AN INTEGRATED APPROACH TO FUSION MATERIAL RESEARCH AT SCK•CEN

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SCK•CEN, Mol (Belgium)

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OUTLINE

- Introduction (short)
- Irradiation and mechanical characterization of EUROFER97
  - Tensile properties
  - Impact properties
  - Fracture toughness properties
- Environmentally assisted cracking in water and Pb-Li
- Multiscale modelling of radiation effects
  - Specific effects on Fe-Cr systems

INTRODUCTION

- Structural materials for fusion
  - superior mechanical and chemical behavior → safe operation of the reactor
  - low activation characteristics → minimization of environmental impact of produced waste
- RAFM steels investigated in EU, Japan and US
  - EU reference steel: EUROFER97 (DEMO design)
- Integrated approach at SCK•CEN, combining:
  - irradiation (base, ODS and joints)
  - characterization mechanical properties (PIE)
  - study of corrosion behaviour (EAC)
  - modelling of radiation damage at the atomic scale

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EU reference RAFM steel: EUROFER97

C	Si	Mn	P	S	Cr	Mo	Ni	V	W	Cu	Co
0.12	0.07	0.44	<0.005	0.004	8.99	<0.001	0.007	0.19	1.10	0.022	0.004
Ti	**Al**	**Nb**	**B**	**N**	**Pb**	**Ta**	**O**	**As**	**Sn**	**Zr**	**Sb**
0.009	0.008	<0.001	<0.001	0.017	<0.0003	0.14	0.0013	<0.005	<0.005	<0.005	<0.005

(weight %)

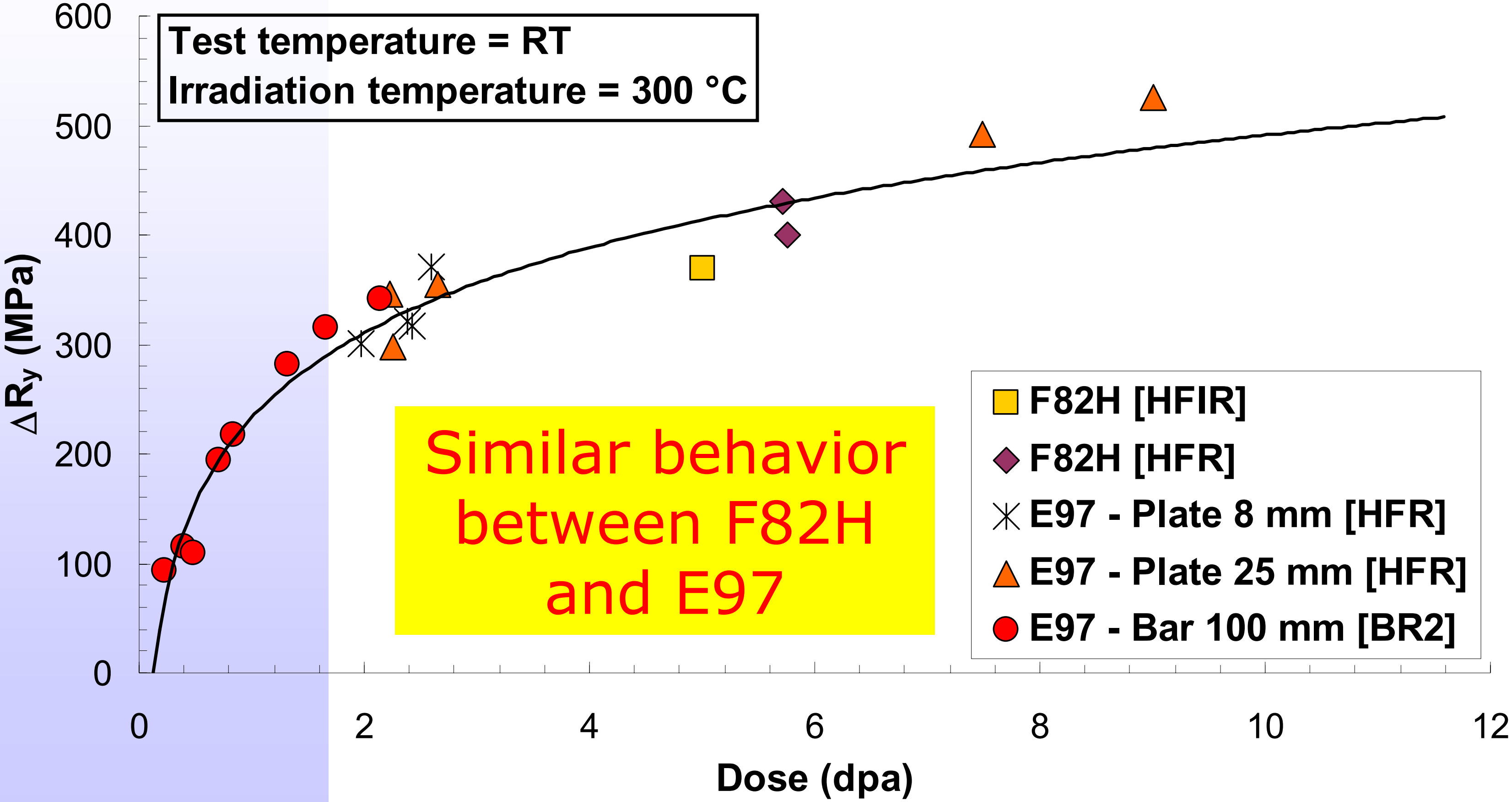
- Produced by Böhler (Germany)
- Heat treatment:
  - normalisation 980 °C
  - tempering 740 °C/3.7 h + air cooling
- Product form: bars with D = 100 mm

Neutron Irradiations

- Test reactor: BR2 (Belgian Reactor 2) in Mol
- Three irradiation campaigns:
  - IRFUMA-I (2000); 0.3 dpa
  - IRFUMA-II (2001-2002); 1 dpa
  - IRFUMA-III (2002-2003); 1.7 dpa
- Irradiation conditions:
  - T = 300 °C
  - Flowing water
  - Extensive dosimetry
- Subsequent PIE (tensile, impact, toughness, EAC) + characterization unirradiated state

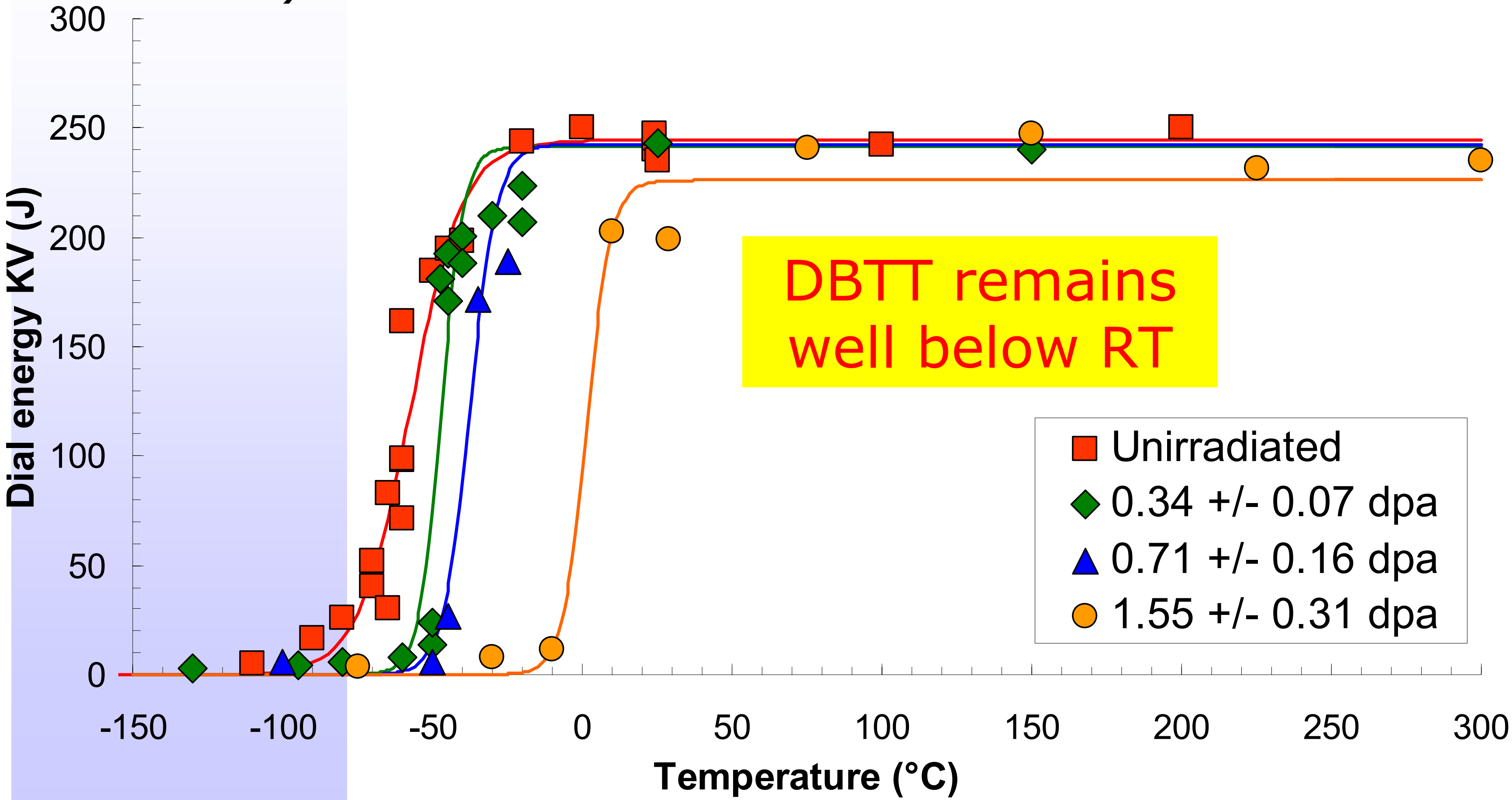
Tensile properties

- Hardening (yield and UTS increase) and loss of ductility (decrease of elongation)



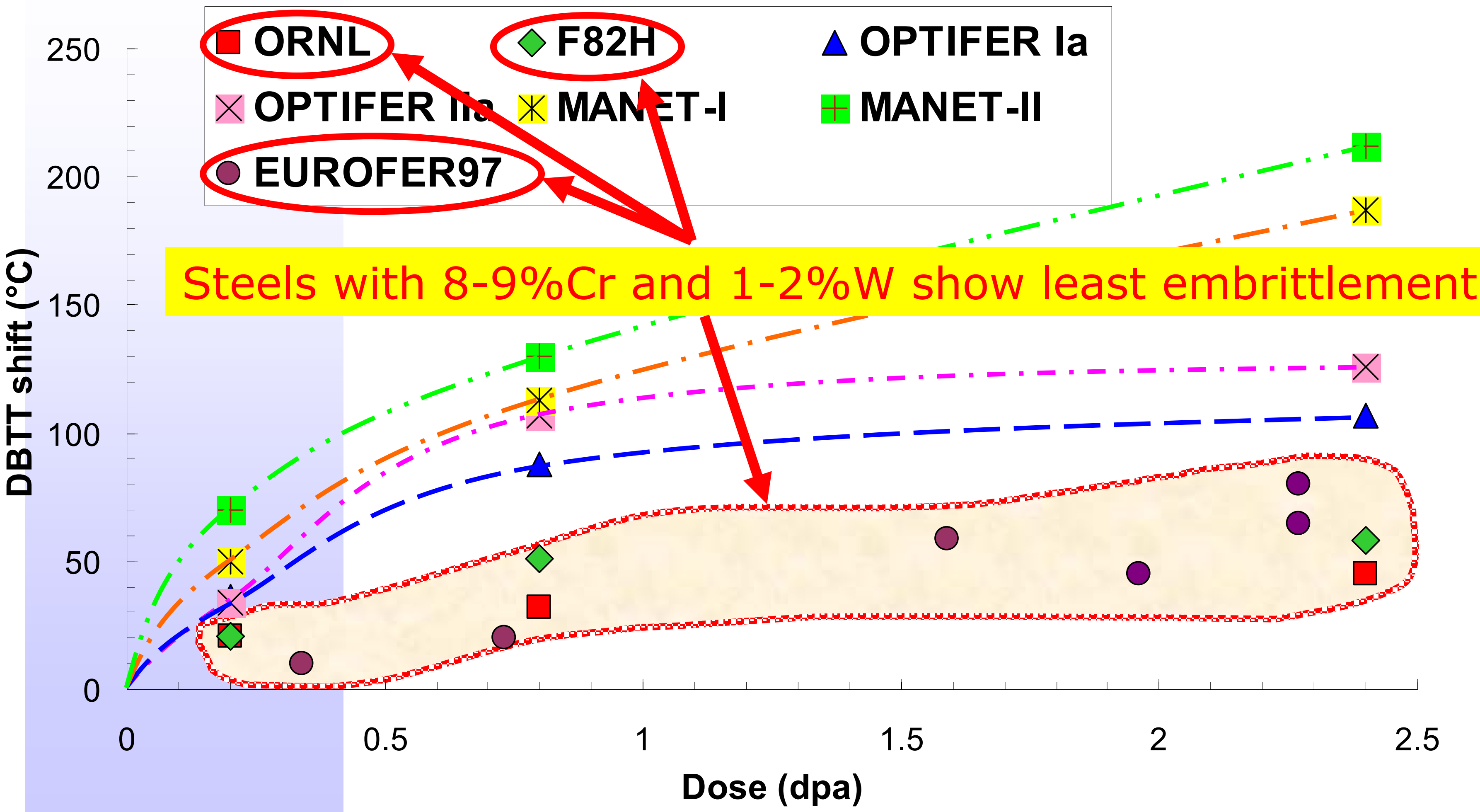
Impact properties (I)

- Material embrittlement (DBTT shift and moderate loss of USE)



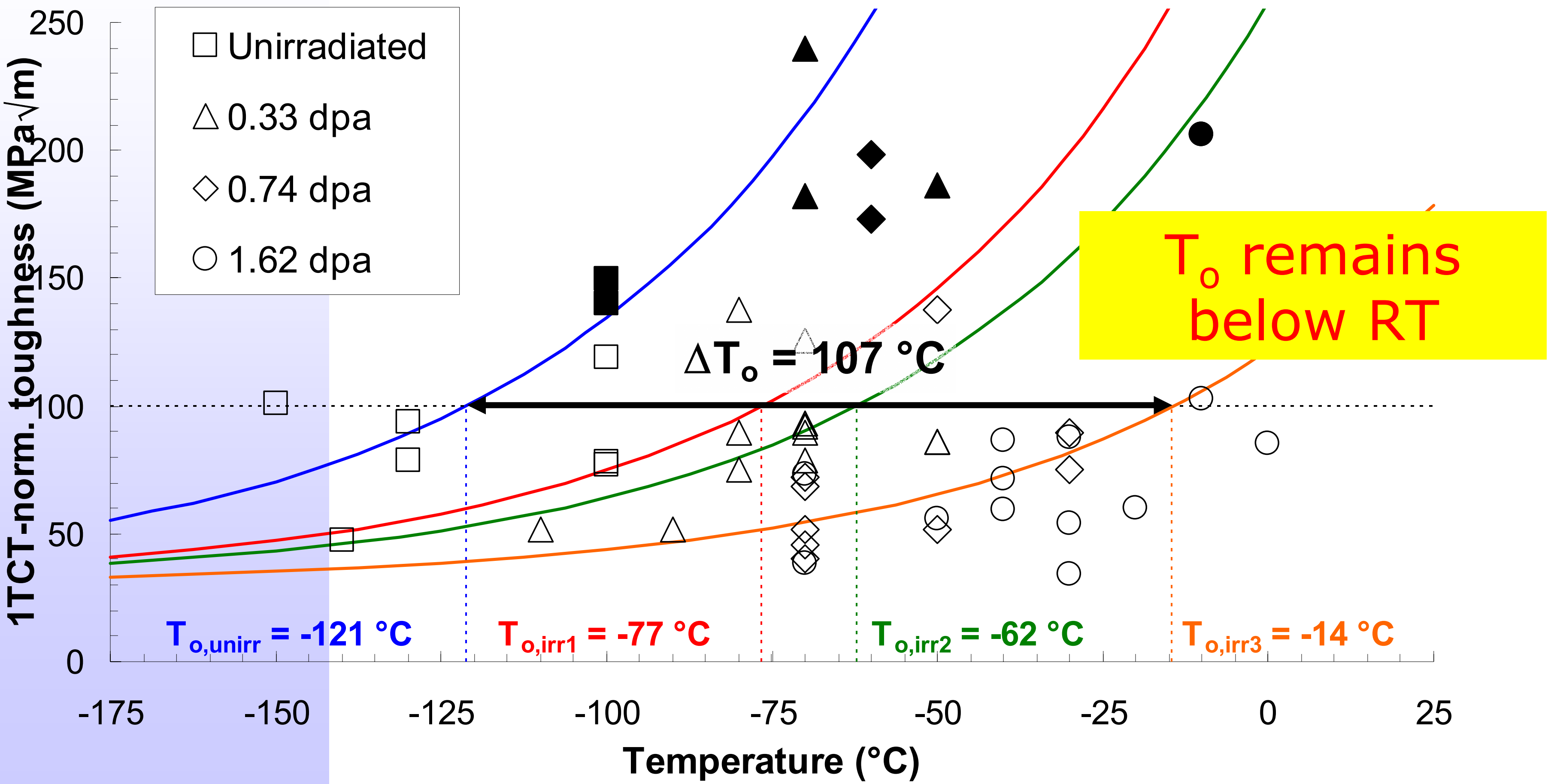
Impact properties (II)

➢ Comparison with other RAFM steels, all irradiated at 300 °C



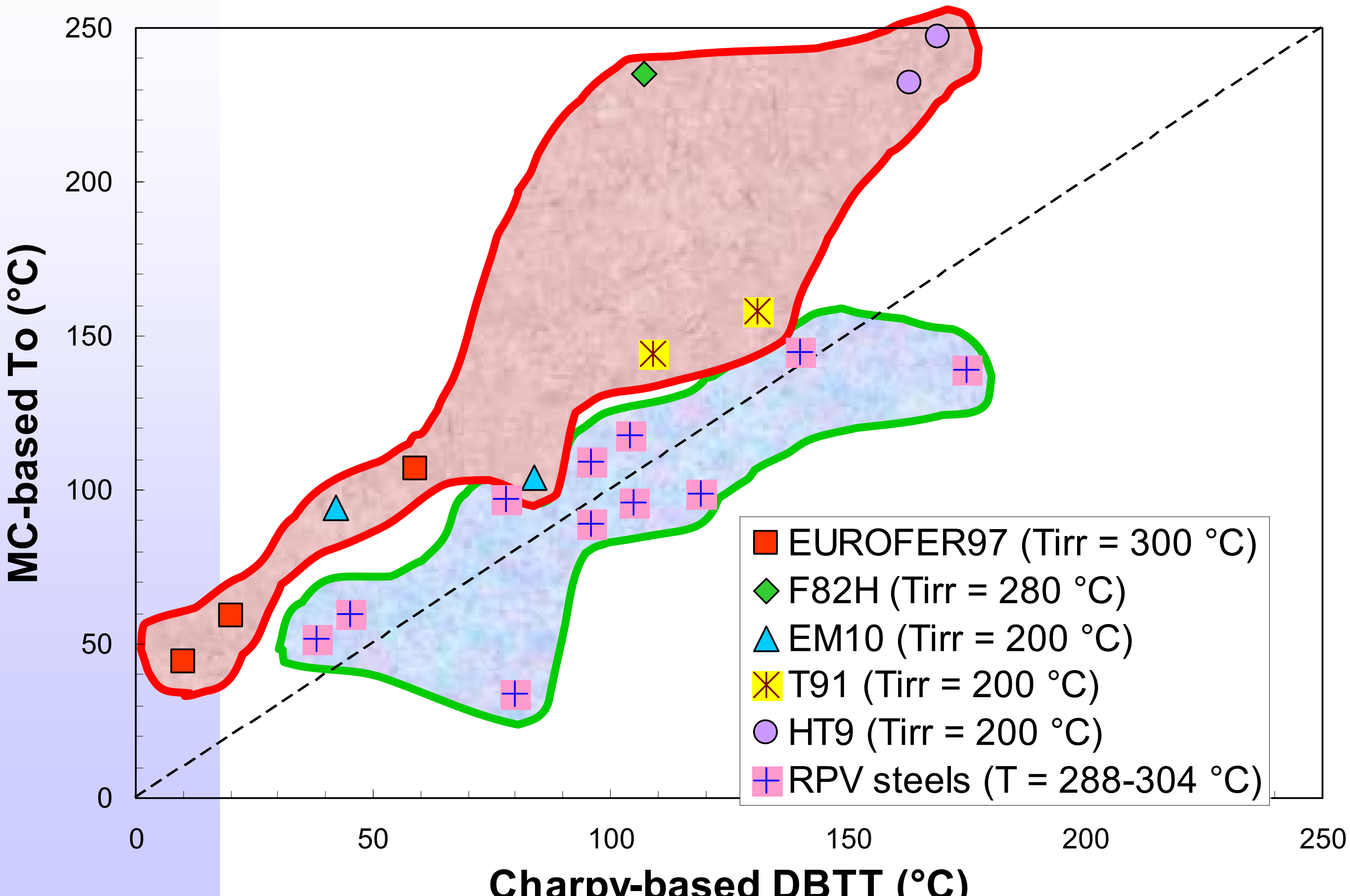
Fracture toughness properties (I)

- Material embrittlement (increase of reference temperature with accumulated dose)



Charpy VS Fracture Toughness: safety implications?

- T_o shifts are consistently and significantly larger than DBTT shifts: observed on F/M steels, but not on RPVS



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Future irradiations & PIE

- IRFUMA-IV campaign
  - Materials: E97 welded joints, E97 ODS
  - Irradiation conditions: T = 300 °C, target dose ~ 1.7 dpa
  - Irradiation planned July 2004-November 2005
  - PIE planned first half of 2006
  - Specimens irradiated:
    - ✓ Sub-size tensile
    - ✓ Sub-size Charpy (KLST)
    - ✓ Precracked sub-size Charpy (PKLST)
    - ✓ Slow Strain Rate Testing (SSRT)

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Environmentally Assisted Cracking in Water and Pb-Li

- Investigation of E97 compatibility with possible fusion reactor environments:
  - water at high temperature
  - liquid Pb-Li eutectic alloys
- Studies have shown that SCC is enhanced by:
  - chloride addition
  - electrochemical potential increase (pitting)
  - increase in hardness and strength
  - Hydrogen embrittlement

Influence of irradiation on the corrosion behaviour of E97

- Occurrence of EAC depends on:
  - environment → increase in potential and corrosion rate
  - material properties → hardening and microchemical modifications induced by irradiation (**IASCC**)
- It's expected that irradiation hardening will enhance the material's susceptibility to LME (Liquid Metal Embrittlement)

Experimental investigation of IASCC in water

- SSRT samples irradiated at 300 °C in BR2
- Test conditions:
  - T = 100 and 300 °C
  - Environments: air, oxygenated water, hydrogenated water
- Post-test SEM examinations (fracture mode)
- Main results:
  - Clear hardening effect
  - No large influence of environment (tendency for flow localization slightly higher in water)
  - Fracture surfaces: general ductile failure, no SCC (well-controlled water chemistry)

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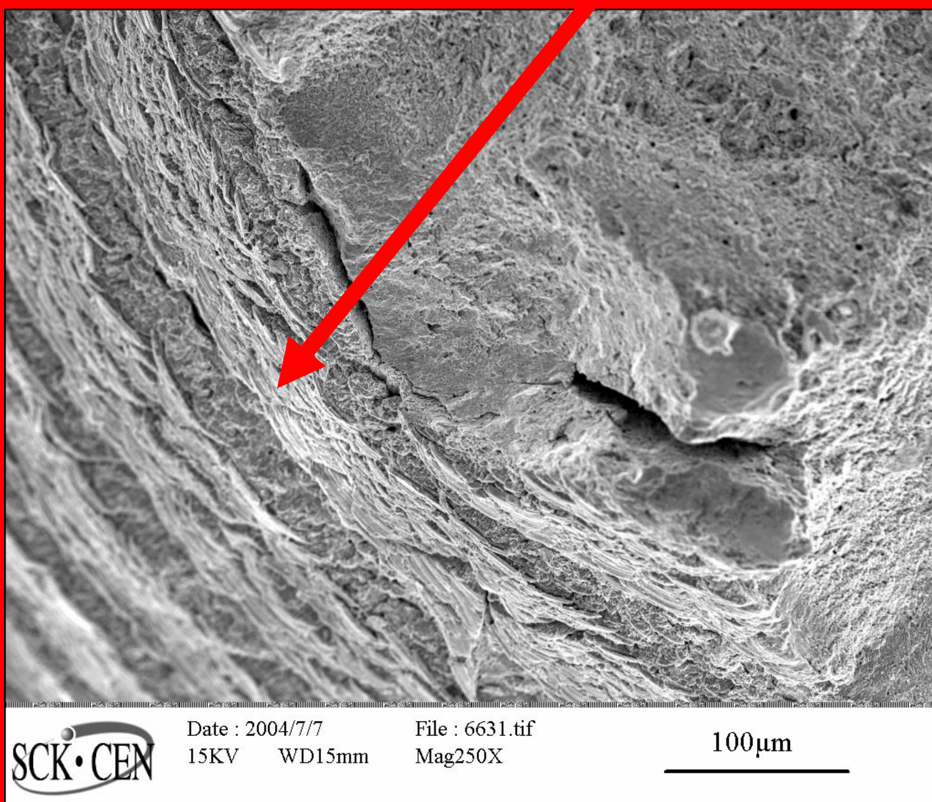
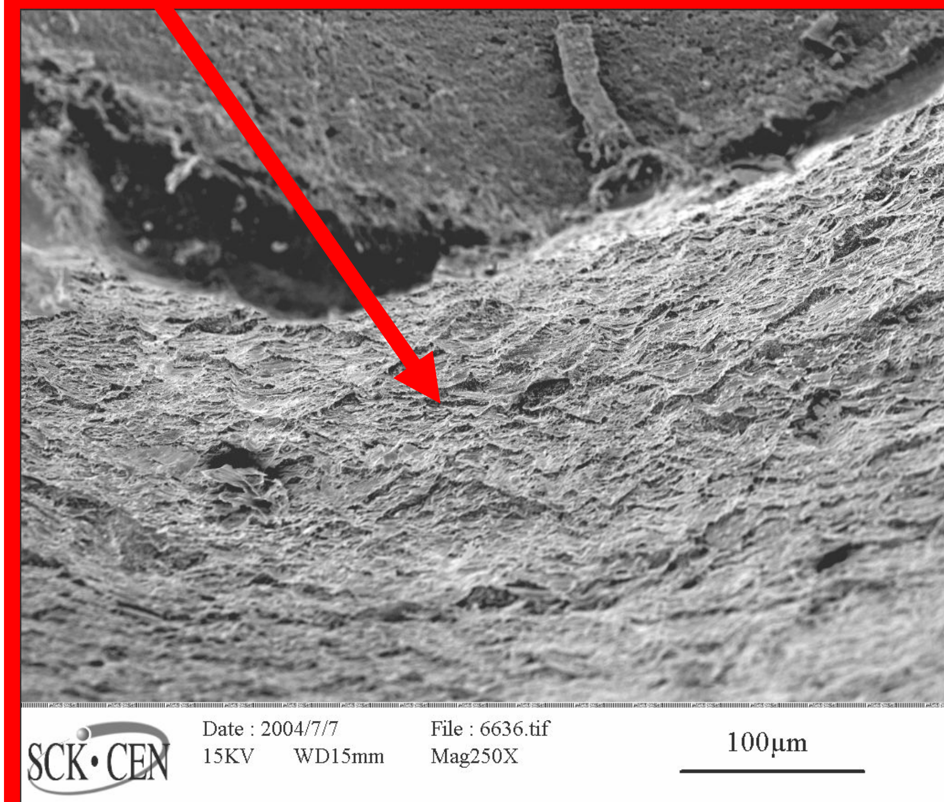
SEM observations of fracture surfaces

- 45° grooves observed (plastic flow localization)
- Evolving to "fish-scale" network for higher doses





Future activities: irradiation effects on LME

- SSRT tests planned on samples submerged in liquid Pb-Li
- Critical parameters to be investigated:
  - temperature (between melting point Pb-Li and T_{irr})
  - strain rate (from 10^{-2} s^{-1} to 10^{-5} s^{-1})
  - pre-wetting time (time of exposure to PbLi before straining specimen)
- Materials considered:
  - EUROFER97 base (irradiated in IRFUMA-III, 2002-2003)
  - EUROFER97 ODS (irradiated in IRFUMA-IV, 2004-2005)

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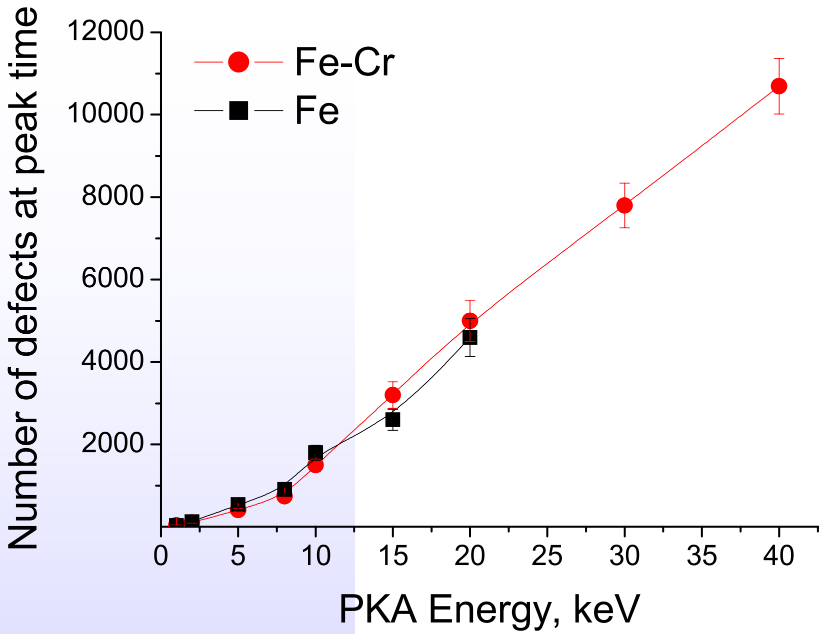
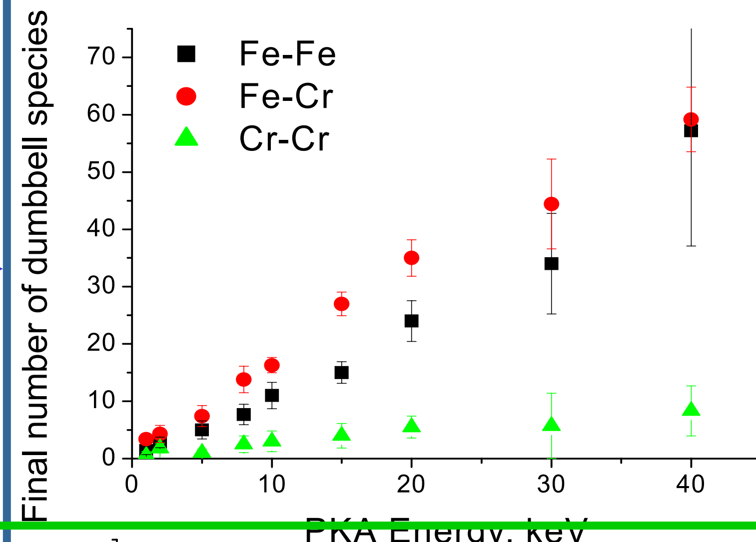
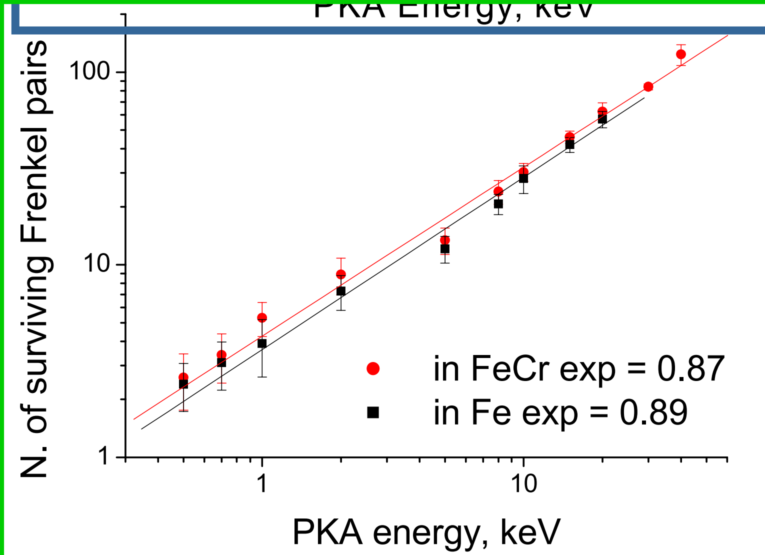
Multiscale Modelling
Atomistic Simulations

- Computer simulation (multiscale modelling) tools are being used all over the world to help understand the behaviour of materials under irradiation from a fundamental point of view (atomic-level)
- At SCK•CEN, two studies are being performed:
  - Molecular dynamics simulation of displacement cascades in Fe-Cr alloys
  - Interstitial motion in Fe-Cr alloys

Cascades in Fe and Fe-Cr (I)
Main results

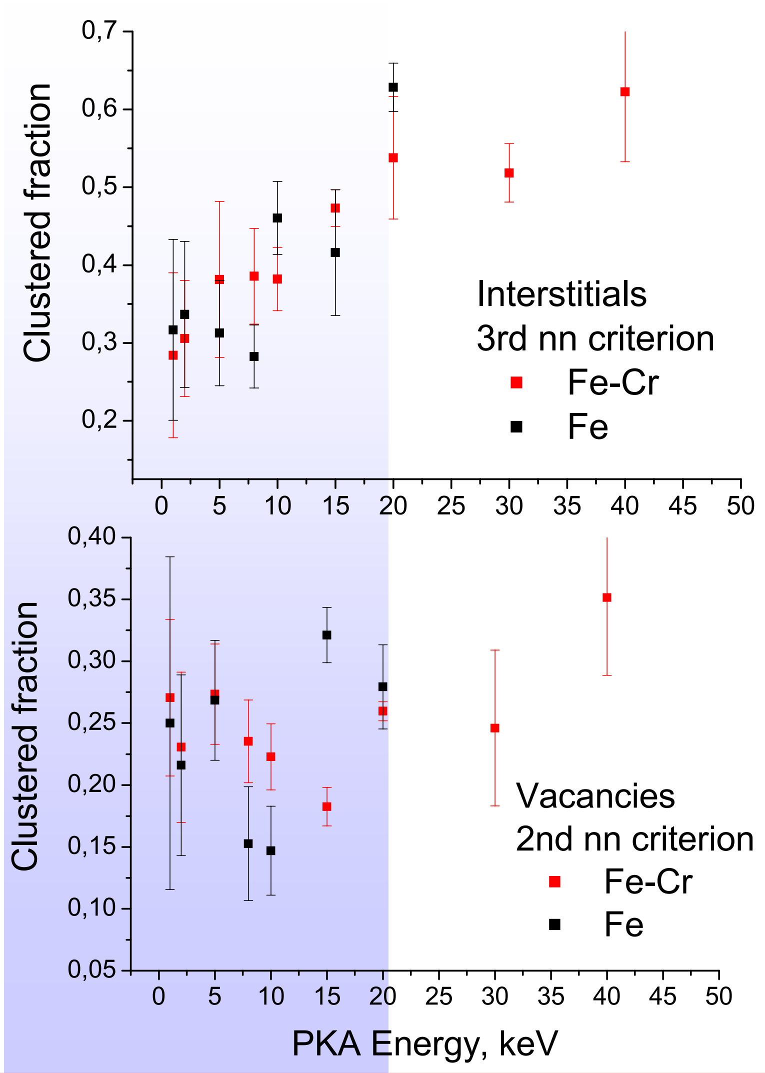


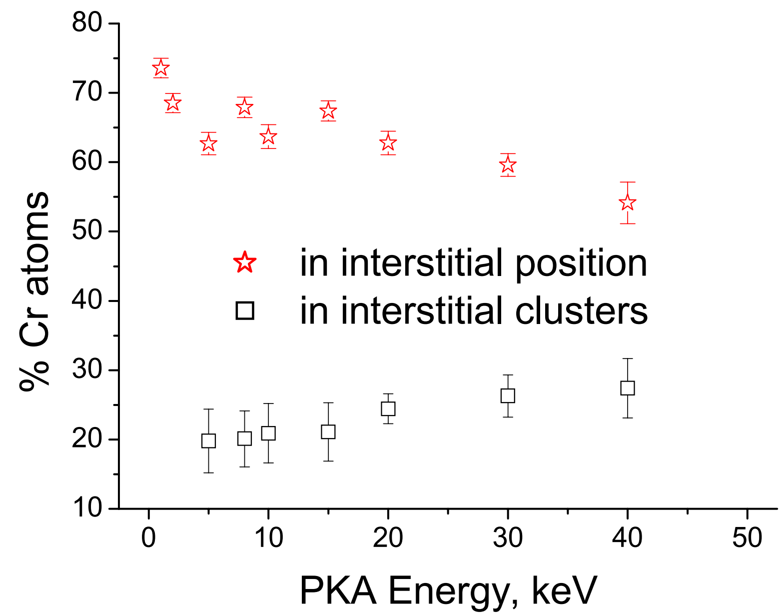




- **No influence of Cr atoms on collisional phase**
- **Post-collisional dumbbell rearrangement: ~60% of the interstitial atoms are Cr atoms**
- **Slightly reduced recombination due to Cr interstitial stability: somewhat higher defect production efficiency in Fe-Cr**

Cascades in Fe and Fe-Cr (II)
Main results





- **No influence of Cr atoms on clustered fraction**

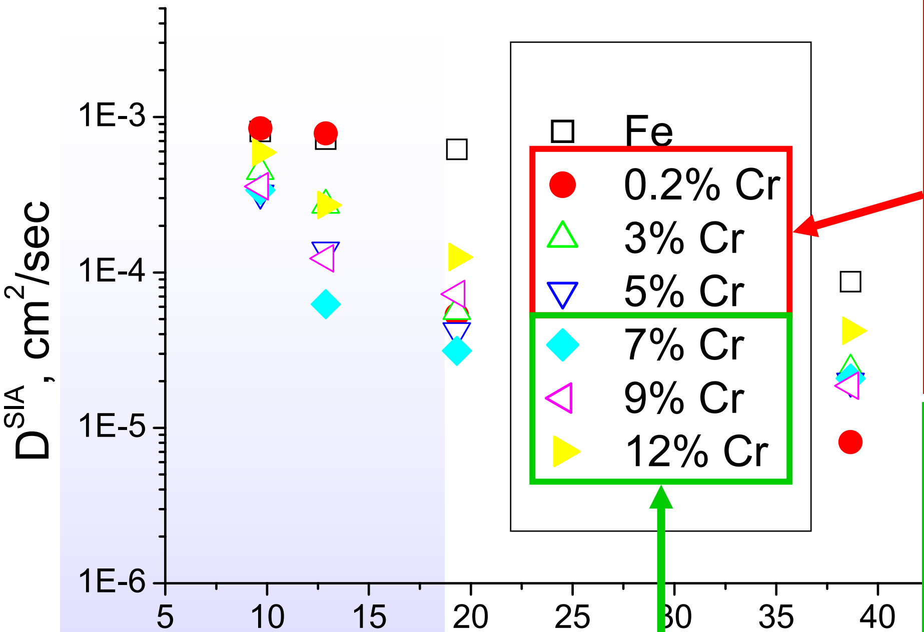
- **Most isolated interstitials are Cr atoms**
- **Yet, interstitial clusters contain a higher concentration of Cr than the alloy, growing with increasing PKA energy!**

Cascades in Fe and Fe-Cr
Some conclusions

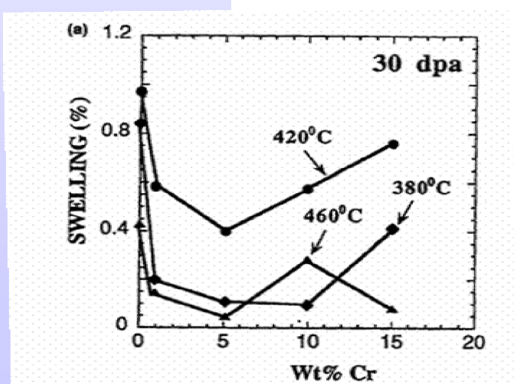
- Main effects of the presence of Cr on primary damage state are that:
  - most interstitial atoms are Cr atoms
  - interstitial clusters contain a fraction of Cr atoms larger than the alloy concentration
- Mixed interstitial clusters are expected to have different mobility from self-interstitial clusters, as supported by experimental observation (**K. Arakawa et al. J. Nucl. Mater. 229-233 (2004) ...**)

⇒ The focus should be moved to the study of the mobility of interstitials and interstitial loops in concentrated (3-12%) Fe-Cr alloys

Mobility of interstitials in Fe-Cr
Cr concentration effect (preliminary)



- <u>Low concentration</u>: pure trapping effect, at low T self-interstitial atoms (SIA) are trapped at Cr atoms and diffusivity is reduced; effect disappears at high T
- <u>High concentration</u>: "jumping from Cr to Cr" the SIA reduces the binding energy to Cr atoms to an effective value, lower than for low concentration: only slight reduction of diffusivity
- Most effective diffusivity reduction for 7% Cr (with this potential …)



F. Garner et al. JNM 276 (2000) 123