PennState
College of Engineering

NucE 497: Reactor Fuel Performance

Lecture 31: Mechanical Behavior

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Wikipedia and slides from ANT international

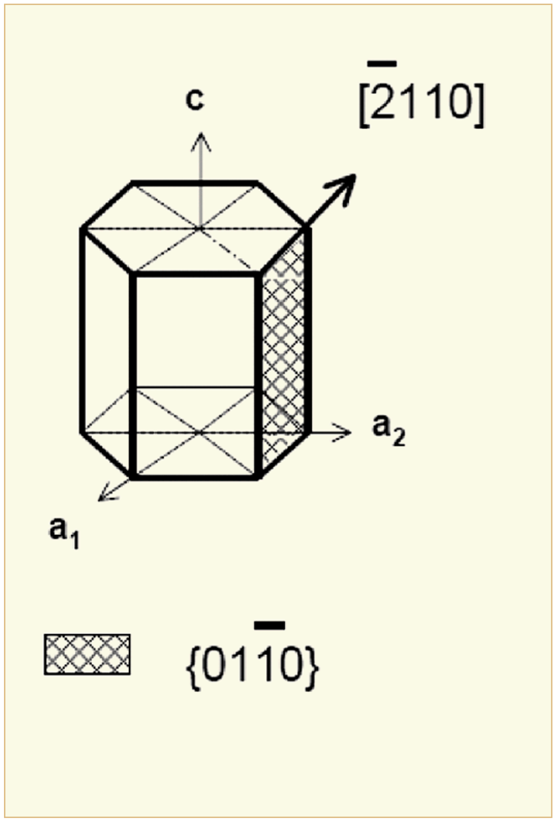
Today we will discuss the mechanical behavior of the cladding and how it changes during operation

- Module 1: Fuel basics
- Module 2: Heat transport
- Module 3: Mechanical behavior
- Module 4: Materials issues in the fuel
- Module 5: Materials issues in the cladding
  - Zirconium alloys and fabrication
  - Cladding creep and growth
  - **Mechanical behavior**
  - Oxidation
  - Hydride formation
  - CRUD formation
- Module 6: Accidents, used fuel, and fuel cycle

Here is some review from last time

- The cladding always creeps inward toward the cladding
  a) True
  b) False
- Growth will cause the cladding to shrink and thicken when
  a) There is a random texture
  b) It gets older than 40
  c) The majority of the grains are oriented such that the center axis of the HCP structure aligns with the axial direction
  d) The majority of the grains are oriented such that the center axis of the HCP structure falls in the radial direction

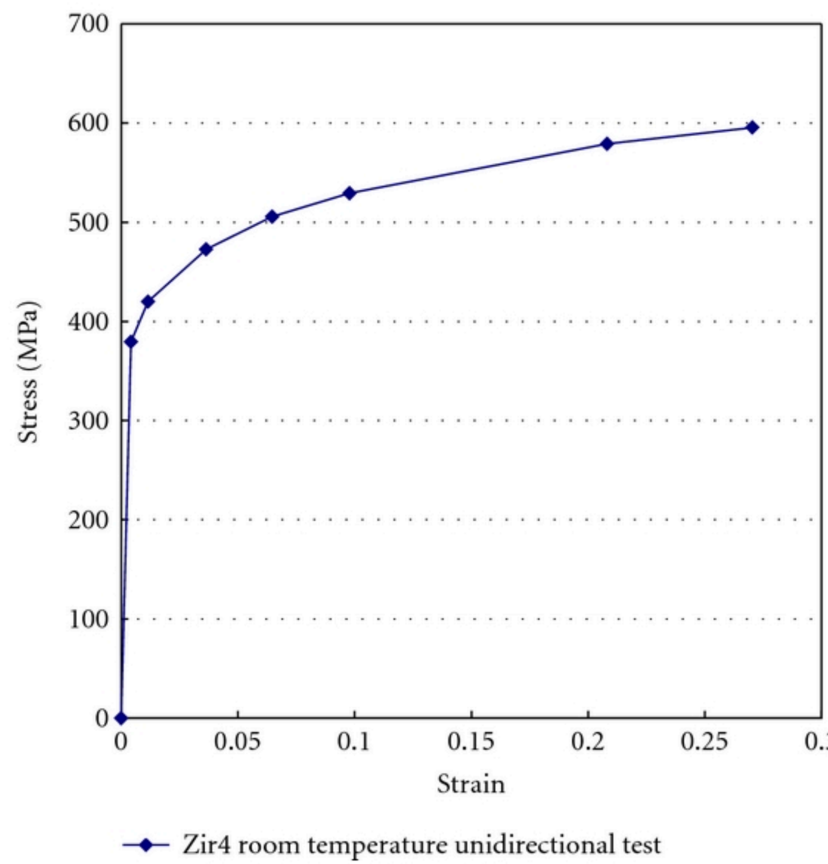
Zirconium alloys plastically deform primarily due to dislocation motion on prismatic planes



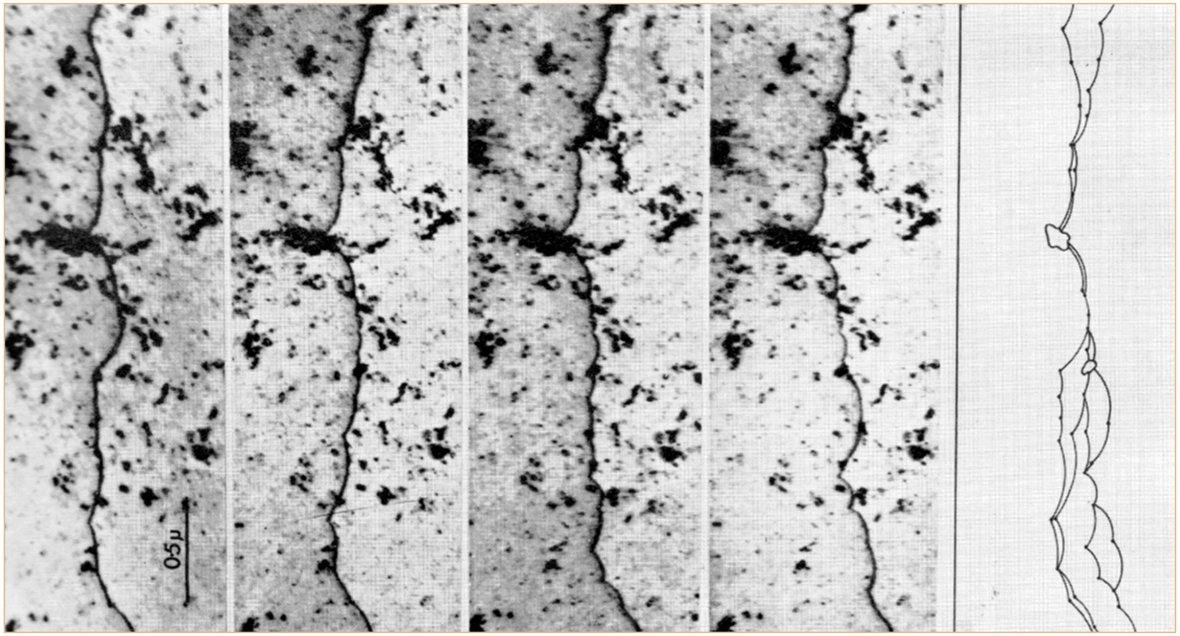
Zircaloy-4:

σ_y = 381 Mpa

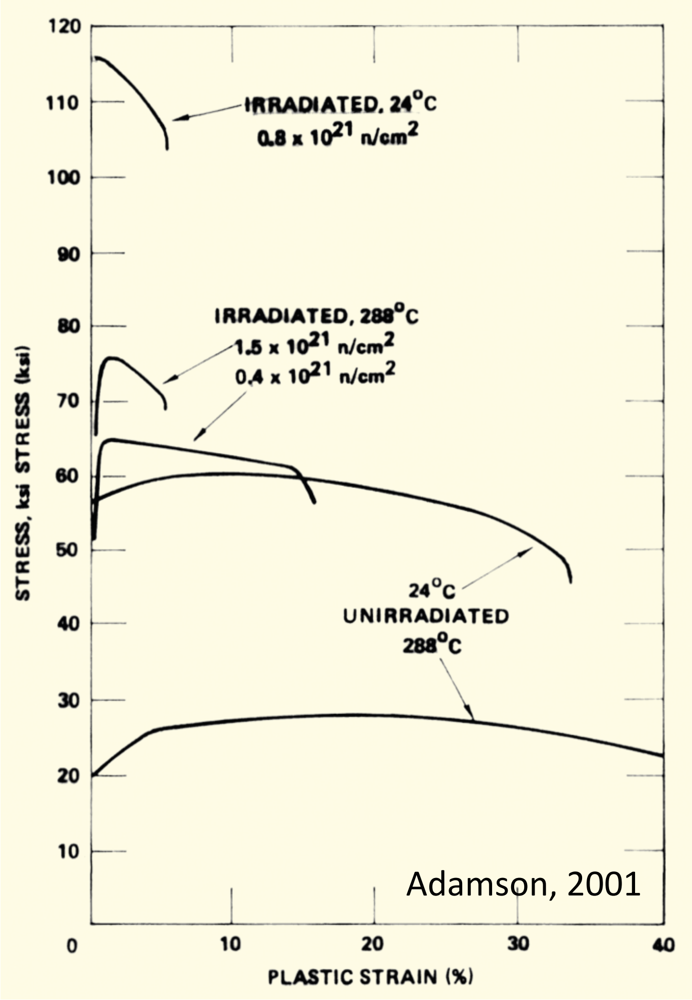
UTS = 514 Mpa



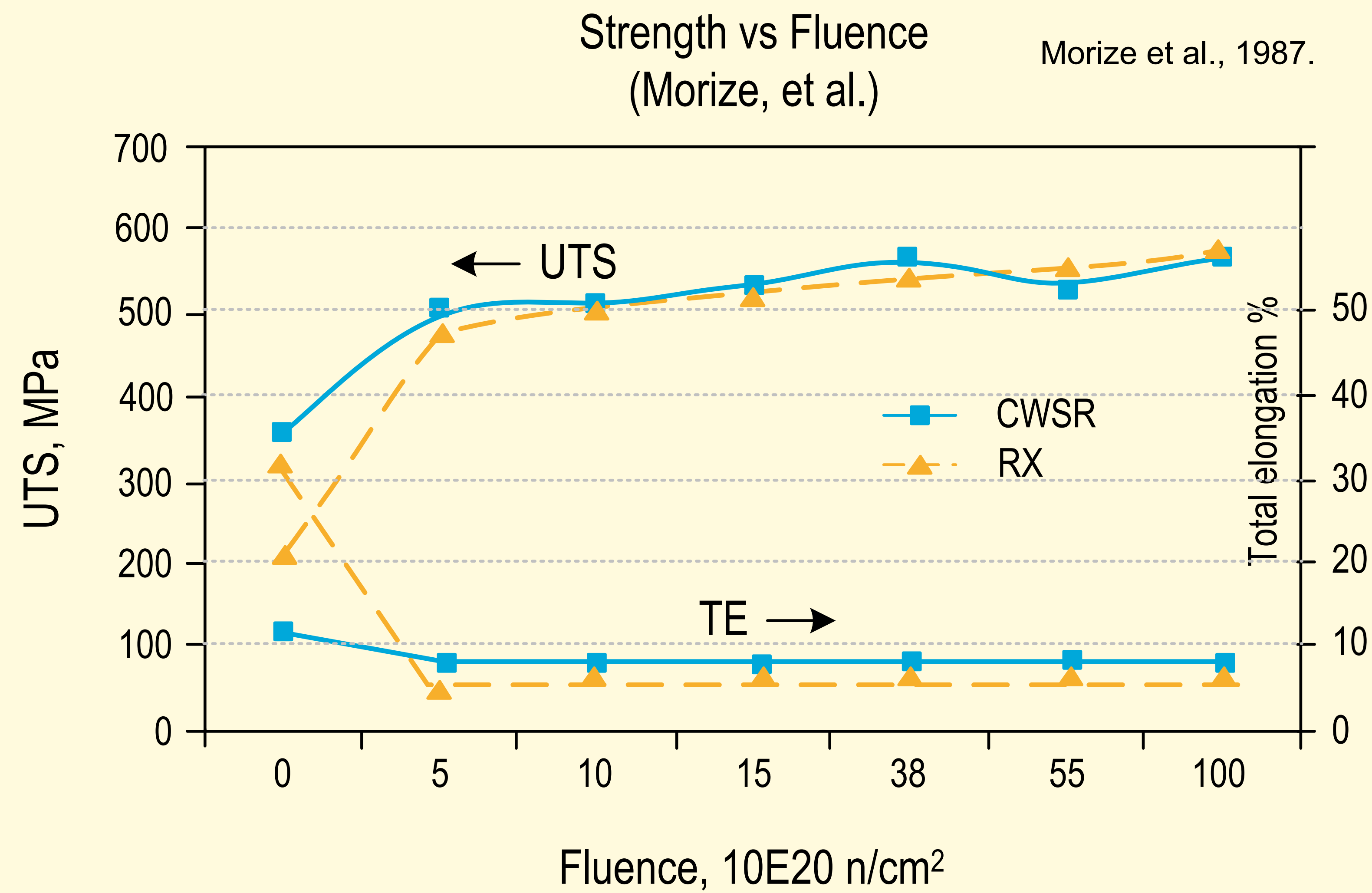
Under irradiation, the interstitial loops that buildup on the prismatic planes cause hardening



dislocation is moving ⟶ [Barnes, 1965]



The UTS goes up with fluence, while the strain before fracture goes down



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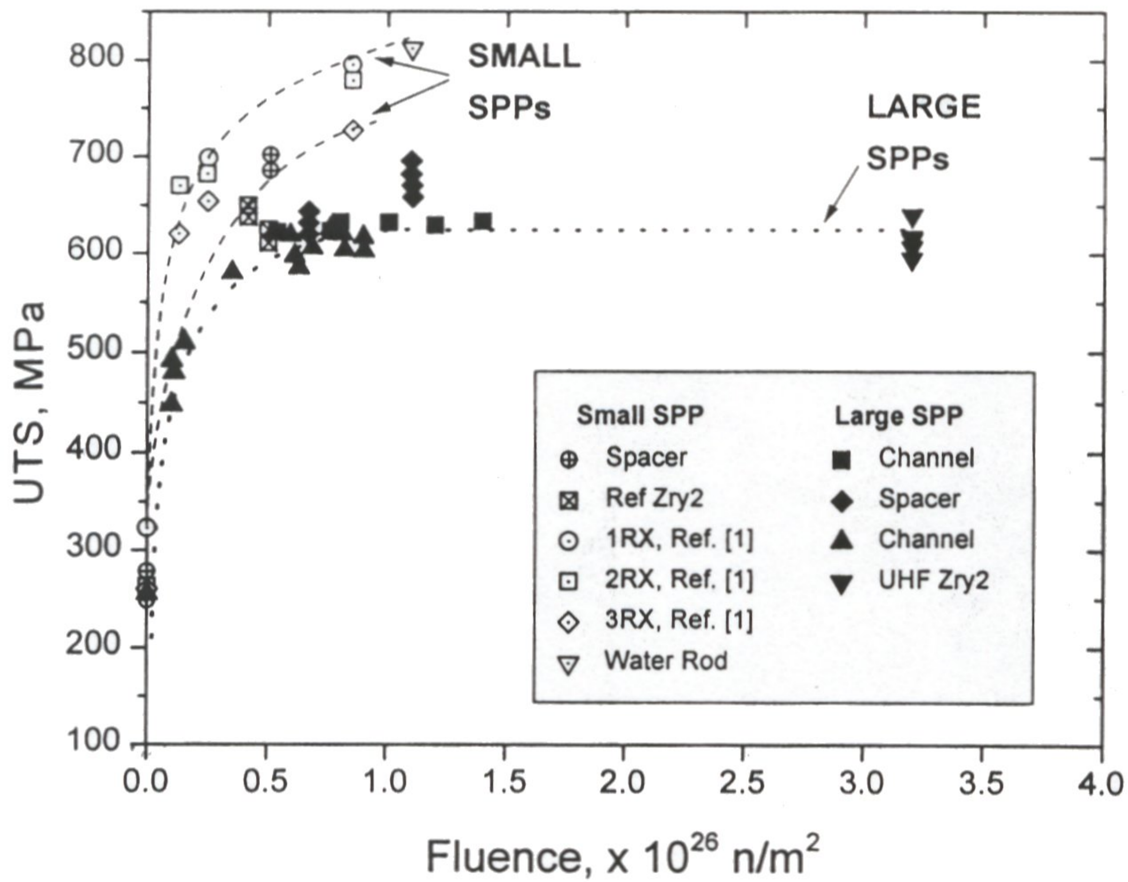
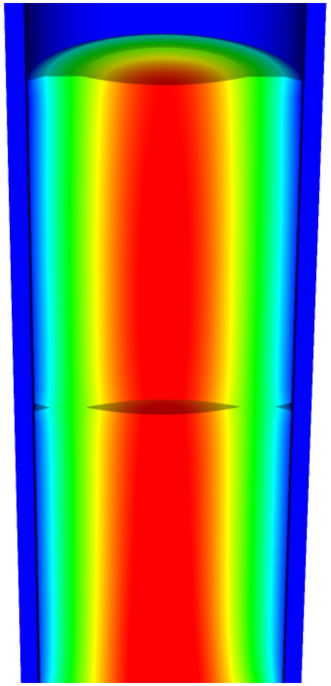
Second phase particle size impact irradiation hardening

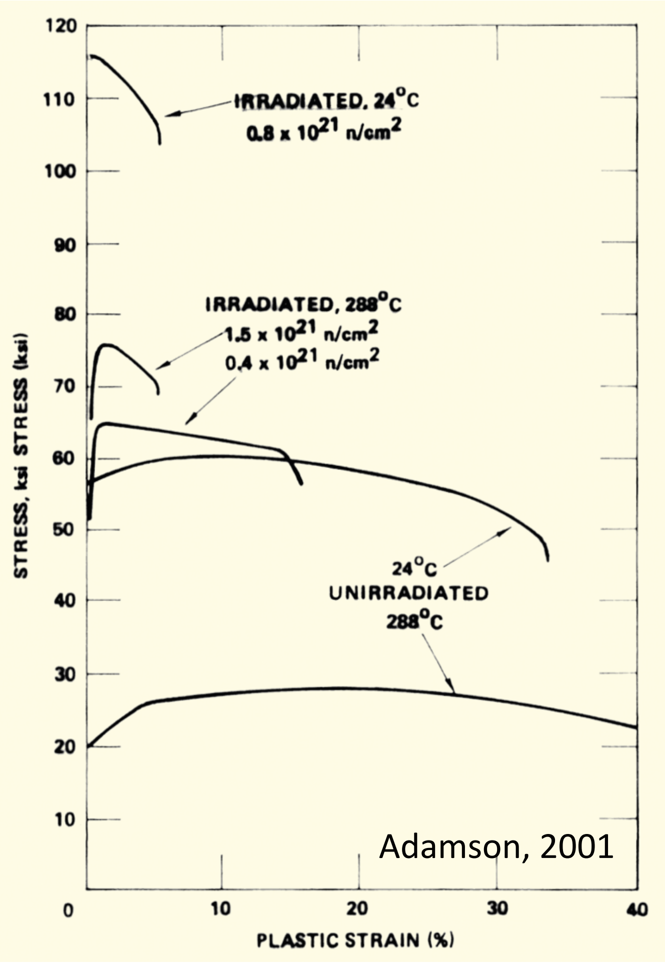


[Adamson,2000; Mahnmood et al., 2000]

How would irradiation hardening impact the performance of the fuel?

- Discuss with your neighbor the impact of dislocation hardening
  - Consider that the yield stress goes up
  - The material gets more brittle

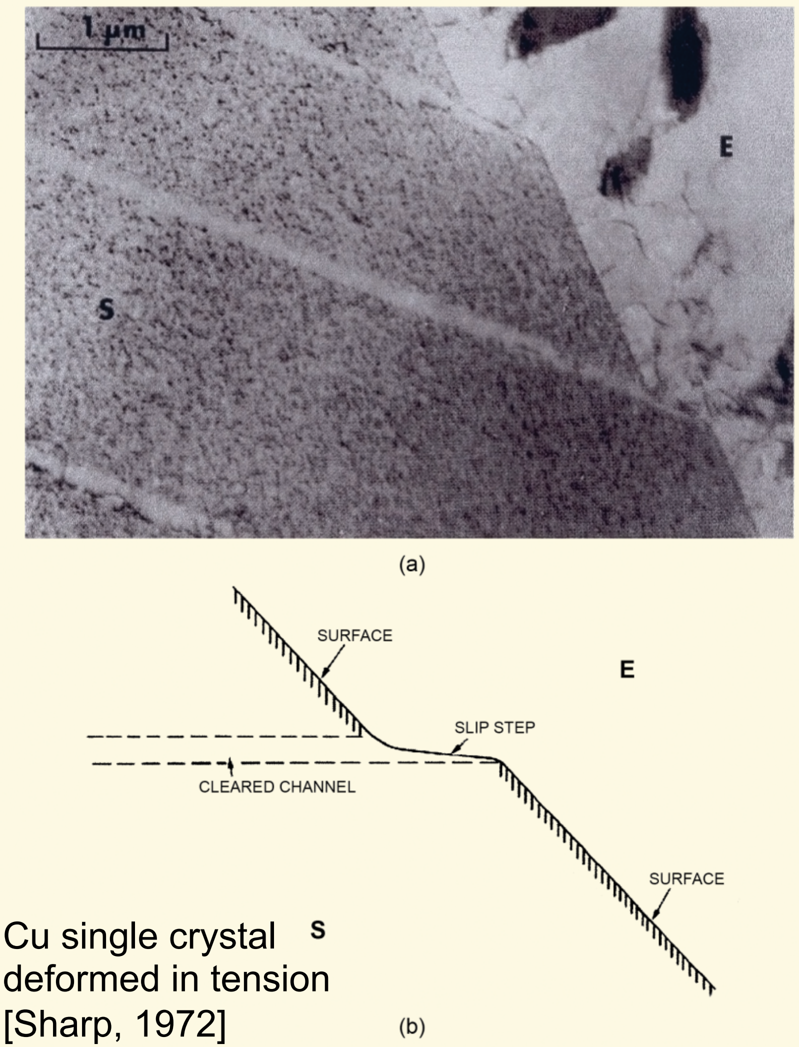




Adamson, 2001

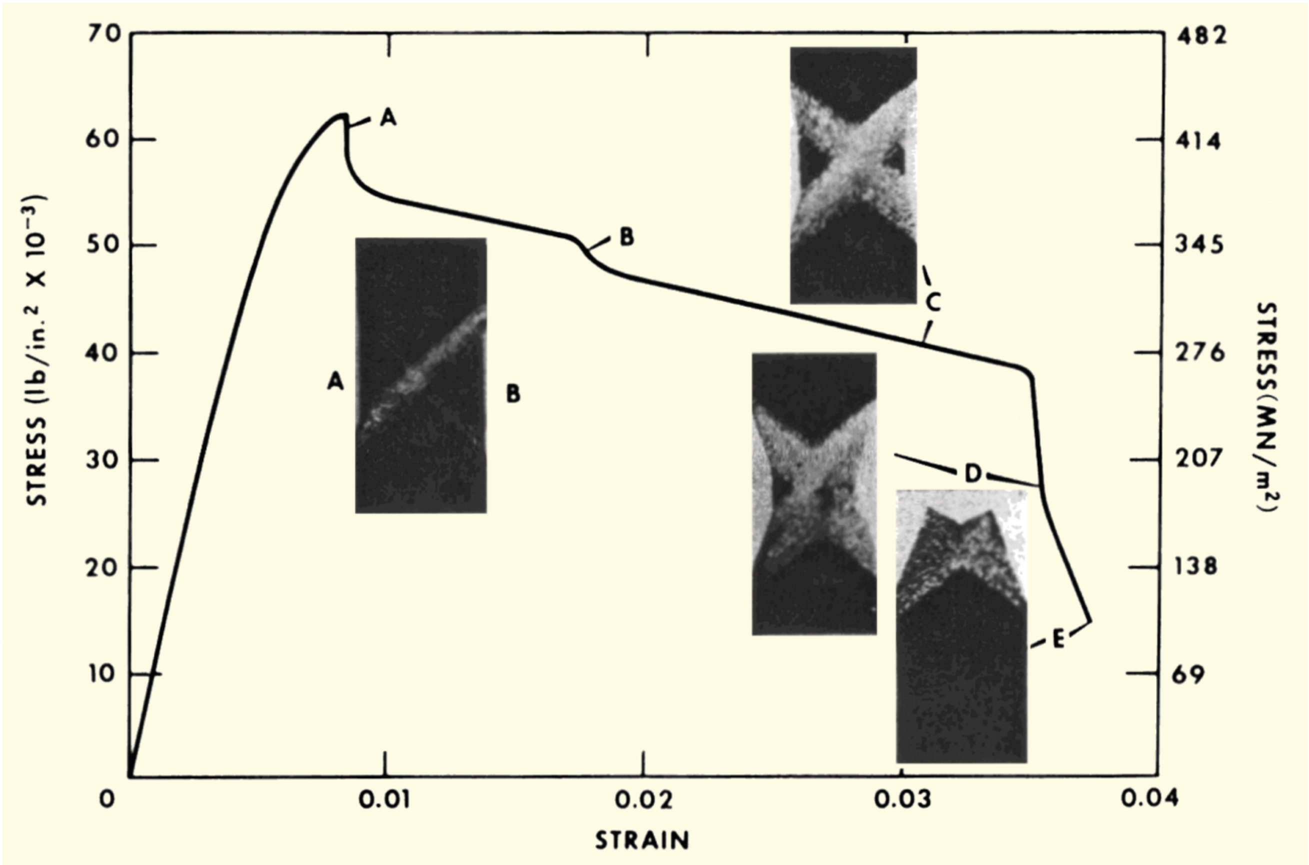
When stress is high enough, loops are "swept away", forming a clean channel

- 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
- HCP zirconium alloys
- In HCP zircaloy, channels form on basal planes or prism planes depending on
    - Load direction
    - Temperature
    - Oxygen content in the alloy
    - Fluence



Cu single crystal deformed in tension [Sharp, 1972]

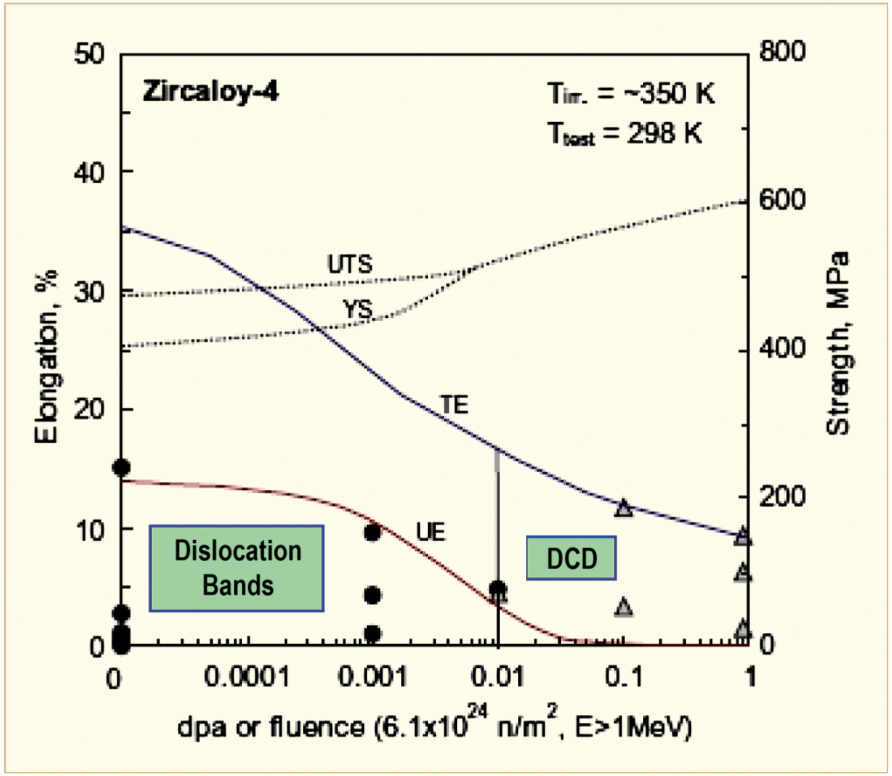
Channel formation is an important behavior in irradiated zircaloy



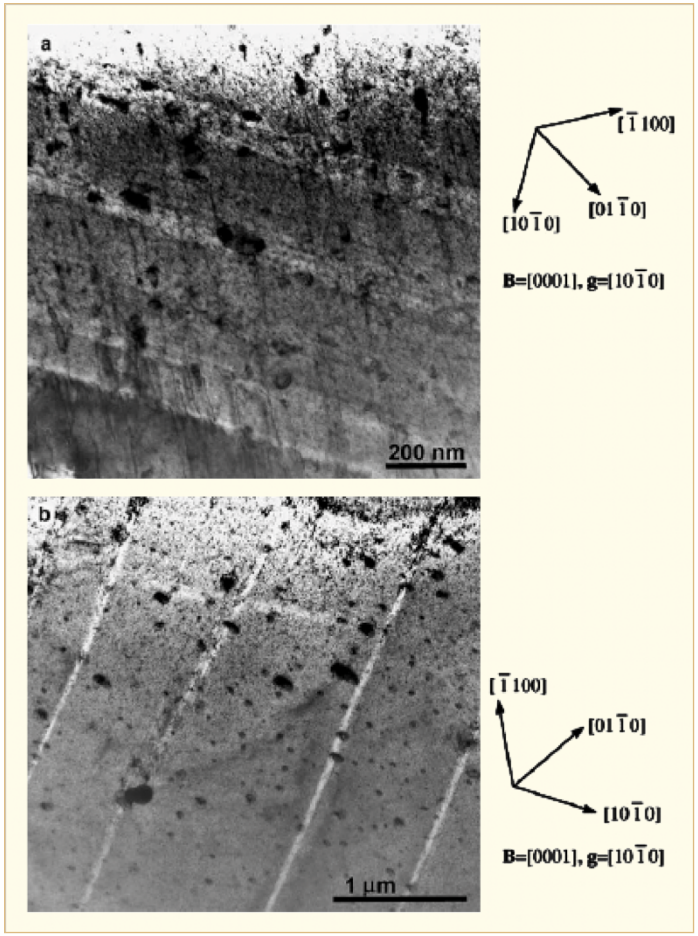
Engineering stress-strain curve for Zircaloy-2 sheet that had been irradiated at 280°C to a neutron fluence of 5 x 10^{20} n/cm^2 and subsequently tested at 300°C, Bement et al., 1965.

Dislocation channel deformation occurs at high load or high fluence

Deformation map
(DCD = dislocation channel deformation)



TEM of prism plane channels

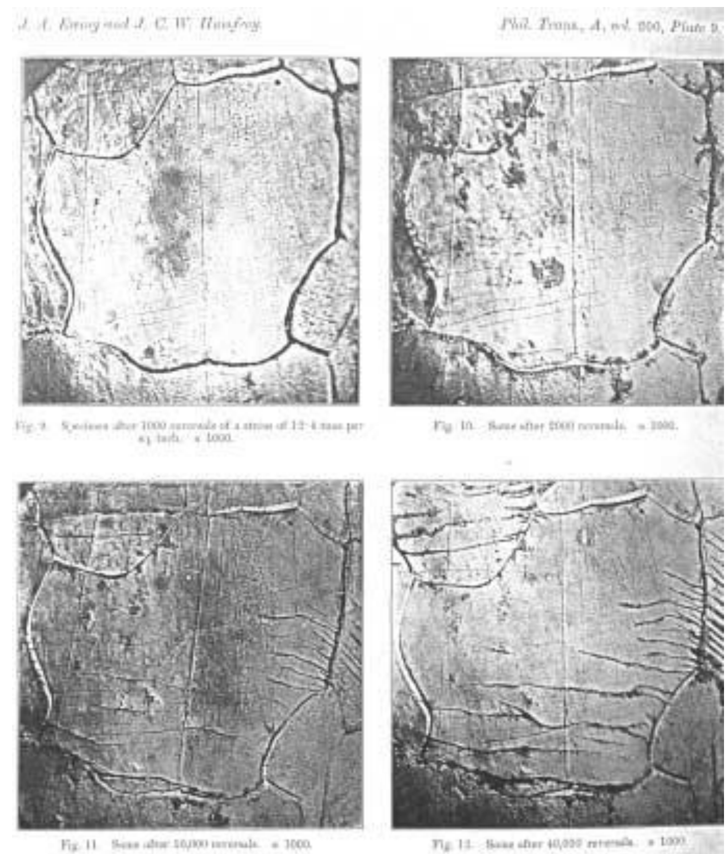


[Farrell,et al.,2004]

Material fatigue is the weakening of a material caused by repeated applied loads

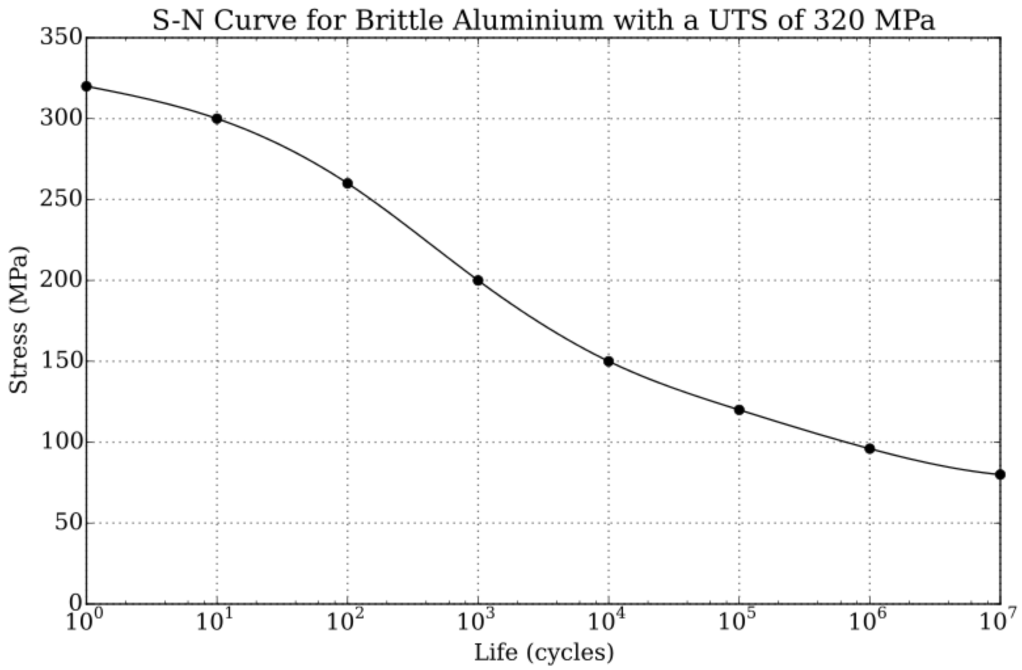
- Materials under cyclic loading can experience brittle like fracture at stresses significantly below their UTS
- It is caused by the slow propagation of microstructure damage

Fracture of an aluminium crank arm. Dark area of striations: slow crack growth. Bright granular area: sudden fracture.



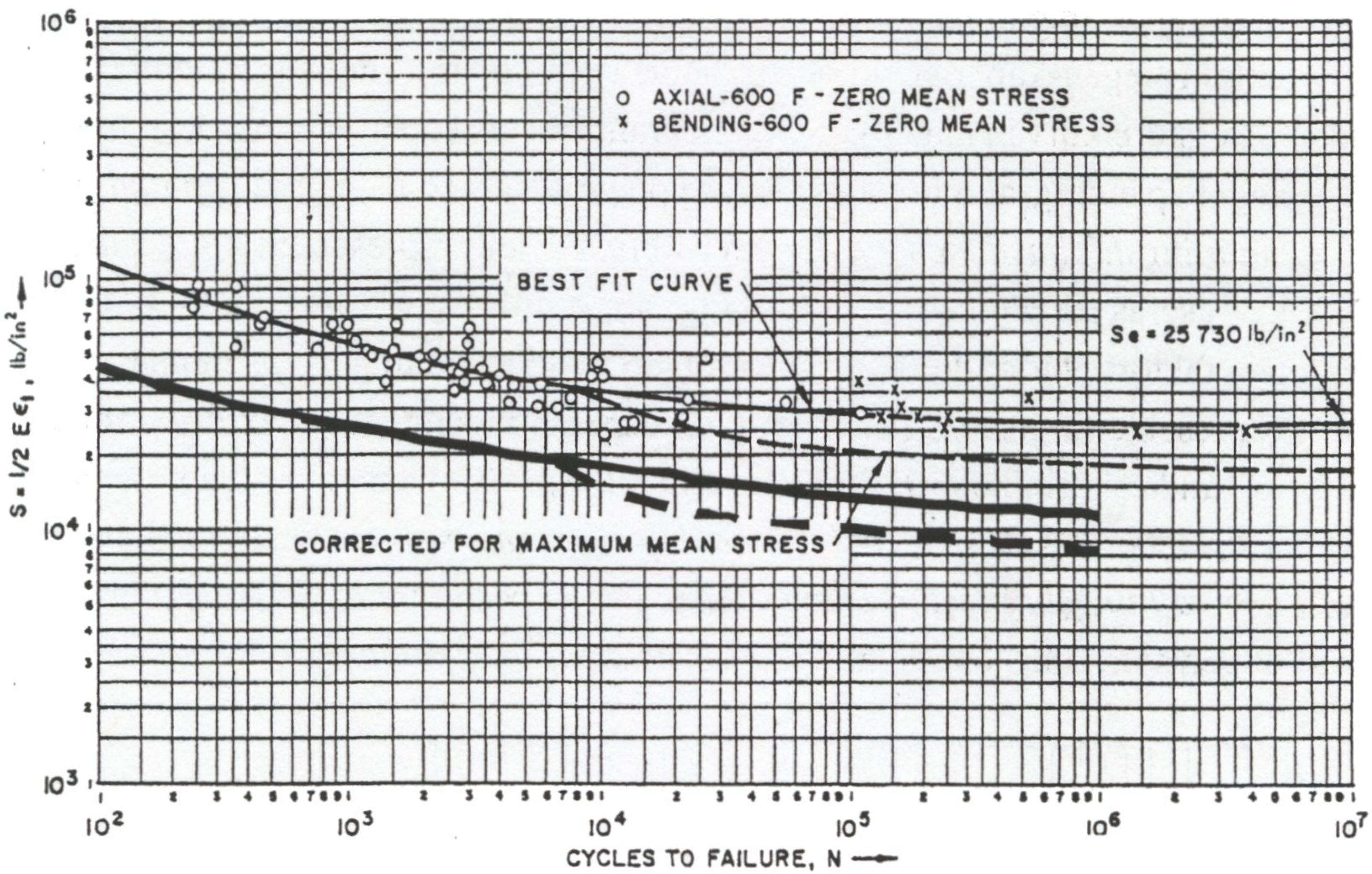
The engineering tool for investigating fatigue is the S-N curve

- An S-N curve is a graph of the magnitude of the cyclic stress (S) against the logarithmic scale of cycles to failure (N).



Does cladding undergo cyclic loading?

Zircaloy experiences fatigue like other metals



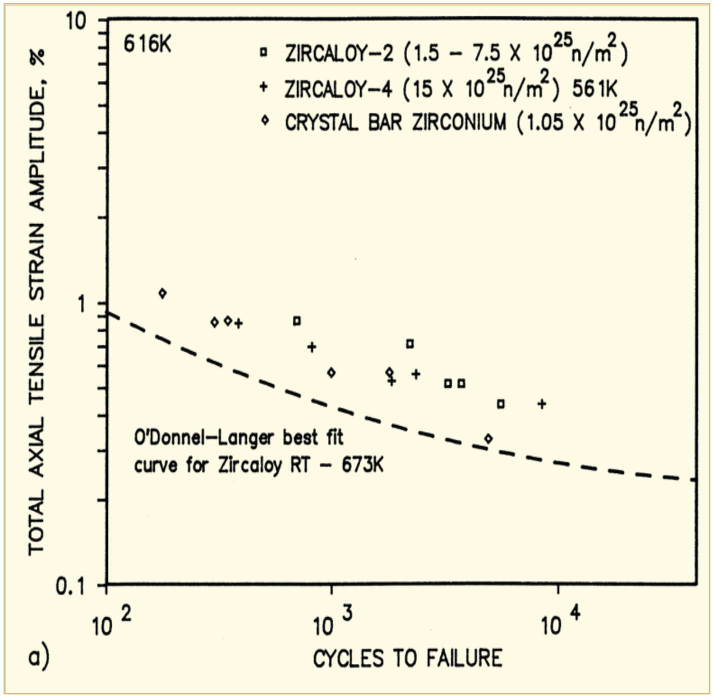
[O'Donnel and Langer, 1964]

Three major conclusions were made from this data from 1964

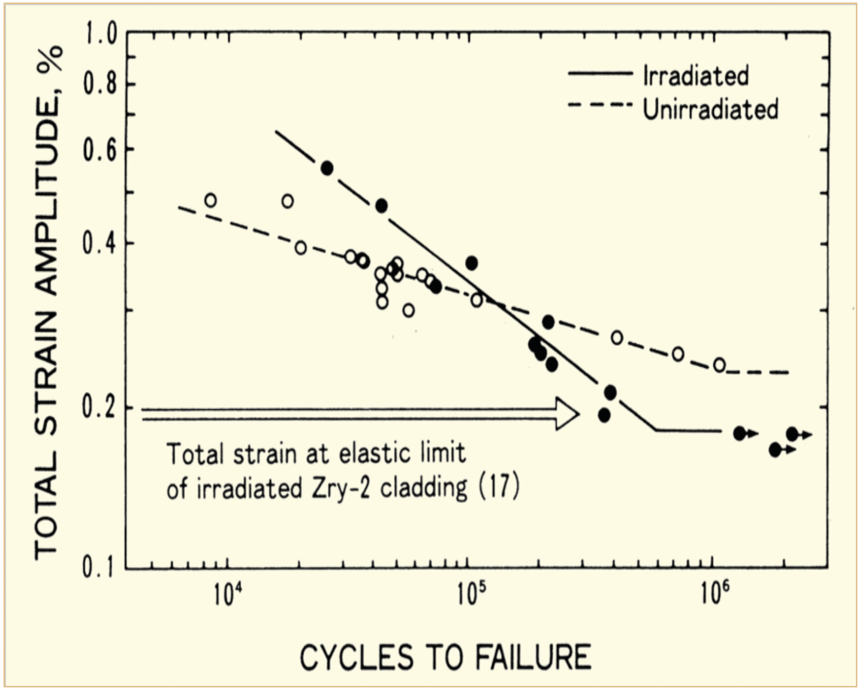
- Zircaloy has a fatigue limit at about 10^5 cycles of 25000 psi (180 MPa). The UTS is >500 Mpa.
- Irradiation slightly lowers the fatigue life in the low cycle range, less than about 10^4 cycles
- Irradiation has no effect on the fatigue life on the high cycle range, greater than about 10^5 cycles.

Design must demonstrate a factor of safety of 2 on stress and 20 on cycles.

What does newer data say?



Tensile total axial strain amplitude versus cycles to failure for irradiated Zircaloy-2, Zircaloy-4 and crystal bar zirconium [Wisner et al., 1994].



Change in failure cycles due to neutron irradiation tested in inert gas environment at 623 K
[Nakatsuka et al., 1991].

This table summarizes data on the effect of irradiation on fatigue life near 300°C (573K).

Author	Year	Fluence, n/cm^2 x 10^{21} (E>1 MeV)	Effect of Irradiation on Fatigue Life	
			low cycle	high cycle
O'Donnel & Langer	1964	2	Reduce	no effect
Wisner et al.	1994	1-7	Reduce	-
Pettersson	1975	0.3	slight increase	-
Nakatsuka et al.	1991	6	Increase	Decrease
Soniak et al	1994	4-8	no effect	Decrease
Chow et al., (Zr 2.5 Nb)	1996	6	slight increase	slight increase
Ishimoto et al.	2003	15	Increase	Increase

- No consensus
- Irradiation probably lowers fatigue life in low cycle regime and increases it in the high cycle regime
- May depend on test technique

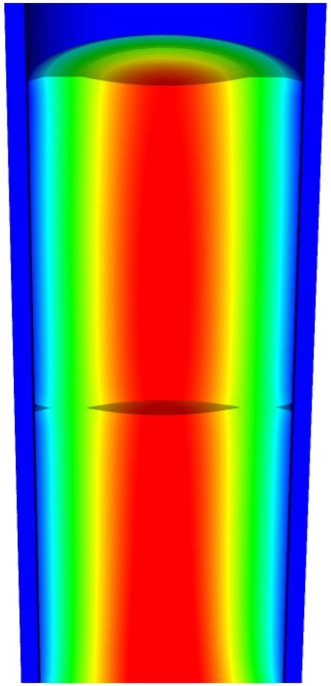
Another consideration in fatigue is the crack growth rate (change in length per cycle)

Author	Year	Effect of Irradiation on crack growth rate 293K	573K	Effect of Water
Wisner et al	1994	-	None	Increase
Pickles & Picker	1981	-	None	increase
Pickles & Picker	1983	-	None	increase
Walker & Kass	1974	small increase	-	-
Chow, et al., (Zr 2.5 Nb)	1996	none	small increase	-

Stress Corrosion Cracking (SCC) occurs when environmental effects accelerate fracture

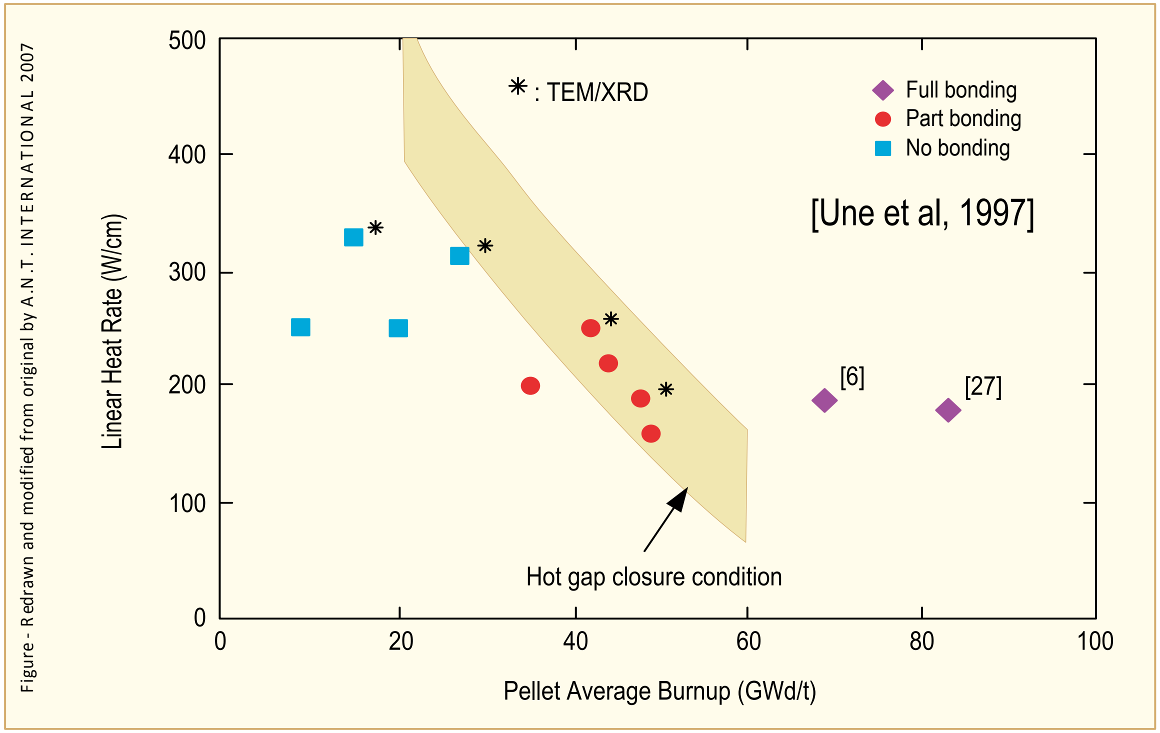
- Zirconium alloy tubes do not react with water, but have SCC from fission products inside tube
  - Early studies identified iodine as a primary SCC agent
  - Later, Cd identified as potent liquid metal embrittling agent

Once pellets contact the cladding, what are possible issues that could occur?

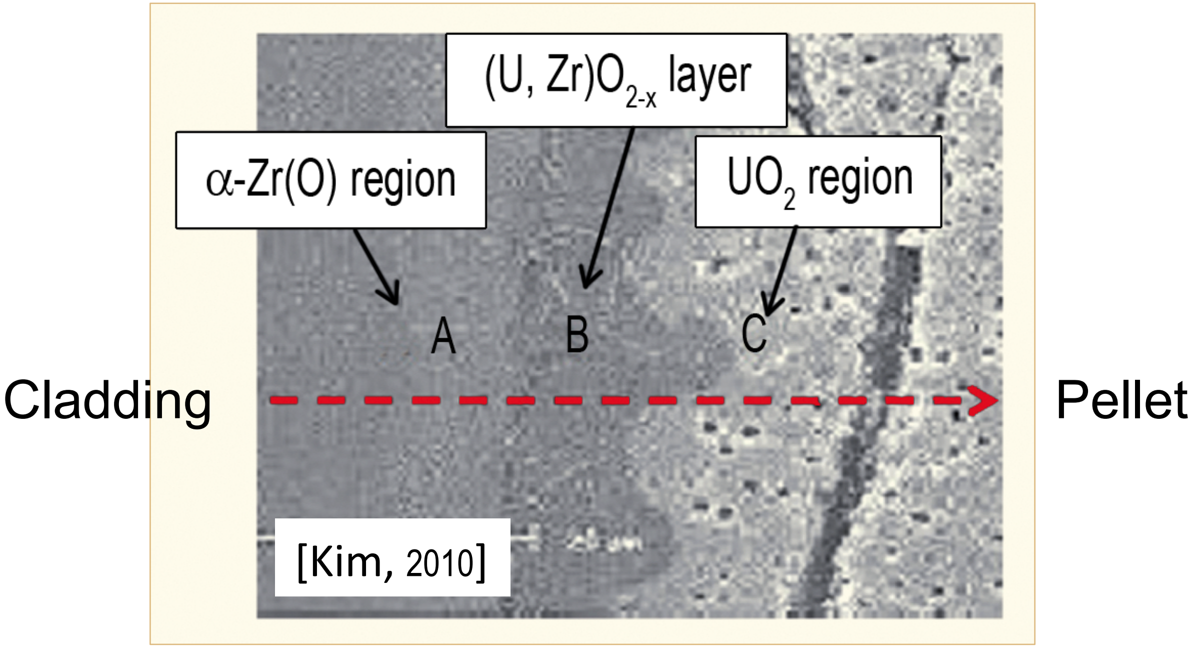


One possible issue with the fuel is fuel and cladding bonding

- 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

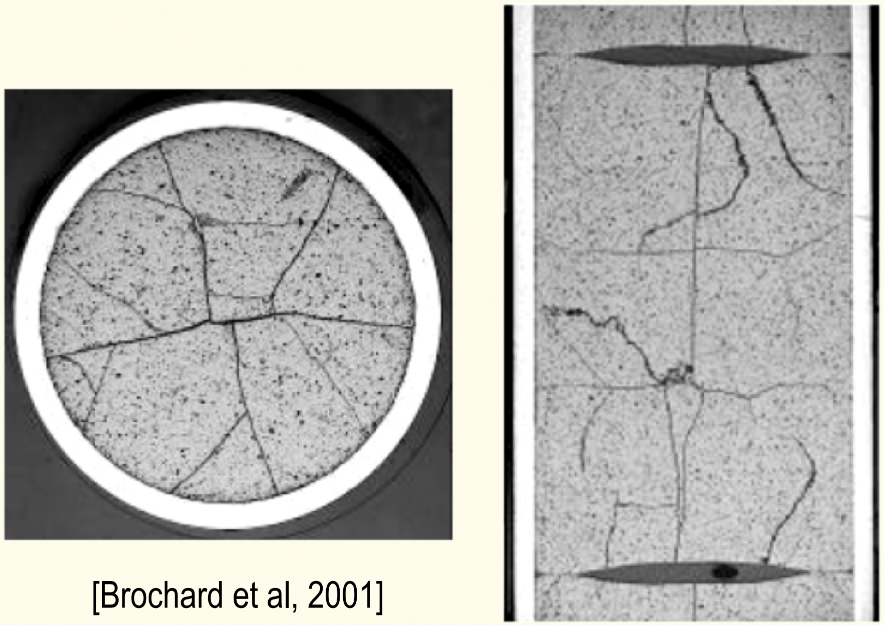


Pellet-clad bonding has both positive and negative effects

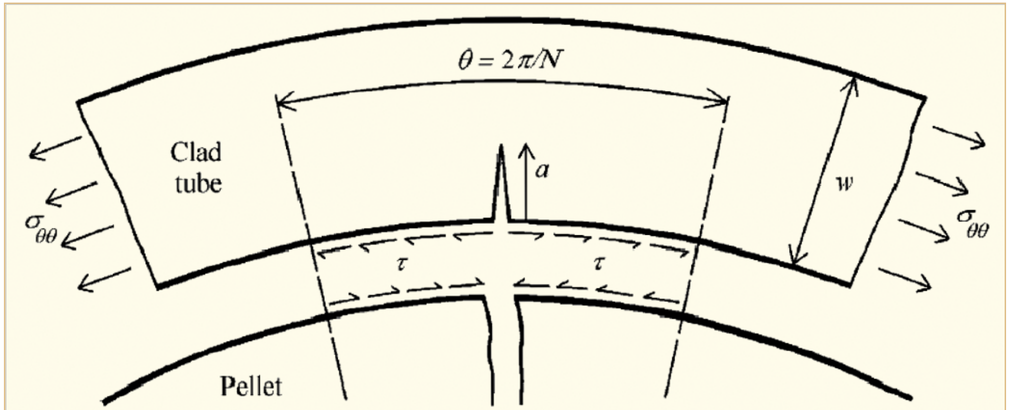


- Bonding eliminates gas gap between pellets and cladding
  → Improves heat transfer
- Bonding connects pellets and cladding
  → Increases PCMI
  - Bond strength ~ pellet strength
  - But, pellet strength degraded by HBS

Pellet-Clad mechanical interaction (PCMI) also occurs

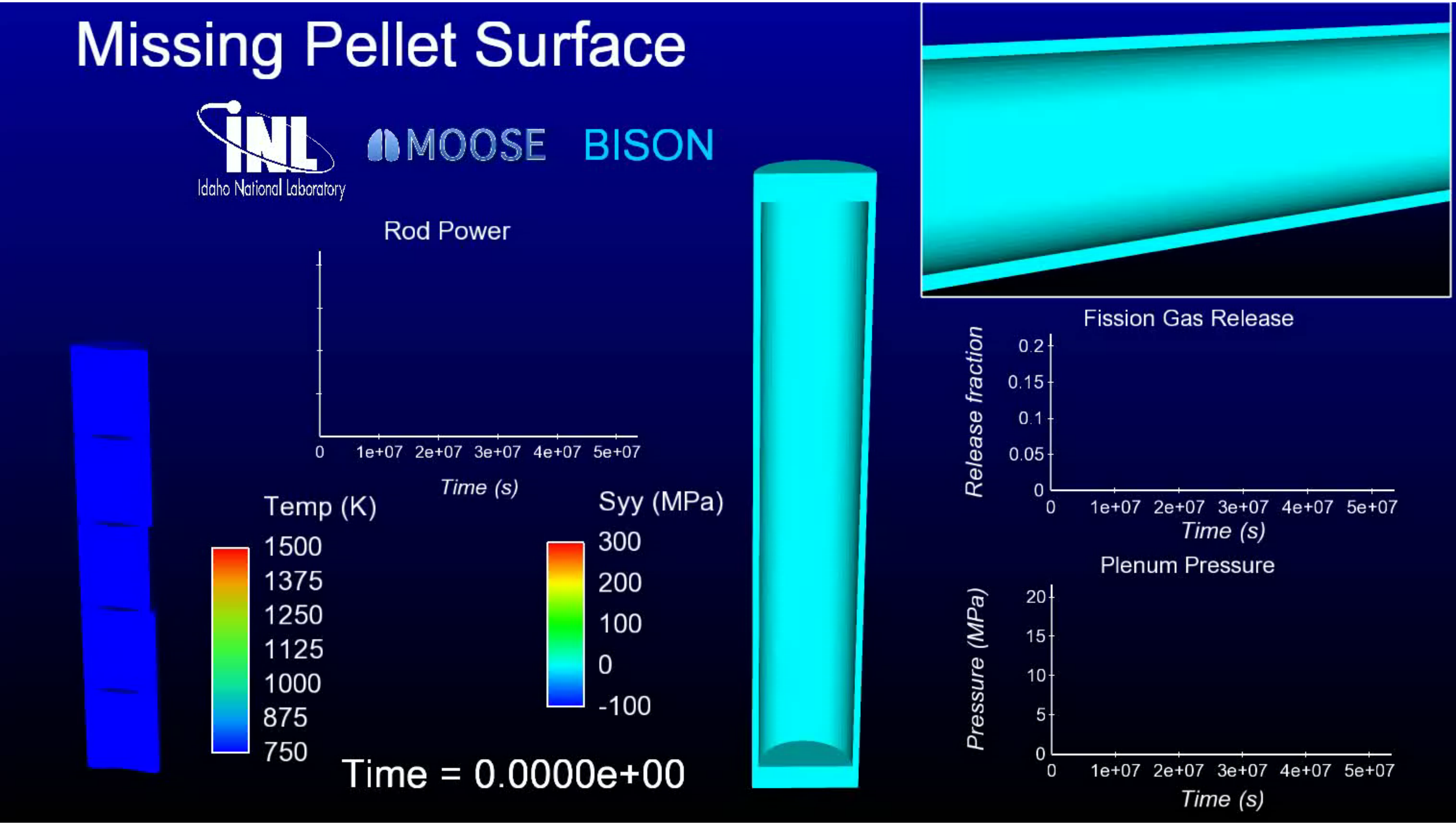


[Brochard et al, 2001]

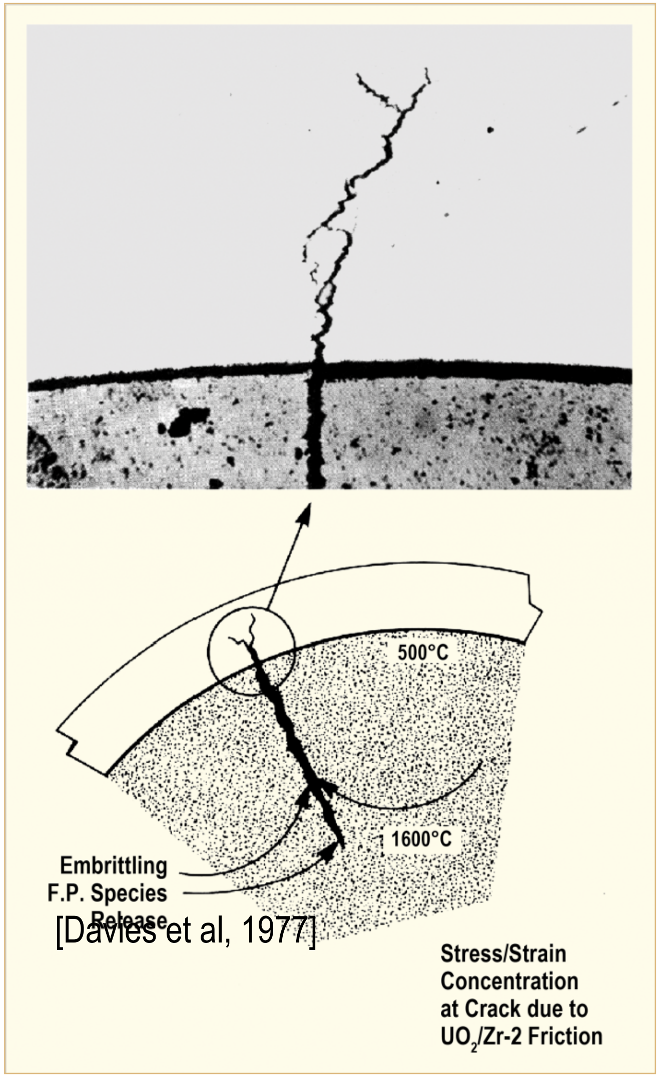


[Zhou et al, 2005

- 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


Missing Pellet Surface
Idaho National Laboratory
MOOSE
BISON
Rod Power
0
1e+07
2e+07
3e+07
4e+07
5e+07
Time (s)
Temp (K)
1500
1375
1250
1125
1000
875
750
Syy (MPa)
300
200
100
0
-100
Time = 0.0000e+00
Fission Gas Release
Release fraction
0.2
0.15
0.1
0.05
0
Time (s)
Plenum Pressure
Pressure (MPa)
20
15
10
5
0
Time (s)

Pellet-clad mechanical interaction can have significant negative impacts



[Davies et al, 1977]

PCMI issues

- Differential elongation among rods and structural elements
  - Axial growth increased by PCMI
  - Elongation greater in rods with more PCMI
- Localized loading in hoop direction
  - PCI-SCC or LME damage
  - Hydrogen assisted cracking

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, but there is no conclusive conclusion on the impact of irradiation on fatigue
- Pellet-clad interaction (PCI) takes two forms
    - Pellet-clad chemical interaction, PCCI (bonding occurs)
    - Pellet-clad mechanical interaction, PCMI (pellet pushes and drags cladding)