U.S. DEPARTMENT OF ENERGY | Nuclear Energy

Fuel Cycle Research and Development

Advanced Cladding Materials for Fuels

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Outline

- Advanced Fuels Campaign in FCRD
- Materials Grand Challenges
- Clad Materials Issues
- Nanofeatures to improve radiation tolerance
- New research techniques at the nanoscale
- Summary

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Advanced Fuels Campaign Mission & Objectives

Mission

Develop and demonstrate fabrication processes and in-pile performance of advanced fuels/targets (including the cladding) to support the different fuel cycle options defined in the NE roadmap.

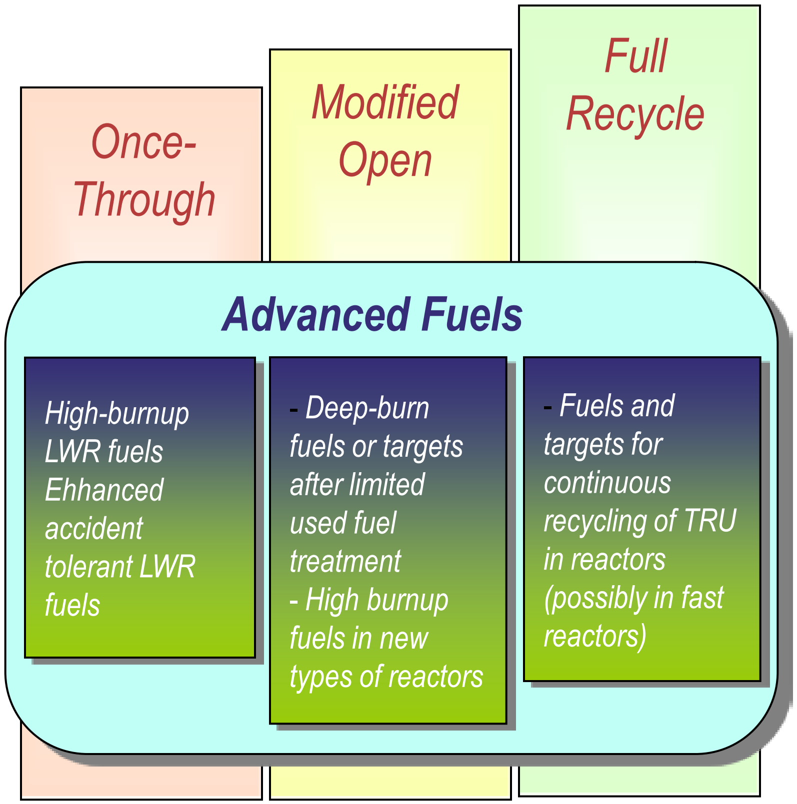
Objectives

Development of the fuels/targets that

- Increases the efficiency of nuclear energy production
- Maximize the utilization of natural resources (Uranium, Thorium)
- Minimizes generation of high-level nuclear waste (spent fuel)
- Minimize the risk of nuclear proliferation

Grand Challenges

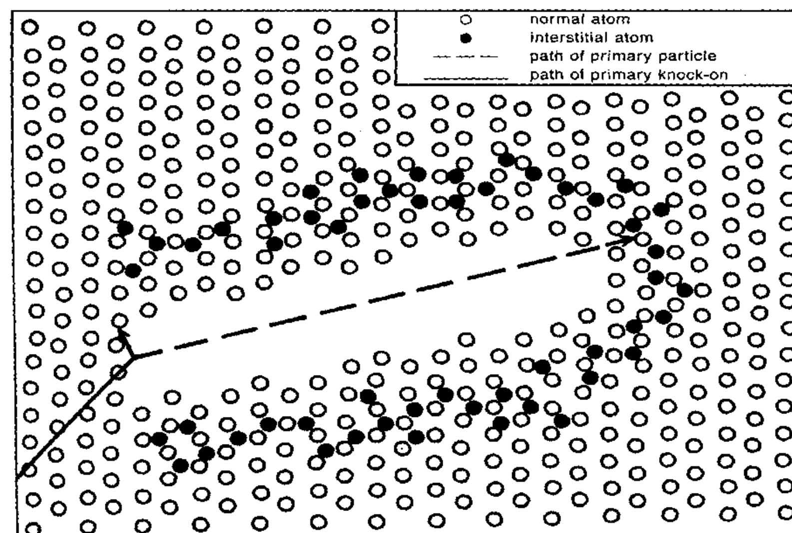
- Multi-fold increase in fuel burnup over the currently known technologies
- Multi-fold decrease in fabrication losses with highly efficient predictable and repeatable processes
- Develop and test advanced alloys for Next Generation LWR Fuels with Enhanced Performance and Safety and Reduced Waste Generation



Grand Challenge for Core Materials to Enable Multifold Increases in Burnup for Fuels

Develop and test advanced alloys suitable for clad and duct and other high dose core components to >400 dpa over the clad /duct operating conditions

- Irradiation tolerant
  - *Resists swelling and creep*
  - *Does not accumulate damage (resists hardening and embrittlement)*
  - *Stable microstructure (resists radiation induced segregation)*
  - *Manages helium or other gas buildup*
  - *Stable with Transmutation impurity buildup*
- Resist chemical interaction with fuel (for the cladding)
  - *Not reactive with fuel*
  - *Prevent diffusion into cladding*
- Corrosion resistance with coolant
  - *Protective oxide layer*
  - *Non reactive with coolant*
- Weldable and Processed into tube form

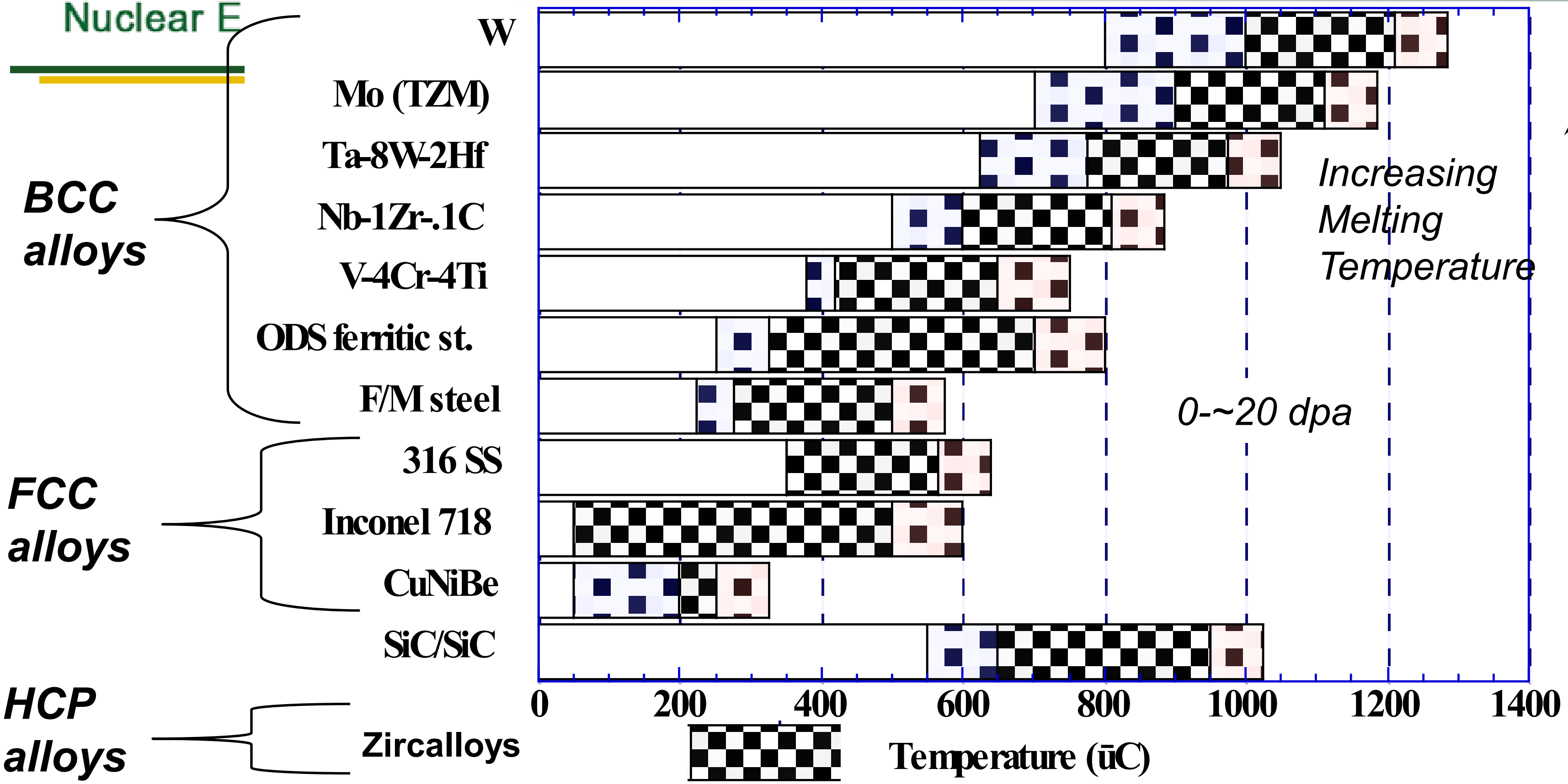


Challenges for Developing Core Materials For Next Generation LWR Fuels

Develop and test advanced alloys for Next Generation LWR Fuels with Enhanced Performance and Safety and Reduced Waste Generation

- Low Thermal Neutron Crossection
  - *Element selection (e.g. Zr, Mg)*
  - *Reduce cladding wall thickness*
- Irradiation tolerant to 20-40 dpa
  - *Resists swelling and irradiation creep*
  - *Does not accumulate damage (resists hardening and embrittlement)*
  - *Stable microstructure (resists radiation induced segregation)*
- Mechanically robust under loading and transportation conditions
- Compatibility with Fuel and Coolant
  - *Resists stress corrosion cracking*
  - *Resists accident conditions (e.g. high temperature steam)*
  - *Resists abnormal coolant changes (e.g. salt water)*
- Weldable and Processed into tube form
  - *Maintain hermetic seal under normal/off-normal conditions*

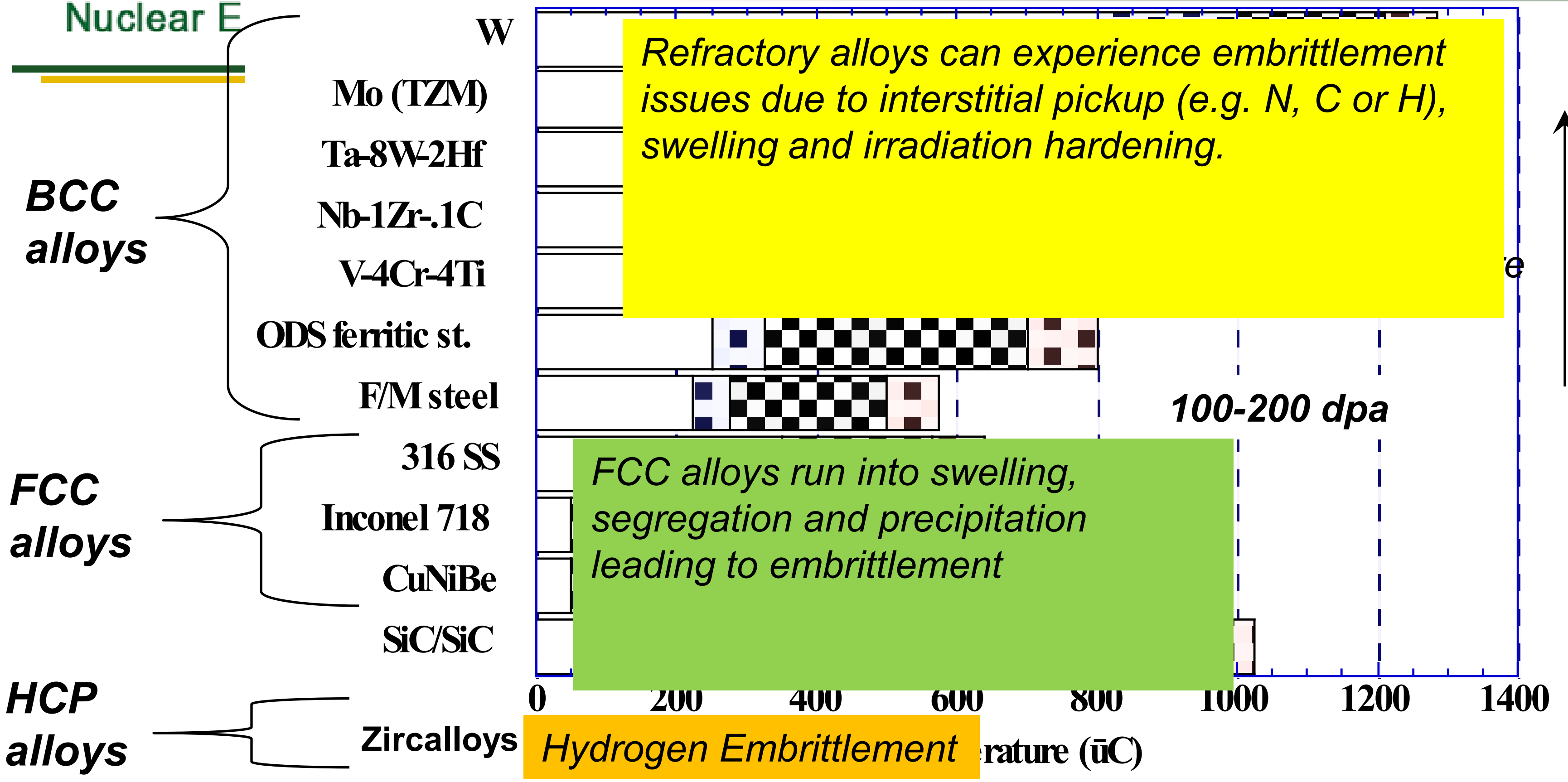
Survey of Materials Limits over Reactor Irradiation Temperatures



- *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, precipitation and radiation induced segregation.*
- *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

Survey of Materials Limits over higher doses to 200 dpa



Nanofeatures to improve radiation tolerance-
Can we vary alloy composition to improve radiation tolerance (e.g. add precipitates or solutes)?
Can we reduce grain size for improved radiation tolerance?
Aim to decrease low temperature embrittlement, reduce swelling or increase high temperature creep strength

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

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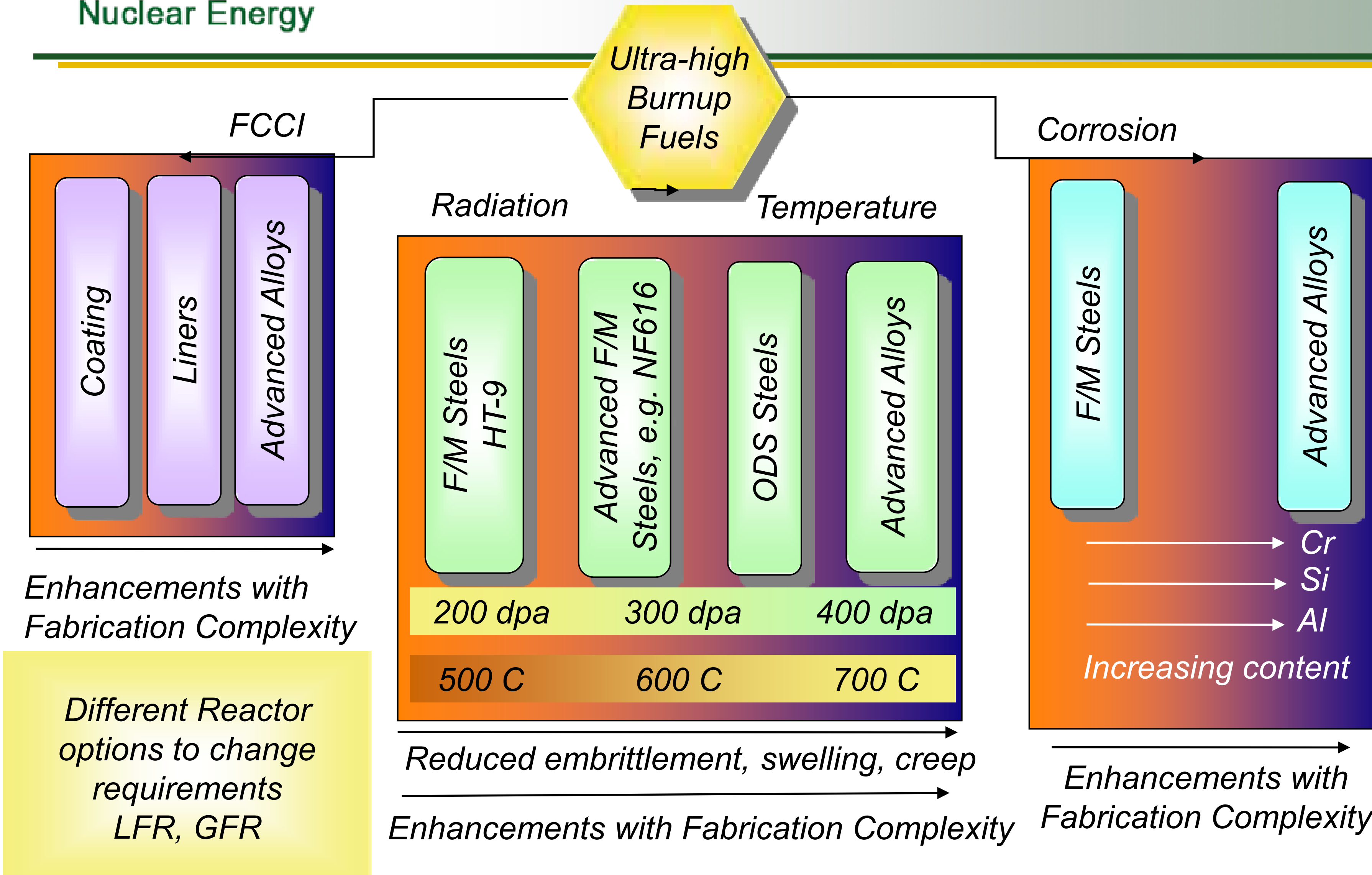
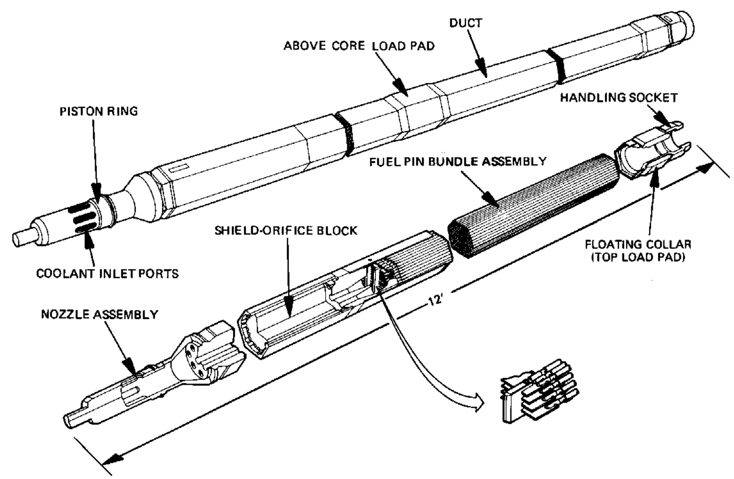
Science-based vision to Core Materials Development

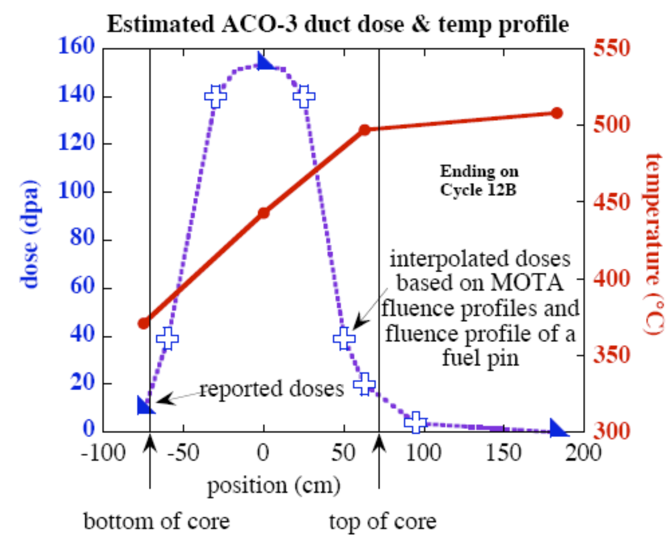


Core Materials Research and Development

Develop the knowledge base up to 200 dpa- High Dose Core Materials Irradiation Data

- ACO-3 Duct Testing
  - *Fracture Toughness testing*
  - *Charpy Testing*
  - *Tensile Testing*
  - *SANS measurements*
- FFTF/MOTA Specimens and Testing

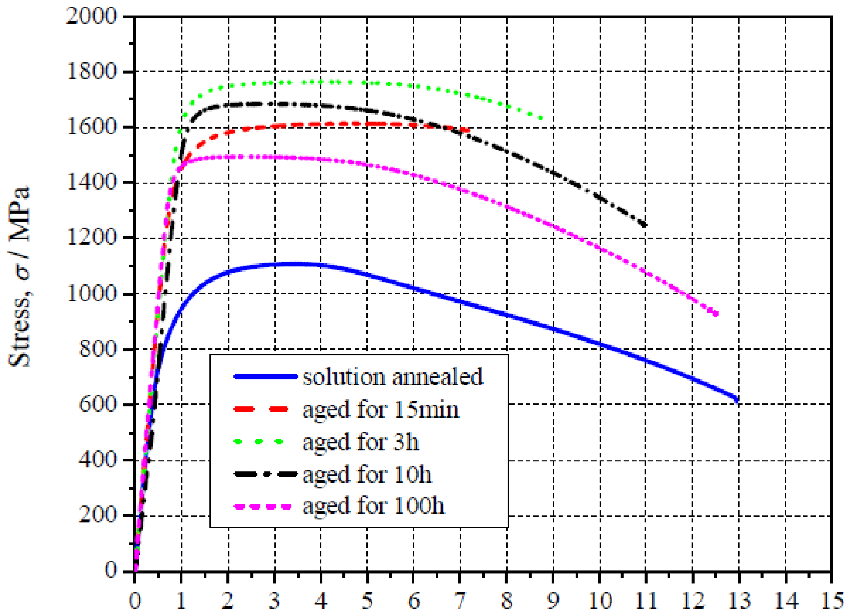


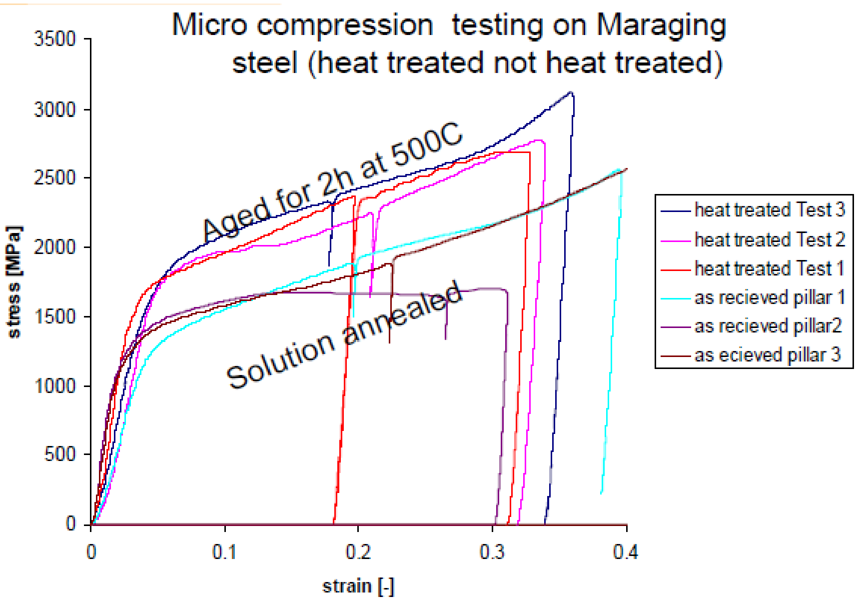


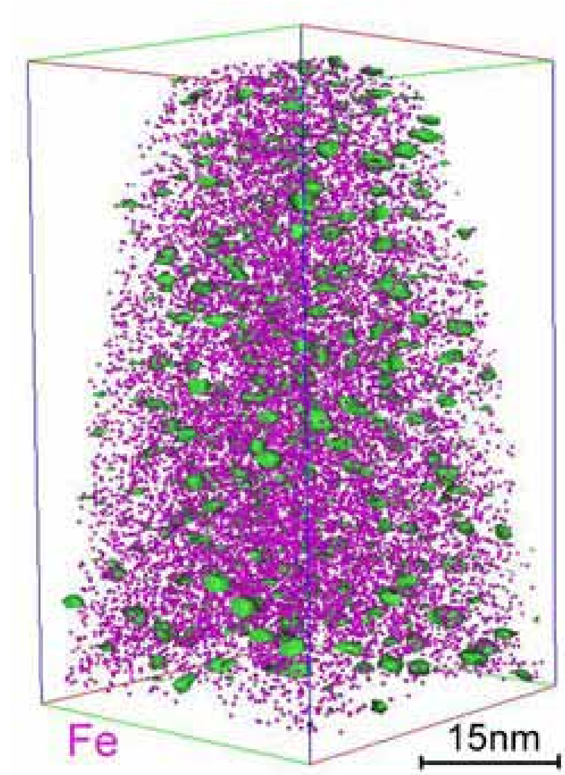
Advanced Material Development

- Develop coatings/liners to Mitigate FCCI
- Develop and test Advanced Cladding materials
  - *Improved Processing of Advanced ODS Alloys*

Strength can be Improved by 60% by adding nanofeatures in a Fe-Cr-Ni steel



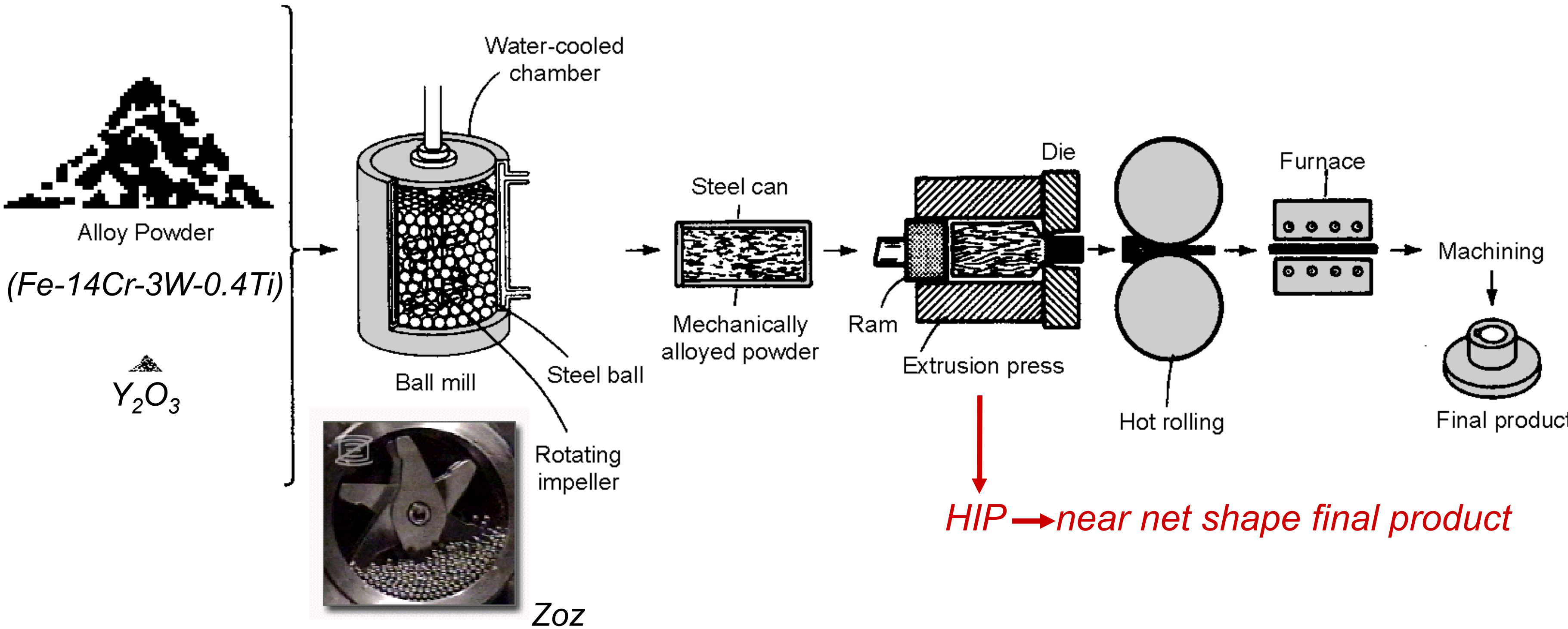




M. Schober et al

A ferritic ODS alloy, 14YWT, is produced by Mechanical Alloying

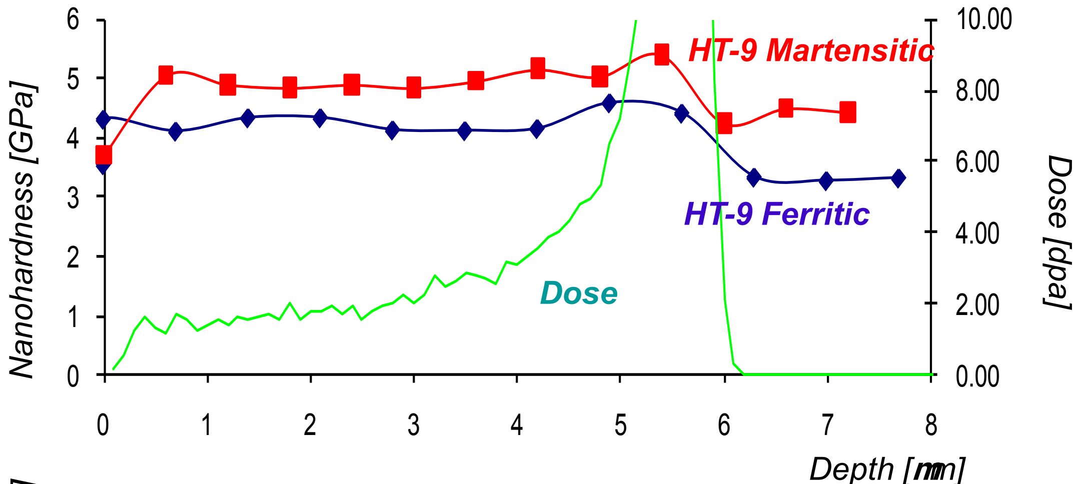
- *Any desired combination of powders: metals, alloys, and dispersoid, such as oxides, carbides, borides, etc.*

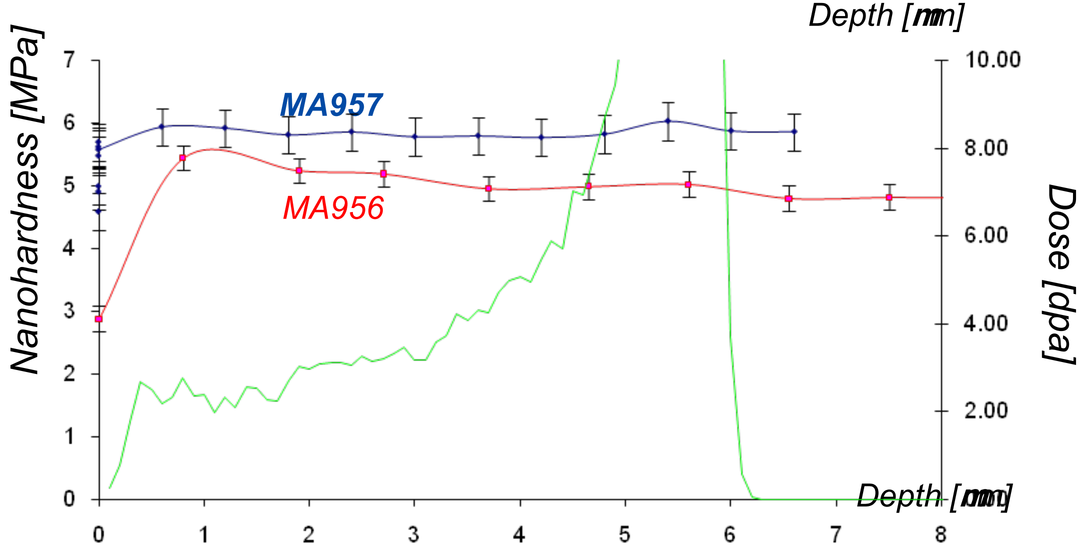


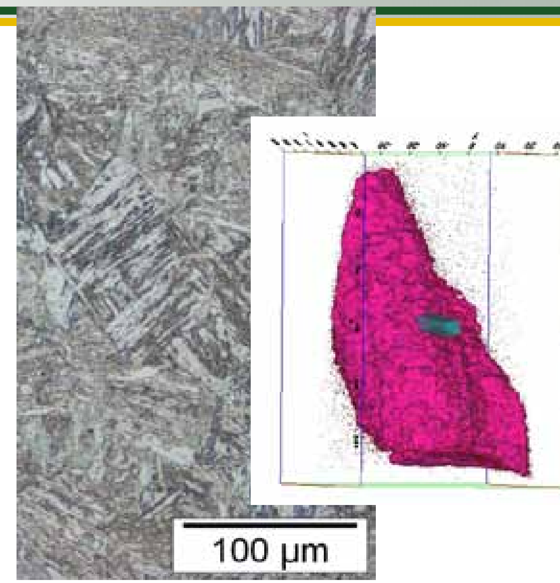
The conventional approach is to ball mill Fe-alloy and Y_2O_3 powders together

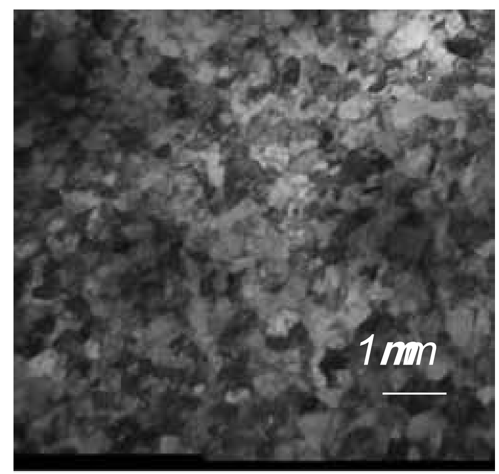
Nanostructured materials such as ODS alloys show no or little radiation induced hardness change.

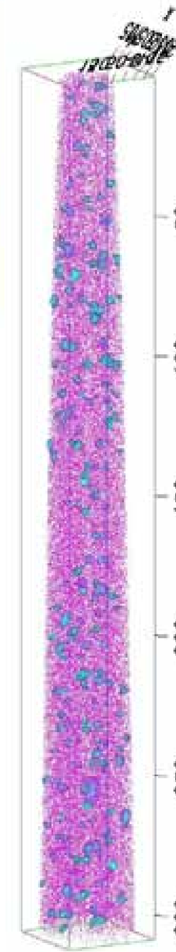
Room temperature irradiation (1.5dpa)









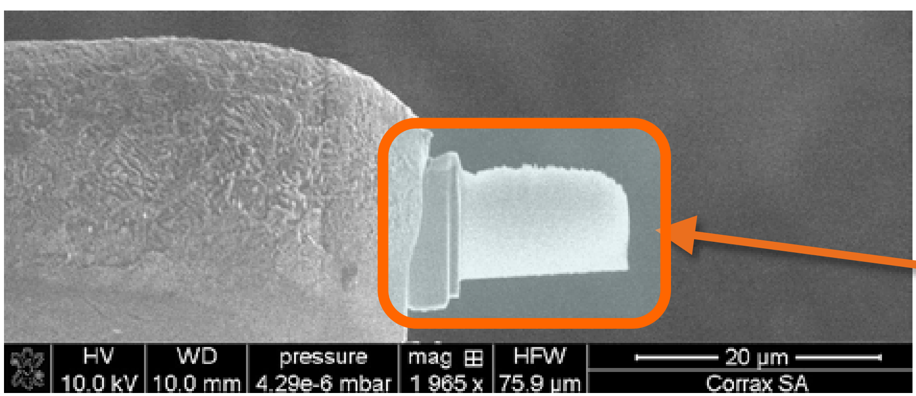


PNNL High Dose MA957 ODS Steel Examination

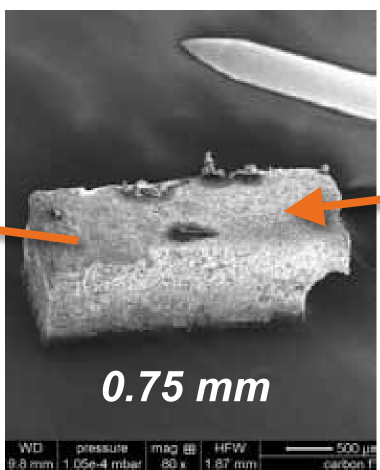
Analysis of MA957

1. Characterization of in-reactor creep response
2. TEM microstructural analysis to study creep mechanisms
3. Tensile testing (and subsequent microstructural examination) after irradiation to 100 dpa in FFTF.
4. APT analysis of oxide particles
5. 300+ dpa ion irradiations to study high dose swelling response

Machined tensile, ring pull, and TEM disk specimens.



Completed 8 μm x 12 μm TEM foil



Material for fabricating APT and TEM specimens

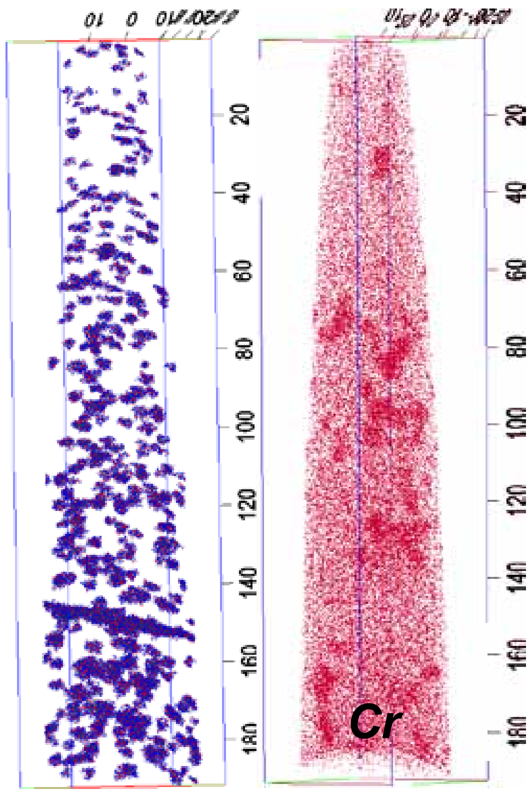
PNNL High Dose MA957 ODS Steel Examination – Early APT Results

$TiYO_x$ oxide particles identified after irradiation from 412°C to 750°C to 100+ dpa.

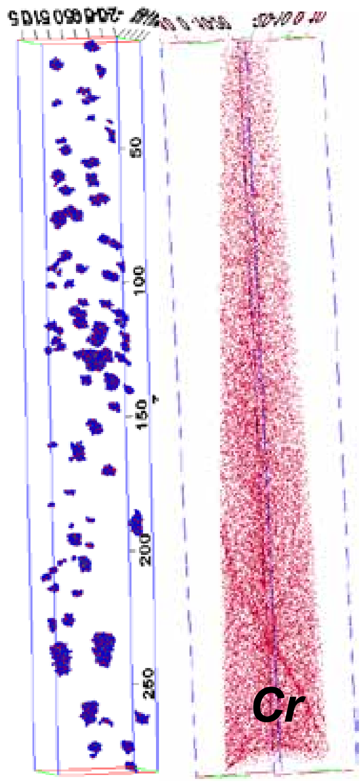
Chromium clustering observed only at 412°C, indicates a'.

Oxide constituents observed on boundaries

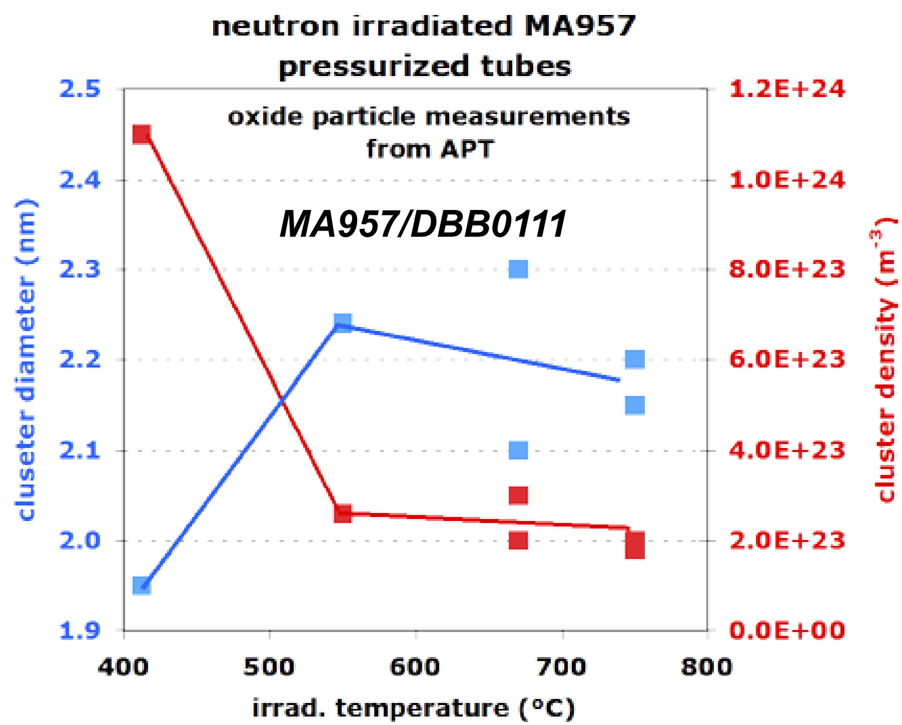
Draft oxide particle counts show 5x higher density at 412° C and suggest an irradiation effect on oxide particles, likely ballistic-dissolution based. (Unirradiated material not yet meas.)



412°C, 109 dpa

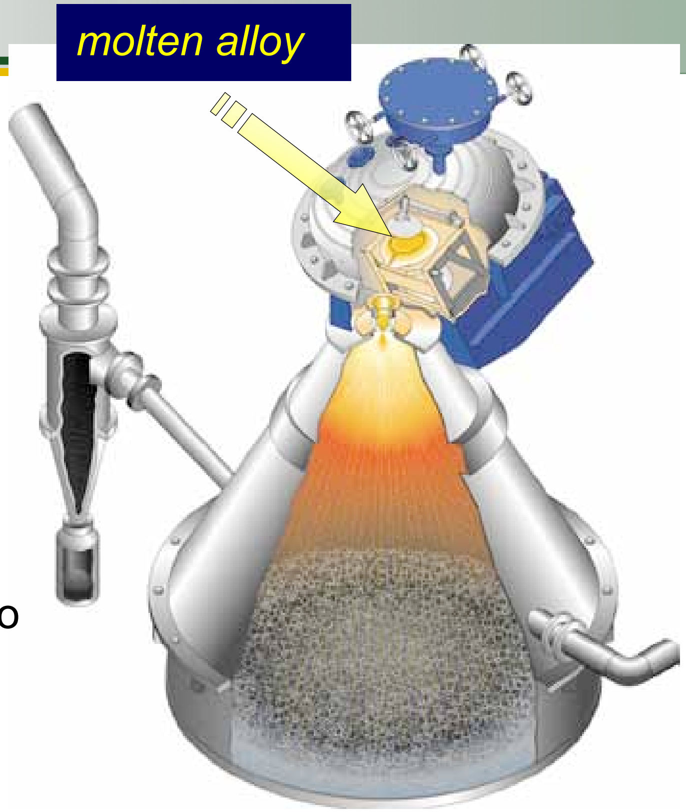


750°C, 121 dpa



draft oxide particle measurements

New atomization approach for producing powder



Y added to molten Fe-alloy and then gas atomized to produce powders

- O can be incorporated to some extent
- Team with ATI Powder Metals to produce a series of experimental powder heats

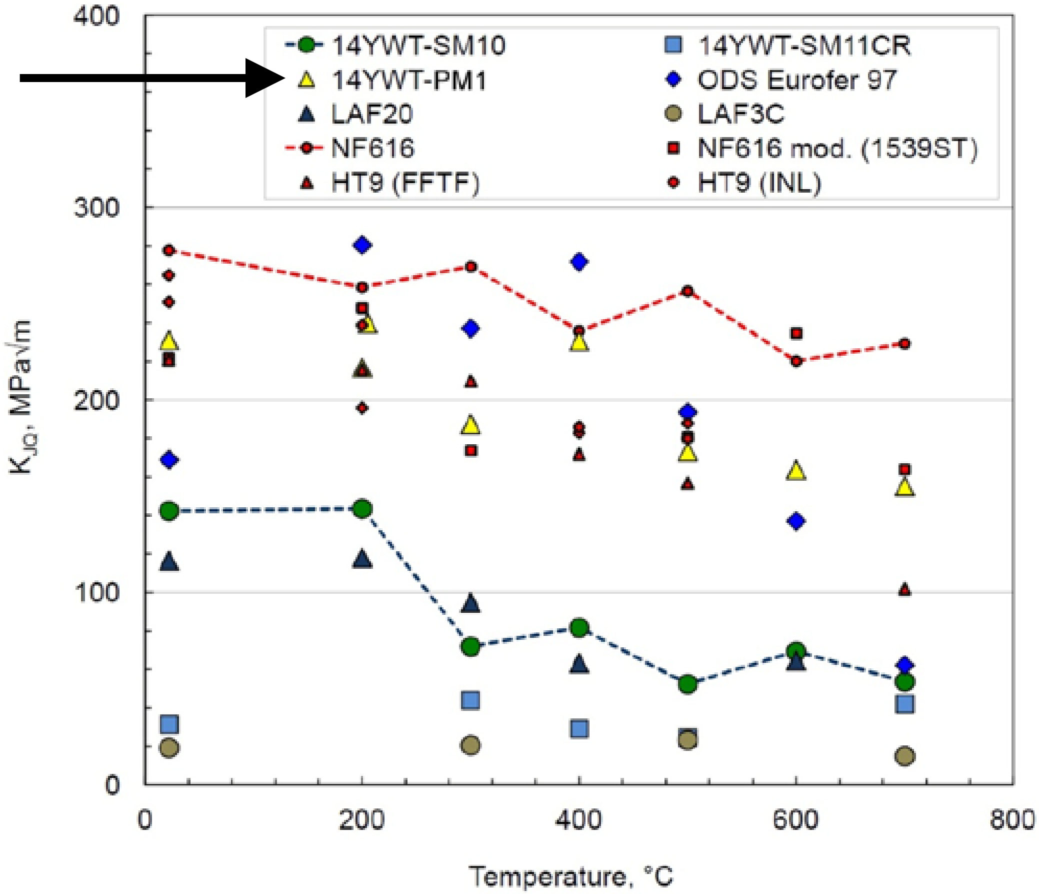
Objective of new MA approach

- more uniform distribution of NC
- reduce contamination by using shorter milling times to uniformly distribute Y and O

Experimental heats

Heat	Chemical Analysis from ATI Powder Metals (wt.%)						
	Cr	W	Ti	Y	O	C	N
L2311	14.0	3.04	0.34	0.20	0.0140	0.006	0.003
L2312	14.0	3.10	0.39	0.23	0.0960	0.003	0.010

14YWT-PM1 exhibited outstanding fracture toughness properties

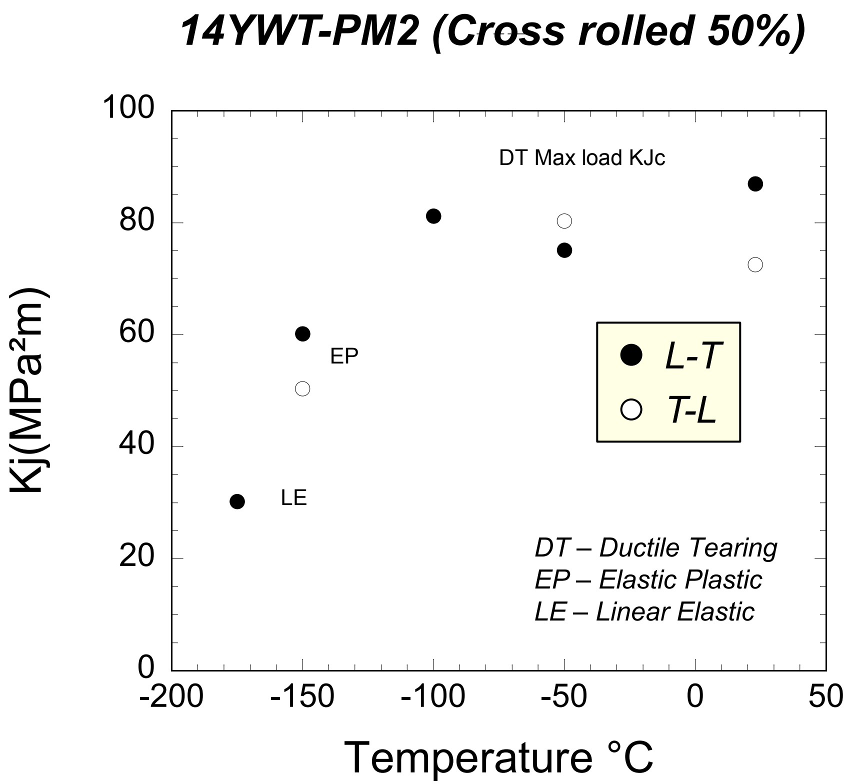


Fracture toughness of PM1 is >200 MPa√m up to ~400ºC and remains >150 MPa√m up to 750ºC

Fracture toughness of PM1 is comparable to non-ODS tempered martensitic steels

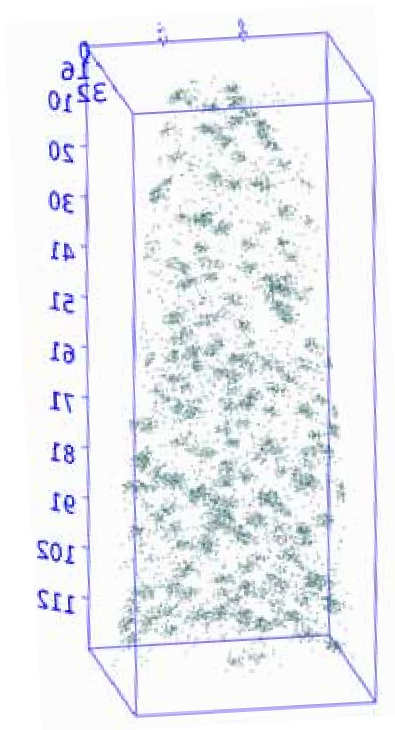
Results indicate lowering the interstitial O, C and N levels is important for optimizing the mechanical properties of 14YWT

Excellent fracture toughness for 14YWT-PM2



Number density = $10^{24}/m^3$

Average size = 2.1 nm



The fracture toughness results for 14YWT-PM2 showed:

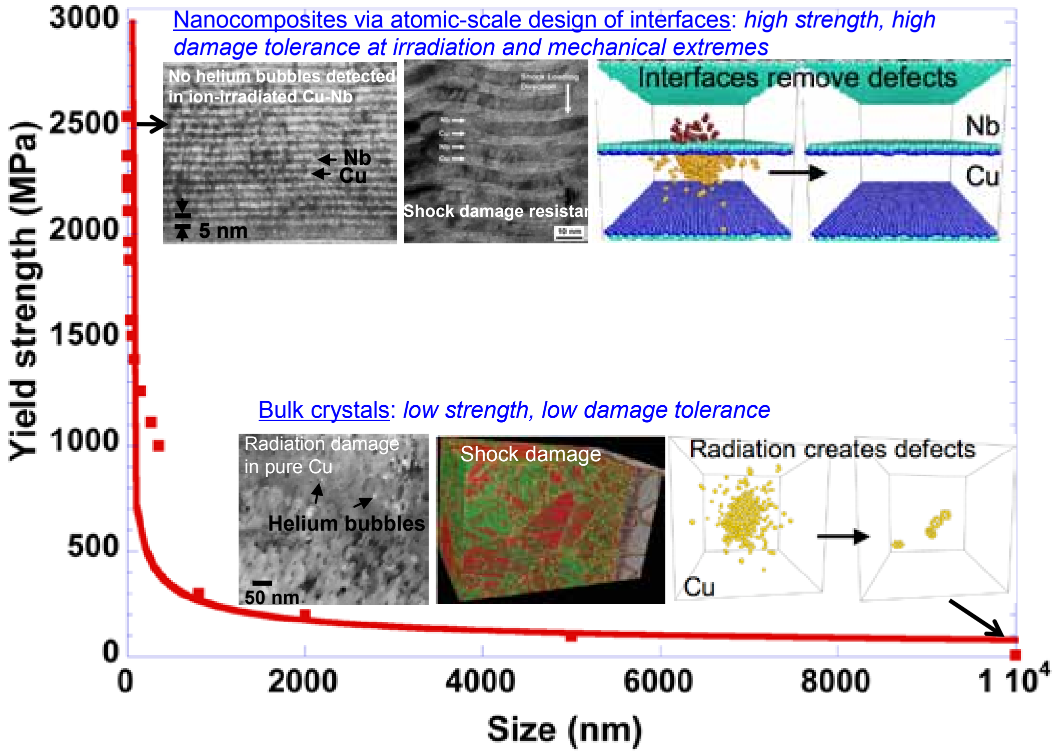
- Higher fracture toughness than OW4 at room temperature (25ºC)
- Very low fracture toughness transition temperature (FTTT)
- No effect of orientation on FTTT, i.e. no anisotropy

Center for Materials at Irradiation and Mechanical Extremes

M. Nastasi, A. Misra (LANL)

Summary statement:

The purpose of this EFRC is to understand, at the atomic scale, the behavior of materials subjected to extreme radiation doses and mechanical stress in order to synthesize new materials that can tolerate such conditions.

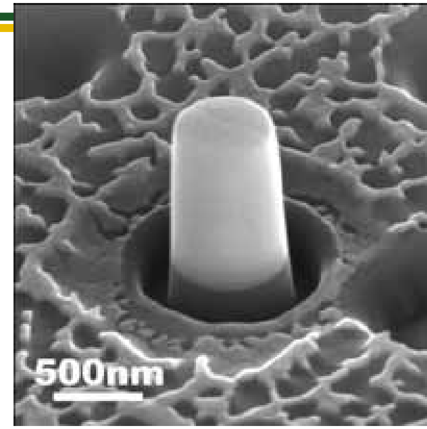


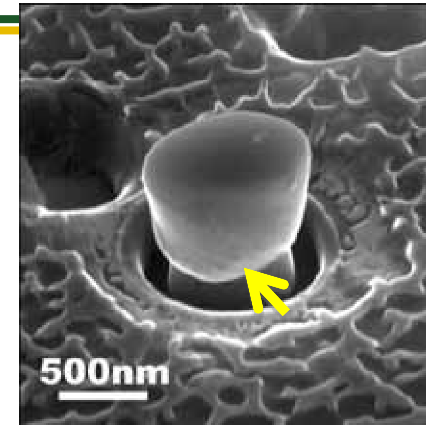
The EFRC is developing a fundamental understanding of how atomic structure and energetics of interfaces contribute to defect and damage evolution in materials, and use this information to design nanostructured materials with tailored response at irradiation and mechanical extremes with potential applications in next generation of nuclear power reactors, transportation, energy and defense.

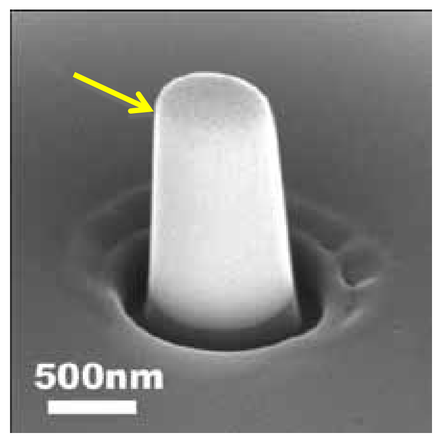
http://cmime.lanl.gov

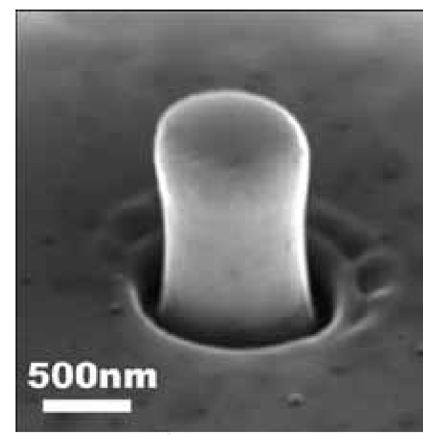
an Office of Basic Energy Sciences Energy Frontier Research Center

CMIME Center for Materials at Irradiation and Mechanical Extremes





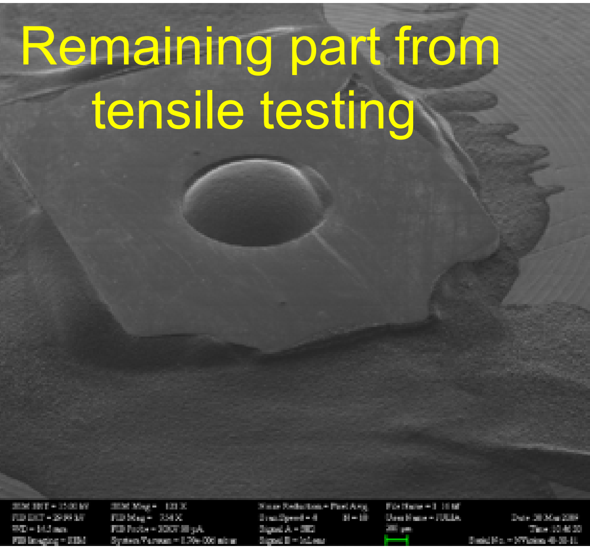


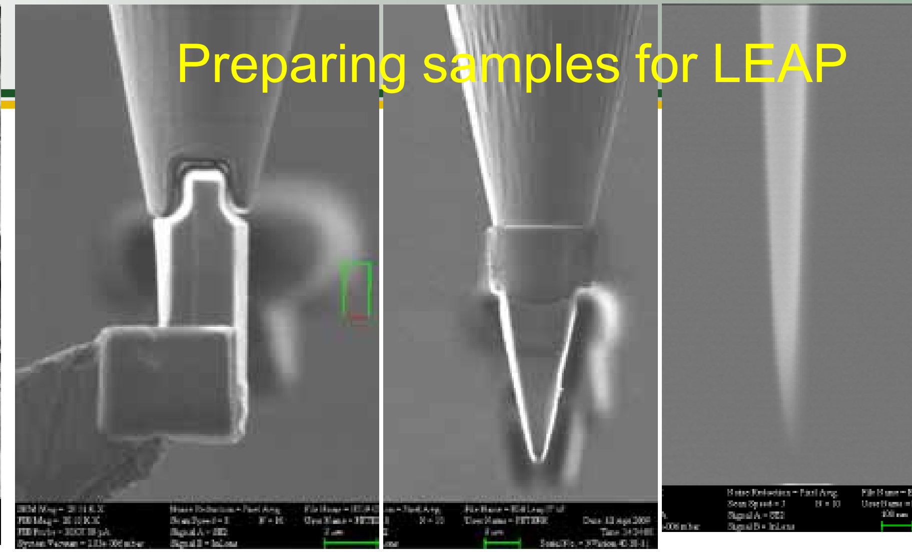


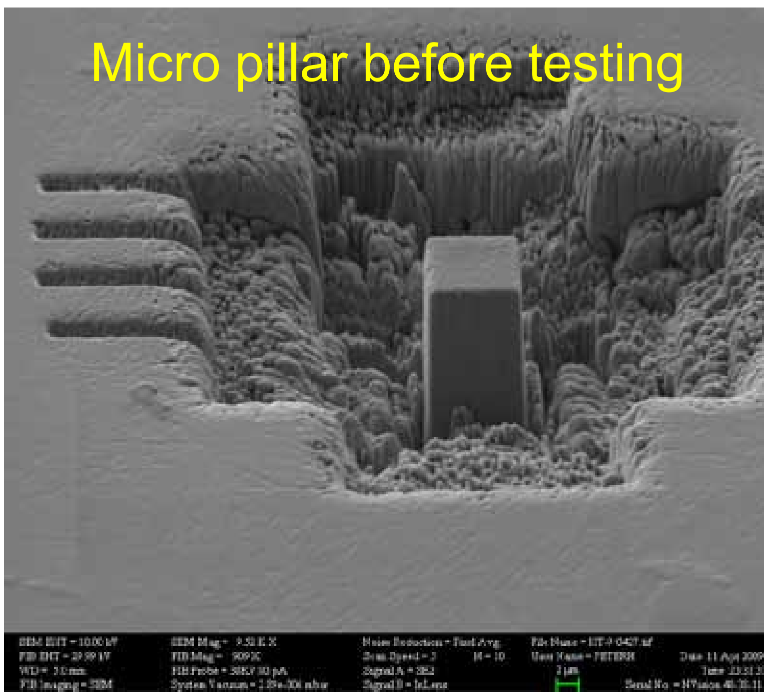
Scripta Materialia, ***64*** *(2011) 974–977.*

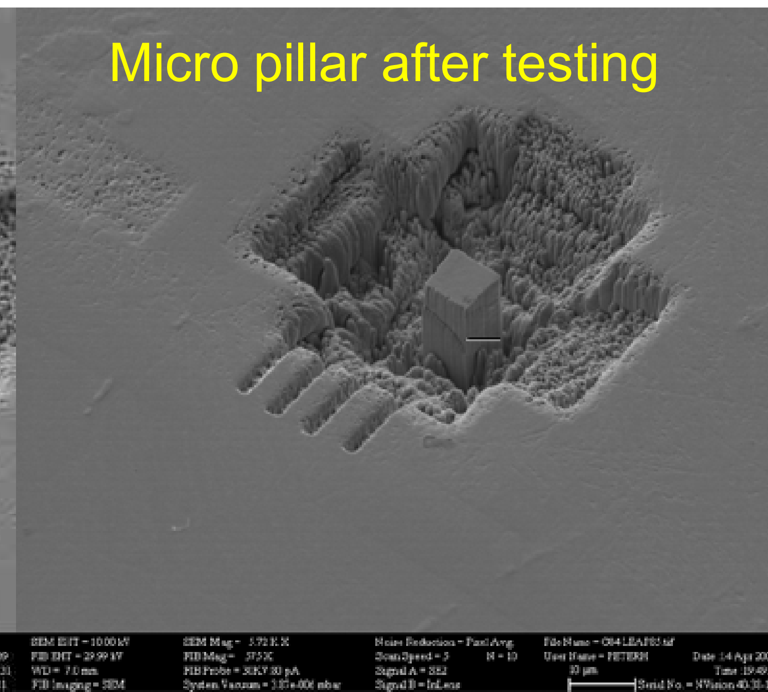
Center for Materials at Irradiation and Mechanical Extremes

Micro-Sample preparation

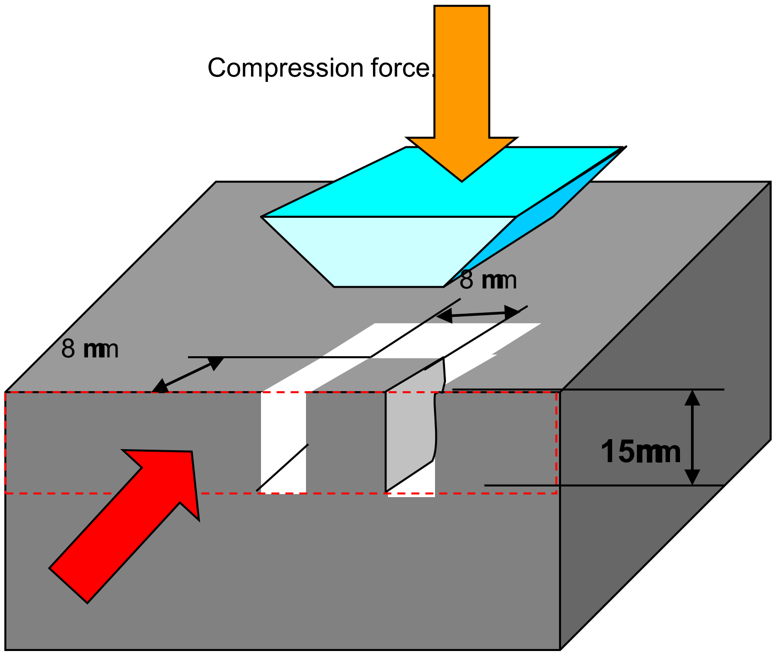




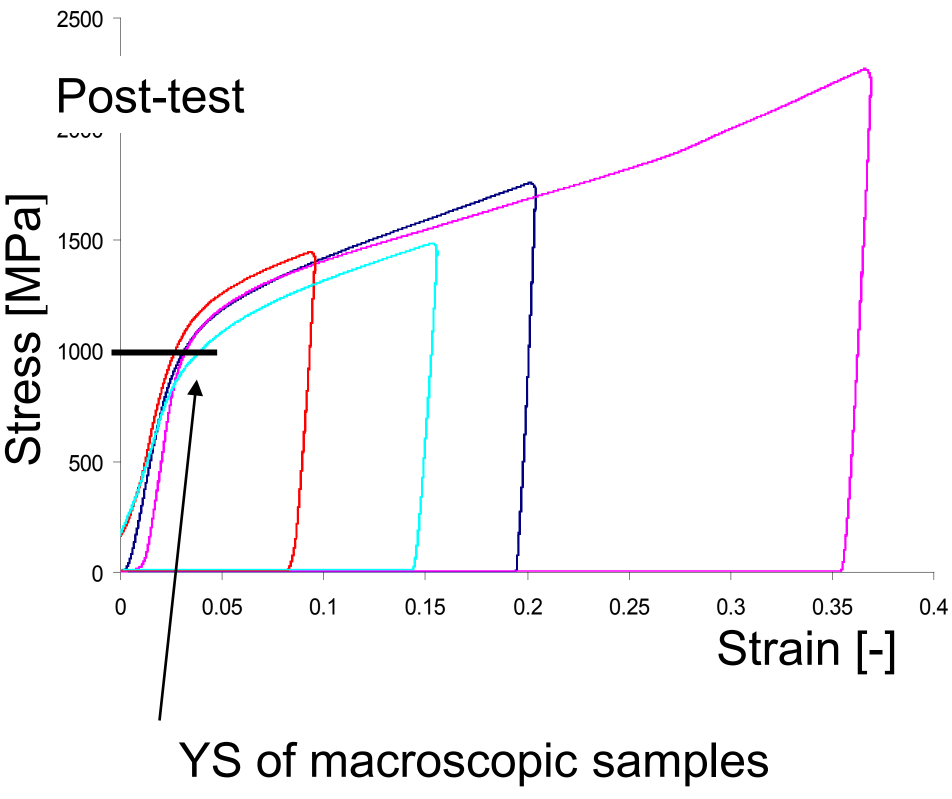




Example of Micro pillar results on MA957



ODS MA957 in the extrusion direction



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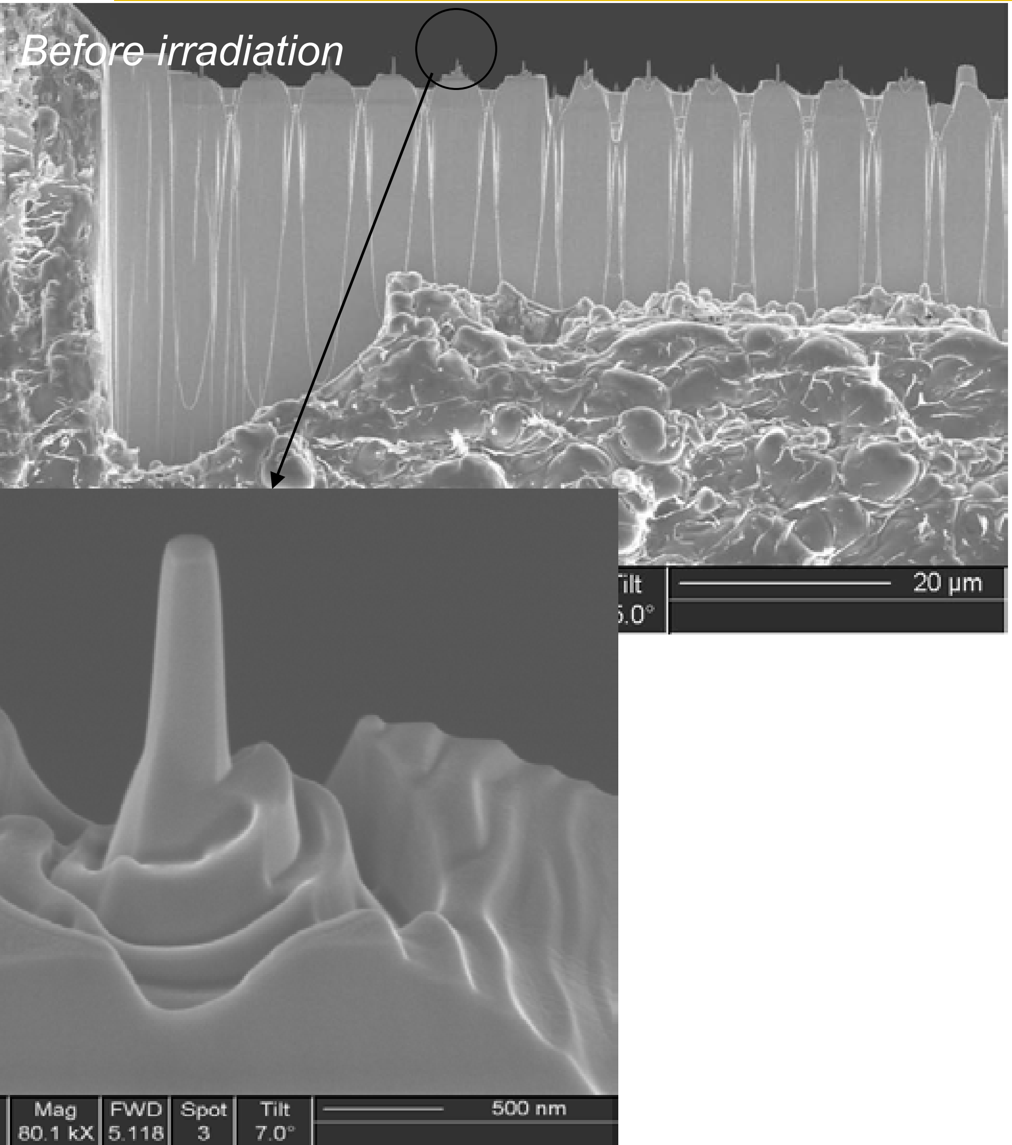
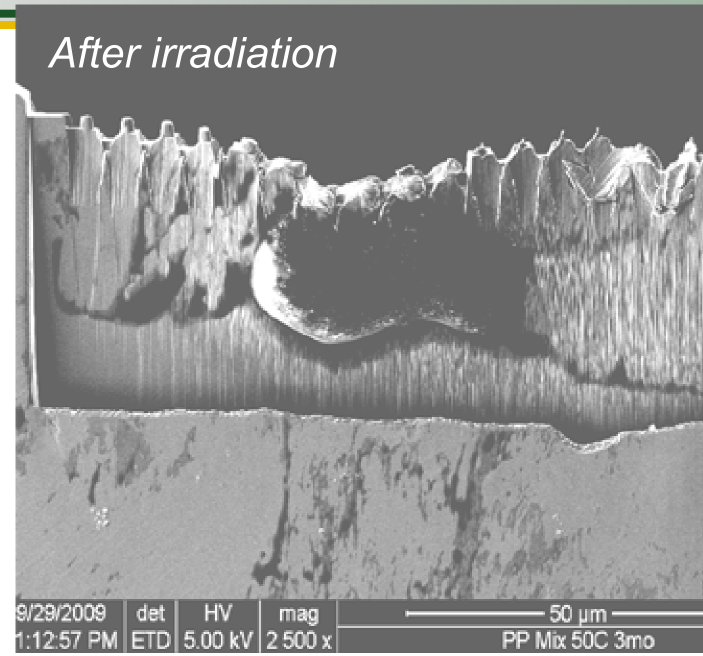
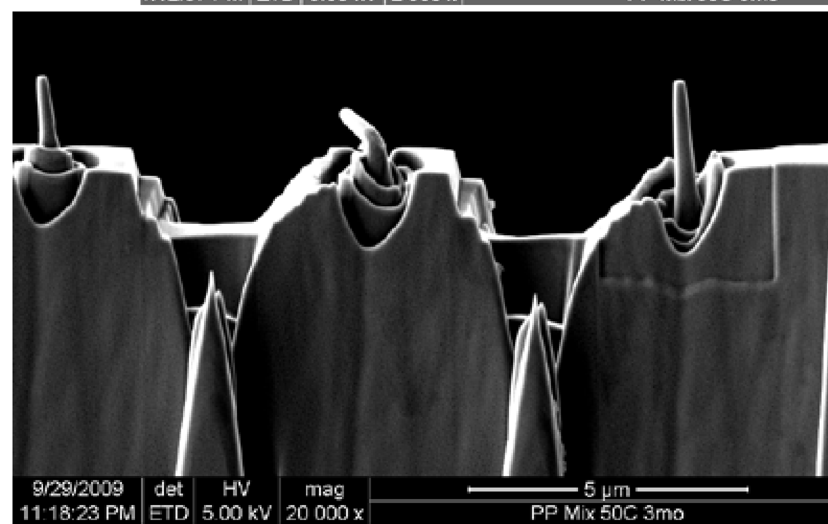
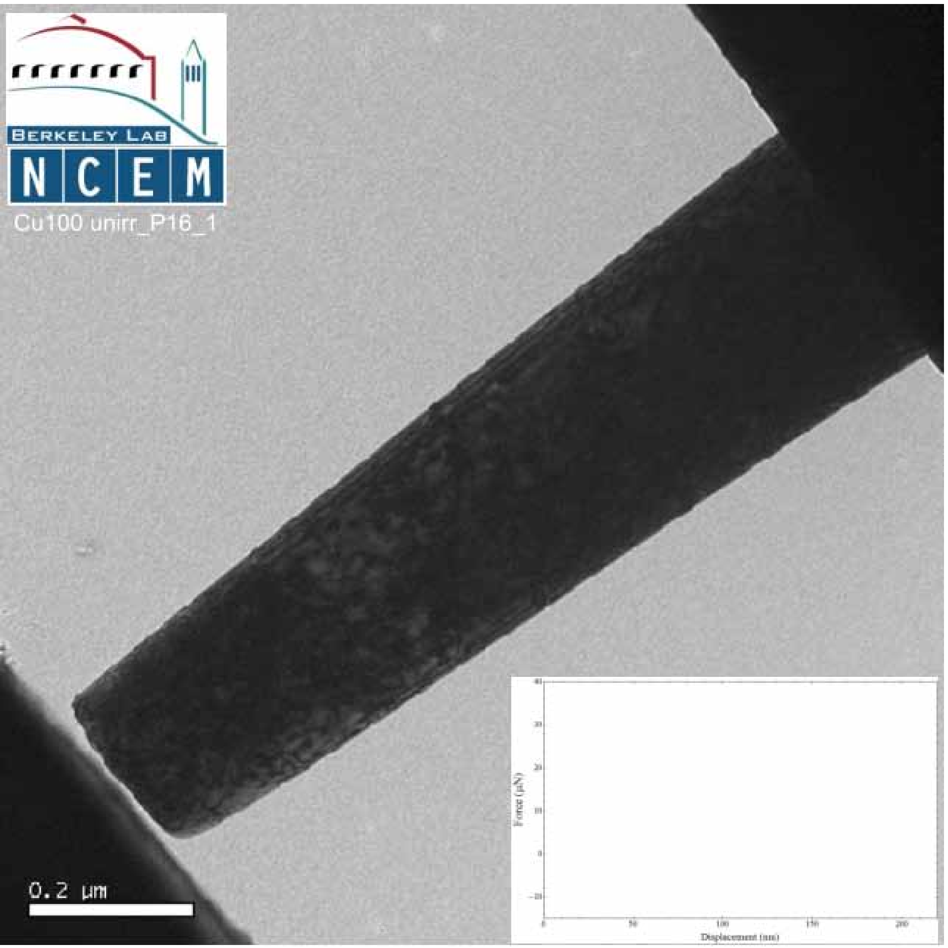
Using ion beam irradiation to learn more on defect-dislocation interaction. First exp.: single crystal Cu

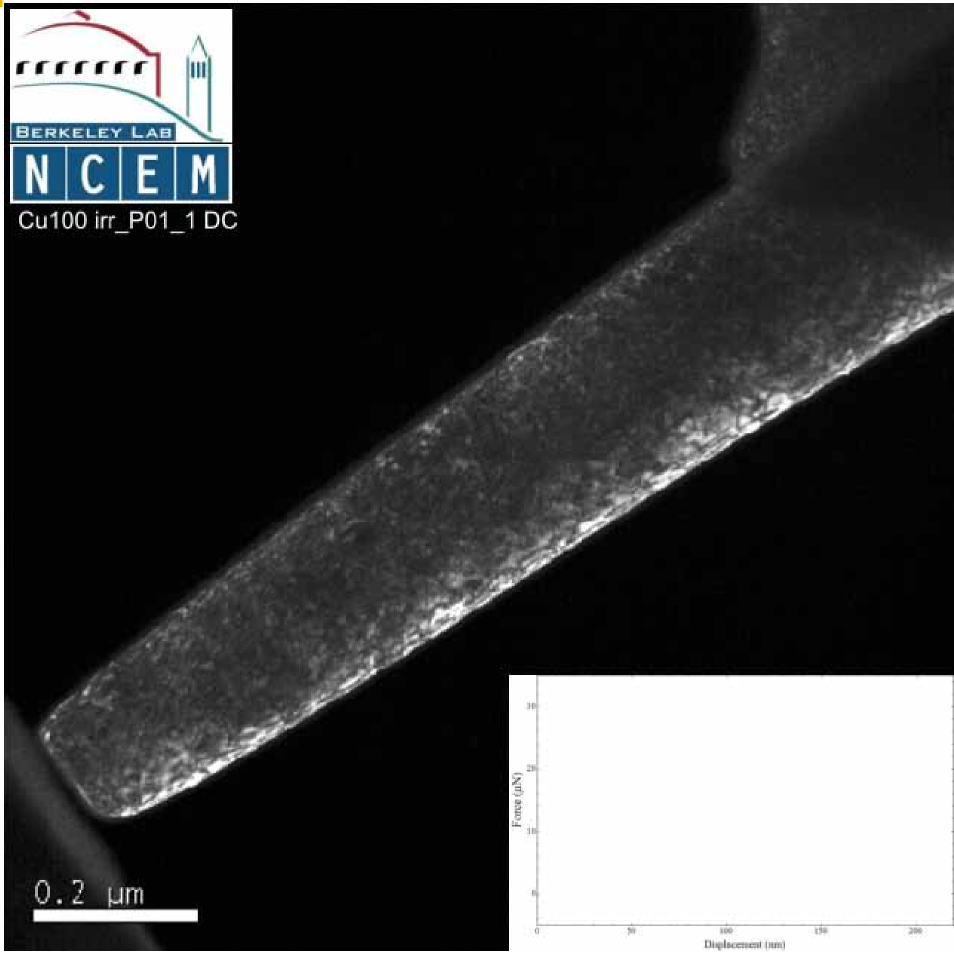






Nano pillar testing irradiated vs. unirradiated





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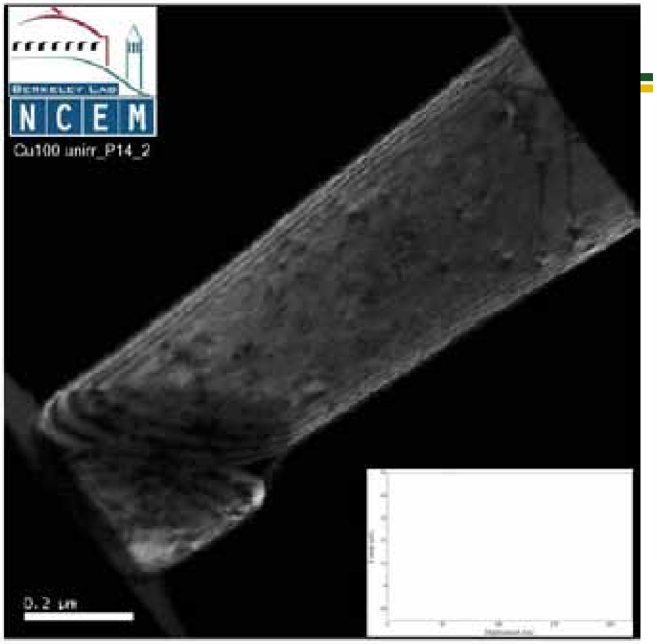
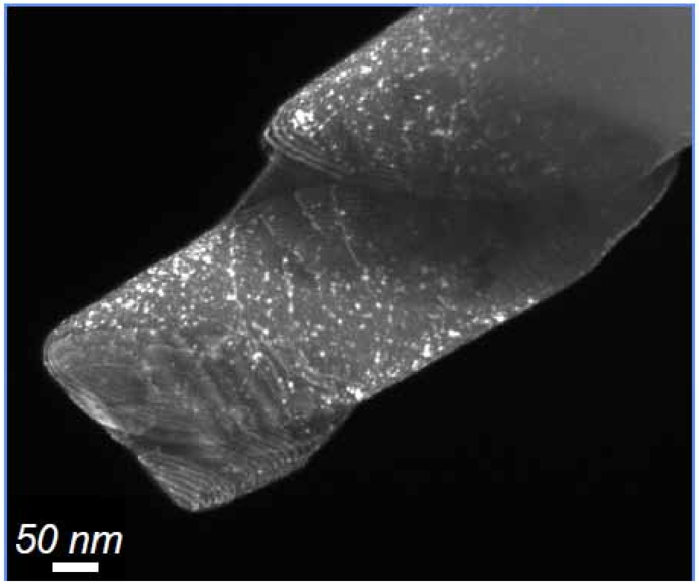
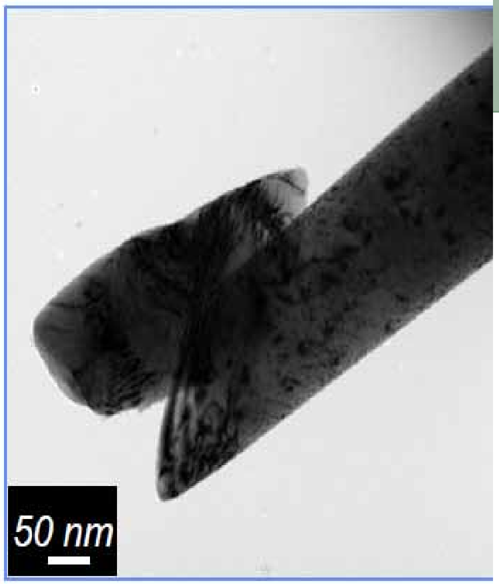
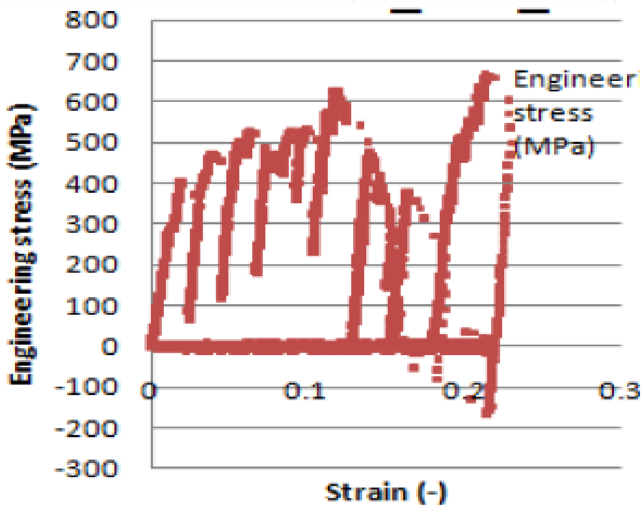
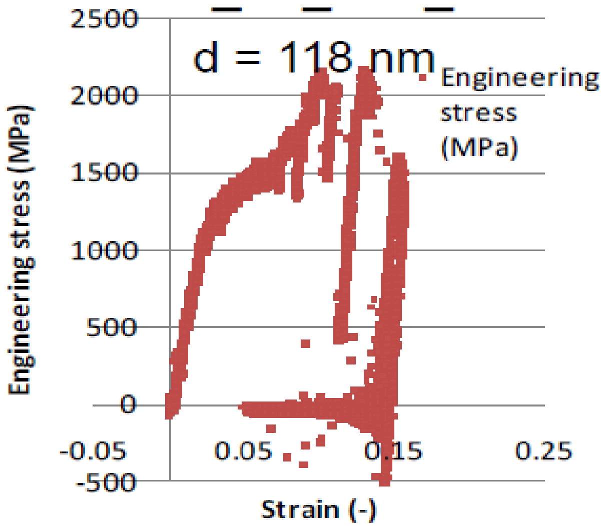








Not irradiated Cu



Irradiated Cu

Summary

Nanoscale Applications in Core Materials for Nuclear applications

- Radiation tolerant materials with nanofeatures
  - *ODS strengthened Steels*
  - *Maraging steels*
  - *Multilayers*
- Nanoscale material preparation/testing
  - *FIB- TEM foils*
  - *Micropillar testing*
  - *Nanopillars for in situ mechanical testing*

New research underway to investigate the application of ODS alloys to LWR improved accident tolerant cladding development.