Multimodal Options for Materials Research to Advance the Basis for Fusion Energy in the ITER Era

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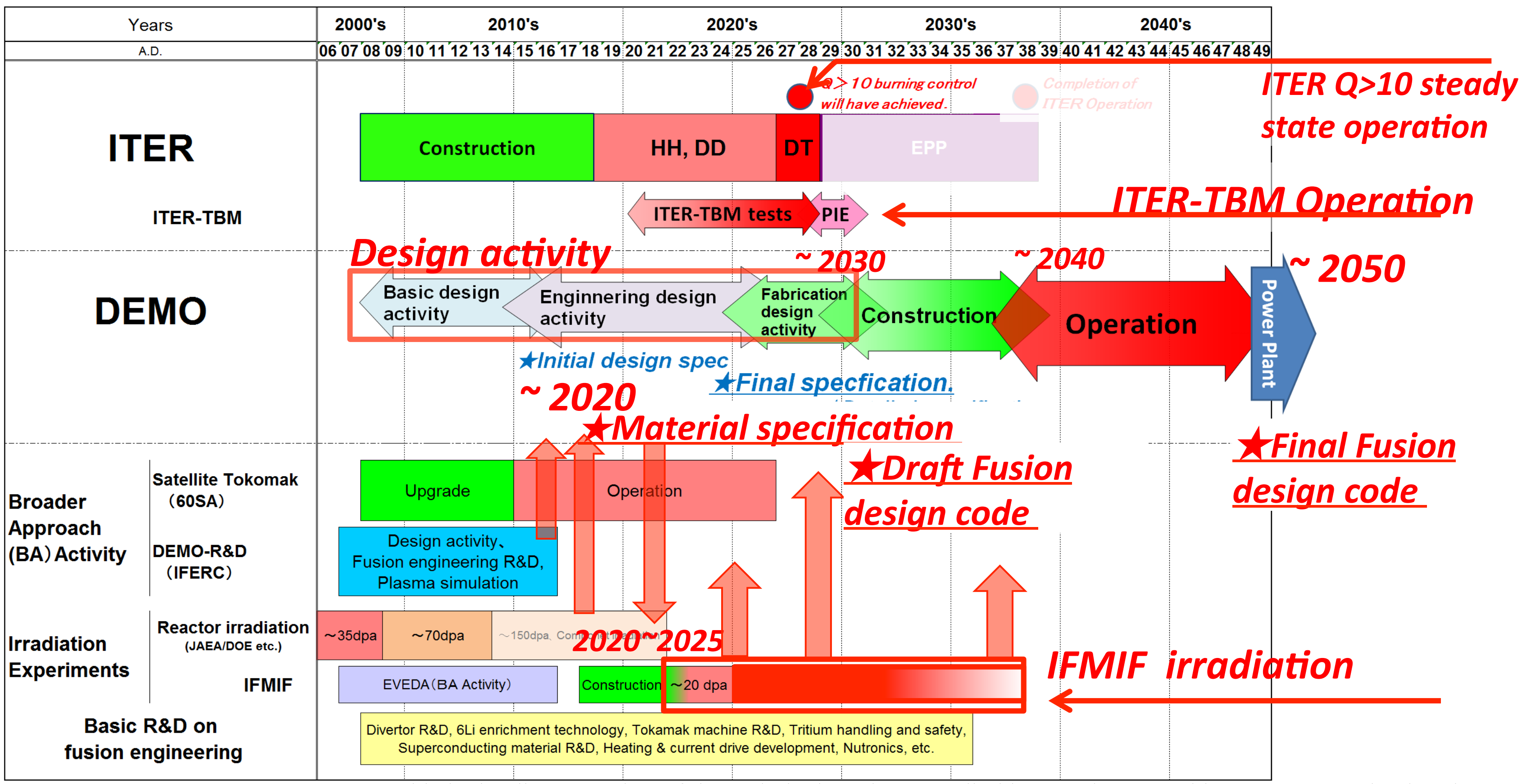
Why Multimodal Research Options?

- Although current international fusion energy roadmaps have many common elements, there are also some key differences
  - Time schedule, post-ITER facility R&D objectives (FNSF, DEMO, etc.), degree of aggressiveness in DEMO design, scientific discovery vs. engineering confirmation emphasis, etc.

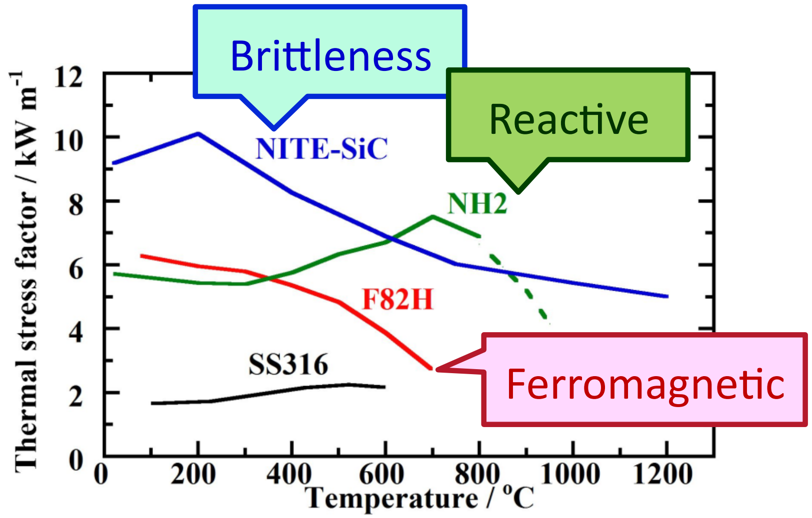
Japan fusion roadmap



Note: This is the digest version of the technical roadmap studied by Roadmap working group organized under ITER/BA technology advisory committee of Fusion energy forum of Japan (and further modified to include recent ITER schedule changes), and is NOT the roadmap authorized by Japanese government.

Why Multimodal Research Options? (cont'd)

- Every fusion blanket and materials system has shortcomings; there is no utopian solution
  - Multiple blanket systems are being explored in an effort to identify at least one viable blanket concept
  - E.g., 6 blanket concepts are being proposed for exploration on ITER (not including DCLL concept, etc.)



$$M = \frac{\lambda \sigma_{UTS}(1-\nu)}{\alpha E}$$

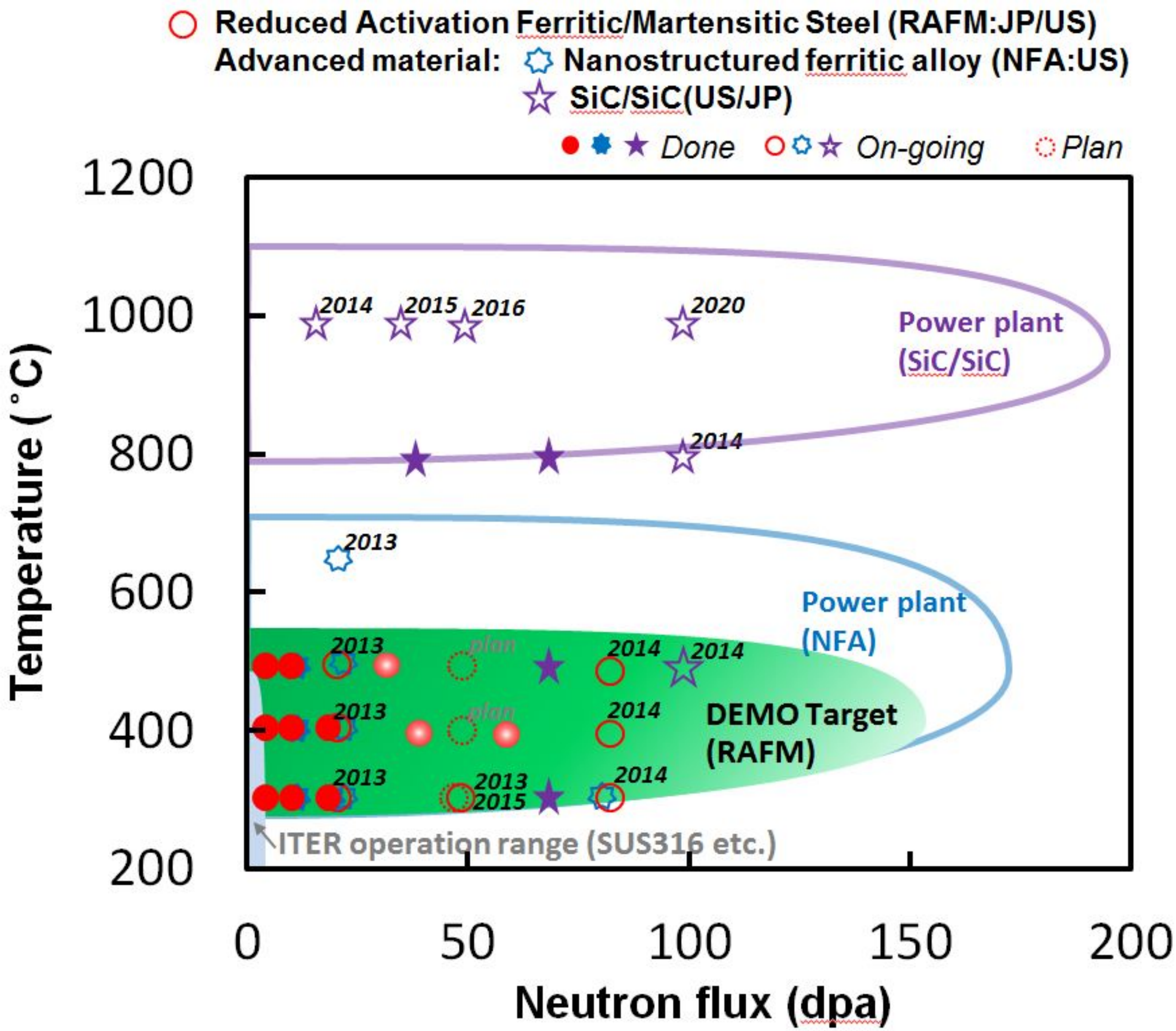
M : Thermal Stress factor
λ : thermal conductivity
σ : tensile strength
ν : Poisson ratio
α : thermal expansion coeff.
E : Young's modulus

(TL = TBM Leader)

Port Number	TBM-1 Concept	TBM-2 Concept
16	HCLL (TL : EU)	HCPB (TL : EU)
18	WCCB (TL : JA)	HCCR (TL : KO)
2	HCCB (TL : CN)	LLCB (TL : IN)

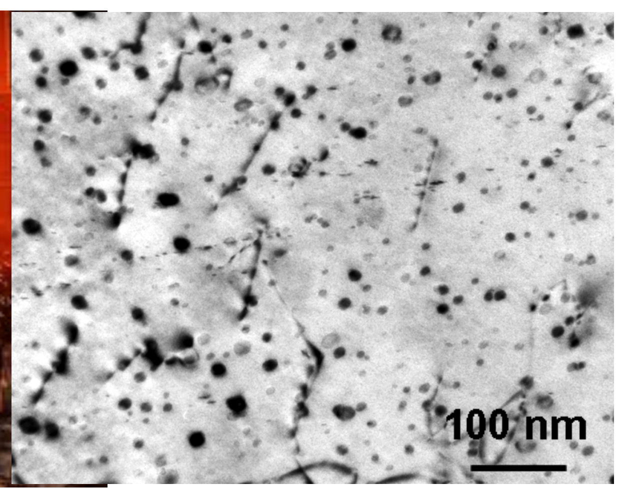
HCLL : Helium-cooled Lithium Lead, **HCPB** : He-cooled Pebble Beds (Ceramic/Beryllium)
WCCB : Water-cooled Ceramic Breeder (+Be), **HCCR** : Helium Cooled Ceramic Reflector
HCCB : He-cooled Ceramic Breeder (+Be), **LLCB** : Lithium-Lead Ceramic Breeder (He/LiPb)

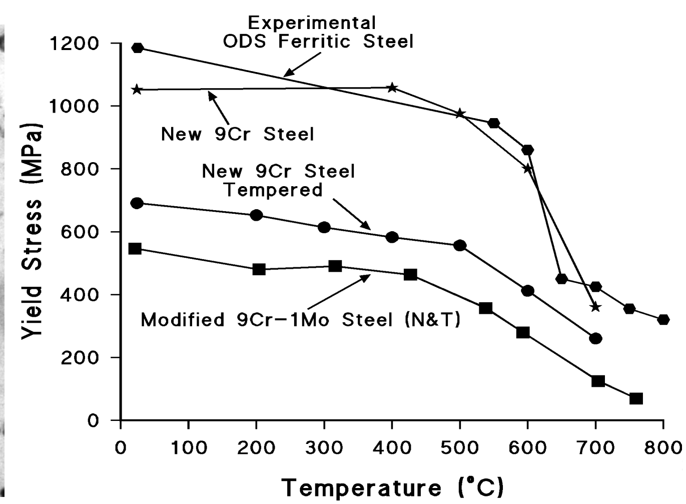
Parameter regimes under investigation in US/JP program on structural materials for fusion



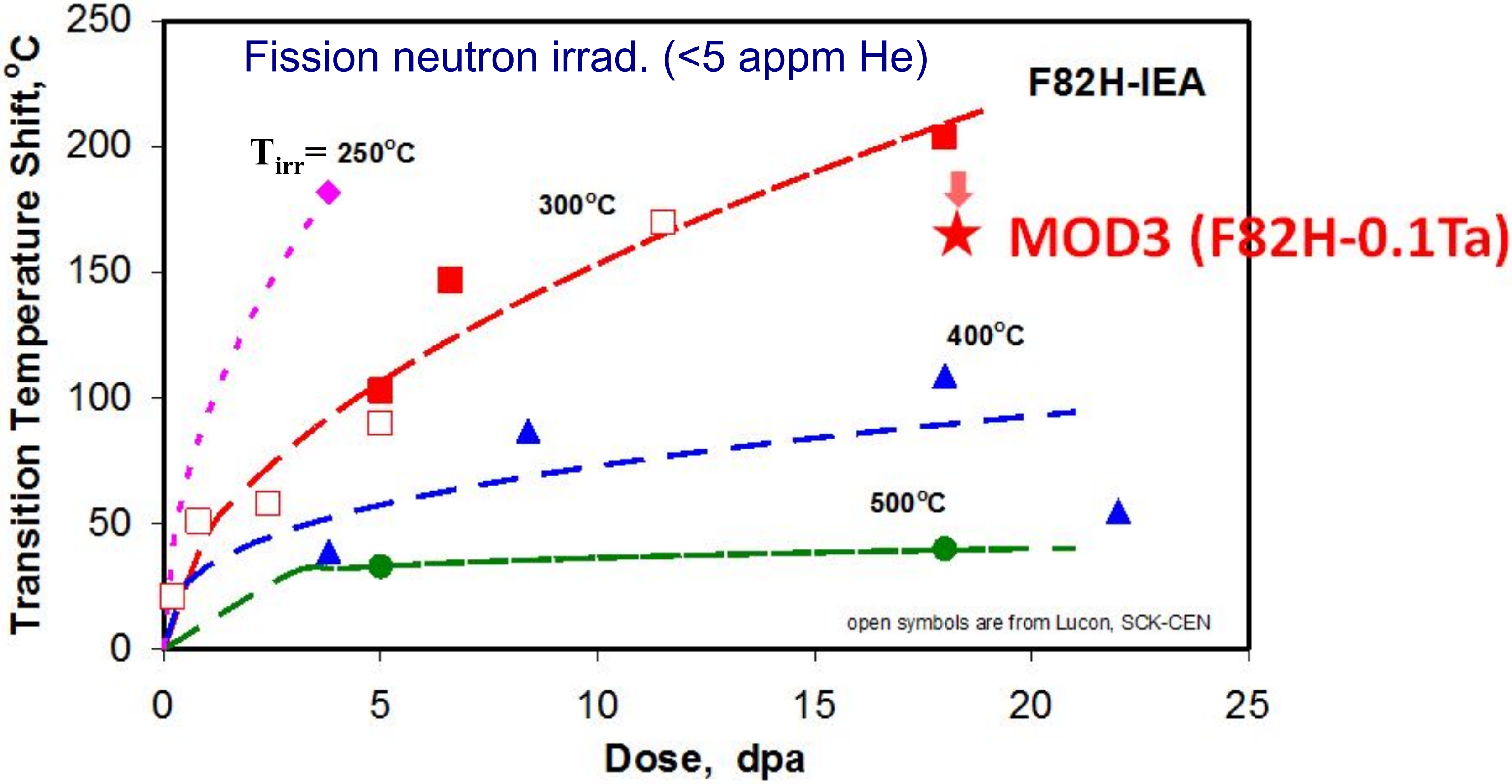
Current status and recent highlights for fusion materials

- Reduced activation ferritic martensitic steels are the leading fusion structural material option worldwide, due to good properties, generally favorable fission neutron irradiation resistance, and extensive industrial capability
  - Key uncertainties include ductile-brittle transition temperature (DBTT) increase due to fusion H, He effects and dose limits in fusion neutron environment
  - Risk mitigation options include oxide dispersion strengthened (ODS) steels and new ferritic/martensitic steels with a very high precipitate density designed with computational thermodynamics tools.





Provisional temperature and dose regimes for radiation-induced embrittlement of current fusion grades of ferritic/martensitic steels have been identified



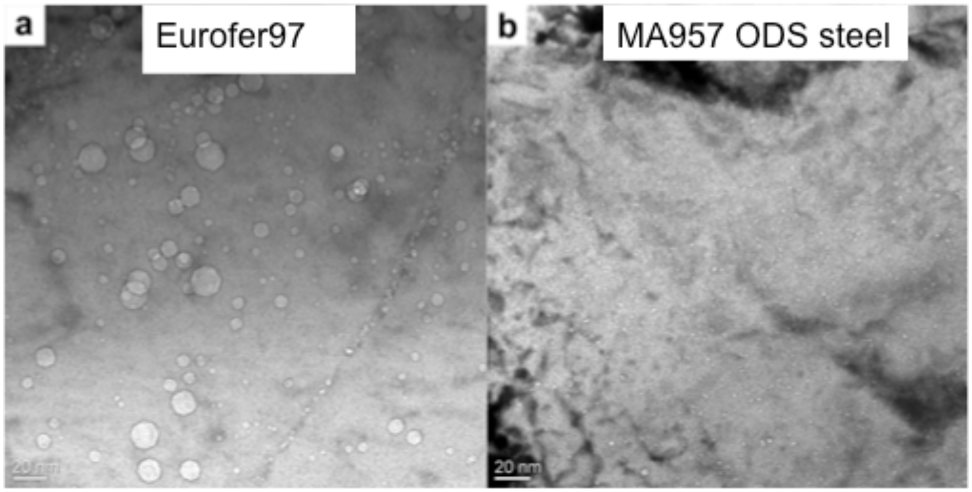
M.A. Sokolov ICFRM15 and H. Tanigawa

Lower operating temperature limit due to neutron embrittlement is ~300°C

Steels with modified thermomechanical treatment can offer slightly improved DBTT

Effect of fusion-relevant He production on ferritic/martensitic steels is being investigated using simulation techniques

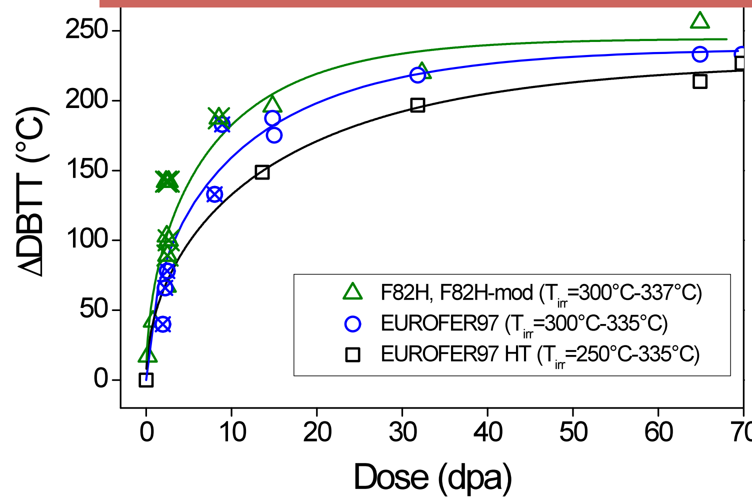
Significant void swelling observed in ferritic/ martensitic steel after 1400 appm He and 25 dpa at 500°C (56 appm/dpa)



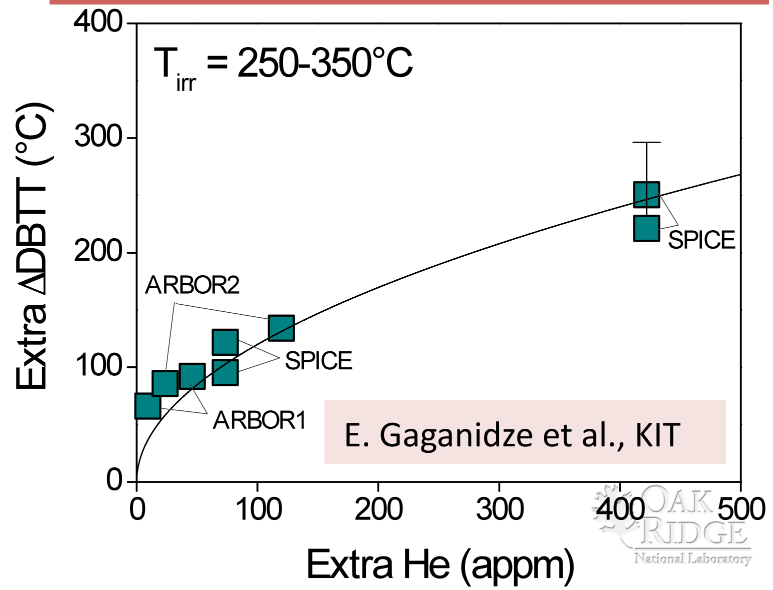
G.R. Odette (ICFRM-15); He injection from Ni foil during fission reactor irradiation

Good resistance to simulated fusion irradiation environment observed up to ~20 dpa
Open question: Are B-doping and He-injector (Ni foil) simulation tests prototypic for actual fusion reactor condition?

EUROFER, <10 appm He



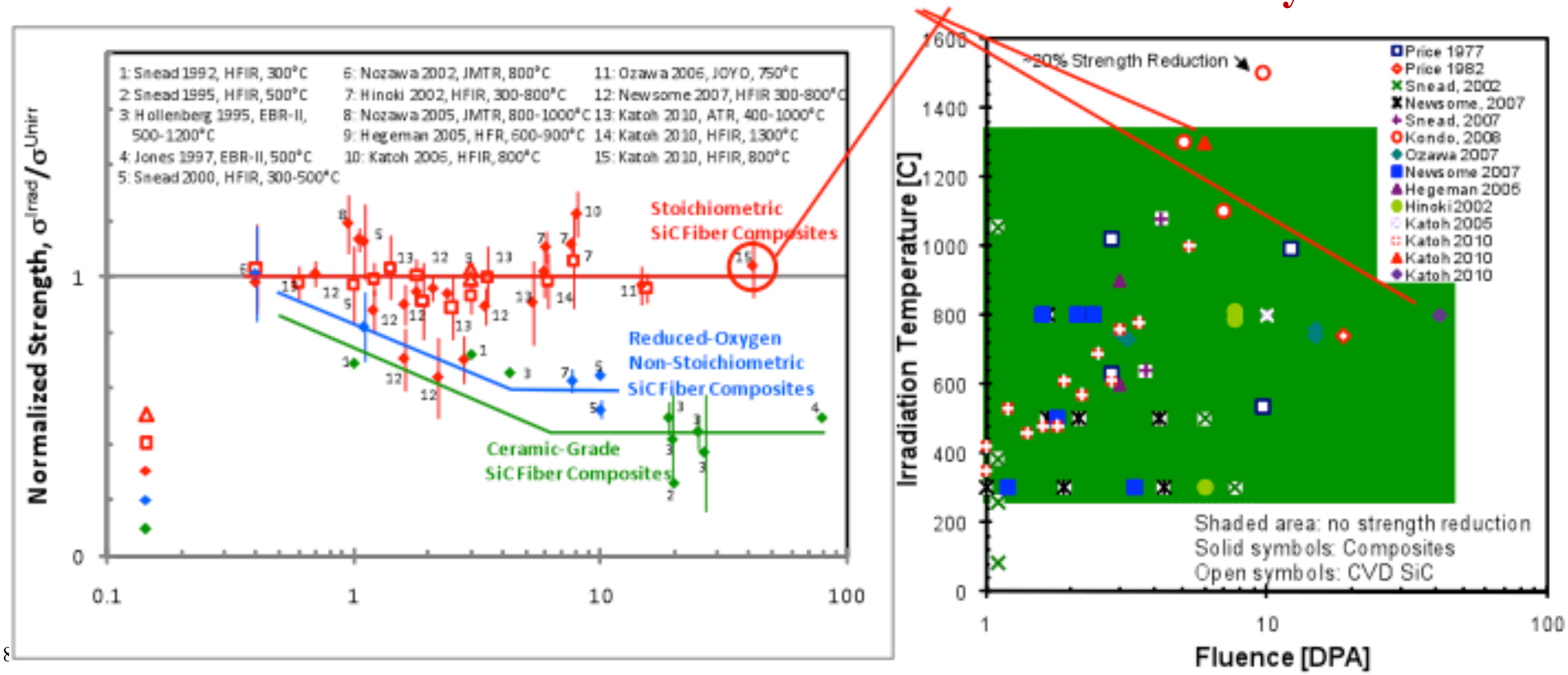
EUROFER, 10-500 appm He



Status and recent highlights: SiC/SiC composites

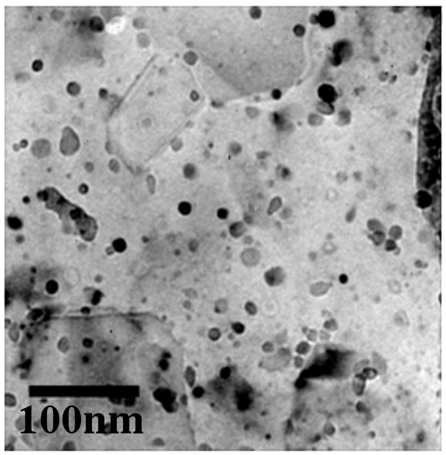
- Fission reactor irradiation stability up to 40 dpa recently confirmed for SiC/SiC composites at 800°C
- Scoping low dose irradiation studies on SiC joints found no degradation
- Outstanding issues include improvements in leak-tightness and fabrication (complexity and cost) and development of structural design criteria

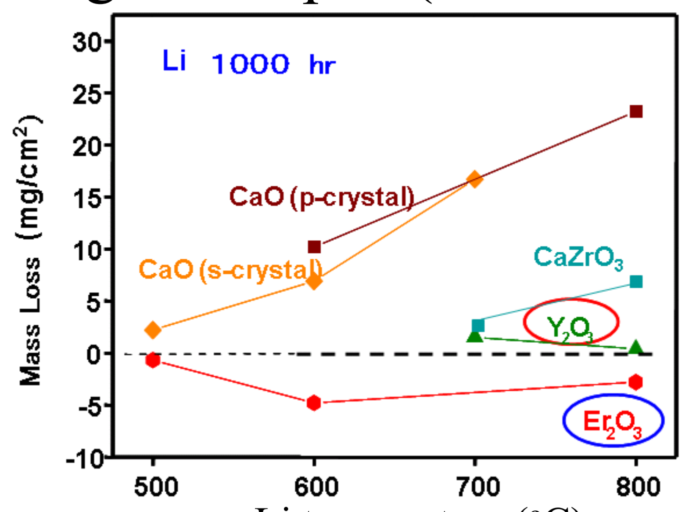
Data from recent US/J HFIR study

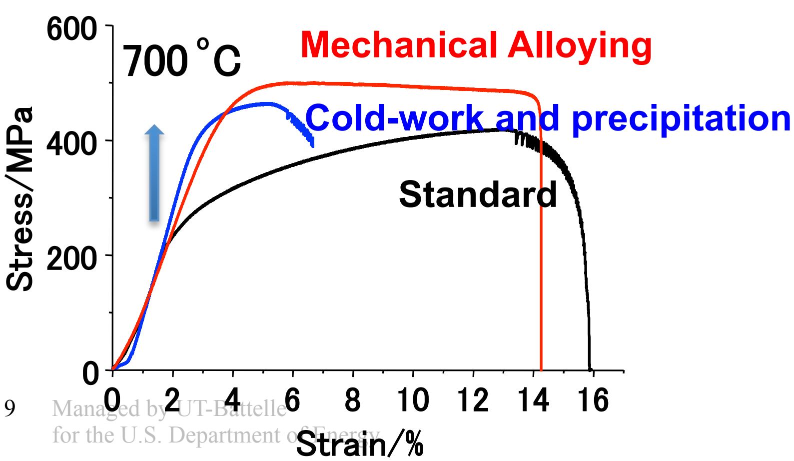


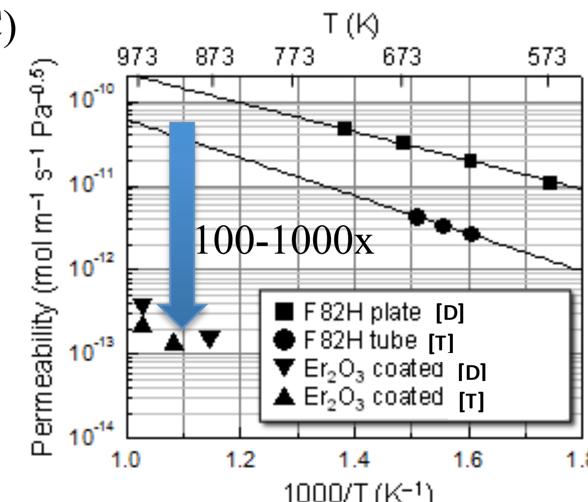
Status and recent highlights: V alloys and coatings

- Higher strength V alloys have been demonstrated using mechanical alloying approach
- New processes for fabricating Er2O3 and Y2O3 MHD insulator and T_2 barrier coatings are being developed (suitable for coating complex geometries)







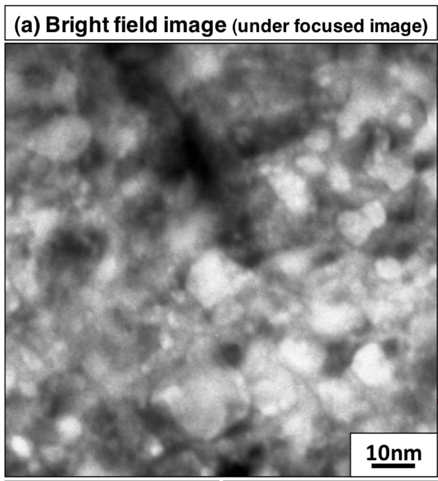


(Chikada, Shimada, J-US TITAN Program)

Status and recent highlights: Tungsten

- Hot wall operation introduces several new phenomena
  - e.g., Nanofuzz surface formation during plasma exposure

~ 600 - 700 K | ~ 900 – 1900 K | > 2000 K



PISCES-A: D_2-He plasma

M. Miyamoto et al. NF (2009) 065035

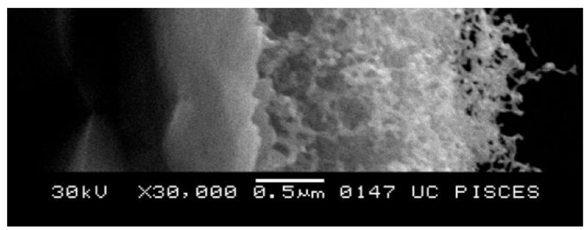
600 K, 1000 s, 2.0x10^{24} He$^+$/m^2, 55 eV He$^+$

- Little morphology
- He nanobubbles form
- Occasional blisters

PISCES-B: mixed D-He plasma

M.J. Baldwin et al, NF 48 (2008) 035001

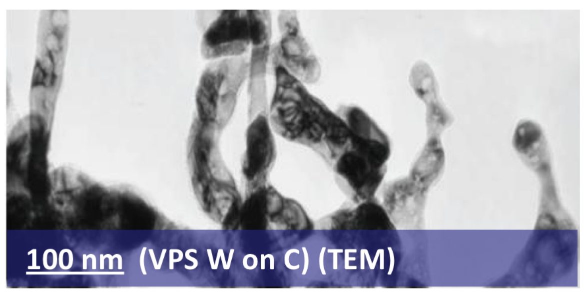
1200 K, 4290 s, 2x10^{26} He$^+$/m^2, 25 eV He$^+$



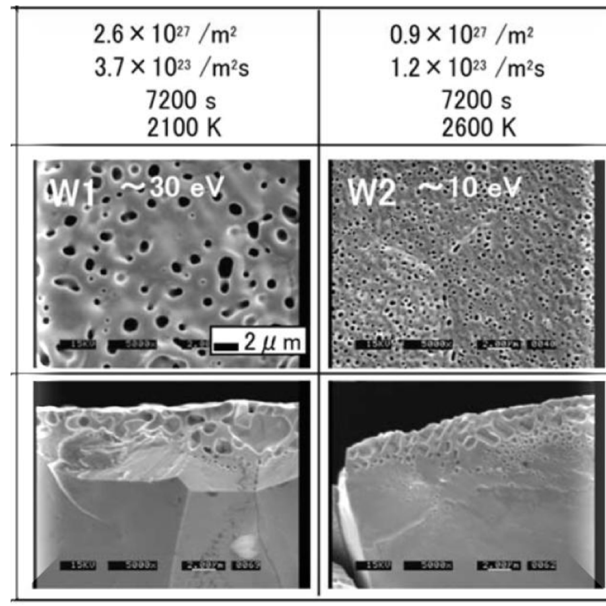
NAGDIS-II: pure He plasma

N. Ohno et al., in IAEA-TM, Vienna, 2006

1250 K, 36000 s, 3.5x10^{27} He$^+$/m^2, 11 eV He$^+$



- Surface morphology
- Evolving surface
- Nano-scale 'fuzz'



NAGDIS-II: He plasma

D. Nishijima et al. JNM (2004) 329-333 1029

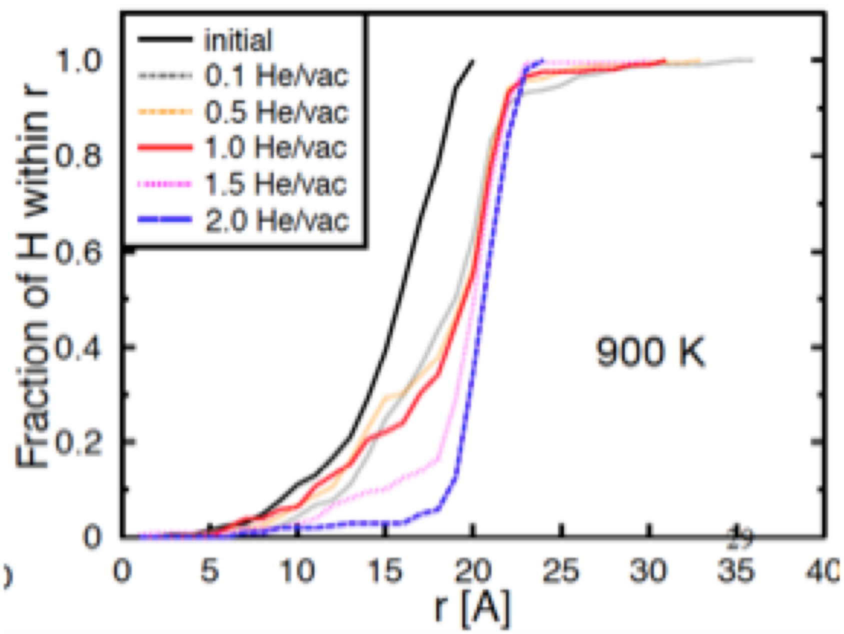
- Surface morphology
- Shallow depth
- Micro-scale

R.P. Doerner, UCSD

Status and recent highlights: Tungsten (cont'd)

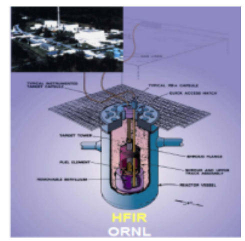
- Hot wall operation introduces several new phenomena
  - enhanced D/T retention after neutron irradiation (due to trapping at defect complexes)

Modeling

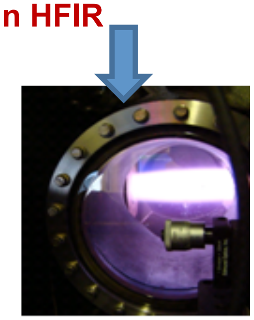


Calculated fraction of hydrogen that is trapped in the vicinity of a 2 nm radius He bubble in tungsten at 900 K (B.D. Wirth).

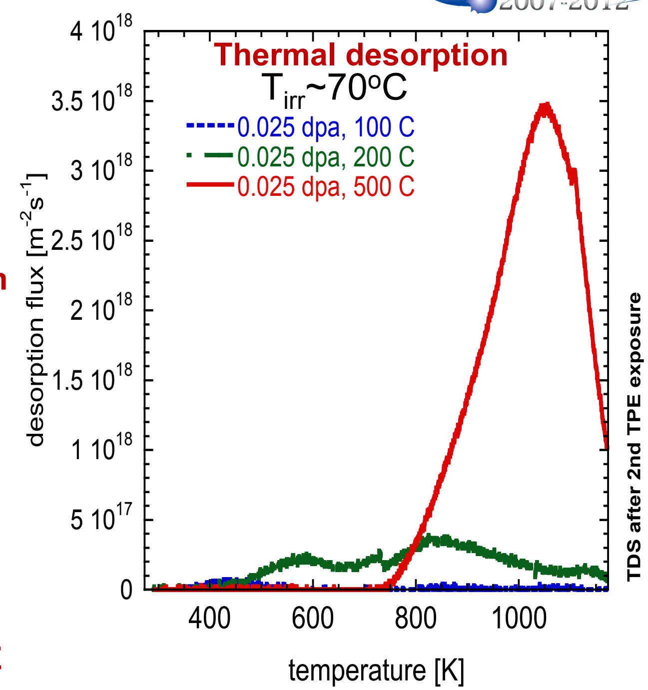
Experiment



Neutron irradiation in HFIR



D plasma exposure in TPE



Hatano et al. FTP 4-1 (Friday)

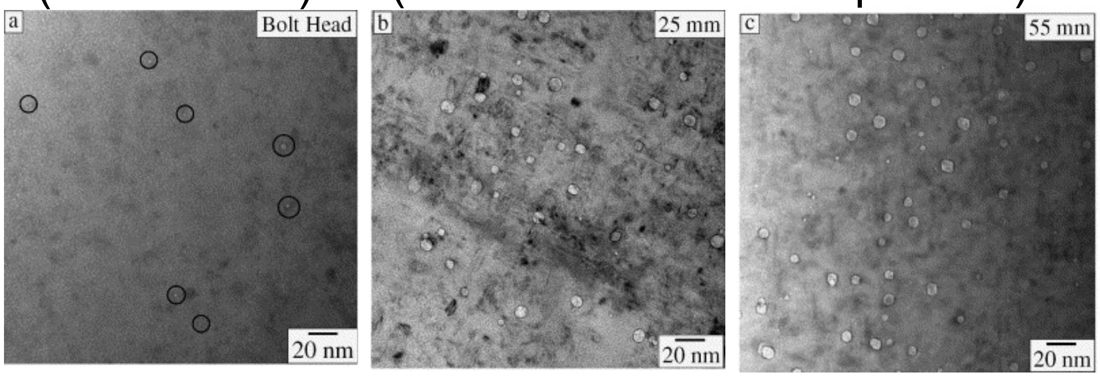
Desorption experiments on W neutron-irradiated at high temperature are scheduled to be performed in the near future

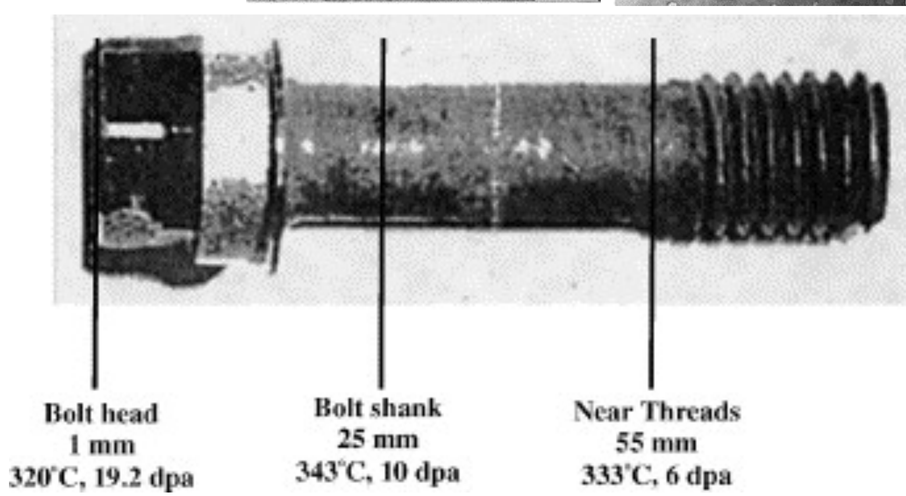
H retention increases dramatically in the presence of cavity formation

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

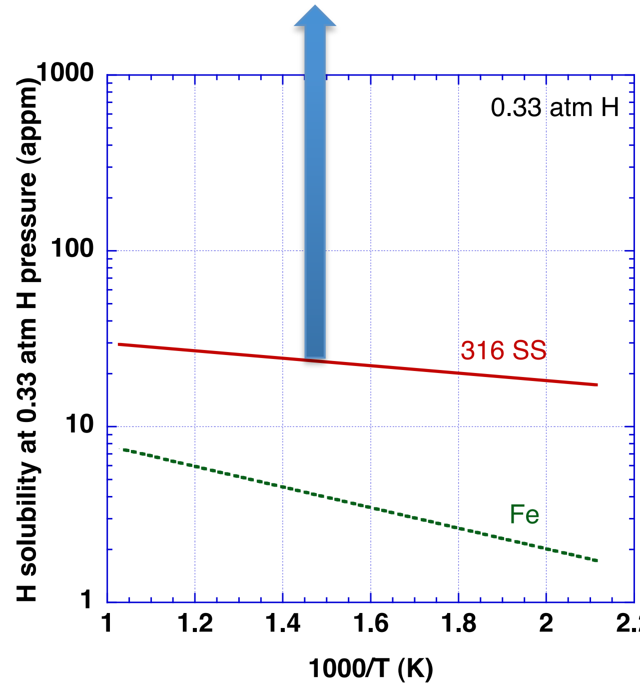
500-700 appm H (few cavities)

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





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

3 High-Priority Materials R&D Challenges

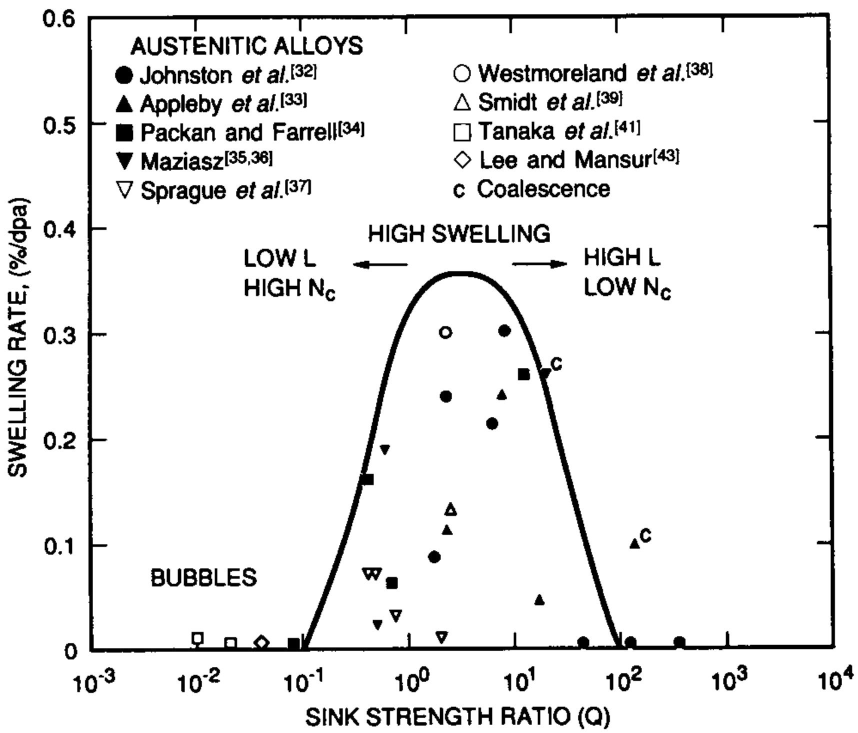
- Is there a viable divertor & first wall PFC solution for DEMO/FNSF?
  - Is tungsten armor at high wall temperatures viable?
  - Do innovative divertor approaches (e.g., Snowflake, Super-X, or liquid walls) need to be developed and demonstrated?
- Can a suitable structural material be developed for DEMO?
  - What is the impact of fusion-relevant transmutant H and He on neutron fluence and operating temperature limits for fusion structural materials?
  - Is the current mainstream approach for designing radiation resistance in materials (high density of nanoscale precipitates) incompatible with fusion tritium safety objectives due to tritium trapping considerations?
  - Is the reduced activation mandate too restrictive for next-step devices, considering that ITER will utilize materials that are not reduced activation?
  - Can recent advanced manufacturing methods such as 3D templating and additive manufacturing be utilized to fabricate high performance blanket structures at moderate cost that still retain sufficient radiation damage resistance?
- What range of tritium partial pressures are viable in fusion coolants, considering tritium permeation and trapping in piping and structures?
  - What level of tritium can be tolerated in the heat exchanger primary coolant, and how efficiently can tritium be removed from continuously processed hot coolants?

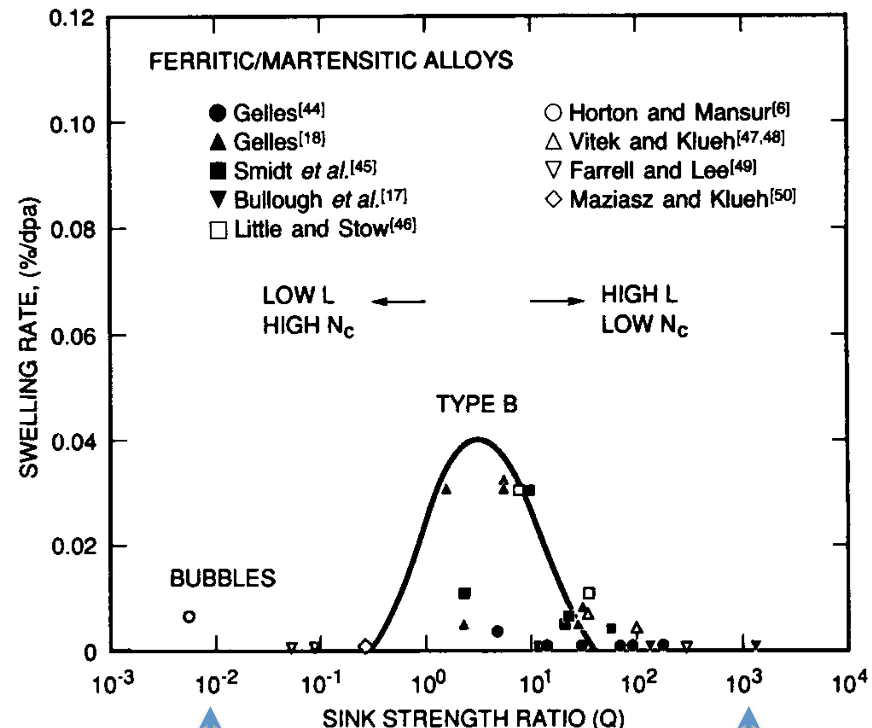
Urgency for a high-intensity fusion-relevant neutron source

- The second materials R&D challenge and parts of the 1st and 3rd R&D challenge listed above require an intense neutron source for their resolution.
  - Scientific studies of radiation degradation phenomena and tritium trapping issues in candidate HHF/blanket materials exposed to prototypical fusion operating environment.
- Obtaining engineering data from an intense fusion neutron source is a significant critical path item for DEMO design and licensing
  - Prioritize research on a limited number of DEMO material and blanket concepts (e.g., is ODS steel or another special material required?)

Void Swelling is typically maximized when the cavity and dislocation sink strengths are comparable

$$Q \sim \frac{\rho_d}{4\pi R N_{cav}}$$





High He/dpa ratio (e.g., spallation neutrons)

Moderate He/dpa ratio (e.g., DT fusion neutrons)

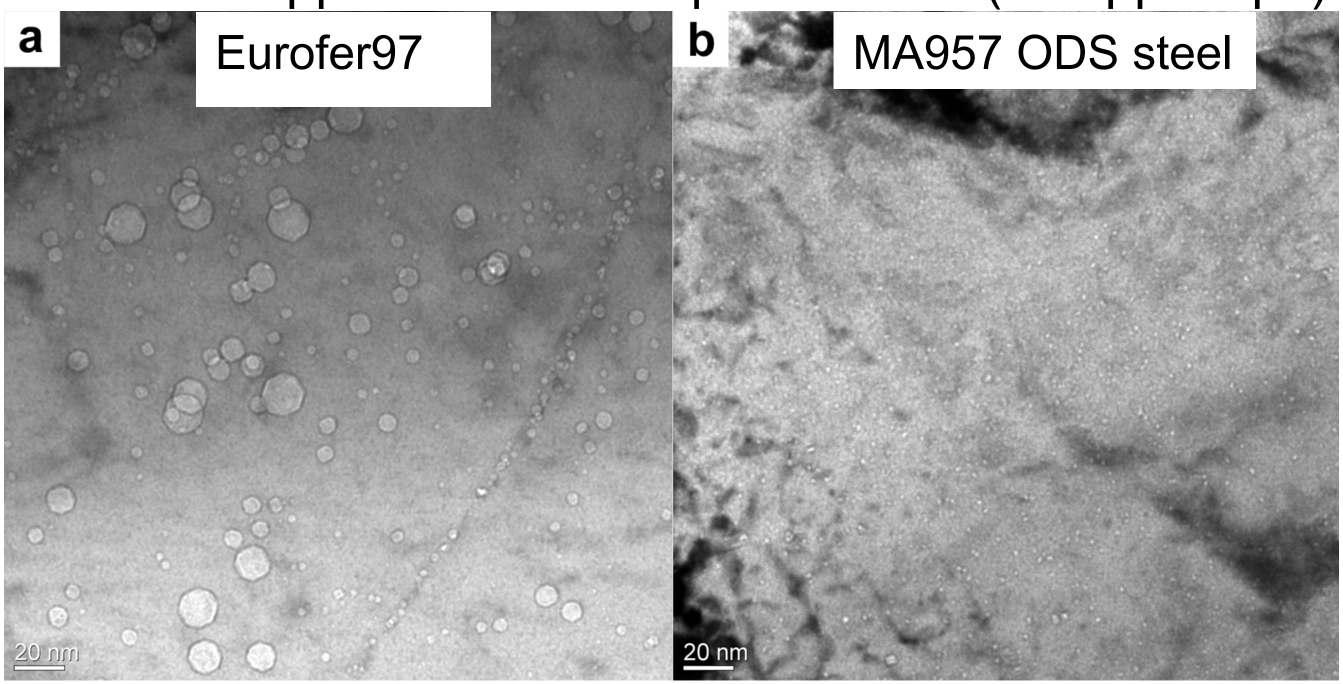
low He/dpa ratio (e.g., fission neutrons)

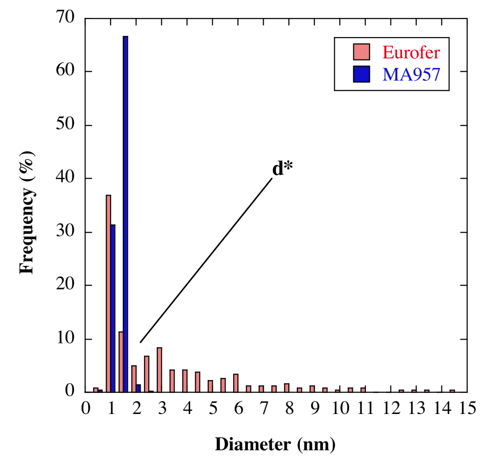
E.H. Lee & L.K. Mansur, Met. Trans. A 21 (1990) 1021

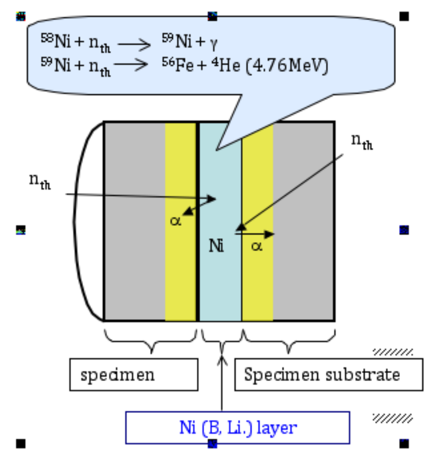
Recent in situ He injector study during fission reactor (HFIR) irradiation suggests void swelling may emerge as an issue at ~25 dpa for ferritic/martensitic steels

- MA957 (ODS steel) and Eurofer97 9%Cr ferritic/martensitic steel
- Eurofer97: 7.5x10^{22} cavities/m^3 with bimodal size distribution (1.3 nm bubbles & 5 nm voids - precursor to significant swelling)
- MA957: 7.8x10^{23} bubbles/m^3 & no voids

1400 appm He and 25 dpa at 500°C (56 appm/dpa)





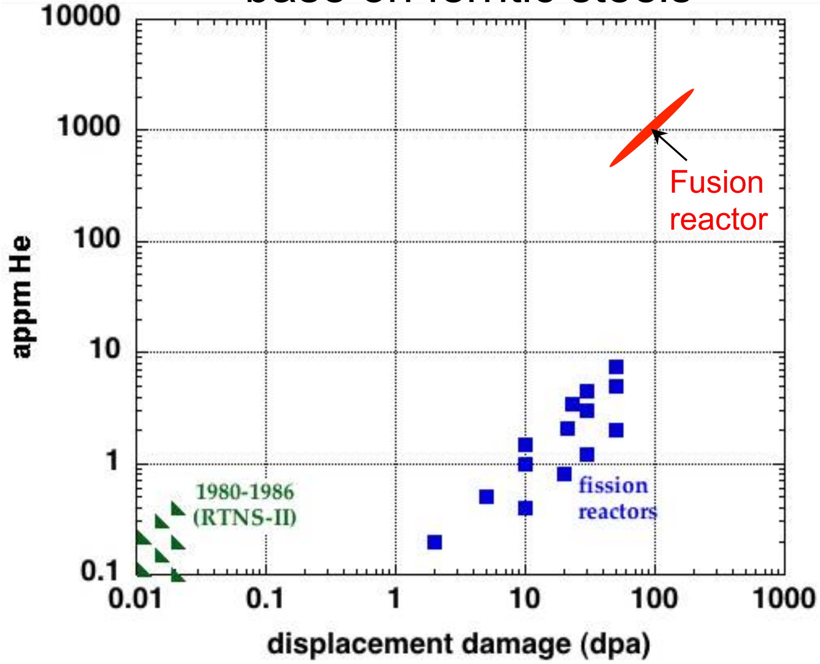


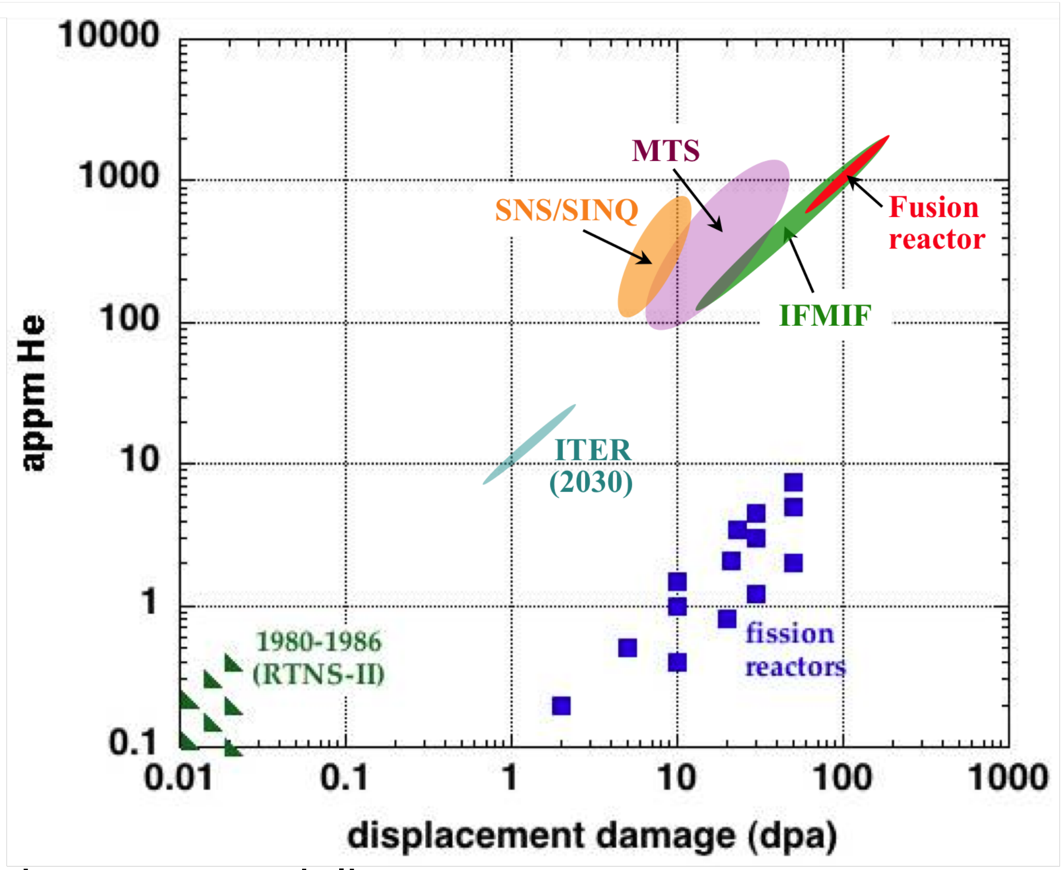
Odette et al. ICFRM-15, Charleston, South Carolina

There are several options to close the current knowledge gap in fusion-relevant radiation effects in materials

- An intense neutron source (in concert with enhanced theory and modeling) is needed to improve understanding of basic fusion neutron effects and to develop & qualify fusion structural materials

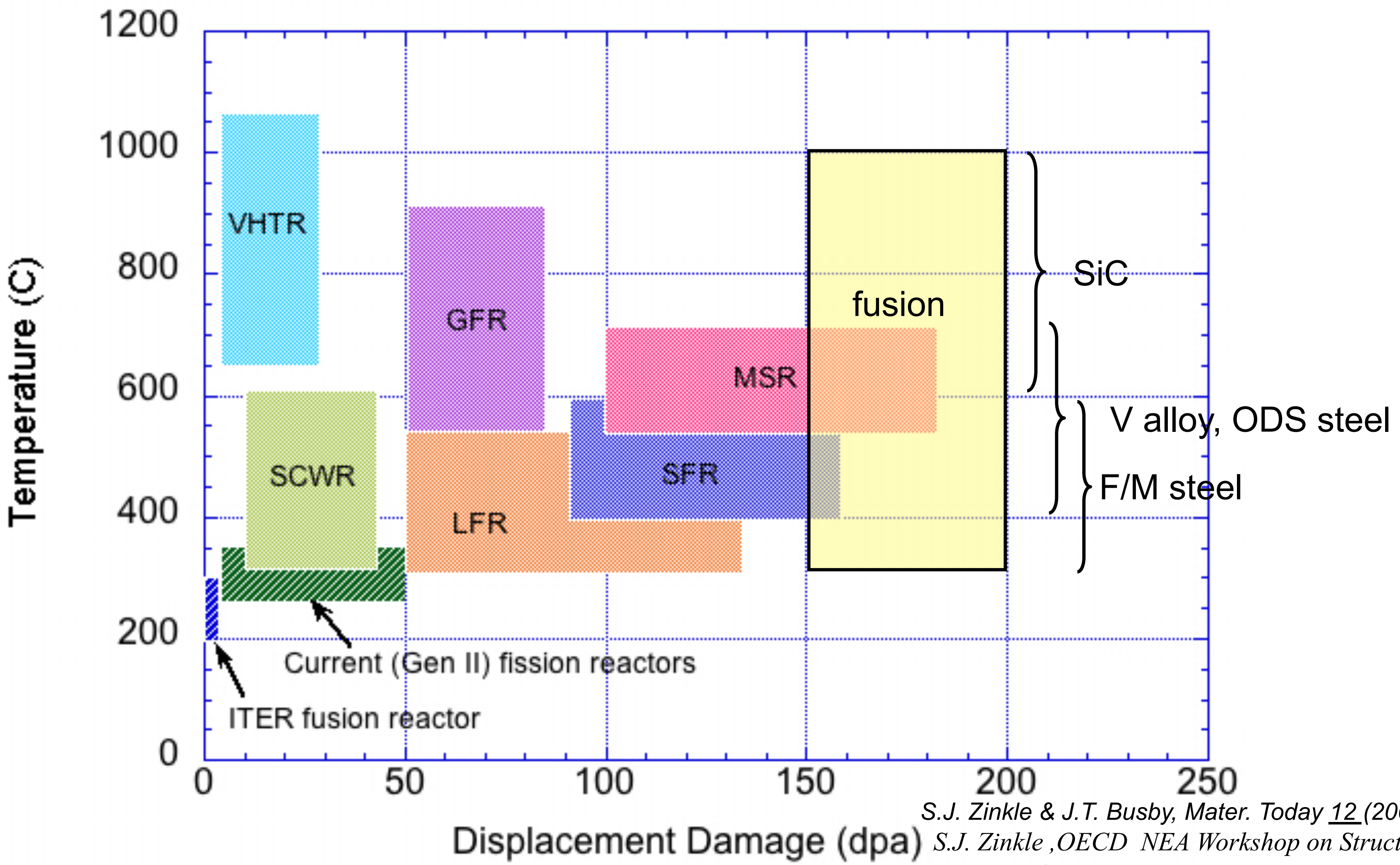
Current knowledge base on ferritic steels





Option A: IFMIF + fission reactors +ion beams + modeling
Option B: robust spallation (e.g., MTS) + fission reactors + ion beams + modeling
Option C: modest spallation (e.g.,SNS/SINQ) + fission reactors + ion beams + modeling

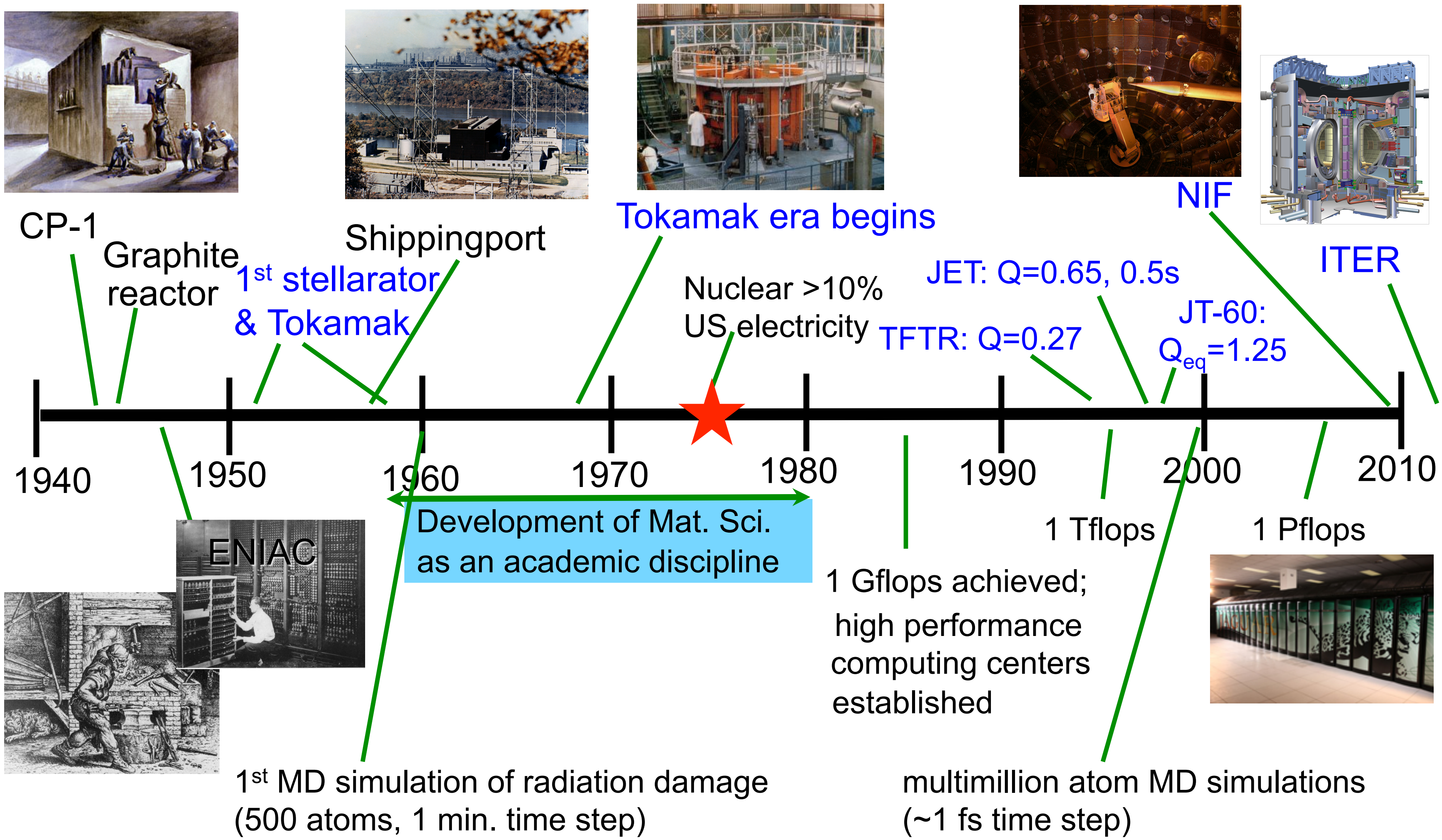
Comparison of Gen IV and Fusion Structural Materials Environments



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

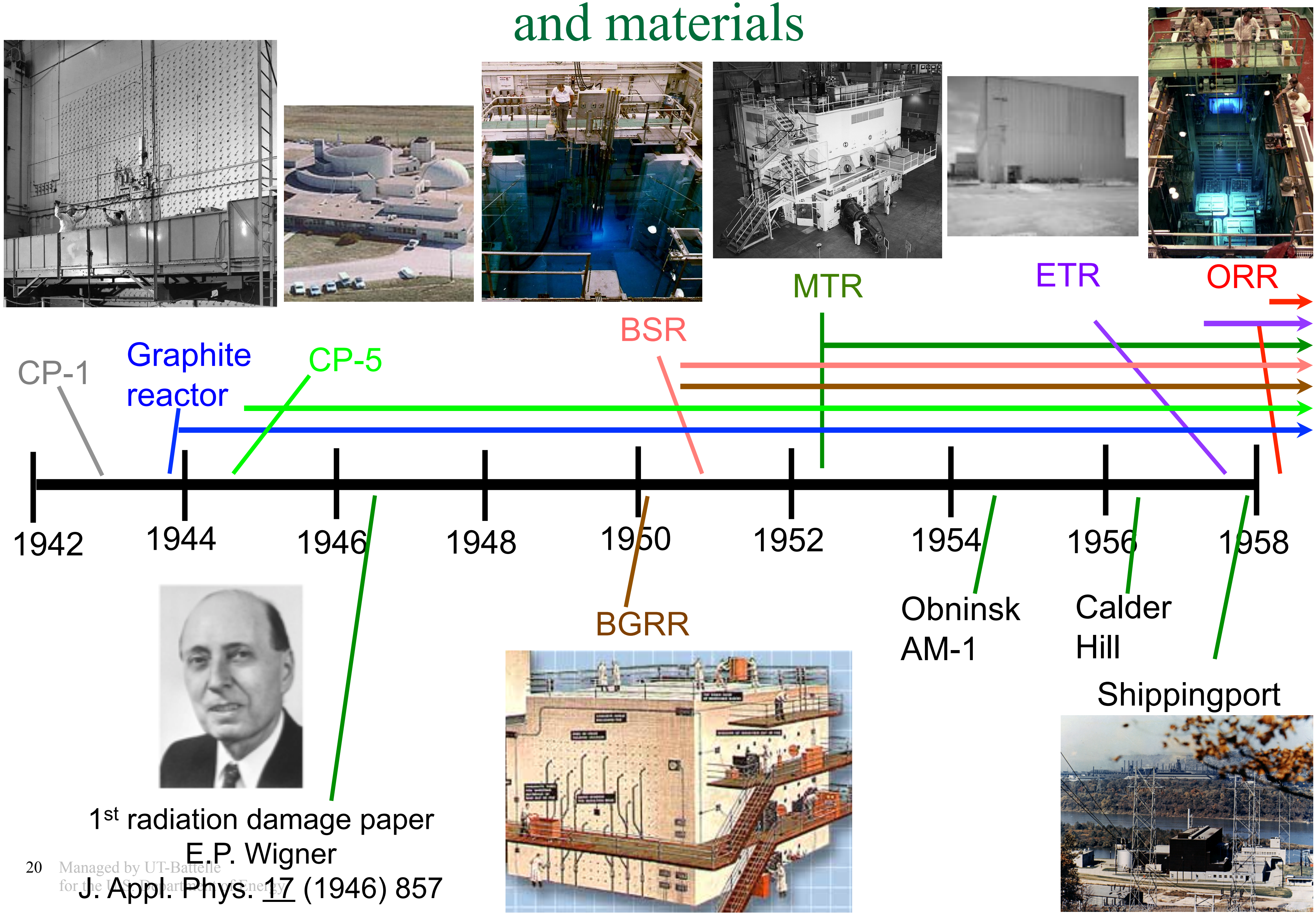
All Gen IV and Fusion concepts pose severe materials challenges

Timeline of some key events for nuclear energy and materials and computational science



Oak Ridge National Laboratory

Detailed timeline of some key facilities for nuclear energy and materials



Contribution of major facilities to Materials degradation science and technology issues

Red: TRL 1-3 issues
Yellow: TRL 4-6 issues
Green: TRL 7-9 issues

Ion & fission irrad. | ITER-TBM | Non-nuclear test stands | Fusion-relevant neutron source | FNSF | Demo

Facility	Non-nuclear Test Stands (thermo-mechanical)	Non-nuclear Test Stands (corrosion)	Ion beams and Fission Reactors	ITER TBM	Non-nuclear Test Stands (partially integrated)	Fusion Relevant Intense Neutron Source	Fusion Nuclear Science Facility	DEMO
First-Wall/Blanket Structural & Vacuum Vessel Materials								
Science-based design criteria (thermo-mechanical strength)	2. Develop high temperature creep-fatigue design rules for nuclear components			4. Proof test verification of blanket module low-dose performance	4. Validate high temperature creep-fatigue design rules w/o irradiation	5. Validate irradiated high temp structural design criteria (50-150 dpa with He, stress)	7. Code qualified designs	7-8. Code qualified designs
Explore fabrication & joining tradeoffs	2. Conventional & advanced manufacturing technologies	2. Loop tests of joints & novel fabrication approaches	2. Rad. stability of joints & novel fabrication approaches	5. Fabricate blanket modules using DEMO-relevant methods	5. Validate near prototypic fabrication and joining technology w/o irradiation	6. Validate near-prototypic fabrication & joining technology (50-150 dpa with He, stress)	7. Demo-relevant fab processes	8. Prototypic advanced fabrication
Resolve compatibility & corrosion issues		3. Basic and complex flow loops			5. Validate corrosion models w/o irradiation		7. Near prototypic operating environment	8. Prototypic extended operating environment
Scientific exploration of fundamental radiation effects in a fusion relevant environment			3. Up to 150 dpa/With He, stress (ion beams, fission reactors)			6. 50 - 150 dpa/With He and stress		
Material qualification: Structural stability in fusion environment (e.g., void swelling, irradiation creep)			3. Up to 70 dpa/no He (fission reactors)	3. Materials behavior in a low-dose env. (Demo-relevant matl. & T <2 dpa)		6. 50 - 150 dpa/With He and stress	7. 10 - 50 dpa, Demo prototypic environment	7-8. Prototypic operation, 50 - 150 dpa/With He/Fully Integrated
Material qualification: Mechanical integrity in fusion environment (e.g., strength, rad resistance, lifetime)	2. Unirrad. mech. prop. data (tensile, creep, fatigue, fract. toughness, da/dN, etc)		3. Up to 70 dpa/no He (fission reactors)	5. Materials behavior in a low-dose fusion env. (Demo-relevant matl.,stress and Temp., <2 dpa)	5. Qualify components w/o irradiation	6. 50 - 150 dpa/With He and stress	7. 10 - 50 dpa, Demo prototypic environment	7-8. Prototypic operation, 50 - 150 dpa/With He/Fully Integrated
Fusion environment effects on tritium retention & permeation		2. Unirradiated diffusion and permeation data	3. Effect of radiation damage at Demo-relevant temperatures	5. Post-irrad. evaluation may provide very useful low-dose info		6. Demo-relevant materials (up to 50-150 dpa with He at correct temp.)	7. System-scale tritium permeation and loss mechanisms	7-8. Prototypic permeation & losses

after DOE/SC-0149 (2012); ITER TBM column revised/corrected by materials degradation subpanel

Conclusions

- Substantial progress continues to be made in understanding and developing high-performance fusion structural materials.
  - Ferritic/martensitic steels appear to be suitable for fusion doses up to ~20 dpa
  - Higher performance options (e.g., ODS steels) may offer significantly better radiation resistance, but joining technology and others issues need to be resolved.
- In order to accelerate the pace for developing practical fusion energy, the construction of an intense fusion neutron source is needed.
  - Tritium retention (fusion safety)
  - Viability of blanket and first wall structural materials
  - Engineering database activities for design and licensing purposes, when a viable candidate has been identified