A Development Strategy for Fusion FW&B Materials and Structures

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Oak Ridge National Laboratory

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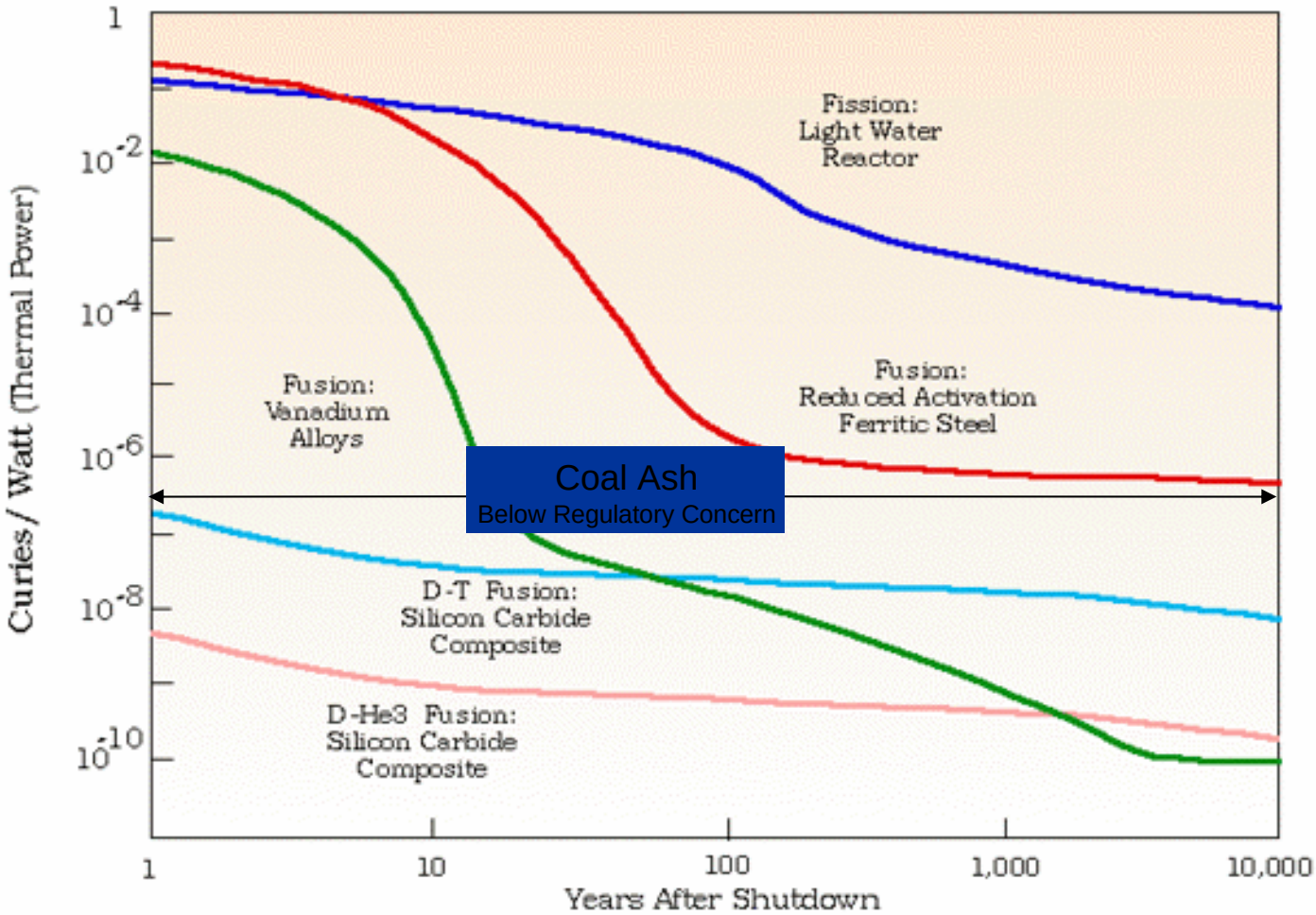
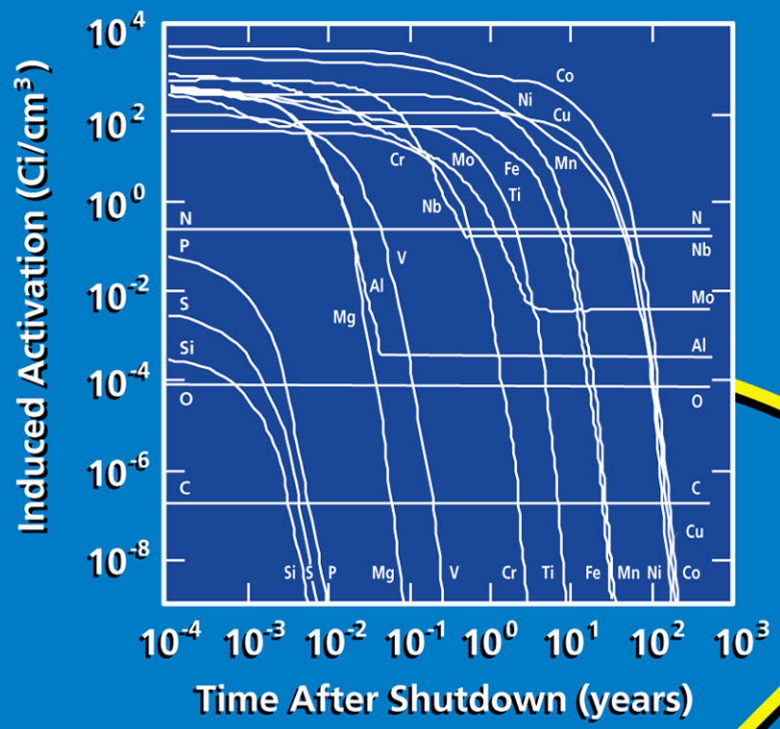
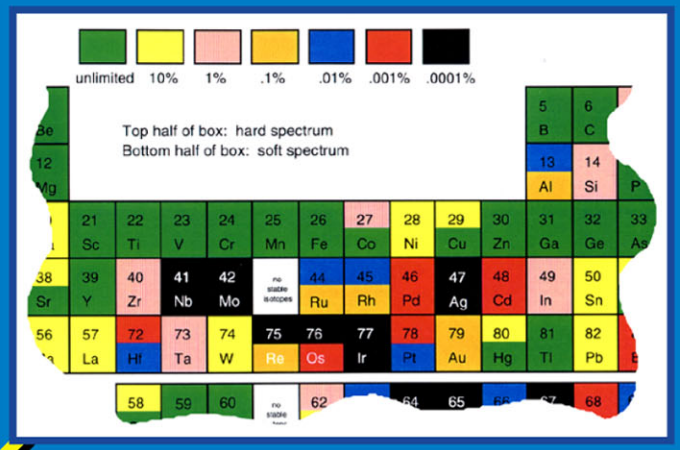
Comparison of Fission and Fusion Radioactivity after Shutdown

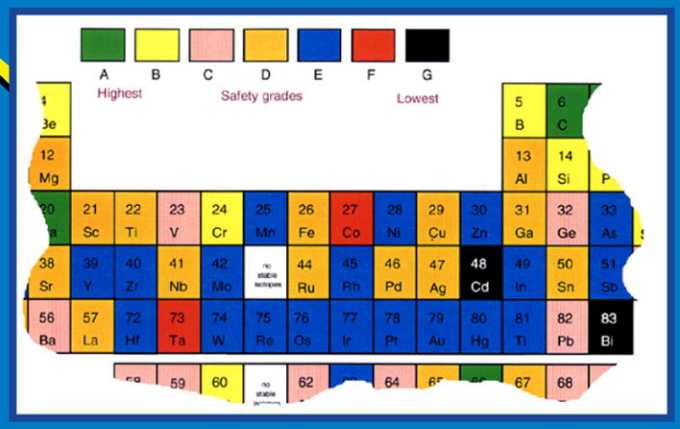


Structural Materials





Class C Waste Disposal



Safety

SiC

V Cr Ti Si

Fe Cr W V Ta

Structural Materials will Strongly Impact the Technological Viability, Safety, and Economics of Fusion Energy

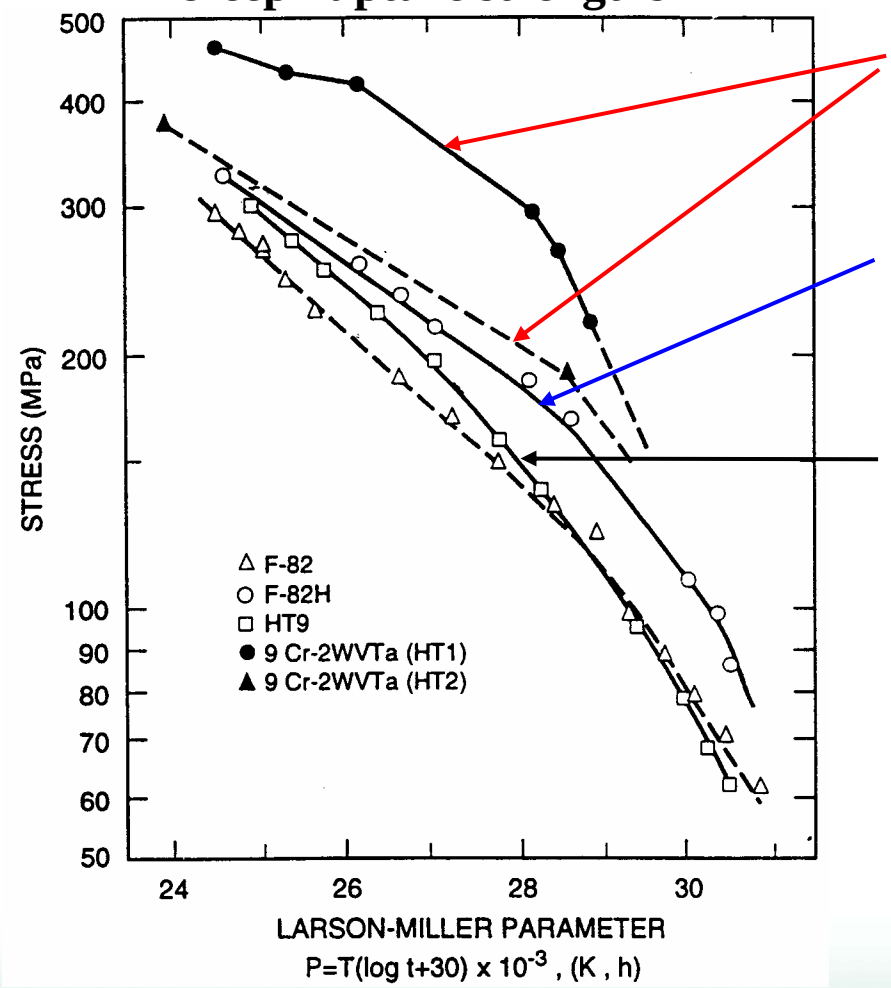
- Key issues include thermal stress capacity, coolant compatibility, safety, waste disposal, radiation damage effects, and safe lifetime limits
- The 3 leading candidates are ferritic/ martensitic steel, V alloys, and SiC/SiC (based on safety, waste disposal, and performance considerations)
  - Commercial alloys (Ti alloys, Ni base superalloys, refractory alloys, etc.) have been shown to be unacceptable for fusion for various technical reasons

Summary of several recent fusion energy blanket concepts

Structural Material	**Coolant/Tritium Breeding Material**					
	Li/Li	He/PbLi	H_2O/PbLi	He/Li ceramic	H_2O/Li ceramic	FLiBe/FLiBe
Ferritic steel						
V alloy						
SiC/SiC						

Reduced Activation Ferritic/martensitic Alloys Developed by Fusion have Properties Comparable or Superior to Commercial (high-activation) Alloys

Comparison of thermal creep-rupture strengths



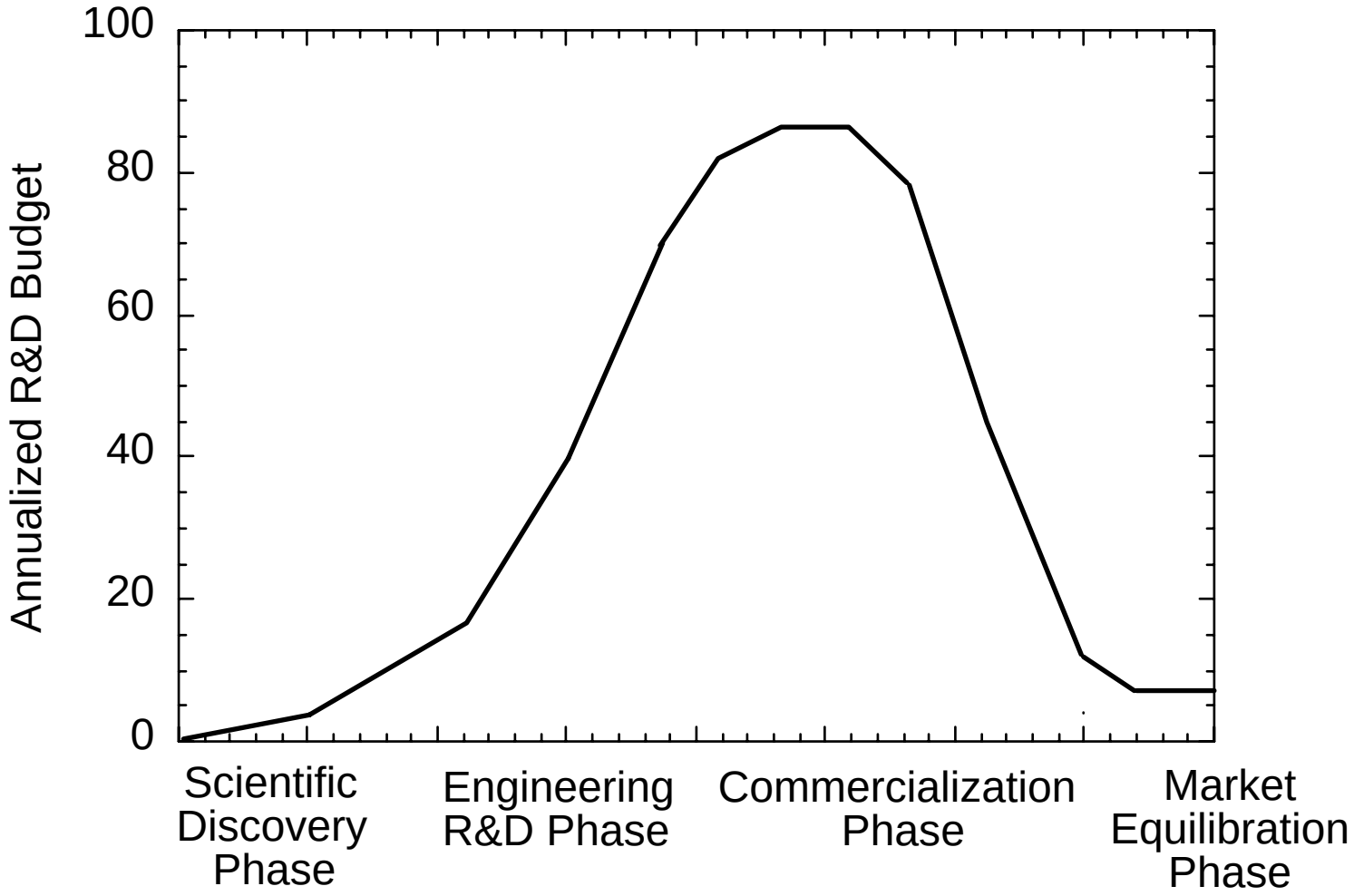
reduced activation steels (small heats)

IEA fusion reduced activation steel

Commercial ferritic steel

Fusion-developed steels also have superior tensile strength, fracture toughness, and thermal conductivity

R&D strategy for introducing new technologies involves an evolution from scientific discovery to applied research



Comparison of R&D approaches

Fundamental Research	Applied Development
	Select reference alloy and blanket concept (e.g. V4Cr4Ti/Li)
Focus research on fundamental understanding of BCC materials and SiC	Focus most research on this alloy/blanket system (narrow scope)
Research planning based on theory and modeling requirements of BCC materials	R&D planning based on need to develop materials to meet device construction schedule
Experiments used to guide and validated scientific theory and models	Experiment-based R&D to build engineering data base
Study model BCC alloys/ surrogate materials encompassing V alloys	

Key Cross-cutting phenomena for Fusion Structural Materials

Mech. props

F/MS	V	Cu	SiC	Phenomena, Issues, Comments
***	***	***	-	hardening and nonhardening embrittlement including underlying microstructural causes and the effects of helium on fast fracture
**	**	**	-	flow localization, consequences and underlying microstructural causes
***	***	*	*	helium effects on high temperature deformation and fracture, and development of improved multiphase alloys for helium control
**	**	**	**	thermal and irradiation creep
**	**	**	**	fatigue
**	***	*	*	hydrogen and interstitial impurity effects on deformation and fracture
*	***	-	**	coatings, multilayers, functionally graded materials
**	**	**	***	swelling and general microstructural stability
**	***	**	***	welding, joining and processing issues
		*	***	physical properties, e.g. thermal conductivity
			***	permeability of gases
**	**	**	**	erosion, chemical compatibility, bulk corrosion, cracking, product transport

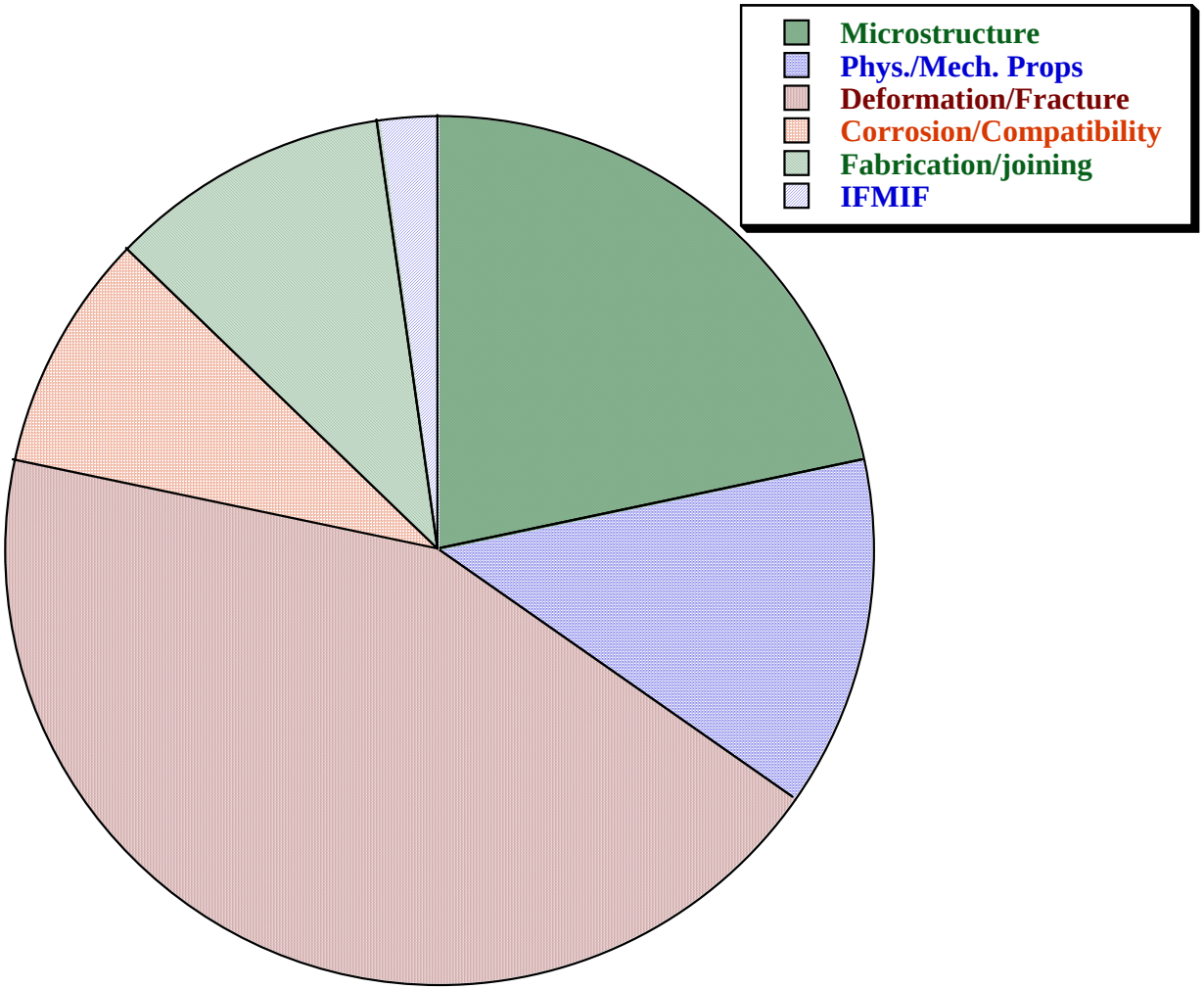
- Much of the R&D on RAFMs, V alloys, etc. can guide the development of advanced ferritic steels

Fusion Materials Science Program

	Theory-Experiment Coordinating Group*				
	Microstructural Stability	Physical & Mechanical Properties	Fracture & Deformation Mechanisms	Corrosion and Compatibility Phenomena	Fabrication and Joining Science
Materials for Attractive Fusion Energy					
• **Structural Alloys*** - Vanadium Alloys - F/M and ODS Steels - High T Refractory Alloys - Exploratory Alloys					
• **Ceramic Composites*** - SiC/SiC, other CFCs					
• **Coatings**					
• **Breeder/multiplier Materials**					
• **Neutron Source Facilities**					
Materials for Near-Term Fusion Experiments					
• **PFMs (Refractory Alloys, etc.)**					
• **Copper Alloys**					
• **Ceramic Insulators**					
• **Optical Materials**					

*asterisk denotes Fusion Materials Task Group

Fusion Materials Sciences R&D Portfolio



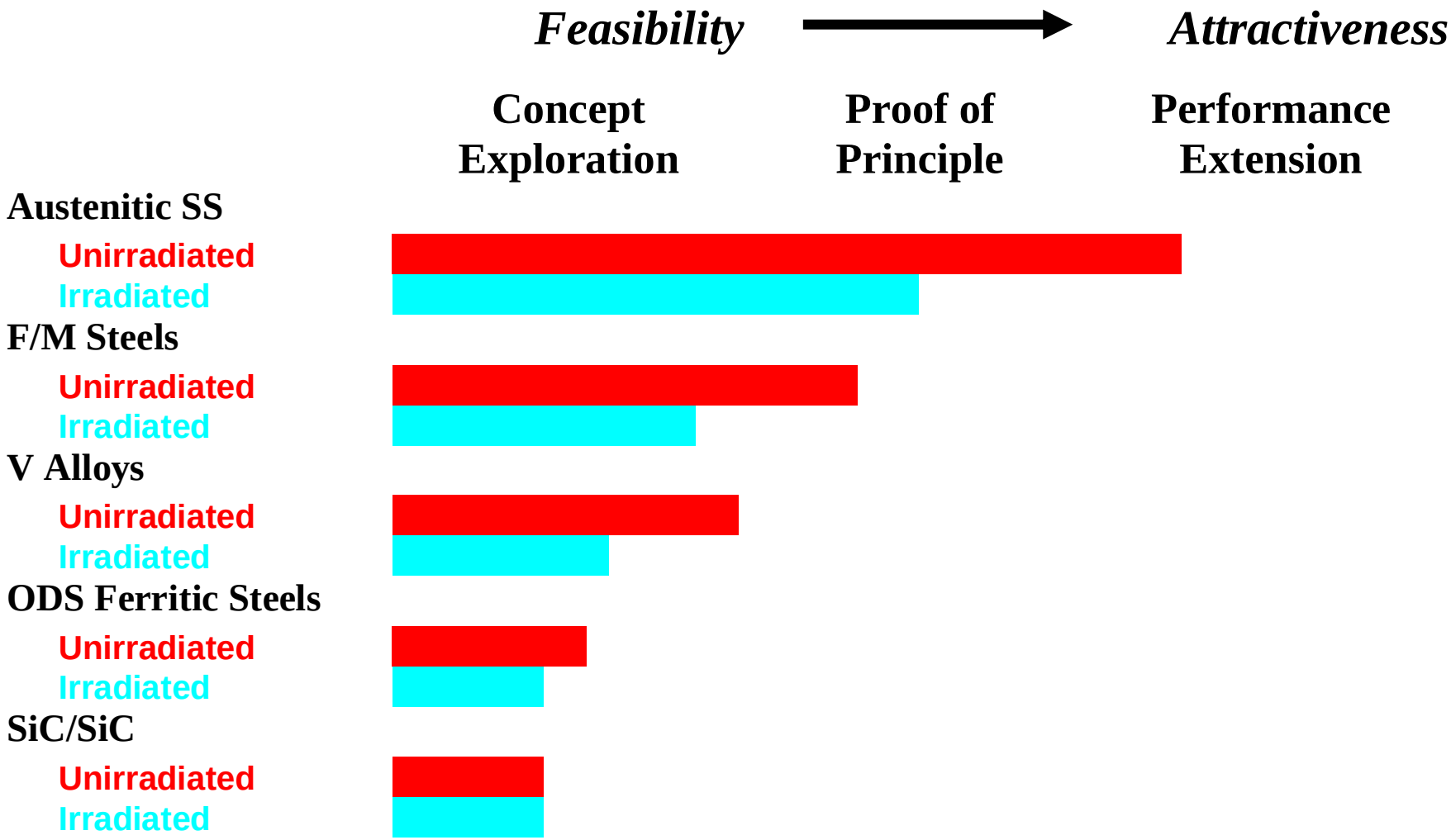
FY01 funding

- Predominant emphasis on mechanical properties

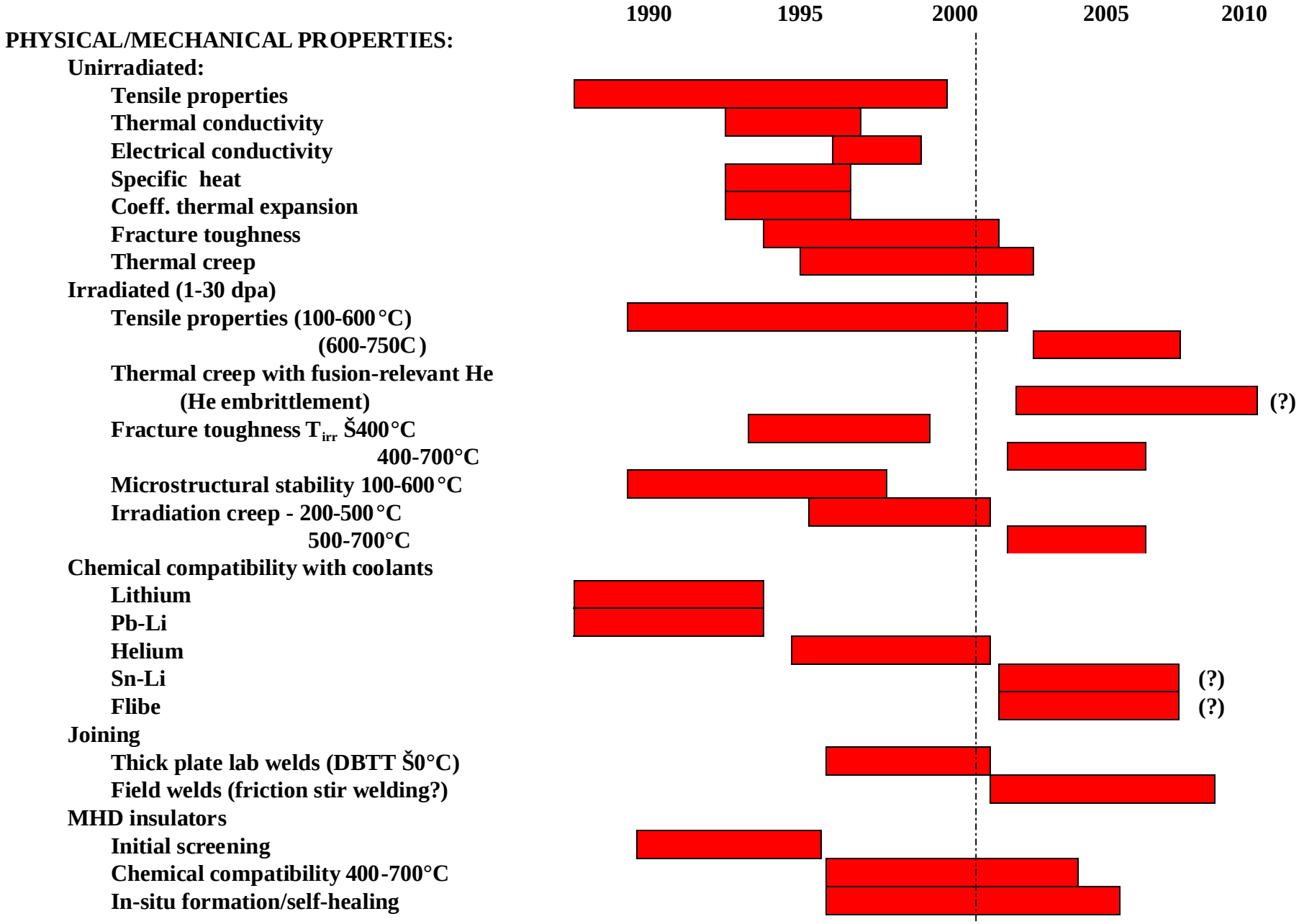
Scientific Research Program Requires a Technological "Rudder" to Maintain Optimal Progress

- Current stage (scientific discovery): Science-based investigations of fundamental materials phenomena
  - Heavy reliance on modeling
- Research techniques evolve as our understanding improves
  - SiC/SiC example: heavy reliance on inexpensive screening tests during initial stage of research (3 point bend bars); current evolution to tensile and fracture toughness test program

Current Status of Structural Materials

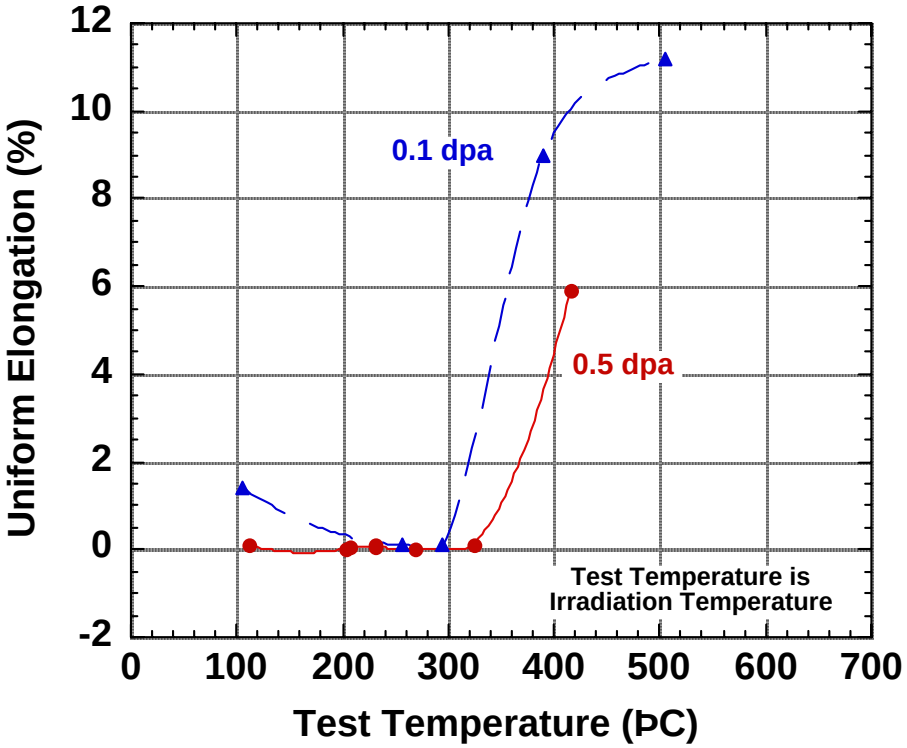


Snapshot of research activities on V alloys examining key feasibility issues

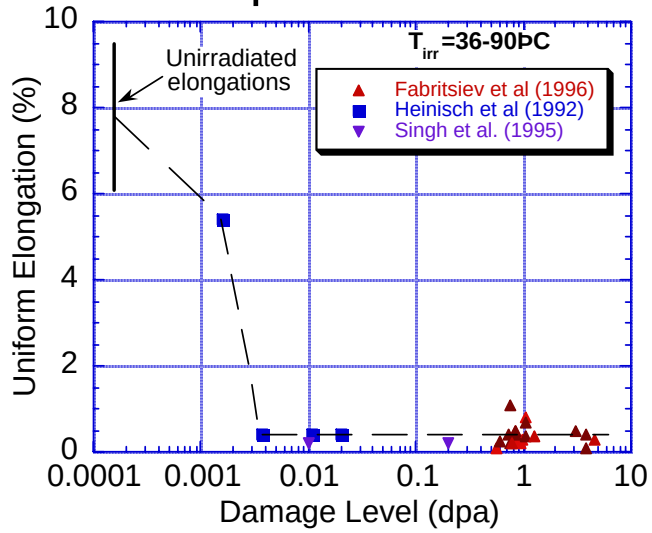


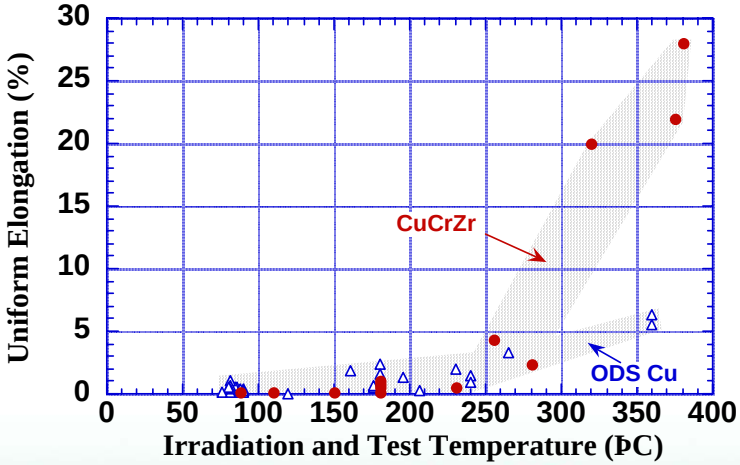
Low uniform elongations occur in many FCC and BCC metals after low-dose irradiation at low temperature

Uniform elongation of neutron-irradiated V-4Cr-4Ti

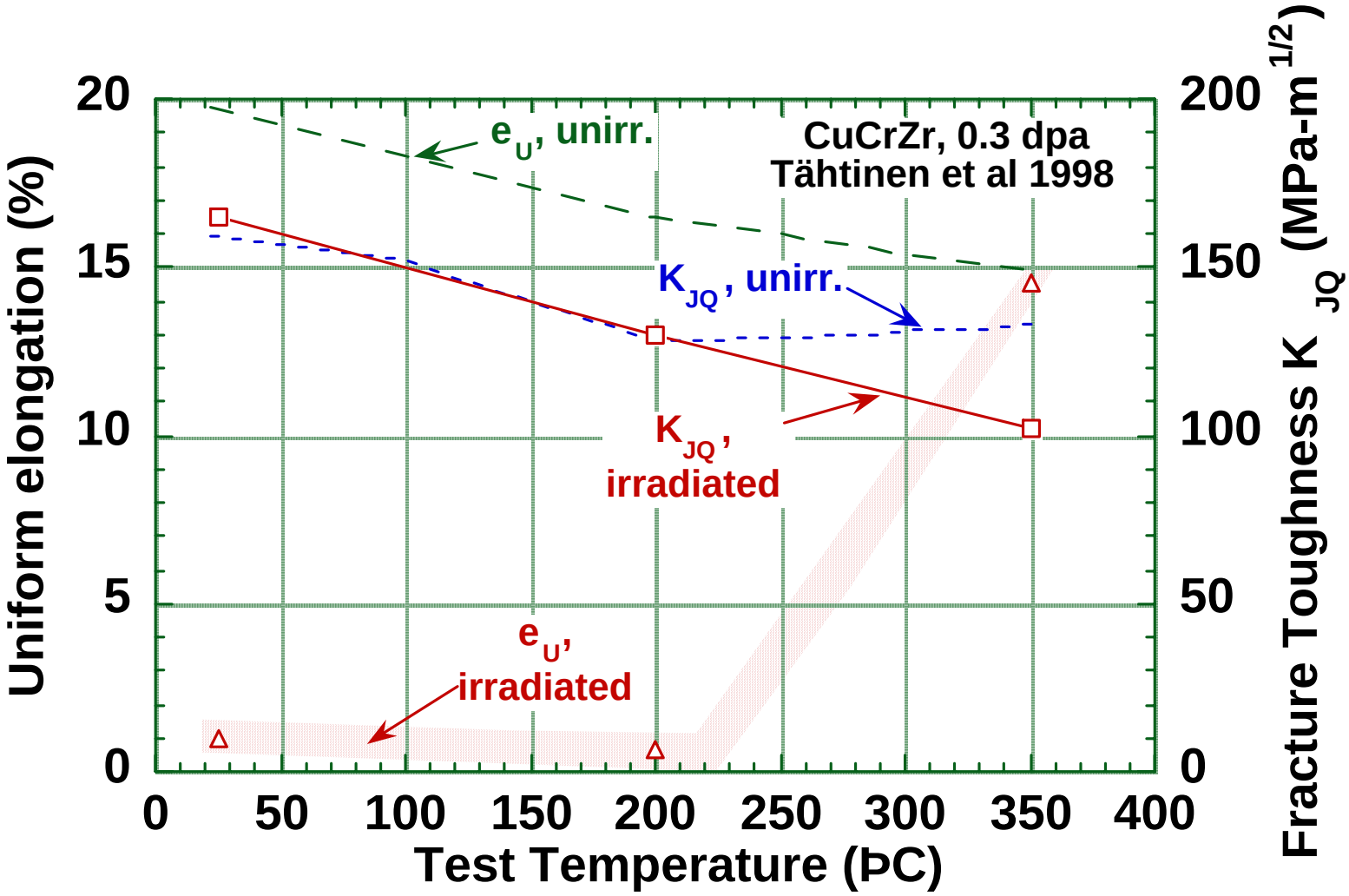


Uniform elongation of neutron-irradiated GlidCop Al25 and CuCrZr



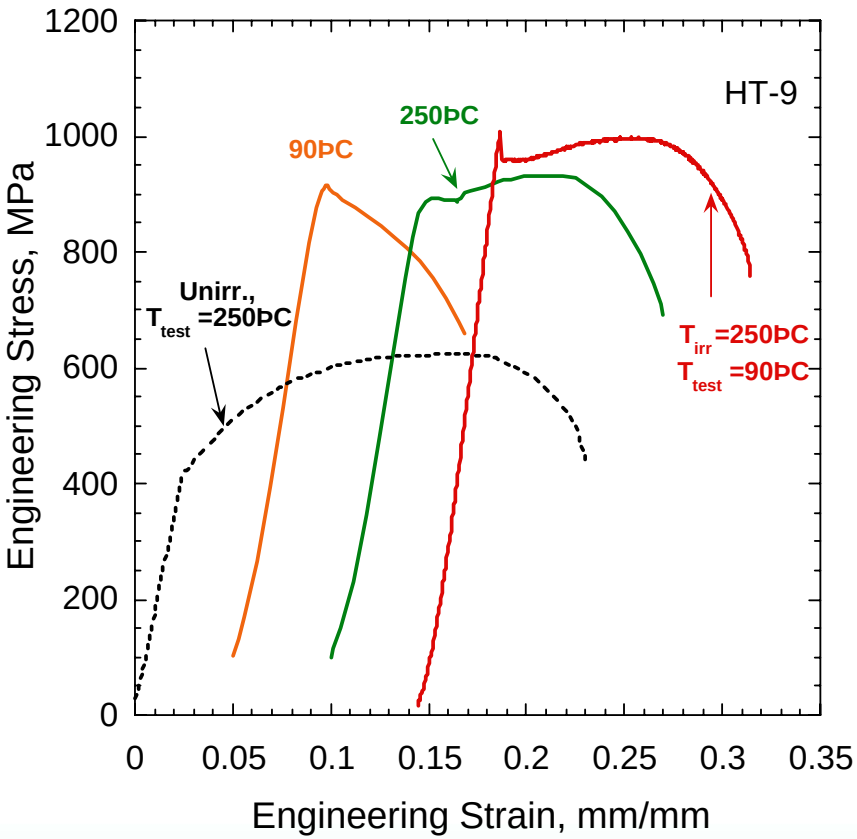


Radiation-induced Tensile "Embrittlement" does not Necessarily Produce Fracture Toughness Embrittlement

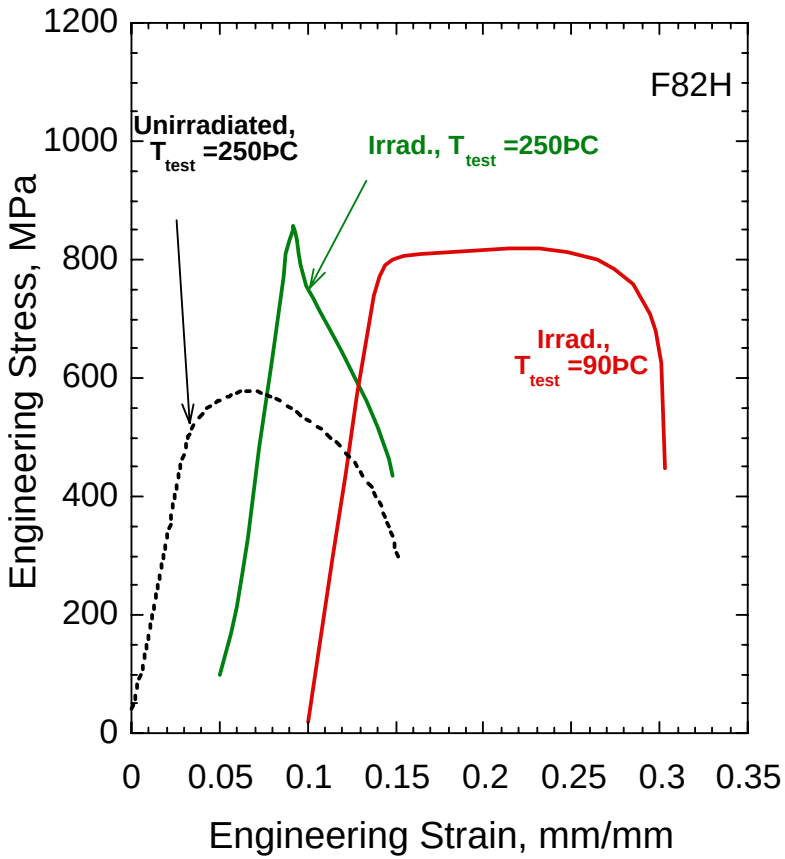


Comparison of Irradiated Tensile Behavior of HT-9 Ferritic Steel and F82H Reduced Activation Martensitic Steel

Uniaxial Tensile Load-Elongation Curves for HT-9 Ferritic/Martensitic Steel Irradiated to 3 dpa at 90 or 250ÞC



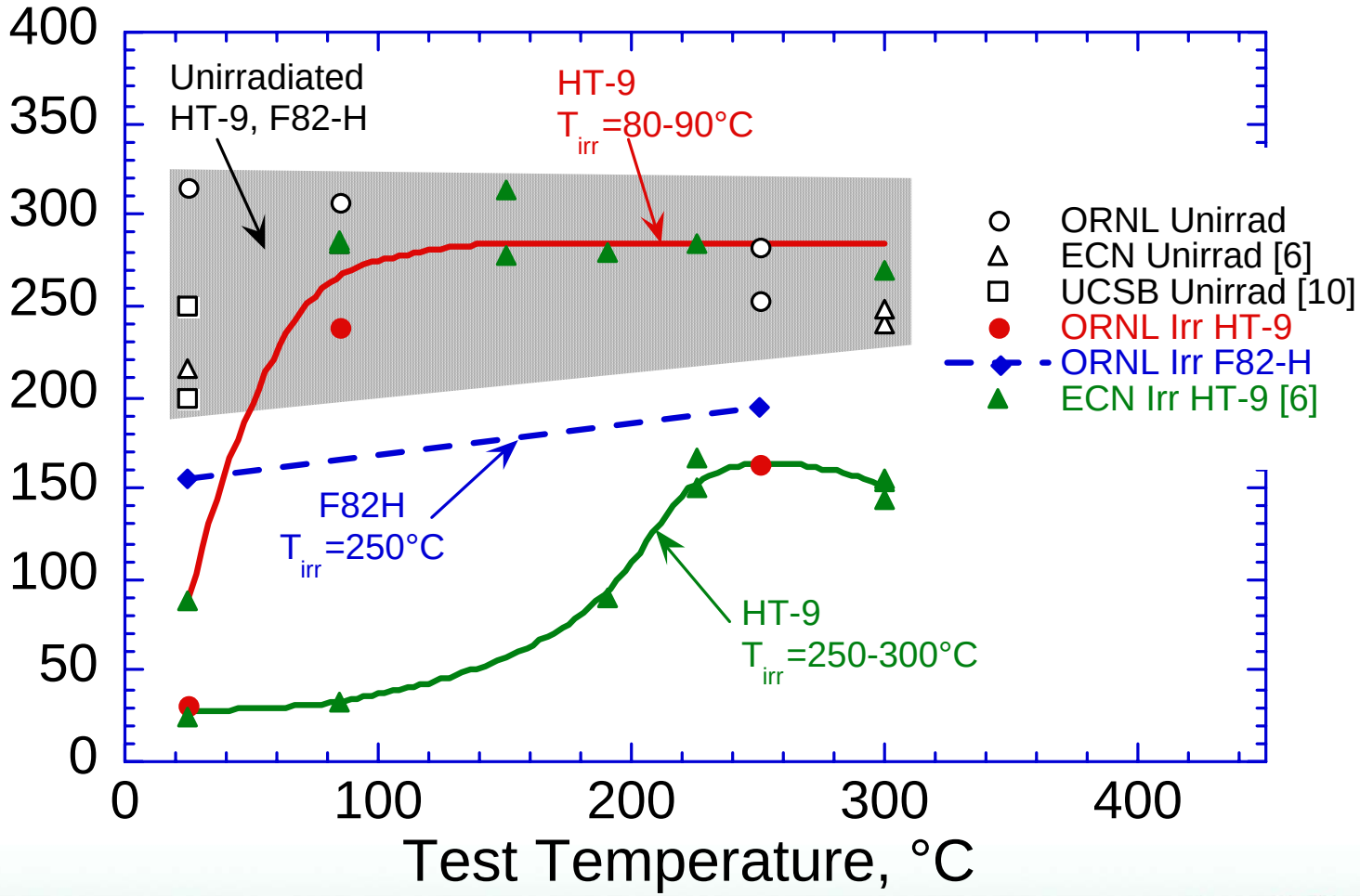
Uniaxial Tensile Load-Elongation Curves for F82H Ferritic/Martensitic Steel Irradiated to 3 dpa at 250ÞC



A.Rowcliffe et al., J.Nucl.Mat. 258-263 (1998)1275

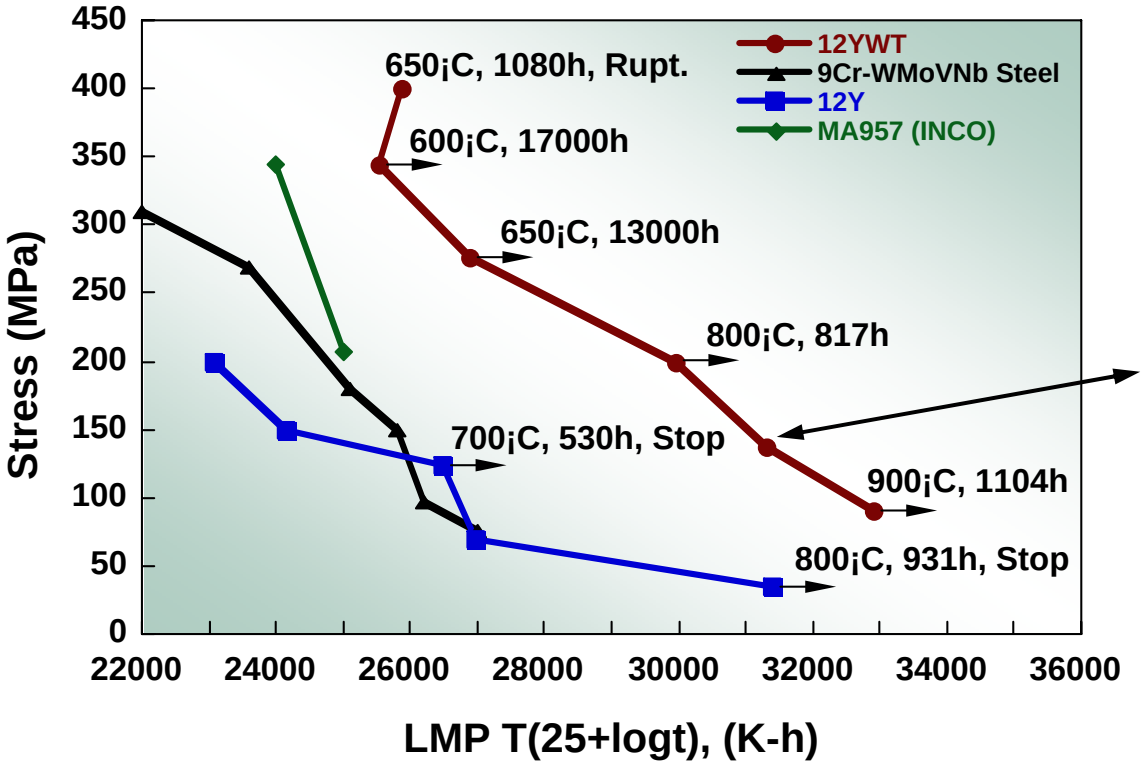
Fusion RAFMs have Superior Low-T Radiation Embrittlement Behavior Compared to HT-9

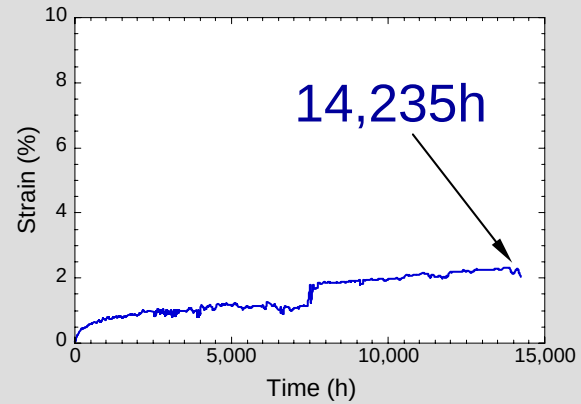
Fracture Toughness of Ferritic/Martensitic Steels Irradiated to ~3 dpa at 90 or ~250ÞC



A.Rowcliffe et al., J.Nucl.Mat. 258-263 (1998)1275 (DOE/JAERI collaboration)

Nanocomposited 12YWT Ferritic Steel Exhibits Excellent High Temperature Creep Strength





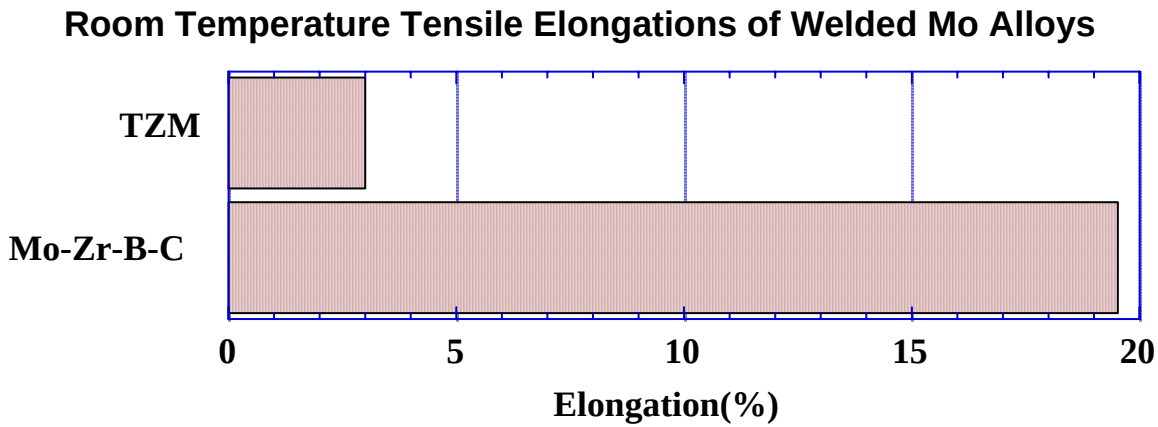
800°C and 138 MPa

- minimum creep rate
  $2.13 \times 10^{-10} s^{-1}$
- total strain
  2.03%

- Time to failure is increased by several orders of magnitude
- Potential for increasing the upper operating temperature of iron based alloys by ~200°C

Recent research offers promise for developing refractory alloys with improved ductility

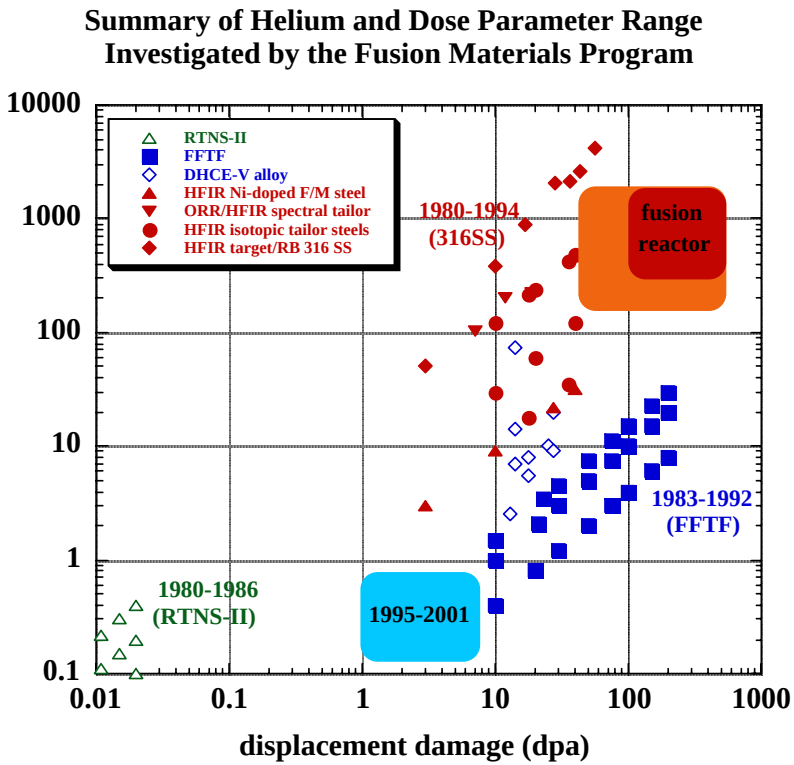
- Controlled (50-1600 appm) additions of Zr, B, C to molybdenum increases room temperature ductility of weldments from nearly zero to e_{tot}~20% (M.K.Miller and A.J.Bryhan, 2001)



- Mechanically alloyed W-0.3wt%Ti-0.05wt%C (H. Kurishita et al., ICFRM-10, Baden-Baden, Oct. 2001)
  - Avoid $(W,Ti)_2C$ brittle phase by limiting max concentration of carbon
  - Small grain size (~2 µm) helps to dilute harmful oxygen grain boundary segregation
  - TiC dispersed particles provide increased toughness (appropriate fracture mechanics tests needed to verify preliminary smooth bend bar results)

Fusion materials research must rely heavily on modeling and surrogate experiments due to inaccessibility of fusion-relevant operating regime

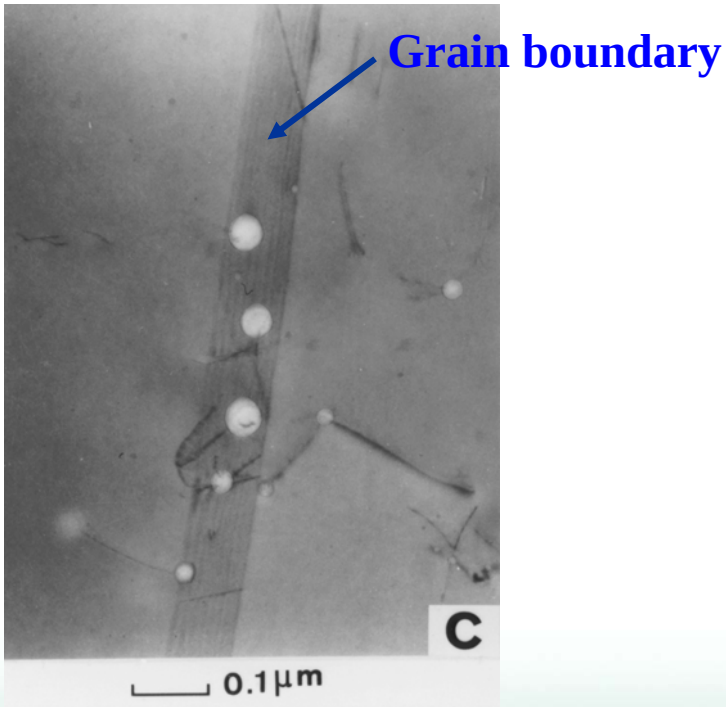
- Extrapolation from currently available parameter space to fusion regime is much larger for fusion materials than for plasma physics program
- An intense neutron source such as IFMIF is needed to develop fusion structural materials



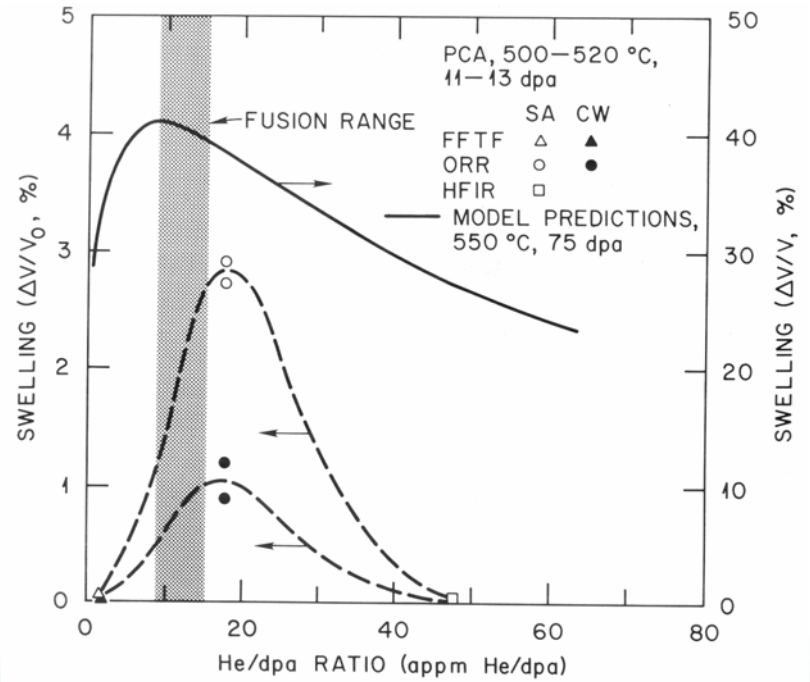
Why is He/dpa ratio an important parameter for fusion materials R&D?

- He generation can alter the microstructural evolution path of irradiated materials
  - Cavity formation
  - Precipitate and dislocation loop formation

He bubbles on grain boundaries can cause severe embrittlement at high temperatures



Swelling in stainless steel is maximized at fusion-relevant He/dpa values

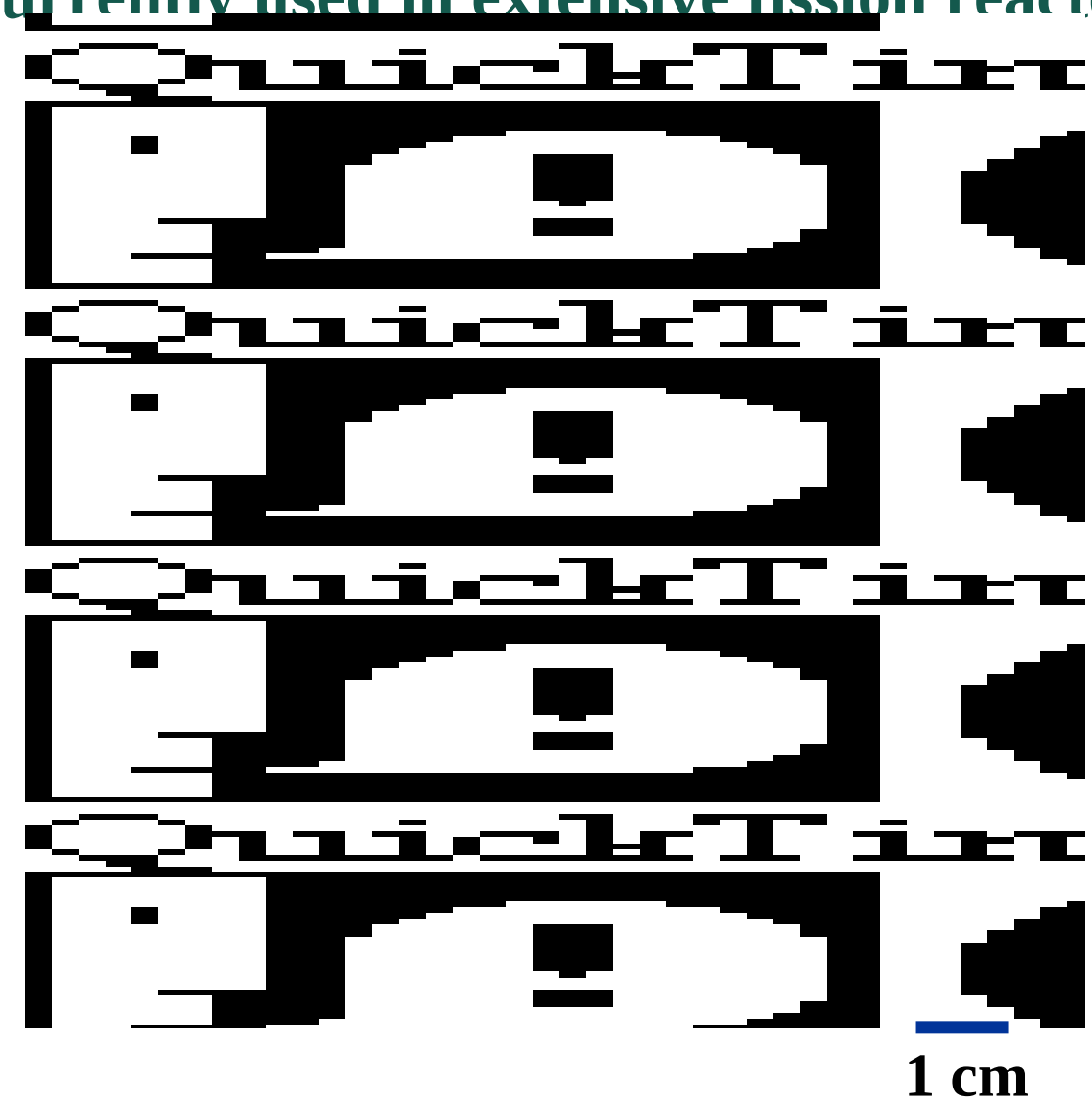


Overview of Fission, Spallation, and d-Li Neutron Sources for Fusion Materials R&D

Neutron Source	Advantages	Disadvantages
Fission Reactors	Well-characterized spectra Allows medium-high damage regimes to be investigated in bulk specimens Operating funds provided by multiple users (non-fusion)	Low He/dpa ratio
Spallation	Allows high-He irradiation conditions to be explored Operating costs may be largely provided by non-fusion agencies	Not designed for materials irradiations (physics/ neutron scattering facility)* He/dpa,H/dpa ratio too high Pulsed irradiation; requires detailed analysis
D-Li	Correct He/dpa ratio, etc. Dedicated materials irradiation facility	Operating funds completely provided by fusion

***feasibility of limited-volume "rabbit" irradiation facility in SNS is being explored**

Validated Miniaturized Specimens Enable Engineering Data to be Obtained from Small Irradiation Volumes (currently used in extensive fission reactor studies)



IFMIF provides capability of accelerated radiation effects testing of structural materials in a fusion-relevant neutron spectrum

- ~0.5 liter volume with >2 MW/m^2 equivalent neutron flux (~5x20x5 cm^3)
  - ~0.1 liter volume with >5.5 MW/m^2
  - (note: typical current fission reactor irradiation capsule volumes are 0.013 to 0.1 liters)
- Strawman specimen matrices indicate that 600-1000 mechanical property specimens (and >>1000 TEM specimens) can be accommodated in 0.5 liter high flux region
- Irradiation to 15-20 MW-a/m^2 of a complete set of mechanical property specimens of two materials at 3 to 4 different temperatures could be accomplished in 4 years in the $\geq$5.5 MW/m^2 flux regions of IFMIF
- Medium and low-flux regions of IFMIF (<2 MW/m^2) are earmarked for in-situ creep fatigue tests, ceramic breeder irradiation studies, diagnostic component irradiations, etc.

Strategy for Deployment Schedule of Fusion Materials intense neutron irradiation facility (IFMIF)

- Acquisition of the fundamental constitutive properties of irradiated materials is essential for the analysis of the performance of irradiated components
- Development of materials will likely require several iterations of materials irradiation & alloy modification (requiring >15-20 years to develop and qualify materials)
- IFMIF should be deployed in advance of a plasma technology test bed (strong interactions between users of each facility are essential)
  - Cf. Snowmass 2002 report