

Zirconium Cladding

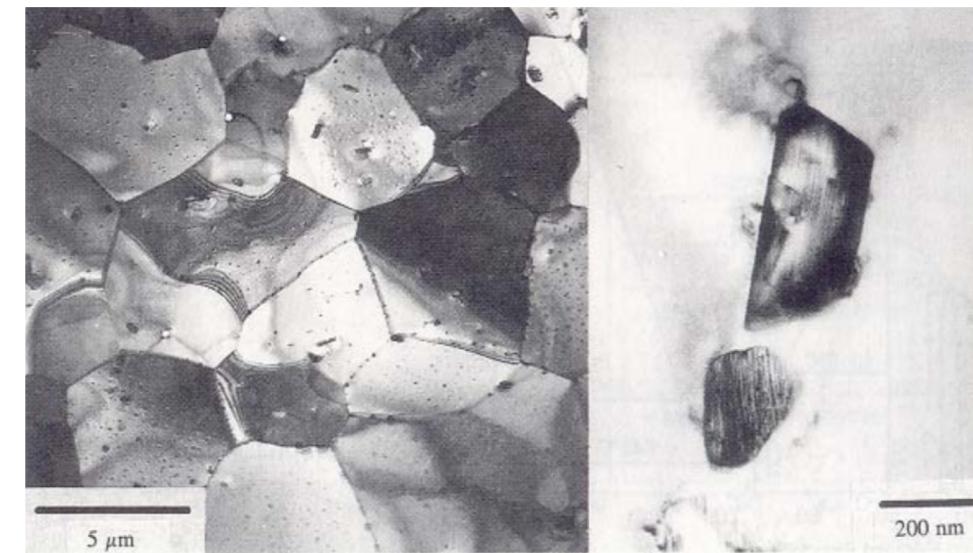
NE 591

Last time

- The fuel thermal conductivity decreases with burnup
- The NFIR model is a fairly accurate empirical model of the fuel thermal conductivity
- Researchers are working to develop materials models for the fuel and cladding that are mechanistic rather than empirical and that are based on the evolution of the microstructure rather than the burnup
- Zirconium alloys used as cladding
- Zirconium fabricated into an alloy to increase corrosion resistance
- There are a number of Zr alloys utilized in PWRs as cladding and structural materials

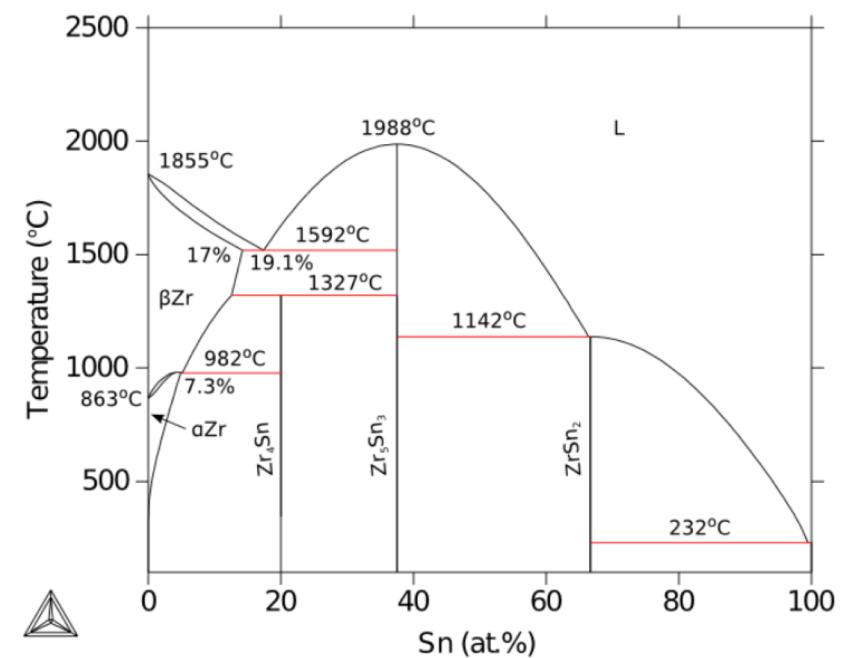
Alloying Elements

- The alloying elements impact the microstructure of the material
- The intermetallic precipitates significantly improve the oxidation behavior
- In Zircaloy 2, the precipitates are
 - $\text{Zr}(\text{Cr, Fe})_2$
 - $\text{Zr}_2(\text{Ni, Fe})$
- In Zircaloy 4, the precipitates are
 - $\text{Zr}(\text{Cr, Fe})_2$
- 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 863°C
 - 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)
- 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

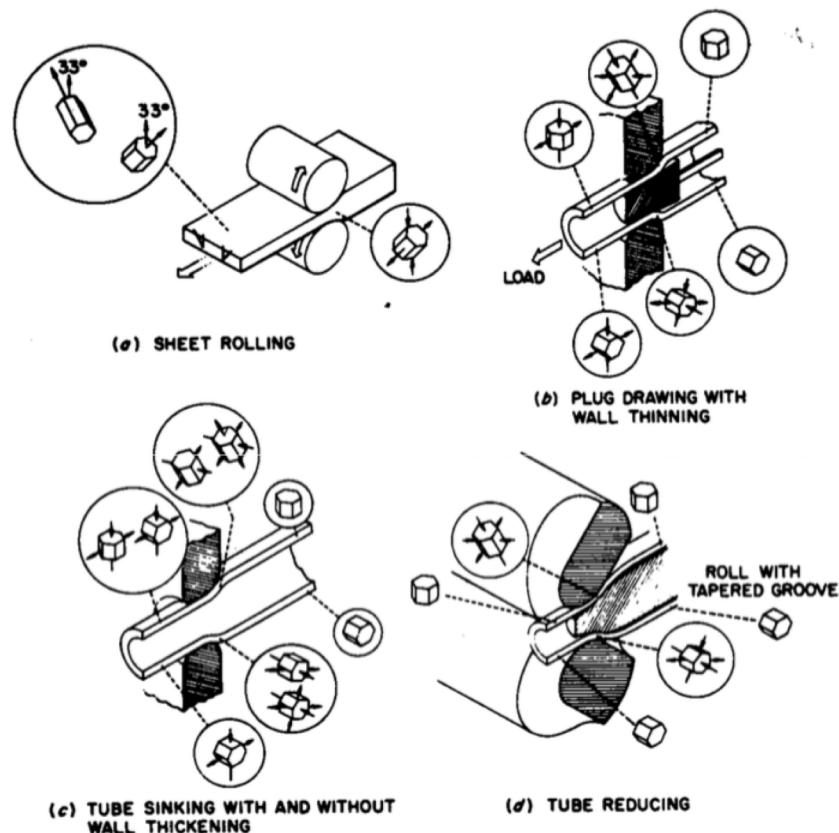
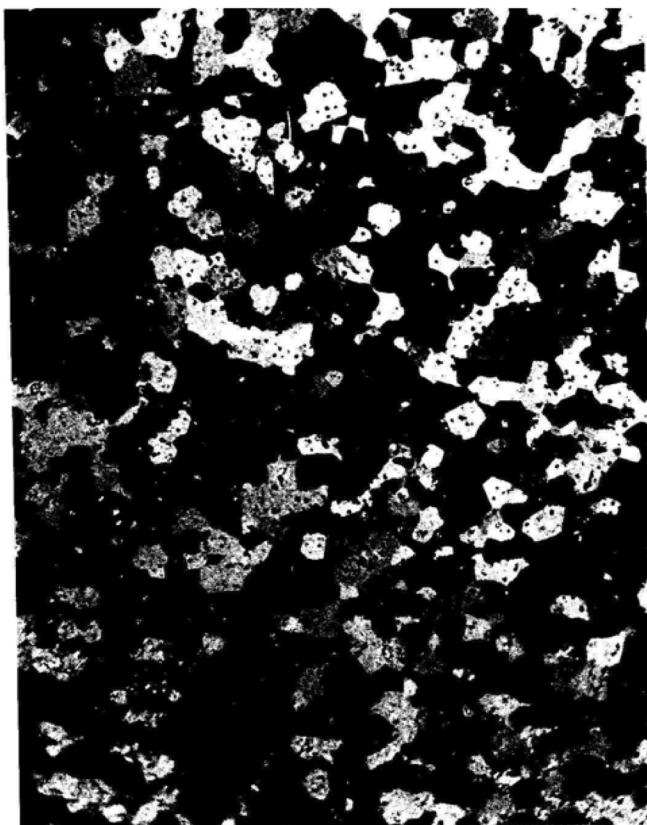


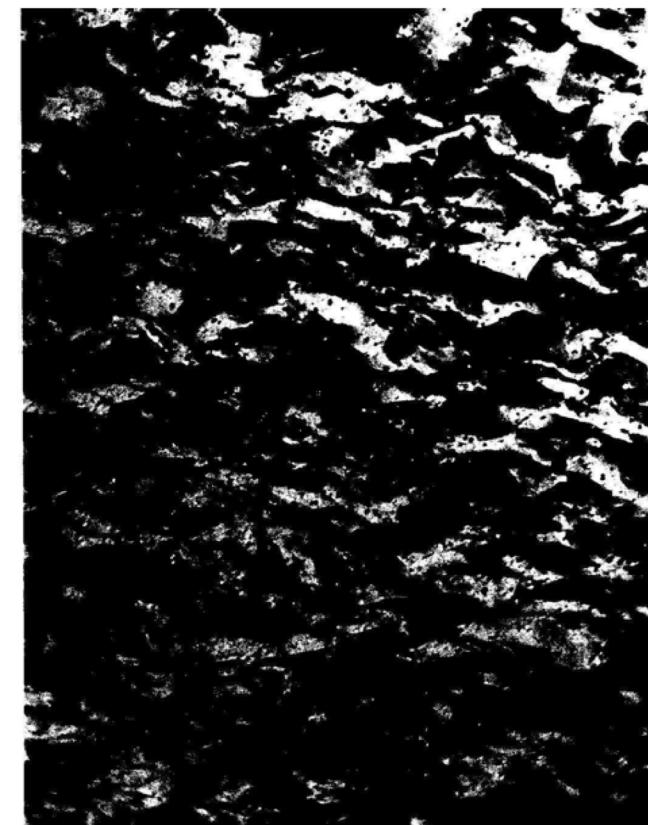
FIG. 48. Strain states for various types of fabrication of Zircaloy-2 [109].

Zr alloy microstructures

Fully recrystallized after quench

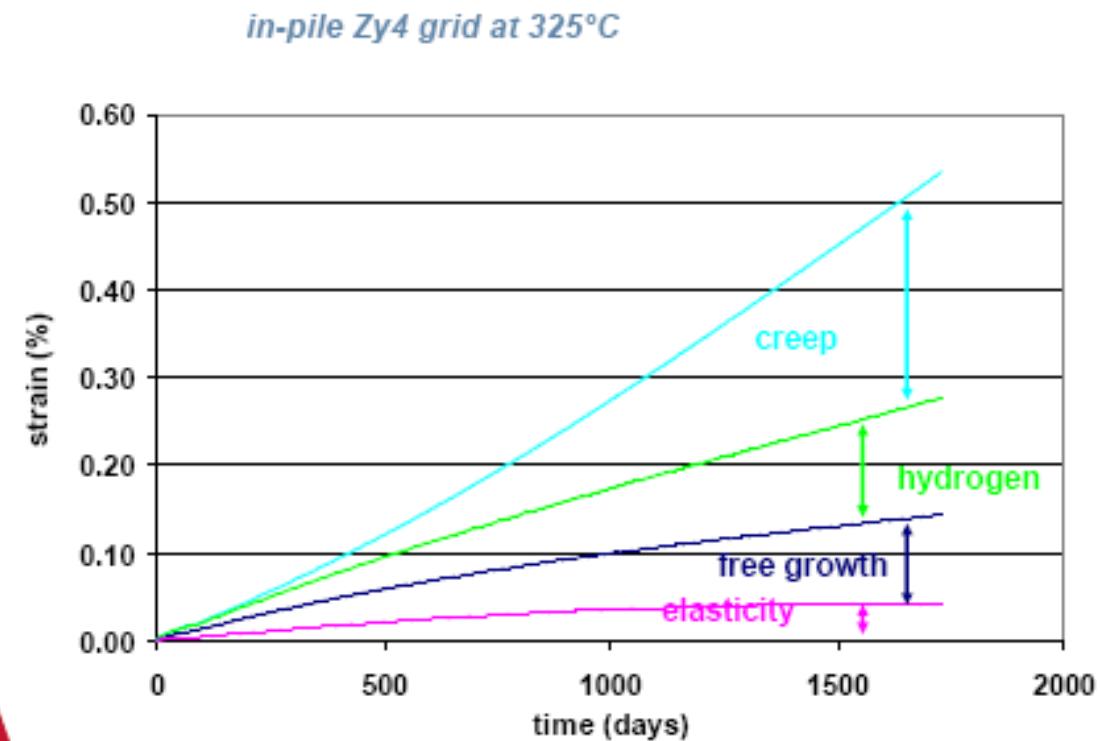


Stress-relieved microstructure



Zirconium Creep and Growth

- The cladding undergoes a very large amount of irradiation during its life, the main effects are:
 - Interstitial loops form on prism 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 $\sim 8 \times 10^{-8}$ 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)



Irradiation Growth

- Irradiation growth results from material anisotropy
- There must be anisotropy in the defect behavior within the unit cell
- There also must be 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

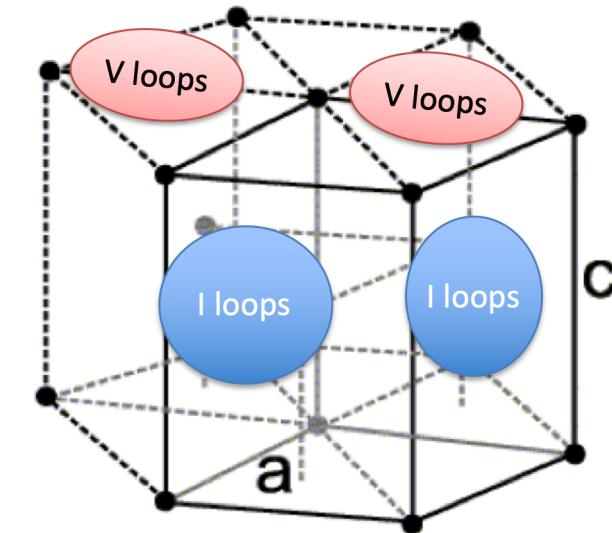


Fig. 1. Typical preirradiation appearance of Zr-Pu alloy specimens. 2 X.

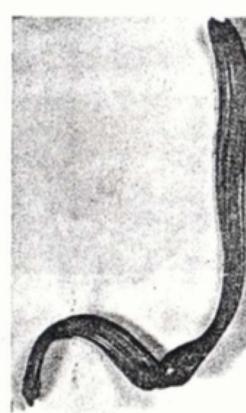


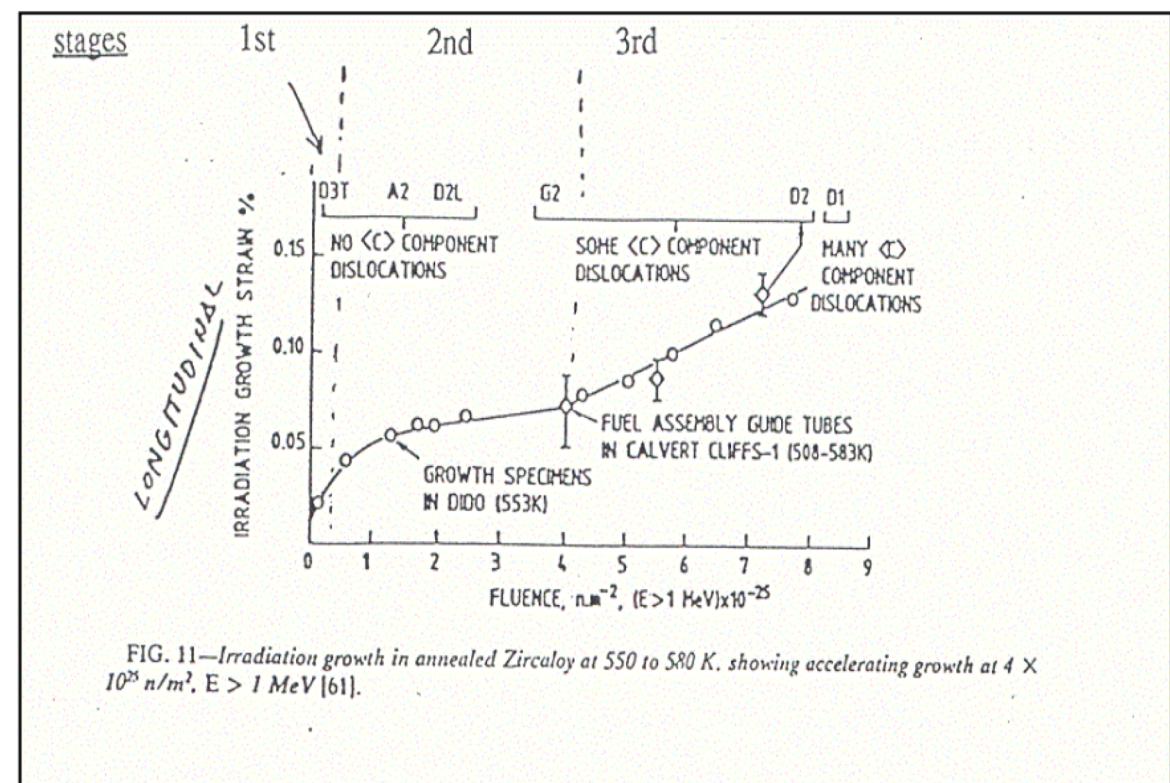
Fig. 2. Typical postirradiation appearance of Zr-5 wt % Pu fuel specimens. 2 X.



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

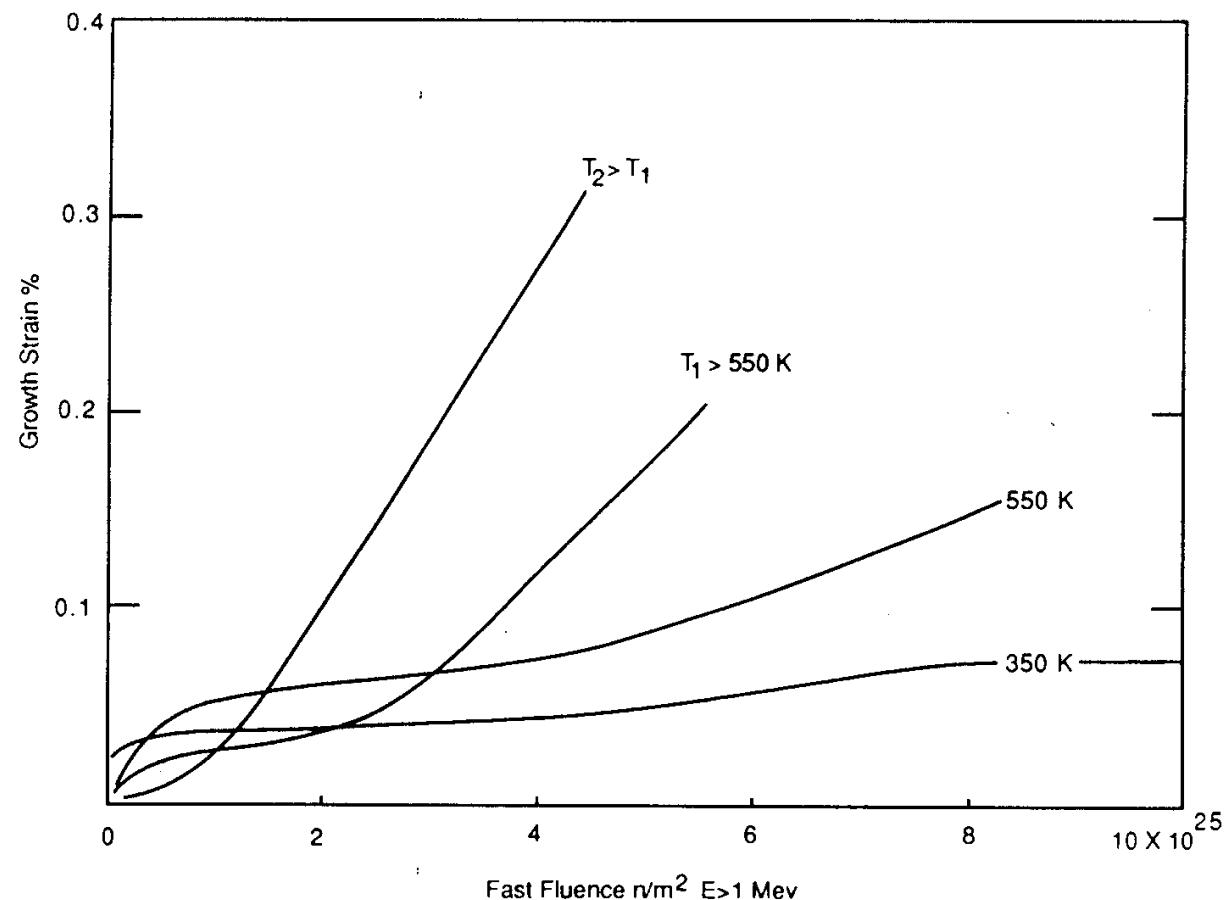
Irradiation Growth

- Three stages of irradiation growth
 - Initial rapid growth to small strains
 - Slow growth, 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



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



Creep

- Empirical models have been developed for thermal and irradiation creep of Zircaloy
- Both based on the Von Mises stress

$$\sigma_m = \sqrt{\frac{1}{2} ((\sigma_{11} - \sigma_{22})^2 + (\sigma_{22} - \sigma_{33})^2 + (\sigma_{33} - \sigma_{11})^2 + 6(\sigma_{12}^2 + \sigma_{23}^2 + \sigma_{31}^2))}$$

- Thermal Creep $\dot{\epsilon}_{ss} = A_0 \left(\frac{\sigma_m}{G} \right)^n e^{\left(\frac{-Q}{RT} \right)}$
 - With $A_0 = 3.14 \times 10^{24}$ (1/s); shear modulus $G = 4.2519 \times 10^{10} - 2.2185 \times 10^7 T$ Pa; $n = 5$; $Q = 2.7 \times 10^5$ J/mol
- Irradiation Creep $\dot{\epsilon}_{ir} = C_0 \Phi^{C_1} \sigma_m^{C_2}$
 - ϕ is the fast neutron flux $n/(cm^2 s) = 3E11 \times LHR$ $n/(cm^2 \cdot s)$
 - Note that SRA stands for stress relief annealed
 - RXA for recrystallization annealed
 - PRXA stands for partially recrystallization annealed

Clad Type	C_0	C_1	C_2
SRA	3.557×10^{-24}	0.85	1.0
RXA	1.654×10^{-24}	0.85	1.0
PRXA	2.714×10^{-24}	0.85	1.0
ZIRLO	2.846×10^{-24}	0.85	1.0

Creep Example

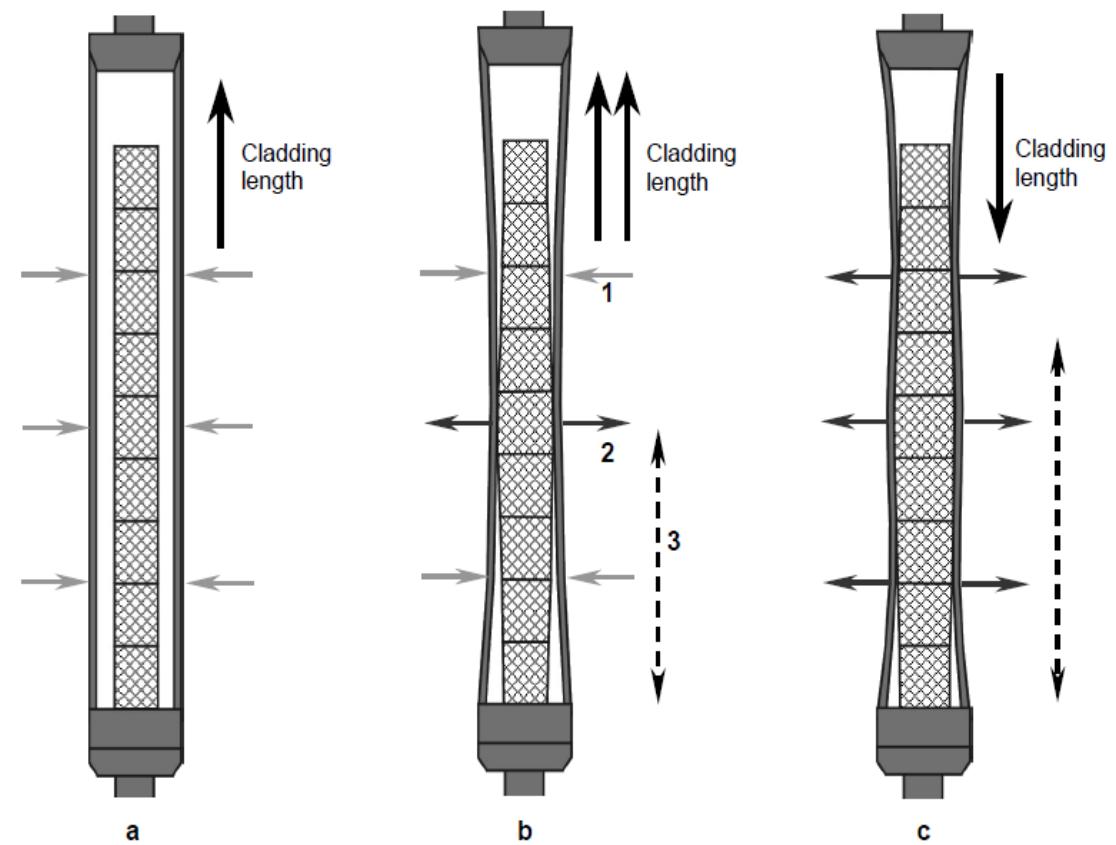
- Consider an SRA cladding tube at $T = 600$ K and LHR = 250 W/cm, with a stress $\sigma_m = 200$ MPa. What is the total creep strain after three years?
- First, we will calculate the thermal creep
 - $A_0 = 3.14 \times 10^{24}$ (1/s)
 - $G = 4.2519 \times 10^{10} - 2.2185 \times 10^7 T$ Pa = $4.2519e10 - 2.2185e7 * 600 = 2.92e10$ Pa
 - $Q = 2.7 \times 10^5$ J/mol, $n = 5$, $R = 8.3144598$ J/(K mol)
 - $3.14e24 * (200/2.92e4)^5 * \exp(-2.7e5/(8.3144598 * 600)) = 1.48e-10$ 1/s
- Now we will calculate the irradiation creep
 - $C_0 = 3.557e-24$, $C_1 = 0.85$, $C_2 = 1.0$
 - $\Phi \approx 3e11$ LHR = $3e11 * 250 = 7.5e13$ n/(cm² s)
 - $3.557e-24 * (7.5e13)^0.85 * 200^1 = 4.43e-10$ 1/s
- The total creep strain rate is $1.48e-10 + 4.43e-10 = 5.91e-10$ 1/s
- The total creep strain after three years is (assuming constant conditions)
 $5.91e-10 * (3600 * 24 * 365 * 3) = 0.056 = 5.6\%$ strain

$$\dot{\epsilon}_{ss} = A_0 \left(\frac{\sigma_m}{G} \right)^n e^{\left(\frac{-Q}{RT} \right)}$$

$$\dot{\epsilon}_{ir} = C_0 \Phi^{C_1} \sigma_m^{C_2}$$

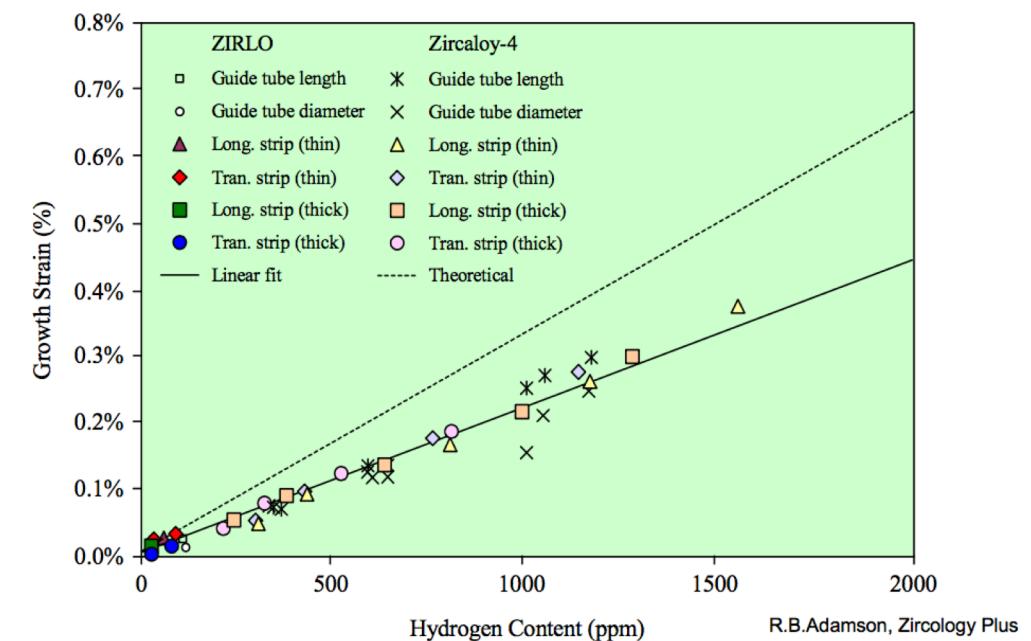
Creep

- Creep impacts fuel performance by shrinking the gap and then conforming to the pellets
- a) before “fuel-cladding” interaction;
b) start of “fuel-cladding” interaction;
c) “fuel-cladding interaction” over most of fuel column
- (1) creep down from water pressure;
(2) creep out from fuel column;
(3) fuel column axial stress



Zirconium Hydrides

- Hydrides cause size change because the hydride lattice is larger than the zirconium lattice
- There are a number of different Zr-H phases and morphologies that can present
- 0.1% H can cause 0.2% growth



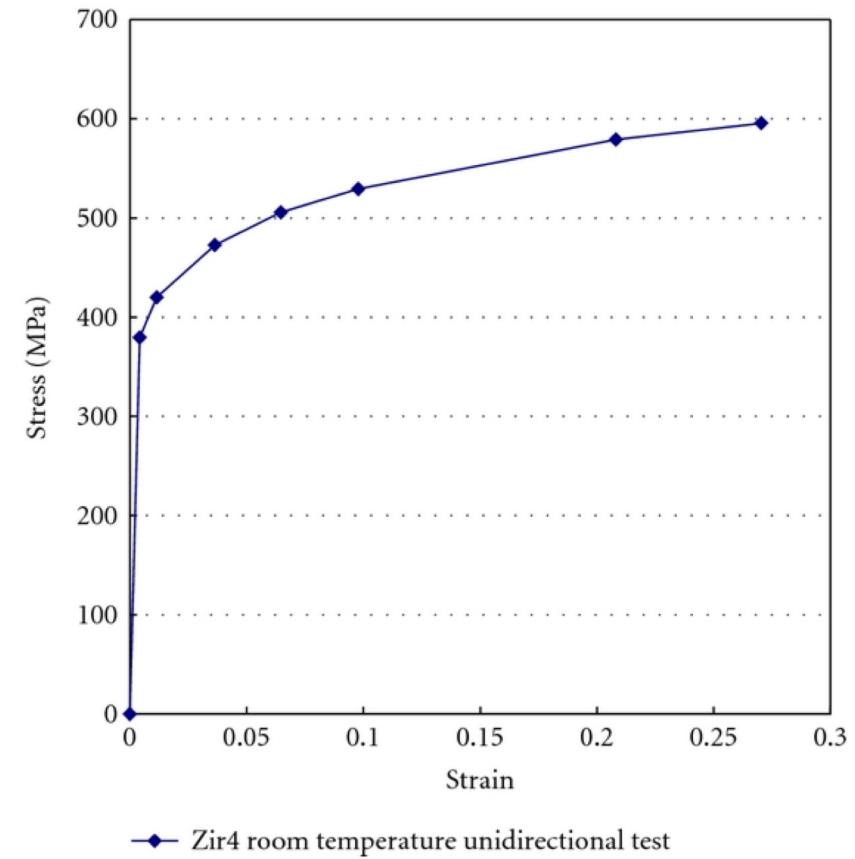
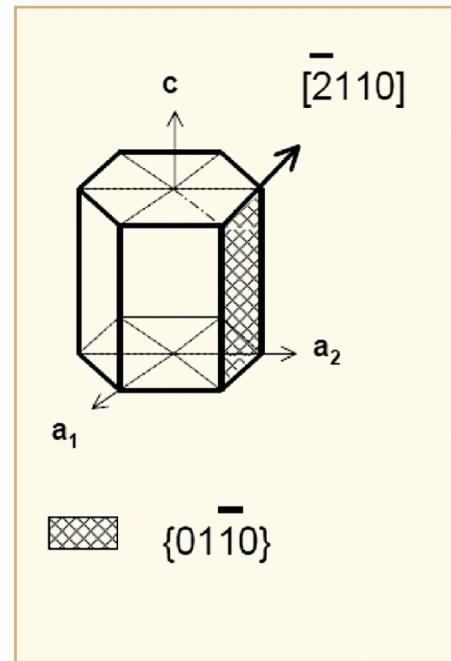
1000 ppm hydrogen can cause 0.2% dimension change

Summary of Creep and Irradiation Growth

- Growth and creep are the major mechanisms for dimensional instability in zirconium alloy cladding
- Growth results from the clustering of interstitials on prismatic planes, and eventually from the clustering of vacancies on basal planes such that the material shrinks in the axial direction
- Growth depends on the fluence, coldwork, texture, temperature, and composition

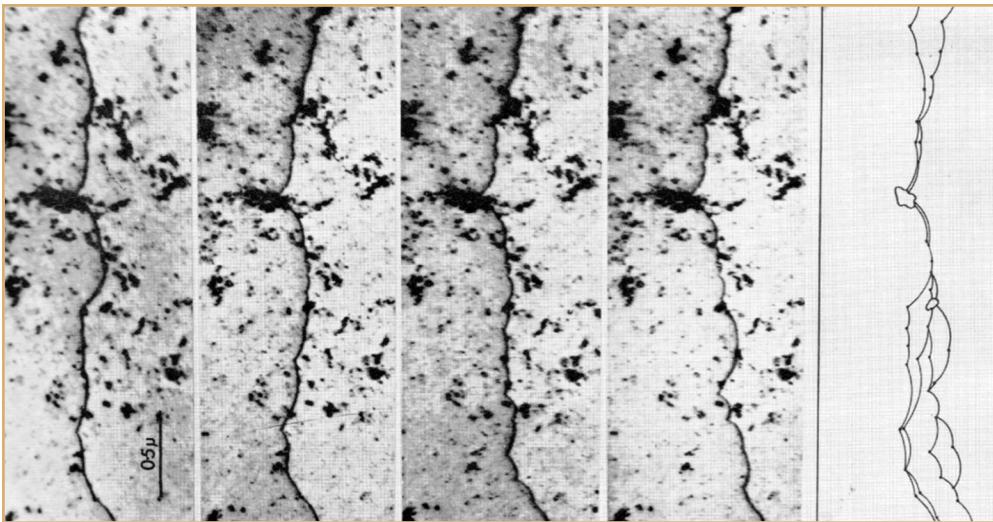
Mechanical Behavior of Zr Cladding

- Zirconium alloys plastically deform primarily due to dislocation motion on prismatic planes

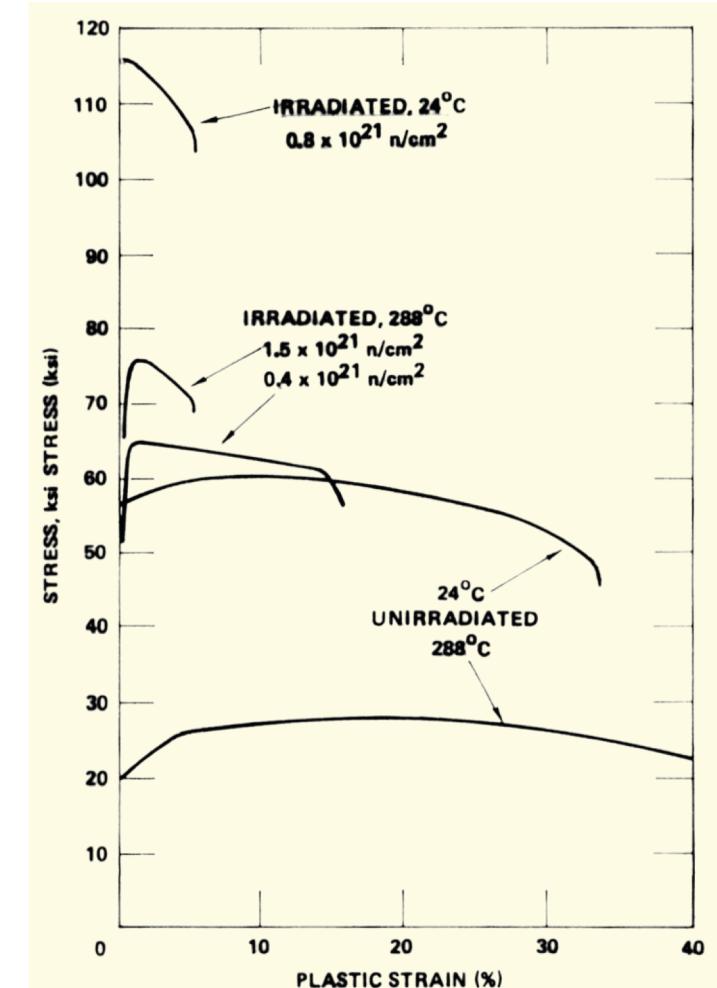


Irradiation Hardening

- Under irradiation, the interstitial loops that buildup on the prismatic planes cause hardening
- The UTS goes up with fluence, while the strain before fracture goes down

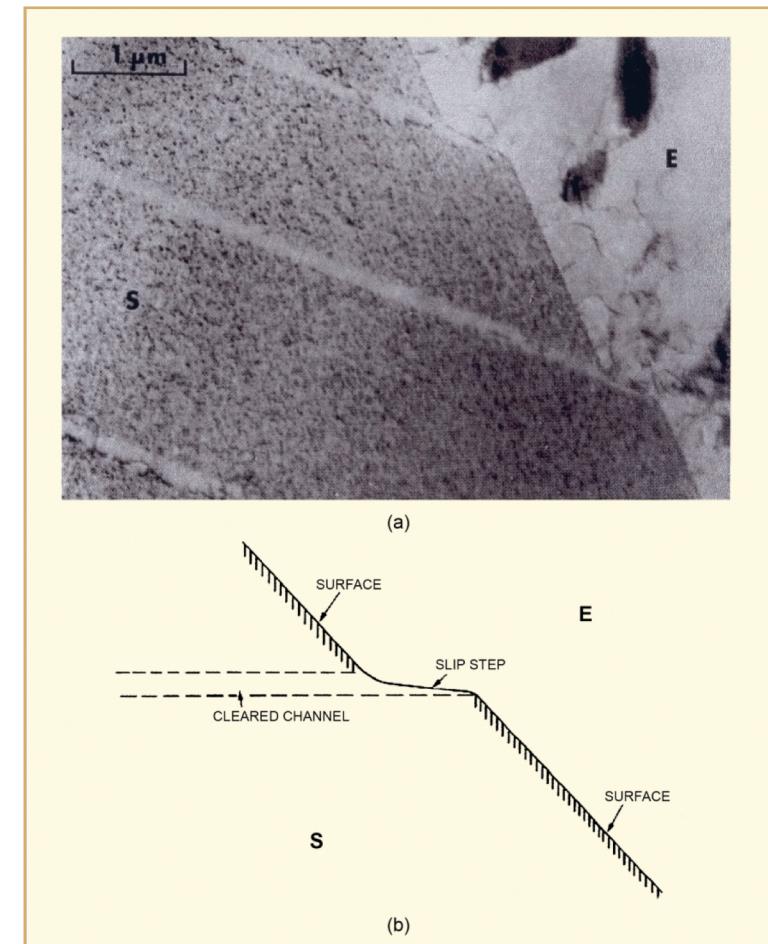


dislocation is moving



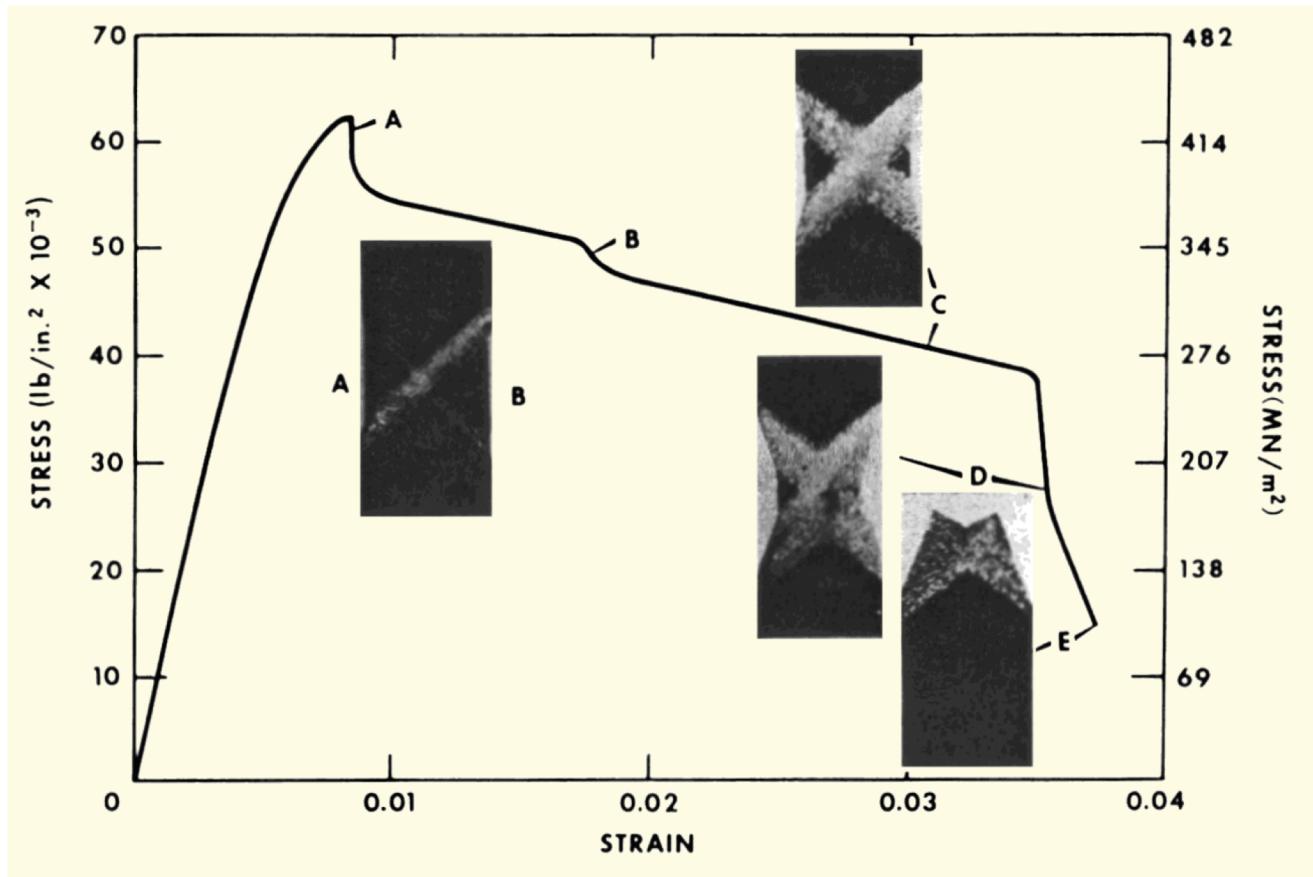
Dislocation Channels

- When stress is high enough, loops are “swept away”, forming a clean channel
- Once a channel is cleaned out, dislocations continue to move along it
- This results in lots of deformation in that one area, creating slip steps
- In HCP zircaloy, channels form on basal planes or prism planes depending on
 - Load direction
 - Temperature
 - Oxygen content in the alloy
 - Fluence



Dislocation Channels

- Channel formation is an important behavior in irradiated zircaloy
- Dislocation channel deformation occurs at high load or high fluence

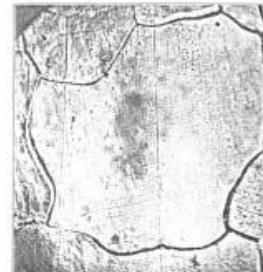
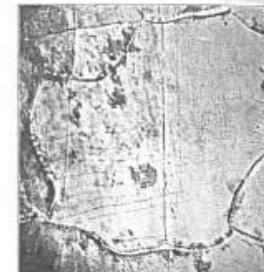
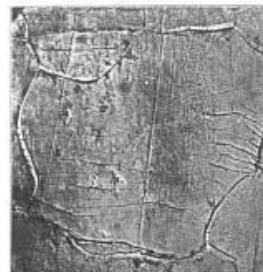
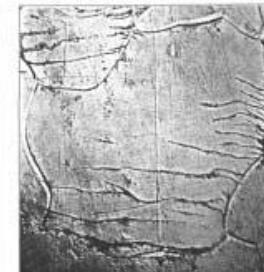
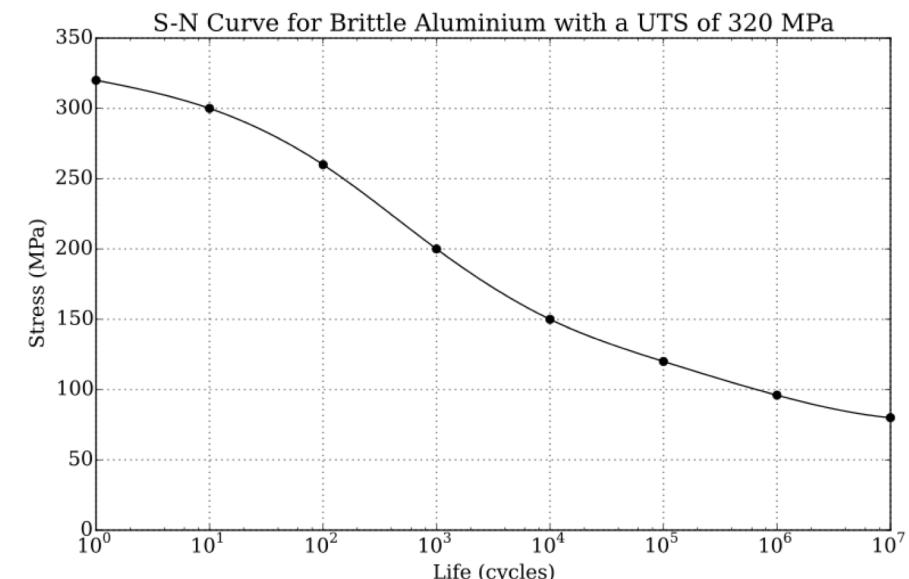


Fatigue

- Material fatigue is the weakening of a material caused by repeated applied loads
- Materials under cyclic loading can experience brittle like fracture at stresses significantly below their UTS
- It is caused by the slow propagation of microstructure damage
- The engineering tool for investigating fatigue is the S-N curve
- An S-N curve is a graph of the magnitude of the cyclic stress (S) against the logarithmic scale of cycles to failure (N).

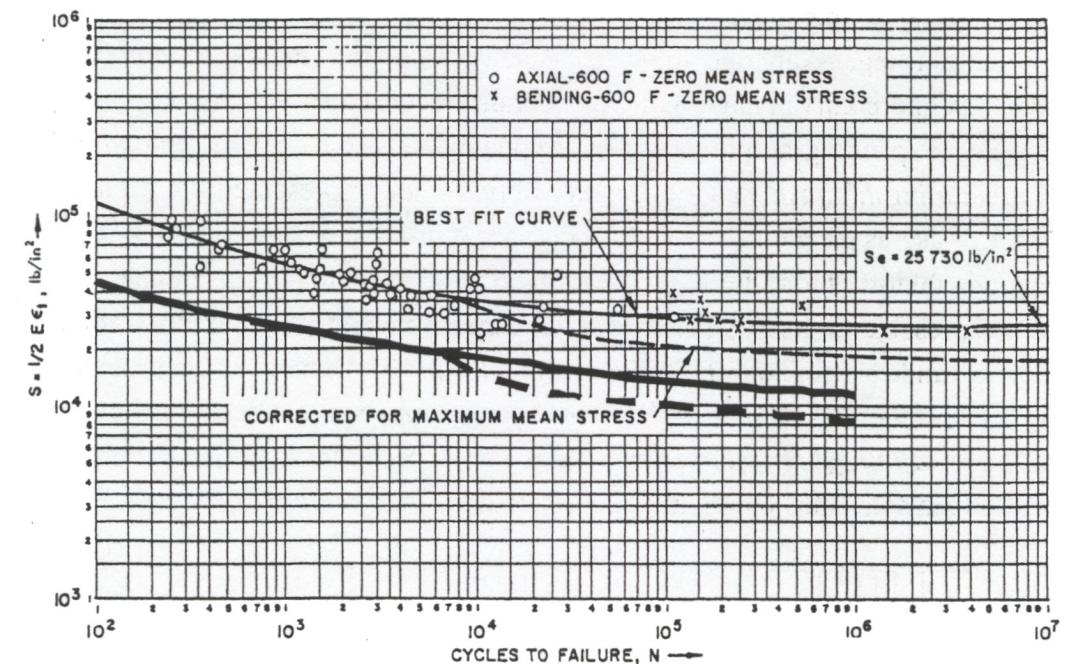
J. A. Ewing and J. C. W. Hinsley

Phil. Trans., A, vol. 300, Plate 9.

Fig. 9. Specimen after 1000 reversals of a stress of 12.4 times per sec. load. $\times 1000$.Fig. 10. Base after 2000 reversals. $\times 1000$.Fig. 11. Base after 10,000 reversals. $\times 1000$.Fig. 12. Base after 40,000 reversals. $\times 1000$.

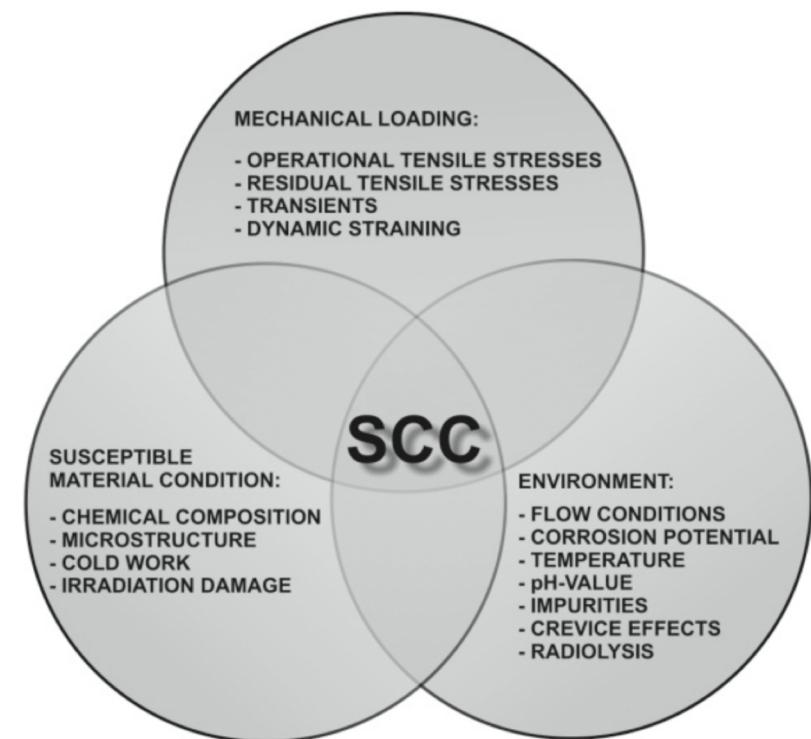
Zr Fatigue

- Zircaloy experiences fatigue like other metals
- Zircaloy has a fatigue limit at about 10^5 cycles with 180 Mpa (The UTS is >500 Mpa)
- Irradiation slightly lowers the fatigue life in the low cycle range, less than about 10^4 cycles
- Irradiation has no effect on the fatigue life on the high cycle range, greater than about 10^5 cycles
- Design must demonstrate a factor of safety of 2 on stress and 20 on cycles



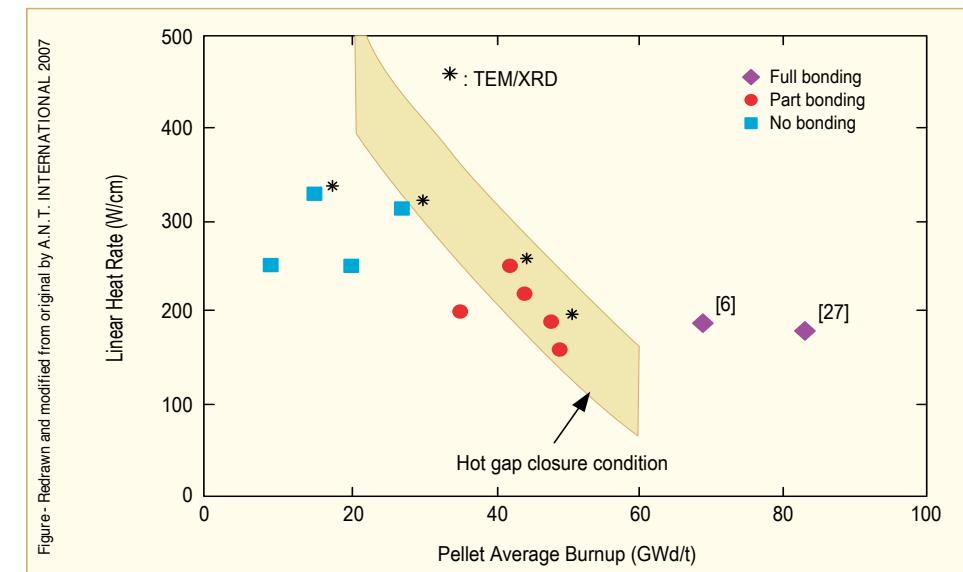
Stress Corrosion Cracking (SCC)

- SCC occurs when environmental effects accelerate fracture
- Zirconium alloy tubes do not react with water, but have SCC from fission products inside tube
 - Early studies identified iodine as a primary SCC agent
 - Later, Cd identified as potent liquid metal embrittling agent



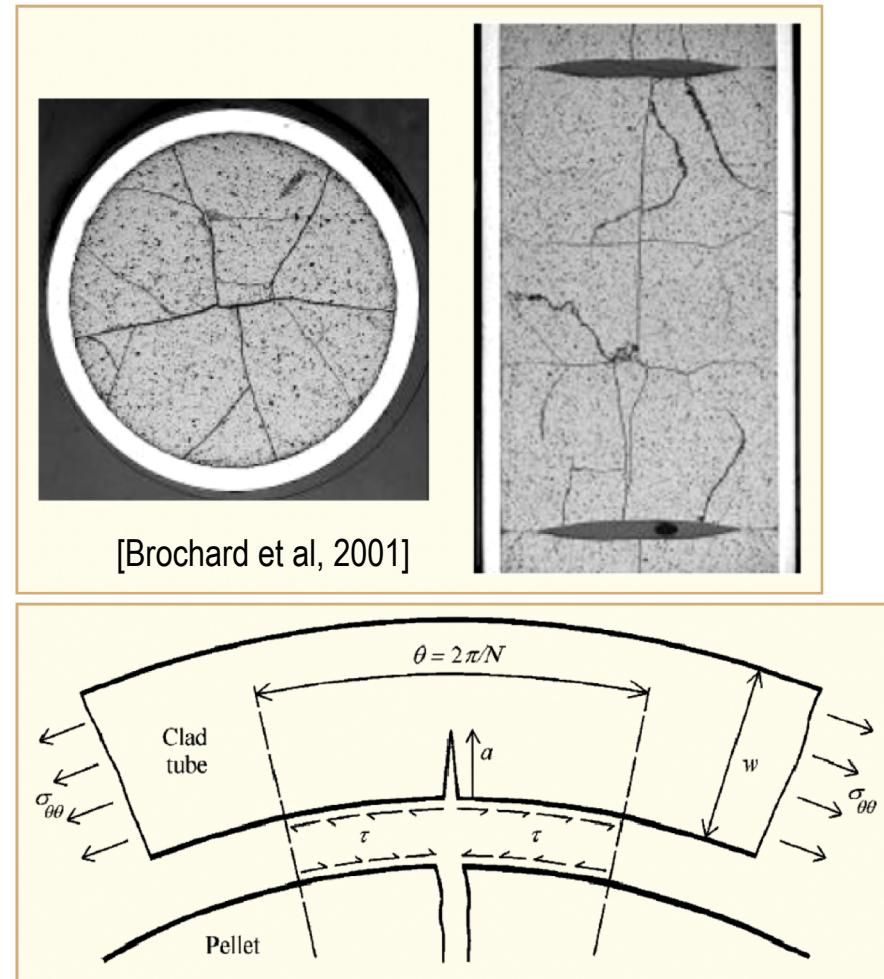
Fuel-Clad Chemical Interaction (FCCI)

- One possible issue with the fuel is fuel and cladding bonding
- Bonding tends to occur with pellet-clad contact at high burnup and high power (pellet-clad chemical interaction or PCCI)
- Bonding results from the inter diffusion of U-Zr-O
- Bonds can also involve cesium in uranates and zirconates
- Bonding eliminates gas gap between pellets and cladding
 - Improves heat transfer
- Bonding connects pellets and cladding
 - Increases PCMI



Fuel-Clad Mechanical Interaction (FCMI)

- Also referred to as Pellet-Clad mechanical interaction (PCMI)
- Mechanical loading arises from interaction among pellet fragments and interaction of fragments with cladding
- Mechanical interaction involves friction or bonding at pellet-clad interface
 - Differential thermal expansion, swelling and creep lead to contact forces in radial direction
 - Friction or bonding convert differential strains into circumferential and axial forces



Summary

- Under irradiation, zirconium experiences irradiation induced hardening due to interstitial loops on the prismatic planes
- Channels form that don't have loops, resulting in localized deformation
- Fatigue can be an issue for cladding, limiting reactor lifetime
- Pellet-clad interaction (PCI) takes two forms
 - Pellet-clad chemical interaction, PCCI (bonding occurs)
 - Pellet-clad mechanical interaction, PCMI (pellet pushes and drags cladding)