Nuclear Fuel Performance

NE-533

Spring 2024

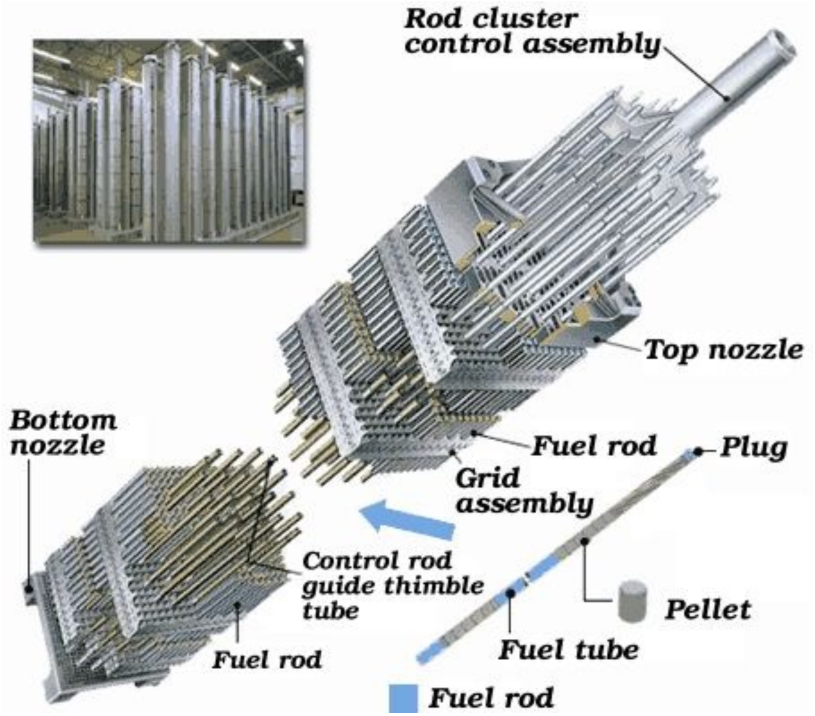
Last Time

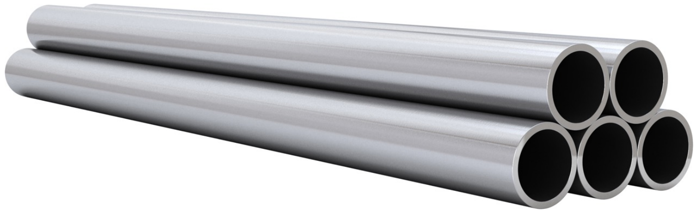
- In-pile release Booth model
- Forsberg-Massih 2-stage FGR model
- Three component diffusion
- Swelling/dimensional change
  - densification, thermal expansion, solid/gas fission product swelling, creep
- Fuel fracture

ZIRCONIUM CLADDING

Cladding

- The purpose of the cladding is to:
  - Hold the pellets together so that coolant can freely flow past
  - Transport heat from fuel to the coolant
  - Contain fission products
  - Contain fuel fragments
- We would also prefer if the cladding had little to no impact on the neutron transport in the reactor





Why Zirconium alloys?

Benefits

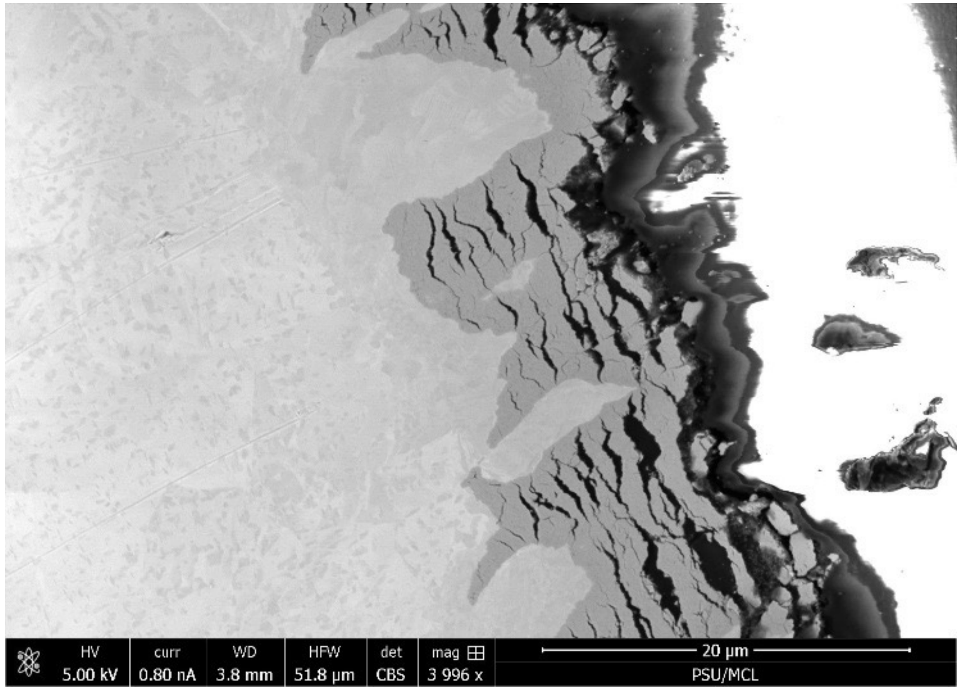
- Low neutron cross section
- Corrosion resistance in 300˚C water
- Resistance to void swelling
- Adequate mechanical properties
- Good thermal conductivity
- Affordable cost
- Available in large quantities

Problems?

- Corrosion under high temperature steam
- Hydride embrittlement
- Anisotropic characteristics lead to creep and growth

Zirconium

- Pure zirconium was not acceptable due to its oxidation behavior
- Oxygen with water reacted to form an oxide layer
- The oxide layer is brittle and flakes off, allowing more oxidation to occur
- Zr alloys were developed to improve corrosion resistance



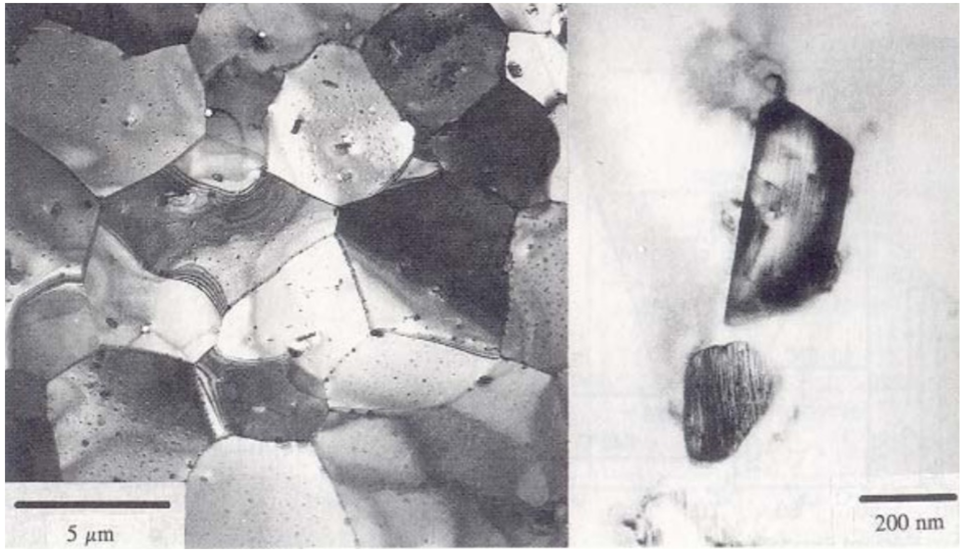
Commercial Zr Alloys in PWRs

Alloy	Sn %	Nb %	Fe %	Cr %	Ni %	O %
PWRs (structural components and fuel rods)						
Zircaloy-4 (SRA)	1.2-1.7	-	0.18-0.24	0.07-0.13	-	0.1-0.14
ZIRLO (SRA)	1	1	0.1	-	-	0.12
Optimized ZIRLO (pRXA)	0.7	1	0.1	-	-	0.12
M5 (RXA)	-	0.8-1.2	0.015-0.06	-	-	0.09-0.12
HPA-45 (SRA/RXA)	0.6	-	Fe+V	-	-	0.12
NDA (SRA)	1	0.1	0.3	0.2		0.12
MDA (SRA)	0.8	0.5	0.2	0.1		0.12

PRXA Partial Recrystallization Anneal
RXA Recrystallization Anneal
SRA Stress-Relief Anneal

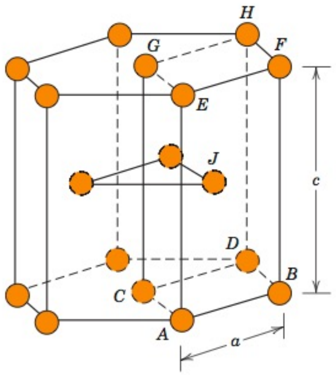
Alloying Elements

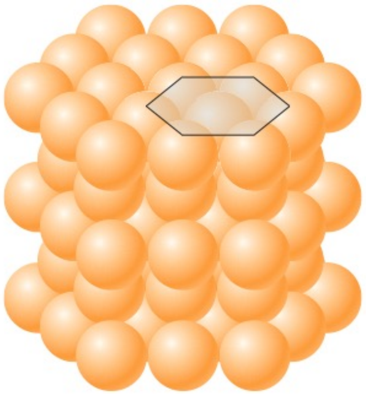
- The alloying elements impact the microstructure of the material
- The intermetallic precipitates significantly improve the oxidation behavior
- In Zircaloy 2, the precipitates are
  - $Zr(Cr, Fe)_2$
  - $Zr_2(Ni, Fe)$
- In Zircaloy 4, the precipitates are
  - $Zr(Cr, Fe)_2$
- Phosphides (Zr_3P) and silicides (Zr_3Si) are also occasionally found
- The precipitate distribution, size, morphology, and composition all impact the properties of the material

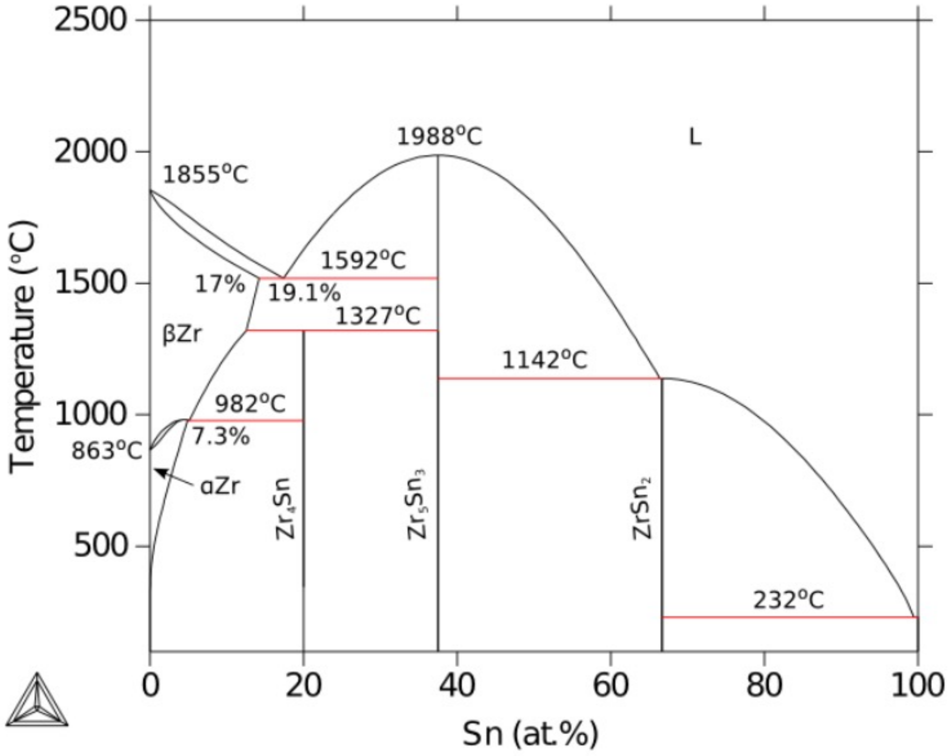


Zirconium Phases

- The α-Zr phase has a hexagonal-close-packed (HCP) structure
  - At temperatures below about 863ºC
  - Has the most desirable properties
- The β-Zr phase has a body-centered cubic (BCC) structure
  - We try to avoid this phase







Zr Tube Fabrication

- The cladding tubes are fabricated using various processes that SEVERELY deform the material
- The severe plastic deformation causes significant dislocation hardening, that can make the tube material too brittle
- Plastic deformation along slip planes results in reorientation of the grains
- The properties of the material are controlled by heat treatment
- Raising the temperature, but below 863˚C, anneals the sample to reduce cold work (stress-relieved)- SRA
- Raising the temperature above 863˚C changes to the β phase. They then quench the sample to create a random texture in the α phase-RXA or pRXA

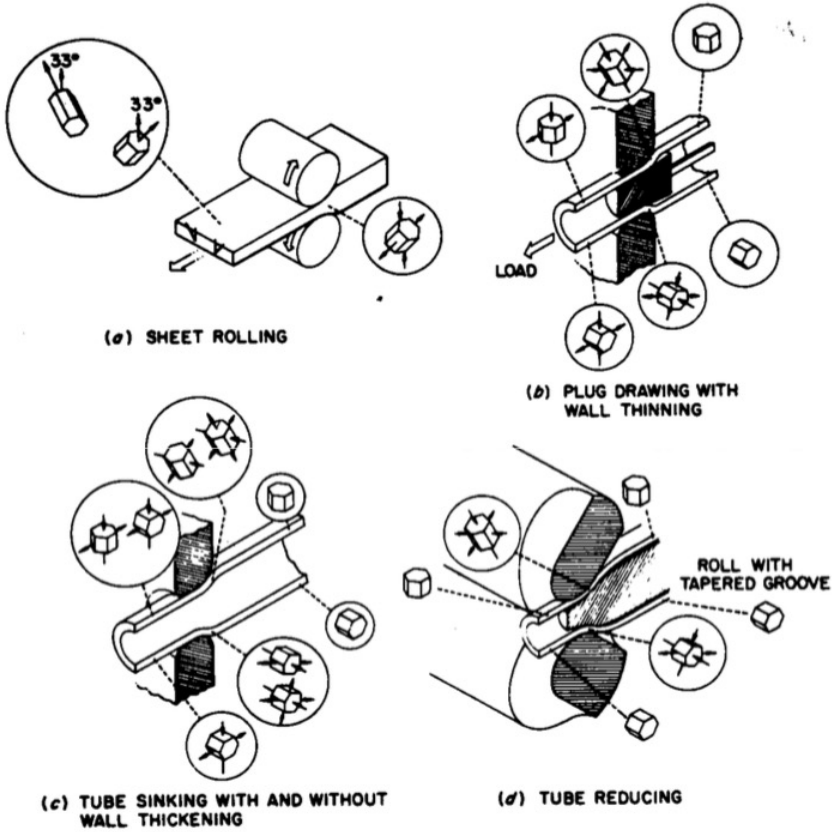


FIG. 48. Strain states for various types of fabrication of Zircaloy-2 [109].

Zr alloy microstructures

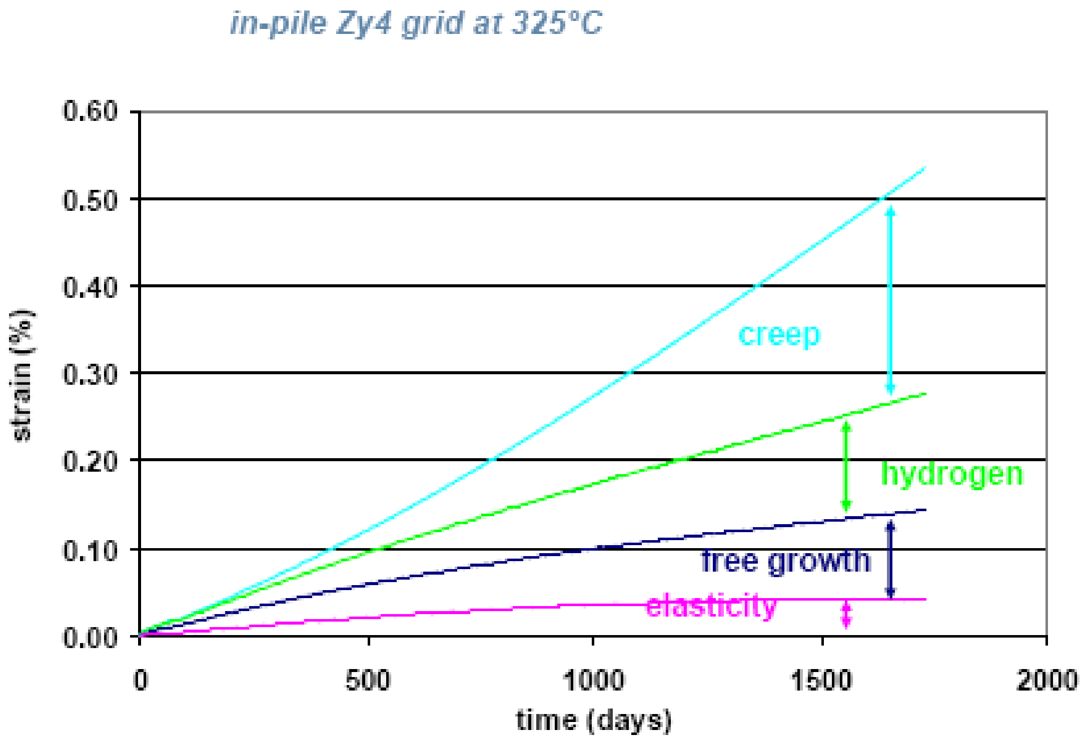
Fully recrystallized after quench

Stress-relieved microstructure

Zirconium Creep and Growth

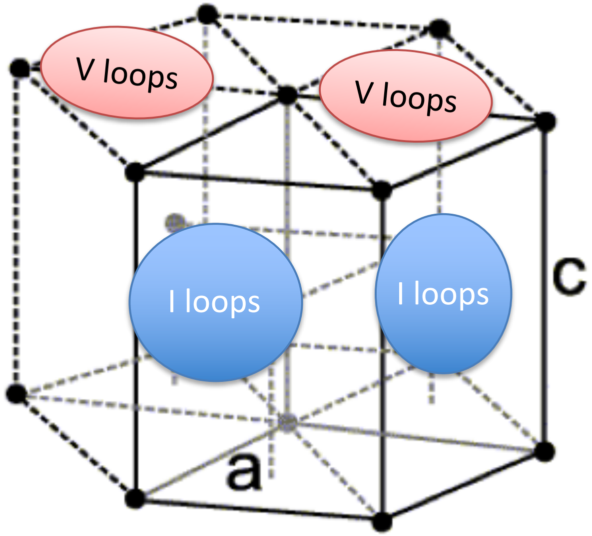
- The cladding undergoes a very large amount of irradiation during its life, the main effects are:
  - Interstitial loops form on prismatic planes
  - Later in the reactor exposure, vacancy loops form on the basal plane
- The overall number of displacements to the cladding can be calculated with the NRT model to be ~ 8 x 10^{-8} dpa/s, which, over 3 years exposure gives a total of ~8 dpa (every atom in the solid is displaced on the average eight times)

$$\nu(T) = \frac{\kappa E_D}{2E_d} = \frac{\kappa(T-\eta)}{2E_d},$$



Irradiation Growth

- Irradiation growth results from material anisotropy
- There is corresponding anisotropy in the defect behavior within the unit cell
- There is also a texture formed in the grain orientations
- Interstitial loops form on pyramidal planes, causing shrinkage along the center axis
- Later, vacancy loops form on basal planes, making it even worse
- The result is a change in shape, shrinking in the 0001 direction and expanding in the pyramidal direction



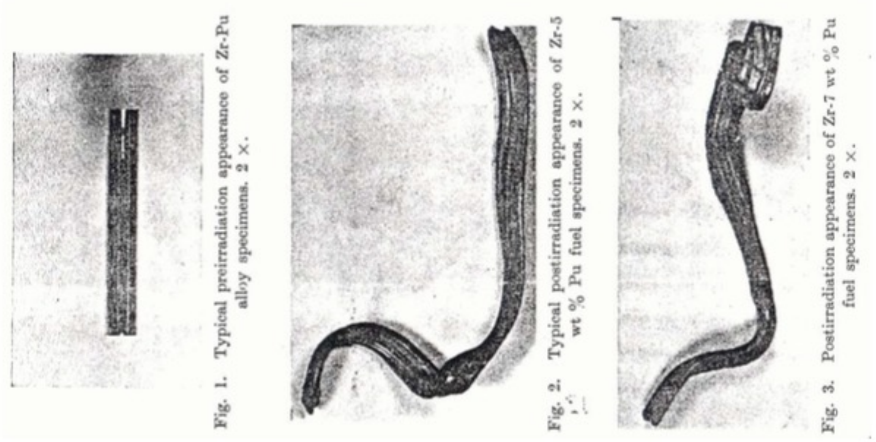


Fig. 1. Typical preirradiation appearance of Zr-Pu alloy specimens. 2 ×.

Fig. 2. Typical postirradiation appearance of Zr-5 wt % Pu fuel specimens. 2 ×.

Fig. 3. Postirradiation appearance of Zr-7 wt % Pu fuel specimens. 2 ×.

Irradiation Growth

- Three stages of irradiation growth
  - Initial rapid growth to small strains – defect generation
  - Slow growth – defect accumulation, gets skipped in cold worked material
  - Accelerated or breakaway growth, caused by formation of vacancy loops on the basal plane
- Irradiation growth is affected by:
  - Fluence, Cold work, Texture, Irradiation temperature, Material chemistry, Hydrogen effects

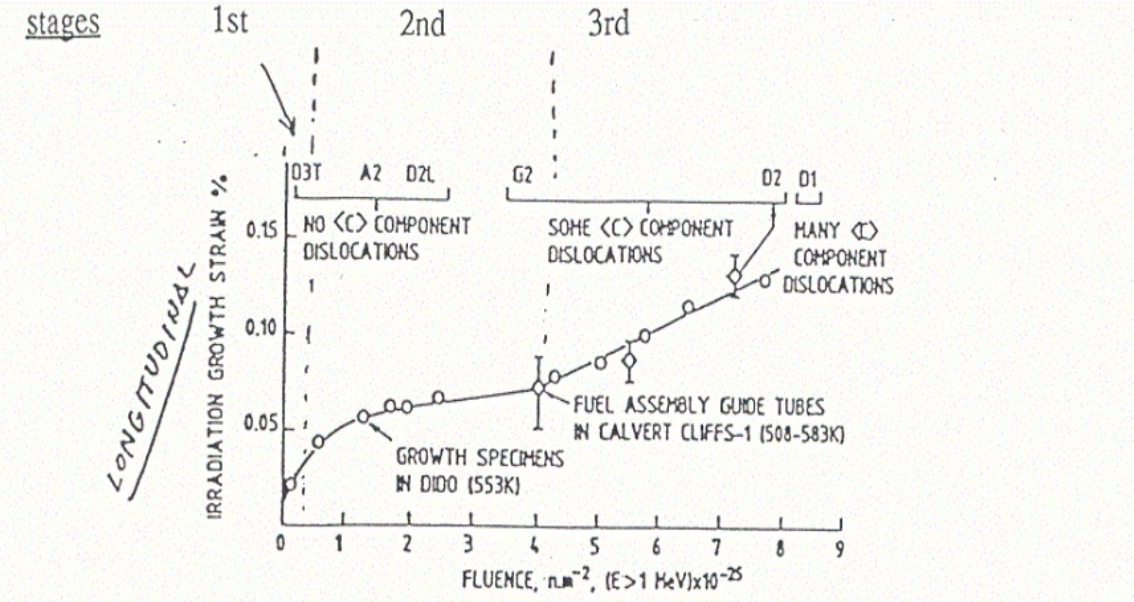


FIG. 11—*Irradiation growth in annealed Zircaloy at 550 to 580 K, showing accelerating growth at 4 ×* 10^{25} *n/m², E > 1 MeV* [61].

Irradiation Growth

- Higher temperature leads to earlier breakaway and higher growth
- Higher temperature adds mobility to defects, allowing them to diffuse faster, form clusters, and exacerbate the defect cluster induced anisotropic growth