Pathways for Fusion Energy: Impact of Materials Options

Steven J. Zinkle[1,2], Lance L. Snead[3], Brian D. Wirth[1,2]

[1]University of Tennessee, Knoxville, TN USA

[2]Oak Ridge National Lab, Oak Ridge, TN USA

[3]Massachussetts Institute of Technology, Cambridge, MA USA

ANS topical meeting on Technology of Fusion Energy

Philadelphia, PA

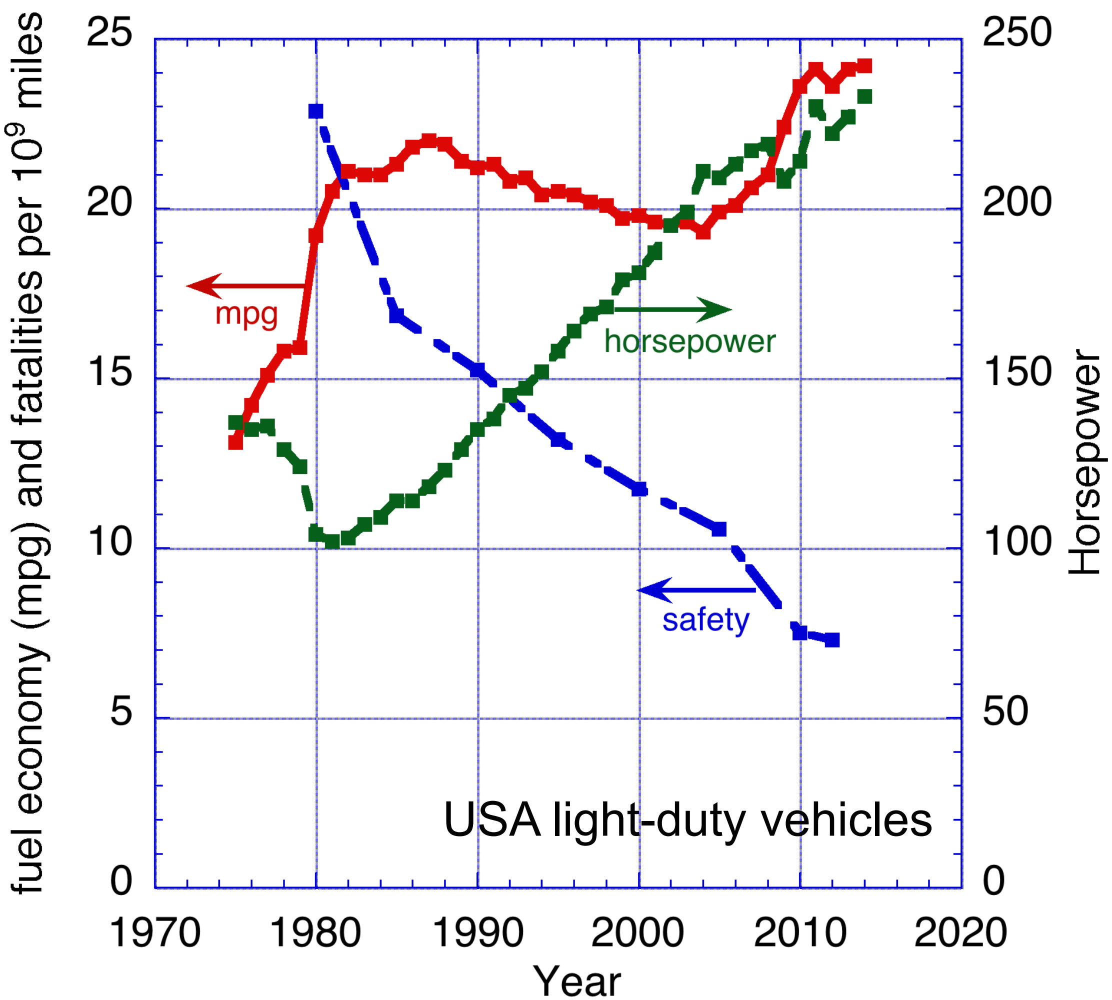
August 22-25, 2016

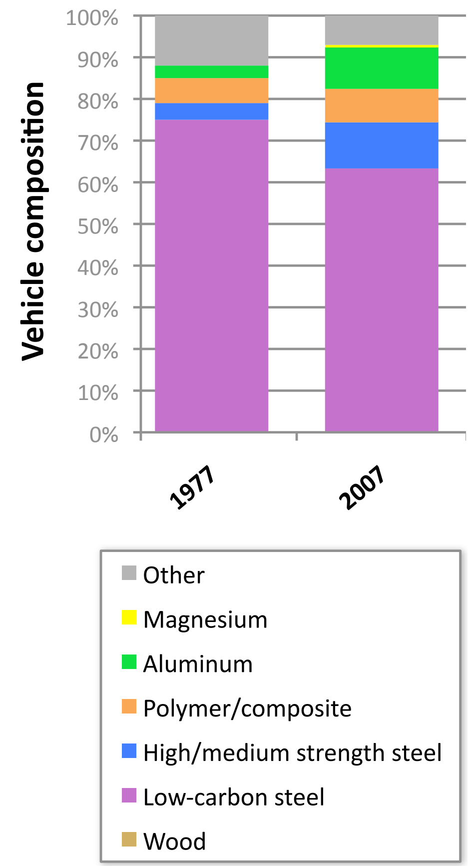
THE UNIVERSITY of TENNESSEE KNOXVILLE

OAK RIDGE National Laboratory

Evolution in light duty personal vehicles

Greatly improved safety, along with improved performance (horsepower, fuel economy)

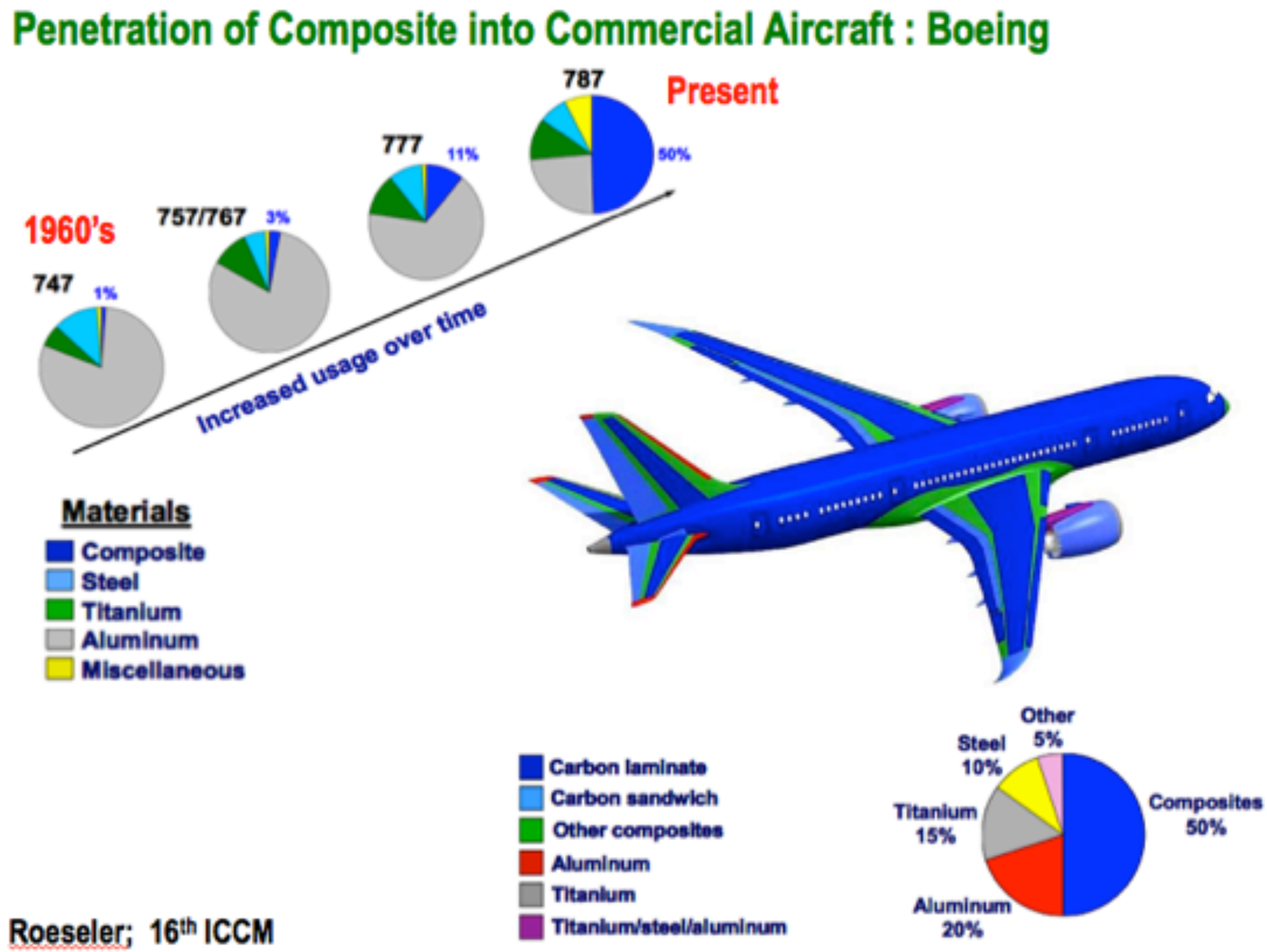




A. Taub et al., JOM 59, 2 (2007) 48

Source: US Dept. of Transportation and Environmental Protection Agency

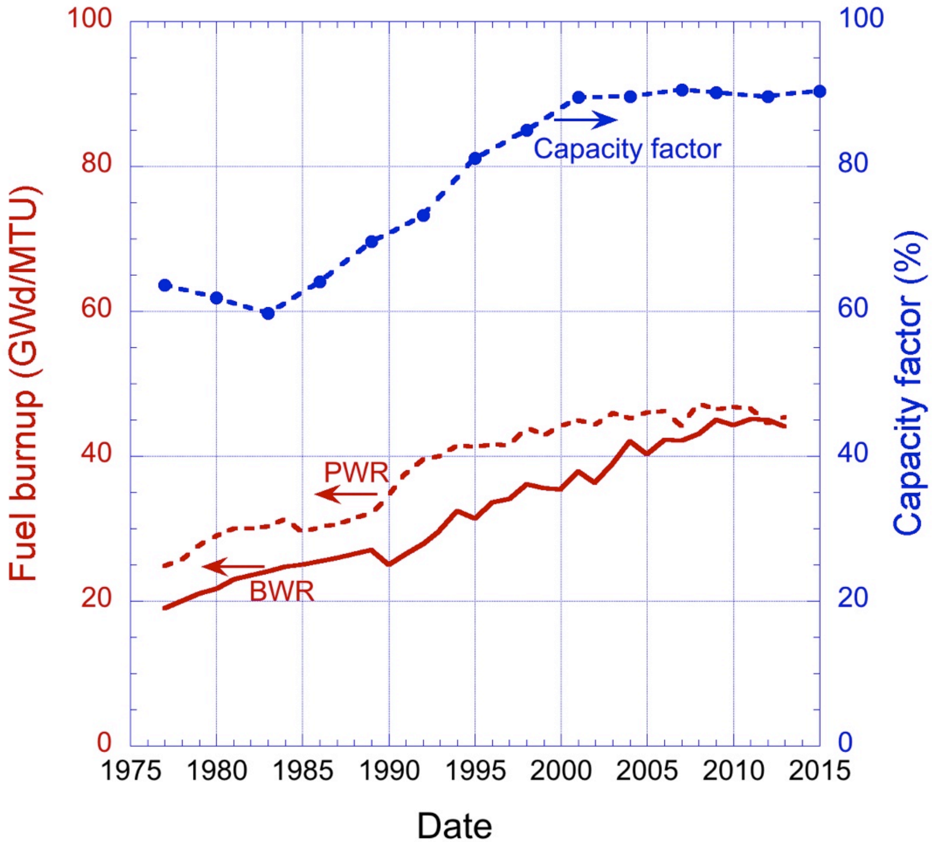
Evolution of materials used in commercial aircraft

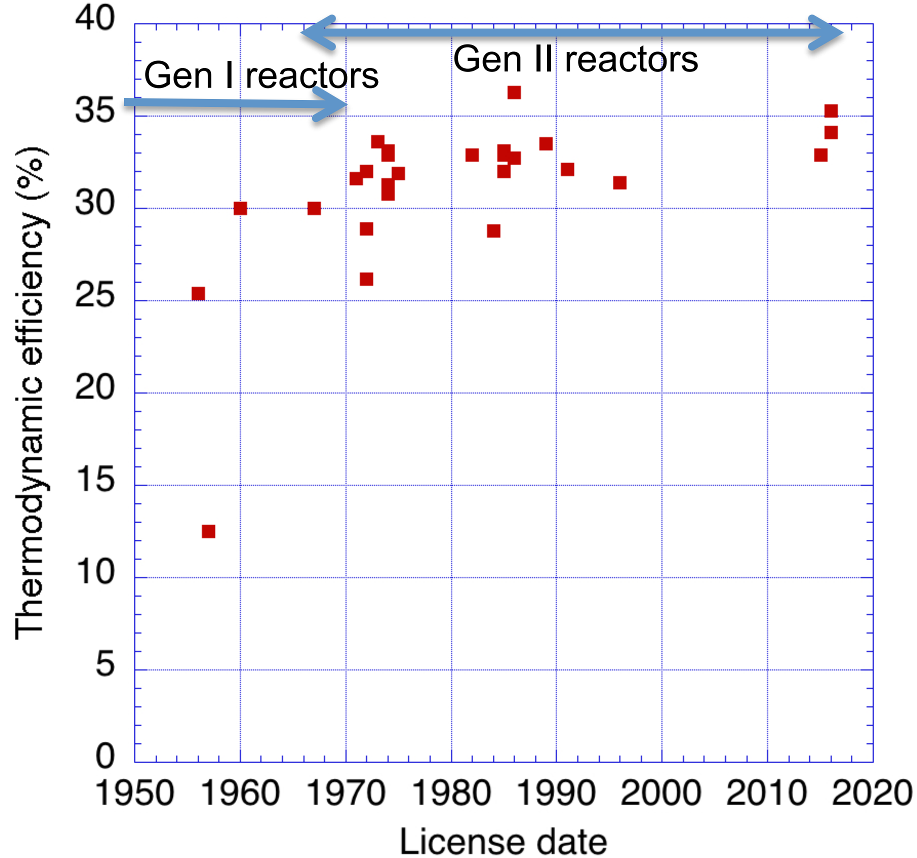


W.G. Roeseler et al., in 16th Int. Conf. on Composite Materials, Kyoto, Japan (Japan Soc. for Composite Mater., 2007)

US Fission Reactor Performance

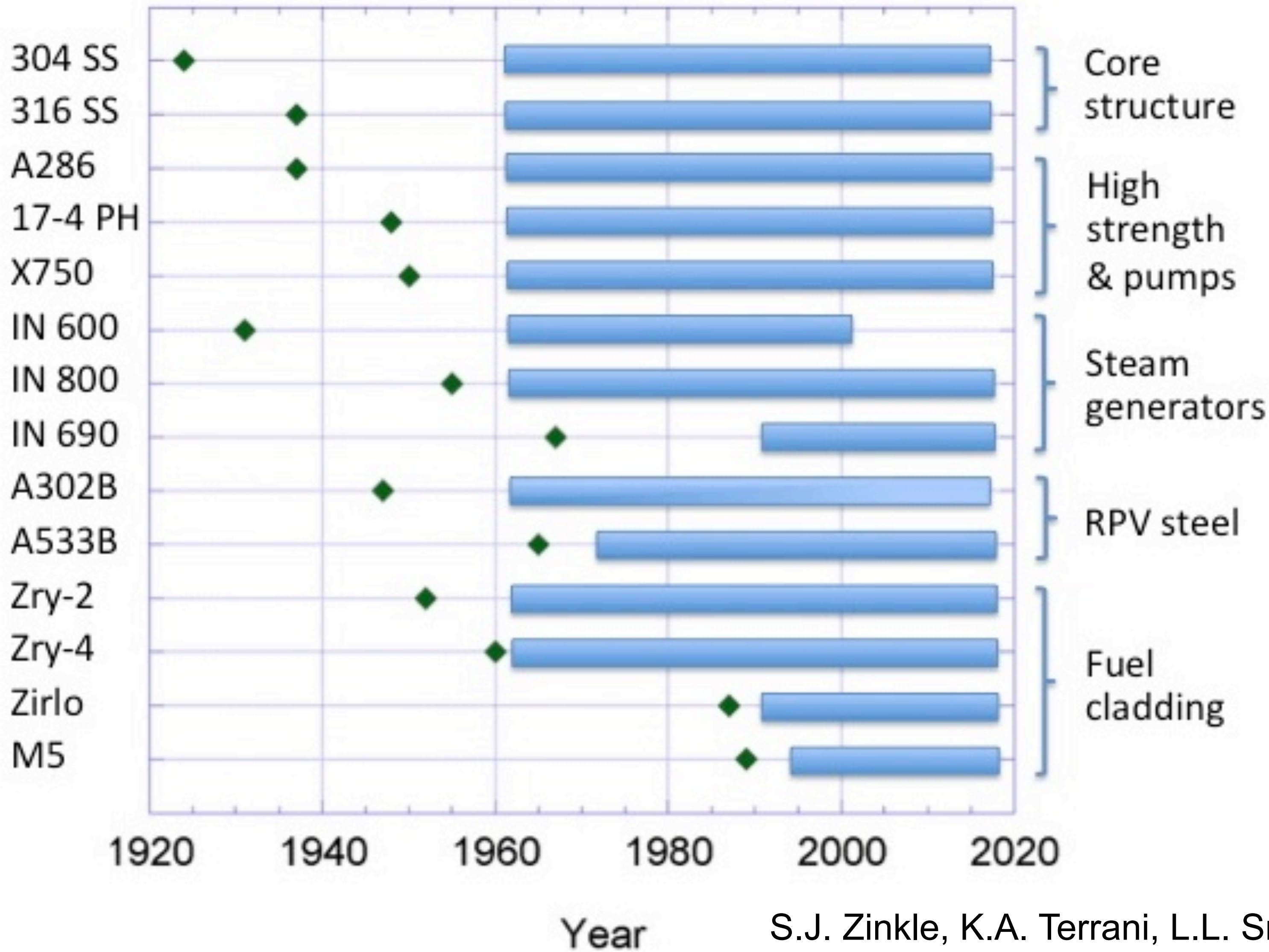
Steady improvement in fuel burnup and capacity factor, but little change in thermodynamic efficiency in past 40 years





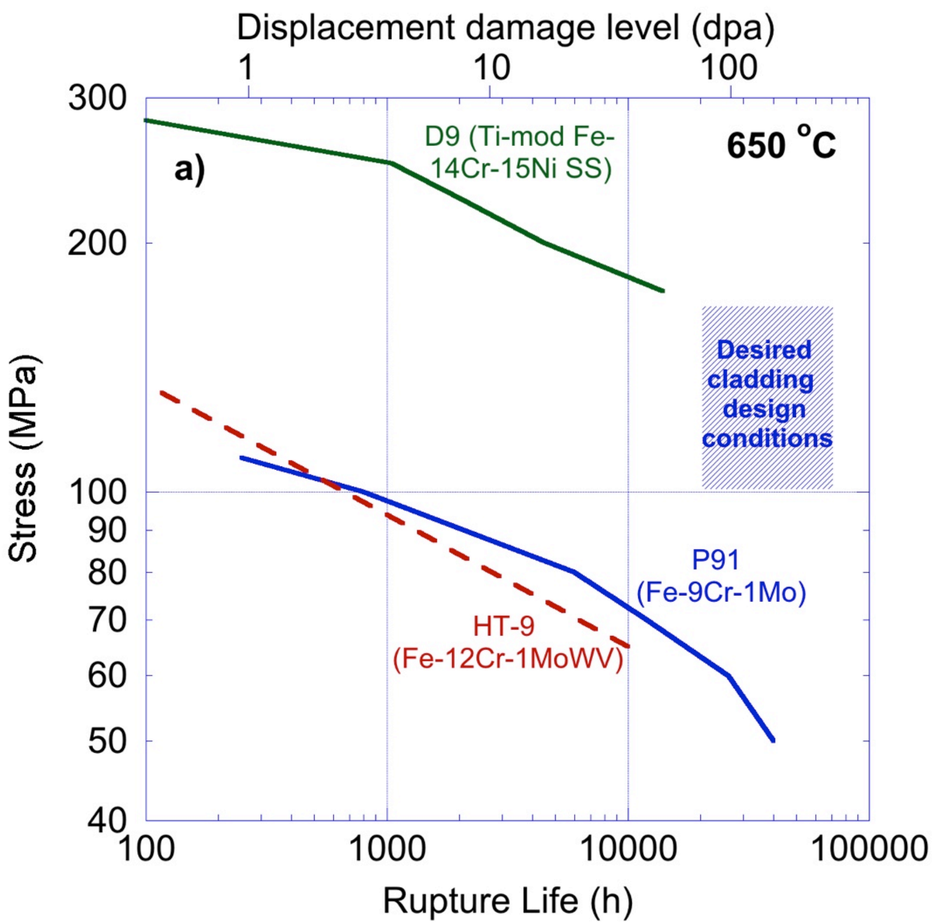
S.J. Zinkle & G.S. Was, Acta Mater., 61 (2013) 735 (updated to include most recent reported data) and NRC list of power reactors (2016)

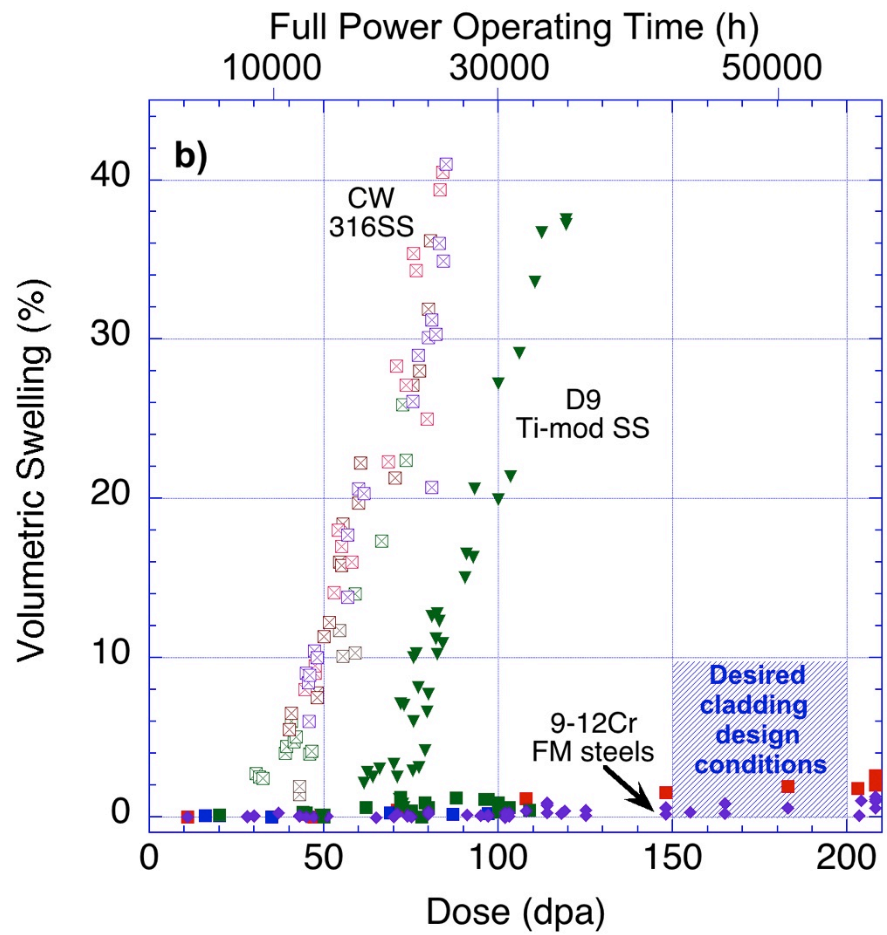
Timeline of structural materials used in light water reactors



S.J. Zinkle, K.A. Terrani, L.L. Snead, Current Opin. Sol. State & Mater. Sci., in press (20th anniv. special issue, 2016)

Thermal creep and void swelling in sodium-cooled fast reactor cladding is problematic for conventional steels





S.J. Zinkle, K.A. Terrani, L.L. Snead, Current Opin. Sol. State & Mater. Sci., in press (20th anniv. special issue, 2016)

Fusion materials R&D and selection will have a major impact on the ultimate viability of fusion energy

- **Robust plasma facing materials**
  - plasma compatibility
  - high heat flux capability
  - minimal net surface erosion
  - low tritium retention
  - moderate lifetime with high reliability
  - use captured heat for power conversion?

Solid vs. liquid wall approaches

- **High operating temperatures (thermodynamic efficiency) and high reliability/ long lifetimes are likely needed to offset the higher capital costs for fusion**
  - Requires high performance structural materials
- **Reduced activation materials for environmental attractiveness**
  - no public evacuation during accident scenarios
  - no long lived wastes requiring geologic disposition
- **Blanket materials for fuel regeneration and heat extraction**
- **Magnet and plasma diagnostic/heating materials**

Translation of Requirements to GOALS for Fusion Power Plants

Larger extrapolation from present

Requirements:

- **Have an economically competitive life-cycle cost of electricity:**
  - Low recirculating power (Increase plasma Q, …);
  - High power density (Increase $P_f \sim \beta^2 B_T^4$, …)
  - High thermal conversion efficiency;
  - Less-expensive systems.

COE has a "hyperbolic" dependence ($\propto$ 1/x) and improvements "saturate" after certain limit

- **Gain Public acceptance by having excellent safety and environmental characteristics:**
  - Use low-activation and low toxicity materials and care in design.

- **Have operational reliability and high availability:**
  - Ease of maintenance, design margins, and extensive R&D.

- **Acceptable cost of development.**

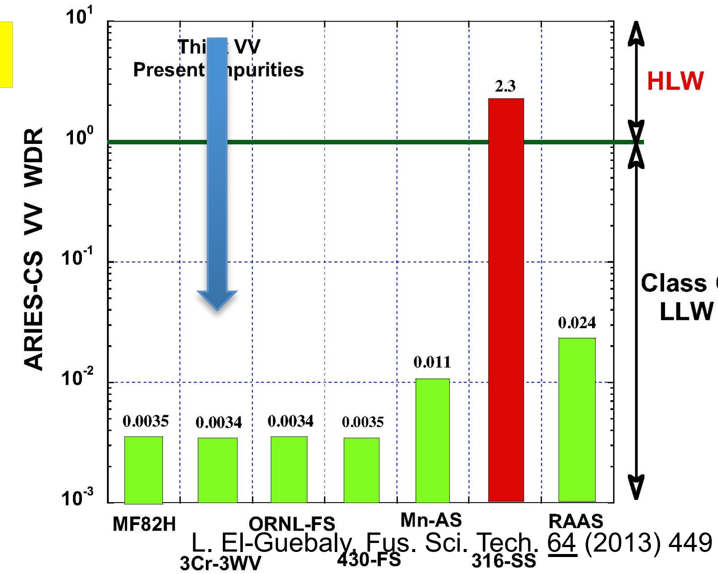
Stay close to present-day

Bainitic steel development for fusion vacuum vessel

- **Target: Fusion structural applications in next-step fusion devices (FNSF or DEMO);**
  - **Vacuum vessel (>400-500°C, relatively low dose)**
  - **Structural ring, magnet shields**

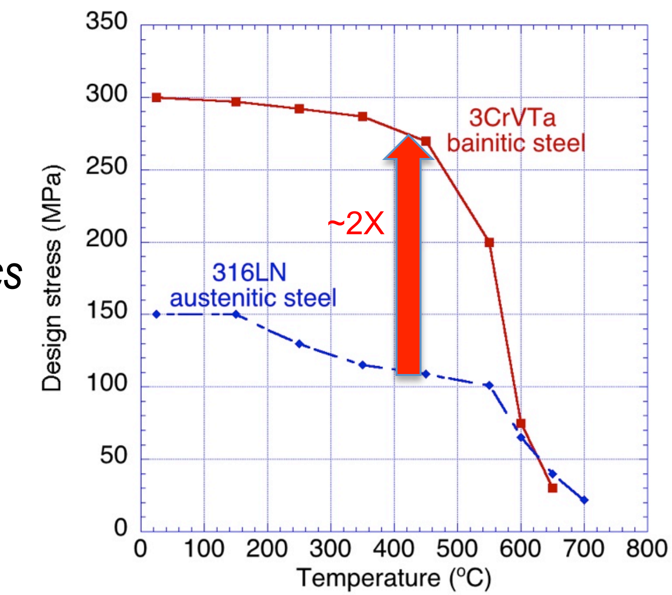
- **Why 3Cr-3WV(Ta) steels?:**
  - **Inexpensive low alloy steel**
  - **Improved tensile and creep properties due to formation of carbide-free acicular bainite ferrite (lower bainitic microstructure)**
  - **potentially no requirement of PWHT, suitable for large volume components**

>100X lower waste disposal burden vs. 316SS



L. El-Guebaly, Fus. Sci. Tech. 64 (2013) 449

- **Approach:**
  - Computational thermodynamics for alloy design
    - *Effect of minor alloying additions on transformation kinetics*
    - *Phase equilibrium of strengthening second-phases*
  - Property evaluation of 3Cr-3WV(Ta) steels
    - *Production of CCT diagrams*
    - *Mechanical property / weldability*

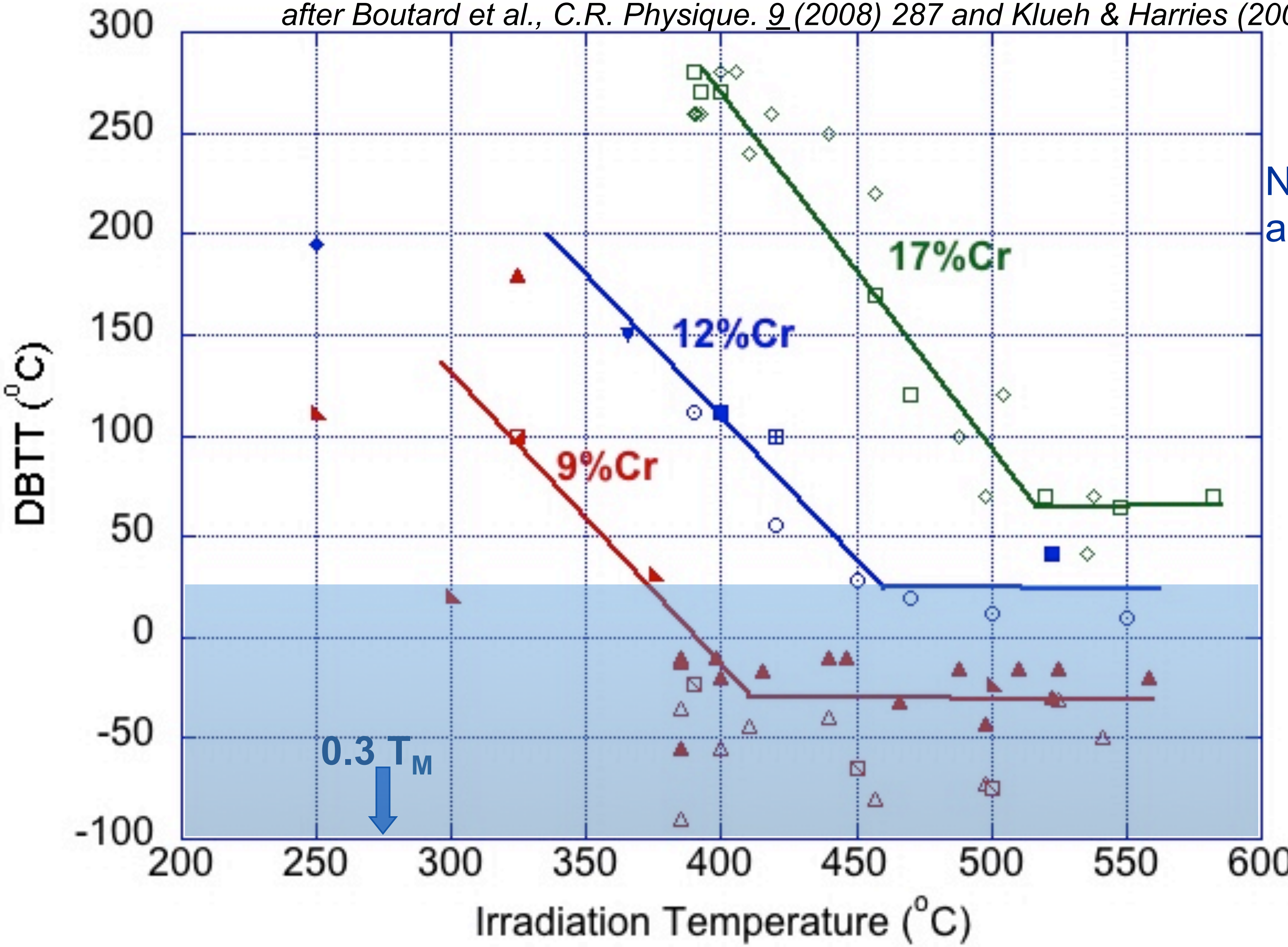


3Cr data: R. Klueh, Met Mat Trans 28A (1997) 335

Y. Yamamoto, ORNL

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

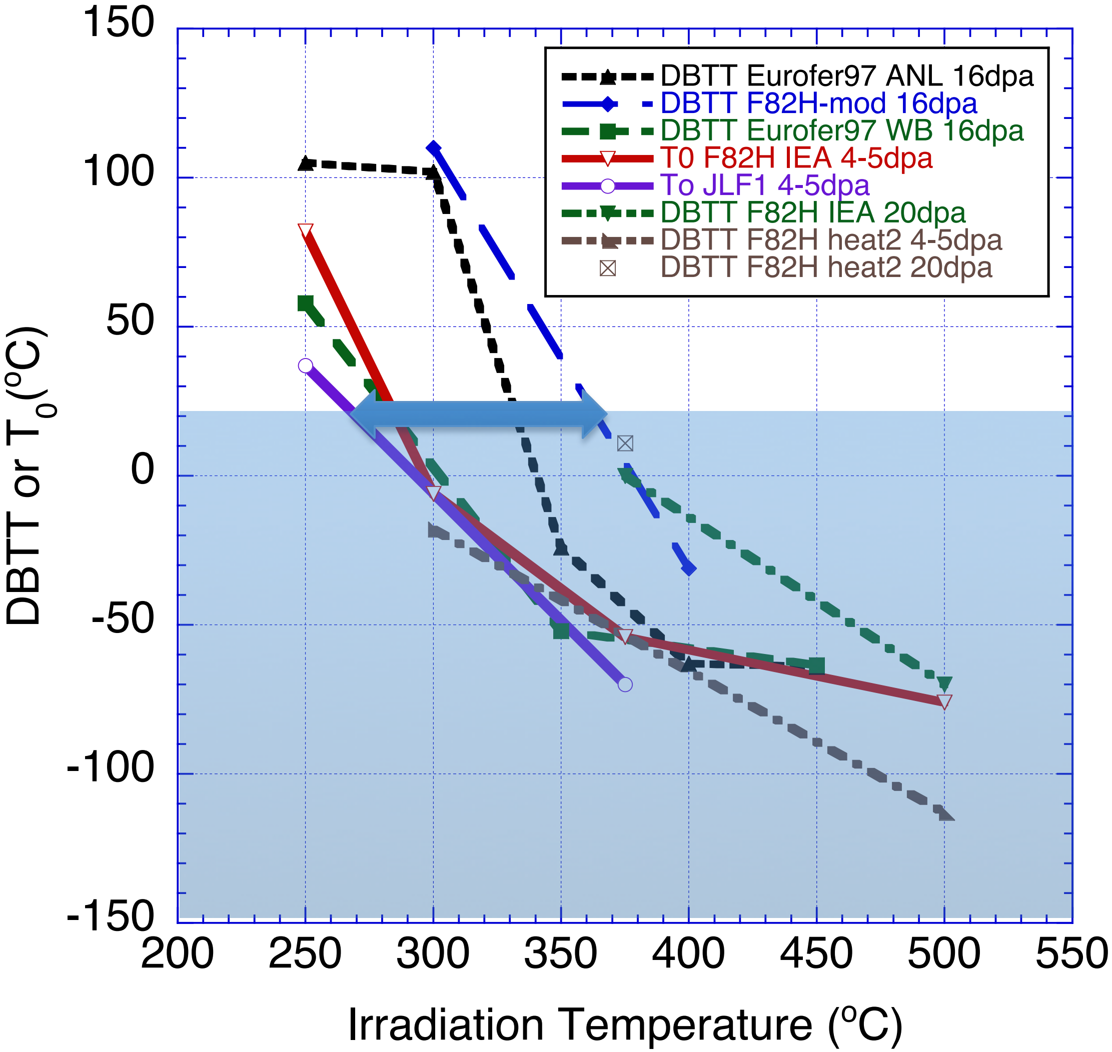
after Boutard et al., C.R. Physique. 9 (2008) 287 and Klueh & Harries (2001)



Need <12%Cr for acceptable DBTT

S.J. Zinkle & N.M. Ghoniem, J. Nucl. Mater. 417 (2011) 2

Effect of Neutron Irradiation on the Ductile to Brittle Transition Temperature of Reduced Activation 9Cr Ferritic/Martensitic Steels

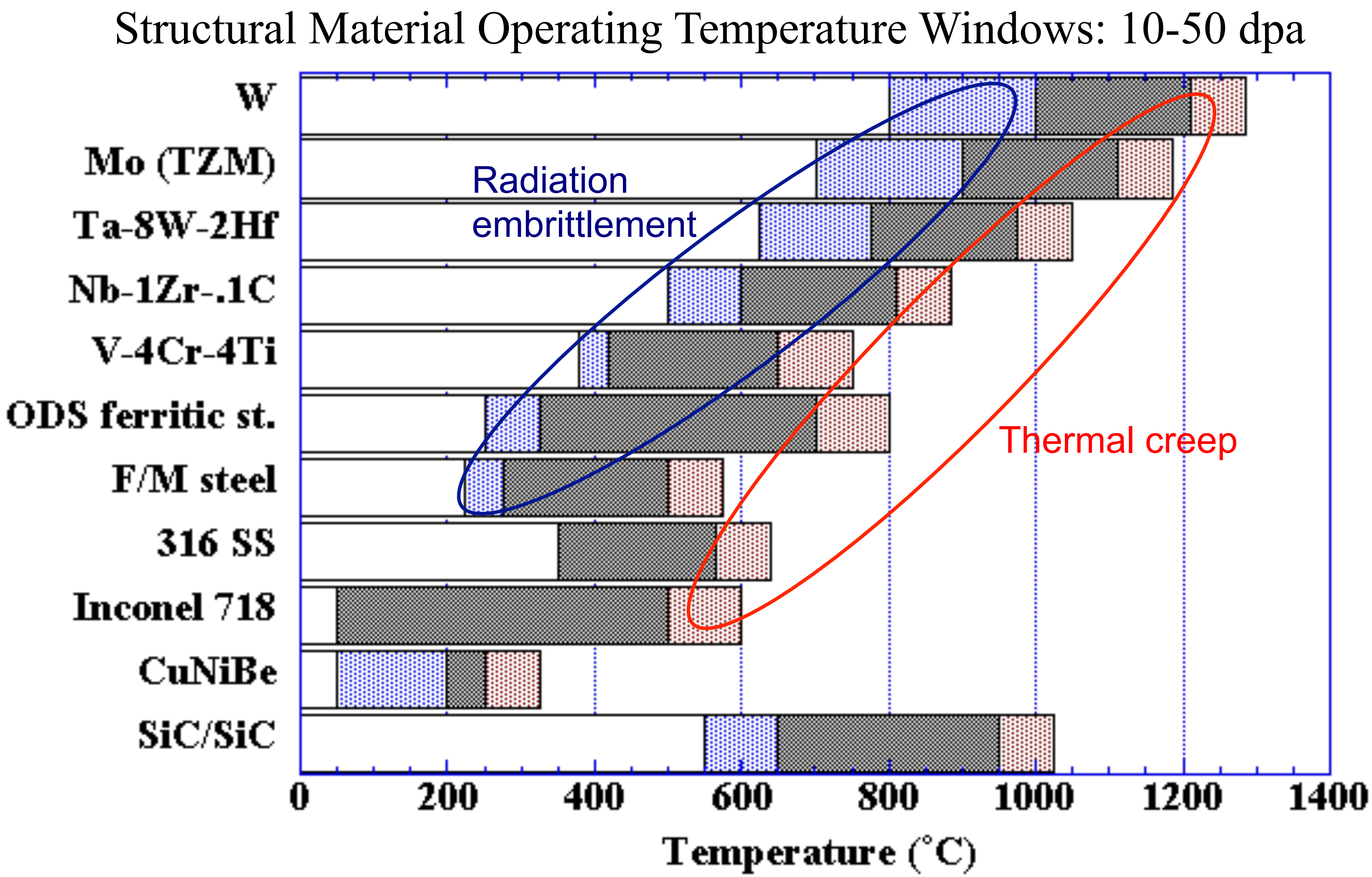


~100°C variation in DBTT for different heats of 9Cr reduced activation FM steel

- what is the cause of this variation?
- Can further reductions be achieved?

S.J. Zinkle, A. Möslang, T. Muroga and H. Tanigawa, Nucl. Fusion 53, no.10 (2013) 104024

Can we break the shackles that limit conventional structural materials to ~300^{o}C temperature window?

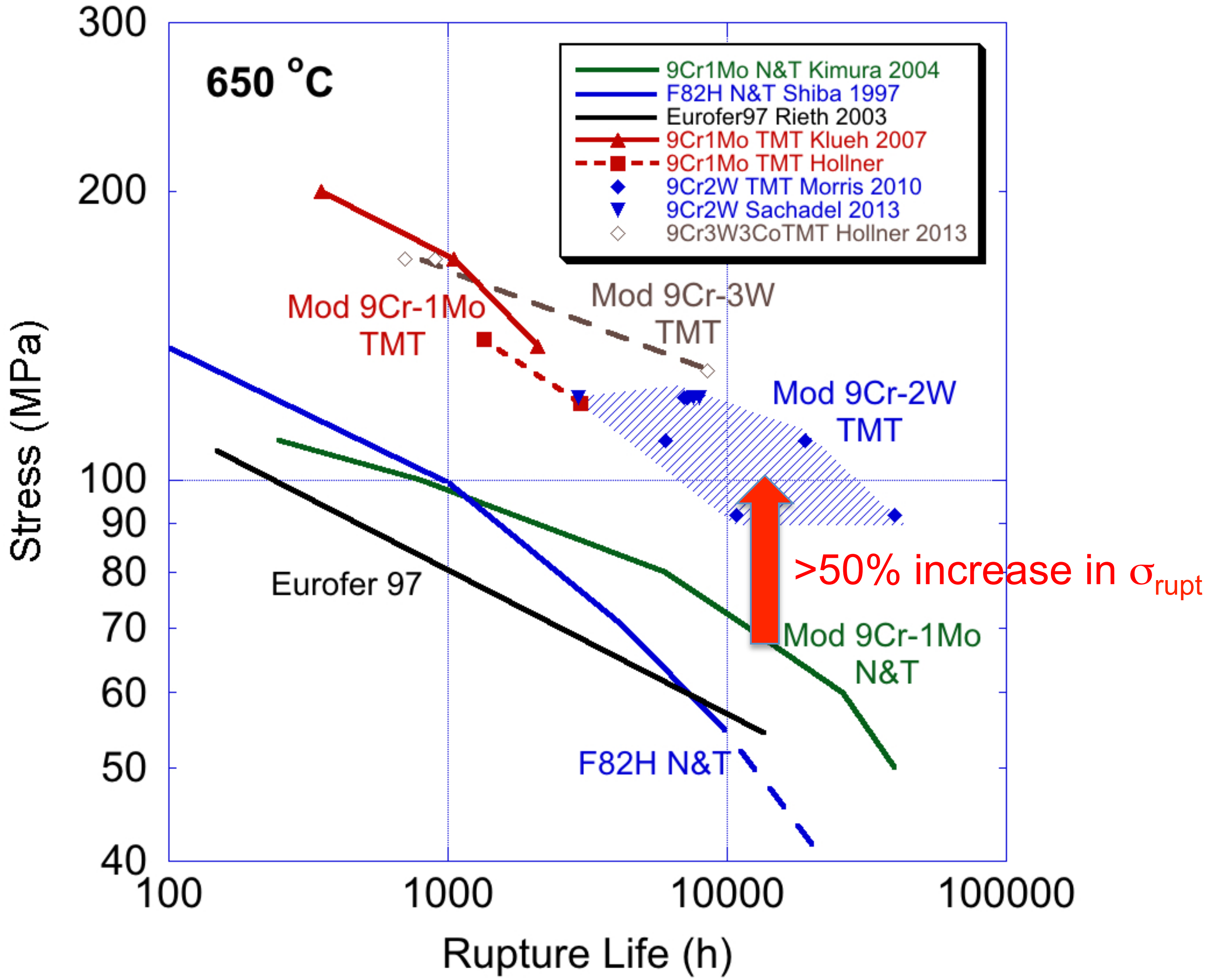


$\eta_{Carnot}=1-T_{reject}/T_{high}$

Additional considerations such as He embrittlement and chemical compatibility may impose further restrictions on operating window

Zinkle and Ghoniem, Fusion Engr. Des. 49-50 *(2000) 709*
S.J. Zinkle & J.T. Busby, Mater. Today 12 *(2009) 12*

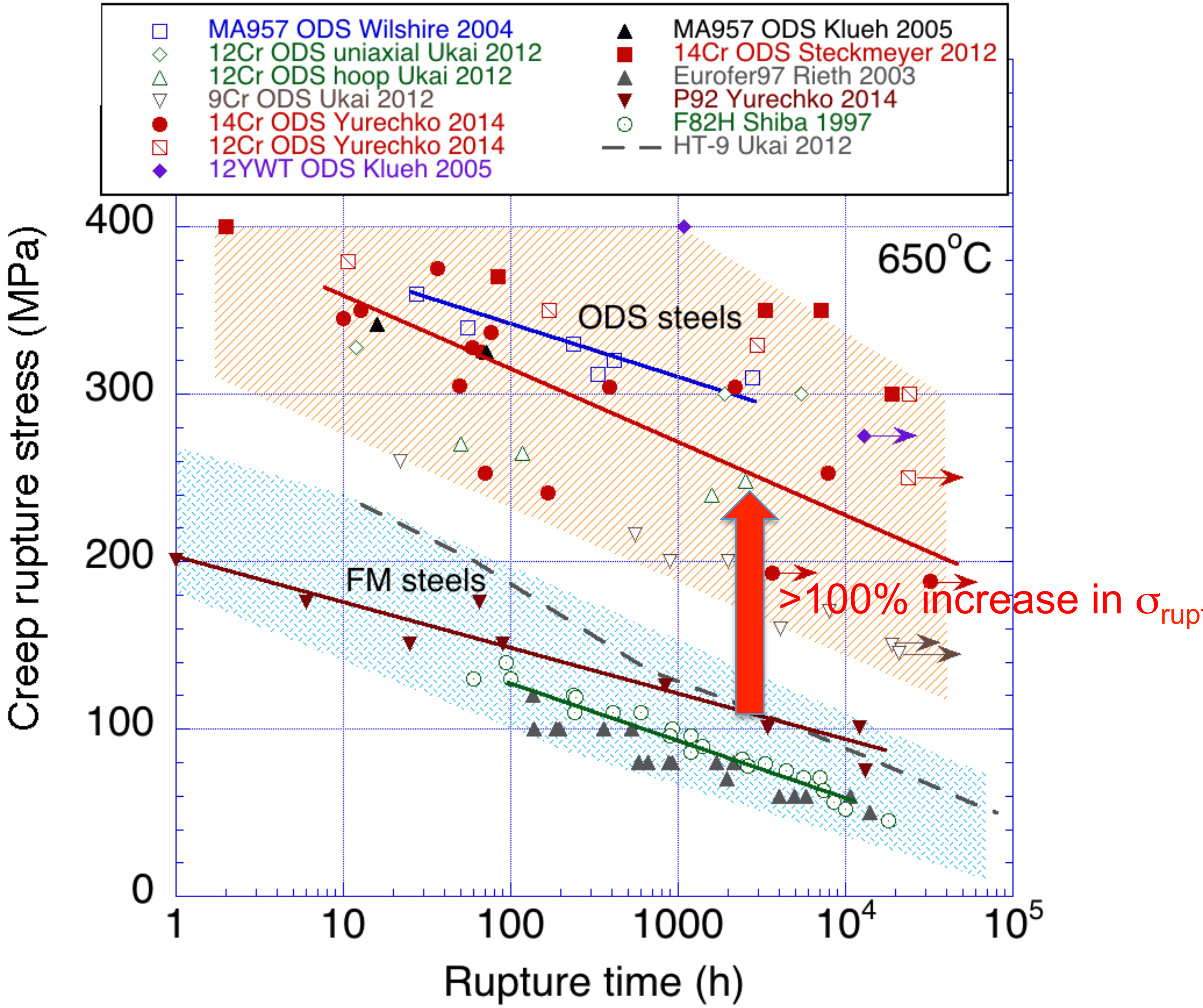
Creep rupture behavior for conventional 9Cr steels vs. next-generation steels (designed by computational thermodynamics)



S.J. Zinkle et al., Nucl. Fusion (2016) Special Issue on Materials for a Fusion Reactor, in press

THE UNIVERSITY of TENNESSEE KNOXVILLE
OAK RIDGE National Laboratory

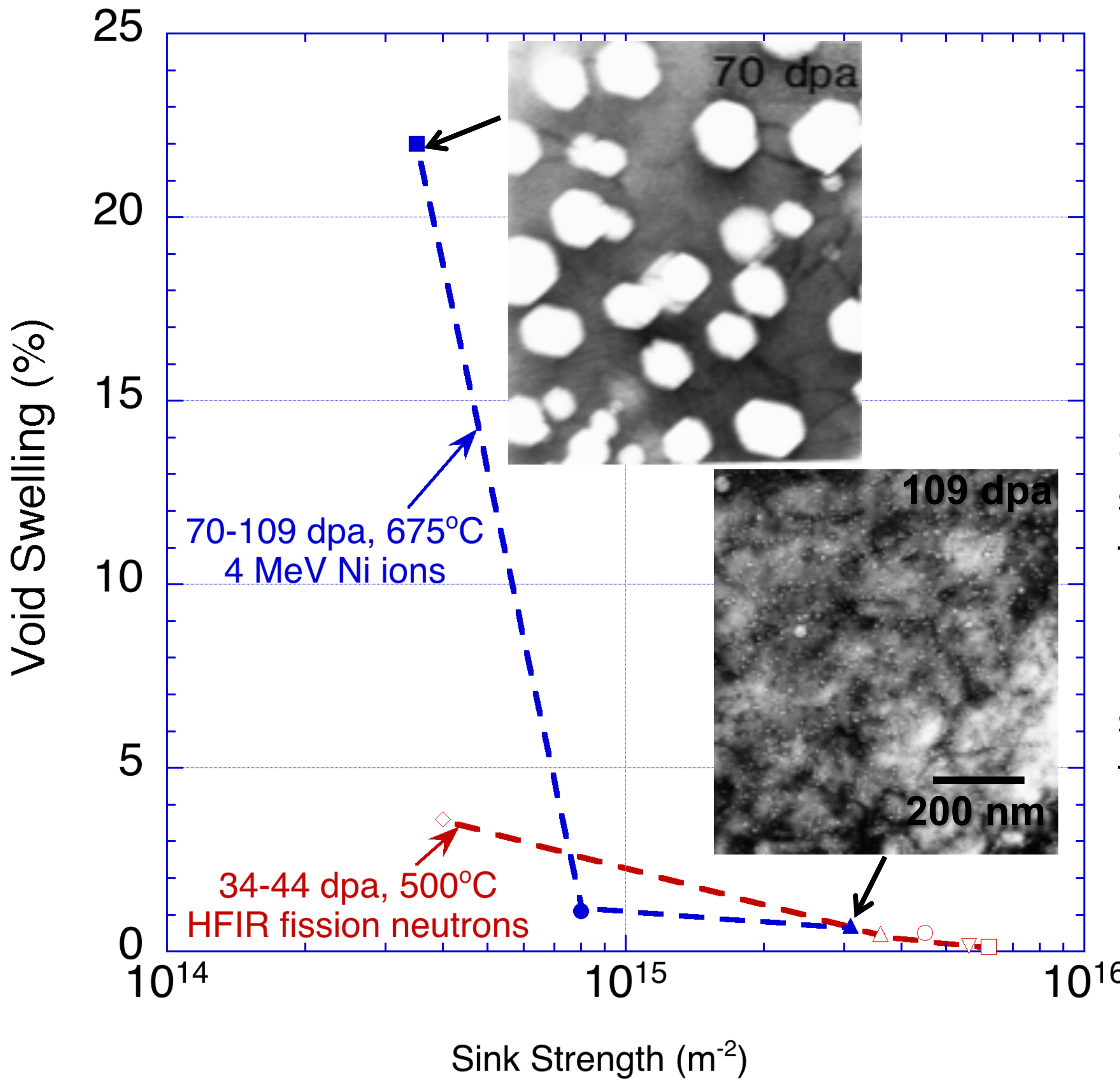
Creep rupture behavior for ODS steels vs. conventional 9Cr steels



S.J. Zinkle et al., Nucl. Fusion (2016) Special Issue on Materials for a Fusion Reactor, in press

Approaches for radiation resistance 3: High Sink Strength

Example: Volumetric Void Swelling of Irradiated FeCrNi Austenitic Alloys



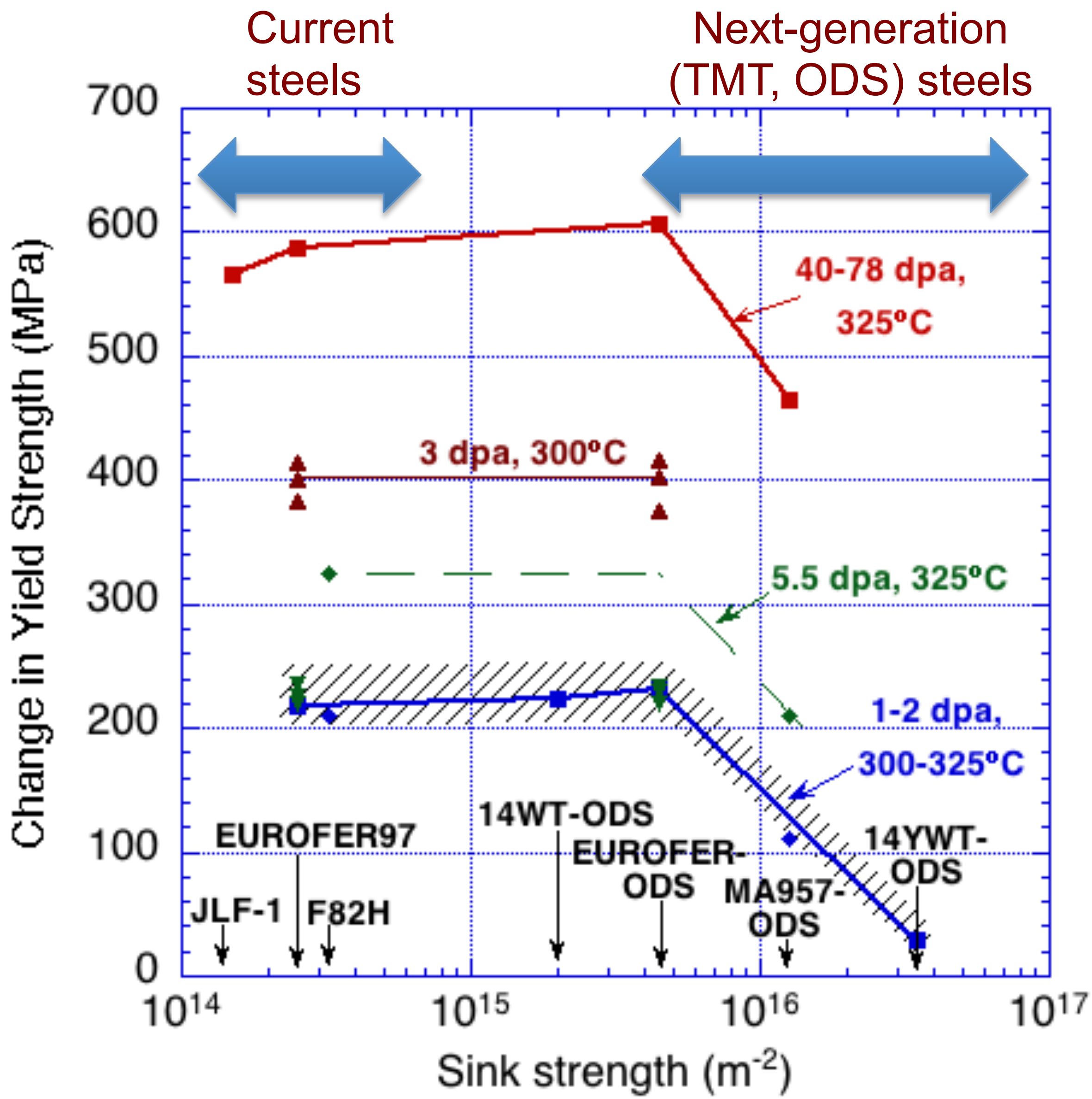
High sink strengths can be achieved via nanoscale precipitation

For void swelling resistance, sink strengths $>10^{15}/m^2$ are generally sufficient for light water fission reactors; fusion reactor irradiation may require even higher sink strengths ($>10^{16}/m^2$?) due to transmutant He production

$S_{ppt} \sim 4\pi RN$

S.J. Zinkle and L.L. Snead, Ann Rev. Mat. Res., 44 (2014) 241

Effect of Initial Sink Strength on the Radiation Hardening of Ferritic/martensitic Steels



Dramatic reduction in radiation hardening occurs when average spacing between defect clusters (dislocation loops, etc.) is >10x average spacing between defect sinks

$N_{loop}^{-1/3} > 10\ S_{tot}^{-1/2}$

or equivalently,

$S_{tot} > 5\text{-}10\ S_{rad\ defects}$

$S_{ppt} \sim 4\pi RN$

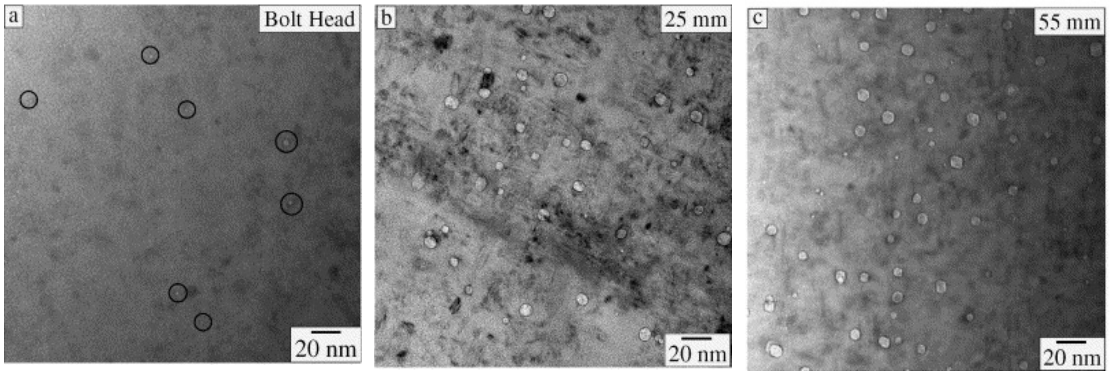
S.J. Zinkle and L.L. Snead, Ann Rev. Mat. Res., 44 (2014) 241
S.J. Zinkle et al., Nucl. Fusion (2016 special issue) in press

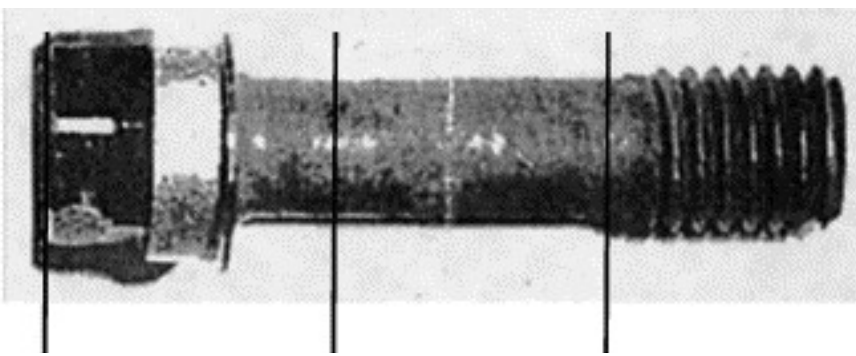
H retention increases dramatically in the presence of cavity formation

3 to 5x increase in retained hydrogen when cavities are present, even with 2-3x reduction in neutron fluence exposure

500-700 appm H (few cavities)

1700-3700 appm H (rad.-induced cavities present)



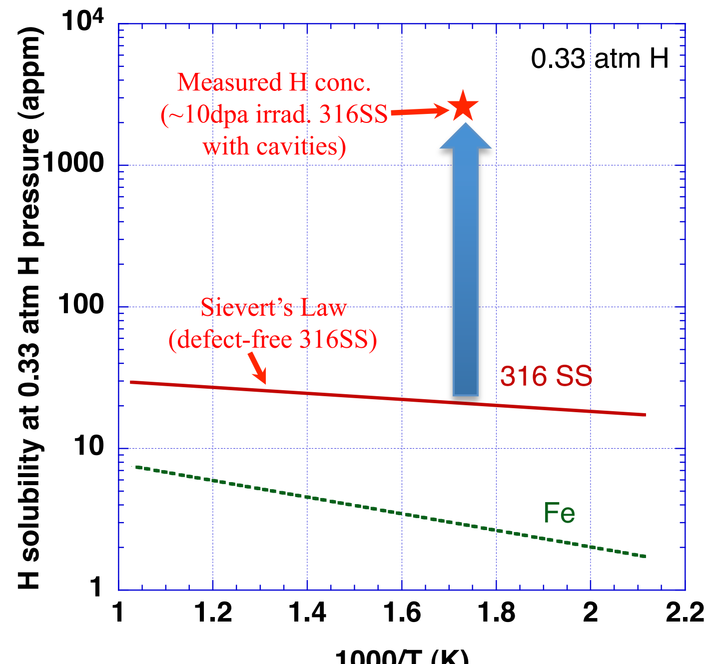


Bolt head
1 mm
320°C, 19.5 dpa

Bolt shank
25 mm
343°C, 12.2 dpa

Near threads
55 mm
333°C, 7.5 dpa

Retained H level is ~100x higher than expected from Sievert's law solubilities



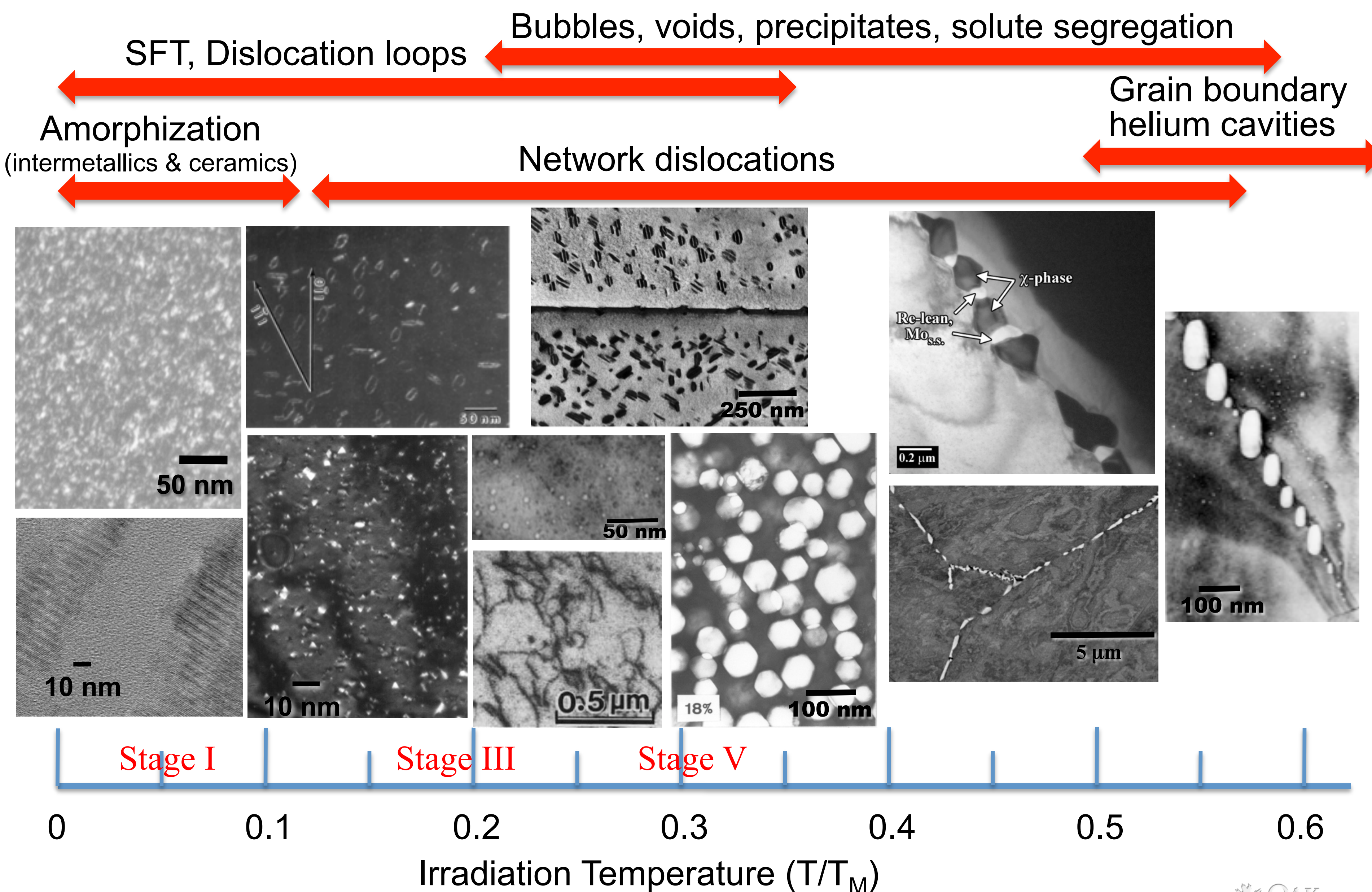
Baffle-former bolt removed from Tihange-1 (Belgium) pressurized water reactor

Type 316 austenitic stainless steel

F.A. Garner et al., J. Nucl. Mater. 356 (2006) 122

S.J. Zinkle, Nucl. Fusion 53 (2013) 104024

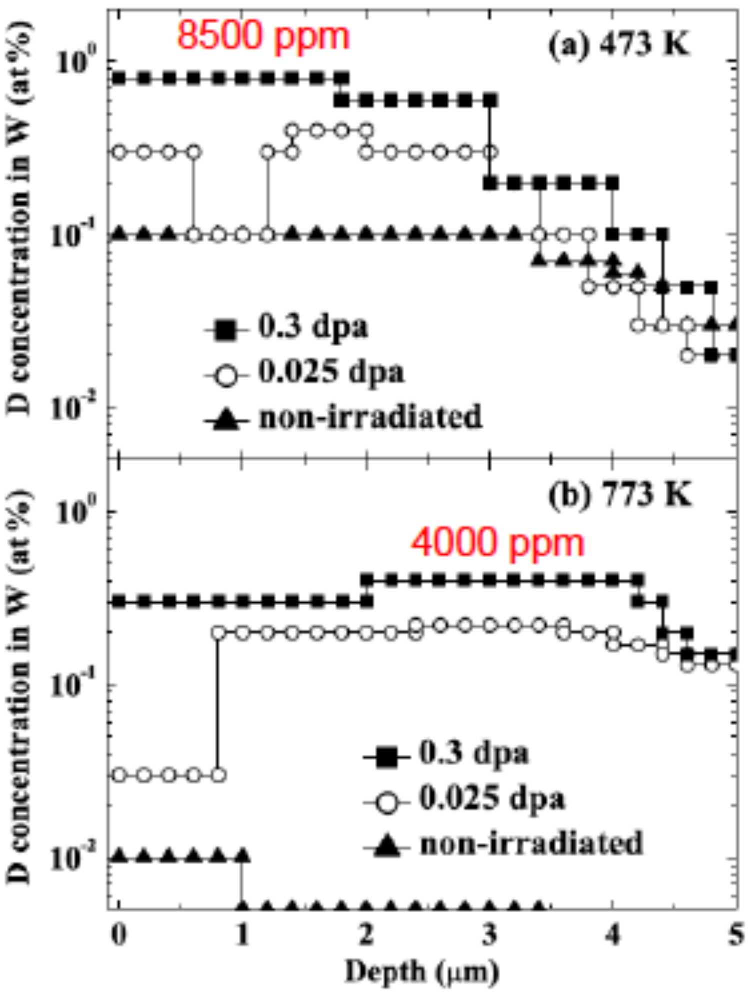
Temperature dependent microstructures in irradiated materials

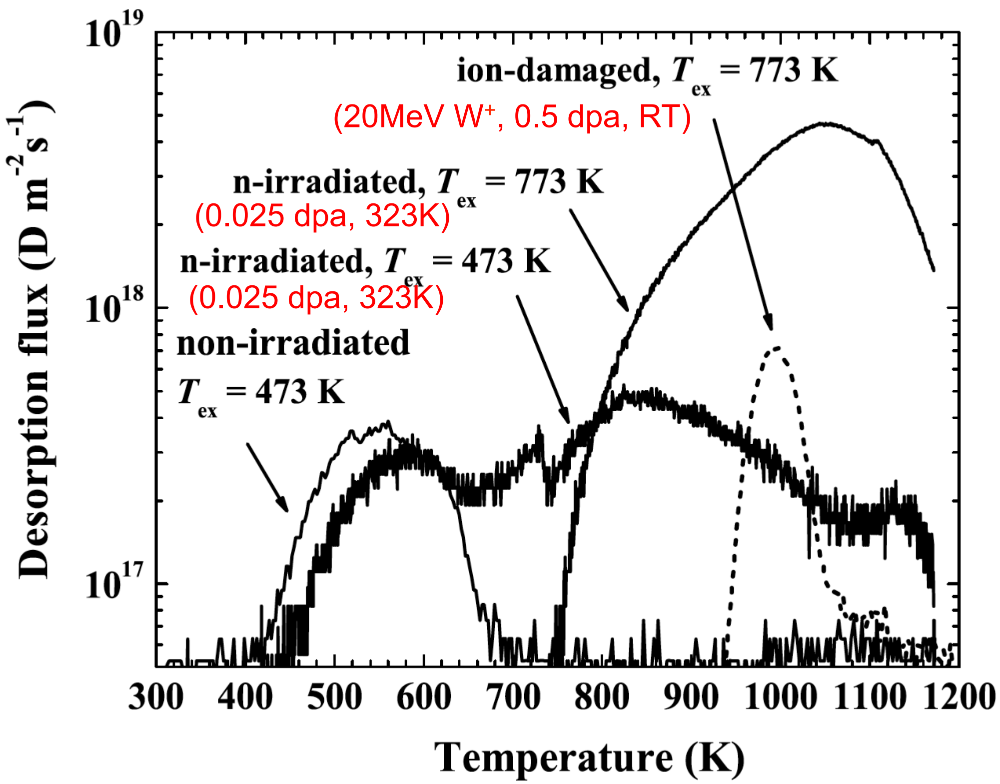


Zinkle & Snead, Ann. Rev. Mater. Res. 44 (2014) 241

Neutron effects can influence D retention in W

• More D retained in damaged W; D distribution relatively uniform in plasma exposed, neutron irradiated W
• D desorption spectra completely different for ion-damaged and neutron irradiated samples --- Temperature of plasma (& neutron/ion) exposure also important factor





H isotope trapping is strongest at irradiation temperatures where cavities form

Y. Hatano et al., Nuclear Fusion, vol. 53, p. 073006, July 2013.

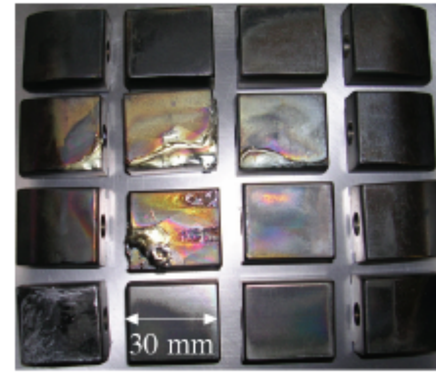
OAK RIDGE National Laboratory

PFC Challenge: No current materials are viable to bridge the significant gap between today's tokamaks & future fusion reactors

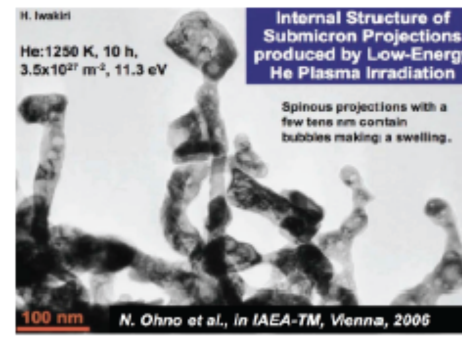
Issue / Parameter	Present Tokamaks	ITER	DEMO	Consequences
Quiescent energy exhaust *GJ / day*	~ 10	3,000	60,000	- active cooling - max. tile thickness ~ 10 mm
Transient energy exhaust from plasma instabilities *$\Delta T \sim MJ / A_{wall}(m^2) / (1\ ms)^{1/2}$*	~ 2	15	60	- require high $T_{melt/ablate}$ - limit? ~ 60 for C and W - surface distortion
Yearly neutron damage in plasma-facing materials *displacements per atom*	~ 0	~ 0.5	20	- evolving material properties: thermal conductivity & swelling
Max. gross material removal rate with 1% erosion yield *(mm / operational-year)*	< 1	300	3000	- must redeposit locally - limits lifetime - produces films
Tritium consumption *(g / day)*	< 0.02	20	1000	- Tritium retention in materials and recovery

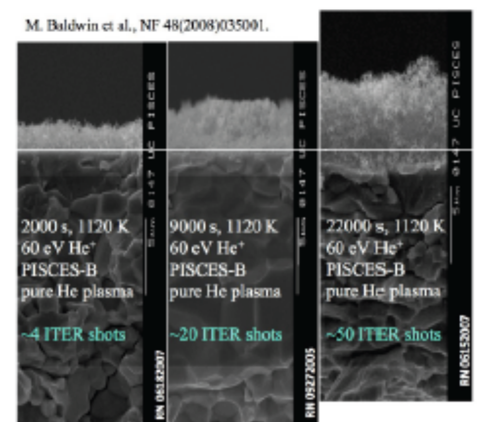
ITER's current operating plans involve a tungsten divertor: significant concern about sub-surface helium bubble formation & surface morphology changes that will influence core plasma performance & tritium retention, and/or tritium-containing dust

C-Mod Molybdenum (T_{melt}=2900 K) limiter melted during disruptions



- Dilute MFE plasma ($n \sim 10^{20}\ m^{-3}$) extinguished by small particulate
  - ➢ 2 mm "drop" of W == $N_{e,ITER}$





*Tritium inventory and control is significant challenge**

Retention in PFC

- Yearly fluence of tritium in divertor

$$\Phi_{ions} \sim 4x10^{31}\ m^{-2}$$
$$\Phi_{Tritium} \sim 10^{5}\, kg\ m^{-2}$$

- So total "recycling" of tritium in divertor

$$\Phi_{Tritium} S_{div} \simeq 10^{7} \frac{kg-T}{year}$$

- In-vessel safety limit: 1 kg

- **< 1 in 10^7 ions can be retained**

Tritium inventory

- Reactor consumes ~0.5 kg/day
- ~1% burn fraction
- 50 kg/day pumped & processed.
- Recycling 100x pumping
- 5000 kg/day recycled in PFC
- Blanket TBR ~ 1.05 → 0.025 kg/day net tritium

- **< 1 in $5x10^6$ can be lost to PFC to not affect supply.**

Worry about tritium retention in He bubbles forming below PFC surface

*Ref: D. Whyte, BPMIC presentation (2015)

Low Readiness of Breeding Blankets (no ideal candidate)

- Large knowledge gaps exist even with a successful ITER TBM programme
- Feasibility concerns/performance uncertainties → **Selection now is premature**
- Strengthen blanket technology R&D. Use System Engineering approach.

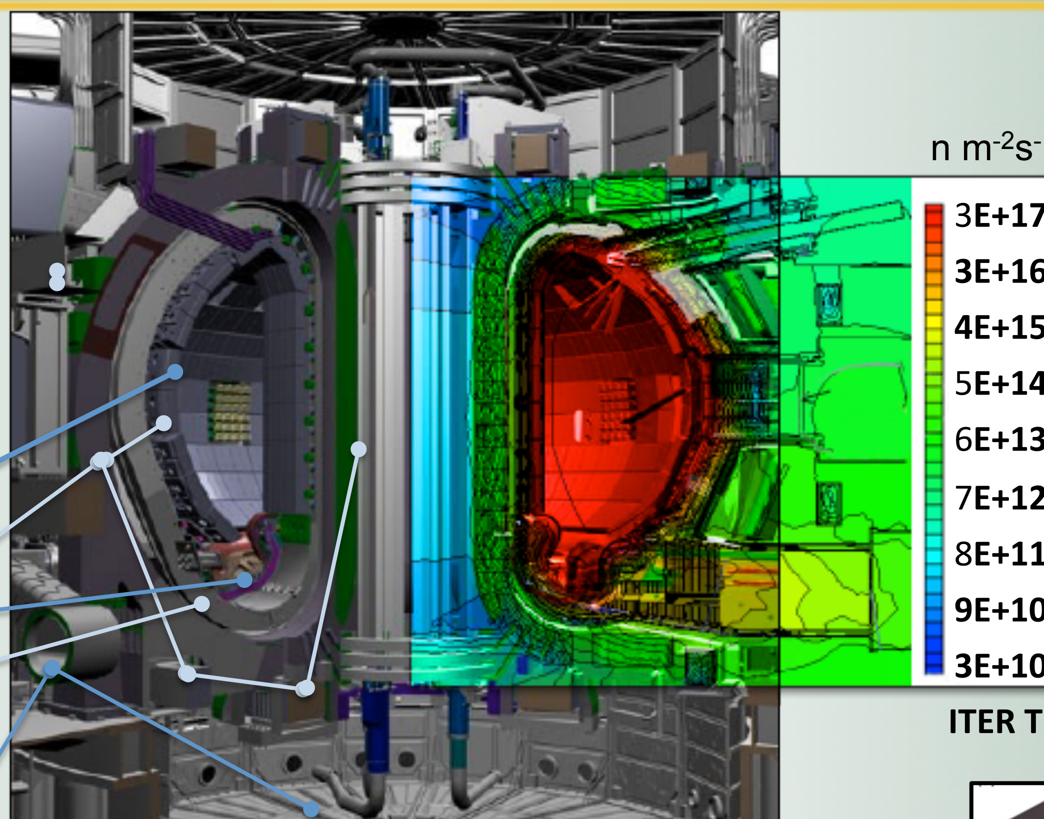
Current designs are based on 1980s-1990s concepts: any room for further innovation?

Concerns	HCPB	HCLL	WCLL	DCLL
Tested in ITER TBM	🙂	🙂	☹	☹
Suitability for Eurofer	🙂	🙂	🤔	🙂
FW heat flux capability	☹	☹	🤔	☹
Safety issues of coolant	🙂	🙂	☹	🙂
Technology readiness BoP	☹	☹	🙂	☹
Potential for high coolant outlet temperature	🙂	🙂	☹	🙂
Coolant pumping power	☹	☹	🙂	☹
Shielding efficiency/ n-streaming void space	☹	☹	🙂	🙂
Activation products in coolant (water)	🙂	🙂	☹	🙂
Breeding efficiency	🙂	☹	☹	☹
Tritium extraction from breeder	🙂	☹	☹	☹
Tritium extraction from coolant	🙂	🙂	🤔	🙂
Tritium permeation through heat exchanger	☹	☹	🙂	☹

G. Federici et al., 2013.

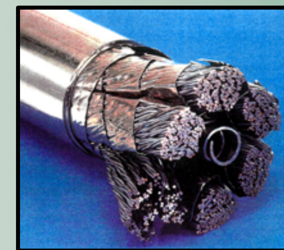
Significant gaps in bridging materials from ITER to DEMO
- virtually no materials systems currently used are reactor viable -

ITER <u>Lifetime</u> Neutron Fluence (nm^{-2})	Fusion Power Reactor <u>Annual</u> Neutron Fluence (nm^{-2})	
3.7E+21	5E+22	Blanket
5.1E+14	7E+15	Magnet
1.9E+21	2.6E+22	Divertor
1.1E+19	1.5E+20	Vacuum Vessel
3.4E+11	4.5E+12	Cryostat



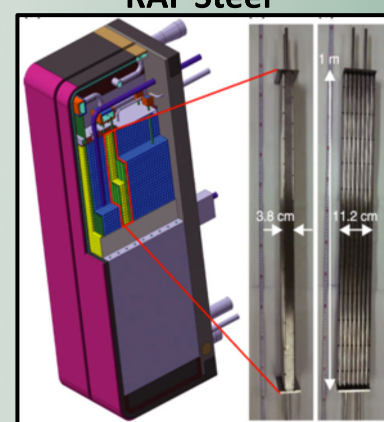
Zinkle & Snead, Ann. Rev. Mater. Res., <u>44</u> (2014) 241

Magnet



Diagnostic

ITER Test Blanket Module: RAF Steel



First Wall : Be-Cu alloy-316 steel

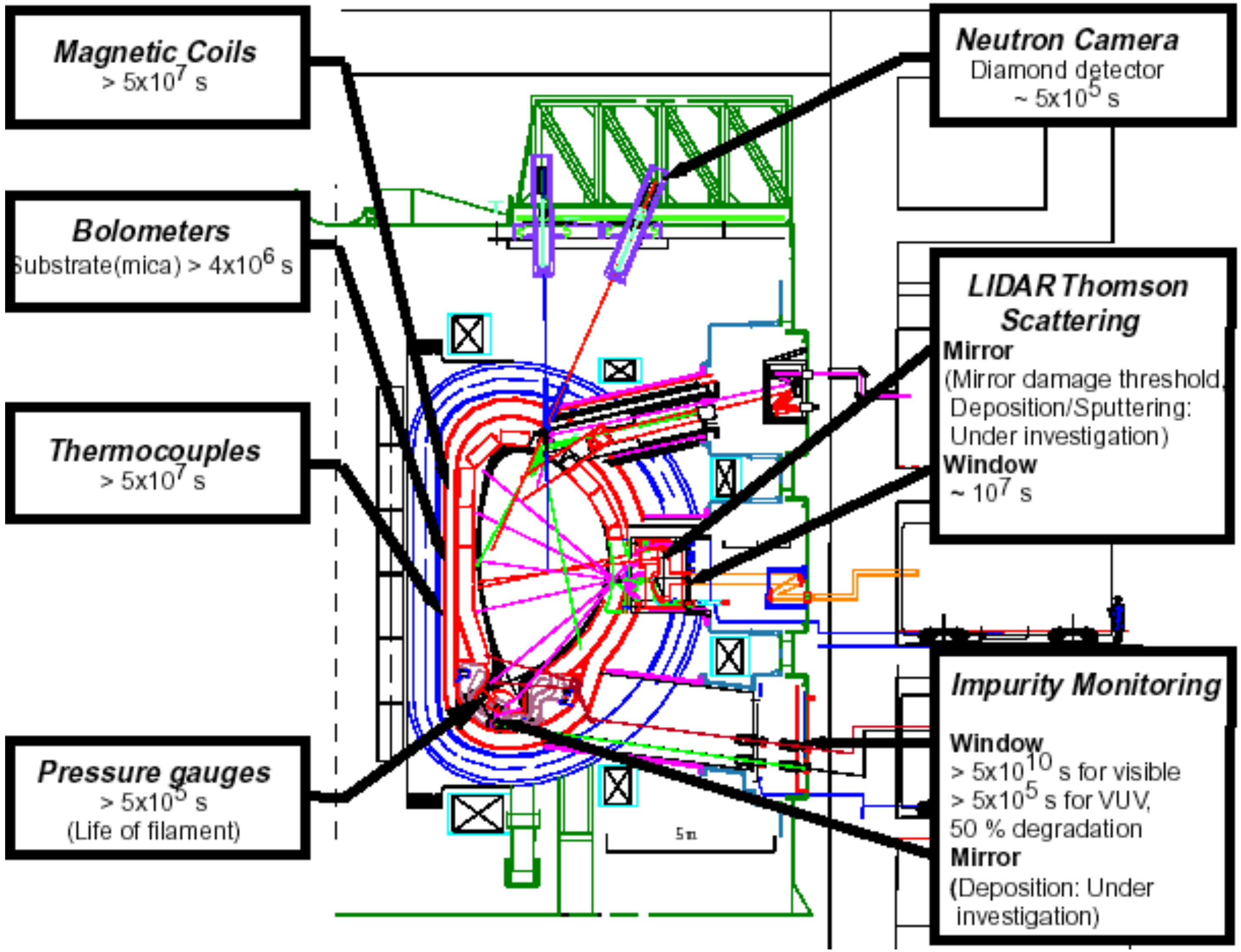
Divertor:W (or CFC)-Cu alloy-316 steel

Current estimated lifetime for some key diagnostics range from 1 week to 1 year

- neutron power density of 0.5 MW / m² or neutron power of 450 MW

- The fluence of 0.3 MWa / m² gives 2 x10⁷ s burn at 450 MW

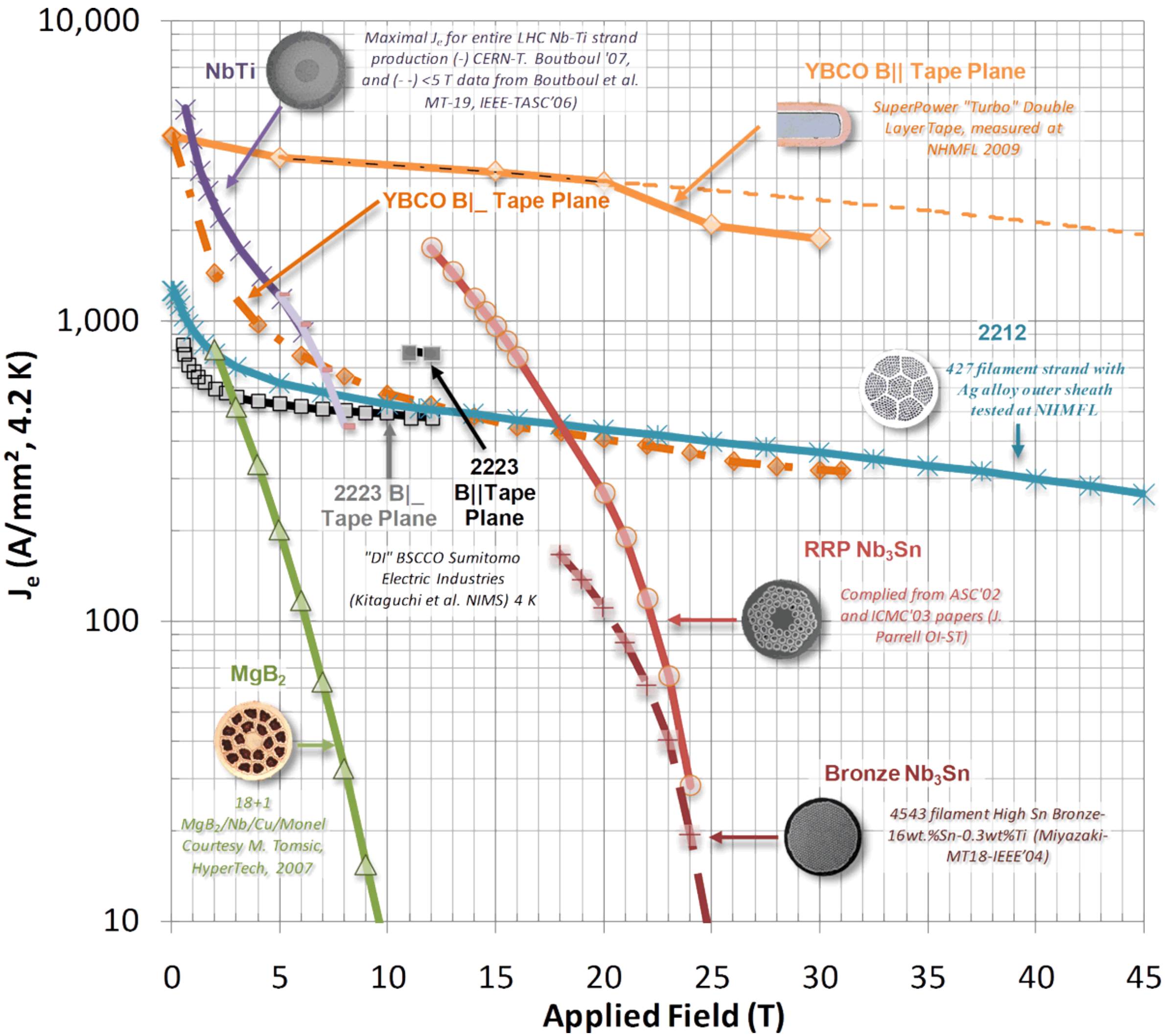
- The lifetime is mainly limited by radiation effects
- except where shown with ().



G. Vayakis, The ITER radiation environment for diagnostics, N 55 RI 38 04-05-06 W 0.1

From A. K.Ebisawa et al, Plasma Diagnostics for ITER-FEAT, HTPD 2000

High Temperature/Field Superconductors: J_e State of the Art



- HTS superconductors can extend the magnetic field range to >> 20 T.
- Stress will be a major concern at very high fields.
- Quench protection and stability have not been addressed.
- HTS wires and tapes are still primitive compared to Nb-Ti and Nb_3Sn (typical commercial batch lengths for YBCO are currently 50 – 150 m).
- MgB_2 is not promising for high field application and J_c is too low at > 20 K.
- Neutron-induced degradation needs further study but HTS appear to be comparable to conventional superconductor limits

Plot maintained at: http://magnet.fsu.edu/~lee/plot/plot.htm

Fusion materials challenges and opportunities

Increasing opportunities for leveraging broader mater. sci. community

- **Challenges**
  - Plasma facing components
    - Will tungsten work?
  - Tritium containment and online extraction/fuel reprocessing
  - Nonstructural materials lifetime in a DT fusion environment
    - Plasma diagnostics (optical fibers, electrical insulators, etc.)
    - Plasma heating feedthrough insulators
    - Next generation magnet systems (insulation, HT_C superconductors?)
    - Ceramic breeders
  - Structural materials
    - Is there a viable option beyond 5 $MW\text{-}yr/m^2$? (50 dpa)