Center for Advanced Nuclear Energy Systems(CANES)

Fatigue and Fracture of Students and Other Materials: Nuclear Engineering- An Exciting Time

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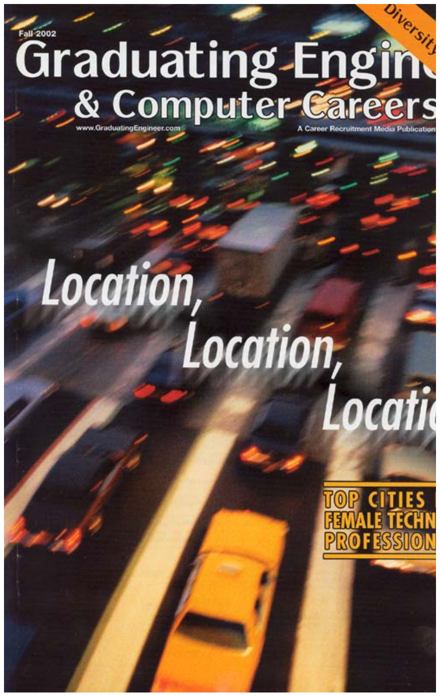
MIT Nuclear Engineering Department

US GROWTH RATES DURING 1990 - 1999

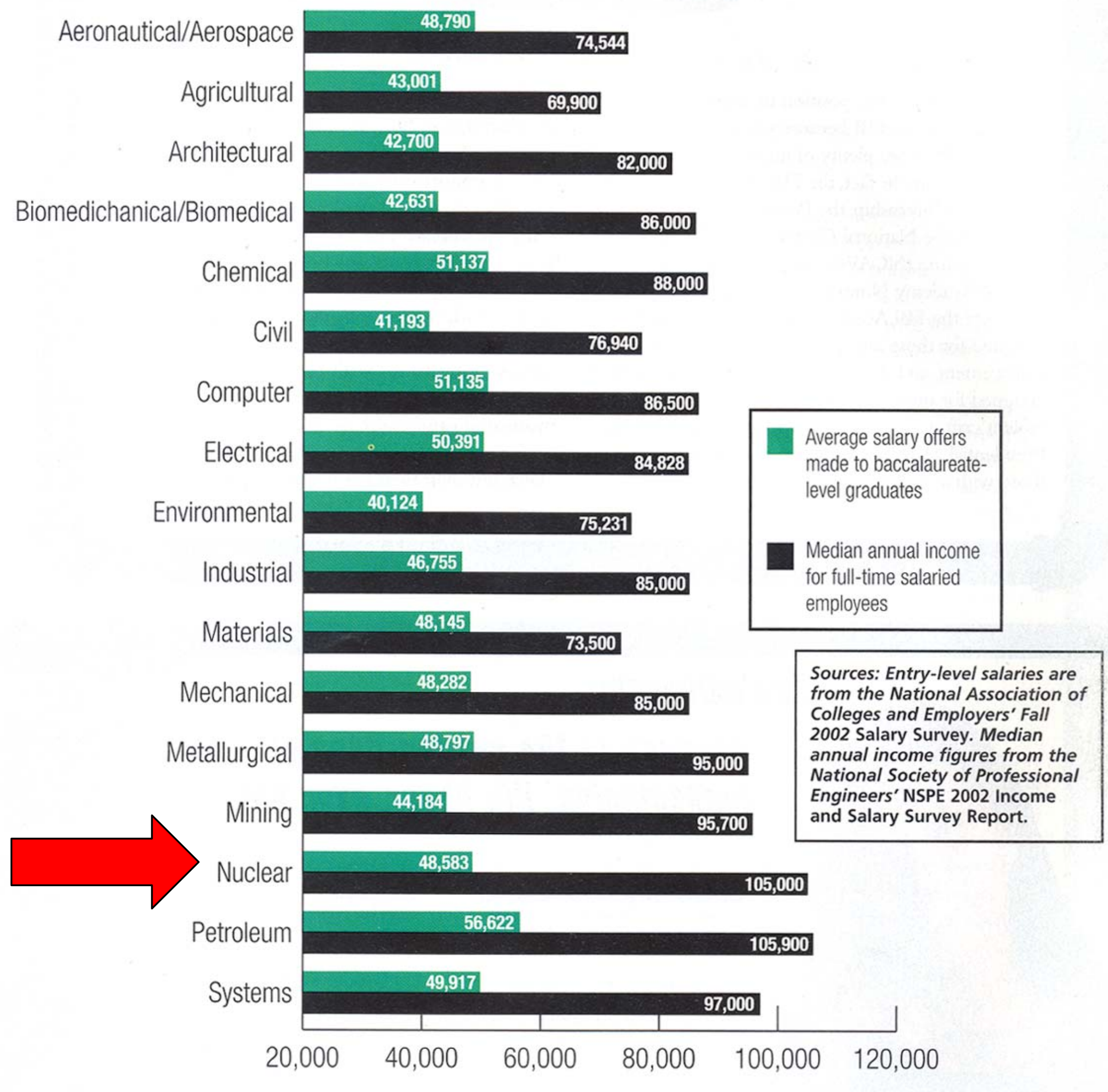
GDP	3.1%
Population	1.0%
Energy Consumption	1.5%
Electricity Consumption	2.2%
Greenhouse Gas Emissions	1.1%
Nuclear Electricity	2.5%

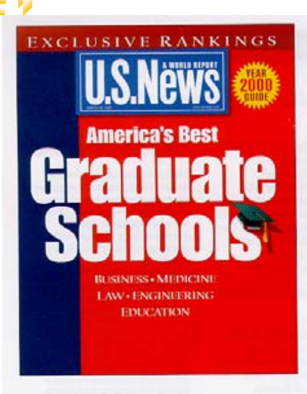
Nuclear Electric Energy

- Nuclear Energy is Now the Cheapest Form of Electricity Generation for Large Power Blocks in the US



Engineering Salaries: Entry-Level vs. Experienced





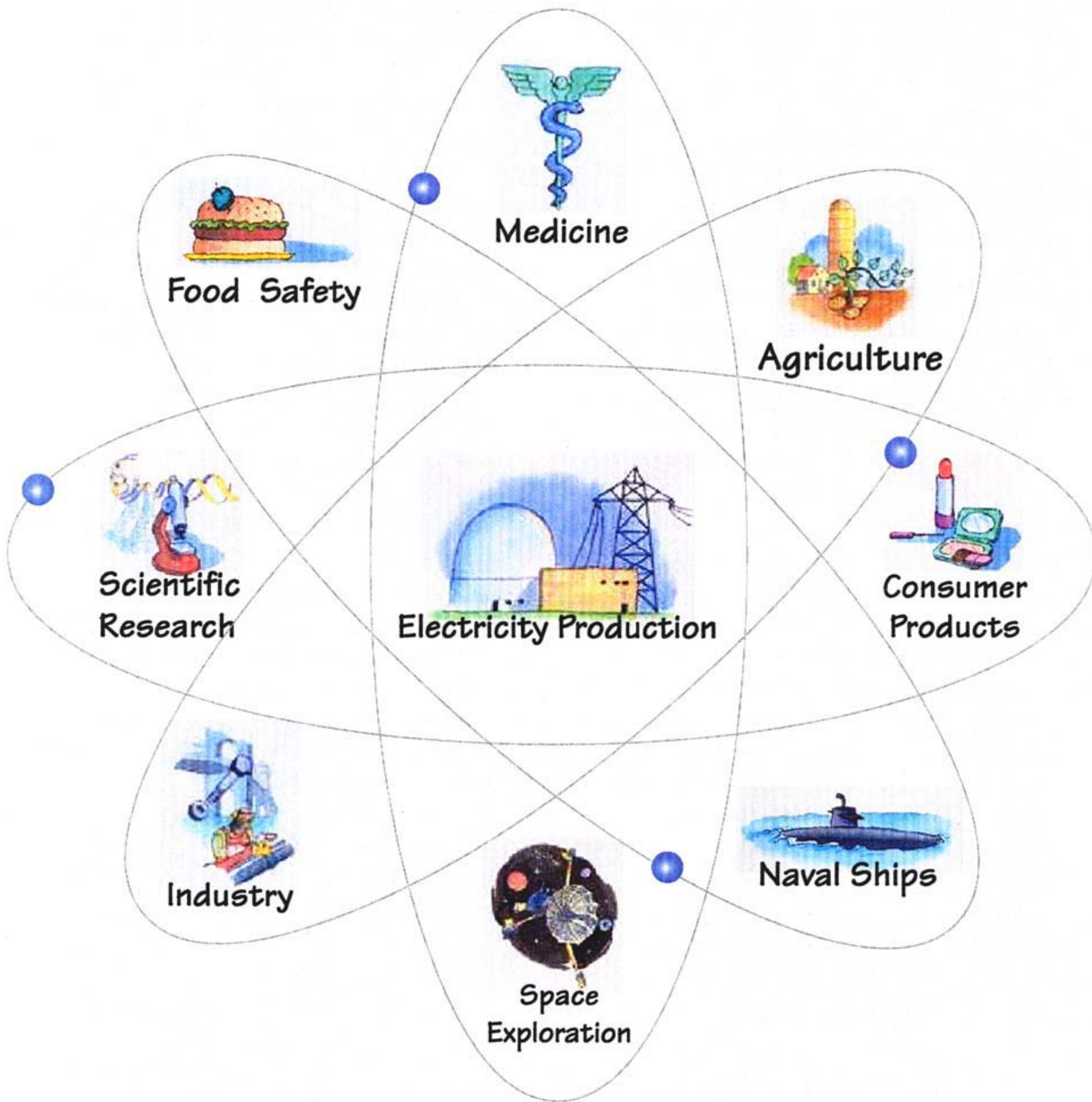
The U.S. News and World Report ranked the Department of Nuclear Engineering at MIT #1 of all Graduate Schools in the country

Also 2002,03,04 !

Income by Engineering Discipline
Median Annual Income (1998 dollars)

Source: National Society of Professional Engineers 1998

CANES



Diversity, Diversity, Diversity

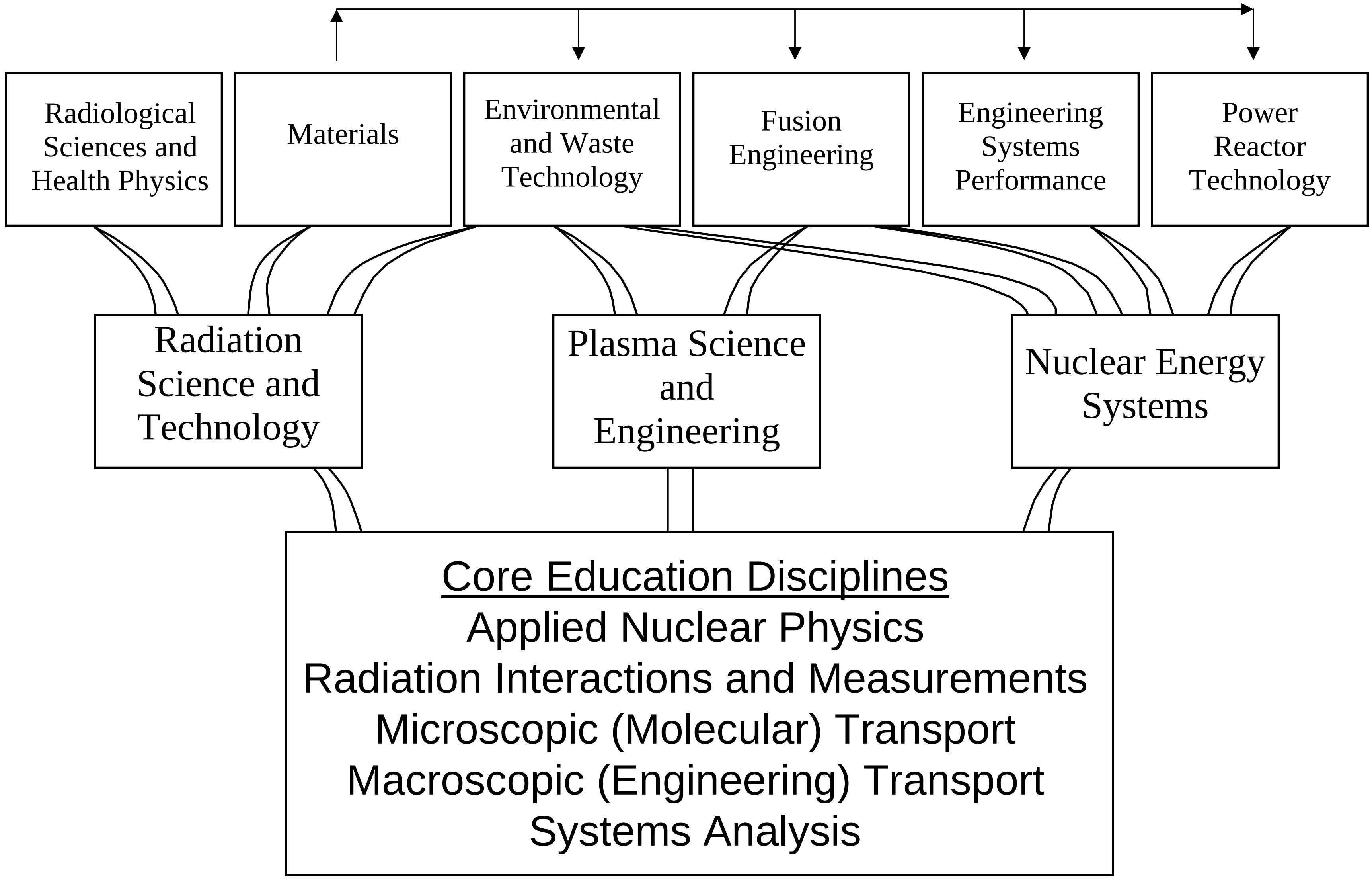
DOE Nuclear Energy Programs

- Generation IV System Development
- Nuclear Energy 2020
- Advanced LWR Development Support
- NGNP
- Space Reactor Program

Nuclear Engineering At MIT

- Small Department focused on personal student attention and Success
- 17 Faculty
- 35 Senior Research Associates
- Plasma Fusion Science Center (PFSC)-Alcator C-Mod
- MIT Research Reactor
- Numerous Accelerators
- 27 Undergraduates - 130 Graduate Students

Department Organization

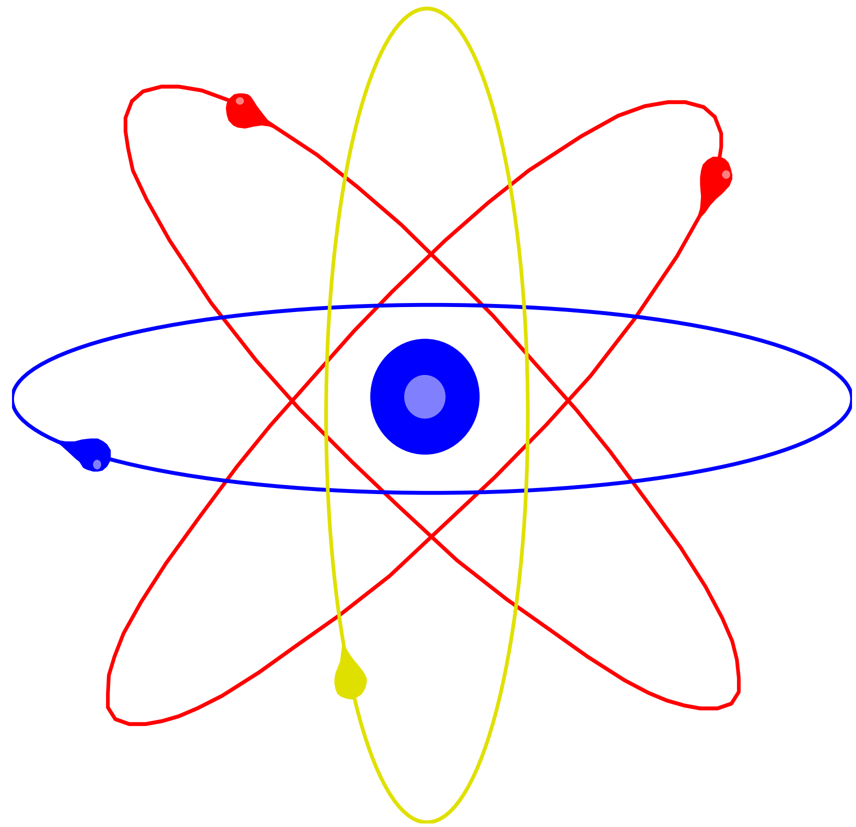


Courses of Study

- Nuclear Energy Systems (Fission)
- Plasma Science and Fusion
- Radiation, Science and Technology
- Systems Design, Policy, and Management

Programs in Nuclear Energy Systems

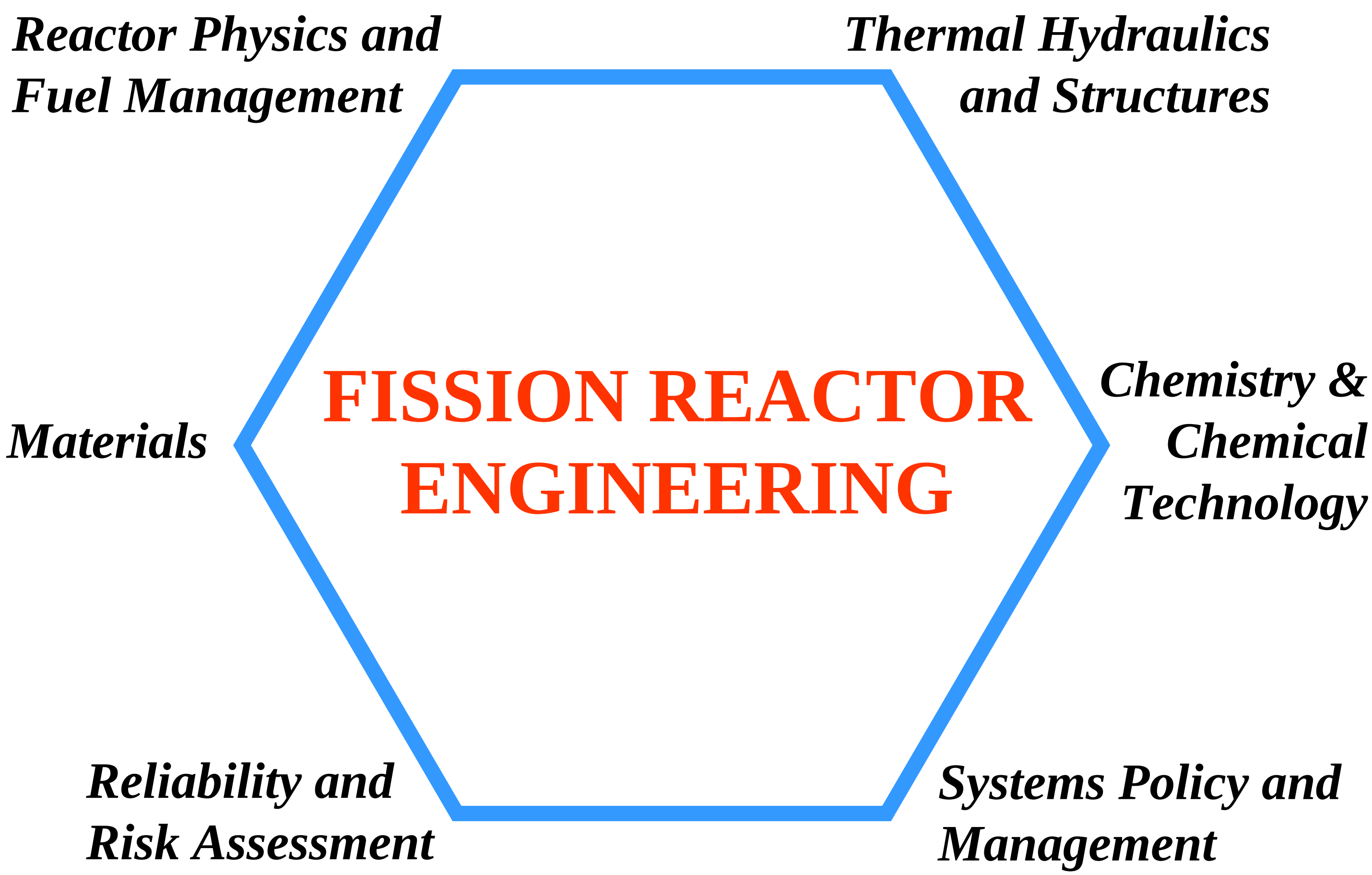
- Advanced Reactors
- Reactor Physics
- Thermal Hydraulics
- Safety Analysis
- Fuel Cycle Issues
- Probabilistic Safety Assessment
- Waste Disposal
- MIT Reactor



Challenges for Next Generation Reactor Systems

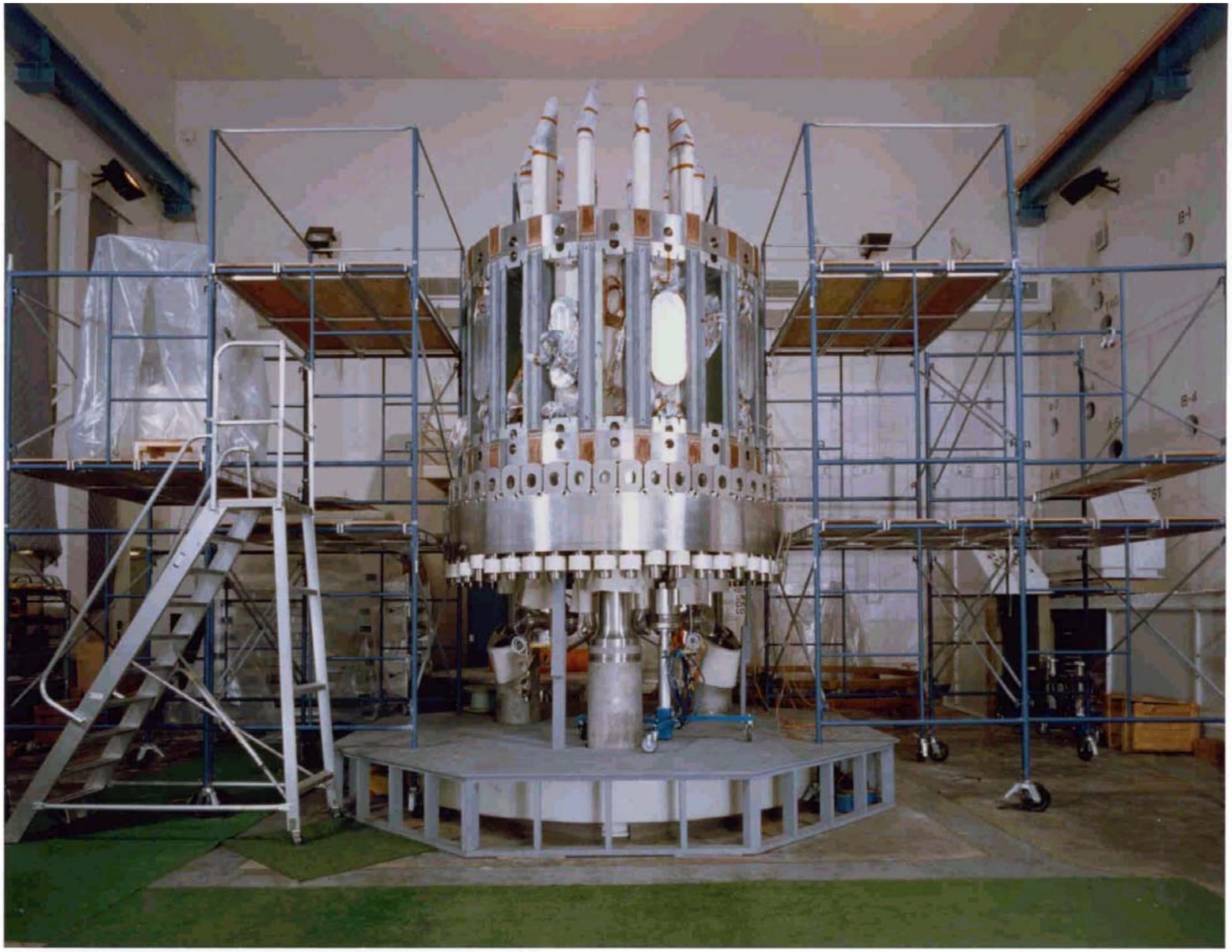
1) Make safety transparent
2) Reduce construction and operating costs
3) Reduce waste burden
4) Enhance proliferation resistance

The Cornerstones of Fission Reactor Engineering

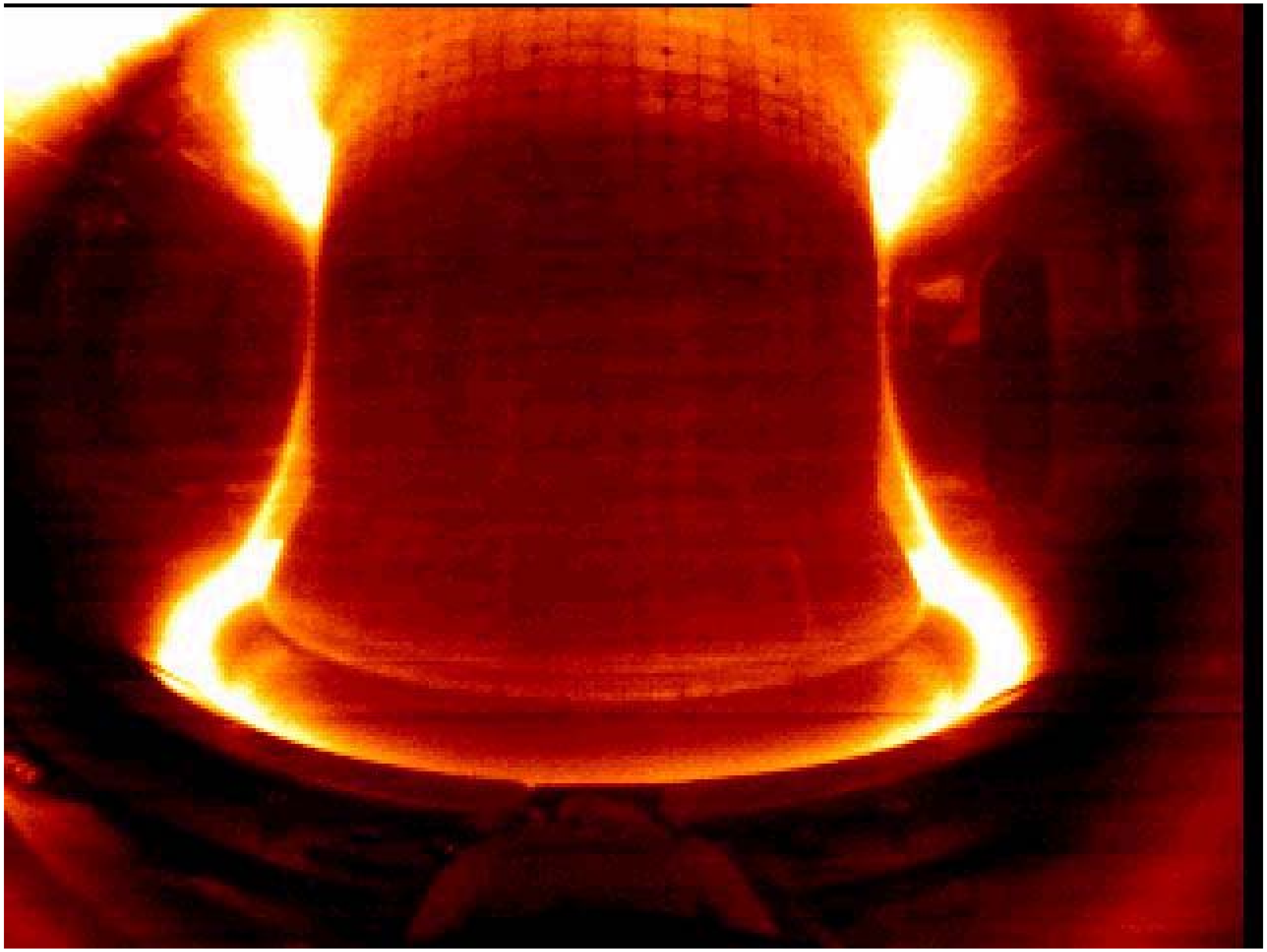


Plasma Science & Fusion

- Plasma Physics
- Basic Science
- Tokomak Research
- Design of Advanced Systems
- Fusion "burner"
- Power Conversion
- Alcator Experiment
- ITER Engineering R&D



MIT’s Tokomak – Alcator C-Mod
Not shown all the instrumentation



Plasma at MIT Alcator C-Mod

Radiation, Science and Technology

- Medical Applications
- Boron Neutron Capture Therapy
- Quantum Computing
- Atomistic Simulations
- Nano/Meso-scale Materials Modeling
- Neutron Imaging
- Accelerator Applications
- Nuclear Chemistry
- Environmental Analysis
- Waste Isolation Studies
- Industrial Applications
- Measurement Devices
- Computational Fluid Dynamics

Materials Materials Materials Materials
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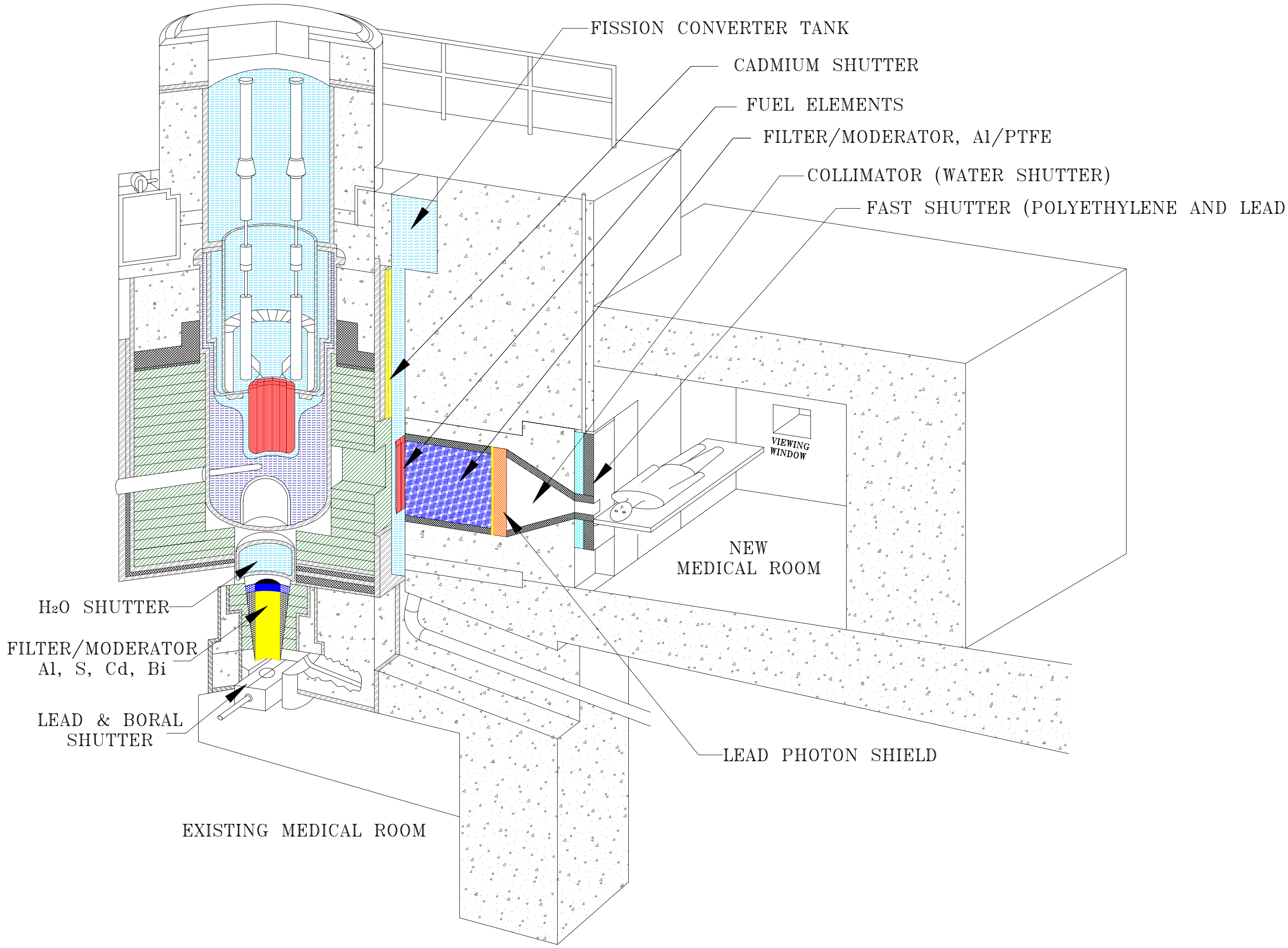
Current Research Areas in Materials

- Environmental Degradation of Materials.
- Advanced Materials for Fusion Applications.
- Materials Development for Advanced Reactor Systems.
- Chemical Reactors for Supercritical Water Oxidation of Military Toxic Wastes.
- Corrosion of Advanced Materials
- Advanced Nuclear Fuels
- Materials Development for Hydrogen Production

Current Programs in Materials & Advanced Fission System Research

- Fuel Performance of Kernel Fuel for Gas Reactors
- Turbo-Machinery for Helium Systems
- Materials Performance & Alloy Development for Pb-Bi Cooled Systems
- Cladding Corrosion in LWR Systems
- Effect of Long Term Aging on Embrittlement & Crack Growth in Stainless Steel Welds for LWR Applications
- High Temperature Materials Irradiations
- Advanced Fuels for LWR Systems
- Radiolysis in Supercritical CO_2 & H_2O
- Corrosion of Materials in Supercritical CO_2 & H_2O
- Materials Development for Hydrogen Production

The MIT Research Reactor



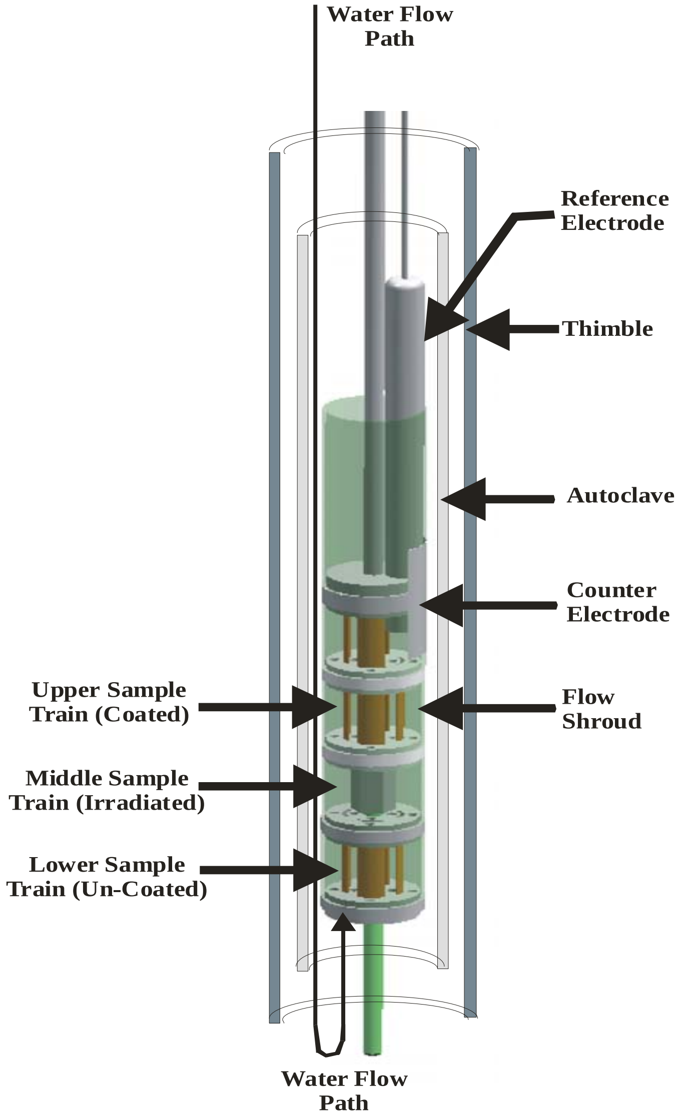
MITR-II showing current and new epithermal beam locations

In-Core Irradiation Capability at MITR

- LWR Environment Irradiation Facility
  - High Pressure Water Loops
  - Complete Chemistry Control
- High Temperature Irradiation Facility (Under Construction)
  - 1400°C Maximum Temperature (10 cm region)
  - Inert Environment
  - Capable of Fuel Irradiations
- LWR Fuel Irradiation Facility
  - Segmented Fuel within Double V   Containment
  - Flexible Fuel Temperature
- Very High Pressure Irradiation Facility (Under Design)
  - 25 MPa Maximum Pressure
  - 750°C Maximum Temperature
  - Supercritical Water
  - Supercritical CO_2

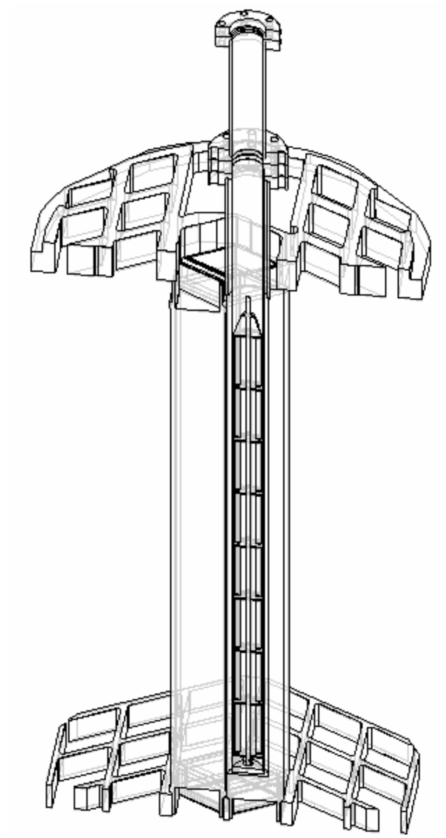
Simulated LWR Environments

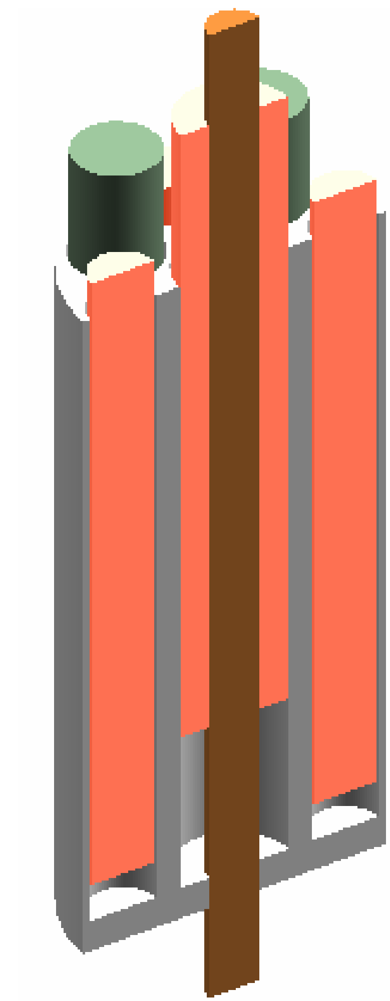
- MITR has extensive experience with materials & corrosion studies @ LWR operating temperatures & pressures.
- Radiolysis models have been developed for LWR environment analysis and experimental design.
- Can be designed for very high pressure



High Temperature Irradiation Facility

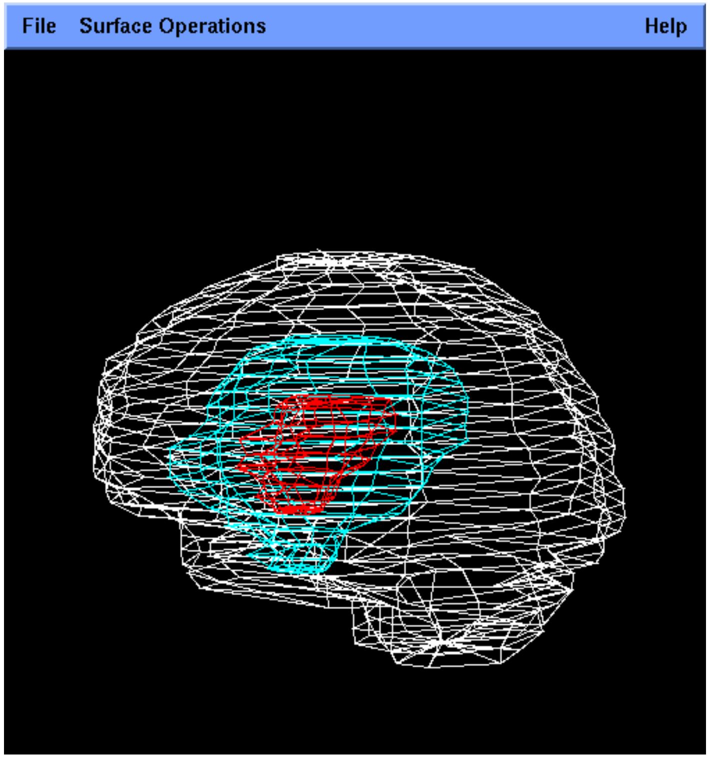
- Maximum Temperature: 1400°C
- Inert Environment
- Tailored Experiments to achieve temperatures from 500-1400°C
- Active Control-Can Run Temperature vs. Time Transients.
- Above facility status can be used to test NGNP & Gen IV materials
  - SiC, Graphite, Composites
- Coated Particle Fuel Irradiations

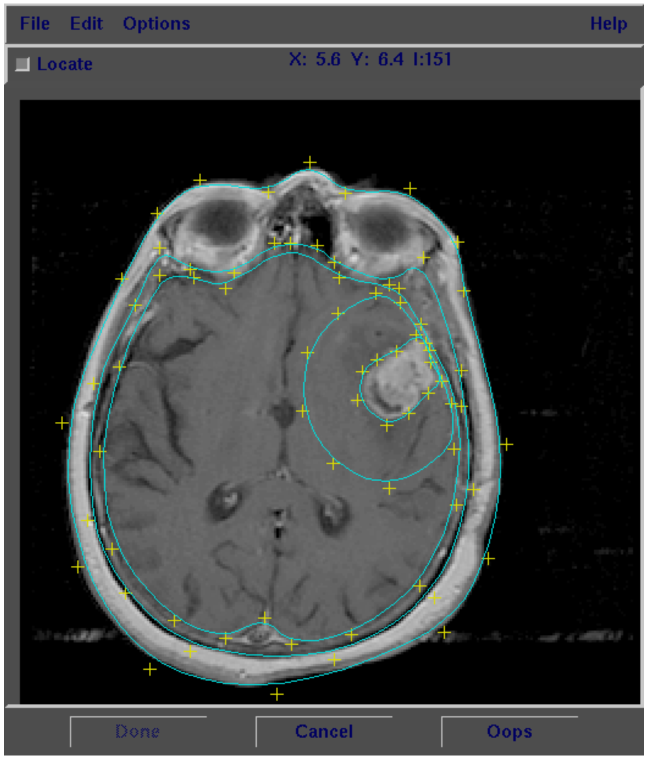




Facility Status: Under Construction

Developing New Cancer Therapies



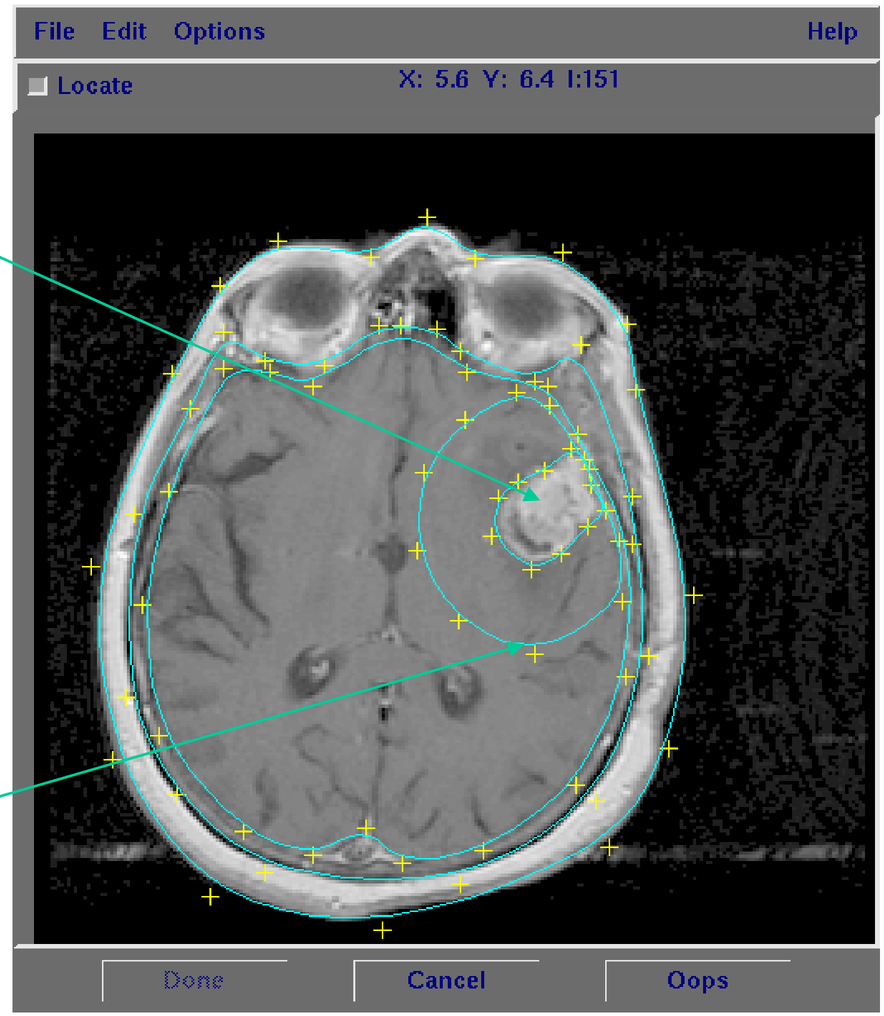


Targeting the Tumor

Tumor

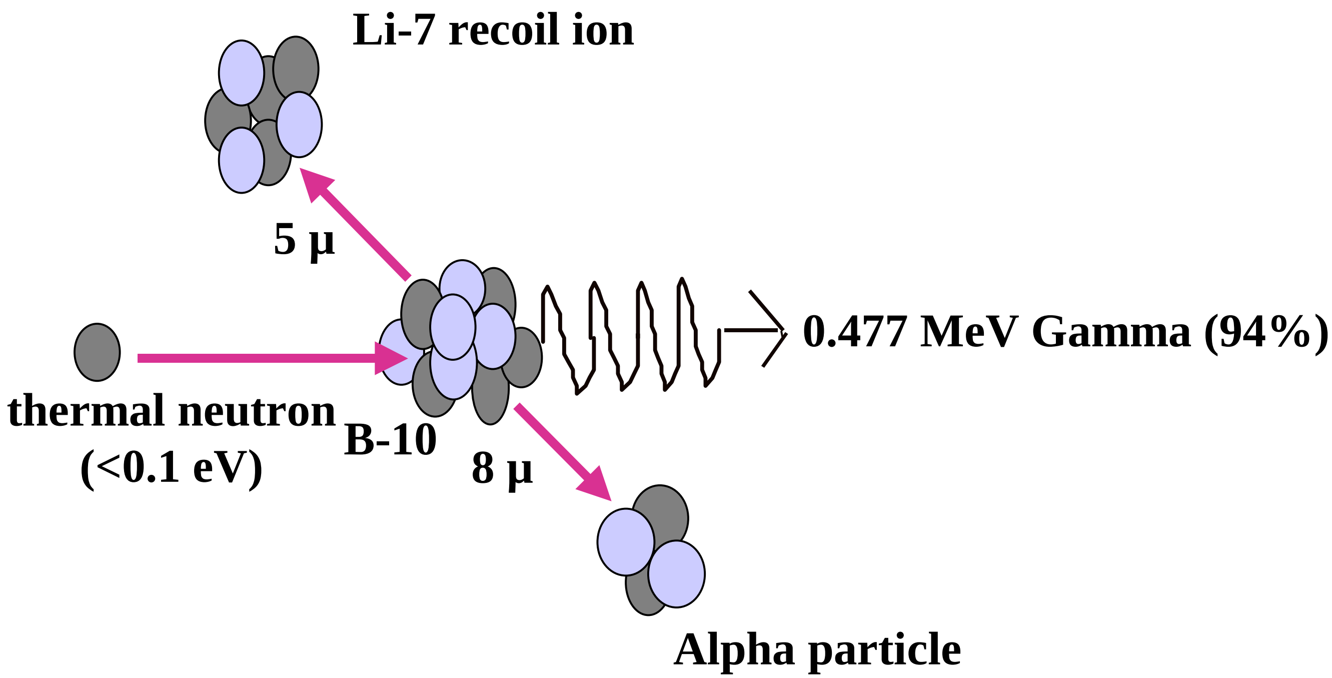
Brain tumors:
A devastating disease

Target volume
(tumor + 2 cm)



The BNCT Reaction

2.33 MeV of kinetic energy is released per neutron capture: initial LET 200-300 keV/µm; RBE v. high; OER v. low

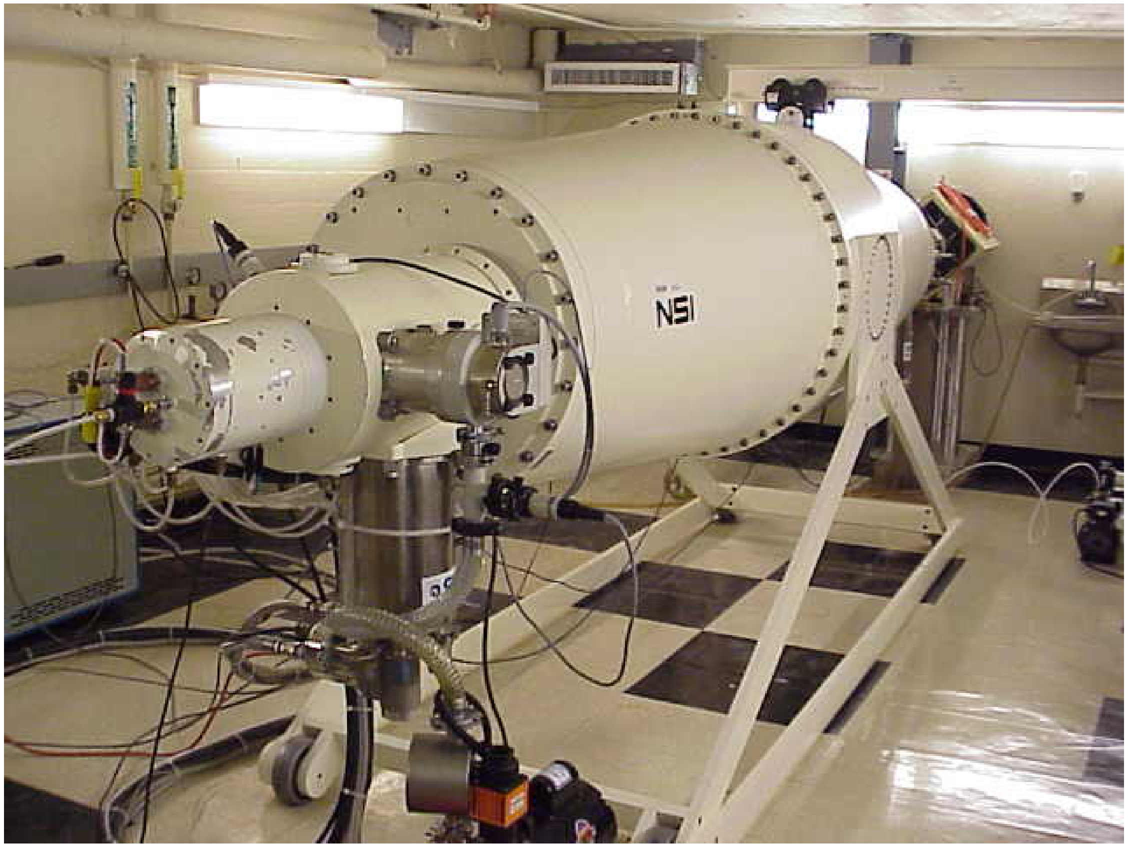


Thermal cross-section = 3837 b (that's *very* big…)

Developing New Cancer Therapies

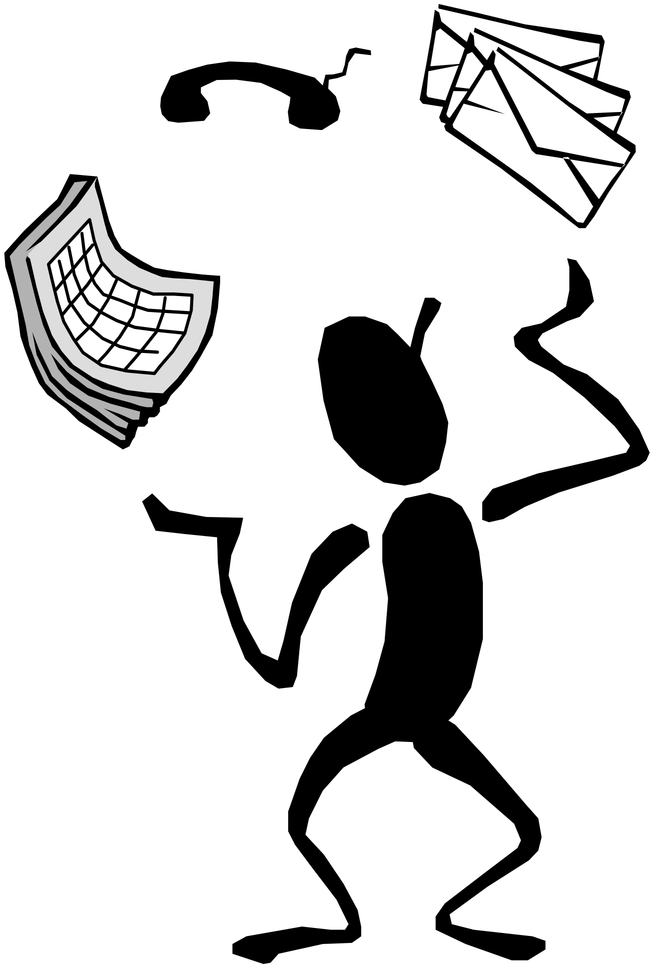
Effects of Radiation on Biological Systems

Particle accelerator in the MIT Laboratory for Accelerator Beam Applications



Systems Design, Policy, and Management

- Operations Research
- Nuclear Policy
- Work Flow Simulations
- Management Strategies
- Non-Proliferation
- International Nuclear Policies



Career Opportunities

Present Shortage of Qualified Scientists and Engineers

- Consulting/ Research
- Radiological Physics
- Government - DOE/NRC
- National Laboratories
- Nuclear Utilities (104)
- Nuclear Vendors
- Fusion Research
- Medical Applications
- National Defense

Nuclear Industry Career Choices

- Nuclear Plant Design - Overall plant
- Operations - Maintenance, inspection
- Advanced Fuels Design- materials
- Safety Analysis - thermal hydraulics
- Core Monitoring Analysis
- Core Reload Analysis - Reactor Physics
- Radiation and Health Physics
- Environmental Monitoring and Analysis
- Materials, Chemistry

National Laboratory Choices

- Basic and applied Research
- Waste Disposal technologies
- Research Reactor Operations and Experiments
- Fundamental science applications
- Nuclear Computer code development
- Decommissioning and site cleanups
- Space Applications

Companies That Have Hired MIT Nuclear Science and Engineering Majors

- Westinghouse
- Silicon Graphics
- Boeing
- Nuclear Utilities
- Texas Instruments
- Institute of Nuclear Power Operations
- Knolls Atomic Power Lab
- Intel
- ANL, LANL, ORNL, INEEL, SNL, LLL
- Goldman Sachs
- Mass General Hospital
- Morgan Stanley
- McKinsey
- Procter and Gamble
- Studsvik
- TRW
- GE, GNF, ABB, Framatome
- Enanta Pharmaceuticals
- Options Trading -Archelon

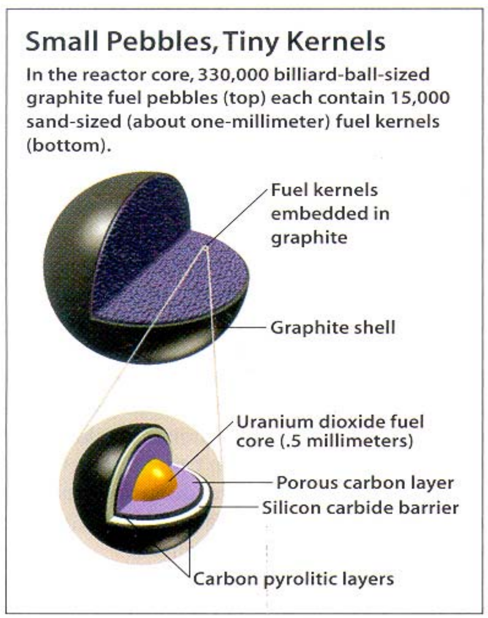
Why Nuclear Engineering?

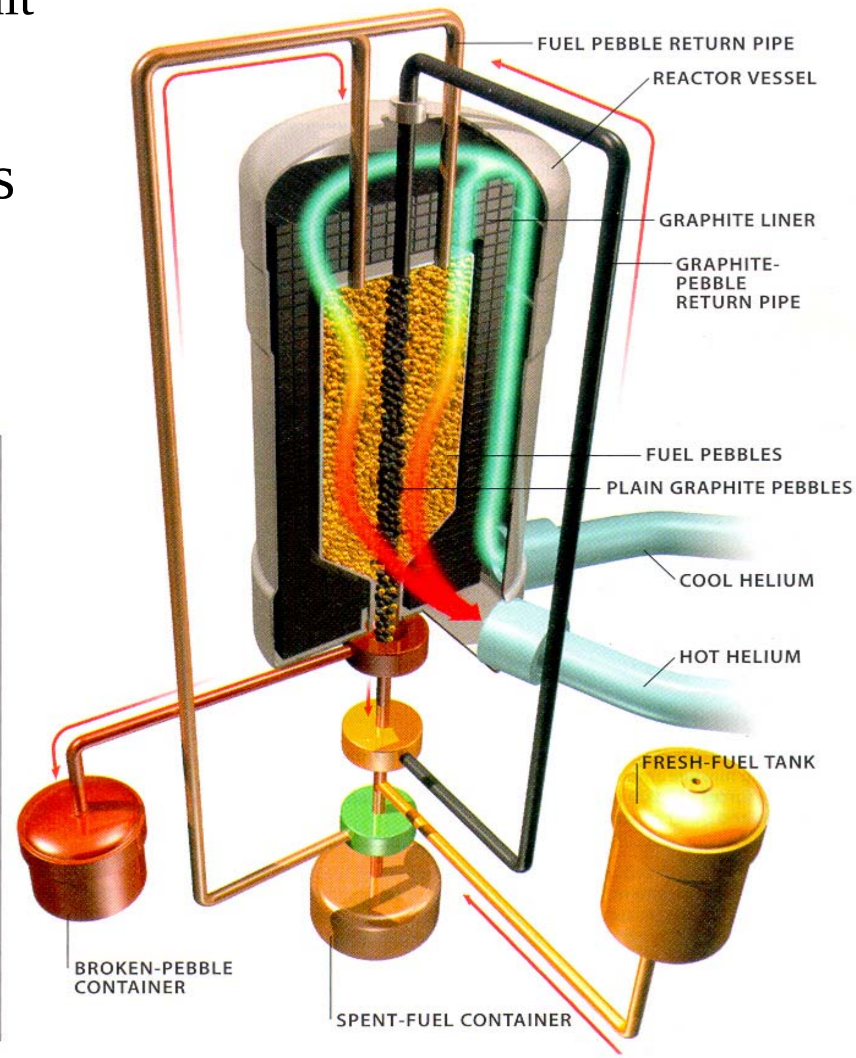
- One of the Highest Paying Engineering Careers.
- Nuclear Science and Engineering are **Hi-Tech** and emerged just 50 years ago.
- The MIT Nuclear Engineering Department offers a Multi-Disciplinary course of study in Nuclear & Radiation Technology applications in Energy, Medicine, & Industry.
- Combines all basic disciplines with a Nuclear Spin.
- A Nuclear Degree allows a great deal of flexibility in your Career Choices because it involves Knowledge of Mechanics, Electronics, Materials & Computer-System Dynamics,
- The Department has a very small student to faculty ratio.
- The Department also offers Opportunities for Research to all its Students, Undergraduates as well as Graduates.

Silver Migration & Release from SiC

Pebble Bed Reactor

- Modular design ~150 MWe per unit
- Helium gas coolant
- Multi-layer coated fuel particles
- Increased efficiency
- Passive safety characteristics





Question

- Does the observed release behavior in the field invalidate the assumption of release by diffusion on its face?

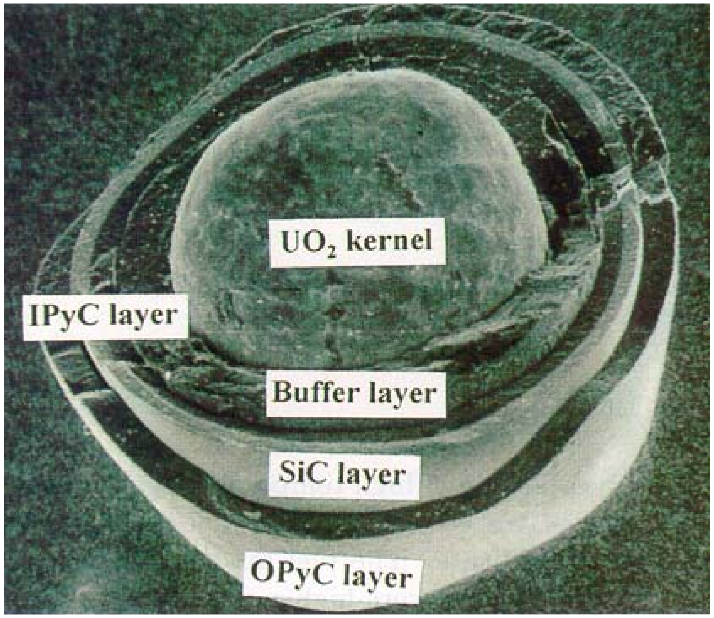
Overview

- Experimental Question
  - By what mechanism does silver migrate through SiC?
    - Diffusion
    - ?
  - Can we observe silver diffusion?
- Methods
  - Silver ion implantation
  - Characterization using Analytical Electron Microscopy (SEM, TEM, STEM, AFM)
- Experimental Conclusions
  - Silver does not diffuse through intact SiC

TRISO Fuel Particles

- Purpose of silicon carbide
  - Structural support for fission product containment
  - Fission product barrier
- Why worry about silver?
  - Highly mobile fission product
  - Increased yield from Pu isotopes
  - Increased activity levels in reactor systems
  - $^{110m}Ag \;\; t_{1/2} = 243$ d
- Previous Hypothesis

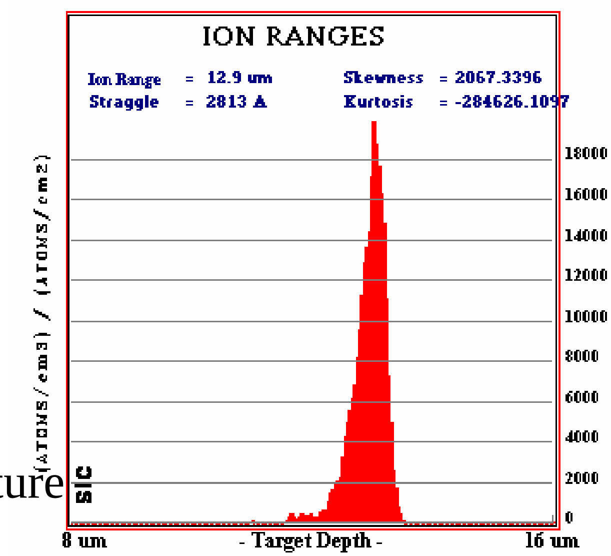
  Silver diffuses through intact SiC layers



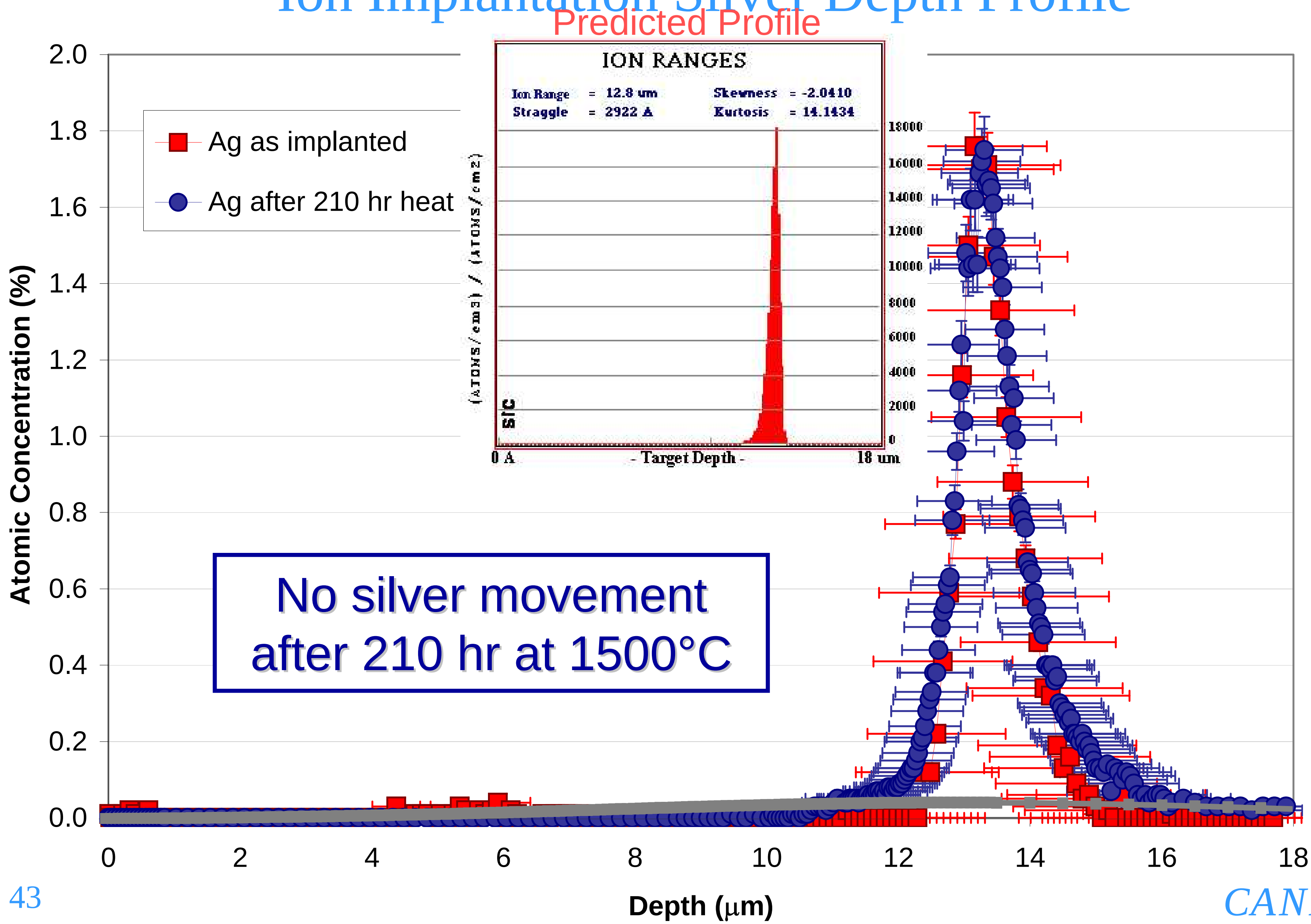
Typical HTGR TRISO fuel particle coatings
K. Sawa, et al. *J. Nucl. Sci. Tech.* **36** (1999) p782.

Silver Ion Implantation

- 161 MeV silver beam, peak concentration at 13 µm
- 93 MeV silver beam, peak concentration at 9 µm
- Implanted ~10^{17} ions = ~2 atomic % silver average, ~25% locally
- Amorphous and recrystallized as-implanted microstructure
- Recrystallized annealed microstructure

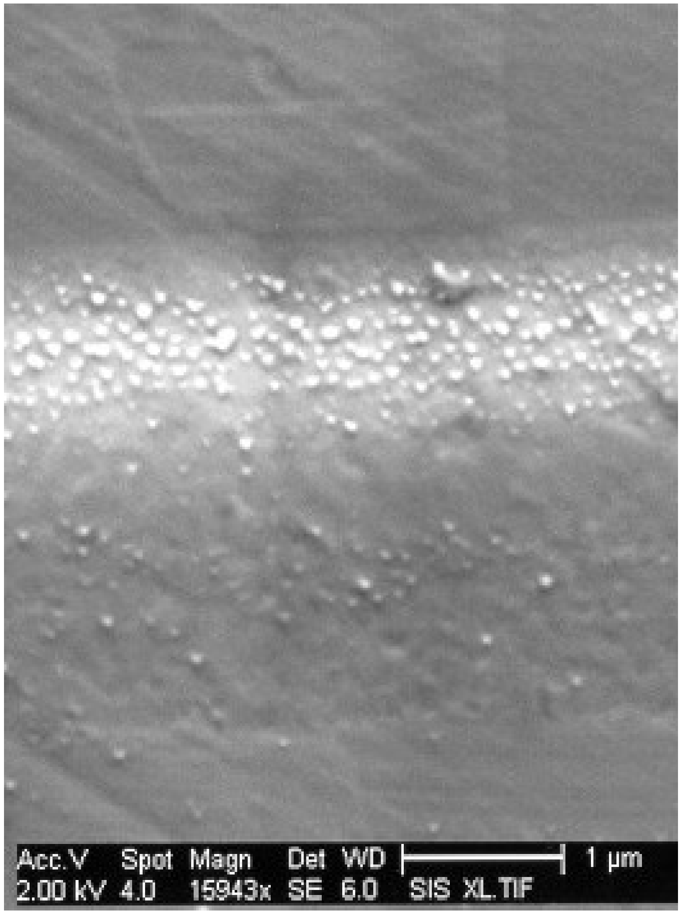


Ion Implantation Silver Depth Profile

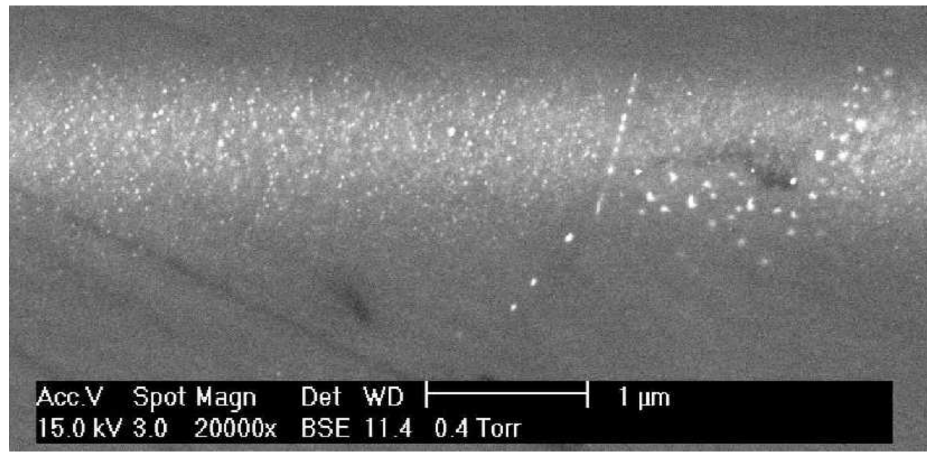


Silver Distribution Before / After Annealing

damaged zone



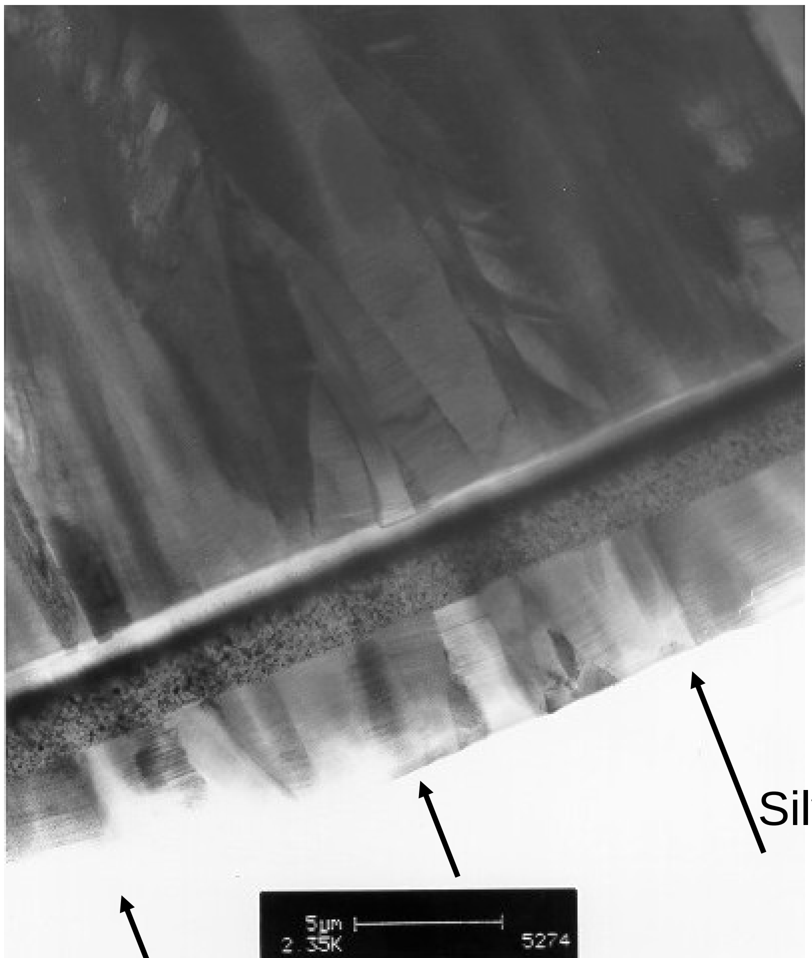
As-implanted
average size = 35 - 95 nm



After heating 1500°C, 390 hr
average size = 10 - 35 nm

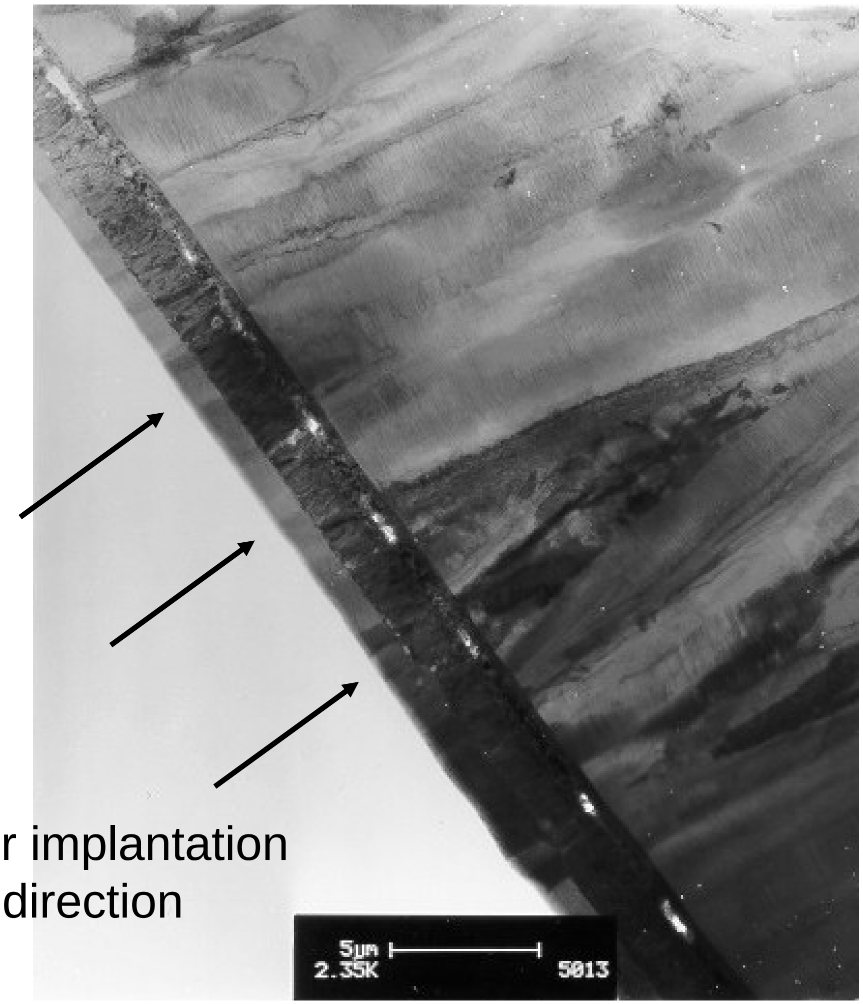
Silver implantation direction

SiC Damage and Recrystallization (TEM)



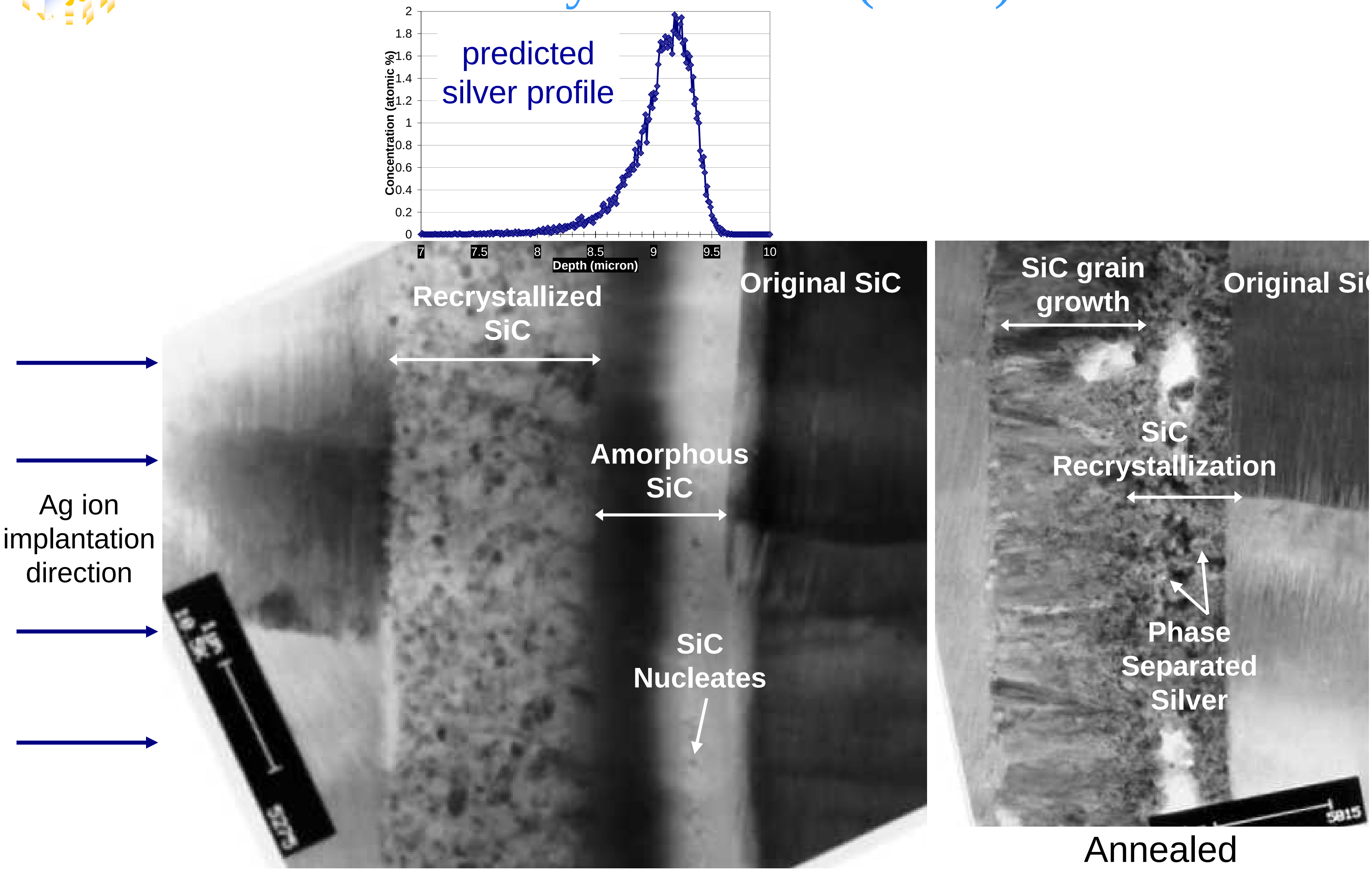
As-implanted

Silver implantation direction



Annealed: 210 hr, 1500°C

SiC Recrystallization (TEM)

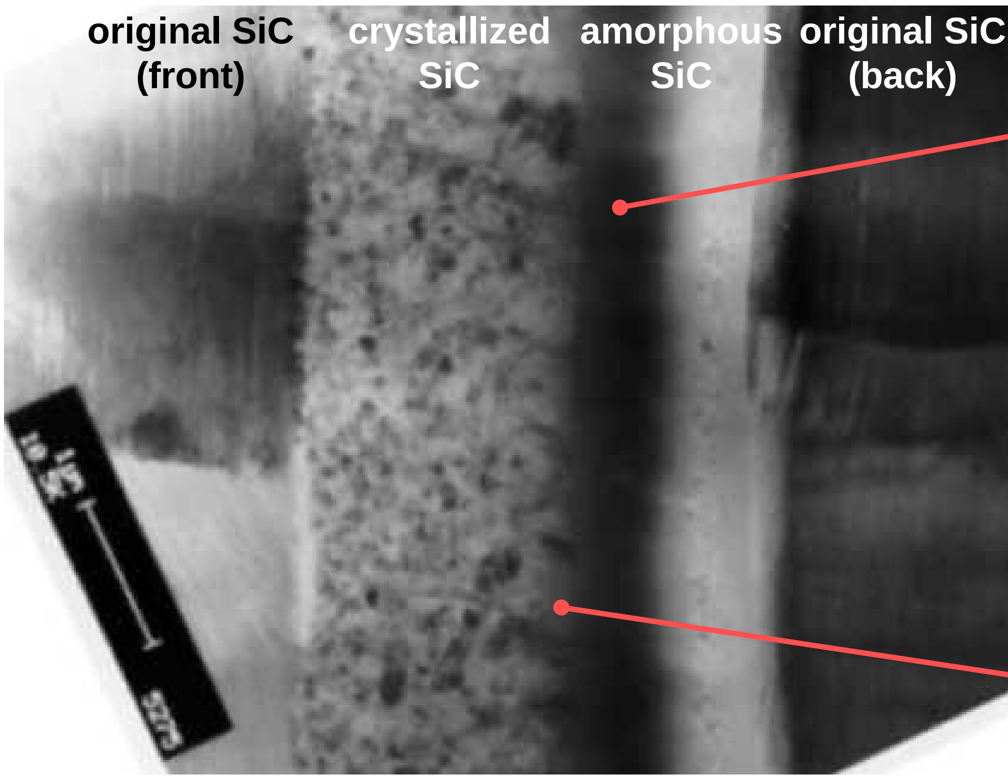


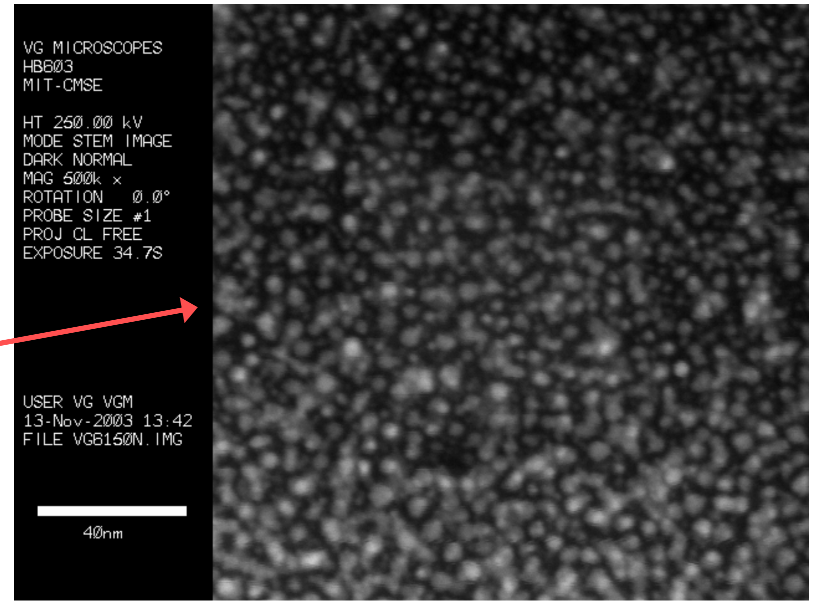
As-implanted

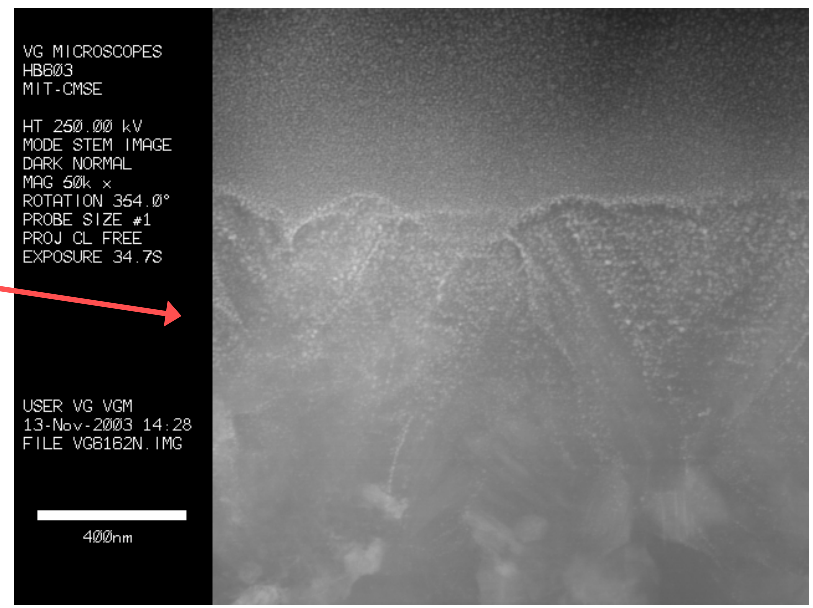
Annealed
210 hr, 1500°C

Silver Distribution During Implantation

Discrete silver in
amorphous SiC zone

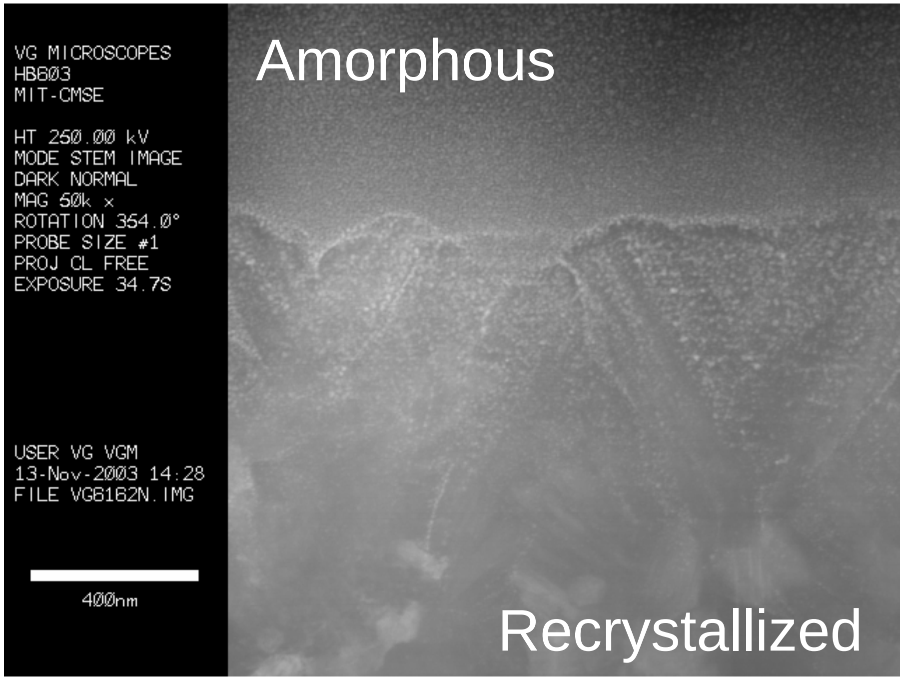




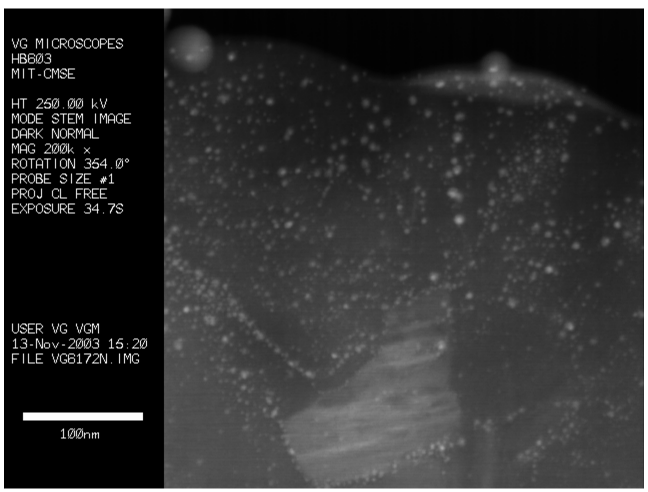


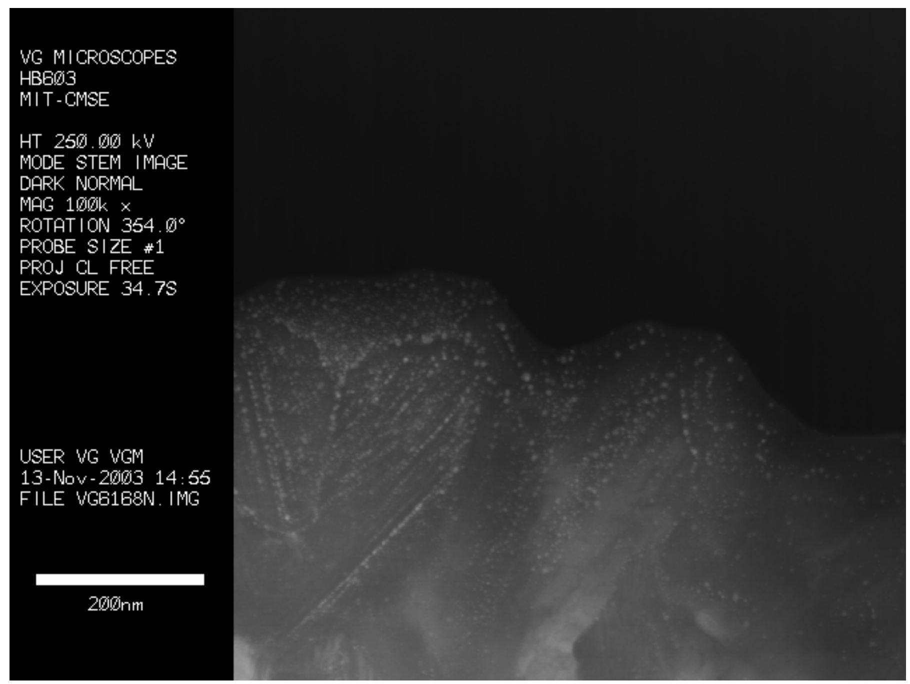
Discrete silver in ~0.5 µm of
recrystallized SiC zone

Silver in SiC Grains (As-implanted)



Silver appears along SiC grains boundaries and faults.

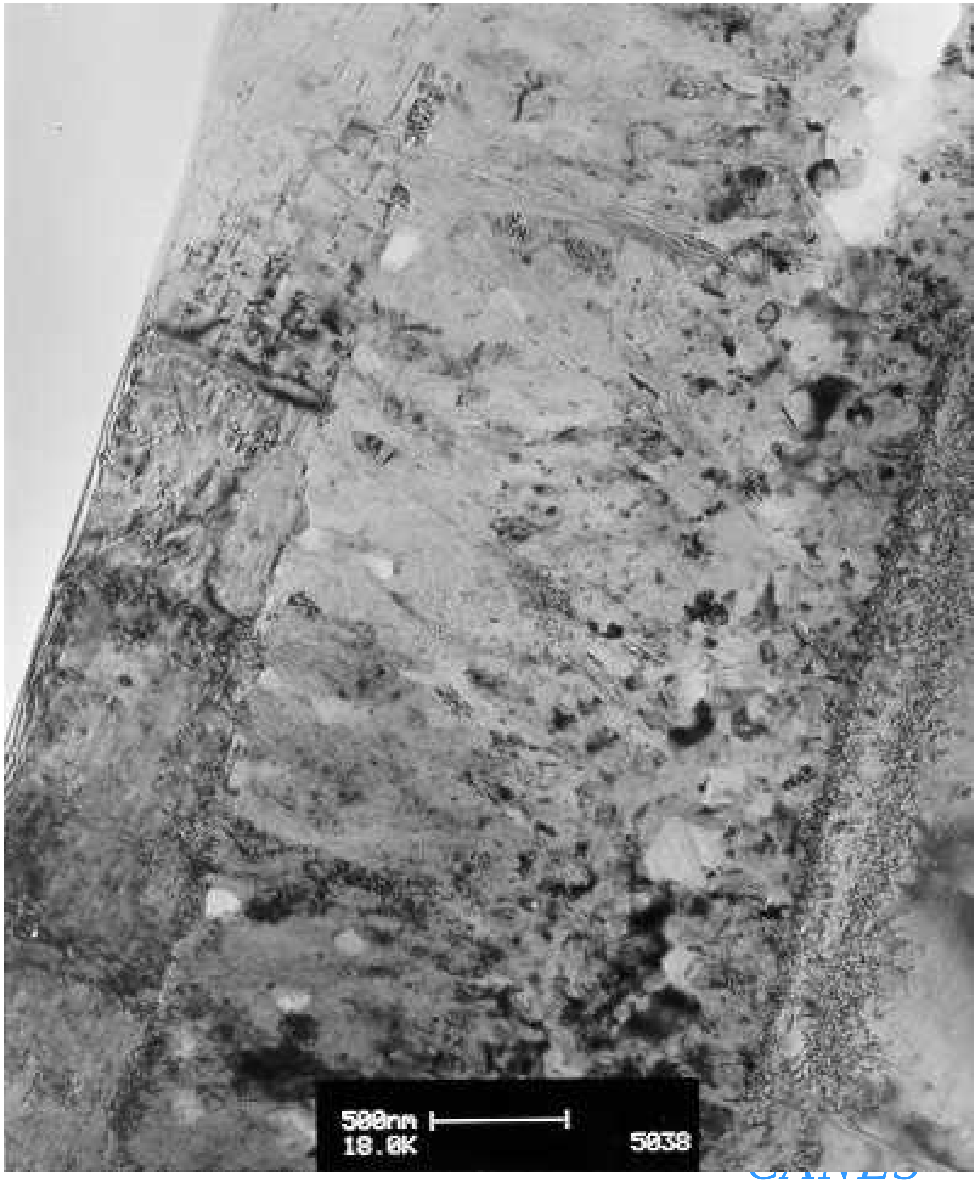




Ion Implantation Summary

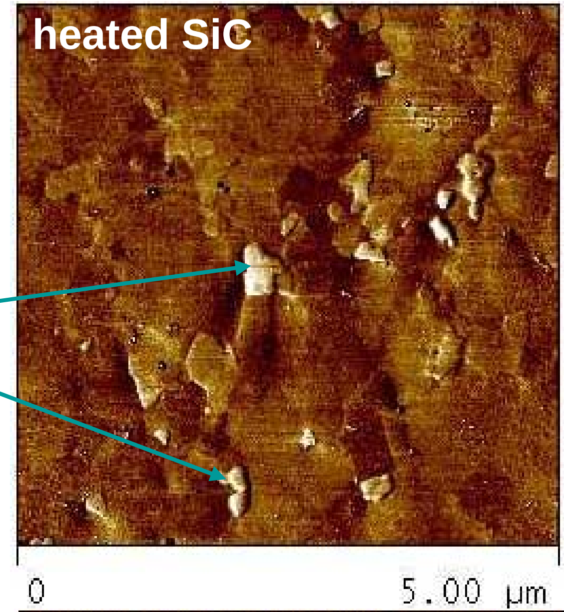
- SiC recrystallization during heating
- Vacancies mobile at 1500°C
- Grain boundary area created during recrystallization
- However …

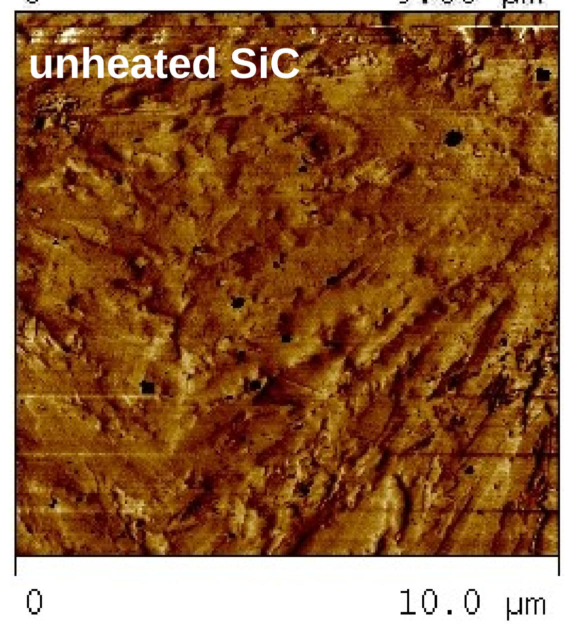
No silver movement observed!



Release Path Hypothesis: Nano-Cracking

- Mechanical pathway
- Nanometer-sized features (cracks) observed in experimental SiC coating in AFM (atomic force microscopy)
- Stresses from differential thermal expansion between individual SiC grains may cause nano-scale cracks
- Aggravated by thermal cycling and radiation damage





Conclusions

- Silver does not diffuse through intact, fine-grained SiC
  - No change in silver concentration profiles
  - No silver movement despite increased grain boundary area
  - Recent PNNL results confirm our results*
- Hypothesis: Vapor migration governs silver release from CVD SiC coatings
  - Mass release observed, but silver profiles not found **(previous results)**
  - Increased leak rates in spherical diffusion couples indicate mechanical cracks **(previous results)**
- Continued SiC development needs to focus on identifying and eliminating crack path
- Coated particle fuel development needs to consider a silver release path other than diffusion

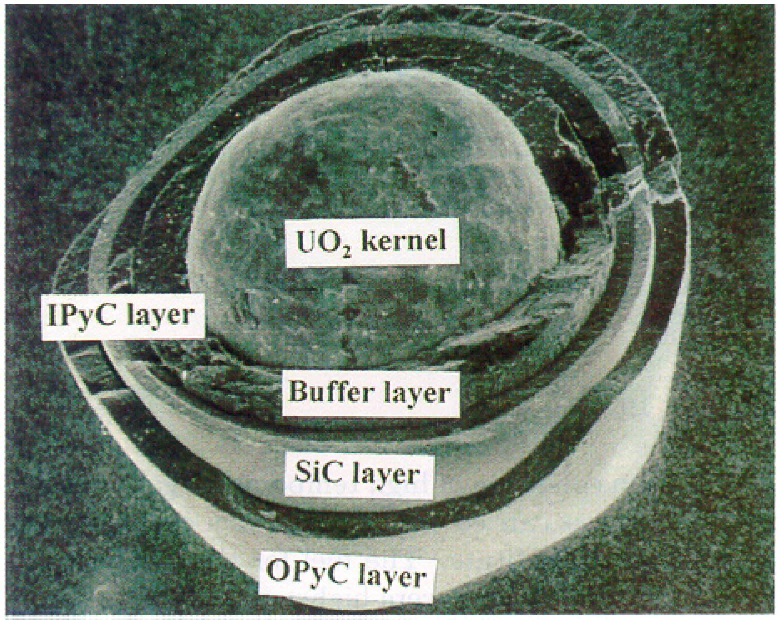
*W. Jiang, W. J. Weber, V. Shutthanadan, L. Li, and S. Thevuthasan, "Thermal and Dynamic Responses of Ag Implants in Silicon Carbide", NIM B, In Press.

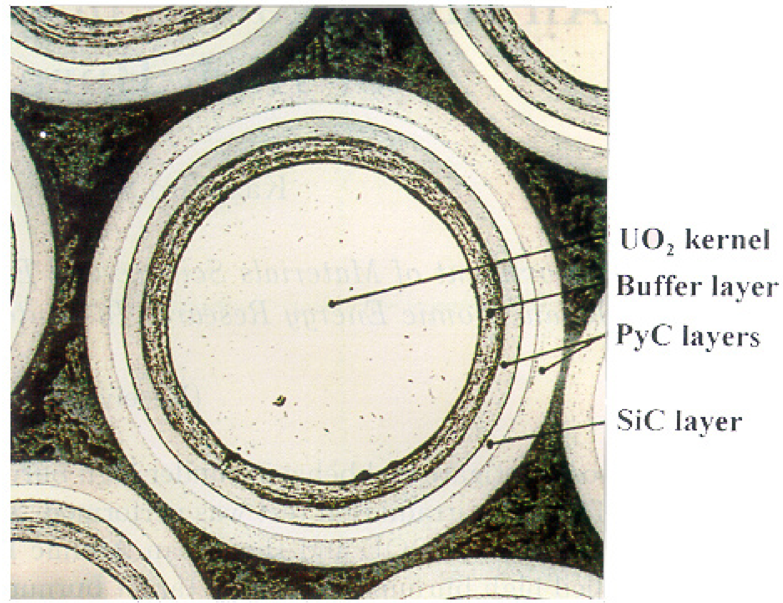
Improvements in Coated Particle Failure Modeling

Outline

- Introduction
  - Coated Fuel Particle
- Improvement on fuel failure model
  - Mechanical failure model – Version 1
  - Real failure scenario
  - Mechanical failure model – Version 2
- Simulations of NPR Irradiations
- Conclusions

Coated Fuel Particle





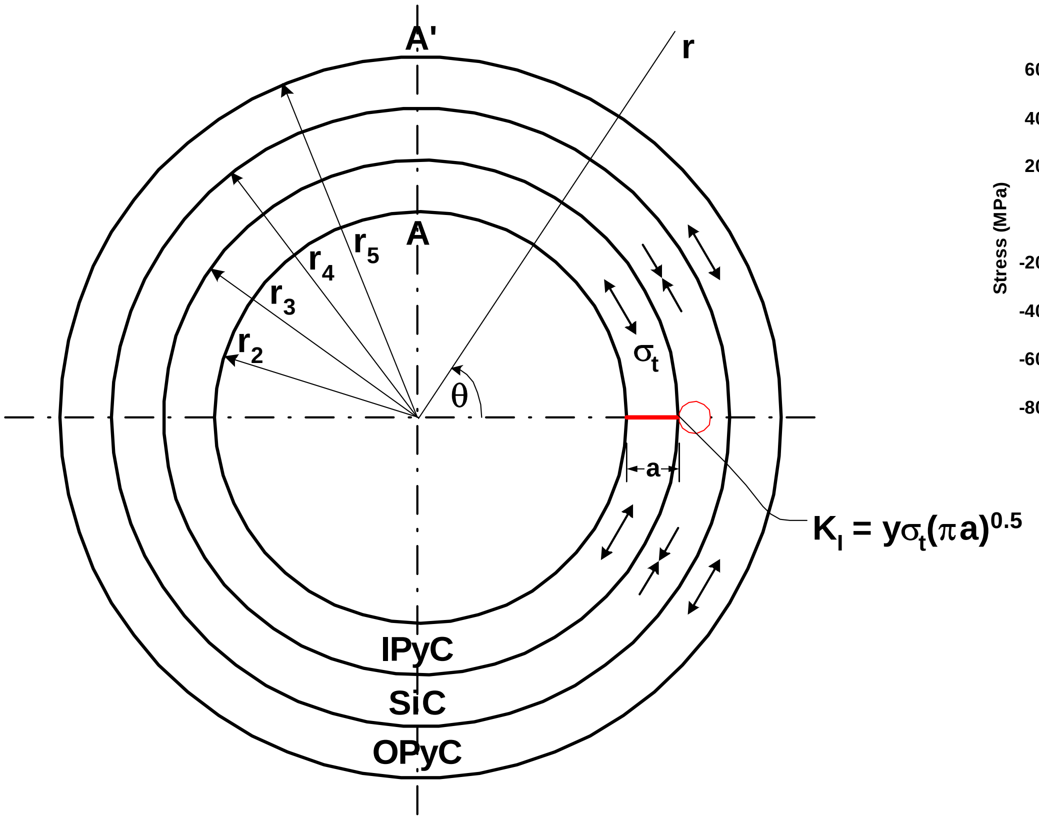
IPyC/SiC/OPyC: Structural layers, pressure vessel and fission product barrier

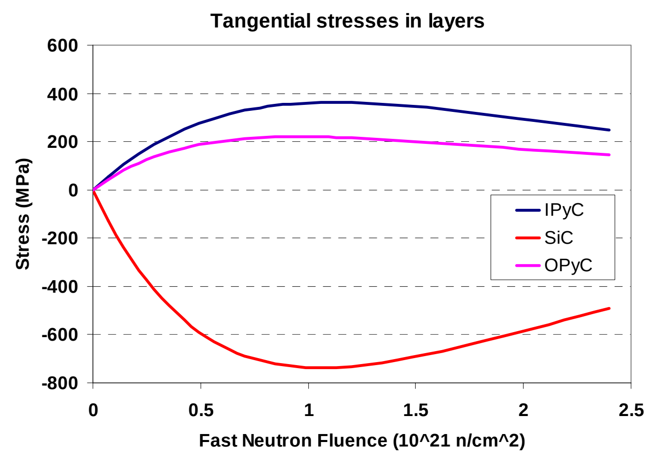
Buffer PyC: Accommodates fission gases and fuel swelling

From Kazuhiro Sawa, et al., J. of Nucl. Sci. & Tech. 36, No. 9, 782 (1999)

The Mechanical Fuel Failure Model

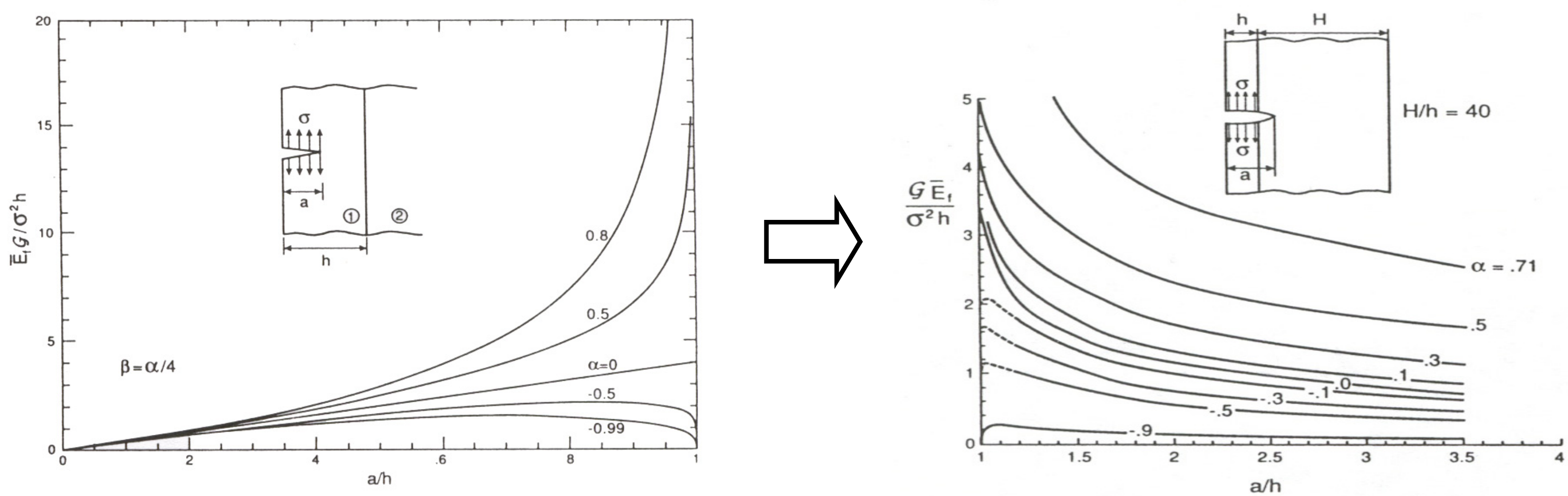
Mechanical Failure Model Version 1





- Does not allow stress relaxation in IPyC and SiC
- Restricted to instant crack propagation into SiC

Realistic Failure Scenario



Driving force available for an edge crack at various depths a/h.

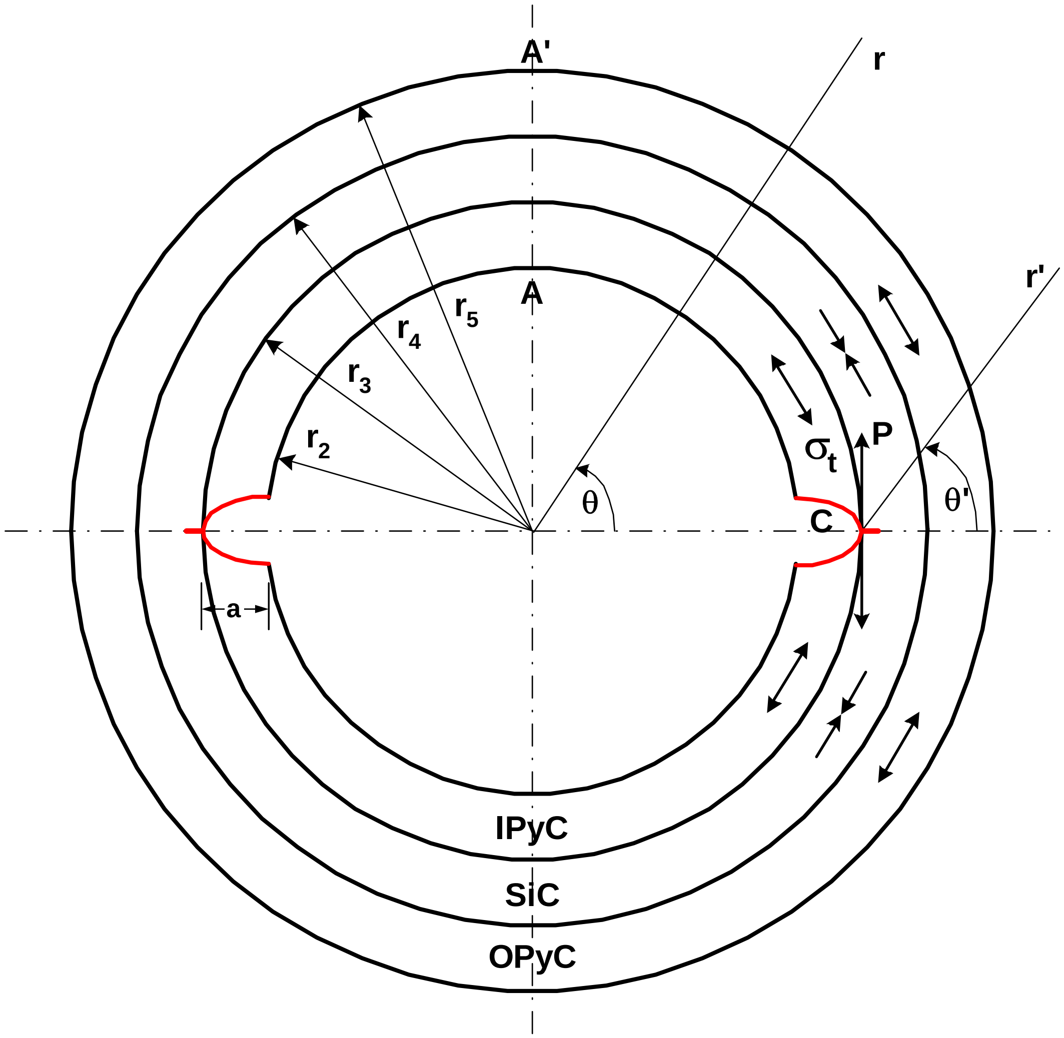
J. Hutchinson and Z. Suo, Advances in Applied Mechanics 29, 133 (1992)

$$\alpha = \frac{\mu_1(\kappa_2 + 1) - \mu_2(\kappa_1 + 1)}{\mu_1(\kappa_2 + 1) + \mu_2(\kappa_1 + 1)} = -0.847$$

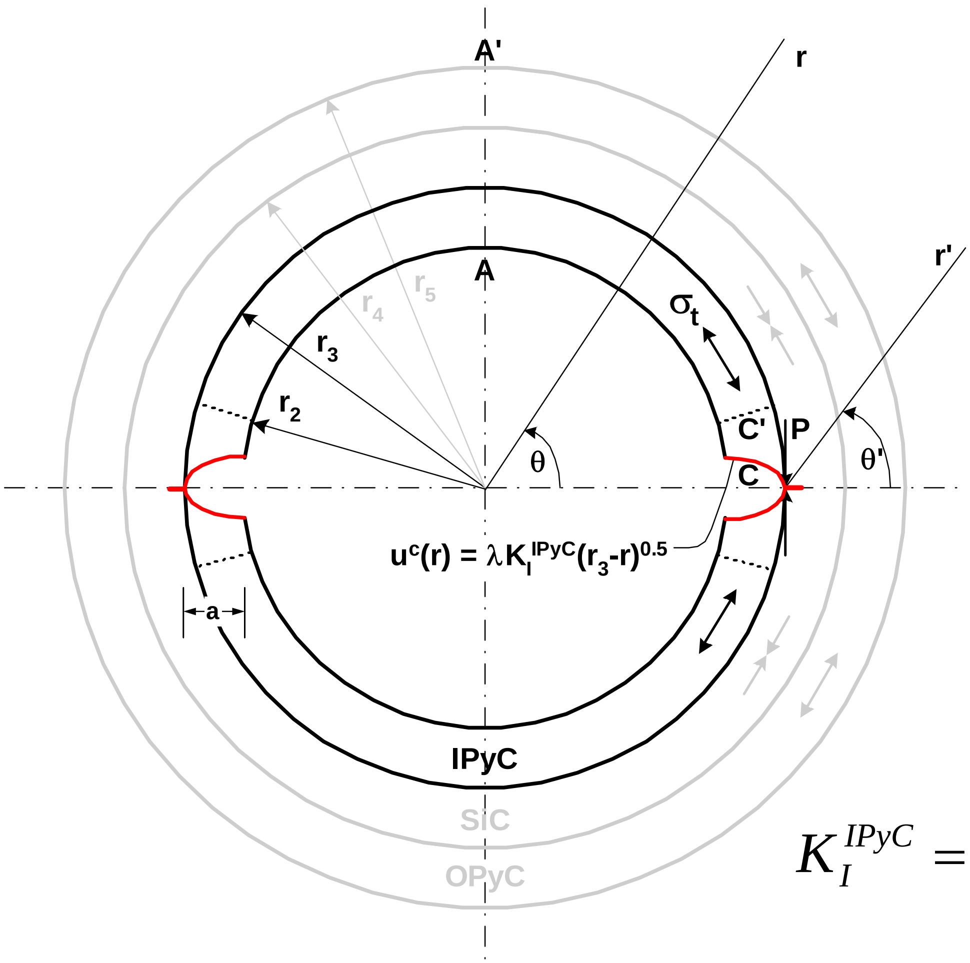
$$\beta = \frac{\mu_1(\kappa_2 - 1) - \mu_2(\kappa_1 - 1)}{\mu_1(\kappa_2 + 1) + \mu_2(\kappa_1 + 1)} = -0.202$$

Improved Mechanical Failure Model: Version 2

Post-Crack Fuel Configuration

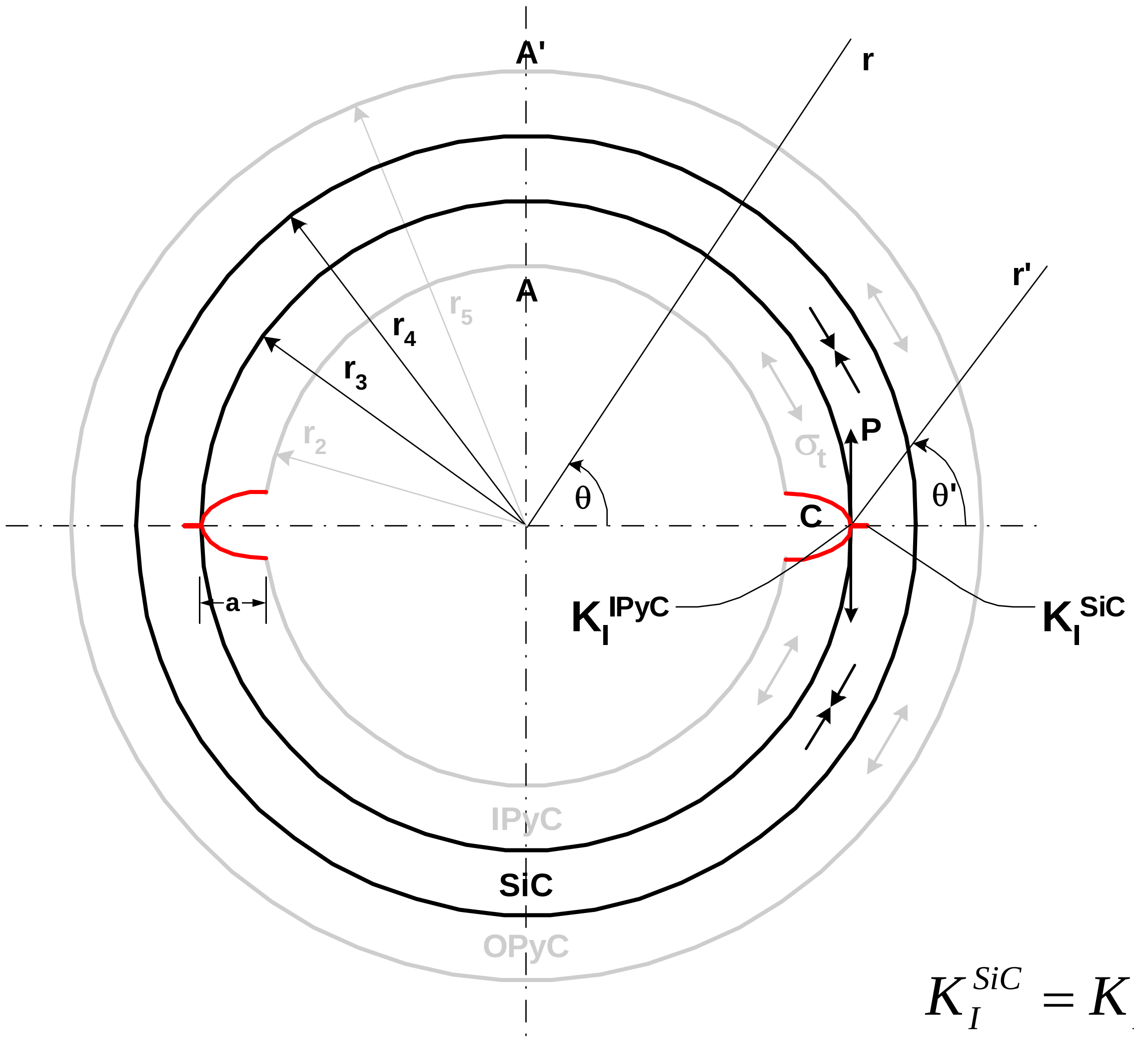


Estimation of Resulting Stresses



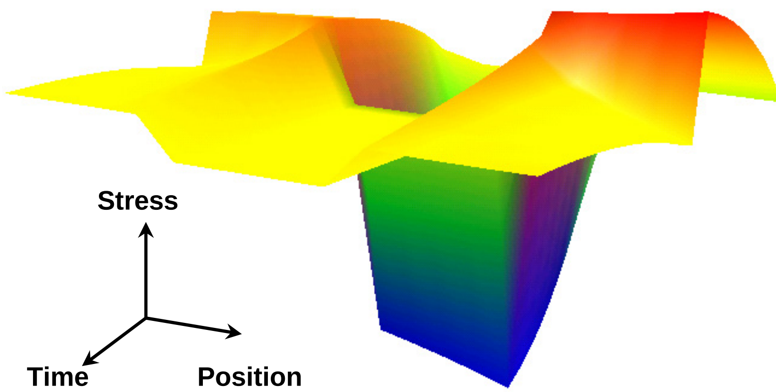
$$K_I^{IPyC} = 0.413\left(1+\frac{r_{IPyC}}{r_{SiC}}\right)\frac{\bar{\sigma}_{IPyC}\sqrt{\pi a_{IPyC}}}{\sqrt{1-\frac{a_{IPyC}}{r_{SiC}-r_{IPyC}}}}$$

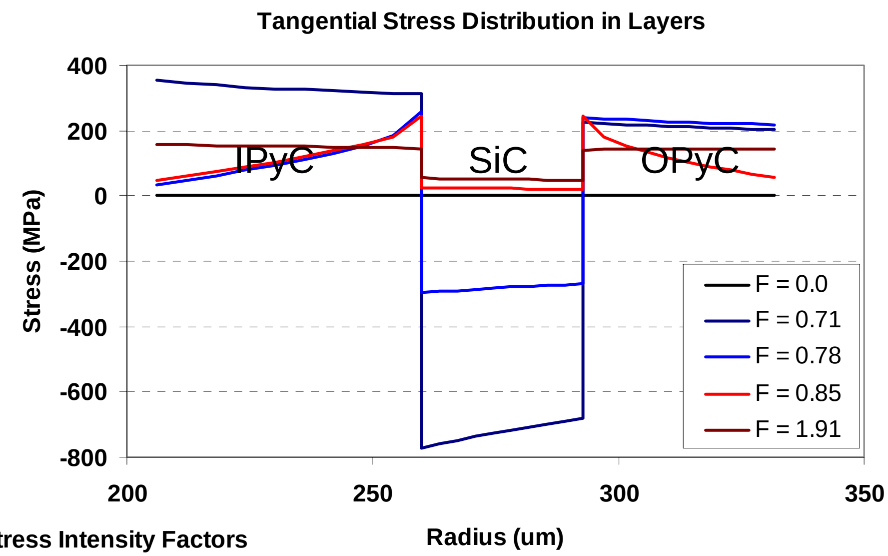
Evaluate Stress Concentration in SiC

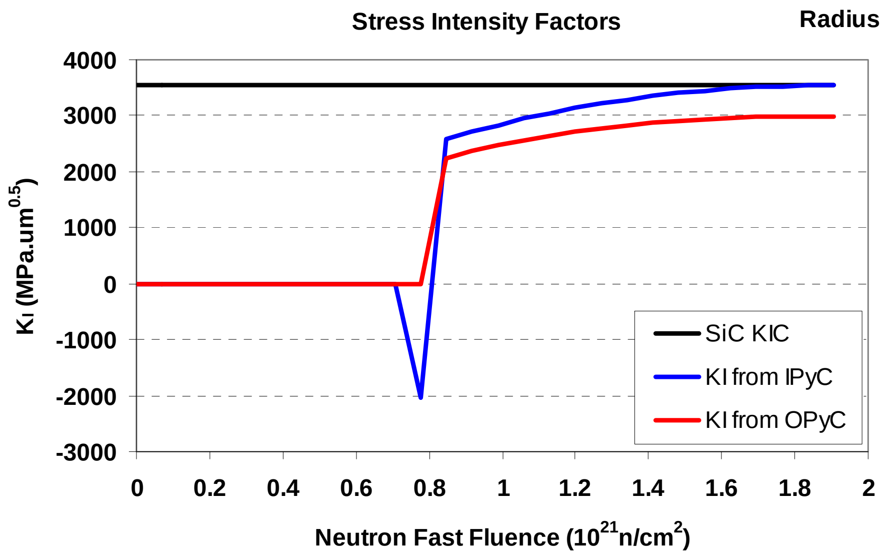


$$K_I^{SiC} = K_I^{IPyC}\sqrt{d/a_{IPyC}} + \overline{\sigma}_{SiC}\sqrt{\pi d}$$

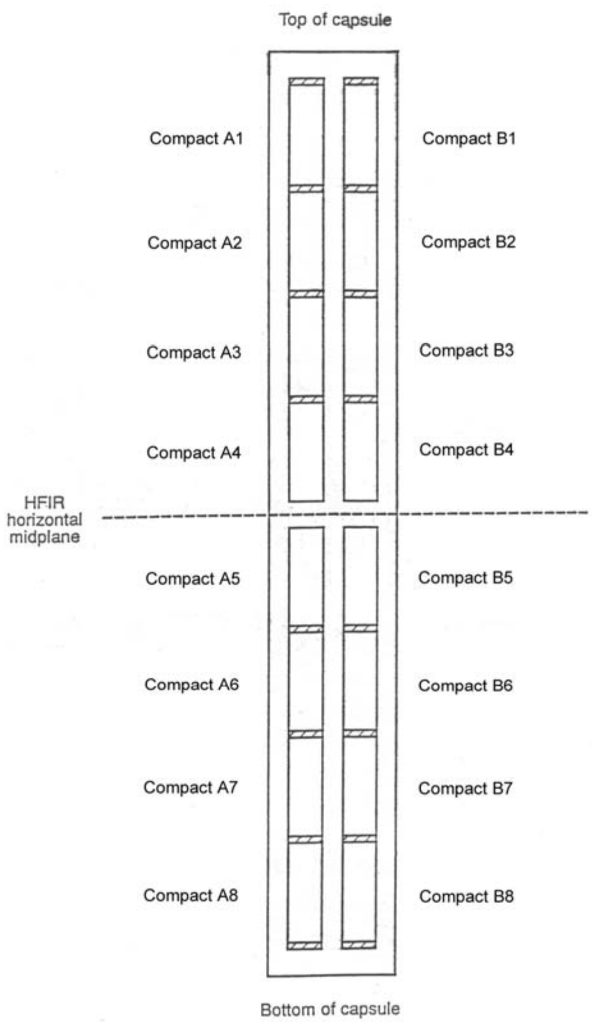
Stress Development in a Failed Particle



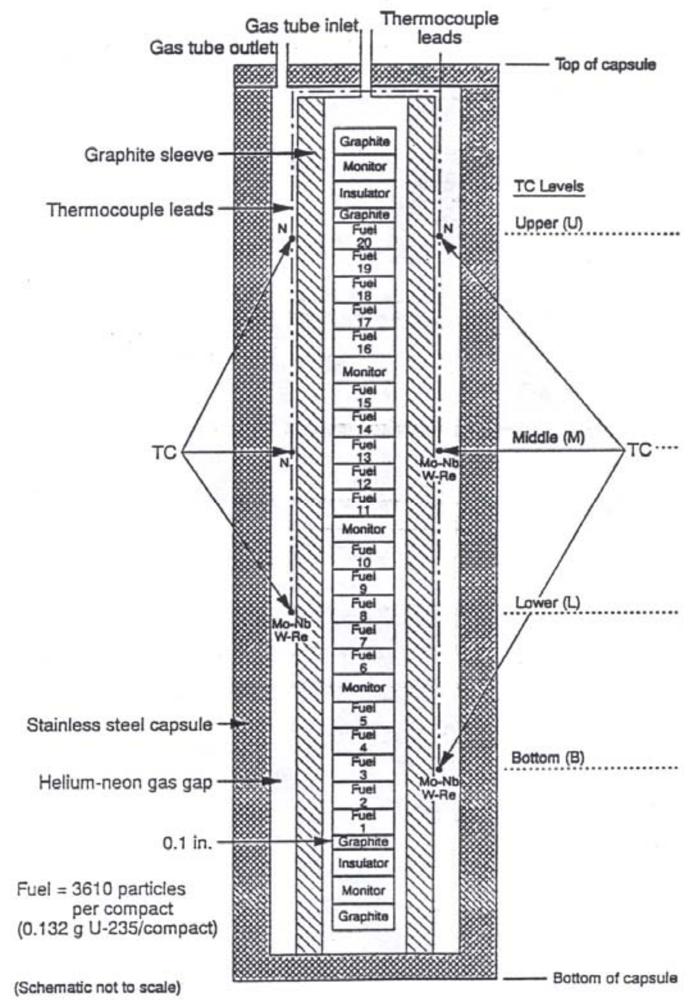




NPR1, NPR2 and NPR1A Capsules



NPR1/NPR2

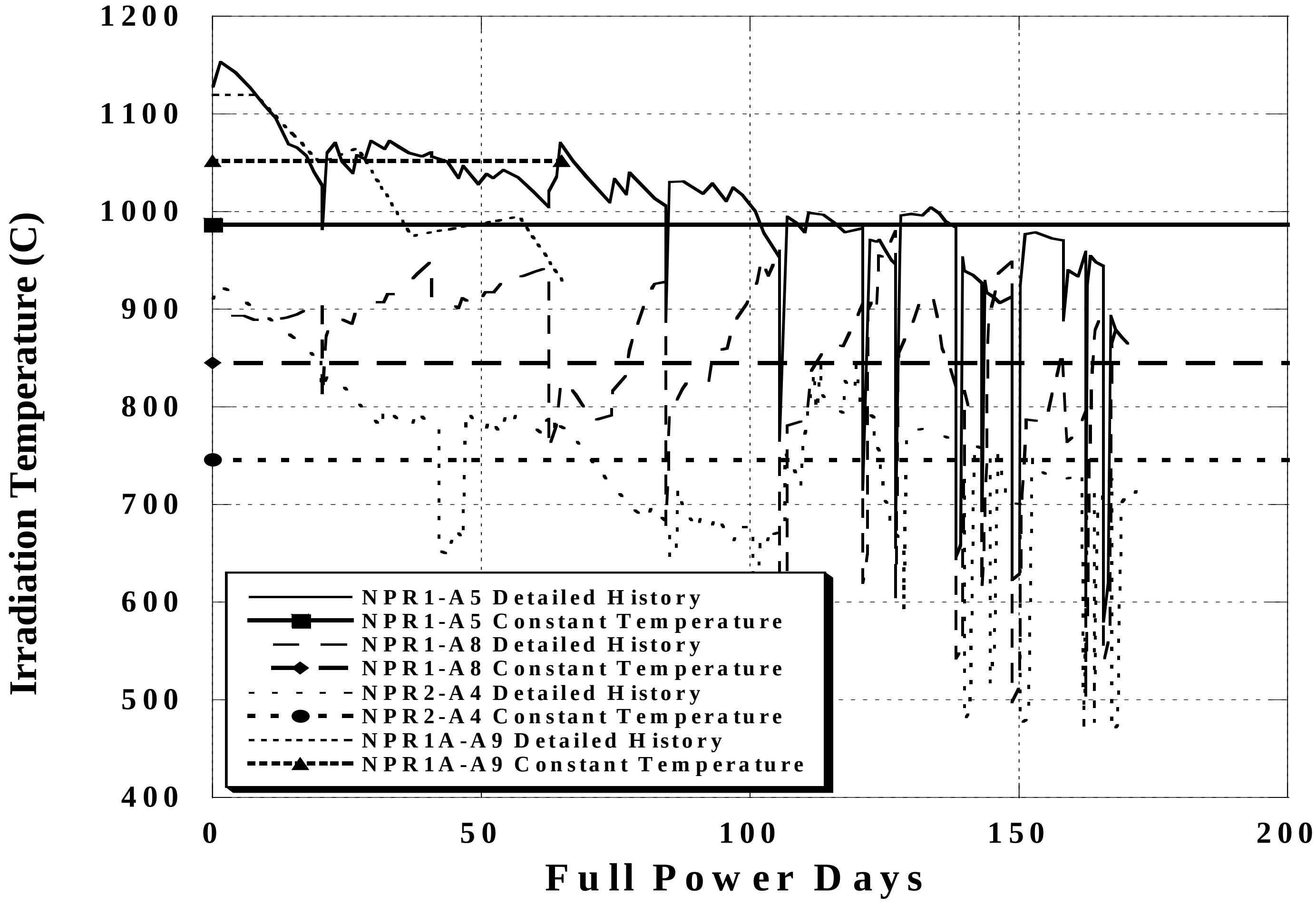


NPR1A

Irradiation Conditions for NPR Compacts

Compact ID	# Particles Contained	Avg. Irr. Temp. (°C)	Fast Fluence (10^{25}n/m^2)	Burnup (%FIMA)
NPR1-A1	6126	874	2.4	74.0
NPR1-A2	5266	1050	3.0	77.0
NPR1-A3	4228	1036	3.5	78.5
NPR1-A4	3755	993	3.8	79.0
NPR1-A5	3755	987	3.8	79.0
NPR1-A6	4228	1001	3.5	78.5
NPR1-A7	5266	1003	3.0	77.0
NPR1-A8	6126	845	2.4	74.0
NPR2-A4	3755	746	3.8	79.0
NPR1A-A9	3610	1052	1.9	64.0

NPR Power Histories

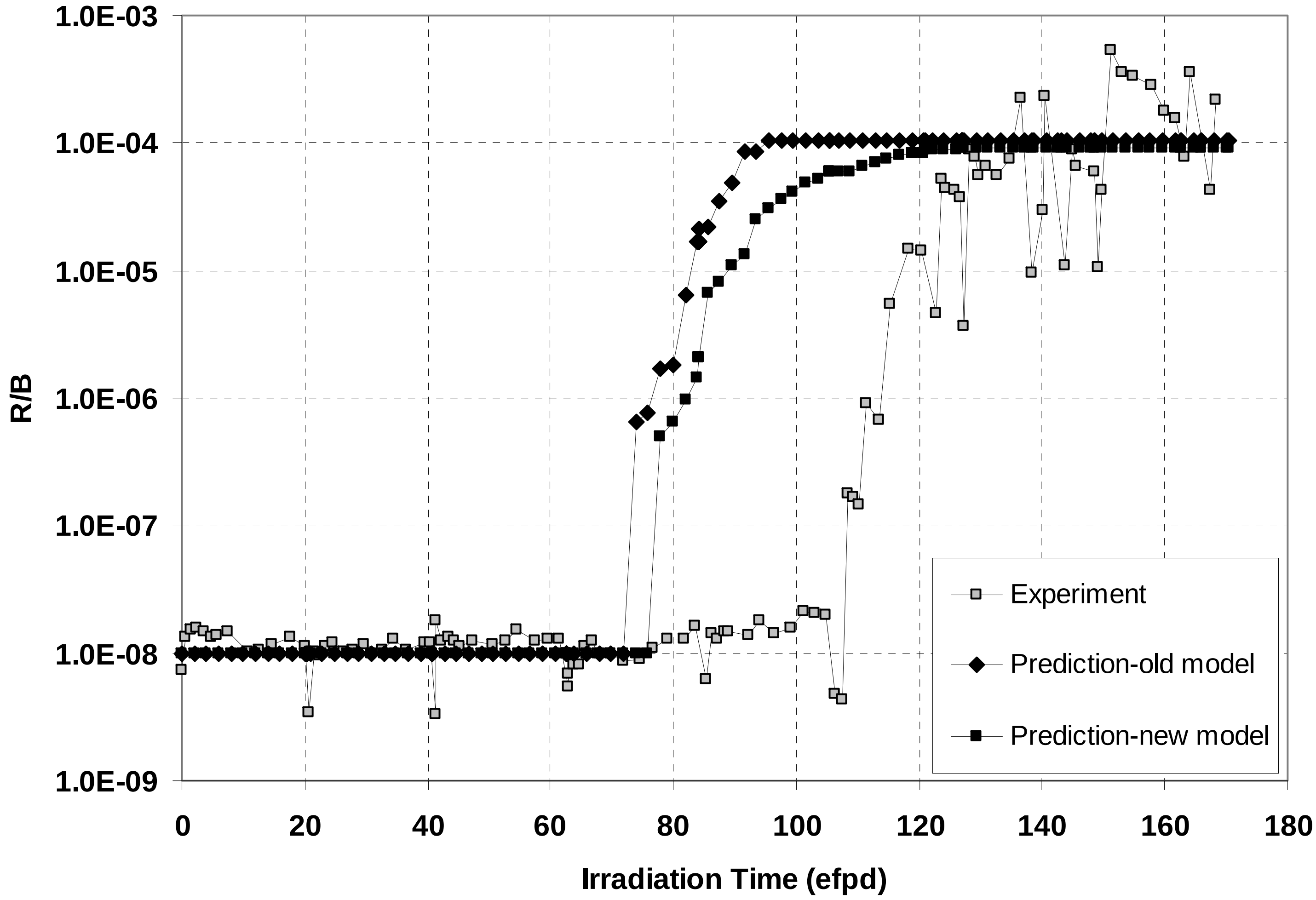


Particle Failure Results

Compact ID	PIE	PIE – 95% Confidence	Old Failure Model	New Failure Model
NPR1-A1	-	-	1.61	1.21
NPR1-A2	-	-	1.00E-4	0.00
NPR1-A3	-	-	0.0250	0.0310
NPR1-A4	-	-	0.857	0.257
NPR1-A5	0.6	0<p<3	0.358	0.122
NPR1-A6	-	-	0.272	0.153
NPR1-A7	-	-	0.0683	0.0383
NPR1-A8	0	0<p<2	2.74	3.01
NPR2-A4	3	2<p<6	13.9	8.84
NPR1A-A9	1	0<p<5	0.492	0.534

(Numbers are in percent)

Kr^{85m} R/B of NPR1 Capsule

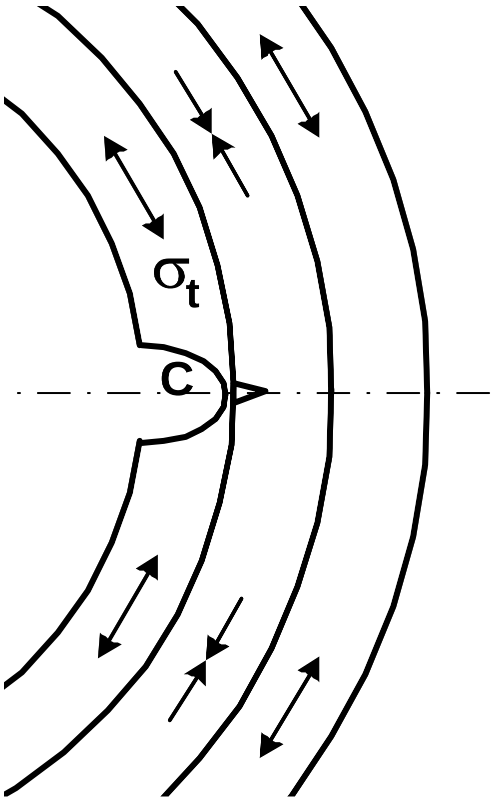


Overall Failure of NPR1 Capsule

	Irradiation Test	Prediction (Old Model)	Prediction (New Model)
No. Particles Contained	77500	77500	77500
No. Failed Particles	526/625 [(a)]	656	565
Failure Probability	0.679%/0.806%	0.846%	0.729%
Peak Fluence at Peak Failure (10^{21}n/cm^2)	2.17	1.08	1.48
Peak Burnup at Peak Failure (% FIMA)	75.3%	67.1%	71.2%
EFPD at Peak Failure	123.6	89.5	101.5
Peak Temperature at Peak Failure (C)	1107	1072	1134

(a): Counts of spikes from ionization chamber/From readings of the Kr^{85m} R/B

Further Improvement on Mechanical Fuel Failure Model



$$\dot{a} = \gamma \left(C^* \right)^{\frac{n}{n+1}}$$

$$C^* = \int_{\Gamma} \left(\dot{w}\, dy - \sigma_{ij} n_j \frac{\partial \dot{u}_i}{\partial x} ds \right)$$

Irradiation Creep Crack Growth causes the macroscopic crack in IPyC to connect with surface intrusions at the interface

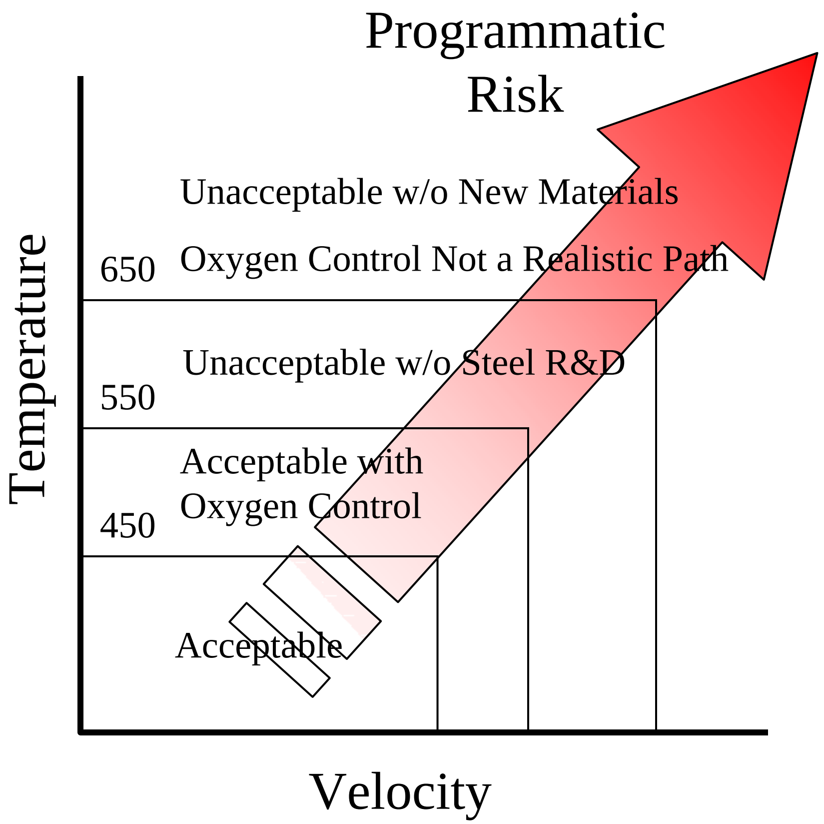
Conclusions

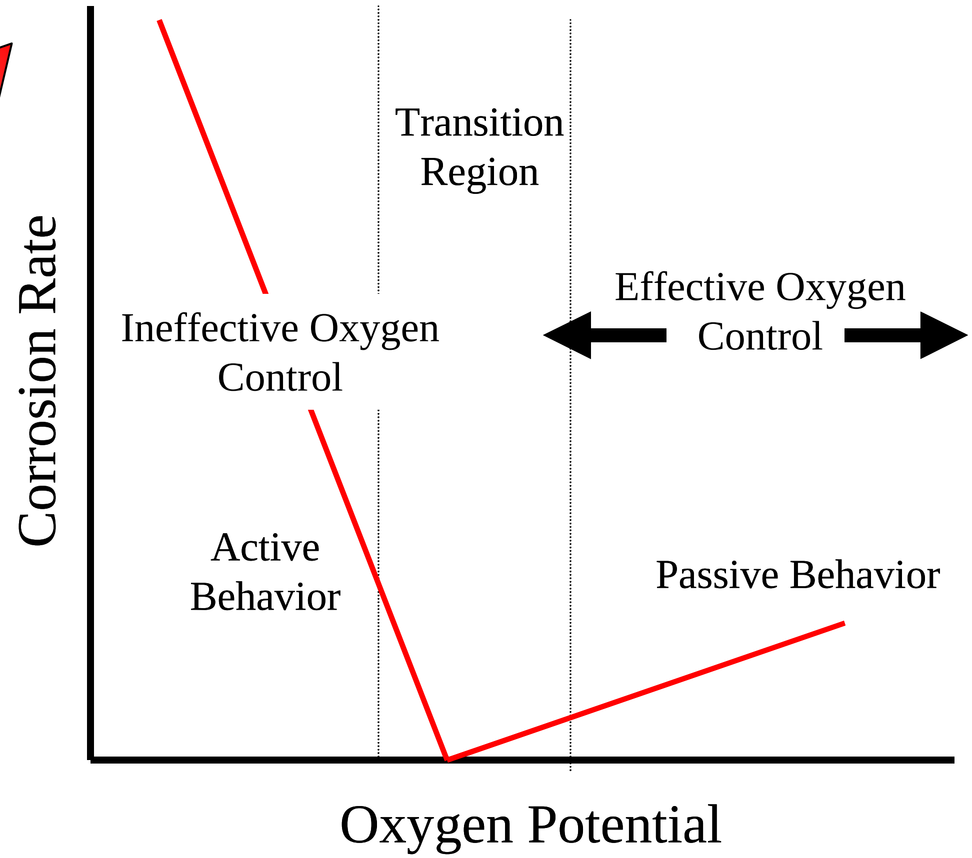
- The improved fuel failure model is a better representation of the physics of cracking in coating layers
  - Accounts for stress relaxation and redistribution
  - Considers post-crack stress development due to irradiation
  - Gives better stress intensity factor calculation in layered structure with material property mismatch

Alloy Development for Pb-Bi Service

Corrosion Issues

- Key Variables: Temperature, Velocity, O_2 Potential
- Location of "Transition" Region a Function of Material System





Program Goals

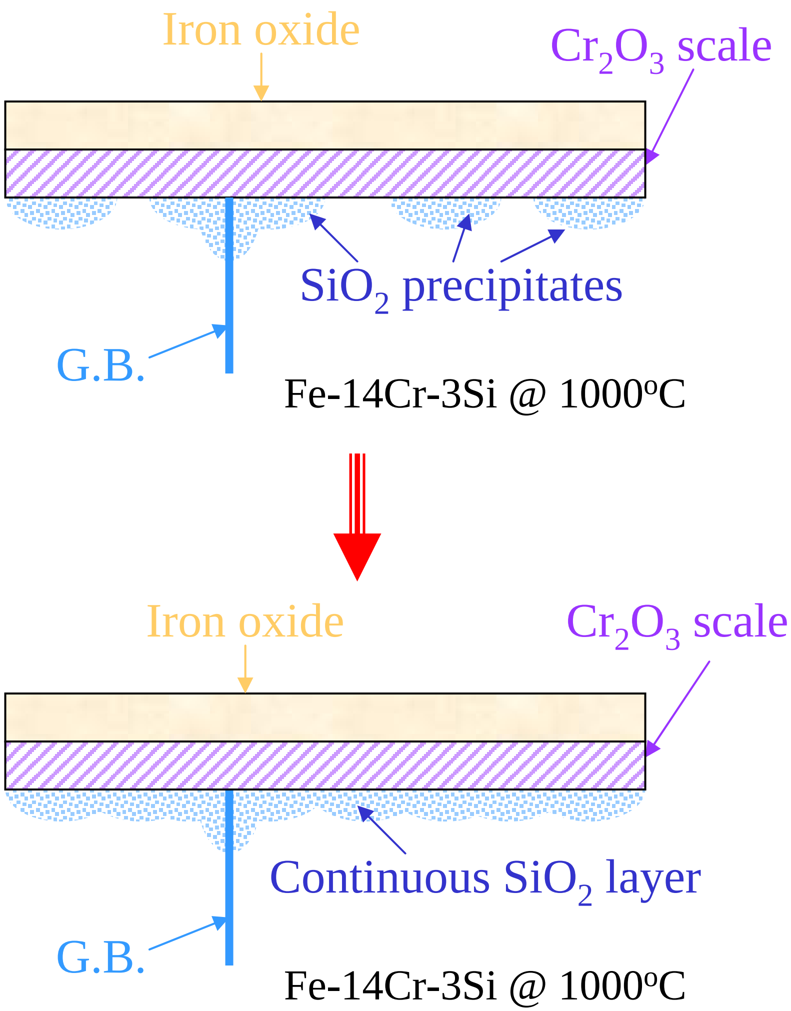
- Development of a fundamental understanding of the behavior of structural materials in Pb and Pb-Bi systems. Establishment of fundamental basis for observed behavior (e.g. EP-823, etc.)
- The development of new materials that are resistant to corrosion in Pb and Pb-Bi systems. Optimized for specific applications
- Interim Temperature Goal: 650°C
- Ultimate Temperature Goal: 1100°C

Performance Requirements: Phase I

- Resistance to Corrosion in Pb and Pb-Bi Eutectic at Temperatures up to 650°C
- Resistance in BOTH Oxidizing (w.r.t iron oxide potential) and Reducing Potentials. (Avoids Fe-Oxide Phase Transformation Issues)
  - *Protection Scheme Must be Self-Healing in Oxidizing Environments*
  - *Low Dissolution Rates in Low (or zero) Oxygen Concentration Regions (unavoidable in actual system applications)*
- High Temperature Strength: $\sigma_{yield} \geq 210$ MPa @ 650°C
- Adequate Creep Rupture Strength
- Microstructural Stability (Implies Solid Solution Strengthening System)
- Fabrication Using Current Technology
  - *Suitable as Structural Material & Cladding Material*

Silicon as an Alloying Element

- Oxides Formed on Fe-Cr-Si alloy Under Oxidizing Conditions
  - Multiple layer structure
    - Outer iron oxide scale
    - Inner Cr_2O_3 scale
    - SiO_2 at scale/alloy interface
  - Synergistic Effect of Cr and Si



Reference: F. H. Stott, *Materials Science and Technology*, August 1989, Vol. 5, 734-740

Hypothesis

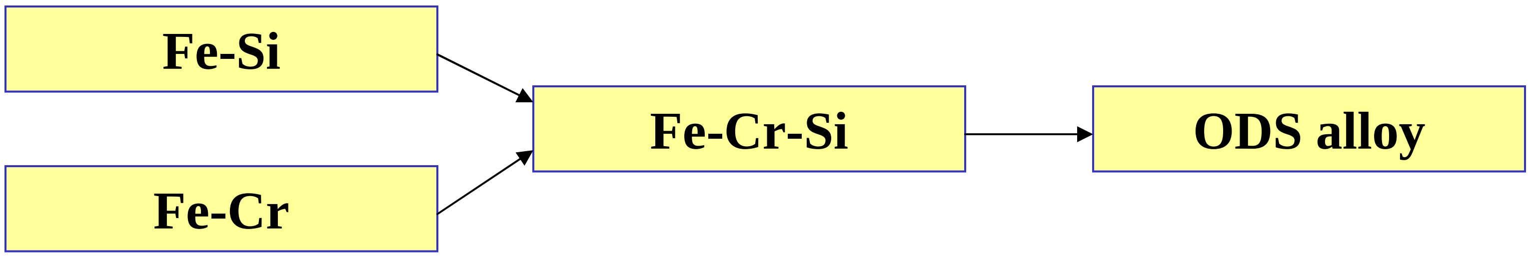
- Fe Based Alloys Have Been Shown to be Resistant to Pb/Pb-Bi Corroison
  - *However, the scientific basis has not been fully developed*
- Fe-Based System Can Be Optimized for Pb, Pb/Bi Service
- Use of a low solubility matrix (Fe) can be augmented by the addition of Si and Cr to enhance corrosion resistance at both extremely reducing conditions, with respect to Fe_3O_4, SiO_2 and Cr_2O_3, mildly oxidizing potentials with respect to Fe_3O_4, but not SiO_2, and Cr_2O_3, and strongly oxidizing potentials with respect to Fe_3O_4, SiO_2, and Cr_2O_3.
- For Extremely Reducing Conditions
  - Fe-Cr-Si matrix will dissolve with preferential enrichment of Si at the surface. Further dissolution will be inhibited by Si enrichment.
- For Mildly Oxidizing Conditions
  - Fe-Cr-Si matrix will dissolve with preferential enrichment of Si at the surface. Depending on oxygen potential, a point will be reached where SiO_2 layer will become stable. Layer effectiveness will be augmented by Cr.
- For Extremely Oxidizing Conditions
  - Multi-Oxide scale layer will form with SiO_2/Cr-rich layers at metal/scale interface which will inhibit further metal dissolution.
- Strength Objectives Can be Achieved
  - Additional strength can be provided by dispersion of fine oxides (~20 nm) (e.g. Y2O3)

Long Term Program Development Paths

- T ≤ 650° C: Phase I
  - Corrosion Resistance: Fe-Cr-Si Solid Solution
  - Strength: ODS

- T≥ 650° C: Phase II *(The Clash of the Titans- Thermodynamics vs. Kinetics)*
  - Ceramic/Composite/Metallic Functionally Graded System
    - Ceramic/Composite-Corrosion Resistance
    - Metallic "Backing"-Structural Requirements
    - Thermal Gradient Design
      - *Use of Thermal Barrier Coatings*

Development Path-Phase I



Current Status of Alloy Development

5 Fe-Cr alloys: 1%, 2.25%, 9%, 12%, 18% Cr

8 Fe-Cr-Si alloys: Cr: 2.25~18%, Si: 0.5~2.55%

Candidate ODS Alloy Chemistries

Fe-18Cr-2.5Si with 0.1% Y_2O_3,

Fe-9Cr-2.5Si with 0.1% Y_2O_3

Commercial ODS Alloys

Results

- Initial Focus Has Been on Fe-Si and Fe-Cr System

- Fe, Fe-1.24Si, Fe-2.55Si, Fe-3.82Si
- Fe-1Cr, Fe-2.25Cr, Fe-9Cr, Fe-12Cr
- Fe-12Cr-0.5Si

- Static Isothermal Testing Condition

  Limited Volume of Liquid Metal Limits Exposure Times

  Limited to Film Formation and Short Term Rate Studies

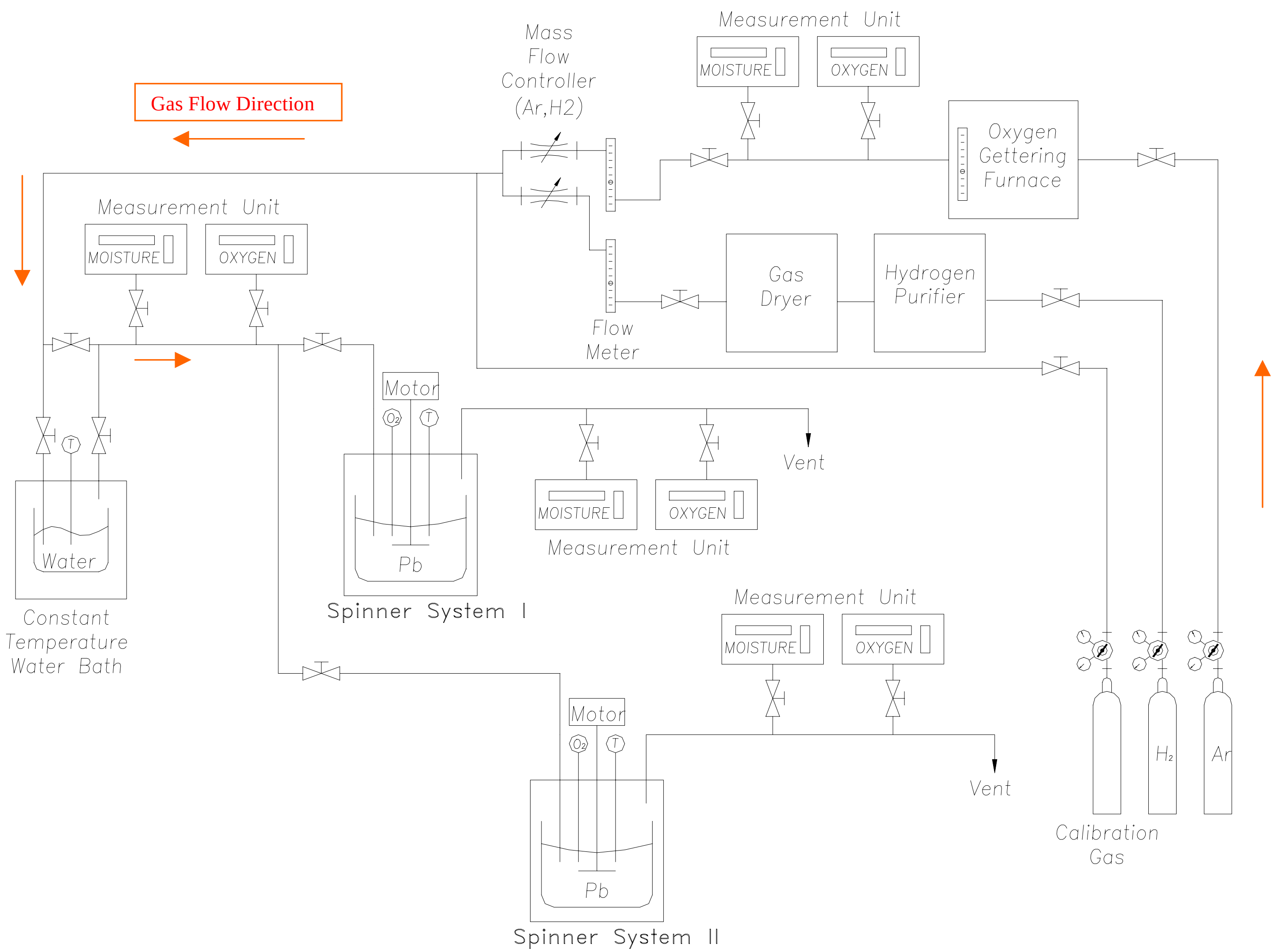
Test Conditions

- Fe-Si alloys
  - Pure Fe, 1.24%Si, 2.55% Si, 3.82% Si
- Fe-Cr alloys
  - 1%Cr, 2.25% Cr, 9% Cr, 12% Cr

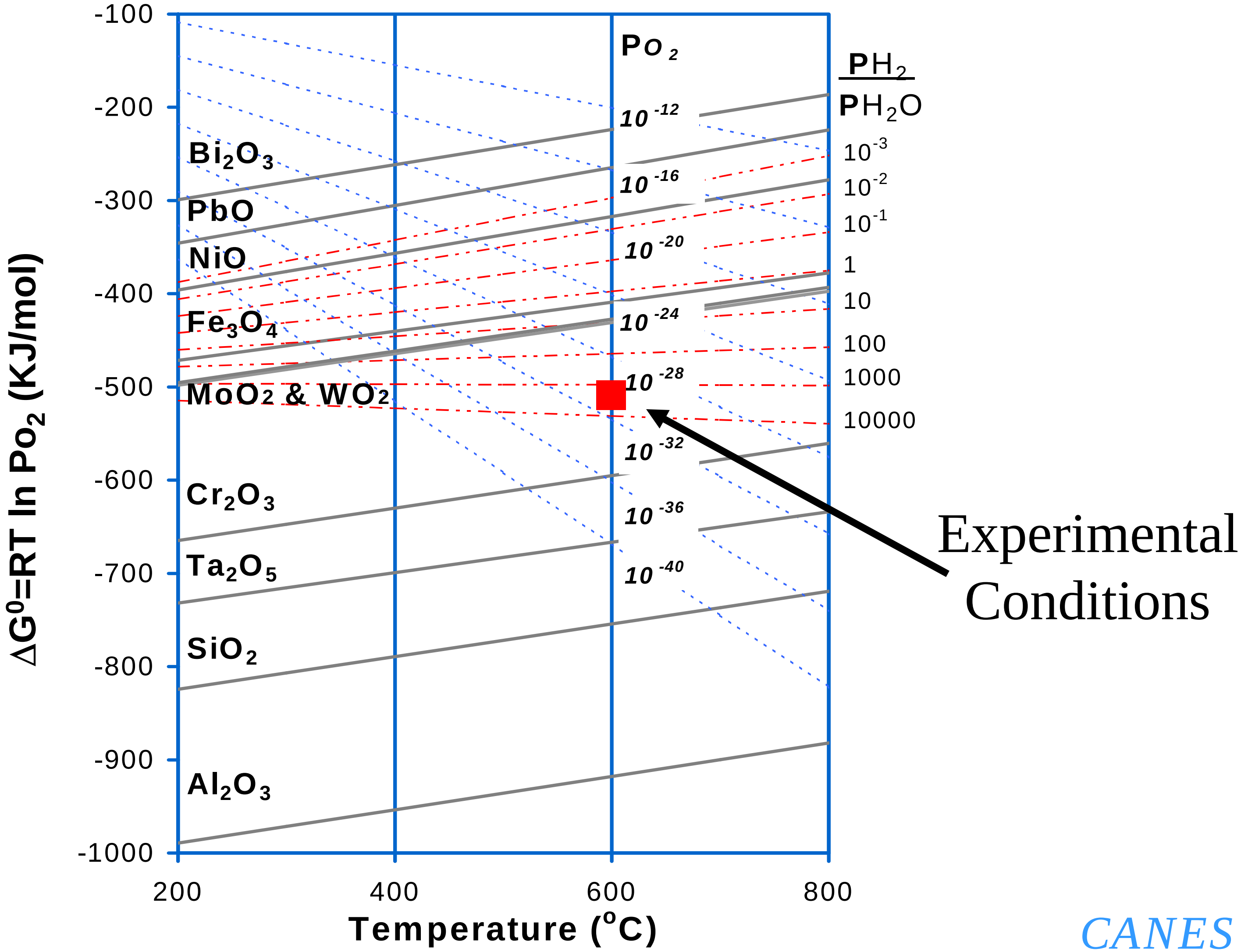
Liquid Metal	Temp(^oC)	Duration (hrs)	Oxygen condition
Pb-Bi	600	100	Fe – Reducing Si – Oxidizing

- Oxygen Potential
  - H_2/H_2O ratio = 3000
  - Reducing condition for Fe
  - Corresponding oxygen partial pressure @ 600^{o}C ~ 10^{-31} atm

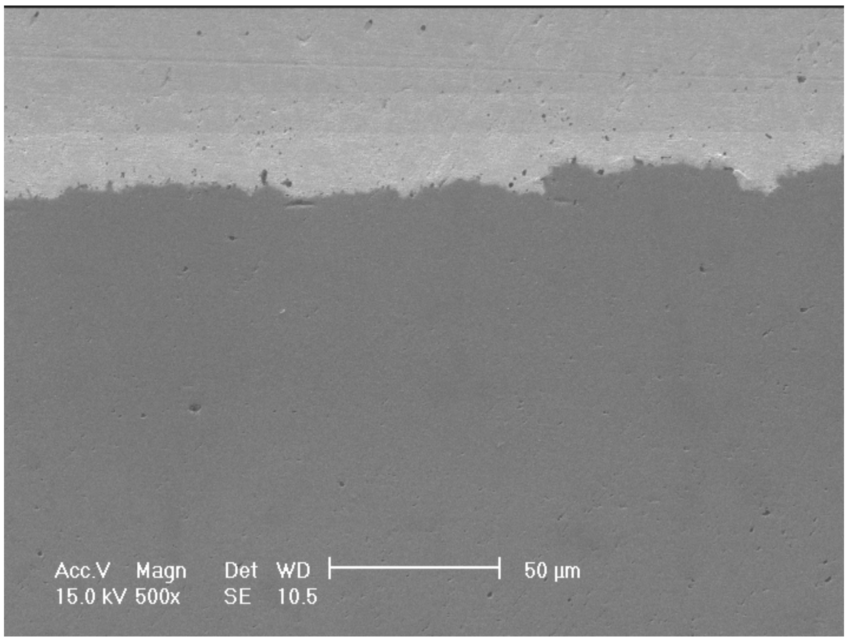
Experimental System

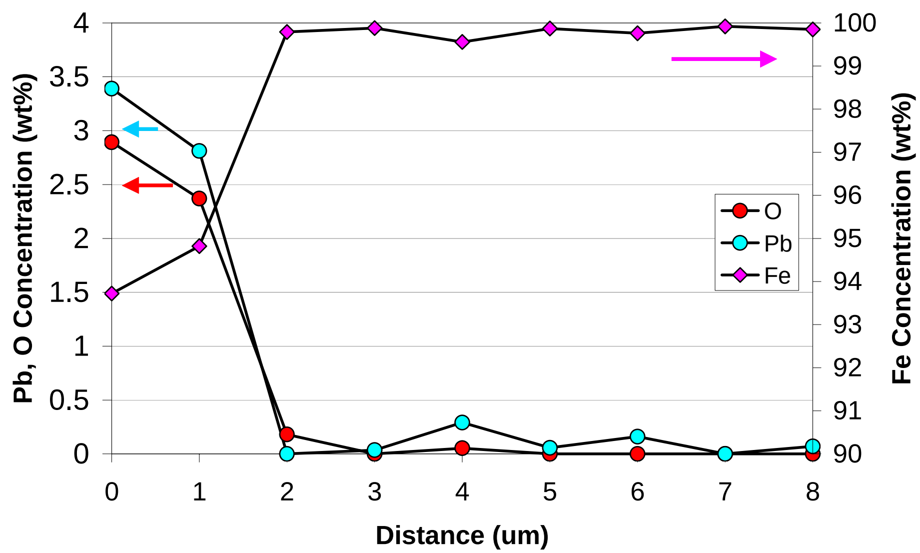


Thermodynamic System

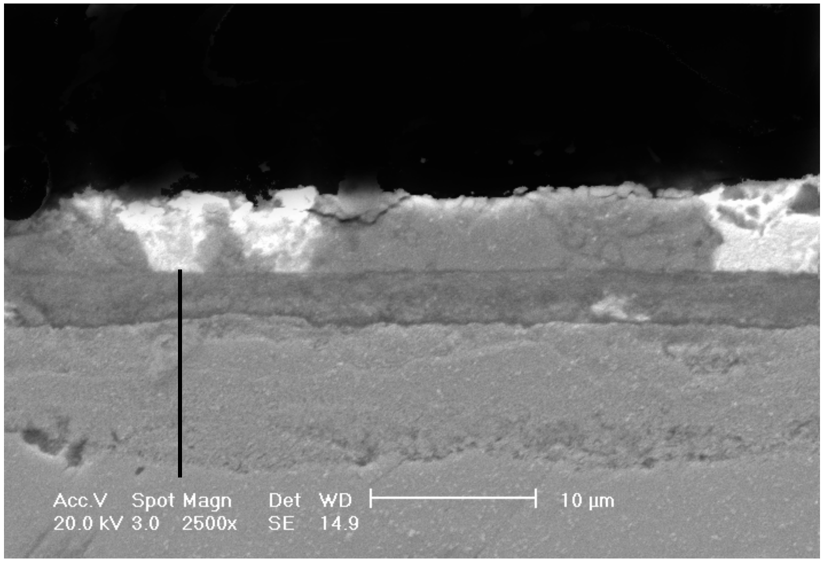


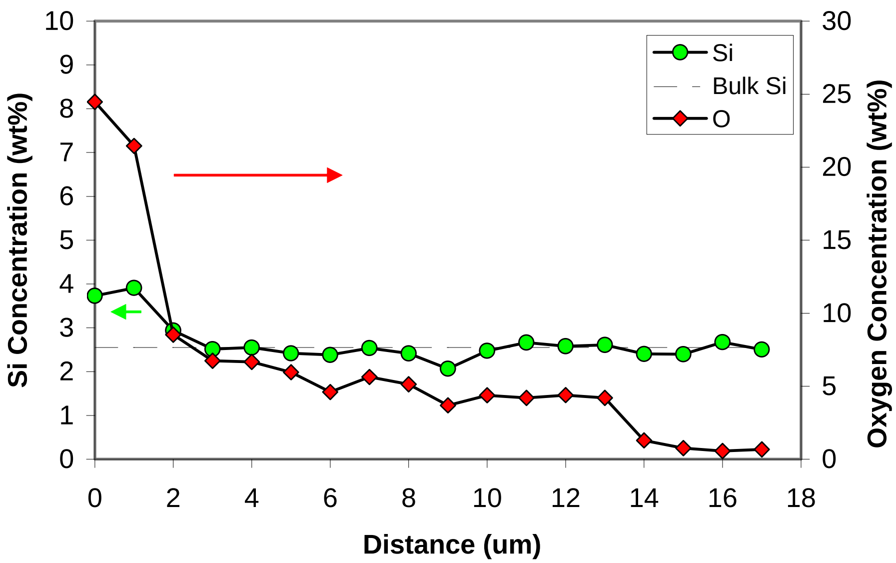
SEM Cross-Section of Fe after 600°C, 100 hr Static Test in Pb-Bi



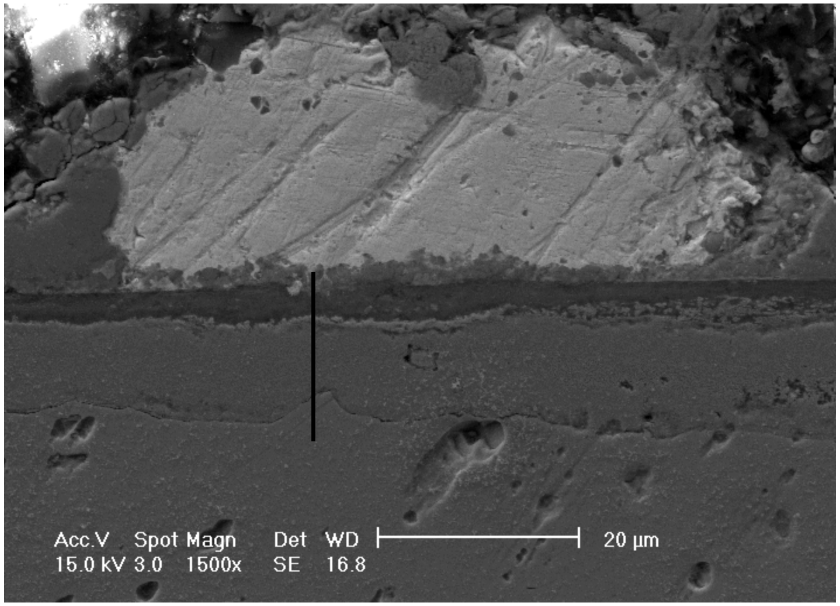


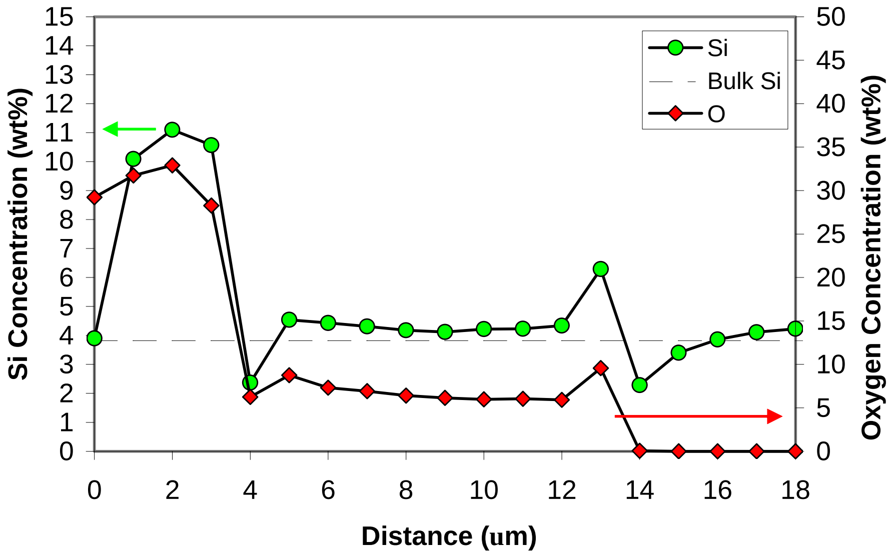
SEM Cross-Section of Fe-2.55wt% Si after 600°C, 100 hr Static Test in Pb-Bi



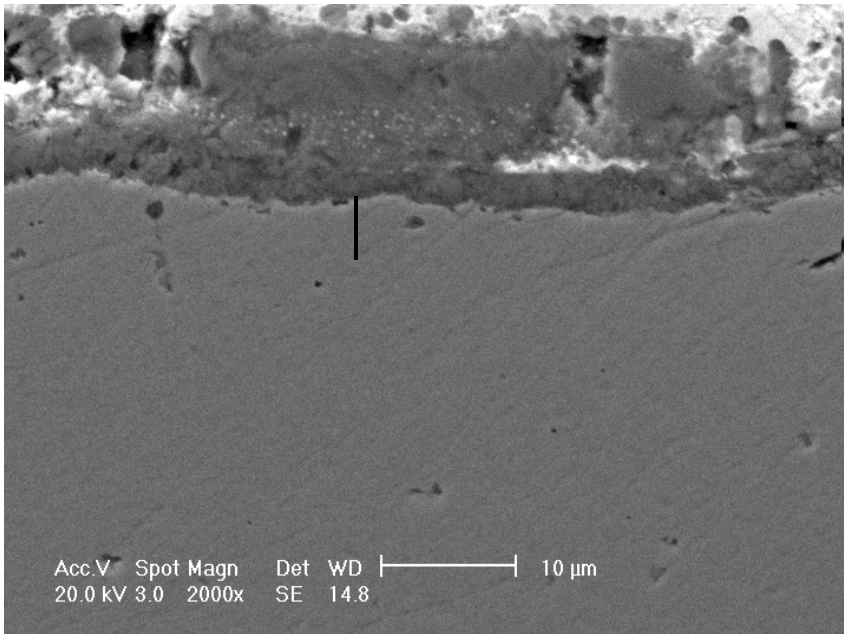


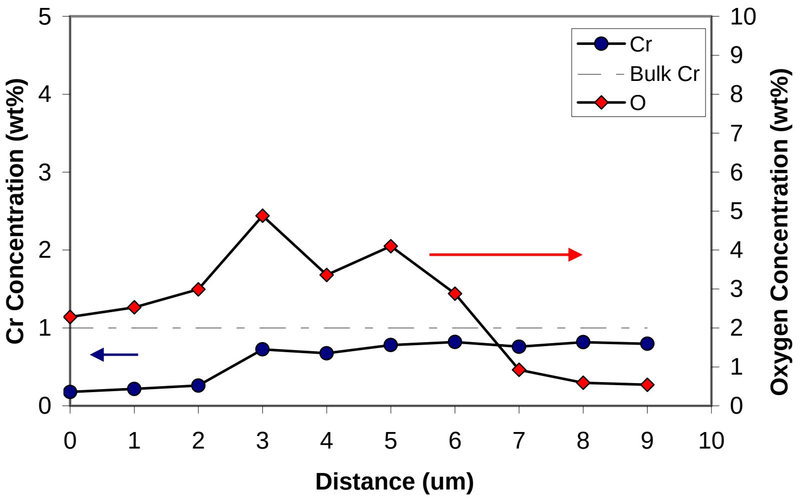
SEM Cross-Section of Fe-3.82wt% Si after 600°C, 100 hr Static Test in Pb-Bi



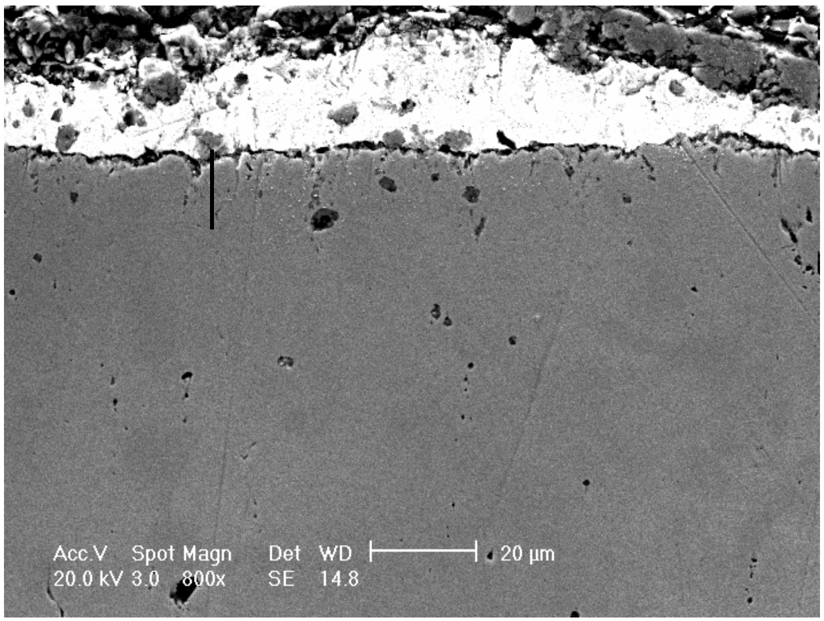


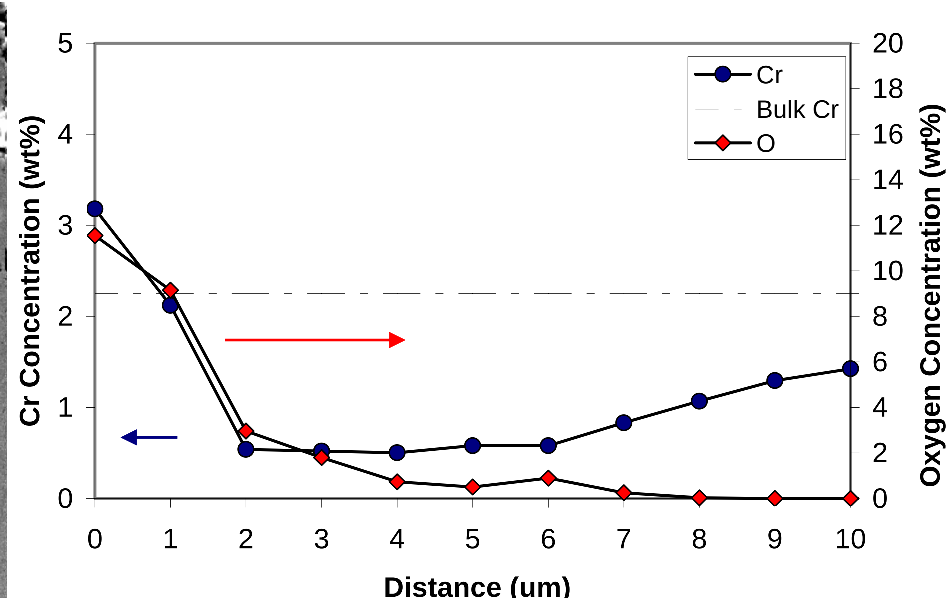
SEM Cross-Section of Fe-1wt% Cr after 600°C, 100 hr Static Test in Pb-Bi



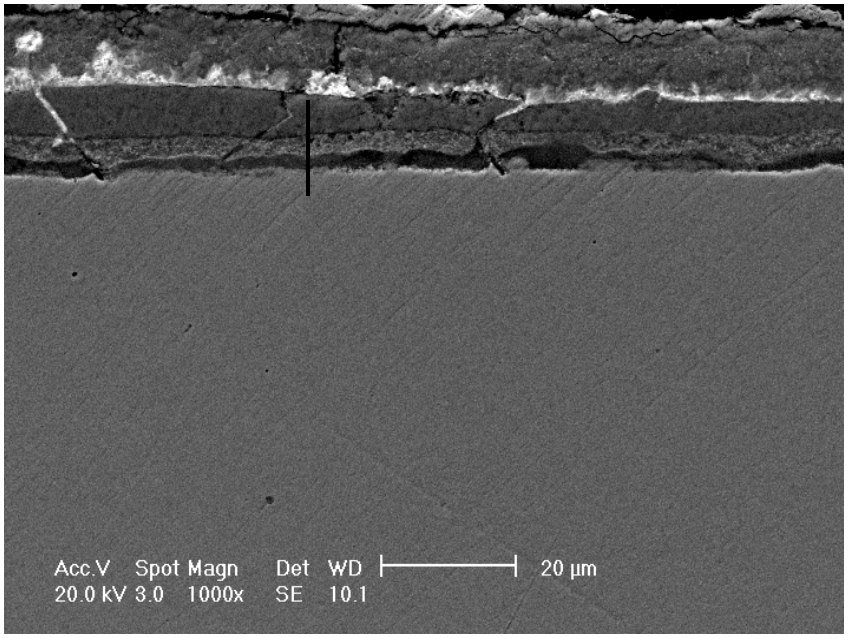


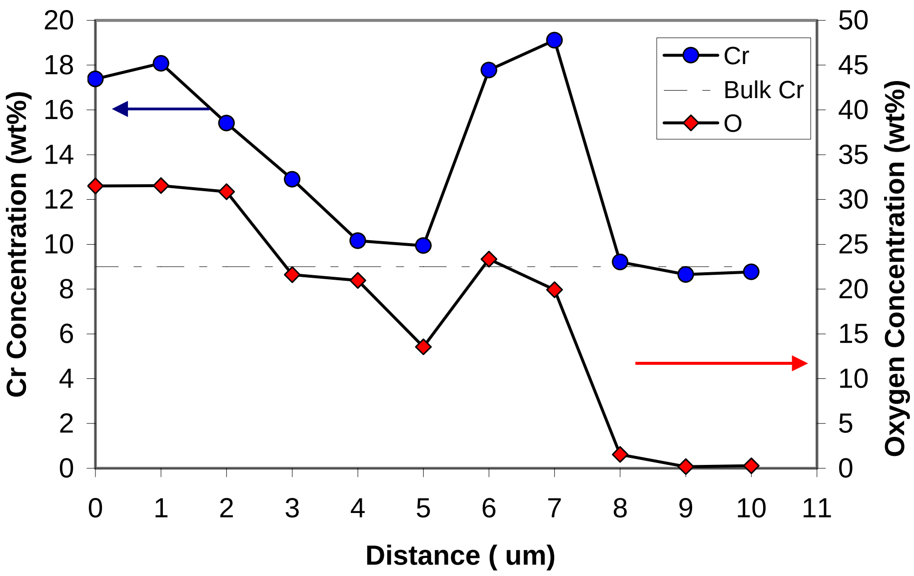
SEM Cross-Section of Fe-2.25wt% Cr after 600°C, 100 hr Static Test in Pb-Bi



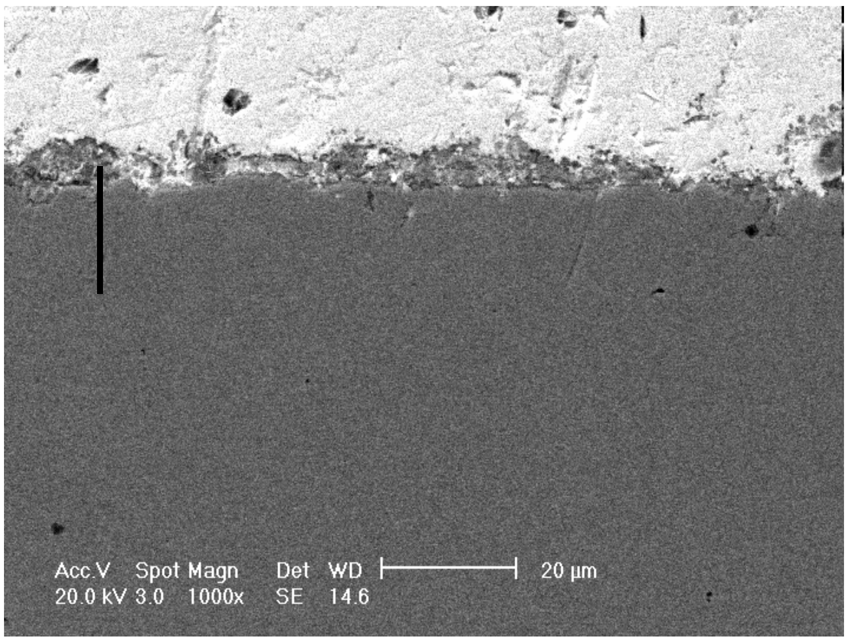


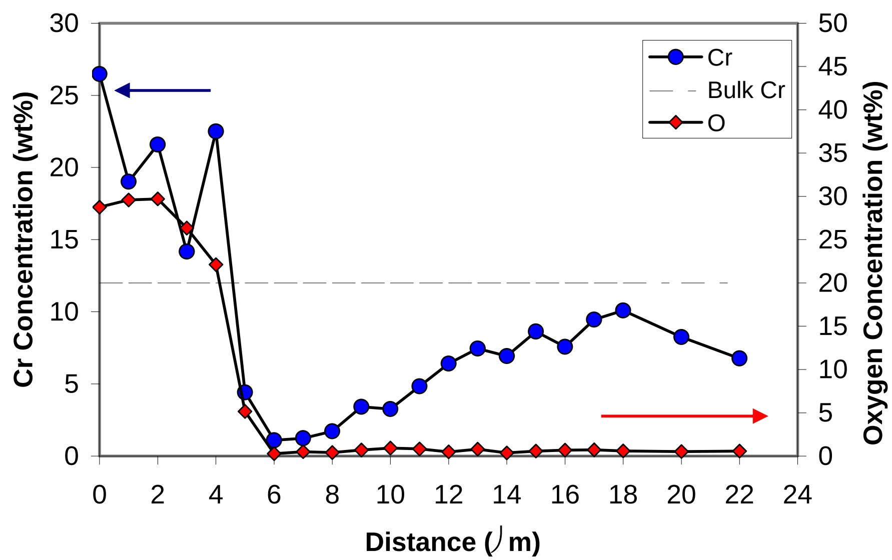
SEM Cross-Section of Fe-9wt% Cr after 600°C, 100 hr Static Test in Pb-Bi



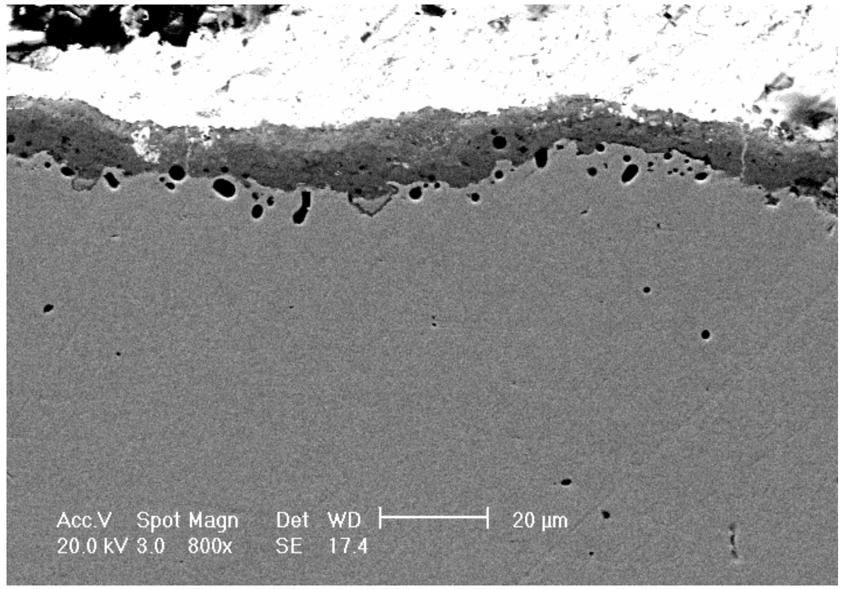


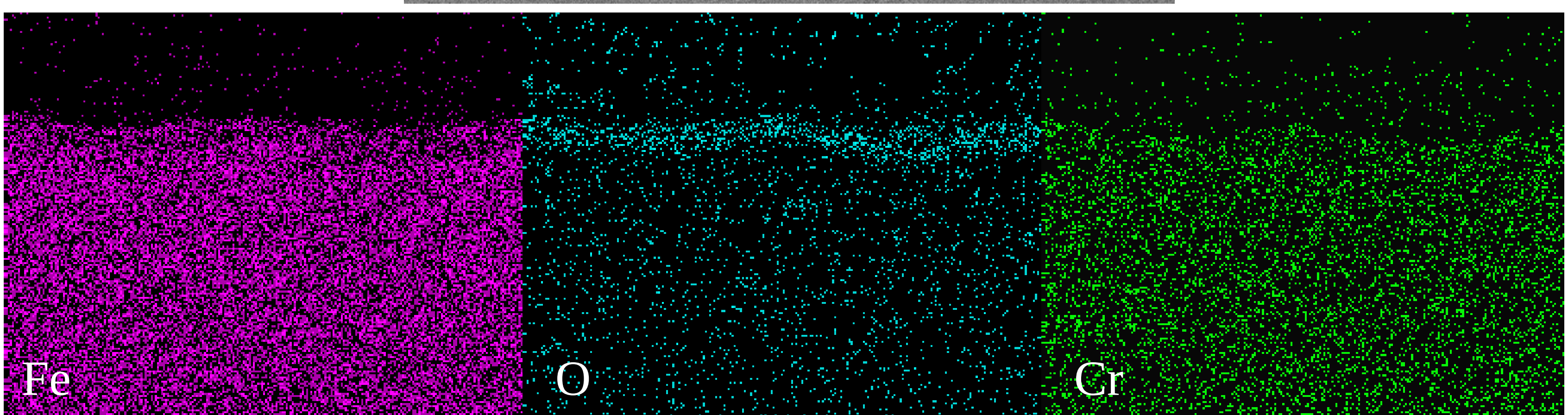
SEM Cross-Section of Fe-12wt% Cr after 600°C, 100 hr Static Test in Pb-Bi





SEM Cross-Section of Fe-12wt% Cr-0.5wt% Si after 600°C, 100 hr Static Test in Pb-Bi





Conclusions

- Selective Fe Dissolution Results in Enrichment of Si at the Surface
- The degree of enrichment is a function of bulk Si concentration.
- Si-Based Oxide Formation Results
- Increased Cr Concentration Results in a Change in Interaction Mode

Future Work

- Exposure of Alloys for Longer Times
- Fabrication of ODS Alloys (2)
  - One High Cr
  - One Low Cr
- Detailed Film Analysis and Modeling
- Very High Temperature Alloy Development