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Joint ICTP-IAEA Workshop on Physics of Radiation Effect and its Simulation for Non-Metallic Condensed Matter

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Nuclear technology applications of ceramics, composites and other nonmetallic materials

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Trieste, Italy

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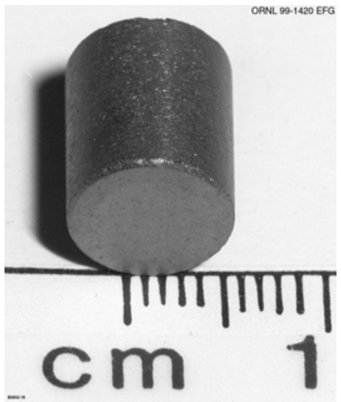
Nonmetallic materials are being used or considered for a wide range of fission and fusion energy applications

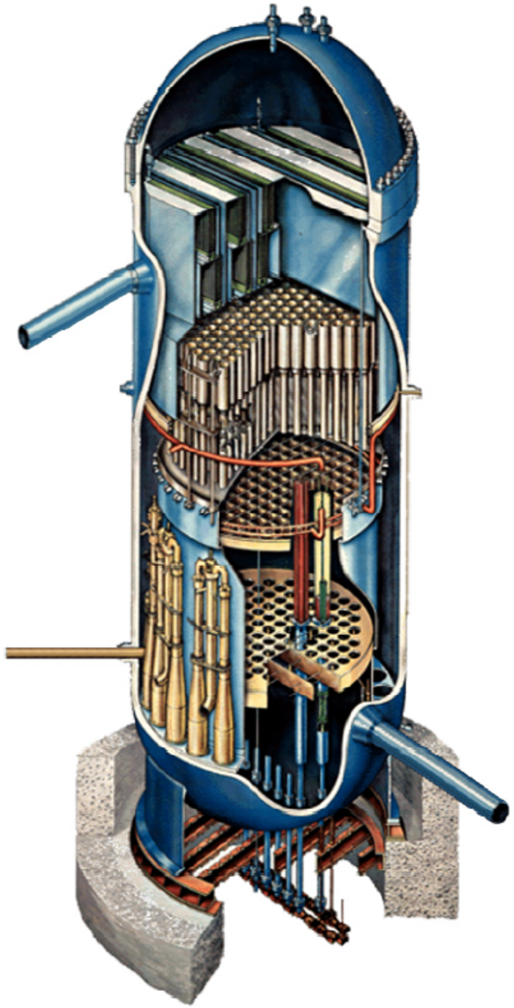
- Microencapsulated fuel systems for gas-cooled and accident-tolerant water cooled reactor designs
- SiC/SiC and CFC composites for structural applications
- Neutron control systems for fission reactors
- Diagnostic applications (optical fibers, windows, bulk insulators, MI cables)
- Plasma heating systems
- Fusion ceramic breeders
- Fission waste storage forms

- This presentation will focus on the first two topics in this list

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





Evolution of coated particle fuel missions

1960s

- Ceramic coated particle fuel developed for extreme performance in nuclear rockets (ROVER/NERVA)
- TRISO used for large graphite reactors (HTRs)

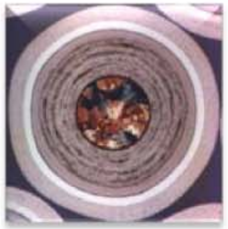
1980s

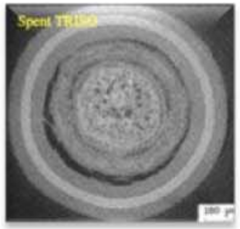
- TRISO fuel adapted for high temperature operation in modular HTR

2000s

- Evidence of exceptional burnup of ceramic TRISO fuel suggests potential use in management/ utilization/ destruction of TRU used fuel

Deep Burn utilizes the high burnup capabilities of TRISO fuel

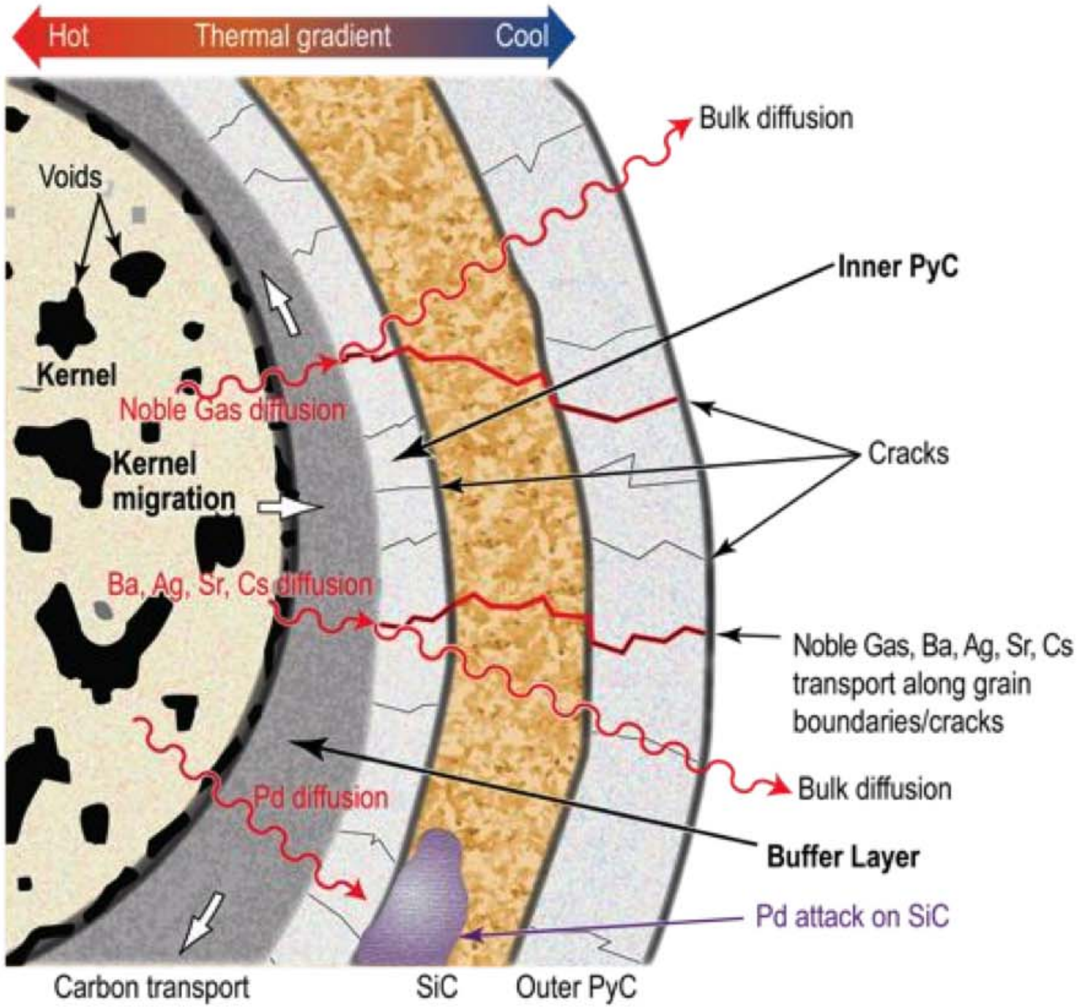




Deep Dive_1006

DOE Deep Burn Project

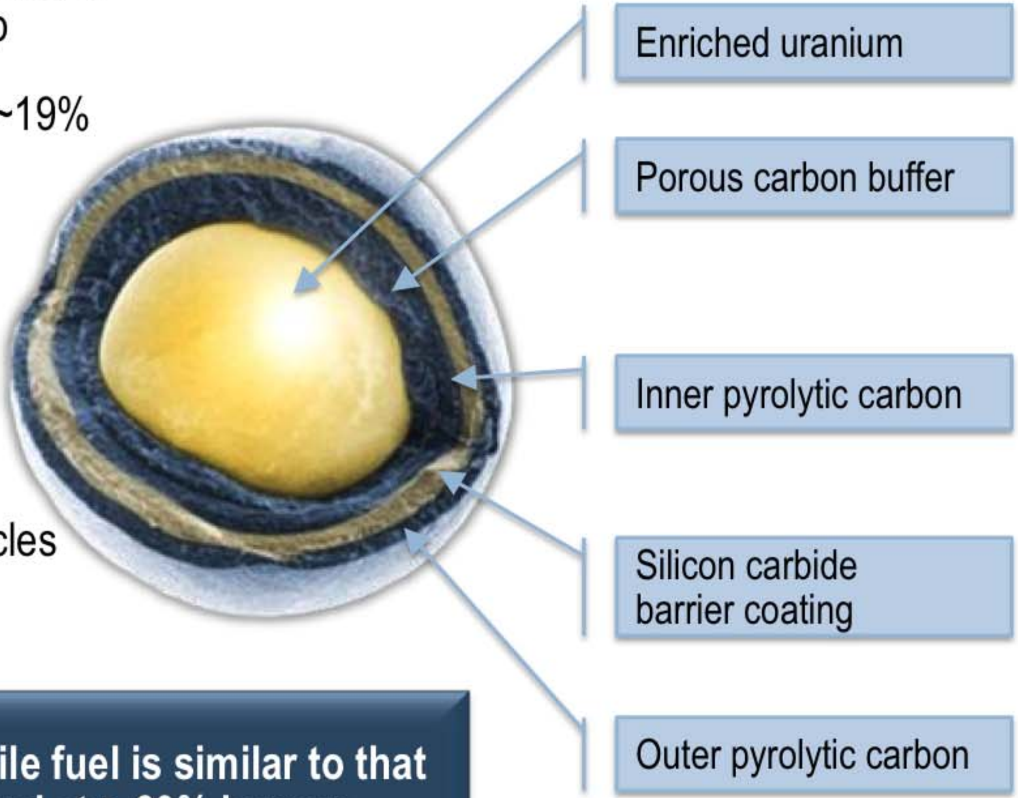
Micro-encapsulated fuels for high-burnup applications



- **Targeted Modeling Activities:**
  - Kernel Stability Through Gettering
  - Ag and Pd Diffusion in SiC & ZrC
  - Thermal Transport
  - Irradiation Effects on PyC & ZrC
- **TRISO Kernel Activities:**
  - Fabrication of uniquely small kernels
  - Inclusion of getter and BP within kernel
- **TRISO Coating Activities:**
  - ZrC Coating Development
  - ZrC Properties and Irradiation Eff.

Recent irradiation tests have demonstrated high reliability of micro-encapculated fuels

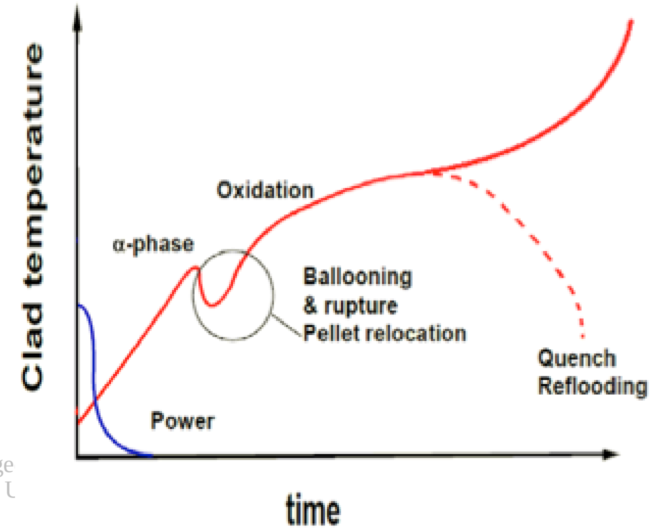
- Fuel recently developed at ORNL and irradiated in ATR at INL under AGR program sponsorship
- Global performance record for particle fuel: ~19% of LEU consumed, **zero** fuel failures
  - More than 2× the previous record
  - More than 3× current LWR fuel
  - Not a single TRISO fuel particle failure since beginning of irradiation ~ 3 year ago
- Significant step for TRISO nuclear fuel particles for use in HTGRs

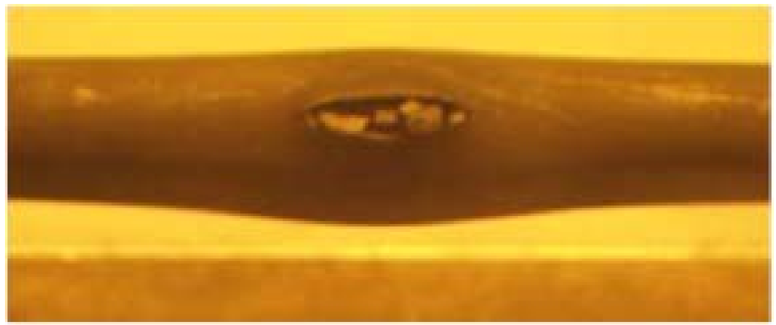


Fluence achieved at 19% burnup with fertile fuel is similar to that expected for Deep Burn TRU TRISO fuel at > 60% burnup

The LOCA at Fukushima Dai-ichi has heightened interest in re-evaluating potential accident-tolerant fuel systems

- Several potential options exist for fuel cladding
  - Oxidation-resistant austenitic steels
    - Relatively high neutron absorption and low melting temperature are concerns
  - Oxidation-resistant coatings on Zr alloy cladding
    - Mechanical robustness of cladding on anisotropic Zr alloy tubing is a concern
  - Ceramic matrix composite cladding
    - Relatively thick cladding requirement, hermetic isolation of fission products, joining techniques and lack of structural design criteria codes (e.g., ASME) are concerns





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)
  - varying water/steam chemistry for light water reactors
  - Normal and transient/accident conditions
- Low oxidation rate (hydrogen production) in water/steam and low oxide heat of formation
- 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

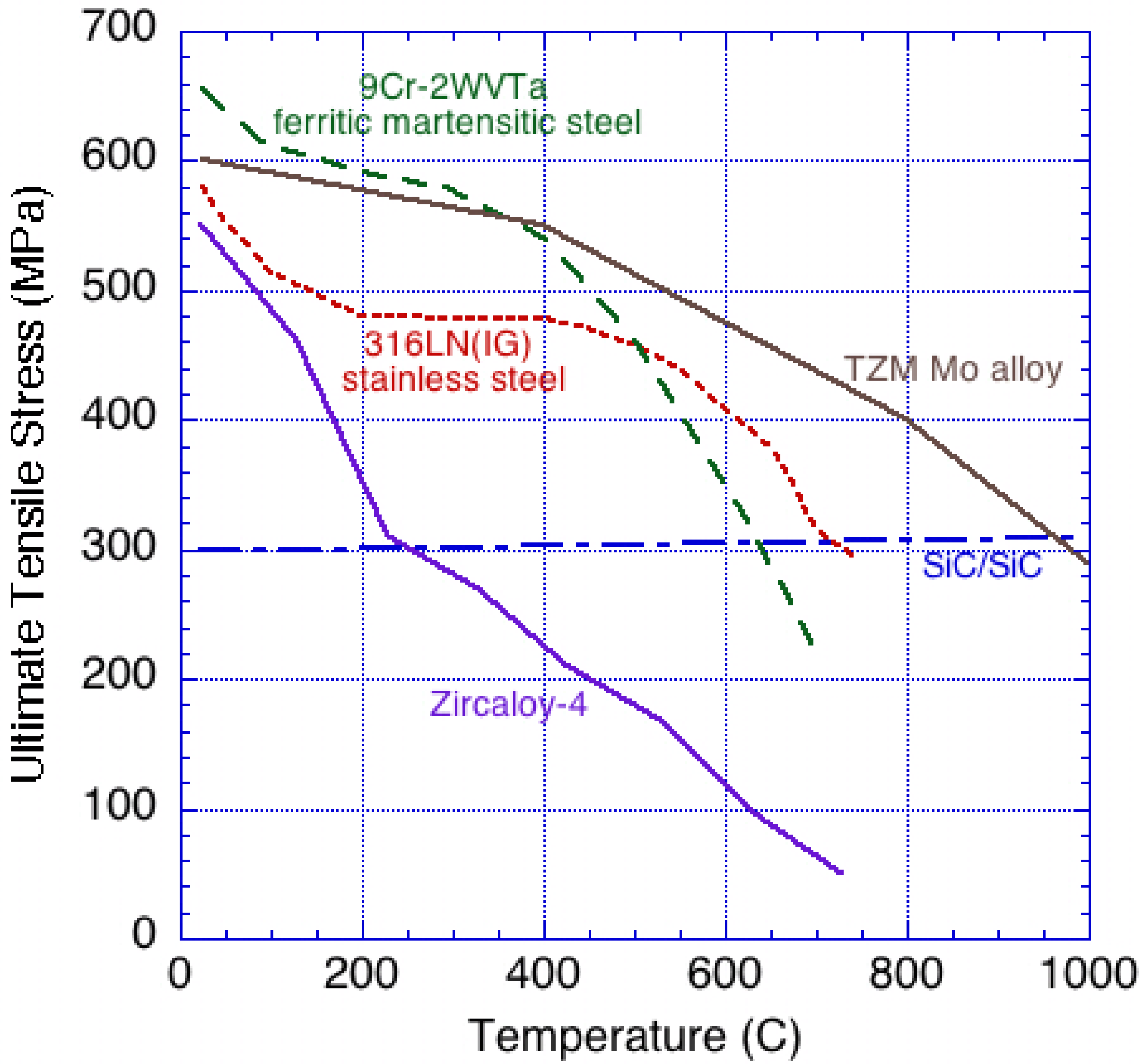
Comparison of properties of candidate cladding base materials

	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	20*
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 through-thickness CVD SiC/SiC composite; high purity SiC has K_{th}~350 W/m-K

Zr has one of the highest heats of oxidation; almost any other candidate is preferable in terms of heat contribution during a high temperature LOCA

Strength of some candidate cladding materials

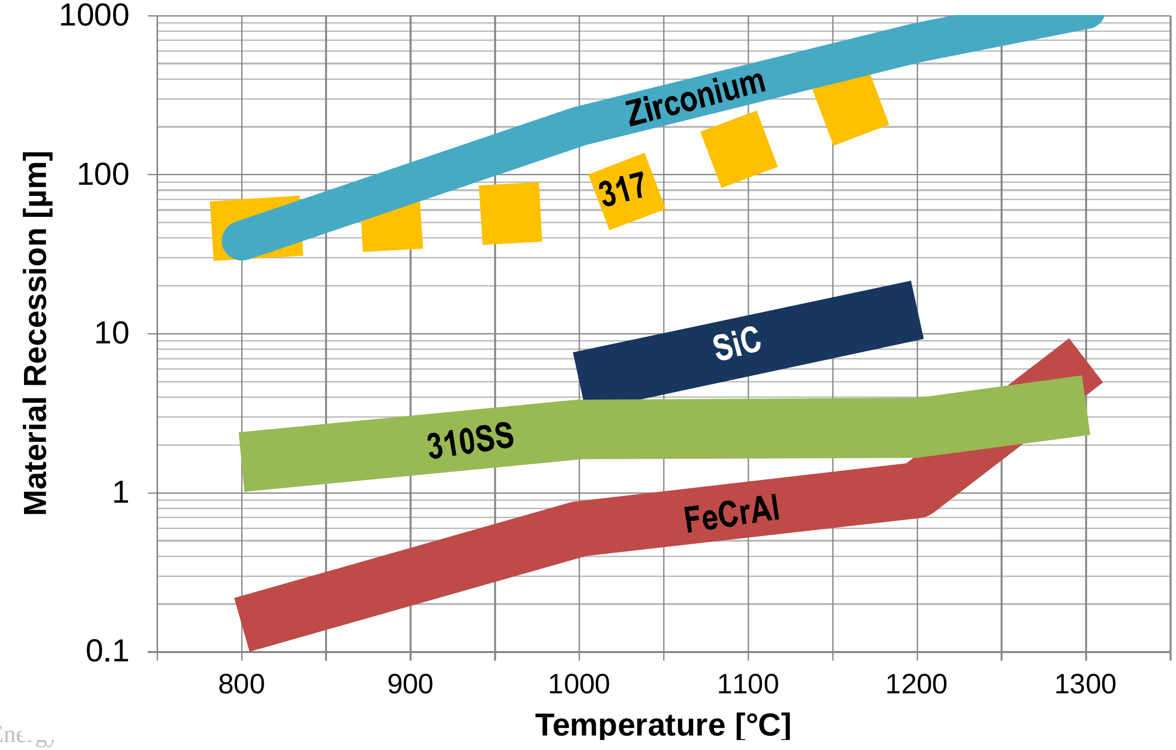


- Mo alloys and SiC/SiC ceramic composite offer improved high temperature strength

High-Pressure Steam Oxidation Tests: Comparison of the Extent of Steam Reaction

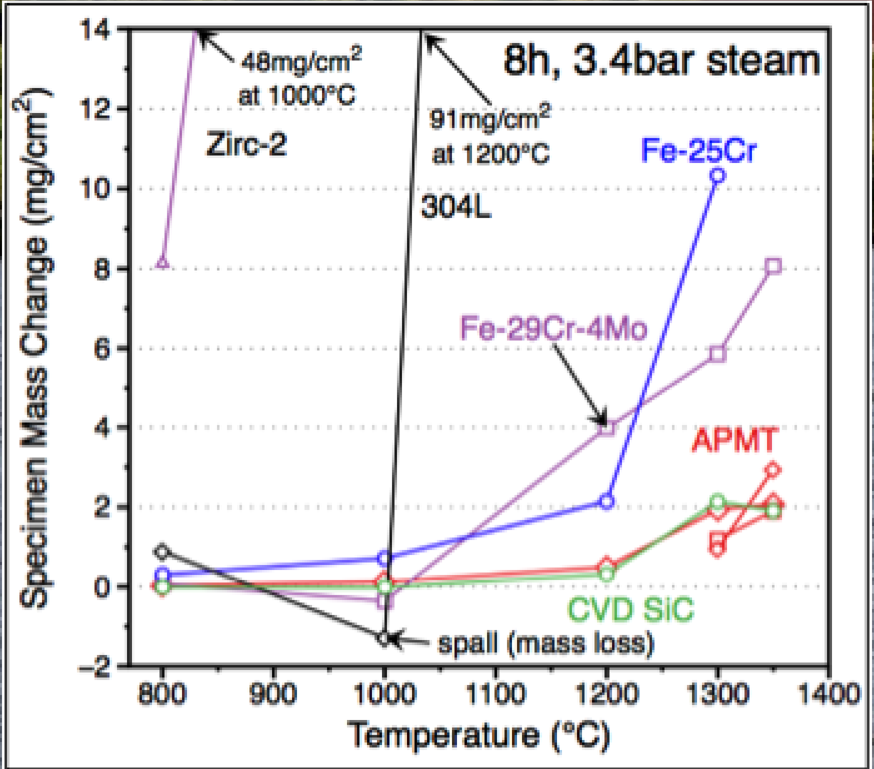
Various materials exposed to pure steam for 8 hours (various flow rates and pressures):

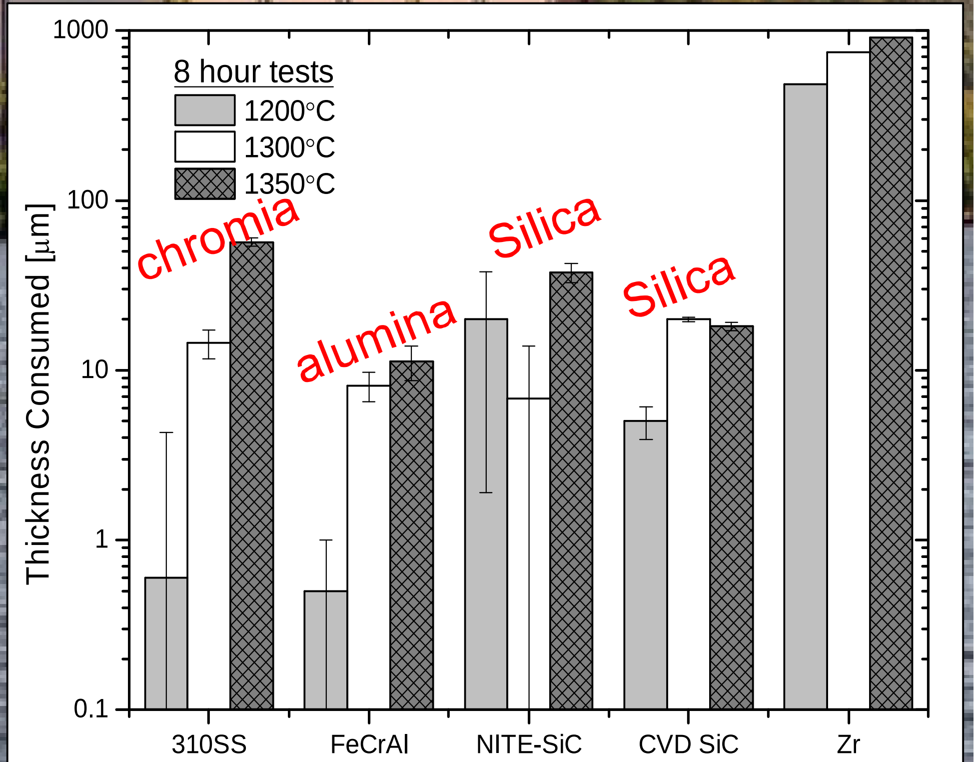
- Zircaloy: Pawel-Cathcart and Moalem-Olander data
- 317 Stainless Steel: ORNL high-pressure tests; thickness loss data
- NITE and CVD SiC: ORNL high-pressure tests; thickness loss data
- 310 Stainless Steel: ORNL high-pressure tests; mass gain data converted to thickness loss
- FeCrAl Ferritic Steel: ORNL high-pressure tests; mass gain data converted to thickness loss



Understanding and Testing Advanced Protection Options for LOCA conditions

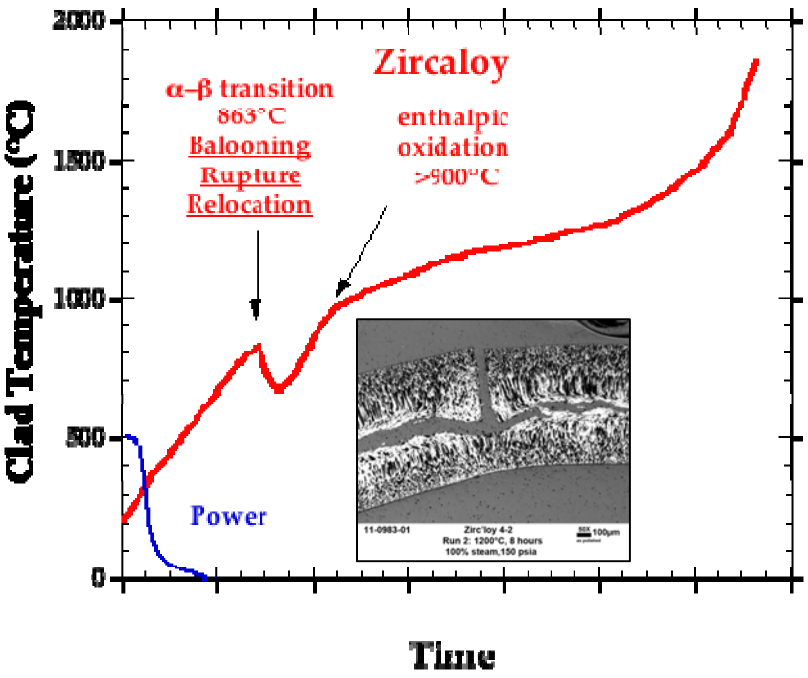
Protection by : chromia (stainless), silica (SiC), or alumina (AFA steels) formers

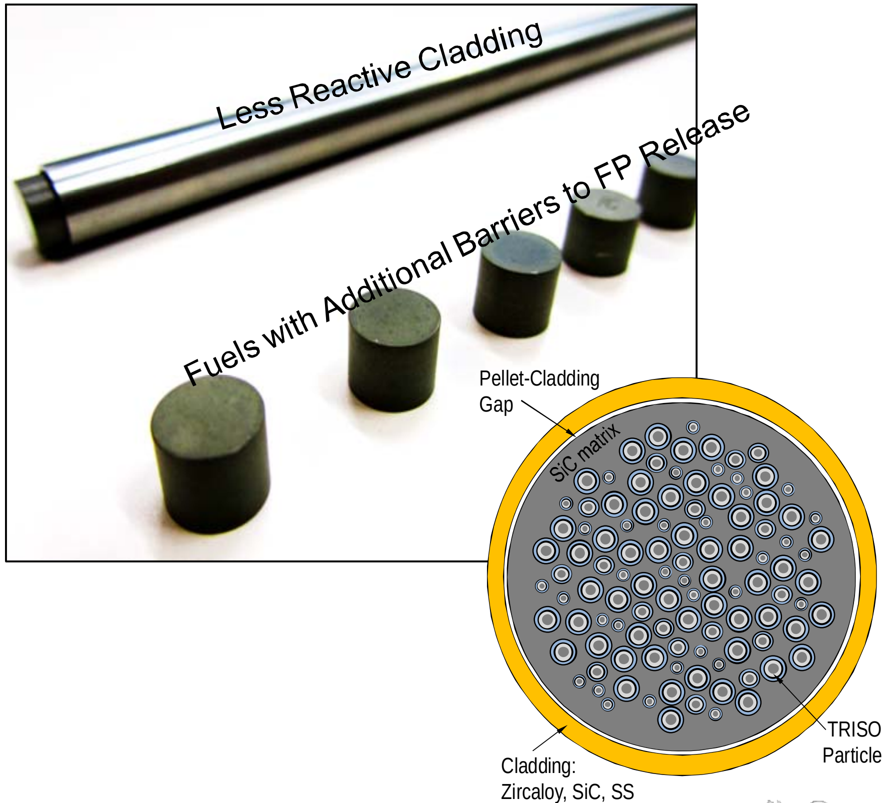




Enhancing accident tolerance of nuclear fuel systems

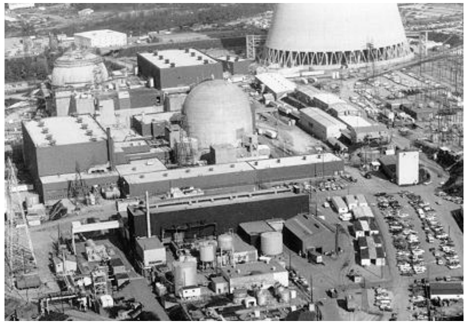
- Reactor safety margin may be improved through new fuel forms with much reduced exothermic reaction, suppressed hydrogen production, and greater time to fission product release.





Evolution of Nuclear Power Systems

Generation I
Early Prototype Reactors

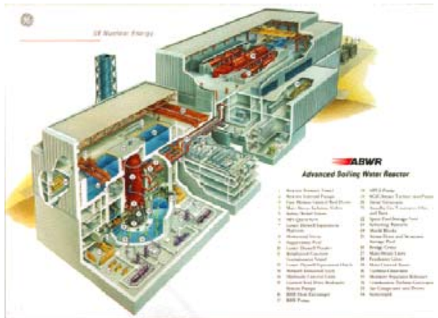


•Shippingport
•Dresden,Fermi-I
•Magnox

Generation II
Commercial Power Reactors

•LWR: PWR/BWR
•CANDU
•VVER/RBMK

Generation III
Advanced LWRs



•System 80+ •AP600
•EPR •ABWR

Generation IV

- Highly economical
- Enhanced Safety
- Minimized Wastes
- Proliferation Resistance

Gen I | Gen II | Gen III | Gen IV

1950 1960 1970 1980 1990 2000 2010 2020 2030

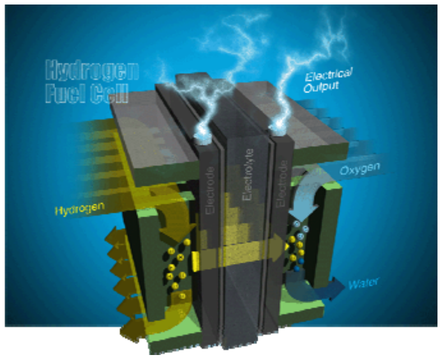
M.L. Corradini, Univ. Wisc.

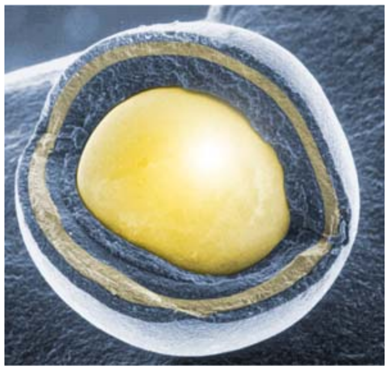
Overview of fission reactor options

- Light Water Reactors (LWRs): pressurized- and boiling-water designs
  - Present fleet of nuclear power plants (UO_2 fuel, zircaloy cladding)
  - Pressing for higher performance, improved fuel reliability and accident tolerance
- "Gen-IV" High Temperature Gas-Cooled Reactors
  - High temperature process heat applications as well as electricity production;
- "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

New technology, New requirements

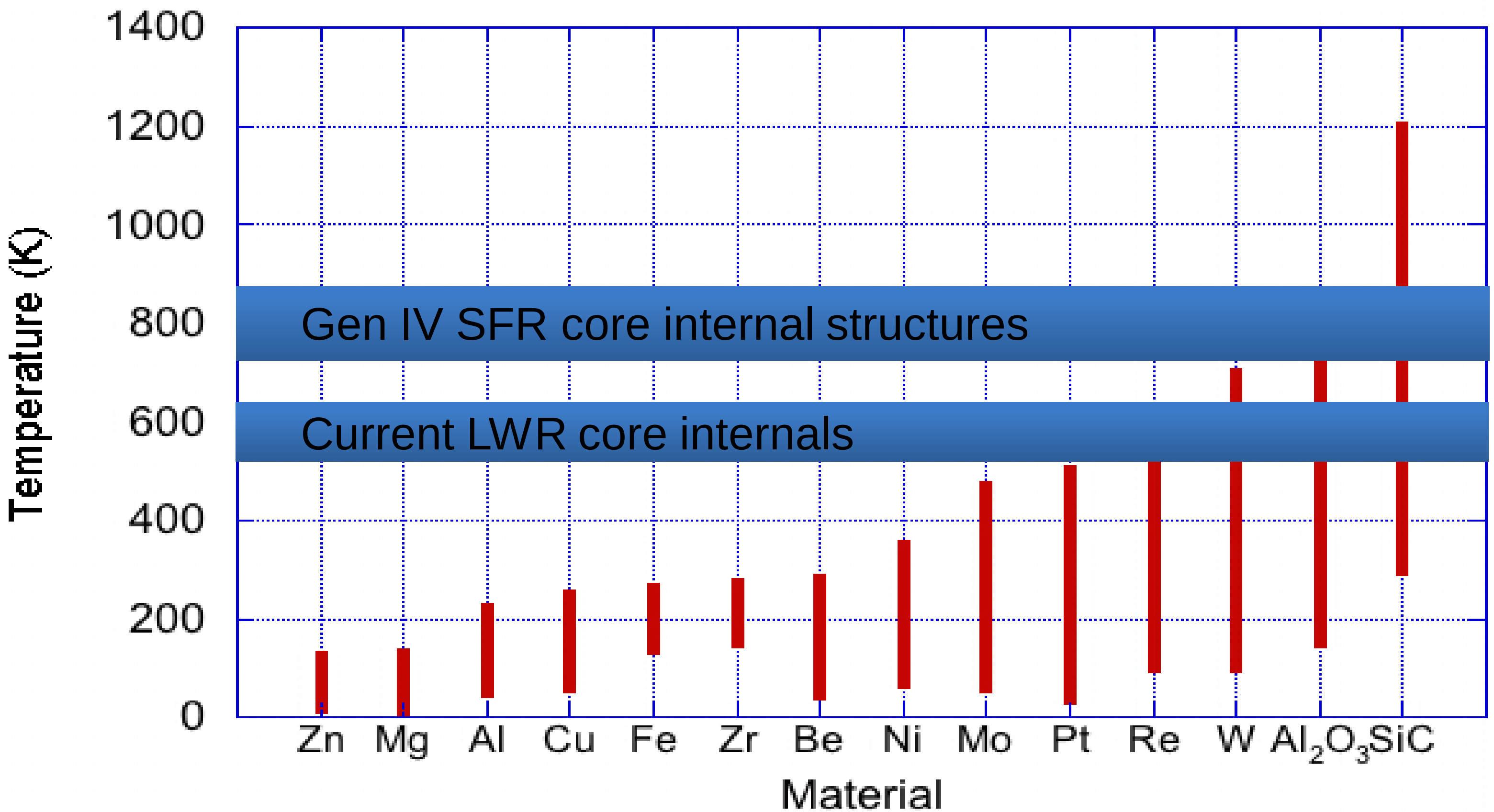
	Gen-III Pressurized Water Reactor	Gen-IV Fast Reactor	Very High Temperature Reactor
Coolant	Water	Sodium	Helium
Power Density (MW/m^3)	100	350	5
Coolant Temp. (C)	330	550	1000
Net Plant Efficiency (%)	34	40	50





Approaches for radiation resistance 2: Immobile defects

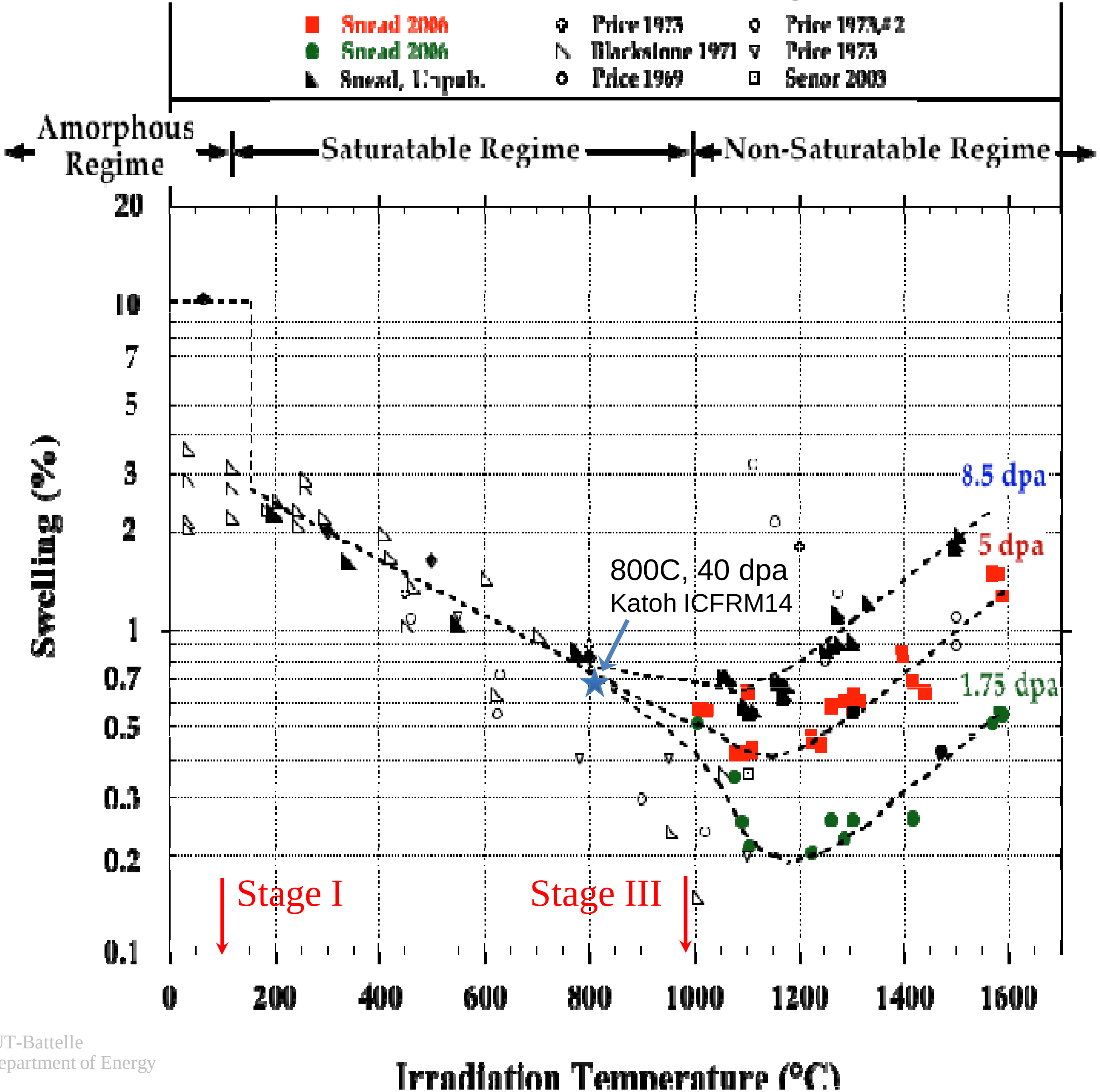
- Defect accumulation is limited if one or more defect types are immobile
  - Utilize materials with negligible point defect mobility at desired operating temperatures
  - A key potential consequence (particularly in ordered alloys and ceramics) is amorphization, with accompanying significant volumetric and property changes



- Regime with intrinsically high point defect recombination typically occurs at too low of temperatures for power generation applications (except SiC and possibly Al_2O_3, W, Re)

after S.J. Zinkle, Chpt. 3 in Comprehensive Nuclear Materials (Elsevier, 2012)

Irradiation-induced swelling in SiC



L.L. Snead

Gen IV NGNP

Candidate Composite Structures

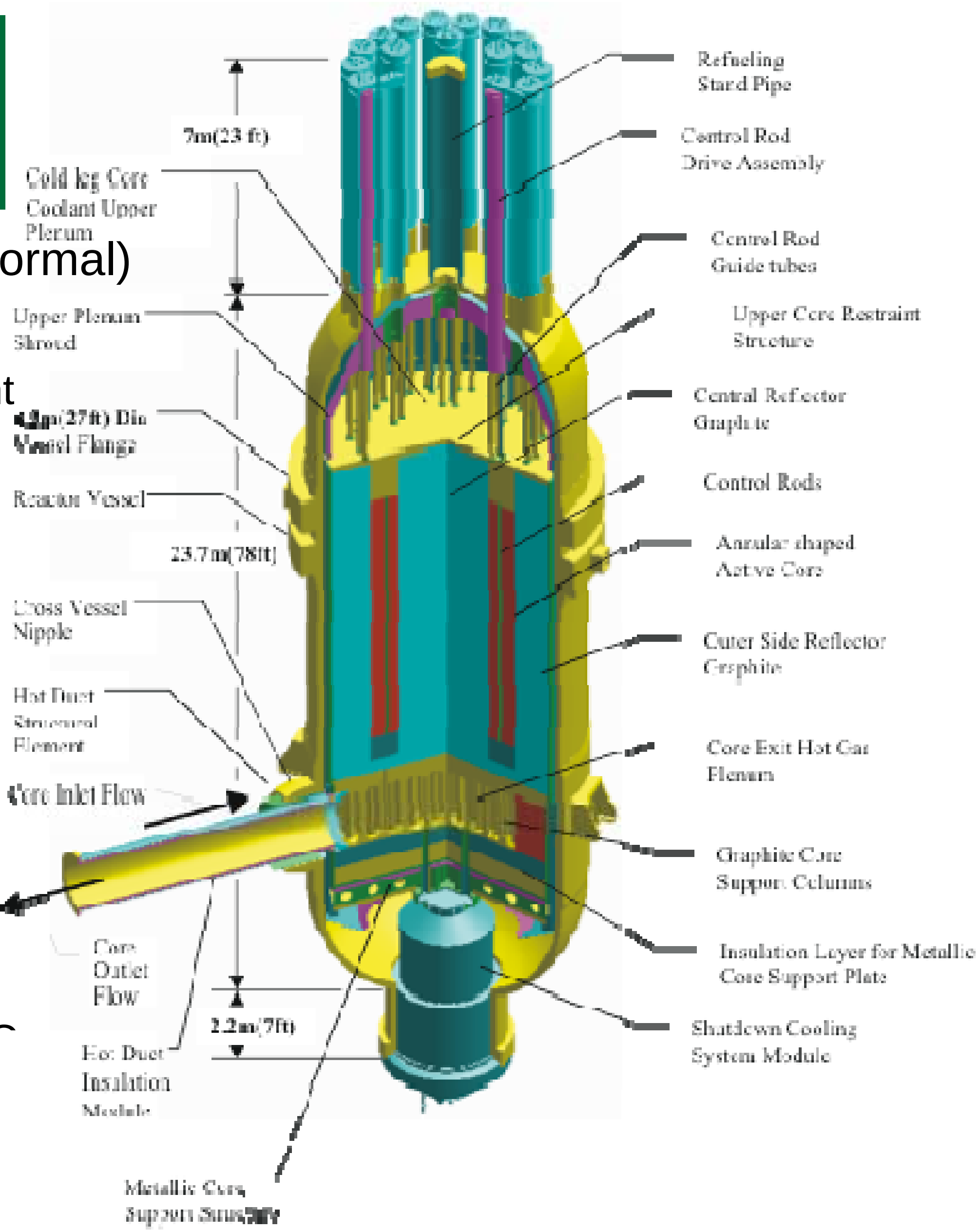
He inlet temp: 100C (normal)/1200C (offnormal)

- Upper Plenum Shroud/Upper Core Restraint
  Normal/Off-normal Temp: 650°C/1300°C
  Neutron damage: 0.05 dpa lifetime

- Control Rods and Guide Tubes
  Normal/Off-normal Temp: 1200°C/1600°C
  Neutron damage: 25 dpa/40 yr lifetime

- Floor Blocks
  Normal/Off-normal Temp: 600°C/600°C
  Neutron damage: <0.05 dpa yr lifetime

- Hot Duct Inner Shell
  Normal/Off-normal Temp: 1000°C/1200°C
  Neutron damage: 0.005 dpa year lifetime



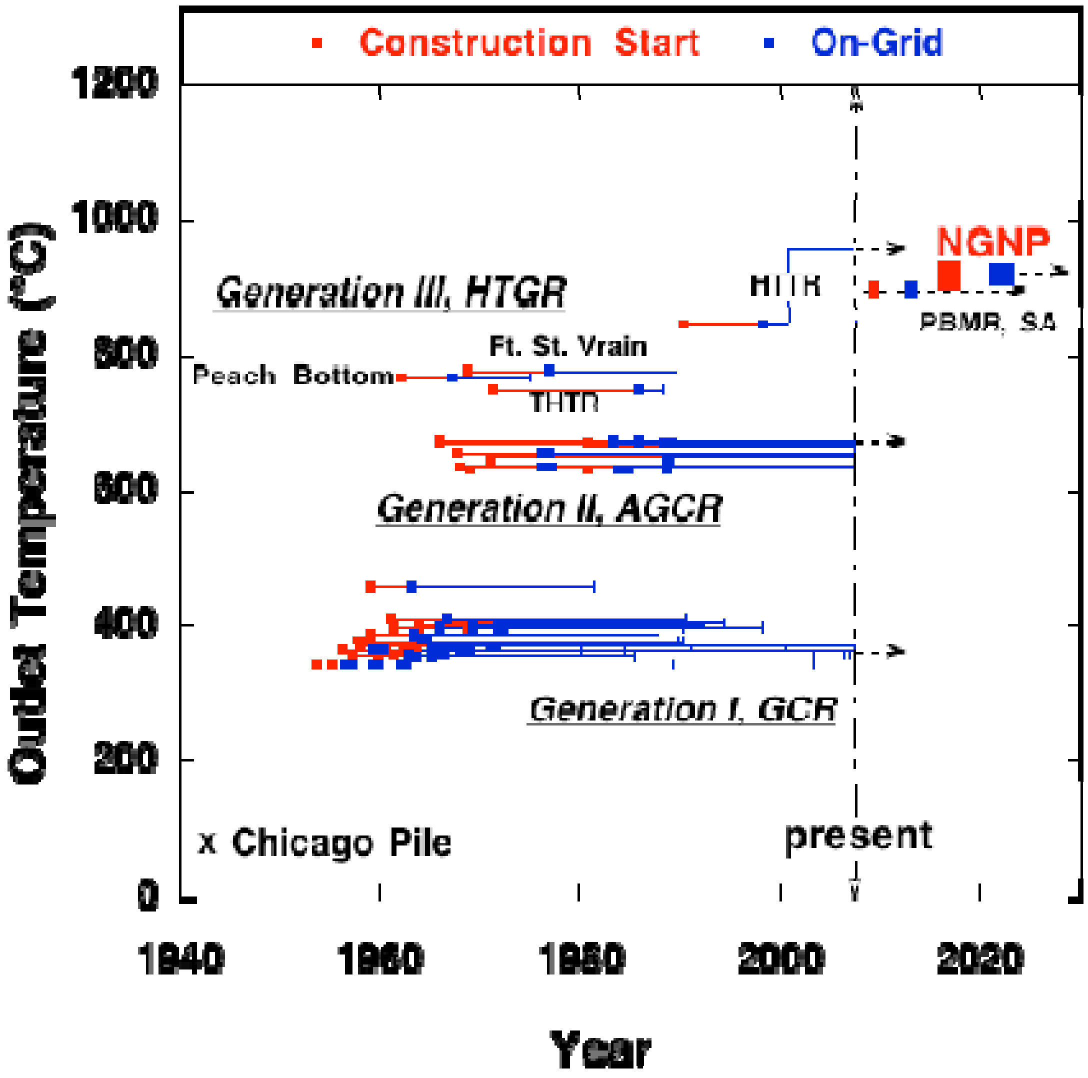
Materials Comparison at 1000°C

Material	Cost $/Kg	Life (dpa)	Irradiation-Induced Property Change @ 1000°C			
			Volume	Strength (MPa)	Modulus	Thermal Conductivity W/m-K
Superalloy	25	~5	-	-	-	-
CFC*	~200	10-15	-5%	150→250	+20%	250→180
SiC/SiC*	~400	>50?	+1%	75→75	-10%	50→20

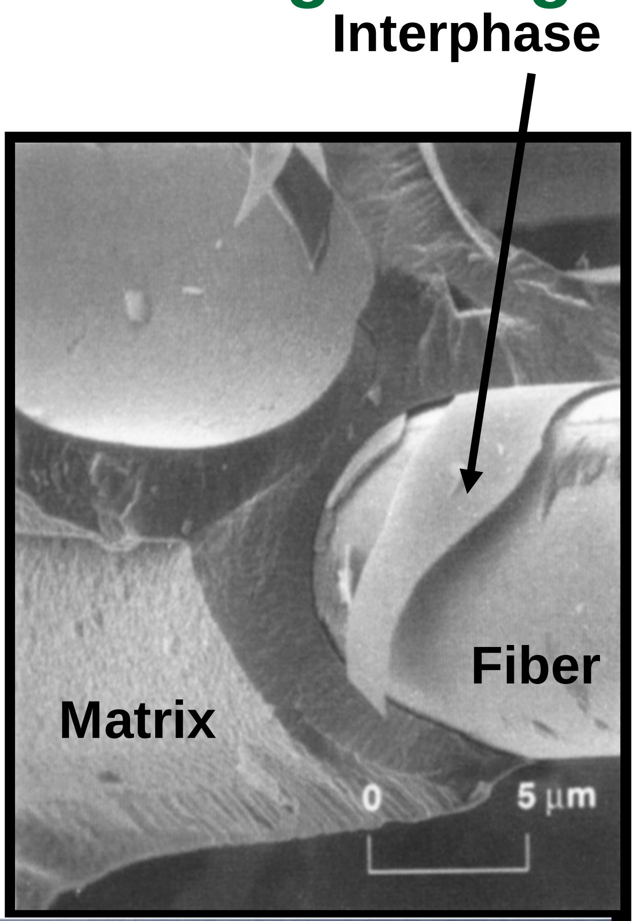
*** does not include prototyping or NDE evaluation.**

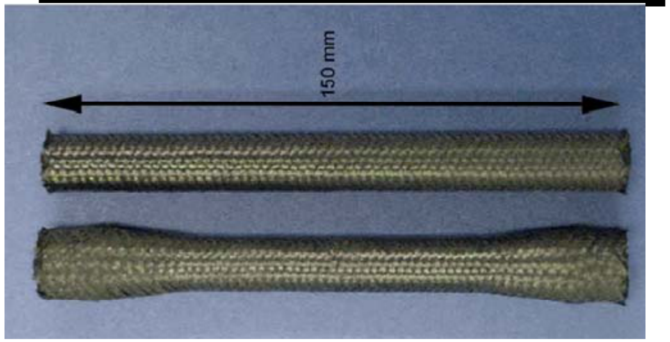
NGNP	Operating Temp	Maximum Temp	Lifetime Dose
Control Rods & Guide Tubes	**1200°C**	**1600°C**	**25 dpa**
Upper Plenum Shroud/Core Restraint	**650°C**	**1300°C**	**0.05 dpa**
Floor Blocks	**600°C**	**600°C**	**<0.05 dpa**
Hot Duct Inner Shell	**1000°C**	**1200°C**	**0.005 dpa**

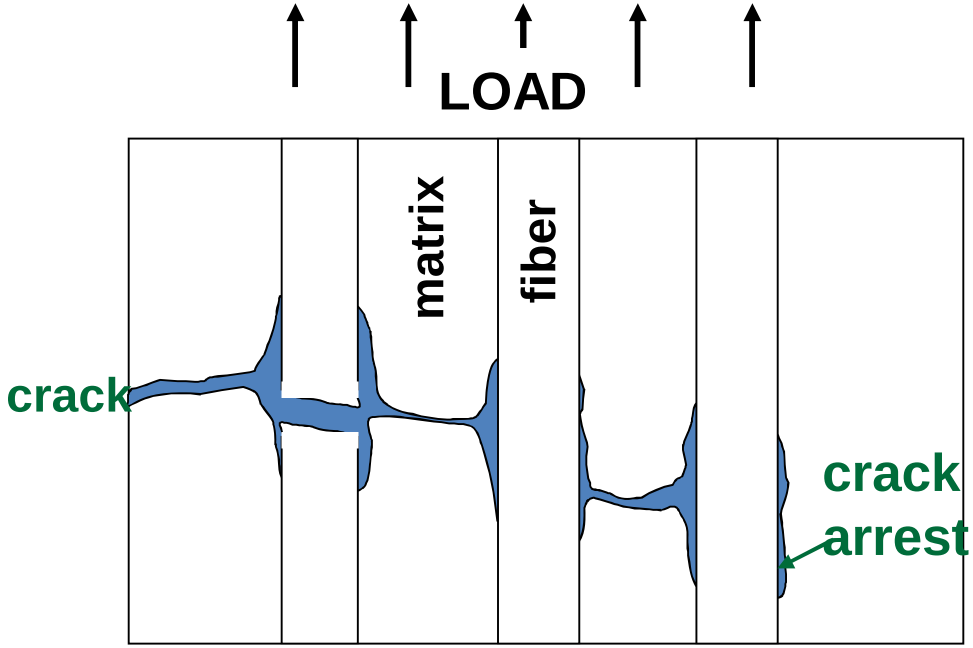
Composite Materials for Generation IV Reactors



Fiber-reinforced Ceramic Composites provide High Toughness vs. Monolithic Ceramics







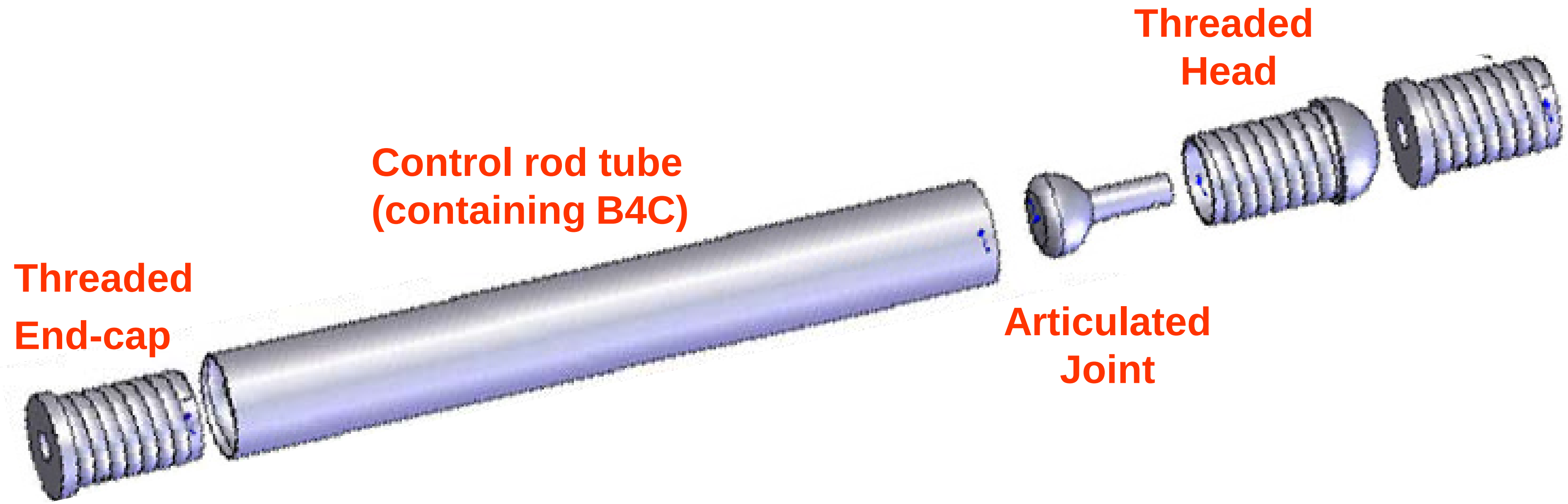
Composite materials, whether platelet, chopped fiber, or continuous fiber reinforced are superior "engineering" materials to monolithics:

- **generally higher strength, especially in tension**
- **higher Weibull modulus (more uniform failure)**
- **much higher damage tolerance (fracture toughness)**

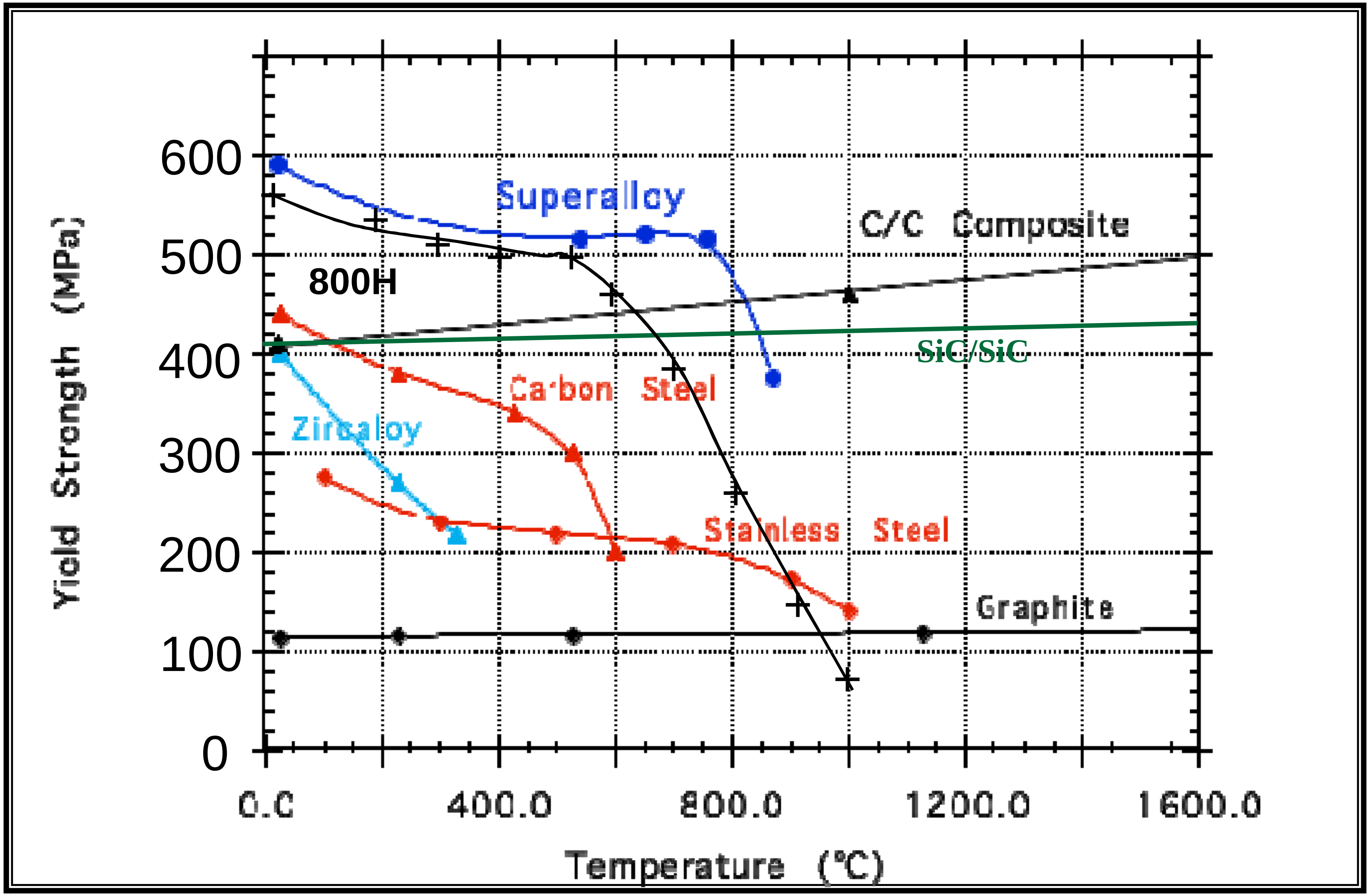
Composite Materials for Generation IV Reactors

- In all gas cooled reactors there is a need for in-core components which are subjected to tensile loading. Nuclear graphite, or other "engineering" ceramics can not be used for such applications.

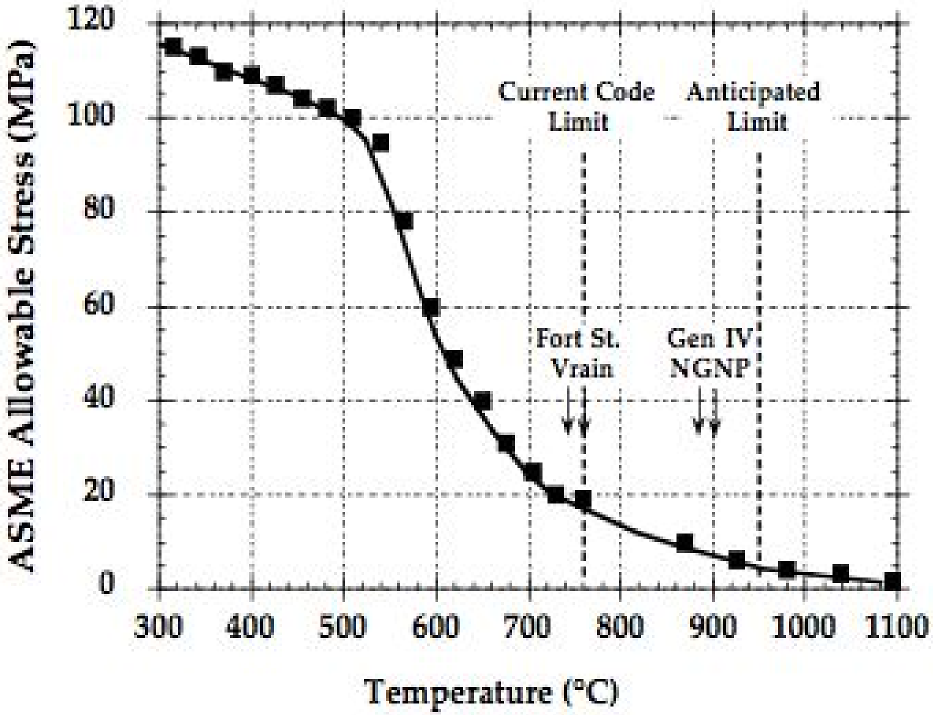
- Components such as hangers and control rods are typically under modest tensile load (tens of MPa) and are not at the highest core fluence.

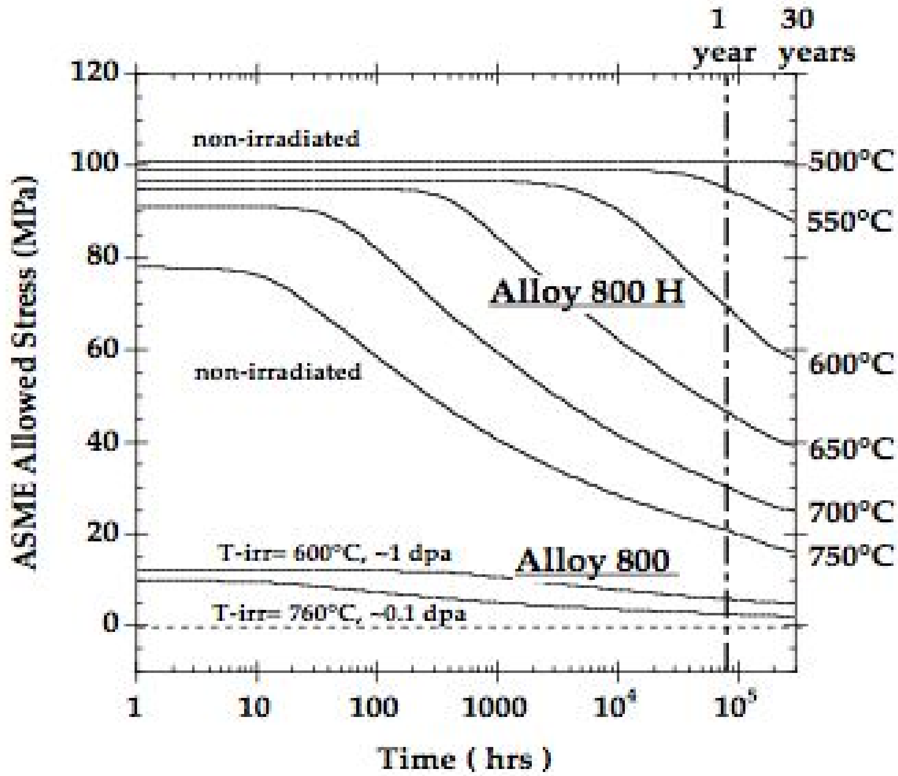


Yield Strength of Various Structural Materials

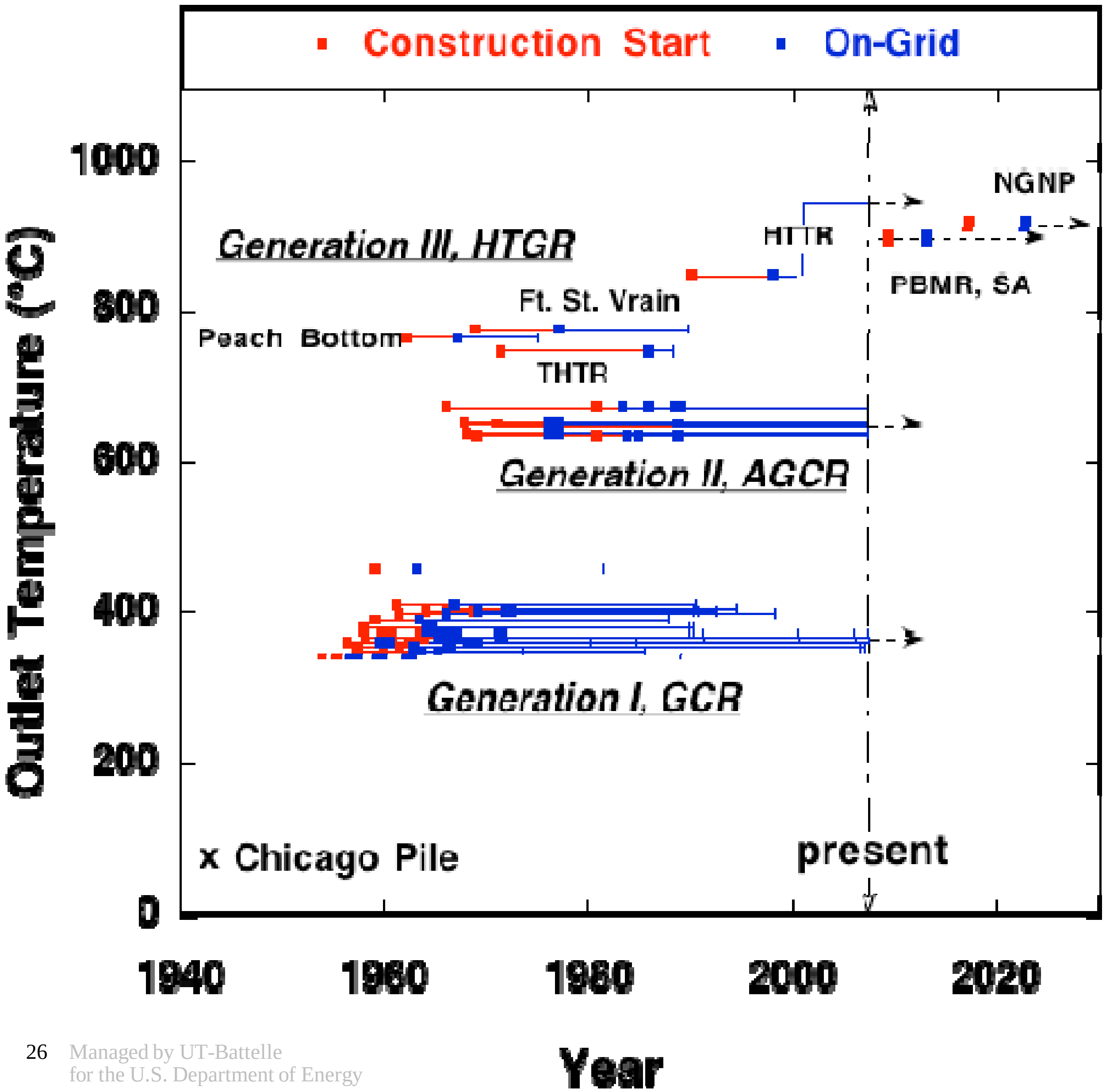


Performance of Candidate Alloy 800H Control Rod





Potential Need for Composite Materials for Gen IV

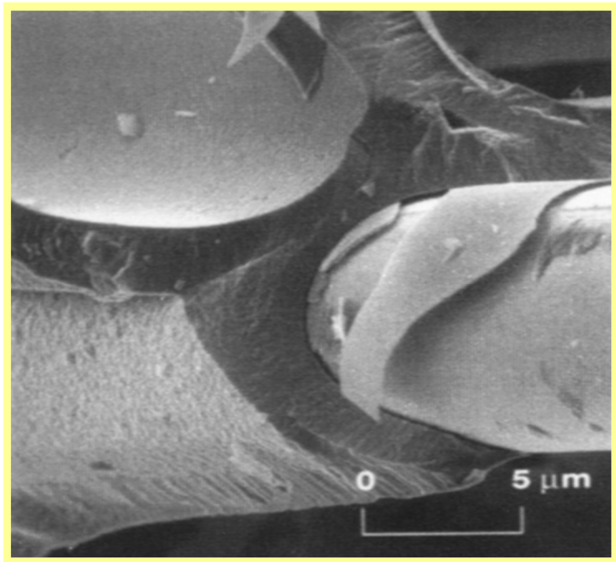


Alloy 800H Code Extension
(allowed stress ~5MPa)

Alloy 800H Code Limit
(allowed stress ~19MPa)

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
- Key Research Topics
  - 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

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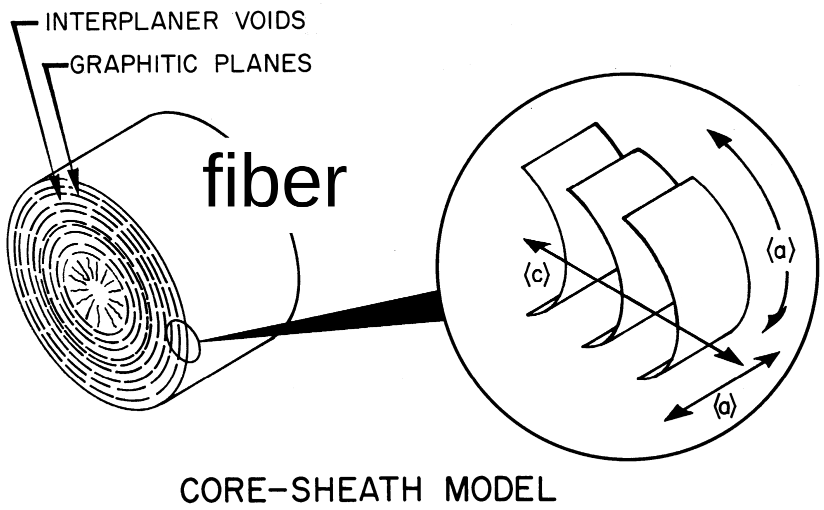
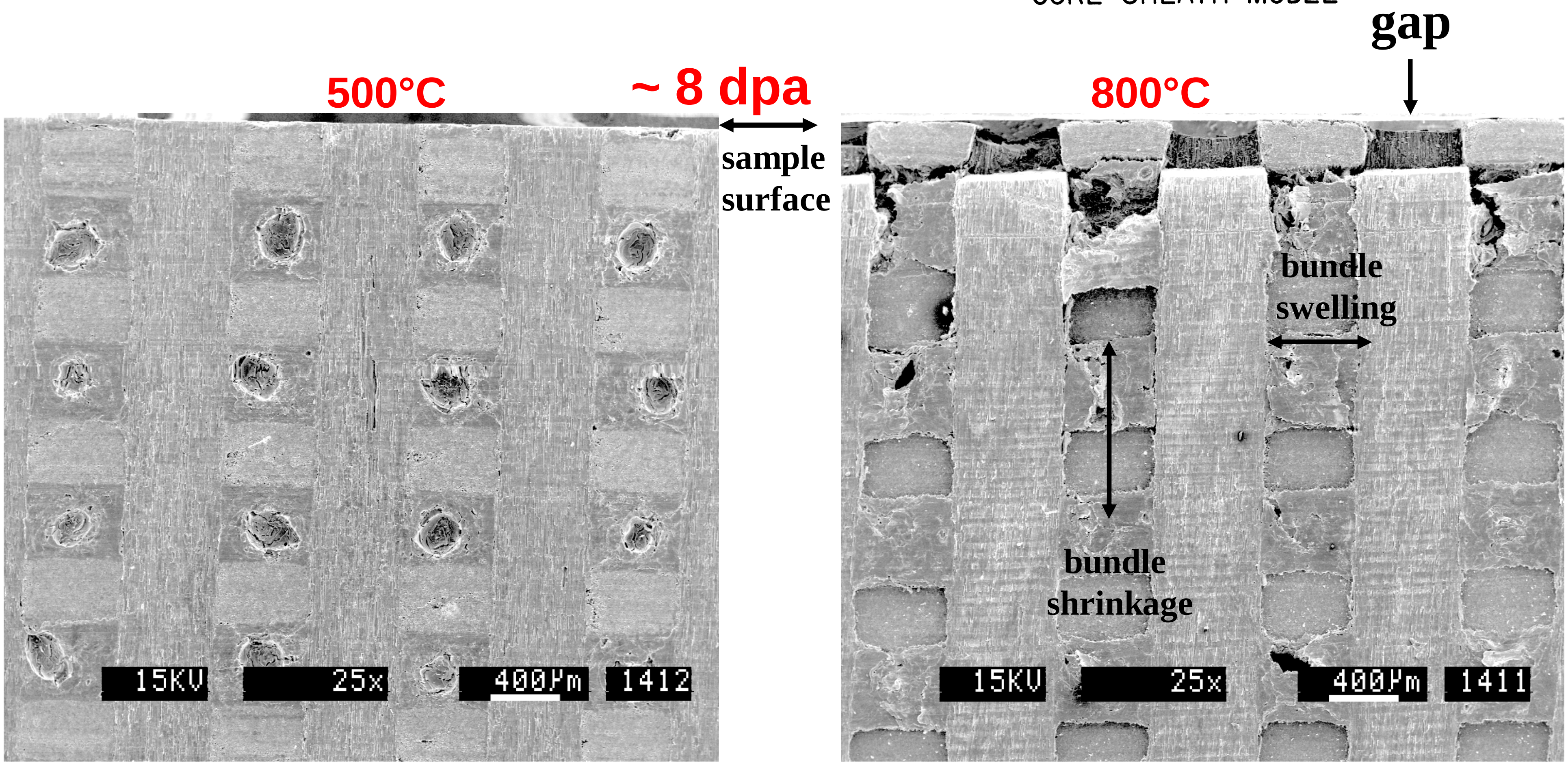
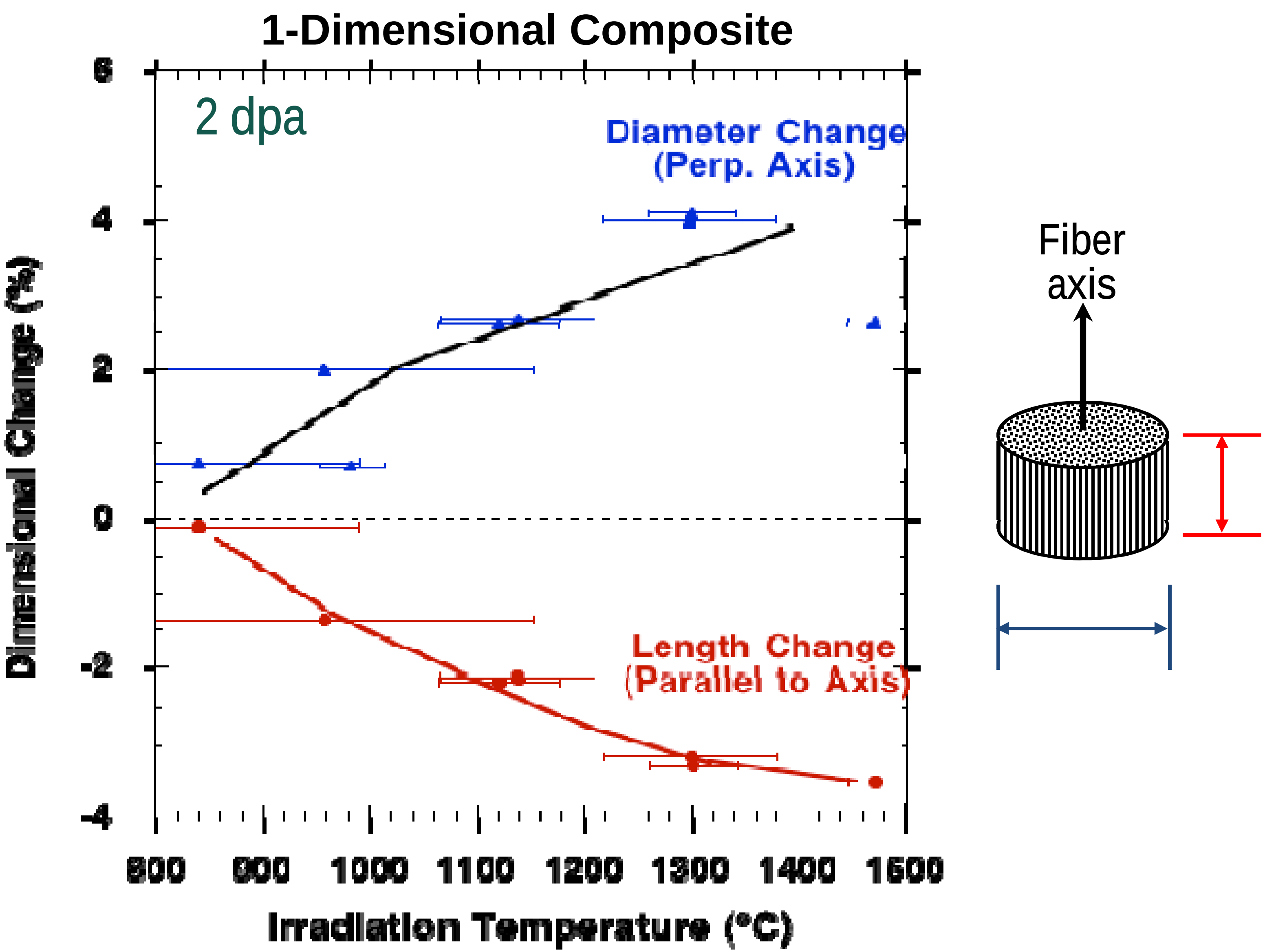
CFC’s Under Irradiation

Elevated temperature irradiation begins to show severe degradation



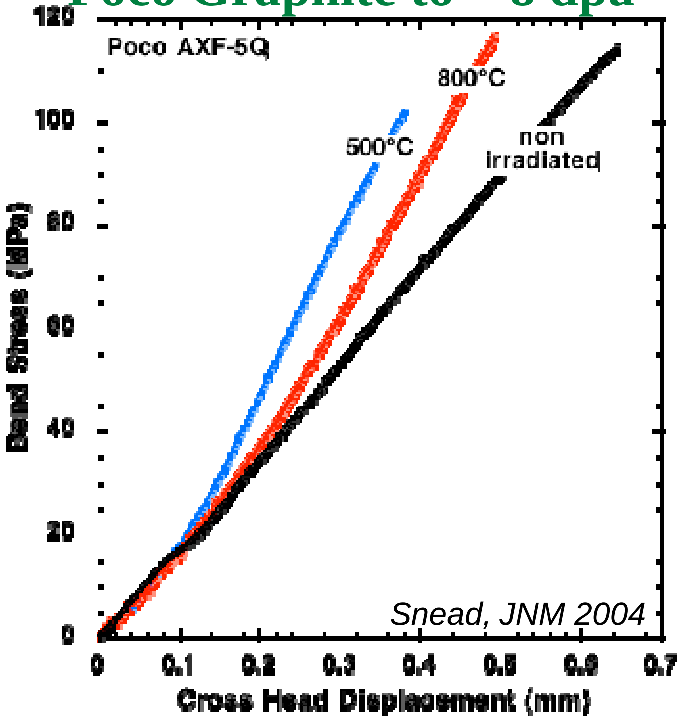


Dimensional Change in 1-D Composite

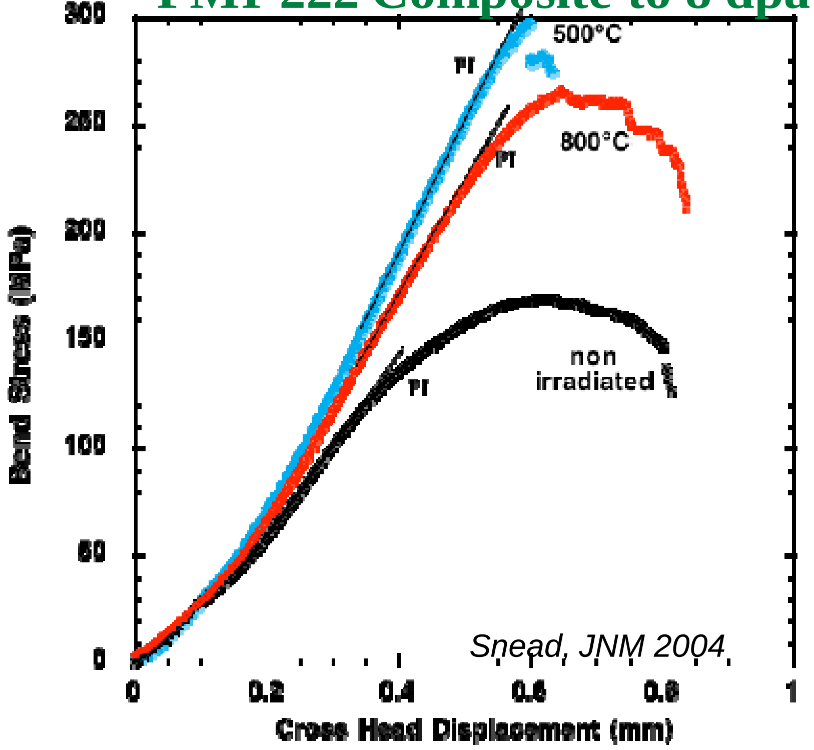


Candidate Nuclear Graphite Compared to CFC

Poco Graphite to ~ 8 dpa



FMI-222 Composite to 8 dpa



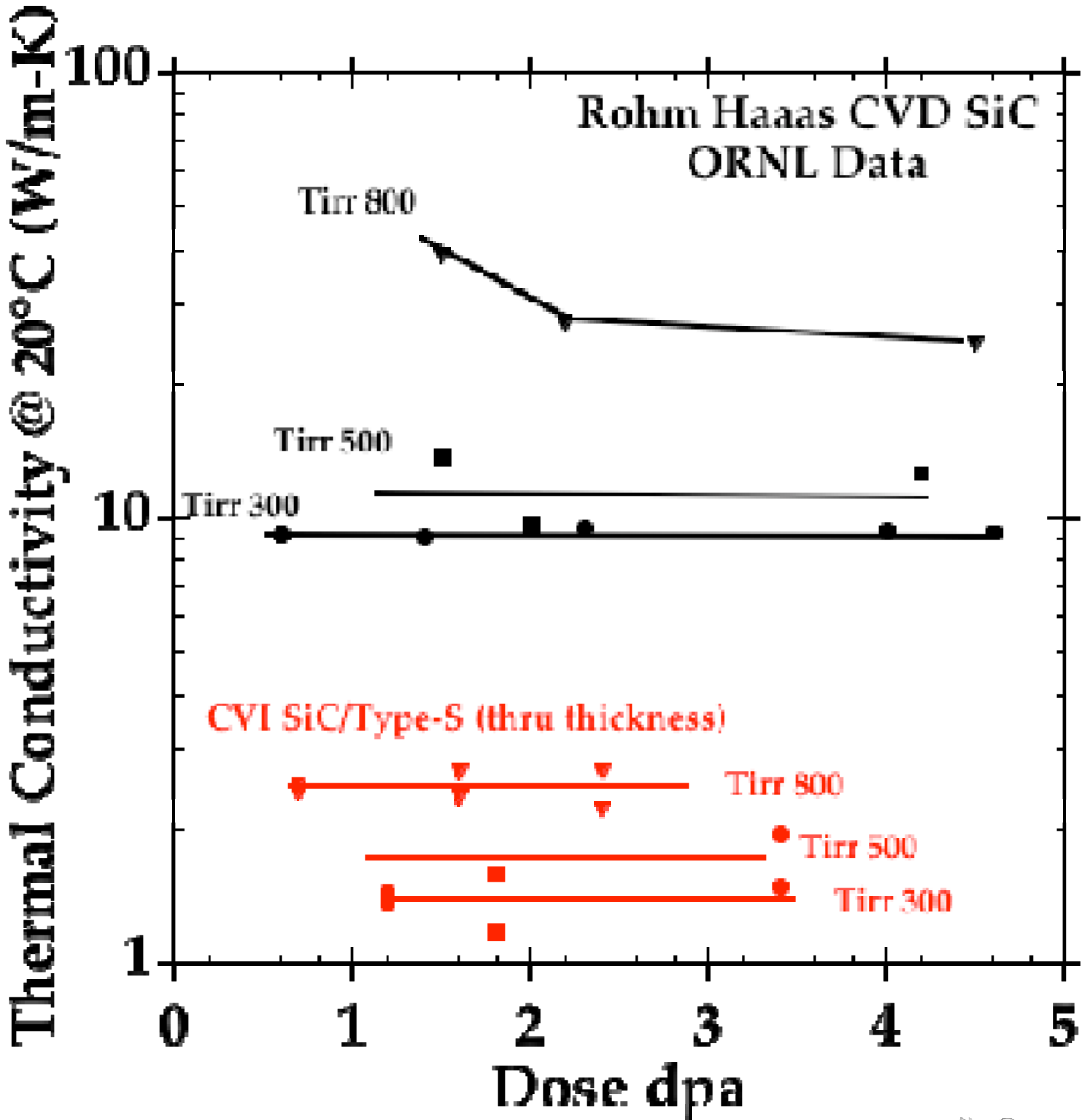
	Virgin	500°C 6 x10^{25} n/m^2	800°C 7.7 x10^{25}n/m^2
Strength MPa	113±20	107 ±7 (-5%)	98 ±11 (-13%)
Length Change	-	0.1 %	1.11 %

	Virgin	500°C 6 x 10^{25} n/m^2	800°C 7.7 x 10^{25}n/m^2
Strength MPa	176 ±20	286 ±25 (+63%)	241±22 (+37%)
Length Change	-	- 1.5 %	-3.6 %

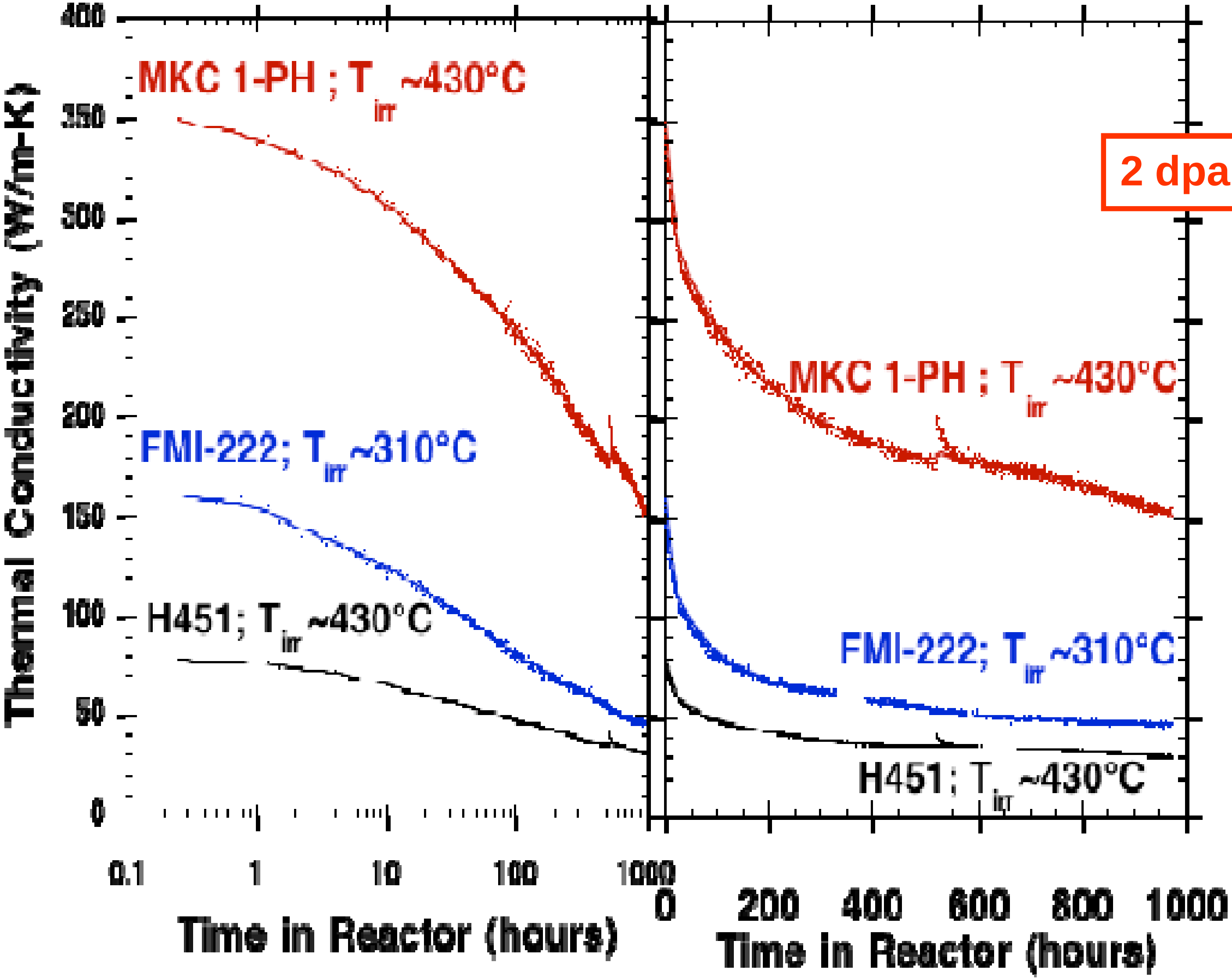
SiC/SiC Composites : Thermal Conductivity

Due to “interfaces” and cracks in SiC composite, thermal conductiivity will necessarily be less than ideal SiC.

Present materials are significanlty lower than ~15 W/m-K reactor study goal.



Comparison of Radiation-induced Thermal Conductivity Degradation of Carbon-fiber composites



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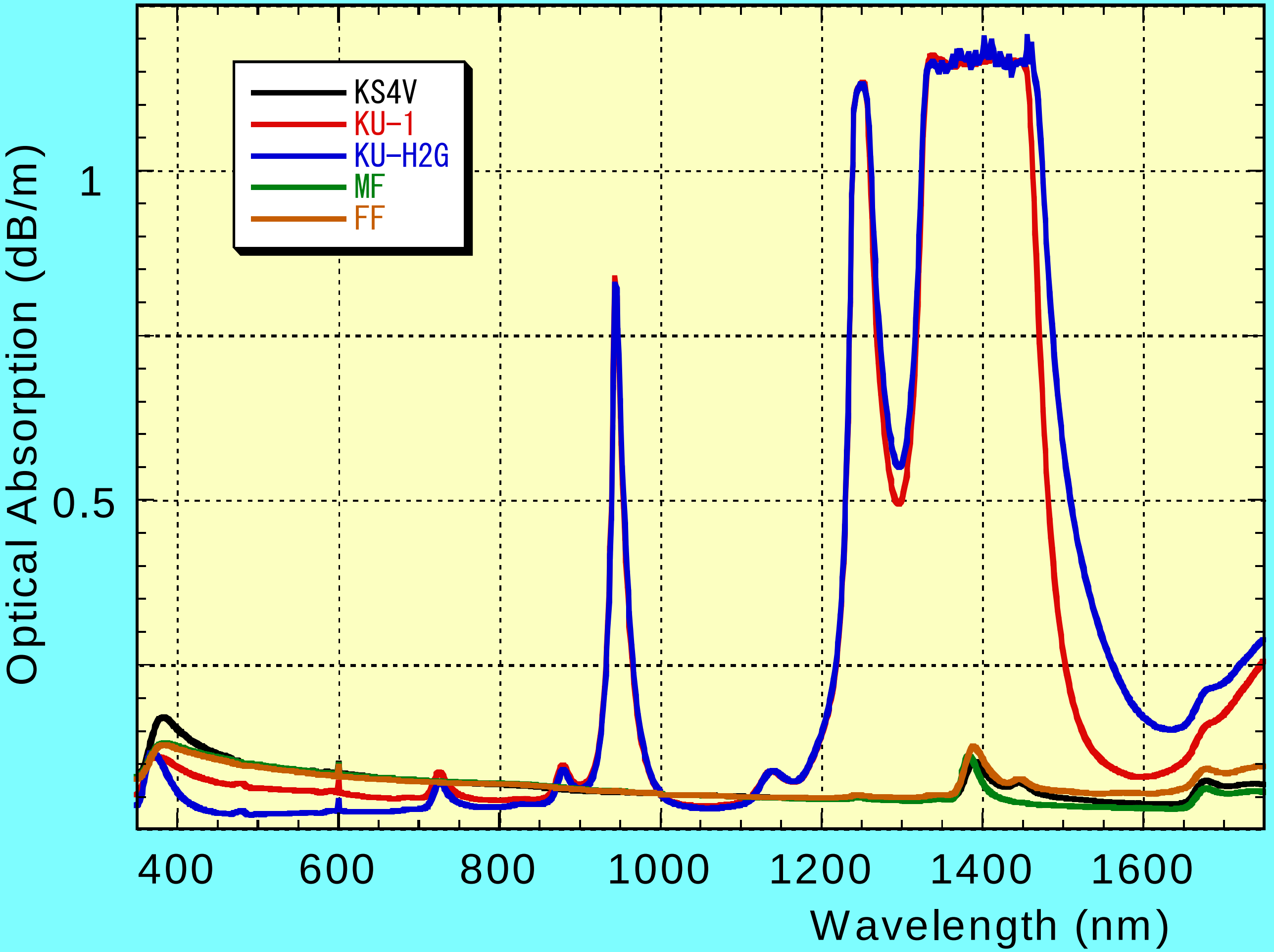
Comments on SiC and carbon fiber composites

- **In order to transition from current proof-of-principal studies to an application phase for ceramic composites, a more comprehensive program is needed:**
  - **Fundamental understanding of engineering properties such as swelling, thermal conductivity, irradiation creep, environmental corrosion, etc.**
  - **Demonstration of scalability and component Q/A.**
  - **Development of ASTM accepted testing standards and coordination with ASME for codification of the material.**

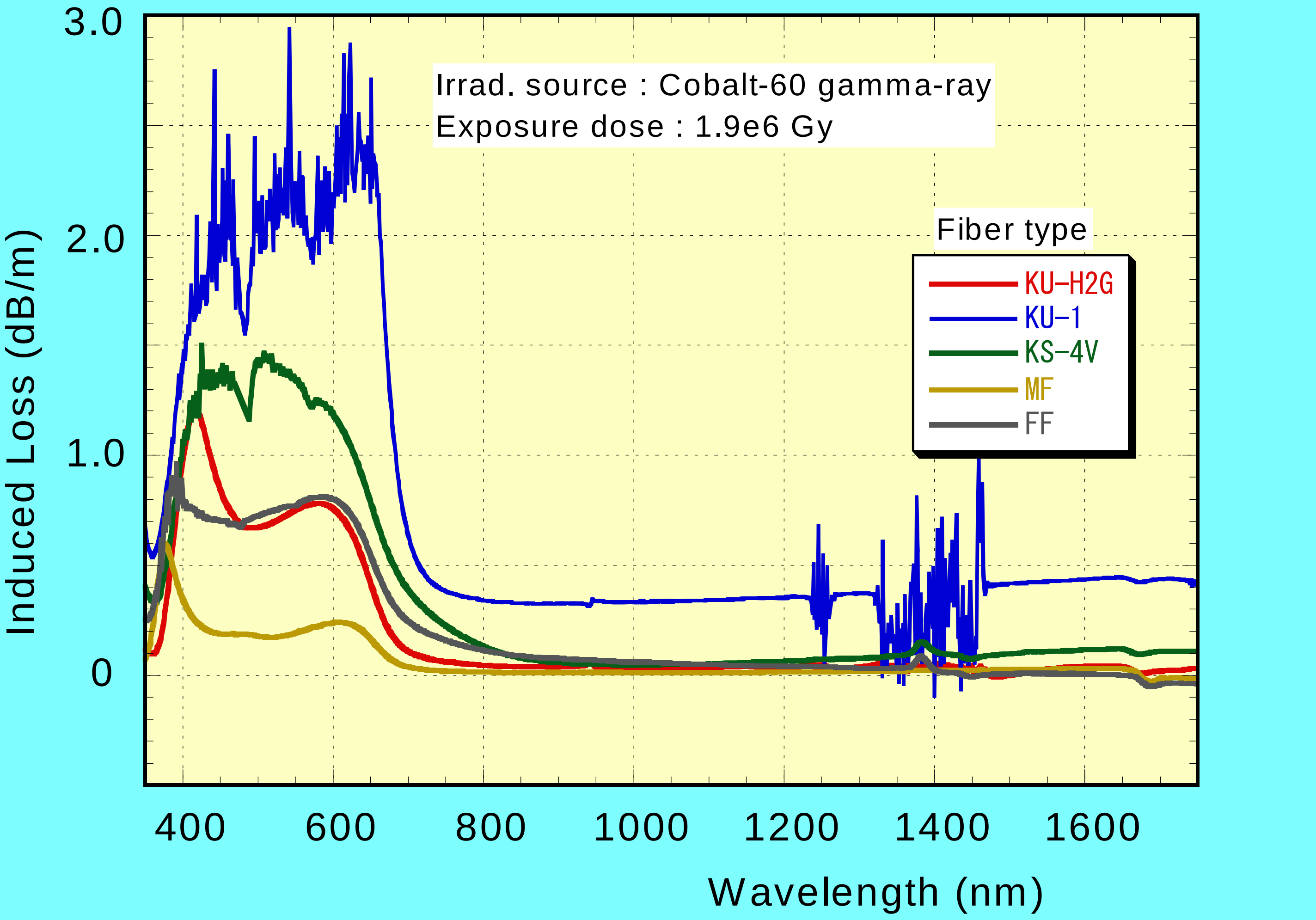
- **The ultimate choice between Carbon Fiber Composite and SiC/SiC will primarily depend on the dose and temperature of application:**
  - **CFC's : High Irradiated Thermal Conductivity (>50 W/m-K)**
    Limited Lifetime (10 dpa for T , ~ 700°C) (~ 1 dpa T>1000°C)
  - **SiC/SiC : Poor thermal conductivity (< 5 W/m-K)**
    Apparent Long Lifetime (> 10 dpa, T< 1300°C)



Initial Transmission Characteristics of ITER Round Robin Fibers

- All fibers have 200μm core and 250μm cladding.
- Fibers KS-4V, KU-1 and KU-H2G are manufacutured by FORC, RF.
- KS-4V fiber has OH-free silica core and both KU fibers have 800ppm OH contained silica core.
- Fibers MF and FF, fluorine doped silica core, were manufactured by Mitsubishi Cable and Fujikura Ltd., JPN.

T. Kakuta et al.



Comparison of increased absorption in ITER round robin fibers at gamma-ray irradiation of 1.9e6 Gy.

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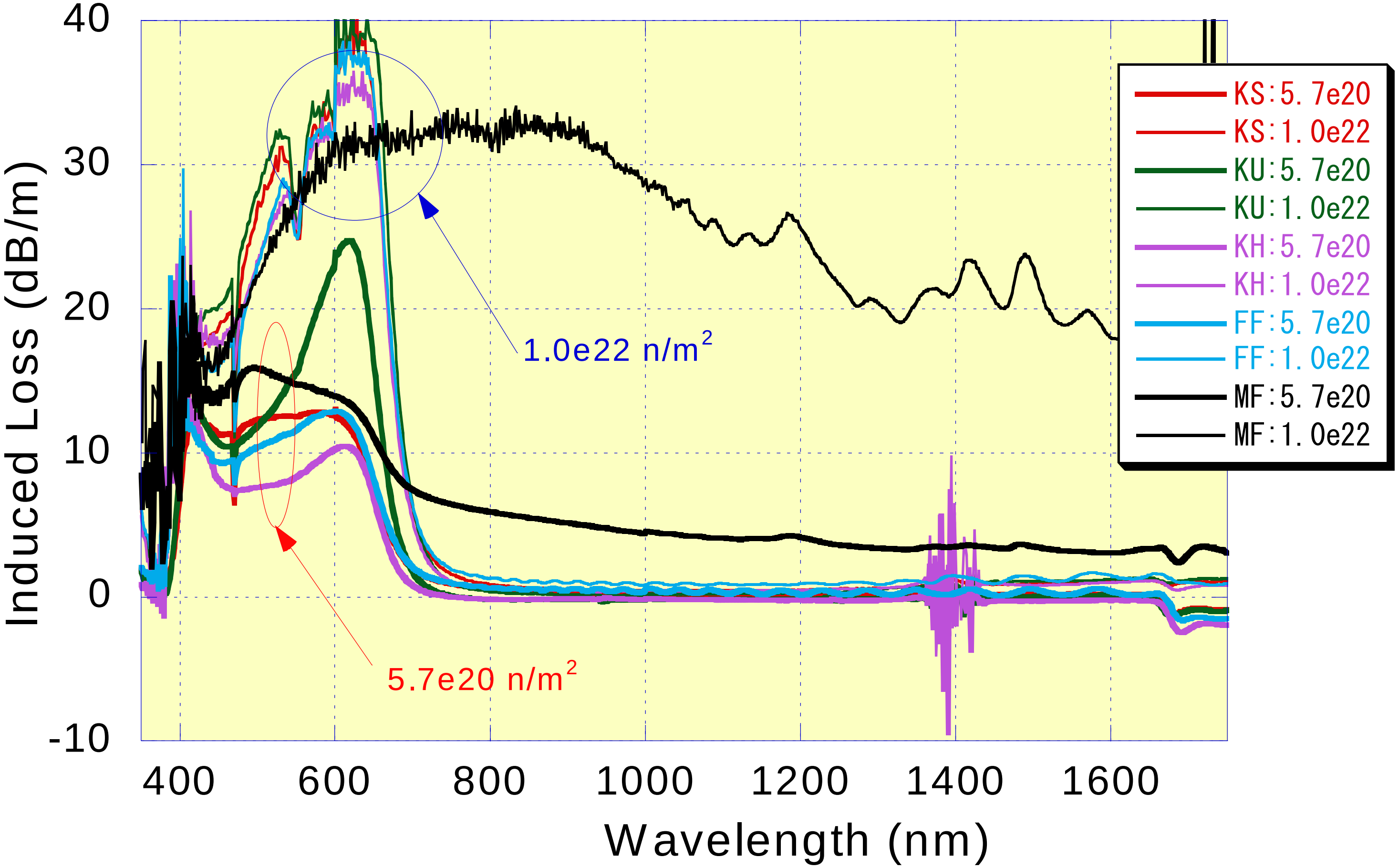

Observed absorption of ITER round robin fibers during fission neutron irradiation.

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Summary

- Nonmetallic materials provide important capabilities for a wide range of current and proposed fission and fusion reactors
  – Generally based on high tolerance to extreme temperatures and radiation levels, or unique optical or insulating properties of ceramics
    - Microencapsulated fuel systems for gas-cooled and accident-tolerant water cooled reactor designs
    - SiC/SiC and CFC composites for structural applications
    - Neutron control systems for fission reactors
    - Diagnostic applications (optical fibers, windows, bulk insulators, MI cables)
    - Plasma heating systems
    - Fusion ceramic breeders
    - Fission waste storage forms