Nuclear Fuel Performance

NE-533

Spring 2025

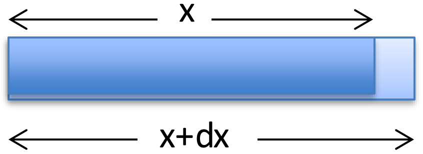
Last Time

- Fission gas release
  - experiments fall into post-irradiation annealing, or in-pile
- Showed Booth model equations for post-irradiation annealing and in-pile release
- Forsberg-Massih 2-stage FGR model
- Three components of diffusion
  - intrinsic, RED, RDD
- Swelling/dimensional change
  - densification, thermal expansion, solid/gas fission product swelling, creep

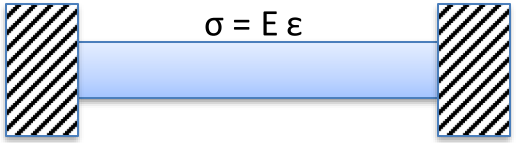
FUEL SWELLING/DIMENSIONAL CHANGE CONT…

Creep

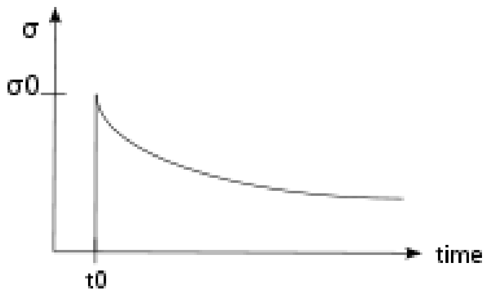
- Creep is a general mechanism for plastic deformation that occurs over time when $\sigma < \sigma_y$
- Consider a heated metal beam so it expands some distance dx



- We then fix it between two walls and let it cool down
- Because $\sigma < \sigma_y$, that stress remains constant



- In creep, defect diffusion is induced by the stress to cause permanent deformation and reduce the stress
- Therefore, creep
  - Occurs over time
  - Increases with increasing number of diffusing defects
    - High temperature (**thermal creep**)
    - Irradiation (**irradiation enhanced thermal creep**)

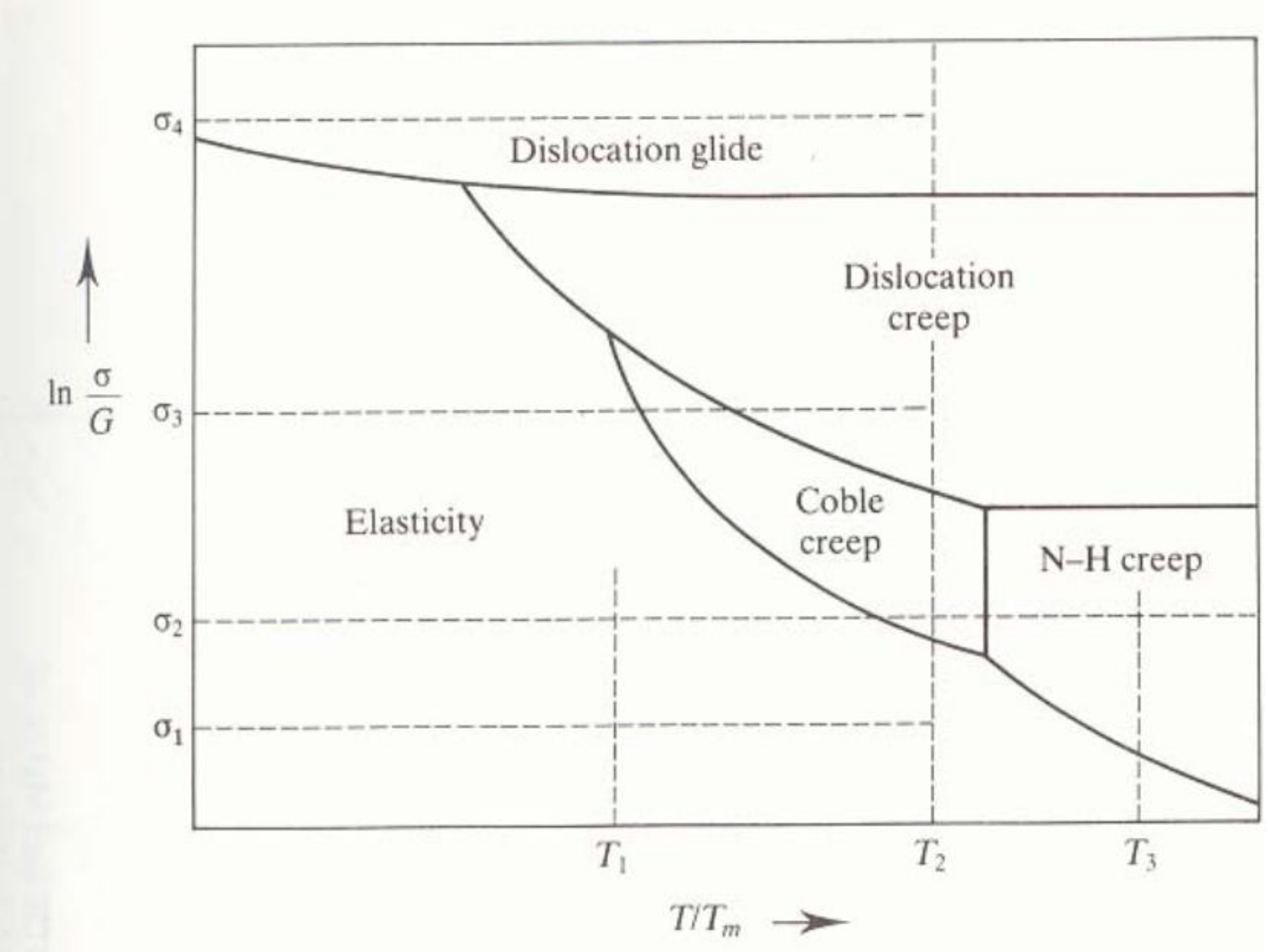


Creep

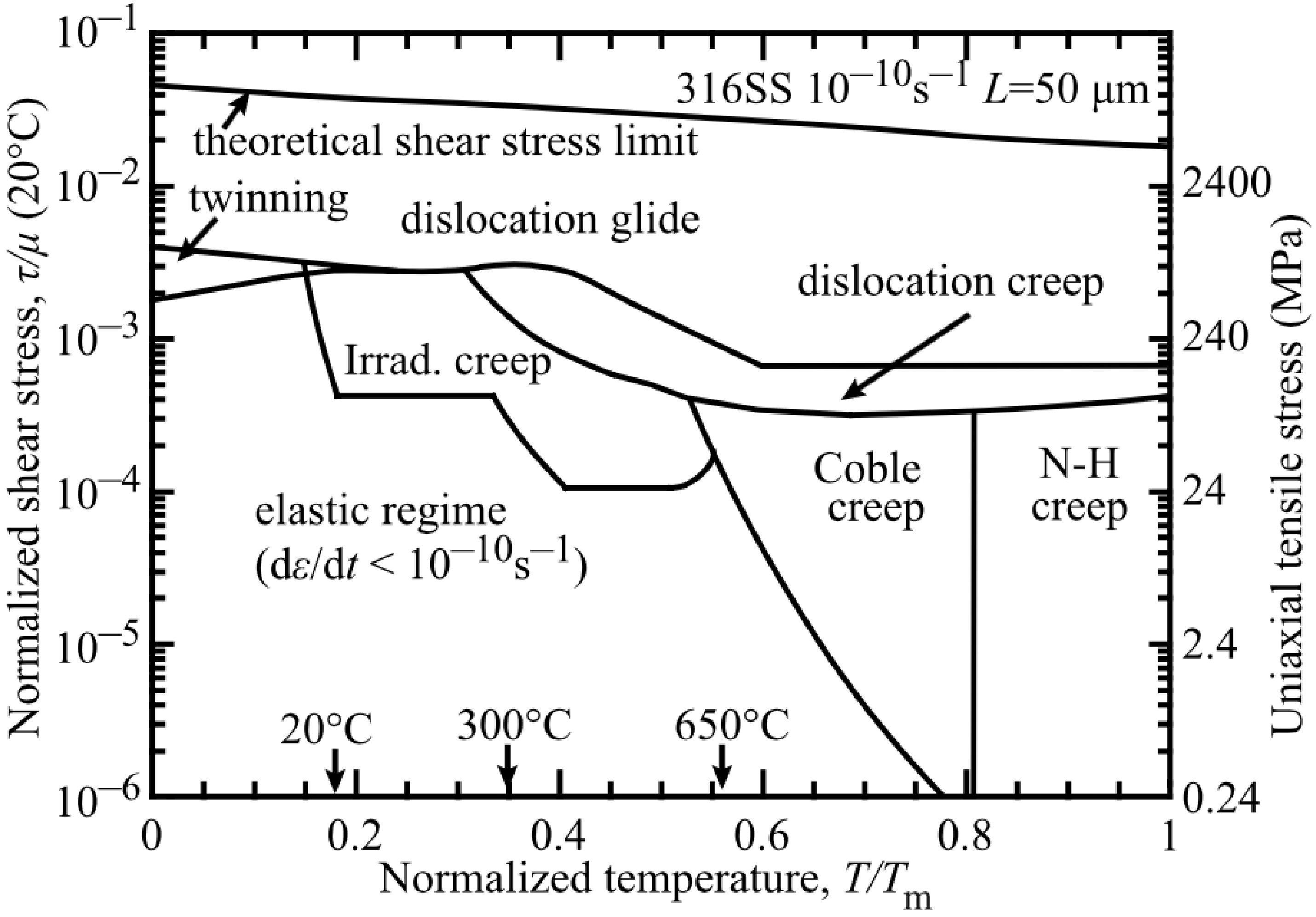
- General creep equation: $\dot{\epsilon} = \frac{C\sigma^m}{D_{gr}^b} e^{\frac{-Q}{k_b T}}$

- Creep can be caused by various microstructural mechanisms
- Bulk Diffusion (Nabarro-Herring creep)
  - Atoms diffuse (high T), causing grains to elongate along the stress axis
  - Q = Q(self diffusion), m = 1, and b = 2
- Grain boundary diffusion (Coble creep)
  - Atoms diffuse along grain boundaries to elongate the grains along the stress axis
  - Q = Q(grain boundary diffusion), m = 1, and b = 3
- Dislocation creep (power law creep)
  - Dislocations glide under a high stress
  - Dislocations climb due to defects to avoid obstacles
  - Q = Q(self diffusion), m = 4–6, and b = 0

Different creep mechanisms are active for different combinations of stress and temperature



The behavior of creep changes in irradiated materials

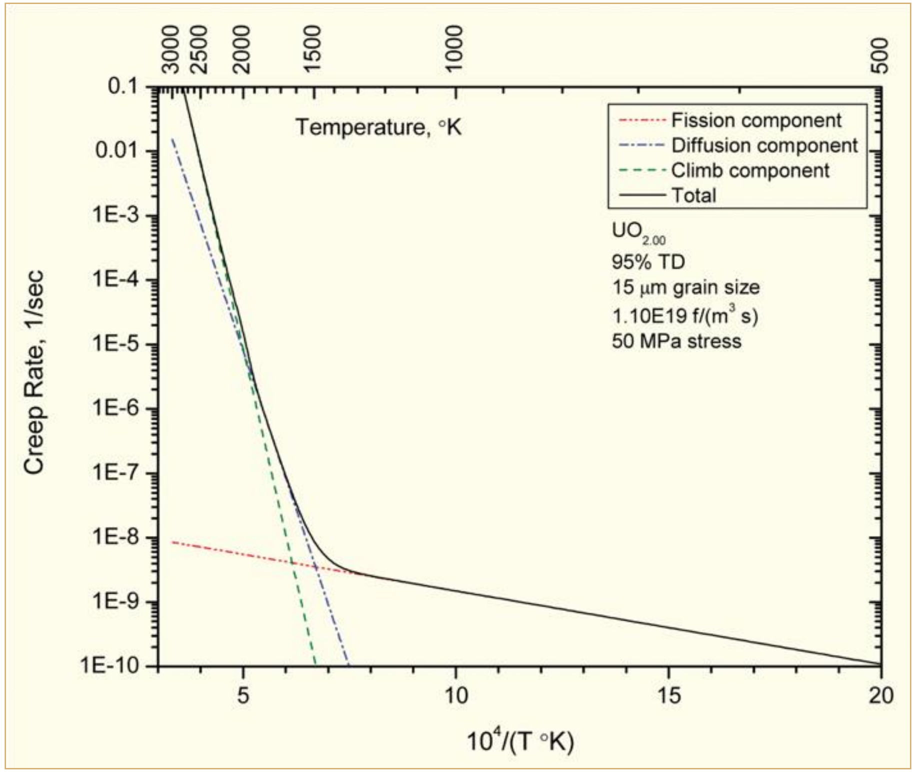


Irradiation and Creep

- Irradiation accelerates creep, causing it to be significant at lower temperatures
- Irradiation typically has little effect on diffusional creep (but can increase defect concentrations)
- Primary impact is accelerating dislocation creep in structural/cladding materials
- The dislocation creep rate can be written as $\dot{\varepsilon} = \rho_d^m b \mathrm{v}_d$
  - $\rho_d{}^m$ is the density of mobile dislocations
  - b is the burgers vector
  - v_d is the dislocation velocity
- Gliding dislocations quickly get pinned by obstacles
- As the dislocations absorb defects created by irradiation, they climb to different slip planes to avoid the obstacles
- More interstitials are absorbed than vacancies due to the higher sink strength for interstitials

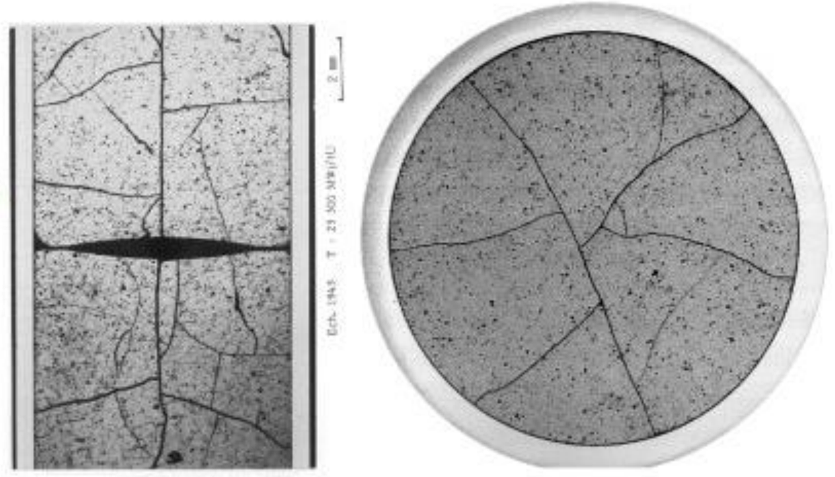
Fuel Creep

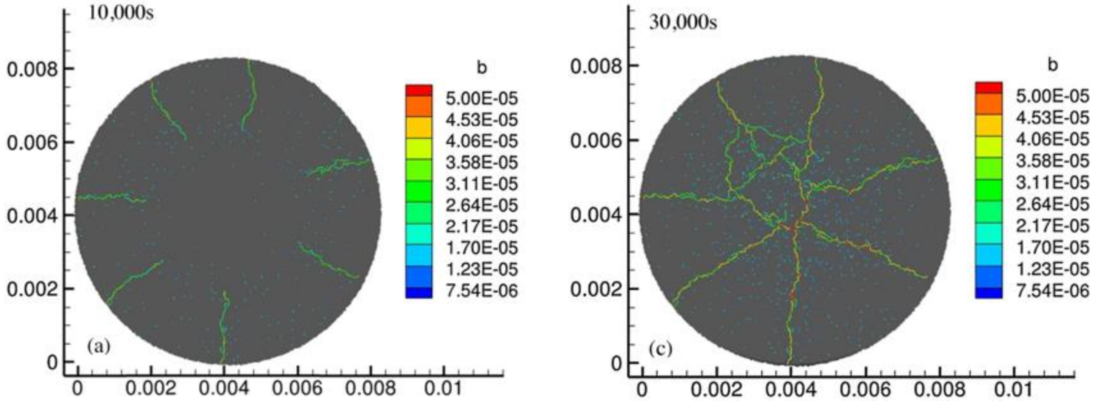
- Like other materials, the fuel also undergoes creep
- The fuel creep (In UO2) is a combination of diffusion creep and irradiation creep
- It is expected that fuel creep plays a major role in dimensional change in metallic fuels, largely via N-H and Coble creep, but still unproven experimentally and no good creep models exist for metallic fuels



Fracture

- UO_2 pellets fracture during changes in temperature due to large thermal stresses
- Fracture results in:
  - Increased gap reduction
  - Reduced thermal conductivity
  - Increased avenues for fission gas release
- Fracture has been typically modeled in two ways:
  - Empirical relocation model that is a function of burnup
  - Semi-empirical smeared cracking model
- Modern methods provide means of modeling discrete cracks

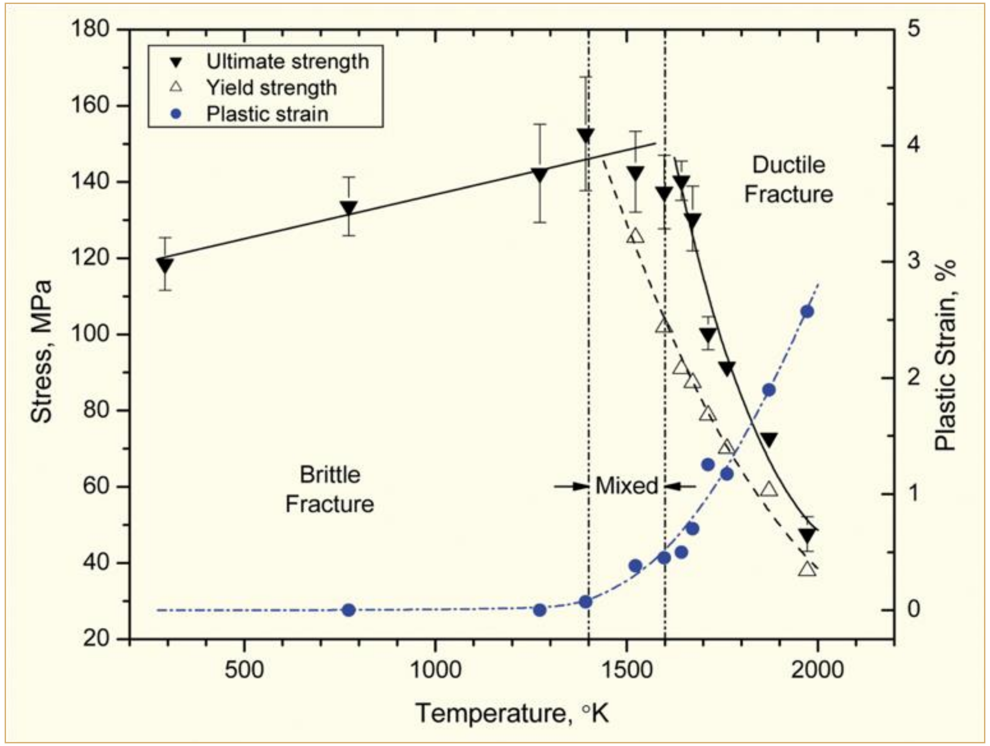




- Radial cracks partially penetrate the pellet during temperature increase
- Full cracking occurs when the temperature decreases

Fracture

- The fracture behavior of the fuel is fairly complicated
- Fracture strength varies with grain size (G)
  - $\sigma_{frac} = G^{-m}\sigma_{frac,\ ref}$, m = 0.04 – 0.05 (vs. m ~ 0.5 for metal)
  - Increasing grain size from 10 μm to 100 μm reduces σ_{frac} by ~10%
- Ductility transition temperature is lower in-reactor than in thermal tests
- Fracture strength is ~10 × higher in compression than in tension
- Load-deformation behavior strongly affected by creep under in-reactor conditions



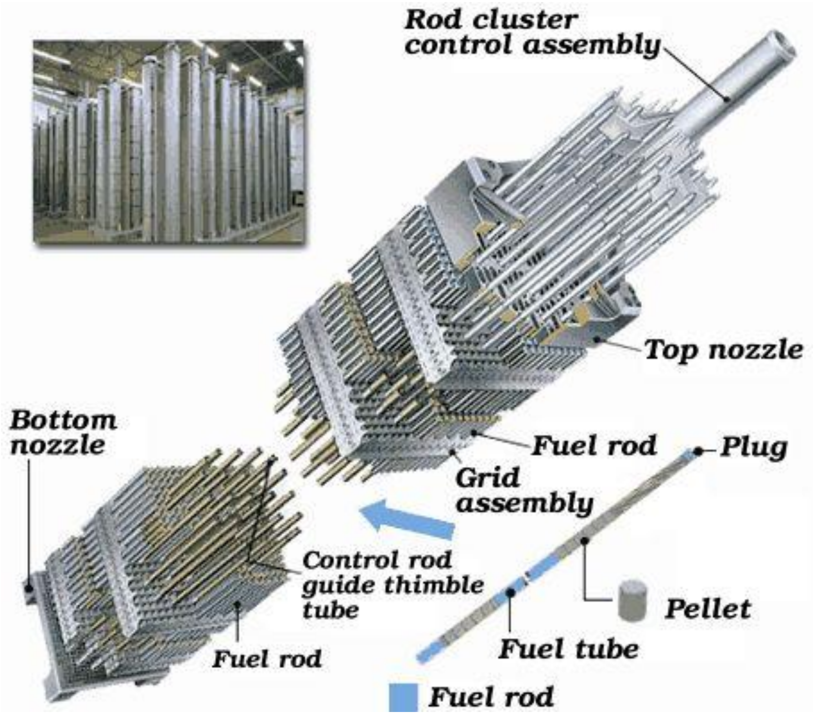
Summary

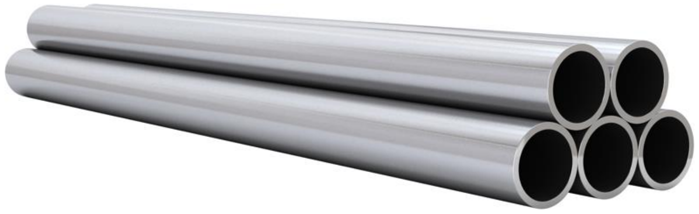
- Many materials models for fuel are empirical and correlated to burnup
- Fuel pellets change shape due to
  - Thermal expansion (increase in volume)
  - Densification (decrease in volume)
  - Swelling (increase in volume)
  - Creep (volume stays the same)
- Fracture also decreases the gap, as fractures pieces shift outward

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

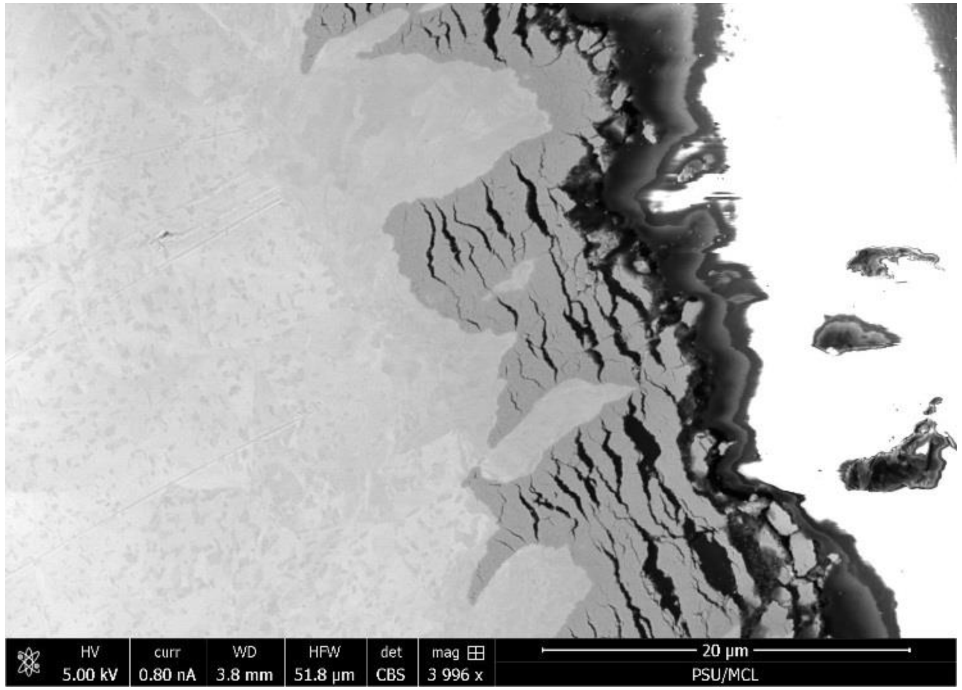
- Low neutron cross section
- Corrosion resistance in 300˚C water
- Resistance to void swelling
- Adequate mechanical properties
- Good thermal conductivity
- Affordable cost
- Available in large quantities

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



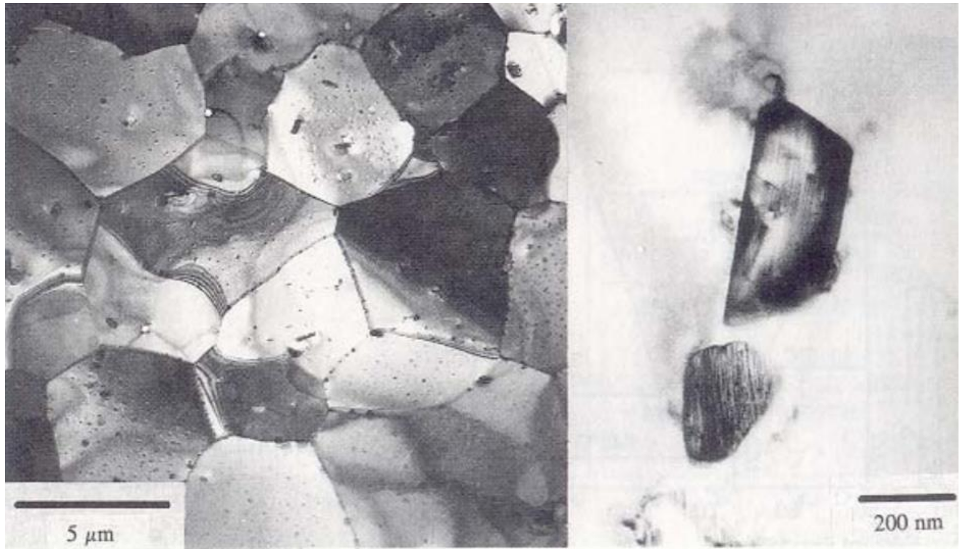
Commercial Zr Alloys in PWRs

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

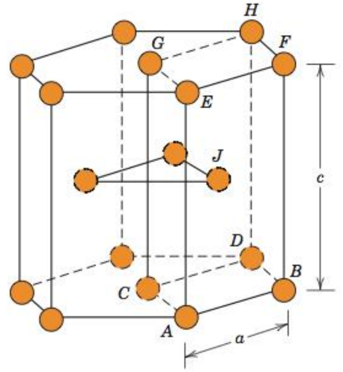
Alloying Elements

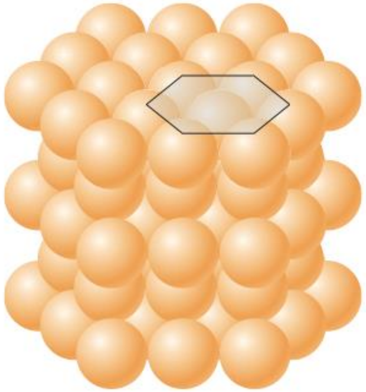
- The alloying elements impact the microstructure of the material
- The intermetallic precipitates significantly improve the oxidation behavior
- In Zircaloy 2, the precipitates are
  - $Zr(Cr, Fe)_2$
  - $Zr_2(Ni, Fe)$
- In Zircaloy 4, the precipitates are
  - $Zr(Cr, Fe)_2$
- Phosphides (Zr_3P) and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

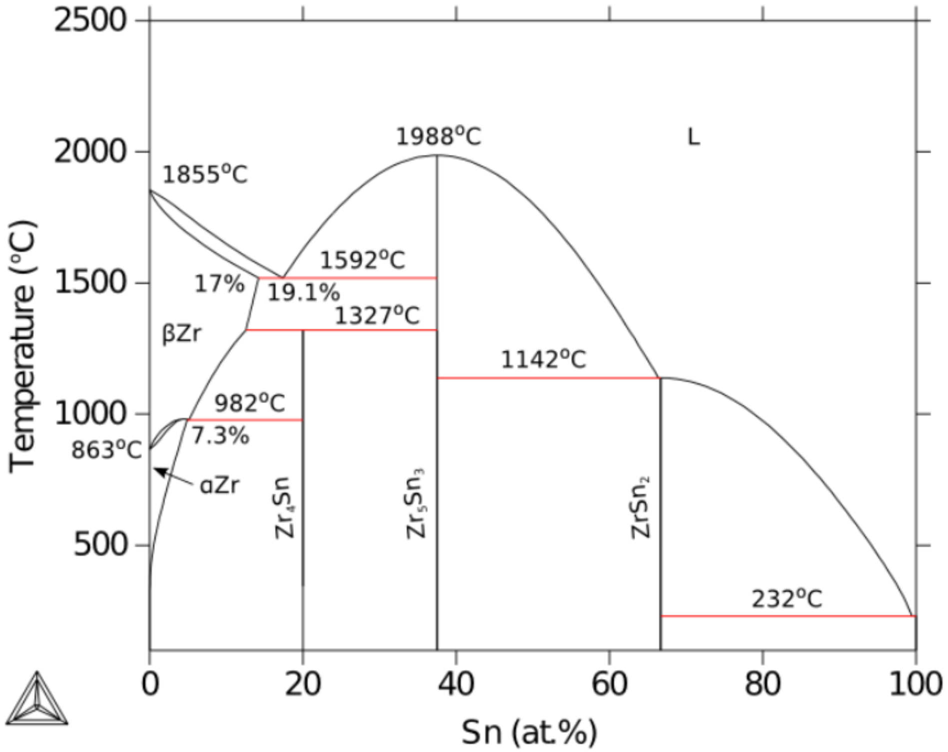


Zirconium Phases

- 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: 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

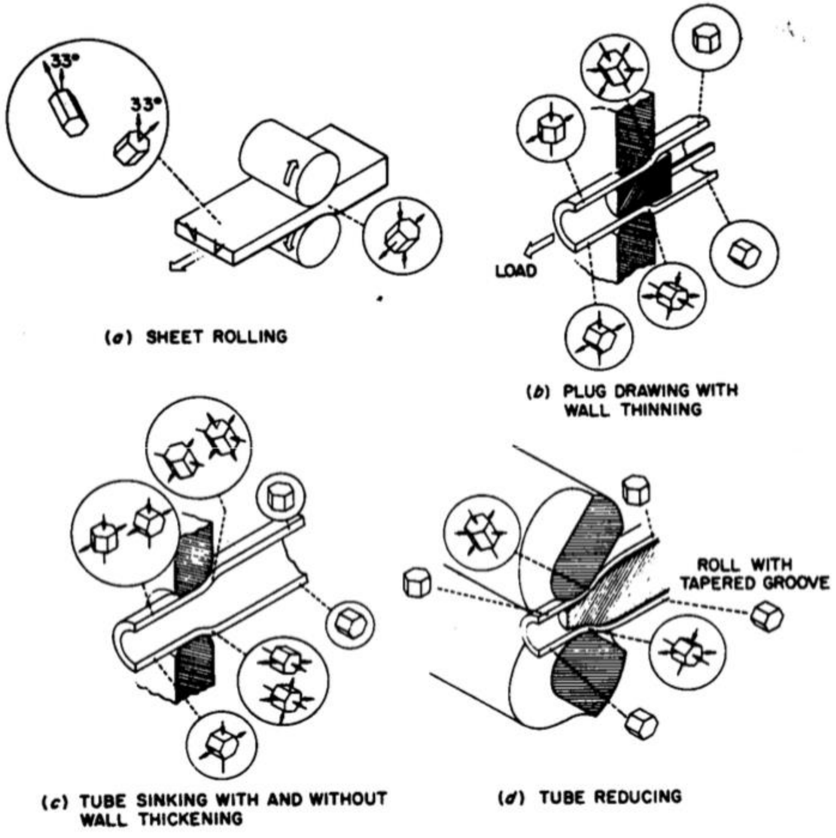


FIG. 48. Strain states for various types of fabrication of Zircaloy-2 [109].

Zr alloy microstructures

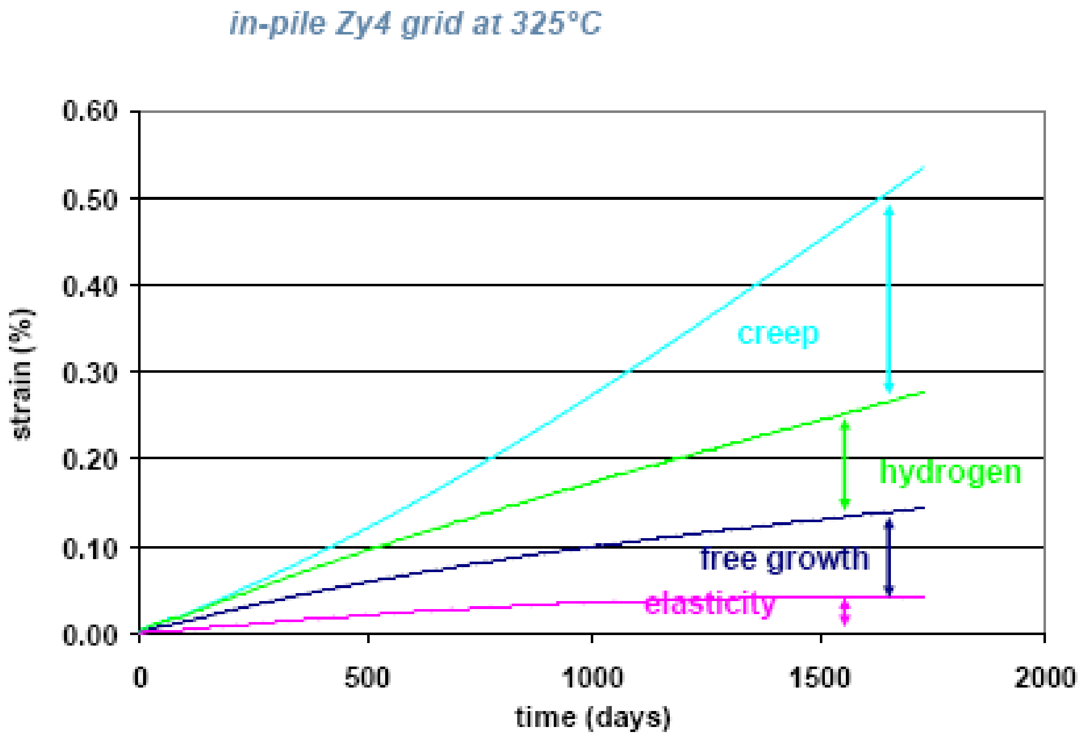
Fully recrystallized after quench

Stress-relieved microstructure

Zirconium Creep and Growth

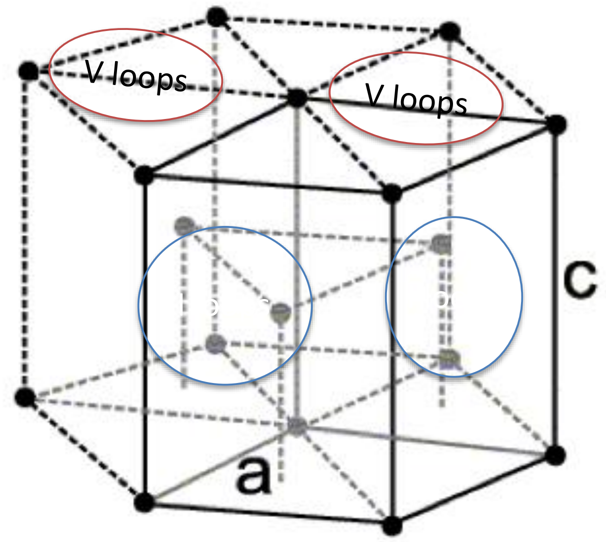
- The cladding undergoes a very large amount of irradiation during its life, the main effects are:
  - Interstitial loops form on prismatic planes
  - Later in the reactor exposure, vacancy loops form on the basal plane
- The overall number of displacements to the cladding can be calculated with the 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)

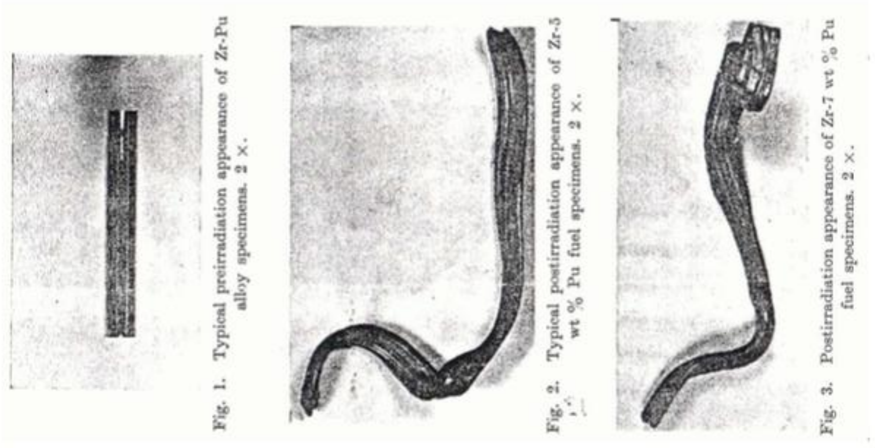
$$\nu(T) = \frac{\kappa E_D}{2E_d} = \frac{\kappa(T-\eta)}{2E_d},$$



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





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

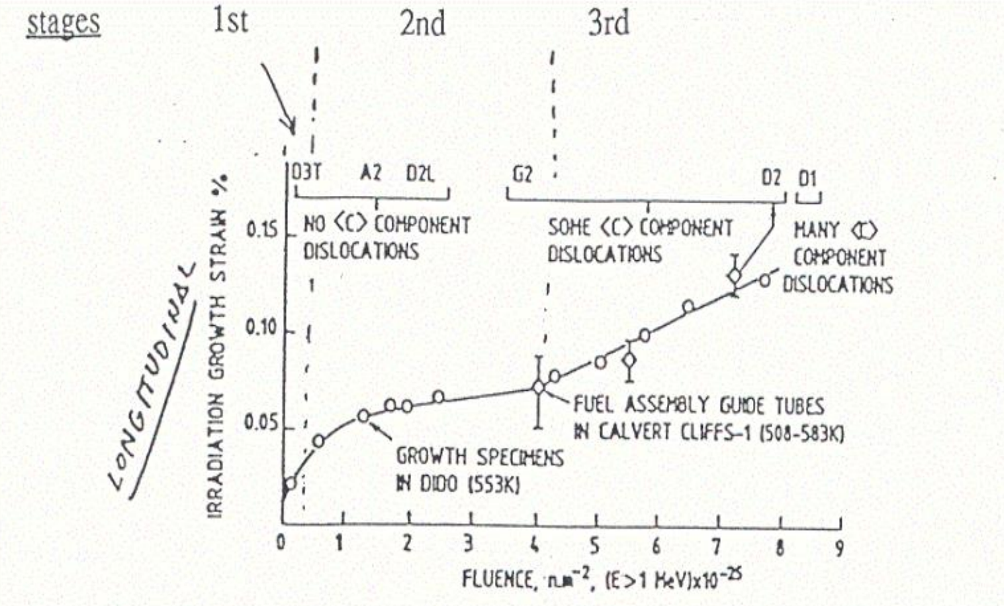


FIG. 11—*Irradiation growth in annealed Zircaloy at 550 to 580 K, showing accelerating growth at 4 ×* 10^{25} *n/m²*, E > *1 MeV* [61].

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