The Abdus Salam
International Centre for Theoretical Physics

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Syllabus for Joint IAEA-ICTP Advanced Workshop on

Development of Radiation Resistant Materials

20 – 24 April 2009

Miramare – Trieste, Italy

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Foreword

The Abdus Salam International Centre for Theoretical Physics (ICTP), Trieste, Italy, in co-operation with the International Atomic Energy Agency (the IAEA), Vienna, Austria, is organizing a Joint ICTP/IAEA Advanced Workshop on Development of Radiation Resistant Materials, to be held at ICTP, Trieste, from 20 to 24 April 2009.

Within the frame of the INPRO and Generation IV initiatives, the next generations of nuclear power reactors are under assessments and in the R&D process. Almost all new reactor concepts are specified by higher efficiency and better utilization of nuclear fuel with minimization of nuclear waste. For the sustainability of the nuclear option, there is currently a renewed interest worldwide in new reactors and closed fuel cycle research and technology development; however, such an approach means that a new class of structural materials with significantly better radiation resistance will have to be introduced. To achieve the high performance parameters, more focused research and testing of new candidate materials are necessary. Recent development of new classes of materials with improved microstructural features, such as composite materials (SiC) and Oxide Dispersed Strengthen (ODS) or advanced Ferritic-Martensitic (FM) steels is quite promising since they have very good radiation resistance properties. In view of the successful and timely implementation of design parameters new structural materials, in particular for primary circuits, have to be developed in next decade. The on-going research has proved that recent progress in material science, supported by computer modeling, can accelerate the R&D process for development of new structural materials.

The scope of the Workshop is education, training and information exchange. Participants will be familiarized with the physics, materials and engineering aspects of structural materials for selected reactor designs. A comprehensive review of fission as well as fusion reactor designs of the innovative material concepts presently under consideration will be given.

PROGRAMME:
The programme will consist of lectures, tutorials and computer demonstrations. Participants will also be invited to make short (10-15 minute) presentations covering their own research activities. The participants will study and discuss the theoretical foundation of all aspects related to the material problems including key issues, as radiation effects on microstructure and properties, advanced post-irradiation methodologies and multi-scale modeling as well as qualification of new structural materials. In addition, the attendees will gain knowledge related to structural materials of selected reactor designs as well as most critical areas from a structural materials point of view. The students will be familiarized with the modern theoretical approaches used to predict a radiation-induced degradation mechanisms and methodologies for the quantification. They will study the principles of the evaluation methodology and become acquainted with the current status of R&D and new challenges in radiation material science. Based on the discussion of the impact of the present uncertainties on the performance of structural materials for innovative reactor systems, the need for theoretical and experimental testing and validation procedures will be justified.

Table 1 - **Time table of the lectures**

	9:00-10:00	10:00-11:00	11:00-11:30	11:30-12:30	12:30-14:00	14:00-15:00	15:00-16:00	16:00-16:30	16:30-17:30	17:30-18:30
Monday (20.04)	**Opening VI/AZ/SS**	**ICTP**	Coffee	**VV1**	Lunch	**LD1**	**SZ1**	Coffee	**SZ2**	Poster session
Tuesday (21.04)	**SZ3**	**LD2**	Coffee	**VV2**	Lunch	**LD3**	**VV3**	Coffee	**VC1**	Poster session
Wednesday (22.04)	**VC2**	**PH1**	Coffee	**AR1**	Lunch	**AR2**	**PH2**	Coffee	**AR3**	Social event
Thursday (23.04)	**JB1**	**AR4**	Coffee	**JB2**	Lunch	**VV4**	**JB3**	Coffee	Evaluation of posters	
Friday (24.04)	**VC3**	**VV5**	Coffee	**Selected participants**	Lunch	**Closing AZ/VI**				

* for reference of the presentation see Table 2.

Table 2 - **List of the lectures**

Ref	Lecturer	Subject	Date
LD1	L.Debarberis IE-JRC (EC)	Effect of radiation embrittlement to reactor pressure vessel steels and its surveillance	20.04.2009 *14:00*
LD2	L.Debarberis IE-JRC (EC)	Phenomenon of (irradiation assisted) stress corrosion cracking for internals of PWR & BWR systems	21.04.2009 *10:00*
LD3	L.Debarberis IE-JRC (EC)	On going issues on structural materials of LFR	21.04.2009 *14:00*
PH1	P. Haehner IE-JRC (EC)	European R&D projects on materials for next generation nuclear systems	22.04.2009 *10:00*
PH2	P. Haehner IE-JRC (EC)	Mechanical testing requirements for advanced nuclear plants	22.04.2009 *15:00*
JB1	J.L.Boutard EFDA (Germany)	Radiation Effects and Major Issues of Materials for Fusion Power Plants	23.04.2009 *09:00*
JB2	J.L.Boutard EFDA (Germany)	Physical Modelling Radiation Effects of Materials for Fusion Reactors	23.04.2009 *11:30*
JB3	J.L.Boutard EFDA (Germany)	Combining experiments & modelling for effective pathways to development of new structural materials	23.04.2009 *16:30*
VC1	V. Chernov SSC Bochvar (Russia)	Radiation Defects and Radiation Properties of Metals and Structural Materials with Different Types of Crystal Lattices (Experiments, Microstructures, Modeling)	21.04.2009 *16:30*
VC2	V. Chernov SSC Bochvar (Russia)	Radiation Damage, Activation and Transmutation of Structural Materials under Long Time Neutron Irradiation (Fusion and Fast Power Reactors)	22.04.2009 *09:00*
VC3	V. Chernov SSC Bochvar (Russia)	Requirements & Problems for Structural Materials for Fusion and Fast Powers Reactors	24.04.2009 *09:00*
AR1	A.Ryazanov RRC-KI (Russia)	Introduction to the multi-scale approach and associated models	22.04.2009 *11:30*
AR2	A.Ryazanov RRC-KI (Russia)	Multi-scale approach in modelling of material properties under charged particle irradiation	22.04.2009 *14:00*
AR3	A.Ryazanov RRC-KI (Russia)	Physical Mechanisms of Radiation Creep Phenomenon in Irradiated Materials	22.04.2009 *16:30*
AR4	A.Ryazanov RRC-KI (Russia)	Qualification of new structural composite materials - SiC under neutron and charged particle irradiation	23.04.2009 *10:00*
VV1	V.Voyevodin KIPT (Ukraine)	The role of structural materials in the assurance of the world nuclear renaissance	20.04.2009 *11:30*
VV2	V.Voyevodin KIPT (Ukraine)	Degradation of physical-mechanical properties of materials under irradiation	21.04.2009 *11:30*
VV3	V.Voyevodin KIPT (Ukraine)	Technological impact on structure phase evolution of materials during irradiation	21.04.2009 *15:00*
VV4	V.Voyevodin KIPT (Ukraine)	Increasing of radiation resistance of structural materials	23.04.2009 *14:00*
VV5	V.Voyevodin KIPT (Ukraine)	Accelerators of charge particles-the power instrument for investigation of radiation resistance and development of new materials of nuclear power	24.04.2009 *10:00*
SZ1	S.Zinkle ORNL (USA)	Qualification of new structural materials	20.04.2009 *15:00*
SZ2	S.Zinkle ORNL (USA)	Development of advanced structural materials	20.04.2009 *16:30*
SZ3	S.Zinkle ORNL (USA)	Ongoing R&D programs in radiation materials science	21.04.2009 *09:00*

Synopsis

Combining Experiments & Modelling for Effective Pathways to Develop New Structural Materials

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Industrial materials are most of the time concentrated and multi-phase alloys, which often undergo phase transformations when in-service at high temperature. Understanding their phase stability and mechanical properties at high temperature and under irradiation is essential for a reliable prediction of their in-service behaviour

The ferritic/martensitic steels are in this situation suffering from α/α' unmixing or even σ phase precipitation for high Cr ferritic steels at high temperature. An essential physical quantity is the Gibbs (constant pressure) or the Helmholtz (constant volume) Free Energy, which is required for phase diagram prediction and chemical potentials, the gradient of which is controlling mass transport. The paper will recall the basic thermodynamics concerning the prediction of phase diagram of alloys with miscibility gap. Starting from ab initio calculations of the enthalpy of mixing of the Fe-Cr alloy performed at 0K, i.e. ground states, the progress made in predicting realistic Free Energy at high temperature including configuration, vibration and magnetic entropies will be sketched and the most recent results including the correct prediction of the α->γ and γ->α phase transitions within α-Fe at high temperature presented.

Concerning the yield behaviour of materials, basic theory about dislocations will be recalled in the framework of the isotropic theory of elasticity. In α-Fe, the tetragonal shear (Bain transformation) stiffness is tending to zero at the α -> γ transus . The consequences of this peculiar behaviour within iron: (i) the anisotropic elasticity of α-Fe, and, (ii) the stabilisation at high temperature of the edge dislocations of <100> Bürgers vector will be discussed. The consequence concerning the softening of α-Fe and bcc steels at high temperature will be underlined.

Finally experimental validation of modelling will be discussed in order to be at the relevant scale and with the pertinent system.

References

[1] The magnetic origin of anomalous high-temperature stability of dislocation loops in iron and iron-based alloys, **J. of Nucl. Mater. In Press, Corrected Proof, Available online 1 January 2009,** S.L. Dudarev, P.M. Derlet, R. Bullough

Physical Modelling Radiation Effects of Materials for Fusion reactors

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The European fusion program on radiation effect modelling has been launched in 2002 to study and correlate the radiation effects under the various spectra used to simulate the D-T fusion neutron spectrum. The main objective is to develop modelling tools to study radiation effects in the reference martensitic steel EUROFER under fusion reactor relevant conditions. Such a modelling is multi-scale in nature. The emphasis has been put on the physics of radiation-induced processes and its experimental validation at every space and time scale. The effort has been devoted to (i) ab-initio determination of the Fe-Cr system cohesion and of the energetics of point defects, He and carbon in α-Fe and Fe-Cr model alloys, (ii) multi-scale modelling of kinetics of radiation effects controlled by diffusion (iii) development of inter-atomic potentials for Molecular Dynamics (MD) simulation of displacement cascades and dislocation dynamics.

The presentation will give a brief review of the various theoretical tools used in multi-scale modelling radiation effects: (i)ab-initio calculation based on the Density Functional Theory, (ii) Molecular Dynamics for fast kinetics, (iii) Monte Carlo methods and Mean Field Theory method such as Rate Theory for diffusion controlled kinetics and (iv) Dislocation Dynamics for the collective behaviour of dislocations.

Recent ab-initio calculations concerning the formation energies of Self Interstitial Atom (SIA) in bcc transition metals showed the essential role of magnetism in α-Fe. Formation energies and diffusion pathways of small vacancy and interstitial clusters ($n<4$) computed in the DFT approximation were used to reproduce via Kinetic Monte Carlo the radiation damage recovery stages in α-Fe. Ab-initio energetics of He and point defects in the system Fe-He-C enabled reproducing He-desorption kinetics via Monte Carlo and Rate Theory. DFT calculations of various configurations of the Fe-Cr system in the Fe-rich range allow understanding the role of magnetism in the behaviour of the system. Exchange Monte Carlo method based on Cluster Expansion fitted on these ab-initio results gives new insight into the phase diagram of the Fe-Cr system and allow reproducing the ordering and clustering tendency of Cr atoms below and above ~9% Cr respectively.

Ab initio data can also be used to fit the empirical potentials required for Molecular Dynamics simulations. The newly developed potentials, based on different assumptions but all reproducing the correct energetics of SIAs in α-Fe, result in reducing considerably the so far unacceptable scatter of Frenkel pairs production obtained in the Molecular Dynamics (MD) simulation of displacement cascades. Semi-empirical potentials were developed to reproduce the negative mixing enthalpy of the Fe-Cr in the low Cr content domain of interest, Cr point defect interaction and primary damage using MD simulations.

The program is now focused on (i) developing reference kinetic tools to predict the Fe-Cr system phase transformation kinetics able to take into account explicitly the magnetism and effect of carbon, under thermal ageing and irradiation (ii) predicting, at the atomic scale, the screw dislocation core structure and glide mechanisms (iii) parameterising Discrete Dislocation Dynamics code to describe the collective behaviour of dislocations at the meso-scale, and (iv) carrying out validation experiments using extensively the multi-beam facility JANNUS. JANNUS will allow irradiating with double (dpa, He) or triple (dpa, He, H) beams and characterizing (TEM, AP-FIM, nano-indentation) volumes of the same order as the ones that can be numerically simulated.

The development of physically based modelling tools to study and reliably predict radiation effects in the Fe-x%Cr-C ferritic model steels is now a realistic medium term objective.

References

[1] The EU programme for modelling radiation effects in fusion reactor materials: An overview of recent advances and future goals, ***J. of Nucl. Mater.*, *In Press, Accepted Manuscript*, *Available online 30 December 2008***, S.L. Dudarev, J.-L. Boutard, R. Lässer, M.J. Caturla, P.M.Derlet, M. Fivel, C.-C. Fu, M.Y. Lavrentiev, L. Malerba, M. Mrovec, D. Nguyen-Manh, K. Nordlund, M. Perlado, R. Schäublin, H. Van Swygenhoven, D. Terentyev, J. Wallenius, D. Weygand, F. Willaime.

Radiation Effects and Major Issues of Materials for Fusion Power Plants

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The decision of constructing ITER has opened the perspective for a fusion reactor demonstrating the feasibility of the thermo-nuclear fusion energy production. The selected D-T fusion reaction releases one 14.03 MeV neutron and one 3.56 MeV helium. Elements of design of the main in-vessel components of a fusion power plant, i.e. tritium-breeding blanket, divertor and first wall, will be presented. The structural materials for these components will have to withstand high doses of ~100 dpa and production of transmutation elements such as He (~10 appmHe/dpa) and H (~45 appmH/dpa) induced by the 14.03 MeV neutrons. In addition the divertor will have to undergo high heat fluxes ~10 MW/m^2.

The irradiation by the 14.03 MeV neutrons will affect the materials at the atomic scale: (i) the crystalline structure is locally destroyed by displacement cascades, (ii) the chemical bonds are strained by He and H transmutation products, and (iii) radiation induces microstructure changes controlled by point defects and impurities diffusion. The basis for the selection of structural materials which will have to be radiation resistant under such condition will be reviewed.

For Tritium (T)–Breeding Blankets, Reduced Activation (RA) 9 % Cr ferritic martensitic steels for temperatures up to ~550 ^{0}C and Oxide Dispersion Strengthened (ODS) ferritic steels up to ~750 ^{0}C have been selected on the basis of their well known metallurgy and high resistance to neutron irradiation in fast reactors. SiC_f-SiC composites for very high operating temperatures are foreseen on the basis of the high stability of the newly developed and nearly stoichiometric β-SiC fibres. For the He- cooled divertor, W-alloys have been selected for their high thermal conductivity, high strength and low sputtering rate to withstand the high heat flux of ~10 MW/m^2.

The most significant experimental results about point defect & He accumulation and phase stability, which control the hardening and embrittlement of ferritic martensitic steels, will be presented. The issues concerning the initial fracture toughness and in–service phase stability of the W-alloys will be underlined.

In the absence of an intense 14.03 MeV neutron source various irradiation techniques are used: (i) alpha particles implantation, (ii) irradiation in fast neutron spectrum or mixed spallation-neutron spectrum, (iii) ion beam irradiation in dual or triple beam configuration, to assess the radiation effects on the in-service properties in the future fusion reactors. The main issues concerning the relevance of these techniques to simulate 14.03 MeV neutron radiation effects will be discussed.

Most of the metallic alloys irradiated at low temperatures show localisation of the plastic deformation in the so-called clear channels when tested out of flux in hot cells. Recent in-pile tensile tests will be presented questioning such behaviour under irradiation.

Qualification of these materials should be carried out in the future International Fusion Material Irradiation Facility (IFMIF) based on D-Li reaction producing a neutron spectrum very similar to the D-T fusion one. The main characteristics of IFMIF and a final overall view of all the irradiation techniques used to simulate radiation effects under fusion reactor conditions will be presented in term of dpa and He production.

References

[1] **Fissile core and Tritium-Breeding Blanket: structural materials and their requirements**, *Comptes Rendus Physique*, *Volume 9, Issues 3-4*, *April-May 2008*, *Pages 287-302*, Jean-Louis **Boutard**, Ana Alamo, Rainer Lindau, Michael Rieth

RADIATION DEFECTS AND RADIATION PROPERTIES OF METALS AND STRUCTURAL MATERIALS WITH DIFFERENT TYPES OF CRYSTAL LATTICES (EXPERIMENTS, MICROSTRUCTURES, MODELLING)

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The main models and results of the R&D investigations of initial and radiation properties of metals and structural materials (SMs – steels and alloys) with different symmetry types of the crystal lattices are discussed (face centered cubical – FCC, body centered cubical – BCC, hexagonal close packed – HCP). The main goals of theoretical, modelling and experimental investigations are to receive, compare and understand the mechanisms of the formation of microstructure and functional properties of SMs under neutron irradiation for further widening of temperature, mechanical and dose application windows for nuclear fission and fusion power reactors (mainly for their cores).

Many functional properties are general for all SMs. In such cases SMs (solid states with some defects) have a similar behavior of initial and irradiation microstructure and properties (impurity segregations, strengthening, creep, fatigue, fracture) and to understand such properties it is enough to use the isotropic approximation (isotropic theory). But some important irradiation properties are very sensitive to the symmetry of crystal lattices. In these cases it is necessary to use the anisotropic theory. The basic SMs for nuclear reactors (especially for cores) have one of mentioned above crystal lattices type.

The symmetry of crystal lattice is necessary condition to realize the typical special irradiation properties of SMs such as (1) swelling (FCC), (2) low temperature embrittlement (BCC), (3) physical yield of a fatigue (BCC), growth (HCP) and some others. Why these specific initial and irradiation properties are realizing basically only for one type of the crystal lattices is a very hard problem of the physical materials science, solid state physics and multi-scale modelling up to day. Generally speaking, today we do not understand why the different types of crystal lattices are formed. Empirical potentials or the first-principal method and the models based on them are used to describe the well known experimental results.

The typical self-point defects (vacancies and interstitials and their clusters) and linear defects (different types of dislocations) in connection with the typical functional properties of SMs with different types of crystal lattices (BCC, FCC, HCP) are discussed. Really up to now there are no physical multi-scale models and ideas for such functional properties of SMs as mentioned above. The pointed out phenomena are absent or very weak in crystals with other symmetry or in isotropic models. Some models and difficulties of their developments are discussed. The symmetry of crystal lattices and internal stresses in crystals with inner structure are the key questions, which must be included in all physical models of irradiation physical and mechanical properties of SMs.

For the practical and education goals the problems of dislocation stresses and anisotropic elasticity theory (with application to FCC, BCC and HCP crystals), many-body interatomic interaction potentials, molecular statics and dynamics and kinetic Monte-Carlo methods are discussed as important parts of radiation materials science and multi-scale modelling. The samples of models and modelling results for formation and kinetic evolution of self-point defects in dislocation stresses are discussed (BCC - α-Fe and V, FCC - Cu and HCP - Zr crystals).

Radiation Damage, Activation and Transmutation of Structural Materials under Long Time Neutron Irradiation (Fusion and Fast Power Reactors)

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The nuclear data base, codes, models and results of investigations of nuclear properties (radiation damage, activation, transmutation and cooling) of metals and based on its structural materials (SMs – steels and alloys) under long neutron exposure (up to 50 years) for ensuring the outstripping development of SMs (mainly for cores) for fission (the RF fast reactor BN-600) and fusion (DEMO-RF – the Kurchatov Institute project) power reactors are discussed. Two types of the RF SMs are considered: the reduced activation ferritic-martensitic heat resistant steel RUSFER-EK-181 (Fe-12Cr-2W-V-Ta-B-C) and low activation vanadium alloy V-4Ti-4Cr. The analysis of the induced activity in SMs after their irradiation was carried out for cooling times up to 1000 years. Used the SMs compositions and neutron spectra are very typical for the international nuclear community for power nuclear fission and fusion reactors.

The known code FISPACT and the new version of the RF basis complex ACDAM have been used in the analyses. The RF complex ACDAM includes three parts (1) ACDAM/ACT – activation/transmutation neutron cross-section for 704 isotopes from the element H (Z=1) to Po (Z=84), (2) ACDAM/DEC – decay data library for approximately 1960 radioactive isotopes from H-3 to Cf-252, and (3) ACDAM/DDL – damage data library to calculate the activation, transmutation (especially for hydrogen and helium production) and primary radiation damage (dpa) of all type of SMs. The complex FISPACT+ACDAM allow to receive all information on nuclear properties of SMs under neutron irradiation with energy up to 20 MeV and presented in a endf-6 format.

Primary radiation damages were calculated follow the well known TRN-standard model. In this model the value of the primary radiation damage (displacement per atom - dpa) is the main parameter to compare the radiation damages of SMs under irradiation of different neutron spectra. Nowadays this standard is very popular, but its physical basis is not enough for real applications. The unsolved questions of the TRN -standard are connected with physics of collision cascades under irradiation in real SMs. Some models of radiation cascades are discussed.

Molecular dynamics simulation of the collision cascades and their influence on the generation of radiation damage areas was carried out in vanadium crystals with inner structure (with grain boundaries). Interatomic interactions were described by known many body empirical atomic potential. Developmental character of the displacement cascades is determined in many respects by the presence of extensive interfaces in materials. Grain boundaries act as barrier for expansion of displacement cascades and accumulate considerable proportion of radiation defects.

As conclusion the summary may be maid (1) the complex FISPACT+ACDAM is practically ready for all nuclear calculations (activation-transmutation-cooling-primary damage in TRN-standard (dpa), (2) it is the actual the necessity of the development of the new radiation damage standard on the basis of the up to now and nearest future knowledge of physics collision cascades, crystal lattice models for metals and alloys, the physical properties of SMs under neutron irradiation and real local neutron spectra of advanced power reactors.

Requirements &Problems for Structural Materials for Fusion and Fast Power Reactors

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The requirements and R&D results of investigations of functional properties of structural materials (SMs) for the RF innovative nuclear power reactors - fission fast type BN-800 (2012) and BN-K(under elaboration) and fusion type DEMO - are presented and discussed. The SMs requirements for the RF innovation power reactors are next: (1) fast reactors – the fuel burning are 17-20-25 % (neutron load on SMs - 100-150-200 dpa), (2) fusion reactors - 10-15 MWt year/sq.m (150-200 dpa). The priorities of SMs is the heat resistance and low activation SMs. The requirements are impossible to carry out using the manufactured SMs. New SMs are required and the investigations are carried out for further widening of temperature, mechanical and dose application windows via the fabrication the special nanostructure states and the mechanisms of the O-C-N-precipitations and sub-structure hardening. The scientific and technological problems of manufacturing of new SMs are discussed. The advanced SMs are heat resistance the ferritic-martensitic (12-14)%Cr steels type the RUSFER-EK-181 (Fe-12Cr-2W-V-Ta-B-C) and vanadium alloys V-(8-4)Ti-(5-4)Cr.

The low-temperature embrittlement as the typical phenomenon of all SMs with BCC crystal lattice is very important functional property (but in correlation with heat resistance) and the fracture toughness measurements (standard and small Charpy V-noch specimens without and with fatigue cracks) to research it (a ductile-to-brittle transition temperature T_{dbtt}) are presented. The microstructure, mechanical properties and embrittlement (T_{dbtt}) of RUSFER-EK-181 with different thermal treatments (TT) before and after irradiation in the RF fast reactor BOR-60 (irradiation temperature 320-330 ^{o}C, dose 15 dpa) are presented and discussed. The post irradiation annealing was performed and the T_{dbtt} was determined. The two types of the initial steel TT were used (traditional and cycling around the critical point A_{c1}). The influence of the cyclic TT is very positive to decrease T_{dbtt} in all cases.

The results of the application of the impact method and the acoustic (ultrasonic) non-destructive method to research the low temperature embrittlement of the SMs as physical problem are discused. The T_{dbtt} temperatures determined in acoustic experiment are close in magnitude to the T_{dbtt} measured by impact testing. The very useful results of relaxation of microstructure and elastic properties of the SMs by acoustic (ultrasonic) method during the irradiation by accelerated protons with the energy 10 MeV are listed and discussed.

As conclusion the to-day recommendations for the temperature windows of irradiation applications of the low activation SMs may be made for steel type RUSFER-EK-181 – (300 – 670(700) ^{0}C) - and for vanadium alloys type V-4Ti-4Cr – (300 – 800(850) ^{0}C. But these recommendations are needed in the additional high dose neutron irradiation tests. The problems of such experiments (2009-2011) on the base of the RF fast power reactor BN-600 are presented and discussed.

Effect of radiation embrittlement to reactor pressure vessel steels and its surveillance

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The importance of understanding radiation embrittlement for the materials which are utilised for nuclear applications is paramount [1]. In fact embrittlement is a factor which can limit the life-time of nuclear components; including critical ones like the reactor pressure vessel [2] of light water reactors for example. The same could be said for the GEN IV design [3] and fusion applications [4]; ITER in particular [5]. For engineering purposes, the full description of the interaction of neutrons with solid matter is too complex to be described in a simple manner and not required in such details probably. In fact, a large number of neutron cross sections would need to be considered and, in addition, the cross sections for the various reactions of elastic or inelastic scattering, absorption and other possible reactions are strongly dependent on the neutron energy spectrum and thus in general varying while the neutron are crossing the matter. Even in the best case when we can describe the whole sets of interactions, still we have a problem in predicting systematically important macroscopic materials properties change for both flow properties, like tensile, and fracture toughness; even for metals, with a very well known chemical composition. In fact, not only the chemical composition but also the metal micro structure and irradiation temperature for example plays a fundamental role when it comes to neutron induced measurable damage. It is not a surprise that with the given boundary conditions for most of the materials used for nuclear applications ad-hoc research programs based direct results are utilised at present to follow irradiation embrittlement trends. For most applications the drawn trends are sufficient to allow safe prediction of nuclear critical components while research at international level, like at IGRDM, is still ongoing to understand and quantify microstructure evolutions

and refine prediction models both by empirical, mechanistic and lately multi-scale modelling of the chain of complex involved processes from ab-inzio conditions [6].
In this paper the various components of radiation embrittlement are discussed in some more details and the state of the art of the modelling is briefly addresses.

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Phenomenon of (irradiation assisted) stress corrosion cracking for internals of PWR & BWR systems

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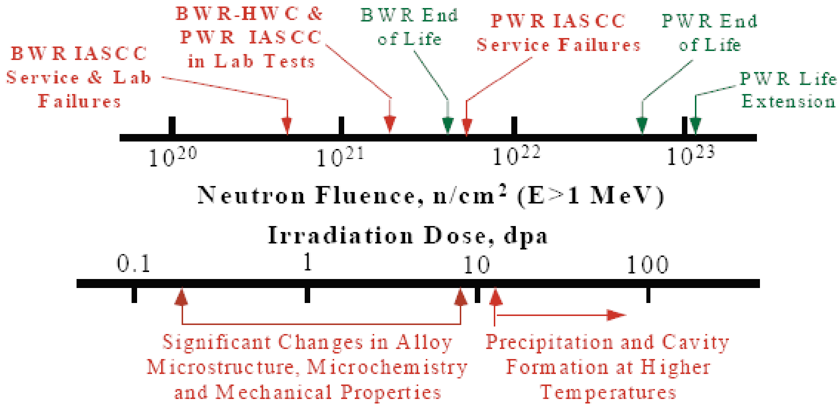
Irradiation assisted stress corrosion cracking (IASCC) is the primary form of core component cracking in boiling water reactors (BWR). It is also an issue of growing importance in pressurized water reactors (PWR). An understanding of the mechanism of IASCC is required in order to provide guidance for the development of mitigation strategies for effective plan life management of ageing NPPs.

Irradiation Assisted Stress Corrosion Cracking (IASCC) is the terminology used to describe cracking of metallic materials exposed to a nuclear reactor coolant and ionizing radiations. Like all stress corrosion phenomena, it requires critical combinations of applied stress or strain, environmental chemistry and metallurgical structure to occur. However, the added feature of IASCC is that, by virtue of atomic displacements, neutron irradiation significantly alters the metallurgical microstructure and ionizing (α, β, γ and neutron) radiation can modify the environmental chemistry. Neutron radiation also causes stress relaxation by irradiation creep, which is a potentially beneficial factor for IASCC resistance. Strong requirements for the reliability of in-core in-vessel structural materials demand a full control of the degradation of the structural materials and guiding replacement campaigns.

IASCC of austenitic stainless steels (used as a material of internals in most cases) is observed in service at lower neutron doses in the oxygenated BWR coolant (i.e. Normal Water Chemistry, usually abbreviated as NWC) compared to the hydrogenated PWR primary coolant.

The preferred parameter for expressing neutron damage is the number of "displacements per atom" or "dpa"; the conversion factor appropriate to the mixed neutron energy spectra of both PWR and BWR is given in Figure 1 where neutron fluences (doses) are given in the alternative units of n/cm^2, the convention adopted here being to give such figures for energy levels greater than 1MeV.



Note : ~15 dpa ≡ 10^{22} n/cm^2 E ≥ 1 MeV (for PWR and BWR neutron spectra)
~7 dpa ≡ 10^{22} n/cm^2 E ≥ 0.1 MeV (for PWR and BWR neutron spectra)

Figure 1 Characteristics of IASCC in Austenitic Stainless Steels [1]

Following observations of cracking of BWR core shrouds and top guides as well as baffle/former bolt cracking in PWR internals that have been attributed to IASCC over the last decade or so, several major research programs were initiated to investigate the causes and to find suitable remedies. The Cooperative IASCC Research (CIR) Program is one of the larger programs in the field.

The CIR program groups together the interests of several international partners drawn from North America, Europe and Japan with project management provided by EPRI. Members include utilities, vendors, nuclear safety authorities and national research laboratories. The program combines both PWR and BWR interests in IASCC. Both types of light water reactor have similar neutron spectra that subject the core support structures to the same types of neutron irradiation damage and, although the water chemistry differs significantly, the resources of the two interests have been combined effectively to mutual advantage. Nevertheless, due to smaller water gaps between the core internal structures and the nearest fuel elements in PWR compared to BWR, the neutron flux to PWR internals can be up to an order of magnitude higher than in a BWR. Consequently, the range of neutron doses of interest to PWR internals is roughly ten times that applicable to BWR.

The objectives of the CIR program were focused primarily on the phenomenon of IASCC and can be summarized as follows:

- Develop a mechanistic understanding of IASCC;
- Derive a predictive model of IASCC, if possible based on a mechanistic understanding;
- Identify possible countermeasures to IASCC.

The CIR program has been organized in two phases plus an extension program in order to complete experimental actions still in progress from the second phase. Phase 1 ran from 1995 to 2000 and Phase 2 to 2005. The extension program should have finished in 2008.

A review of the phenomena of IASCC for the different materials is given in this presentation.

On going issues on structural materials of LFR

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For nuclear energy generation to significantly contribute to greenhouse gas mitigation, very large growth rates in the nuclear market share would be required. To achieve such levels, the available fissile mass becomes a limiting factor. Thus, the use of fast reactors with moderate to high conversion ratios must become a significant factor in the introduction of advanced nuclear energy sources.

The LFR concept [1] is one of the FR concepts which are in developments within GEN IV international cooperation. The LFR core is conceived with fuel as metal or nitride-based, containing fertile uranium and transuranics; with full actinide recycle fuel cycle.

The LFR reactor is cooled by lead by natural convection with a today's reactor outlet coolant temperature of 550 degrees Celsius. The possibility is to progressively increase the outlet temperature, ranging up to 800 degrees Celsius, as soon as advanced materials become available after qualification work. These higher temperatures are interesting for efficiency purposes but also enables the production of hydrogen by thermo-chemical processes.

Amongst fuel efficiency and the reduced production of high-level radioactive waste and actinides (thereby mitigating proliferation risks), one main advantage of the LFR system is its materials management .

The key challenges for the LFR system concern the lead or lead alloy handling and the development of the necessary fuels and materials in the range of 550/800°C.

The main materials issues, including high temperature stability, material characterisation and qualification testing are described in this presentation. Emphasis is also given to high temperature erosion and corrosion assessment and qualification of a wide range of materials in connection with lead.

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European R&D projects on materials for next generation nuclear systems

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The development of next generation nuclear systems, needed to replace or supplement the current designs of nuclear reactors within the next 25 years and beyond, critically depends on the availability of advanced structural and functional materials systems which have to withstand extreme conditions: high temperatures, intense neutron irradiation, and strongly corrosive coolant environments, in combination with complex loading states and cyclic loading histories. International research efforts are required to qualify commercially available materials under the extreme conditions that can be encountered in the innovative concepts, and to develop, characterise and qualify new materials and coatings for longer term perspectives. Generally, the mechanical performance and reliability of those materials will depend in which of the six candidate systems for Generation IV reactors they will be applied. Some issues, however, are also cross-cutting in nature, i.e. common to different GenIV systems, ADS and fusion devices. This presentation gives an overview of ongoing and recently concluded European R&D projects on materials development for next generation nuclear reactors. Using examples taken from projects in which the author's laboratory has participated, the materials qualification high temperature testing for crucial components, like reactor pressure vessel and piping, gas turbines and heat exchangers is described in some detail. Finally pointers are derived as to not only the scale of the remaining research needs but also the mechanisms which are planned to be followed in Europe, to obtain the required data and understanding.

Mechanical testing of advanced materials for GenIV, ADS and fusion systems

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Structural materials issues represent major challenges for all next generation nuclear systems, including the envisaged GenIV concepts, ADS and fusion systems, since reliable data on the performance of candidate materials under the representative conditions (high temperature and extended operation times) and environments (intense neutron irradiation and corrosive coolants) are generally missing. The presentation focuses on the mechanical characterisation and qualification of candidate engineering materials in the relevant environments and conditions. Emphasis is put on the assessment of damage interactions due to the simultaneous exposure to complex thermo-mechanical loads, chemical attack and irradiation. Noting that these complex issues can only be addressed within international collaborative efforts, the need for the development and harmonisation of advanced testing techniques, in particular non-standard environmental tests specifically designed to assess the materials performance under realistic operation conditions, is stressed. The need for careful data management represents another concern to be discussed in the light of lessons learnt from the past.

Charge State Effects of Radiation Damage on Microstructure Evolution in Dielectric Materials under Neutron and Charge Particle Irradiations

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Dielectric materials are required to use in a future fusion reactors as radio frequency windows, diagnostic probes and et.al. The degradation of physical properties of these materials under neutron irradiation is determined by the kinetics of point defect accumulation in defect clusters (dislocation loops, voids and et.al.). Under irradiation due to the ionization process and covalent type of interaction between atoms the point defects in dielectric materials can have an effective charge. The physical situation for microstructure development in irradiated materials having non-charged and charged point defects is completely different.

In this lecture the physical mechanisms of growth and stability of interstitial defect clusters in ceramic material - cubic zirconium are proposed under different types of irradiation conditions: 100-1000 KeV electrons, 100 KeV He^+ and 300 KeV O^+ ions. The anomalous formation of extended interstitial defect clusters under electron irradiation subsequent to ion irradiation is observed using microstructure investigation by TEM. It is demonstrated that the strong strain field (contrast) near interstitial clusters is formed. Under electron irradiation the interstitial clusters grow up to the some critical size and after then become unstable resulting to the multiplication of dislocation network near the interstitial clusters. For the explanation of this phenomenon the several theoretical models of growth and stability of interstitial clusters in ceramic materials are suggested, which based on the consideration of growth of normal and charged dislocation loops, gas-filled platelets, taking into account the charge states of point defects and point defect clusters, the effect of electric and elastic fields formed near interstitial clusters on the diffusion migration process of point defects. On the Based of these theoretical models the electrical field distributions and the elastic stress fields in ceramic materials near charged interstitial clusters are calculated. The obtained theoretical results for distribution of modified strain field contrast, growth rates and critical radius of unstable interstitial clusters are compared with existed experimental data.

In this lecture a new method is proposed also for the investigation of charge states of point defects in irradiated dielectric materials. Experimental investigations show that in irradiated dielectric materials the denuded zone near free surface and grain boundaries is formed. New method is based on the effect of an applied electrical field on the formation of interstitial-type point defect clusters (dislocation loops) in irradiated ceramic materials new grain boundaries and free surface. For this aim a new theoretical model is suggested, which takes into account the effect of applied electrical field on the denuded zone formation. In this model it is shown that the denuded zone size depends on charge of point defects and direction of applied field to free surface. The obtained theoretical results are compared with the experimental data, performed in α-Al_2O_3 irradiated at 760 K by 100 keV He^+ ions. The performed experimental observations shown, that a bigger fraction of interstitials escapes to surface sink in a wedge-shaped thin-foil specimen under irradiation condition and width of denuded zone is also changed in the presence of an applied electrical field. The comparison of theoretical and experimental data allows estimating the effective charge of point defects in irradiated α-Al_2O_3.

Multi-scale Approach in Modeling of Radiation Induced Phenomena in Irradiated Materials

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The multi-scale approaches and modeling of radiation induced phenomena in irradiated materials based on *ab initio* calculations and developed new theoretical models for the investigations of physical mechanisms of radiation resistance of fission and fusion structural materials will be presented in this lecture. Some physical phenomena determined radiation resistance properties with the different time scale are considered here including the modeling of point defects, cascade and sub-cascade formation, defect migration and clustering, radiation swelling, creep and at. al. Molecular dynamic methods, Monte Carlo simulations and developed theoretical models are used for the investigation of primary radiation damage formation in irradiated materials. The analytical relations for the some characteristics of sub-cascades are determined including the average number of sub-cascades per one primary knock atom (PKA) in the dependence on PKA energy, the distance between sub-cascades and the average sub-cascade size as a function of PKA energy. Based on the developed model the numerical calculations for main characteristics of sub-cascades in different materials: Be, Fe, V, W, Cu and C are performed using the neutron fluxes and PKA energy spectra for fusion reactors: ITER and DEMO. The numerical results for main characteristics of sub-cascade formation under fusion irradiation conditions are compared with the same results obtained using neutron energy spectrum of fission reactor HFIR.

Follow the formation of self interstitial (SIA) dislocation loops in perspective fusion structural materials - binary vanadium alloys: V-A (A=Fe, Cr and Si) are analysed here using a reaction rate theory, including the effect of undersized solute atoms on SIA loop nucleation and growth. In this model undersize solutes can act as the loop nucleation sizes. The suggested model takes into account also the effect of solute segregation to loops and dislocation lines. Such bias modification affects the nucleation and growth SIA loops too. The influence of these two factors on nucleation and growth dislocation loops in binary vanadium alloys is discussed here. It is shown that under irradiation the density of dislocation loops is increased with the increasing of concentration of undersized solute atoms and growth kinetics of SIA loops in these alloys has some peculiarities too. The comparison of numerical modelling with the observed experimental data related with the dislocation loop formation and growth under electron irradiation show that the suggested model is able to describe the main features of the experimentally observed results for nucleation and growth of SIA loops in binary vanadium alloys.

The results of numerical modeling of radiation swelling in graphite and SiC, that are considered as structural materials for VHTR reactors, are presented here and they are based on kinetic consideration of point defect accumulation and kinetic growth of defect clusters (dislocation loops and voids) in the matrix. The theoretical model for description of radiation swelling in graphite takes into account the anisotropic microstructure of graphite and the effect of grain size on radiation swelling. The numerical modeling of radiation swelling in polycrystalline graphite is performed in the dependence on grain size and concentration of impurity atoms. The influence of helium atoms on radiation swelling of SiC is considered here too. The obtained theoretical results for radiation swelling are compared with the existed experimental data for irradiated graphite and SiC materials.

Physical Mechanisms of Radiation Creep Phenomenon in Irradiated Materials

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The radiation resistance of fission and fusion reactor structural materials under neutron irradiation is determined by many physical phenomena. One of the important problems from them is an irradiation creep. This phenomenon is determined by radiation induced plastic deformation in stressed structural materials which is increased with the increasing of neutron irradiation dose. The radiation induced plastic deformation is characterized by a strain rate and it depends on many parameters: initial microstructure, chemical composition of allays, irradiation temperature, cascade efficiency, generation rate of point defects, and accumulation of defect clusters (dislocation loops, voids and precipitates) under neutron irradiation. The investigation of influence of these parameters on the irradiation creep is very important for understanding of physical mechanisms of this phenomenon and chose of best radiation resistance fission and fusion structural materials.

The main aim of this lecture is the comparative analyses of physical mechanisms of irradiation creep of fission and fusion structural materials based on modern theoretical models and last experimental results concerning the peculiarities of irradiation creep behaviour under neutron irradiation. This lecture is oriented on the clearing of the effect of elemental composition of structural materials, type of crystal lattice (BCC and FCC), neutron flux, generation rate of point defects, temperature and dose dependencies of irradiation creep under neutron irradiation.

The modern theoretical models of irradiation creep of structural materials are presented here, which take into account the some peculiarities of defect microstructure evolution in these materials under irradiation. The presented data in this lecture include the critical review of last experimental results concerning irradiation creep behaviour in different types of structural materials under neutron and ion irradiation including ferritic-martensitic and austenitic stainless steels. The special part of this report is devoted to the comparison of some theoretical models for the calculations of irradiation creep module with experimental data obtained under neutron irradiation in fast atomic reactors and ion irradiation.

The presented here results allow clarifying the some physical mechanisms of irradiation creep phenomenon in fission and fusion structural materials.

Qualification of new structural composite materials - SiC under neutron and charged particle irradiation

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Ceramic materials produced on the basis of SiC and SiC/SiC composites are considered, due to their high-temperature strength, pseudo-ductile fracture behavior and low-induced radioactivity, as candidate materials for fusion and high temperature gas cooled reactors. The radiation swelling and creep of SiC are very important problems that determine the radiation resistance and using of them in these reactors. In fusion reactor environment helium atoms will be produced in SiC in the first wall region up to very high concentrations (15000-20000 at.ppm) and therefore it is very important to understand a helium effect on radiation swelling of SiC.

In this lecture a compilation of non-irradiated and irradiated properties of SiC are provided and reviewed and analyzed in terms of application to fusion and high temperature gas cooled reactors. In addition to a compilation and review of literature data are included, in the different temperature irradiation regimes.

Special topic of this lecture is oriented on the micro structural changes in chemically vapor-deposited (CVD) high-purity beta-SiC during neutron (to $4.5–7.7 \cdot 10^{25}$ n/m^2 (E > 0.1 MeV)) and self-ion irradiation (5.1 MeV Si^{2+} up to 200 dpa) at elevated temperatures. The evolution of various irradiation-produced defects including dislocation loops, network dislocations and cavities is discussed as a function of irradiation temperature and fluencies. These observations are discussed in relation with the known irradiation phenomena in SiC, such as low temperature swelling and cavity swelling. It is shown also here that the compressive stress following the anisotropic swelling in ion-irradiated specimen may affect the loop evolution.

One of the main difficulties in the radiation damage studies of SiC materials lies in the absence of theoretical models and interpretation of many physical mechanisms of radiation phenomena including the radiation swelling and creep. The point defects in ceramic materials are characterized by the charge states and they can have an effective charge. The internal effective electrical field is formed due to the accumulation of charged point defects in matrix which affects on the diffusion process of charged point defects under irradiation.
In the present lecture the general physical mechanisms of radiation swelling in irradiated SiC materials are summarized. The new theoretical models are suggested for the investigation of radiation swelling including the explanation of helium effect on this phenomenon in irradiated ceramic materials. The theoretical models are based on kinetic consideration of charged point defect accumulation and kinetic growth of point defect clusters (dislocation loops and voids) in the matrix taking into account the charge state of point defects and the effect of internal electric field formed under irradiation near dislocation loops on diffusion processes of point defects in the matrix.

The recent experimental results concerning the helium effect on radiation swelling of SiC under neutron, single-ion and dual-beam irradiation are presented here too. The obtained theoretical results for radiation swelling are compared with the existed experimental data for irradiated ceramic materials under neutron and ion irradiations. It is shown that helium atoms increase the radiation swelling of SiC especially at high temperatures.

The role of nuclear reactor materials in the assurance of nuclear renaissance

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Nuclear reactor materials are the very important part of nuclear renaissance. Generally, in any reactor, the core component materials are subjected to demanding conditions of temperature, stress and neutron irradiation. Such conditions demand more efforts in the development of core component materials Intense R&D efforts in the area of radiation damage have led to the development of a wide spectrum of core component materials.

1. Modern status of structural reactor materials for exploitated and future nuclear reactors.

1.2. Philosophy of the selection and designing of materials for certain types of nuclear reactors.

1.3. Development of chemical compositions and optimization of structure-phase conditions of base materials, which are used in the cores and for reactor's pressure vessels

1.4. The main irradiation facilities, methodics of investigation of microstructure evolution

2. Interaction of irradiation with matter - structure and composition effects.

The macroscopic effects are the concern of the plant designer and operator arise from microstructural change which are initiated at the atomic level

2.1. Frenkel pairs and their behaviour. Unit of radiation damage.

2.2.Cascades of atomic collisions and their influence on the microstructure evolution. Influence of spectrum of irradiated particles.

2.3. Point defects behaviour in irradiated FCC, BCC and HCP structures and reactions between them.

2.4. Kinetics of freely migrating defects, clustering and formation of defect complexes.

2.5. Mechanisms of radiation-induced segregation Inverse Kirkendall effect

2.6. Role of transmutation effects –in different materials and different reactors.

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Technological impact on structure phase evolution of materials during irradiation

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Purpose of this lecture to present arguments that justify the study of micro structure phase evolution processes, which are responsible for property changes. It will be described in more details selection of key technological issues concerning nuclear plant performance.

2.1. Pressure vessel steels.

2.1.1. Matrix defects in irradiated pressure vessel steels.

2.1.2. Phase transformation in irradiated pressure vessel steels-formation and evolution.

2.1.3. Role of nickel and alloying elements in structure-phase transformations.

2.2. Zr-base alloys.

2.2.1. Features of dislocation structure evolution.

2.2.2. Second phases behaviour in zirconium alloys.

2.2.3. Transmutation effects and their influence on defect structure evolution.

2.3. Austenitic and ferritic-martensitic steels.

2.3.1. Nucleation and evolution of components of dislocation ensemblies in austenitic and ferritic-martensitic steels at different and technologically important structure states.

2.3. Solid solution decay during irradiation-features and similarities in different steels.

2.3.3. Precipitates in FCC and BCC materials .Mechanisms of second phase evolution in austenitic and ferritic-martensitic steels.

2.3.4. Mechanisms of void swelling as a last step of micro structure evolution.

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Degradation of physical-mechanical properties of materials under irradiation

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This lecture briefly reviews the main technological problems that result from radiation damage phenomena. These problems of nuclear materials determine the safety and economy of appropriated nuclear power plant work.

3.1. Radiation embrittlement of pressure vessel steels-the most important factor in plant safety and the protection of the investment.

3.1.1. Embrittlement, which is determined by strengthened mechanisms.

3.1.2. Phosphorus segregation and intergranular embrittlement.

3.1.3 Comparison of microstructure features of radiation embrittlement of pressure vessel steels in different reactors (PWR, WWER-440, WWER-1000).

3.2. Radiation behavior of Zr-base alloys-key core components in water –water reactors

3.2.1. Growth of Zr-base alloys.

3.2.2. Voids in Zr –alloys .Transmutation effects in zirconium alloys and their influence on degradation.

3.3. Void swelling and void structure parameters in austenitic and ferritic-martensitic steels-main structural steels for many nuclear applications.

3.3.1. Temperature and dose dependencies of swelling for technologically important metals and alloys.

3.3.2. Similarity and distinction of processes of swelling of metals, of steels and of alloys with FCC and BCC lattices.

3.4. Features of radiation embrittlement in austenitic and ferritic-martensitic steels.

3.5 Degradation of properties in pressure vessel internals (PVI) -low temperature swelling and embrittlement.

3.6. The role of gaseous and solid transmutants in the degradation of physical-mechanical properties of irradiated materials.

3.7. Influence of irradiation conditions (dose rate, stress, temperature history etc) on degradation of initial materials properties.

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Increasing of radiation resistance of structural materials

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Looking forward, it is projected that structural components of Generation IV fission reactors will operate at 500ºC–1000ºC and reach damage levels of up to 100–200 dpa. Radiation doses in future commercial fusion power reactors might be significantly higher. Such high doses and temperatures will most certainly require the development of improved materials.

This lecture is devoted to description of possible ways of improvement of radiation resistance of materials on the background of knowledge of structure-phase transformation in materials under irradiation.

4.1. Change of composition and increasing of radiation resistance of pressure vessel steels.

4.2. Radiation resistance of zirconium alloys.

4.2.1. Optimization of Zr-base composition and increasing their radiation resistance.

4.2.2. Influence of phase transformation on improvement of properties.

4.3. Link of structure and composition of materials with their radiation resistance.

4.3.1. Influence of structure factors and microchemical evolution on radiation resistance of austenitic, ferritic and ferritic-martensitic steels.

4.3.2. Influence of alloying on evolution of structure and swelling of steels of base compositions.

4.3.3. Dynamic stability of second-phase precipitates. Precipitates as key points in creation of steels resisted to void swelling.

4.3.4. Synergetic and self-organization in irradiated materials.

4.4. Grain boundary engineering and minimization of segregation processes, swelling and intergranular embrittlement The role of alloying components in the variation of the segregation level.

4.5. Philosophy of creation and status of development and investigation of nano-(precipitation-strengthened) materials. Mechanisms of radiation resistance increase in nano-structures. Stability of nano, meso and macro-levels of nanostructures in conditions of thermal and radiation exposure

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Accelerators of charge particles-the power instrument for investigation of radiation resistance and development of new materials of nuclear power

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Problems of life extension for exploitation of nuclear reactors and development of new type reactors demand receiving a lot of data for properties of structure and fuel materials under irradiation, that is practically impossible without using of accelerators. Since, accelerators play a very important role in technological development and industrial applications, further coupling accelerator studies with modeling can have tremendous potential to increase understanding of radiation damage in high dose materials, validation of complex materials models and increased use of novel characterization techniques.

Understanding of radiation damage mechanism of nuclear materials and development of technology for estimating and predicting radiation damage-are main tasks for accelerators using.

It is necessary to say that now really exist new era (also renaissance)for studies on ion-accelerators - with high-tech instrumentations.

How these tasks are solved now and which instrumentation is useful in ion simulation experiments-is a predmet of this prseentation.

Briefly main tasks, which are needed in accelerators using are such:

- Investigation of fundamental processes (Simulation of particle collisions, Quantification of kinetic properties of radiation defects, Simulation of formation & growth of defects, defect characteristics depending on radiation dose (type, size, density, etc.)
- R&D materials for fast reactors (swelling and embrittlement) Observation of radiation-induced microstructure such as segregation and hardening\Radiation behaviour of Zr-base alloys Microstructural predicting for possibilities of Life extension for exploitating reactors; RPV steels (dpa rate), RVI (low temperature embrittlement)

- Synergetic effect on helium and hydrogen in fusion and spallation systems.

The short schedule of prented lecture is:

5. History of the accelerators using in investigations of radiation damage of materials

5.2. Classifying of accelerators-for study of which phenomena they may be used

5.3. Methodological aspects of using accelerators

5.4. Advantages and artifacts of accelerator using.

5.5. Correlation with the results of reactor tests.

5.6. Use of accelerators in modern nuclear power –new methods and new tasks.

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Qualification of New Structural Materials

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Qualification of structural materials for nuclear energy applications typically involves two sequential steps: qualification of the material for general-purpose (non-irradiation environment) structural engineering applications by a recognized national or international regulatory agency, and then qualification of the material for the demanding radiation environment. In the United States, the first step typically involves the development of a code case presented to the American Society of Mechanical Engineers (ASME) for approval, which requires acquisition of a wide range of mechanical and physical property data on multiple (three or more) large-scale heats of the alloy. The second step involves examination of the dimensional, mechanical and structural stability of the alloy following exposure to prolonged neutron irradiation at prototypic conditions. Each of these steps tend to be very time-consuming; for example, the typical time required to prepare a successful ASME code case for a new structural material intended for nonirradiation applications is on the order of 5 to 10 years. As a result, there are currently only four alloys (Types 304 and 316 austenitic stainless steels, nickel-base alloy 800H, and 2 ¼ Cr-1Mo bainitic steel) that are currently qualified for nuclear reactor service at elevated temperatures under the ASME Boiler and Pressure Vessel Code Section III, subsection NH (a partially completed Code case for 9Cr-1Mo ferritic-martensitic steel is also drafted). A summary of the key mechanical properties measurements for structural materials will be given. Potential strategies to shorten the time needed to obtain regulatory approval for a new structural material will be discussed.

Development of Advanced Structural Materials

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Structural materials represent the key for safe containment of nuclear fuel and fission products as well as reliable and thermodynamically efficient production of electrical energy from nuclear reactors. Advanced materials can enable improved reactor performance via increased safety margins and design flexibility. In particular, increased strength and thermal creep resistance would provide greater design margins leading to improved safety margins, longer lifetimes, and higher operating temperatures, thus enabling greater flexibility. In many cases, a key strategy for designing high-performance radiation-resistant materials is based on the introduction of a high, uniform density of nanoscale particles that simultaneously provide good high temperature strength and neutron radiation damage resistance.

Development of structural materials for nuclear energy applications is historically a long and costly process, due to the long proof testing period to validate the performance of the material in prototypic environments for appropriate licensing authorities. Materials science tools such as computational thermodynamics and multiscale radiation damage computational models in conjunction with focused experimental validation studies (nonirradiation and irradiation environments) may offer the potential for a significant reduction in the time period to develop and qualify structural materials for advanced nuclear energy systems. Validation of the performance of these advanced materials in prototypic operating environments will be a key step to obtain acceptance of these advanced materials by reactor vendors, utilities, and the licensing authorities. Examples of the potential for rapid development of high-performance structural materials will be given, including both evolutionary ingot-based steel metallurgy and alternative processing techniques such as powder metallurgy production of oxide dispersion strengthened steels.

Ongoing R&D Programs in Radiation Materials Science

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Research programs for radiation effects on structural materials are currently being stimulated by growing interest in improved materials for fission energy (existing light water reactors as well as potential Generation IV reactors), fusion energy, and spallation neutron sources. The major categories of materials being investigated include austenitic stainless steels, ferritic/martensitic stainless steels, zirconium alloys, oxide dispersion strengthened ferritic/martensitic steels, refractory alloys, and ceramic composites including carbon-carbon and SiC-SiC composites. In all cases, underlying fundamental issues associated with defect production and accumulation are being investigated using a combination of computational modelling and experimental tests. Material-specific issues such as radiation-induced changes in dimensions (void swelling, etc.) and changes in the ductile to brittle transition temperature are being addressed with specific testing programs and accompanying modelling. This presentation will summarize some of the major materials candidates for fission and fusion energy systems, and highlight some of the recent research accomplishments on structural alloys and ceramic composites.

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Participants' Synopsis

Efficiency Calibration for a 4π NaI(Tl) Gamma-ray Detector

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Abstract

A straightforward mathematical method for the efficiency calibration of 4pi NaI (Tl) detectors especially for environmental and low activity samples is presented. The method is based on the model, developed earlier by Y.S. Selim and M.I. Abbas for gamma detector efficiency calculations. Furthermore, the attenuation of photons by the source it self (self-absorption) is determined by calculating the photon path length through the source material. Especially for environmental samples with large volumes, this method is very useful, because it takes into account the self-absorption of photons in the sample. The theoretical and the published experimental efficiency values are in good agreement.

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3- **Mahmoud I. Abbas** (2006), Applied Radiation and Isotopes 64, 1048.

4- S. H. Byun, W. V. Prestwich, K. Chin, F. E. McNeill, D. R. Chettle, Nucl. Instr. And Meth. A535 (2004) 674.

5- **Mahmoud I. Abbas** (2001), Applied Radiation and Isotopes 55, pp. 245-252.

6- **Mahmoud I. Abbas** and Younis S. Selim (2002), Nuclear Instruments & Method in Physics Research –A480, 649.

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TOMOGRAPHIC ATOM PROBE CHARACTERIZATION OF REACTOR STRUCTURAL MATERIALS

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RF SSC Institute for Theoretical and Experimental Physics, Moscow

It is known that nanostructural features play a crucial role in material defining its macroscopic parameters. In some cases specially created high number density fine clusters increase radiation resistance and high-temperature strength, in other formation of clusters under irradiation is considered to be the cause of embrittelment. Understanding the influence of such clusters on material behaviour under heat- and radiation exposure requires not only information about their size and composition but also about internal structure and interface. As the size of these peculiarities does not exceed few nanometers one of the most appropriate techniques for their characterization is tomographic atom probe. It allows to reconstruct a 3D image of the investigated material with nearly atomic resolution and simultaneously to determine chemistry.

In this presentation an overview of tomographic atom probe (TAP) investigation of constructional materials carried out in the laboratory of atomic-scale investigations of condensed matter (ITEP) will be presented. TAP results on initial structure of a precipitation-hardening steel Rusfer EK-181 and an oxide-dispersion strengthened steel Eurofer ODS will be showed. A number of fine clusters enriched with V-N-C and Y-V-O, found in EK-181 and Eurofer ODS correspondingly, will be described.
Comparison of our results with other TAP data for similar steels will be shown focusing on clusters' structure and composition. Review of TAP investigation of VVER-440 weld and application of this technique for detailed study of vessel steel degradation will also be presented.

Evaluation of gas and radiation damage generation for structural materials irradiated in fusion and fission reactors

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[2]*Institute of Physics and Power Engineering, Obninsk, Russian Federation*

Comprehensive neutronic analyses have been carried out for different reactor materials like the ferritic-martensitic steel (Fe-12Cr..), zirconium and vanadium alloys (Zr-1Nb, V-4Ti-4Cr) to assess hydrogen and helium generation and atomic displacement production in during of their irradiation in different neutron fields as the VVER-1000, BN-600, Bor-60, ITER, DEMO-RF, IFMEF and so on.. The analyses have been performed utilizing calculational models close to reality. The code FISPACT and cross-sections from the ACDAM data library have been used in the analyses. Theoretical estimates of total tritium, hydrogen, helium production rate and displacement formation have been obtained.

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Influence of Impurities and Alloying Elements on Nanostructure and Radiation Embrittlement of VVER Pressure Vessels Steels

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Russian Research Center "Kurchatov Institute", Moscow, Russia

The aim of this study is investigation of influence of impurity and alloying elements on nano-structure parameters and operating characteristics of pressure vessel steels at various irradiation stages for optimization of chemical composition of new pressure vessel steels.

Actual conditions of VVER pressure vessels operation induce considerable changes in the nanostructure of its materials, which are accompanied by phase transitions, formation of nanosize precipitates, radiation defects – dislocation loops, intergranular and intragranular impurity segregations. These structural changes lead to significant radiation embrittlement which appears in the shift of the ductile-to-brittle transition temperature (ΔT_k). The values of ΔT_k in otherwise identical irradiation conditions depend on the contents of impurity elements, such as phosphorus and copper, in the steel of VVER-440 pressure vessels and contents of alloying elements, first of all nickel – in VVER-1000. Electron-microscopy and fractofraphic structural studies showed that high nickel and low copper content in the steels of VVER-1000 pressure vessels in comparison with VVER-440 changed the evolution of nanostructure and relative contribution of different mechanisms to their radiation embrittlement. Early stages of irradiation of VVER-440 pressure vessel steels are dominated by radiation-induced precipitates, whose density grow with the irradiation dose first and then sharply drop. Closer to the design and beyond-design operation period values of neutron fluence, the main contribution to radiation hardening and embrittlement is made by dislocation loops, and radiation embrittlement of the base metal changes from transcrystalline to mainly intergranular embrittlement. As for base metal VVER-1000 – early stages of irradiation are characterized by steady monotonous growth of the radiation-induced precipitates and dislocation loops density. These changes in weld metal are 2–3 times more intensive than in the base metal. Upon reaching the design values of neutron fluence in weld metal, the density of dislocation loops jumps up abruptly, that results in linear growth of ΔTk without saturation typical for the base metal. Besides, the intergranular segregation of phosphorus, and, consequently, brittle intergranular fractures was observed even at low phosphorus content (less than 0.009%).

Thus to reduce the rate of radiation embrittlement of pressure vessel steels the content of impurities (Cu and P) and Ni through their harmful effect should be decreased.

Characterization and property evaluation of advanced cladding materials for fast reactors

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Abstract

The next generation fast reactor fuels are expected to reach a burnup greater than 20 at%. The cladding and duct materials for these fuels must be able to withstand significant radiation damage of the order of greater than 200 dpa. Until the 1970s, the primary material for fast reactor fuel cladding were austenitic stainless steels. Void swelling limits the use of the high swelling austenitic steels for fuel cladding. The appeal of the 9–12%Cr–Mo steels for in-core applications for cladding, for fast reactors was their better void swelling resistance, higher thermal conductivities and lower expansion coefficients than those of austenitic stainless steels. Among them, T91 alloy showed improved irradiation resistance primarily because of the lower carbon concentration. In particular, under irradiation conditions where HT9 develops an increase in the ductile-brittle transition temperature of 120–150°C at 36 dpa, T91 develops a shift of only 52–54°C for the same dpa.

In this study, the thermophysical properties like thermal diffusivity and thermal conductivity of fast reactor cladding materials like AISI 316 stainless steel, Ti modified stainless steel (alloy D9) and ferritic steel like T91alloys are studied in detail. The thermomechanical properties like hot hardness and tensile strength of the above was estimated using hot harness and automated ball indentation techniques, respectively. The creep properties are determined by impression creep technique. Since accurate thermal property data are needed from the point of view of enabling a confident design of engineering components, the above data will be useful for the fuel designers. Data on thermophysical properties are also required by the regulatory authorities to prove that fuel can be burned safely in the reactors. This poster deals with the evaluation of thermophysical and thermomechanical properties of advanced cladding materials developed in BARC and their properties are compared and presented.

Development of Process route for the production of Fe-0.12C-9Cr-2W-0.24 Y_2O_3 ODS alloy tubing for Indian FBR

(**P. K. Maity,** K.V.K.Deshpande, A. Surender, A. K. Dubey, S. K. Jha, Dr Kamal Kapoor,
C Phani Babu & B. Lakshminarayana)

Abstract

In the work of Nuclear Renaissance, India is playing key role in generation of clean & green Nuclear Energy. It has entered into its second stage Nuclear Power Program on commercial scale with the commencement of construction of 500 MWe Prototype Fast Breeder Reactor (PFBR) at Kalpakkam. Nuclear Fuel Complex (NFC), Hyderabad is playing a crucial role in the manufacture of all the critical sub-assemblies in D9 grade materials for this reactor.

The D9 material with controlled cold work is having very good void swelling resistance and high temperature properties, which can sustain up to 100 dpa. After successful development of production of D9 grade fuel clad tube, it is envisaged to use ODS alloy clad tube for the future FBR under planning. ODS alloy tubing is the promising clad tube material to achieve very high burn up, as ODS alloy clad tube can sustain up to 250dpa with excellent high temperature mechanical properties.

NFC has successfully demonstrated the trial production of these tubes. The paper covers the manufacturing process and characterization of the final tube. The manufacturing process includes powder canning, consolidation upsetting & extrusion of rods, machining of rods, cold reduction schedule considering the formability of the material, establishing annealing cycles and evaluation of mechanical and metallurgical properties. The experimentation with various percentages of reduction concluded that the reduction should be limited within 35% in each pass. This called for multipass reduction with intermediate heat treatment. Manufactured clad tubes exhibited an isotropic grain structure and equivalent tensile strength in both the direction. Further characterizations with respect to mechanical and nuclear properties are being carried out.

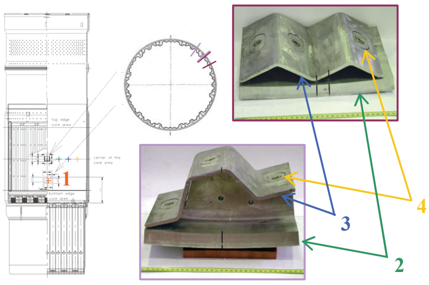
TEM Analysis of Irradiation Induced Defects in WWER Type Reactor Vessel after Long Term Service

Jan Michalička, Eliška Kejlová, Jan Kočík
Nuclear Research Institute plc., 250 68 Řež, Czech Republic

Introduction:

This poster is a part of experimental studies of NPP WWER-440 type reactor vessel internals material behaviour after long term in-service irradiation. Service period of the reactor was 15 years.
Studies were aimed to complex material characterizations of the unique non-active (referential) and active materials.
Non-active studied parts were core shroud basket, baffle and bolts.
Active studied parts were core barrel, core shroud basket, baffle and bolts.
Each of the part was made from Ti-stabilized austenitic steel 08Ch18N10T (A321 equivalent) common for reactor internals of WWER.
Radiation defects observed in structure belong to 3 groups: **dislocation loops, cavities, radiation induced precipitates and black dots.**

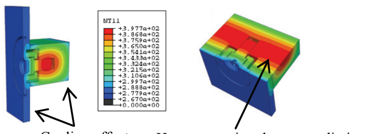
Arrangement – specimen preparation:



Evaluated conditions:

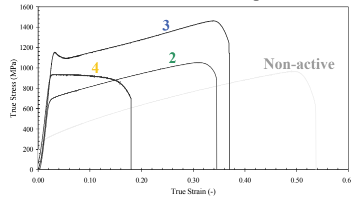
	Part of the vessel	dpa	E
1	Core barrel	2.4	> 0.5 MeV
2	Core shroud basket	5.2	
3	Baffle	5.2	
4	Bolts	11.4	

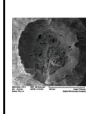
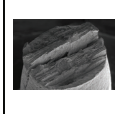
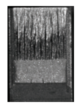
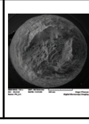
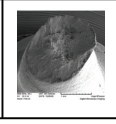
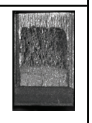
Simulated temperature field:



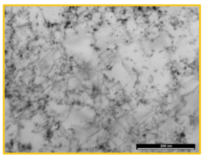
Basic mechanical and fracture properties:

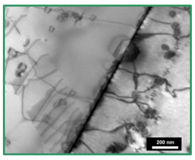
Tensile test of non-active and active specimens at 24°C

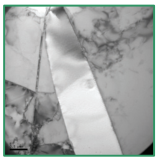


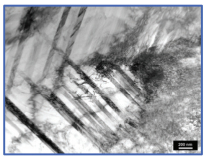
	Tensile test		Fracture toughness t.	
	0 dpa	11 dpa	0 dpa	11 dpa
24°C				
320°C				

Non-active material

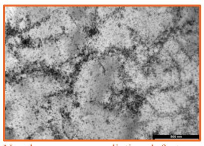

Dislocation substructure and fine particles of TiC

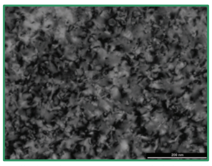

Fine particles TiC at dislocations and grain boundaries

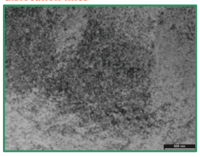

δ-ferrite grain

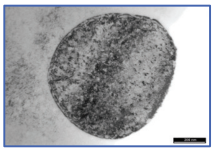

Cold-worked regions, deformation bands, dense dislocation tangles

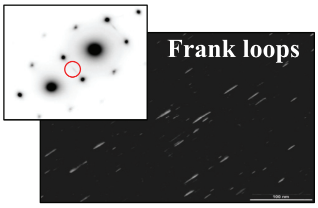
Active material


Non-homogenous radiation defects (caused by low fluence < 3 dpa) near subgrain and twin boundaries, dislocation walls and cells and dislocation lines

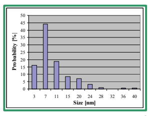

Radiation damage structure; bright-field, dynamical conditions; dense population of hardly discernible radiation-induced defects


In highly irradiated materials (> 5 dpa) the dense homogenous population of defects replaces original dislocation substructure

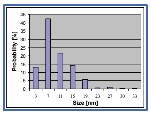

The δ-ferrite grain, dynamical conditions; radiation-induced defects: black dots, small precipitates of G phase, small loops and **no cavities**



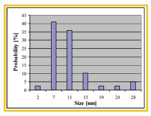
Frank loops shown in dark-field by so called **rel-rod technique** – visible are only loops in one (111) plane perpendicular to a foil surface („edge-on" position). This is useful for their statistical distribution evaluation.



Density: 5,72E+21 m^{-3}

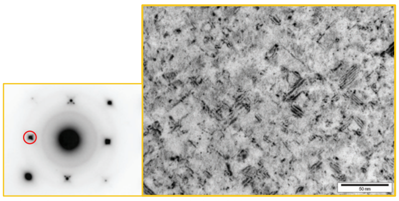


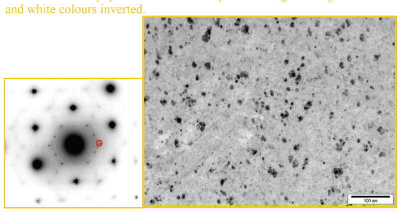
Density: 3,19E+21 m^{-3}



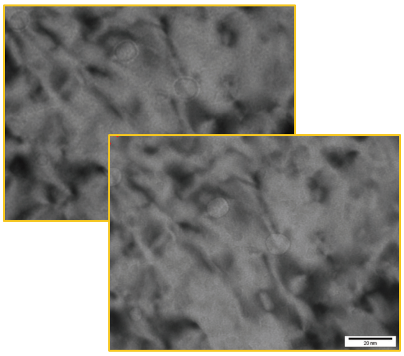
Density: 1,82E+21 m^{-3}

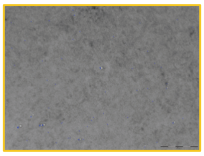
NOTE: In case of core barrel Frank loops were not evaluated.

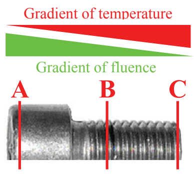

Selected area diffraction, z~[100], dark-field image in austenite reflection. There is shown a population of Frank loops containing stacking fault. Black and white colours inverted.

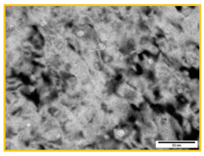

Selected area diffraction, z~[114], dark-field image in reflection of small dispersed precipitates of γ´ prime probably. Black and white colours inverted.

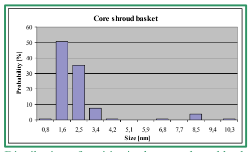
Cavities


Over / under focusing method to a cavities determination


Cavities in bolt from cross-section A

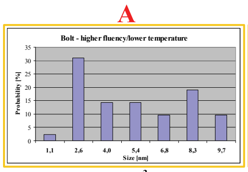



Cavities in bolt from cross-section C



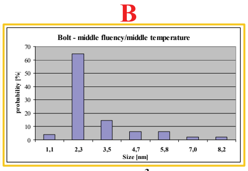
Distribution of cavities in the core shroud basket
Density: 6.93E+21 m^{-3}

A



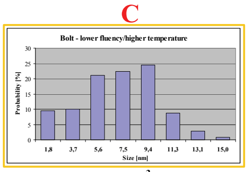
Density: 4.80E+21 m^{-3}

B

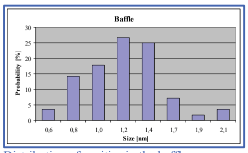


Density: 4.38E+21 m^{-3}

C



Density: 2.33E+21 m^{-3}



Distribution of cavities in the baffle
Density: 3.61E+22 m^{-3}

NOTE: No cavities were found in case of the core barrel – small fluence (2.4 dpa).

Conclusions:

- Neutron irradiation induces in the studied material after 15 years of in-servis reactor operation hardly discernible dense population of radiation-induced defects which replaces the original dislocation-stacking fault-twin microstructure
- Frank loops (faulted dislocation loops with Burgers vector 1/3[111] lying on the {111} planes) are the principal component of damage structure. In addition some perfect loops and some segments of dislocation lines are seen too.
- Radiation-induced particles of the new phases were observed; γ´ prime phase probably in the case of austenite, G phase in the case of δ-ferrite.
- Occurrence of cavities has been identified. The population of cavities 1 - 2 nm in size is homogeneous in most cases but large cavities 8 – 15 nm in diameter were observed in some cases (basket, bolds) too.
- Black-dot damage manifests the occurrence of very small defects ≈ 1 nm in size, they may represent small dislocation loops, stacking-fault tetrahedra, point defect clusters and precipitate embryos.

Nuclear reaction models for systematic analysis of the fast neutron induced (n,p) reaction cross sections

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A systematical study of known experimental cross sections for fast neutron induced (n,p) reactions is important to both nuclear reactor technology and the understanding of nuclear reaction mechanisms. In addition, it is often necessary in practice to evaluate the neutron cross sections of the nuclides for which no experimental data are available. Because of this, in last decade we carried out the systematic analysis of the known (n,p) cross sections and observed a systematic regularity in the wide energy interval of neutrons and
for the broad mass range of target nuclei [1]. However, consistent theoretical substantiation on existence of the systematical regularity for the (n,p) cross sections up to now no apparently is available. Several formulae have been suggested to describe the isotopic dependence of the (n,p) cross sections around the neutron energy 14-15 MeV, only. In this paper to explain the systematic regularity in (n,p) cross sections simple and
convenient formulae are deduced using the statistical model, exciton model of Griffin and PWBA. For the statistical model the constant nuclear temperature approximation and the evaporation model are used [2]. In the cases of the exciton model and PWBA the depletion factor, the nucleon emission rate [3] and Coulomb field [4] are neglected, respectively. It was shown that theoretical total (n,p) cross sections are satisfactorily in agreement with experimental values. Also, main conclusions of Levkovsky [5] for the isotopic effect at the 14.5 MeV were analyzed and expanded in the wide energy range of 6-16 MeV.

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Nanoring formation by ion irradiation and thermal annealing:

A comparative study

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Zinc sulfide has been extensively investigated as a luminescent material having various applications in opto-electronic and display devices. ZnS is a direct wide band gap (3.91 eV) compound semiconductor and shows outstanding properties in nano-dimensional structures. Thin films of SiO_2 with pre-determined fraction of ZnS were prepared by RF co-sputtering method. The thicknesses of the films were 170 nm as measured by ellipsometry method. The samples were irradiated at room temperature with 100 MeV Ni ions and the vacuum of irradiation chamber was of the order of 10^{-6} torr during the experiment. The films were irradiated normal to sample surface at different doses (Φ) ranging from 5 x 10^{11} to 5 x 10^{13} ions/cm^2. The pristine sample does not show any rings like structure on the surface of the film. On the other hand, the irradiated samples shows the nano-rings like structures on the surface. At low fluence, particles changed their shape from spherical to oblate perpendicular to the beam direction. As fluence increases, the formation of rings like structure has takes place. In case of thermal annealing, the films were annealed at 800 ^{0}C and 900 ^{0}C for different time. The formation of ring has been observed at temperature of 800 ^{0}C for 2 hours. In both cases, the ring formation is due to the agglomeration of the particles. For the agglomeration of the particles, the extra energy is given in form of thermal energy in case of annealing and electronic energy transferred by heavy ions in case of irradiation.

Multiscale Modeling Creep-Fatigue Damage in Irradiated Materials

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Abstract

The development of radiation resistant materials for high-temperature nuclear applications presents relevant design challenges. In ferritic/martensitic steels, one of the main difficulties arises from the synergic effects of possible failure mechanisms, such as low and high cycle fatigue, thermal and irradiation creep and creep-fatigue interaction. These failure mechanisms originate at the microscopic level (from the interaction between the stress field, dislocation
motion and radiation-induced point defects), and manifest at the mesoscale as intragranular and transgranular cracks and voids.

Rigorous modeling of this intrinsically multiscale damage accumulation process is a fundamental step in the development of radiation resistant polycrystalline materials at high temperatures. As opposed to phenomenological modeling, the approach proposed in this work incorporates the evolution of continuum dislocation and defect fields into a rigorous finite-deformation mechanics framework. The problem is formulated as a boundary initial value problem governed by a set of non-linear partial differential equations. The solution is found by a Finite Element Method that employs mesh adaptation in order to accurately resolve high gradients of the analyzed fields and eventual formation of defect patterns and stress localization. Material parameters and average reaction rates necessary for the proposed coarse-grained continuum description are obtained from ab initio, Molecular Dynamics, Kinetic Monte Carlo, and Dislocation Dynamics simulations. Some preliminary results in 2D are presented in this poster.