Idaho National Laboratory

Basics of Nuclear Fuels

Steven L. Hayes, PhD

Manager, Fuel Performance & Design Department

ATR-NSUF Users Week
Idaho Falls, Idaho
June 8, 2010

Outline of Presentation

- ***What is a Nuclear Fuel?***
- **Types of Nuclear Fuels**
  - Fuel Element Designs
  - Fuel Element Materials/Forms
  - Fuel Assemblies
  - Fabrication Issues
- **Irradiation Performance Phenomena**
  - High temperature gradient
  - Burnup and fission product accumulation
  - Irradiation growth
  - Fuel swelling and fuel-cladding mechanical interaction (FCMI)
  - Fission gas release
  - Fuel constituent redistribution
  - Fuel restructuring
  - Fuel-cladding chemical interaction (FCCI)
  - Fuel-coolant compatibility
  - Cladding swelling, creep, corrosion

What is a Nuclear Fuel?

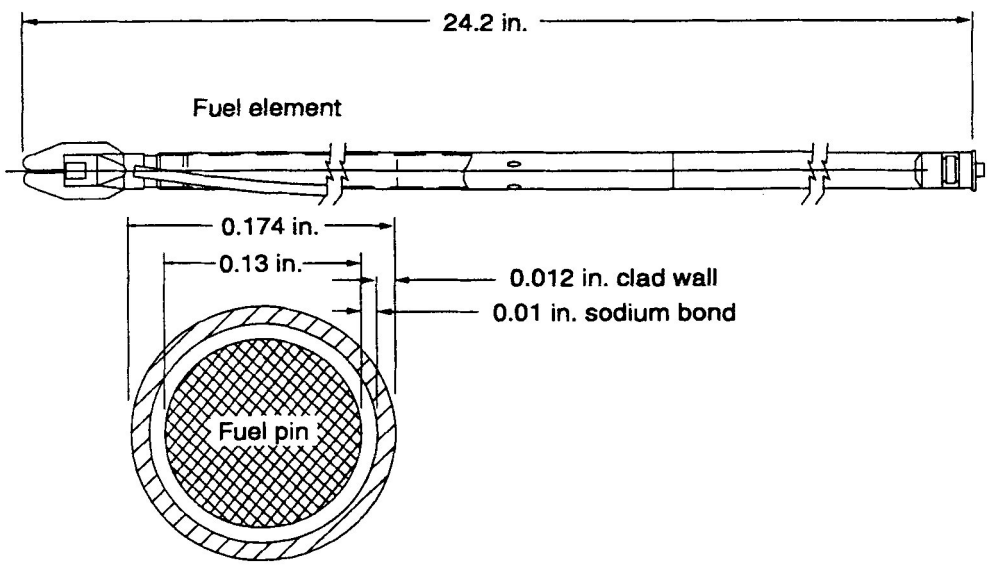
- **Nuclear fuel is a (usually removable) component that includes fissile and/or target material used as the power source to achieve and sustain a controlled nuclear chain reaction**
  - It must survive the reactor environment without allowing any significant release of radioactive materials
- **Fissile Materials:**
  - U^{235} is the only naturally occurring fissile isotope
  - **Natural uranium** contains 0.7 wt% U^{235} and 99.3 wt% U^{238}
  - Targets of U^{238} produce fissile Pu^{239} by neutron capture
  - Targets of Th^{232} produce fissile U^{233} by neutron capture
  - Other actinides also include fissionable isotopes
- **Nuclear fuel elements *normally* include:**
  - The fissile and/or target material in a stable form
  - A cladding barrier to contain the fissile material and fission products and prevent interaction with reactor coolant
  - An assembly structure to fit the reactor design allowing load and unload

Types of Nuclear Fuels

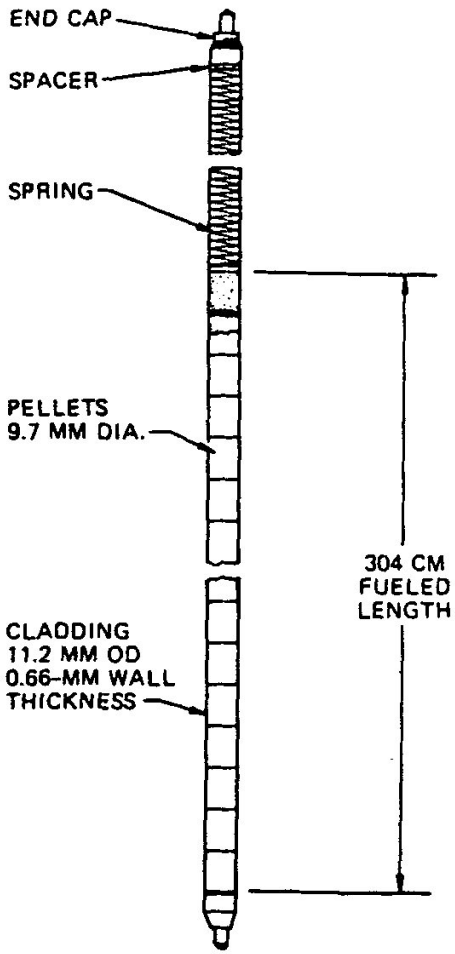
- **Nuclear fuels differ widely from reactor to reactor**
  - Geometrical configuration of fuel and cladding
    - *Fuel rods*
    - *Fuel plates*
    - *Particle fuels*
  - Materials used for U-bearing (or Pu) fuel
    - *Ceramic compounds*
    - *Metallic alloys*
  - Materials used for cladding

Fuel Element Designs

- **Rod-Type Fuels**
  - Most common fuel type (i.e., LWRs, LMRs, TRIGAs)
  - Cylindrical fuel in cladding tube
  - Plenum for fission gas



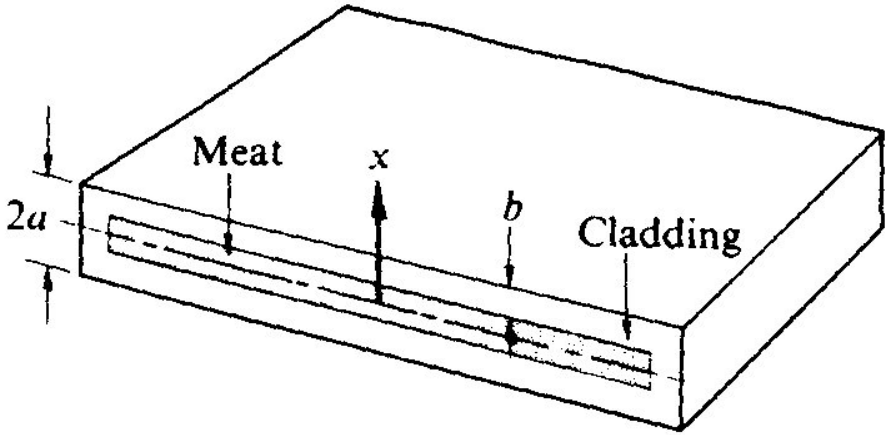
EBR-II Mark-II driver-fuel element

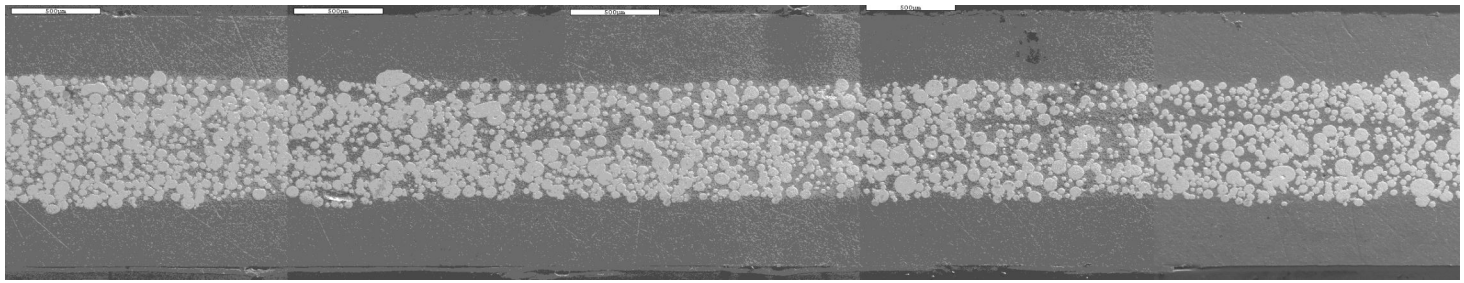


Fuel rod of a pressurized-water reactor.

Fuel Element Designs

- **Plate-Type Fuels**
  - Research and test reactors (HFIR, MTR, ATR)
  - Dispersion fuels (i.e., fuel particles embedded in a metal matrix)
  - No plenum





Fuel Element Materials

- **Desirable Properties**
  - High thermal conductivity
  - High melting point
  - Low thermal expansion
  - Chemically stable
  - Resistant to radiation damage
  - High fissile density
  - Economical fabrication

- **There is <u>No Perfect Fuel</u>**
  - **Compromise is always required**

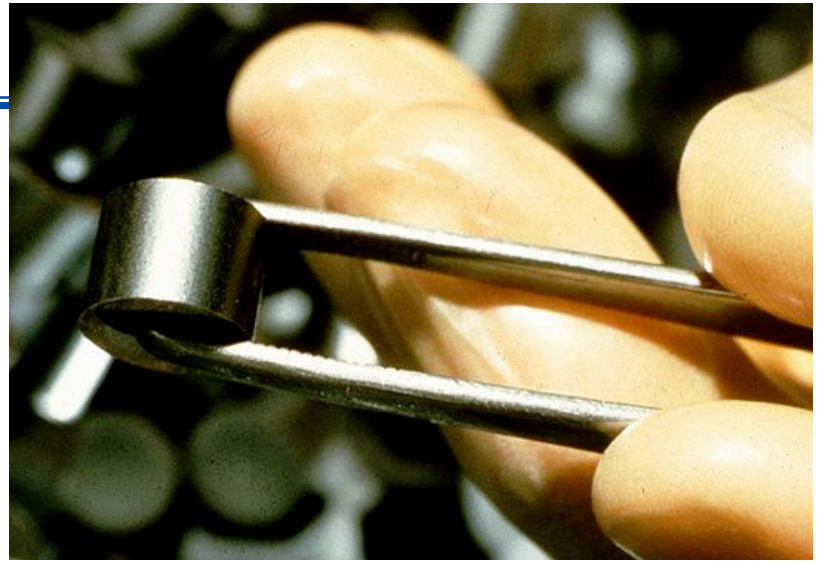
- **Fuel Materials**
  - Ceramic Compounds
    - *Oxides {UO_2, $(U,Pu)O_2$}*
    - *Carbides {UC, (U,Pu)C}*
    - *Nitrides {UN, (U,Pu)N}*
  - Metal Alloys (U-Pu-Zr-Mo)
  - Others (UAl_x, U_3Si_2, U/Zr hydride)

- **Cladding Materials**
  - Zirconium Alloys for LWRs
  - Stainless Steels for Fast Reactors
  - Aluminum Alloys for Research and Test Reactors
  - SiC for Gas Reactors
  - Refractory Alloys for High Temperature Applications (i.e., W, Ta, Nb, Mo, V)

- **Bond (Gap) Materials**
  - Helium gas
  - Liquid sodium
  - Metallurgical bond (i.e., no gap)

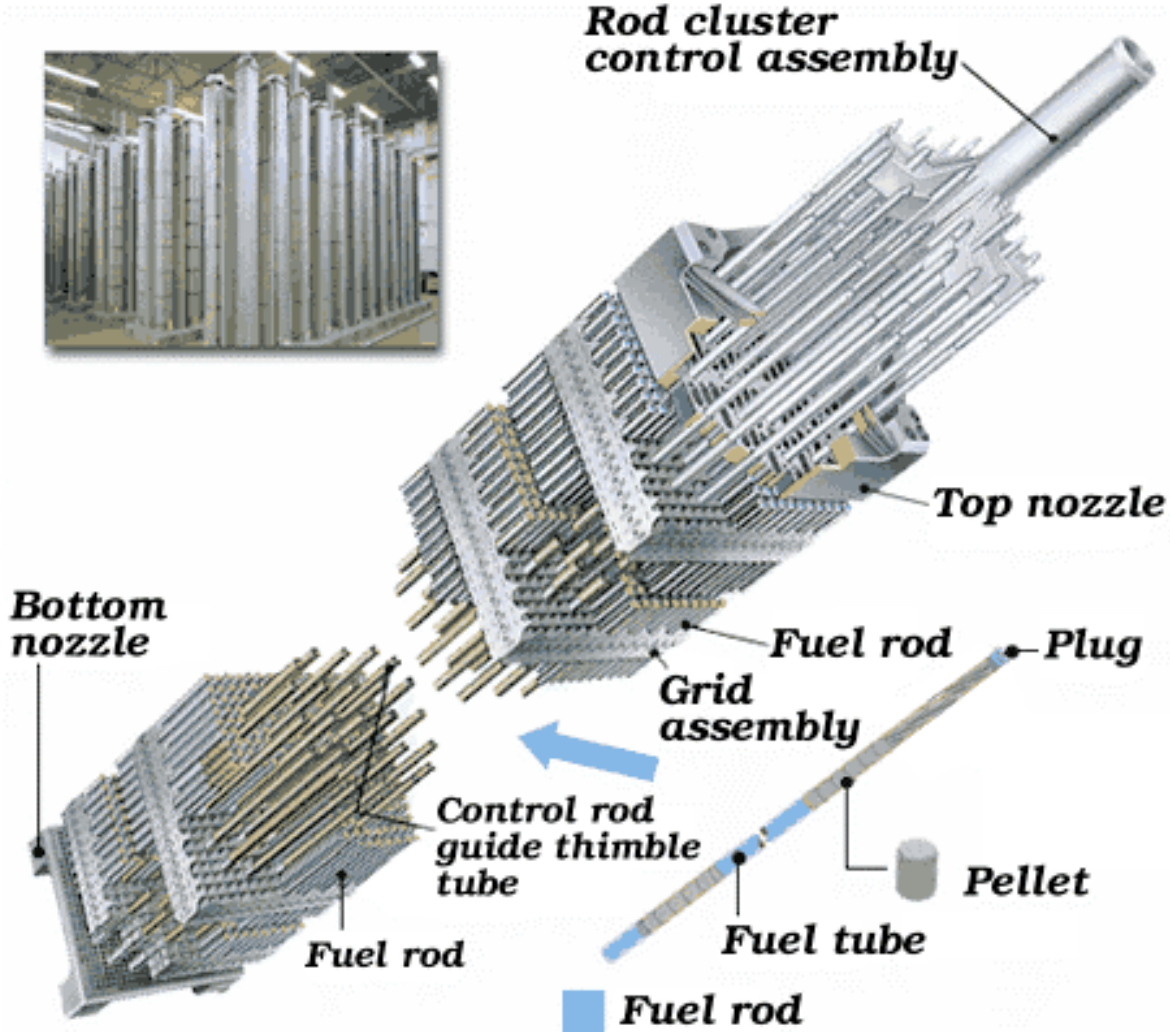
Fuel Forms

- **Oxide Fuels** – Nominally UO_2
- **Metallic Fuels**
  - Pure U metal
  - U – Al alloys
  - U – Zr alloys
  - U – Mo alloys
- **Dispersion Fuels** [metallic compounds or ceramics in a metal matrix]
  - UAl_x-Al
  - U_3Si_2-Al
  - U-ZrH
- **Particle Fuels** – UO_2 or UO_2+UC_2 [ceramic spherical particles with ceramic barrier coatings in a graphite or ceramic matrix]



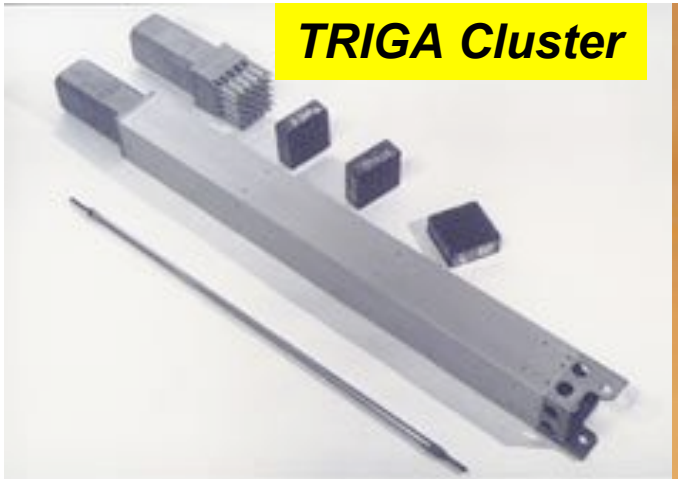
Fuel Assemblies – Reactor Core Fuel Arrays

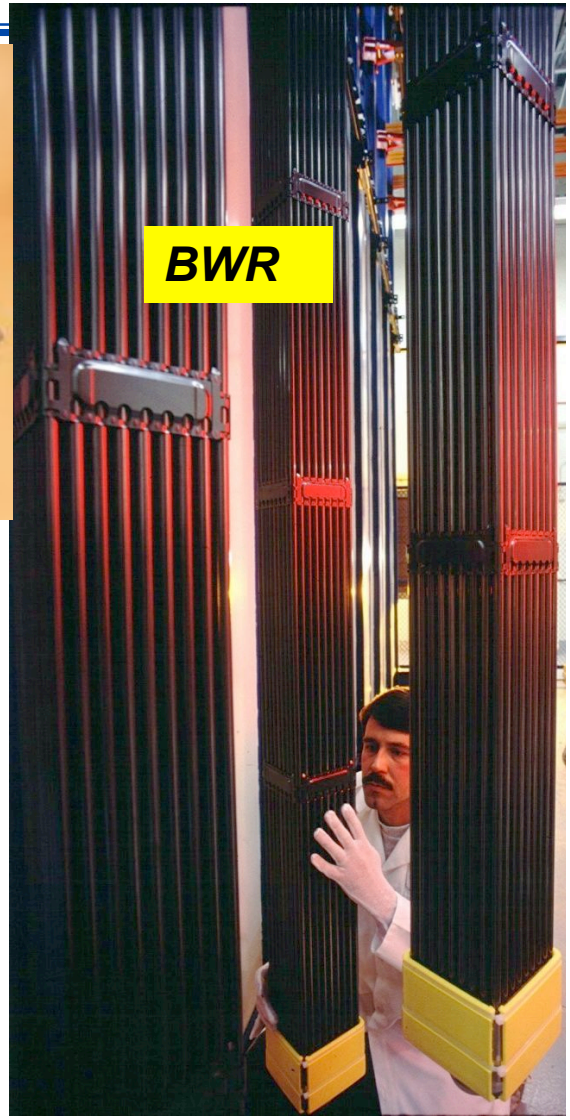
- **Fuel Assemblies are arrays of fuel pins or rods spaced and framed with hardware, sometimes with control rods, for direct insertion into reactor cores.**
- Fuel assemblies are specific to the reactor design involved.

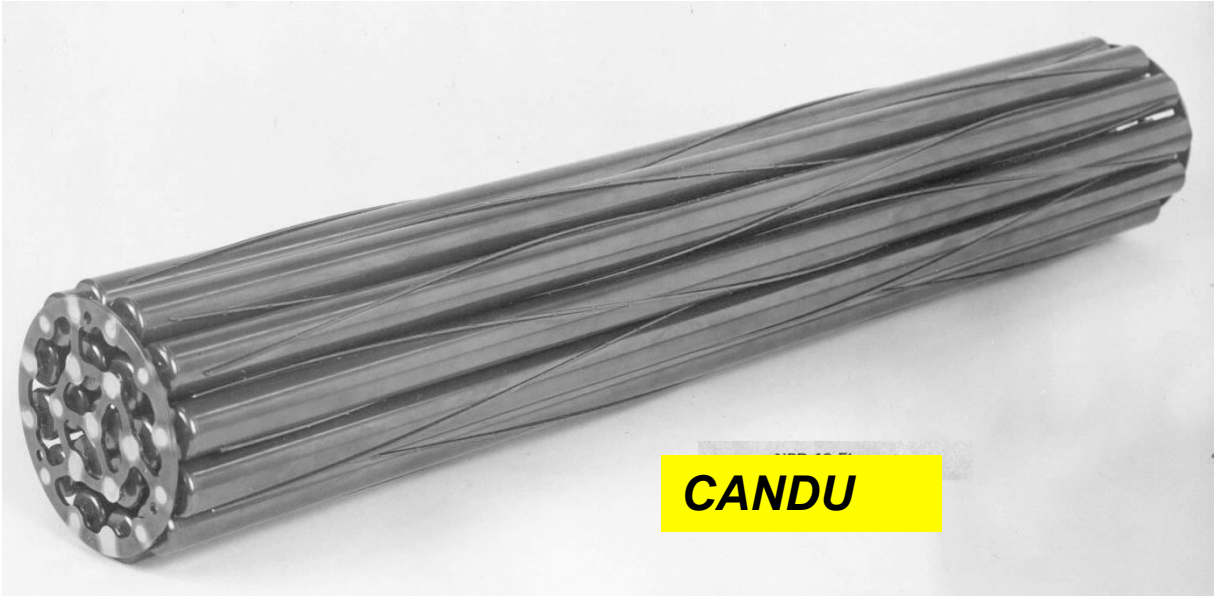


Mitsubishi PWR Fuel Assembly Configuration

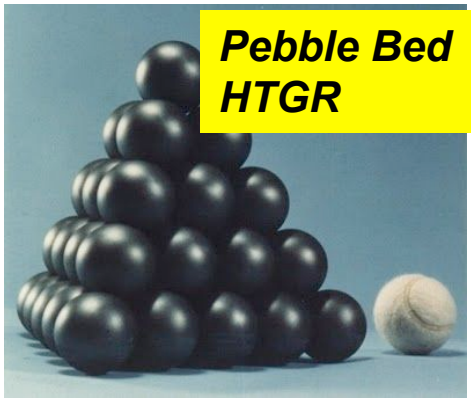
Other Fuel Assemblies

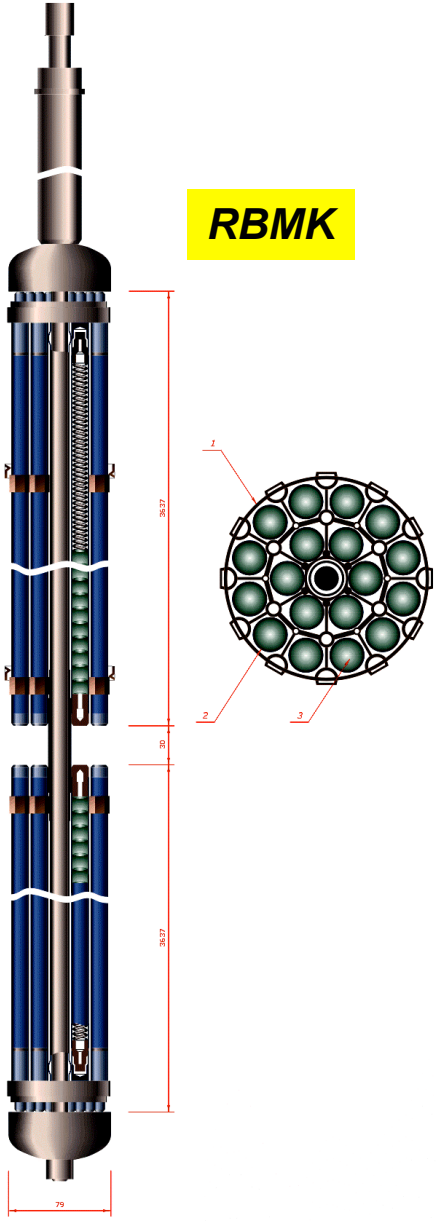


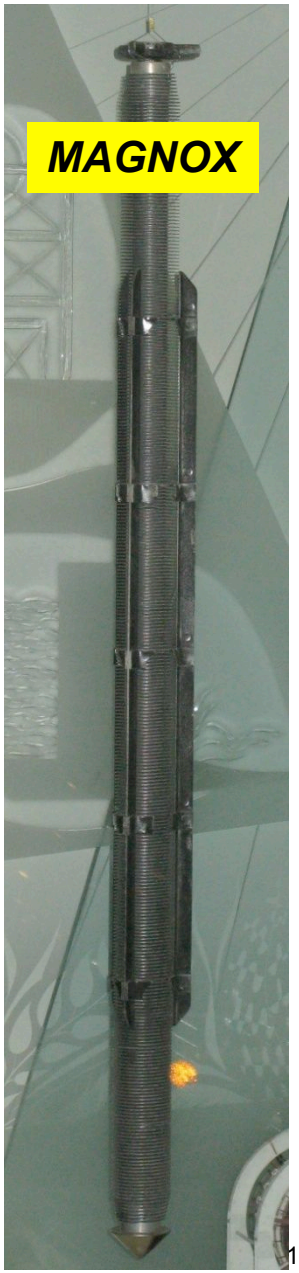




Other Fuel Assemblies

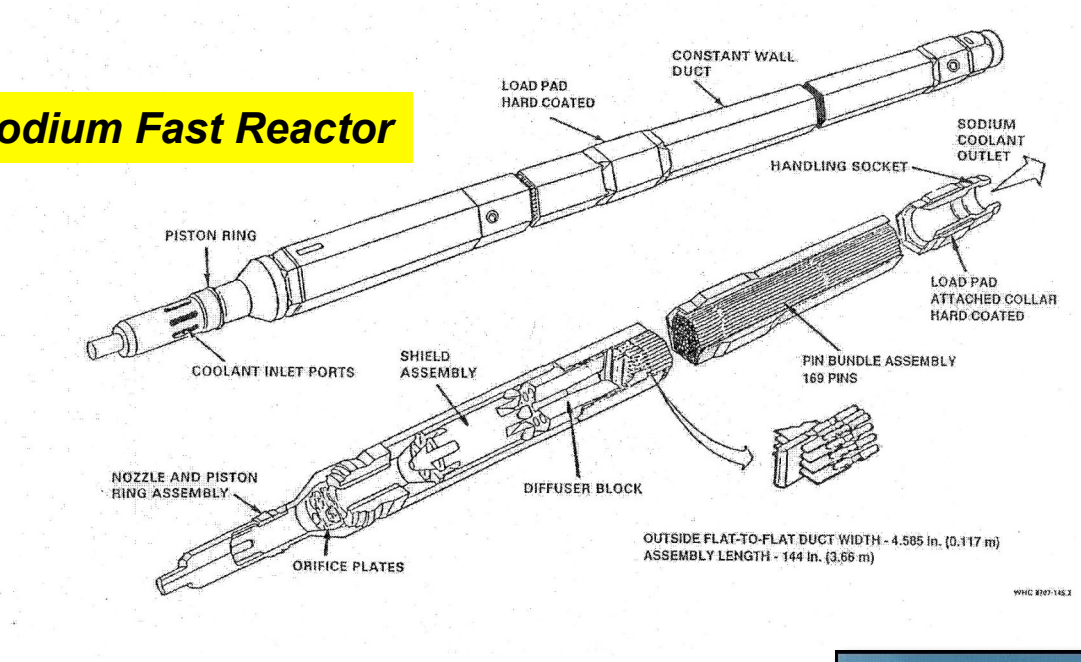


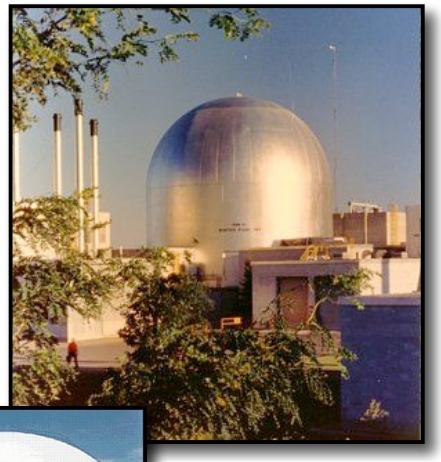


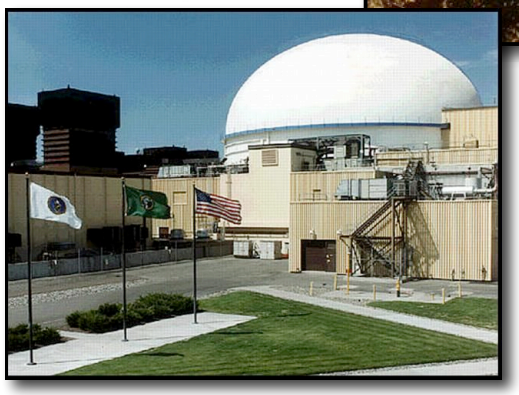


Other Fuel Assemblies

Sodium Fast Reactor

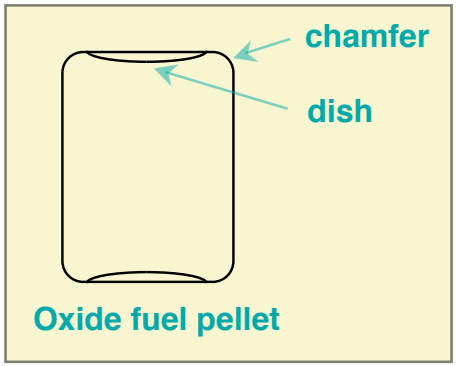






Fabrication Issues

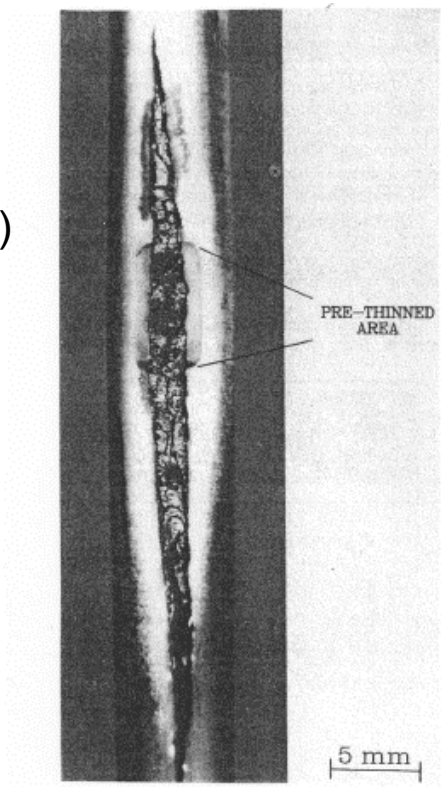
- **Ceramic Fuels**
  - Pellet fabrication non-trivial
  - Powder processing
    - *pressing, sintering, grinding of pellets*
    - *tight tolerances*



- **Metallic Fuels**
  - Relatively easy to fabricate by melting/casting processes
- **Dispersion Fuels**
  - Make, mix and press fuel and matrix powders
  - Roll or co-extrude with cladding
- **Particle Fuels**
  - Complex fabrication process
  - Aqueous synthesis of fuel kernels
  - CVD application of coatings
  - Compacting with matrix material

Irradiation Performance Phenomena

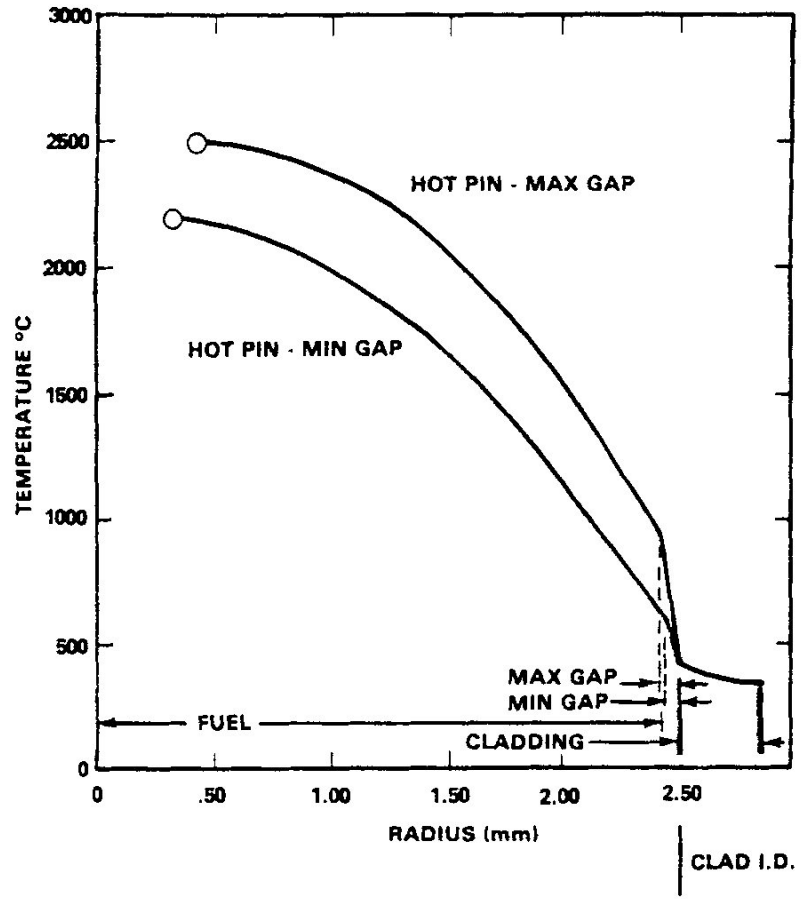
- **During irradiation of a nuclear fuel, many complex and interrelated phenomena occur**
  - High temperature gradient
  - Burnup and fission product accumulation
  - Irradiation growth
  - Fuel swelling and fuel-cladding mechanical interaction (FCMI)
  - Fission gas release
  - Fuel constituent redistribution
  - Fuel restructuring
  - Fuel-cladding chemical interaction (FCCI)
  - Fuel-coolant compatibility
  - Cladding swelling, creep, corrosion



- **These phenomena degrade the nuclear fuel eventually requiring its discharge from the reactor**

Fuel Temperatures/Temperature Gradient

- **Oxide Fuels**
  - Low thermal conductivity
    - *High central temperature*
    - *Large thermal gradient*
  - High melting point
    - *> 2800°C*
- **Metallic Fuels**
  - High thermal conductivity
    - *Low central temperature*
    - *Small thermal gradient*
  - Low melting point
    - *1100-1200°C*
    - *Eutectics even lower*

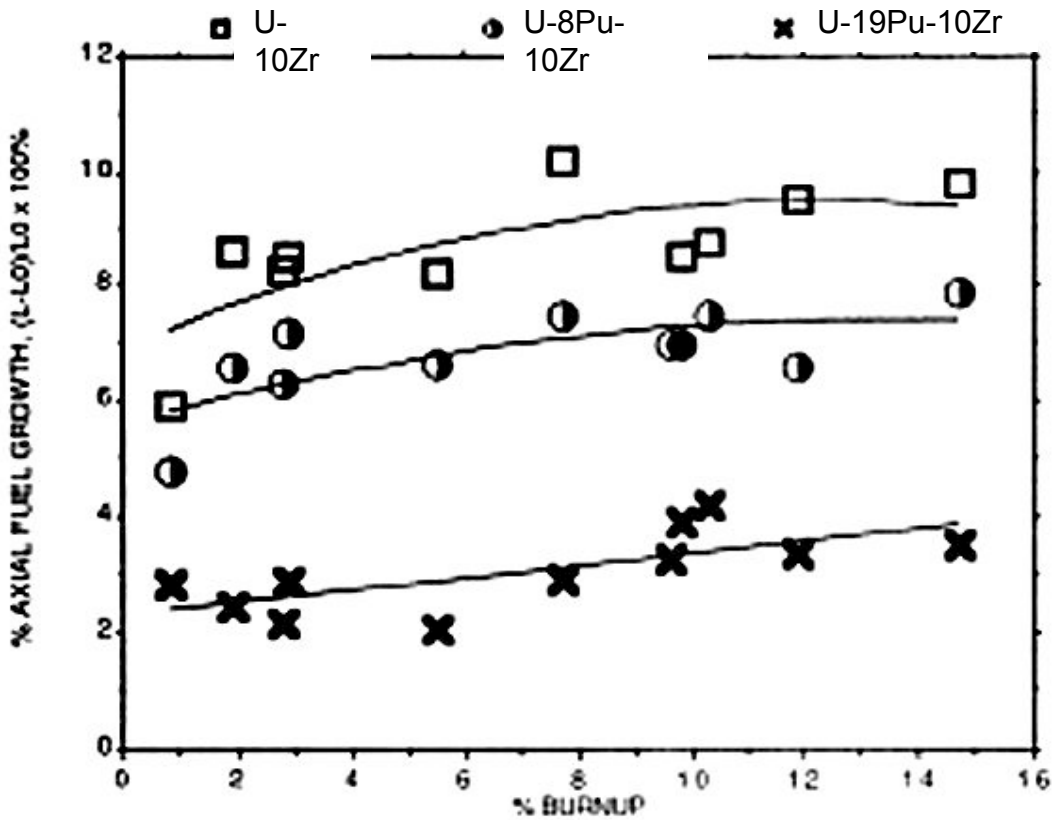


Burnup and Fission Products

- **Burnup**
  - A measure of how much U (or Pu) has been fissioned
    - *Units of MW-days/ton-U or atomic-%*
    - *LWR fuel currently limited to ~50,000 MWD/ton; experiments to >70,000 MWD/ton*
    - *Metallic & oxide fuel (fast reactors) limited to ~10 at.-%; experiments to 20 at.-%*
    - *Dispersion fuel (HEU research reactors) limits ~50 at.-%*
    - *50-90% of useful U (Pu) atoms not burned → motivation for reprocessing*
- **Fission Products**
  - Two atoms replace every U (or Pu) atom that fissions
  - More than 30 chemical elements produced by fission; chemical state of fuel can evolve substantially during irradiation
  - 25% of fission products are gas atoms (Kr, Xe)
  - Fuels with high minor actinide (Am, Cm) content also produce significant quantities of He during irradiation

Metallic Fuel Behavior—Axial Growth

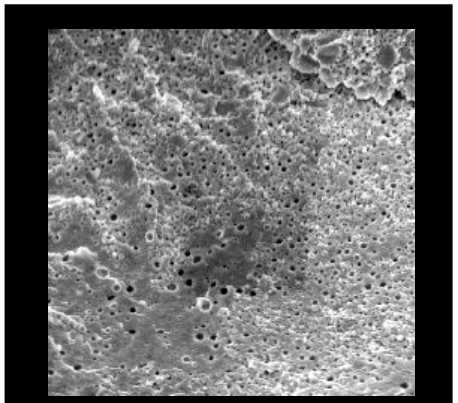
- **Axial growth of fuel column can be significant reactivity effect**
  - Can influence with alloying additions
  - Must understand to ensure adequate excess reactivity for desired cycle length



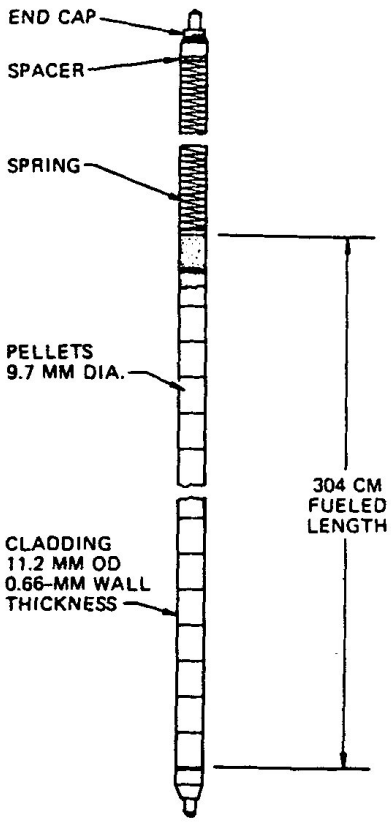
Axial Fuel Growth, from Pahl et al, 1990

Swelling/Fission Gas Release

- **Fuel Swelling**
  - Fuel swells due to generation of fission products
  - Gas atoms coalesce into bubbles, accelerating swelling
  - Fuel swelling tends to reduce or close gap
- **Fission Gas Release**
  - Some fission gas escapes fuel
  - Pressurizes plenum
  - Percent of gas escaping fuel
    - *< 10% in LWR fuel*
    - *> 50% in fast reactor fuel*

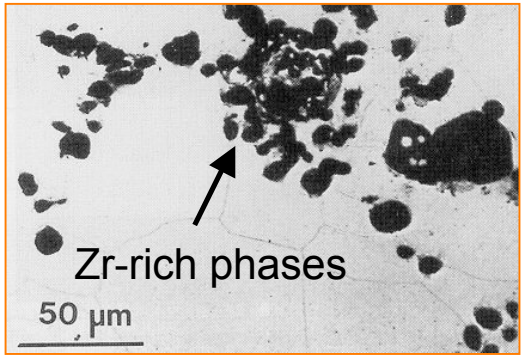


Bubbles in metallic fuel



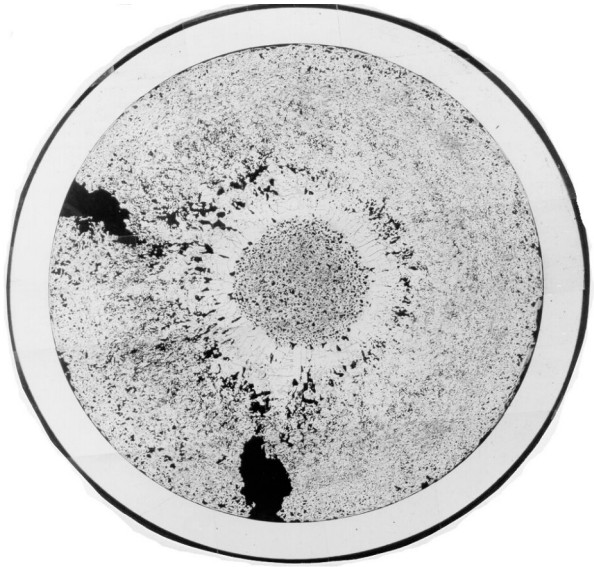
Fuel rod of a pressurized-water reactor.

Metallic Fuel Behavior—Swelling & Restructuring

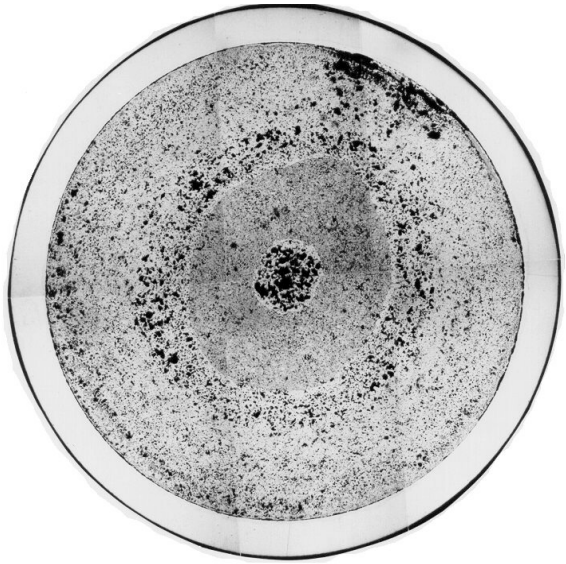


As fabricated U-20Pu-10Zr

X423A at 0.9% BU



X419 at 3% BU

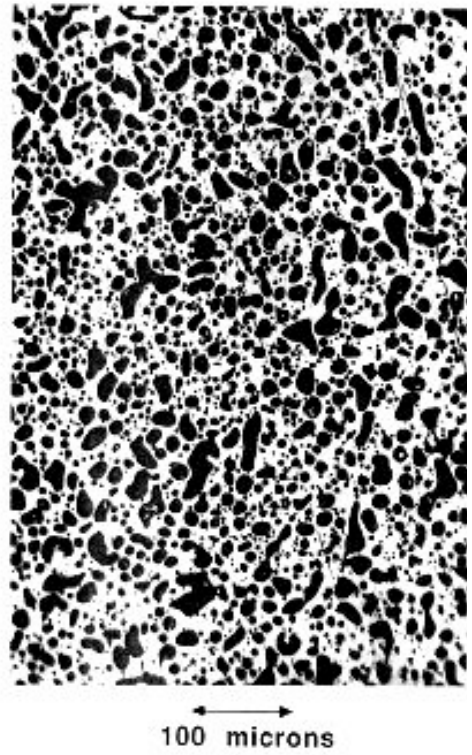


X420B at 17% BU

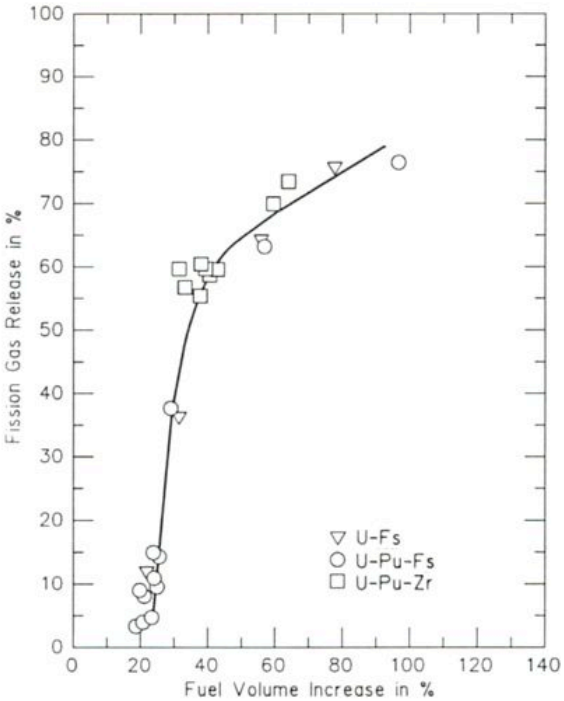
- Redistribution of U and Zr occurs early
- Inhomogeneity doesn't affect fuel life

Metallic Fuel Behavior—Swelling & Gas Release

- **Swelling**
  - Low smear density fuels
  - Rapid swelling to 33 vol% at ~2 at.% burnup
- **Gas Release**
  - Inter-linkage of porosity at 33 vol% swelling results in large gas release fraction
  - Decreases driving force for continued swelling

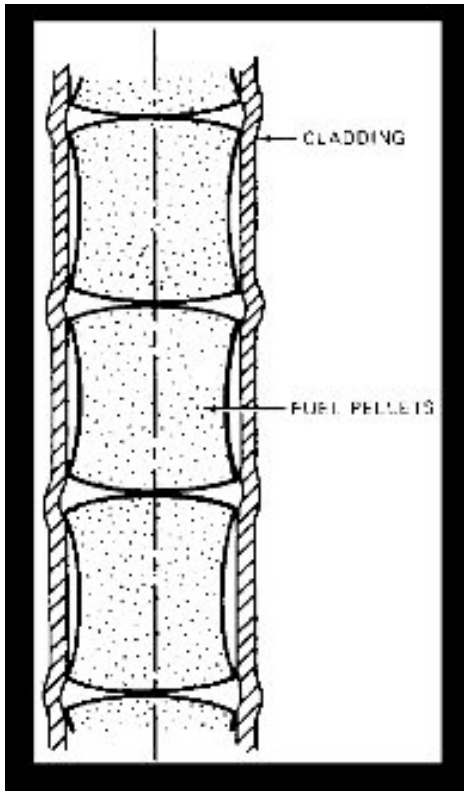


U-19Pu-10Zr (γ-phase) at 2 at.% burnup



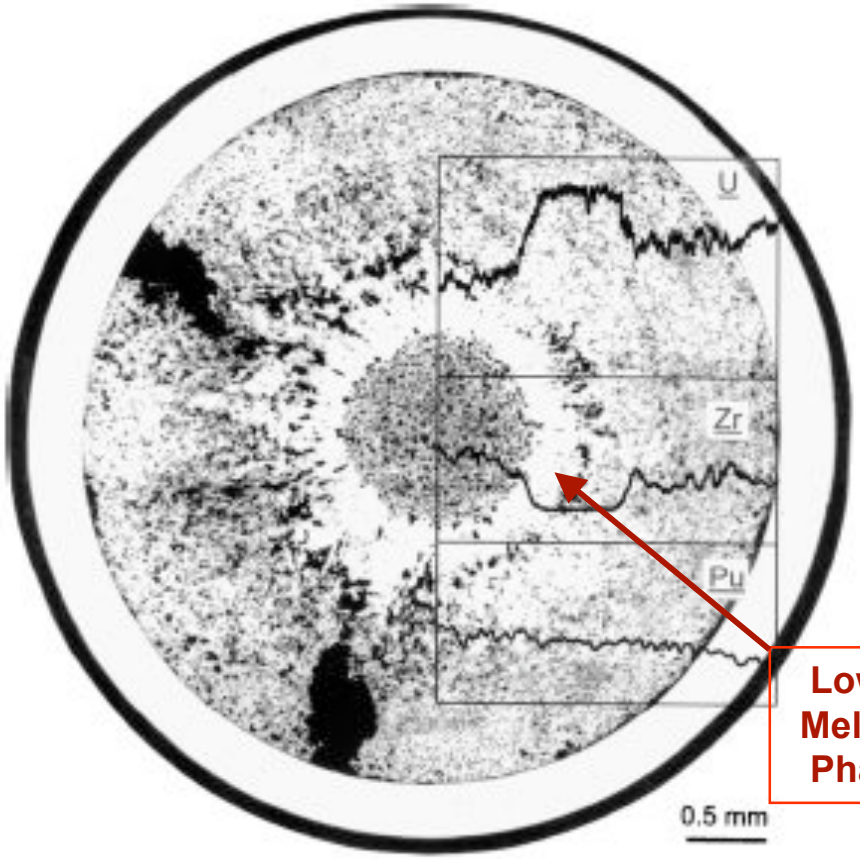
Fuel-Cladding Mechanical Interaction

- **Fuel-Cladding Mechanical Interaction**
  - Fuel swelling and/or cladding creepdown closes gap
  - Continued swelling/creep stresses cladding
  - Cladding strain eventually results in failure

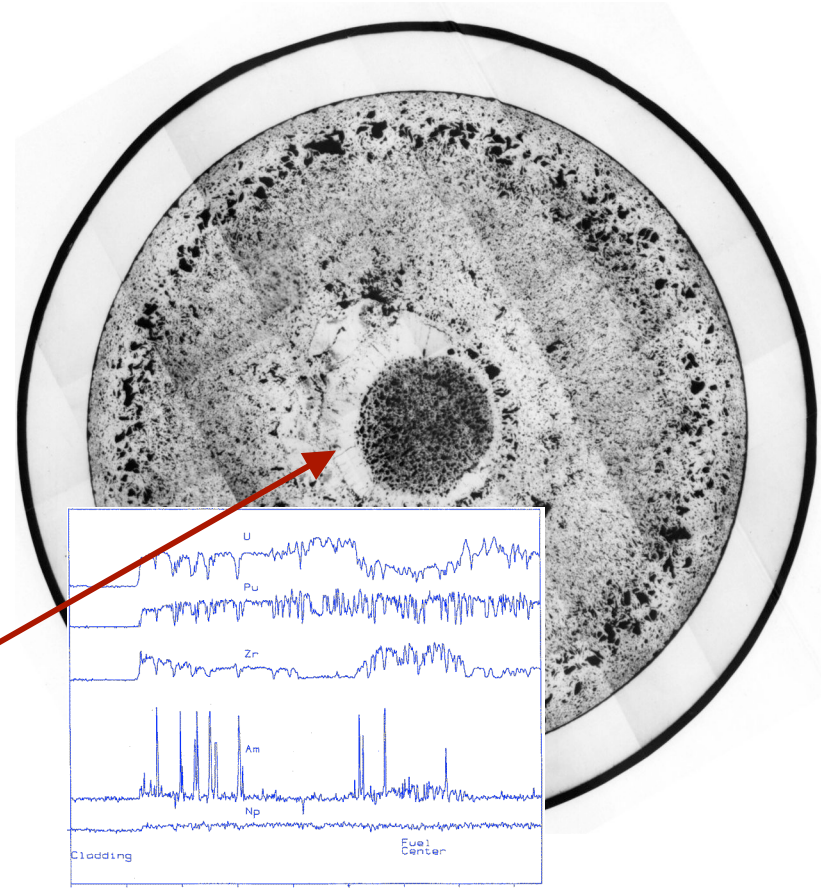


Metallic Fuel Behavior—Fuel Constituent Redistribution

U-Pu-Zr

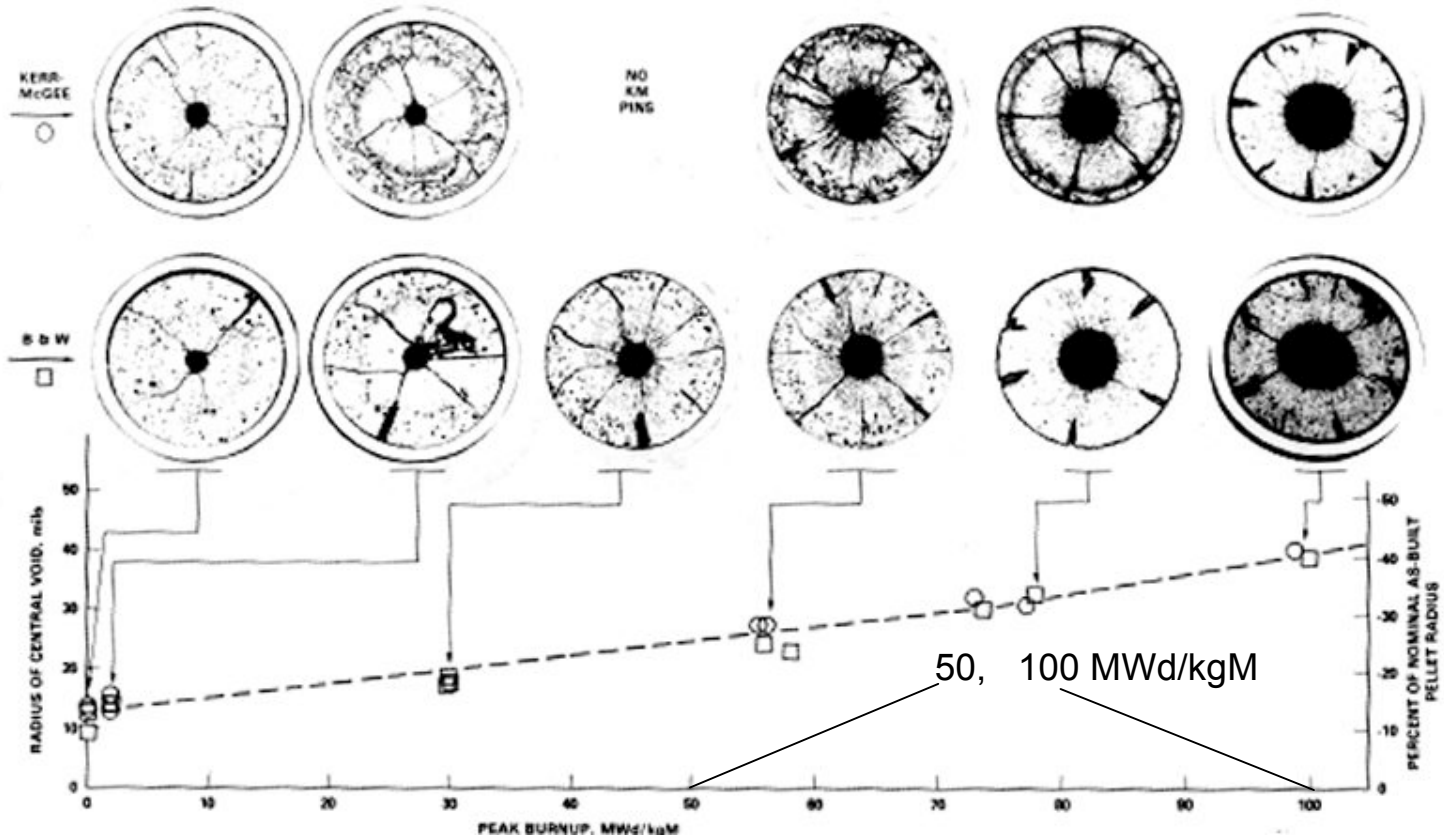


U-Pu-Am-Np-Zr



Lower Melting Phase

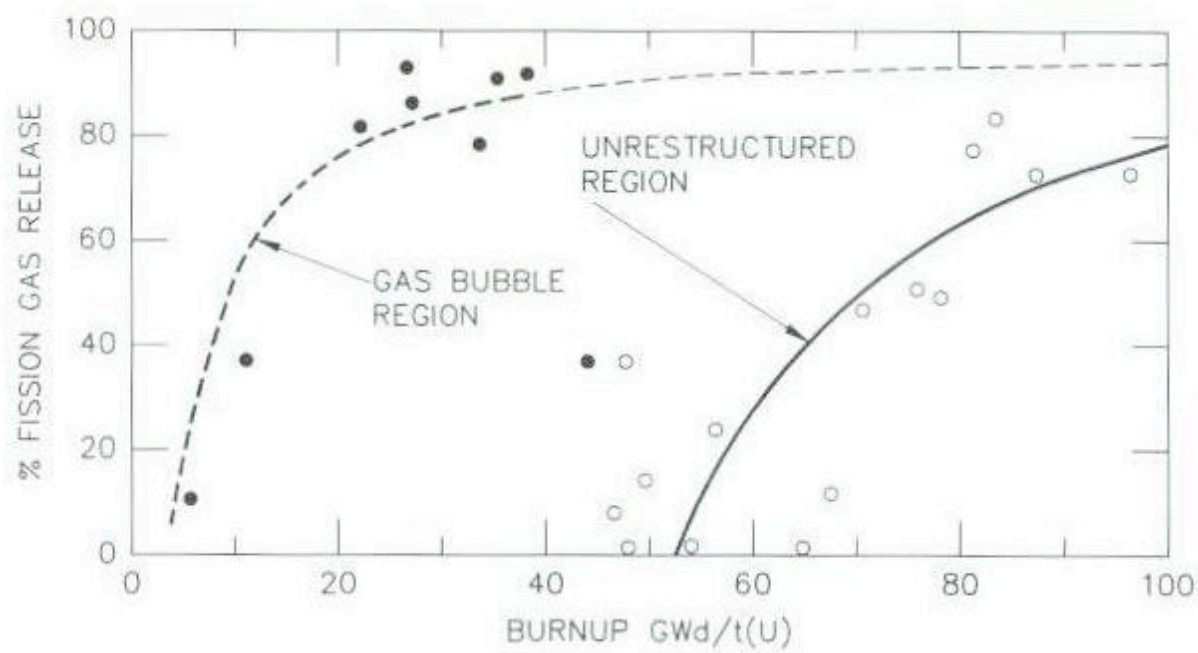
MOX Fuel Behavior—Restructuring



MOX fuel ceramography of FFTF driver fuel produced by Kerr-McGee and Babcock and Wilcox, showing restructuring as a function of burnup. **(from Hales, et al, 1986)**

MOX Fuel Behavior—Gas Release

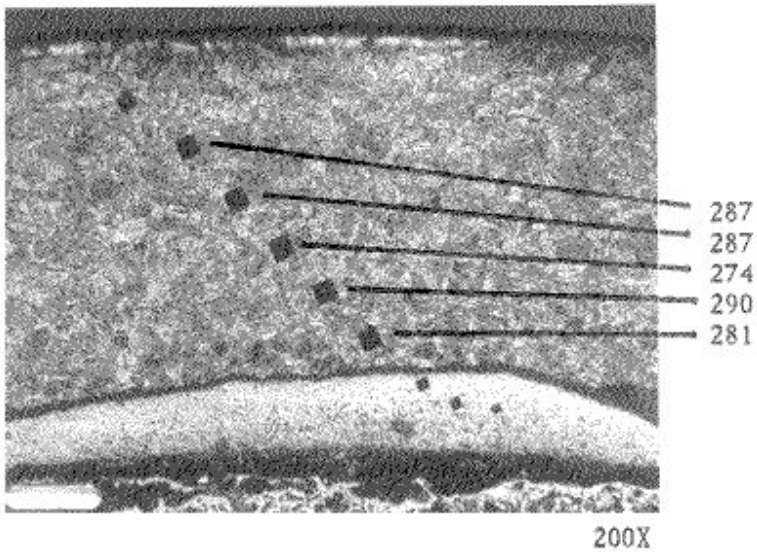
- **MOX fuel operated at high temperature and undergoing restructuring exhibits high gas release.**



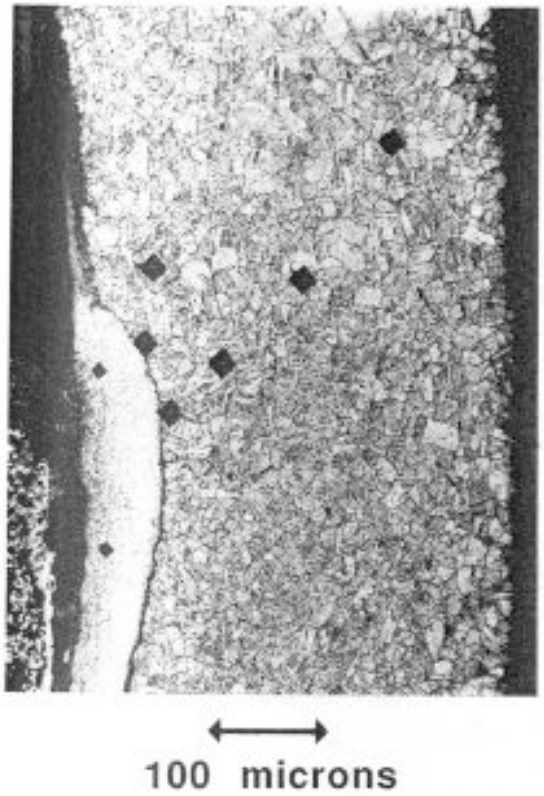
(from Lambert, et al, 1994)

Metallic Fuel Behavior—Steady-state FCCI

- **Fuel-Cladding Inter-diffusion**
  - RE fission products (La, Ce, Pr, Nd) and some Pu reacts with SS cladding
  - Interaction product brittle
  - Considered as cladding wastage



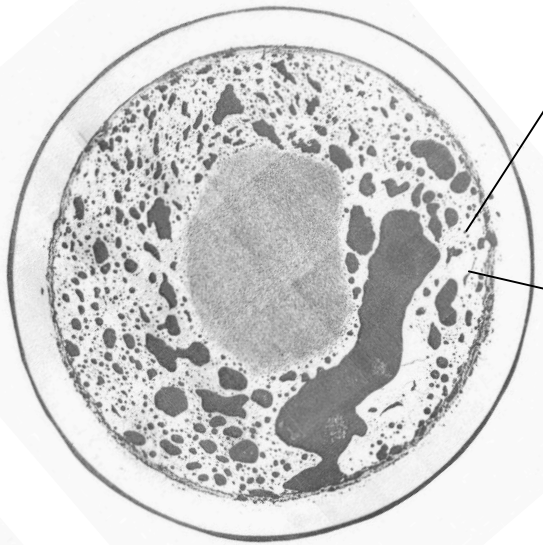
U-19Pu-10Zr with D9;
12 at.% burnup
(from Pahl, et al, 1990)

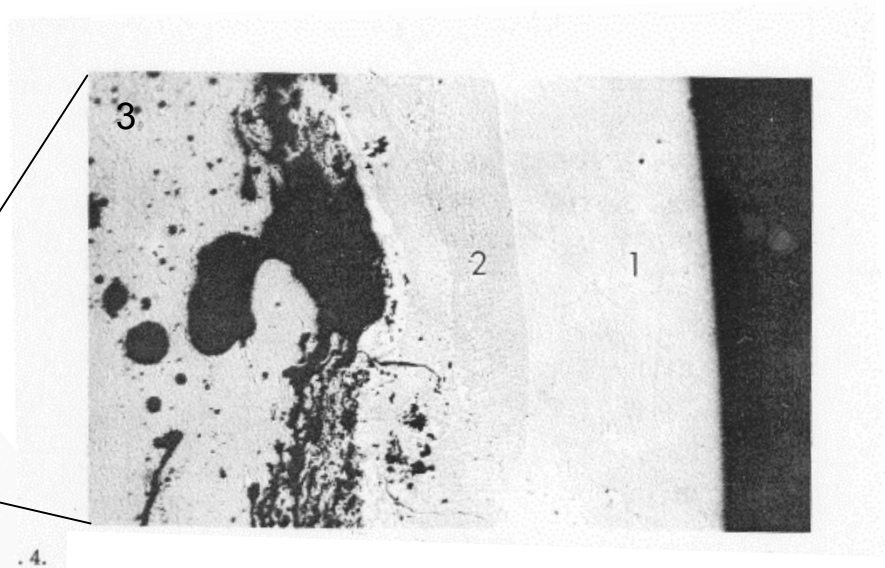


Transient Phenomena—Metallic Fuels
Fuel/Cladding 'Eutectic' Formation

U-10Zr / HT9 at 800°C, 1 hr

(from H. Tsai, et al, 1990)

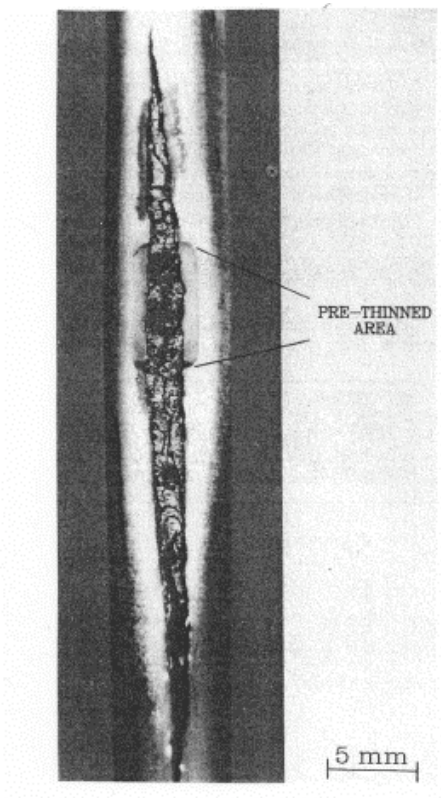




1 - Unaffected cladding
2 - Fuel/cladding solid-state interaction
3 - Fuel/cladding liquid phase

MOX Fuel Behavior—Fuel-coolant Compatibility

- Run-beyond-cladding-breach (RBCB) of MOX accompanied by fuel/Na reaction and initial crack extension
- Fuel loss can be related to degree of interaction
- Reactant layer can becomescoherent and mitigate further reaction with coolant



Typical breach extension in induced midlife failure, EBR-II K2B test.
(from Lambert, et al, 1990)

Cladding Performance

- **Cladding integrity assures fission product containment**
  - Breach of cladding referred to as fuel "failure"
  - Failure generally precludes continued use of fuel element/bundle
- **Cladding integrity degrades during irradiation**
  - Temperature, pressure and neutron flux cause "creep"
    - *High coolant pressure causes creep<u>down</u> (LWRs)*
    - *High fission gas release causes outward creep (LMRs)*
  - Radiation damage causes swelling (embrittlement)
  - Corrosion by coolant
  - Interaction with fuel

Life-Limiting Phenomena

- **Cladding breach ends a fuel element's use**
- **Cladding breach occurs due to:**
  - Embrittlement of zirconium cladding due to corrosion/hydriding by water coolant and stresses induced by FCMI (LWRs) → motivates development of corrosion-resistant cladding alloys
  - Creep rupture of cladding due to fission gas pressurization, accelerated by cladding thinning due to FCCI (LMRs) → motivates development of creep-resistant cladding alloys
- **Fuel burnup limit established to preclude cladding breach during irradiation**

QUESTIONS?