

# USING OF CHARGE PARTICLE ACCELERATORS FOR INVESTIGATION OF PHYSICAL MECHANISMS RADIATION RESISTANCE OF FISSION AND FUSION STRUCTURAL MATERIALS

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Radiation resistance of fission and fusion structural materials is determined by such physical phenomena as radiation swelling and creep, radiation hardening, helium and hydrogen embrittlement. All these physical phenomena depend on microstructure changes in irradiated structural materials. The key problem for prolongation of life time of fission atomic reactors and for fusion reactors is to obtain some experimental data about the microstructure changes in structural materials changes at different temperatures at high doses of neutron irradiation and at high level produce displaced atoms. The experimental tests of structural materials in atomic reactors are very expensive and they take long time. The irradiation of these materials by charge fast particle beams on accelerates allow to get such experimental data for physical mechanisms of radiation resistance of them more easy during short time without high radioactivity level on irradiated samples. The some results concerning the investigation of effect of charged particle irradiation on materials are presented here.

## 1. INTRODUCTION

The complex studies of radiation damage in materials and subsequent selection of the most radiation resistant materials for the fission and fusion reactors (ITER, DEMO) require the studies of radiation-induced physical mechanisms of their physical phenomena that determine their properties under irradiation. The radiation resistance of structural materials in fission and fusion reactors is limited by such physical phenomena as radiation swelling (material volume growth under irradiation), irradiation creep (enhanced plastic deformation of stressed materials under irradiation), radiation induced phase instability, as well as radiation hardening and irradiation embrittlement, manifested in the materials plasticity degradation and crack formation due to irradiation.

The irradiation of materials on the fast particle accelerates by light and heavy ions allow investigating the some physical mechanisms of these phenomena, which determine the radiation resistance of these materials.

## 2. MAIN PROBLEMS AND TASKS IN THE INVESTIGATIONS OF RADIATION RESISTANCE OF FISSION AND FUSION STRUCTURAL MATERIALS USING CHARGE PARTICLE ACCELERATORS

The fast particle irradiation of different types of materials results in considerably changes of their physical and mechanical properties. All physical phenomena that determine the radiation resistance of structural fission and fusion materials are related with the formation of primary radiation defects (vacancies and interstitials) in atomic collision cascades created as a result of fast particle scattering on material atoms, as well as to the kinetics of nucleation and growth of defect clusters in the form of voids, gas bubbles, dislocation loops and dislocation movement. The kinetics of damage accumulation in fusion materials is substantially influenced by the 14-MeV neutrons that create large collision cascades and result in the gas (hydrogen and helium) atom accumulation in the material due to the (n,p) and (n, $\alpha$ ) nuclear reactions. The structural materials proposed for the first wall

of a fusion reactor are metallic alloys, ceramic materials where neutron irradiation stimulates radiation swelling and degradation of mechanical properties, precipitation of new phases and segregation of various elements along grain boundaries and dislocations. In turn the phase instability and point defect clusters affect noticeably on the variation of physical and mechanical properties of materials.

The theoretical investigations of the kinetics of defect cluster accumulation and defect structure evolution under light and swift heavy ion irradiation are very important and necessary also for the adequate understanding and interpretation of simulation experiments using the light and heavy ion irradiation widely performed on charged particle accelerators.

The production of new advanced materials (ceramic materials, high temperature superconductor materials, semiconductor materials based on silicon) for future new technologies and many applications (space, microelectronics and at al.) is connected with the understanding of the physical mechanisms of the effect of fast particle irradiation on the physical and mechanical properties too. The investigations of an effect of fast heavy ion irradiation on semiconductor materials are very important for the development of the space and microelectronics technologies. Two kinds of interactions between incident heavy ions and these materials are known as the elastic collisions and the electronic excitations. Many experimental results show that damage production is realized not only by the elastic collisions but also some times mostly due to the electronic excitation effects. The physical mechanisms of damage production including radiation annealing processes due to electronic stopping power of fast heavy ions still are not understandable. So for the using advanced materials in such new technologies it is important to understand the physical mechanisms of defect production and especially the mechanisms of energy transfer from fast heavy particles to materials.

The increasing requirements for the improving of safety of modern atomic reactors and extension of the fuel's life leads to a need a better understanding of the physical

mechanisms of phenomena which limit the safety work of atomic reactors due to dimensional instability of structural materials under neutron irradiation. The more important problems for them are the irradiation creep, radiation swelling, radiation hardening and embrittlement. The phenomenon of irradiation creep arises in stressed structural materials under neutron irradiation. This phenomenon is determined by radiation induced plastic deformation in structural materials which is increased with the increasing of irradiation dose. The radiation creep characterizes by the strain rate and it depends on many parameters: initial microstructure, chemical composition of alloys, irradiation temperature, cascade efficiency, generation rate of point defects, and accumulation of defect clusters (dislocation loops, voids and precipitates) under neutron irradiation. The investigations of influence of these parameters on the irradiation creep and swelling are very important for the understanding of physical mechanisms of these phenomena.

The other main important topic is the investigation of radiation-induced processes in structural ceramic and metallic materials for fusion reactor (ITER, DEMO) including the development of new theoretical models for the calculation of primary point defect and defect cluster formation in ceramic materials under heavy ion irradiation for future advanced technologies. Recent studies have proved that electronic excitation play an important role for the aggregation and recombination processes of point defects in oxide ceramics which are applied to nuclear industries and fusion reactor ITER. For example, retardation of dislocation loop formation has been reported in such ceramic oxides if there is existing high ionizing radiation with production of new displacement damage. In contrast, dislocation loop formation has been observed in spinel irradiated with very little displacement damage ( $10^{-5}$ dpa) under very high-density electronic excitation ( $\sim 600$  MeV Xe ions). The influence of the electronic excitation, which takes place also under 14 MeV neutron irradiation on the formation process of defects in oxides is not fully understood at this moment. So using accelerators can help to understand many

physical mechanisms of radiation resistance in fission and fusion structural materials.

### 3. INVESTIGATION OF RADIATION RESISTANCE OF GRAPHITE MATERIALS FOR FISSION AND FUSION REACTORS

The investigations of physical-mechanical property changes of graphite materials for RBMK atomic and fusion (ITER, DEMO) reactors have been performed on cyclotron of RRC-KI using proton beam with the energy up to 30 MeV and carbon ions with the energy up to 50 MeV. Accomplished analysis includes comparative investigations of physical-mechanical properties of irradiated and non irradiated graphite materials, including analysis of degradation of such main physical values of these materials: coefficient of thermal conductivity, changes of electrical resistivity, thermal expansion coefficient, changes of density, changes of mechanical properties of these materials: strength analysis of elastic modulus changes, stress to rupture, yield stress, ultimate tensile, deformation to

rupture. Such experimental tests were performed using the 30 MeV protons when the penetration depth of them is higher the thickness of irradiated samples. These irradiation tests have been made at different temperatures up to maximum irradiation dose  $10^{19}$  proton/cm<sup>2</sup>. The dose dependencies for the main physical-mechanical properties of different types of irradiated nuclear graphite materials have been obtained. The numerical calculations of radiation damage accumulation in the graphite materials under proton irradiation in the energy interval from 1 MeV up to 35 MeV and up to irradiation doses  $10^{19}$  proton/cm<sup>2</sup> allow comparing degradation of physical mechanical properties under ion irradiation with neutron irradiation.

The carbon beams with the energies up to 50 MeV were used for the investigation of radiation swelling behaviour at high dpa doses and at different temperatures in nuclear graphite materials. The scheme of irradiation on RRC KI cyclotron by carbon ions of graphite samples is shown in Fig.1.

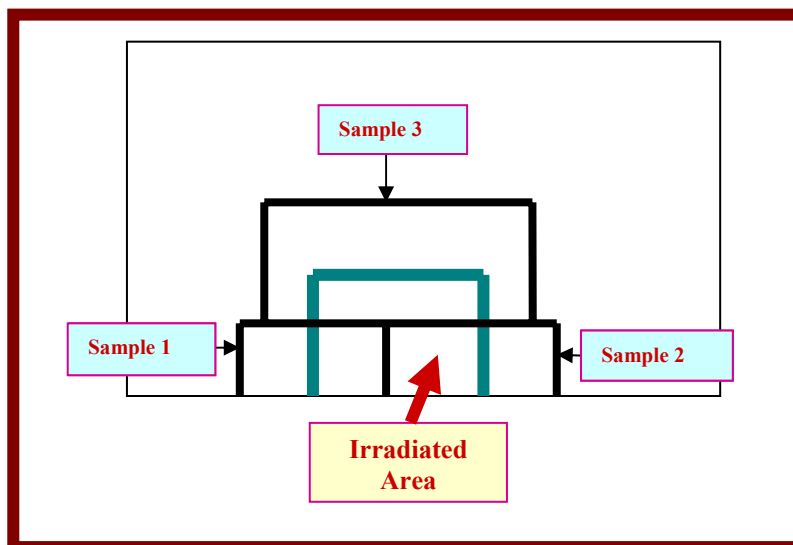


Fig.1. Scheme of irradiation on RRC KI cyclotron of graphite samples by carbon ions

The analysis of radiation swelling behaviour in irradiated graphite materials were performed using profilometry measurements of step height between irradiated and non irradiated areas (see Fig.1) as was suggested early [1]. The typical dose dependence of radiation swelling reactor two types of

graphite: WPG and WPG-KP obtained after carbon ion irradiation with the energy  $E_C = 3$  MeV at temperature  $T=450$  °C is shown in Fig. 2. It should be remarked that 1 dpa is accumulated in atomic reactor at neutron dose  $\Phi = 10^{21}$  n/cm<sup>2</sup>.

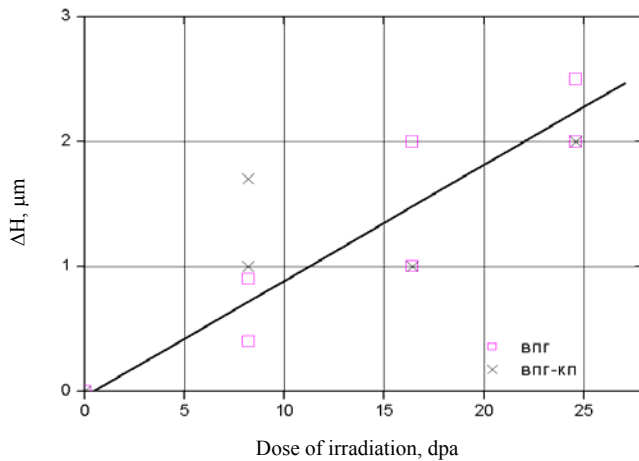


Fig.2 Dose dependence of radiation swelling reactor graphite: BИИГ (WPG) and BИИГ-КП (WPG-KP) obtained after carbon ion irradiation with the energy  $E_C = 3$  MeV at temperature  $T=450$  °C

#### 4. INFLUENCE OF RADIATION DAMAGE OBTAINED UNDER FAST CHARGED PARTICLE IRRADIATION ON PLASMA-FACING EROSION OF FUSION STRUCTURAL MATERIALS

Plasma-facing materials (PFM's) in a fusion reactor will be subjected to high heat and fast particle irradiation. Fusion neutrons can produce in materials high level of radiation damage (up to 100 dpa). Usually plasma erosion of PFM's is studied at non-irradiated samples. Here we suggest a new experimental procedure and present first experimental results concerning the effect of high-level radiation damage in graphite materials on erosion of them under stationary plasma flux. The experimental procedure is performed in two stages. In the first stage the material samples are irradiated by energetic ions to obtain radiation damage. On the second stage, these irradiated samples were exposed to the steady-state plasma stream to induce erosion of material. For the production of high-level of radiation damage we irradiated graphite materials on cyclotron in RRC KI by the carbon ions with the energy of 5 MeV. The irradiation conditions allow to produce in graphite materials the radiation damage up to tens dpa in a few days, that is equivalent to dose of fast 14 MeV neutrons near  $10^{26}$  neutron/m<sup>2</sup>. Plasma exposure of irradiated and non-irradiated materials is

performed on the linear plasma facility LENTA in RRC KI. The following graphite materials were studied: pyrolytic graphite (as a reference grade), MPG-8 graphite (Russian grade) and CFC SEP NB (ITER candidate material). The results of the surface modification of irradiated graphite materials due to erosion processes in deuterium plasma in the dependence on radiation damage level are presented here.

Carbon ions  $^{12}\text{C}^+$  accelerated on the cyclotron to 5 MeV were taken to produce radiation damage in samples of three grades of graphite: SEP NB-31, MPG-8 and pyrographite [2]. Three levels of radiation damage 1 dpa, 5 dpa and 10 dpa in average over the damaged layer were obtained in the samples of each carbon material. The  $^{12}\text{C}^+$  ion doses necessary to reach these values were  $10^{17}$  ion/cm<sup>2</sup>,  $5 \cdot 10^{17}$  ion/cm<sup>2</sup> and  $10^{18}$  ion/cm<sup>2</sup> correspondingly. Distribution of the accumulated defects in the surface layer is given in Fig.3 where calculated curve of primary radiation defects produced in graphite by 5 MeV  $^{12}\text{C}^+$  ions as a function of depth is shown (SRIM program [3]).

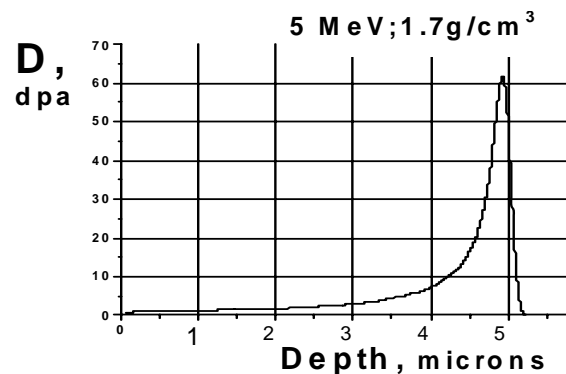


Fig.3. Calculated profile of the primary radiation defects  $D$  (dpa) as a function of depth for a carbon material with density  $\rho=1.7$  g/cm<sup>3</sup> after irradiation with 5 MeV carbon ions to the dose  $10^{18}$  ion/cm<sup>2</sup> ( $D_{max}=65$  dpa lies at  $\sim 5$  μm,  $\langle D \rangle=9.7$  dpa)

Sputtering effect and radiation-induced deformation (swelling) took place on the materials along with the radiation damage produced by the ions. Modification of surface structure of the SEP NB-31 sample is shown in Fig.4 where the boundary between irradiated and non-irradiated areas is presented for the 10 dpa sample. The development of the surface is seen on the irradiated part (left).

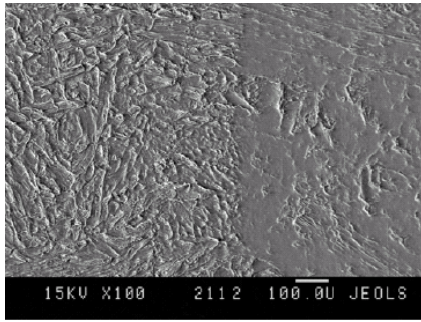
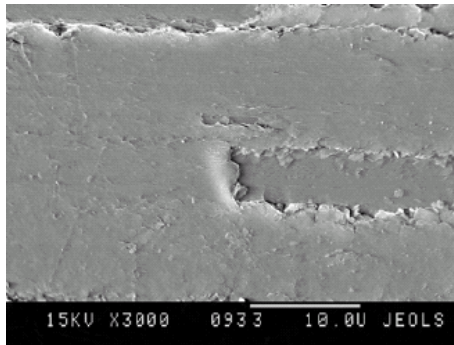


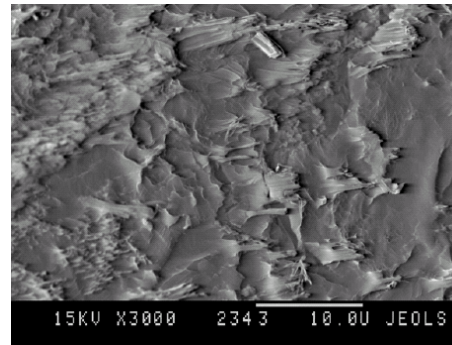
Fig.4. Border of irradiated area on SEP NB-31 surface (10 dpa): damage zone is to the left (Scale: 100  $\mu\text{m}$ )

The graphite samples irradiated on the cyclotron were then exposed to steady-state deuterium plasma on the LENTA plasma simulator. The exposure parameters were

taken as follows: deuterium ion current density about 10 mA/cm<sup>2</sup>, ion energy 100 eV (negative bias), and sample temperature during plasma operation  $\leq 40$  °C. The carbon materials were processed in the plasma in two steps 1 hour each. The layer of about a half fast ion penetration depth was eroded during the first step (about 2-3  $\mu\text{m}$ ), and the layer of the maximal radiation damage was eroded in the second plasma exposure ( $\sim 3\text{-}7$   $\mu\text{m}$ ) [4]. Weight losses were measured after plasma runs and surface microstructure was analyzed. The changes were found both after each plasma exposure. Example of the SEP NB-31 surface irradiated to 10 dpa after plasma bombardment is shown in Fig. 5b.



a



b

Fig.5. Surface of the SEP NB-31 sample: a) in initial state and b) after irradiation to 10 dpa and plasma exposure (scale 10  $\mu\text{m}$ )

Weight loss due to the plasma bombardment was registered and erosion rate  $G$  was measured. Taking into account deuterium ion current to the surface the erosion yield was evaluated. Fig.6 shows erosion rate of the SEP NB-31 composite in plasma as a function of plasma current to material sample irradiated with high-energy ions and for comparison to the non-irradiated one. Erosion yield  $Y$  was deduced from these measurements as the slope of the curves presented on the picture. Though scattered data (for CFC), the result gives evidence that erosion rate is higher for irradiated (damaged) material. Enhancement factor of the erosion yield given by appropriate ratio of  $Y$  values for irradiated to non-irradiated materials. It was  $Y_{\text{SEP irradiated}}/Y_{\text{SEP}} = 2.6 \pm 0.6$  for SEP NB-31 and even larger  $Y_{\text{pyro irradiated}}/Y_{\text{pyro}} = 4.8 \pm 0.4$  for pyrographite while the lower value  $Y_{\text{MPG irradiated}}/Y_{\text{MPG}} = 1.6 \pm 0.4$  was found for MPG-8.

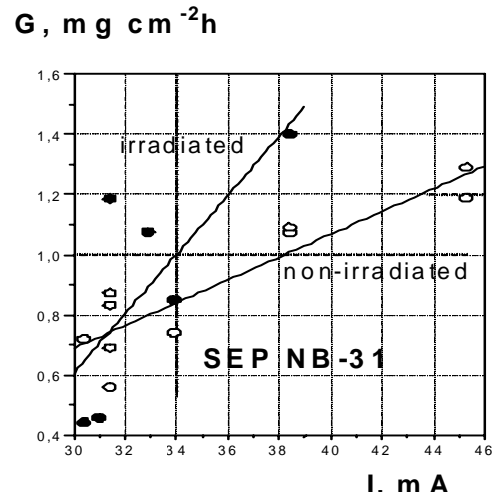


Fig.6. Erosion rate of SEP NB-31 graphite vs plasma ion flux

The results of double successive plasma exposure of carbon materials are illustrated in Fig.7 showing erosion depth and erosion rate of three types of graphite under study after the first and after the second plasma exposures.

All materials exhibit increase of erosion rate in the second plasma corresponding to the layer

of maximal defect density involved in the erosion.

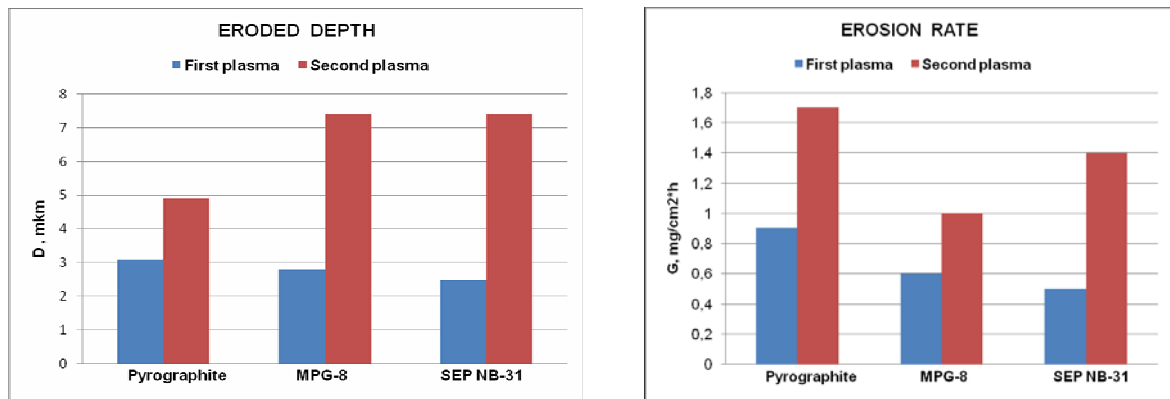


Fig.7. Erosion depth and erosion rate in double plasma experiment for 1 dpa samples of SEP NB-31, MPG-8 and pyrographite

## 5. INVESTIGATION OF MICROSTRUCTURE CHANGES OF ZR ALLOYS UNDER CHARGE PARTICLE IRRADIATION

### 5.1. Status of Zr alloys in atomic reactors

Zr alloys are considered as very important structural materials in atomic reactors. The physical and chemical properties of these materials are modified due to the microstructural changes that occur in the reactor environment. It is therefore necessary to assess how the microstructure evolves during irradiation in the nuclear reactors to predict future changes in physical-mechanical properties including radiation resistance of them.

The two main alloy types used for nuclear reactor components are either based on Zr-Nb (Zr-2.5Nb) or Zr-Sn (Zircaloy-2 and -4). Zircaloy-2 was used for pressure tubes in the Winfrith Steam Generating Water Reactor (SGHWR) and early CANDU (CANada Deterium Uranium) reactors; however, the current standard material for pressure tubes in CANDU reactors is Zr-2.5Nb. Most of the structural core components in pressurized water reactors (PWRs) and boiling water reactors (BWRs) are fabricated from Zircaloy-2 or Zircaloy-4. The behavior and radiation resistance of Zr alloys under neutron irradiation are determined by such physical phenomena as radiation growth and radiation corrosion and they limit their use in nuclear reactors at high irradiation doses. Here the some results related with the microstructure

changes of Zr alloys under charged particle irradiation are presented.

### 5.2. General aspects of dislocation loop formation and growth in Zr alloys

Zr is unusual material in that both vacancy and interstitial  $\langle \alpha \rangle$  - type dislocation loops, i.e. those having Burgers vector of

$\mathbf{b} = 1/3 \langle 11\bar{2}0 \rangle$ , are stable and can co-exist during irradiation [5]. Under neutron irradiation of Zircaloy-2 and Zircaloy-4 there is the correlation between measured irradiation growth of these materials and process of formation and growth of dislocation loops of two types: interstitial and vacancy  $\mathbf{c}$ -component dislocation loops. Vacancy  $\mathbf{c}$ -component loops have observed at temperatures between 560-773 K and range in size from about 0.1-1.0  $\mu\text{m}$  diameter depending on the temperature. Analysis has shown that they have Burgers vector

$\mathbf{b} = 1/6 \langle 20\bar{2}3 \rangle$  and, in many cases, they are directly responsible for increased irradiation growth and creep rates [6]. Increasing the alloying or impurity additions has an effect on the microstructure of neutron irradiated zirconium. It was observed [9] also that the concentration of vacancy  $\mathbf{c}$ -component faulted

$\frac{1}{6} \langle 20\bar{2}3 \rangle$  dislocation loops is highest in alloyed or impure Zr. The formation of  $\mathbf{c}$ -component loops appears to be suppressed in cold-worked material at temperatures  $\leq 580$  K, at least for the fluencies up to about  $8.10^{25}$  n/m<sup>2</sup>. The

absence of *c*-component loops has been related to the lack of solute redistribution (which promotes *c*-component loop formation) in the cold-worked material [6, 7]. This observation may be relevant to differences in the radiation growth behavior reported for annealed and cold-worked fuel channel components of PWR and BWR reactors [8]. The preferred loop habit plane is clearly determined by the *c/a* ratio and the presence of impurities. Stress could also influence loop habit plane and may even dominate, as appears to be case for the formation of *c*- component loops in electron-irradiated Zr [9].

Possibly [10, 11], the irradiation-induced dissolution of precipitates plays a role in enhancing basal plane vacancy loop formation in grains with transverse *c*-axes.

The experimental results show [12] that the dislocation loop structure in Zircaloy-2 and - 4 irradiated with fast neutrons at low temperatures (330 and 350 K) is *a*-type only for the fluence up to  $2.5 \cdot 10^{26} \text{ n/m}^2$ . Although *c*-component loops are not apparent in annealed material up to the high fluencies, irradiation of cold-worked Zircaloys containing *c*-component network dislocations (about  $0.4 \cdot 10^{14} \text{ m}^{-2}$  initially) results in an additional *c*-component, i.e., *c/2*, dislocation density increase of about  $1.6 \cdot 10^{14} \text{ m}^{-2}$  after a fluence of about  $7.5 \cdot 10^{25} \text{ n/m}^2$  due to helical limb on the existing network.

The *a*-type dislocation loop structure generated in annealed Zircaloy-4 at about 560 and 680 K increases rapidly in the early stages of irradiation ( $<0.5 \cdot 10^{25} \text{ n/m}^2$ ) from about  $0.1 \cdot 10^{14} \text{ m}^{-2}$  to achieve approximate steady-state values of about  $8 \cdot 10^{14} \text{ m}^{-2}$  [12]. The *c*-type dislocation loop structure develops over a longer period. After an initial incubation period lasting about  $3 \cdot 10^{25} \text{ n/m}^2$ , the *c*-component loop density is increased from an initial value of about  $0.001 \cdot 10^{14} \text{ m}^{-2}$  up to  $0.1 \cdot 10^{14} \text{ m}^{-2}$ .

The experimental investigations [13] in neutron irradiated zirconium alloy (Zr-1%Sn-1%Nb-0.4%Fe) show that the irradiated alloy microstructure contains irradiation-induced clusters and dislocation loops. All dislocation loops are *a*-type with Burgers vector  $1/3\langle 11\bar{2}0 \rangle$ ; *c*-type dislocation loops which

dominate in Zircaloy and Zr-Nb alloy were not observed. In this neutron irradiated alloy to a fluence of  $1 \cdot 10^{24} \text{ n/m}^2$  ( $E > 0.1 \text{ MeV}$ ) the mean size of *a*-type loops is 8 nm (the maximum loop size is 25 nm) and their concentration is  $8 \cdot 10^{21} \text{ m}^{-3}$ . At high irradiation dose up to  $4 \cdot 10^{26} \text{ n/m}^2$  the loop size was increased to 12nm, their concentration being lowered to  $4 \cdot 10^{21} \text{ m}^{-3}$ . At high fluencies the loops become aligned into rows or layers parallel to the basal plane. The observed structural changes in the irradiated alloy are accompanied by second phase redistribution.

The vacancy and interstitial dislocation loops contribute to the observed irradiation growth (typically along *a*-axis expansion and *c*-axis contraction). The magnitude of irradiation growth in annealed Zr and the Zircaloys is generally higher at 353 K compared with 553 K. One explanation for this is that only interstitial *<a>* type loops are formed during irradiation at the lower temperature and the growth strain would be higher than if both vacancy and interstitial dislocation loops are formed. In the latter case the strain could arise from a mixed population of small vacancy and interstitial *<a>* type loops. In both cases saturation of radiation growth is determined by the saturation of loop density.

Under neutron irradiation zirconium alloys are subject to structure transformations, elements redistribution and acceleration of uniform corrosion, particularly, for intermetallic type Zr-alloys. Correlation between these processes has the interest for investigations of new Zr-materials to operate in-pile for long time. Neutron irradiation in reactors and difficulties with corrosion tests of irradiated samples raise the using accelerators of charged particles to achieve high damage levels during shorter periods, that makes experiments faster and reduces their cost.

Approaches to methodology of applying accelerator irradiation and procedure of simulating damage up to 10–20 dpa those parallel typical VVER burners, subsequent corrosion and TEM examinations are discussed here.

The developed procedure on cyclotron in RRC KI allowing receiving radiation damage at 300–350 °C up to 100 μm deep

to 10 dpa doses with the optimal irradiation conditions and obtained experimental results by the scientists from RRC KI and VNINM on irradiated Zr-alloys are shown here [14].

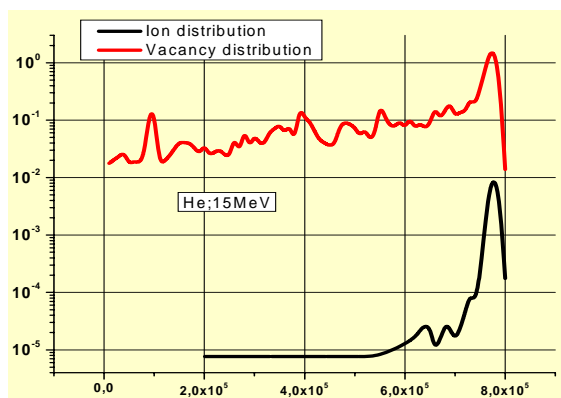


Fig.8. The distribution profile of generation rate of point defects under irradiation of Zr alloy by 15 MeV helium ions at irradiation dose  $10^{17} \text{ cm}^2$

For the choosing of the best irradiation conditions and regimes for the irradiation of Zr alloys by different types of ions on cyclotron RRC KI it is necessary to make the some numerical calculations of the distribution profiles of generation rate of point defects in Zr alloy under irradiation by different types of charge particles (protons, helium ions) with the different energies. The typical results of the numerical calculations of distribution profile of generation rate of point defects under irradiation of Zr alloy by 15 MeV helium ions at irradiation dose  $10^{17} \text{ cm}^2$  are shown in Fig.8.

The comparisons of microstructure changes in none irradiated and irradiated by helium ions with the energy 10 MeV Zr alloys:  $\text{Zr}_{110}$  and  $\text{Zr}_{635}$  (Distribution of  $\beta$  – Nb phases, Laves phase  $\text{Zr}(\text{Nb}, \text{Fe})_2$  and dislocations c-type) are shown in Fig.9 and Fig.10 respectively. In Fig.11 we can see the denuded zone formation near grain boundary (Fig.11 (a)) and distribution of dislocation loops in Fig.11 (b) with the average size  $D=5-10 \text{ nm}$  and bulk density  $N = 10^{17} \text{ cm}^{-3}$ .

For the comparison in Fig.12 we can see microstructure changes in neutron irradiated Russian Zr alloys:  $\text{Zr}_{635}$  (a) и  $\text{Zr}_{110}$  (b) up to neutron dose:  $F_n=0.5 \cdot 10^{26} \text{ n/m}^2$  and damage dose 2 dpa. The obtained results show the distribution of precipitates and ordered dislocation loops (a-type) with the average diameter of them  $D=10-20 \text{ nm}$  and bulk density  $N=10^{16} \text{ cm}^{-3}$  loops. The same physical picture has been observed after charged particle irradiation on microstructure of Zr alloy  $\text{Zr}_{635}$  irradiated on cyclotron by protons with the energy 4 MeV at temperature  $T=350^\circ\text{C}$  and irradiation dose 1 dpa ( $2 \cdot 10^{17} \text{ p/cm}^2$ ), where has been observed also the ordered structure of dislocation loops (a-type). These obtained experimental results confirm the good correlation between microstructure changes in irradiated Zr alloys after neutron and charged particle irradiations at comparable dpa level (1-2 dpa) produced radiation damage.

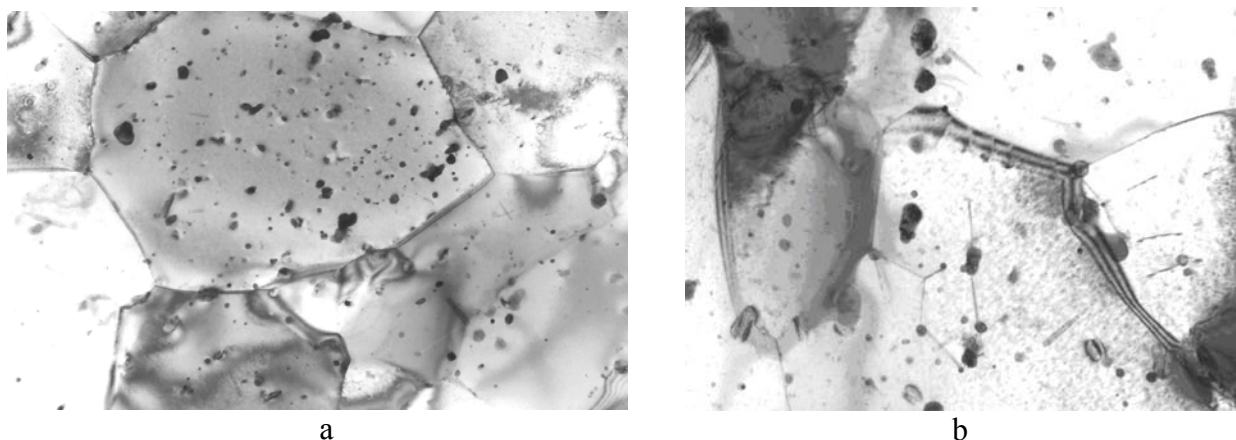


Fig.9. Microstructure of none irradiated Zr alloys:  $\text{Zr}_{110}$  (a) and  $\text{Zr}_{635}$  (b). Distribution of  $\beta$ -Nb-phases, laves phases  $\text{Zr}(\text{Nb}, \text{Fe})_2$  and dislocation loops c-type. (Magnification: X 10000)



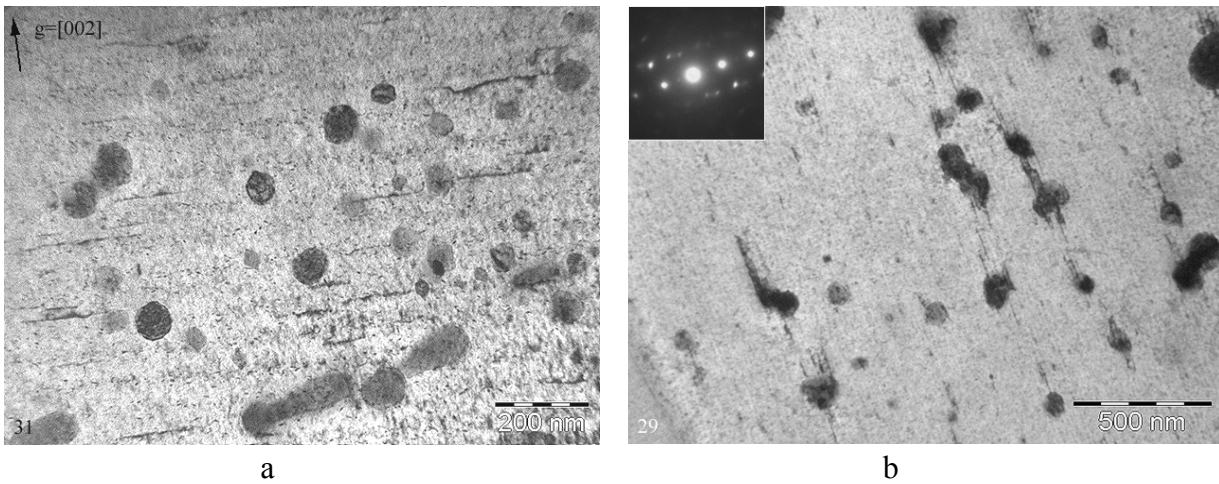


Fig.10. Microstructure of irradiated Zr alloys: 3110 (a) and 2635 (b). Distribution of  $\beta$ -Nb phases, laves phases  $Zr(Nb,Fe)_2$  and dislocations c-type. (Magnification: X 50000)

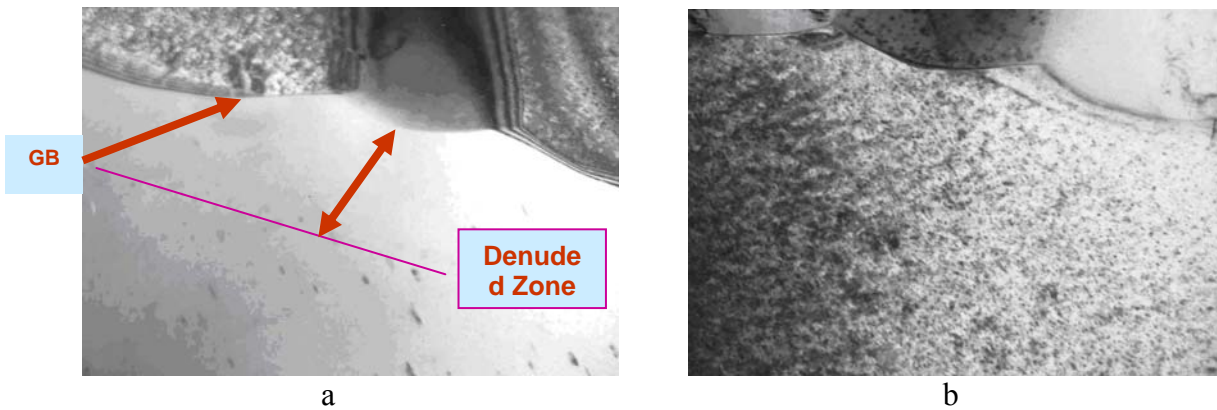


Fig.11. Microstructure of Zr alloy 3110 irradiated on cyclotron by helium ions with the energy 10 MeV at temperature  $T=350^\circ C$  and irradiation dose 1 dpa ( $D=2 \cdot 10^{17} p/cm^2$ ). The distribution of dislocation loops (a-type) has the average diameter of them  $D = 5-10 \text{ nm}$  and bulk loop density  $N=10^{17} \text{ cm}^{-3}$

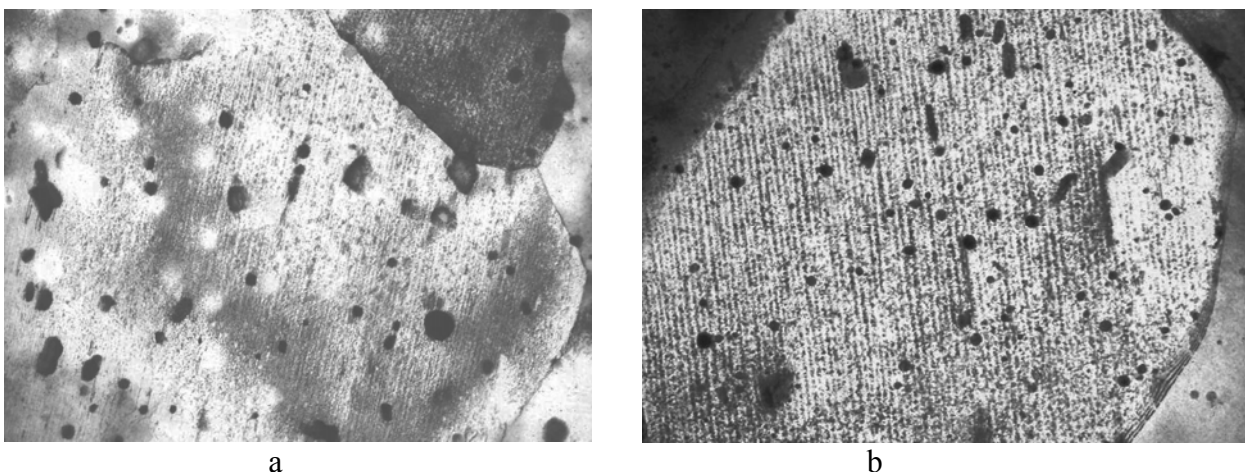


Fig.12. Microstructure of neutron irradiated Russian Zr alloys: 2635 (a) u 3110 (b) up to neutron dose:  $F_n=0.5 \cdot 10^{26} \text{ n/m}^2$  and damage dose 2 dpa. Distribution of precipitates and ordered dislocation loops (a-type) with the average diameter of them  $D=10-20 \text{ nm}$  and bulk density  $N=10^{16} \text{ cm}^{-3}$  loops.

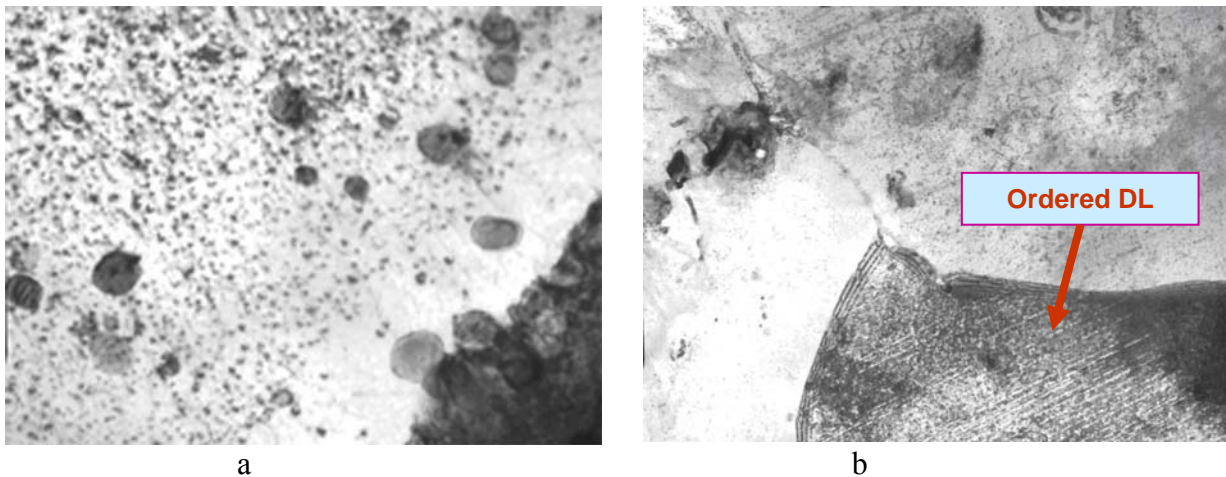


Fig.13. Microstructure of Zr alloy Z635 irradiated on cyclotron by protons with the energy 4 MeV at temperature  $T=350^{\circ}\text{C}$  and irradiation dose 1 dpa ( $2\cdot 10^{17}\text{ p/cm}^2$ )

Finally during these investigations of microstructure changes in Zr alloys it was chosen and founded the optimal irradiation regimes allowing to irradiate Zr alloys and to analyzing the microstructure changes in these materials in the temperature interval  $T=300\text{--}500^{\circ}\text{C}$  by protons with the energies up to 30 MeV and by helium ions with the energies up to 20 MeV up to irradiated doses 10 dpa. The obtained experimental results for microstructure changes in irradiated Zr alloys are comparable with the data obtained at the same dpa level after neutron and charged particle irradiations.

### CONCLUSION

Using the charged particle irradiation on accelerators of different types of materials allows clarifying many physical mechanisms of radiation resistance of fission and fusion structural materials and investigating at high irradiation doses and different temperatures the some physical phenomena in irradiated fission and fusion structural materials.

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## **ИСПОЛЬЗОВАНИЕ УСКОРИТЕЛЕЙ ЗАРЯЖЕННЫХ ЧАСТИЦ ДЛЯ ИССЛЕДОВАНИЯ ФИЗИЧЕСКИХ МЕХАНИЗМОВ РАДИАЦИОННОЙ СТОЙКОСТИ ЯДЕРНЫХ И ТЕРМОЯДЕРНЫХ КОНСТРУКЦИОННЫХ МАТЕРИАЛОВ**

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Радиационная стойкость ядерных и термоядерных конструкционных материалов определяется такими физическими явлениями как радиационное распухание и ползучесть, радиационное упрочнение, гелиевое и водородное охрупчивание. Все эти физические явления зависят от микроструктурных изменений в облученных конструкционных материалах. Ключевая проблема для продления ресурса ядерных реакторов деления и термоядерных реакторов заключается в получении экспериментальных данных относительно микроструктурных изменений в конструкционных материалах при различных температурах и высоких дозах облучения нейтронами и высоких уровнях образования смещенных атомов. Экспериментальные испытания конструкционных материалов в атомных реакторах являются очень дорогостоящими и занимают много времени. Облучение этих материалов пучками заряженных частиц на ускорителях позволяет получить экспериментальные данные о физических механизмах их радиационной стойкости значительно легче в течение коротких периодов времени без высокого уровня радиоактивности облученных образцов. Приводятся некоторые результаты исследования влияния облучения заряженными частицами на материалы.

## **ВИКОРИСТАННЯ ПРИСКОРЮВАЧІВ ЗАРЯДЖЕНИХ ЧАСТОК ДЛЯ ДОСЛІДЖЕННЯ ФІЗИЧНИХ МЕХАНІЗМІВ РАДІАЦІЙНОЇ СТІЙКОСТІ ЯДЕРНИХ ТА ТЕРМОЯДЕРНИХ КОНСТРУКЦІЙНИХ МАТЕРІАЛІВ**

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Радіаційна стійкість ядерних та термоядерних конструкційних матеріалів визначається такими фізичними явищами, як радіаційне розпухання та повзучість, радіаційне зміцнення, окрихчення гелієм та воднем. Усі ці фізичні явища залежать від микроструктурних змін в опромінених конструкційних матеріалах. Ключова проблема для подовження ресурсу ядерних та термоядерних реакторів полягає в отриманні експериментальних даних щодо микроструктурних змін в конструкційних матеріалах при різних температурах і високих дозах опромінення нейтронами і високих рівнях утворення зміщених атомів. Експериментальні іспити конструкційних матеріалів в атомних реакторах є дуже дорогими і займають багато часу. Опромінення цих матеріалів пучками заряджених часток на прискорювачах дозволяє отримати експериментальні дані щодо фізичних механізмів їх радіаційної стійкості значно легше протягом коротких відрізків часу без високих рівнів радіоактивності опромінених зразків. Наведені деякі результати дослідження впливу опромінення зарядженими частками на матеріали.