therefore, it is important to know levels of tritium release, consumption and also probable expenses for disposal of radioactive waste contaminated with tritium and the economic efficiency of fusion reactors.

6. Taking part in the project of the international ITER thermonuclear reactor creation, the Kazakhstan scientists had no opportunity to use their own facilities. At present, work has begun to create Kazakhstan's own tokamak type reactor for investigation of the behaviour of various first wall materials and divertor plates during normal and accident conditions. The concept of spherical tokamak will be used in the construction of KTM reactor. This will allow the achievement of higher plasma parameters and testing of stressed elements of the first wall and divertor at the loads which are similar to probable loads in experimental thermonuclear reactors (5-10МВ/м^2). Even with this, it is possible for the basic facility to remain compact and cost-effective. The design of this reactor follows this basic configuration: mega-ampere spherical tokamak with low aspect ratio and the plasma confinement time of approximately five seconds and parameters of plasma sufficient for providing investigations in the area of thermonuclear structural materials as a support of ITER Project. 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 of KTM facility is shown in Fig. 3.

KTM will be the only mega-ampere facility in the world with aspect ratio A=2, that will be able to expand database on the physics of plasma confinement in the intermediate range between spherical and classic tokamaks. Technical and economical verification of site selection, draft project and its review are recently completed. The work is carrying out in close cooperation between Kazakhstan organizations (National Nuclear Center, Kazakh National University, KATEP, KazNIPIEnergoProm) and Russian ones (Russian Science

Center "Kurchatov Institute, TRINITI, Efremov Institute NIIEFA, Leningradsky Severny Zavod). The start of the facility manufacturing is planned for the next year.

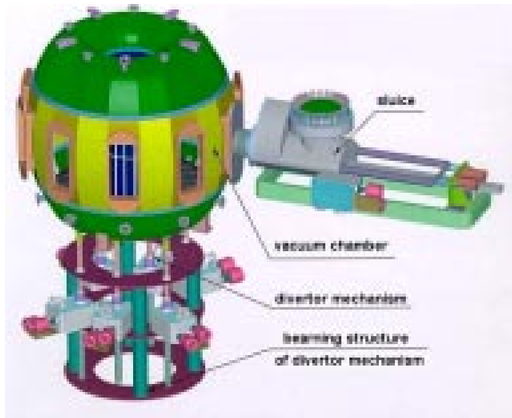

FIG. 3. Appearance of KTM facility.

The construction of the structural material research reactor in the Republic of Kazakhstan will allow for more fruitful and successful participation of the Kazakhstan scientists in the structural materials program of works at the ITER and DEMO reactors.

3. Plans and prospective

For near 2-3 years it is planned to continue investigations to verify the operational capacity of ITER and DEMO structural materials in hydrogen isotopes atmosphere, including the conditions of reactor irradiation and presence of protective coatings, and also to continue works aimed to verification of computer codes for simulation of LOCA accidents in ITER reactor. New activity for DEMO project – irradiation tests of lithium ceramics Li_2TiO_3, enriched by ^{6}Li isotope, with 20% of Li burnup level and registration of releasing tritium "in situ" at different temperatures with the following post-irradiation investigations of physical and chemical properties.

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