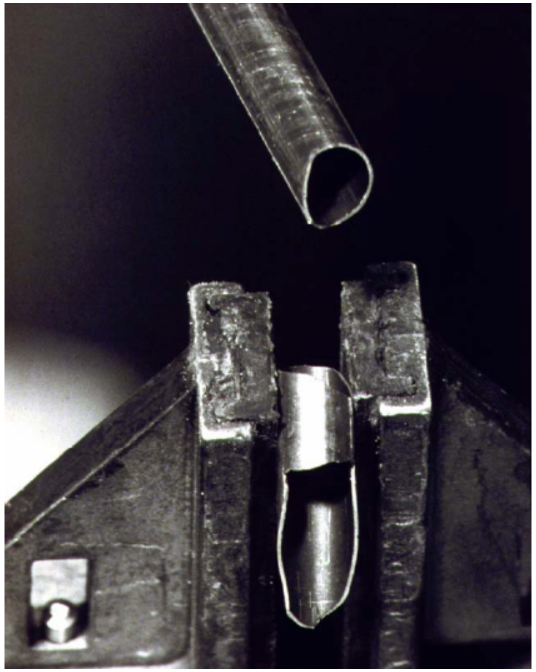
Structural materials for liquid metal cooled reactors



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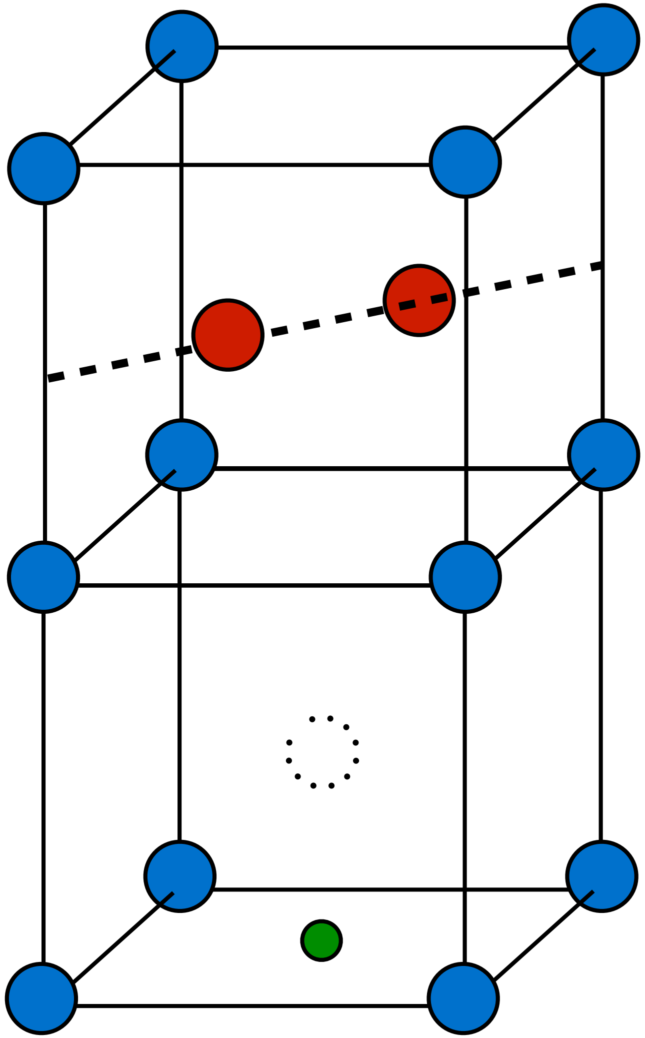
Outline

- Point defects and their production
- Swelling
- Austenite versus ferrite
- Embrittlement
- Thermal creep
- Corrosion

Point defects



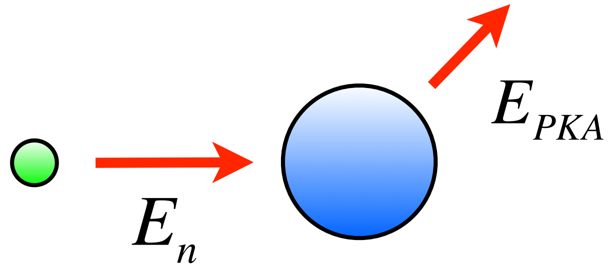
BCC lattice with point defects

- When an atom in a crystal is knocked out of a lattice position by energetic neutrons, interstitial-vacancy pairs are created.
- Dislocations are large clusters of self-interstitials.
- Impurities like He reside either in substitutional or interstitial positions.
- Cavities are clusters of vacancies.
- Bubbles arise when He atoms aggregate in cavities.

Displacement damage

- For isotropic elastic scattering one has

$$E_{PKA} = \frac{2m_n m_{PKA}}{(m_n + m_{PKA})^2} E_n \approx \frac{2m_n}{m_{PKA}} E_n$$



- Thus, neutrons typically transfer 1/29 of their energy to iron atoms
- Average recoil energy required for displacing an atom and creating an interstitial-vacancy pair in BCC iron: $E_{dis} \approx 40$ eV.
- Average neutron collision energy in fast reactor: 300 keV -> $E_{PKA} \approx 10$ keV
- Damage production rates are orders of magnitude higher than in light water reactors.

Displacement production rate

- Kinchin & Pease expression for total number of Frenkel defects produced by single recoil:

$$\nu_d = \frac{E_{dam}}{2E_{dis}}$$

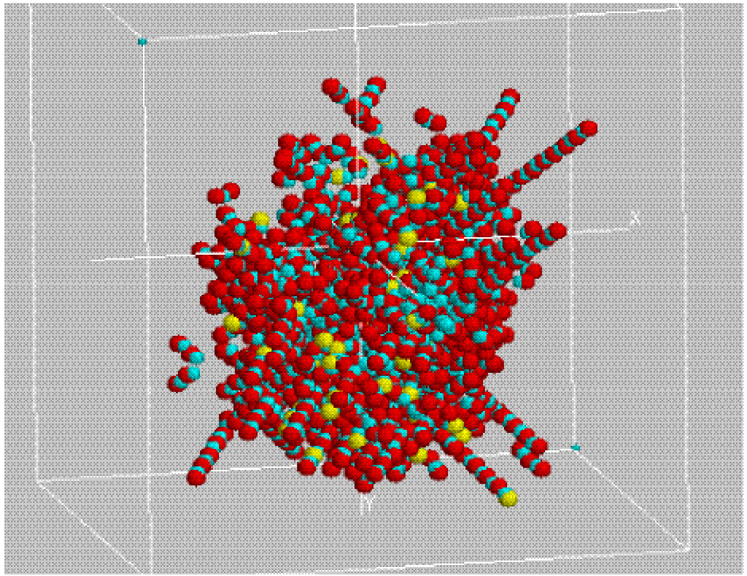
- The available damage energy E_{dam} is lower than E_{PKA} due to ionisation of atoms.

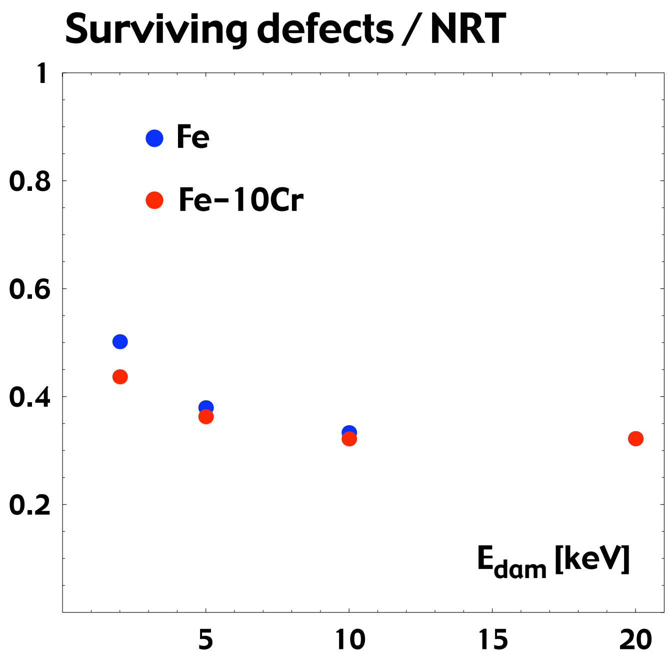
- A damage dose of 1 dpa (displacement per atom) is obtained when all atoms in a material have been displaced from their original position once (on average).

- Usually calculated by a modified Kinchin & Pease formula called ”NRT standard”, after Norget, Torrens and Robinson.

- Typical displacement production rate in fast reactors: 40 dpa/year

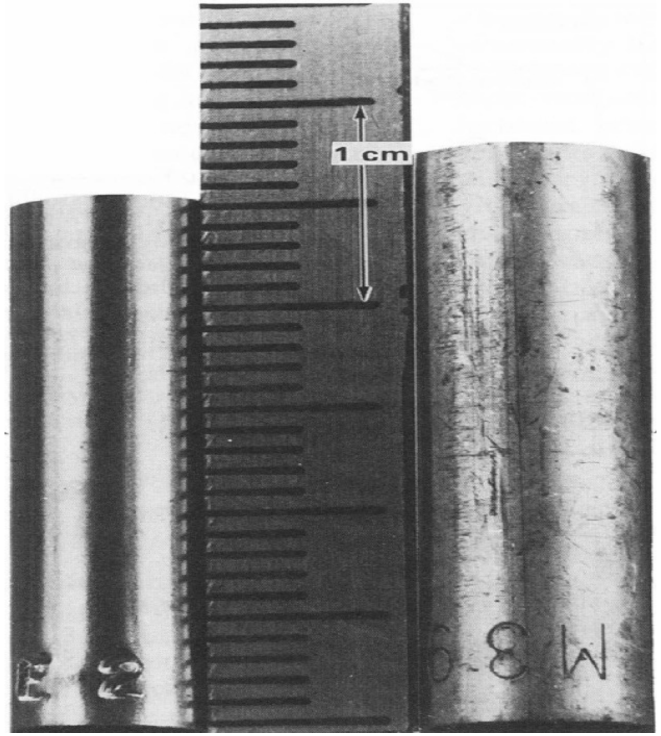
Damage production: simulation





- Kinchin-Pease & NRT do not take into account recombination of defects during cascade cooling down.
- Simulations show that the number of surviving defects in iron is ≈ 1/3 of the NRT standard, in agreement with experiment.

Irradiation induced swelling

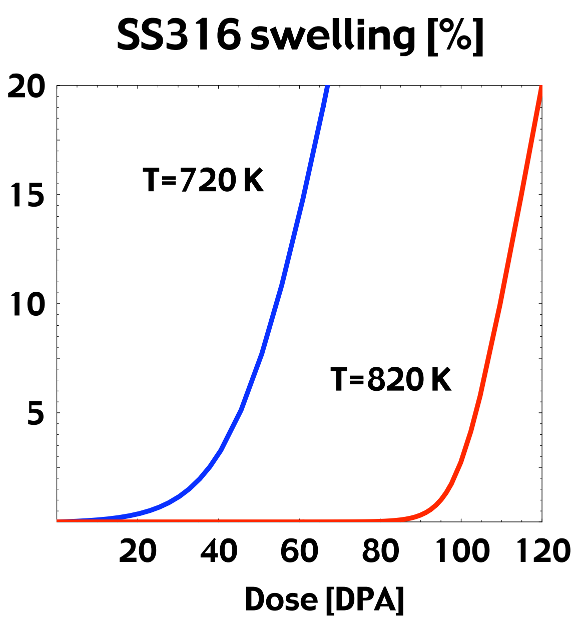


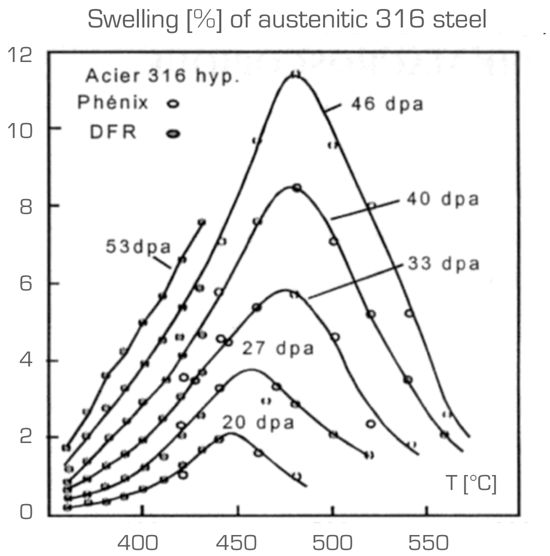
Before irradiation

After irradiation

- SS316 cladding tubes irradiated to 80 dpa at 510°C
- 33% increase in volume
- Swelling due to formation of voids under irradiation
- Leads to void induced embrittlement

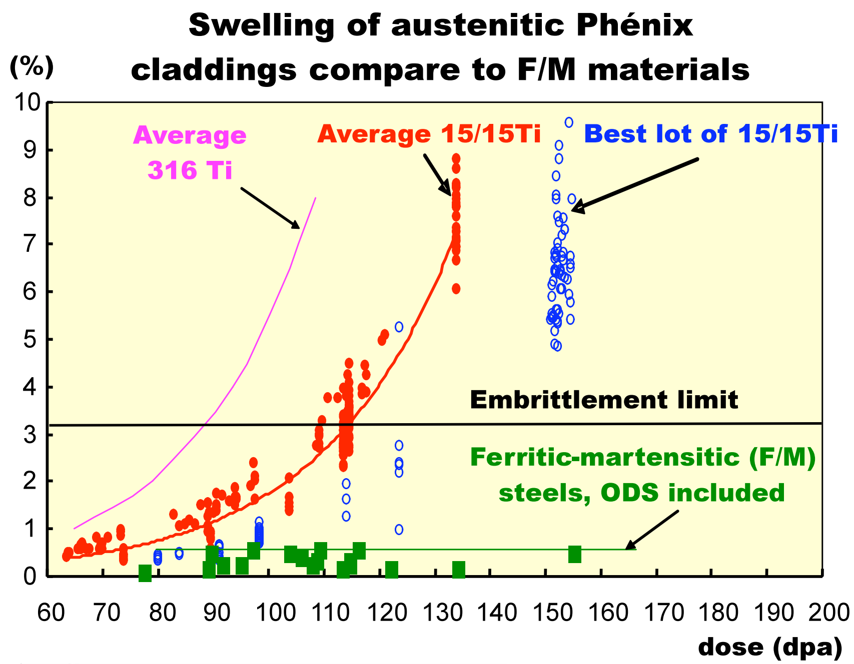
Swelling as function of temperature and dose

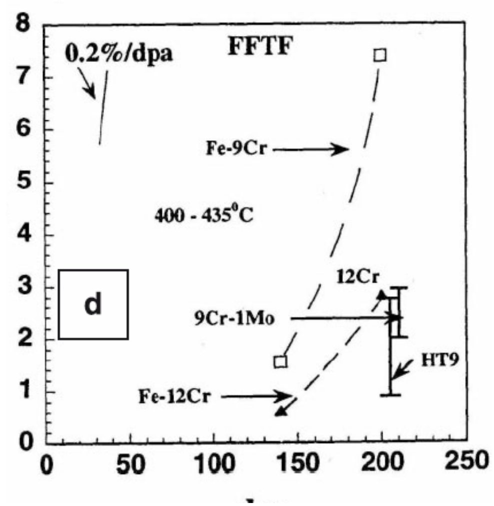




- Initial swelling rate due to cavity formation and grain growth, small in magnitude.
- Accelerated swelling phase driven by helium bubble formation.
- Minimum incubation period for SS316 observed for T = 450 - 500°C.
- At lower T, vacancies are immobile and do not form clusters/cavities.
- At higher T, vacancy emission rate from clusters is too large for cavities to reach “critical radius”.

Austenitic versus ferritic-martensitic steels

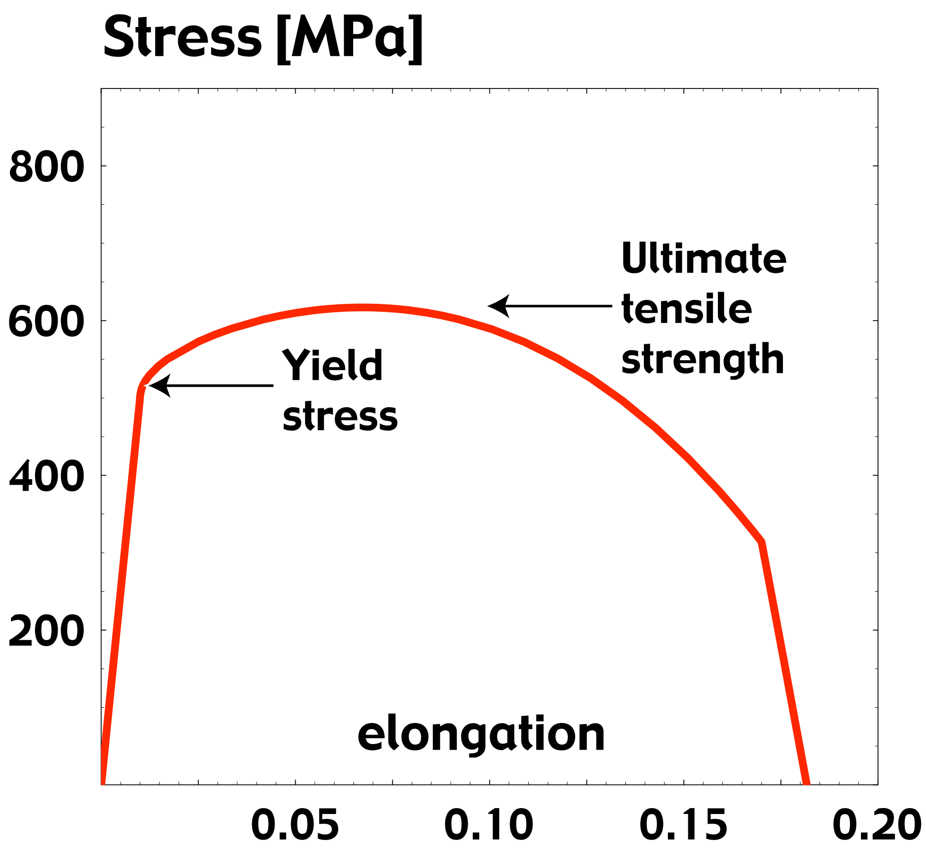




- Helium production due to (n,α) reaction twelve times larger in Ni (fcc structure stabiliser) than in Fe or Cr.
- Austenitic steels have an fcc structure, in which dislocations (interstitial clusters) are less mobile.
- These dislocations preferentially absorb individual interstitial defects, allowing vacancies to form voids and cavities.
- Threshold dose strongly dependent on chemical composition & microstructure. Addition of titanium and cold work reduces swelling.
- Maximum allowed swelling: 3%
- Dose limit for 15/15Ti $\approx$ 120 dpa
- Optimised ferritic-martensitic steels resist swelling up to 200 DPA!

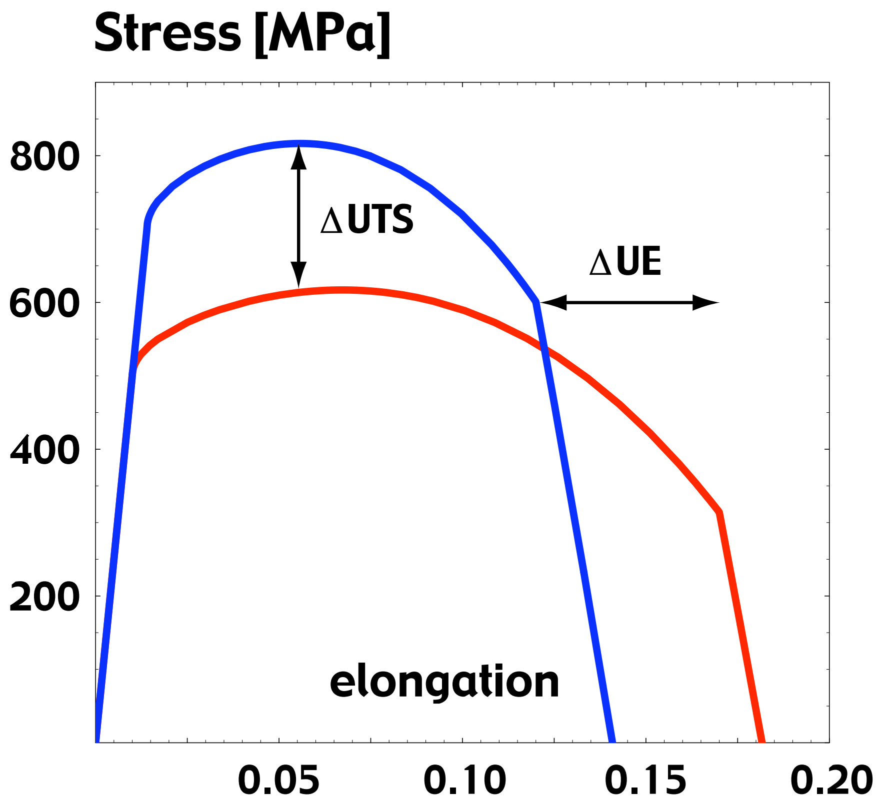
Mechanical properties: tensile test

- Steels subjected to sufficiently high mechanical loads or pressure will fracture
- The fracture may be of ductile or brittle character.
- Zero plastic elongation = brittle
- Combination of high yield stress with large elongation (ductility) is desired.



Irradiation hardening and embrittlement

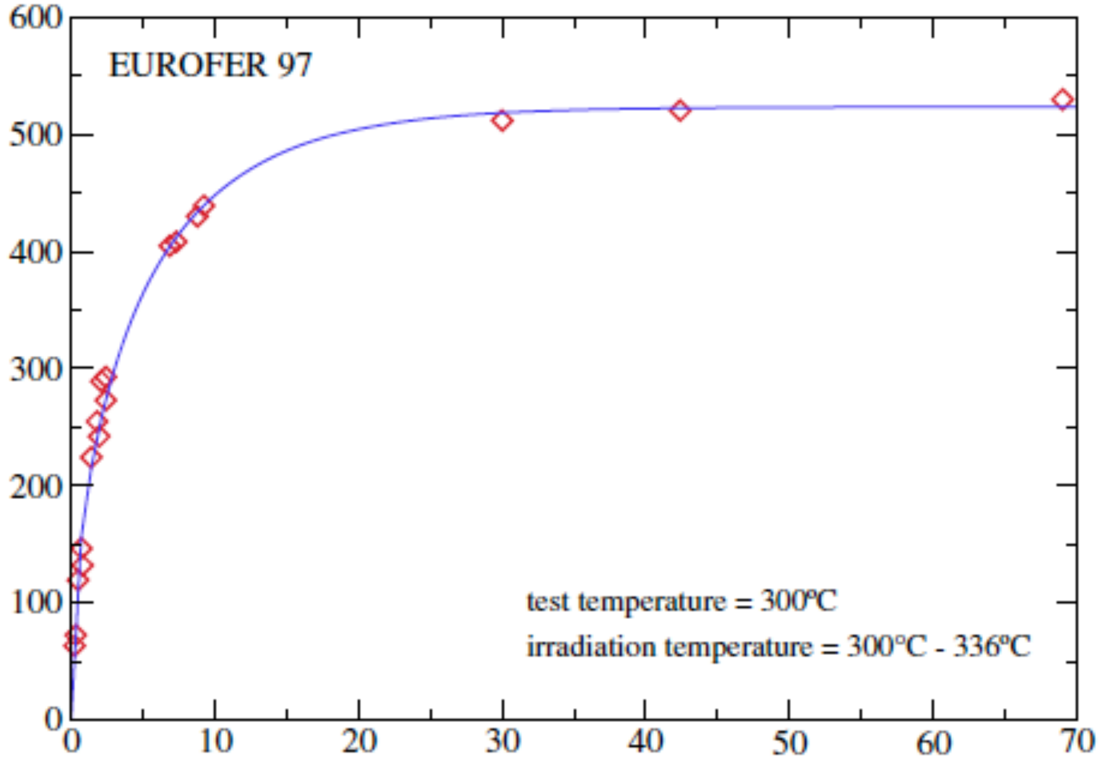
- Irradiation leads to changes in Yield Strength (YS) and Ultimate Tensile Strength (UTS).
- Increase in strength = hardening. Increased density of defects acting as obstacles for dislocation movement.
- Reduction in Ultimate Elongation (UE) = loss in ductility = embrittlement.
- Hardening usually combines with loss of ductility. Loss in ductility may occur even without hardening.



Saturation of hardening

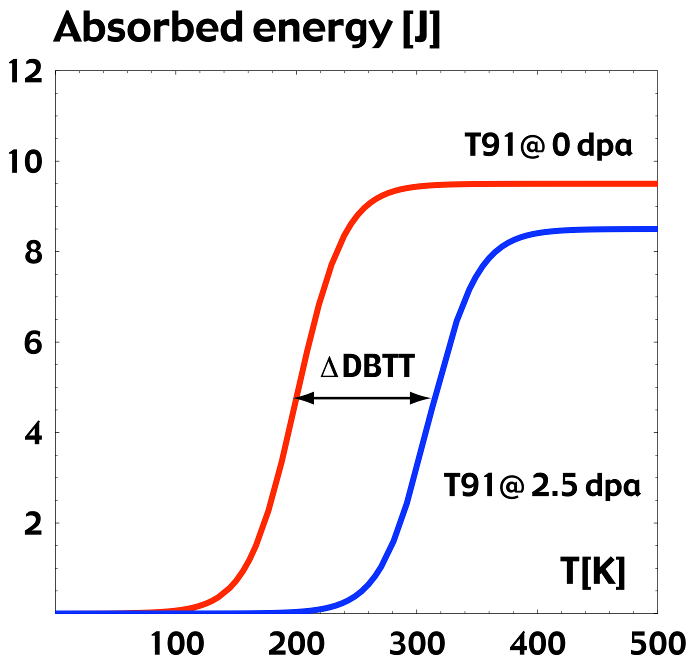
- Hardening saturates after a few tens of dpa in a fast spectrum

Increase in yield strength [MPa]



Dose [dpa]

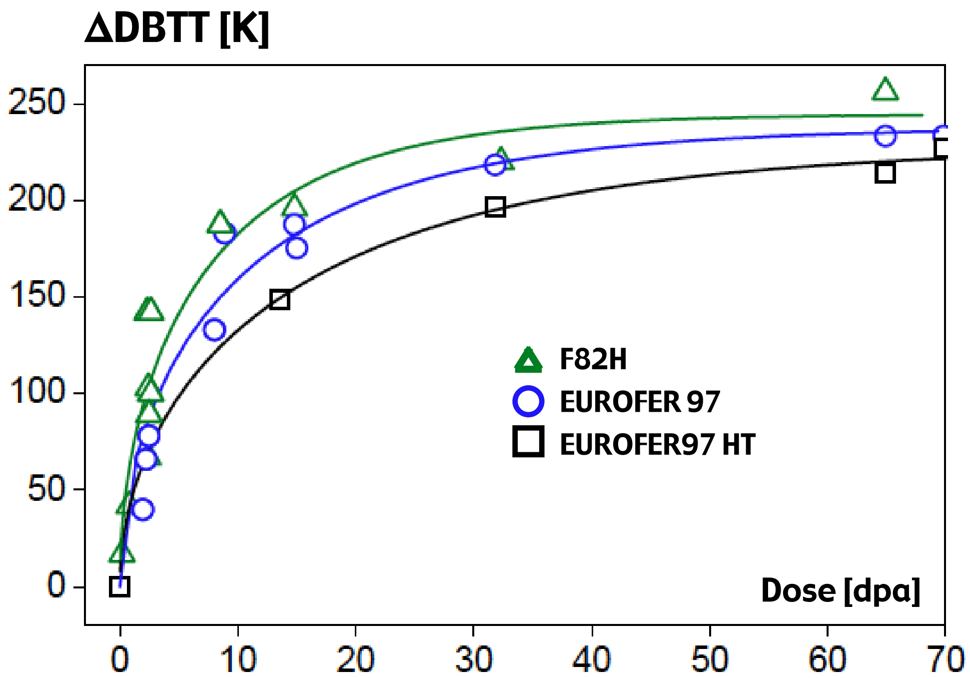
Ductile to brittle transition



Charpy test

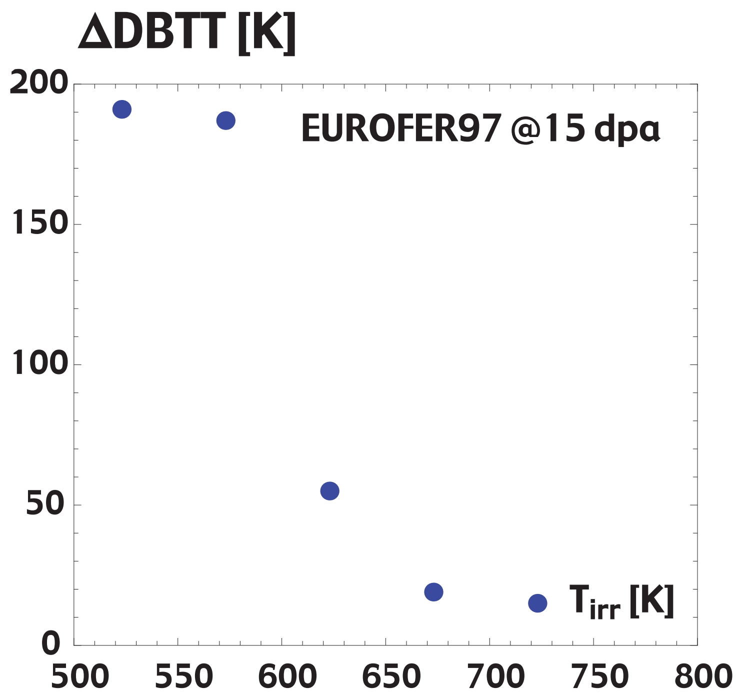
- Austenitic steels are ductile for doses up to the swelling threshold
- Ferritic steels (and bcc metals in general) are brittle at T < 200 K.
- The ductile to brittle transition temperature (DBTT) increases with irradiation dose.
- Dose of few dpa shifts DBTT above room temperature

DBTT shift as function of radiation damage dose



- 9Cr1Mo steels irradiated at T = 330°C exhibit ΔDBTT > 200 K!
- Major part of shift occurs within first 20 dpa.

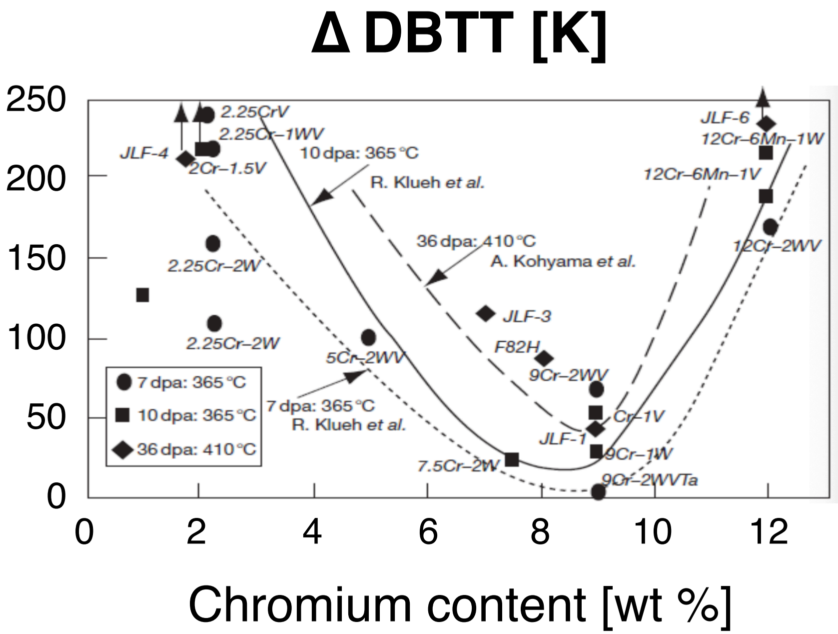
DBTT shift as function of irradiation temperature

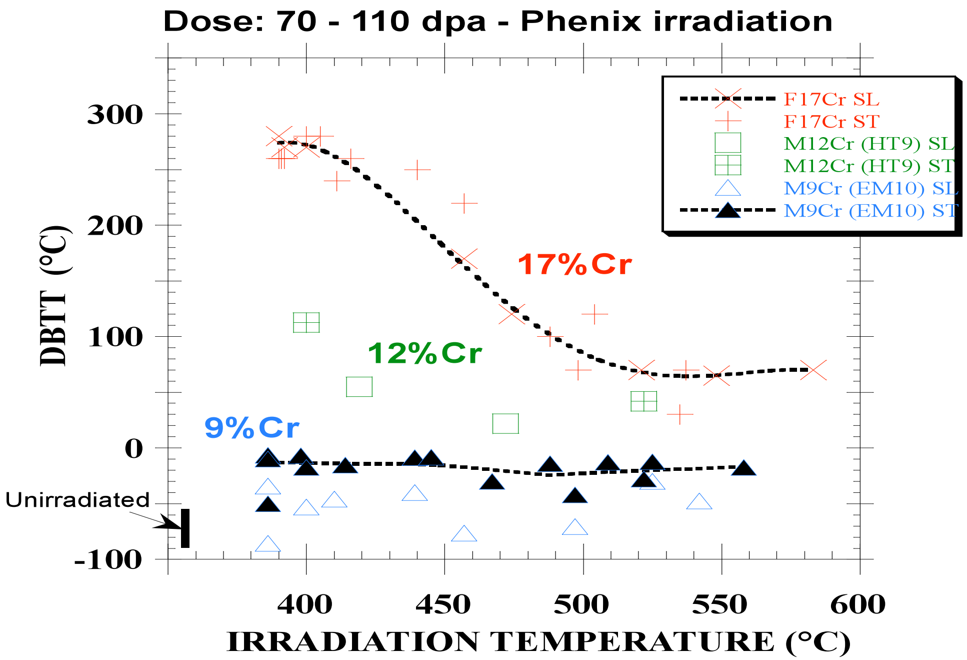


- For 9Cr steels, ΔDBTT is significant below irradiation temperatures of 600±20K
- For irradiation temperatures above 670 K, the shift is of no concern.
- The threshold will depend on Cr content

DBTT shift as function of steel composition and temperature

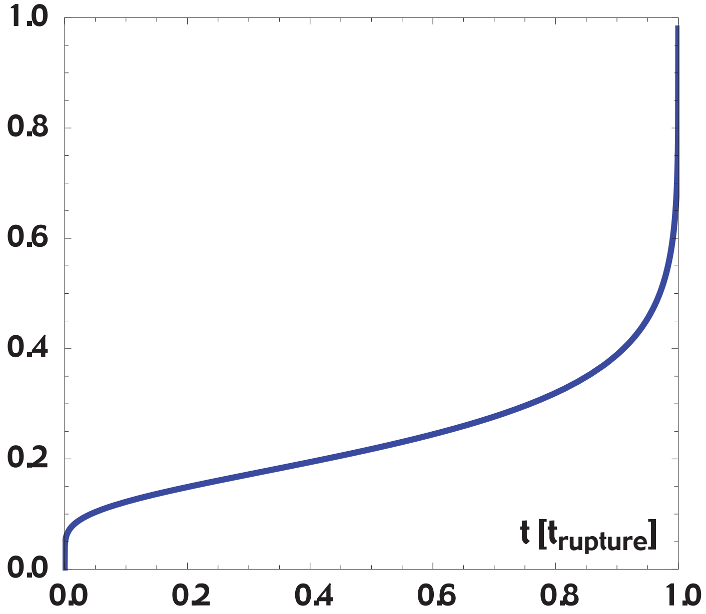
- The increase of DBTT is minimal for 9% Cr concentration
- 12Cr steels exhibit a significant ΔDBTT at $T_{irr} \geq 670$ K!



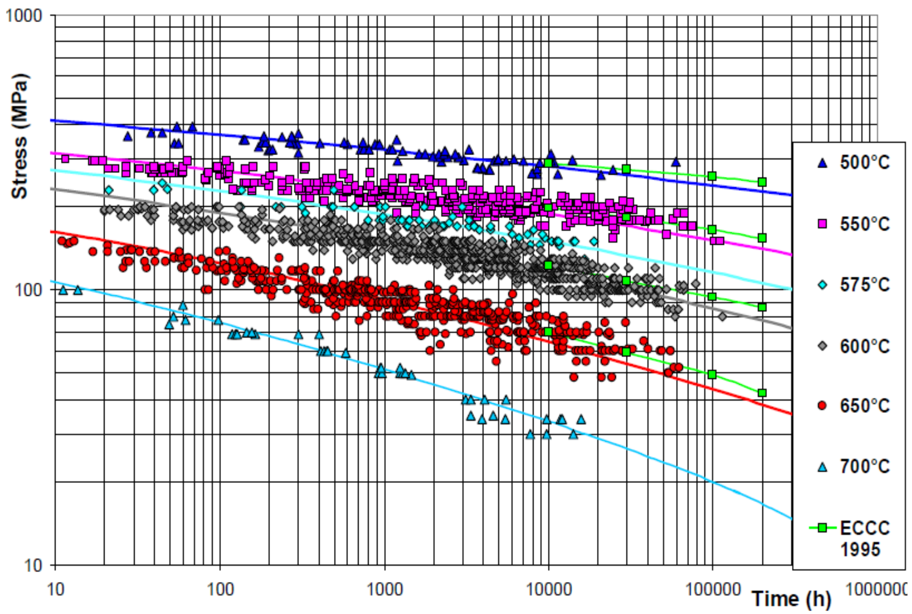


Thermal creep (≠ irradiation creep!)

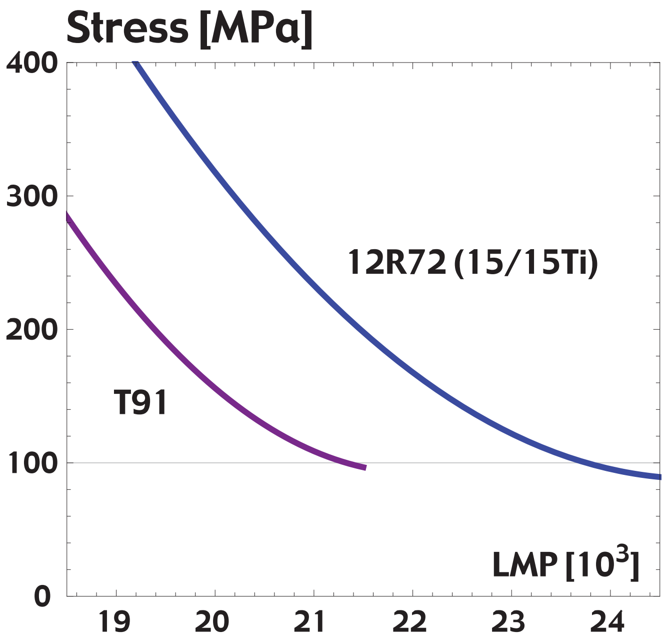
- Fission gas production in Gen-IV fuels leads to internal gas pressure in cladding tubes of up to 10 MPa at end of life, with corresponding Hoop stress ≈ 100 MPa.
- Rapid creep sets ultimate limit to clad temperature under transients.



Creep rupture life of 9Cr-1Mo steel



Thermal creep: Larson-Miller parameter

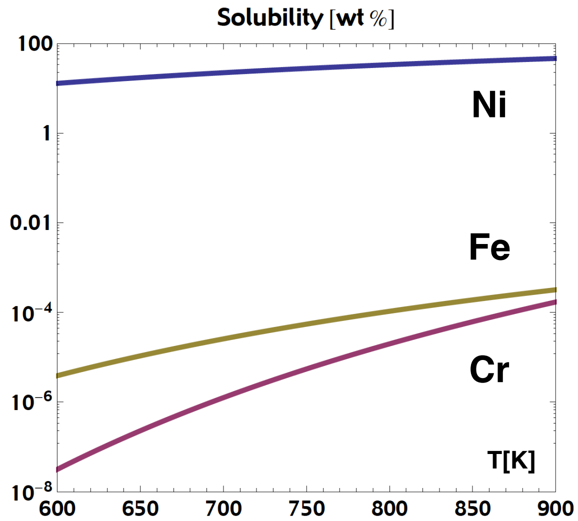


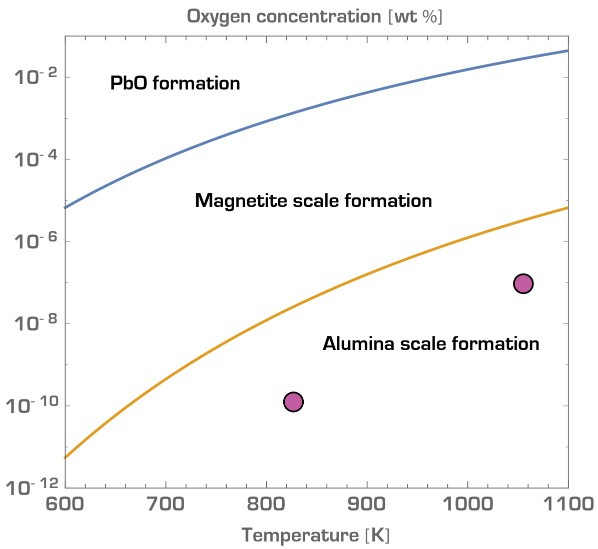
- Creep rupture time and temperature may be correlated through the Larson-Miller parameter P:
- $P = T\,(\mathrm{const} + \log[t_r])$
- Time for rupture t_r is expressed in hours, temperature T in Kelvin
- May be used for extrapolation to temperature and time ranges where experimental data do not exist!
- Creep rupture life time of best Ferritic Martensitic steels orders of magnitude lower than for austenitic steels

Exercise

- Calculate the time for clad rupture to occur for a 15-15 titanium stabilised austenitic steel and a T91 ferritic-martensitic steel subjected to 100 MPa load at
  - T = 820 Kelvin (normal operating temperature)
  - T = 1020 Kelvin
  - T = 1220 Kelvin
- Larson-Miller parameter P:
  - $P = T\,(A + log[t_r])*10^{-3}$, where t_r is expressed in hours, T in Kelvin. A = 20.
  - P = 21.5 for T91 at 100 MPa
  - P = 23.8 for 15-15Ti at 100 MPa

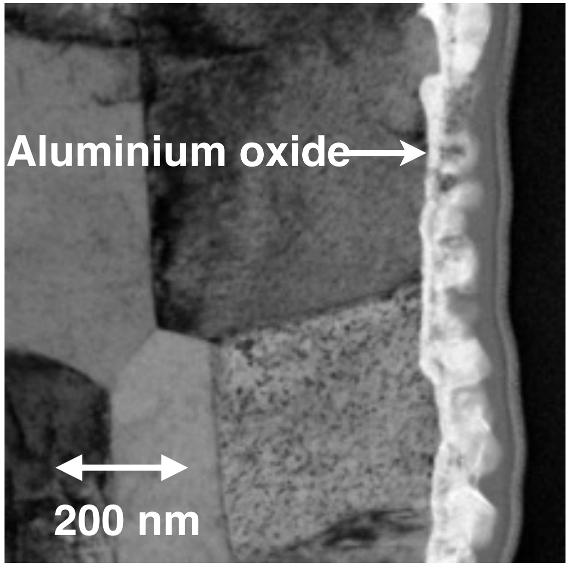
Corrosion of stainless steels



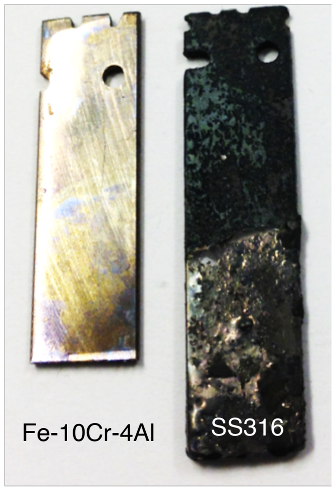


- Liquid sodium does not attack stainless steels under nominal operating conditions.
- Nickel dissolves in liquid lead.
- Formation of a metal oxide film on the surface of stainless steel prevents dissolution of nickel
- A minimum amount of oxygen must be dissolved into the coolant for the oxide film to form.
- A concentration of oxygen above its solubility limit leads to formation of lead oxide particles.
- Aluminium oxides (alumina) form at lower oxygen concentration than (Fe,Cr) oxides

Alumina forming alloys for lead-cooled reactors



AFA on SS316



- Chromium oxide films grow thick (≈ 100 microns), are mechanically unstable, and do not remain protective above 450°C in lead beyond exposure times of ≈ 1 year.
- Fe-10Cr-4Al-RE forms < 100 nm thin, protective, and ductile aluminium oxide film on its surface.
- RE = Reactive Elements: optimized blend of Ti, Zr, Nb
- Developed by KTH & Kanthal.
- Successfully exposed to stagnant lead at 550°C for two years and 850°C for ten weeks.
- 10 ton batch produced by Sandvik/Kanthal.
- Low creep strength. To be used as protective overlay on pressure boundaries, such as fuel cladding tubes and steam generator tubes.
- Alumina forming austenitic steels (AFA) are mechanically stronger, can be welded on SS316 vessels and are corrosion tolerant in liquid lead up to 650°C.

Choice of steels for liquid metal cooled reactors

Component	Sodium	Lead
Cladding tube	15-15Ti	15-15Ti + Fe-10Cr-4Al-RE overlay weld
Heat exchanger/Steam generator tube	Alloy 800	Alloy 690 + Fe-10Cr-4Al-RE compound tube
Hex-cans	Fe-9Cr-1Mo	Fe-10Cr-4Al-RE
Primary vessel	SS316L	SS316L + AFA overlay weld