Radiation material science. Neutrons vs heavy ions

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- Irradiation parameters in nuclear facilities
- Main Candidate Materials
- Material research in spallation environment

Irradiation parameters in nuclear facilities

Reactor type	**Parameters**		
	Damage rate	Helium generation rate	Hydrogen generation rate
Thermal reactors	3 dpa/year	~ 280 appm/year	~ 60 appm/year
Fast reactors	30…40 dpa/year	20…30 appm/year	
Fusion reactors	20 dpa/year	300 appm/year	800 appm/year
Reactors of IV generation, **electro nuclear systems**	5…40 dpa/year	950…3500 appm/year	3000…4000 appm/year

Radiation damage dose: dpa - displacement per atom

Irradiation parameters in nuclear facilities

Defect poduction (in steels)	Fusion neutrons (3-4 GW reactor)	Fission neutrons (BOR 60 reactor)	Mixed spectrum of high energy protons and spallation neutrons (SINQ)
Damage rate [dpa/year]	20-30	~ 20	~ 10
Helium [appm/dpa]	10-15	~ 1	~ 50
Hydrogen [appm/dpa]	40-50	~ 10	~ 450

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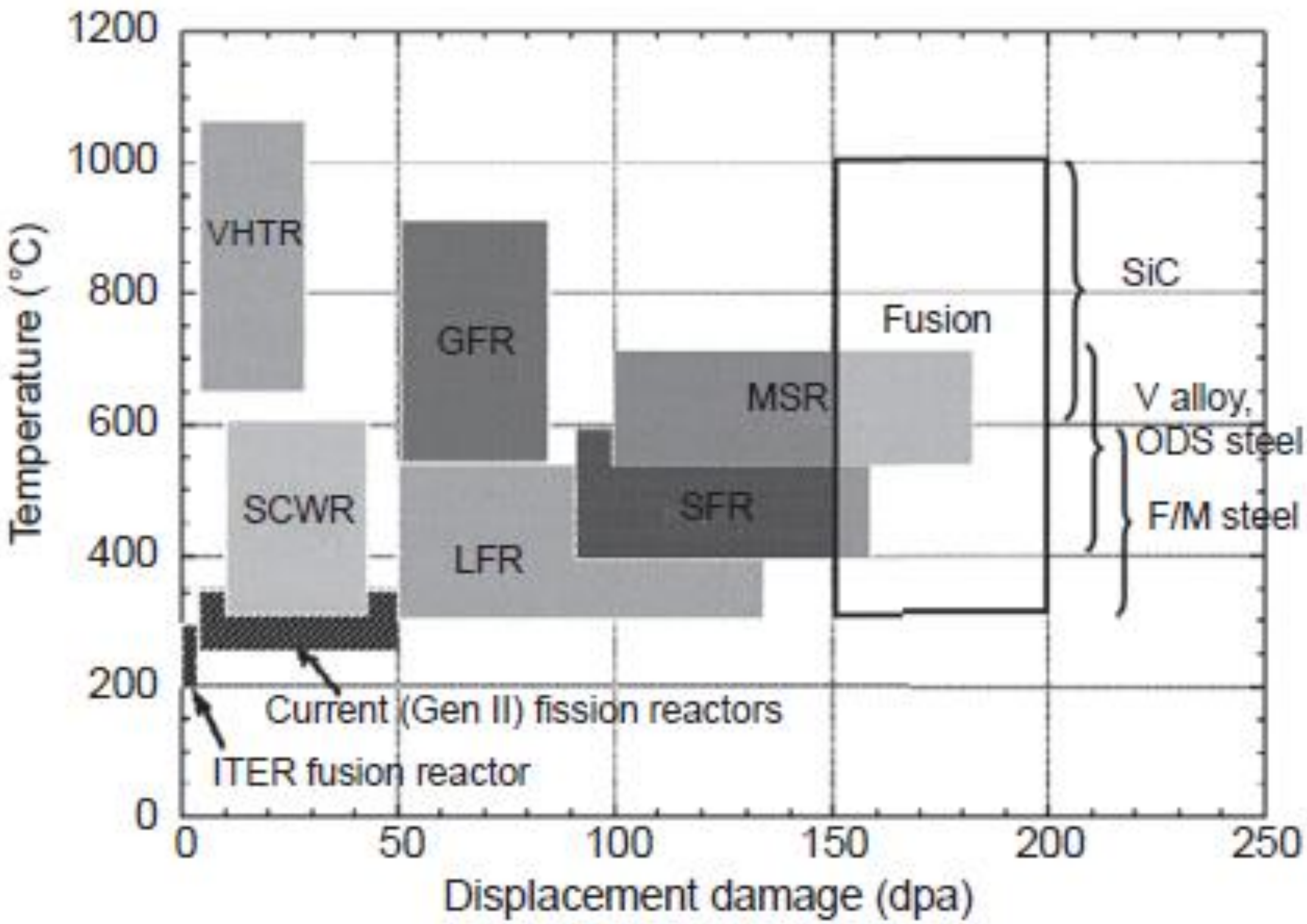
Summary of operating parameters for fusion, fission and spallation facilities

Parameters	Technology		
	Fusion	Fission (Generation IV)	Spallation
Energy	<14 MeV	<1-2 MeV (most n's)	≤ 1MeV GeV (p and n)
He/dpa	10	0.1-50	50-100
Stresses	Moderate, slowly varying	Moderate, slowly varying	High, pulsed

L.K. Mansur et al. / Journal of Nuclear Materials 329–333 (2004) 166–172

Operating temperatures and displacement damage dose regimens for structural materials in generation II and Generation IV reactors



S.J. Zinkle, J.T. Busby, Mater. Today 12 (11) (2009) 12-19

Main phenomena limited long-term radiation stability of reactor materials

- Void Swelling

- Irradiation Creep

- Radiation Embrittlement/Helium Embrittlement

Specific types of hydrogen-induced damage of metals and alloys:

- hydrogen embrittlement;
- cracking from precipitation of internal hydrogen;
- cracking from hydride formation;
- hydrogen-induced blistering

Main Candidate Materials

Main candidate materials have a chemical composition that is based on low activation Fe, Cr, V, Ti, W, Si, C

- Reduced activation ferritic/martensitic (RAFM) steels
- Oxide dispersion strengthened (ODS) RAFM steels
- Oxide dispersion strengthened RAF steels
- Vanadium-base alloys
- C/C, SiC/C, SiC/SiC ceramic composites

ODS = Oxide dispersion strengthened alloys

Ferritic matrix + 5÷50 nm size thermally stable oxides dispersed within it

Strengthening principle in ODS alloys:
Nanoparticles are obstacles to dislocation glide

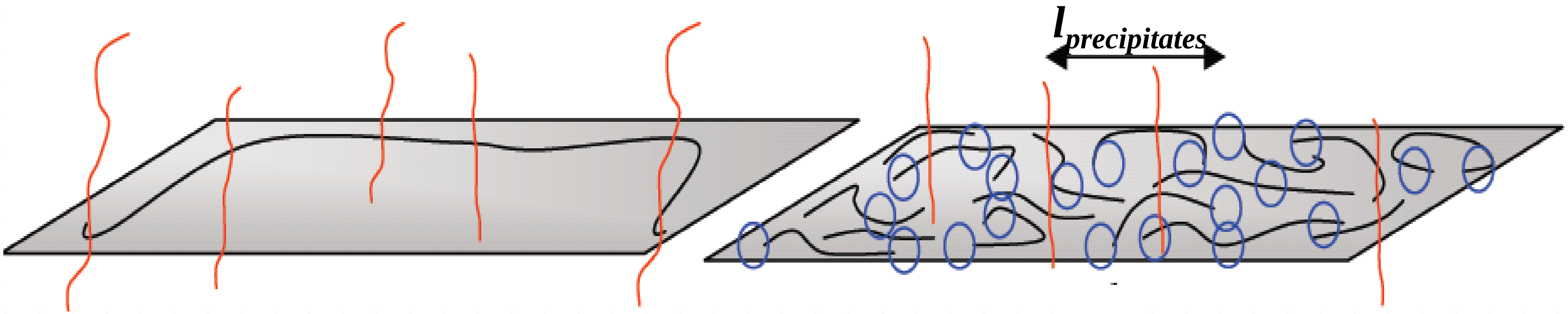
ODS steels are promising candidates for fuel cladding

Strengthening of alloys: ODS principle

Increase obstacles to dislocation glide

- Precipitates or other dislocations
- Finer dispersoides and higher number density

$$\Delta\sigma \propto \quad A + \frac{2\,\mu\, b}{l_{precitates}}$$

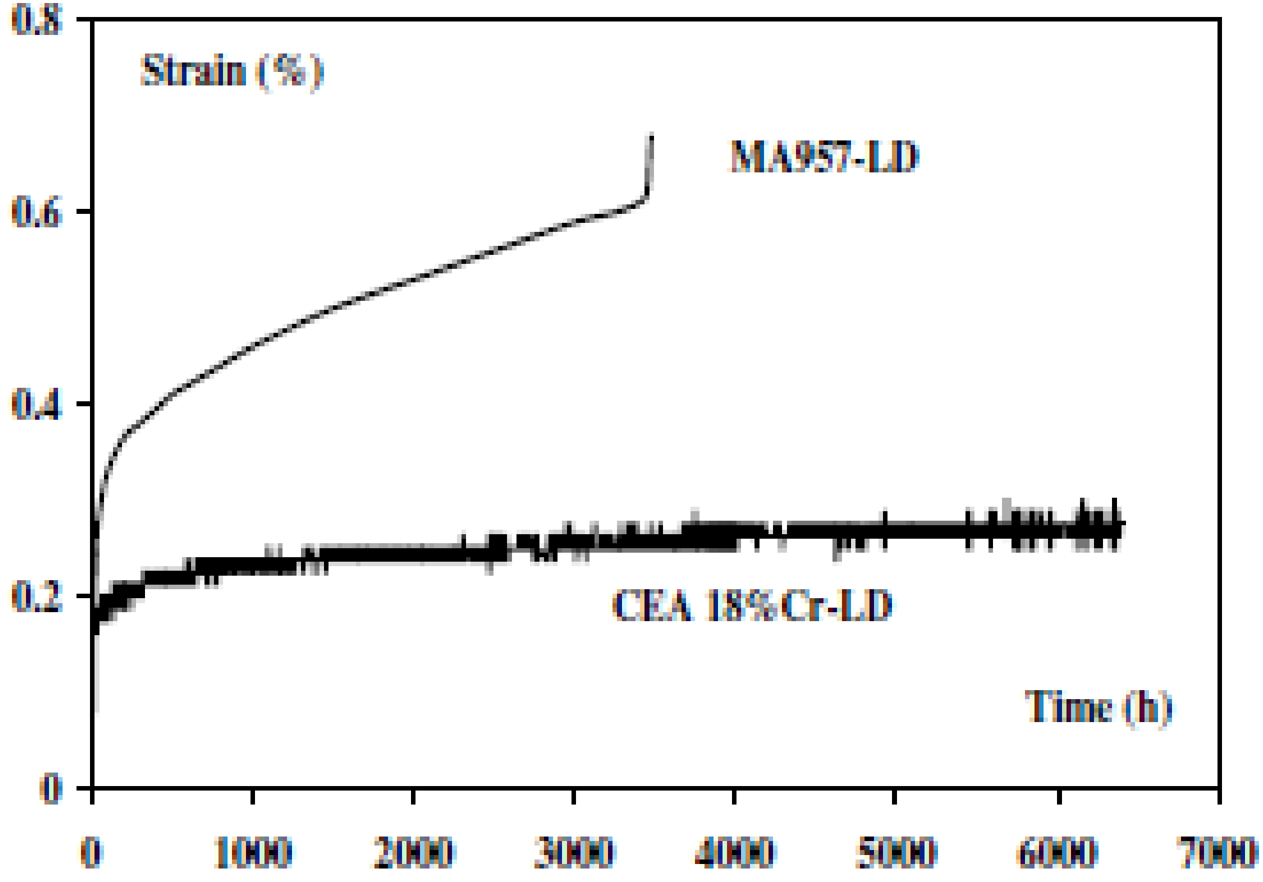


Clement, CEA

Example of commercial ODS

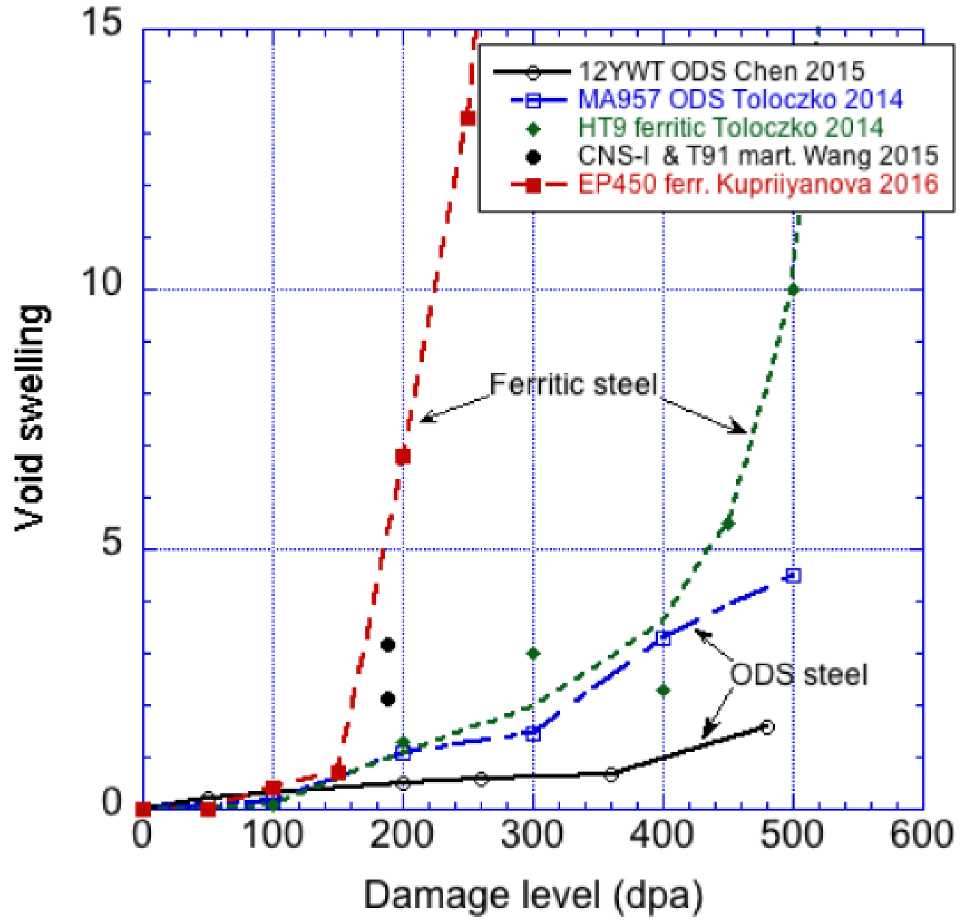
chromia-forming **alumina-forming**

	from	Fe	Ni	Cr	Al	Ti	Mo	W	others	C	Y_2O_3
MA 956	INCO	base		20	4,5	0,5					0,5
PM 2000	Plansee	base		20	5,5	0,5					0,5
ODM 751	Dour Metal	base		16,5	4,5	0,6	1,5				0,5
MA 957	INCO	base		14		1	0,3				0,25
MA758	INCO		base	30	0,3	0,5				0,05	0,6
MA754	INCO		base	20	0,3	0,5				0,05	0,6
PM 1000	Plansee		base	20	0,3	0,5					0,6
MA760	INCO		base	20	6		2	3,5	Zr 0,15	0,05	0,95
PM 3030	Plansee		base	17	6		2	3,5	Ta 2 Si 0,95		1,1
MA757E	INCO	0,5	base	16,8	4	0,5				0,06	0,7
HDA-8077	Cabot		base	15,7	4,2					0,06	Y :1,6
MA6000 (γ')	INCO		base	15	4,5	2	2	2	Ta 2 Zr 0,15	0,05	1,1
MA753 (γ')	INCO		base	20	1	2,2				0,05	1,3



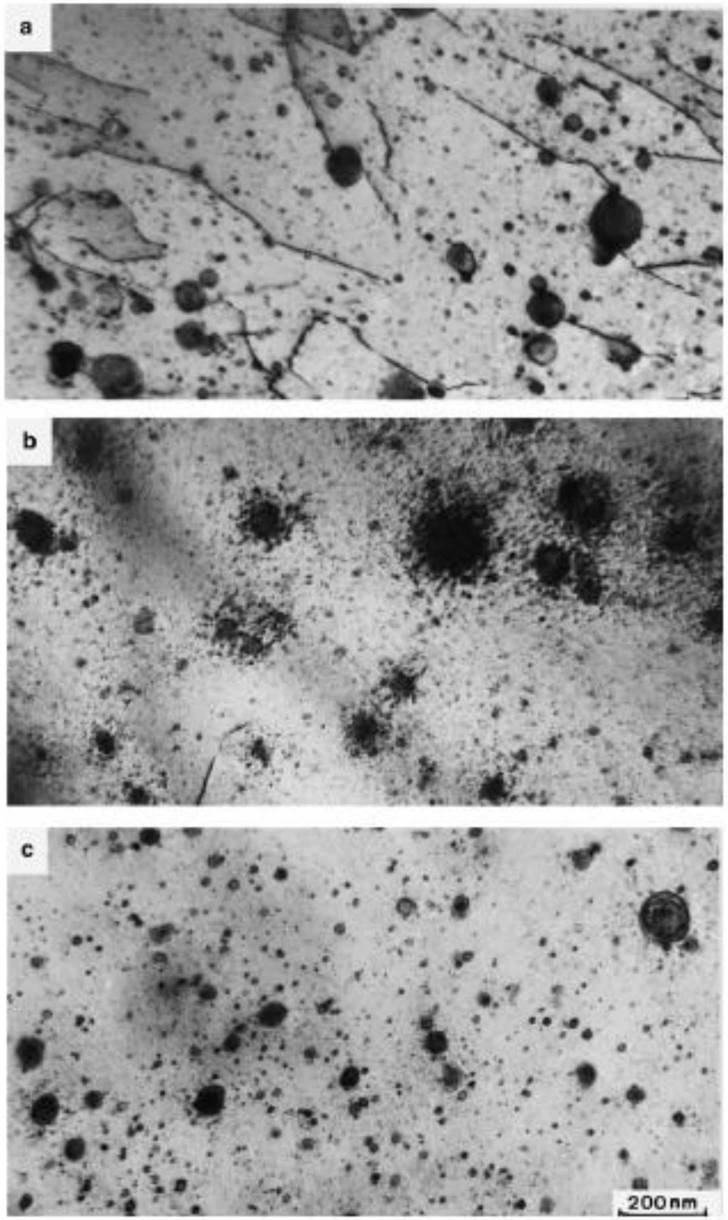
Thermal creep of MA957 and 18%Cr-ODS alloys at 650º C / 250 MPa

J. Malaplate et al., Journal of Nuclear Materials 417 (2011) 205–208



Comparison of void swelling at 450-480^{o}C in conventional ferritic/martensitic and ODS steels

S. Zinkle et. al. Development of Next Generation Tempered and ODS Reduced Activation Ferritic/Martensitic Steels for Fusion Energy Applications



Eurofer ODS steel

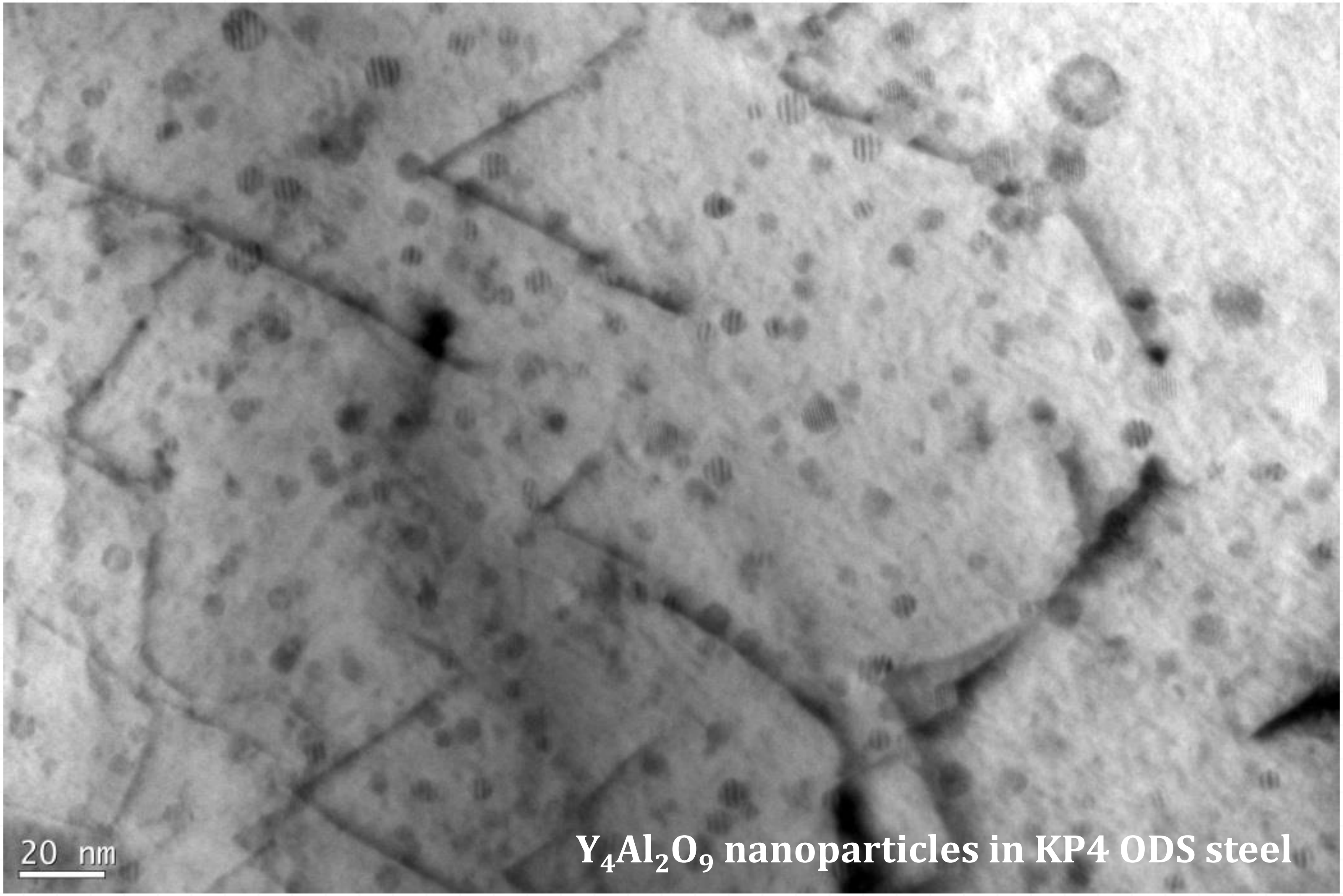
unirradiated

1 MeV He irradiated to 78.8 dpa at 532°C

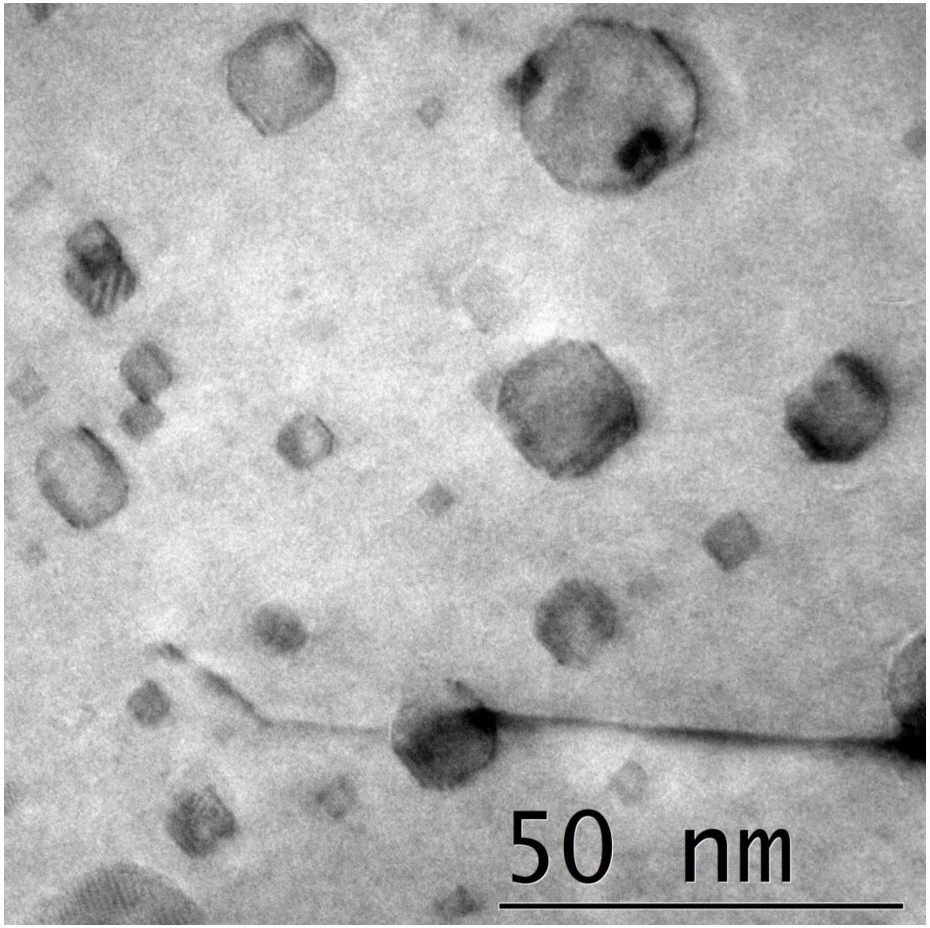
1 MeV He irradiated to30.5 dpa at 580° C

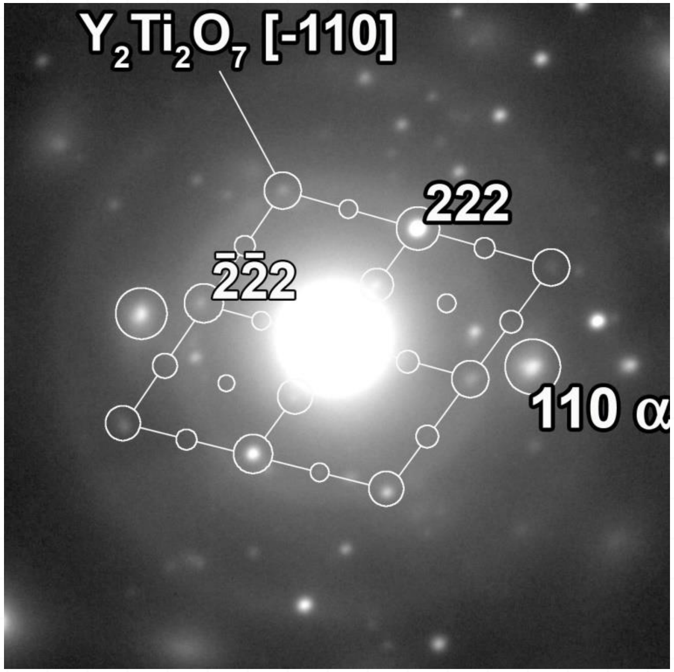
I. Monnet et al., Journal of Nuclear Materials 335 (2004) 311–321

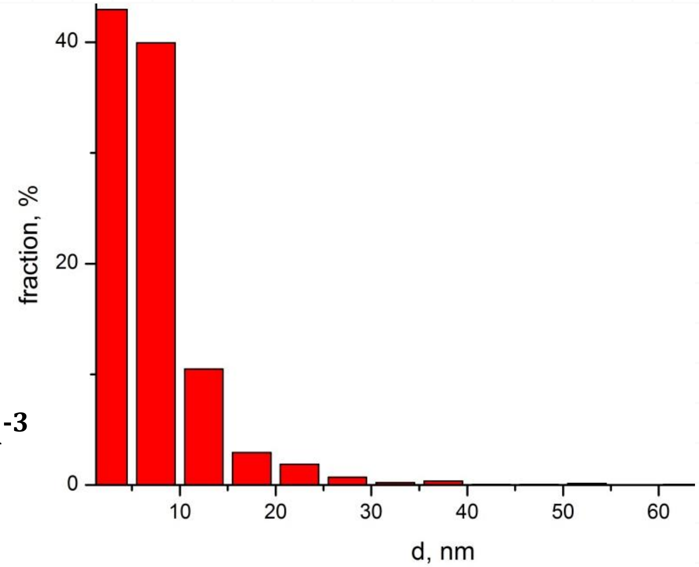
Radiation stability of ODS alloys against fission fragment impact



EP450 ODS alloy





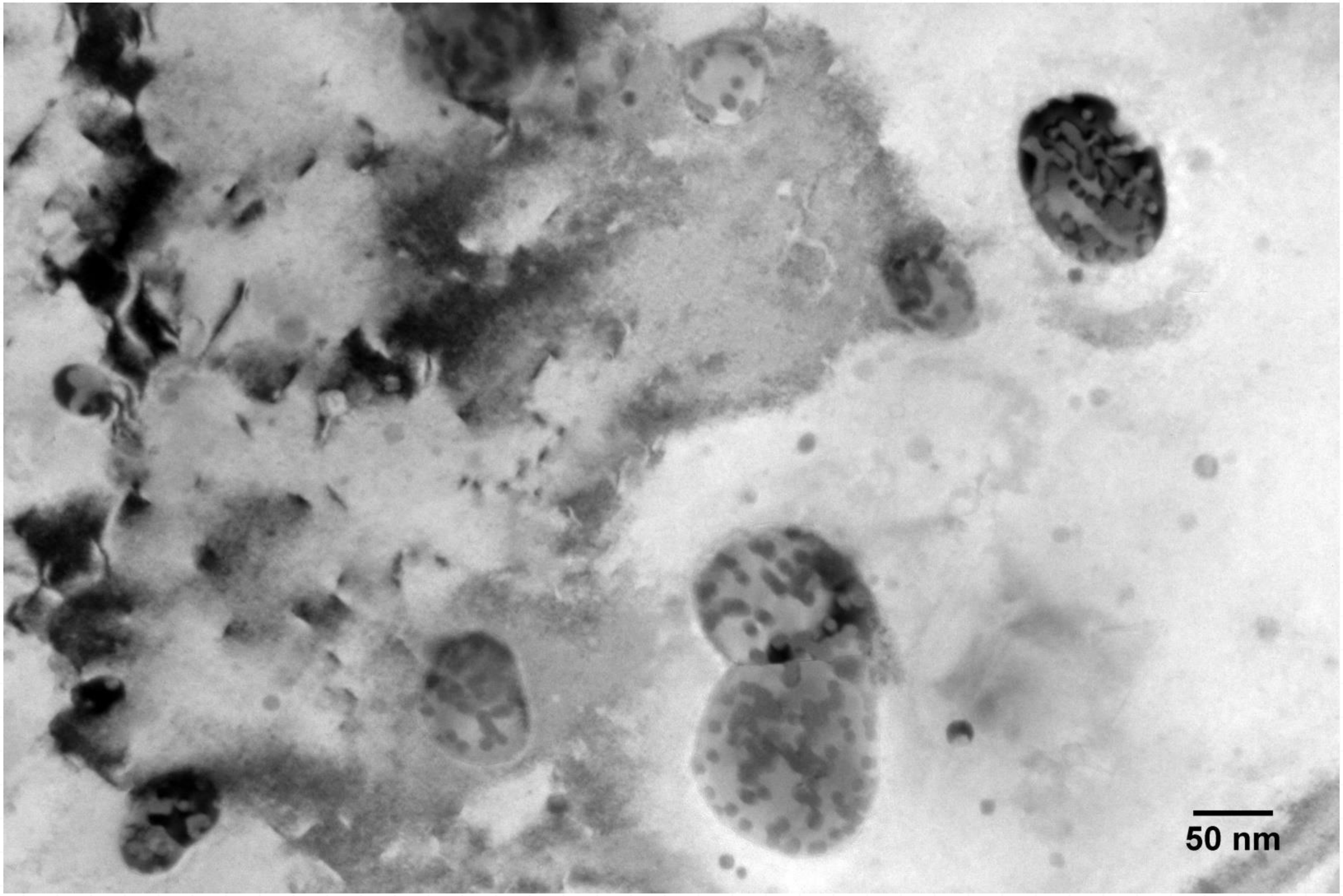


Coherent $Y_2Ti_2O_7$ (cubic) oxides in α-Fe matrix

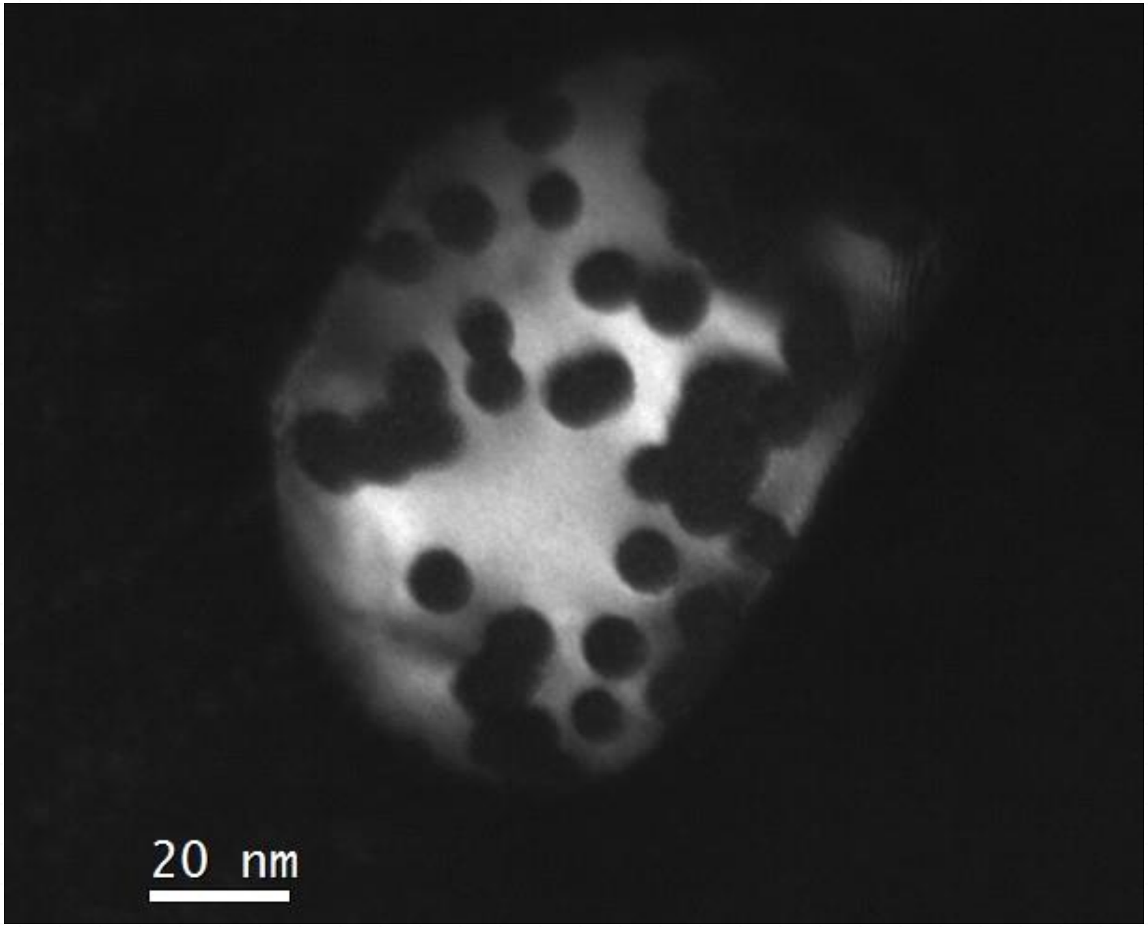
Oxide size distribution →

d = 5 - 200 nm; ‹d› = 10 nm; concentration ~ 10^{15} cm^{-3}

EP450/ Xe 167 MeV/ 1×10^{12}cm^{-2}

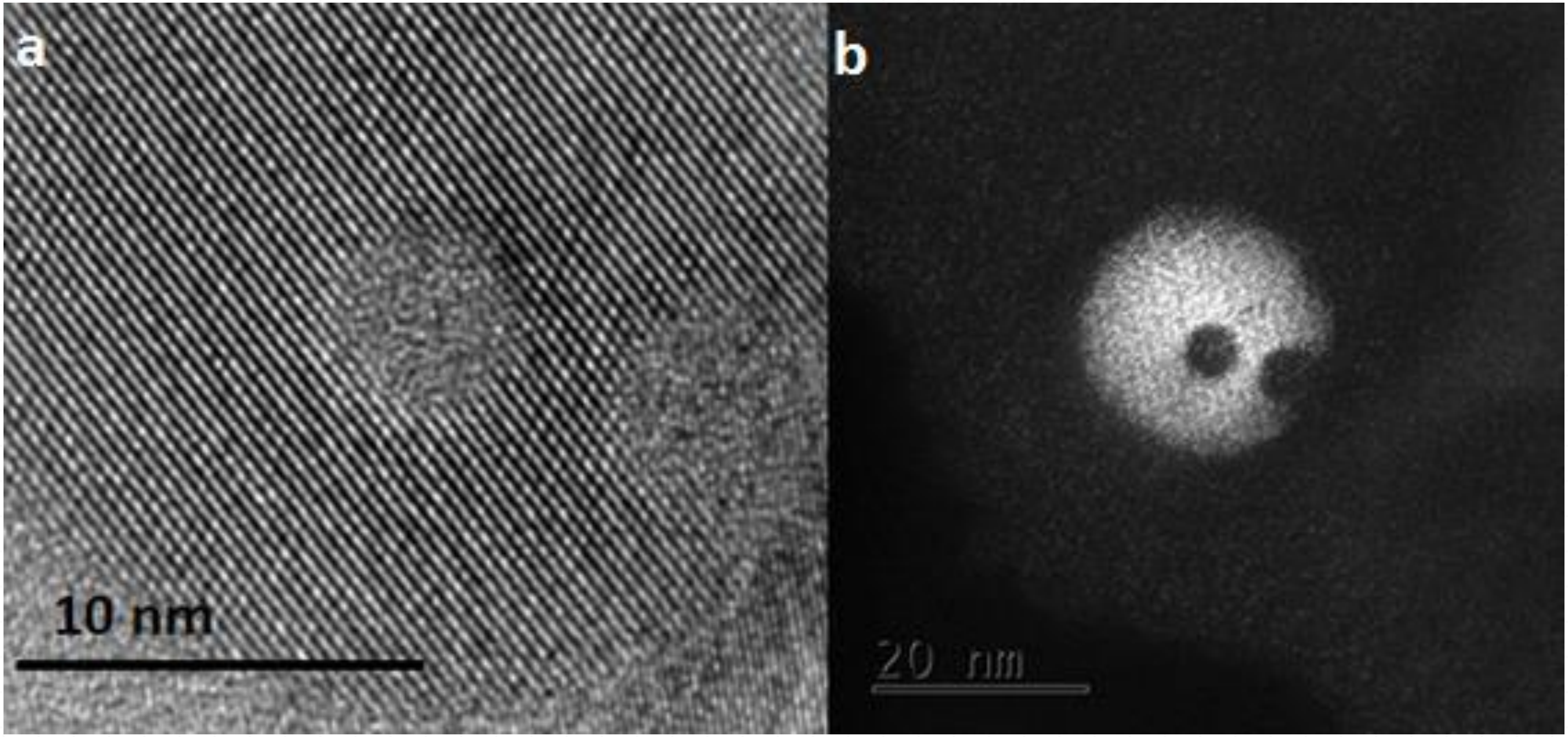


15CRA-3/ Xe 167 MeV/ $1 \times 10^{12} cm^{-2}$ S_e = 23 keV/nm



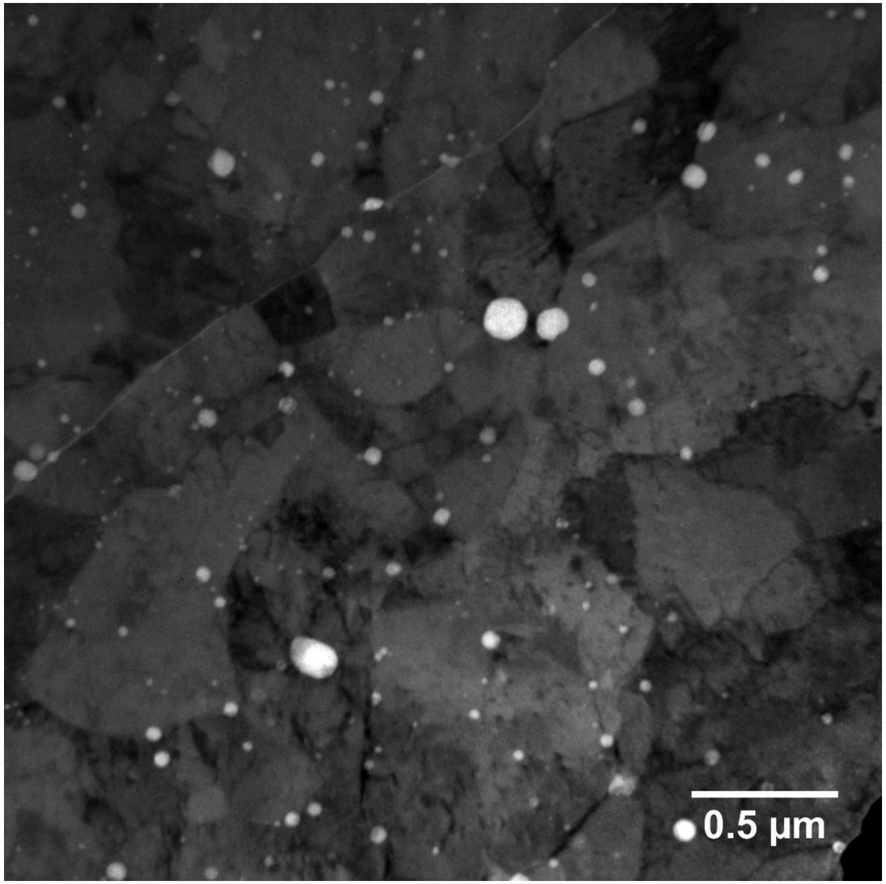
Mean track diameter is 7.5 nm

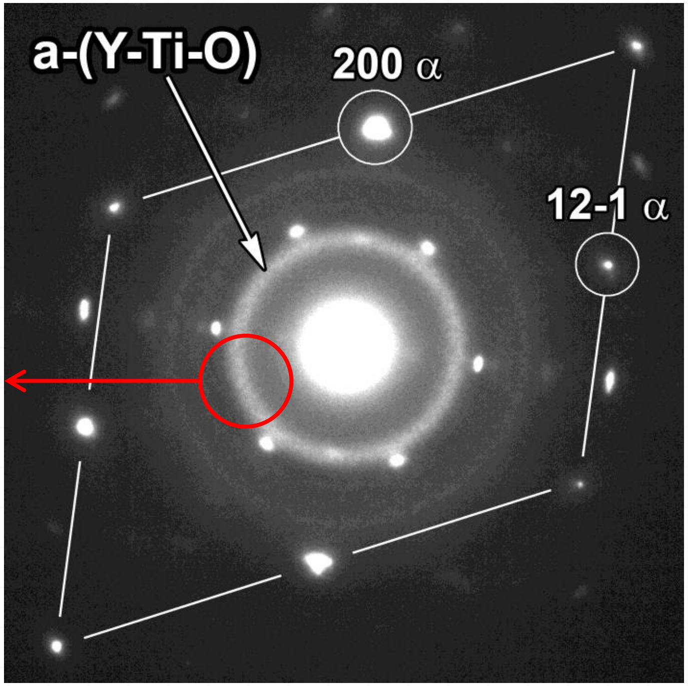
EP450/ Xe 167 MeV/ $1\times10^{12}cm^{-2}$ S_e = 23 keV/nm



HRTEM and DF TEM micrographs of latent tracks in $Y_2Ti_2O_7$

Cr16/ Bi 700 MeV/ 1.5×10^{13}cm^{-2}





Amorphization of Y-Ti oxides induced by irradiation in the ion track overlapping regime (plane view DF, SAD)

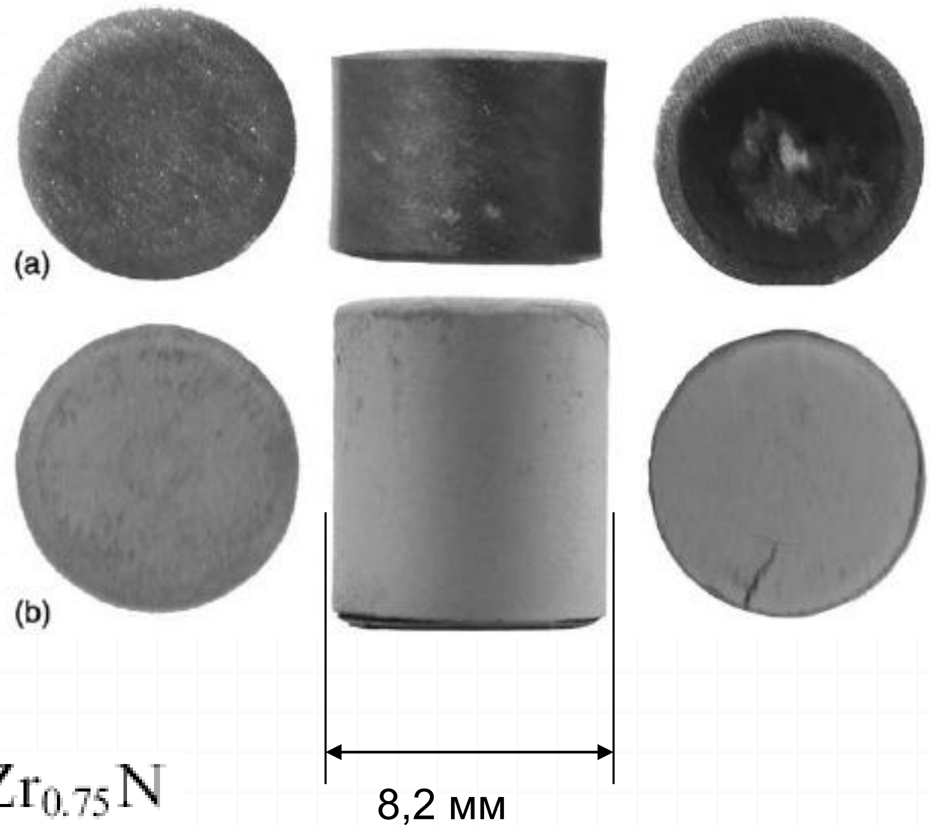
Accelerator for applied research – IC100

B^{+2}, Ne^{+4}, Ar^{+7}, Kr^{+17}, Xe^{+26} ions with energy ≈ 1.2 MeV/amu

Ion fluence range – up to 10^{15}cm^{-2}

Radiation tolerance of (Am,Np,Pu,Zr)N as inert matrix fuel

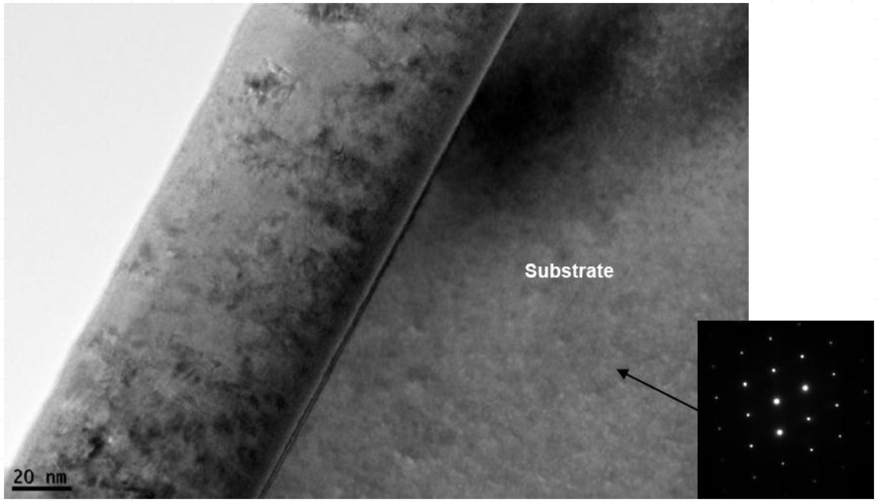
Inert matrices - ceramics with a high melting point and with low neutron absorption cross sections to be used as hosts for transmutation of actinides via nuclear reactions



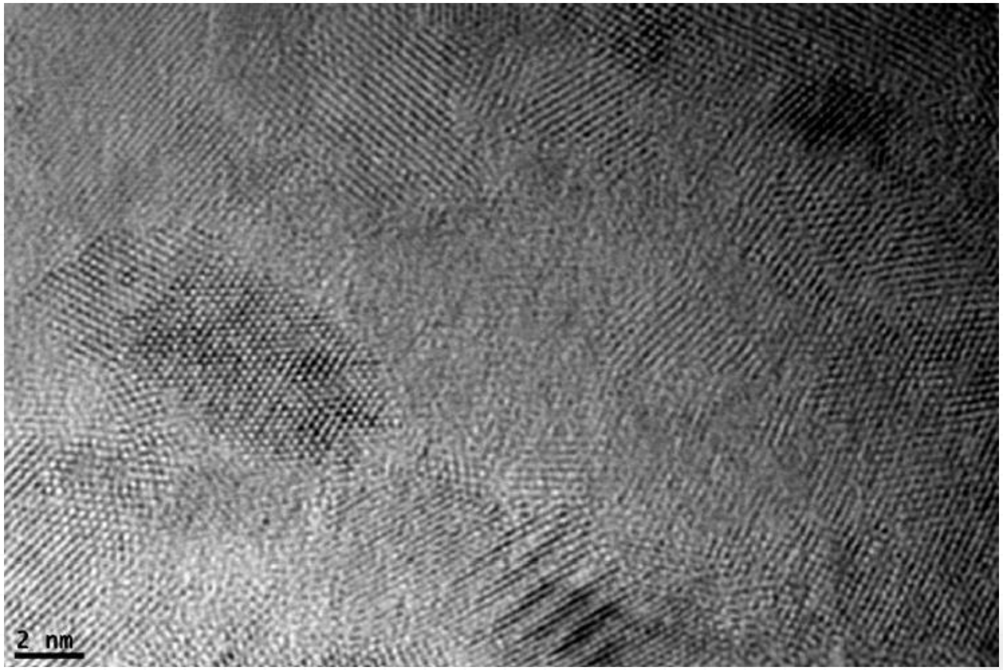
$Pu_{0.25}Zr_{0.75}N$

M. Streit et al. Journal of Nuclear Materials 319 (2003) 51

Radiation tolerance of nanostructured ZrN coating on cladding materials simulating fission fragments impact

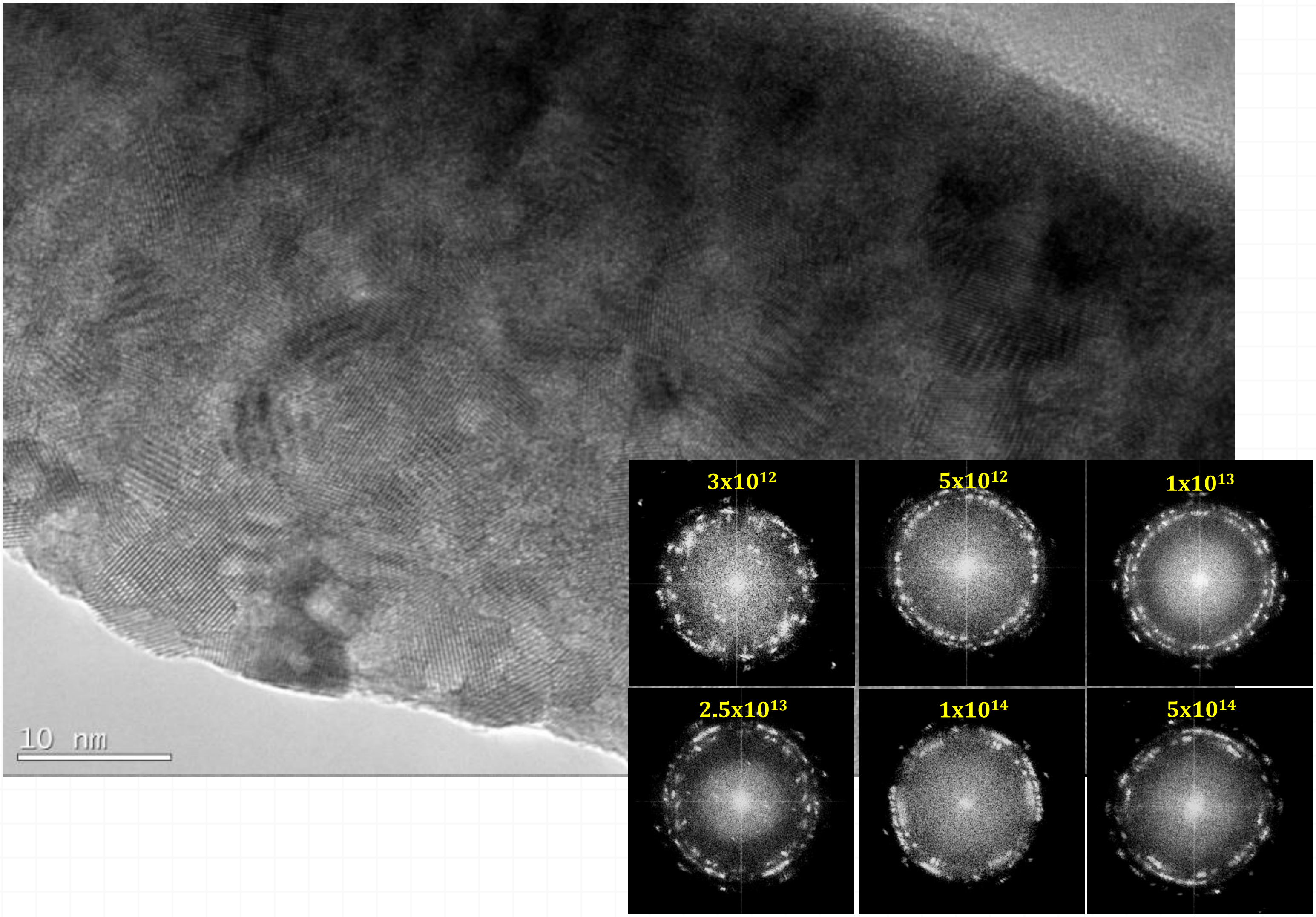


TEM image of as-grown 80 nm ZrN layer



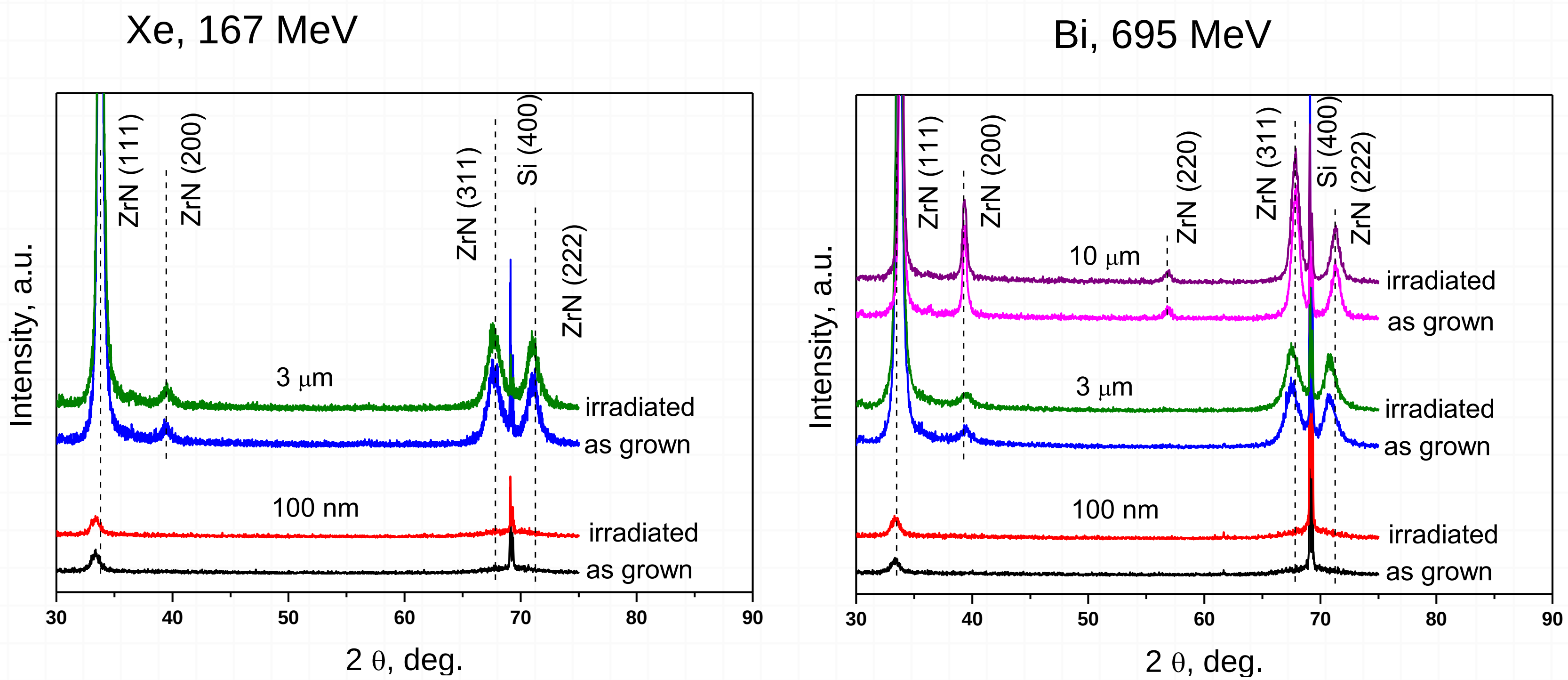
Bright field TEM micrograph of virgin ZrN sample

TEM (100 nm / 167 MeV Xe)



FFTs of the ZrN after irradiation with Xe ions to fluences from $3x10^{12}$ to $5x10^{14}$ ions/cm^2

Results (XRD)



X-ray diffraction patterns recorded on virgin and swift heavy ion irradiated nanocrystalline ZrN samples

- **The phase composition of nanocrystalline ZrN is not changed after heavy ion irradiation at electronic stopping power up to 49 keV/nm**
- **TEM examination does not reveal latent tracks formation in ZrN irradiated with heavy ions of fission fragments energy**

- Irradiation of constructive reactor materials for postradiation studies (structure, mechanical properties)

- Real-time examination of irradiating materials

Identification of small irradiation induced defects (type, size, density) by combining SANS and transmission electron microscopy (TEM) observations with molecular dynamics (MD) simulations and TEM image simulations

Thank you for your attention!