Microstructural Evolution in Cladding

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EFRC Summer School on the Evolution and Impact of Microstructural Defects on In-Reactor Material Response

Idaho Falls, Idaho

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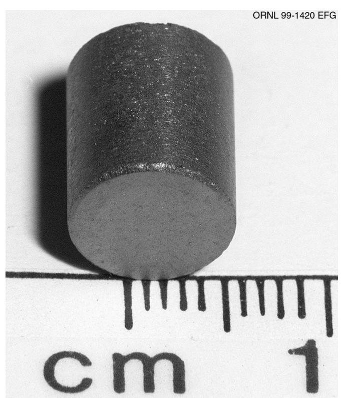
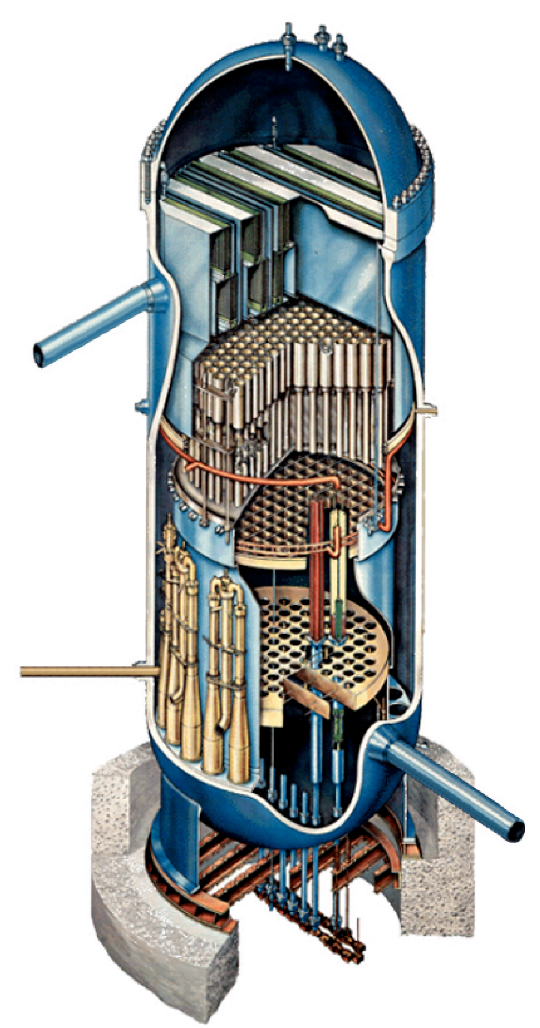
Outline

- **Overview of fuel cladding functions and candidate materials**
  - Light Water Reactors, fast reactors)
  - Pressing for higher performance and improved fuel reliability
- **Key challenges for fuel cladding**
- **Microstructure of irradiated Zr alloys**
- **Microstructure of irradiated stainless steels**
- **Microstructure of irradiated ferritic steels**
- **Brief comments on SiC composites**

Materials performance is key for economic and safe fission reactor operation in current LWRs

- **Heat generation in UO_2-based fuel pellets**
- **Heat transfer across Zr alloy cladding**
- **Numerous core internal structures to securely position core**
- **Reactor pressure vessel for containment of fission products**
- **Piping and steam generator equipment for heat conversion to electricity**



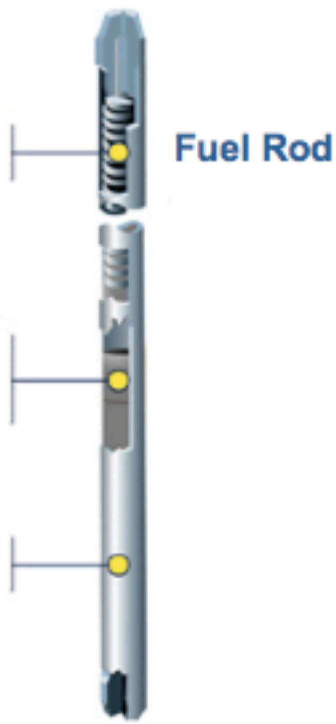


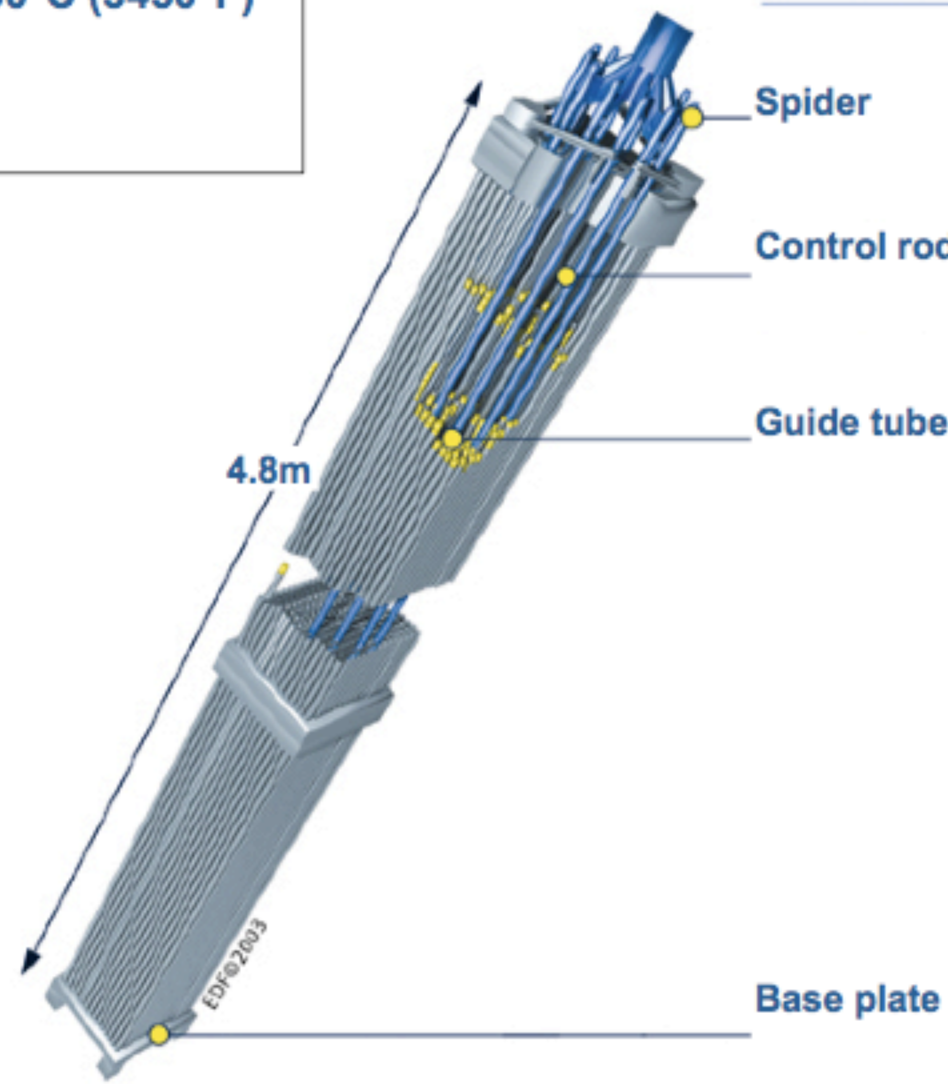
1300MW FUEL ASSEMBLY

Incore Fuel pellets temperature = 1900°C (3450°F)

Linear power density = 170W/cm

Medium burn up = 38000 MWd/t

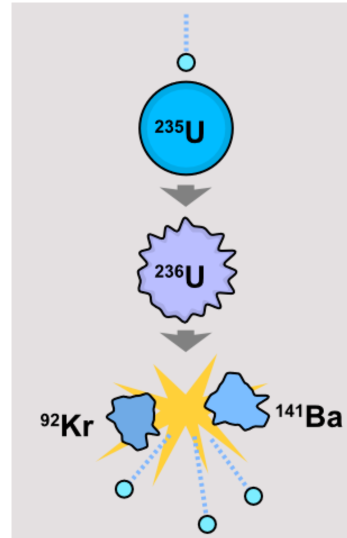


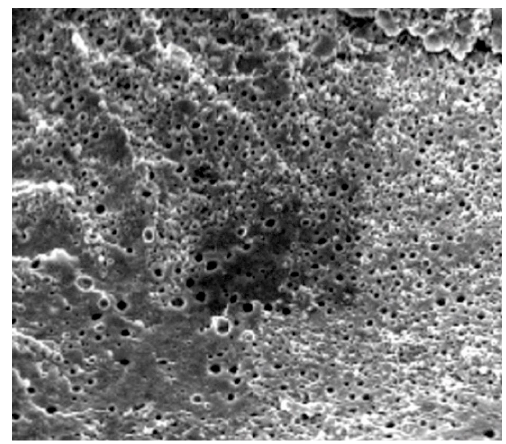


Source: R. Clement

Overview of Swelling/Fission Gas Release Issues

- **Fission Products**
  - Two atoms replace every U (or Pu) atom that fissions
  - 25% of fission products are gas atoms (Kr, Xe)
- **Fuel Swelling**
  - Fuel swells due to generation of fission products
  - Gas atoms coalesce into bubbles, accelerating swelling
  - Fuel swelling tends to reduce or close gap
- **Fission Gas Release**
  - Some fission gas escapes fuel
  - Pressurizes plenum and decreases thermal conductivity
  - Percent of gas escaping fuel
    - < 10% in LWR fuel
    - > 50% in fast reactor fuel





Bubbles in metallic fuel

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Light Water Reactor Average Fuel Cycle Lengths have increased by ~30 to 40% since 1990

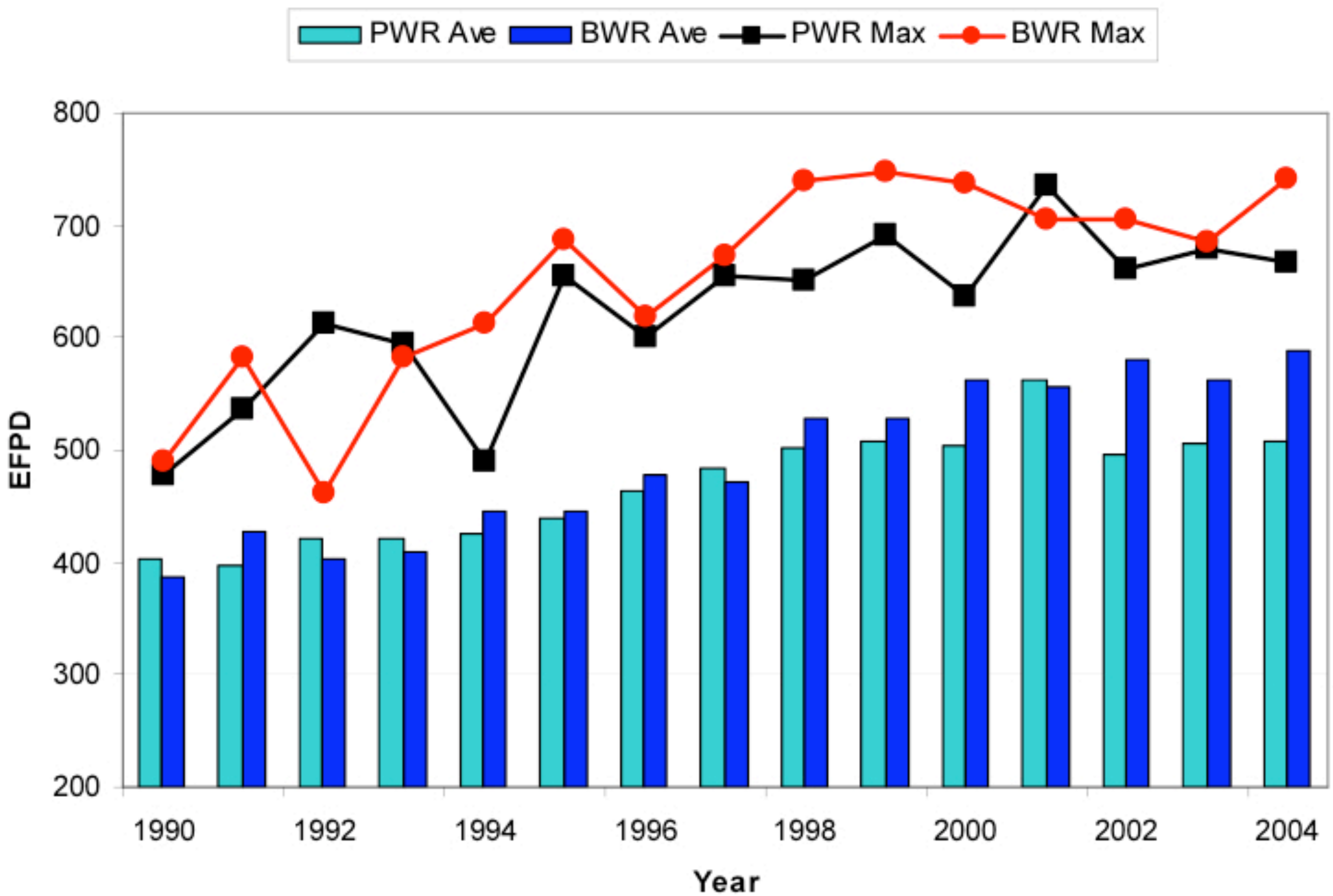


Figure 1. Increasing average cycle lengths for fuel elements in PWRs and BWRs in the USA.

Nuclear technology review 2007 supplement, 51st IAEA General Conference, (Vienna, 2007)

Light Water Reactor Average Fuel Discharge Burnup has doubled since 1970

Corresponding typical dose to LWR fuel cladding has increased from ~10 dpa to >20 dpa

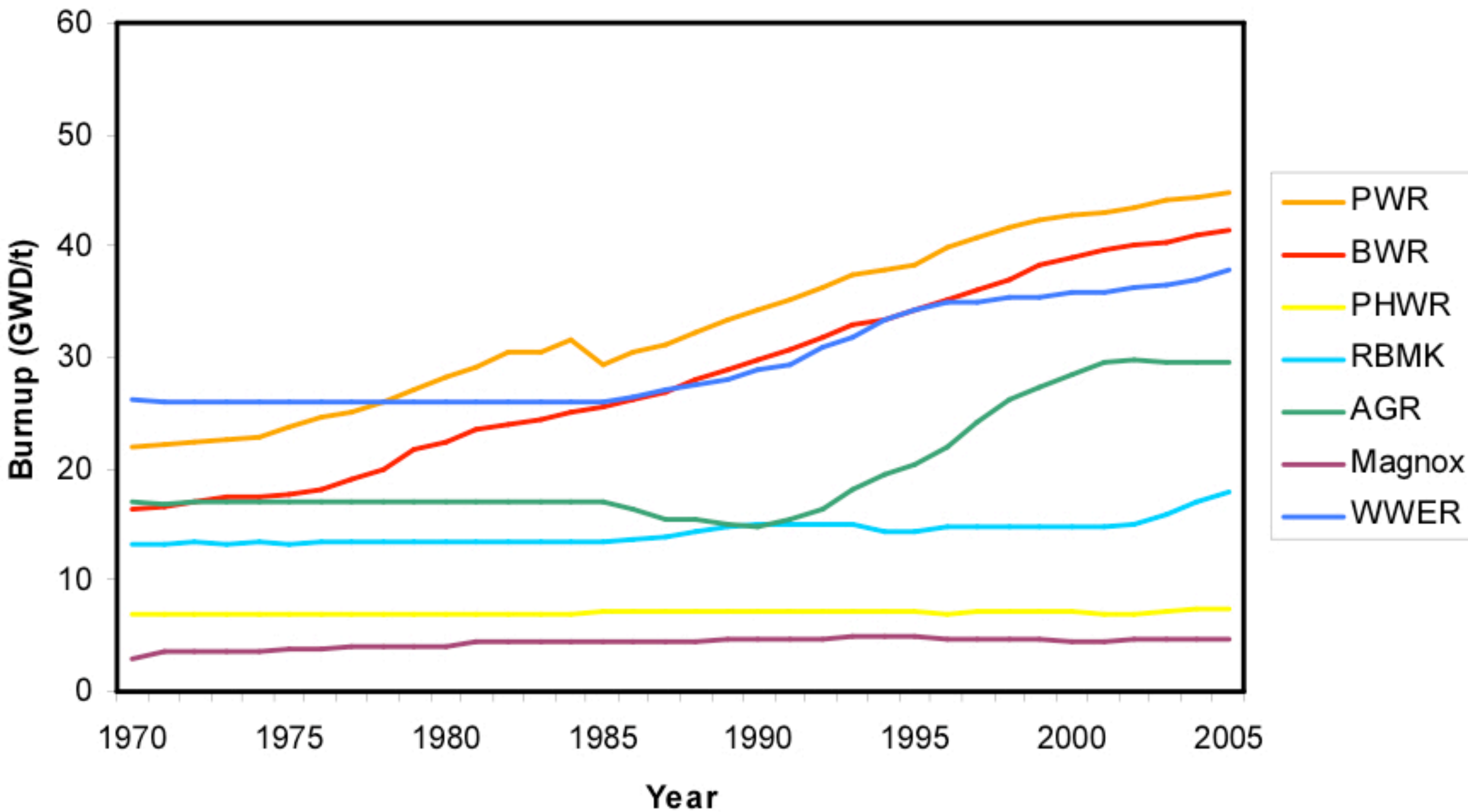
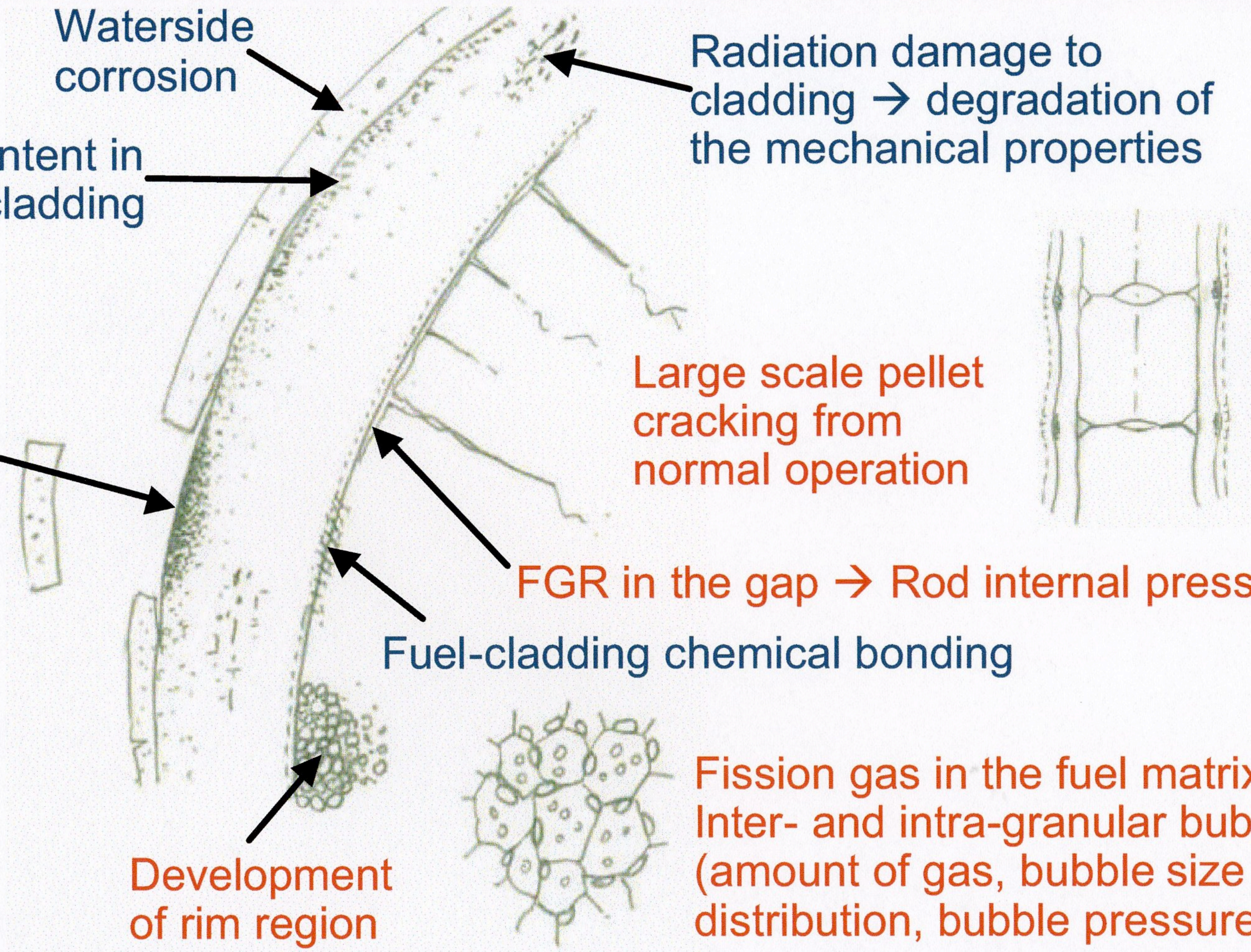


Figure 2. Trends in fuel burnup for different types of reactors

Nuclear technology review 2007 supplement, 51st IAEA General Conference, (Vienna, 2007)

What is changing with Burnup ?



Fuel Operational Challenges: Increased fuel duty cycle

- **Total residence time (burnup) has increased (from 3 (30GWd/t) to ~ 5 years) => *higher radiation damage* and longer exposure => *more corrosion***
- **The NRC Burnup limit is 62 GWd/t, some utilities have expressed interest in increasing this to 75 or 100 GWd/t => decrease waste and increase availability factor.**
- **Primary water chemistry is different (presence of Li, Zn injection, CRUD formation, hydrogen water chemistry)**
- **Fuel is operating at higher temperatures (power uprates)**
- **Fuel cycles have been increased (18 or 24 month cycles)**

=> Utilities and fuel vendors would like to have the flexibility of increasing burnup further, while operating economically and safely

Fuel failure rates in Light Water Reactors have decreased dramatically since 1980

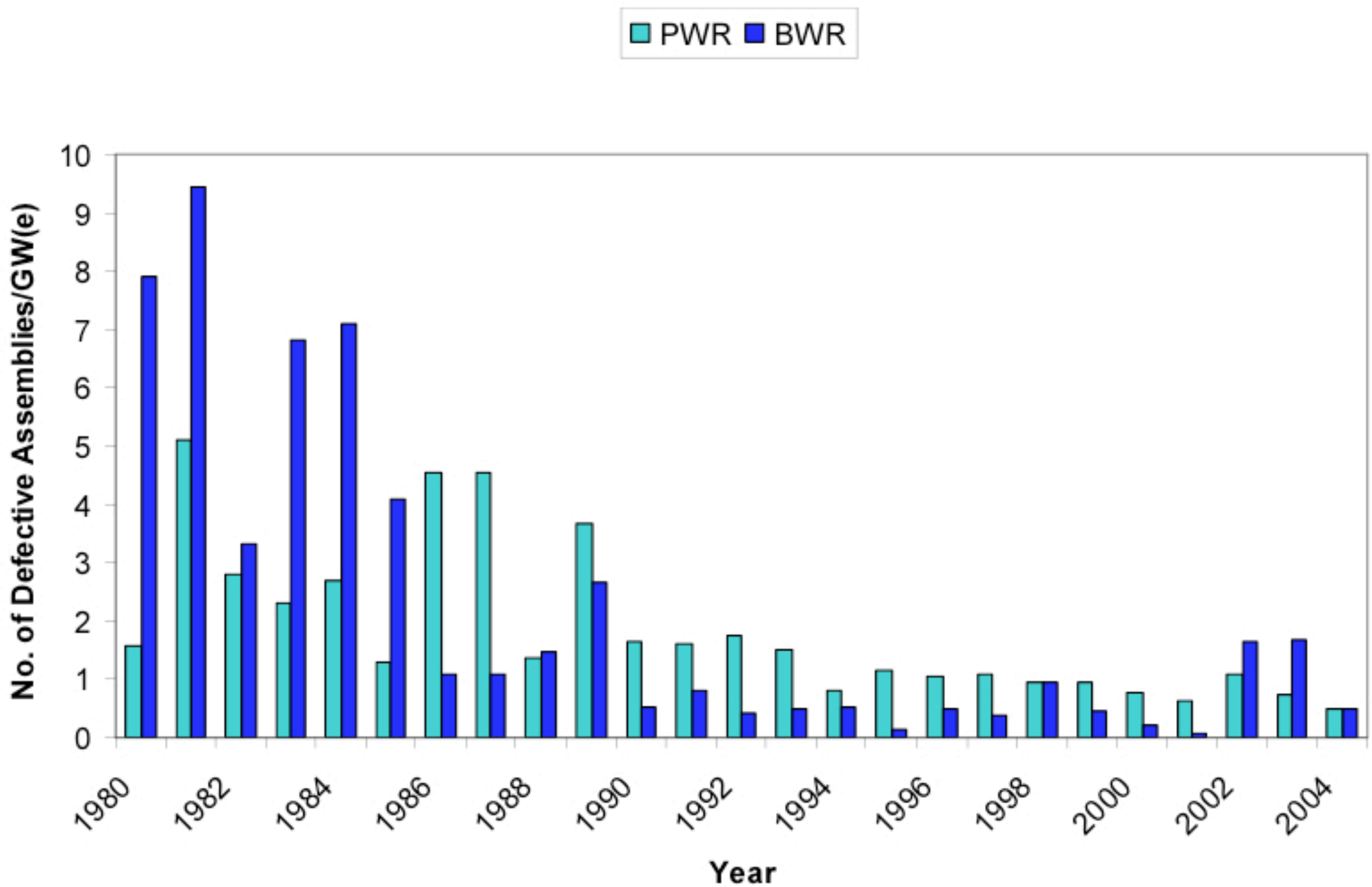
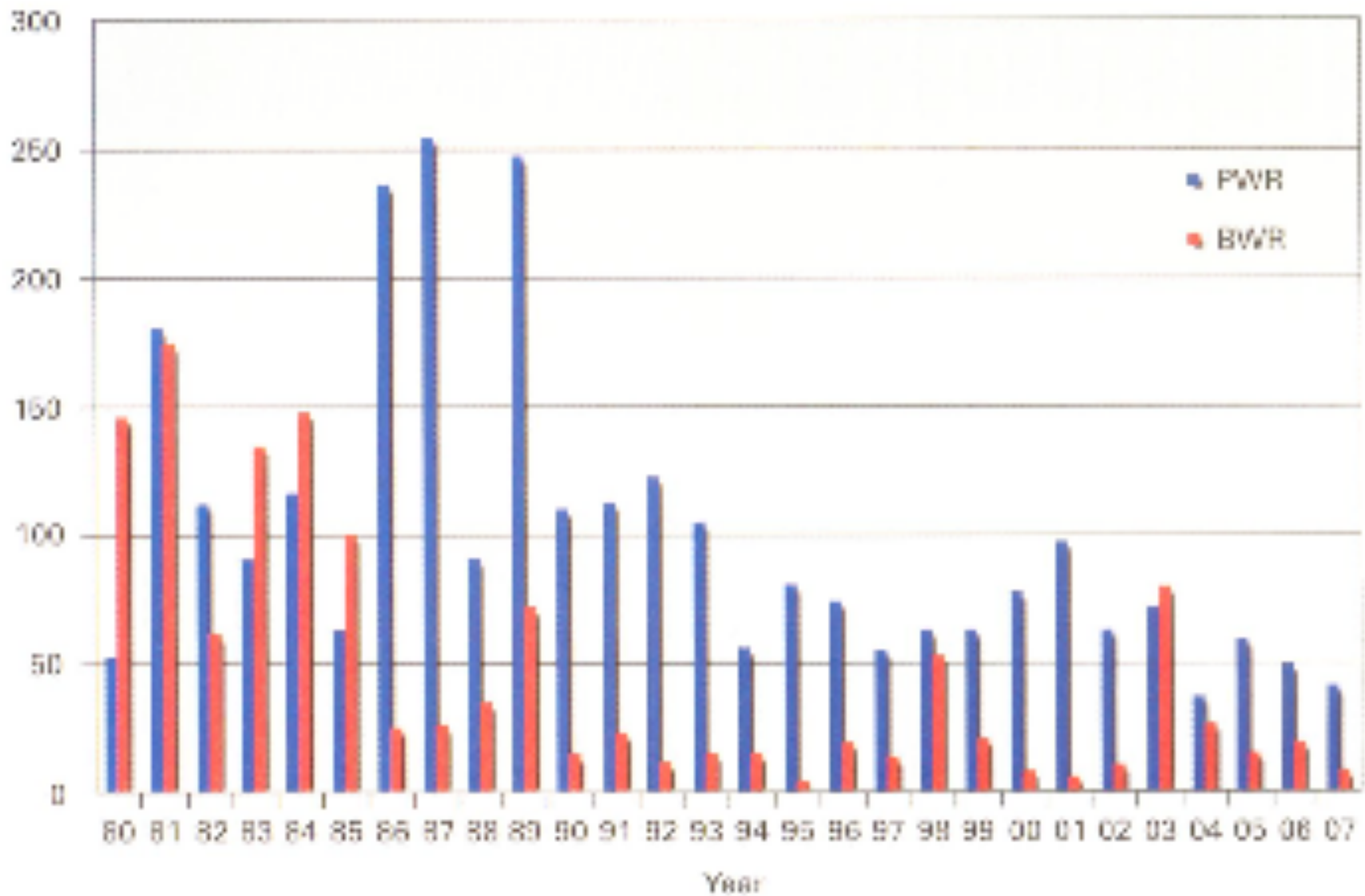


Figure 5. Fuel failure rates in US plants.

Nuclear technology review 2007 supplement, 51st IAEA General Conference, (Vienna, 2007)

Fuel failures continue to be an issue, although greatly reduced compared to 1980s

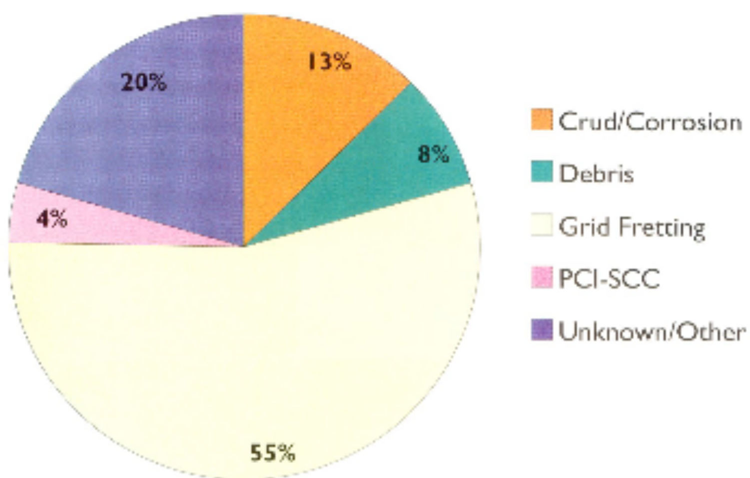


Fuel failures in U.S. PWRs and BWRs, 1980–2007 (Graphics: EPRI)

Nuclear News, 2008

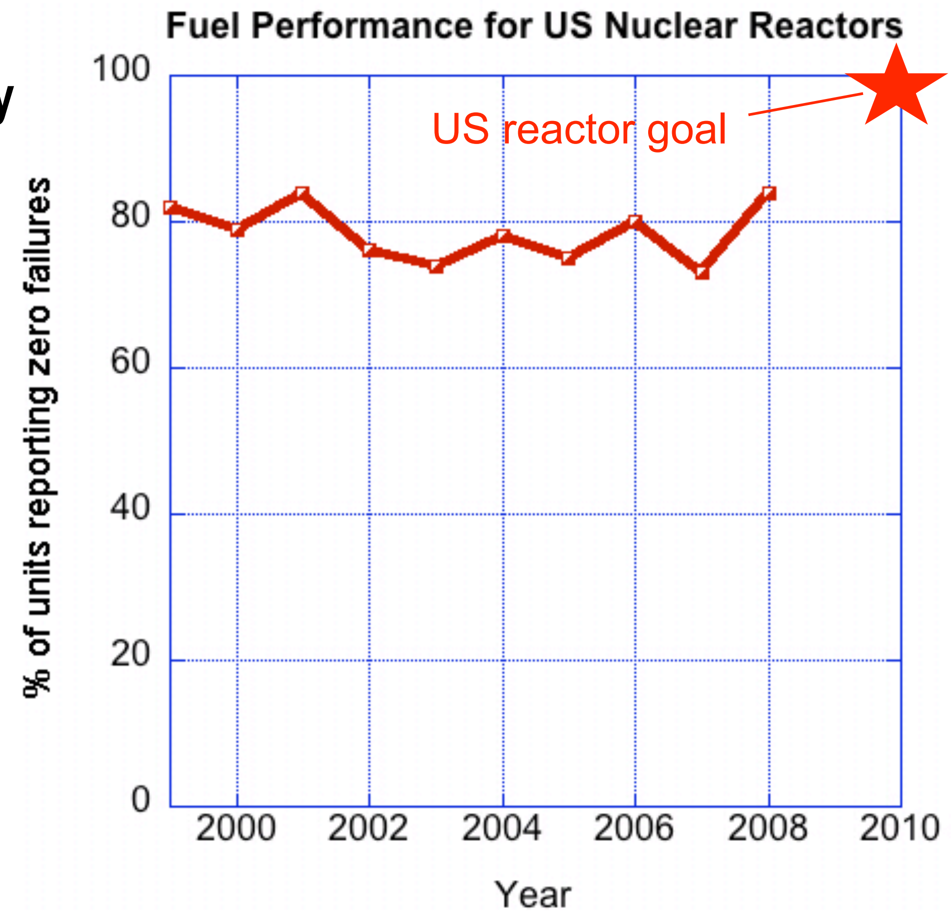
The goal of US reactor utilities was zero failures by 2010

- **Even while higher fuel burnups are being achieved, the industry wanted to completely eliminate fuel failures by 2010 (cladding failures cause increased exposure to workers in outages, may force unplanned shutdowns).**
- **This involves thousands of fuel assemblies and millions of fuel rods and none can fail.**



Percentage of fuel failures by mechanism for PWRs and BWRs, 2000–2007

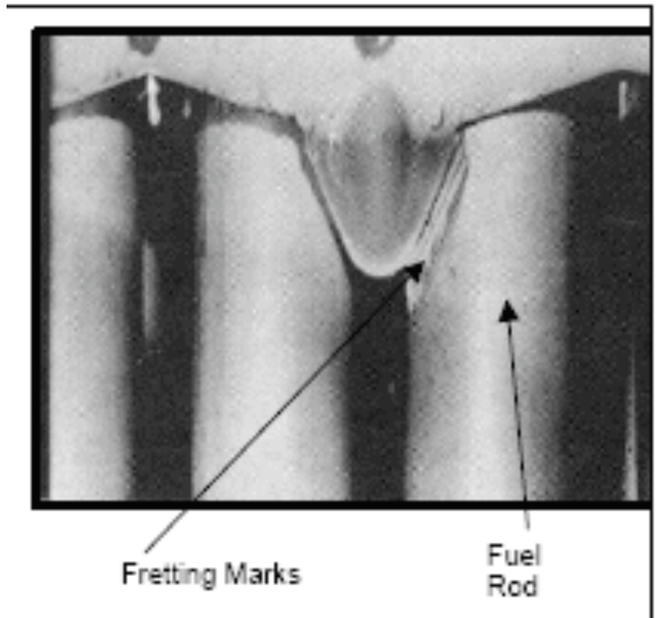
- Fuel failure encompasses diverse failure mechanisms

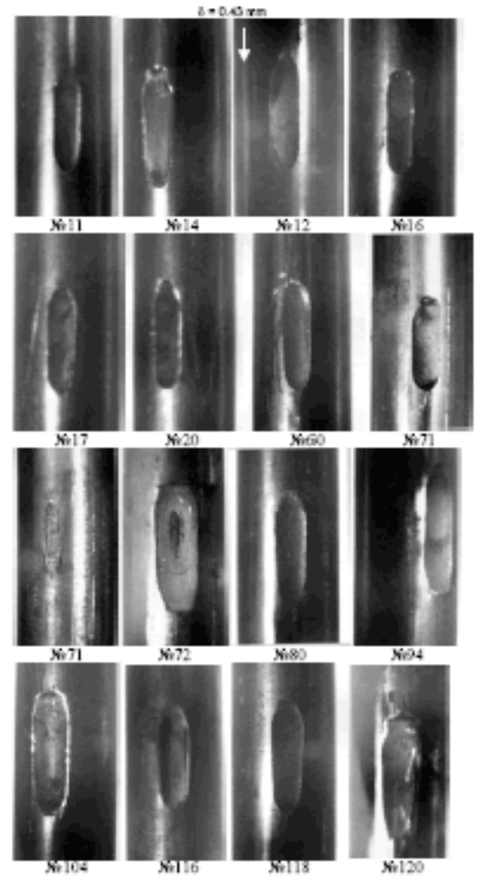


after Nuclear News (June 2009) p. 24 and B. Tompkins, Nuclear News (May 2008) p. 34

Grid-rod fretting is a major Zr alloy cladding issue

Typical marks of contact interaction of VVER fuel rods with spacer.





Visual appearance of cladding fretting at the places of contact with spacer grids.

Totju Totev

Slide compiled by Todd Allen

Pellet-clad Interaction (PCI)

Pellet chip

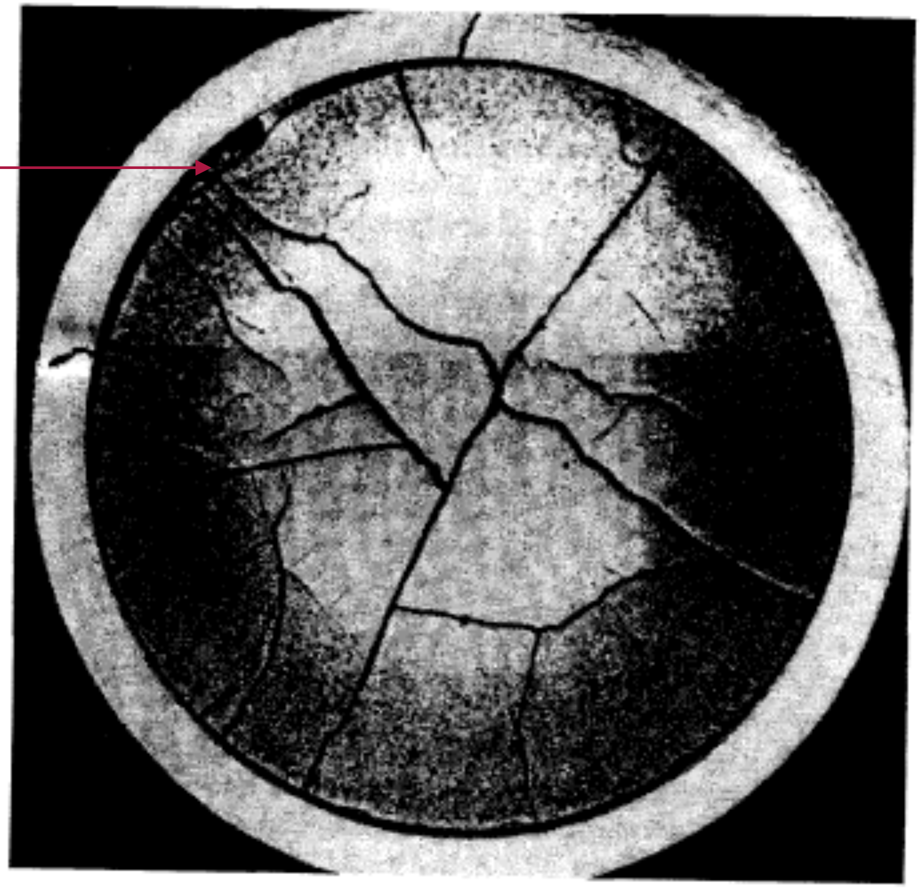


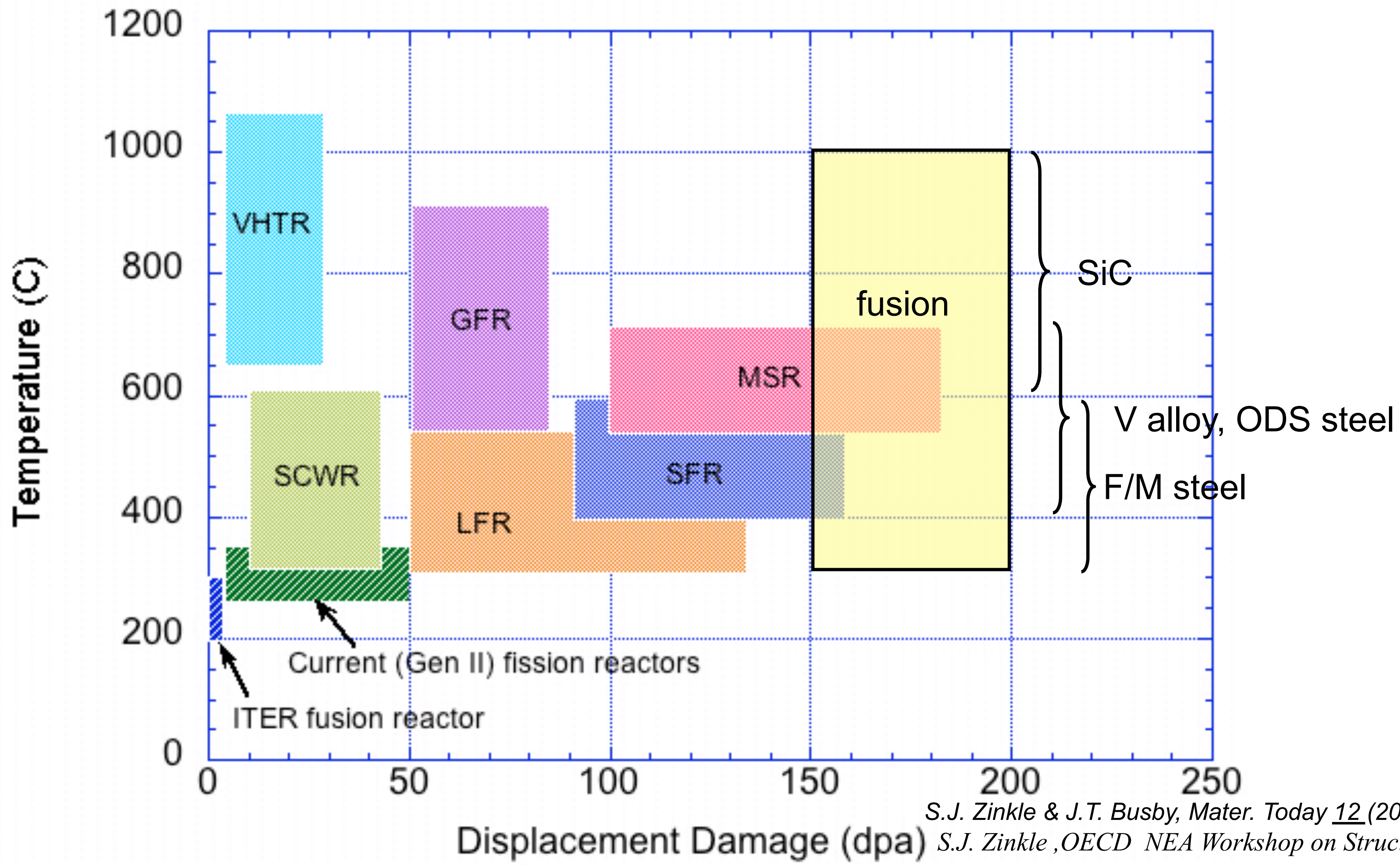
FIG. 5.8. PCI failure due to a wedge shaped pellet chip lodging between pellet and cladding [5.30].

IAEA Technical Report #388

Overview of fission reactor options

- **Light Water Reactors (LWRs): pressurized- and boiling-water designs**
  - Present fleet of nuclear power plants (UO_2 fuel, zircaloy cladding)
  - Pressing for higher performance and improved fuel reliability
- **"Gen-IV" High Temperature Gas-Cooled Reactors**
  - High temperature process heat applications as well as electricity production
  - Selected for Next Generation Nuclear Plant (NGNP)
- **"Gen-IV" Na-cooled Fast Reactors**
  - Close the nuclear fuel cycle by "burning" transuranic isotopes and fission product wastes from LWR plants
- **Other "Gen-IV" reactor concepts**
  - Supercritical water reactor
  - Molten salt reactor
  - Pb-cooled fast reactor
  - Gas-cooled fast reactor

Comparison of Gen IV and Fusion Structural Materials Environments



S.J. Zinkle & J.T. Busby, Mater. Today 12 (2009) 12

S.J. Zinkle ,OECD NEA Workshop on Structural Materials for Innovative Nuclear Energy Systems, Karlsruhe, Germany, June 2007

All Gen IV and Fusion concepts pose severe materials challenges

Overview of desired cladding attributes

- Very low parasitic neutron absorption
  - (dependent on spectrum; Zr is very good for LWRs)
- Good mechanical strength
  - Normal and transient high temperature conditions
- Good compatibility with coolant and fuel (including fission products)
  - (water for most current nuclear reactors)
  - Normal and transient/accident conditions
- Good thermal conductivity
- Good radiation resistance
  - (lifetime dose of ~20 dpa for zircaloy after 40-50 MWd/kgHM)
- High melting temperature
  - Provides additional safety margin for accident conditions
- Isotropic properties

Comparison of properties of candidate cladding base materials

	Mg	Al	Be	Zr	Fe	Cr	Ni	V	Mo	SiC
Thermal neutron absorption cross section (barns)	0.063	0.23	0.009	0.185	2.5	3.1	4.5	5.08	2.6	0.087
Thermal conductivity (W/m-K)	156	237	201	22	80	94	91	31	138	30*
0.5 T_M (°C)	183	194	502	790	630	792	590	808	1170	1278
Crystal structure	hcp	fcc	hcp	hcp	bcc	bcc	fcc	bcc	bcc	fcc

*value for commercial purity SiC; high purity SiC has K_{th}~350 W/m-K

Fuel Assembly Performance

- **Design Functions**
  - **Provide support and protection for the fuel-pin bundle and other components of the subassembly**
  - **Provide a controlled path for the primary coolant**
  - **Provide a compact structural unit that can be easily moved in and out of the core by a refueling machine**
  - **Interact with adjacent subassemblies, retaining ring, and core support plates in a manner that assures safe and predictable reactor geometry**
- Design Issues
  - **Swelling, creep, fatigue, toughness**
  - **Reduced limits for weldments**

Todd Allen/Jeremy Busby

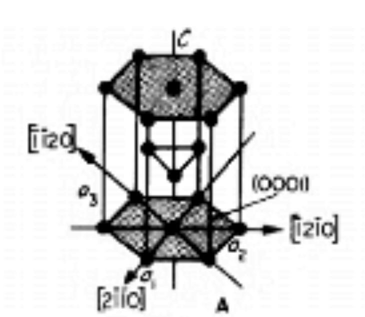
Cladding Performance Issues

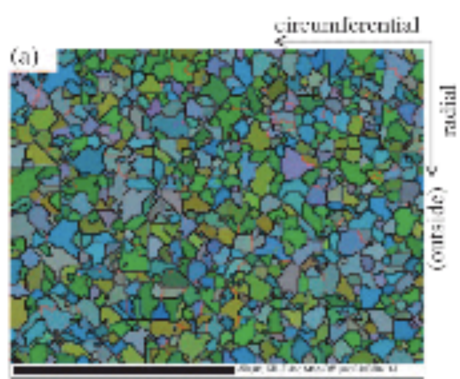
- **Survival of cladding must be predictable**
  - **Wastage-**
    - **Corrosion by the coolant**
    - **Fuel cladding chemical interaction (FCCI)**
  - **Strain**
    - **Fission gas pressurization**
    - **Swelling and associated creep of constrained components**
    - **Fuel cladding mechanical interaction (FCMI)**
  - **Microstructural stability during unplanned transients**
    - **Loss of coolant accident (LOCA), reactivity insertion accident (RIA)**

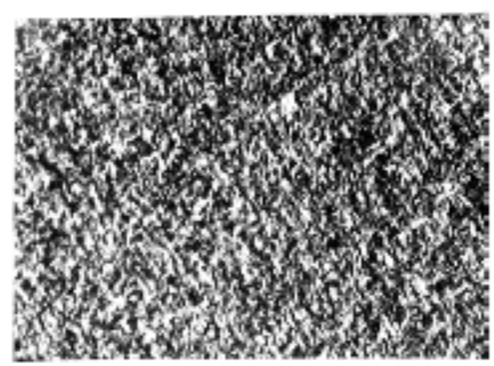
Zry-2 and Zry-4 are the standard LWR cladding materials

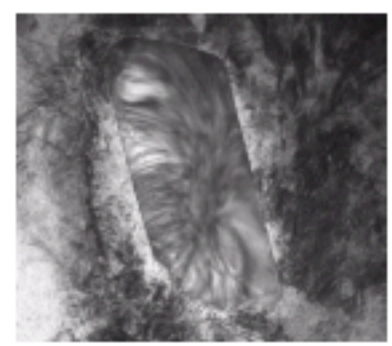
- Zirconium alloys are commonly used as material for cladding tubes in LWR because of their inherent resistance to a wide variety of environmental conditions and their neutron transparency
- The most used alloys are Zircaloy 2 in BWR and Zircaloy 4 in PWR

	Sn (%)	Fe (%)	Cr (%)	Ni (%)	O (%)	Structure
Zircaloy 2	1.2 – 1.5	0.07 – 0.2	0.05 – 0.15	0.03 – 0.08	0.09 – 0.16	RXA
Zircaloy 4	1.2 – 1.5	0.18 – 0.24	0.07 – 0.13	-	0.09 – 0.16	CWSR









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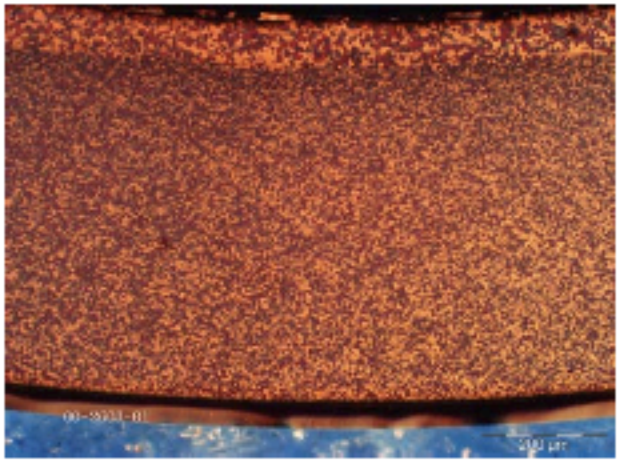
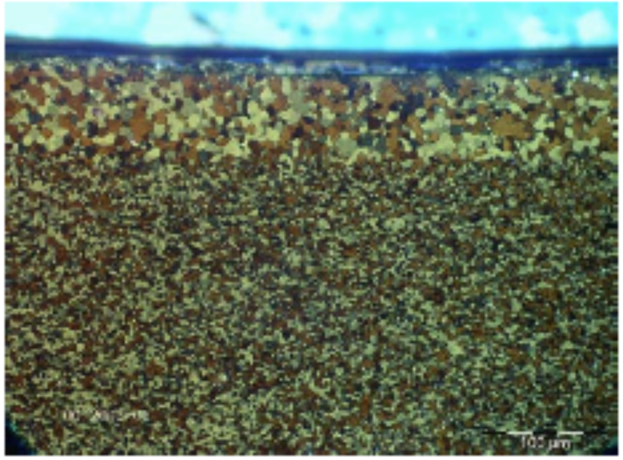
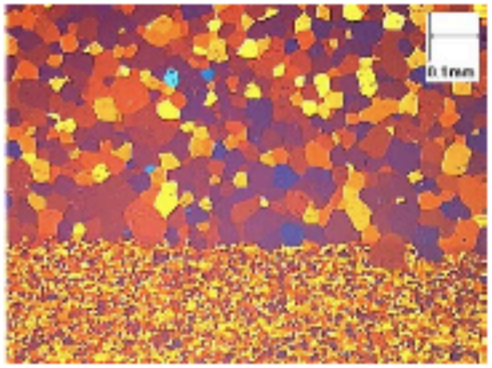
Source: B. Cheng

Zry-2 has been optimized for BWR performance via liners

- In order to mitigate the risk of failure during power transients in BWR, the fuel vendors have developed a barrier cladding by co-extrusion of Zircaloy 2 bulk with an alloyed inner liner

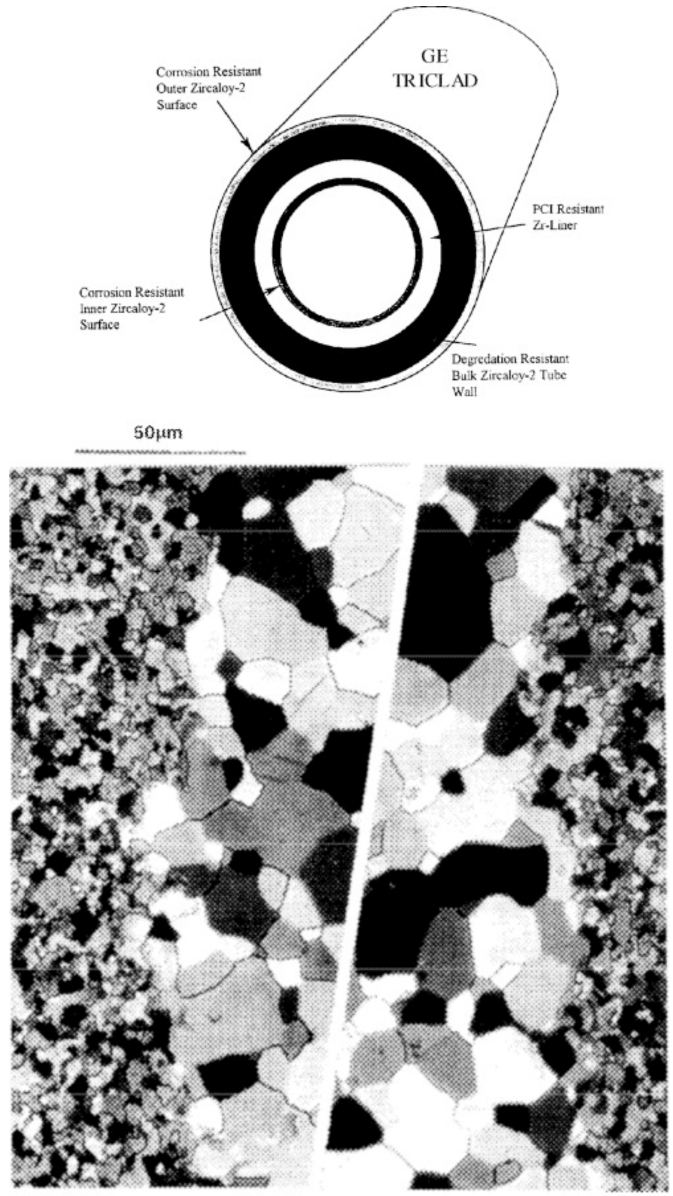
	Sn (%)	Fe (%)	Cr (%)	Ni (%)	O (%)	Structure
Zircaloy 2	1,2 – 1,45	0,15 – 0,20	0,05 – 0,15	0,03 – 0,08	0,09 – 0,16	RXA
Liner	0,1 – 0,4	0,04 – 0,07	Max 0,02	Max 0,007	Max 0,06	RXA

Todd Allen/Jeremy Busby

Source: B. Cheng

"Triclad" architecture of Zircaloy tube

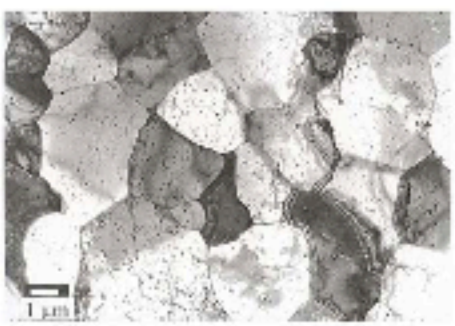


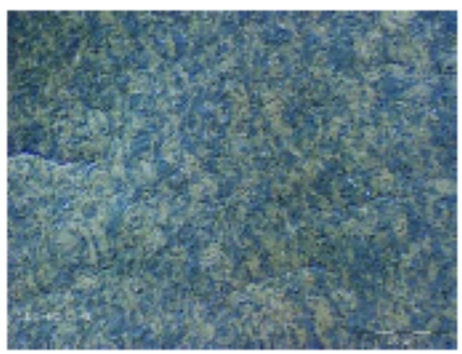
K.L. Murty and I Charit, Prog. Nucl. Energy 48 (2006) 325

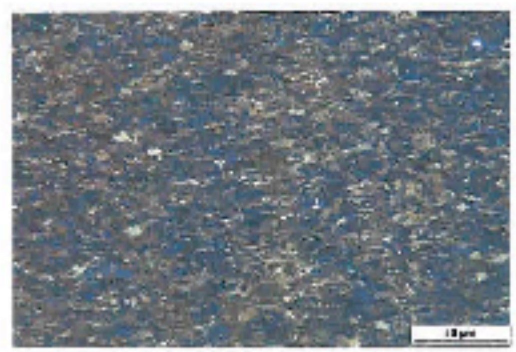
Zry-4 is being replaced with modern cladding alloys

- According to the evolution of PWR operating conditions (load follow, high burn up), the fuel vendors have developed new Zr alloys with improved performances
  - *Nb based binary and quaternary alloys*

	Sn (%)	Nb (%)	Fe (%)	O (%)	Structure
M5	-	1	0.015 – 0.037	0.118 – 0.148	RXA
Zirlo	0.8 – 1.1	0.8 – 1.2	0.09 – 0.13	0.105 – 0.145	CWSR
Opt.Zirlo	0.6 – 0.8	0.8 – 1.2	0.09 – 0.13	0.105 – 0.145	PRXA







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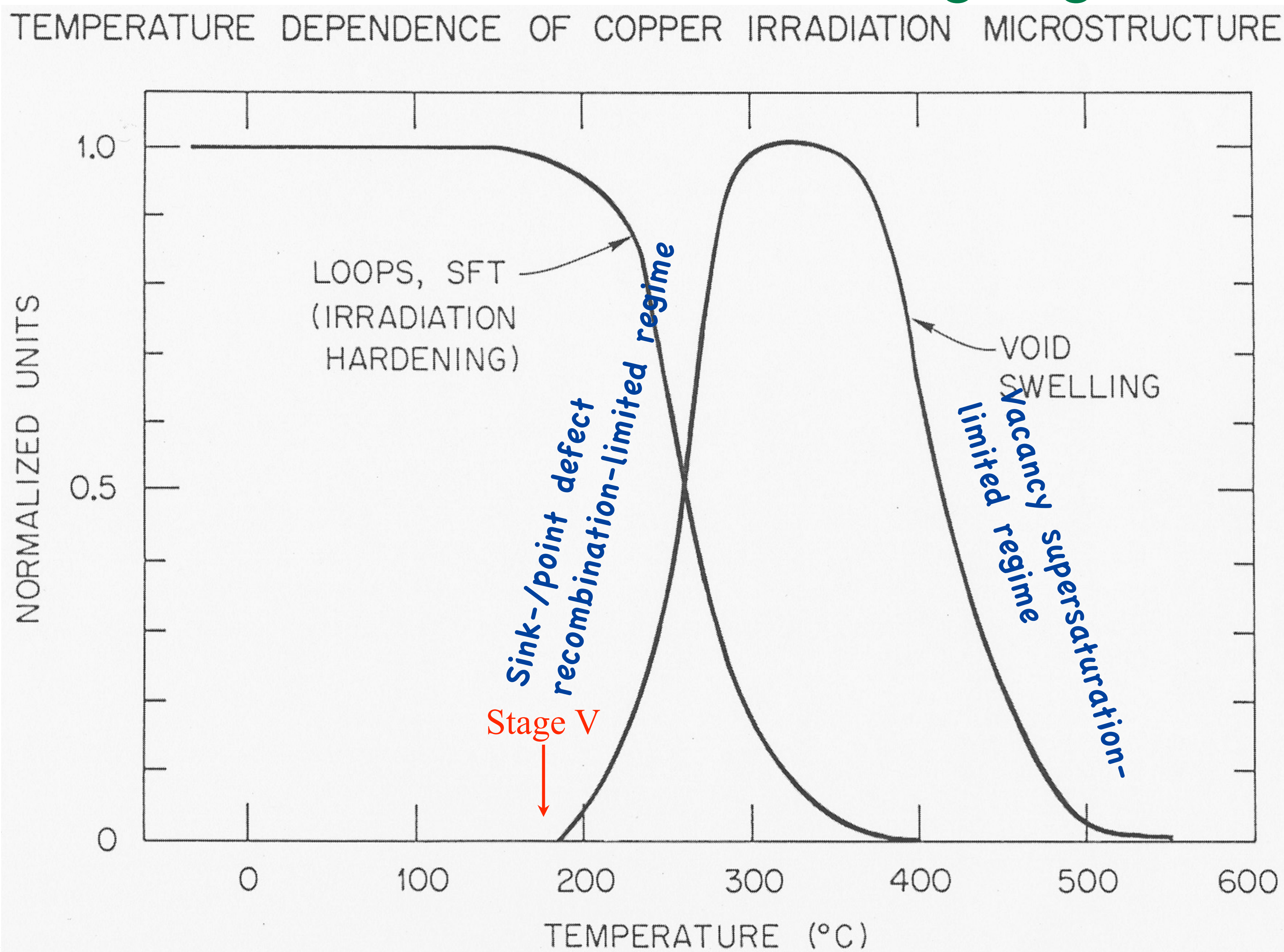
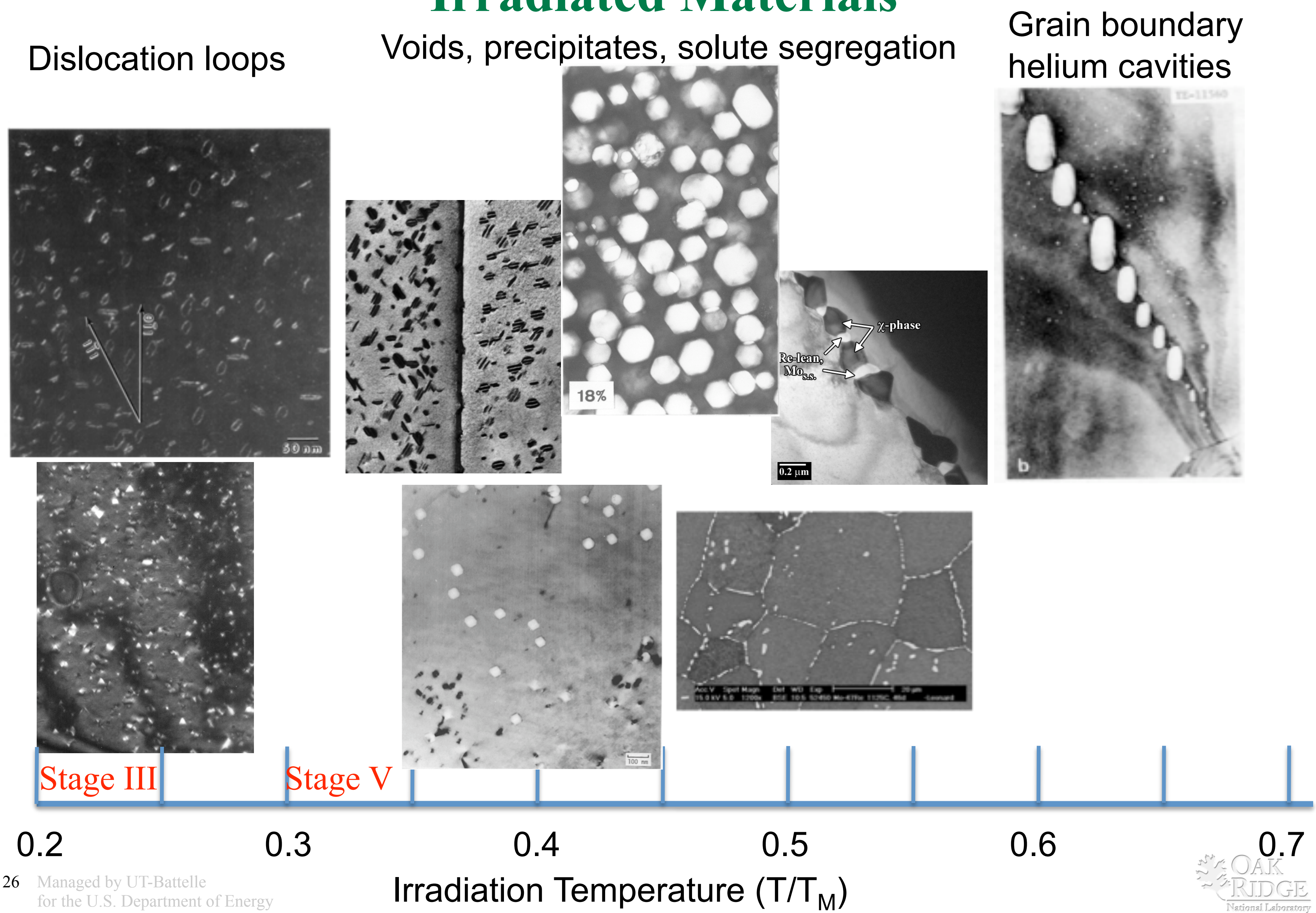
Source: B. Cheng

General overview of main radiation damage regimes



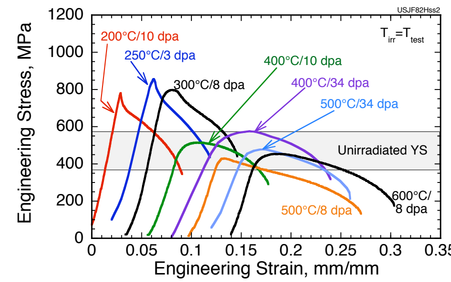
Zinkle, ASTM STP 1125 (1992) p. 813

Overview of Defect Microstructures in Irradiated Materials

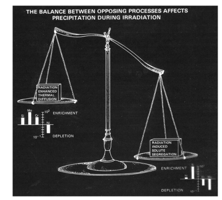


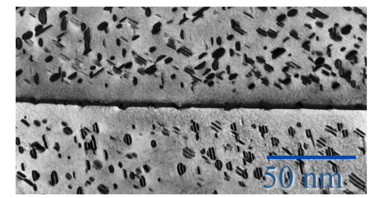
Radiation Damage can Produce Large Changes in Structural Materials

- **Radiation hardening and embrittlement (<0.4 T_M, >0.1 dpa)**



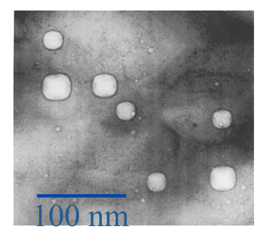
- **Phase instabilities from radiation-induced precipitation (0.3-0.6 T_M, >10 dpa)**



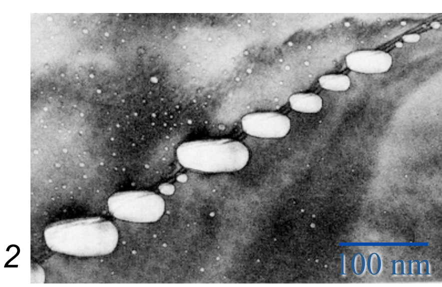


- **Irradiation growth and creep (<0.45 T_M, >10 dpa)**

- **Volumetric swelling from void formation (0.3-0.6 T_M, >10 dpa)**

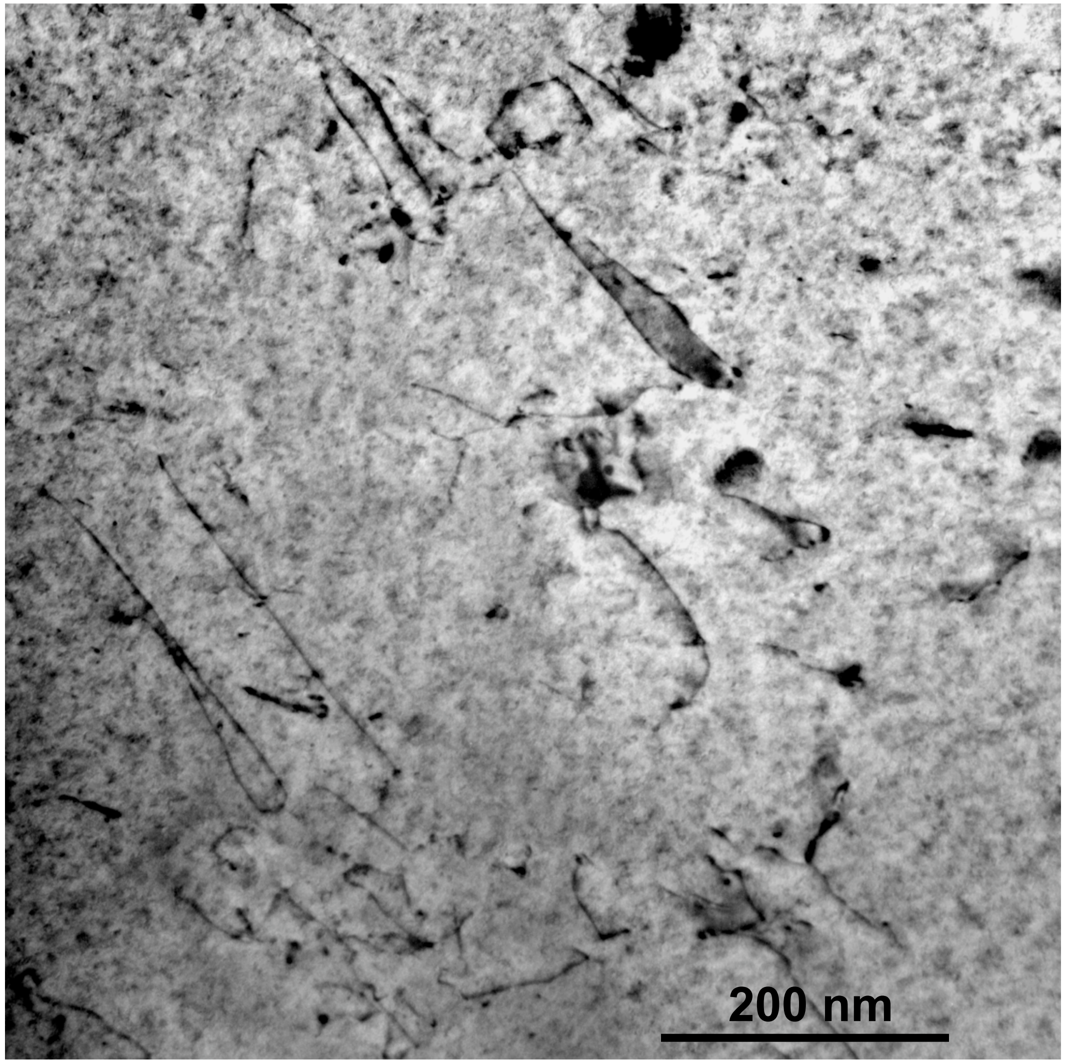


- **High temperature He embrittlement (>0.5 T_M, >10 dpa)**



S.J. Zinkle, Phys. Plasmas 12 (2005) 058101; Zinkle & Busby, Mater. Today 12 (2009) 12

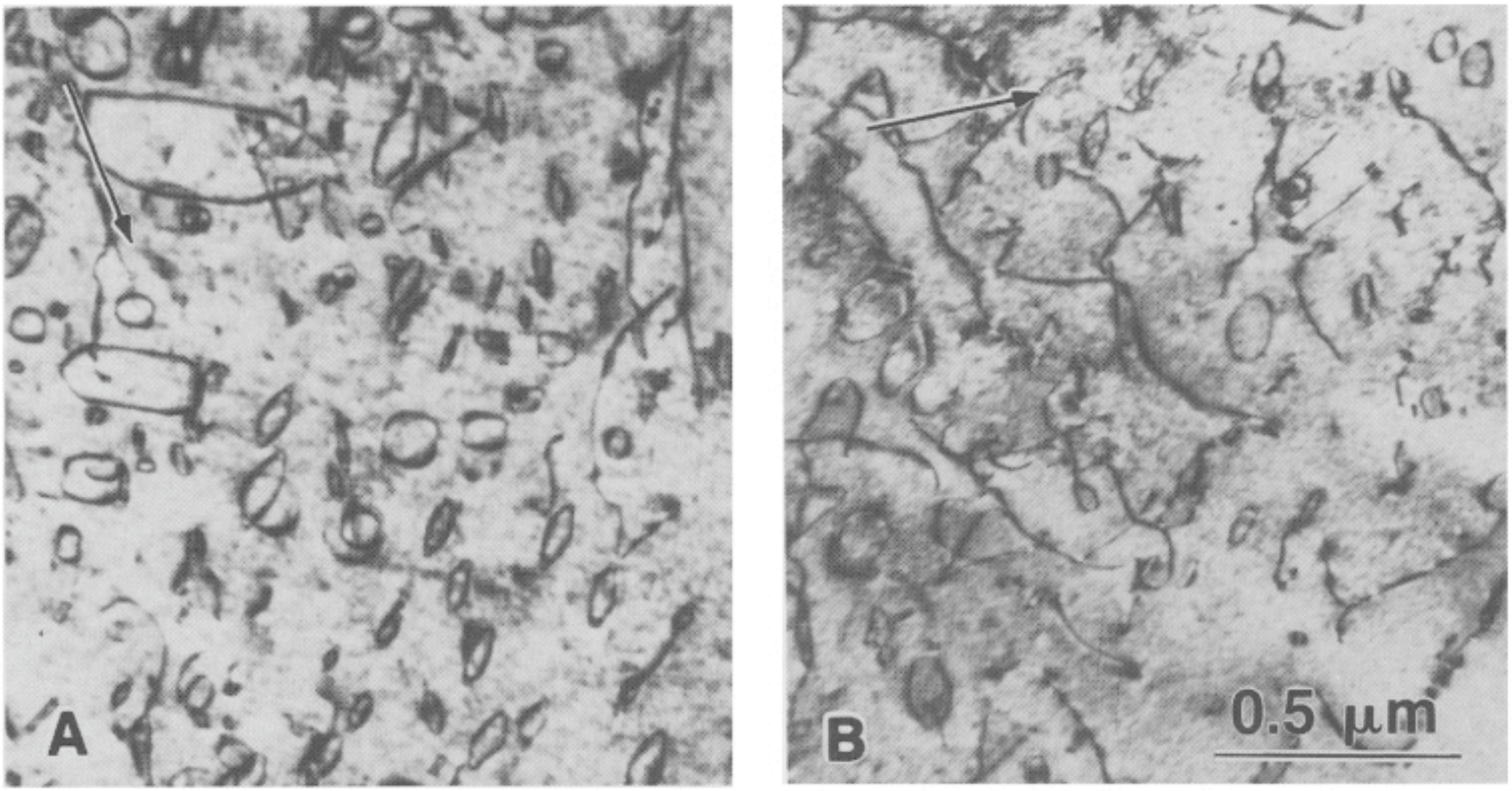
Microstructure of Zircaloy-4 after 0.1 dpa at 350°C



K. Farrell et al., ORNL/TM-2003/63 (2003)

<a> type dislocation microstructure in neutron irradiated zircaloy-4 at 425^{o}C

3.7 dpa　　　　　　　　50 dpa



Micrographs illustrating the evolution of $\langle \boldsymbol{a} \rangle$ type dislocation microstructure during irradiation in EBR-II at 700 K: (a) 1.1×10^{25} n m^{-2}; (b) 1.5×10^{26} n m^{-2}. Diffracting vector $\boldsymbol{g} = 10\bar{1}1$ and beam direction $\boldsymbol{B} = [0\bar{1}11]$ in each case.

M. Griffiths, J. Nucl. Mater. 159 (1988) 190

Dislocation loops form in Zr-alloys in similar sizes and densities as other irradiated metals such as stainless steel

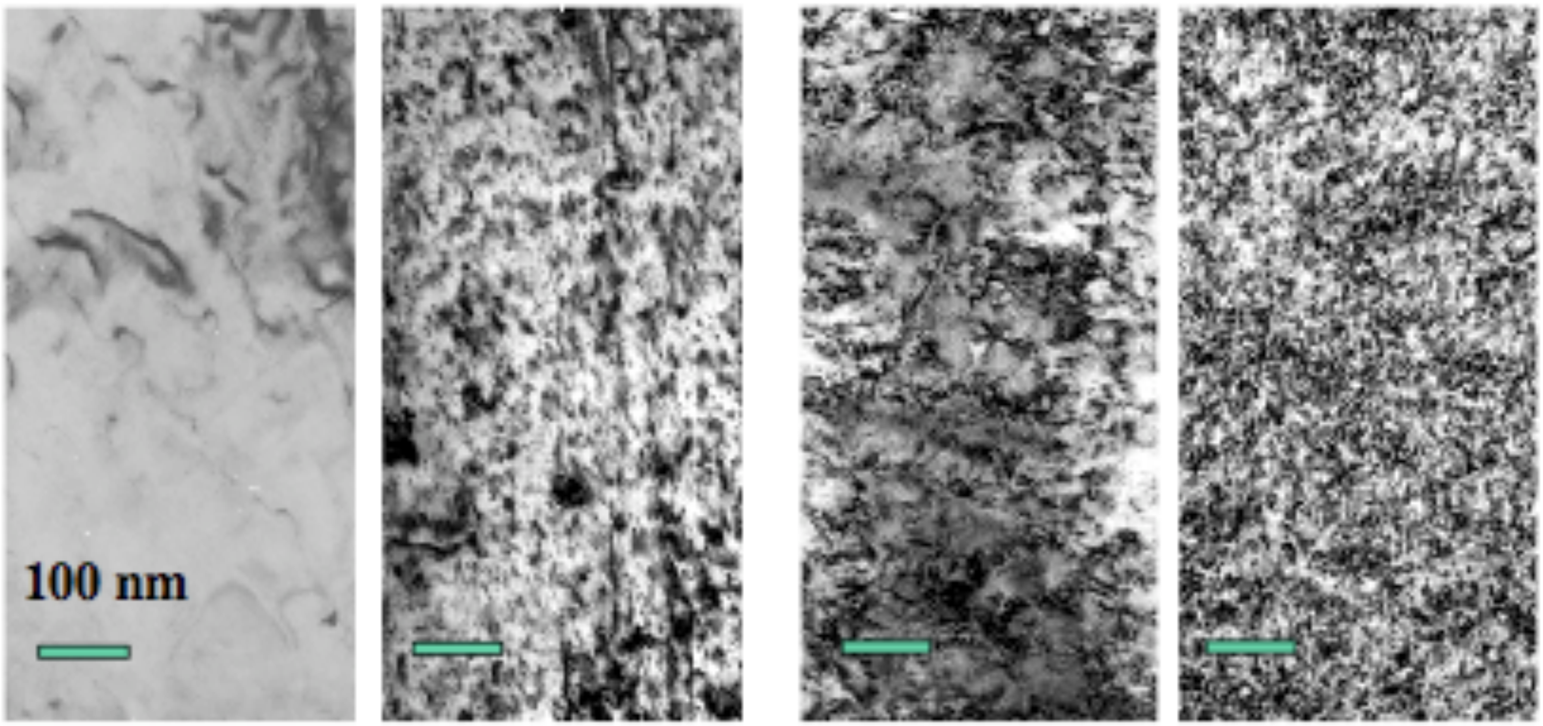


Figure 5 – *Bright-field tran*
evolution of dislocation loc

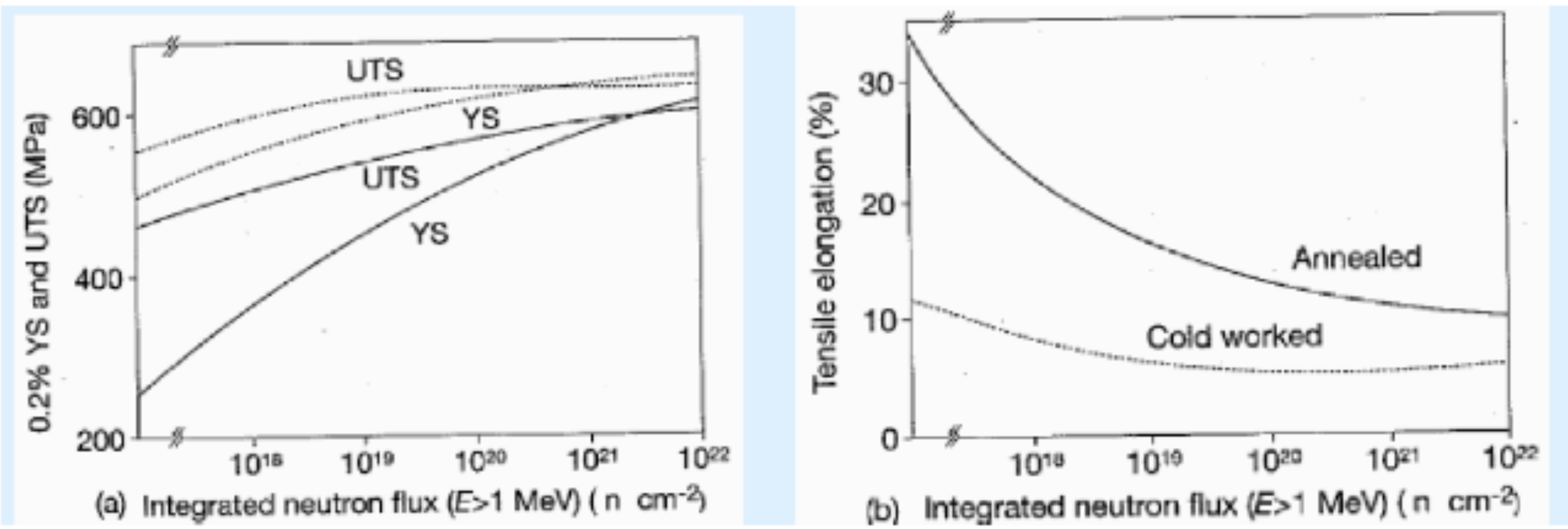
Table I Effect of proton irradiation on loop density and average diameter

	As received	2 dpa	5 dpa	7 dpa
Average diameter (nm)	-	7	11	11
Density (10^{21} m^{-3})	0	7	8	15

Todd Allen/Jeremy Busby

Irradiation damage influences Zr-alloys in the same manner as steels

- Irradiation damage has a significant effect on burst and tensile properties by
  - inducing hardening (For CWSR material, the initial high dislocation loop density attenuates the irradiation effect)
  - reducing the ductility

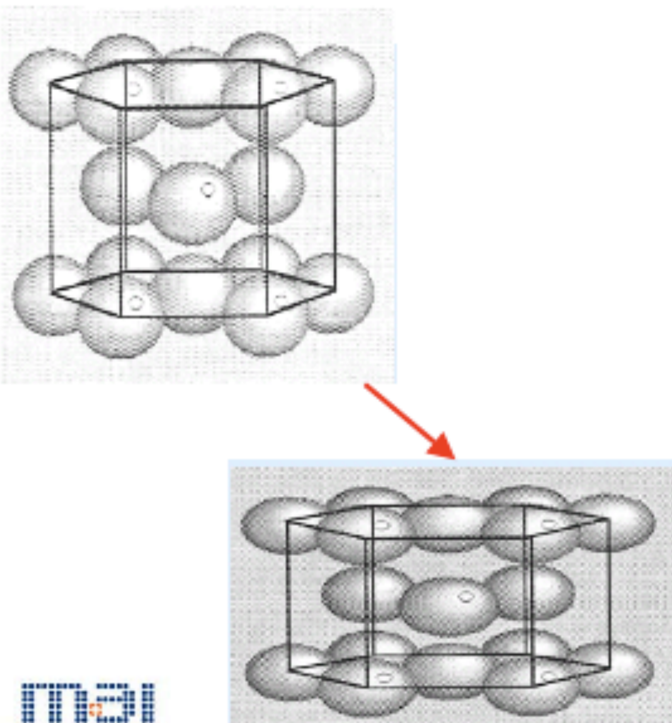


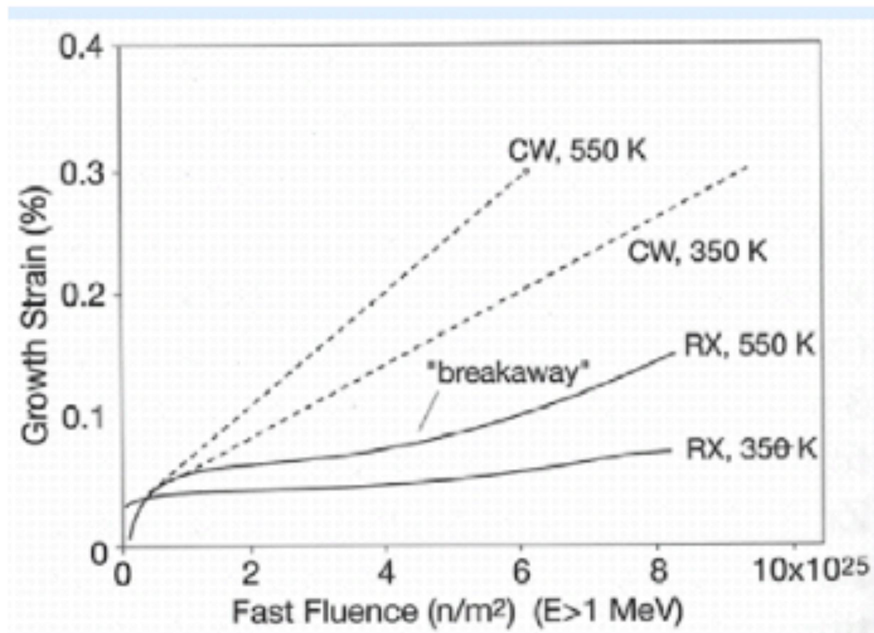
Todd Allen/Jeremy Busby

Source: B. Cheng

Growth in Zr-based alloys is likely due to partitioning of dislocation loops

- Dimensional changes of the cladding at constant volume in the absence of stress application
  - Expansion along the <a> axis of the hexagonal crystallite concomitant with contraction along <c> axis
  - According to texture, irradiation growth evinces axial length increasing



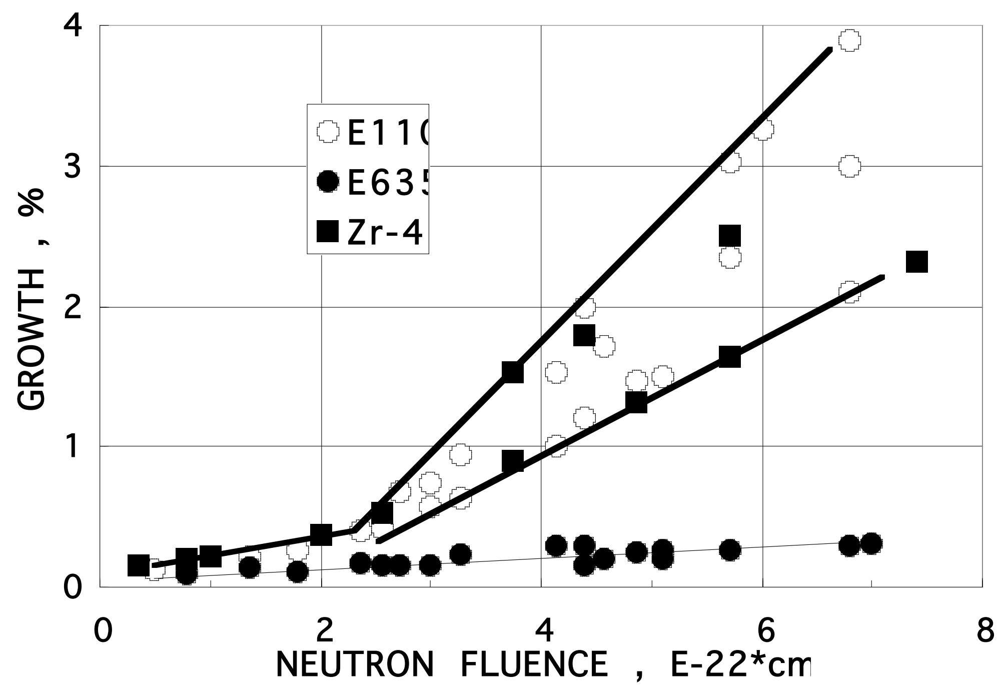


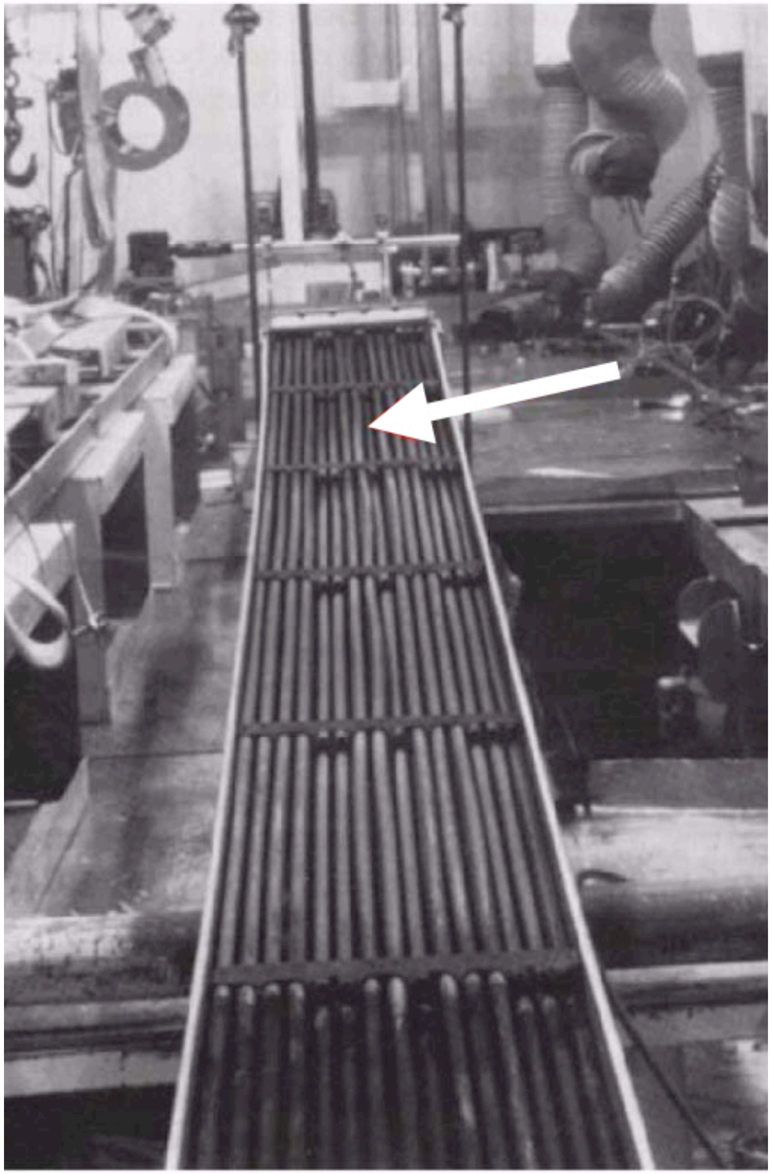
Todd Allen/Jeremy Busby

Source: B. Cheng

Fuel assembly bowing

- Fuel rod bow – due to interaction with top and bottom nozzles
- Assembly bow from variation in growth of Zircaloy as a function of flux gradients.
- Consequences: difficulties in positioning fuel during loading and unloading operations, and insertion of control rods.

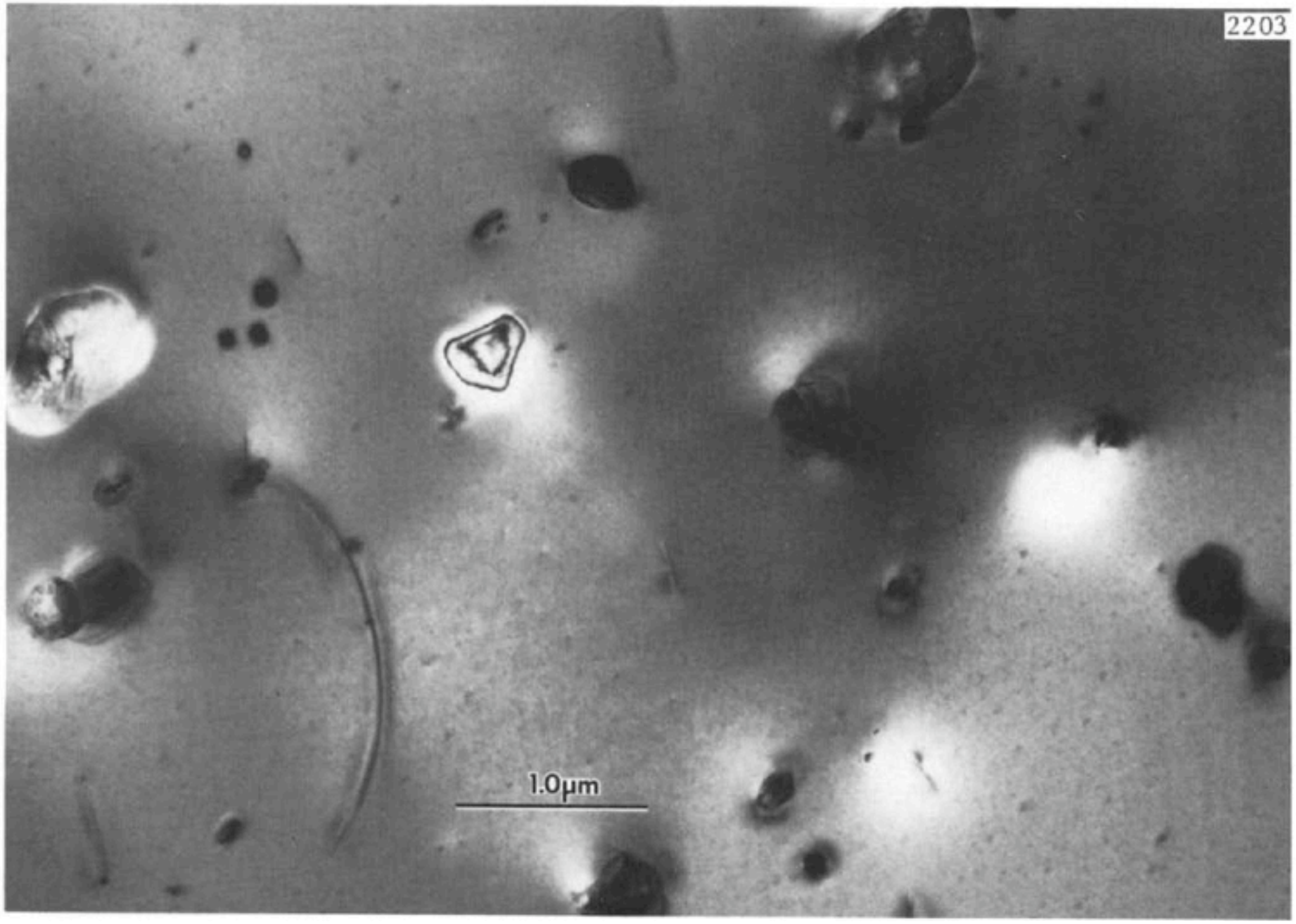




D.G. Franklin & R.B. Adamson, 1988

Totju Totev

$Zr(Fe,Cr)_2$ precipitates in unirradiated Zircaloy-4



W.J.S. Yang, J. Nucl. Mater. 159 (1988) 71

Precipitate amorphization has been observed in LWR cladding

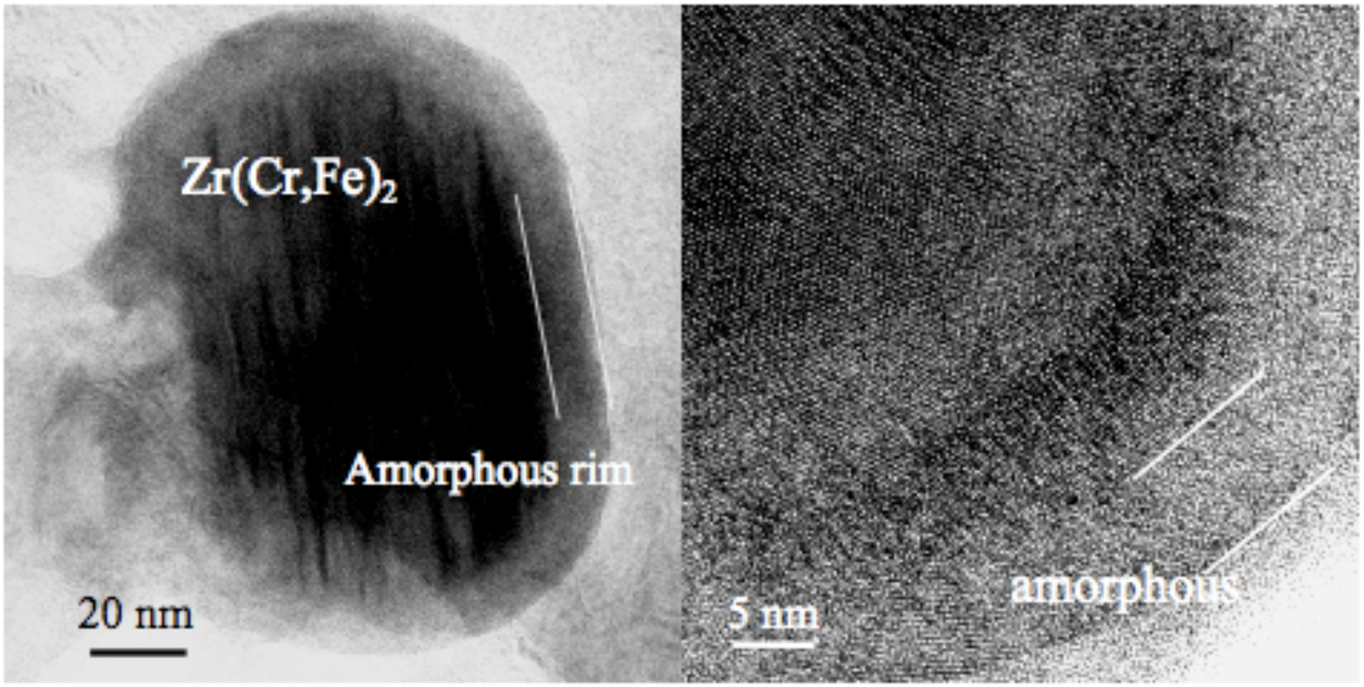


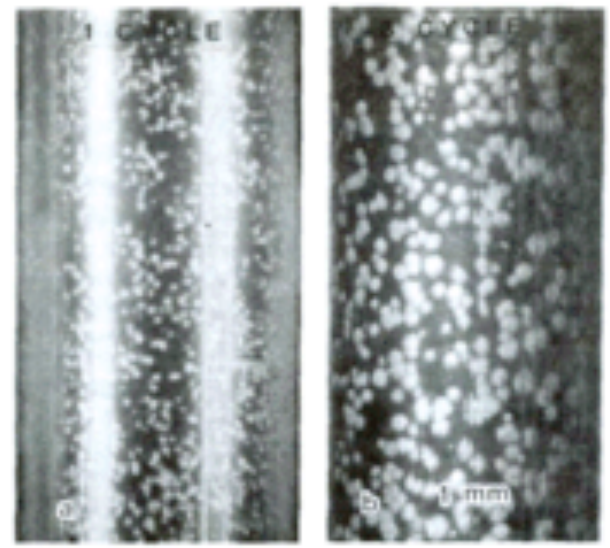
Figure 7 – conventional bright-field (left) and high-resolution (right) image of a $Zr(Cr,Fe)_2$ precipitate after irradiation to 5 dpa at 310 °C. The amorphous structure was confirmedfor the bright-field image based on absence of diffraction contrast upon changing sample orientation.

Todd Allen/Jeremy Busby

Amorphization of Zr(Fe,Cr) particles can influence a number of cladding properties

- **Breakdown of particles can influence mechanical performance**
- **Breakdown of particles may influence oxide formation**
- **Release of Cr and Fe to matrix or oxide may influence growth and stability of oxide layers.**

Overview of Aqueous Corrosion of Zr alloys



Nodular corrosion

- BWR
  - *Nodular corrosion*
  - *Unfavorable water chemistry*
  - *Crud induced localized corrosion (CILC)*
- PWR
  - *Higher temperature than BWR so uniform corrosion more of an issue*
  - *Uniform corrosion-increases at high burnup*
    - *Decrease in thermal conductivity and associated increase in temperature*
    - *Zirconium hydrides form brittle region*

Todd Allen

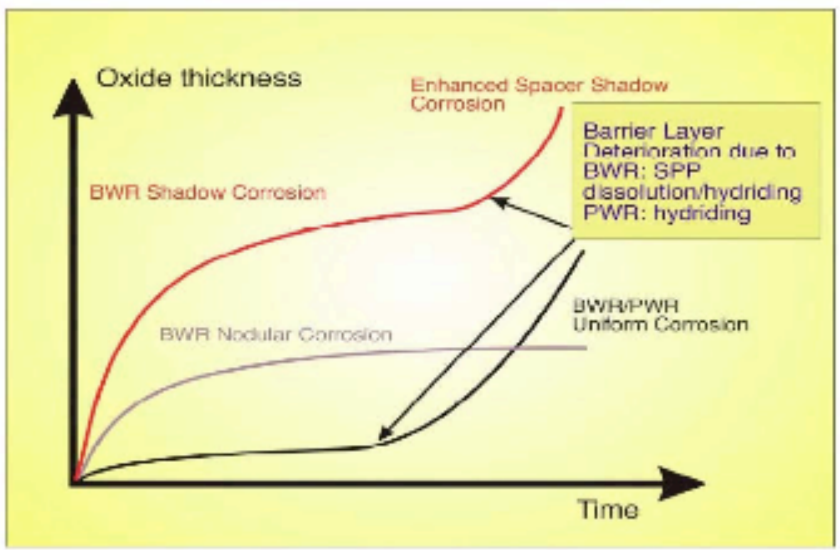
General corrosion is the dominant form of degradation

- **In primary environment (water or steam), Zr alloy cladding undergoes corrosion according to following chemical reaction**

$$Zr + 2H_2O \rightarrow ZrO_2 + 2(1 - w)H_2(\text{coolant}) + 4w\ H(\text{metal})$$

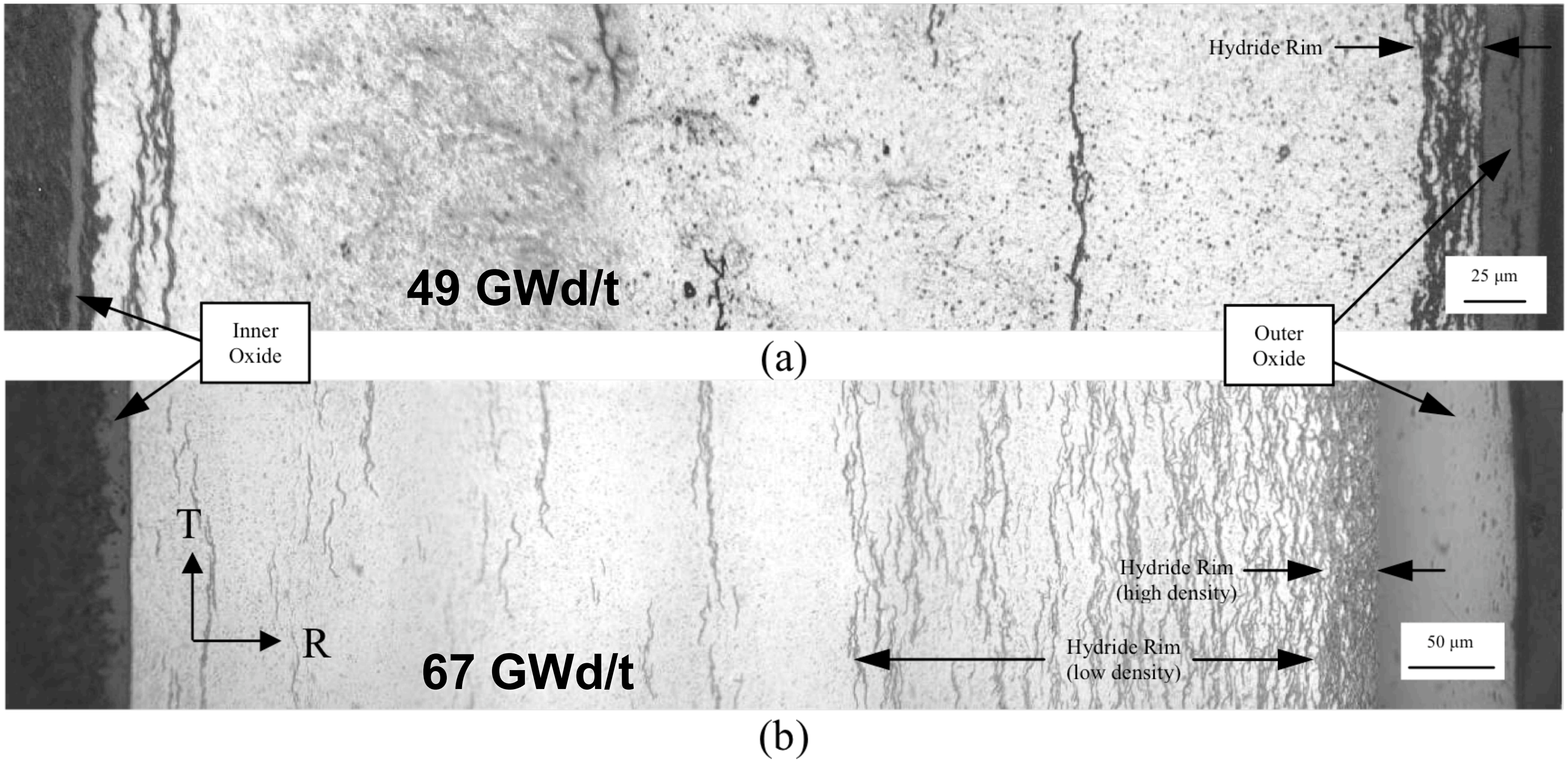
w : fraction of reaction produced hydrogen absorbed by the metal

- **Progressive formation of a ZrO_2 layer**
- **Hydriding of the cladding metal bulk**



Source: B. Cheng

Optical microstructure of irradiated Zircaloy-4 cladding



Inner surface

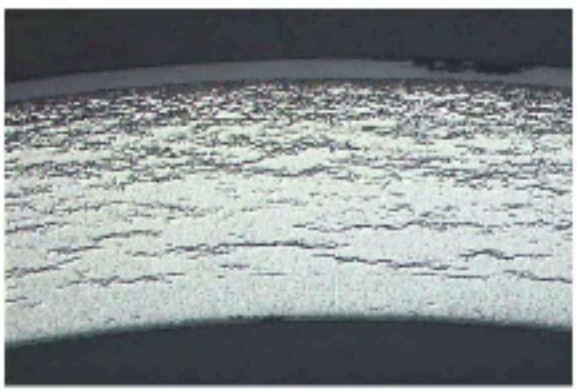
outer surface

R.S. Daum (ANL)

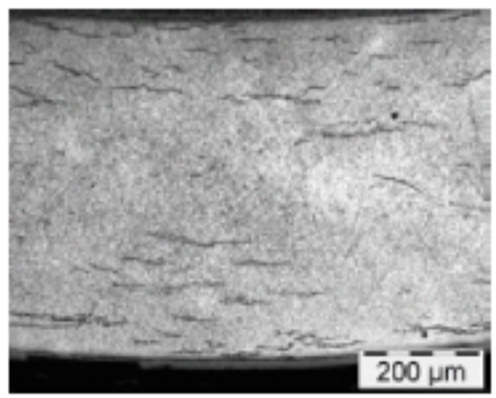
Hydrides form platelets within the metal matrix

➢ Zr hydrides are normally distributed over the whole cladding thickness, precipitating under the form of platelets, preferentially aligned along the hoop direction according to texture

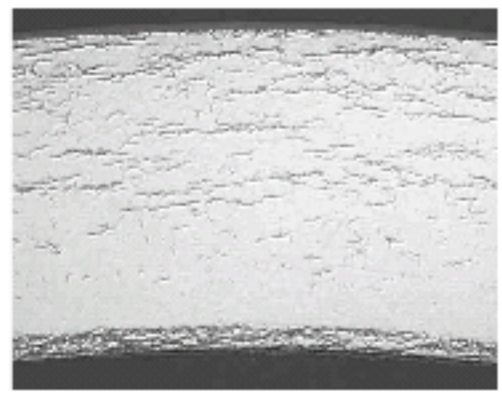
- PWR (High Burn up)
  - CWSR Zy4 (ZrO_2 > 50 µm), a hydride rim of about 30 µm to 60 µm is observed close to the colder outer surface of cladding
  - M5 cladding (ZrO_2 ≈ 20 µm), according to low hydrogen pick-up, no hydride rim is observed
- BWR
  - ➢ Barrier Zy2 cladding (ZrO_2 ≈ 20 µm), most of the hydrides tend to precipitate in the liner



PWR – CWSR Zy4 - ZrO_2 > 50 µm



PWR – RXA M5 – ZrO2 ≈ 20 µm



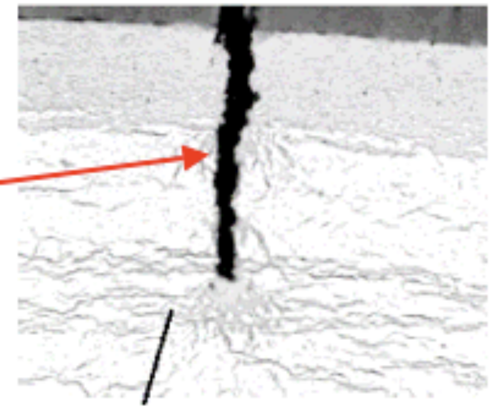
BWR – Zy2 with liner - ZrO_2 ≈ 20 µm

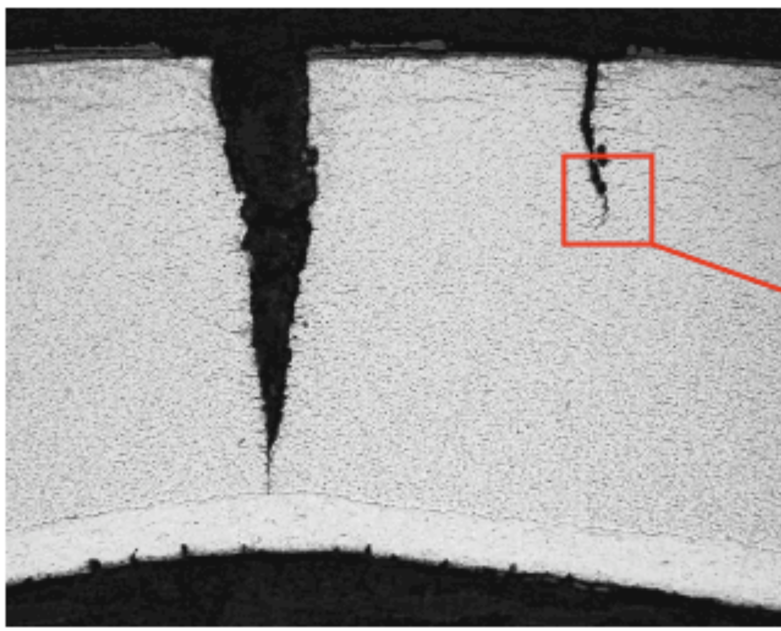
Todd Allen/Jeremy Busby

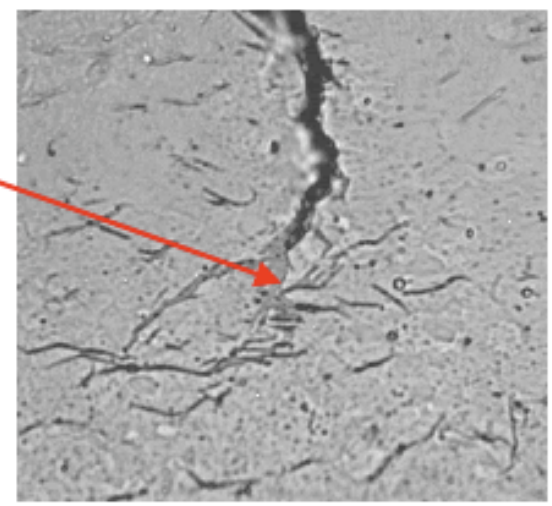
Source: B. Cheng

Hydride failures may be long-term or delayed

- Fracture proceeds by Delayed Hydride Cracking (DHC) mechanism
  - Phenomenon might be activated under decreasing temperature, for instance during Dry Storage
  - The pre-existence of a crack is required (For instance, initiation in hydride rim under reactor operation)
  - Propagation of the crack is assisted by hydrogen diffusion and hydride precipitation at the crack tip





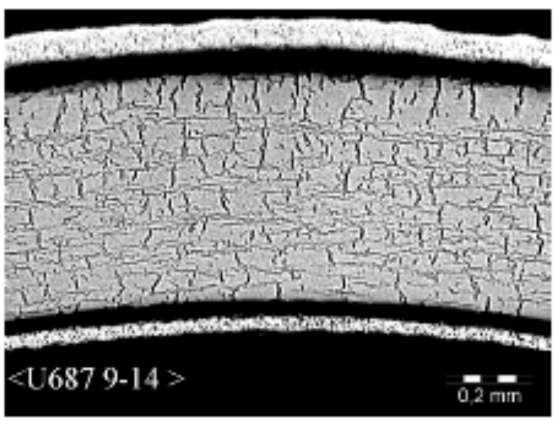


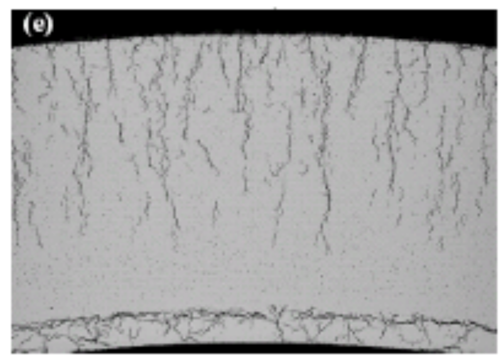
Todd Allen/Jeremy Busby

Source: B. Cheng

Stress (magnitude and direction) can influence existing hydrides

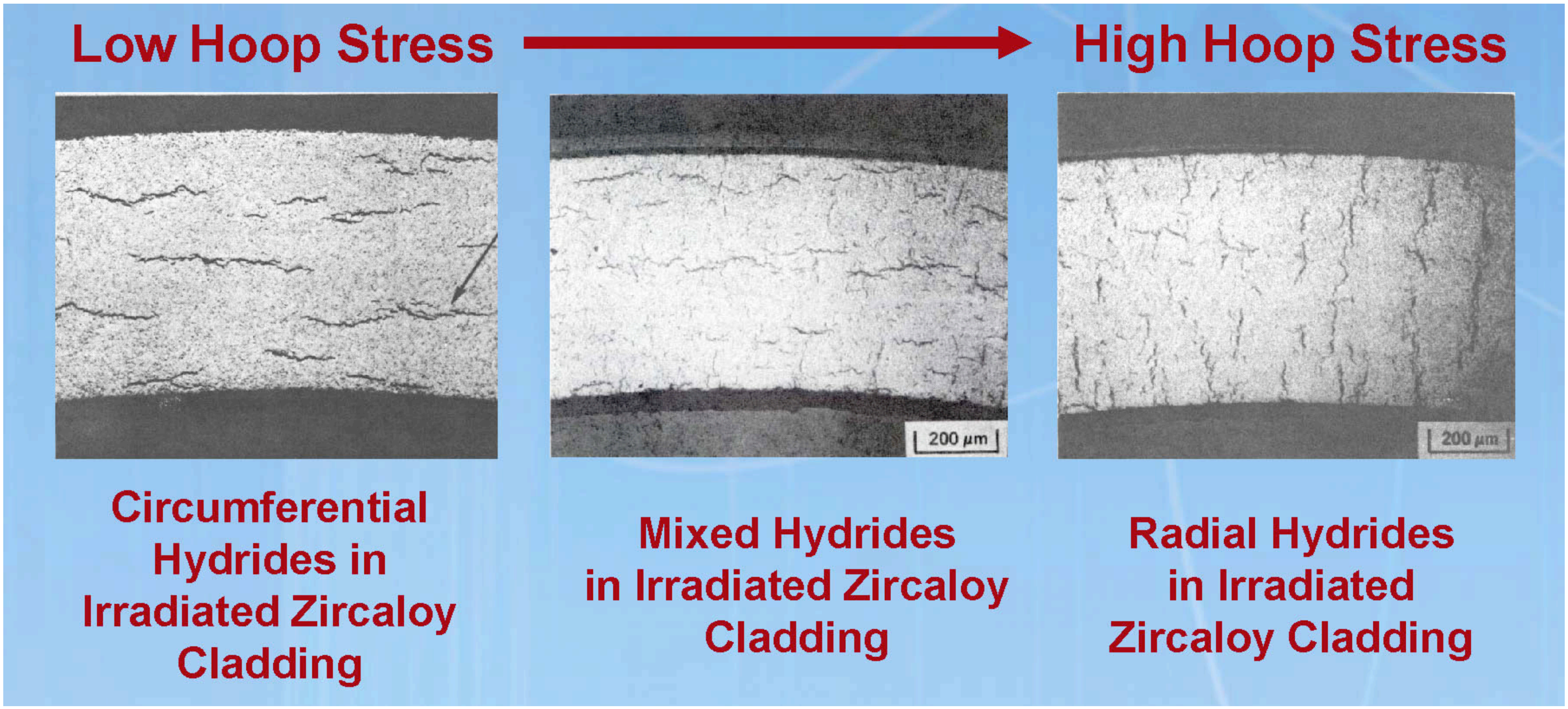
- Hydride reorientation (HRO) from hoop to radial direction may occur when the cladding is cooled down under the tensile hoop stress generated by the internal pressure of the fuel rod
- The precipitation of radial hydrides is observed whether the cladding hoop stress exceeds some critical value
- The radial hydrides have a deleterious impact on the mechanical properties of the cladding i.e Ductility drop
- HRO phenomenon is a key issue with respect to the Spent Nuclear Fuel (SNF) integrity during dry transportation and storage





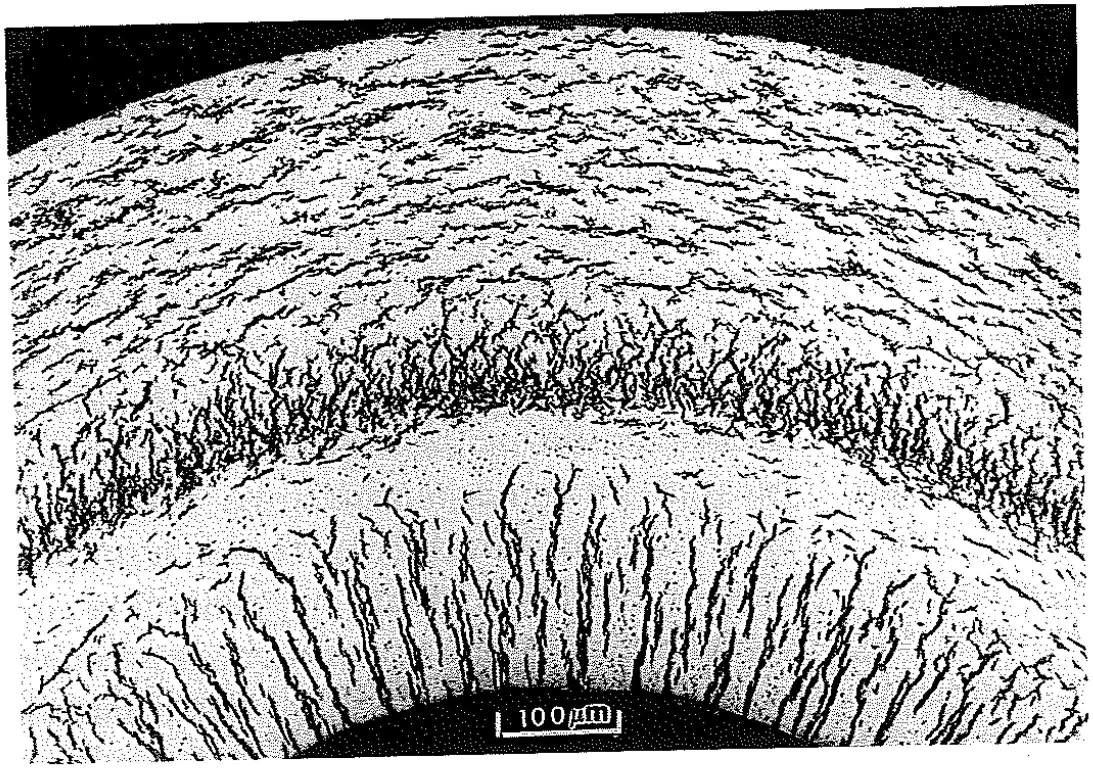
Source: B. Cheng

Role of Stress on Hydride formation in Zircaloy cladding



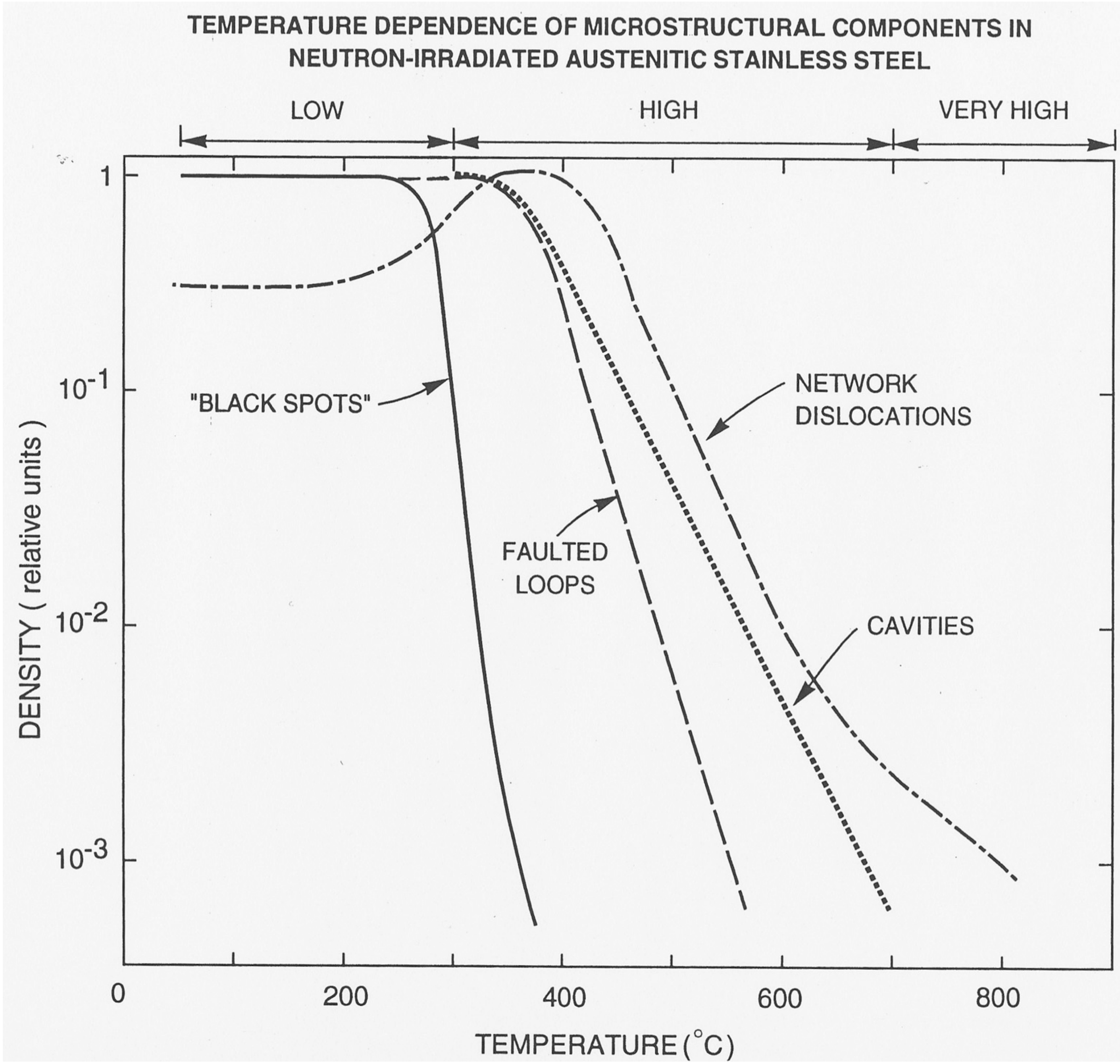
J. Voglewede, NRC (PIE workshop, March 2011)

Role of Stress on Hydride formation in Zircaloy cladding



Tensile stress affects hydride formation on the inner diameter of the cladding;
Typical hydride habit planes are {1010} in pure Zr and {1017} in zircaloys, epitaxial with the matrix: $(111)_{ZrH}||(0001)_{Zr}$

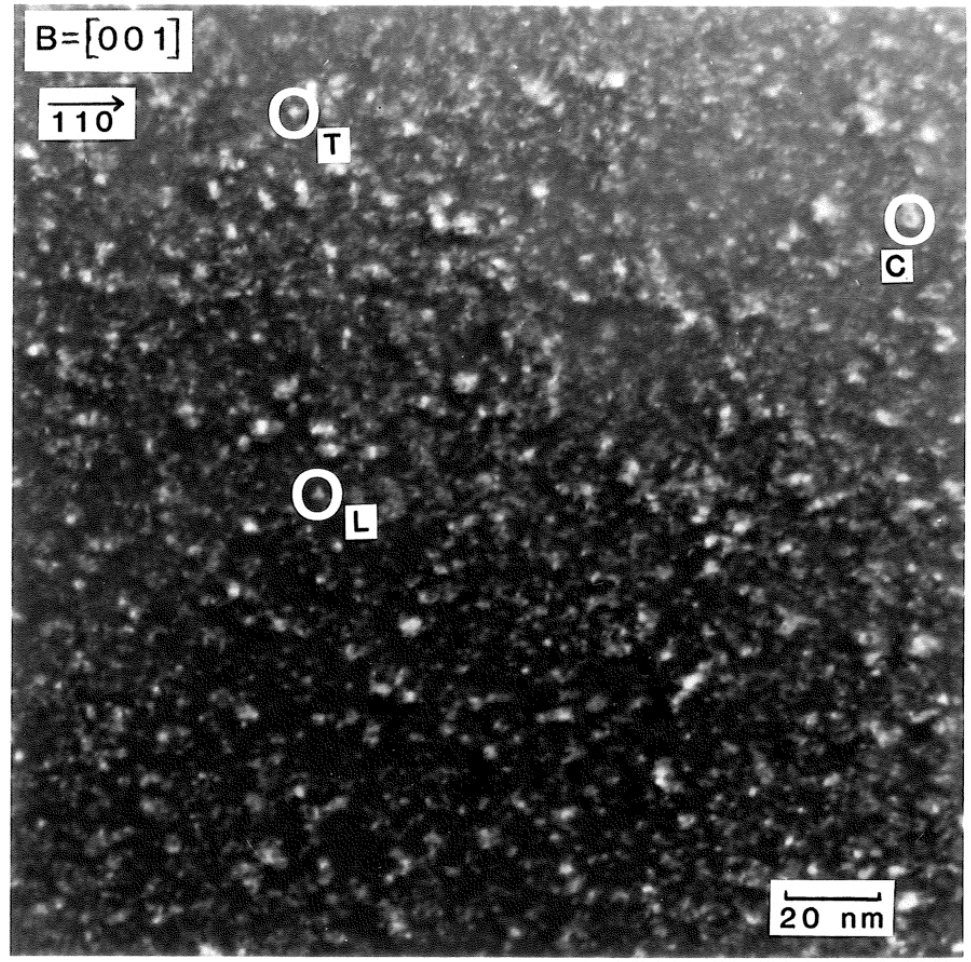
C. Lemaignan, A.T. Motta, Nuclear Materials, vol. 10B (Wiley-VCH, 1994) p.1
(from C.E. Coleman, AECL)



S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

Defect clusters in Type 304L stainless steel following neutron irradiation near 120^{o}C

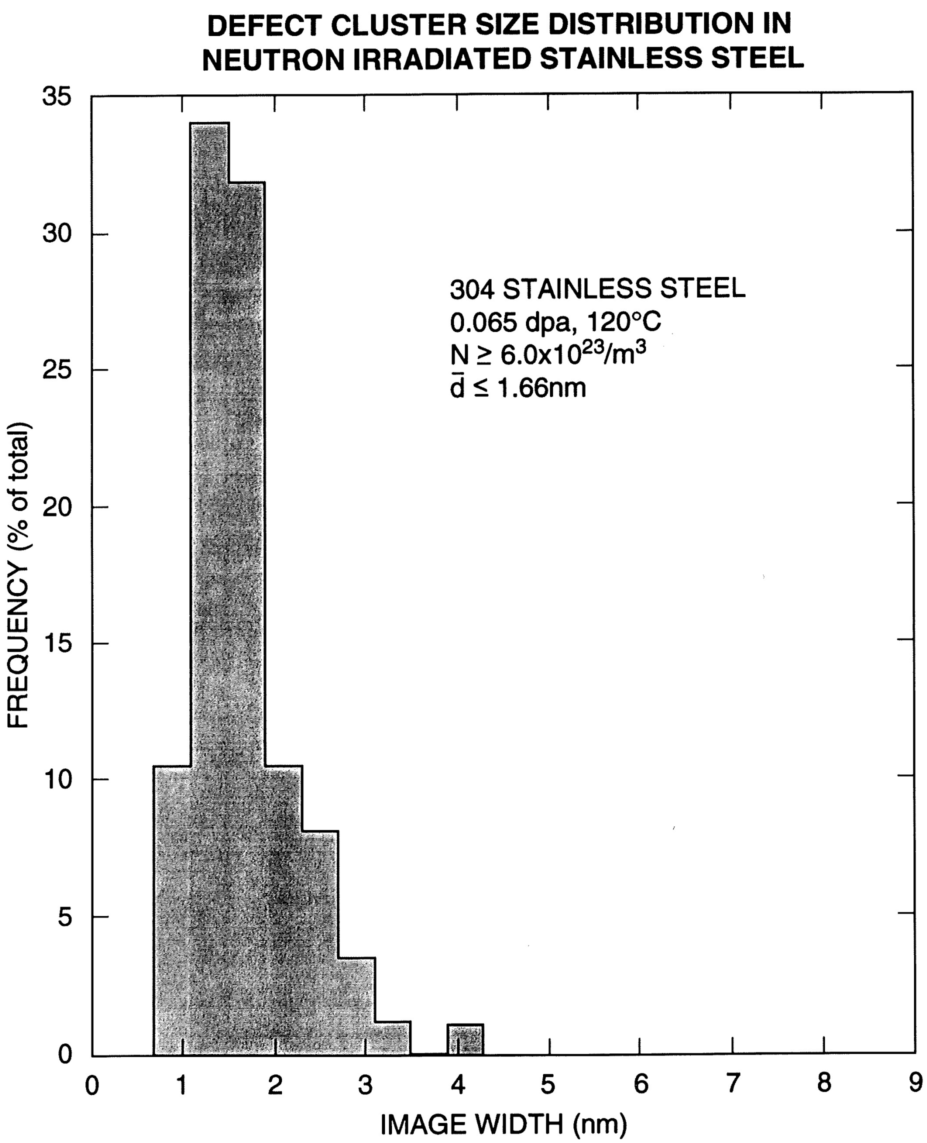
0.065 dpa

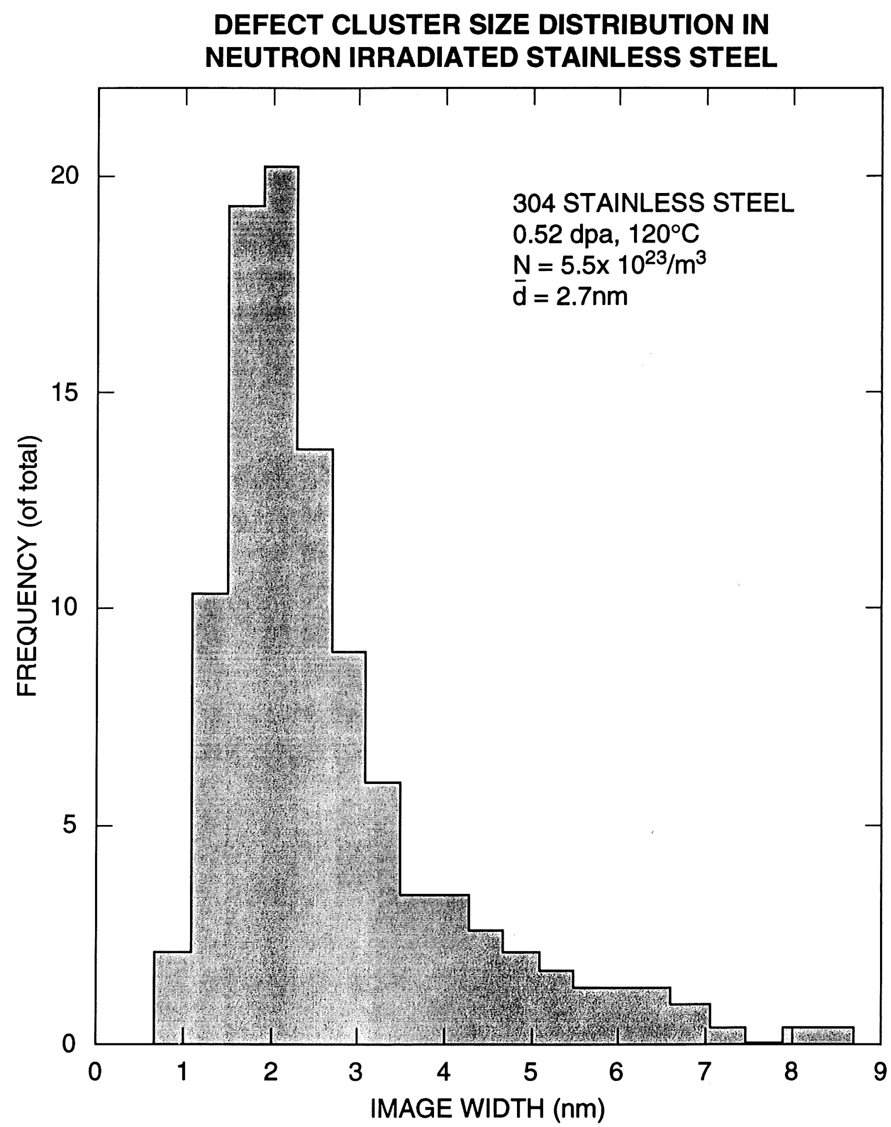


0.52 dpa

S.J. Zinkle, R.L. Sindelar, J. Nucl. Mater. 155-157(1988) 1196 and unpublished results

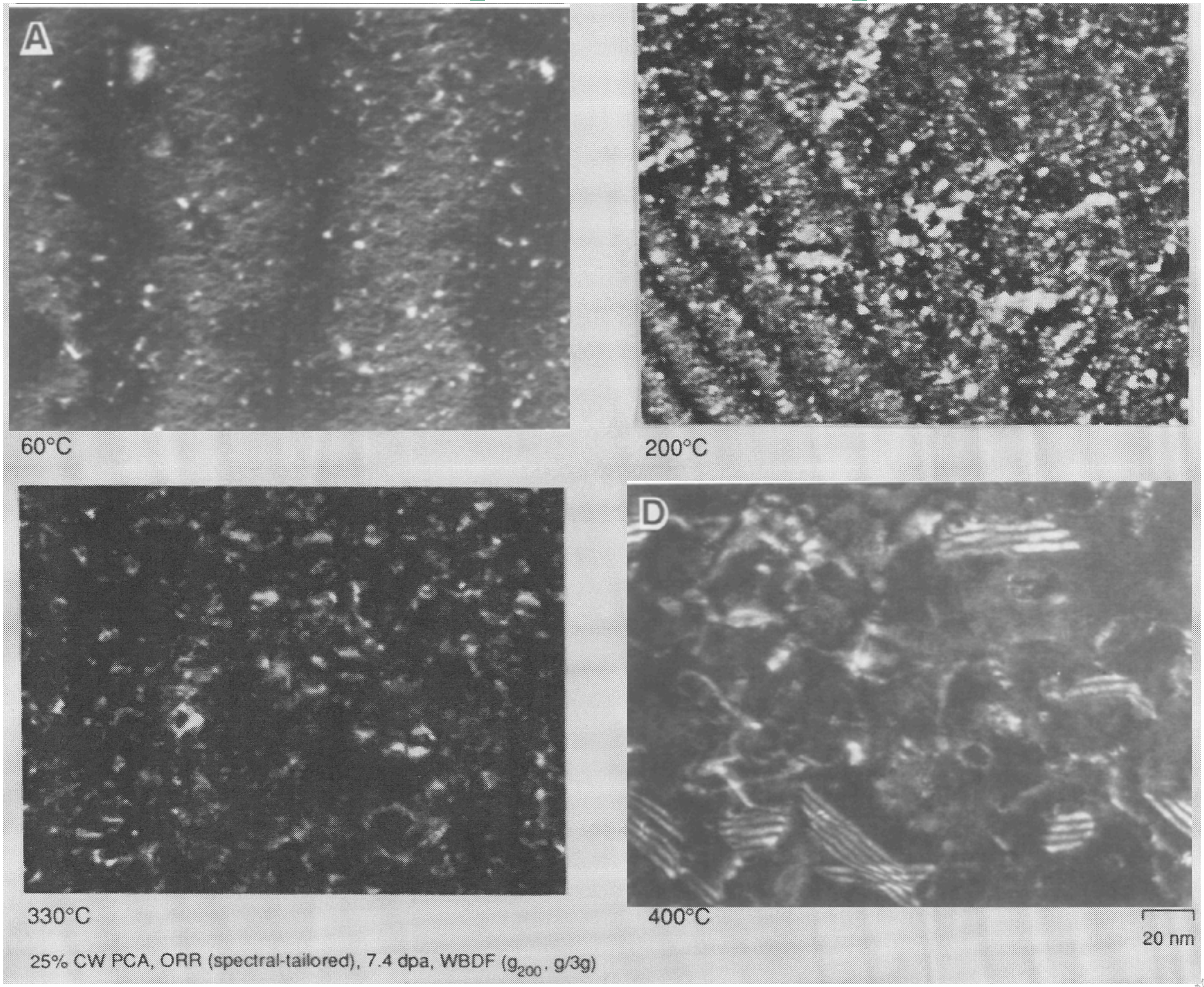
Evolution of defect cluster size in irradiated stainless steel





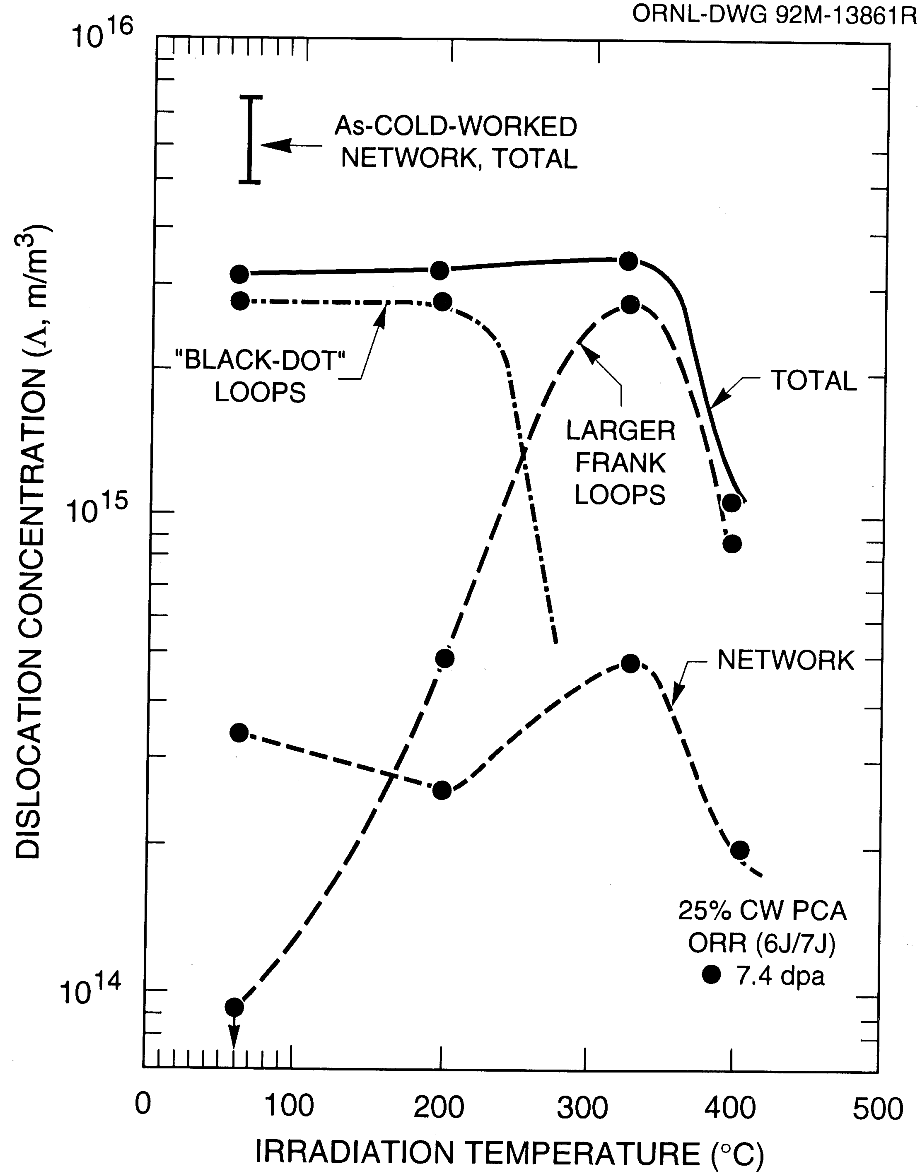
S.J. Zinkle, R.L. Sindelar, J. Nucl. Mater. 155-157(1988) 1196
and unpublished results

Effect of irradiation temperature on loop microstructure



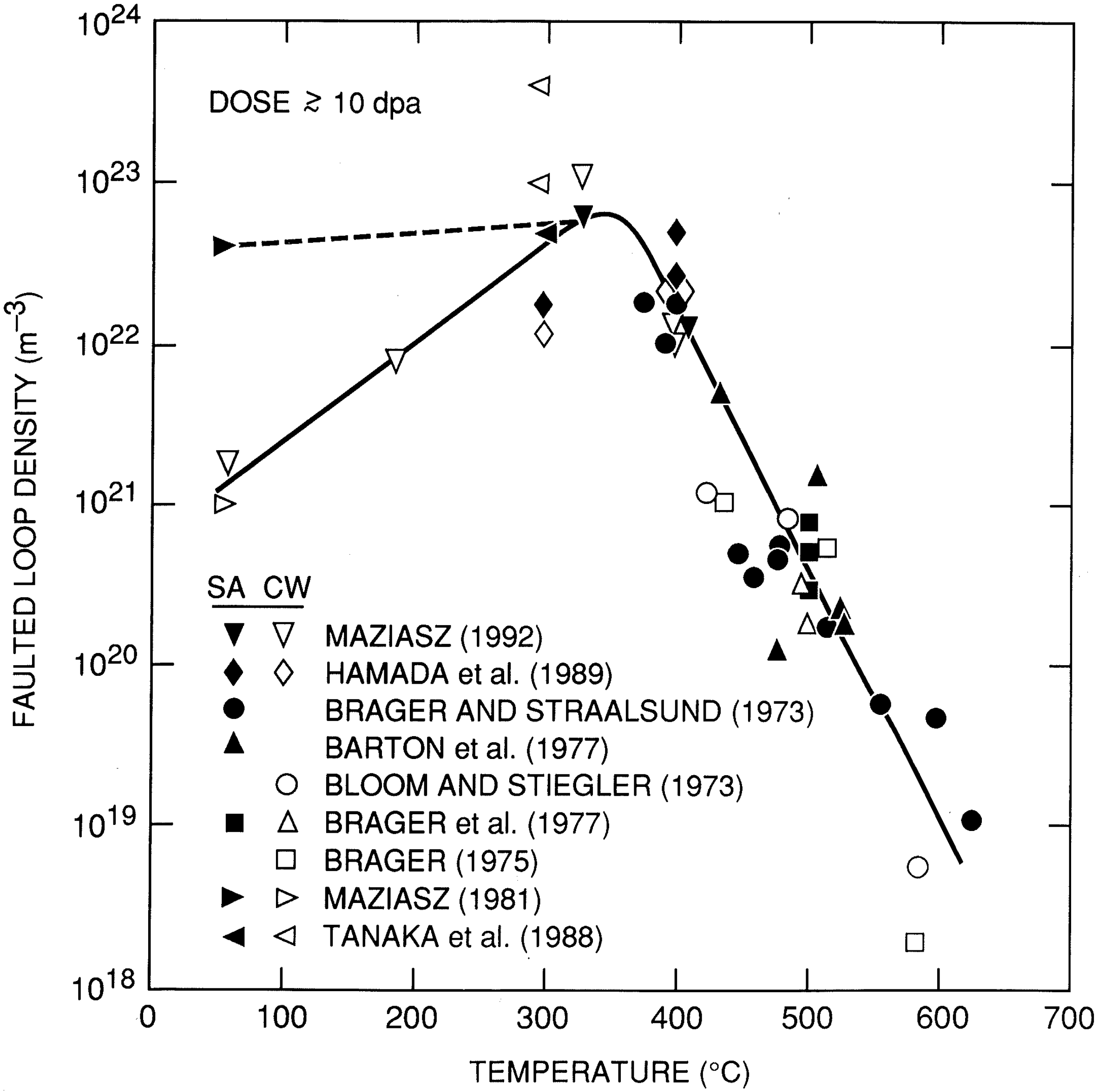
P.J. Maziasz, J. Nucl. Mater. 191-194 (1992) 701

Temperature dependence of loop microstructure in stainless steel



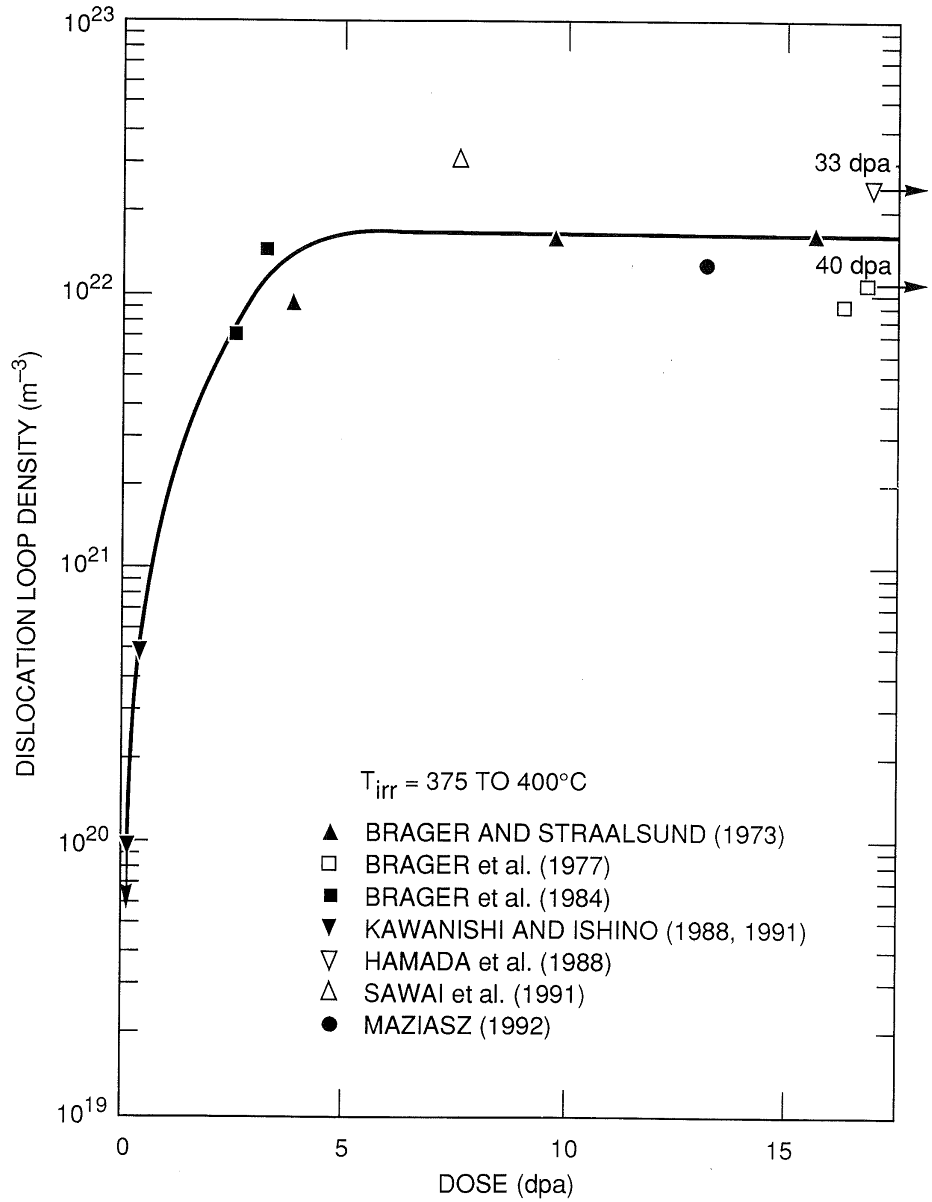
S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

Temperature dependence of faulted loop density in neutron irradiated 316 stainless steel



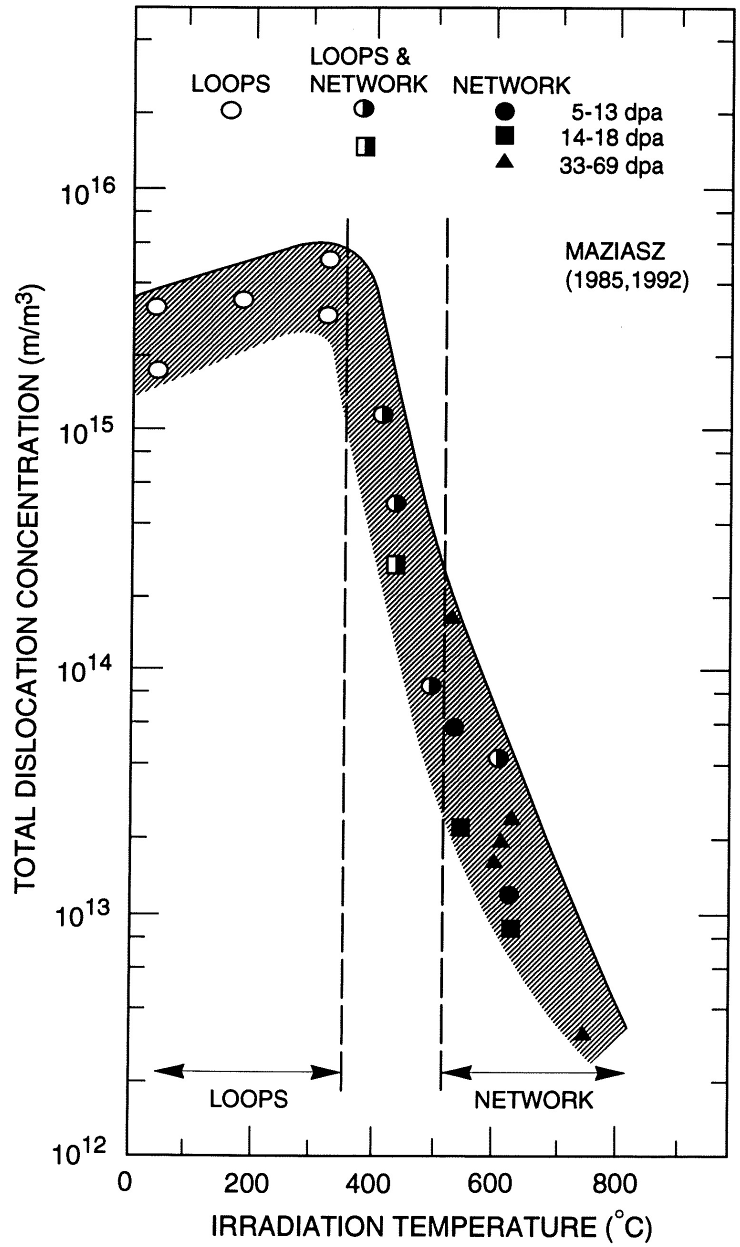
S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

Dislocation loop density due to neutron irradiation in solution annealed 316 stainless steel



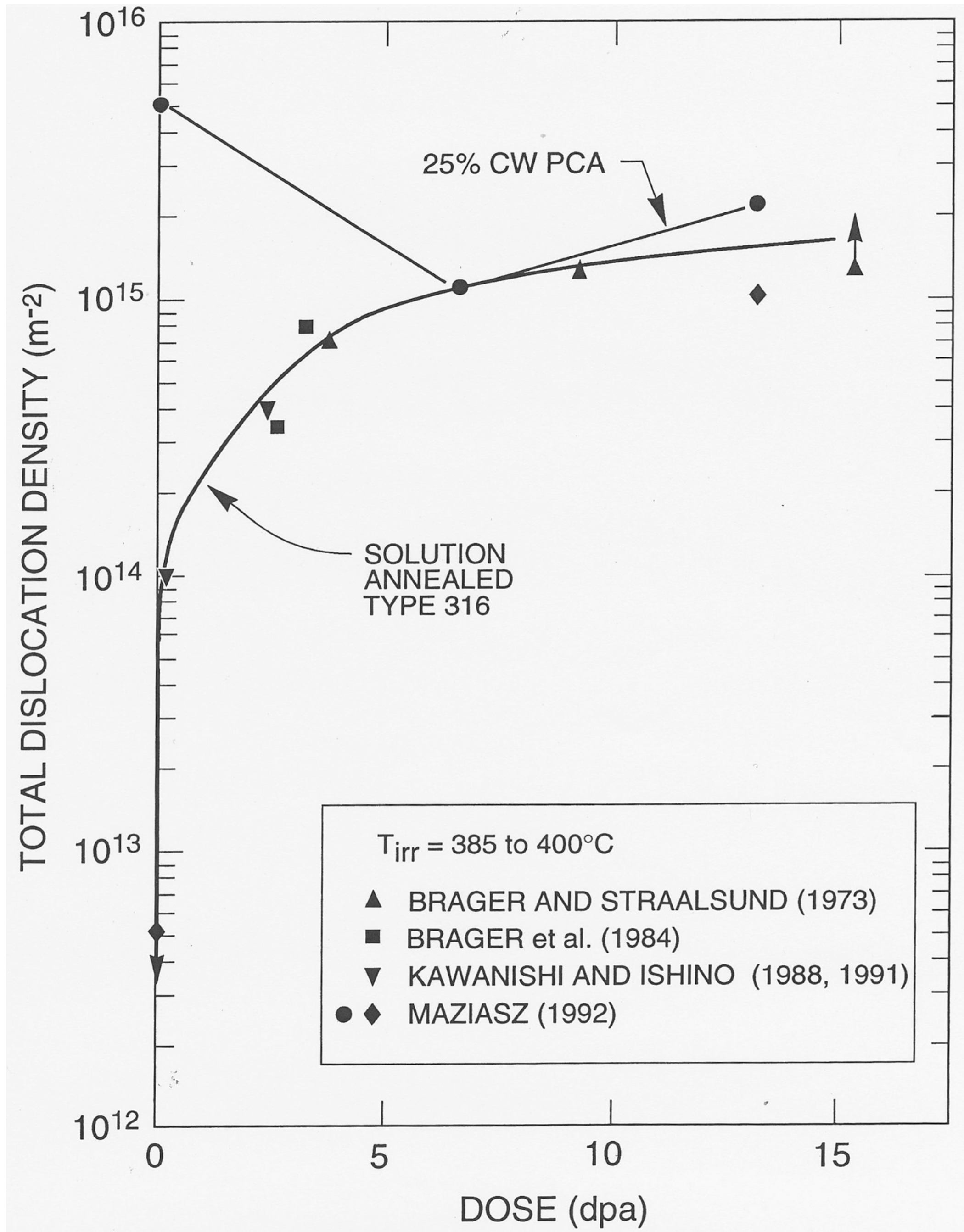
S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

Total dislocation density in neutron irradiated austenitic stainless steel



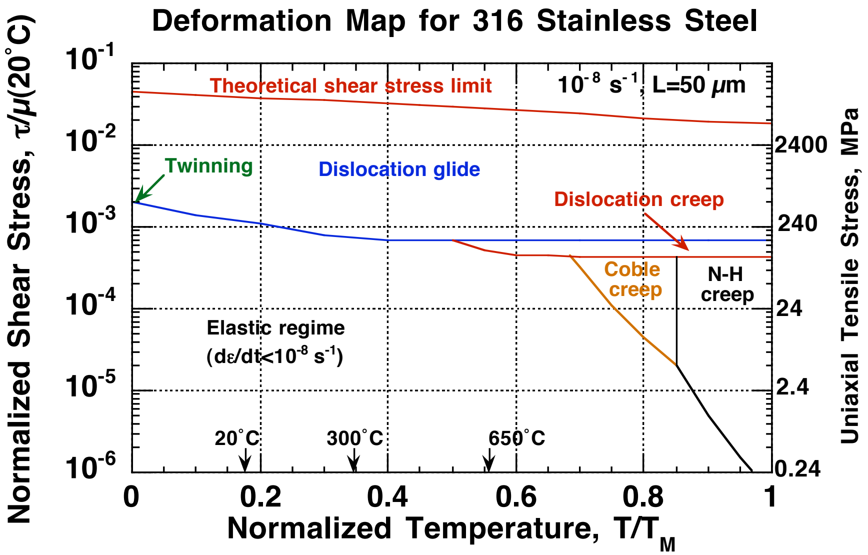
S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

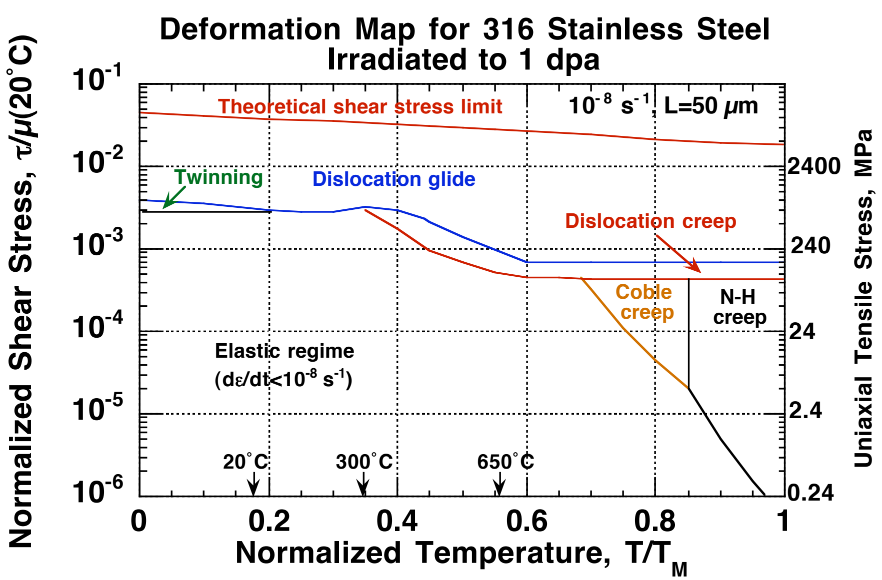
Effect of dose on the total dislocation sink strength of neutron irradiated Type 316 stainless steel



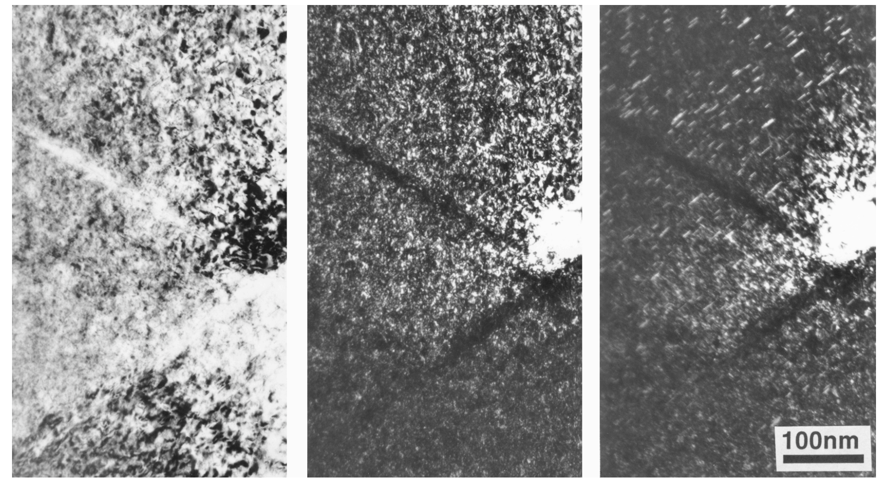
S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

Deformation mechanisms in FCC metals

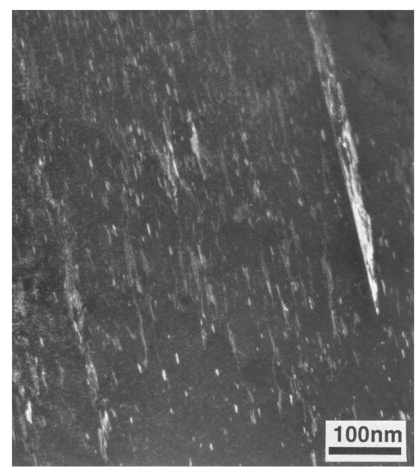




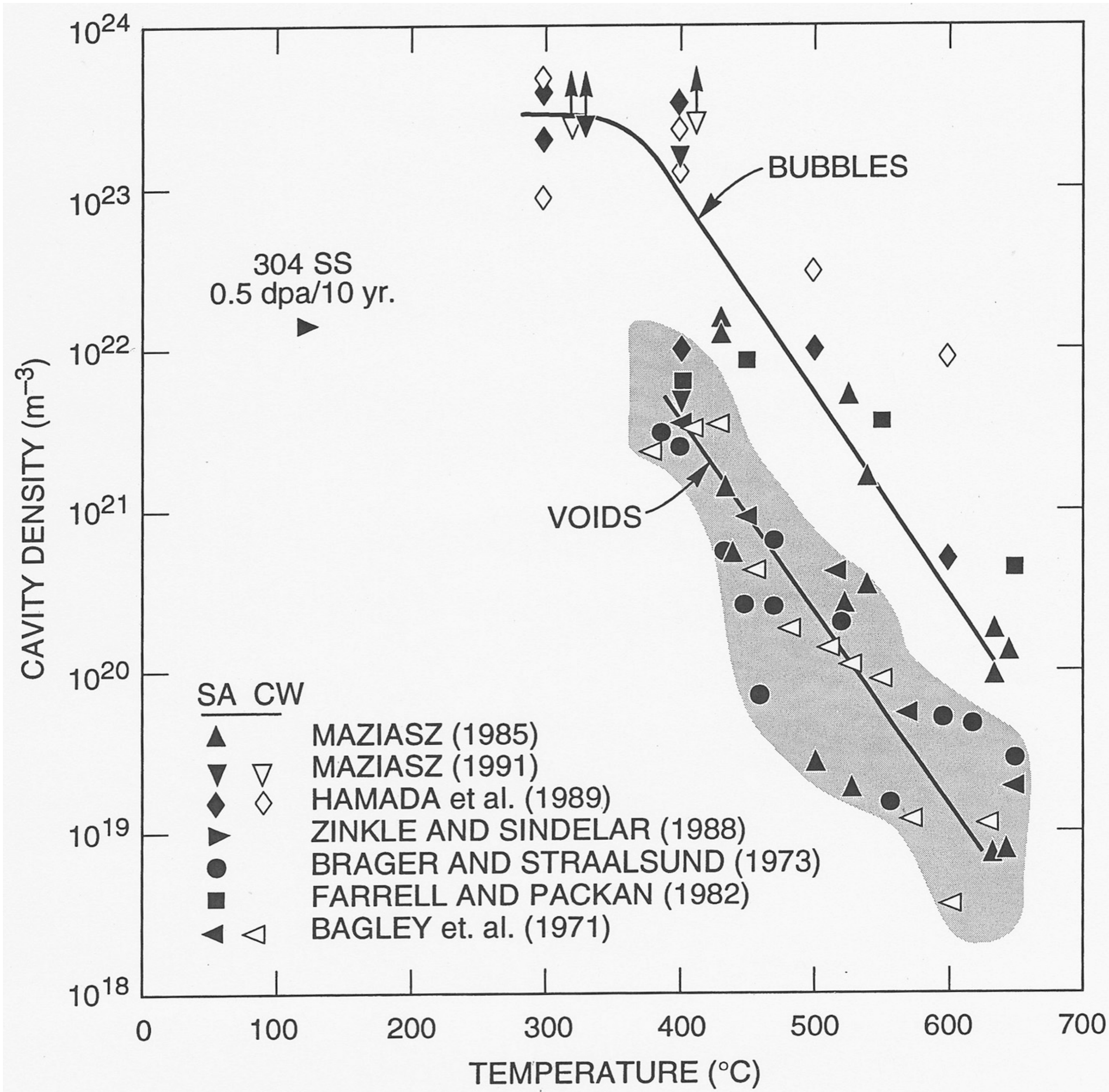
Dislocation channels, twins and martensite are observed in irradiated 316SS after deformation



Channeling (Disln glide) occurs at higher temperatures (~300˚C)

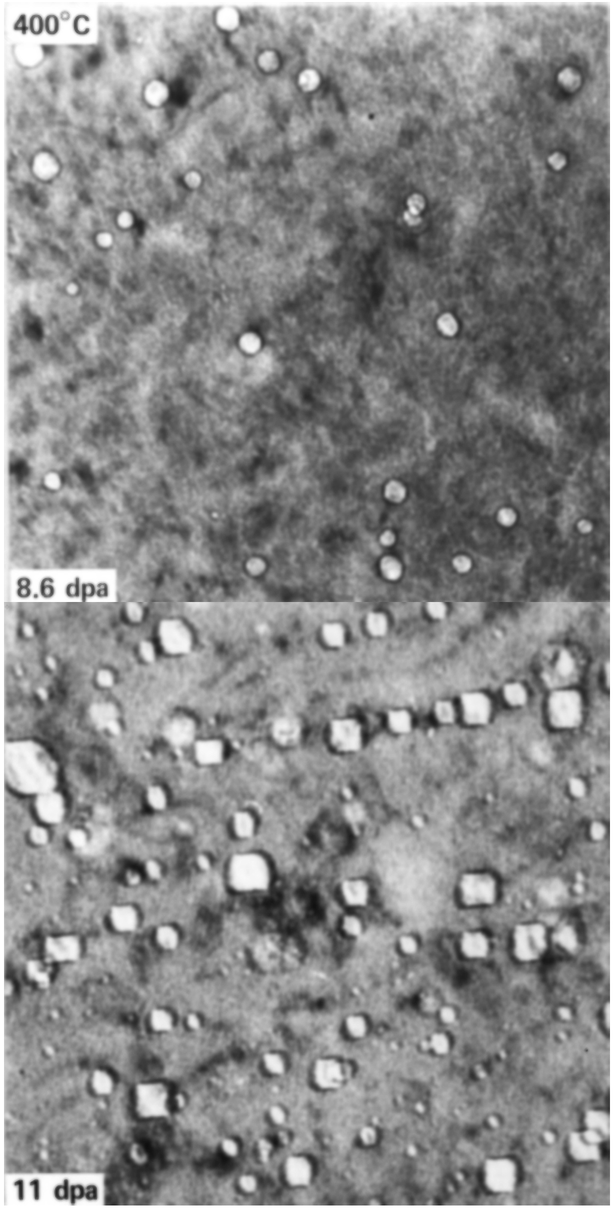


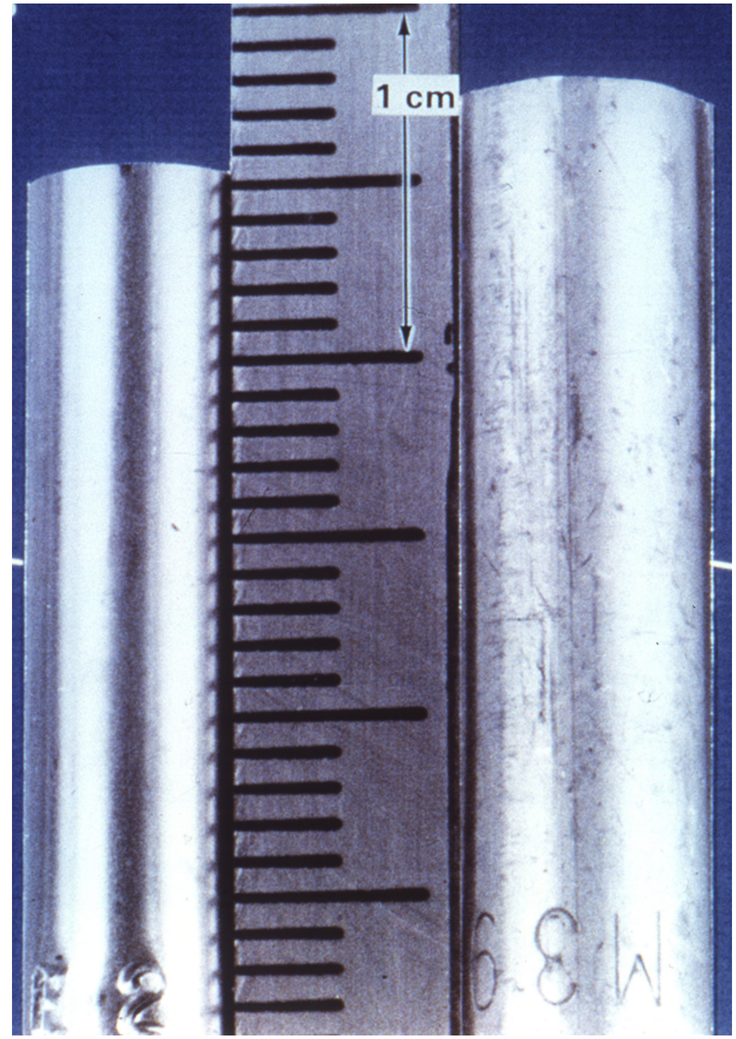
Twinning occurs at lower temperatures (<200˚C) and high strain rates



S.J. Zinkle, P.J. Maziasz, R.E. Stoller, J. Nucl. Mater. 206(1993) 266

Radiation-induced swelling in stainless steel

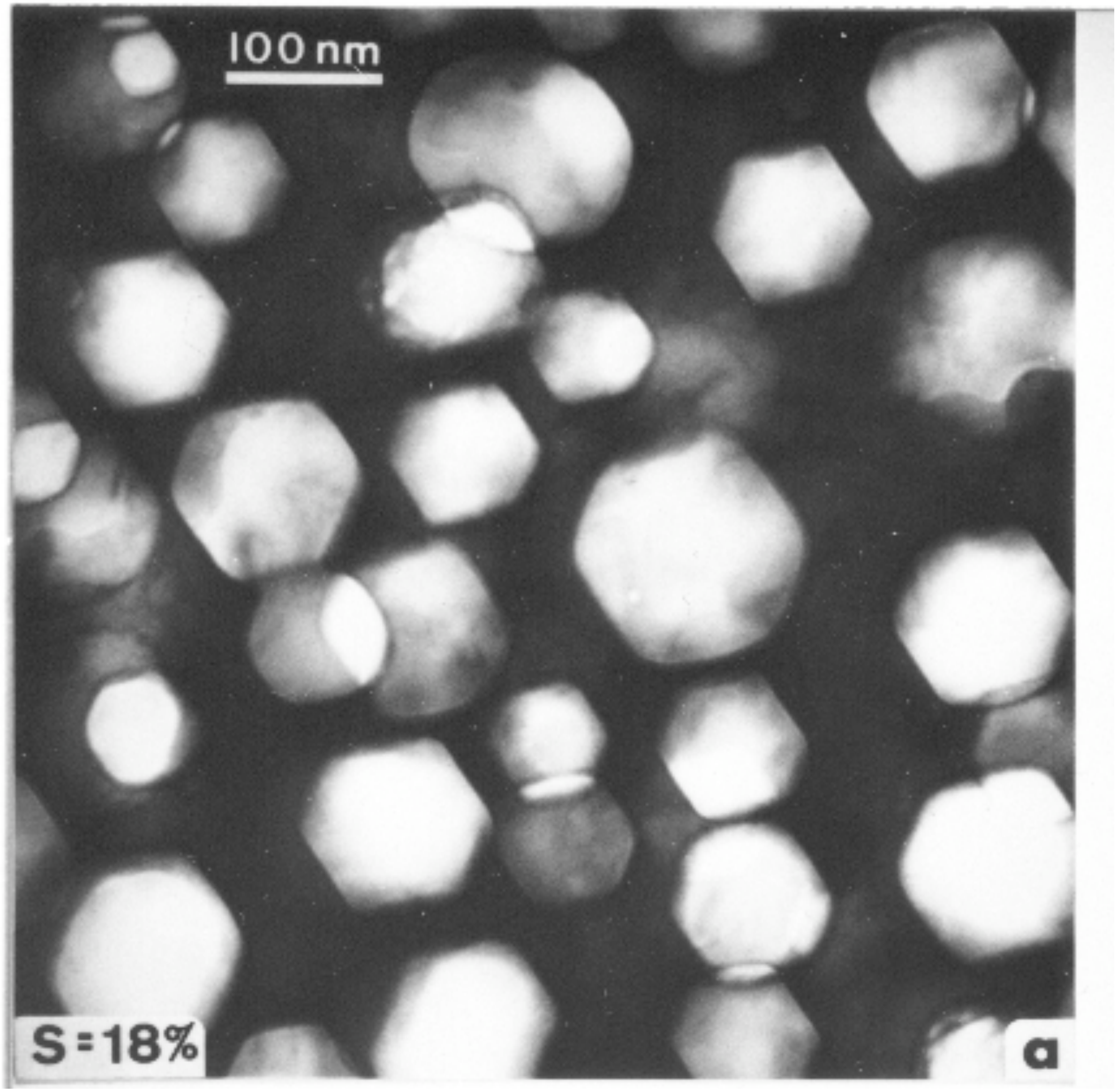




20%CW 316 steel irradiated at $T=523^{0}C$, $1.5x10^{23}n/cm^{2}$

J.L. Straalsund et al., J. Nucl. Mater. <u>108&109</u> (1982) 299

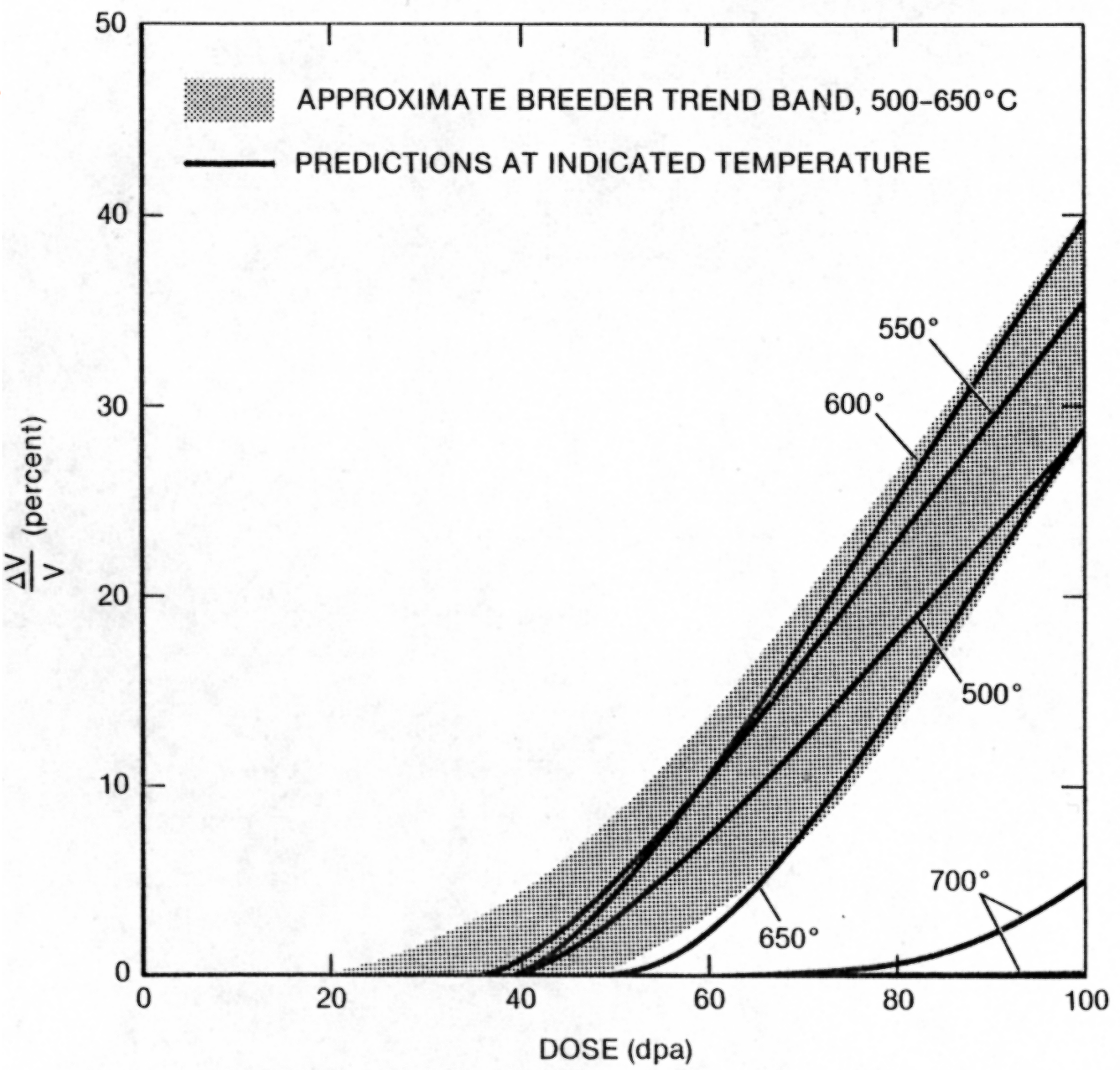
Void formation in ion-irradiated austenitic stainless steel (625˚C, 70 dpa)



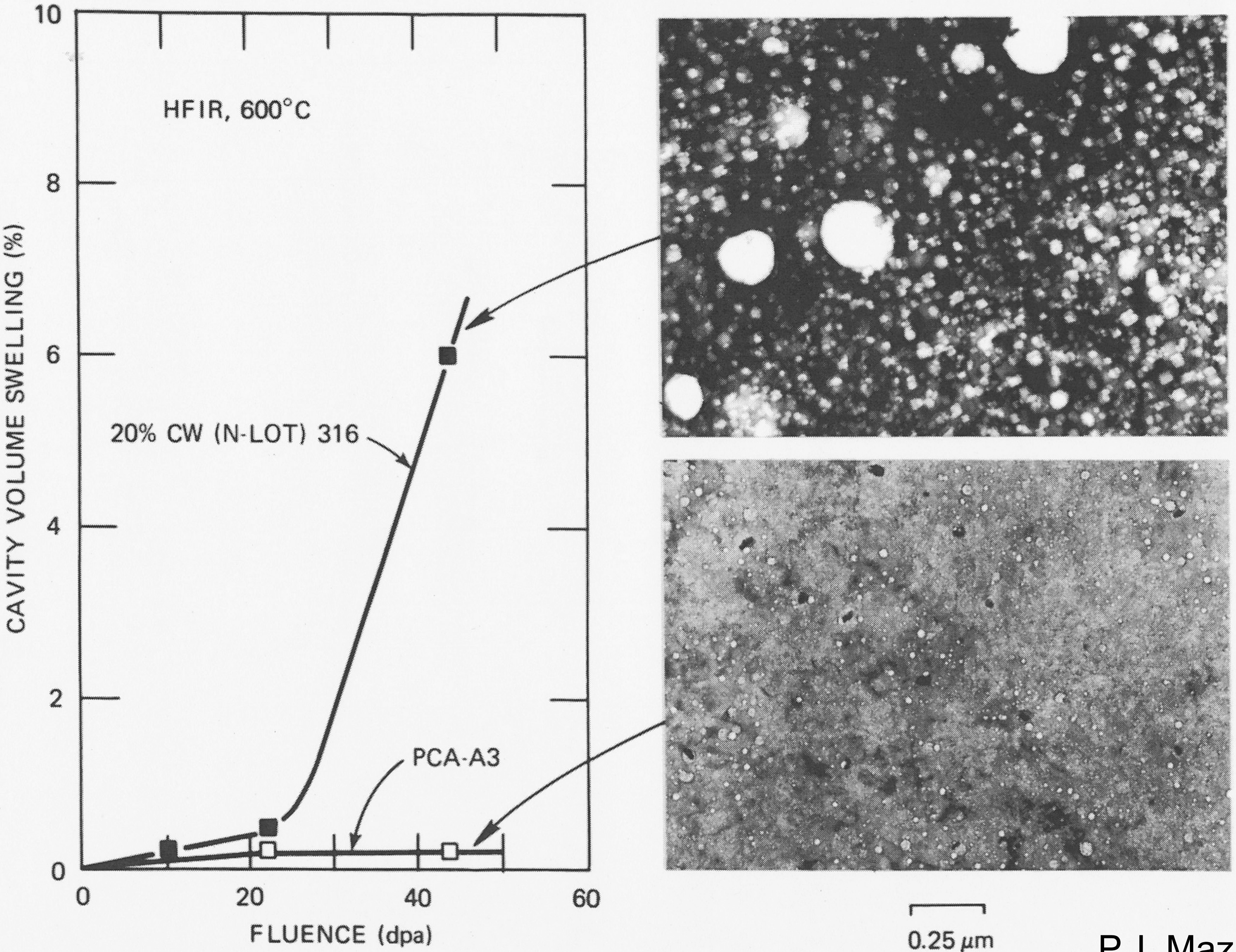
Packan & Farrell (1982)

Swelling of Stainless Steel:

Maximized between 500 and 650^{o}C

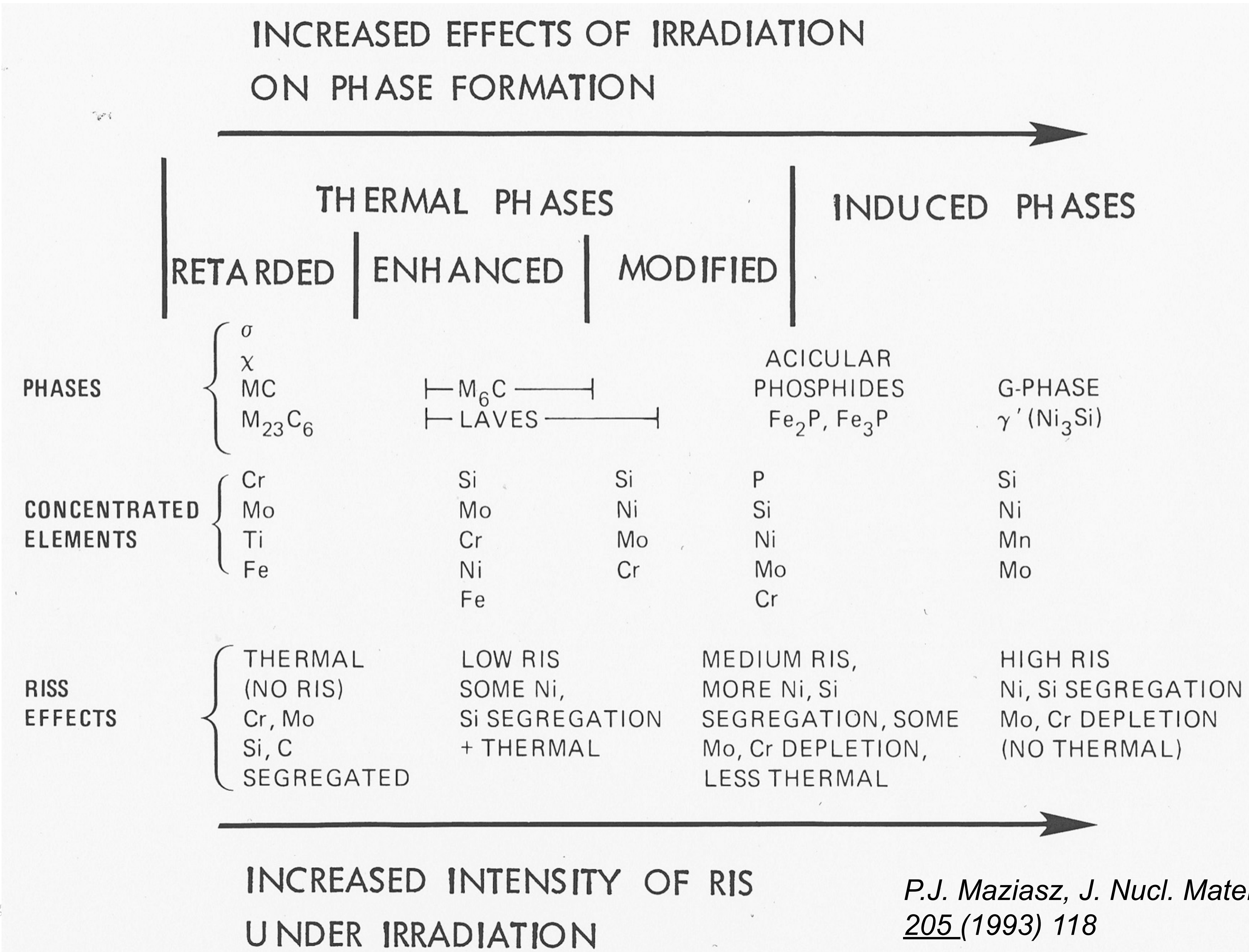


HELIUM TRAPPING AT MC PARTICLES IMPROVES SWELLING RESISTANCE OF AUSTENITIC STAINLESS STEELS IRRADIATED TO 45 dpa AND 2500 appm HELIUM

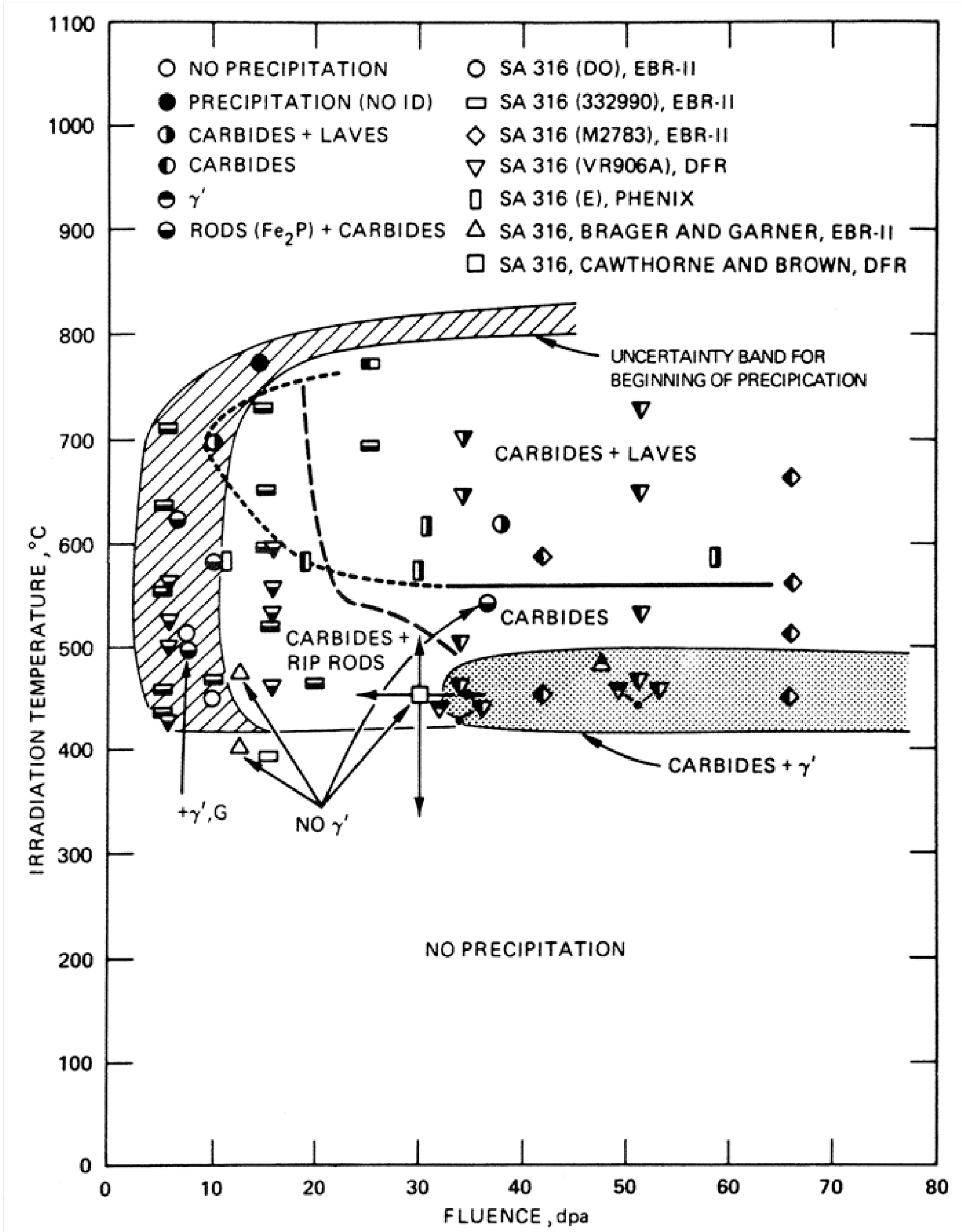


P.J. Maziasz

Summary of precipitation in irradiated stainless steel



P.J. Maziasz, J. Nucl. Mater. <u>205</u> (1993) 118



Summary of precipitate formation in irradiated austenitic stainless steel

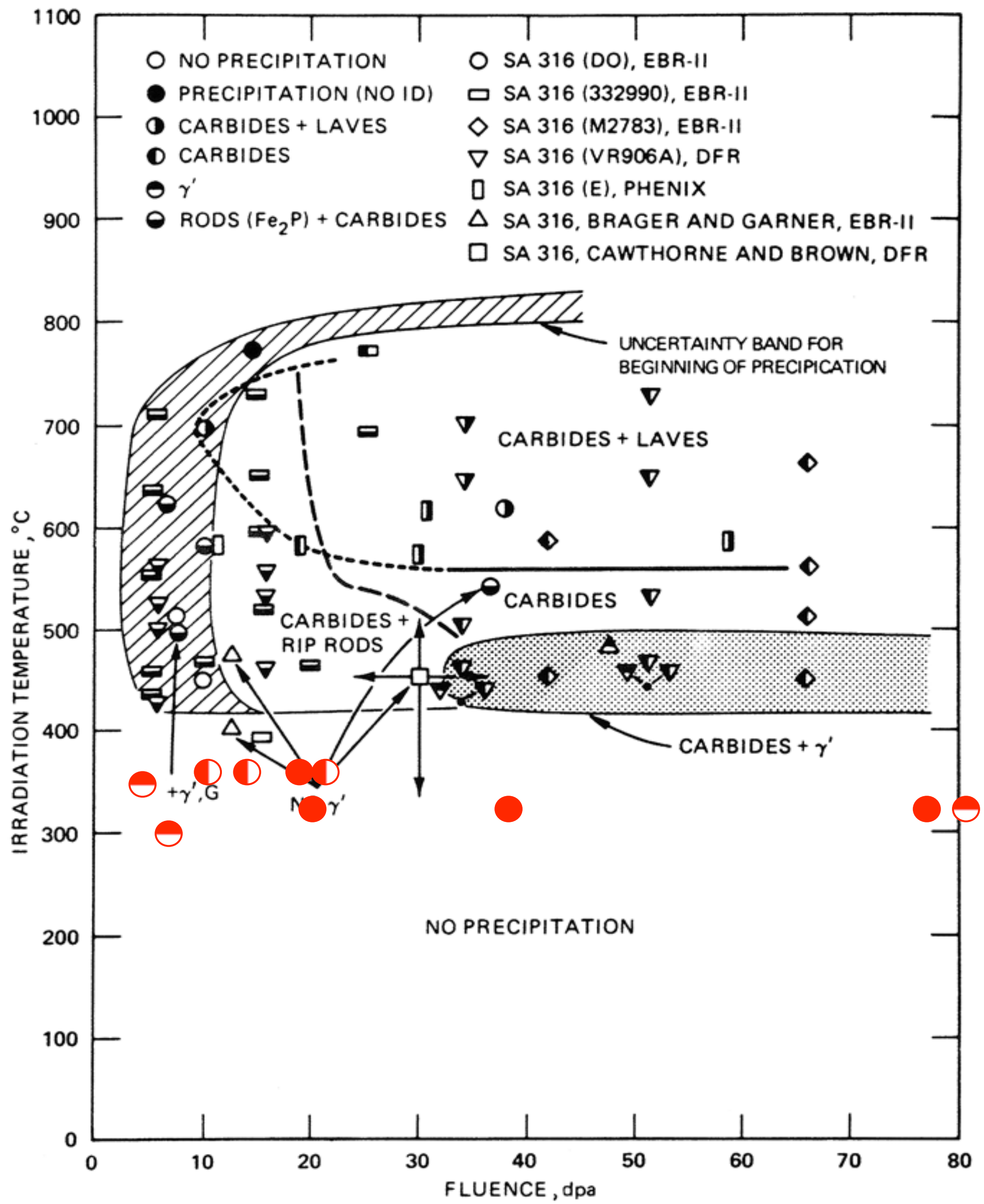
Onset of precipitate formation occurs at low doses (<5 dpa)

Main precipitate formation occurs between 400 and 750˚C

Some recent observations of precipitation at high doses at temperatures below 400˚C

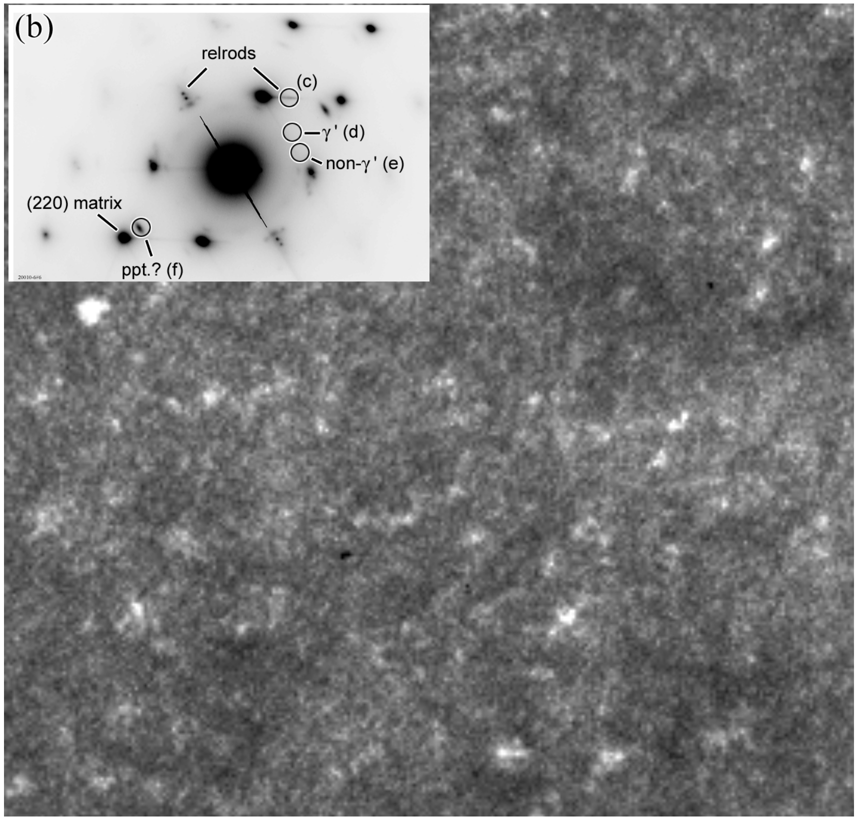
P.J. Maziasz, J. Nucl. Mater. 205 (1993) 118

Precipitate formation in irradiated stainless steel

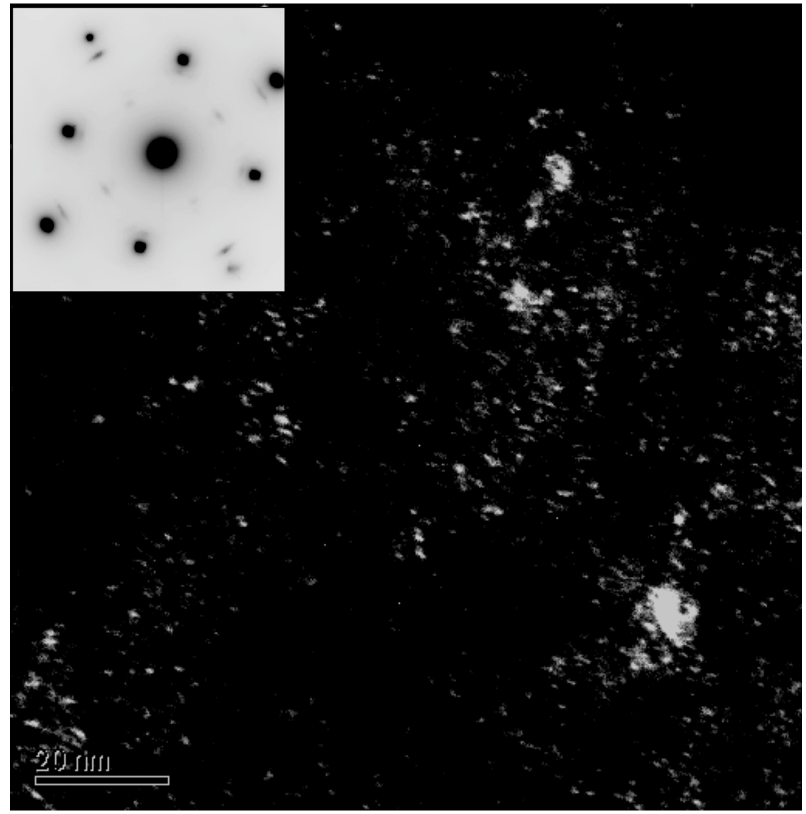


Kenik & Busby, Mat. Sci. Eng. A, submitted

Figure 13



Tihange baffle bolt:
neutron-irradiated to ~7 dpa at 299°C*.



304+Si proton-irradiated to 5.5 dpa at 360°C.

Kenik & Busby, Mat. Sci. Eng. A, submitted

Room Temperature Radiation Hardening in 9Cr FM Steels

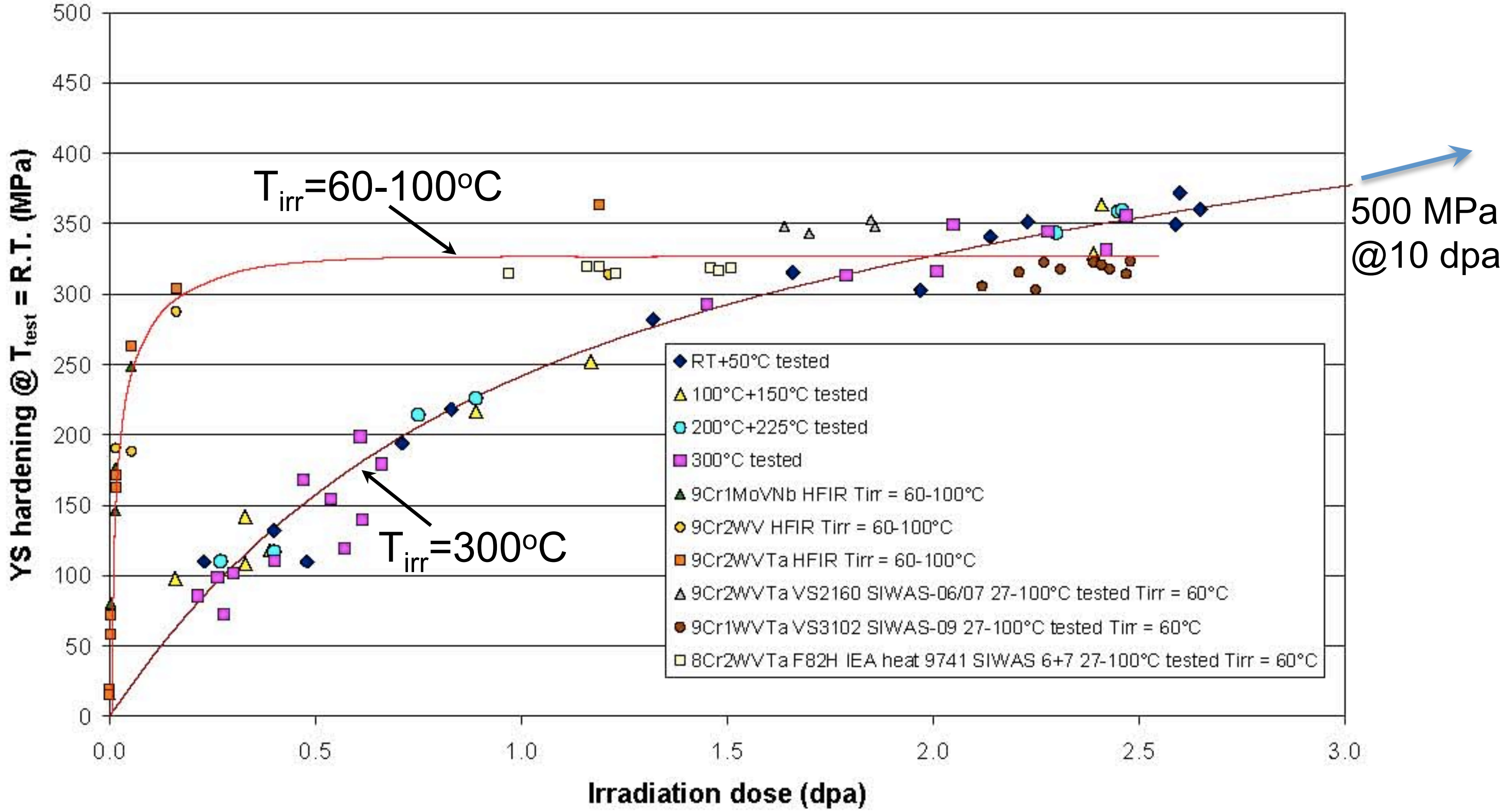
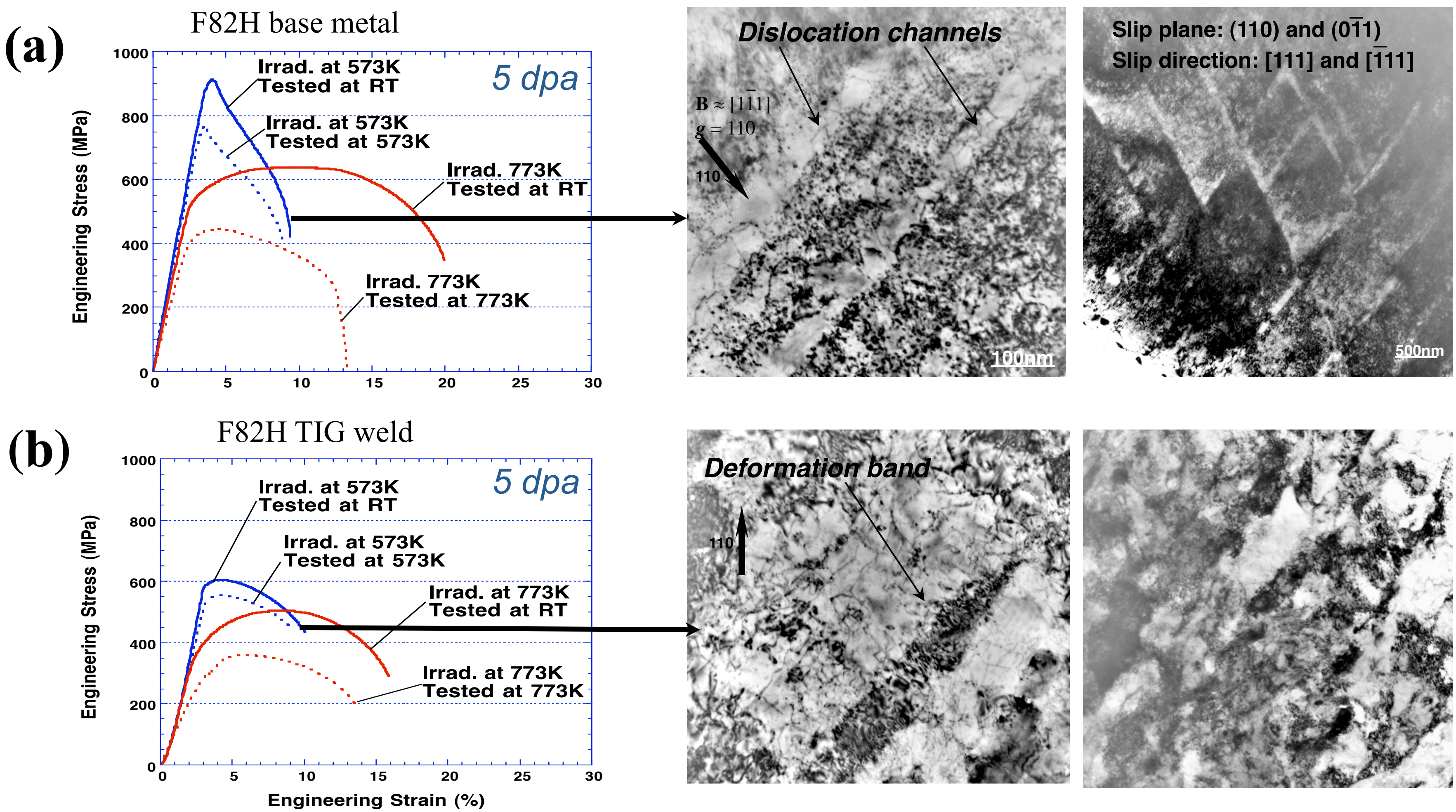


figure 86 Yield stress hardening at 60-80 °C (est.) and 300 °C irradiation temperatures at RT test temperature. NRG data from current report and previous NRG report [4]. HFIR data from [26]

J. Rensman, 2005

Deformation microstructures in neutron-irradiated Fe-8Cr-2WVTa ferritic/martensitic steel (F82H)



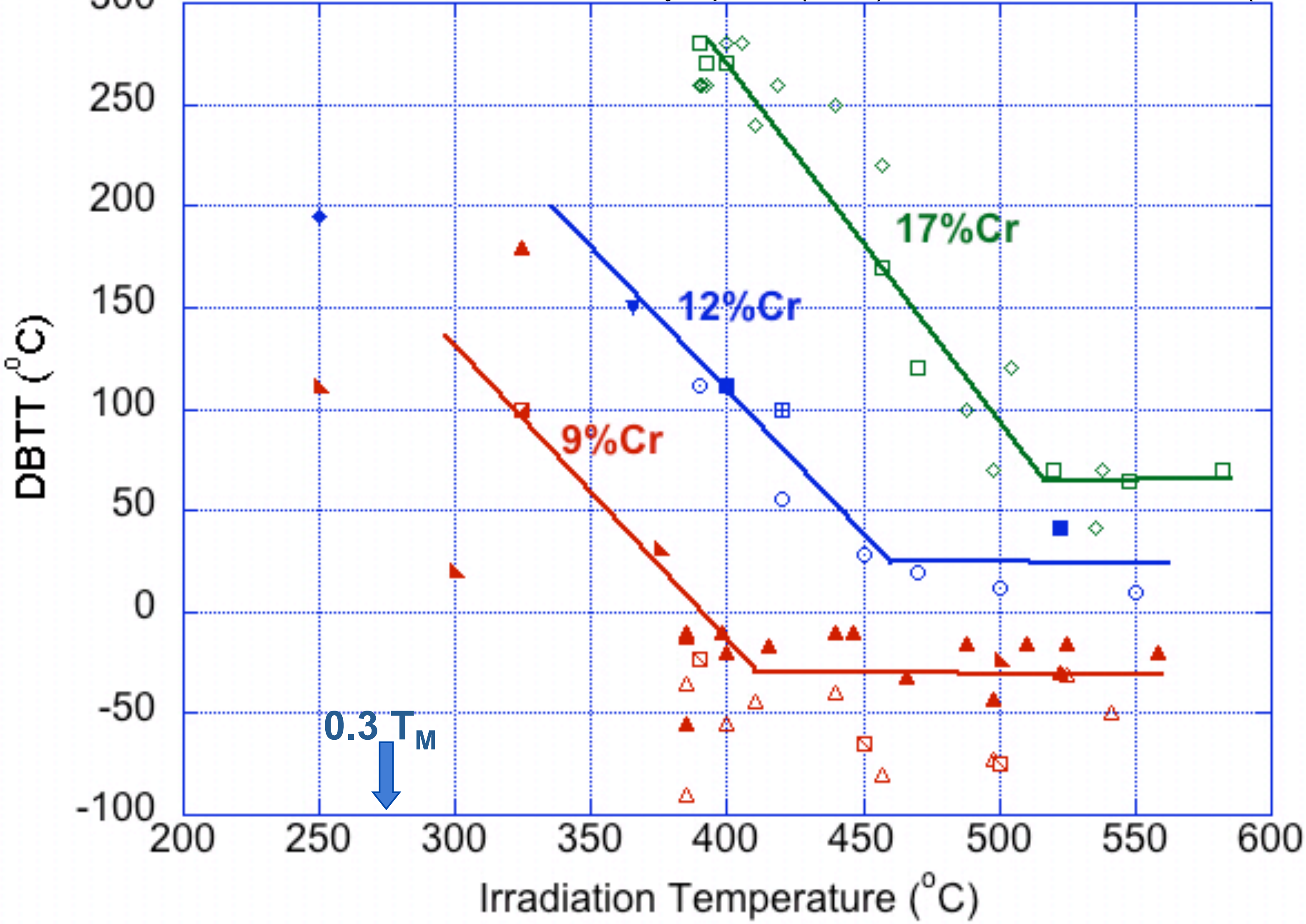
Irradiated weld metal (lower radiation hardening) did not exhibit dislocation channeling after deformation

Fig. 1 Stress-strain curves of F82H BM (a) and TIG (b) irradiated at 573K and 773K in tests at RT

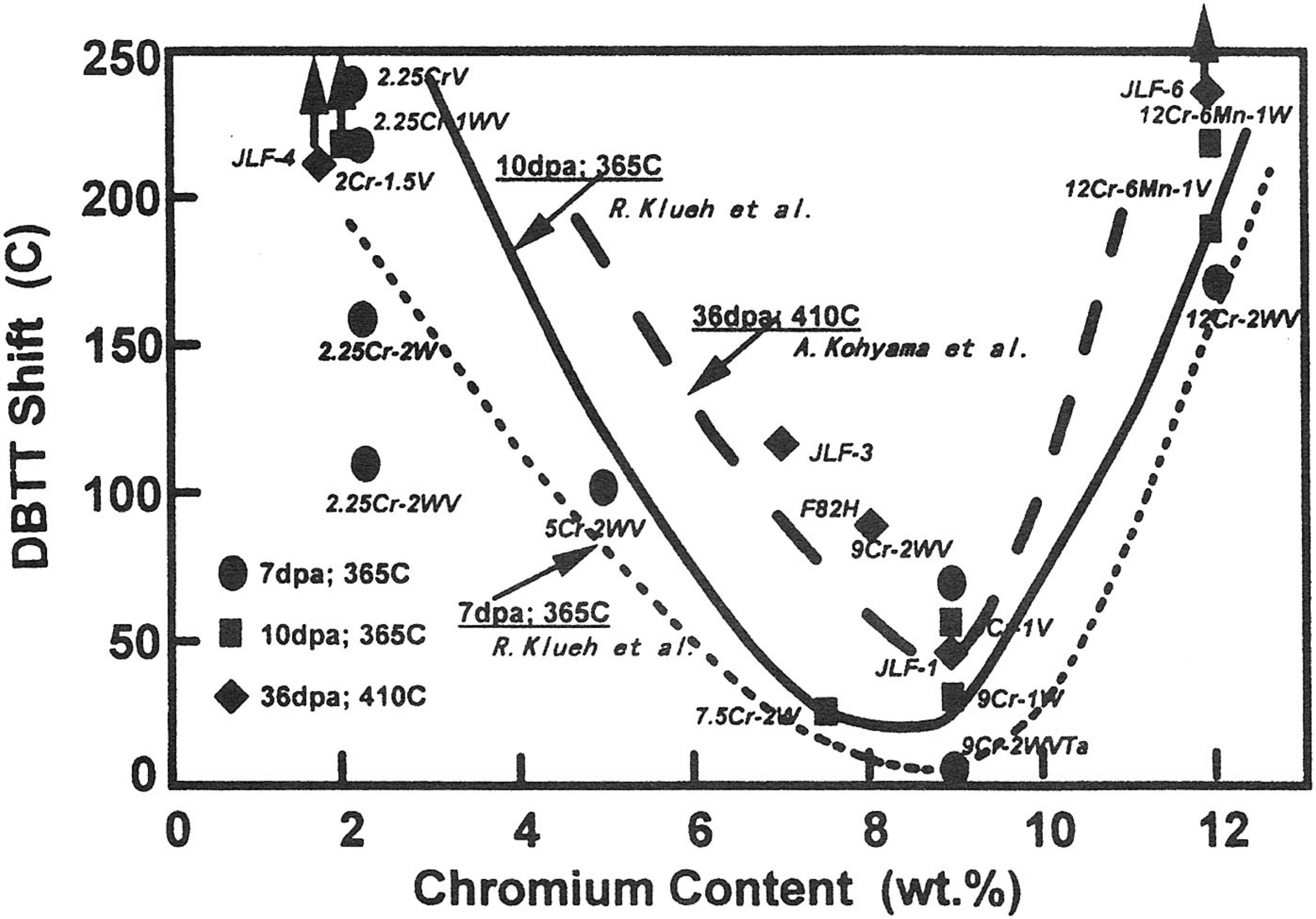
N. Hashimoto et al., Fus.Sci.Tech. 44 (2003)

Effect of Neutron Irradiation on the Ductile to Brittle Transition Temperature in Ferritic/martensitic Steels

after Boutard et al., C.R. Physique. 9 (2008) 287 and Klueh & Harries (2001)

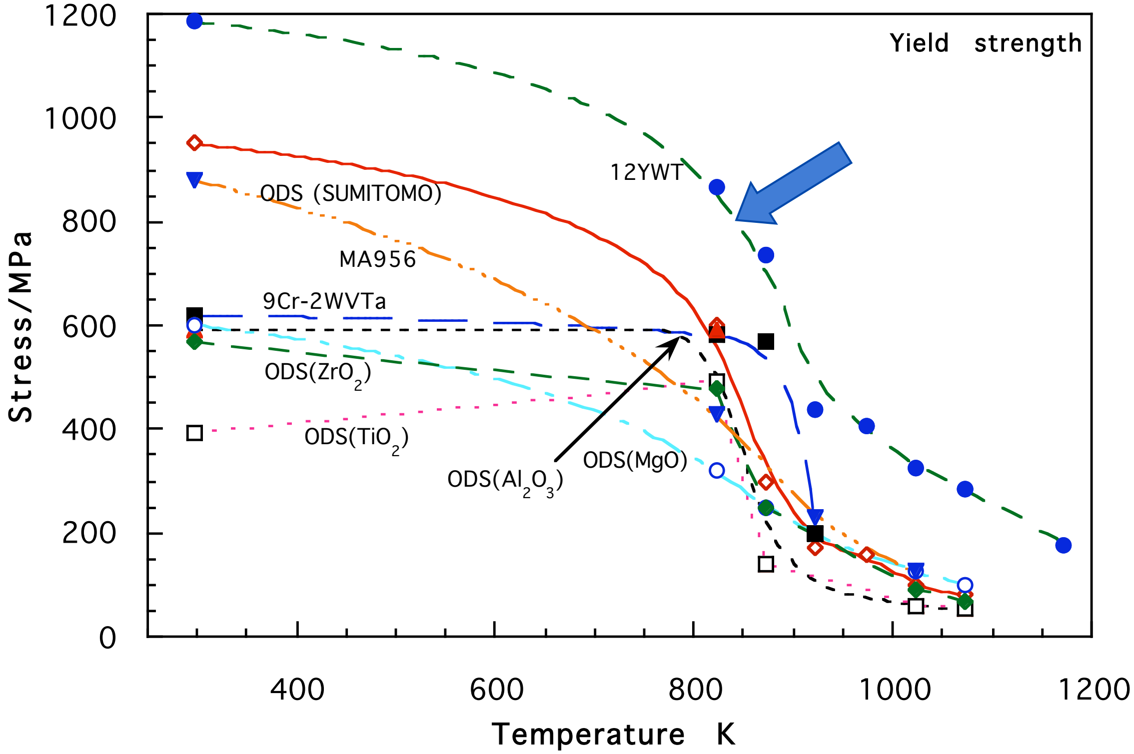


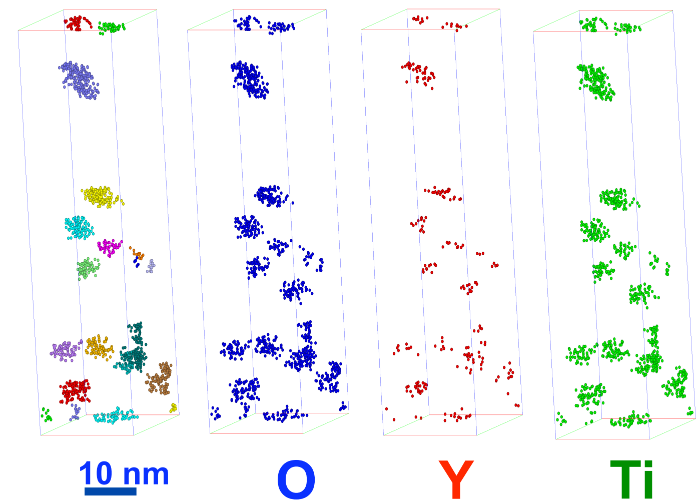
Effect of Cr Content on the Ductile-Brittle Transition Temperature of Irradiated Ferritic/martensitic Steels



Kohyama et al. JNM, 233-237 (1996) 138

New 12YWT Nanocomposited Ferritic Steel has Superior Strength compared to conventional ODS steels



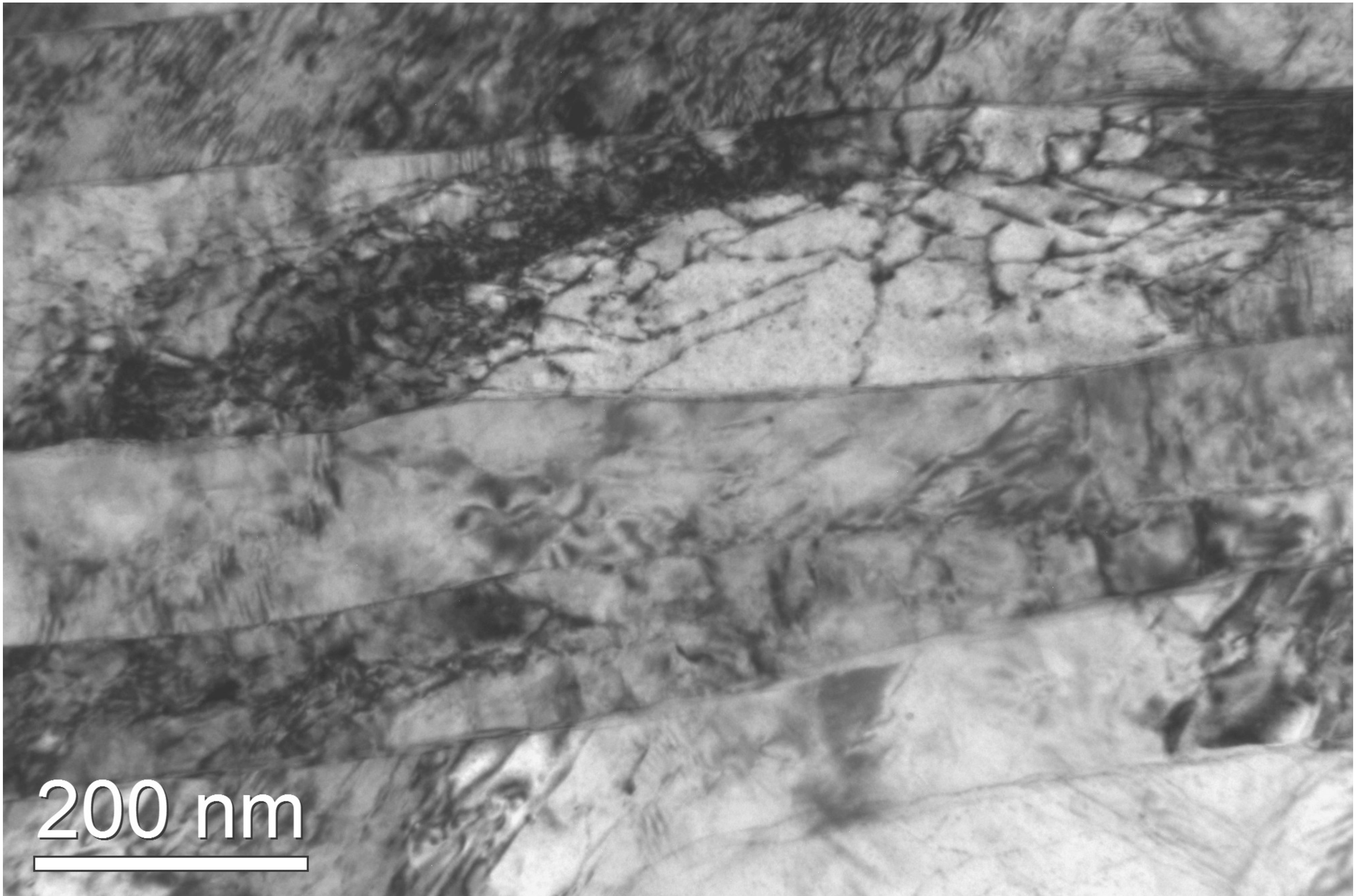


- Atom Probe reveals nanoscale clusters to be source of superior strength
  - Enriched in O(24 at%), Ti(20%), Y (9%)
  - Size : $r_g = 2.0 \pm 0.8$ nm
  - Number Density : $n_v = 1.4 \times 10^{24}/m^3$
- Original Y_2O_3 particles convert to thermally stable nanoscale (Ti,Y,Cr,O) particles during processing
- Nanoclusters not present in ODS Fe-13Cr + $0.25Y_2O_3$ alloy

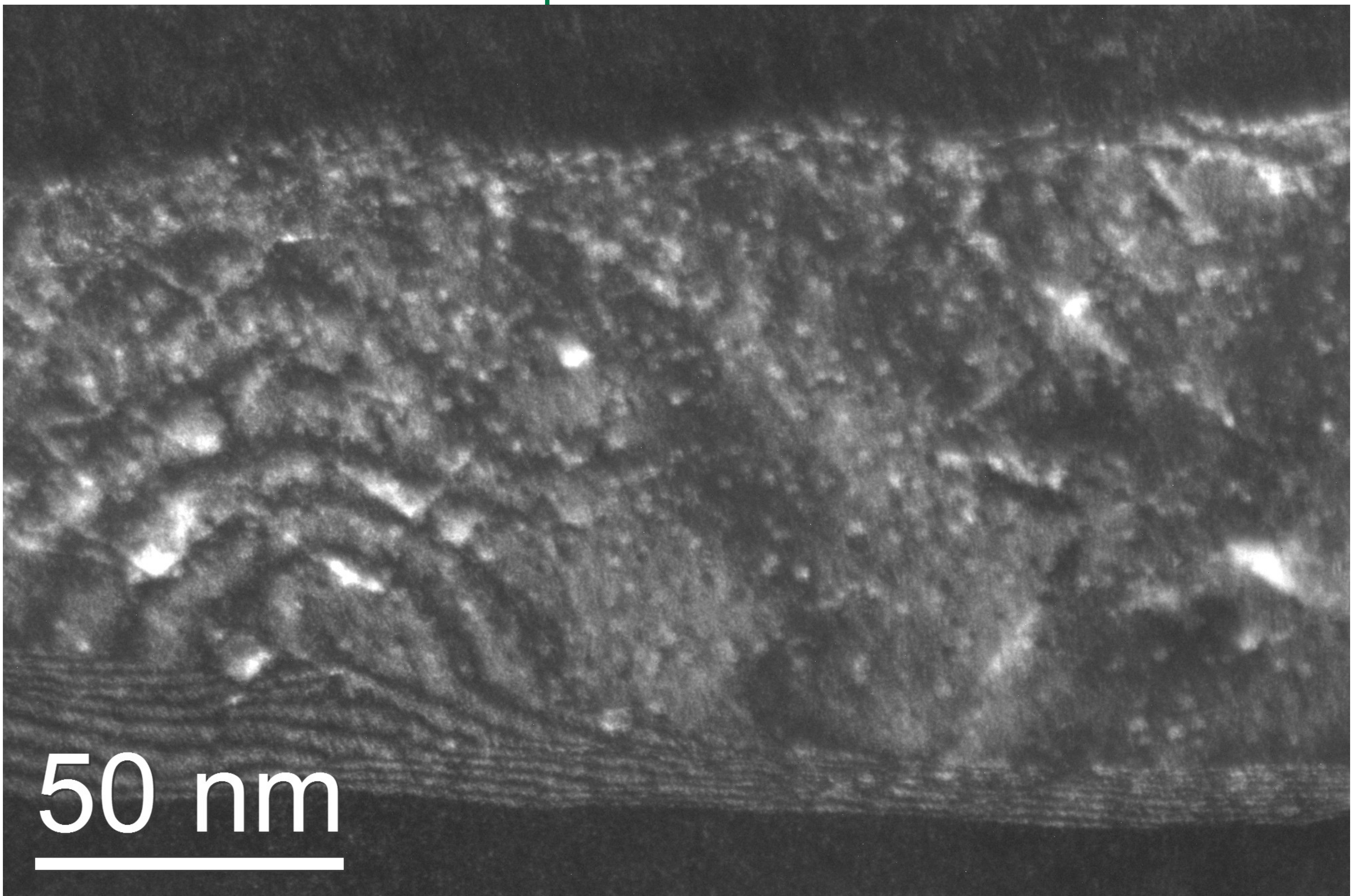
- **Thermal creep time to failure is increased by several orders of magnitude at 800˚C compared to ferritic/martensitic steels**
  - **2% deformation after ~2 years at 800ºC, 140MPa**
- **Potential for increasing the upper operating temperature of iron based alloys by ~200°C**
- **Acceptable fracture toughness near room temperature**

D.T. Hoelzer, M.K. Miller, et al.

General microstructure of ODS ferritic steel (12YWT)



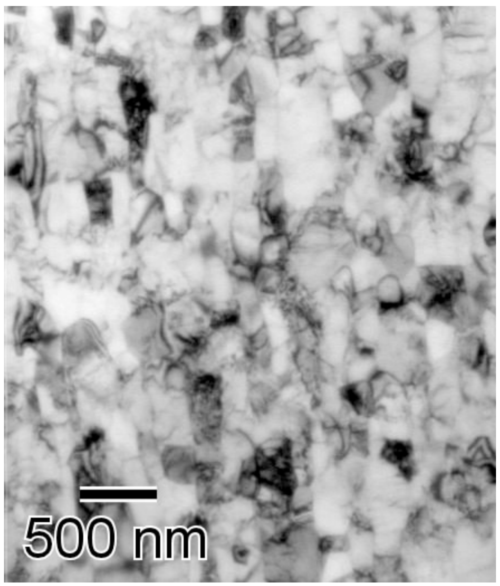
Microstructure of ODS ferritic steel (12YWT) after neutron irradiation at 300C to 9 dpa



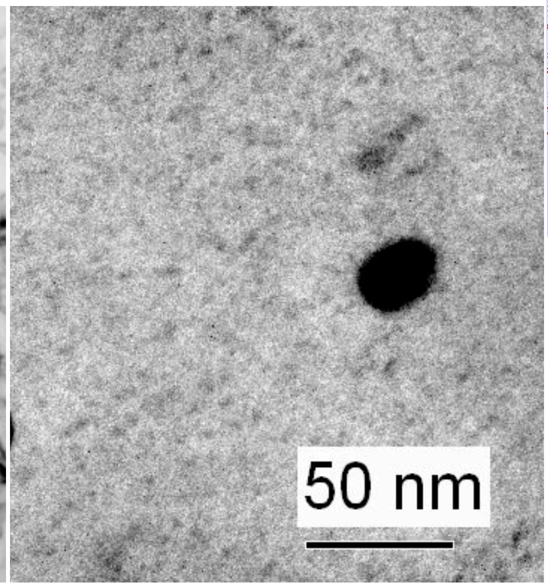
Nanostructuring Achieves Good Fracture Toughness and High-Strength Properties

NFA 14YWT Developed at ORNL

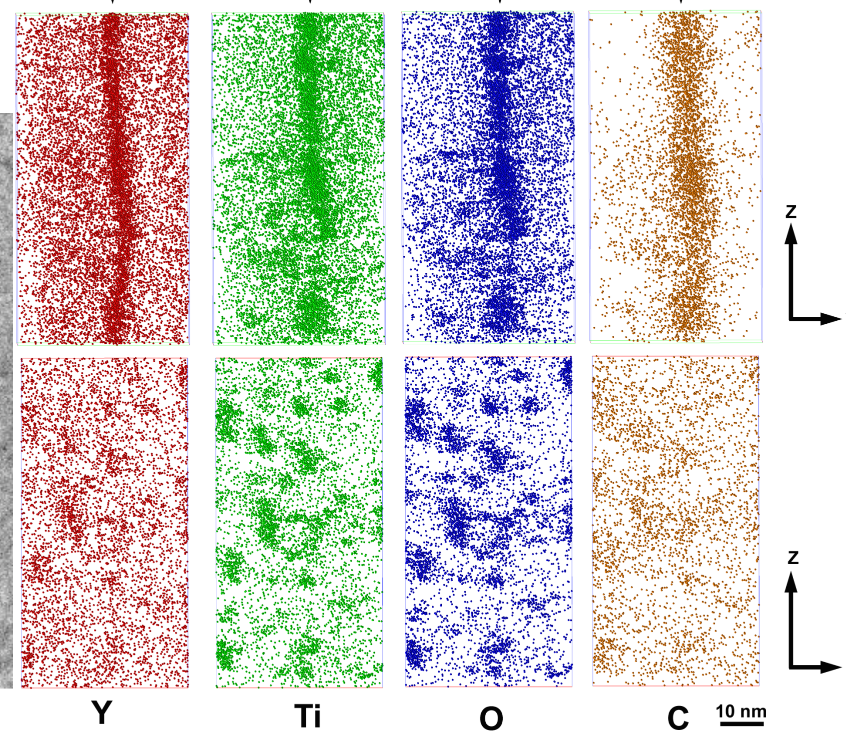
BF TEM



Fe M Jump Ratio



Grain boundary nucleation of NC

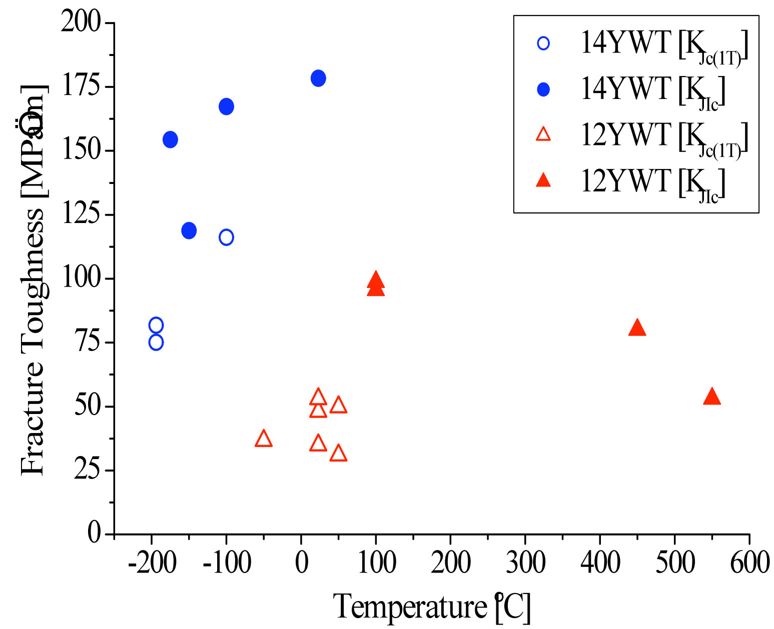


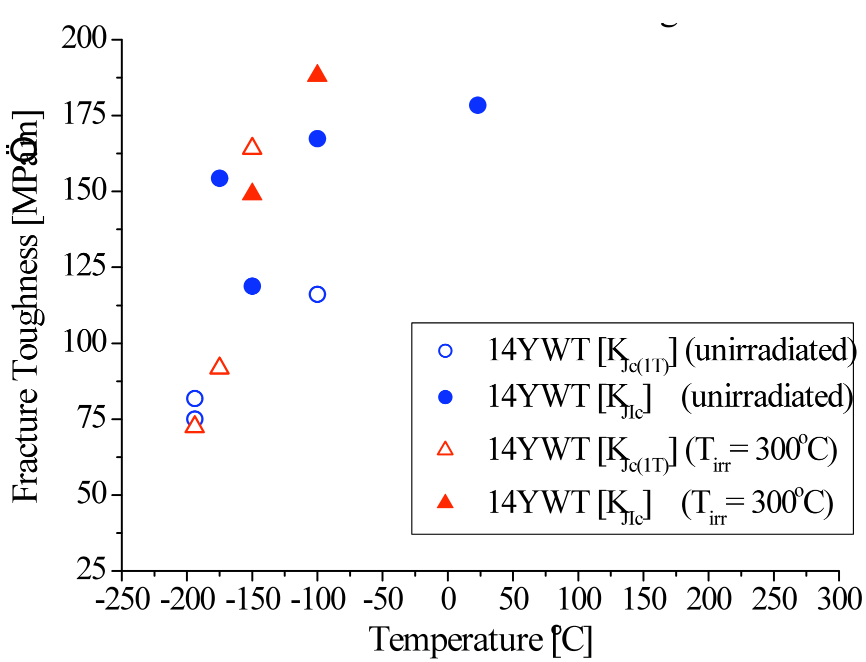
- Nano-size grain size with very high grain boundary interfacial area
- High number density of NC in-matrix with λ = 10-15 nm
- High number density of NC decorating grain boundaries

D.T. Hoelzer, ICFRM-13

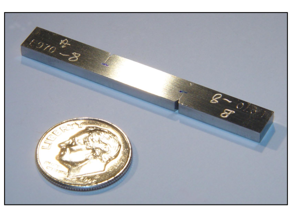
High Fracture Toughness Achieved in 14YWT

- The fracture toughness of 14YWT is much better than that of 12YWT
- The DBTT is shifted from ~75 ºC for 12YWT to -150 ºC for 14YWT





- Neutron irradiation to 1.5 dpa at 300ºC appears to slightly improve the fracture toughness above -150ºC
- More testing is required though…



- L-T Orientation
- Pre-cracked: crack length to width (a/w) ratio of 0.5
- Tested using the unloading compliance method (ASTM 1820-06)
- K_{Jc} for brittle cleavage calculated from critical J-integral at fracture, adjusted to 1-T reference specimen $K_{Jc(1T)}$
- K_{JIc} for ductile deformation behavior calculated from critical J-integral at onset of stable crack growth

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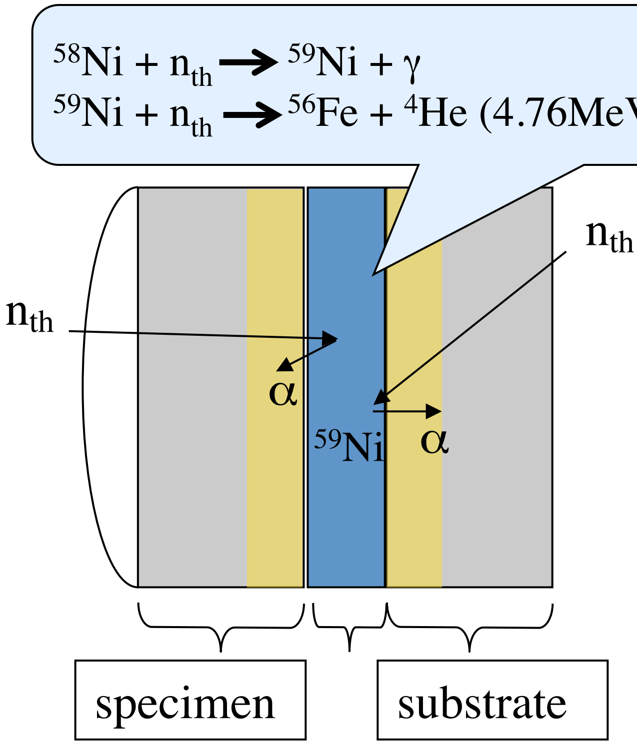
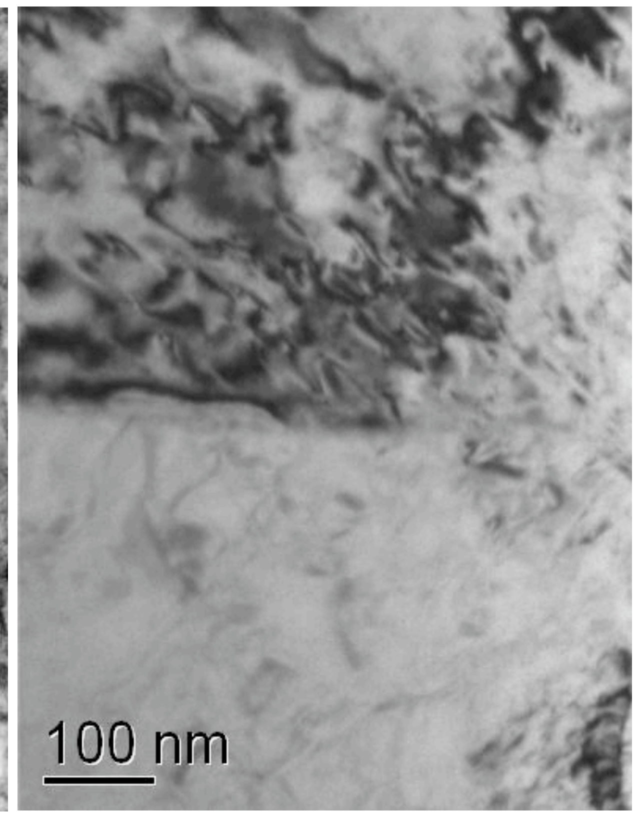
Proof of Concept: Results of In-situ Neutron + He Implantation of NFA MA957



- Simultaneous neutron and He implantation
- HFIR: 9 dpa and up to 380 ppm He at 500ºC

BF TEM

BF TEM



- *No visible defect damage*

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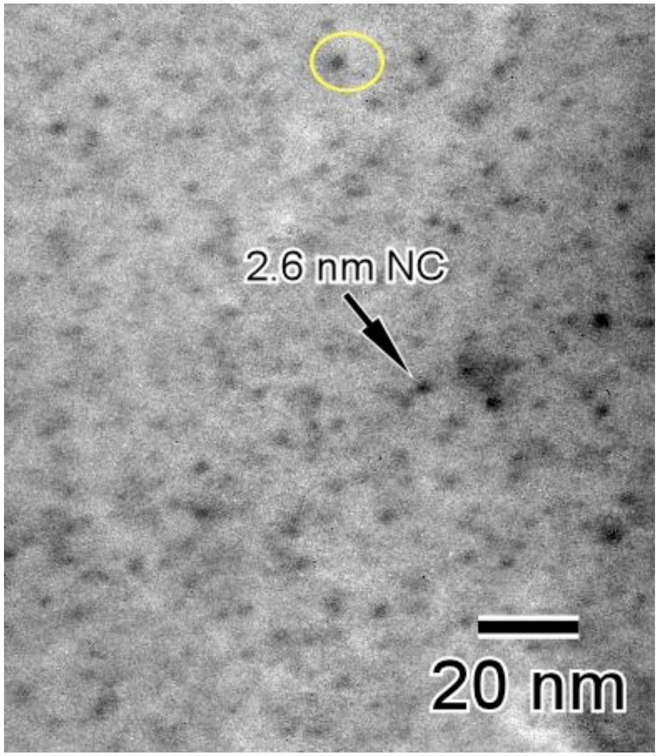
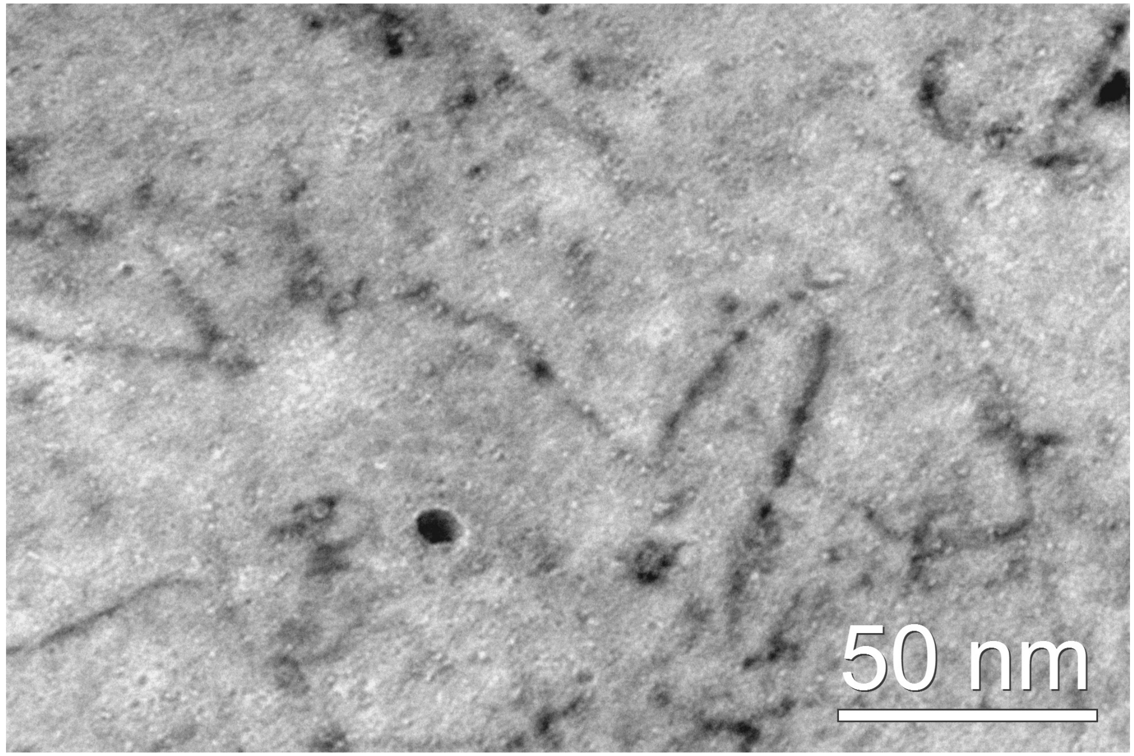
Survival of NC and He Trapping in irradiated MA957

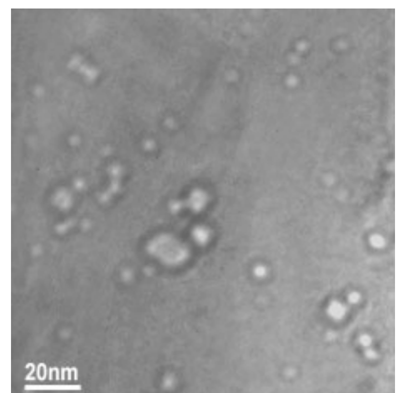
EFTEM Fe M Jump Ratio



-512 nm Under Focus



- Compared to Eurofer 97, the results indicate that NC play an important role in mitigating the accumulation of point defects as well as trapping He into small ~1-2 nm bubbles



Eurofer 97 (9Cr steel)

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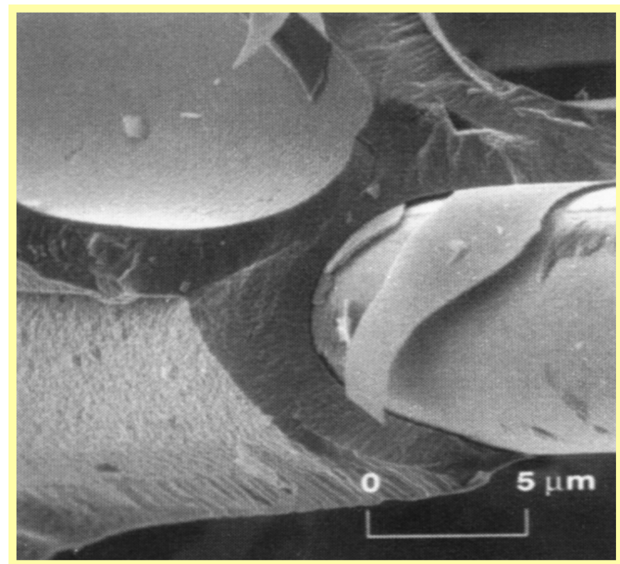
Development of SiC Composites for Nuclear Reactor Structural Applications: Difficult & High Risk But High Payoff

- **SiC Composites Offer**
  - Low radioactivity and afterheat; chemically inert (eases safety and waste disposal concerns)
  - High operating temperatures (greater thermodynamic efficiency) and low thermal neutron absorption
- **The Feasibility Issues**
  - Conventional SiC composites are not a hermetic barrier to fission gases
  - Thermal conductivity is reduced by irradiation
  - Little is known about mechanical property response to irradiation
  - Technology base for production, joining, design of large structures is very limited
- **Research Approach**
  - Understand the magnitude and cause of radiation effects on key properties such as thermal conductivity and strength

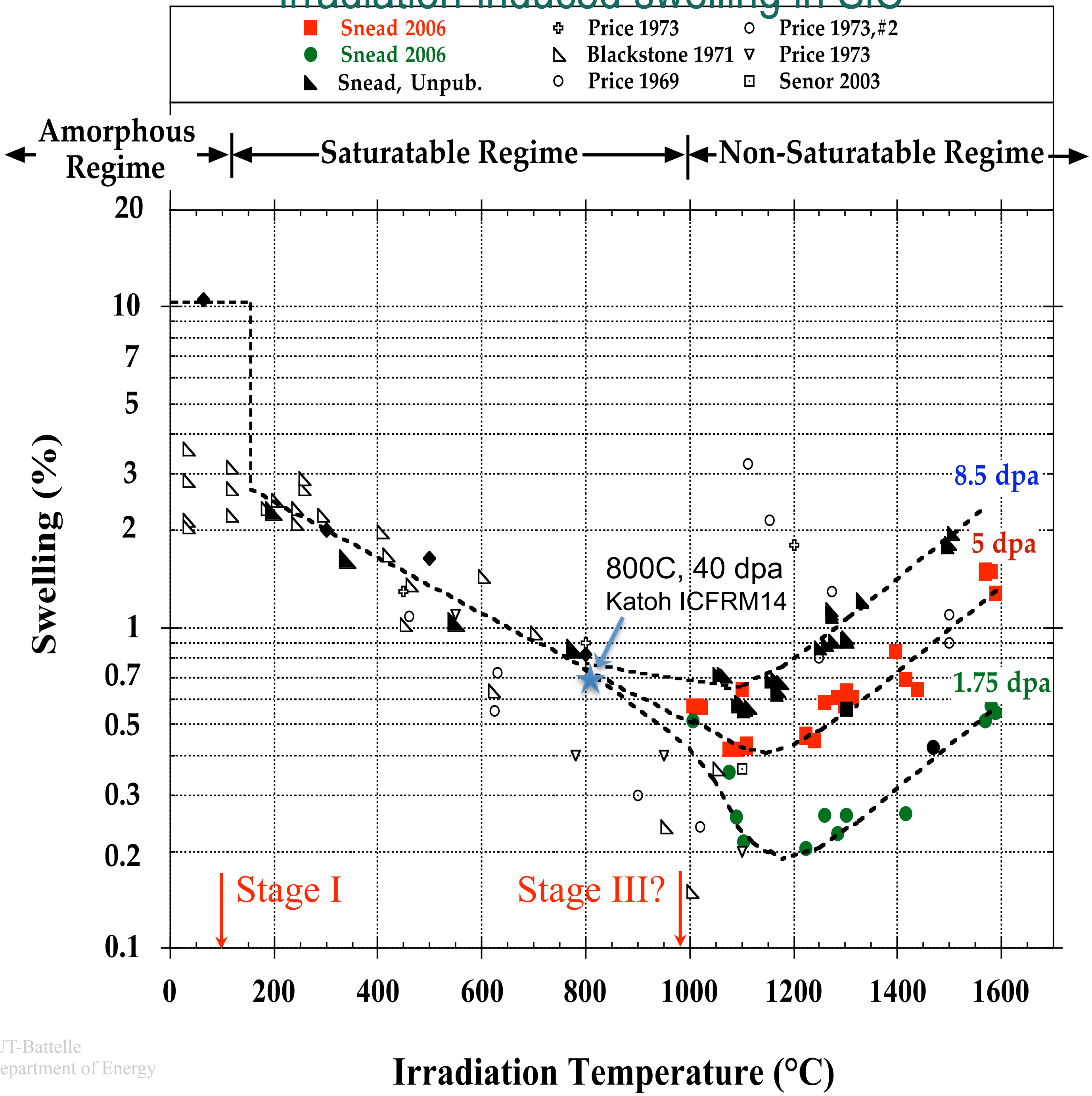
- Design composite structures (fiber, fiber-matrix interphase and matrix) with improved performance
- Develop the required technology base with industry and international partners



Silicon carbide composites offer engineered structures for extreme environments through tailoring of the fiber, matrix, and interphase structures

75 Managed by UT-Battelle
for the U.S. Department of Energy

Irradiation-induced swelling in SiC



L.L. Snead

Conclusions

- All cladding options have some shortcomings
  - Further materials research is important for improving the current candidate materials
- Zr alloys are a reasonable option for light water reactor systems
  - Near term alternate: austenitic steels (neutronics penalty)
  - Longer term alternate: SiC/SiC composites (fabrication/joining, fission product containment, and low ductility are some key challenges)
- Design of nanoscale features in structural materials confers improved mechanical strength and radiation resistance
  - Further research on fundamental mechanisms and experimental validation of performance are needed to develop improved materials for advanced nuclear energy systems