



U.S. DEPARTMENT OF
ENERGY

Nuclear Energy

Fuel Cycle Research and Development Program

Candidate ATF Clad Technologies and Key Feasibility Issues

Lance Snead
Oak Ridge National Laboratory



Enhanced Accident Tolerant Fuel

Delay and/or Reduce
Heat and Hydrogen
Production

Reduced Clad
Reaction Kinetics

ATF

Improved Clad
Properties

Coolable Geometry
Time to Rupture
Time to Melt

Matrix retention,
Entrainment

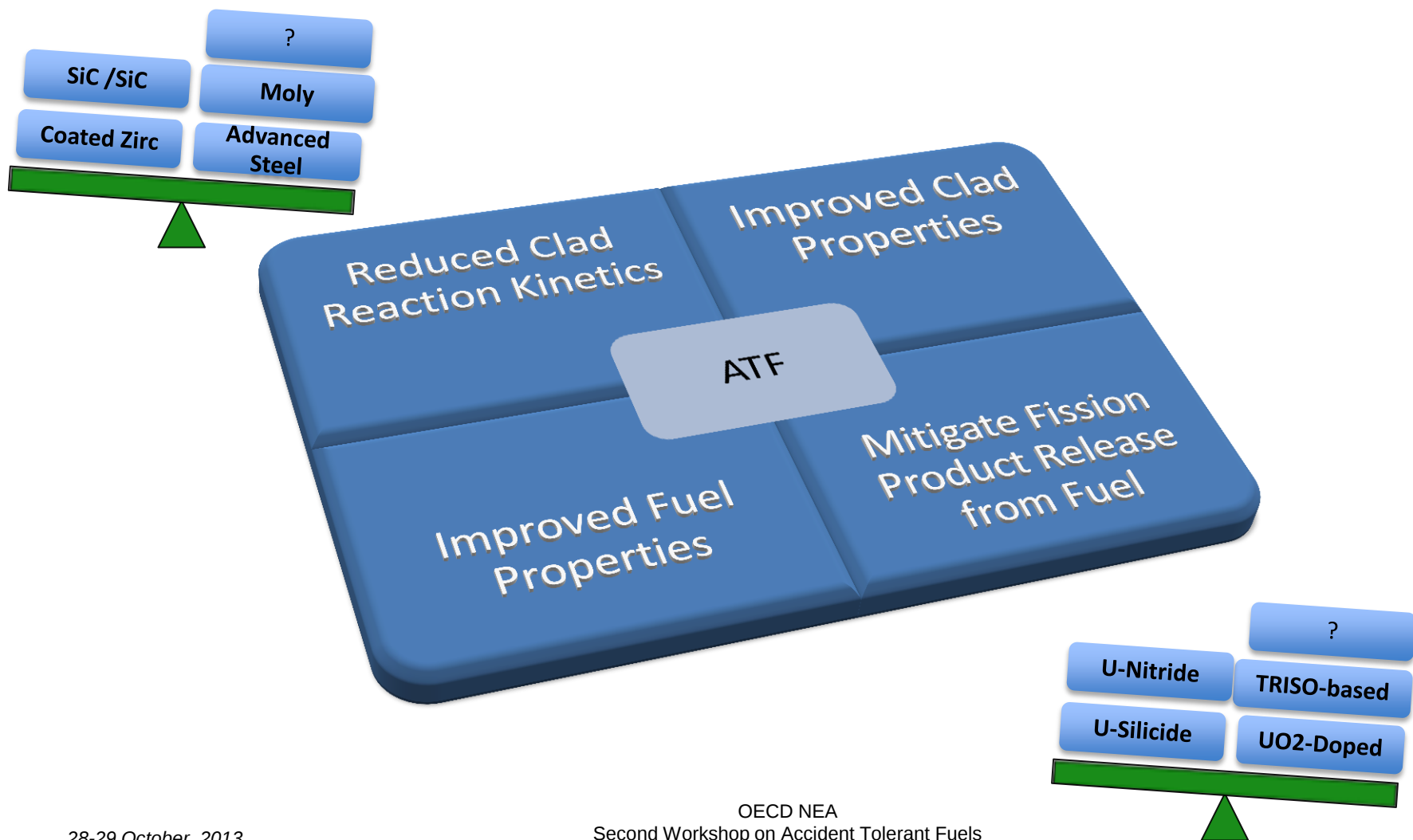
Mitigate Fission
Product Release
from Fuel

Improved Fuel
Properties

Enhanced K-th
Improved toughness
Improved Irr. Perf.

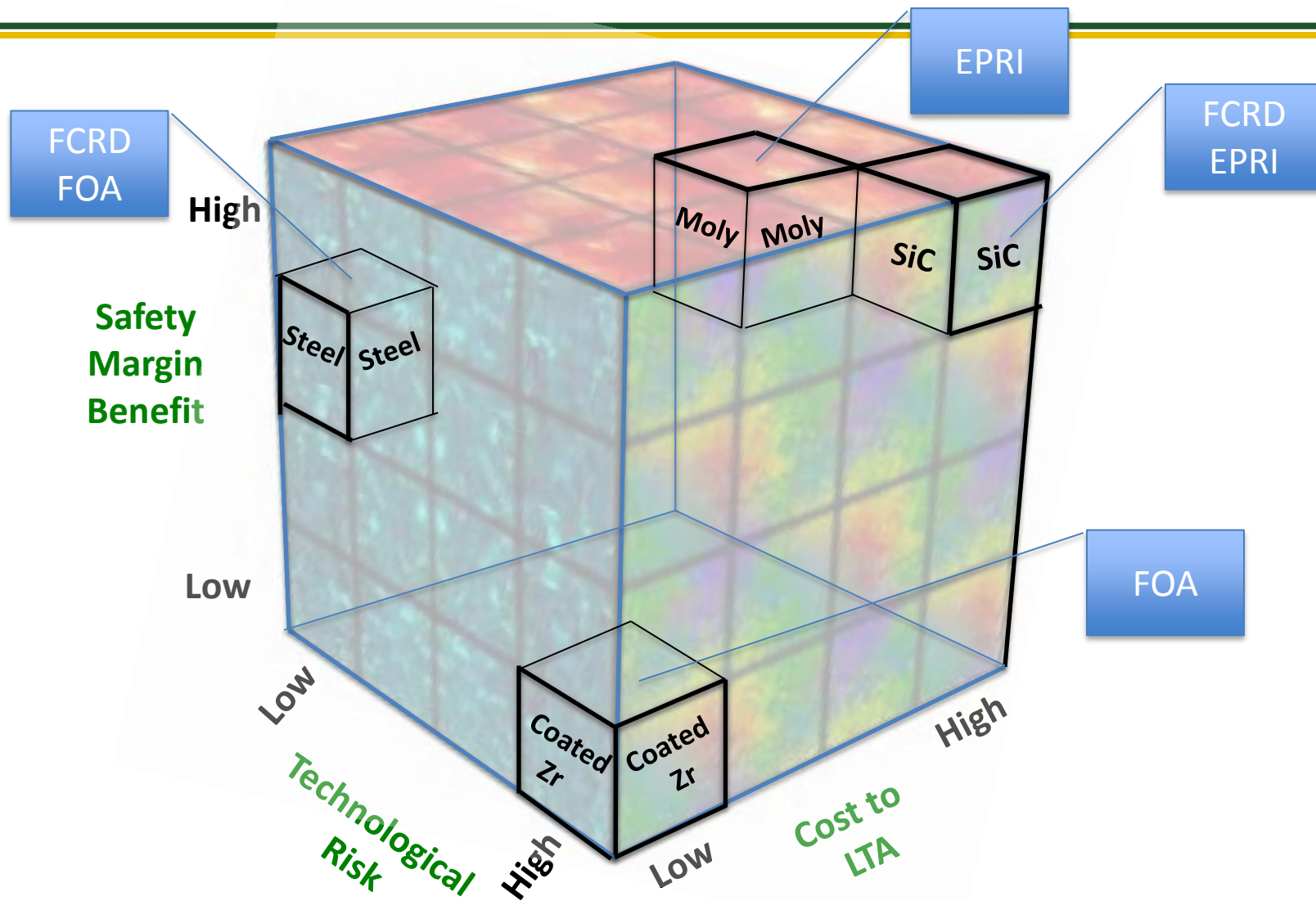


Enhanced Accident Tolerant Fuel



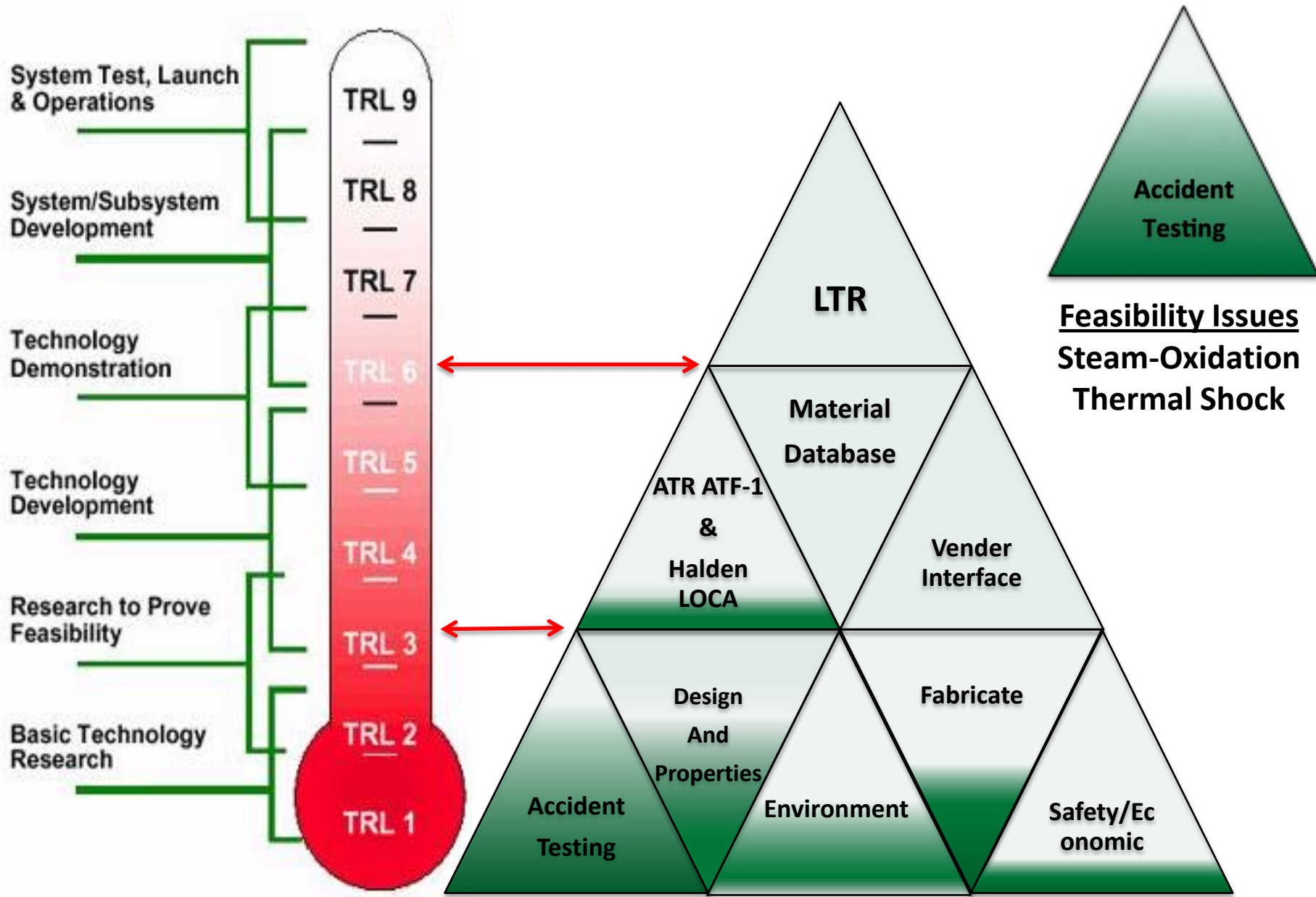


Enhancing Safety Margin with Advanced Fuels - Clad Options -



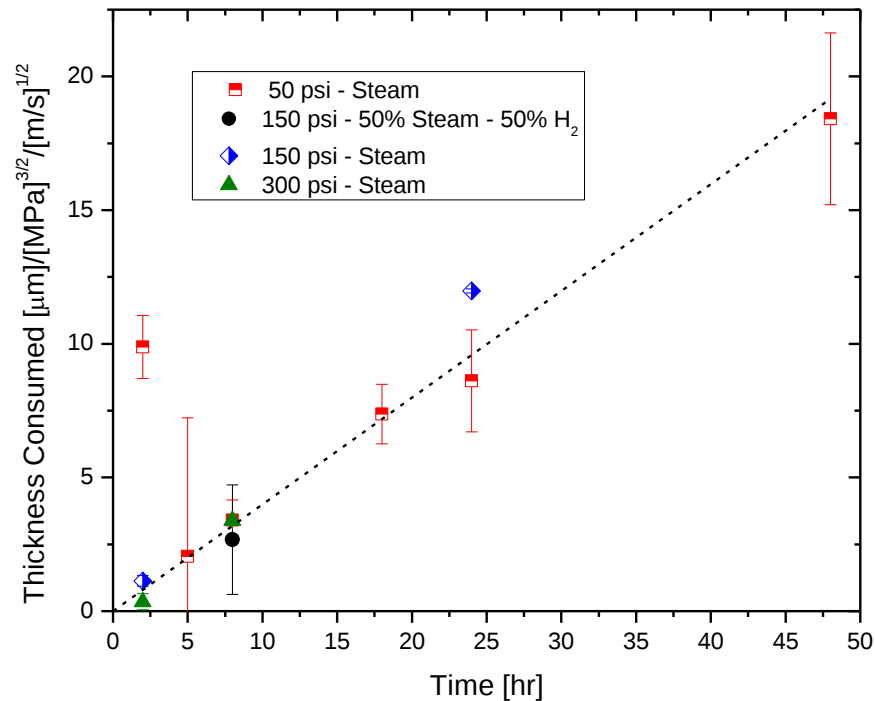
SiC/SiC : Path to ATR Irradiation and Commercial LTR of ATF

- Key performance issues are addressed on path to high-value irradiation and development.



Progress Since Last OECD Meeting

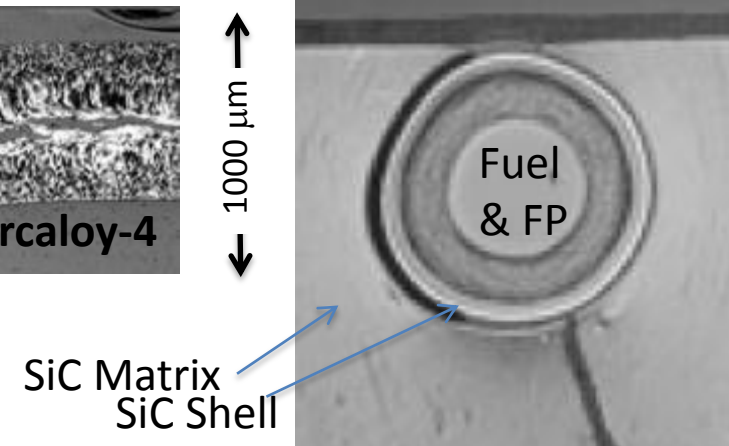
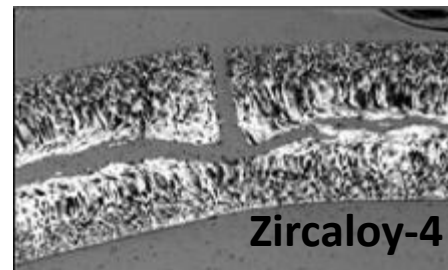
- Mapping of SiC steam Attack -



Feasibility Issues
Steam-Oxidation
Thermal Shock

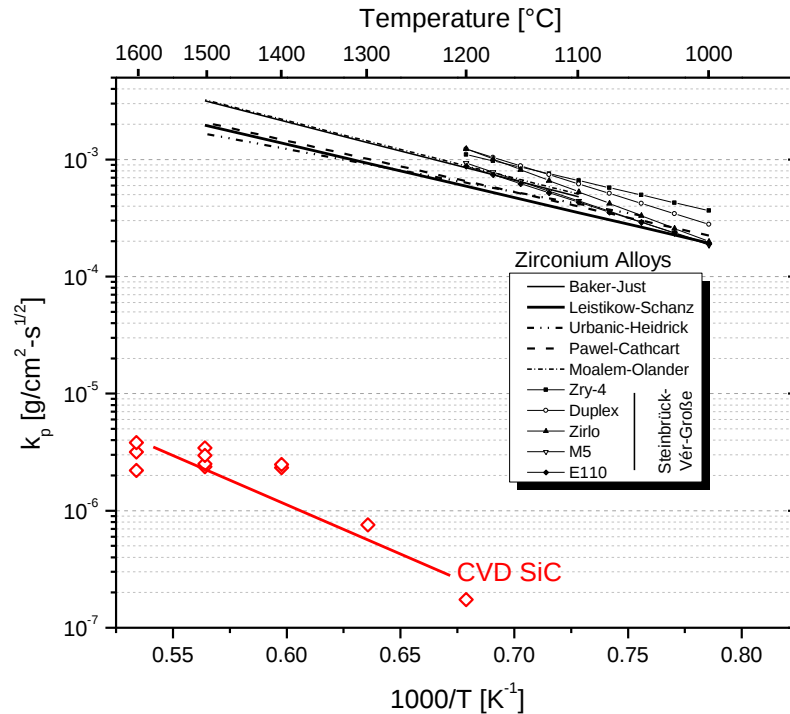


- Steam attack on SiC has been catalogued from 900-1600°C as a function of pressure and mechanisms determined.
- Publication in Journal of Nuclear Materials



Progress Since Last OECD Meeting

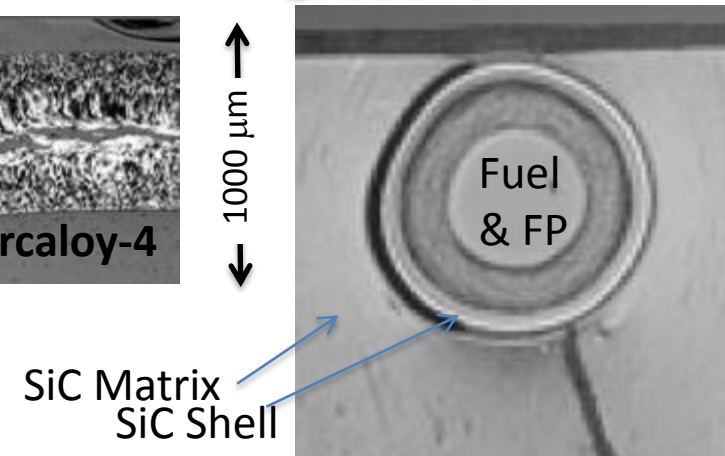
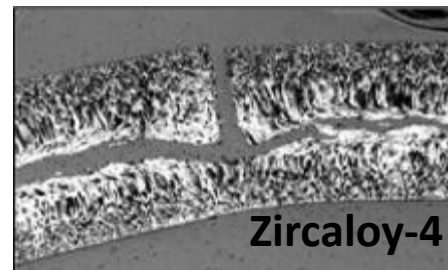
- Mapping of SiC steam Attack -



Feasibility Issues
Steam-Oxidation
Thermal Shock

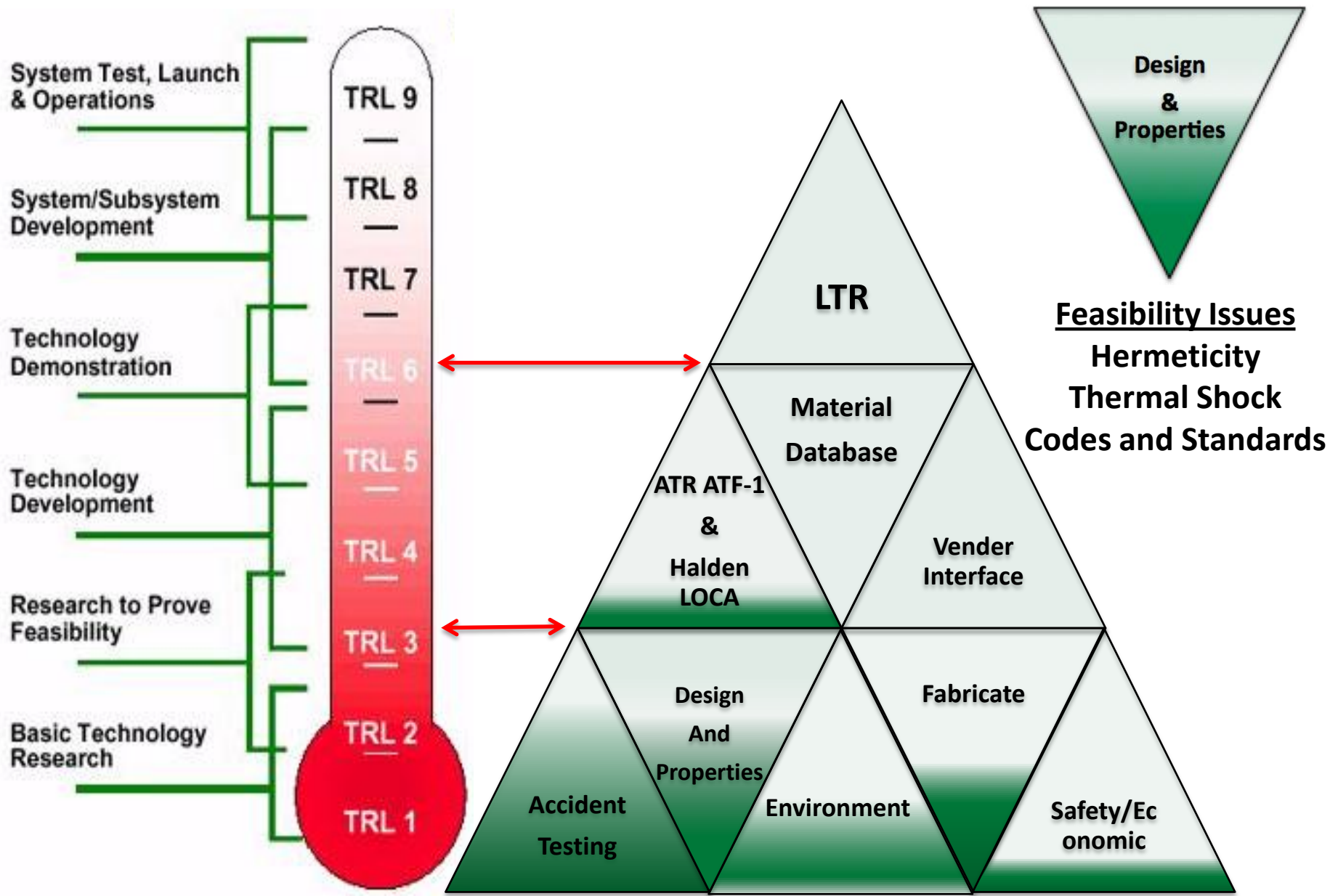
SiO₂ **Attack and Recession**

- Steam attack on SiC has been catalogued from 900-1600°C as a function of pressure and mechanisms determined.
- Publication in Journal of Nuclear Materials



SiC/SiC : Path to ATR Irradiation and Commercial LTR of ATF

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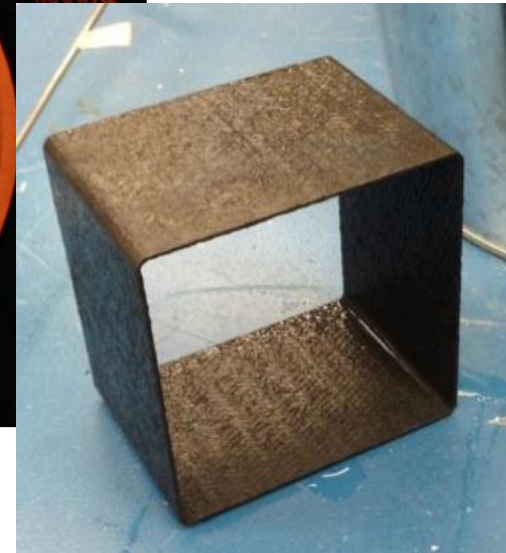
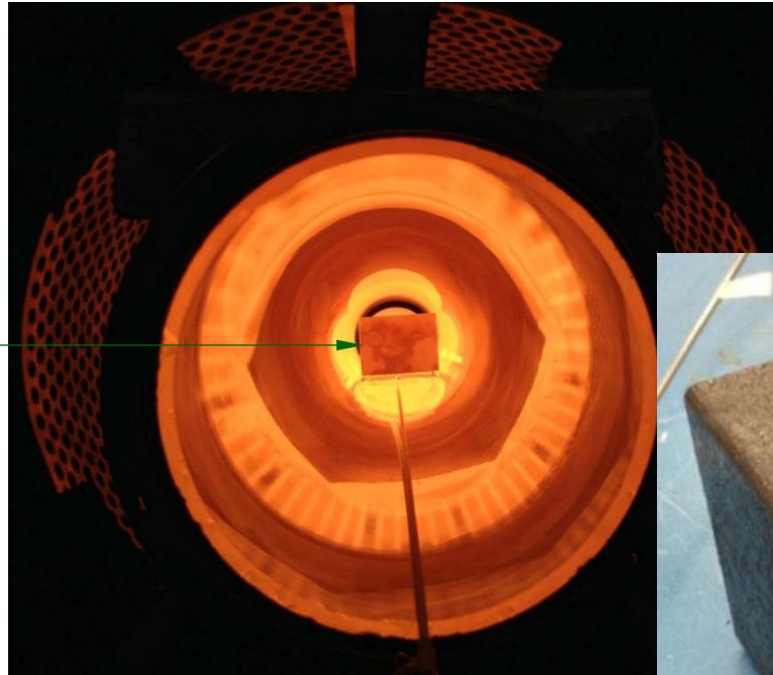


Progress Since Last OECD Meeting

- Thermal Shock -

Design
&
Properties

SiC/SiC
composite
channelbox
inside infrared
(IR) furnace at
~1150°C

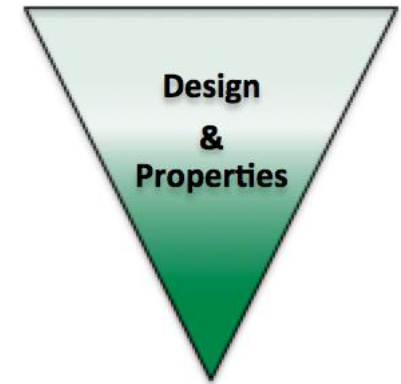
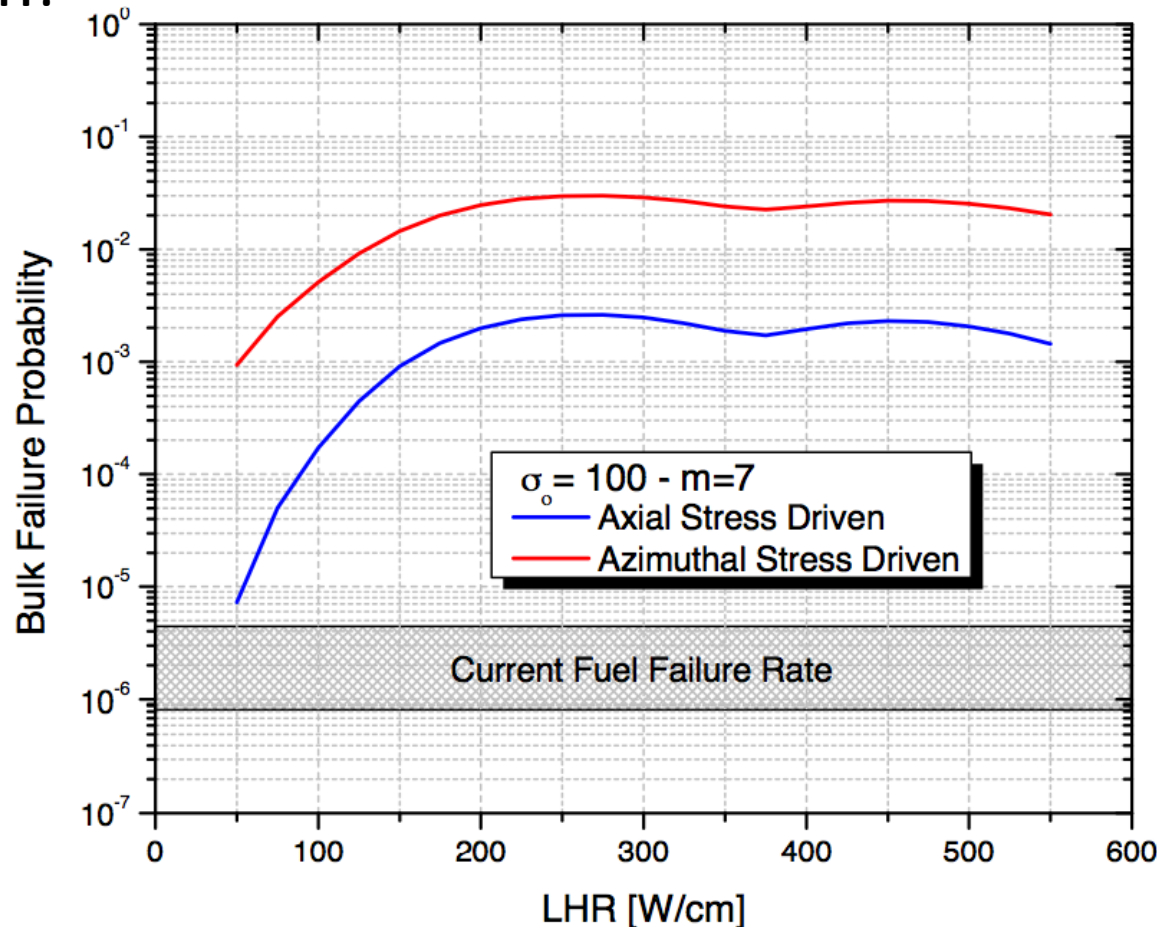


- Thermal shock viability of SiC/SiC channel box prototype successfully demonstrated.

Progress Since Last OECD Meeting

- Design-

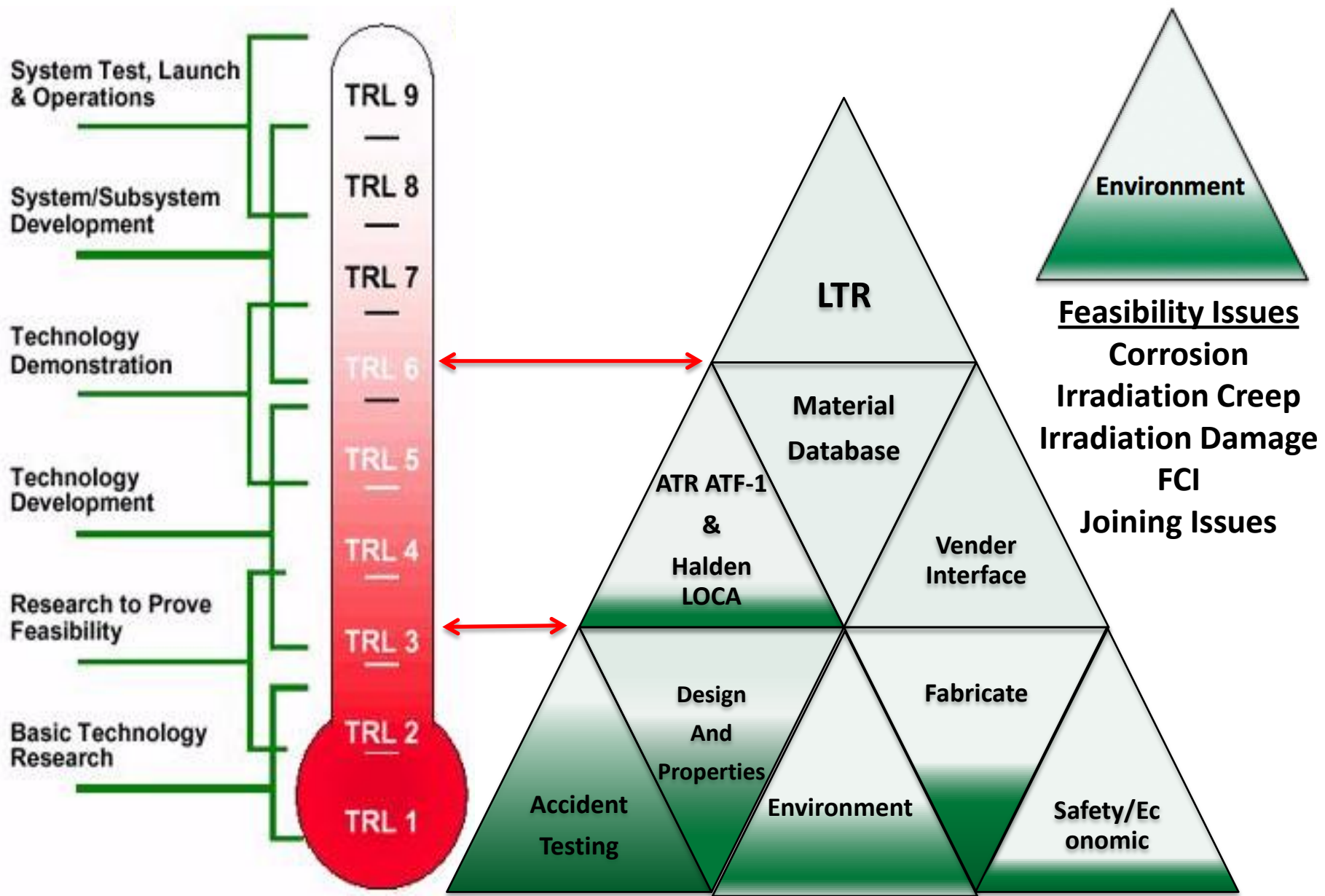
- JNM publication of an analysis to determine the possibility/probability of failure, as determined by FP release of SiC tubes. Analysis suggests fully ceramic designs will not hold FP.



Feasibility Issues
Hermeticity
Thermal Shock
Codes and Standards

SiC/SiC : Path to ATR Irradiation and Commercial LTR of ATF

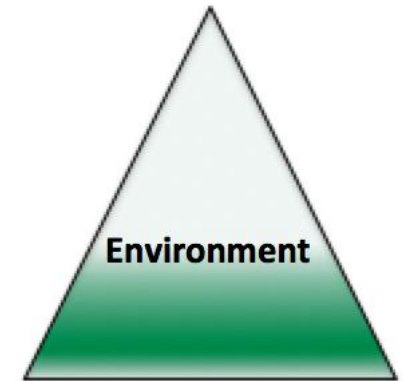
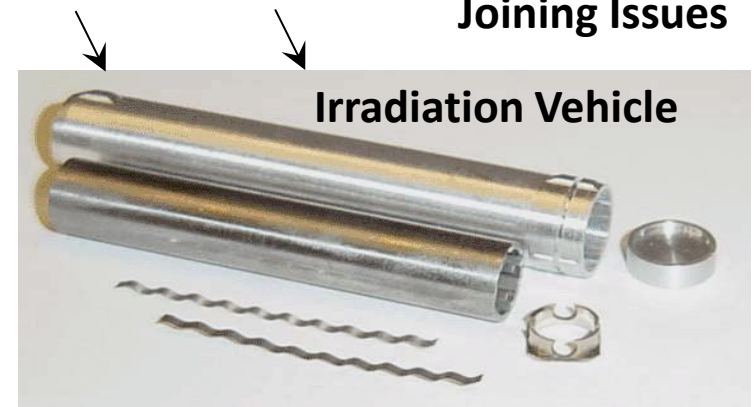
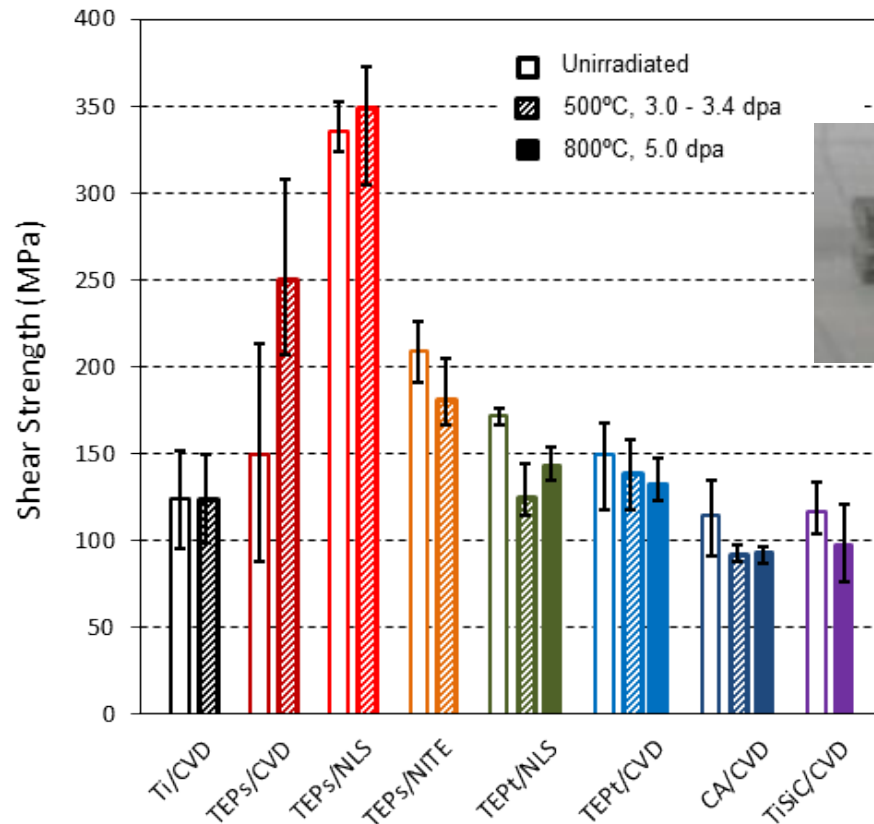
- Key performance issues are addressed on path to high-value irradiation and development.



Progress Since Last OECD Meeting

- Environment-

- **Irradiation Creep** : Fundamental mechanisms published JNM.
- **Irradiation Damage** : Handbook of SiC composite properties published in JNM.
- **Results of successful joining study** published in JNM. Study to be completed to full LWR lifetime dose.

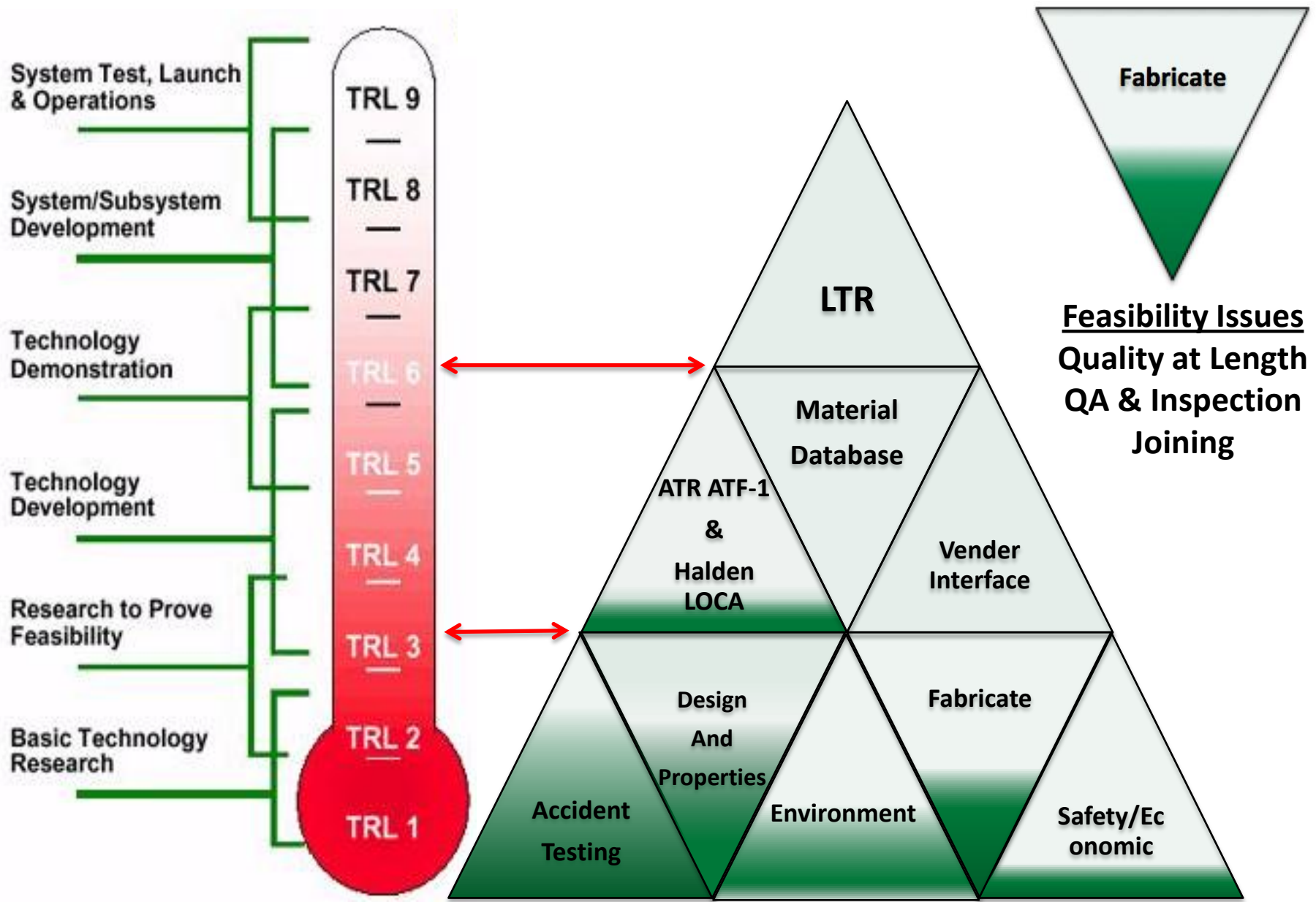


Feasibility Issues

Corrosion
Irradiation Creep
Irradiation Damage
FCI
Joining Issues

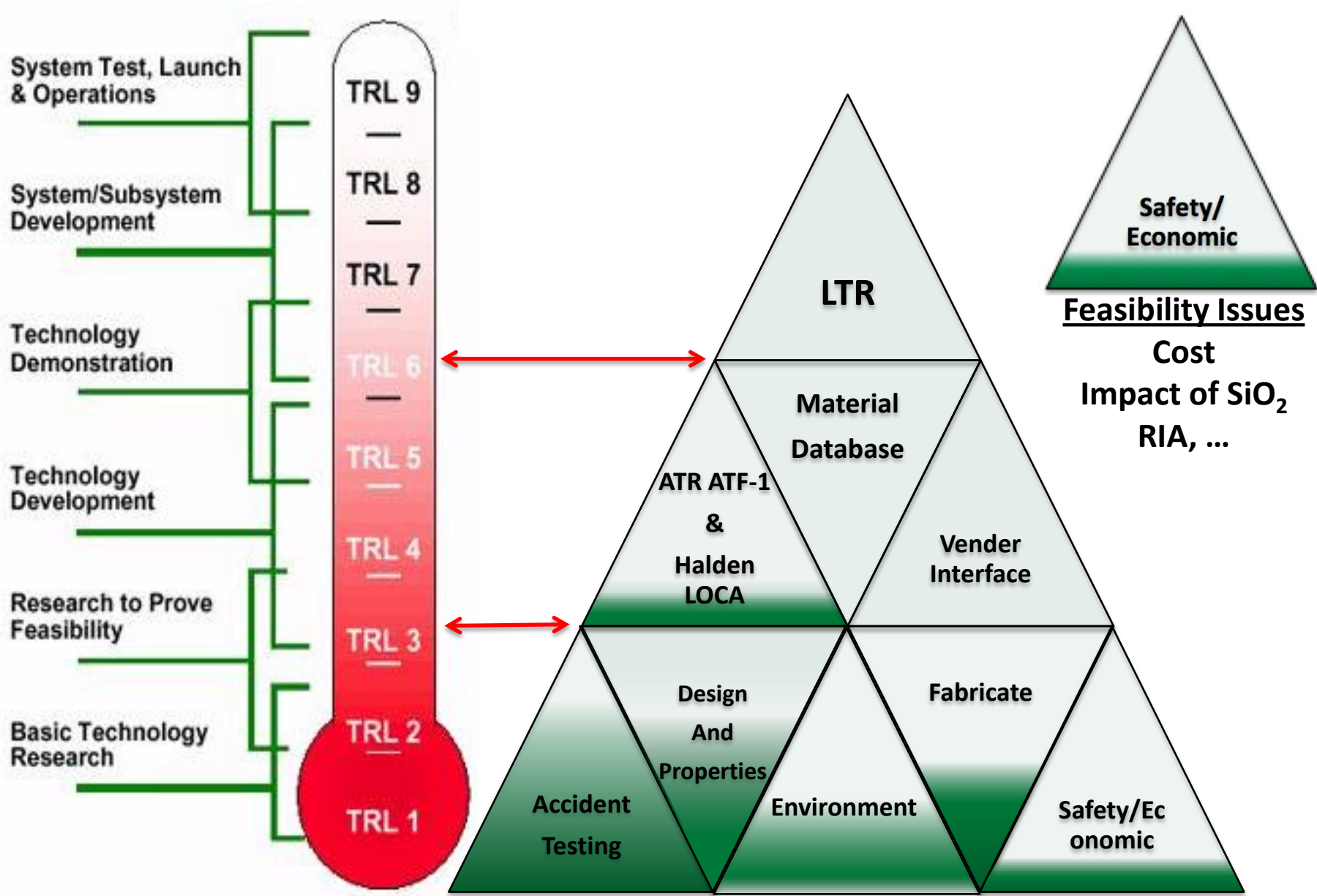
SiC/SiC : Path to ATR Irradiation and Commercial LTR of ATF

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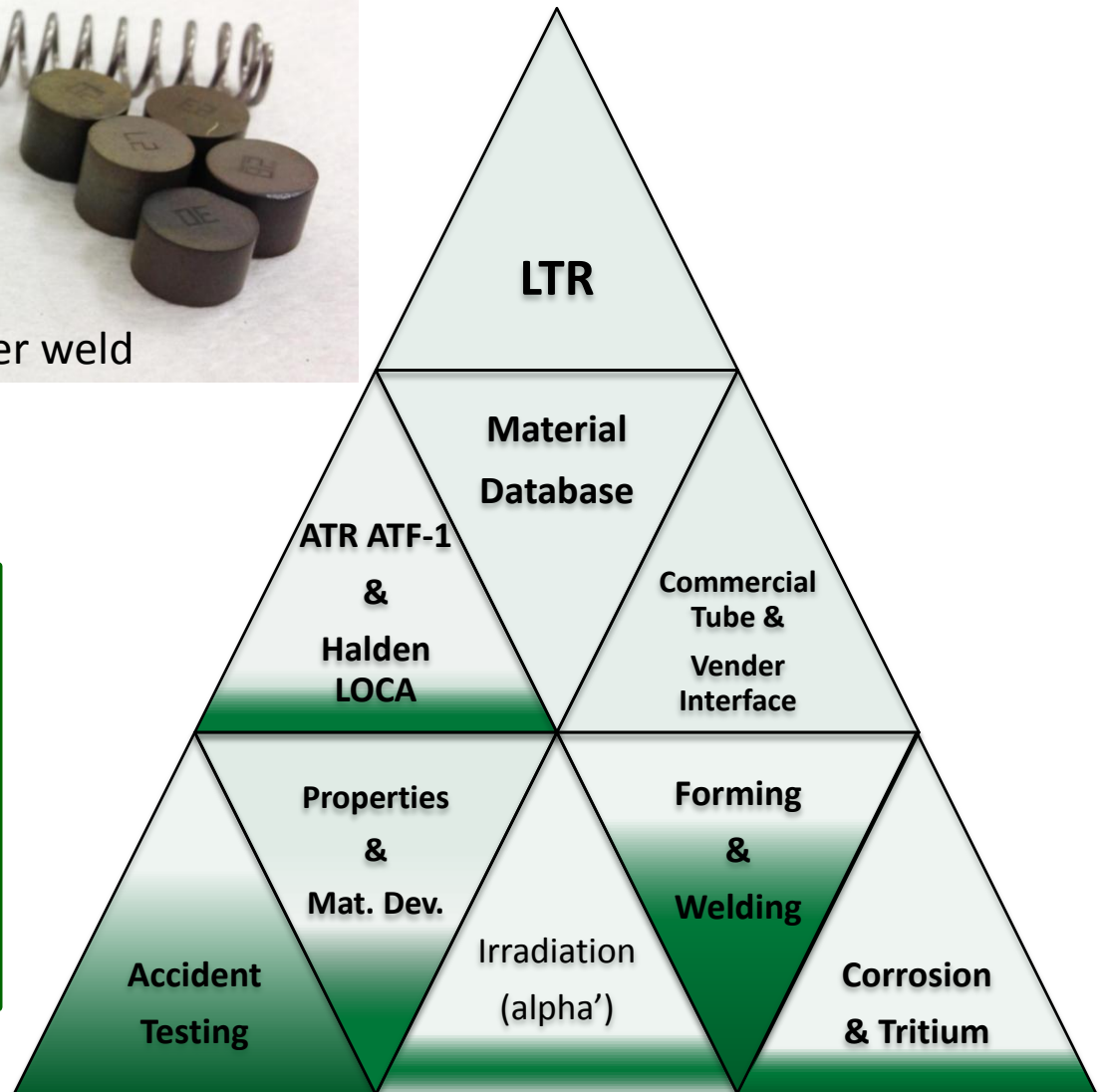
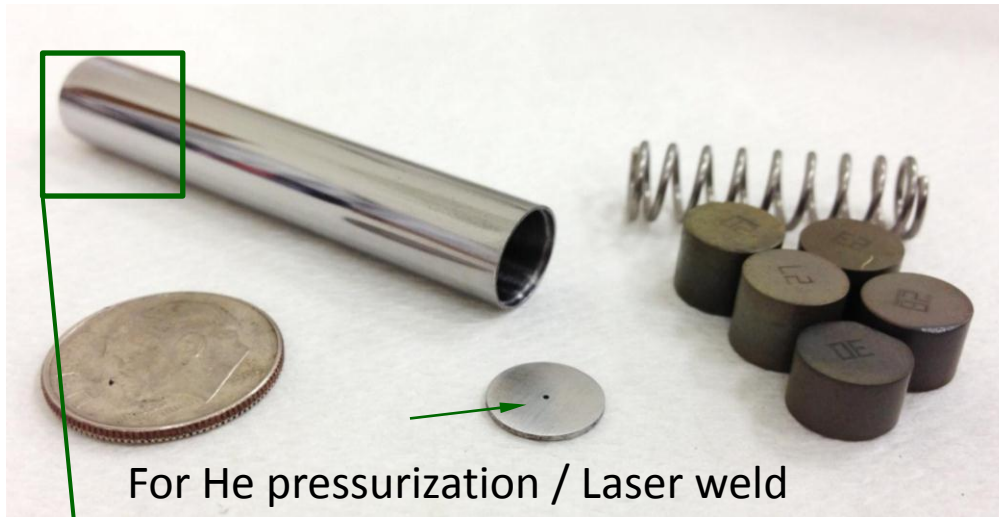
SiC/SiC : Path to ATR Irradiation and Commercial LTR of ATF

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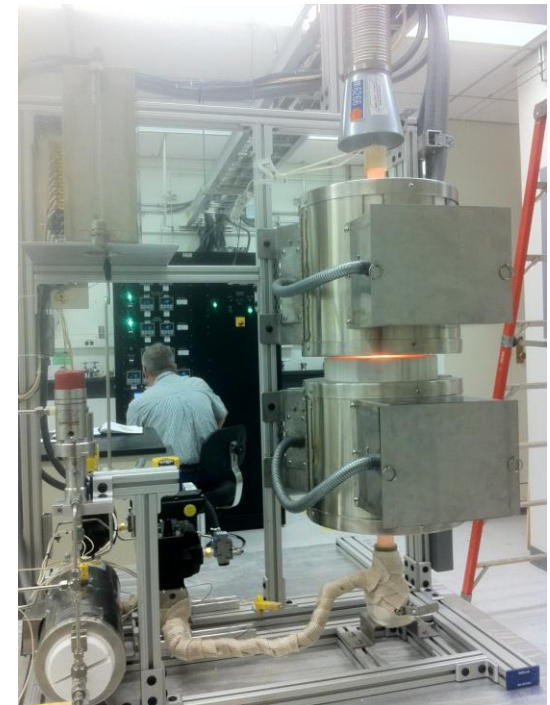
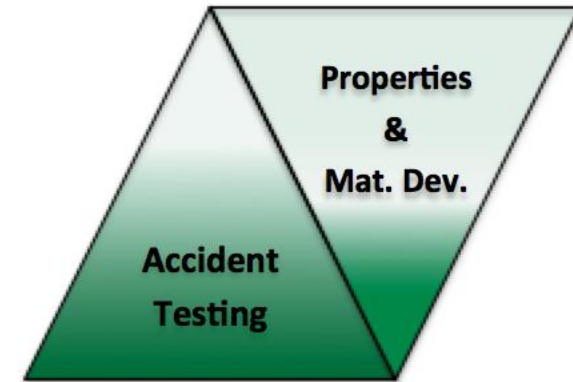
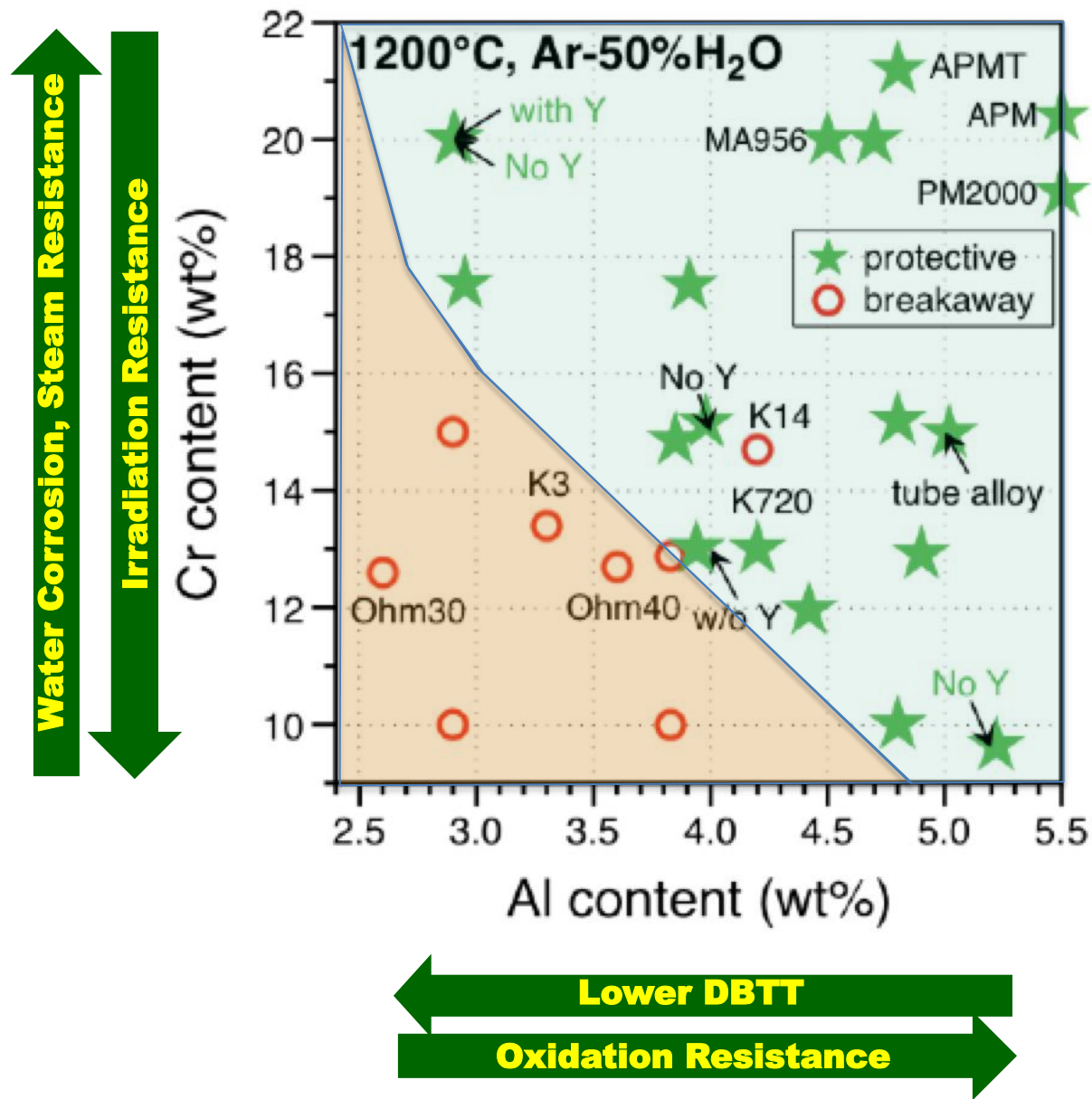
Advanced Steel: Path to ATR Irradiation and Commercial LTR of ATF

- Key performance issues are addressed on path to high-value irradiation and development.

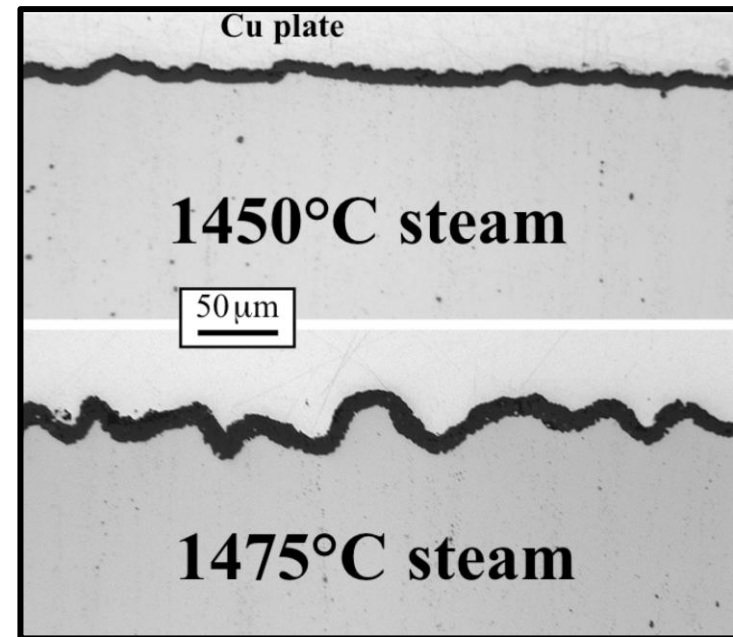
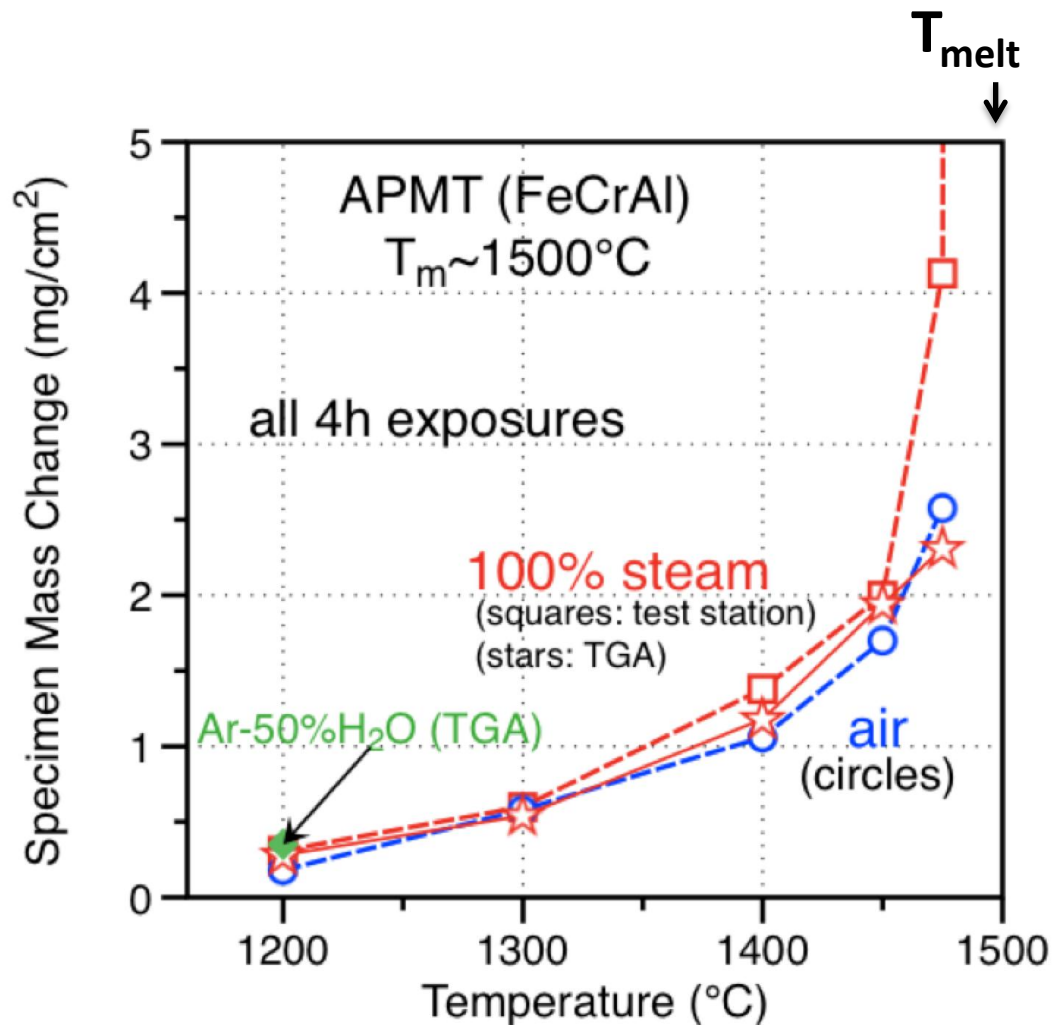


Advanced Alloys: Progress Since Last OECD Meeting

- Accident Testing, Properties & Material Development-



Phase 1 Alloys : Establish kinetic boundary to 1500° C steam oxidation



Phase I: Tensile Properties

- Cr and Al levels do not significant impact tensile properties.

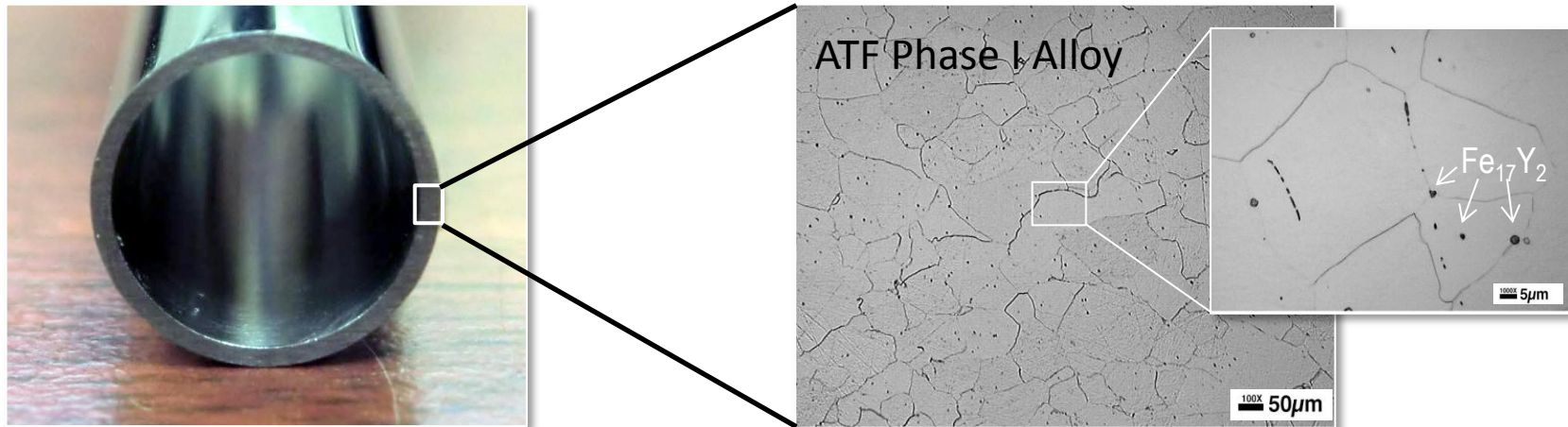


Table: Tensile properties of the FeCrAl alloys and min. requirements or Zircaloy 2 or 4 in ASTM B351

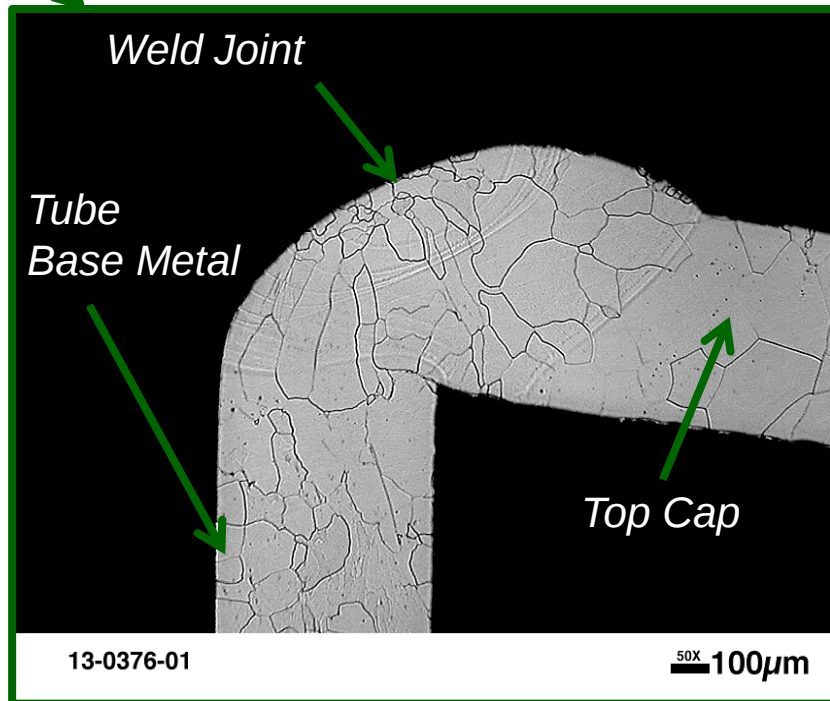
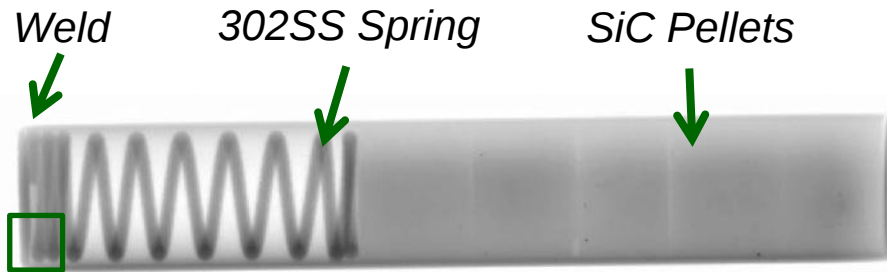
Alloy	Tested at	YS, MPa	UTS, MPa	Uniform EL, %
ORNL FeCrAl (Phase I) >30-50 micron grain	RT	389 ± 38	483 ± 33	15 ± 3
	300°C	262 ± 8	399 ± 29	12 ± 2
ORNL FeCrAl (Phase II) ~1 micron grain	RT	629	811	12
APMT (Fw-22Cr-5Al base)	RT	576	608	0.7
	300°C	379	680	20
Min. requirements for Zircaloy 2 or 4	RT	240	415	14
	316°C	105	215	24

Advanced Alloys: Progress Since Last OECD Meeting

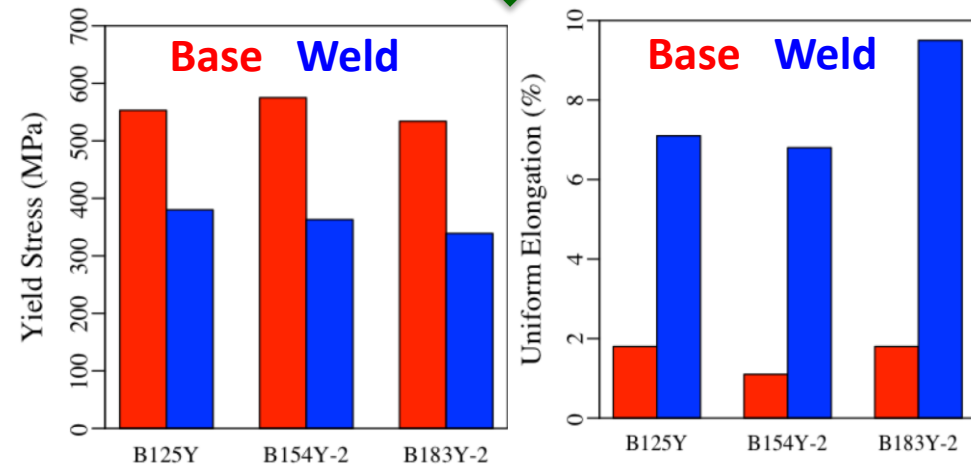
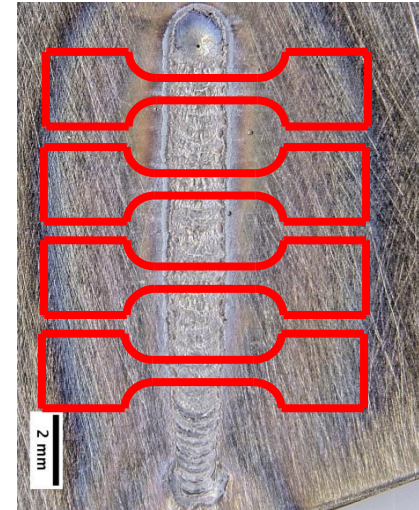
- Welding and Tube Fabrication -

Forming
and
Welding

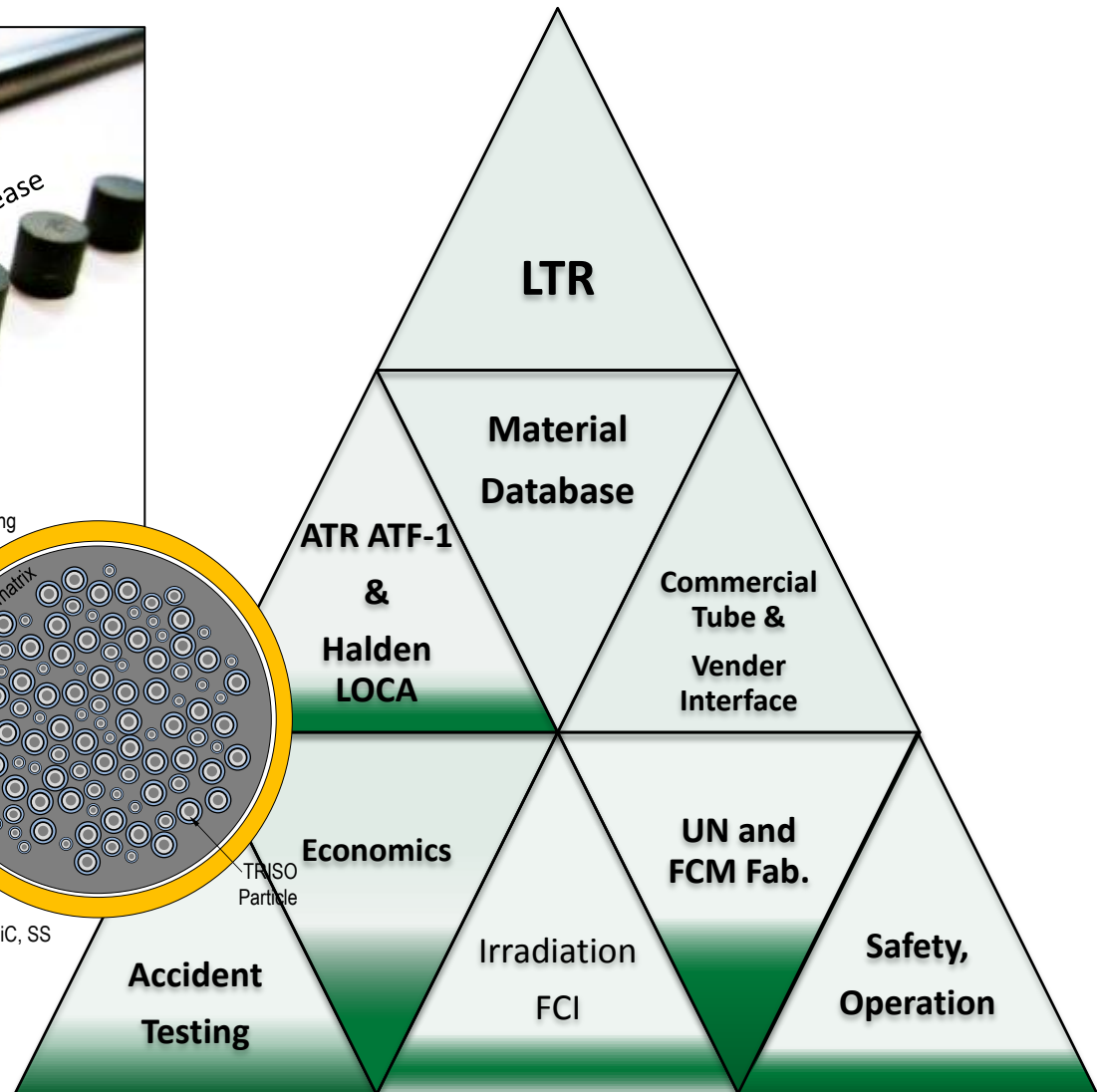
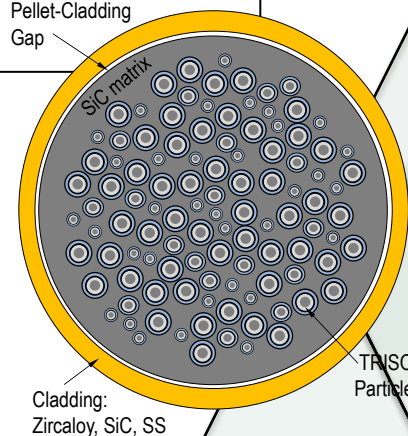
Laser Spot and E-beam Closure Weld



Laser Weld Studies

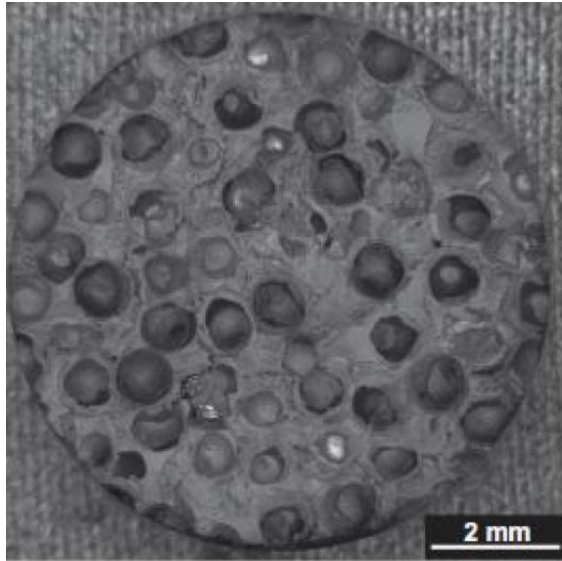


Fully Ceramic Microencapsulated (FCM) Fuel



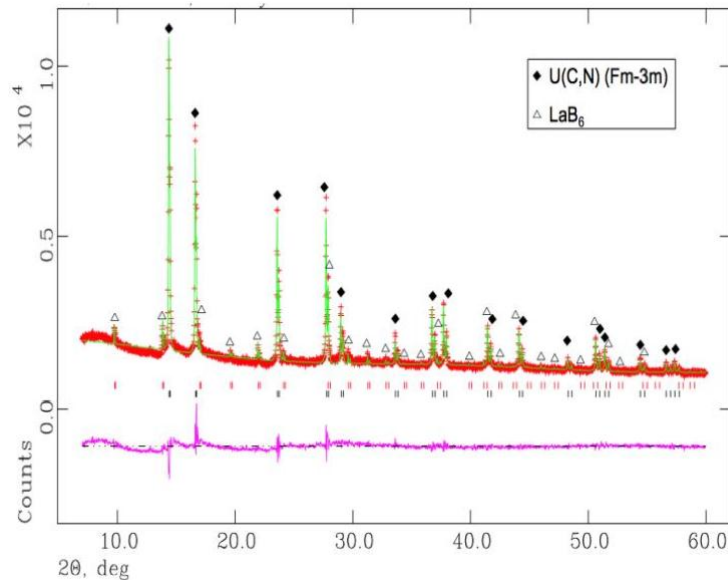
FCM : Progress Since Last OECD Meeting

- Fabrication -



- Routine fabrication of 40-45 v/o material with protective overcoating.

UN and
FCM Fab



- Kernel fabrication to 92 % density.
- Mechanisms and properties published in JNM articles.

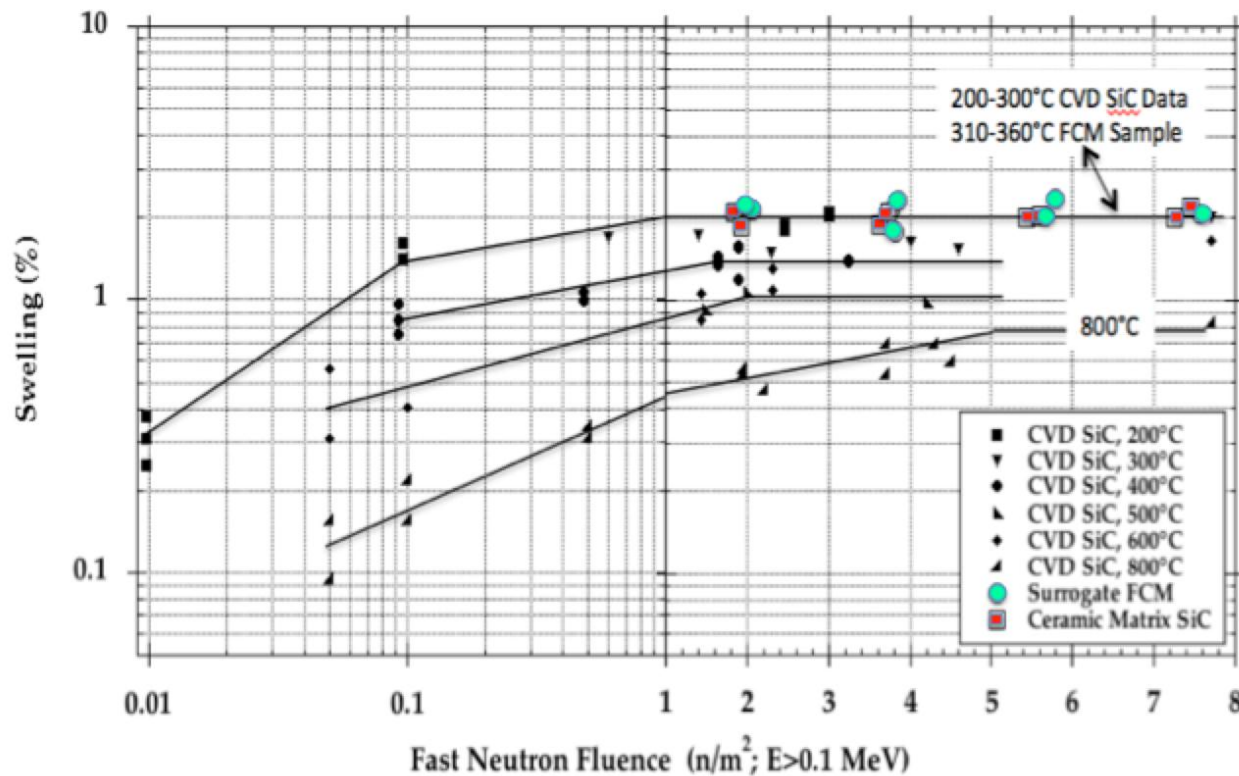
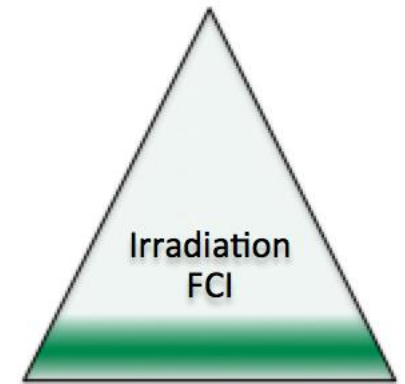
FCM : Progress Since Last OECD Meeting

- Fabrication -

- Irradiation completed to 2/3 lifetime to determine key feasibility:

1) Fuel Clad Interaction : No Interaction with Zircaloy

2) Swelling and Conductivity of Matrix Similar to CVD SiC.



FCM : Progress Since Last OECD Meeting

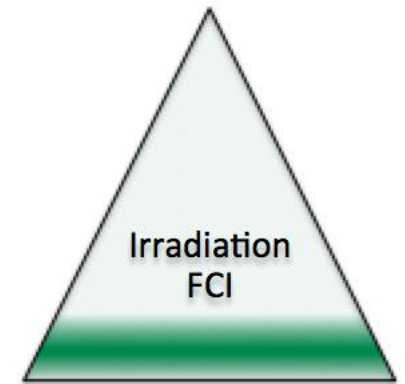
- Fabrication -

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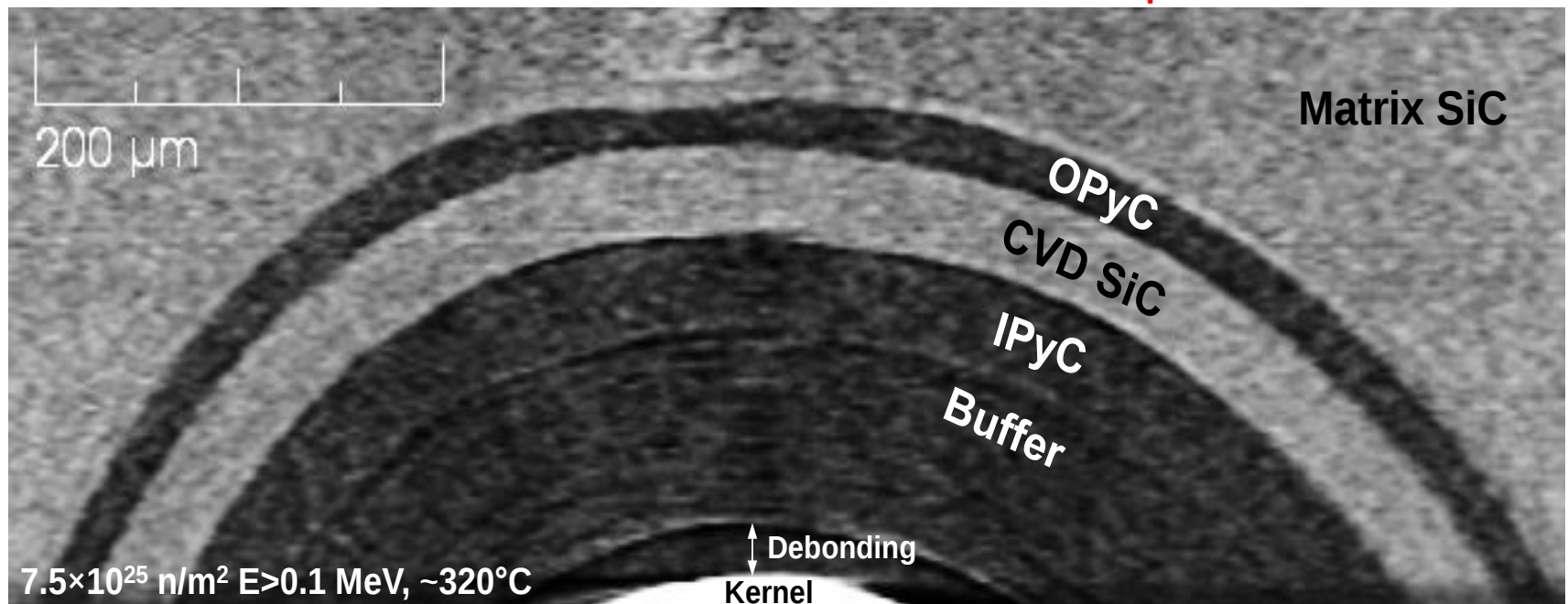
1) Fuel Clad Interaction : No Interaction with Zircaloy

2) Swelling and Conductivity of Matrix Similar to CVD SiC.

3) Integrity of TRISO constituents under LWR irradiation OK.

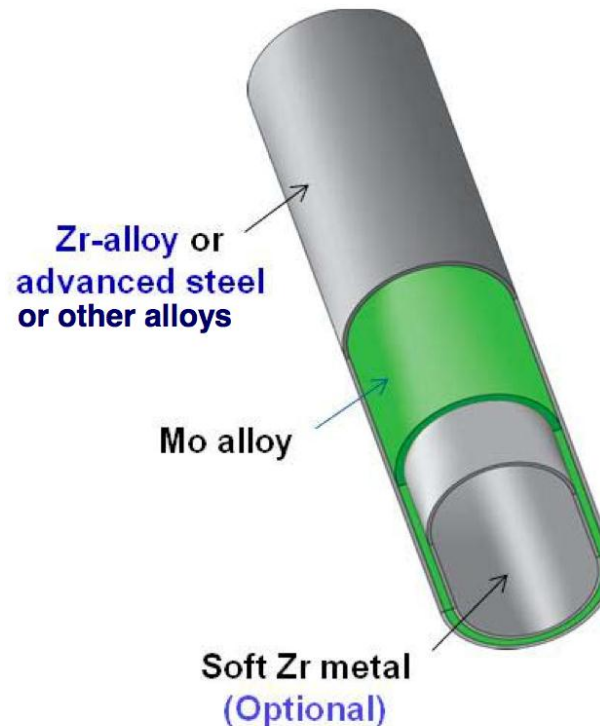


Published in Special ATF Edition of JNM

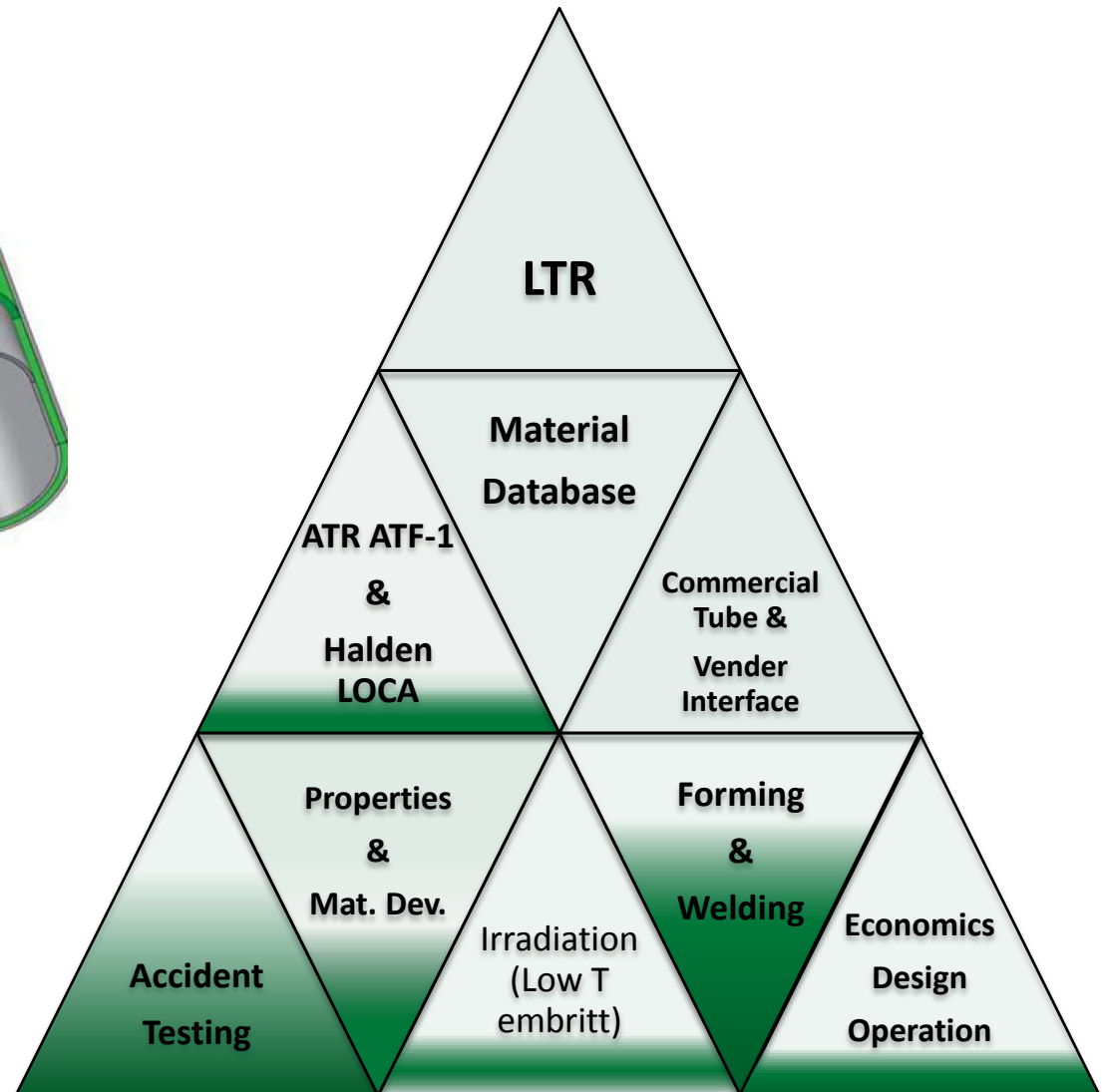


Moly: Path and Progress to ATR Irradiation and Commercial LTR of ATF

- Key performance issues are addressed on path to high-value irradiation and development.



- Thin tube forming drawing
- Steel coating
- Steam attack





Publication of ATF Papers in Special ATF Edition of Journal of Nuclear Materials

Paper No.	Journal	Lead Author information		Lead Author Organization	Status:	doi Link
		Last name	First name			
1	Journal of Nuclear Materials	Hunt	Rodney D.	ORNL	Accepted	http://dx.doi.org/10.1016
		Preparation of UC0.07–0.10N0.90–0.93 spheres for TRISO coated fuel particles				
2	Journal of Nuclear Materials	van Rooyen	Isabella	INL	under revision	Not Available
		Effects of a hot water corrosion flow test on a hybrid SiC-CMC-Zircaloy-4 fuel cladding tube.				
3	Journal of Nuclear Materials	Terrani	Kurt	ORNL	Accepted	http://dx.doi.org/10.1016
		Advanced Oxidation-Resistant Iron-Based Alloys for LWR Fuel Cladding				
4	Journal of Nuclear Materials	Yan	Yong	ORNL	Accepted	http://dx.doi.org/10.1016
		Post Quench Ductility Evaluation of Zircaloy-4 and Select Iron Alloys under Design Basis and Extended LOCA Conditions				
5	Journal of Nuclear Materials	Katoh	Yutai	ORNL	Accepted	http://dx.doi.org/10.1016
		Continuous SiC Fiber, CVI SiC Matrix Composites for Nuclear Applications: Properties and Irradiation Effects				
6	Journal of Nuclear Materials	Koyanagi	Takaaki	U of Kyoto	Accepted	http://dx.doi.org/10.1016
		Effects of neutron irradiation on mechanical properties of silicon carbide composites fabricated by nano-infiltrated transient eutectic process				
7	Journal of Nuclear Materials	Katoh	Yutai	ORNL	Accepted	http://dx.doi.org/10.1016
		Radiation-Tolerant Joining Technologies for Silicon Carbide Ceramics and Composites				
8	Journal of Nuclear Materials	Terrani	Kurt	ORNL	Accepted	http://dx.doi.org/10.1016
		The effect of fuel thermal conductivity on the behavior of LWR cores during loss-of-coolant accidents				
9	Journal of Nuclear Materials	Kondo	Sosuke	U of Kyoto	Accepted	http://dx.doi.org/10.1016
		Irradiation creep of 3C SiC and microstructural understanding of the underlying mechanisms				
10	Journal of Nuclear Materials	Yueh	Ken	EPRI	under revision	Not Available
		Silicon Carbide Composite for Light Water Reactor Fuel Applications				
11	Journal of Nuclear Materials	Besmann	Theodore	ORNL	w/ editor	Not Available
		Fission Product Release and Survivability of UN-Kernel LWR TRISO				
12	Journal of Nuclear Materials	Hunt	Rodney D.	ORNL	under revision	Not Available
		Carbothermic synthesis of 900 μm uranium nitride kernels: literature review, thermodynamics, analysis, and vacuum conversion of uranium dioxide and carbon				
13	Journal of Nuclear Materials	Ott	Larry J.	ORNL	Accepted	http://dx.doi.org/10.1016
		Preliminary Assessment of ATFs on LWR during Normal Operation and under DB and BDB Accident Conditions				
14	Journal of Nuclear Materials	Snead	Lance L.	ORNL	Accepted	http://dx.doi.org/10.1016
		Stability of SiC-Matrix Microencapsulated Fuel Constituents at Relevant LWR Conditions				
15	Journal of Nuclear Materials	Nelson	Andrew T.	LANL	under revision	Not Available
		High Temperature Oxidation of Molybdenum in Water Vapor Environments				
16	Journal of Nuclear Materials	Farmer	Mitch T.	ANL	under review	Not Available
		Scoping Assessments of ATF Impact on Late Stage Accident Progression Including Molten Core-Concrete Interaction				
17	Journal of Nuclear Materials	Zinkle	Steven J.	ORNL	under review	Not Available
		Accident Tolerant Fuels for LWRs: A Perspective				



Nuclear Energy

Journal Publications specifically with
FCRD acknowledgement

18	Journal of Nuclear Materials	Terrani	Kurt A.	ORNL	Published	http://dx.doi.org/10.1016/j.jnucmat.2013.03.006
		Protection of zirconium by alumina-and chromia-forming iron alloys under high-temperature steam exposure				
19	Journal of Nuclear Materials	Silva	Chinthaka M.	ORNL	Published	http://dx.doi.org/10.1016/j.jnucmat.2013.08.007
		Evaluation of sintering effects on SiC-incorporated UO2 kernels under Ar and Ar-4%H2 environments				
20	Journal of Nuclear Materials	Terrani	Kurt A.	ORNL	Published	http://dx.doi.org/10.1016/j.jnucmat.2013.02.042
		In situ ceramic layer growth on coated fuel particles dispersed in a zirconium metal matrix				
21	Journal of Nuclear Materials	Byun	Thak S.	ORNL	Accepted	http://dx.doi.org/10.1016/j.jnucmat.2013.10.007
		Process development for 9Cr nanostructured ferritic alloy (NFA) with high fracture toughness				
22	Journal of Nuclear Materials	Ben-Belgacem	Michel	ORNL	under review	Not Available
		Thermo-mechanical analysis of LWR SiC/SiC composite cladding				
23	Microscopy and Microanalysis	Parish	Chad M.	ORNL	under review	Not Available
		Combining EDS and EBSD to resolve ambiguous phase identifications in a Zr-stainless steel diffusion couple				
24	Journal of Nuclear Materials	Katoh	Yutai	ORNL	Accepted	http://dx.doi.org/10.1016/j.jnucmat.2013.05.037
		Properties of Zirconium Carbide for Nuclear Fuel Applications				
25	Journal of Nuclear Materials	Pint	Bruce A.	ORNL	Published	http://dx.doi.org/10.1016/j.jnucmat.2013.05.047
		High Temperature Oxidation of Fuel Cladding Candidate Materials in Steam-Hydrogen Environments				