PennState
College of Engineering

NucE 497: Reactor Fuel Performance

Lecture 30: Cladding Growth and Creep

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Professor Motta's notes and slides from ANT international

Today we will discussion two types of dimensional change in the cladding: creep and growth

- 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

- We use zirconium alloys for cladding rather than pure zirconium because
  a) The alloys have a lower neutron absorption cross section
  b) The alloys have improved mechanical properties
  c) The alloys are more resistant to radiation damage
  d) The alloys are more resistant to oxidation
- Which is true about the heat treatments used when fabricating the cladding?
  a) The temperature is raised but held below the phase transition temperature to reset the texture
  b) The temperature is raised above the phase transition temperature to reset the texture
  c) The temperature is raised above the phase transition temperature to reduce the dislocation hardening (stress-relief)
  d) The temperature is raised to improve the cladding surface finish

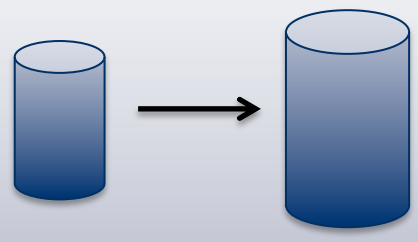
The microstructure changes within the fuel directly impact the fuel centerline temperature

Microstructure Change	T increase or decrease?
• Thermal expansion	Decrease
• Densification	Increase
• Swelling	Decrease
• Fission gas release	Increase
• Decrease in fuel thermal conductivity	Increase

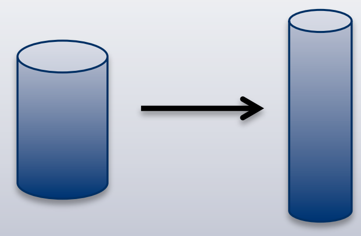
Today we will talk about dimensional changes in the cladding that also impact the temperature

- Zirconium alloys are resistant to swelling, but not to irradiation growth and creep

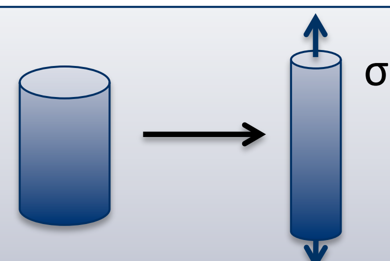
Swelling
- Increase in volume
- Decrease in density
- Caused by irradiation induced cavities



Irradiation Growth
- Anisotropic change in shape
- Constant density
- Occurs without stress, just irradiation



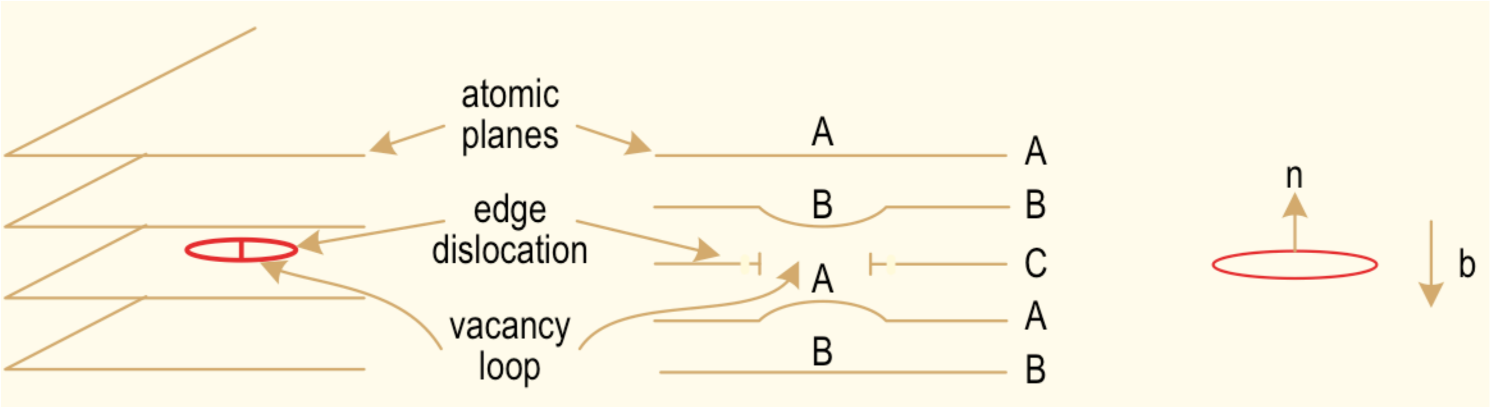
Irradiation Creep
- Change in shape
- Constant density
- Occurs under stress with $\sigma < \sigma_y$

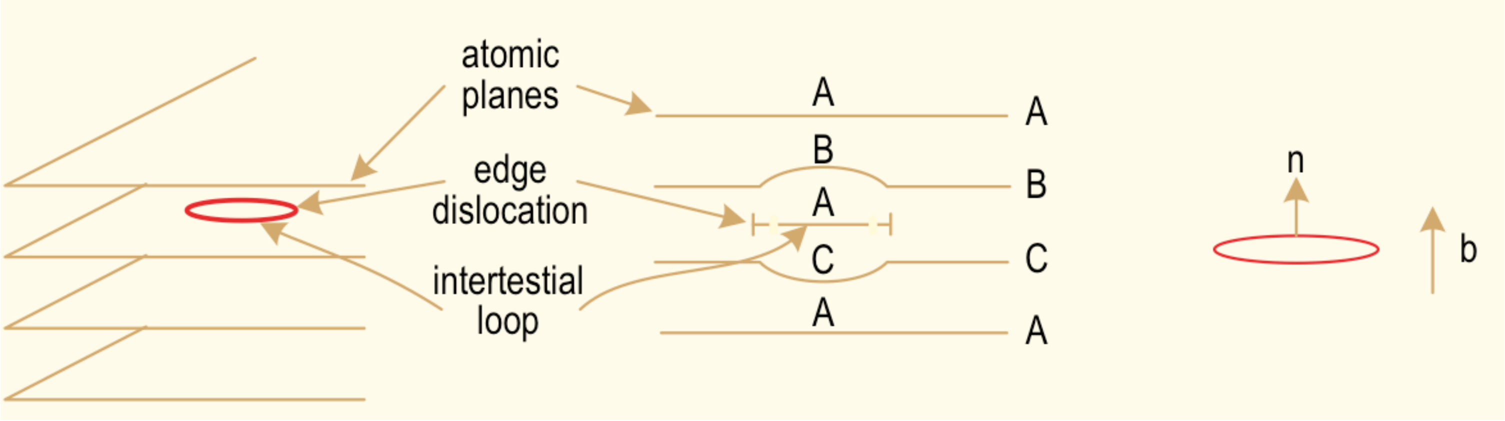


The cladding undergoes a very large amount of irradiation during its life

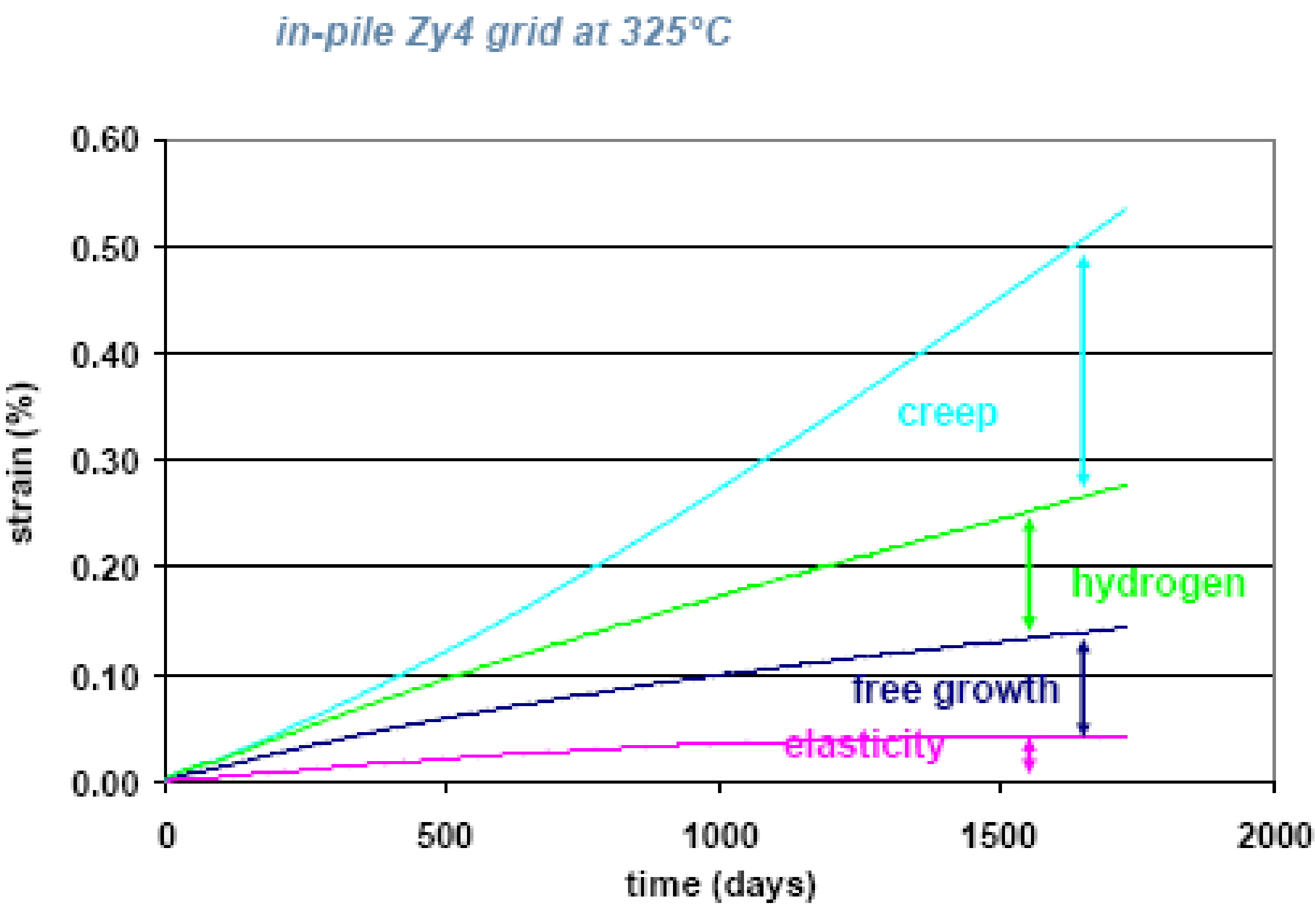
- The overall number of displacements to the cladding can be calculated with the NRT model to be G ~ 8 x 10-8 dpa/s, which, over 3 years exposure gives a total of ~ 8 dpa (every atom in the solid is displaced on the average eight times)
- Potential for extensive atomic rearrangements, and irradiation effects
- In the case of Zircaloy cladding, the main effects are:
    - Interstitial loops form on prism planes
    - Later in the reactor exposure, vacancy loops form on the basal plane

Vacancy and interstitial loops are partial planes missing or added to the lattice



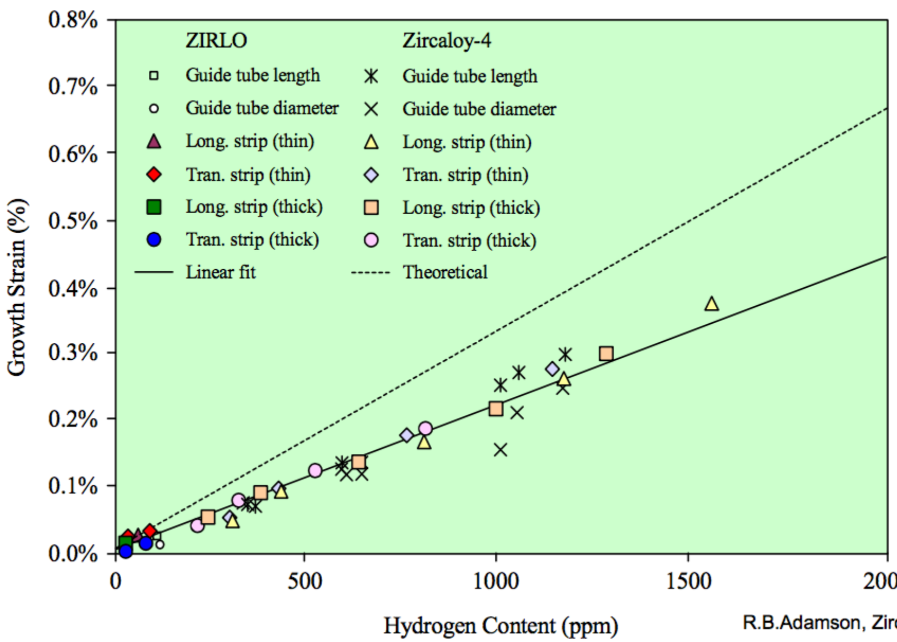


Zirconium alloy components change shape due to several mechanisms

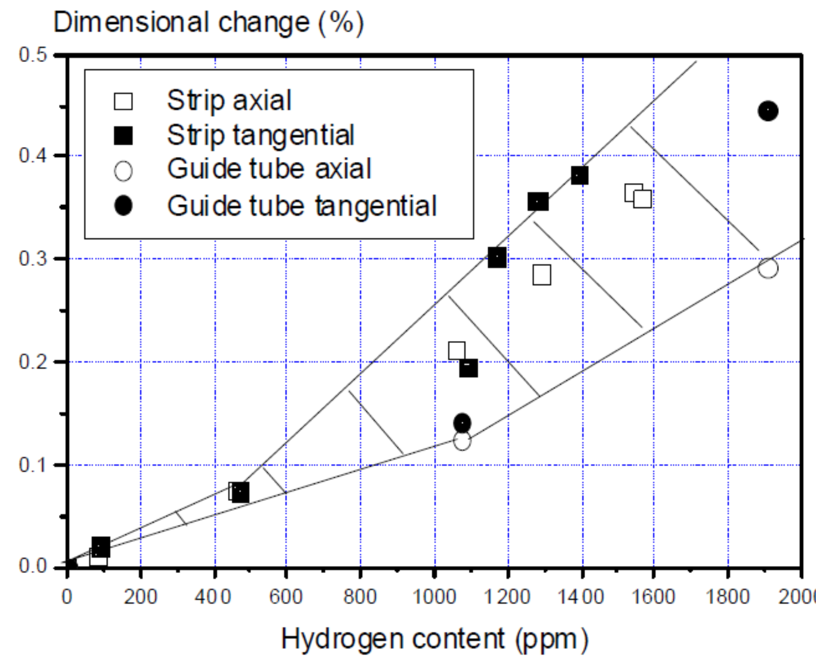


Barberis, et al., 2007

Hydrides cause size change because the hydride lattice is larger than the zirconium lattice



R.B.Adamson, Zircology Plus



Seibold et al. 2000

1000 ppm hydrogen can cause 0.2% dimension change

Both temperature dependent and radiation dependent creep take place in the cladding

- An empirical model has been developed for the irradiation creep
  - $\dot{\epsilon}_{ir} = C_0 \Phi^{C_1} {\sigma_m}^{C_2}$
  - Φ is the fast neutron flux (n/(m^2 s))
    $\Phi \approx$ 3e11 LHR n/(cm^2 s)
  - σ_m is the Von Misses stress (MPa)
  - Note that SRA stands for stress relief annealed, RXA for recrystallization annealed, and 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

- The Von Mises stress is calculated from the stress according to

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

- The empirical model for thermal creep of zircaloy is
  - $\dot{\epsilon}_{ss} = A_0 \left(\frac{\sigma_m}{G}\right)^n e^{\left(\frac{-Q}{RT}\right)}$
  - With $A_0 = 3.14 \times 10^{24}$ (1/s)
  - The 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

The creep strain is another strain that doesn't directly cause stress

- The equation for the total stress is $\boldsymbol{\sigma} = \boldsymbol{\mathcal{C}}\left(\boldsymbol{\epsilon} - \boldsymbol{\epsilon_{vol}} - \boldsymbol{\epsilon_{cr}}\right)$
- After computing the creep strain increment, you integrate with time to get the total creep strain
- The creep strain is applied in the same direction as the stress causing the creep, so the tensor is
  - $\boldsymbol{\epsilon}_{cr} = \frac{\boldsymbol{\sigma}}{|\boldsymbol{\sigma}|}\epsilon_{cr}$

Now we will work a problem for cladding creep

- Consider an SRA cladding tube at T = 600 K and LHR = 250 W/cm, with a stress σ_m = 200 MPa. What is the total creep strain after three years?
- First, we will calculate the thermal creep $\dot{\epsilon}_{ss} = A_0 \left(\frac{\sigma_m}{G}\right)^n e^{\left(\frac{-Q}{RT}\right)}$
  - $A_0 = 3.14 \times 10^{24}$ (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 $\dot{\epsilon}_{ir} = C_0 \Phi^{C_1} {\sigma_m}^{C_2}$
  - $\Phi \approx$ 3e11 LHR = 3e11*250 = 7.5e13 n/(cm^2 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

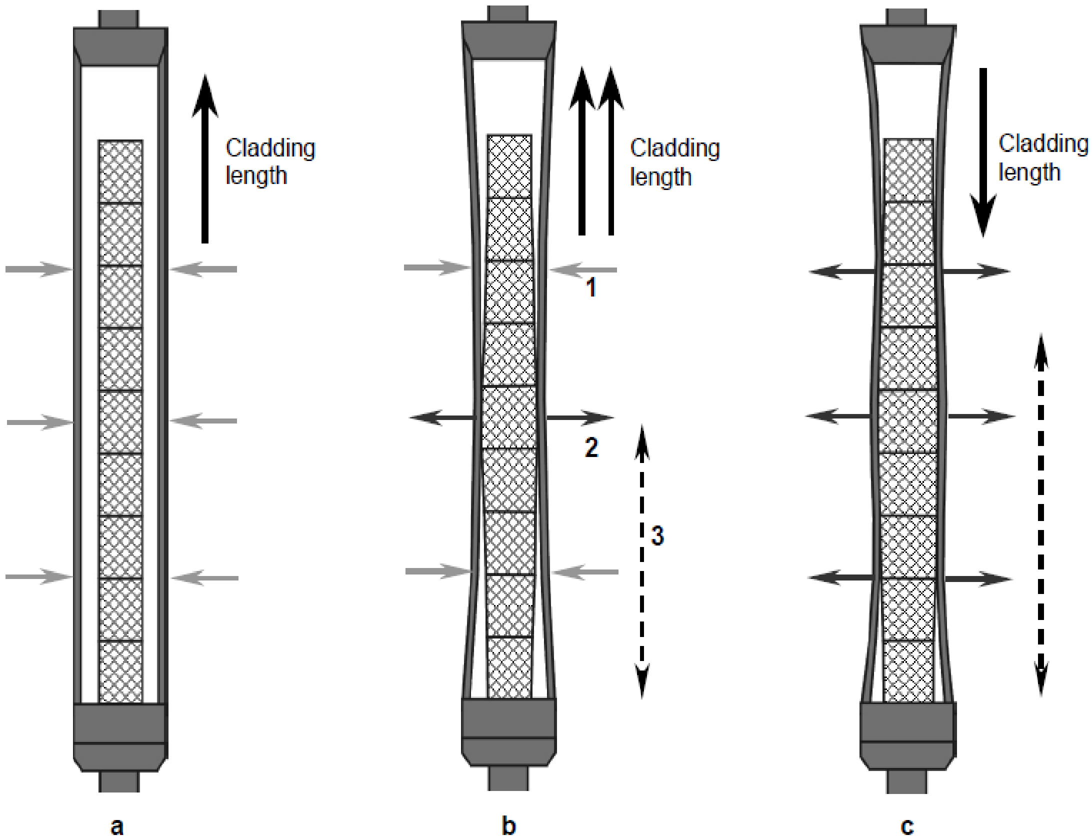
Now you do a problem

- What would the total creep after three years if LHR changed to 300 W/cm (T = 600 K with a stress σ_m = 200 MPa)
  - Thermal creep rate is 1.48e-10 1/s, $\dot{\epsilon}_{ir} = C_0 \Phi^{C_1} {\sigma_m}^{C_2}$
  - C_0 = 3.557e-24, C_1 = 0.85, C_2 = 1.0
  - $\Phi \approx$ 3e11 LHR

- The temperature is unchanged, so the thermal creep strain rate is unchanged
- First, calculate a new fast neutron flux
  - $\Phi \approx$ 3e11 LHR = 3e11*300 = 9e13 n/(cm^2 s)
  - Strain rate = 3.557e-24*(9.0e13)^0.85*200^1 = 5.17e-10 1/s
- Calculate the total strain rate
  - Total strain rate = 1.48e-10 + 5.17e-10 = 6.65e-10 1/s
- The total creep strain after three years (assuming constant conditions) is
  - 6.65e-10*(3600*24*365*3) = 0.0629 = 6.29% strain

Creep impacts fuel performance by shrinking the gap and then conforming to the pellets

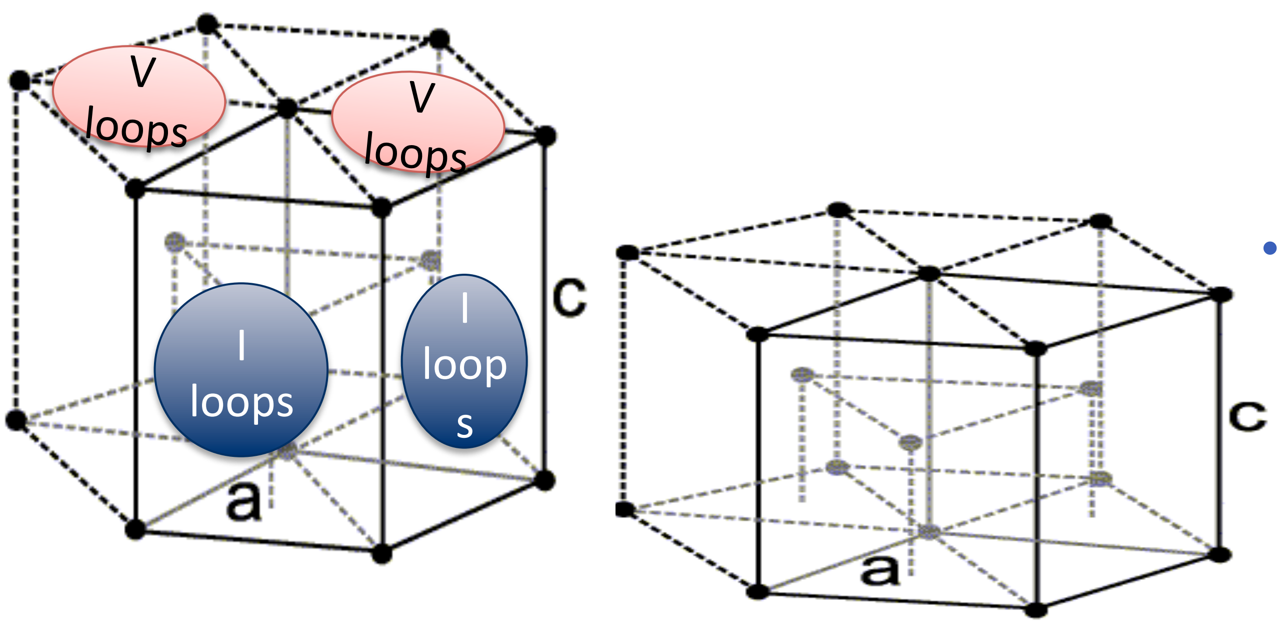
a) before “fuel-cladding” interaction, b)for a moment of “fuel-cladding” interaction, c) “fuel-cladding interaction” over most of fuel column, [Rogozyanov et al. 2005].



(1)creep down from water pressure; (2)creep out from fuel column; (3)fuel column axial stress

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 texture in the cladding results in elongation of the cladding and thinning of the walls

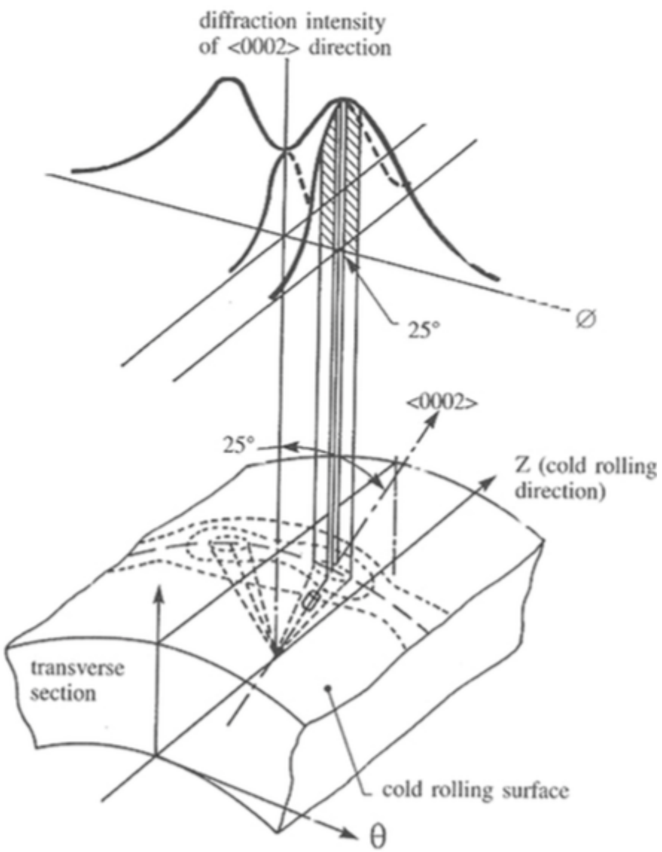
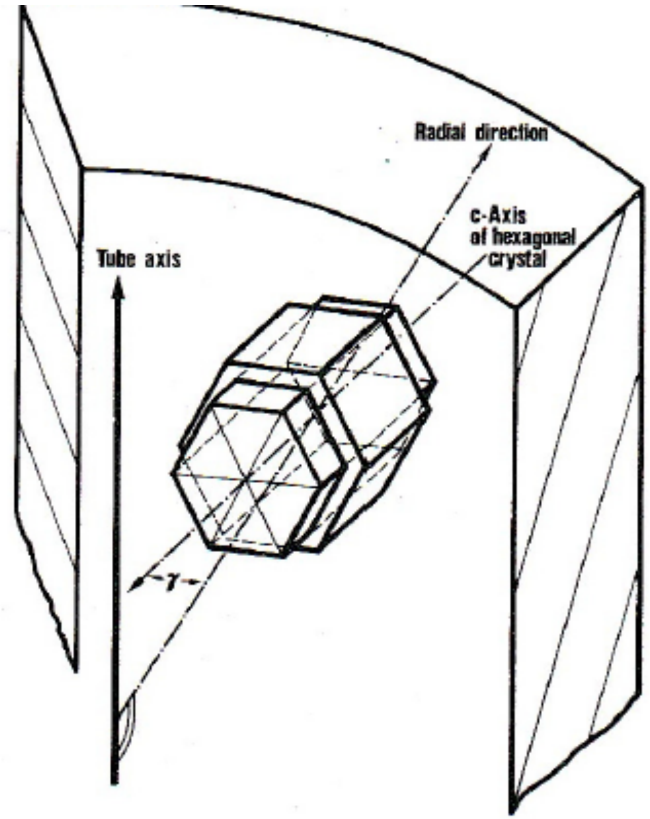


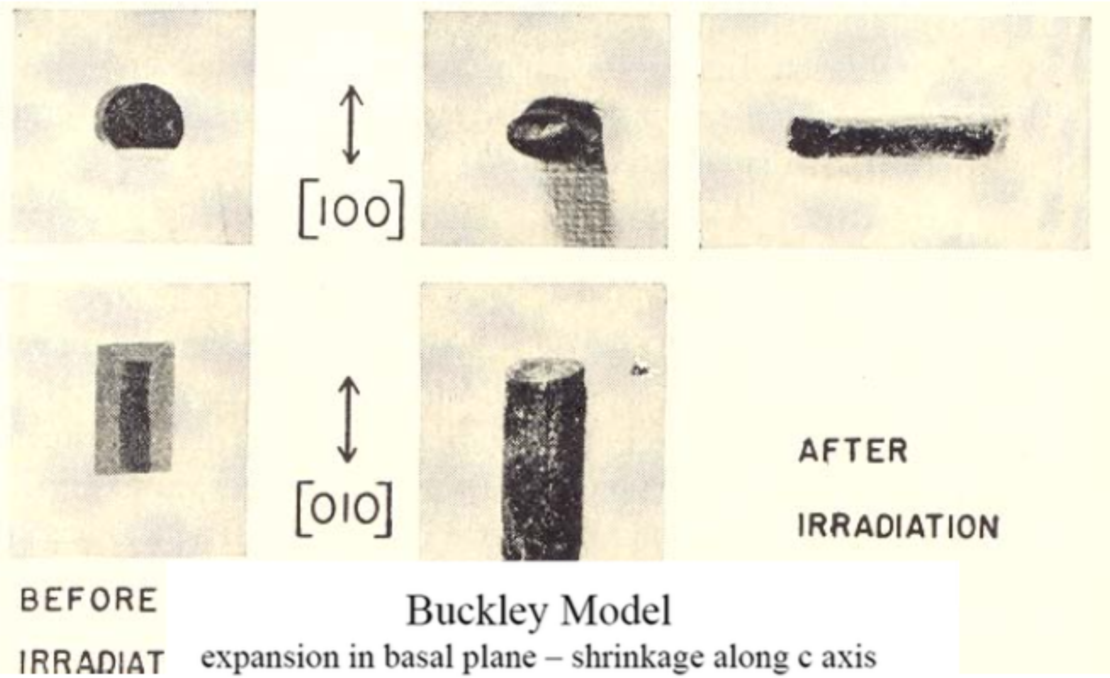
FIG 5.6. – Cladding material texture.
Exemple of texture with preferred orientation at 25 °.

Berthet et al, 1999



Schematic of irradiation growth in a tube having extreme radial texture ($f_c \approx .1$, $f_t \approx .1$, $f_N \approx .8$) showing expansion in the tube axis and circumference and contraction in the radial (through-wall) directions, Stehle et al., 1975.

The result is a change in shape, shrinking in the 0001 direction and expanding in the pyramidal direction



Buckley Model
expansion in basal plane – shrinkage along c axis

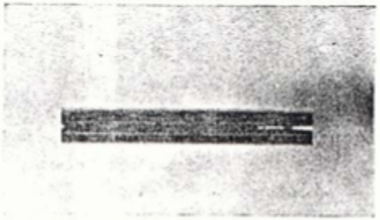


Fig. 1. Typical preirradiation appearance of Zr-Pu alloy specimens. 2 ×.

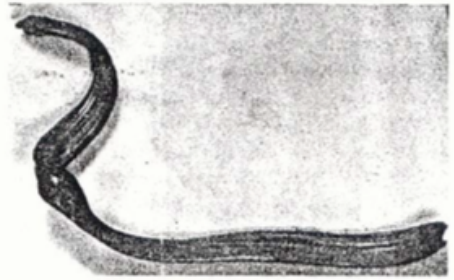


Fig. 2. Typical postirradiation appearance of Zr-5 wt % Pu fuel specimens. 2 ×.

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

Zr-Pu alloys at 500°C (irradiated with fission fragments)

There are three stages of irradiation growth

1. Initial rapid growth to small strains
2. Slow growth, gets skipped in cold worked material
3. Accelerated or breakaway growth, caused by formation of vacancy loops on the basal plane

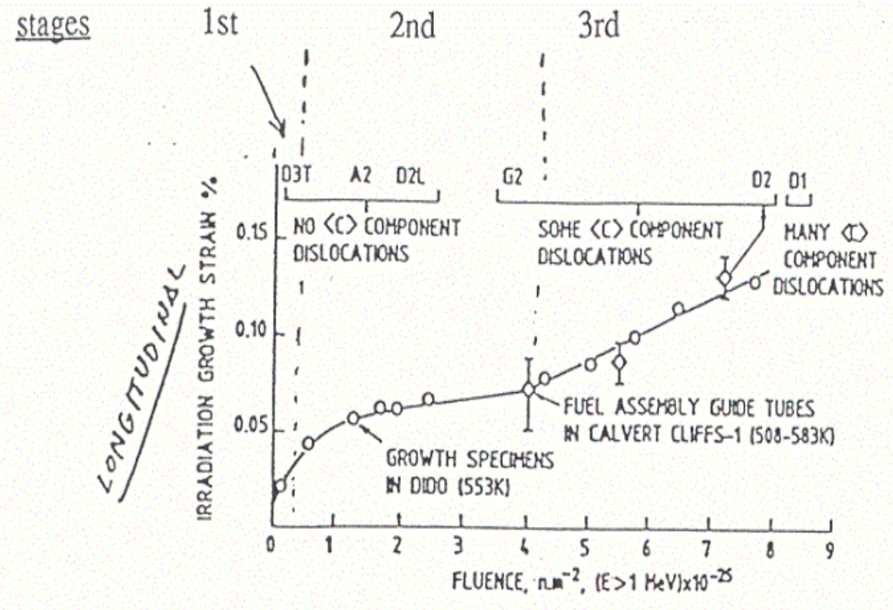
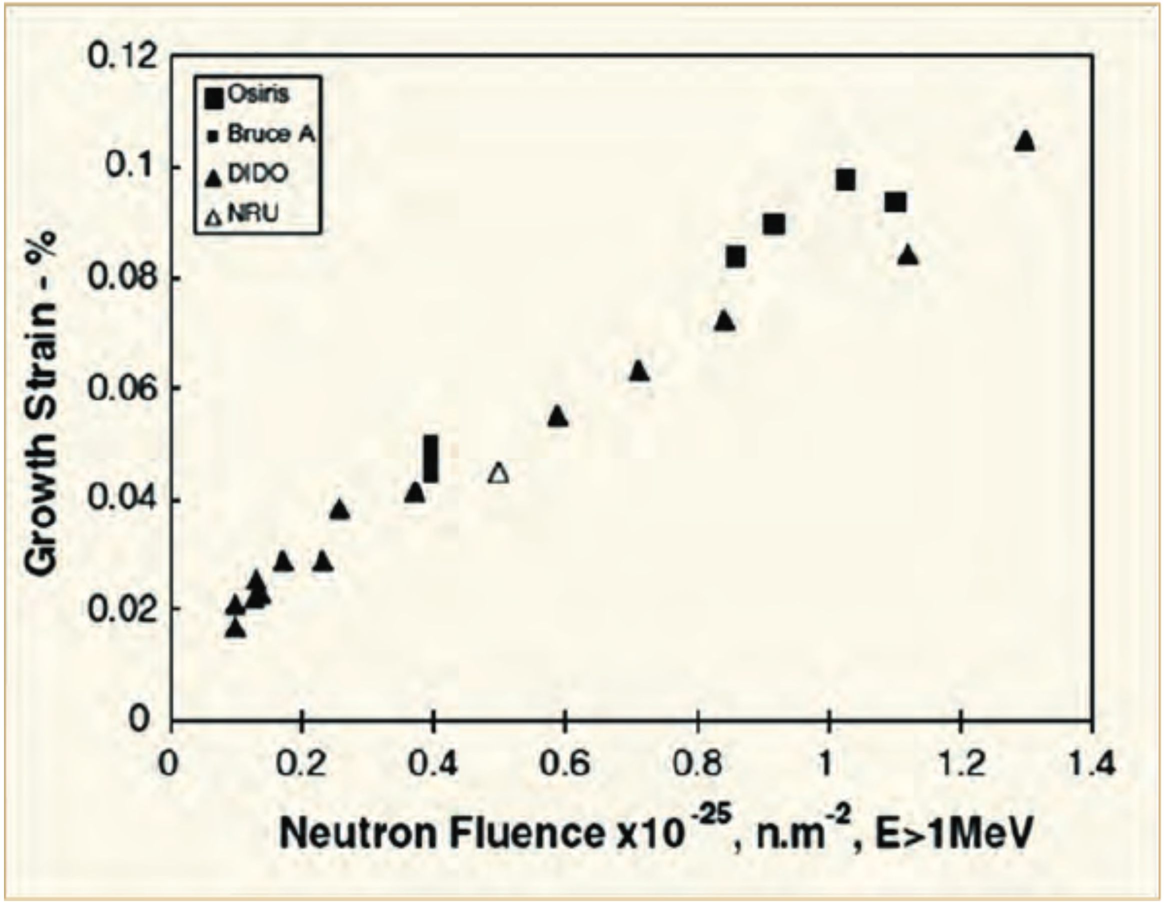


FIG. 11—*Irradiation growth in annealed Zircaloy at 550 to 580 K, showing accelerating growth at 4 × 10^{25} n/m^2, E > 1 MeV* [61].

Irradiation growth is strongly affected by several factors

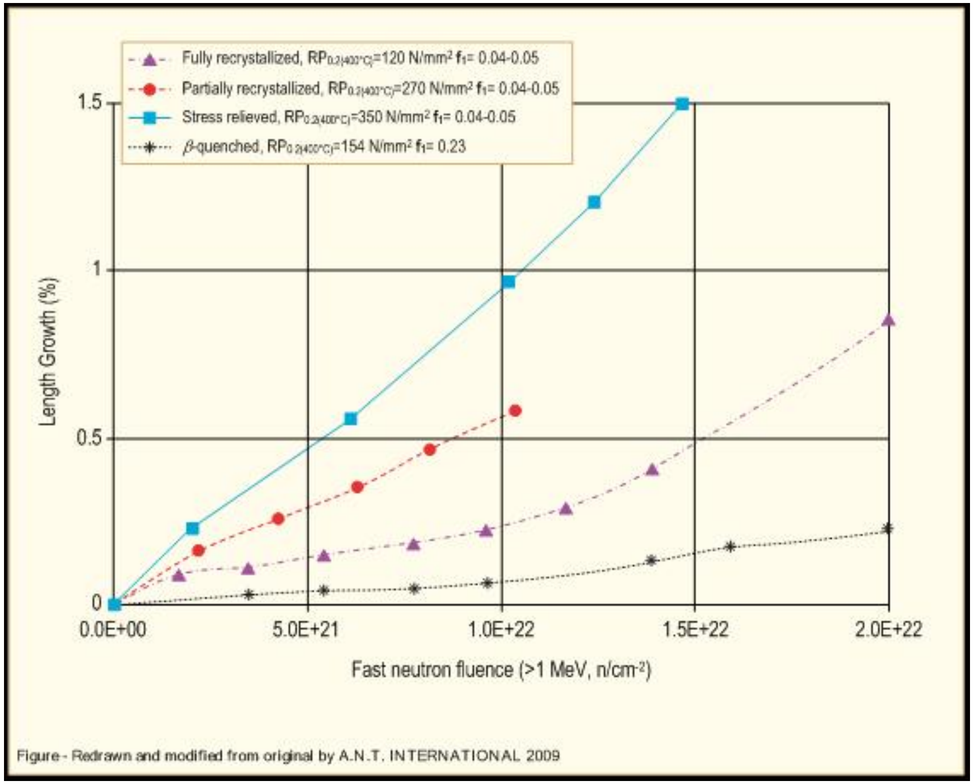
- Fluence
- Cold work
- Texture
- Irradiation temperature
- Material chemistry (alloying and impurity elements)
- Hydrogen effects

Growth is a linear function of fluence after break away



$$\varepsilon_G = K\phi^n \quad \text{where } n = 1$$

The amount of cold work also has a large effect



[Garzarolli et al., 1996]

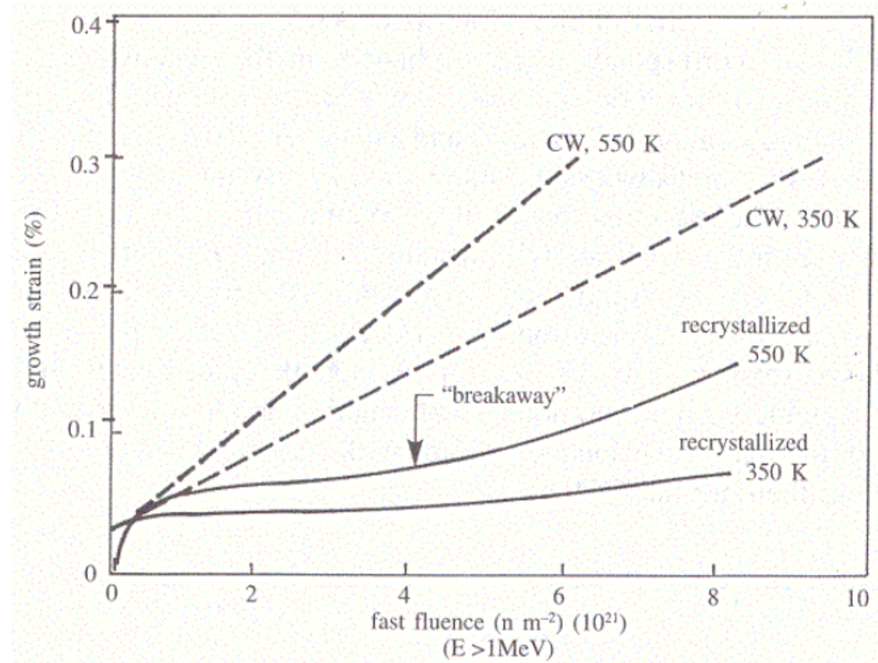
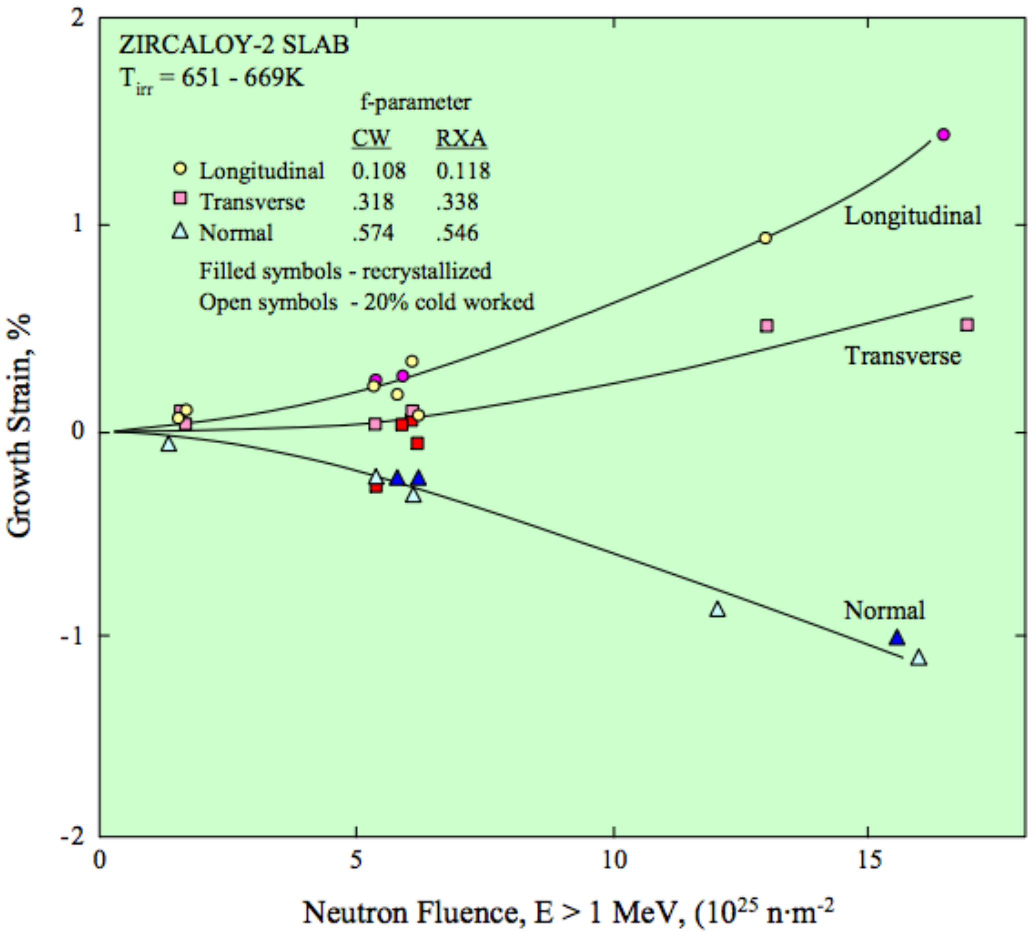


FIG. 4.27. – Growth strain versus dose and temperature.

The growth behavior depends on the texture and can be estimated with a simple equation

- The growth strain along macroscopic direction x is
  - $\varepsilon_{gx} = \varepsilon_0 (1 - 3 f_x)$
- where f_x is the resolved fraction of basal planes aligned with direction x.



Example problem

At a fluence of $0.4*10^{-21}$ n/cm^2 and with 10% of the grains oriented with the basal plane aligned to the axial direction, the growth strain in the axial direction of a cladding tube is 0.02 after set amount of time.

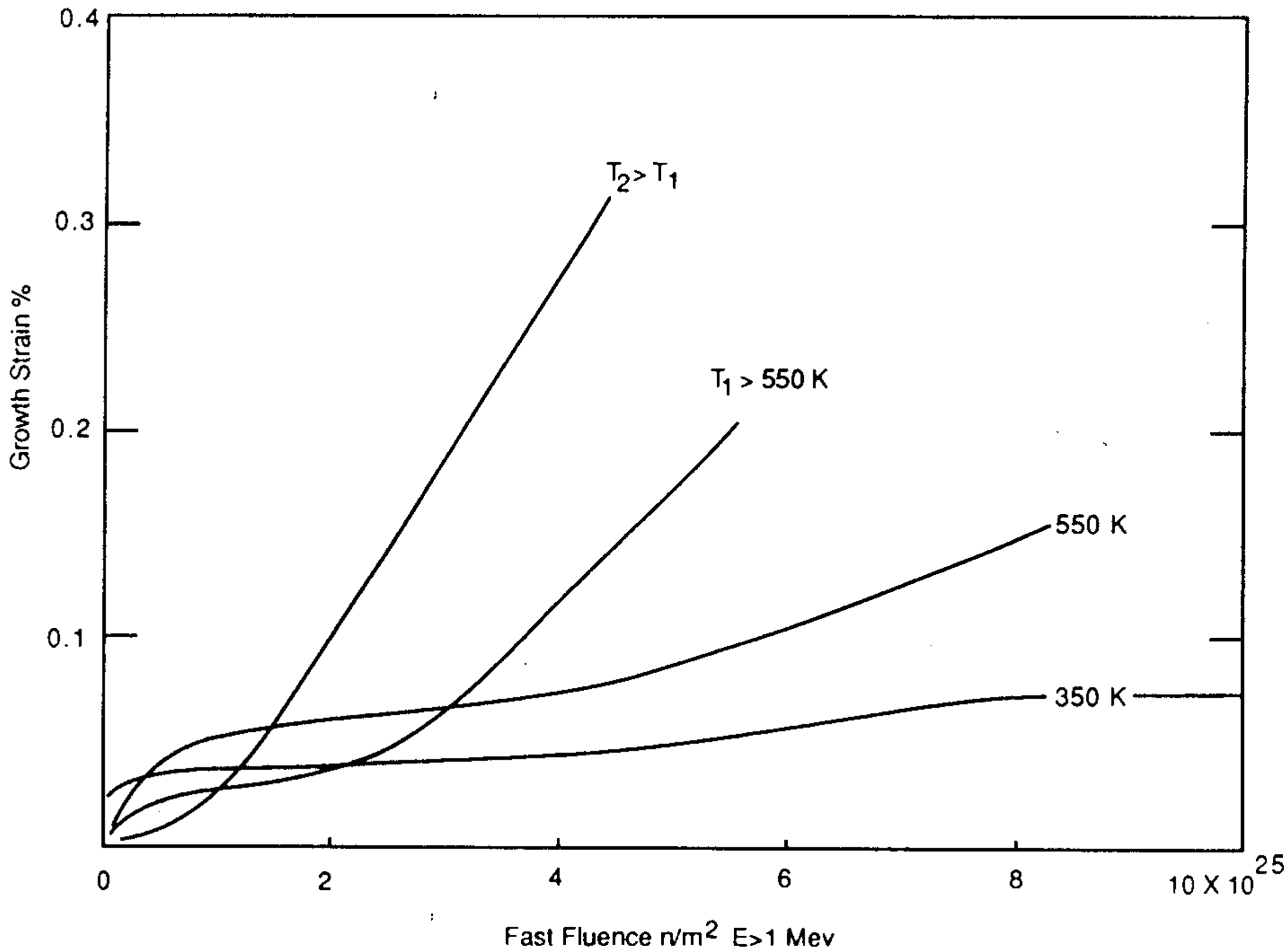
- What would the axial growth strain be in the same material after the same amount of time if 75% of the grains were oriented with the basal plane aligned to the axial direction? Remember $\varepsilon_{gx} = \varepsilon_0 (1 - 3 f_x)$

- First, we need to solve for ε_0,
  - $\varepsilon_0 = \varepsilon_{gx}/(1 - 3 f_x) = 0.02/(1 - 3*0.1) = 0.0286$
- Then, solve for the new strain
  - $\varepsilon_{gx} = \varepsilon_0 (1 - 3 f_x) = 0.0286 (1 - 3*0.75) = -0.0358$

- What would the growth strain be in the original material if the fluence was $0.7*10^{-21}$ n/cm^2? Remember that the strain varies linearly with the fluence.

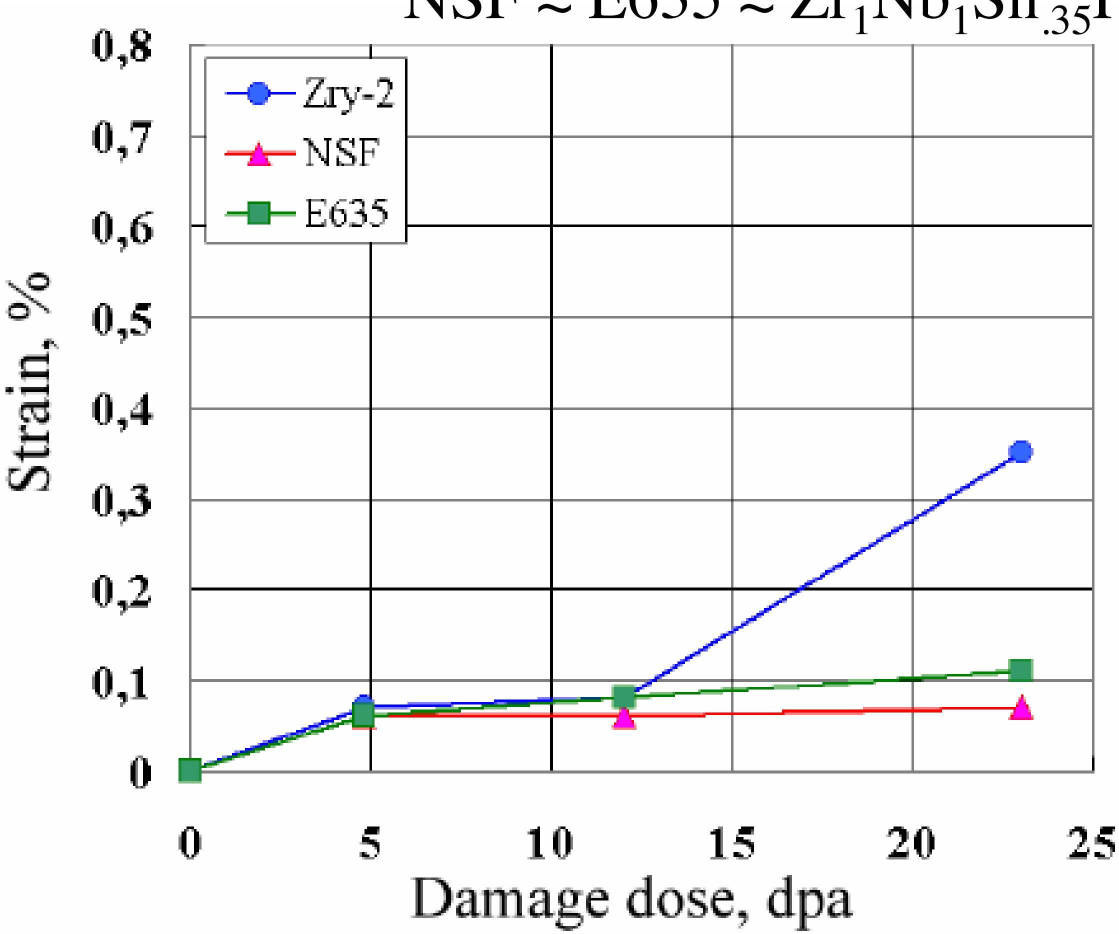
- It varies linearly with fluence, so
  - $\varepsilon_0 = 0.02(0.7/0.4) = 0.035$

Higher temperature leads to earlier breakaway and higher growth



Different zirconium alloys can have drastically different growth rates

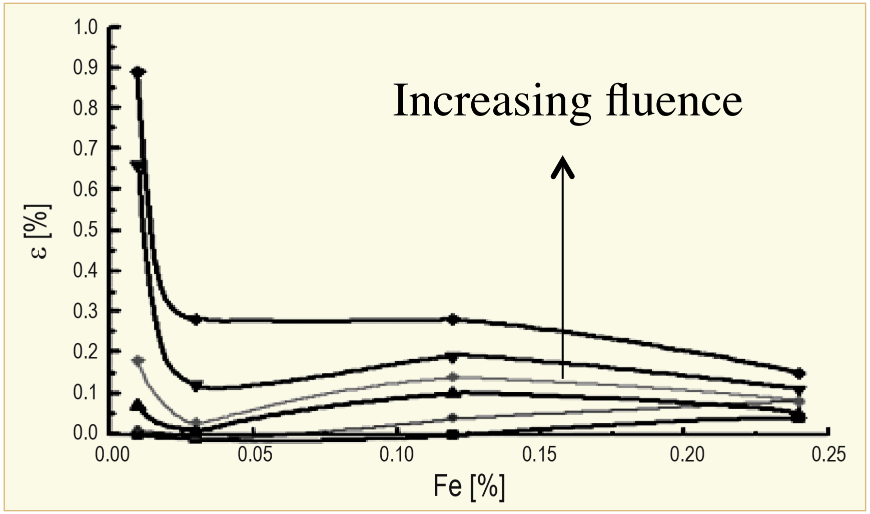
NSF ≈ E635 ≈ $Zr_1Nb_1Sn_{.35}Fe$



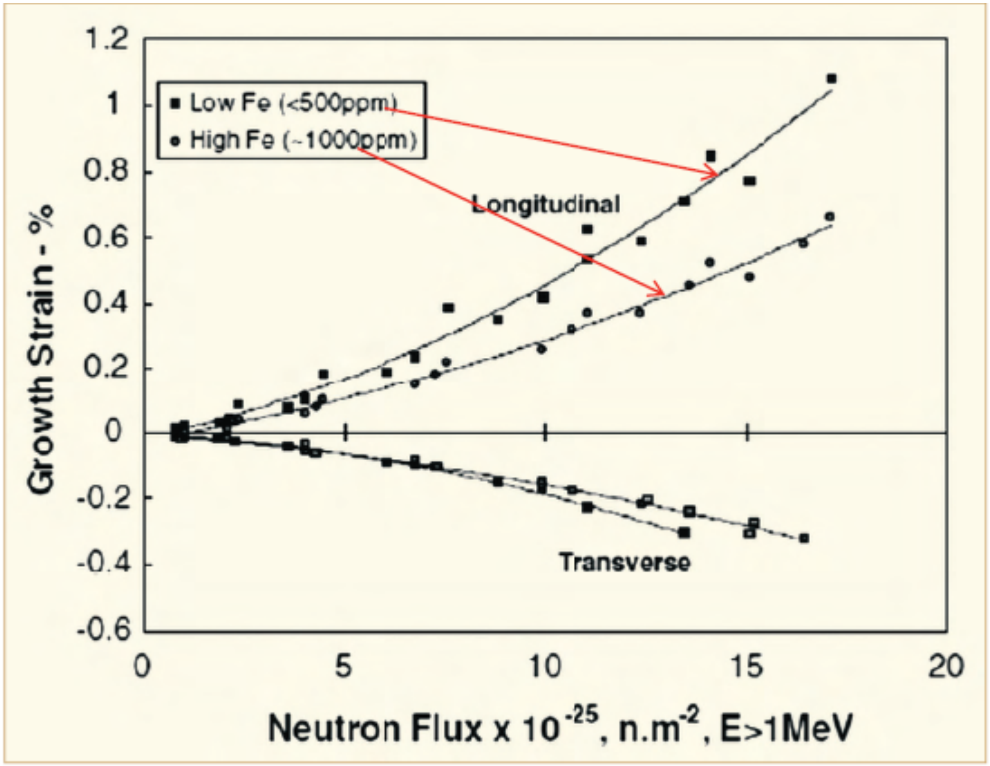
Nb-containing alloys like E635, ZIRLO, M5 have low growth

RX

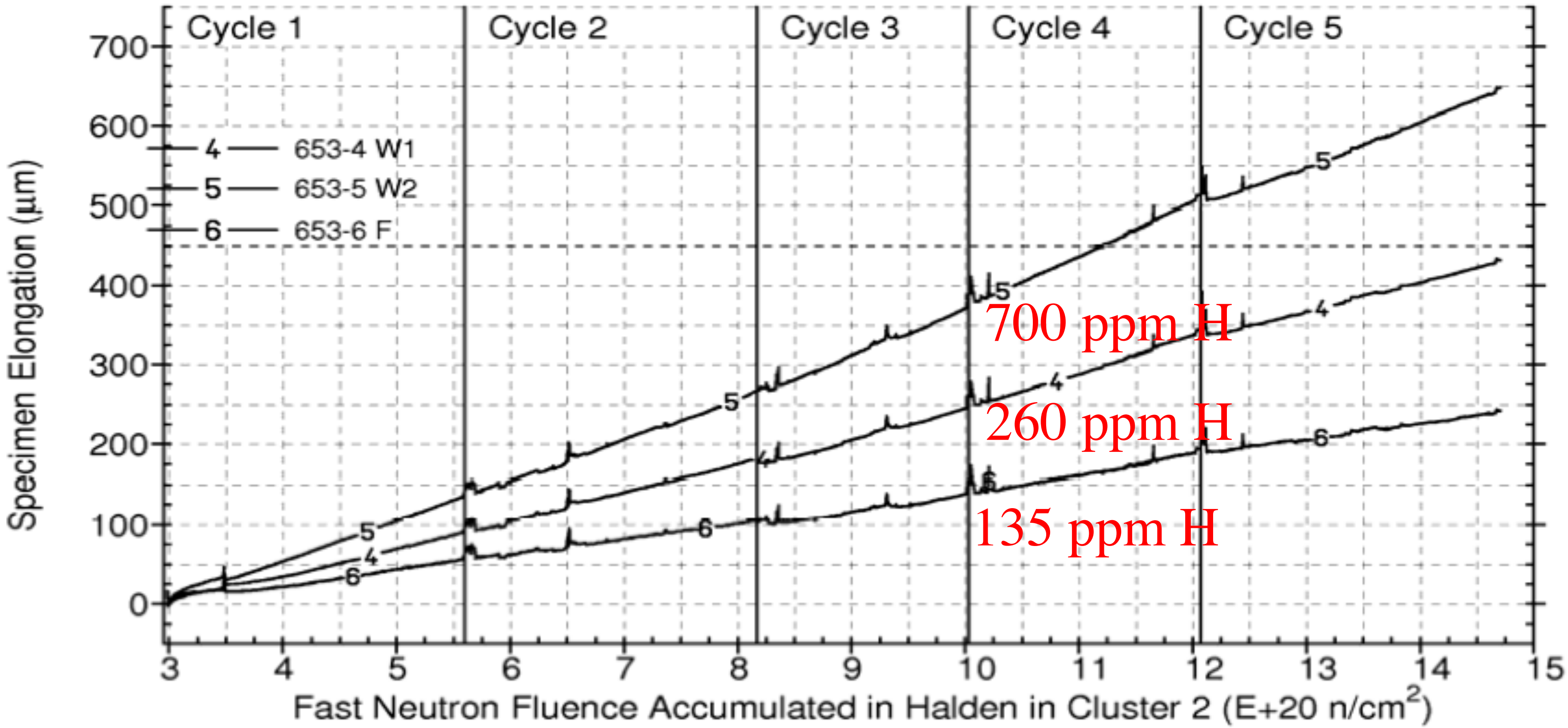
Growth also goes down with increasing Fe content



Shishov, 2010



The growth increases as the hydrogen concentration increases



McGrath et al., 2010

Gradients in neutron fluence result in gradients in growth that cause bowing

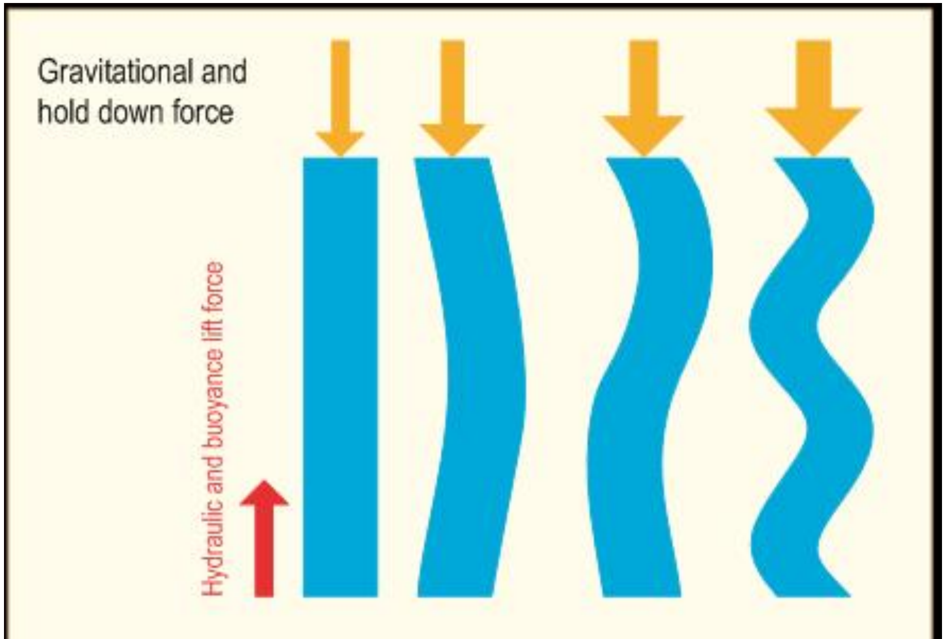
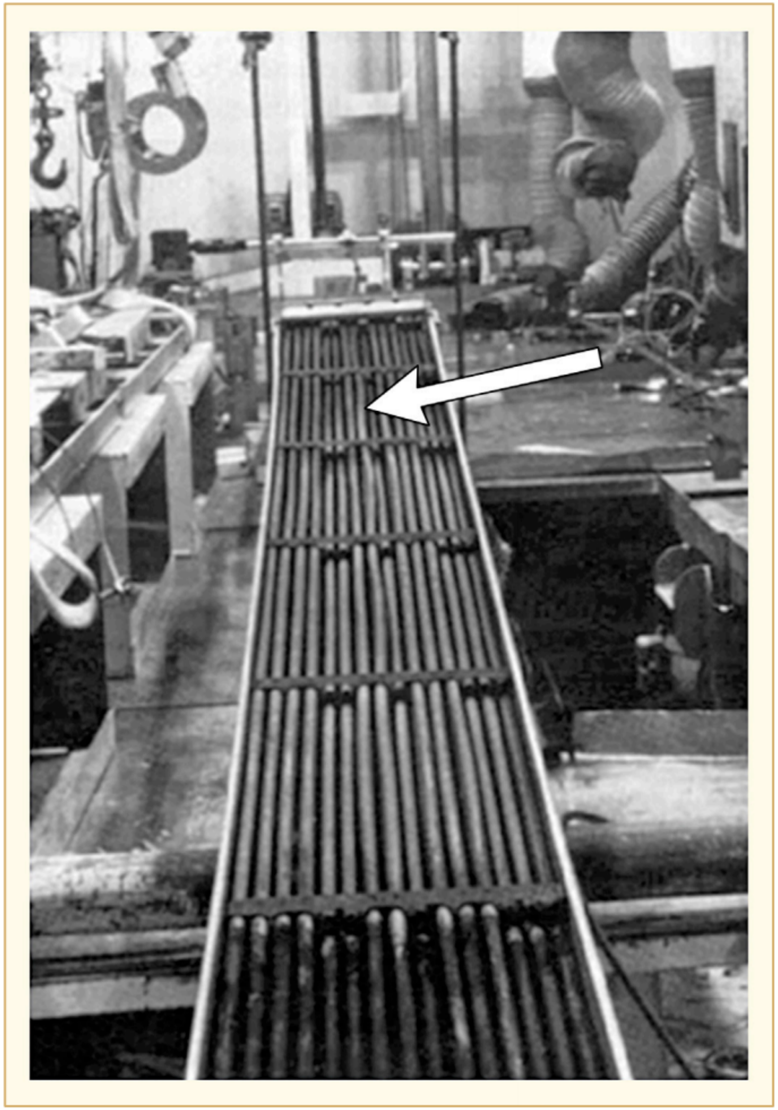
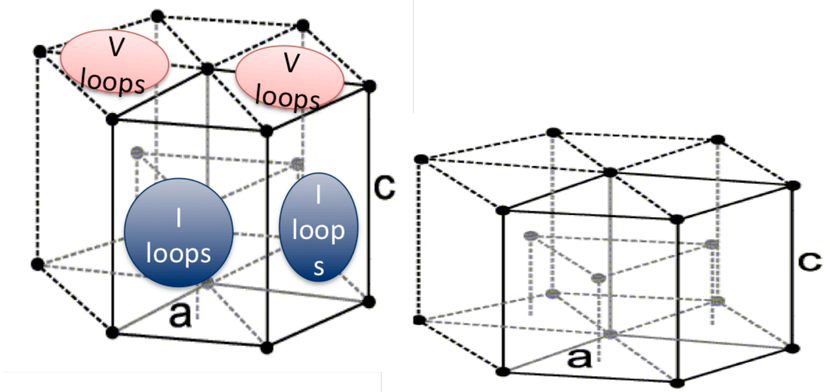


Figure - Redrawn and modified from original by A.N.T. INTERNATIONAL 2007



Summary

- 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