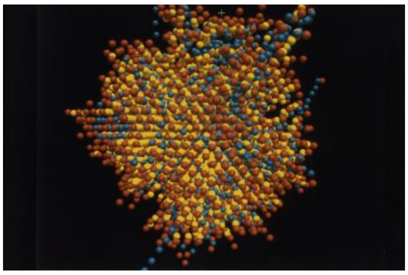
International School in Nuclear Engineering

A 5-day Doctoral Level Short Course on

MATERIALS FOR NUCLEAR REACTORS, FUELS AND STRUCTURES

October 6 to 10, 2014

Saclay – FRANCE



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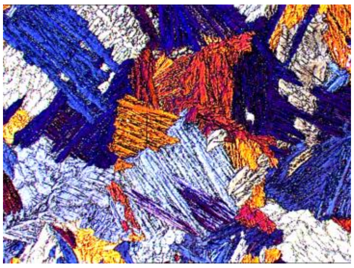

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Objectives of the course

The main objectives of this doctoral course are:

- to describe the mechanisms of irradiation damage
- to identify the available techniques to investigate the evolution of materials under irradiation
- to compare the properties of steels used for reactor core structures
- to explain the evolution of fuel and cladding during operation.
- to describe ceramic materials for GenIV reactors and investigate their structural and microstructural evolution under irradiation

Public

The course is designed for young researchers, PhD students, post-doctorates and engineers from nuclear industry companies, research centers, Universities, Technical Safety Organizations (TSO), regulatory bodies.

Pre-requisite

Minimum background: Master in Materials Science (basic knowledge in phase diagrams, diffusion, steels are strongly required).

Course content

I. Mechanisms of irradiation damage

- Description of the various irradiation sources in nuclear materials (charged particles, neutrons, photons); theoretical framework (binary collisions).
- Physics of the particle-solid interaction; specific case of ions: elastic and electronic slowing-down (statistical nature of processes, trajectories, collisions, range and range straggling, interaction cross section); use of the SRIM simulation code.
- Physics of the radiation damage: various type of defects; basic mechanism of defect creation by electronic and elastic processes; annealing of defects; theory of Kinchin and Pease (displacement threshold, collision cascades, number of displaced atoms); Coulomb explosion and thermal spike models; molecular dynamic examples.

II. Introduction to fuel materials

- Types of nuclear fuels (metallic, UO_2, MOX, for research applications); behaviour under irradiation: evolution of fuel composition, evolution of fission gases, formation of radiation defects, microstructural phase transformation (High Burn up Structure); radiation-enhanced diffusion and creep.

This section is reduced to a short introduction of the topics of concern. Details of the fuel behaviour are given in the lectures presented by J. Noirot and D. Parrat (module 2).

III. Behavior of in core metallic materials

- Steels for LWR, FBR and fusion:
  - ***Austenitic, ODS, 9%Cr martensitic steels for fuel Assembly in FBR :*** Metallurgy (phase diagrams, fabrication route and thermal treatments), mechanical properties, thermal ageing and behaviour under neutron irradiation (swelling, embrittlement ...)
  - ***Austenitic stainless steels for internals of LWR***: Irradiation hardening/embrittlement, IASCC...
  - ***Low activation 9Cr steels for fusion reactors***: Illustration of the "low activation properties", behavior under neutron irradiation (DBBT evolution...)
  - ***Ferritic steels for pressure vessel of LWR***: Embrittlement of pressure vessel steels, ductile-brittle transition temperature, irradiation effects and mechanisms, survey program and non destructive analyses.
- Zr alloys for Fuel Assembly in LWR:
  - ***Phase's diagrams***: binary phase diagrams including the main chemical alloying elements (Sn, Nb, Fe, Cr), oxygen and hydrogen, extension to "real" multi-alloyed industrial materials.
  - ***In service behaviour***: elaboration of Zr alloys, physical and mechanical properties before/after irradiation, growth under irradiation, irradiation creep, oxidation and hydriding.
  - ***Accidental conditions***: focusing mainly on LOCA behavior (influence of phase transformations, High Temperature (HT) creep properties, HT oxidation, secondary hydriding..., and relationship with the Post-Quenching mechanical properties.
  - ***Accident-tolerant fuel claddings*** *:* illustration of on-going R&D efforts on "enhanced-accident-tolerant-claddings" for LWRs, with the objective to increase significantly the materials high temperature mechanical resistance and/or to delay the high temperature steam oxidation, including coated Zr base claddings and SiC/SiC innovative clad concepts

IV. Carbides, SiC and ZrC high temperature materials for GenIV reactors

- Description of the ceramic materials for GenIV. Evaluation and qualification: required properties under irradiation. Simulation methods: experimental simulation of the ceramic behavior under irradiation by the use of energetic ion beams (radiation damage and doping).
- Presentation of the JANNuS facility. Main irradiation sources depending on the ceramic use (fuel, transmutation matrix, confinement matrix). Investigation of the structural and microstructural evolution under irradiation: (i) radiation effects due to elastic collision; (ii) radiation damage due to electronic interactions; (iii) synergy between the elastic and electronic processes.
- Examples of recent works performed on SiC, TiC and ZrC.

V. Technical tour

A tour of CEA nuclear materials laboratory is planned

Contacts - Registration

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