MATERIALS CHALLENGES FOR GENERATION IV REACTORS

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Meet the presenter

Stuart Maloy is a Team Leader for MST-8 (materials at radiation and dynamic extremes) at Los Alamos National Laboratory(where he has worked for 28 years) and is the advanced reactor core materials technical leader for the Nuclear Technology Research and Development's Advanced Fuels campaign and the NEET Reactor Materials Technical Lead for DOE-NE. He earned his Bachelors Degree ('89), Masters Degree ('91) and PhD ('94) in Materials Science from Case Western Reserve University and is a registered PE in Metallurgy. He has applied his expertise to characterizing and testing the properties of metallic and ceramic materials in extreme environments such as under neutron and proton irradiation at reactor relevant temperatures. This includes testing the mechanical properties (fracture toughness and tensile properties) of Mod 9Cr-1Mo, HT-9, 316L, 304L, Inconel 718, Al6061-T6 and Al5052 after high energy proton and neutron irradiations using accelerators and fast reactors. Characterization of materials after testing includes using transmission electron microscopy for analyzing defects such as dislocations, twins and second phases, using high resolution electron microscopy to characterize defects at an atomic level and nanoscale mechanical testing. Stuart has >190 peer reviewed technical publications and numerous presentations.

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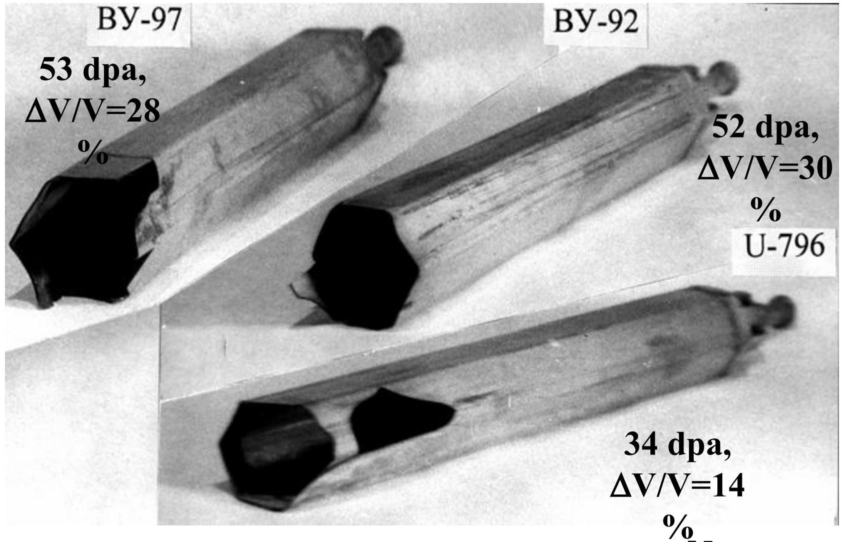
Outline

- Radiation Effects in Materials
- Materials in Nuclear Reactors – FCC, BCC alloys
- Reactor Conditions/Materials Performance Issues
  - LWR (BWR/PWR)
  - Typical materials
  - Advanced Reactors (VHTR, SCWR)
  - Advanced Fast Reactors (SFR, GFR, MSR)
- Summary of Reactor Operating Conditions
- Summary of Performance Issues

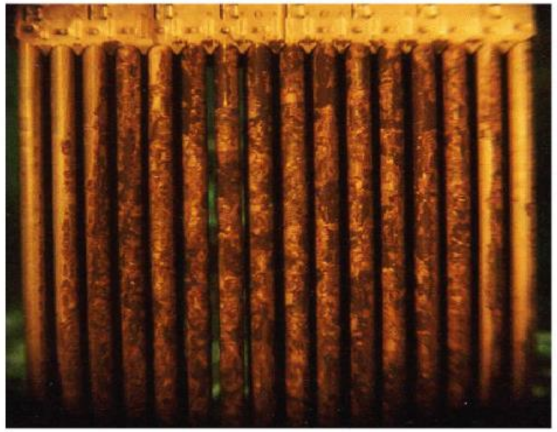
Materials in nuclear systems can fail

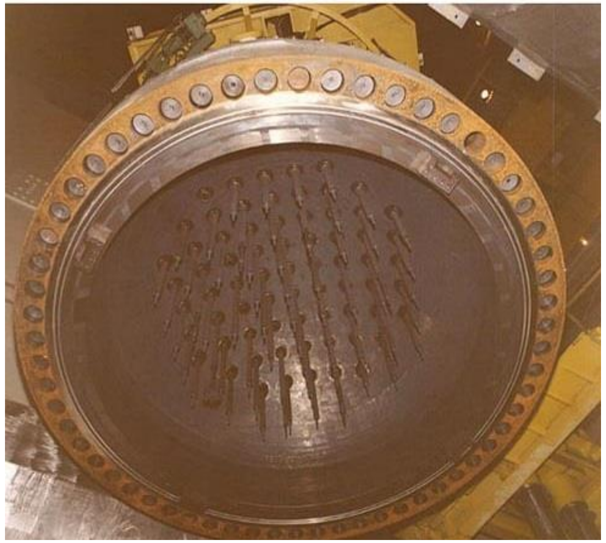


Fast Reactor Duct Failure

Grid-to-Rod Fretting

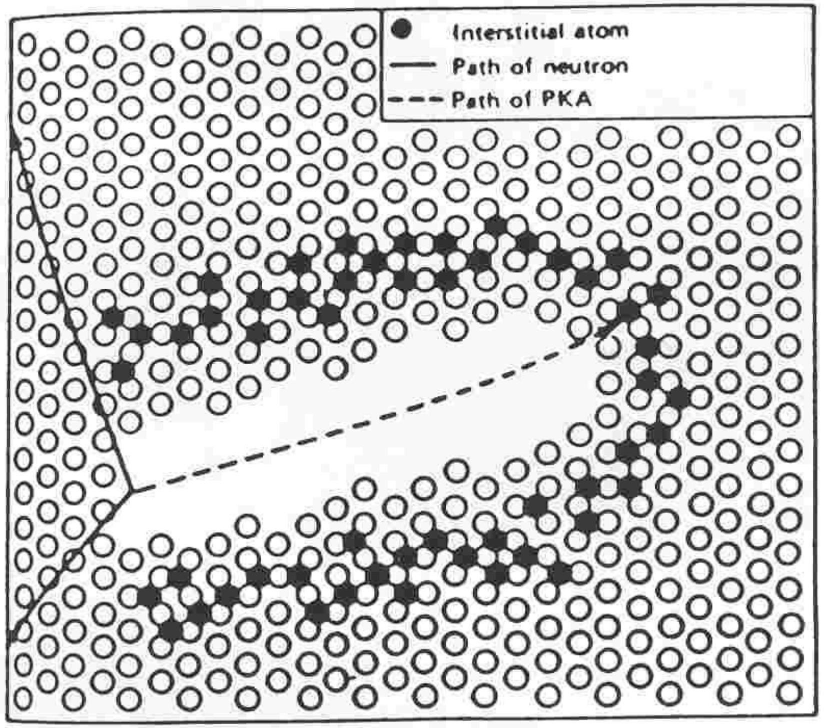
CRUD



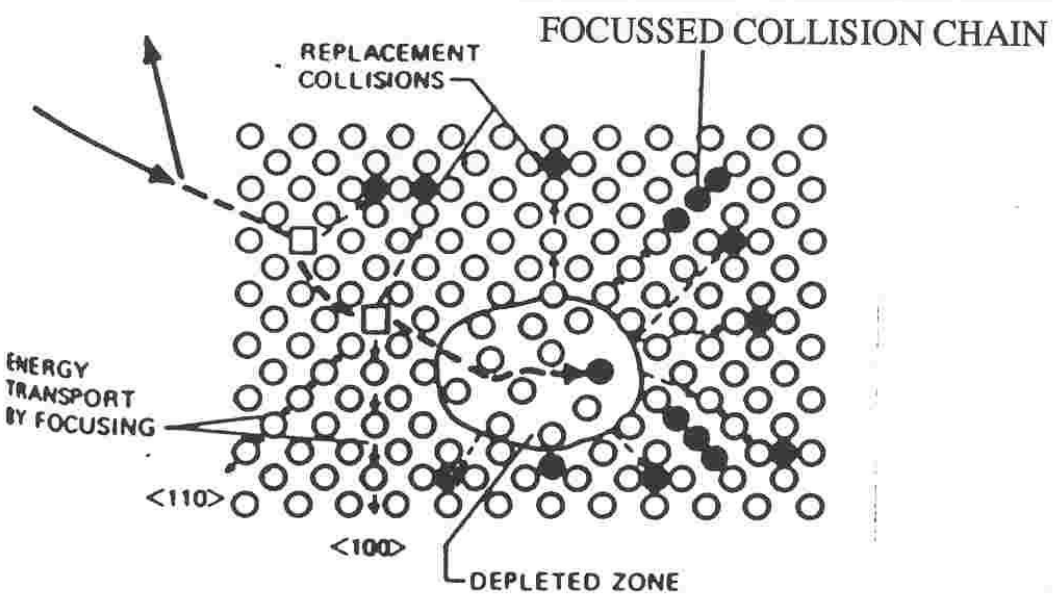


Davis-Besse Reactor Vessel Head Degradation

Radiation Damage can have Significant Effects on Mechanical Properties



Original model of the displacement spike, J.A. Brinkman, Amer. J. Phys 24 (1956) 24.



Later, but still qualitative, version of the displacement spike (also called depleted zone).

Displacement damage occurs when enough energy (approximately 25 eV) is transferred to an atom producing a or many Frenkel defects. Energy can be transferred from many different particles (neutrons, protons, fission products, etc).

Particles with higher energies (>5 MeV) can cause spallation in materials.

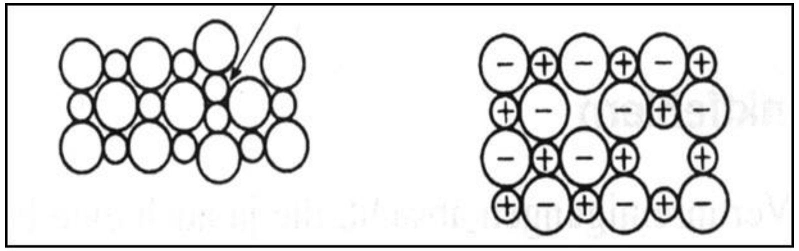
Background on radiation damage

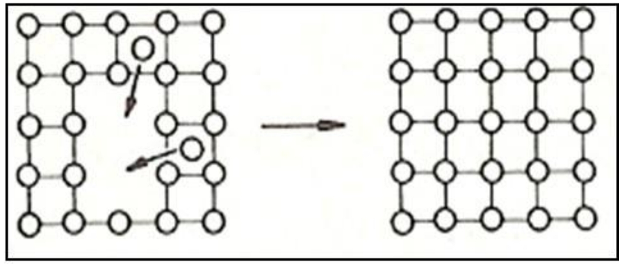
1st Transmutation (changing elements)

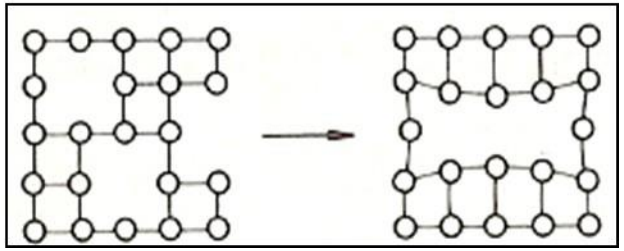
- Activating the material
- Producing new elements leading to He bubble formation and embrittlement e.g. 56Fe(n,a)53Cr or 58Ni(n,a)55Fe

2nd Materials property changes due to atomic displacements:

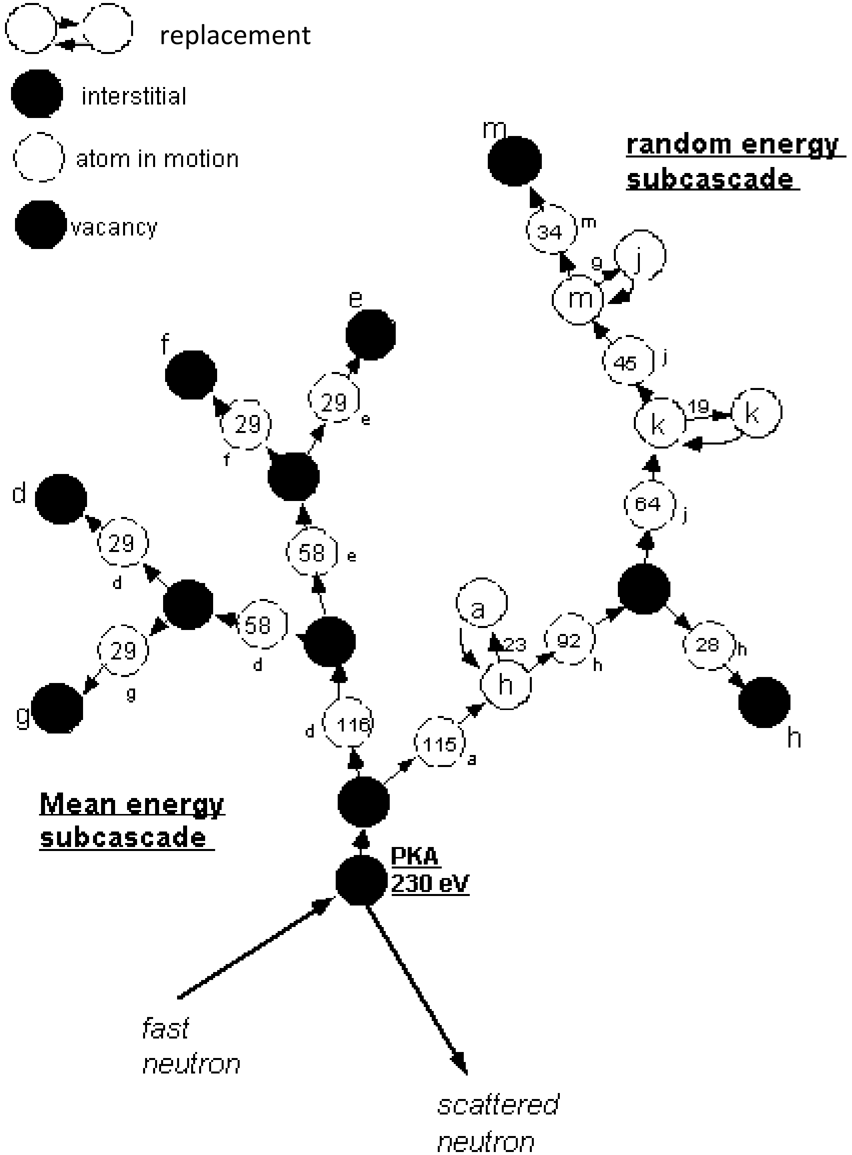
- Creation of Frankel pairs can lead to:
  - **Increase of dislocation density**
    → embrittlement
  - **Formation of voids**
    → swelling
  - **Increased diffusivity**
    → local segregation
  - **Amorphisation or crystallization**
    → unexpected phase changes

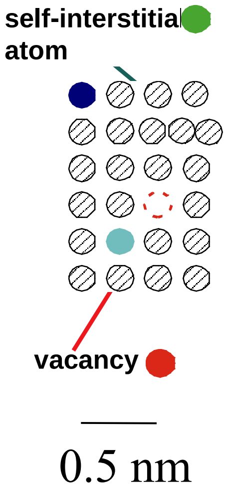




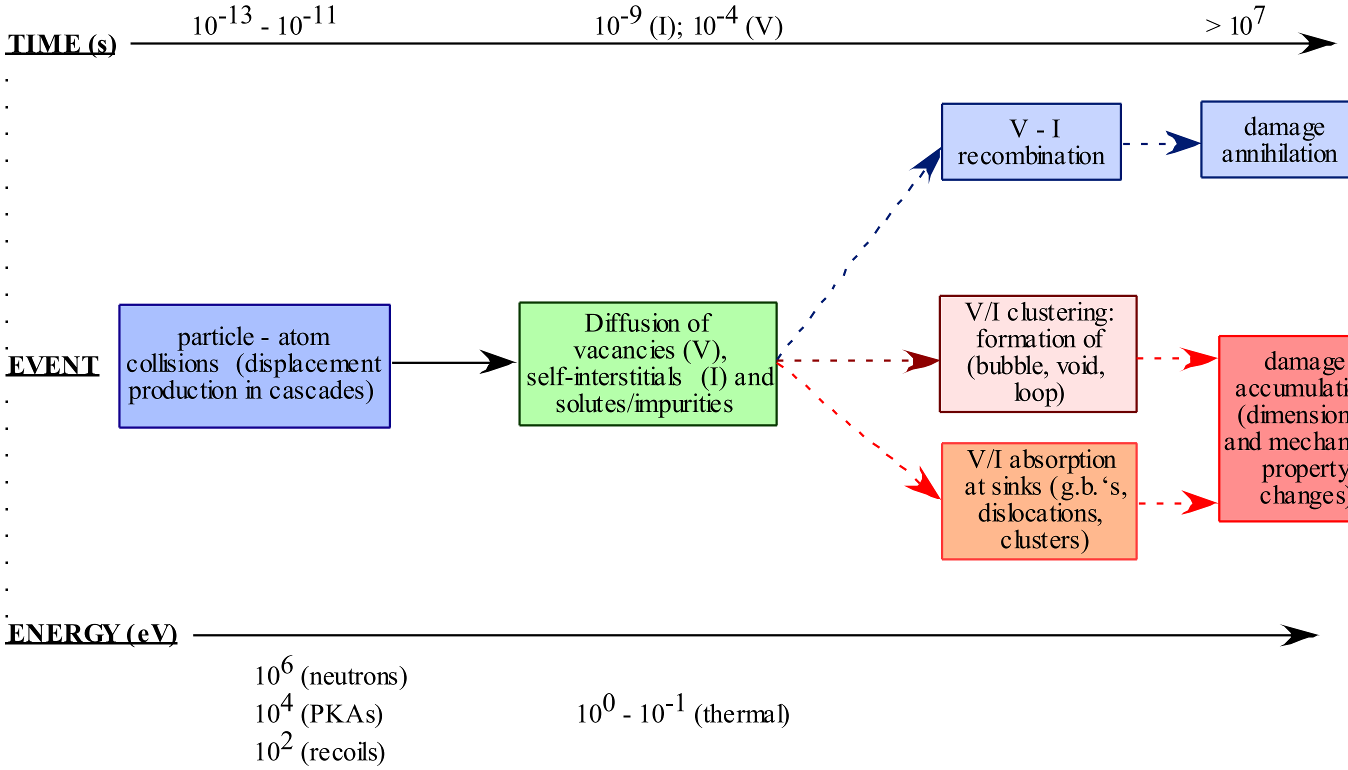


Example of cascade by low energy PKA’s

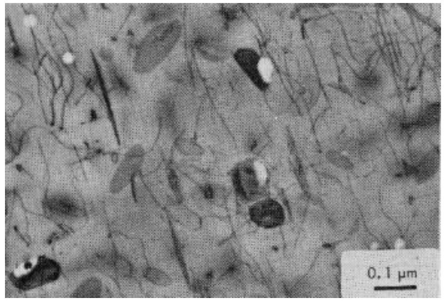




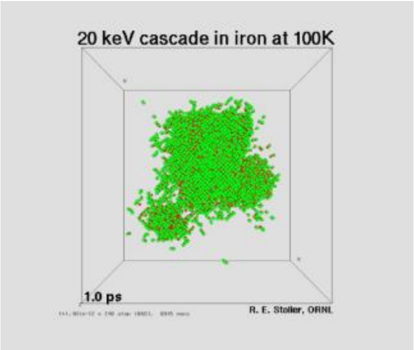
Radiation effects are governed by the ultimate fate of point defects (& transmutants)



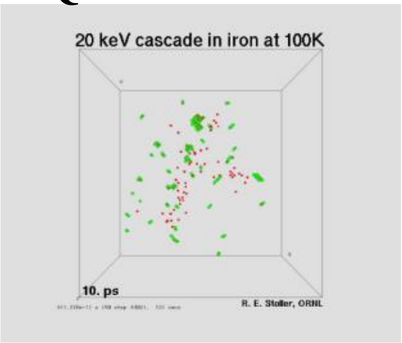
Voids, I-loops & precipitates in stainless steel



Nascent cascade



"Quenched" cascade

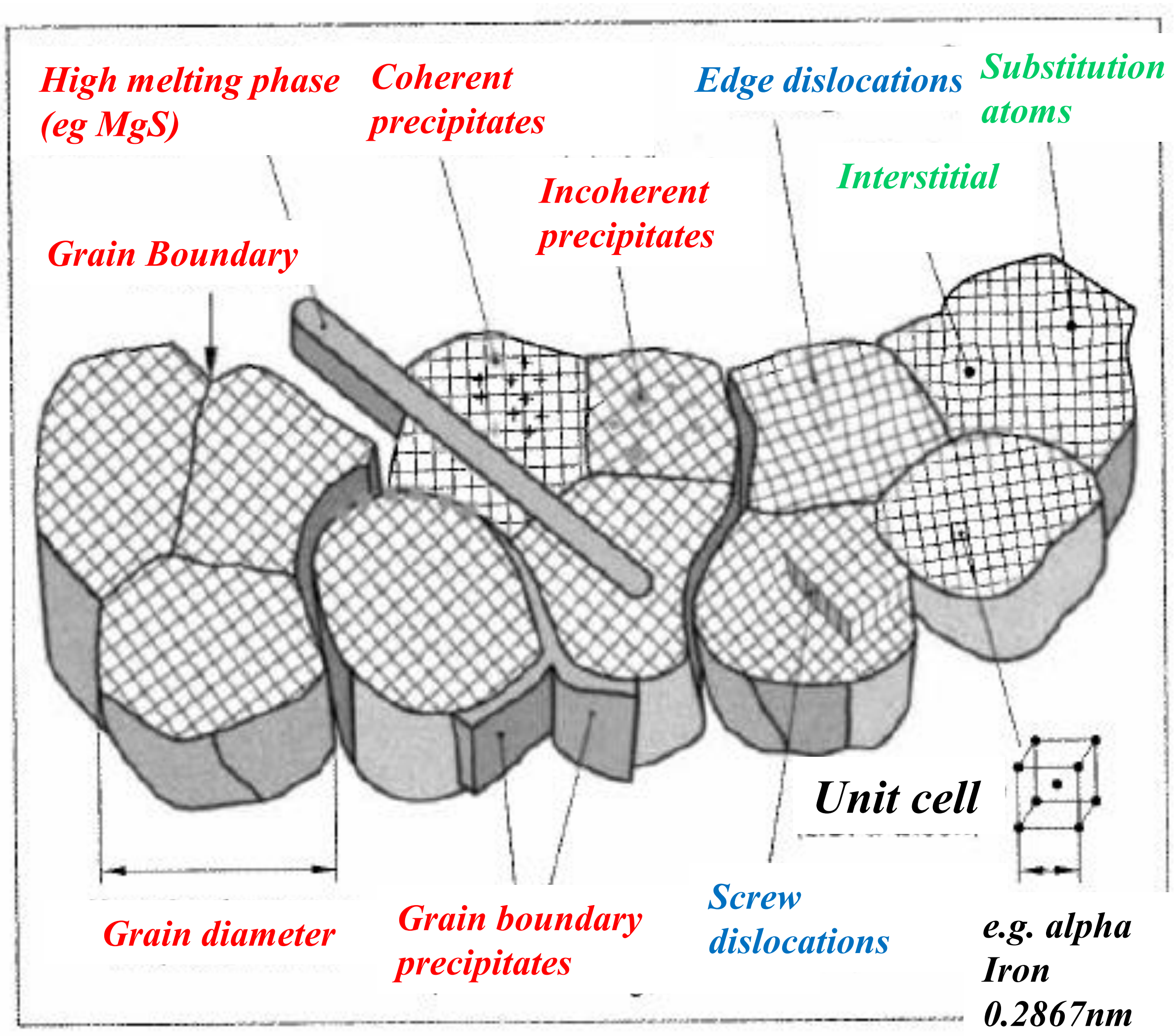


Number of defects:

Nascent cascade: $N_i = N_v = \nu(T)$

"Quenched" cascade: $N_i' = N_v' < \nu(T)$

A wide range of materials properties are determined on the mesoscale



Green=nanoscale defects

Blue= nanoscale-meso scale

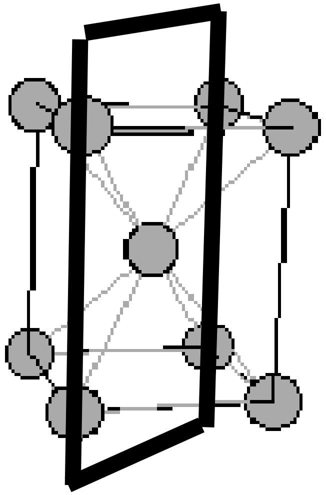
Red=mesoscale defects (actually defect dependent

Typical Alloy Compositions for Nuclear Applications

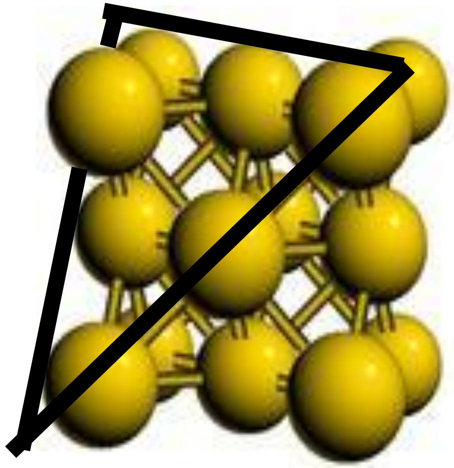
- Austenitic Steels (Face Centered Cubic, FCC)
  - 316L (Fe, 18Cr, 10Ni, 2 Mo), 304L Stainless steel (Fe, 18Cr, 8 Ni, 1.5 Mo)
  - Inconel 718 (55 Ni, 21Cr, 24Fe, 5Nb, 3Mo, 1Ti, 0.8Al)
  - Alloy 600 (72 Ni, 17Cr, 9Fe, 1 Mn)
- Ferritic Steels (Body Centered Cubic, BCC)
  - Mod 9Cr-1Mo (Fe, 9Cr, 1Mo, 0.1C, 0.5Mn)
  - HT-9 (Fe, 12Cr, 1Mo, 0.2C, 0.5W, 0.6Mn, 0.3V, 0.5Ni)
- Zirconium Alloys (Hexagonal Close Packed, HCP)
  - Zr-Sn alloys (Zircaloy 2, Zircaloy 4)
  - Zr-Nb alloys (Zr-1Nb; Zr-2.5Nb, M5 (Zr-1Nb))
  - Zr-Sn-Nb-Fe alloy (ZIRLO)

Radiation Effects in Metals

- Defect formation
  - Basically Frenkel Defects (self interstitials and vacancies)
  - No charge compensating defects
  - Very little effect from gamma irradiation
  - Amorphization is uncommon at typical irradiation temperatures 25 to 600C
- Close Packed Structures
  - FCC (face centered cubic)-close packed plane is (111)-Frank loops- atomic packing factor is 0.74
  - BCC (body centered cubic)-close packed plane is (110)-atomic packing factor is 0.68
  - HCP (hexagonal close packed)-for ideal c/a ratio of 1.633, atomic packing factor is same as FCC, 0.74

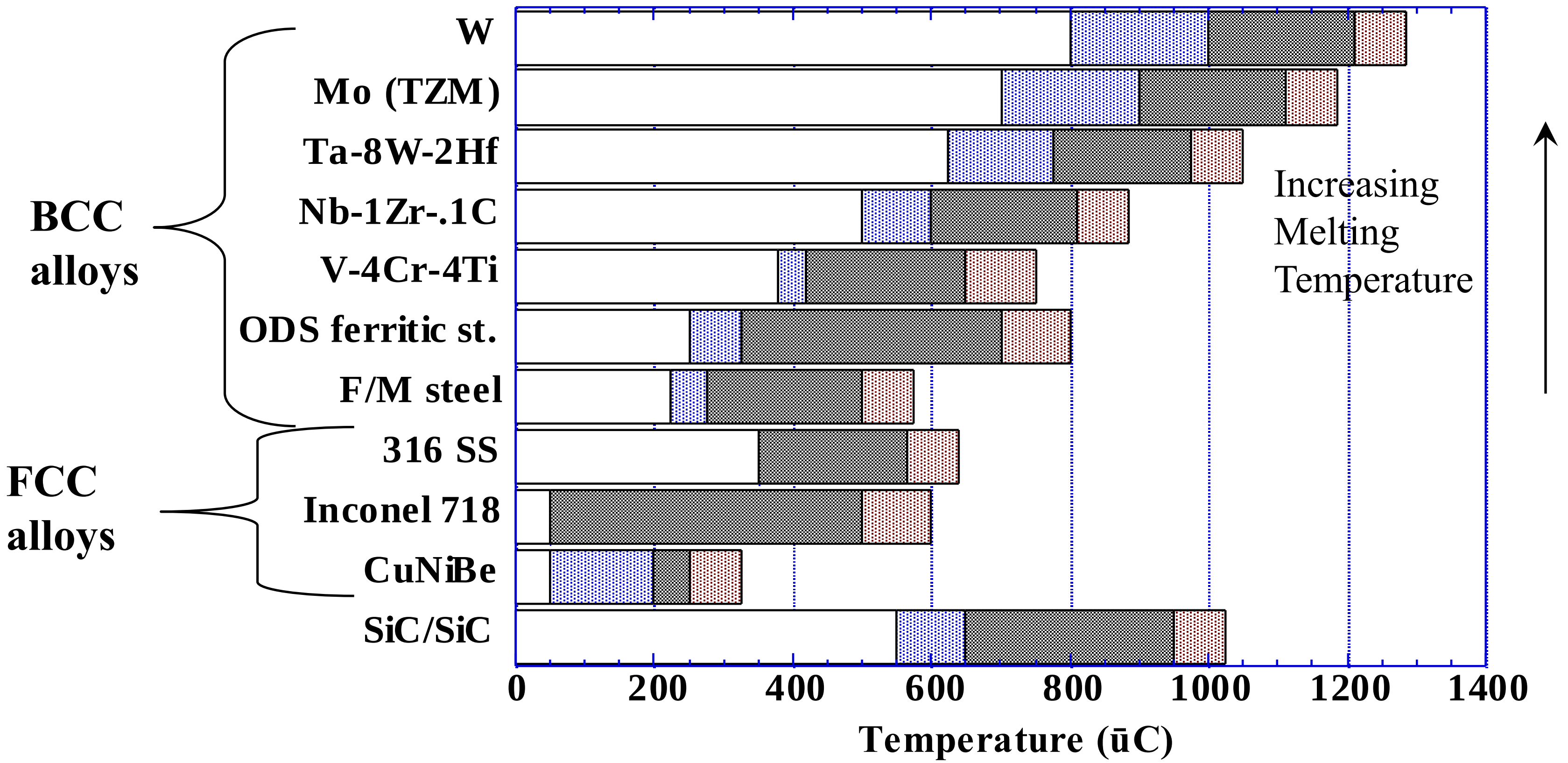


BCC



FCC

Useful Operating Temperature Dependent on Melting Temperature and Crystal Structure



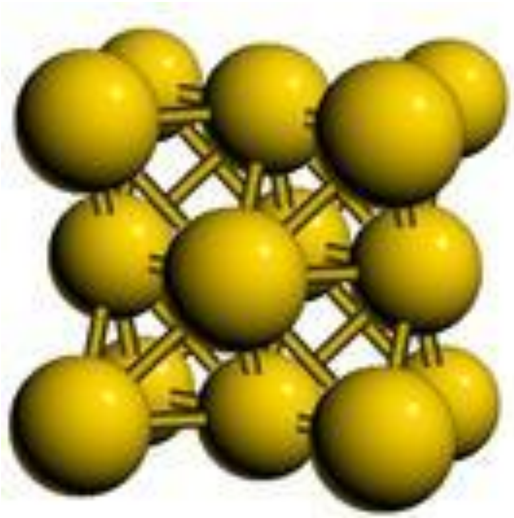
- At lower temperature (blue region) vacancies are immobile and interstitials are mobile resulting in interstitial clusters and loops and small vacancy clusters.
- At medium temperature (gray region) vacancy mobility increases resulting in more self annihilation of defects (vacancy finds interstitial) and possibility of swelling.
- At higher temperature (red region) vacancy and interstitial mobility are high leading to problems with creep or helium embrittlement.

S.J. Zinkle and N.M. Ghoniem, Fus. Eng. Des. 51-52 (2000) 55; S.J. Zinkle et al. STAIF2002

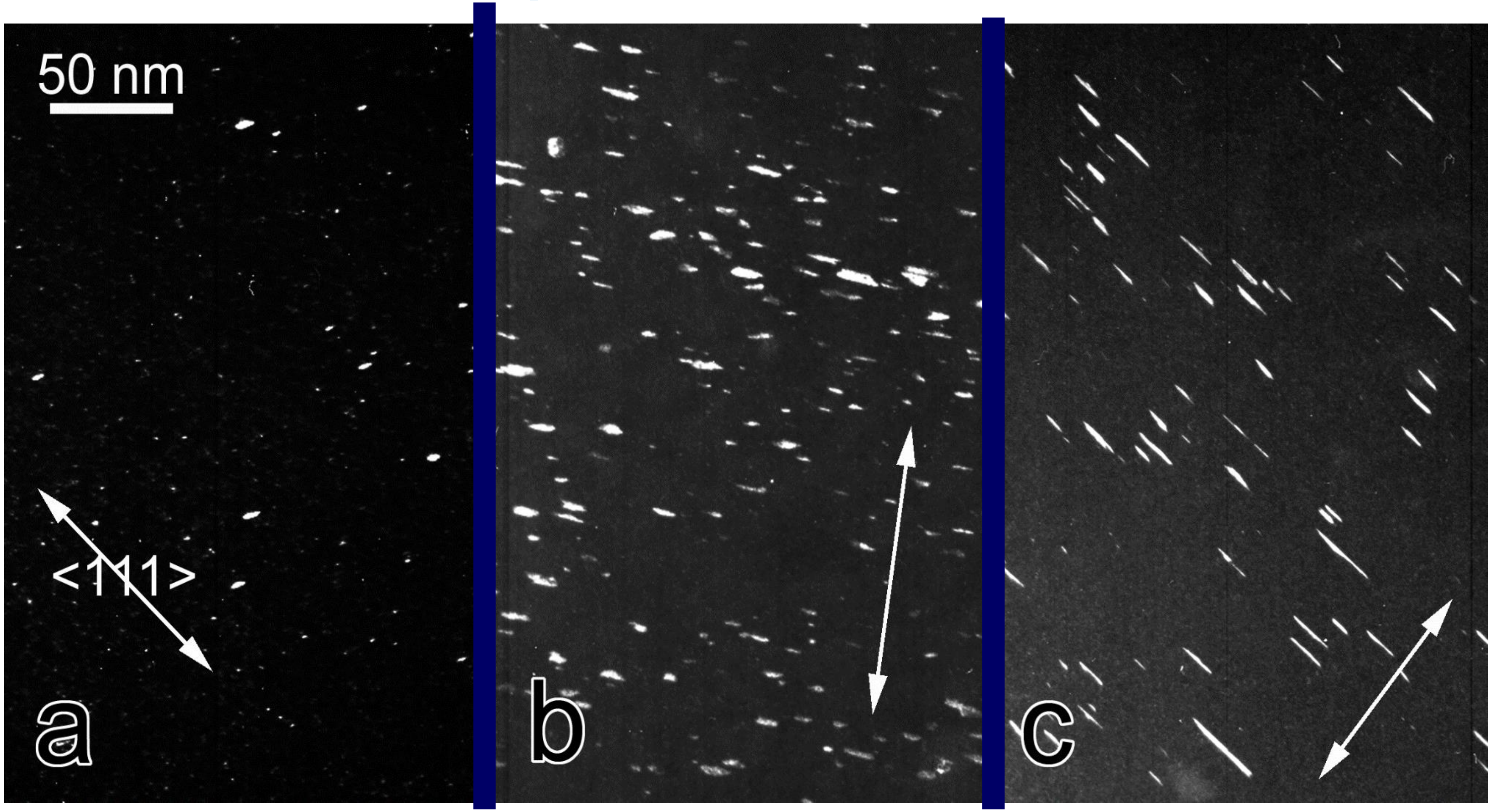
Irradiation Effects in Metals

A. 316L/304L Stainless Steel (FCC)
B. Alloy 718 (FCC)
C. Ferritic Steels (BCC)

Irradiation Effects in 316L/304L Stainless Steel (FCC) at 50-200C



TEM images Showing the Growth of Frank Loops in 304L



0.7 dpa 3.8 dpa 9.8 dpa

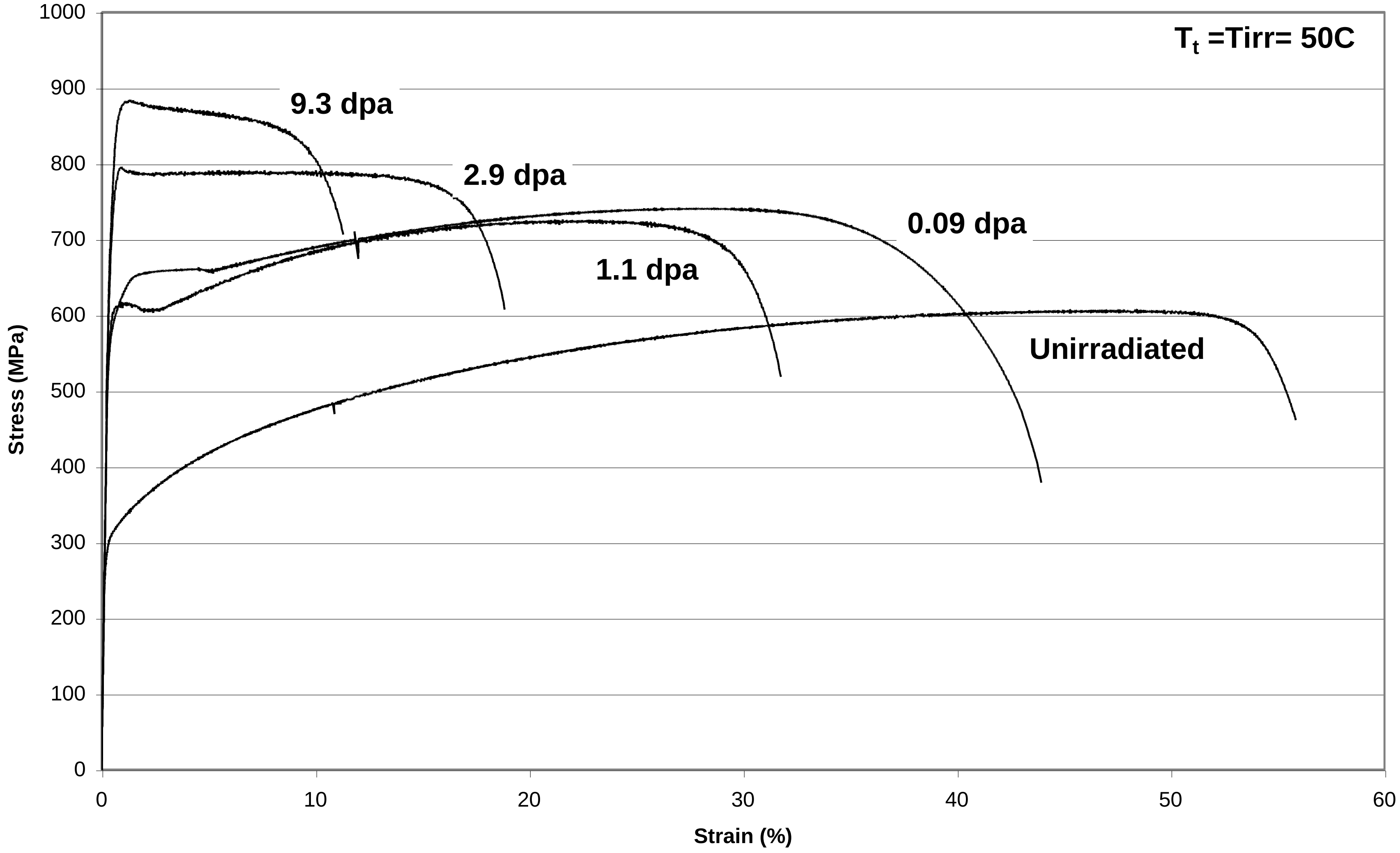
Irradiated with 800 MeV proton beam at 30-50C

- no helium clusters are observed
- loops mainly from collection of interstitials

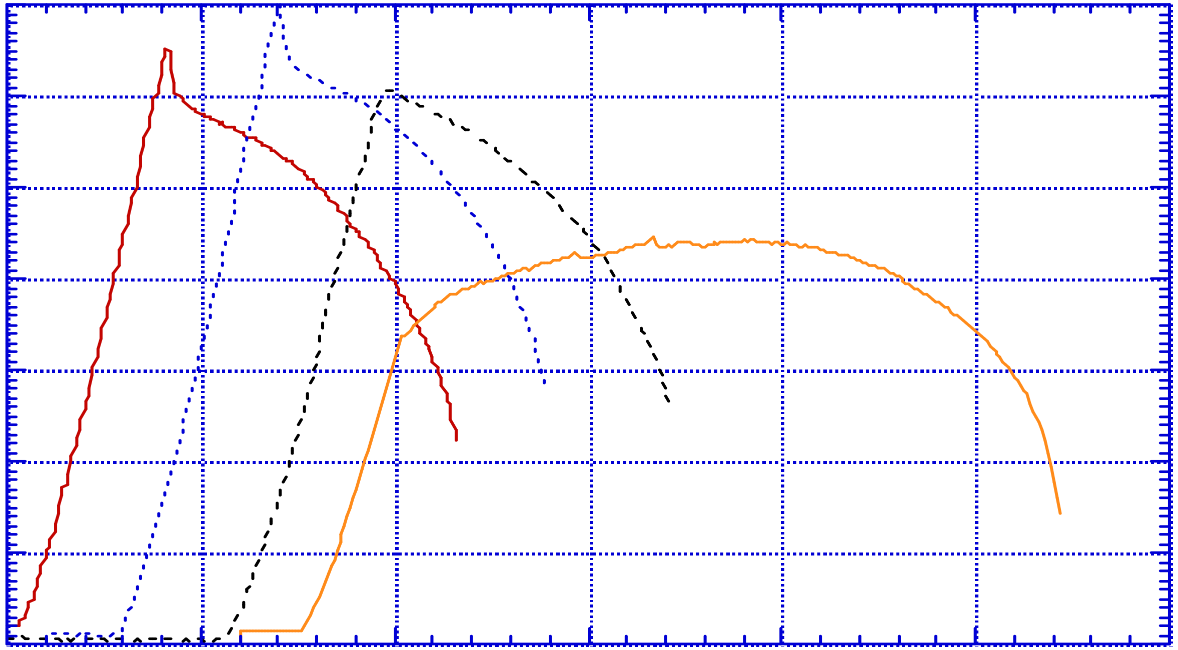
Dislocation Loops Show a Saturation in Density around 4 dpa

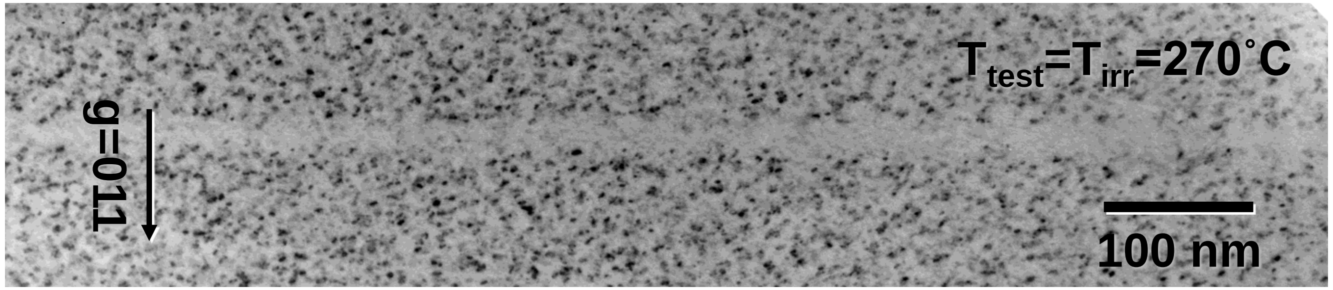
Dose dpa	Loop Number Loop Density m^{-3}	Mean Loop Diameter nm	Total Dislocation Density m^{-2}
0.7	$1.6x10^{22}$	1.8	$9.0x10^{13}$
3.8	$5x10^{22}$	9.6	$1.51x10^{15}$
9.8	$2.1x10^{22}$	20.1	$1.32x10^{15}$

Stress/Strain Curves show increase in yield stress and decrease in elongation in 316L Stainless steel after irradiation

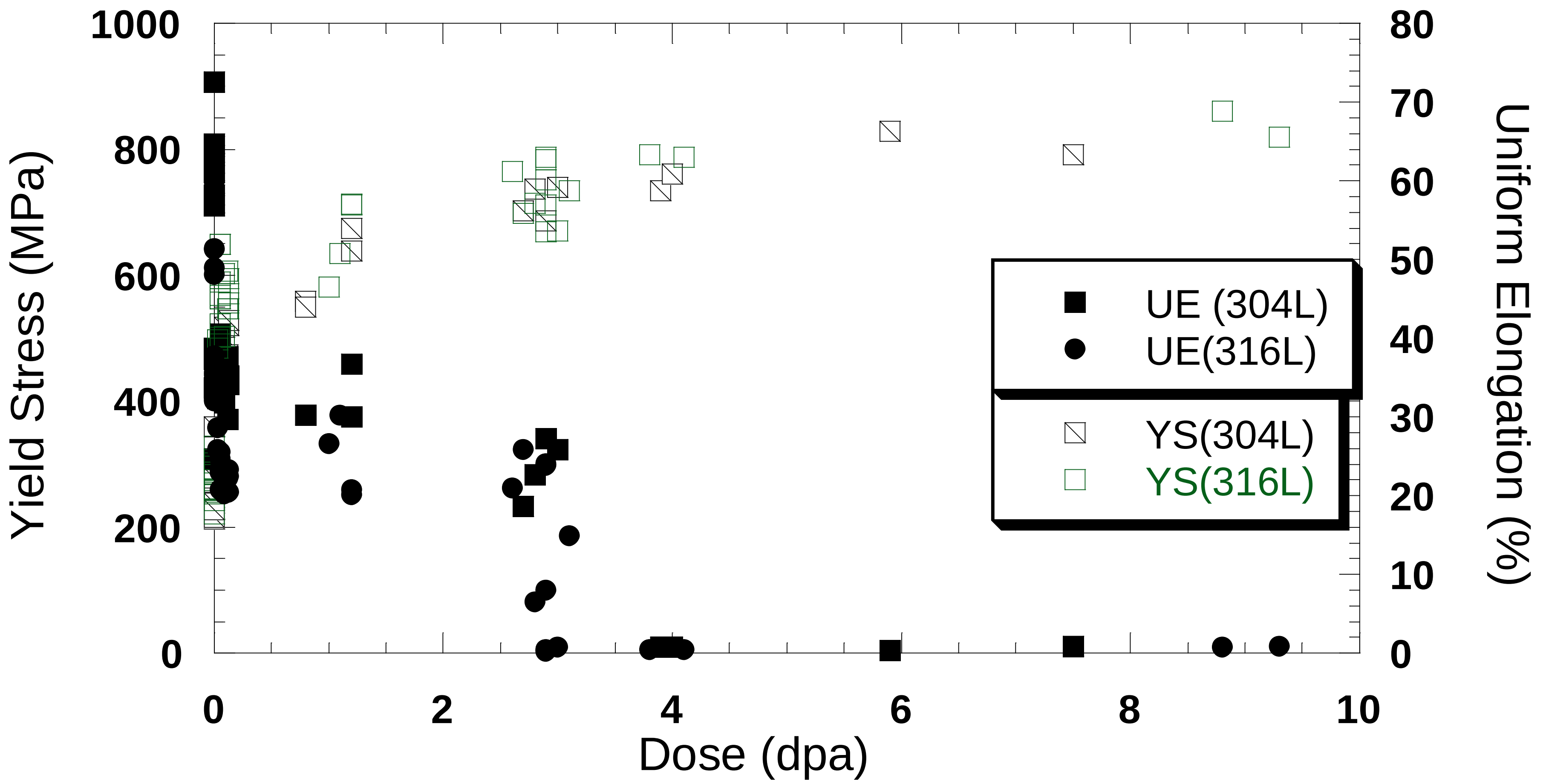


Irradiated Materials Suffer Plastic Instability due to Dislocation Channeling

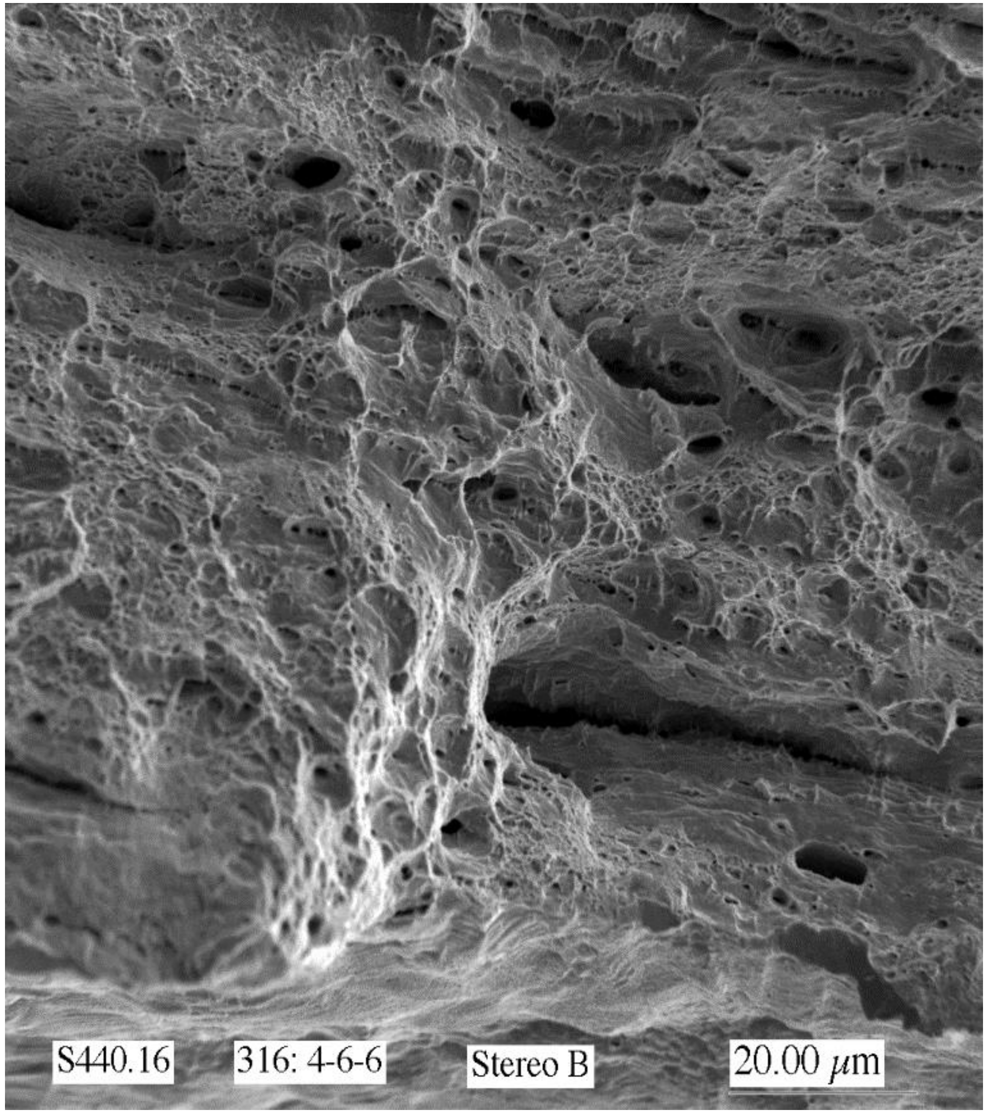




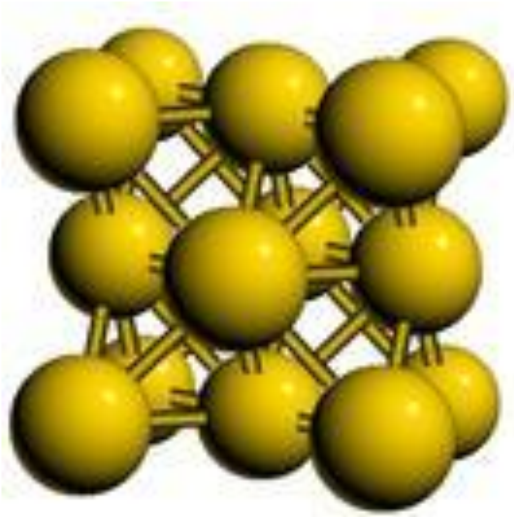
Change in the Tensile Properties with Dose for 304L and 316L Stainless Steel for Tirr=50-160C



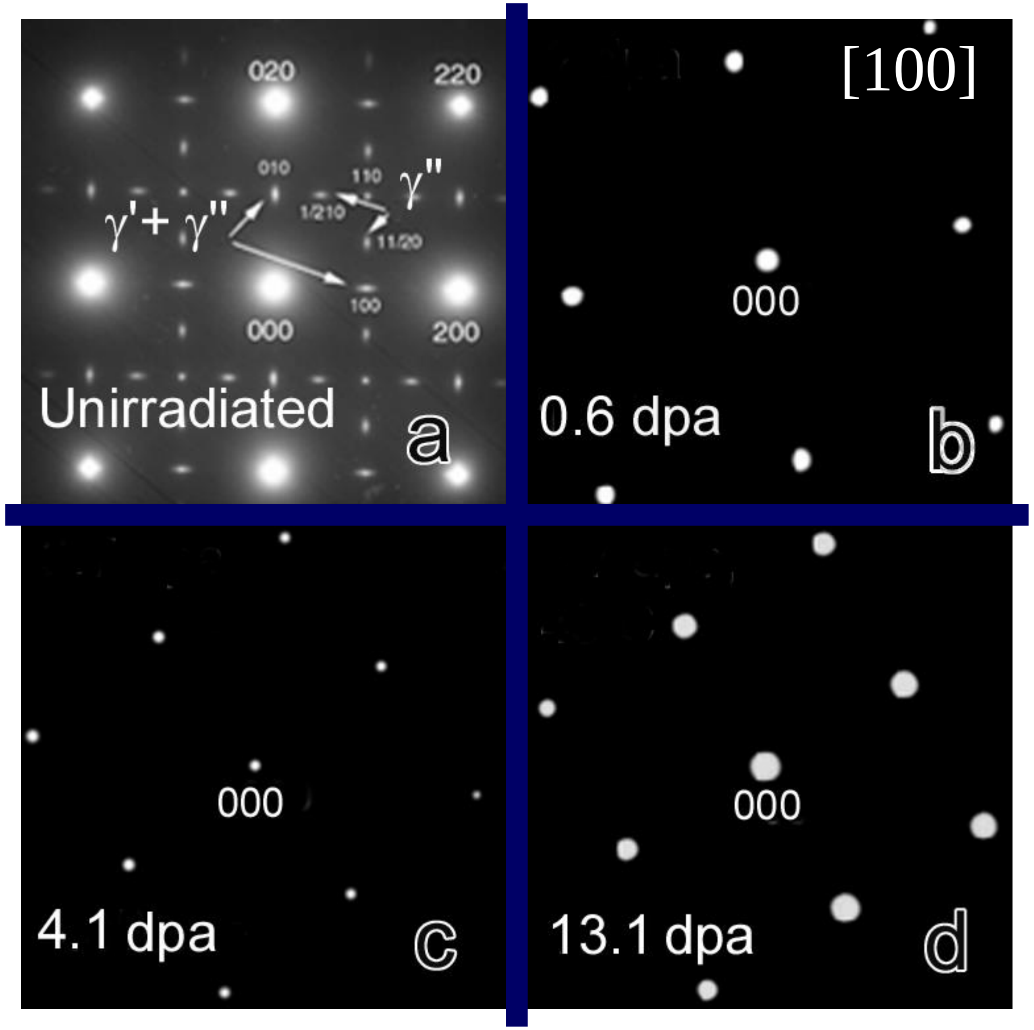
SEM Images of Fracture Surfaces of 316L Stainless Steel (dose = 9dpa) tensile specimen



Irradiation Effects in Inconel 718 (FCC) at 50-200C

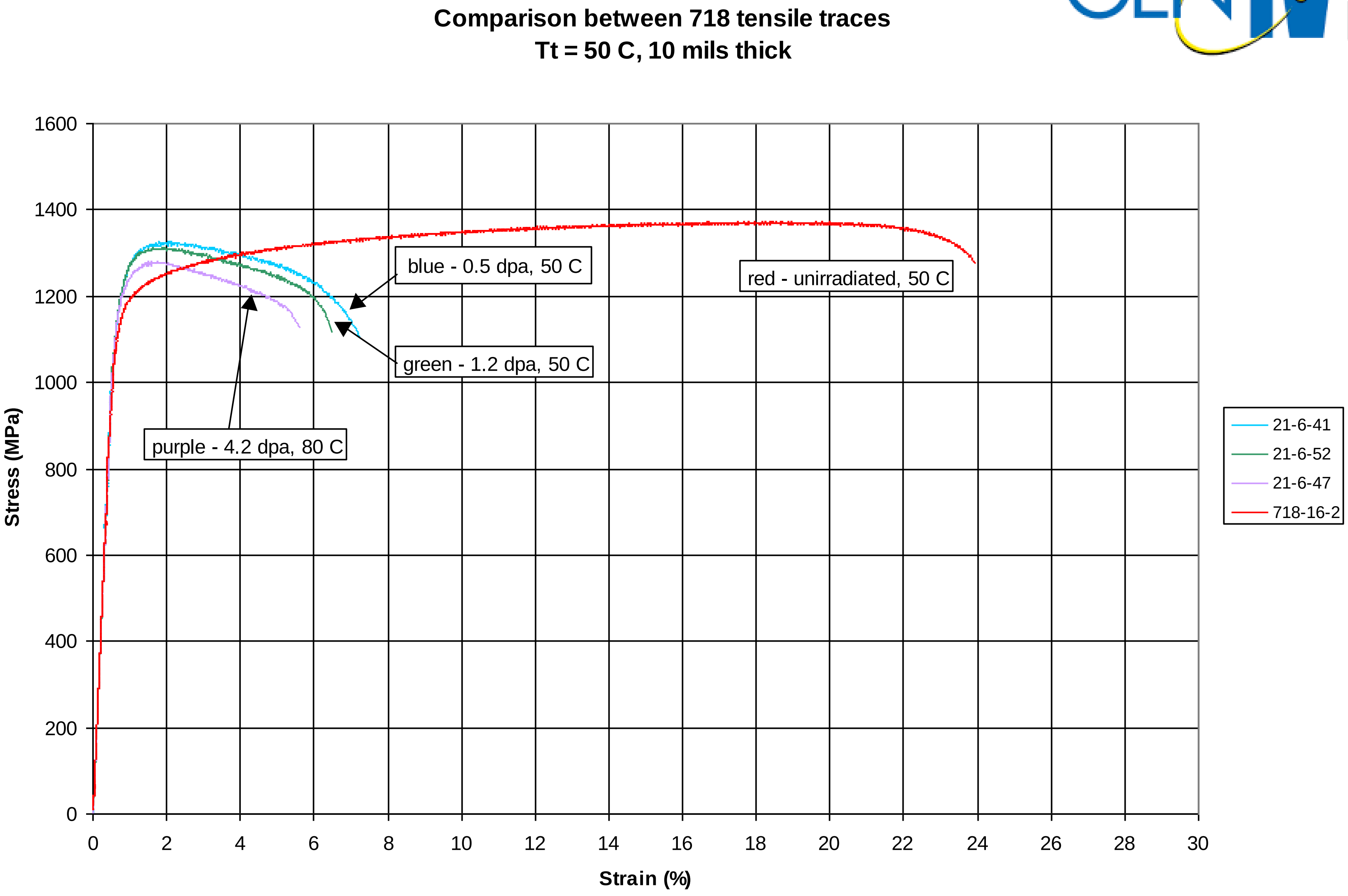


Disordering of γ ′/γ ″ Precipitates under Proton Irradiation

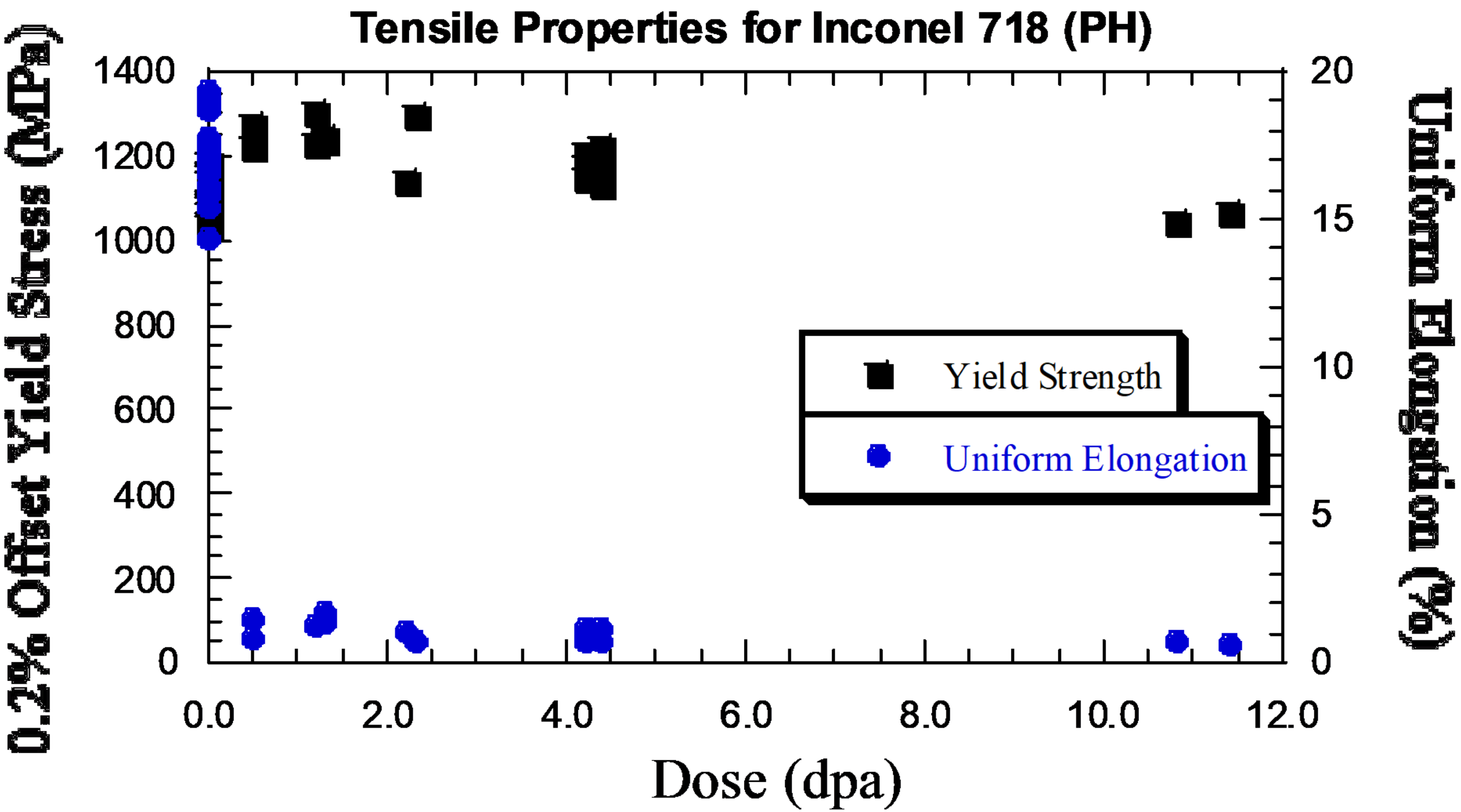


γ ′ and γ ″ disorder under proton irradiation at all dpa levels in Inconel 718.

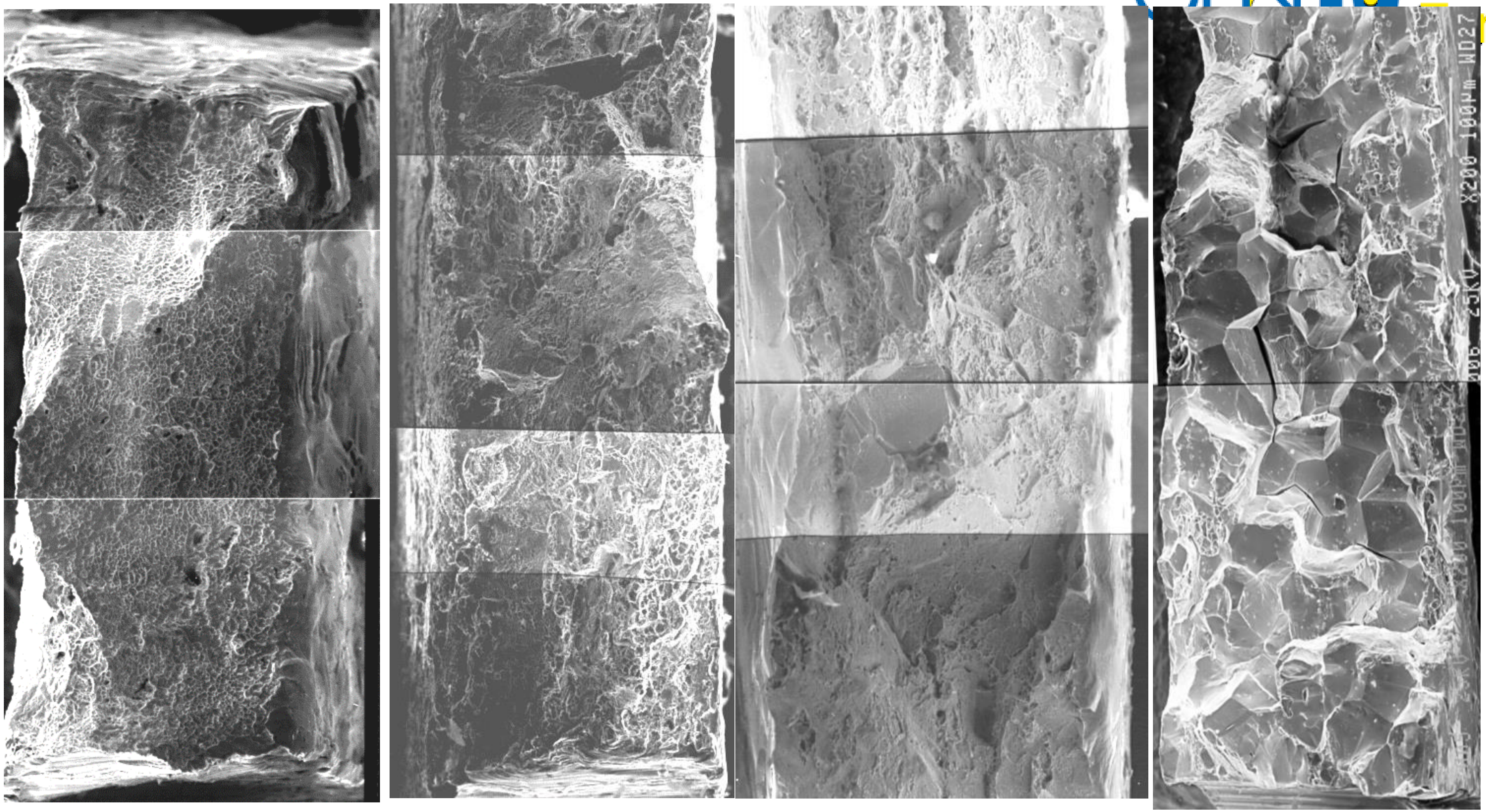
Inconel 718 (precipitation hardened) exhibits small change in yield stress during irradiation



Loss of Ductility is Quick in Inconel 718 (precipitation hardened) after irradiation.



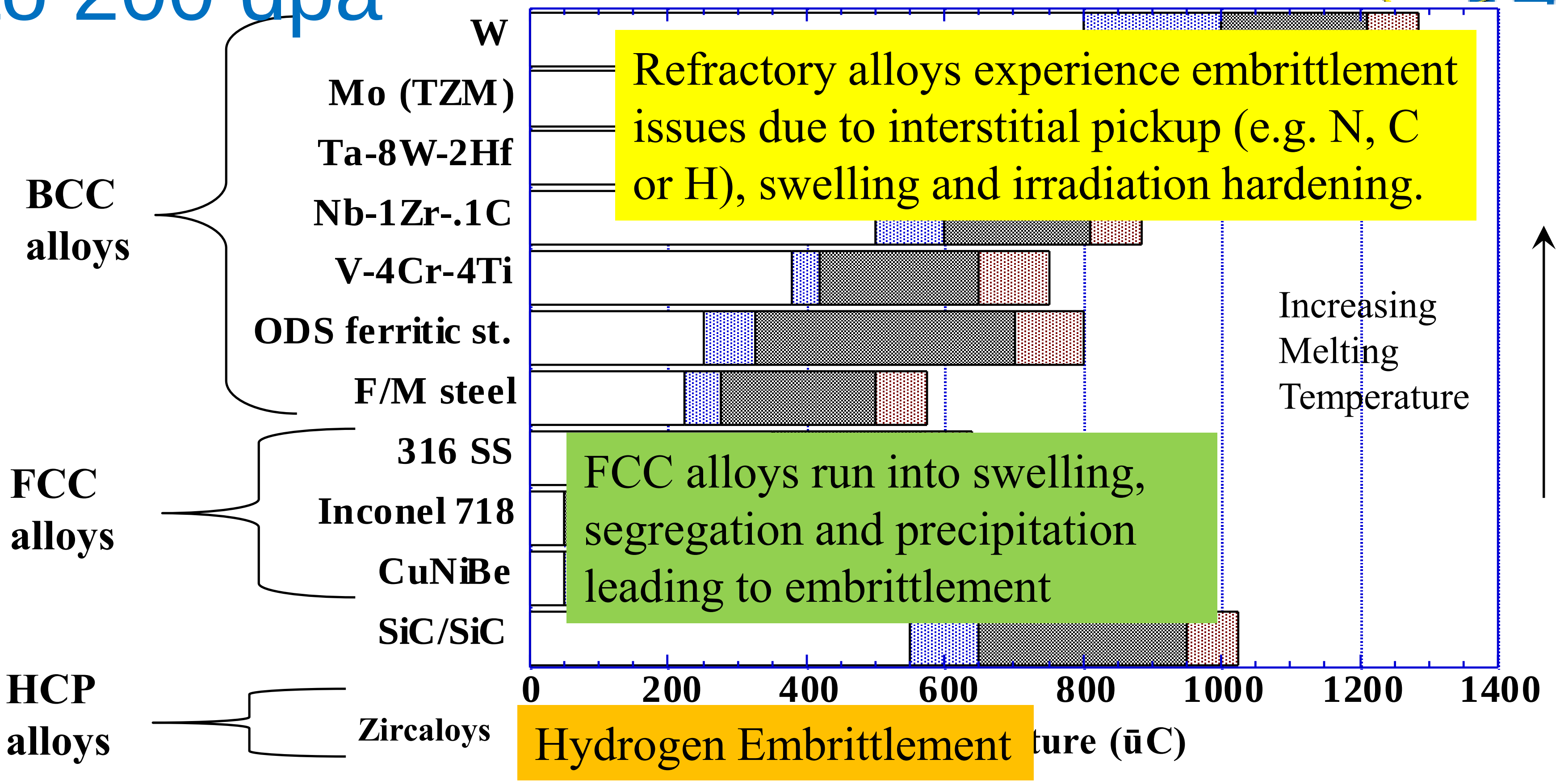
At higher doses fracture appearance changes to intergranular failure in Irradiated Alloy 718



0 dpa 4.6 dpa 10.5 dpa 19.8 dpa

Total elongation is zero at 19.8 dpa

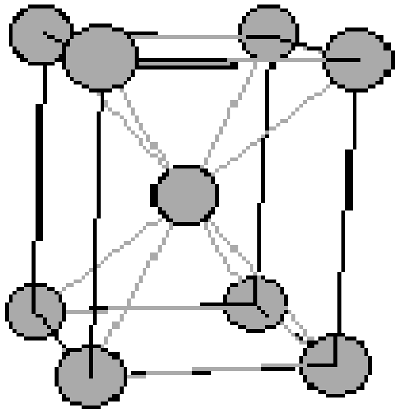
Survey of Materials Limits over higher doses to 200 dpa



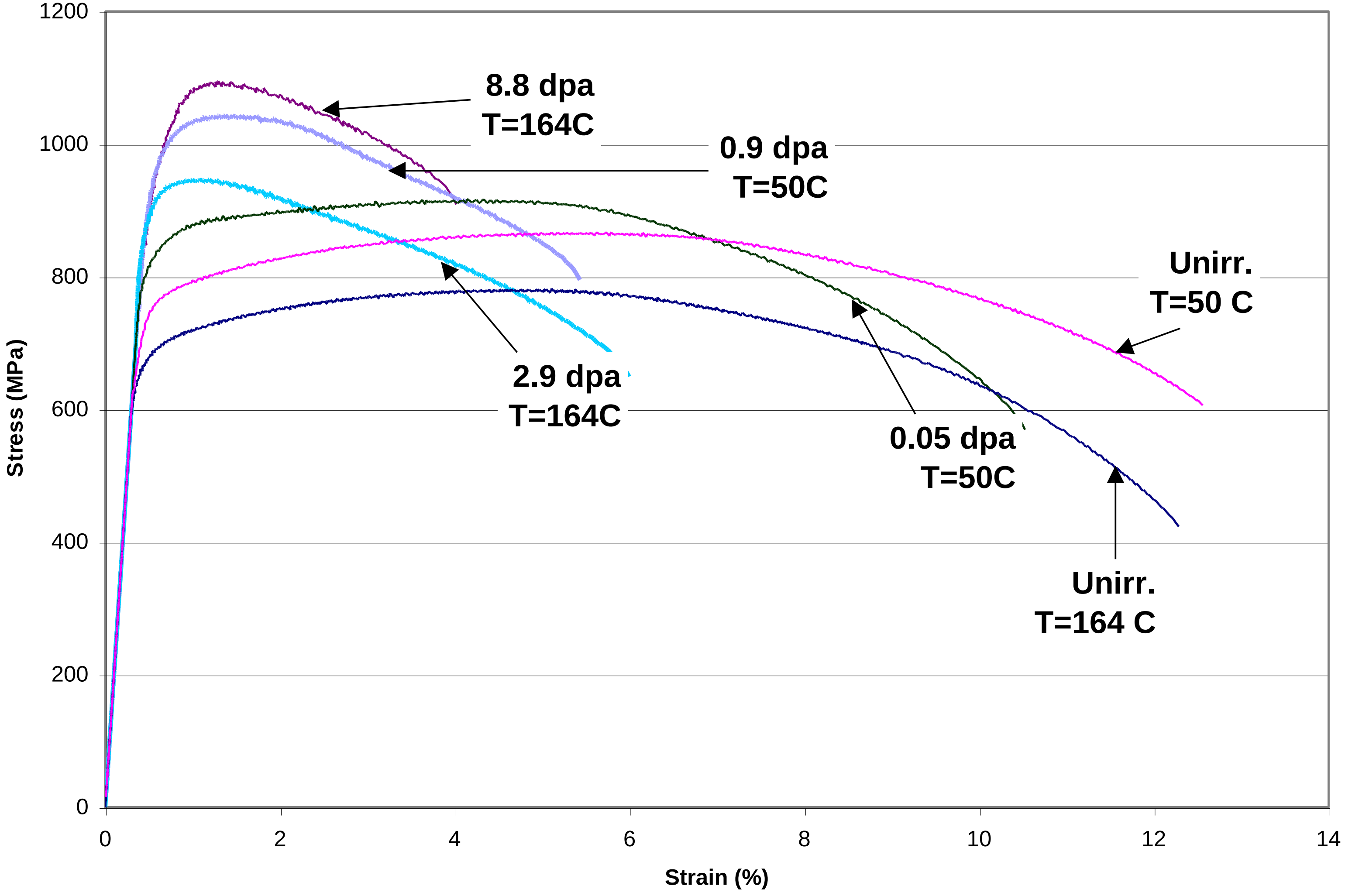
- Can we vary alloy composition to improve radiation tolerance (e.g. add precipitates or solutes)?
- Do other metal alloys show promise?

S.J. Zinkle and N.M. Ghoniem, Fus. Eng. Des. 51-52 (2000) 55; S.J. Zinkle et al. STAIF2002

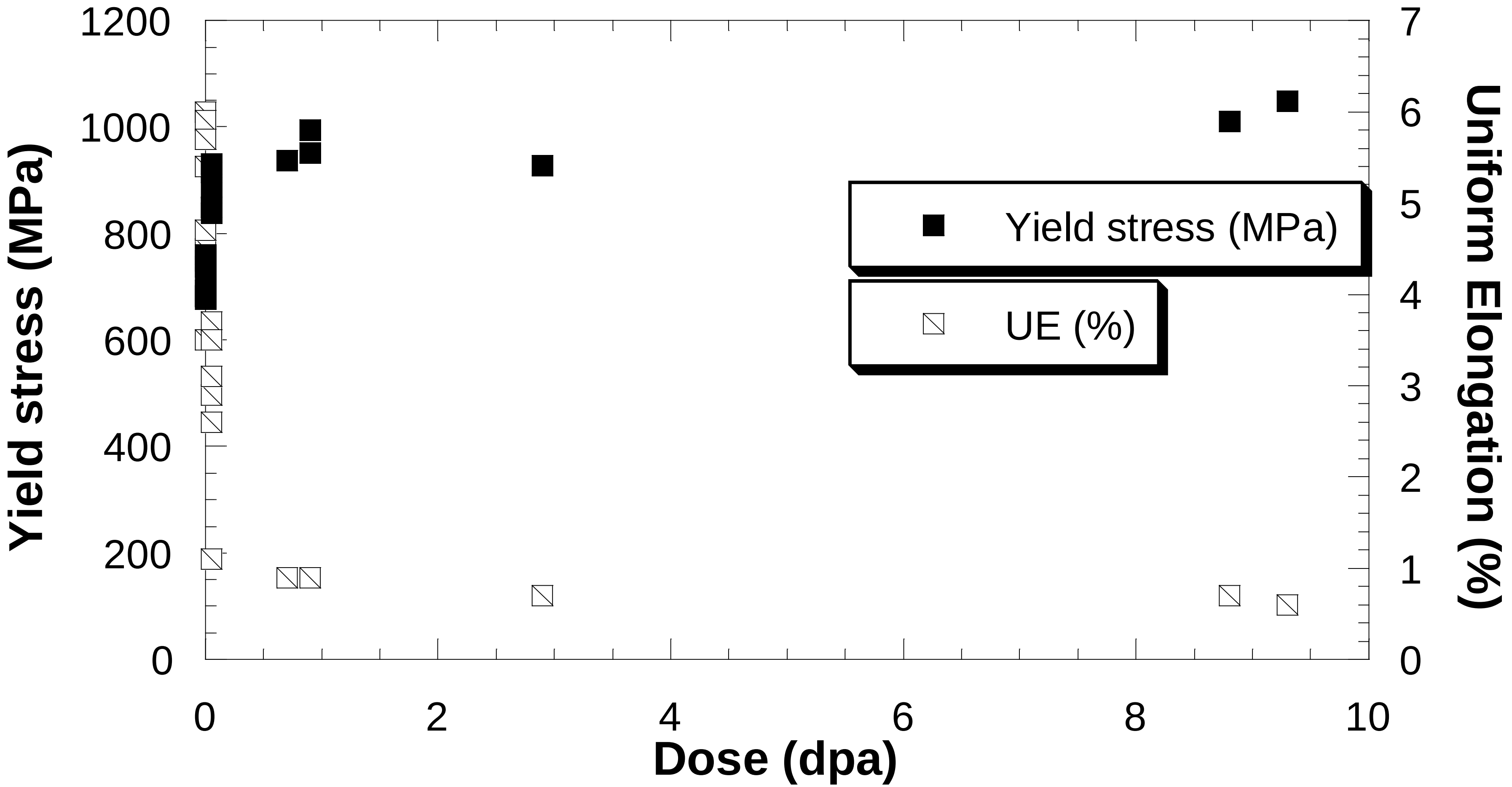
Irradiation Effects in Mod 9Cr-1Mo and HT-9 (BCC) at 50-400C



Stress/Strain Curves for Mod 9Cr-1Mo after irradiation in a Spallation Environment



The Change in the Tensile properties with dose for Mod9Cr-1Mo



Change in DBTT with Irradiation Dose for Mod 9Cr-1Mo

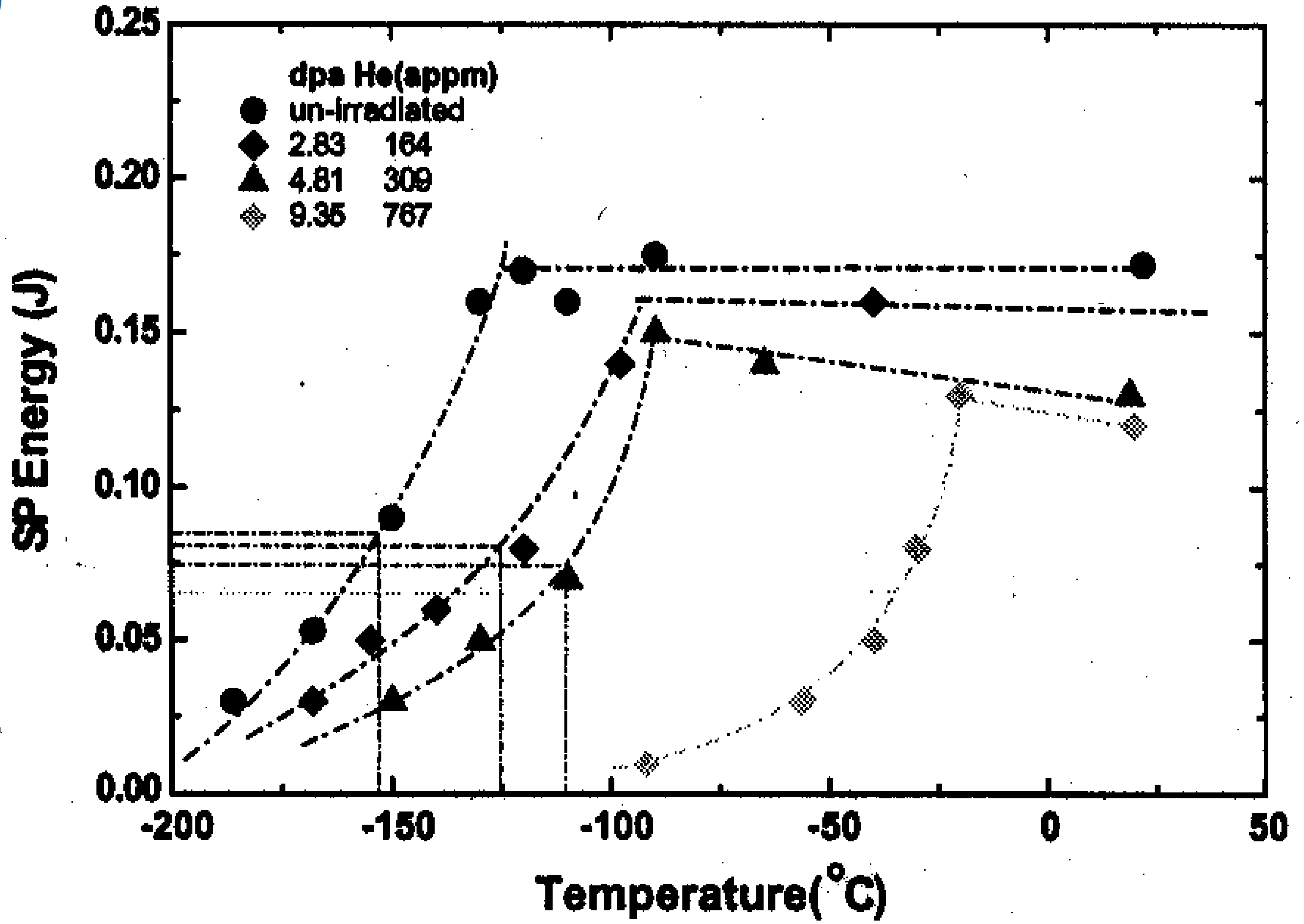
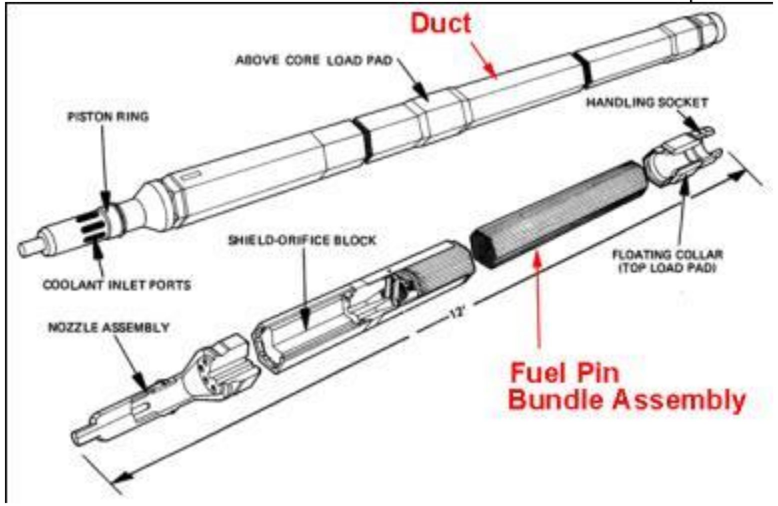


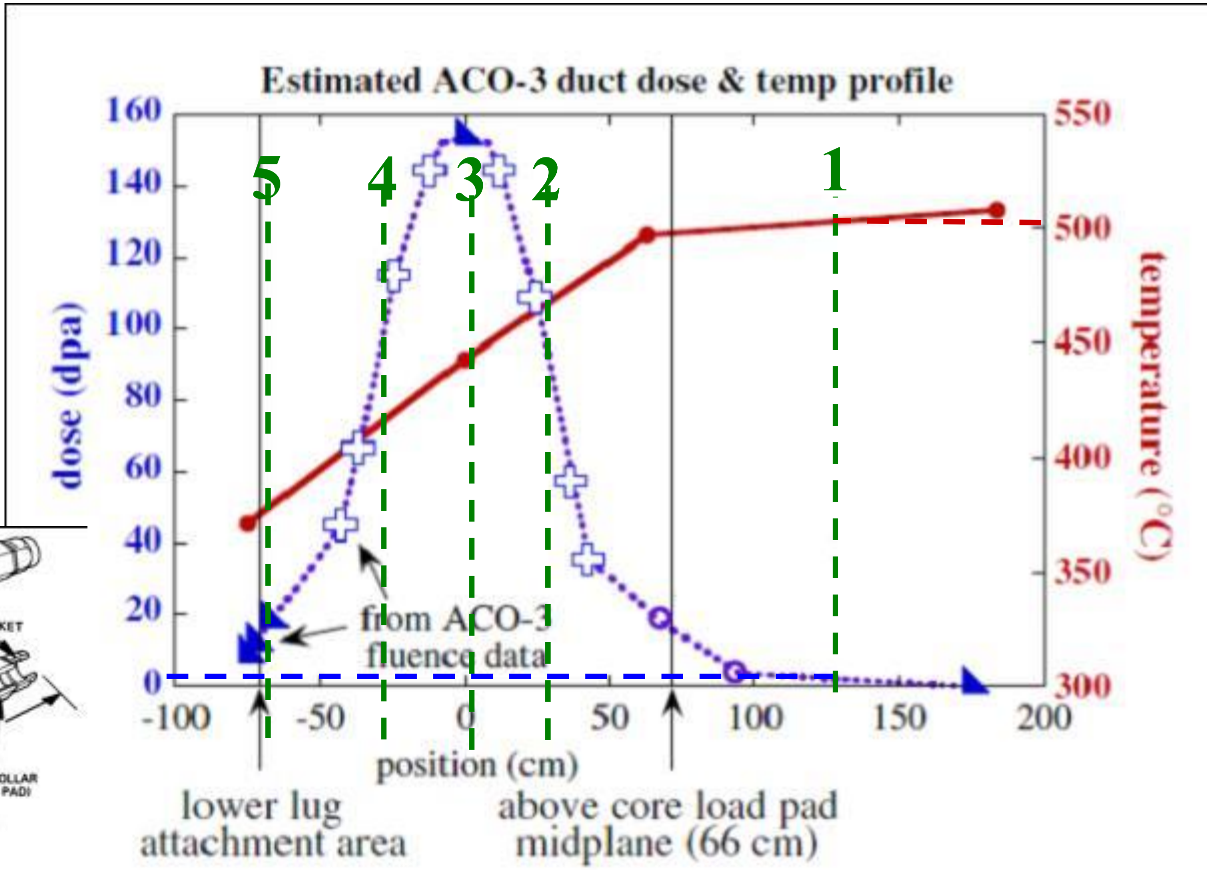
Fig. 6. Test temperature dependence of the SP energy for specimens of different irradiation doses.

From Dai et al., JNM, v. 318 (2003), 192-199

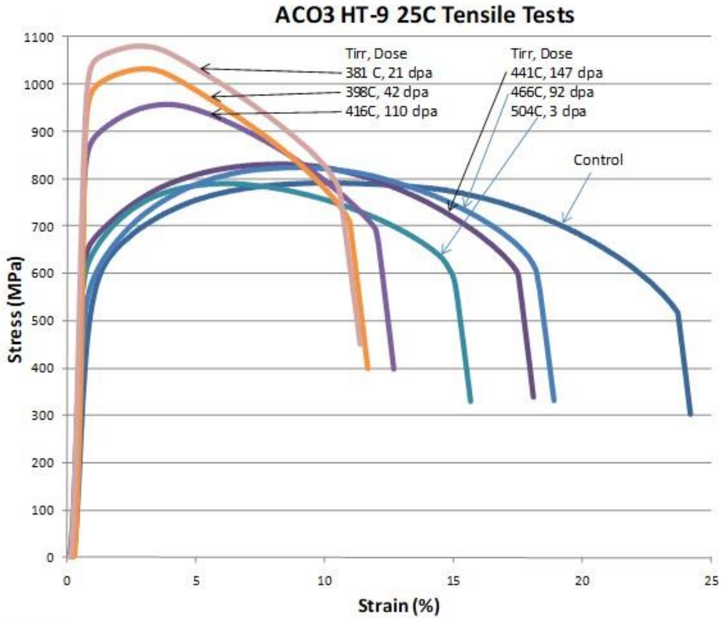
The ACO-3 duct was analyzed after irradiation in the Fast Flux Test Facility

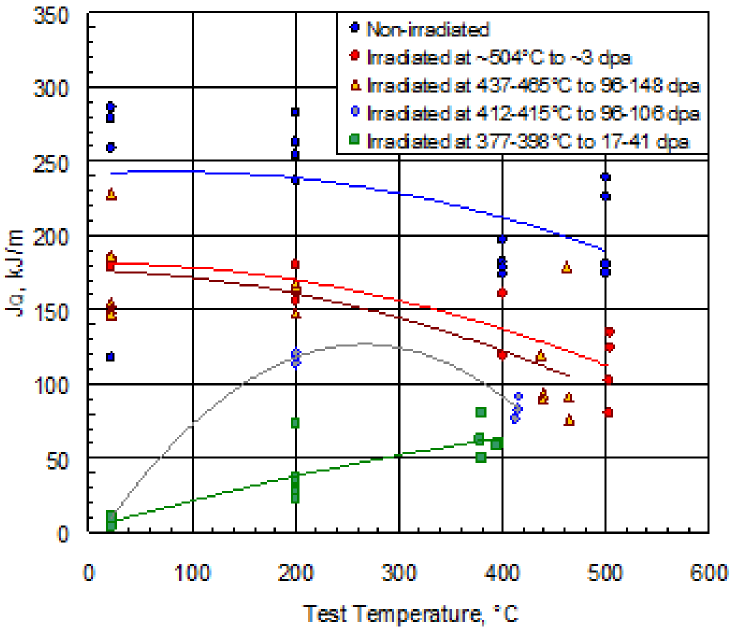
FFTF, Hanford site, WA

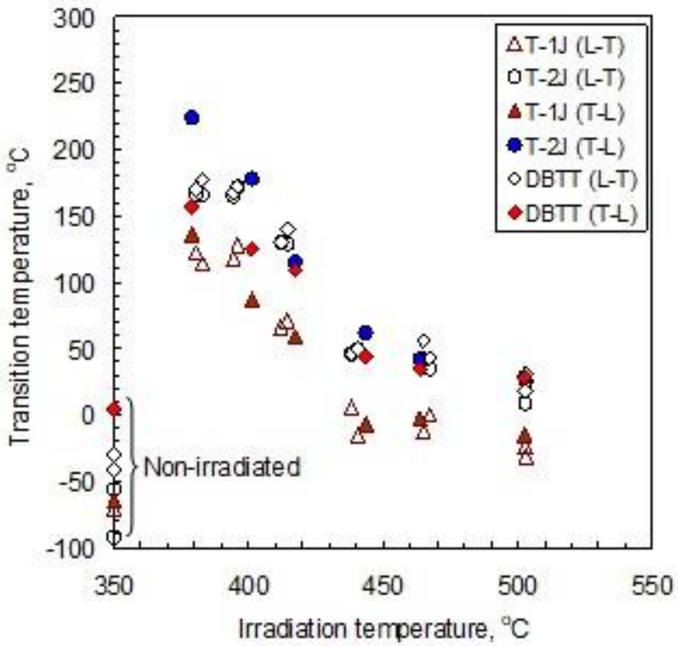


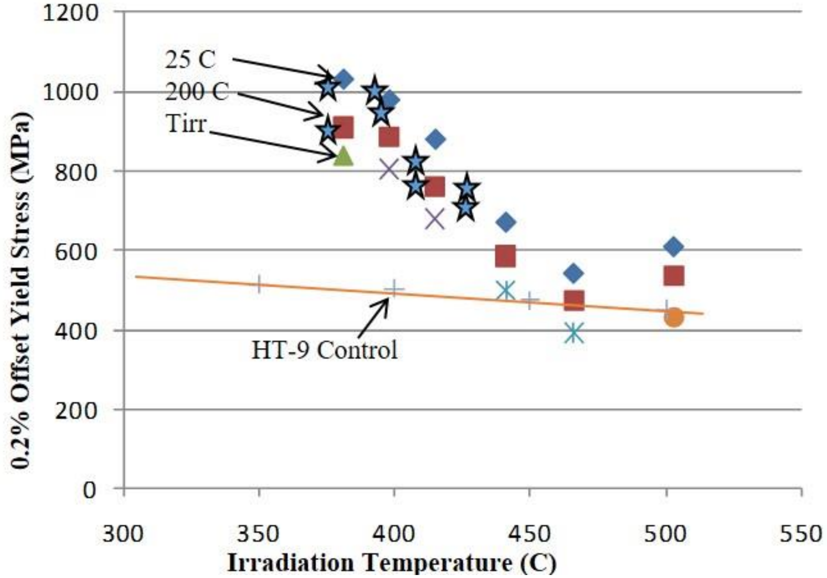


Mechanical Test Results on ACO-3 Duct Show Strong Effects of Irradiation Temperature

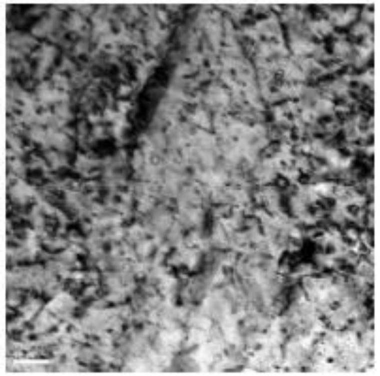








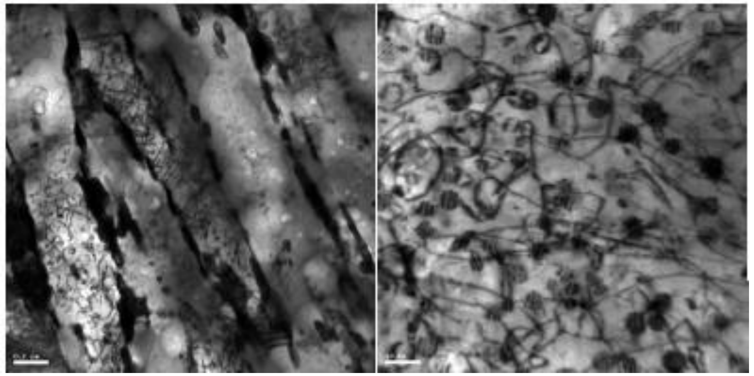
TEM analysis of ACO-3 Duct Material (B.H. Sencer, INL, O. Anderoglu, J. Van den Bosch, LANL)

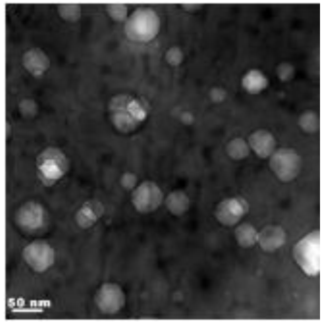


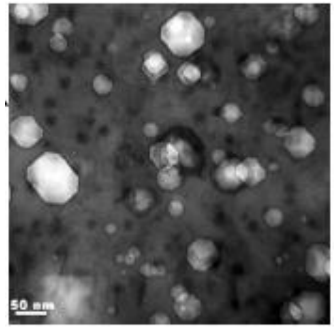
T=384C, 28 dpa

• G-phase precipitates and alpha prime observed

•No void swelling observed.







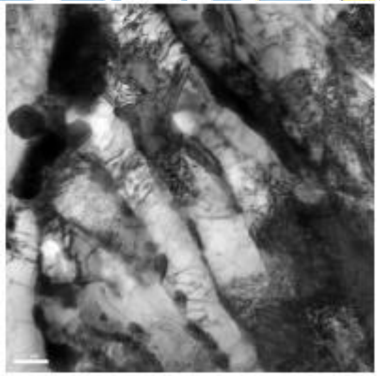
T=450C, 155 dpa

• Precipitation observed

• Dislocations of both a/2<111> and a<100>

• Loops of a<100>

• Void swelling observed (~0.3 %)



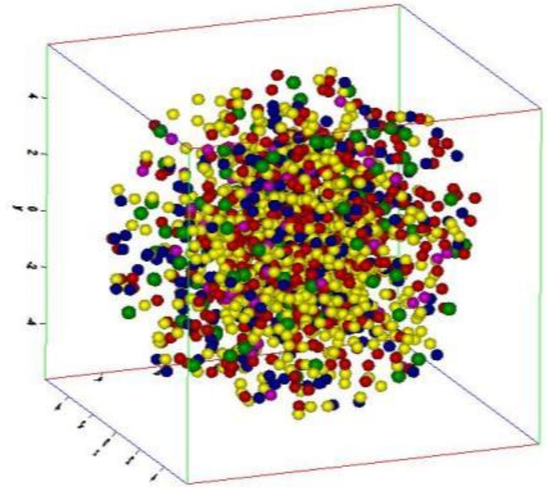
T=505C, 4 dpa

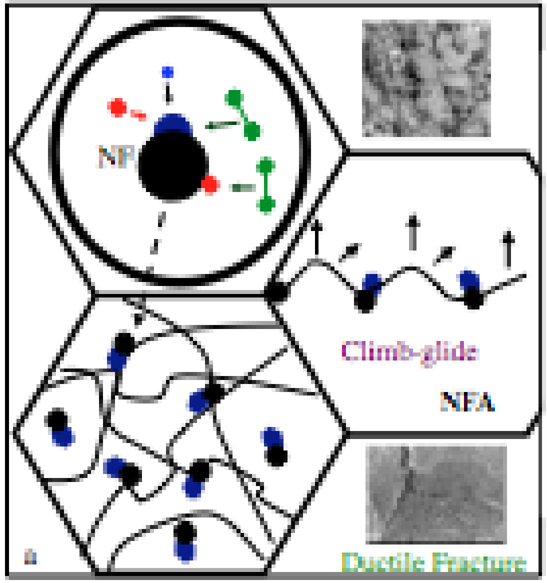
•No precipitation or void swelling observed.

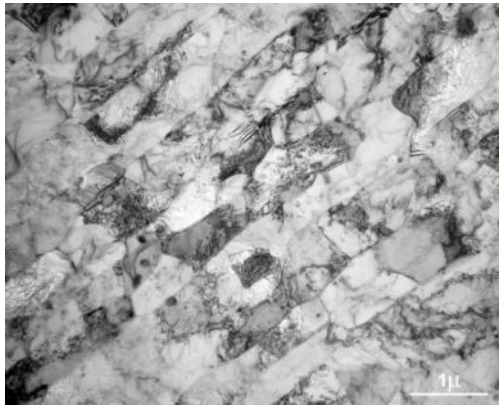
Nanostructured Ferritic Alloys Show Promise as Advanced Radiation Tolerant Materials

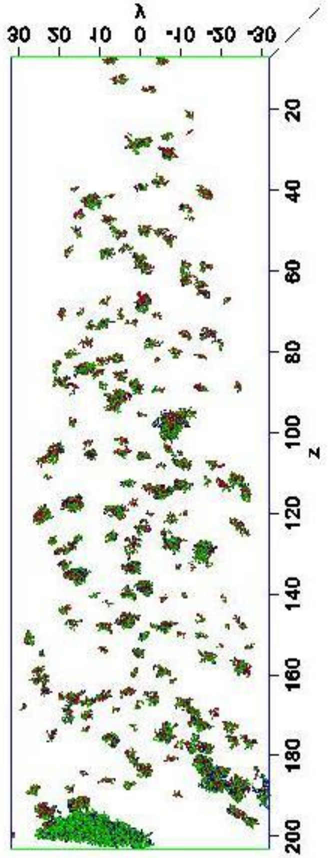
- Strength & damage resistance derives from a high density Ti-Y-O nano-features (NFs)
- NFs complex oxides ($Y_2Ti_2O_7$, Y_2TiO_5) and/or their transition phase precursors with high M/O & Ti/Y ratios (APT)
- MA dissolves Y and O which then precipitate along with Ti during hot consolidation (HIP or extrusion)
- Oxide dispersion strengthened alloys also have fine grains and high dislocation densities

Y-YO-Ti-TiO-O



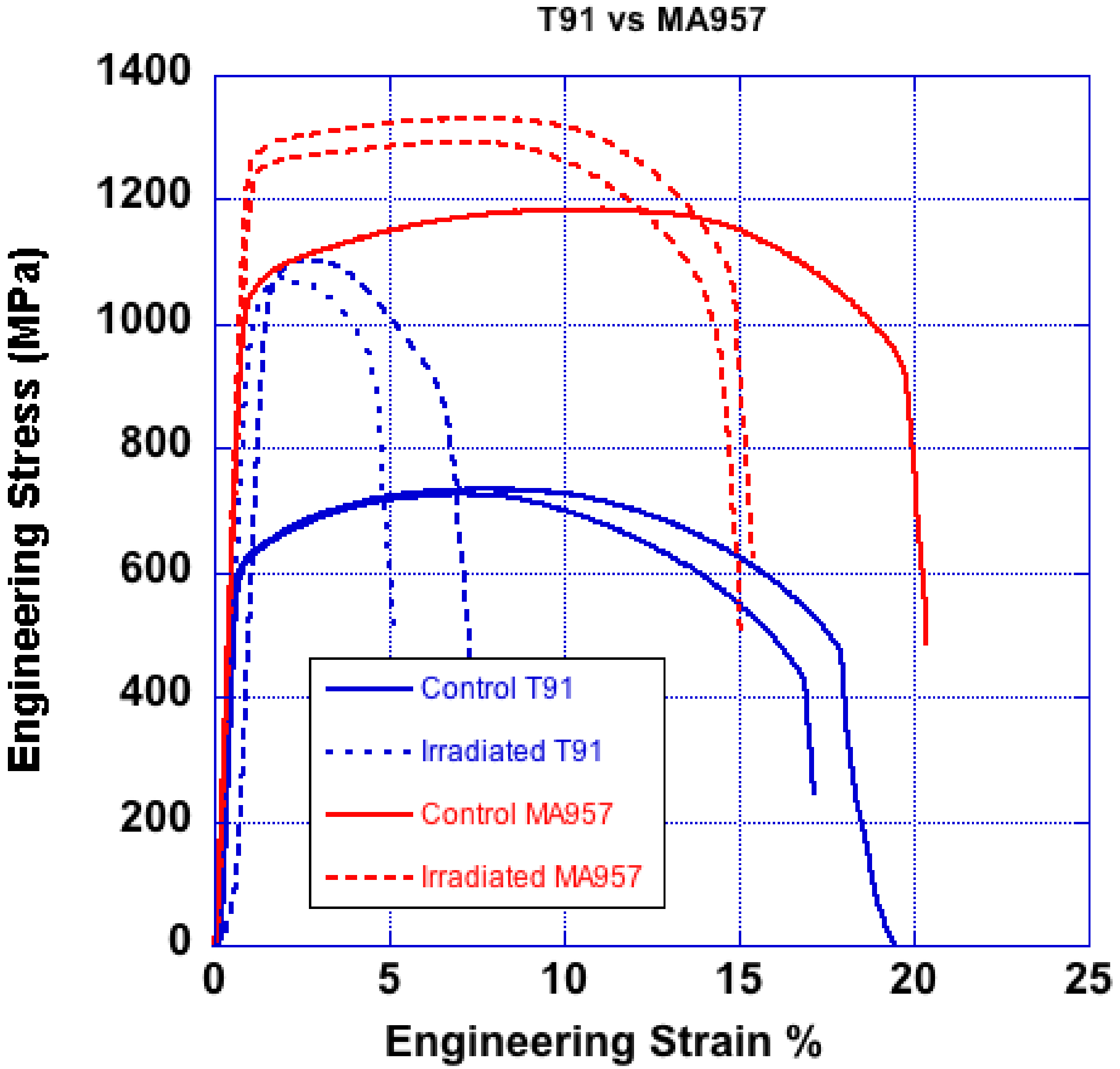






UCSB, LANL, ORNL

Ductility Retention observed in MA957 after irradiation to 6 dpa at 290C



LANL, UCSB, NSUF

Reactor Conditions/Materials Performance

BWR/LWR

Gen IV Reactors

VHTR- SCWR

SFR-LFR-MSR

Pressurized water reactors (PWR) and Boiling Water Reactors (BWR)- Present Reactor Fleet

GEN IV International Forum

Reactor Purpose:

Power production (1.5GWe/reactor)

Reactor Conditions

Water coolant (~288-360C)

Thermal and Fast neutron spectrum

2-4 dpa/year

Materials Issues

Cladding-

- Fuel clad chemical interaction
- Hydride formation
- Zircaloy corrosion

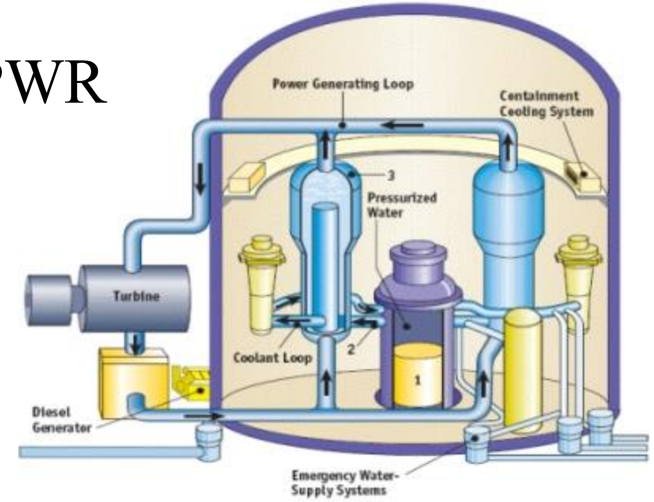
Water coolant piping

- Stress Corrosion Cracking /IASCC

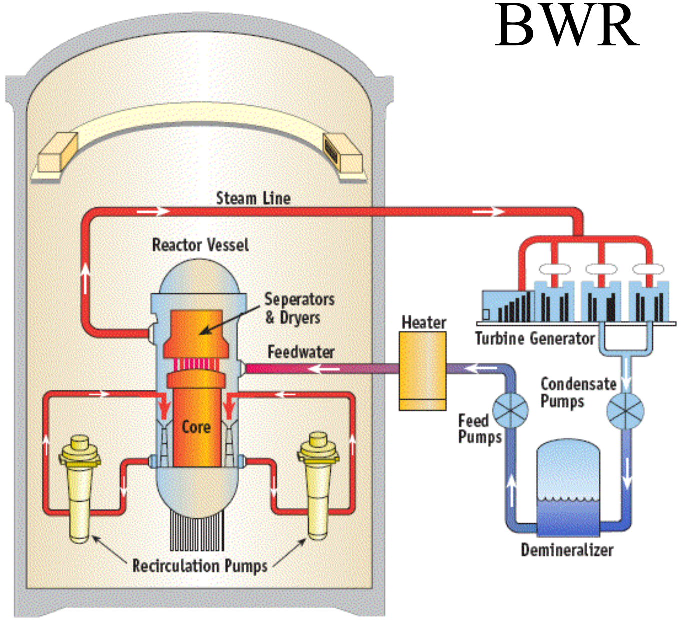
Pressure Vessel

- Aging effects

PWR



BWR



Construction materials for current reactor designs are diverse

	LWR	SFR	GFR/VHTR
Coolant	Water	Sodium	Helium
Temperature	288-360° C	400-550° C	550-1100° C
Cladding	Zirconium-based	9 or 12Cr steels	SiC/SiC
Core Internals	304/316 SS	316 SS	SiC/Alloy 800H
Vessel	Steel/308 SS	316 SS	Steel/316 SS
Heat Exchanger	Alloy 600/690	9-12Cr/316 SS	Alloy 617
Piping	304/316 SS	9-12Cr/316 SS	Alloy 617

- Despite considerable differences in operating parameters, there are common material uses between LWR and SFR applications

Very High Temperature Reactor VHTR- NGNP

Reactor Purpose:

More Efficient Power production

Inherent passive safety features

Reactor Conditions

He coolant

950C outlet temperature

600 MWt

Solid graphite block core

2-4 dpa/year

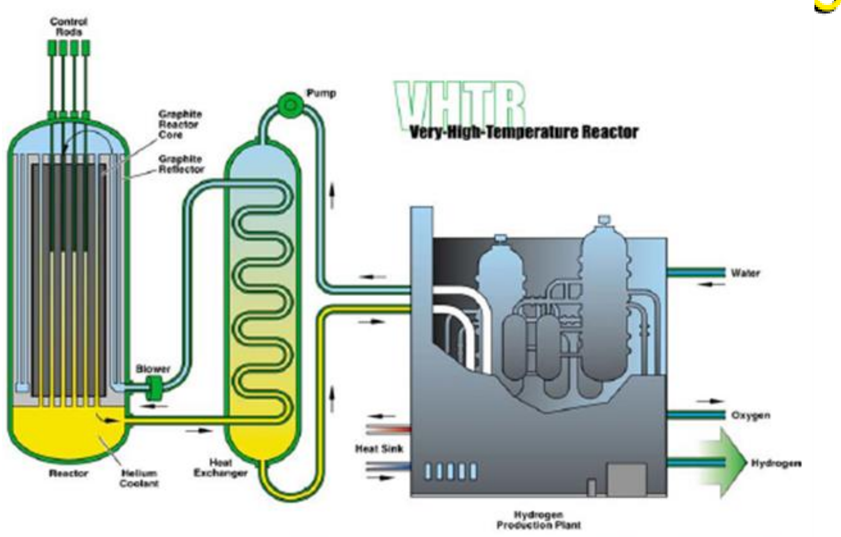
Materials Issues

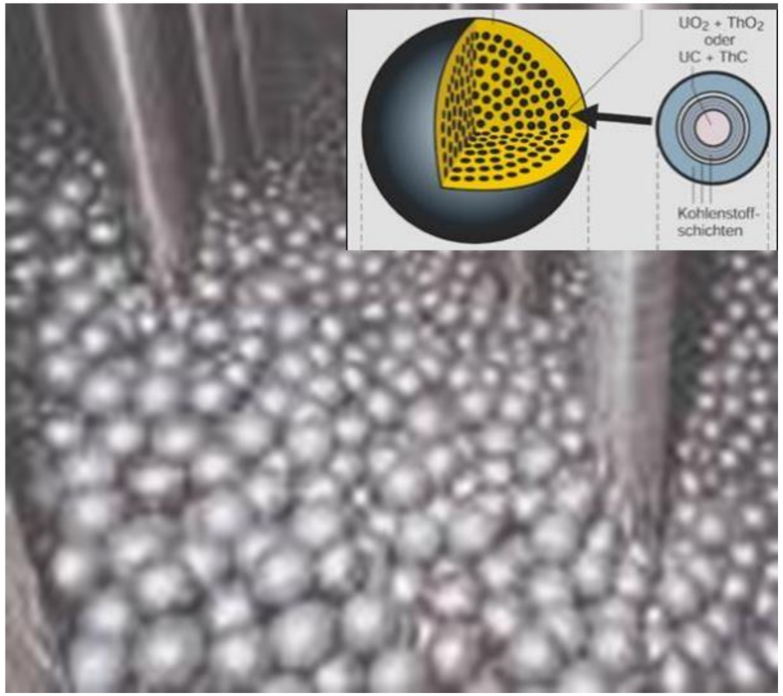
Improved metallic materials for VHTR pressure vessels (operating temperature ~450C).

Improvements in graphite properties (oxidation resistance and structural strength)

High Temperature Mechanical properties of coolant piping (e.g. Inconel 617)

Development of materials for the intermediate heat exchanger





Supercritical Water Reactor

Reactor Purpose:

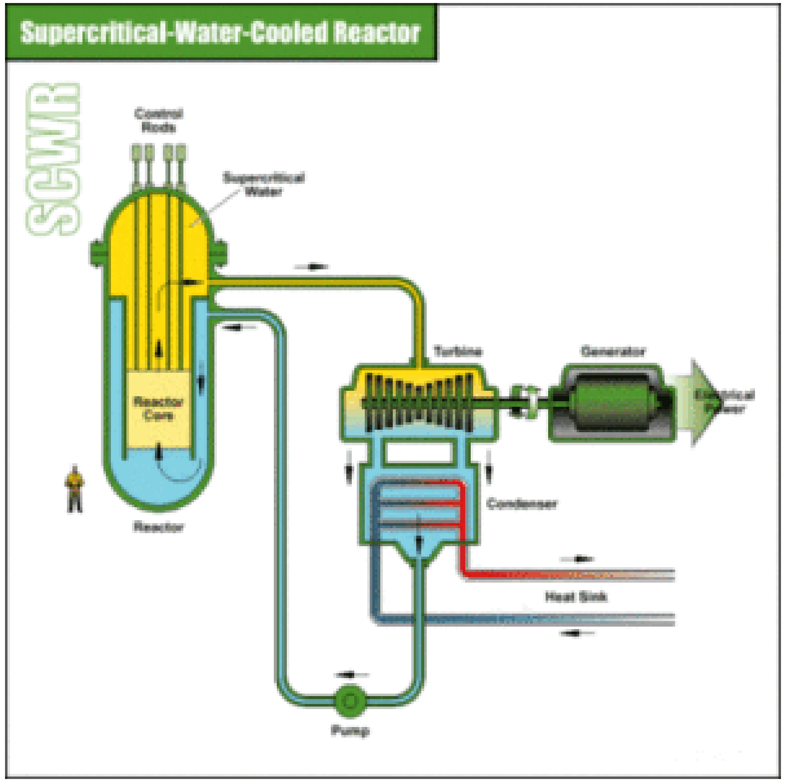
More Efficient Power production

Reactor Conditions

supercritical water coolant
550C outlet temperature
1700 MWe
>20 MPa
2-4 dpa/year

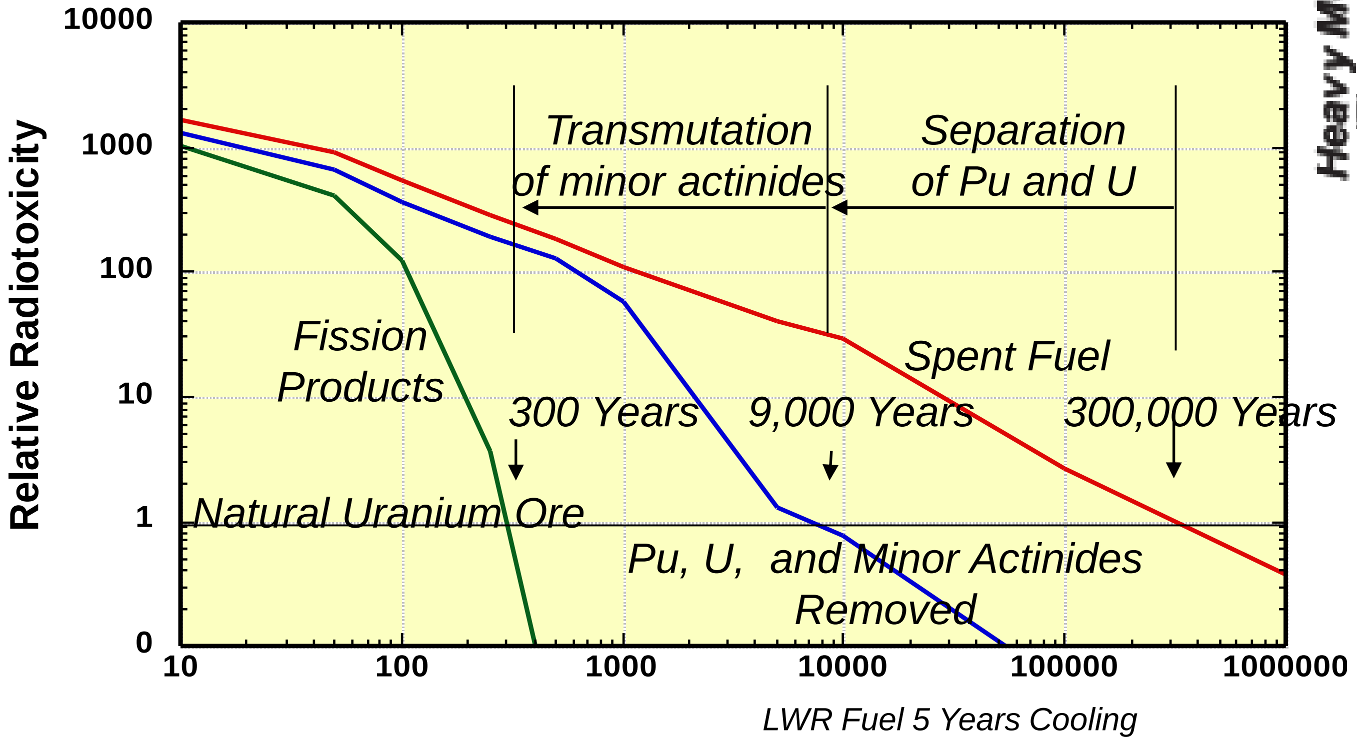
Materials Issues

corrosion and stress corrosion cracking,
radiolysis and water chemistry
dimensional and microstructural stability and strength,
embrittlement and creep resistance of fuel cladding and structural materials.
temperature range of 280–620°C and irradiation damage dose ranges of 10–30 displacements per atom (dpa) (thermal spectrum) and 100–150 dpa (fast spectrum)

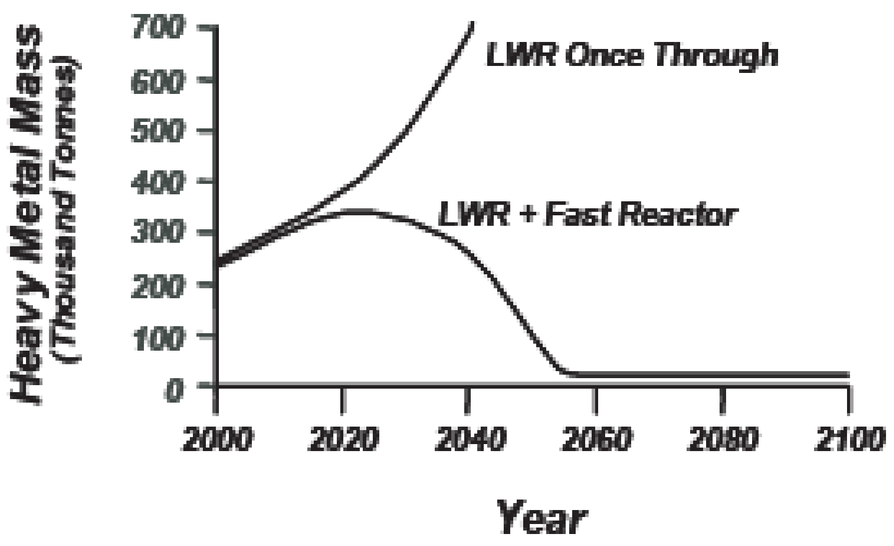


Why Fast reactors?

- Fast reactors are able to address the back end of the fuel cycle
- Produce energy out of waste
- Extend the fuel resources



Worldwide Spent Fuel

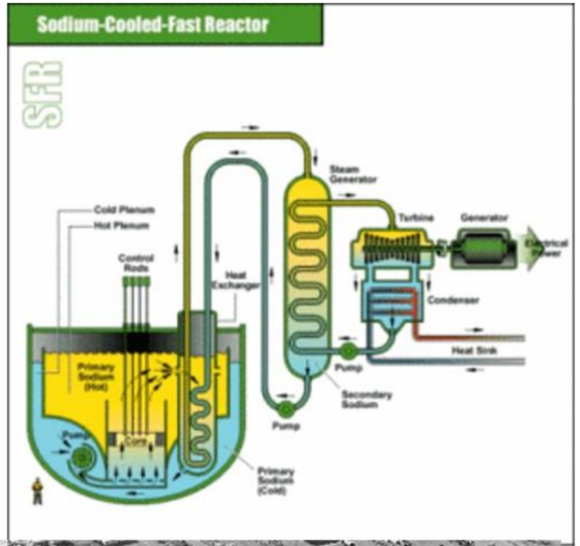


Sodium cooled fast reactor/ Lead Fast Reactor

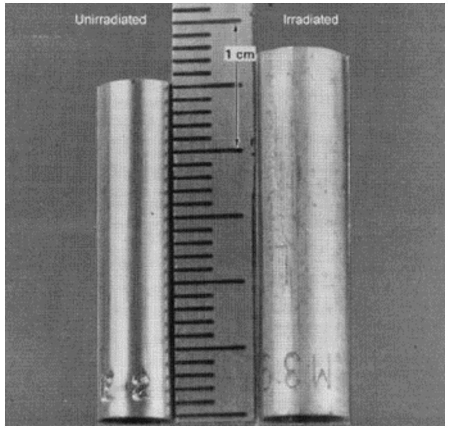
Reactor Purpose:
- High level nuclear waste Transmutation
- Power production
- Actinide management

Reactor Conditions
- Na, Pb or Pb/Bi coolant
- 550C to 800C outlet temperature
- 20-30 dpa/year

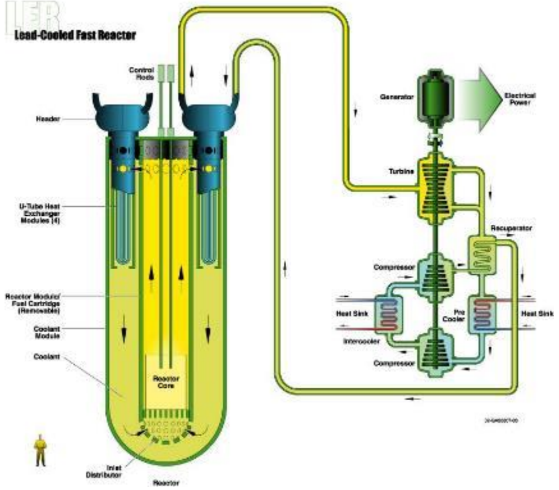
Materials Issues
- In Core-
  - High dose irradiation effects
  - FCCI
- Liquid metal corrosion
  - Lead corrosion of materials
  - Liquid metal embrittlement



FFTF research reactor in Hanford (1980-1993)



Swelling in 316L SS



Gas cooled fast reactor

Reactor Purpose:

High level nuclear waste Transmutation
More efficient Power production
Actinide management

Reactor Conditions

He or Supercritical CO_2 coolant
850C outlet temperature
Several Fuel options and core configurations
20-30 dpa/year

Materials Issues

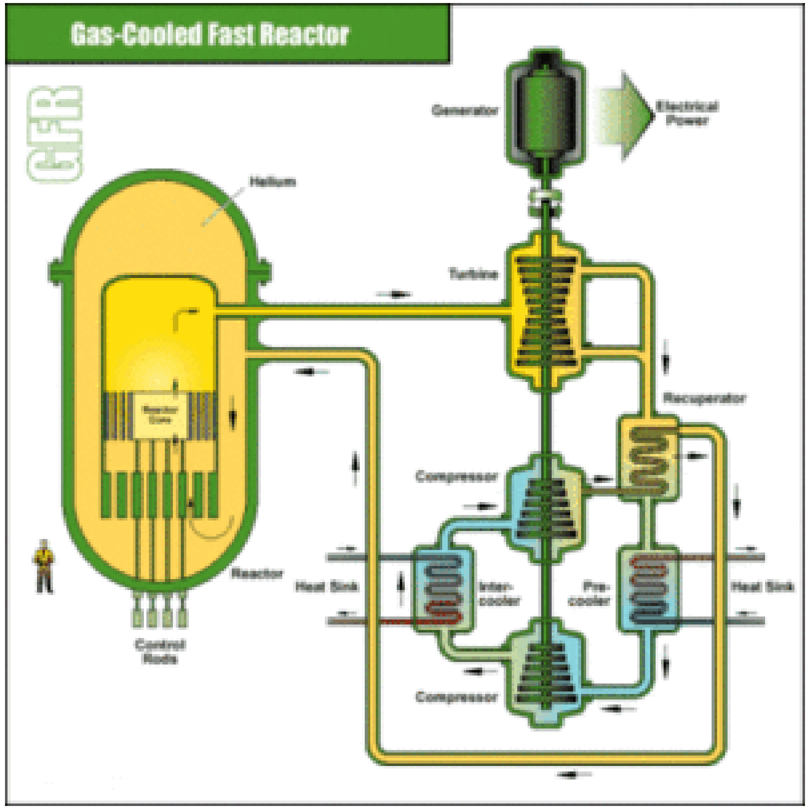
Fuel development (must achieve high-power density and retain fission gases at high burnup and temperature)

Proposed fuel is a composite ceramic (CERCER) with closely packed and coated actinide carbide kernels or fibers.

Alternative fuel concepts

fuel particles with large kernels and thin coatings and ceramic-clad solid solutions.

Nitride compounds, enriched 99.9% in N-15



Molten Salt Reactor

Reactor Purpose:

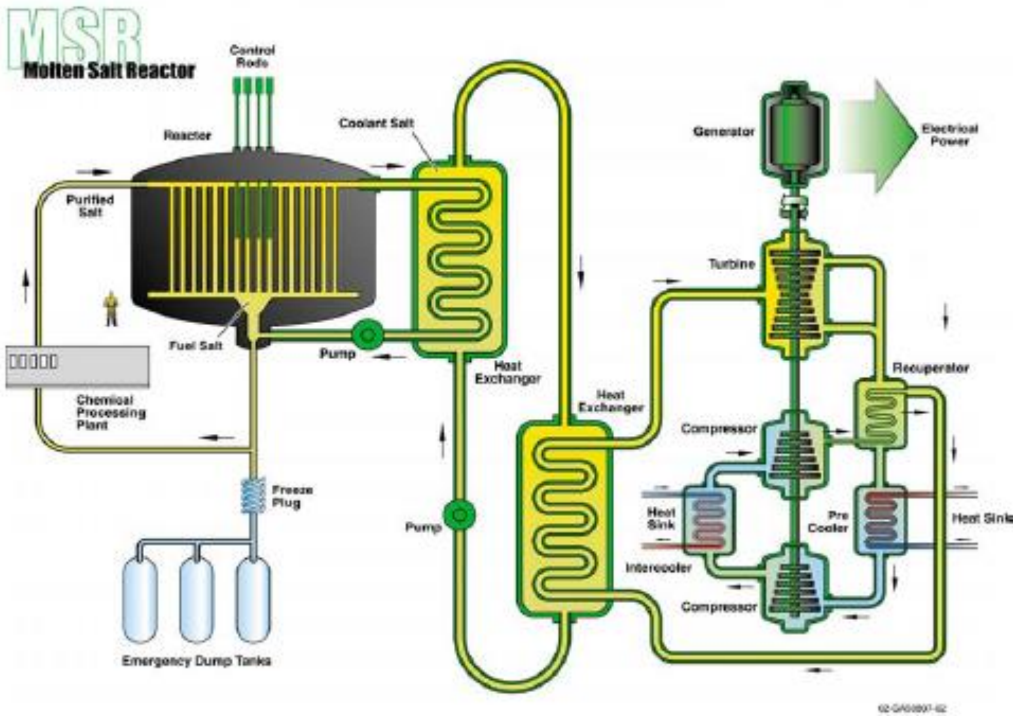
High level nuclear waste
Transmutation
Fast Reactor power
Liquid fuel core removes radiation effects concerns in the fuel

Reactor Conditions

Fuel: liquid Na, Zr, U and Pu fluorides or chlorides
700-800C outlet temperature
1000 Mwe
Core materials -Ni-based alloys (pressure vessel), graphite, SiC (solid fuel))
Low pressure (<0.5 MPa)
20-30 dpa/year for solid fuel clad

Major Materials Issues

- Materials compatibility testing in a controlled chemistry test loop
- Materials compatibility testing in a controlled chemistry test loop under irradiation.
- Radiation damage to pressure vessel and coolant piping

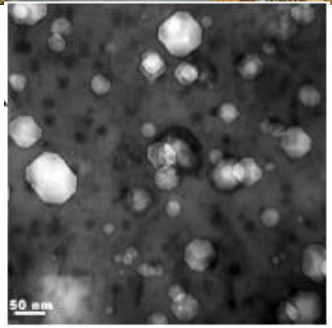


Summary Reactor Operating Conditions

Reactor Type	Fuel Materials	Fuel Temperature	Pellet to Clad bond	Coolant Type	Structural Materials for Core Internals	Lifetime Dose (dpa)	Structural Temperatures
Gen IV/ Lead Fast Reactor LFR	U/PuN; TRUN (enriched to N^{15})	500-600C	Lead	Pb or LBE	Ferritic/Martensitic Steel alloys	150-200	400-600C
Gen IV/ Sodium Fast Reactor SFR	Metal(U-TRU-10%Zr Alloy), MOX(TRU bearing)	600-800C (metal fuel) 800-2000C (Oxide fuel)	Sodium	Sodium	Ferritic/Martensitic Steel alloys	150-200	400-550C
Gen IV/ Gas cooled Fast Reactor GFR	UPuC/SiC (50/50%) with 20% Pu content ; Solid Solution fuel with SiC/SiC cladding	2000 +	Helium	Helium	Nickel Superalloys /Ceramic Composites	80	500-1200C
Fusion Energy	N/A	N/A	N/A	Pb-Li	F/M steels; Vanadium alloys; Ceramics	150	300-1000C
LWR – PWR, BWR	UO2	800-1600C	Helium	Water	316L.ferritic pressure vessel, Zircalloy cladding	Cladding ~10 dpa, Internals up to 80 dpa	200-300C
Very High Temperature Reactor (VHTR, NGNP)	TRISO	800-2000C	Intimate contact	Helium	Ni-based alloys, ceramics and graphite	~10 dpa	700-1000C
Supercritical Water Reactor (SCWR)	UO2	800-2000C	Helium	Water	F/M steels, austenitic steels	10-30 thermal 100-150 Fast	300-600C
Molten Salt Reactor (MSR)	Na, Zr, U, Pu fluorides	700-800C	N/A	N/A	Ni-based alloys, graphite	100-150 dpa	600-800C

Summary of Materials Performance Issues

Reactor type	Primary Materials	Performance Issues
Light Water Reactors (PWR/BWR)	Ferritic pressure vessel steels, Fe-based austenitic stainless steels, zirconium alloys	IGSCC, IASCC, Fuel clad mechanical interaction, hydriding, Radiation embrittlement (DBTT), hydrogen embrittlement
Very High Temperature Reactor (VHTR)	Ni-based superalloys, Graphite, ferritic/martensitic steels, W/Mo Alloys, SiC/SiC composites	Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation
Sodium Fast Reactor (SFR)	Fe-based austenitic SS, Ferritic/martensitic steels,	Radiation Embrittlement (DBTT), toughness, helium embrittlement, swelling, RIS, corrosion, FCCI
Lead Fast Reactor (LFR)	Fe-based austenitic SS, Ferritic/martensitic steels,	Radiation Embrittlement (DBTT), toughness, helium embrittlement, swelling, RIS, corrosion, FCCI, liquid metal embrittlement
Supercritical Water Reactor (SCWR)	Ferritic pressure vessel steels, Fe-based austenitic stainless steels, zirconium alloys, ferritic/martensitic steels	IGSCC, IASCC, Fuel clad mechanical interaction, hydriding, Radiation/helium embrittlement (DBTT), swelling, RIS, corrosion, toughness
Gas Fast Reactor	Ceramics (carbides, nitrides), ceramic composites, nickel superalloys	Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation
Molten Salt Reactor	Ni-based alloys, graphite, coatings	Corrosion, Helium embrittlement, creep strength, swelling, RIS, transmutation, toughness, oxidation



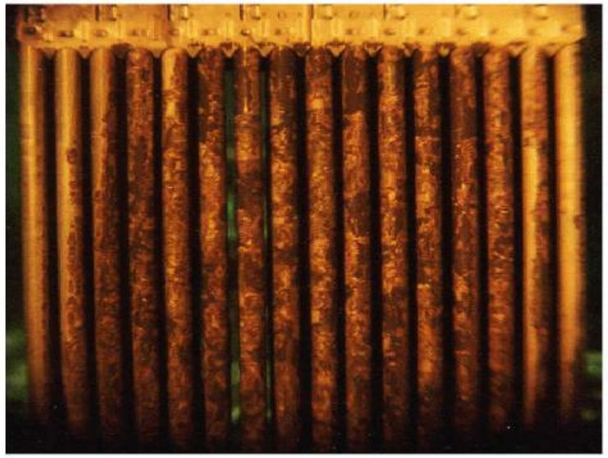
Void development in HT-9, 155 dpa

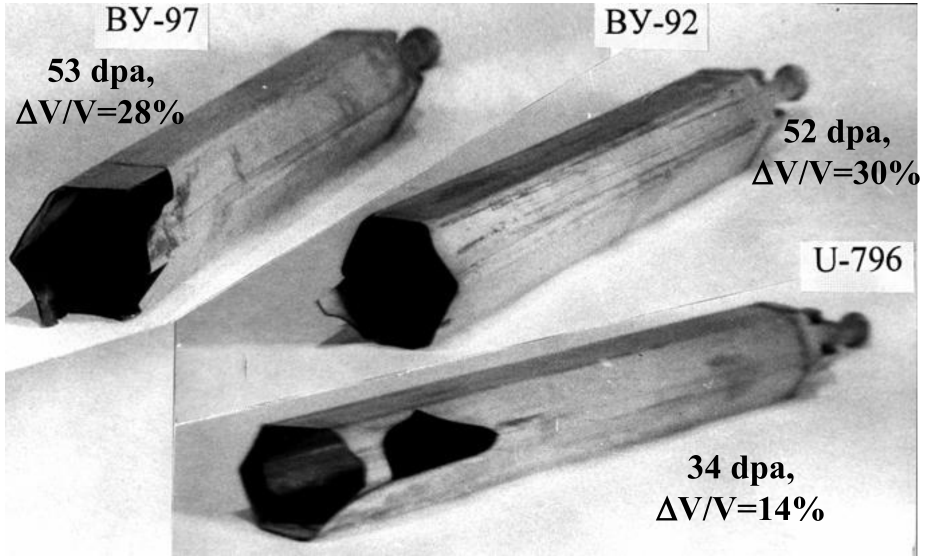
316l steel tube after irradiation

Questions???

Grid-to-Rod Fretting

CRUD





Fast Reactor Duct Failure

Upcoming webinars

21 March 2018	SCK•CEN's R&D on MYRRHA	Prof. Dr. H.C. Hamid Ait Abderrahim, SCK-CEN, Belgium
18 April 2018	Russia BN 600 and BN 800	Dr. Iiuri Ashurko, Institute of Power and Engineering, Russia
23 May 2018	Proliferation Resistance of Gen IV Systems	Dr. Robert Bari, Brookhaven National Laboratory, USA

4th GIF Symposium

16-17 October 2018

at the 8th edition of Atoms for the Future
UIC Paris, France

GEN IV International Forum

http://gifsymposium2018.gen-4.org/

Call for abstracts Extended Deadline - 31 March 2018

Track 1 & 2: Progress on Gen IV systems
Track 3: Human capital development
Track 4: Research infrastructures
Track 5: Safety and security
Track 6: Fuels and materials
Track 7: Advanced components and systems for Gen IV reactors
Track 8: Integration of nuclear reactors in low carbon energy systems
Track 9: Decommissioning & Waste Management
Track 10: Operation, Maintenance, Simulation & Training
Track 11: Construction of nuclear reactors

The symposium has **two major objectives**:

- to review the progress achieved for each system against the R&D goals of the 2014 Technology Roadmap Update.
- to identify the remaining challenges and associated R&D goals for the next decade necessary for the demonstration and/or deployment of the Gen IV systems, and the goal of establishing nuclear energy as a necessary element in the world's long-term sustainable carbon-free energy mix.

MSc and PhD students, young professionals, policy makers and nuclear stakeholders are encouraged to participate