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IAEA Atoms for Peace and Development

Role of Research Reactors in Material Research and Development

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Research Reactor Section
International Atomic Energy Agency

November 2017

Outline of Presentation

- Nuclear Material Challenges
- Fuel Development
- Fuel and Material Experiment Examples
  - Advanced Gas Reactor Fuel Development
  - Neptunium Fueled Nuclear Data
  - Advanced Gas Reactor Graphite Compressive Tests
  - Mixed Oxide Fuel
  - Magnox Reactor Graphite Aging
  - Advanced Fuel Cycle Initiative
  - U-Mo Fuel Development
  - Irradiation Assisted Stress Corrosion Cracking

Reactor Material Challenges

- Difficult conditions inside operating reactor – high temperature, vibration, mechanical stress, coolant chemistry, and intense fields of high energy neutrons
- Current operating reactor material failures have enabled better material understanding, but it is important to understand weaknesses and material phenomena before failures occur
- Original LWR licenses were 40 years – most have applied for (and received) a 20 year extension
  - Need to demonstrate material service lifetimes
  - LWR vessel test coupon show preliminary effects

Need Testing in Material Test Reactors

PWR Component Materials

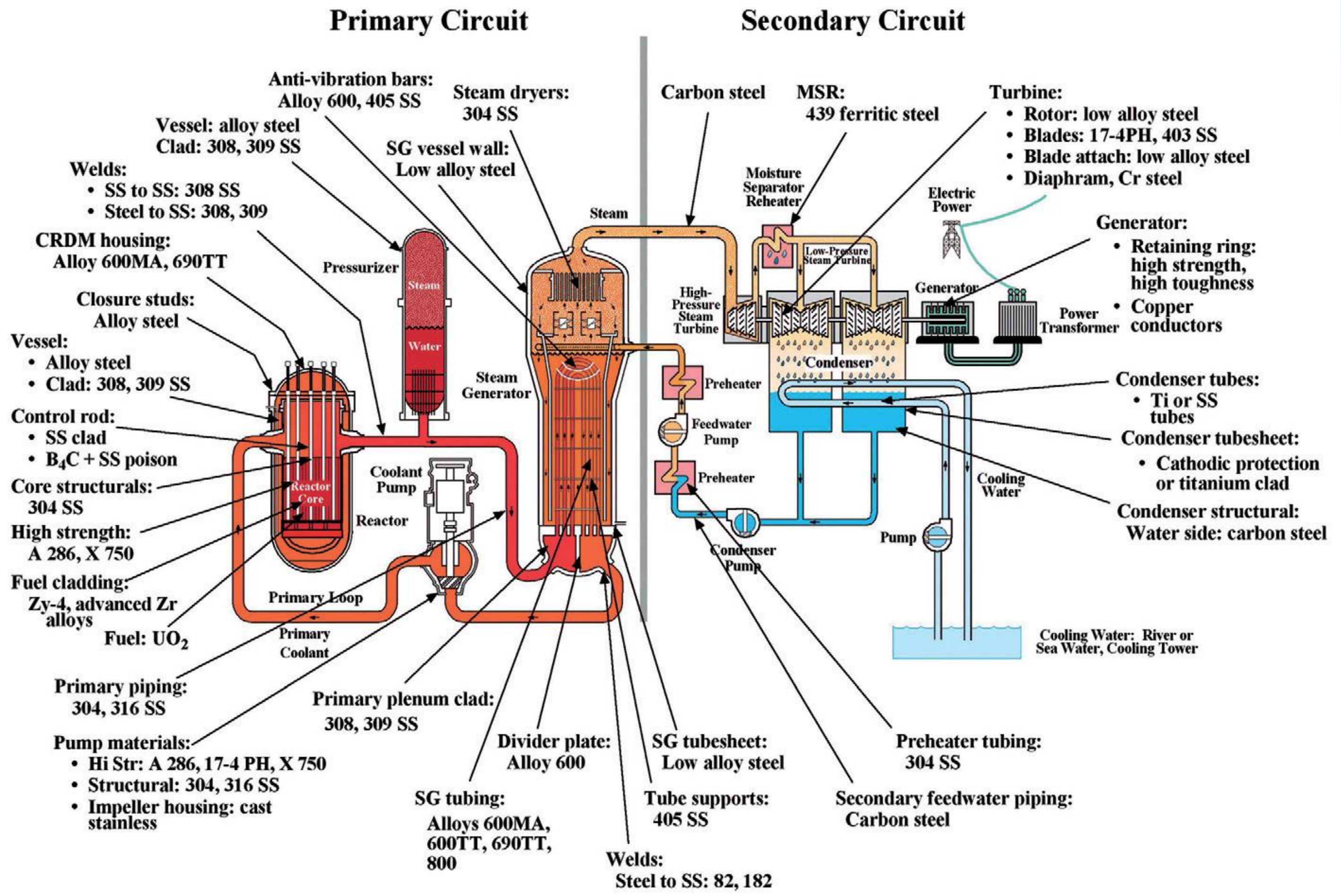


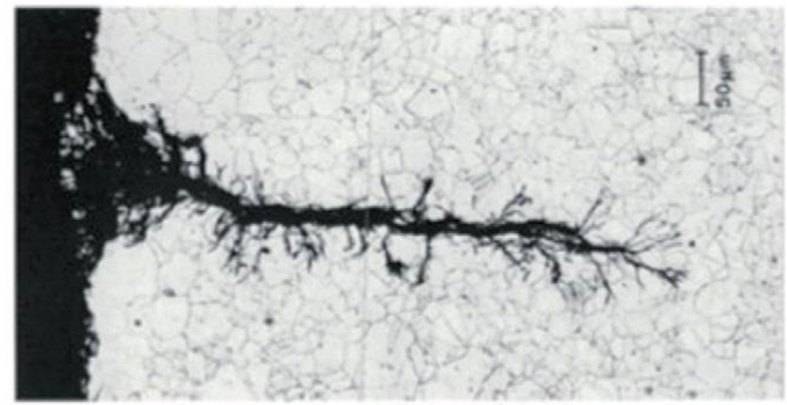
Figure originally by Roger Staehle. "Material Challenges for Nuclear Systems," Todd Allen, Jeremy Busby, Mitch Meyer, David Petti, Materials Today, December 2010, Volume 13, Number 12.

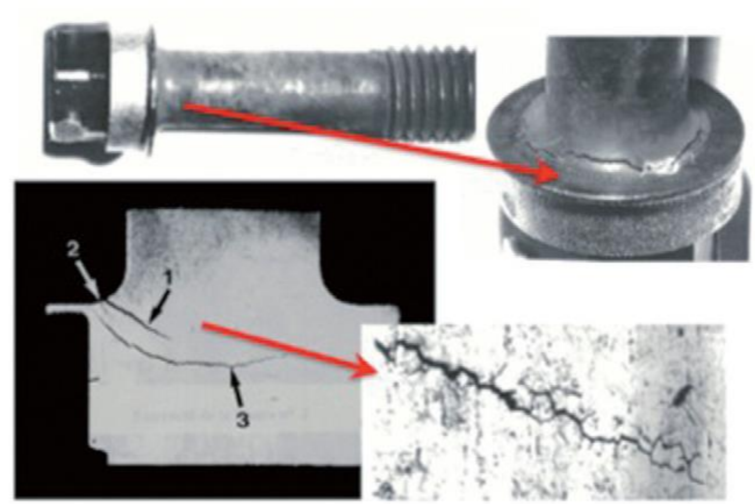
Reactor Material Degradation

Examples of stress corrosion cracking in a light water reactor

- Primary water stress corrosion cracking in a steam-generator tubing

- Irradiation assisted stress corrosion cracking in a pressurized water reactor baffle bolt





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Materials Degradation Phenomena in Austenitic Stainless Steel

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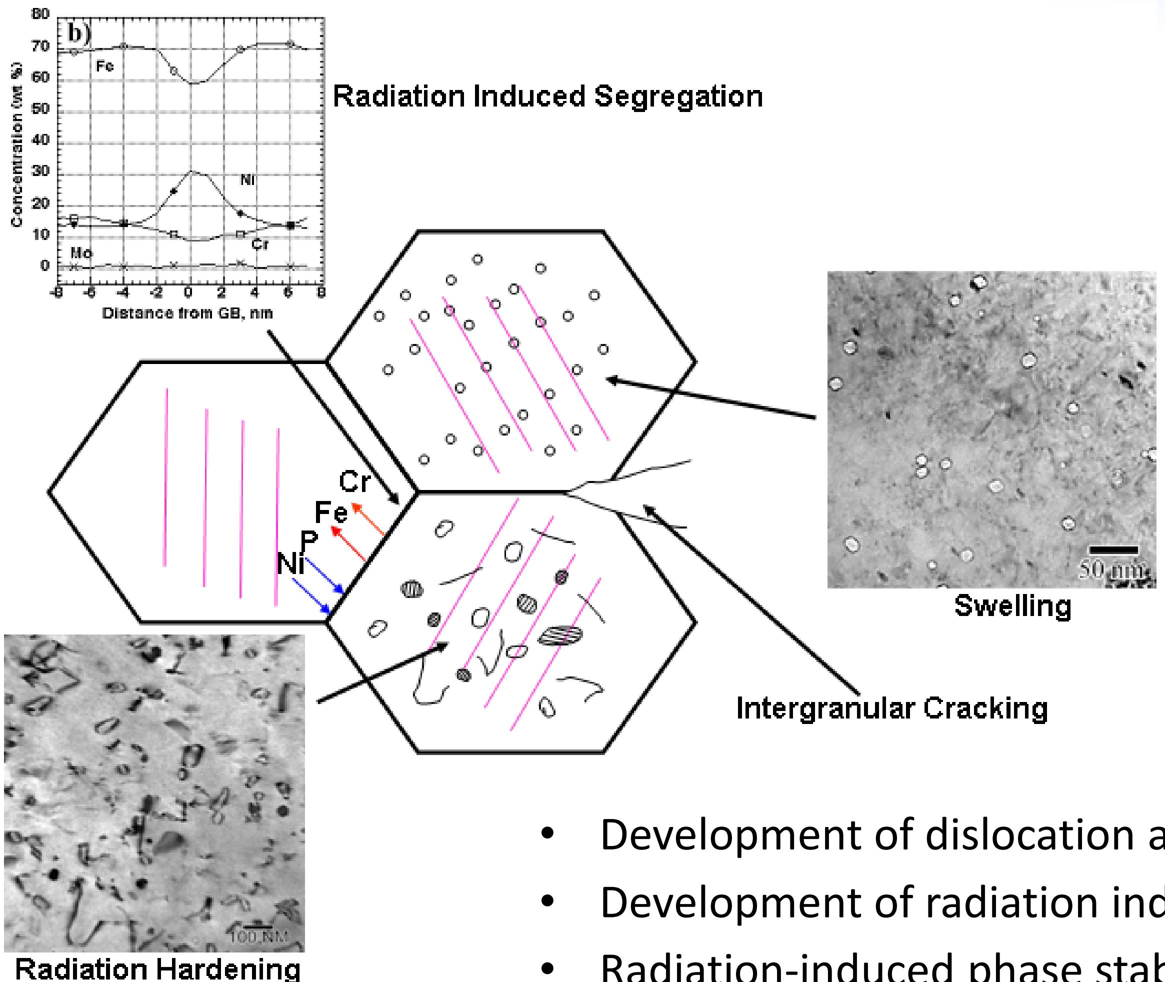
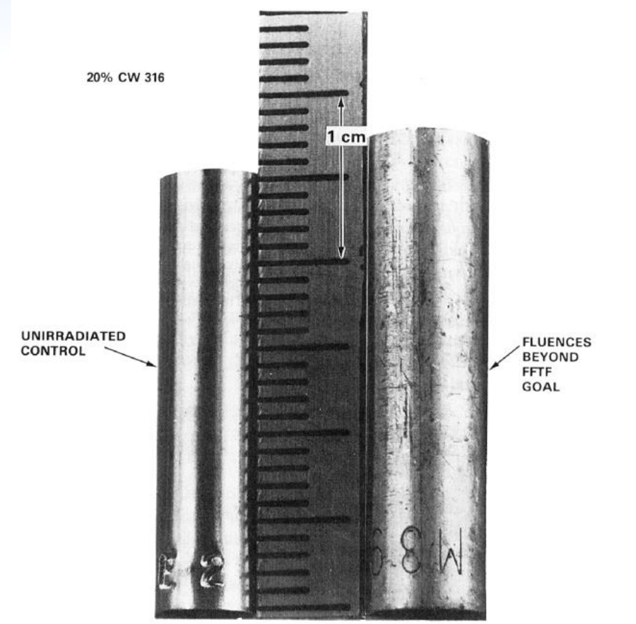




Straalsund, J.S., R.W. Powell, and B.A. Chin, Journal of Nuclear Materials, 1982. 108-109: p. 299-305.

- Development of dislocation and void structures
- Development of radiation induced segregation
- Radiation-induced phase stability
- Radiation-induced dimensional changes

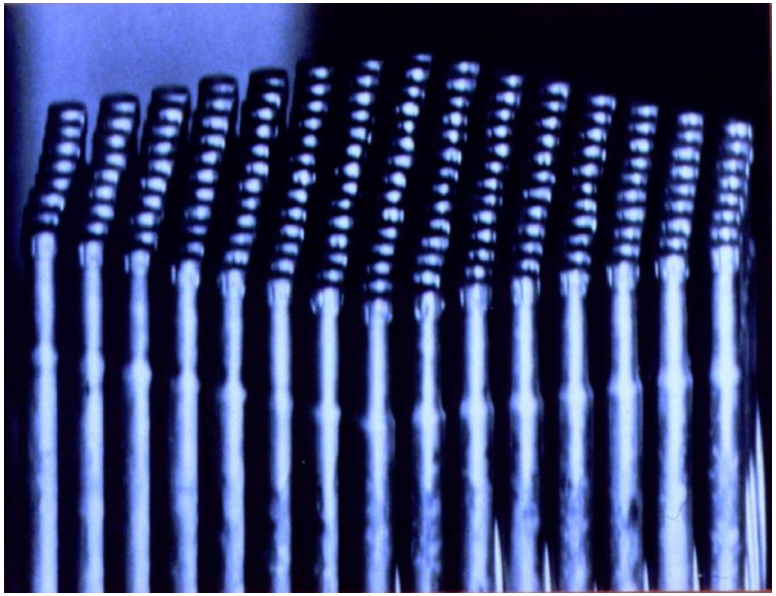
Nuclear Fuel Life Challenges

- Trend is for higher fuel burnup – longer time in the reactor, subject to more fissions.
- Fuel failures typically due to clad failures rather than actual fuel failures
- BUT the operating impact is similar – more frequent shutdowns for fuel changes, leading to less economical situation
- Search is underway for accident tolerant fuel, with new, more durable, cladding materials
- Deployment of new fuel required an extensive qualification program, beginning with irradiation testing

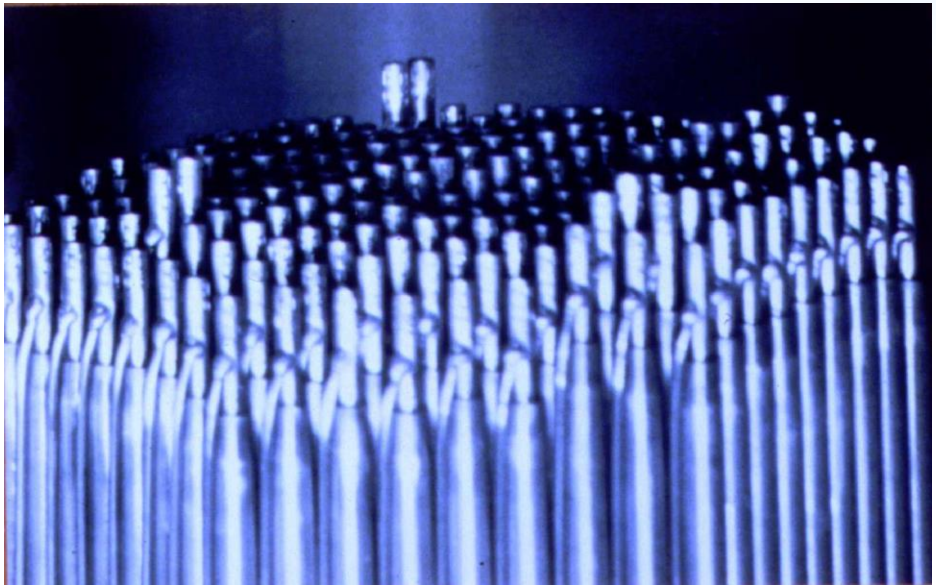
Need Testing in Material Test Reactors

Macroscopic Effects of Void Formation

FFTF Fuel Pin Bundles



HT-9, no swelling



316-Ti stainless, swelling

Future Reactor Conditions

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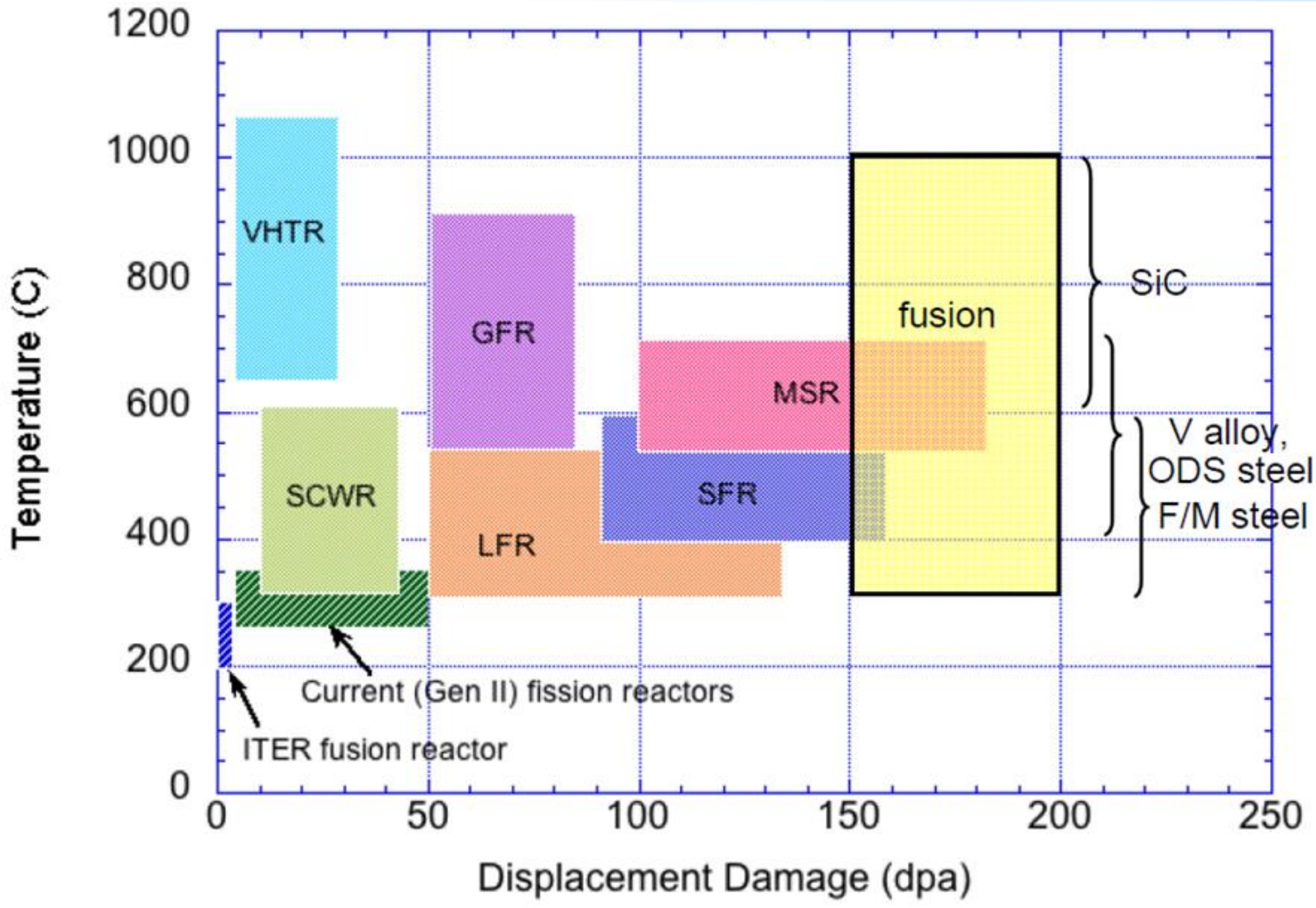
Approximate operating environments for Gen IV systems

Reactor Type	Coolant Inlet Temp (°C)	Coolant Outlet Temp (°C)	Maximum Dose (dpa*)	Pressure (MPa)	Coolant
Supercritical Water-cooled Reactor (SCWR)	290	500	15-67	25	Water
Very High Tmpearature Gas-cooled Reactor (VHTR)	600	1000	1-10	7	Helium
Sodium-cooled Fast Reactor (SFR)	370	550	200	0.1	Sodium
Lead-cooled Fast Reactor (LFR)	600	800	200	0.1	Lead
Gas-cooled Fast Reactor (GFR)	450	850	200	7	Helium/SC CO_2
Molten Salt Reactor (MSR)	700	1000	200	0.1	Molten Salt
Pressurized Water Reactor (PWR)	290	320	100	16	Water

* dpa is displacement per atom and refers to a unit that radiation material scientists used to normalize radiation damage across different reactor types.

For one dpa, on average each atom has been knocked out of its lattice site once.

Future Reactor Material Service



S.J. Zinkle, 2007

Material Testing and Qualification

- Instrument development, testing, calibration, qualification
- Fuel/material testing (ageing, corrosion, irradiation)
- Fuel/material qualification (temperature, pressure, irradiation)
- Development of new fuels/materials (actinide fuels, high temperature reactors, fast reactors, fusion reactors, …)
- Phenomena Studied
  - Swelling
  - Void Formation
  - Grain boundary Effects
  - Embrittlement

He bubbles on grain boundaries can cause severe embrittlement at high temperatures (S. Zinkle, ORNL)

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Radiation Damage Research Methods

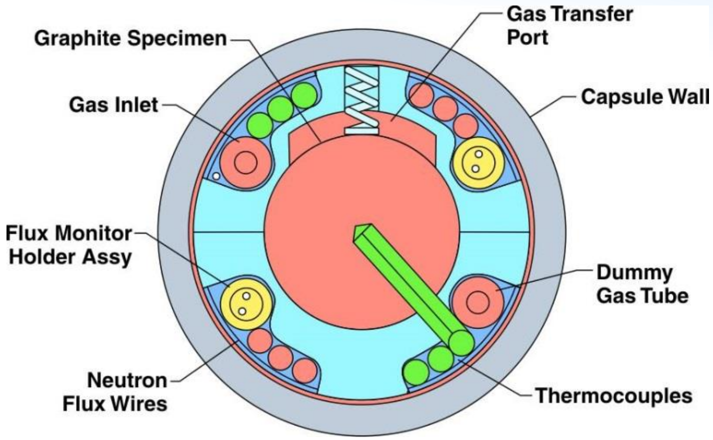
- Experimentally
  - Materials Characterization
    - Electron Microscopy
    - Field-ion microscopy and Atom Probe Tomography
    - X-Ray, Neutron Diffraction
    - Synchrotron Light Sources
  - Mechanical Properties
    - Tensile, Fracture Mechanics, Creep
    - SCC tests in autoclave systems

- Modeling
  - Atomistic
  - Molecular dynamics
  - Kinetic Monte Carlo
  - Diffusion and Rate Theory
  - Empirically developed models

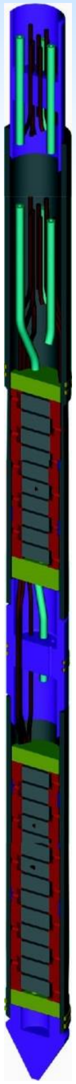
Because of the complex nature of radiation damage in materials our understanding is continually evolving

Magnox Graphite Irradiation

- Experiment Purpose - Extend data base on Magnox graphites for life extension support for UK Magnox power stations
- On-line temperature indication and control
- Two equal size capsules - one oxidizing & one inert, mirror images about ATR core centerline
- Inert Capsule
  - 99.996% pure helium (< 1 ppm O_2)



Capsule Cross Section

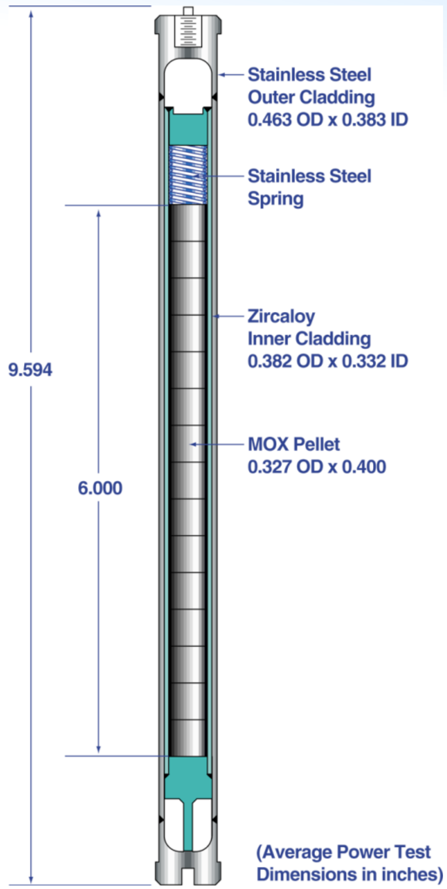


Vertical Section

Mixed Oxide (MOX) Fuel Irradiation

Purpose of the experiment was to obtain Mixed Oxide Fuel (MOX) fuel and cladding irradiation performance data on fuel pins made with weapons grade plutonium downblended with low enriched uranium

- PWR temperature at surface of fuel pin cladding
- Linear heat rate requirements
  - 6 KW/ft minimum
  - 10 KW/ft maximum
- Fuel burn-up levels
  - 8 GWd/t minimum
  - 50 GWd/t maximum
- Maintain orientation of irradiation basket in relation to reactor core center
- Maintain orientation of fuel pins relative to reactor core center



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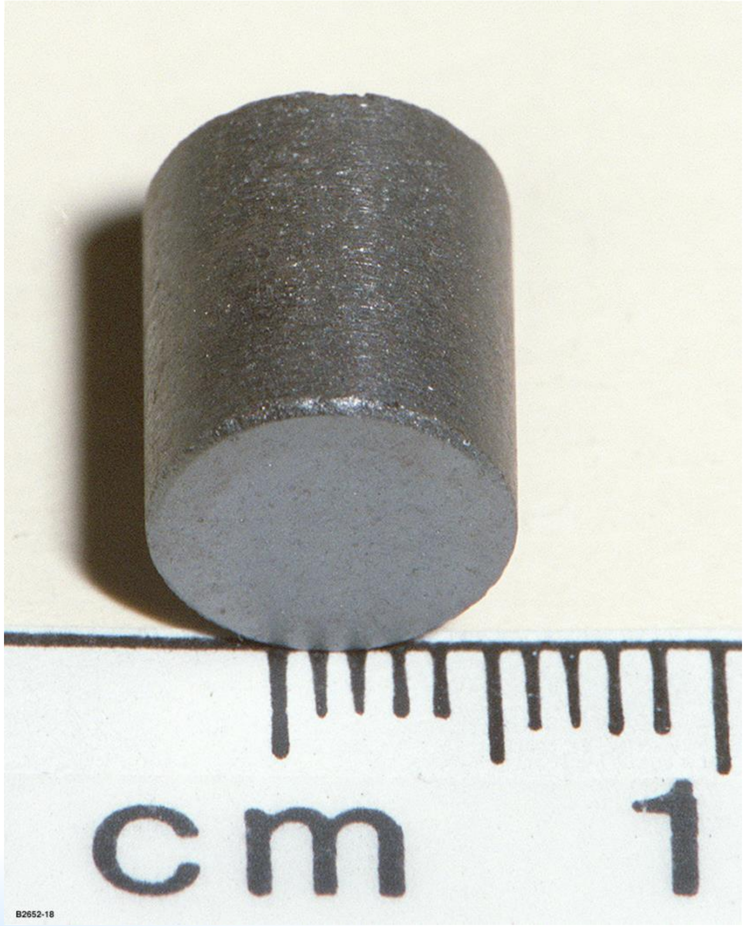
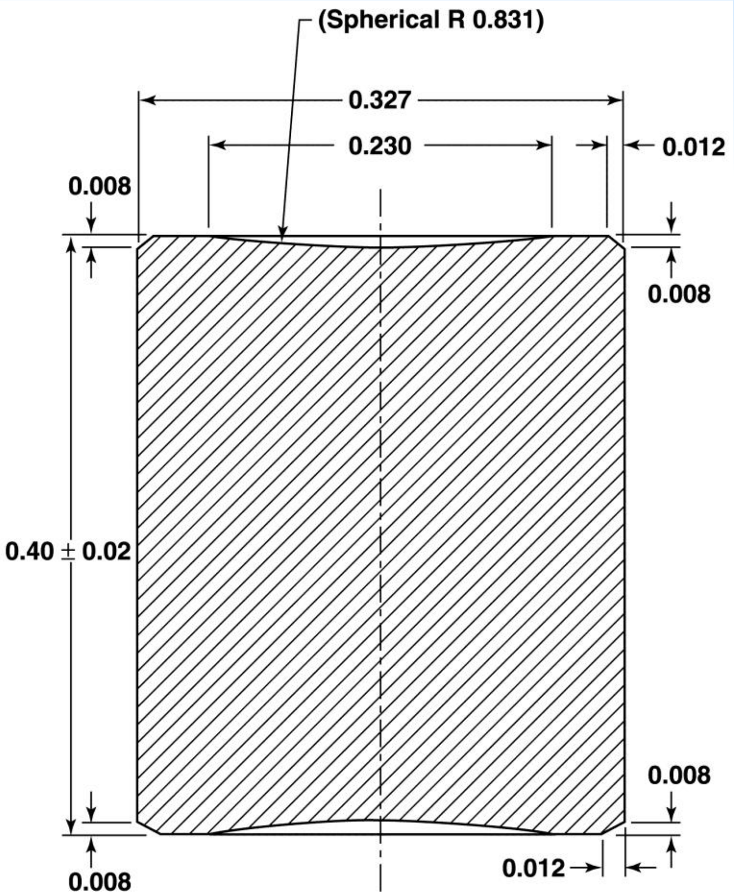
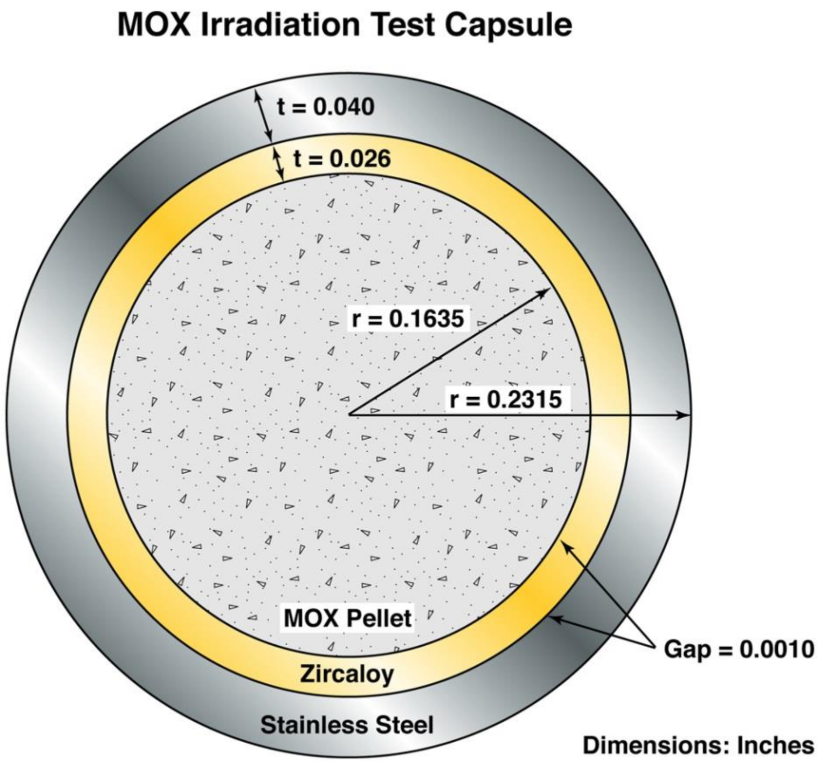
MOX Test Fuel Pellets





Test Fuel Employed Typical
PWR Pellet Dimensions with Normal
Dish and Chamfer

MOX Fuel Capsule Cross Section

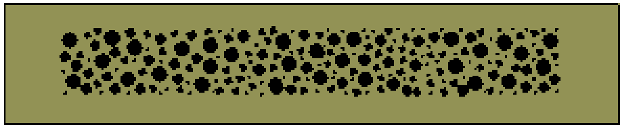


- Capsule designed to ASME Section III Class 1 requirements
- Small (0.025 mm) insulating gas gap between fuel pin and capsule provided desired temperatures
- Zircaloy fuel pin outer surface protected from
  - Corrosion
  - Hydrogen pickup (hydrides)

Results of irradiation testing were satisfactory. Led to lead test assemblies being fabricated and irradiated in commercial PWRs, also with satisfactory results. Not in full scale production yet.

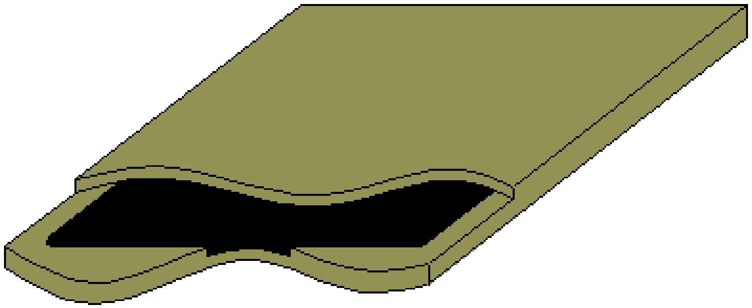
U-Mo Fuel Testing

- Dispersion fuel: consists of fuel alloy powder in an aluminum matrix clad with aluminum
- Monolithic fuel: contains a single fuel foil in place of the dispersion of fuel particles
- Highest possible uranium loading
  - 15.3 g-U/cm^3 with U-10Mo
  - 16.3 g-U/cm^3 with U-7Mo
- Smaller surface area for reaction with aluminum
- Fuel aluminum interface is in the cooler region of the fuel zone



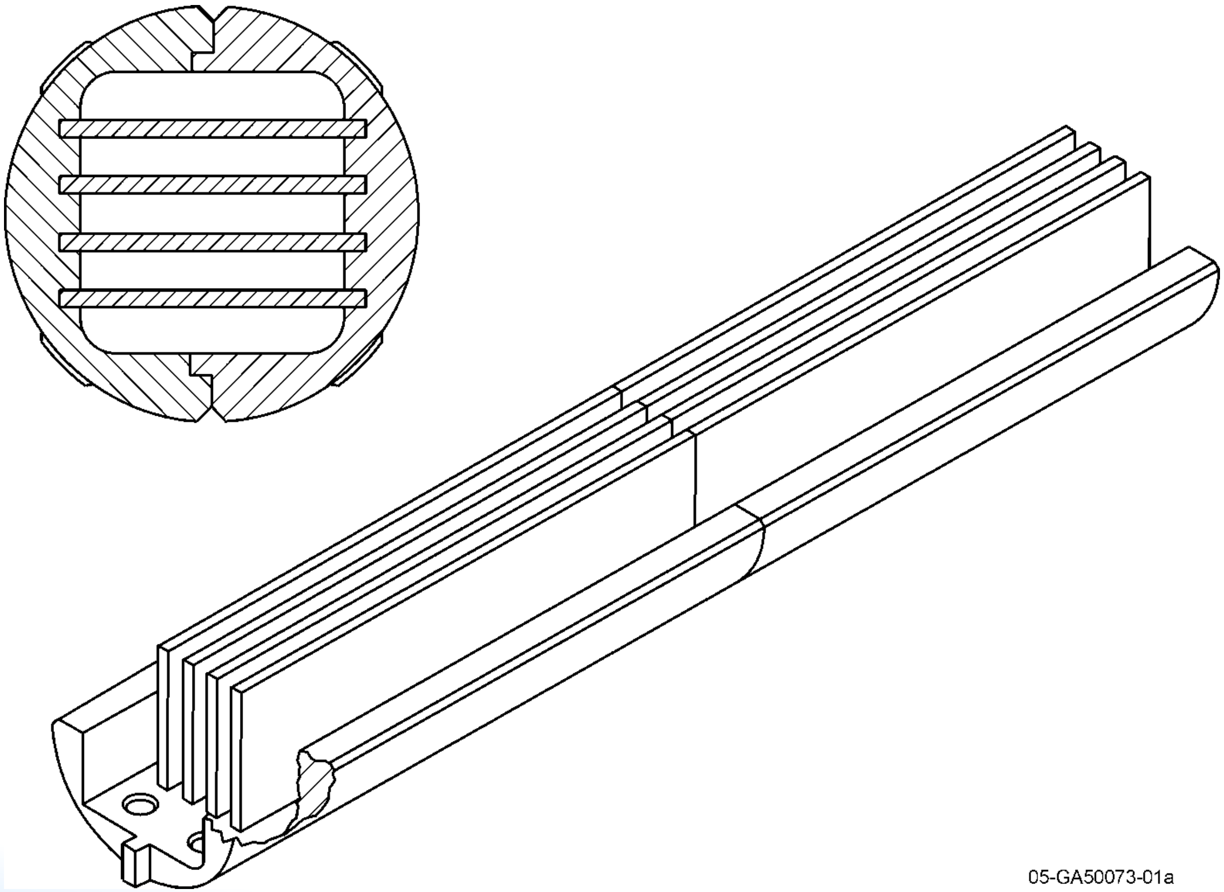
Dispersion Fuel

Monolithic Fuel

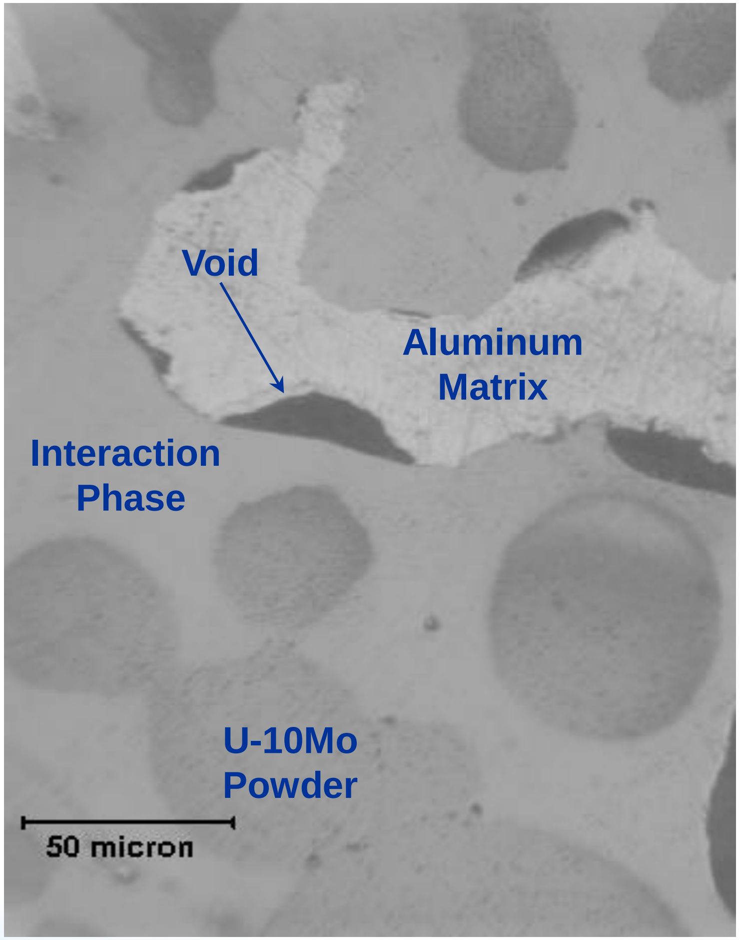


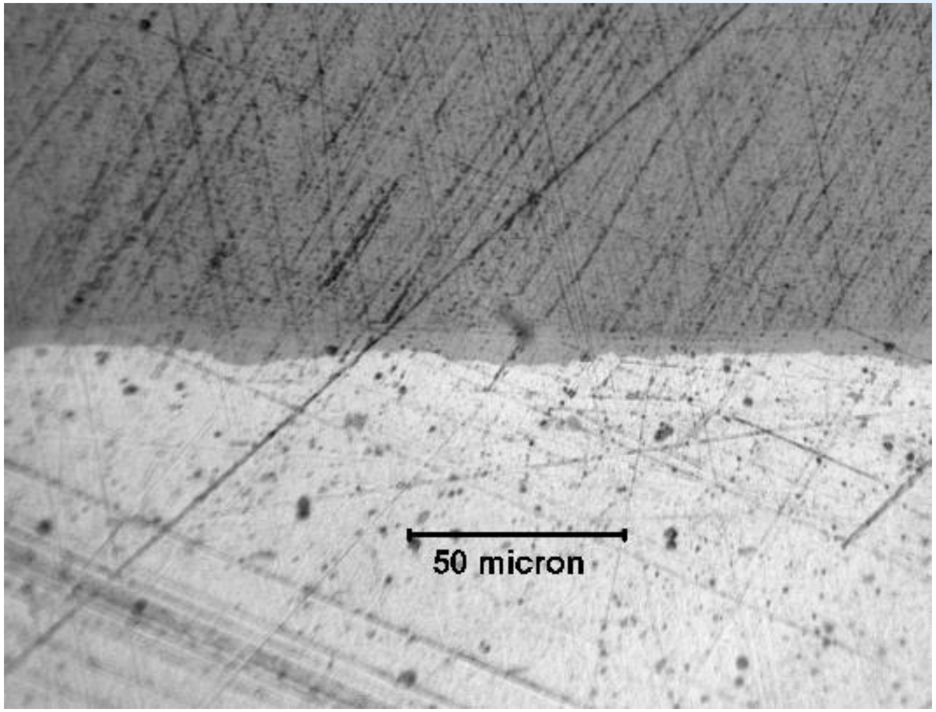
Monolithic Fuel Assembly Cutaway

Mini-plate (Eight-Plate) Capsule Configuration



U-Mo Irradiation Performance Comparison





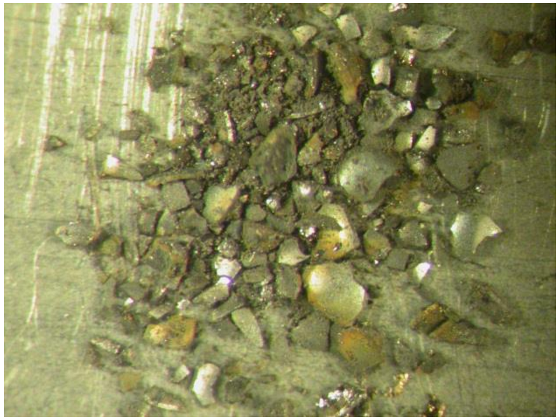
RERTR-4 Monolithic Plate

RERTR-4 Atomized Dispersion Plate

Fuel Tests in the BIGR Reactor

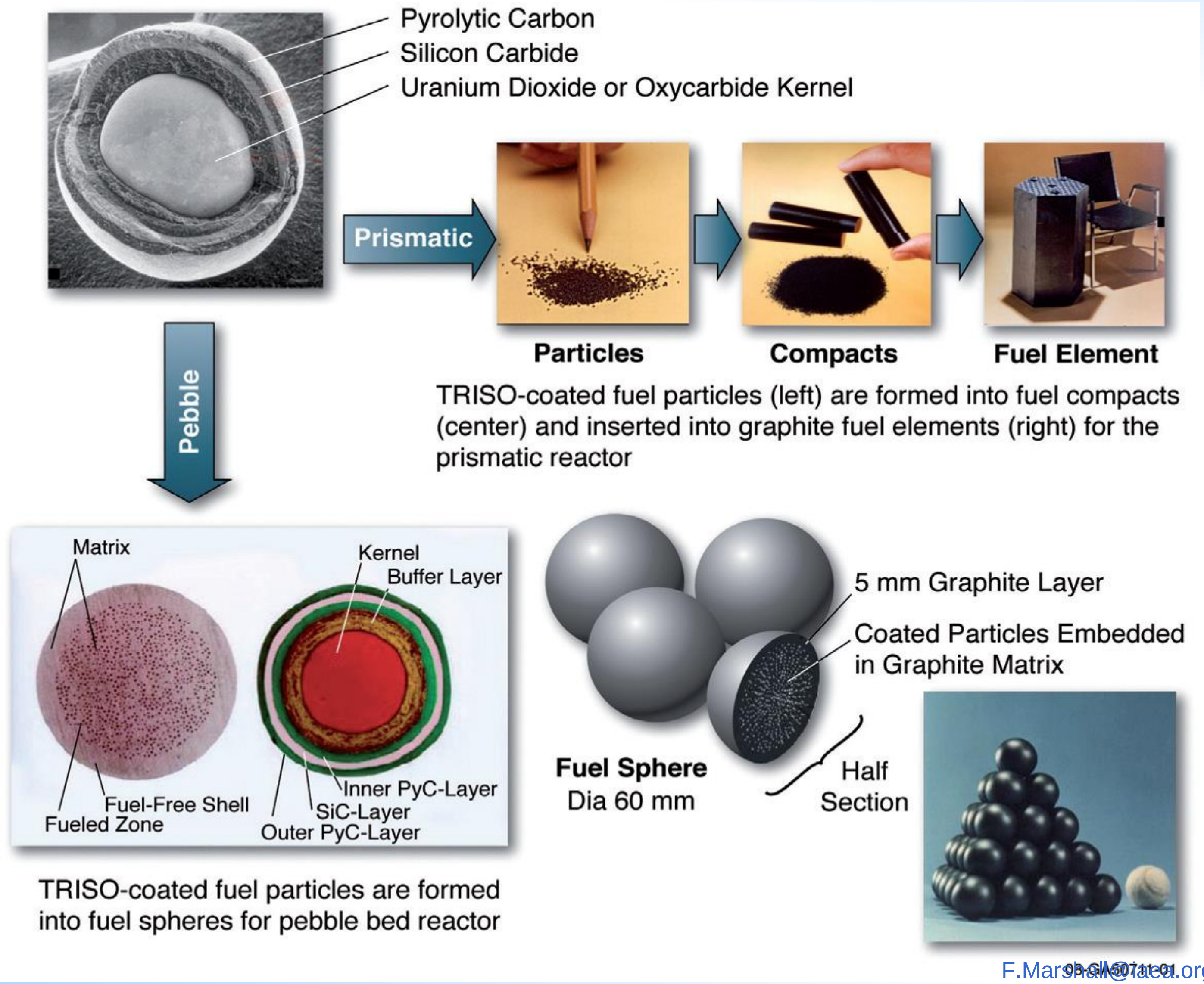
Fast Pulse Graphite
Reactor, Russian Federation

WWER Fuel Samples after irradiation in BIGR reactor



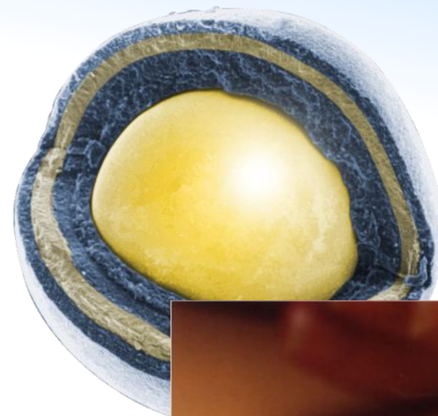
Coated particle fuel before and after irradiation in BIGR

High Temperature Gas Reactor Fuel



AGR Fuel Development Program

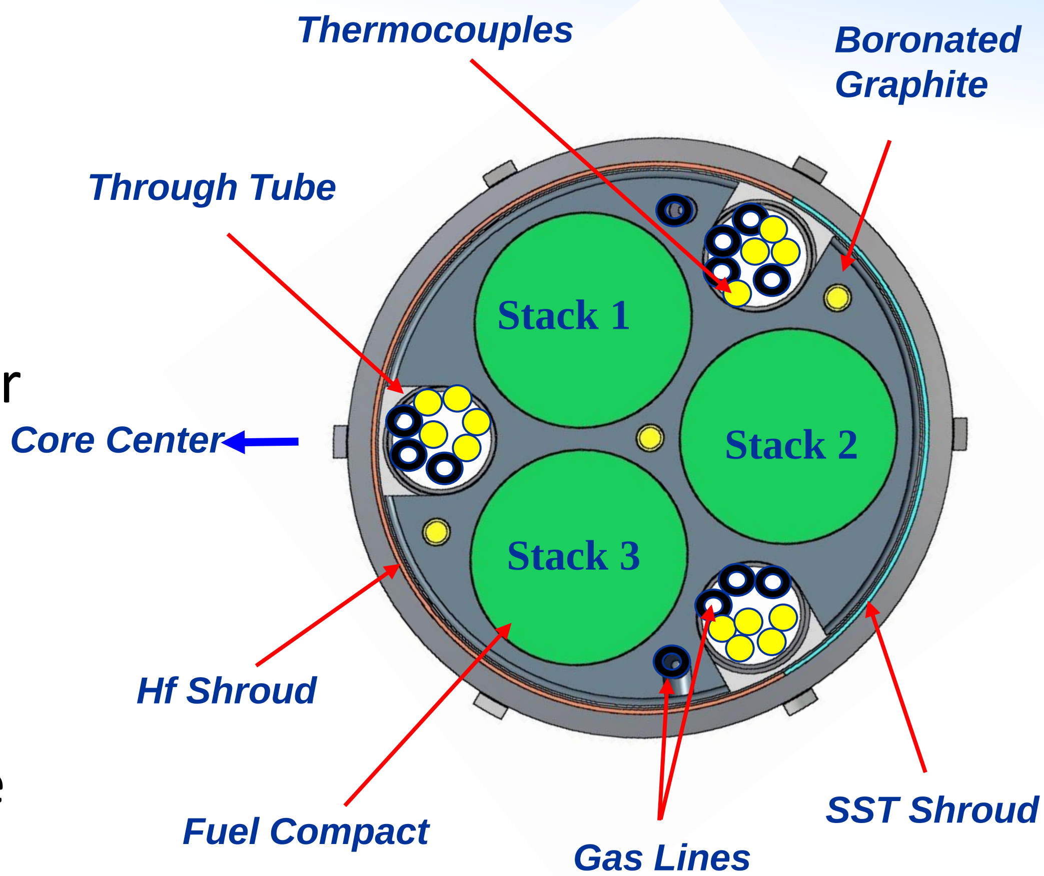
- Objective - support development of next generation Very High Temperature Reactors - near term for the Next Generation Nuclear Plant
  - Provide irradiation performance data to support fuel process development
  - Support development & validation of fuel performance & fission product transport models and codes
  - Provide irradiated fuel & materials for post irradiation examination & safety testing
- Purposes of AGR-1 Experiment are:
  - Shakedown of test design prior to fuel qualification tests
  - Irradiate early fuel from laboratory scale processes
- TRISO-coated, Uranium Oxycarbide (UCO)
- Low Enriched Uranium (LEU), <20% enrichment



Fuel Particles

AGR-1 Capsule Design Features

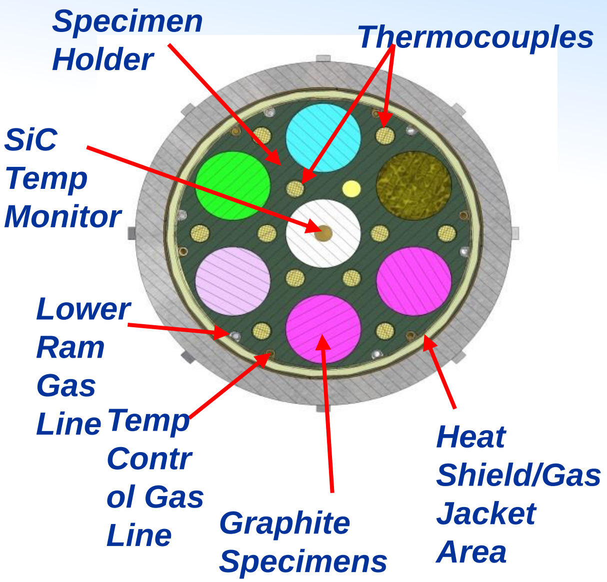
- Fuel Stacks
  - 3 fuel compacts/level
  - 4 levels/capsule
  - Total of 12 fuel compacts/capsule
  - Surrounded by nuclear grade graphite
- Through Tubes
  - Provide pathway for gas lines & TC's between capsules
  - Maintain temperature control gas jacket



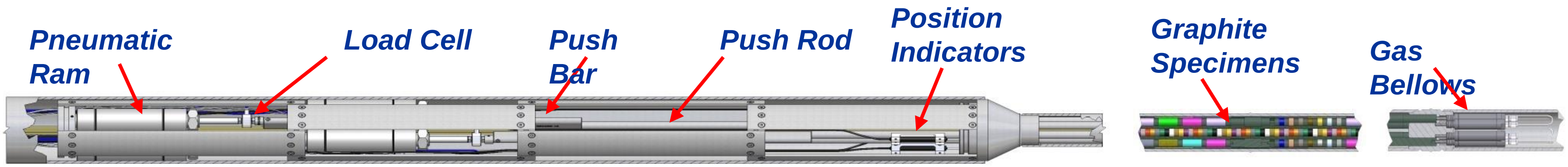
AGR-1 Capsule Cross Section

AGC-1 Compressive Load System

- Nuclear grade graphites used in previous gas reactors unavailable due to loss of feedstock
- Experiments will be conducted at:
  - 600, 900, and 1200ºC
  - 4 to 7 dpa fast neutron damage levels (5.5 and 9.6 x 10^{21} n/cm^2 for E > 0.1 MeV)
  - Compressive loads of 2 to 3 ksi (14 to 21 MPa)
- 6 Pneumatic rams above core to provide compressive load on specimens in peripheral stacks during reactor operation
- >500 individual specimens in the test capsule



AGC-1 Capsule Cross Section



AGC-1 Test Train

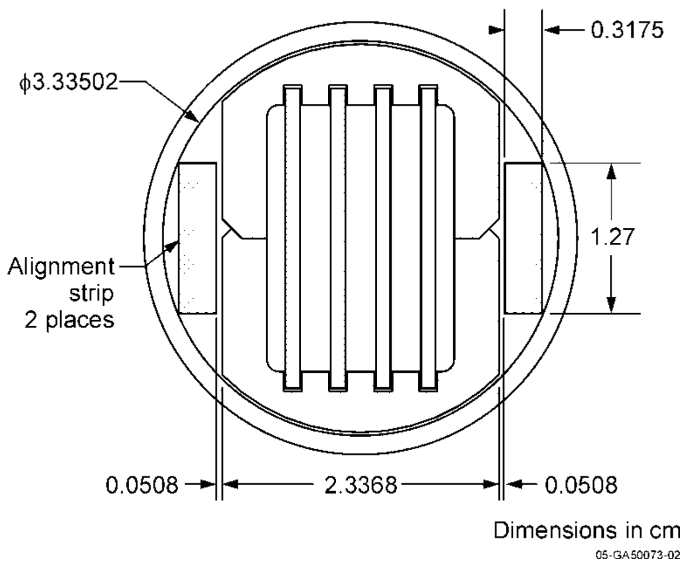
Protected Plutonium Production (PPP) Experiment Objective

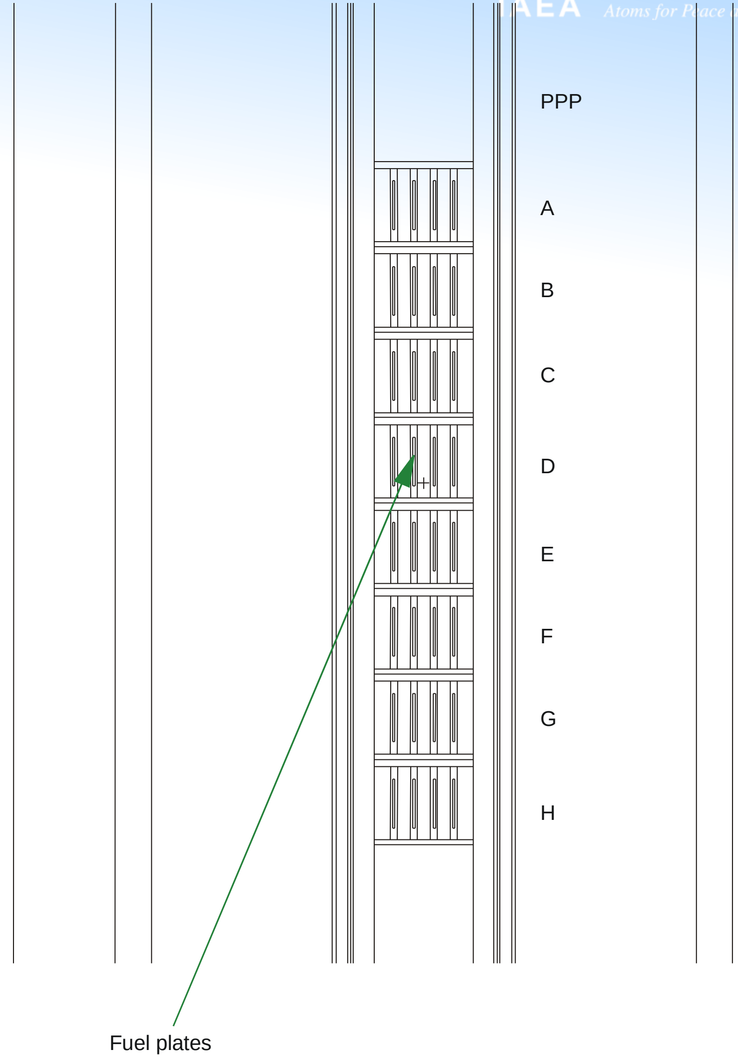
- Determine the accuracy of neptunium-237 and plutonium-238 cross sections by conducting integral data measurements - irradiation in ATR, followed by radiochemical analysis
- Accurate cross sections are needed to analyze new fuel cycles containing neptunium in light water reactor fuel (with protected plutonium production)
- Evaluate the amount of Pu-238 generated to investigate the option of "seeding" power reactor fuel with Np-237 to generate sufficient Pu-238 with the Pu-239 - proliferation-resistant spent fuel

PPP Experiment Description

- 2 rows of 4 plates (fuel and 'dummy')/capsule
- Four capsules/basket - total of 32 plates
- 22 mini fuel plates - aluminum cladding
- Capsule anti-rotation in basket is included in design
- Plates to be cooled by primary coolant
- Flux wires will be added inside 'dummy' mini-plates
- Capsules to be repositioned between 100, 200, and 300 EFPD Burnup phases

Schematic View of PPP Experiment Capsule



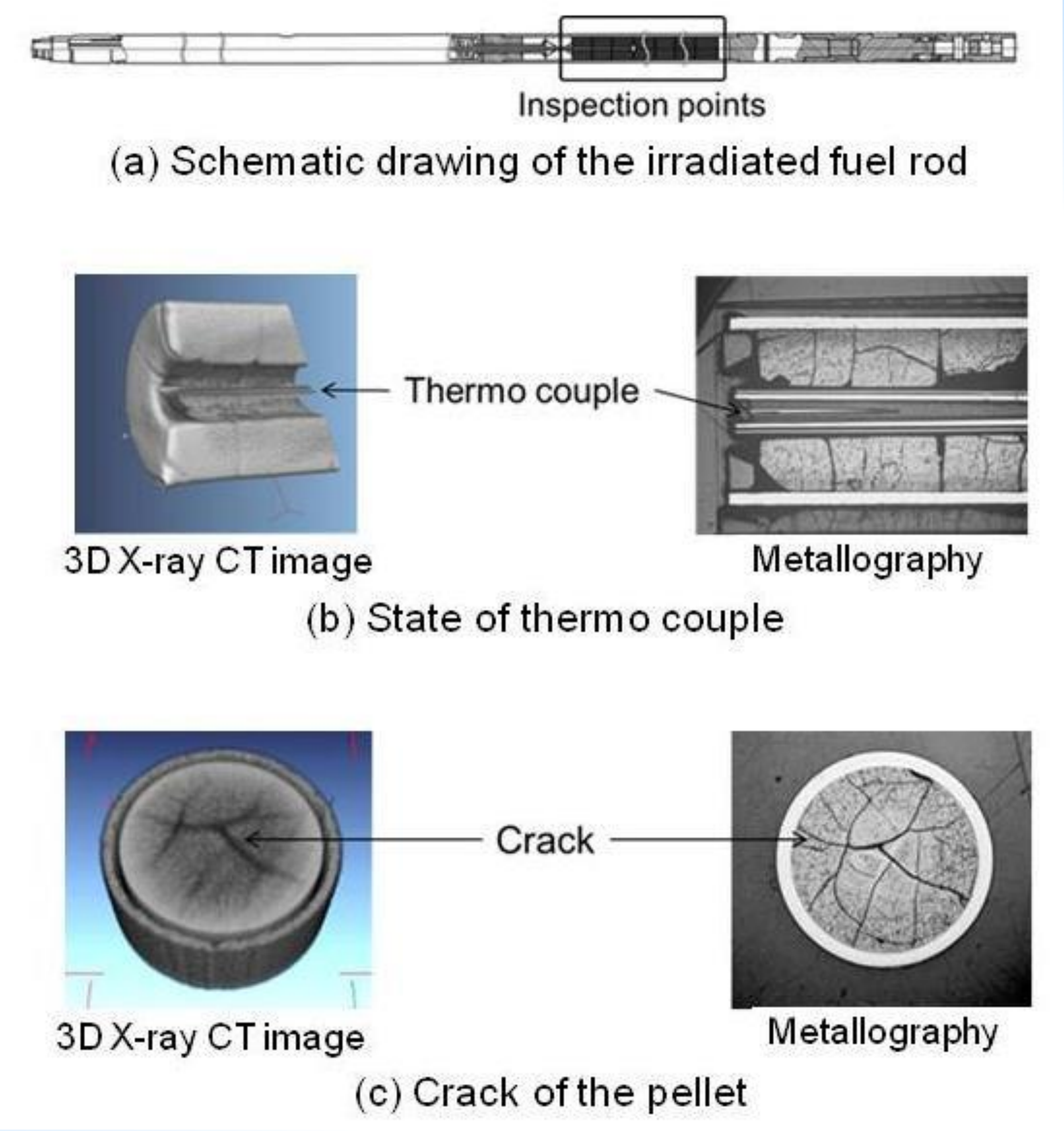


Mini-Plate Fabrication

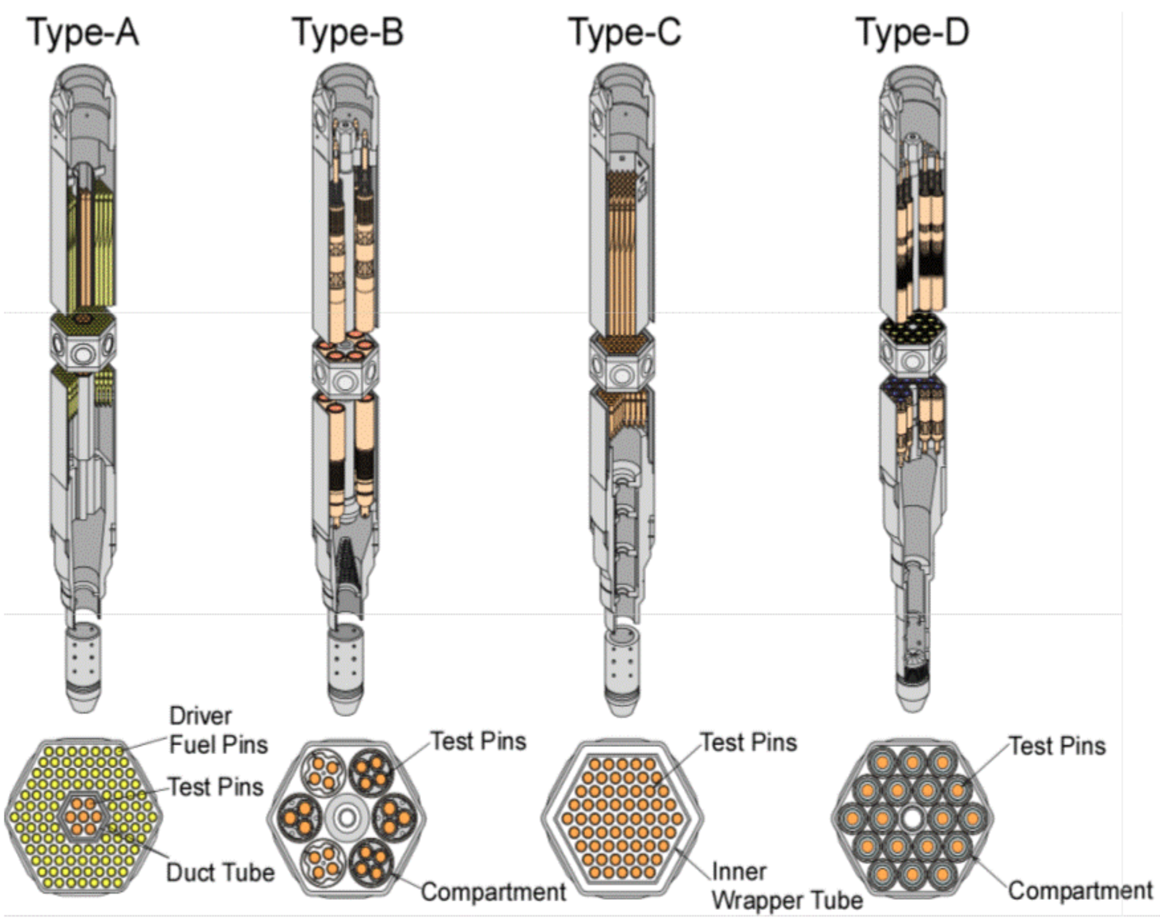
- Powder compacts are welded into picture frame assembly (top)
- Assembly is hot-rolled to thickness (middle)
  - Assembly is heated to 485 C and rolled in six passes
  - Final thickness 1.4 mm
- Plates are sheared into final size (bottom)

Fuel Tests in JMTR

- Resolution performance test in JMTR, Japan



Fuel Tests in JOYO, Japan



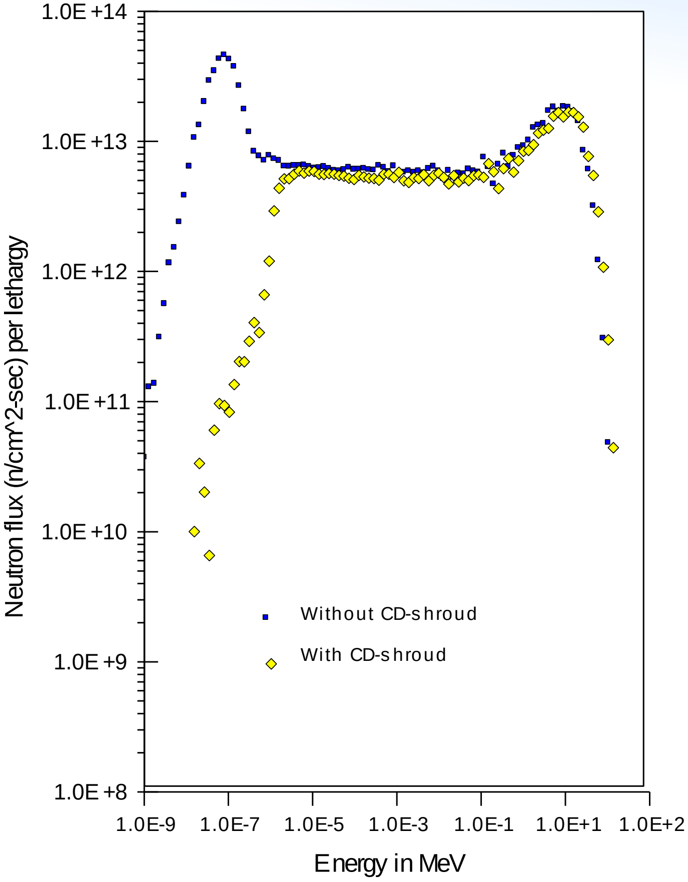
- All experiment capsules can be inserted into any of the fuel assembly positions

Actinide Transmutation Fuel Development

AFCI Flux Spectra with Cadmium Sleeved Basket

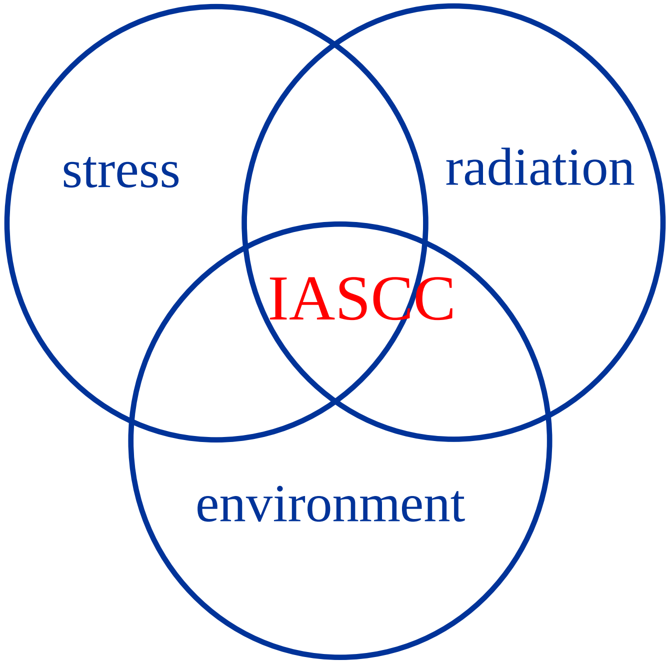
- Hard Spectrum Achieved in ATR by Use Of .045 inch Thick Cadmium
- > 97% of Thermal Flux is Removed

	Thermal neutron flux (E < 0.625 eV) n/cm^2-sec	Fast neutron flux (E > 1.0 MeV) n/cm^2-sec
With CD-shroud	8.46E+12	9.31E+13
Without CD-shroud	3.71E+14	9.39E+13
Ratio	2.28%	99.14%
Note: the flux tallies are normalized to a E-lobe power of 22 MW.		



Irradiation Assisted Stress Corrosion Cracking

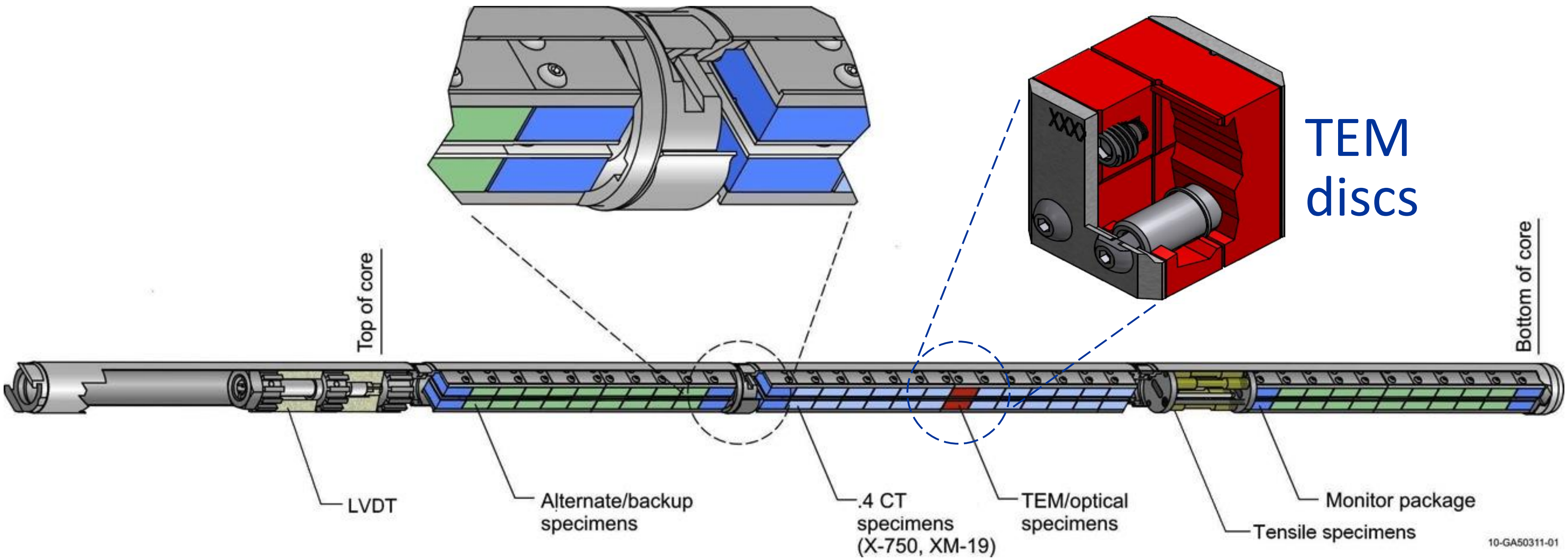
- IASCC occurs in Fe, and Ni base austenitic reactor materials
- Component cracking occurs at stress levels well below design stress



Intergranular cracking

PWR Loop Test: IASCC

- 0.4T compact tension specimens
- X-750 and XM-19
- 54 CT specimens + tensile specimens
- TEM disc specimens embedded in ‘dummy’ CT specimens



Material is this test was used for Irradiation Assisted Stress Corrosion Cracking (IASCC) crack growth rate measurements

Shielded IASCC Test Systems

Specially designed hot cells used to conduct stress corrosion crack growth rate measurements and fracture toughness testing in simulated BWR and PWR environments (and changing conditions)

Description:

- Testing cell with two 4 liter autoclaves
- 0.4T and 0.5T compact tension specimens for IASCC
- DCPD crack growth rate measurement (Direct Current Potential Drop)
- Utility cell with SEM for fracture surface examination
- 100 kN testing capacity
- Lead shielding for 40,000 R source term
- Accepts GE-100 cask transfers

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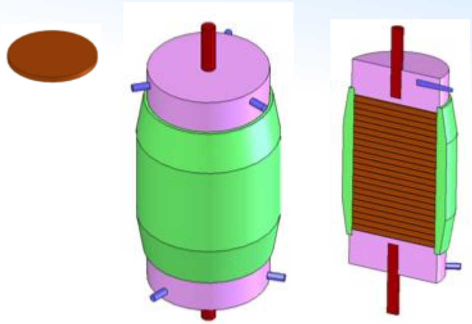
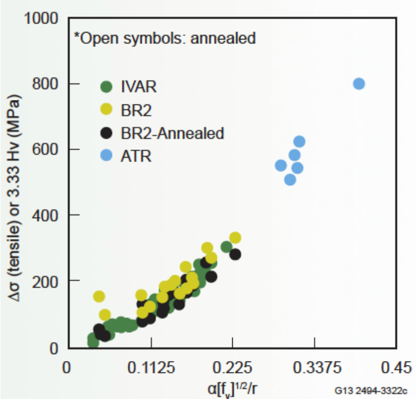
“Characterization of the Microstructures and Mechanical Properties of Advanced Structural Alloys for Radiation Service”

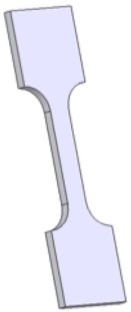
Prof. G. Robert Odette, UCSB, Dr. Jim Cole, INL (Peter Wells, Graduate Student)

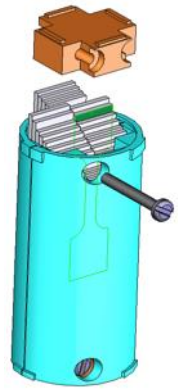


Scientific Goal:

Large matrix or “Library” of samples (~1300) consisting of 39 advanced reactor structural materials. Testing conditions and sample geometries were selected to gain insight into a variety of outstanding questions on irradiation behavior in this important class of materials.



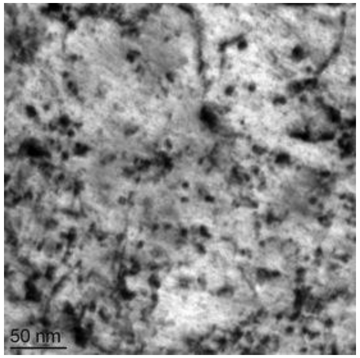




Significant Outcomes:

- Formation of late blooming phases in model RPV steels.
- High temperature strength and fracture behavior of SFR relevant F-M cladding alloys after irradiation.
- Radiation induced segregation behavior in model Fe-Cr alloys.
- Micromechanical behavior of FIB produced model Fe-Cr cantilevers.

Model RPV Steels



Project participants also include ORNL and PNNL

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Future Needs for Materials Prototyping

- The past approach is based on exhaustive parametric testing ('heat and beat' or 'bake and break') to see if it performs
- The future should be based on modeling/simulation with extensive experimental validation, *in-core*
  - Transmutation fuels with recycled fuel waste, for destruction of long-lived minor actinides
  - High temperature fuels for developing nuclear energy into a low-emission, high-temperature heat source
  - Fuels and materials for advanced concepts for nuclear energy systems, including fast reactors with considerably improved economics and performance
  - Highly reliable fuels for light water reactors
  - Materials for service in extreme nuclear conditions
  - Materials for advanced waste forms and decay storage systems

The Direction of Future Capabilities

- Focus on *phenomena*, not performance
  - Measure time-dependent changes in the fuel/clad system under extreme temperature and irradiation environments (both *in-core,* and *in-cell* during PIE)
  - Diagnose and enhance control of fuels/materials fabrication processes to optimize performance
- High-radiation field, real-time measurement systems
  - Fuel temperature and temperature gradient
  - Neutron (and gamma) flux and dose
  - Macro- and microstructure of fuel and clad
  - Chemical potential of fuel under irradiation
  - Wide range of thermomechanical, thermophysical and physiochemical properties
  - Elemental composition and phase identification
  - Dimensional changes

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