# *Neutron irradiation effects – point defect balance equations, swelling, Rector pressure vessel*

P. Hosemann, Nuclear Engineering Department
University of California, Berkeley

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1. ***Materials selection strategie***
2. ***Defect balance equations***
3. ***Observed microstructural defects in fcc & bcc metals***
4. ***Void swelling***
5. ***Reactor pressure vessel***

UC Berkeley
NUCLEAR ENGINEERING

***References:***

1. ***Chapter 17 & 18*** *of D. R. Olander,* *Fundamental Aspects of Nuclear Reactor Fuel Elements,* NTIS (1976)

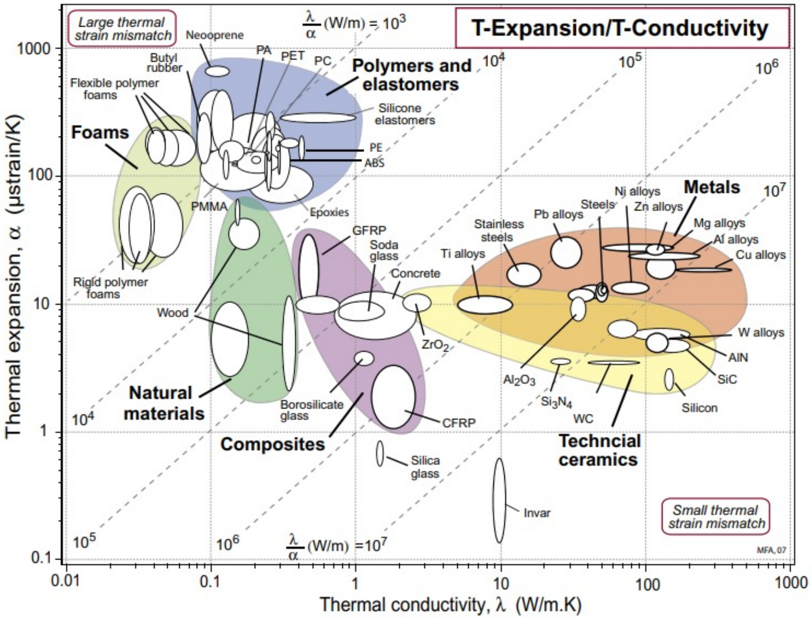
## *Engineering material selection for (nuclear) engineering application*

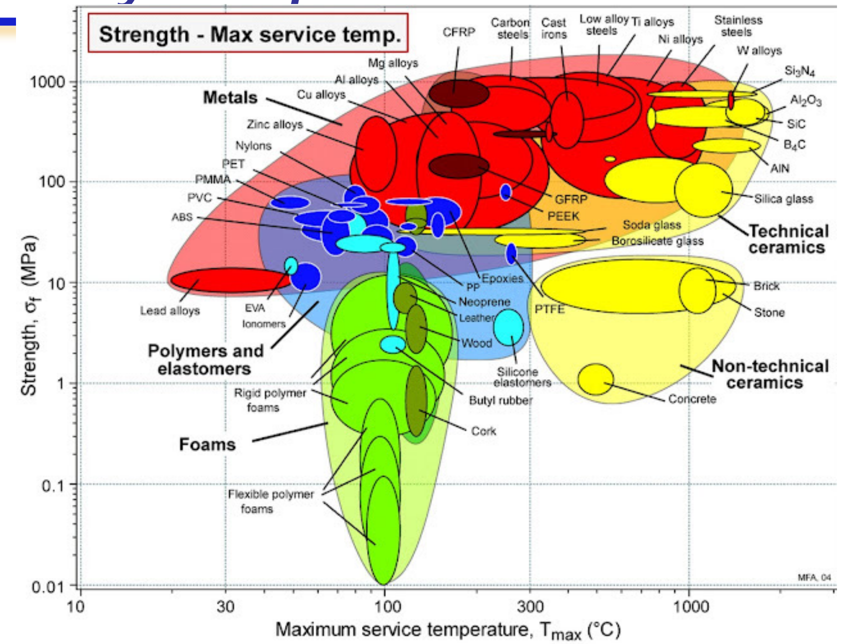
# Traditional engineering materials selection

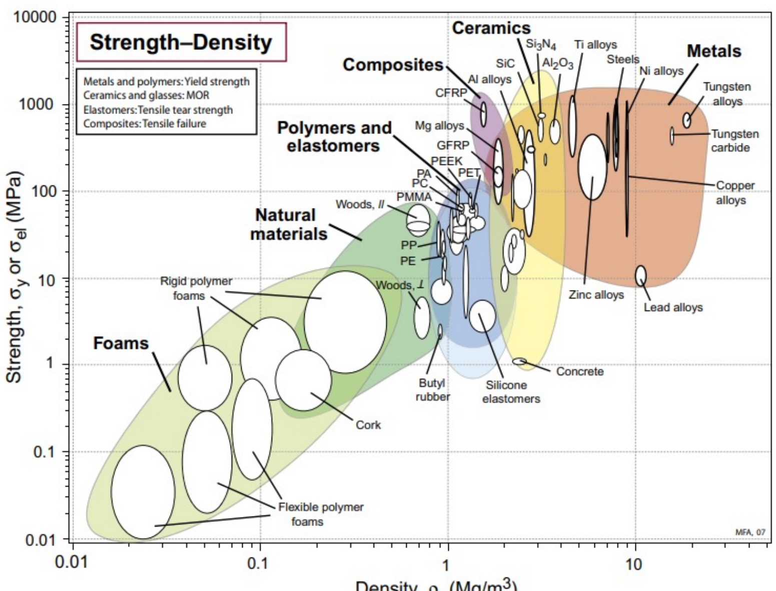
*P.Hosemann, M. Fratoni, D. Frazer, M.F. Ashby*
*Scripta Mat 2018*

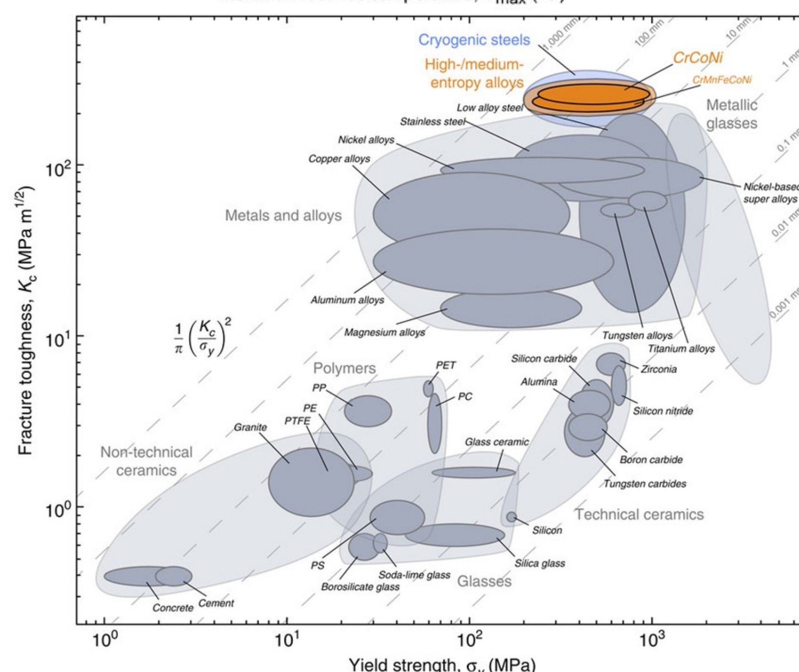
| *Requirement defined by the engineers* | *Materials Property* |
|---|---|
| ***Stiffness*** | ***Elastic Modulus*** |
| ***Temperature*** | ***Melting point, creep, etc*** |
| ***Stress*** | ***Yield stress*** |
| ***Lifetime*** | ***Creep,*** |
| ***Load cycles*** | ***Cycles to failure*** |
| ***Environment*** | ***Corrosion rates*** |
| ***Accident scenario*** | ***Elongation, fracture toughness*** |
| ***Light weight, X-ray absorption*** | ***Density*** |

# The concept of ashby maps

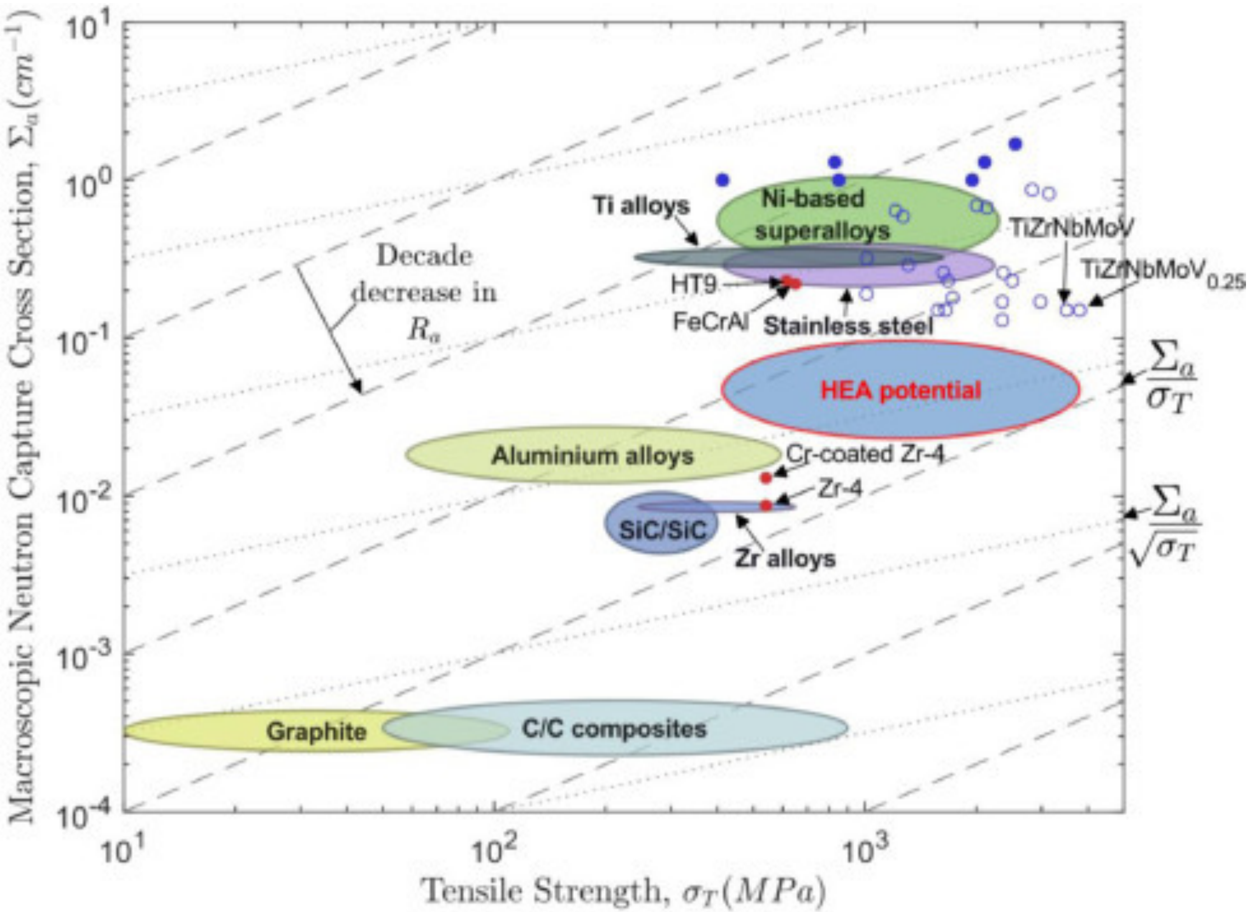


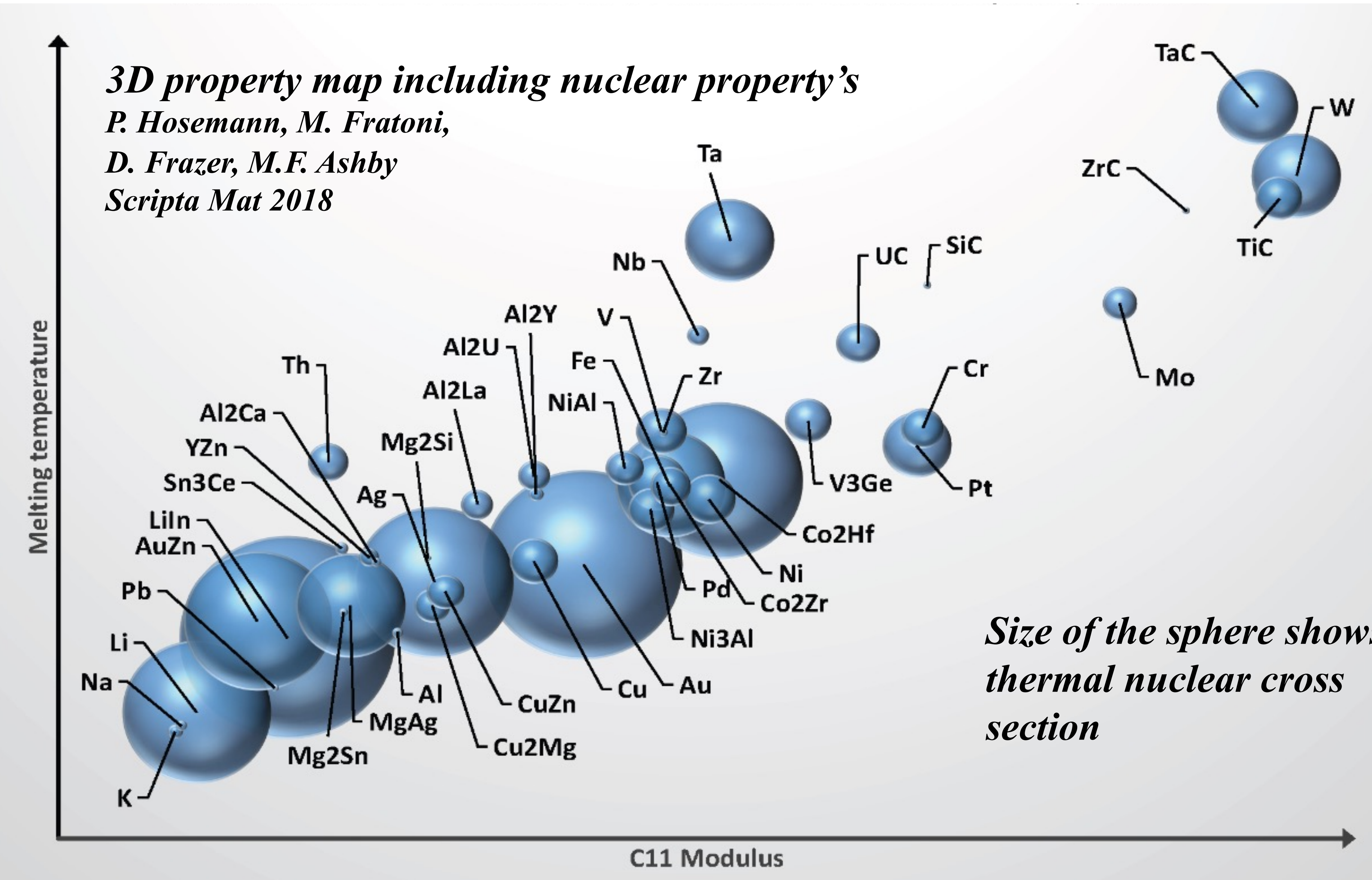
# Materials properties considering neutron capture




Michael Moschetti, Patrick A.Burr, Edward Obbard, Jamie J.Kruzic, Peter Hosemann, Bernd Gludovatz

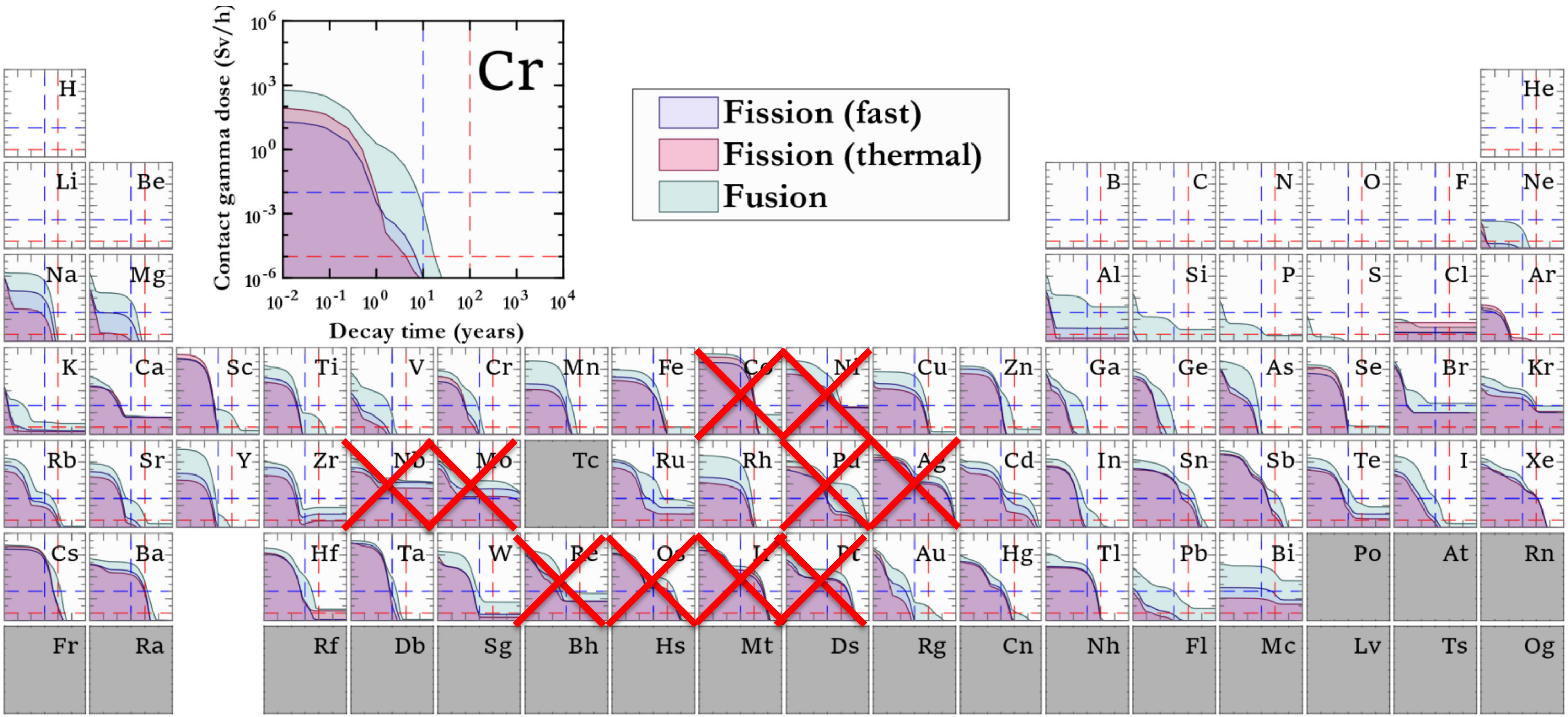
*Design considerations for high entropy alloys in advanced nuclear applications*
JNM 2022

# *Materials selection for nuclear applications*

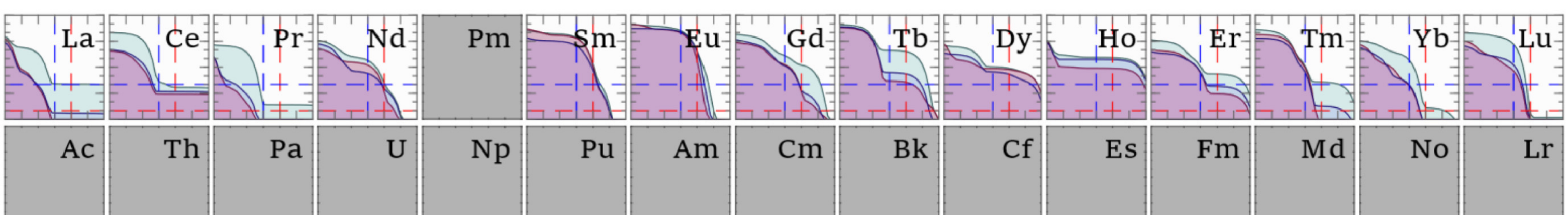


# Selection of elements for LA materials. Decay time after service

M. Moschetti, et al. JNM 2022

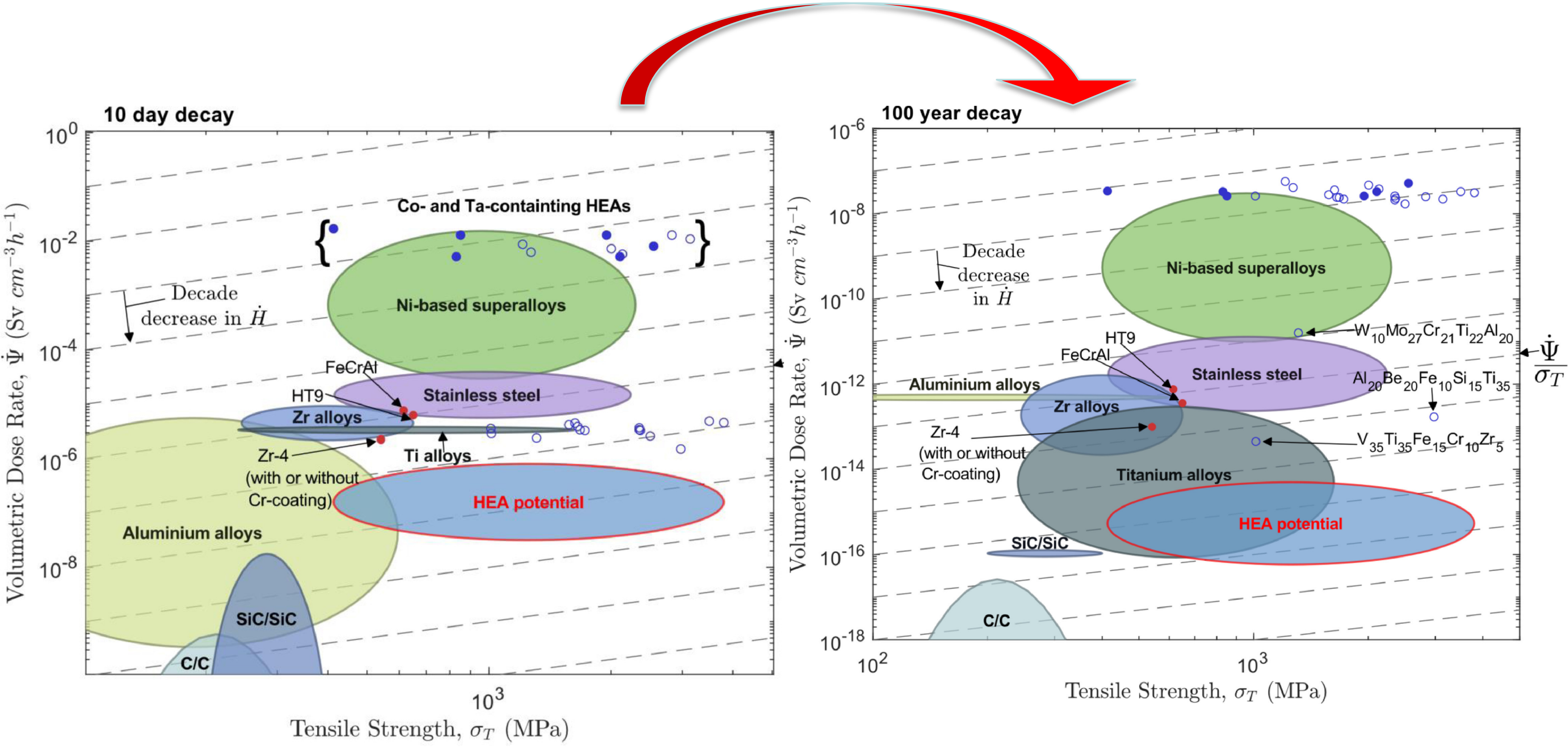



PWR and SFR, fusion first-wall (5 years of service)



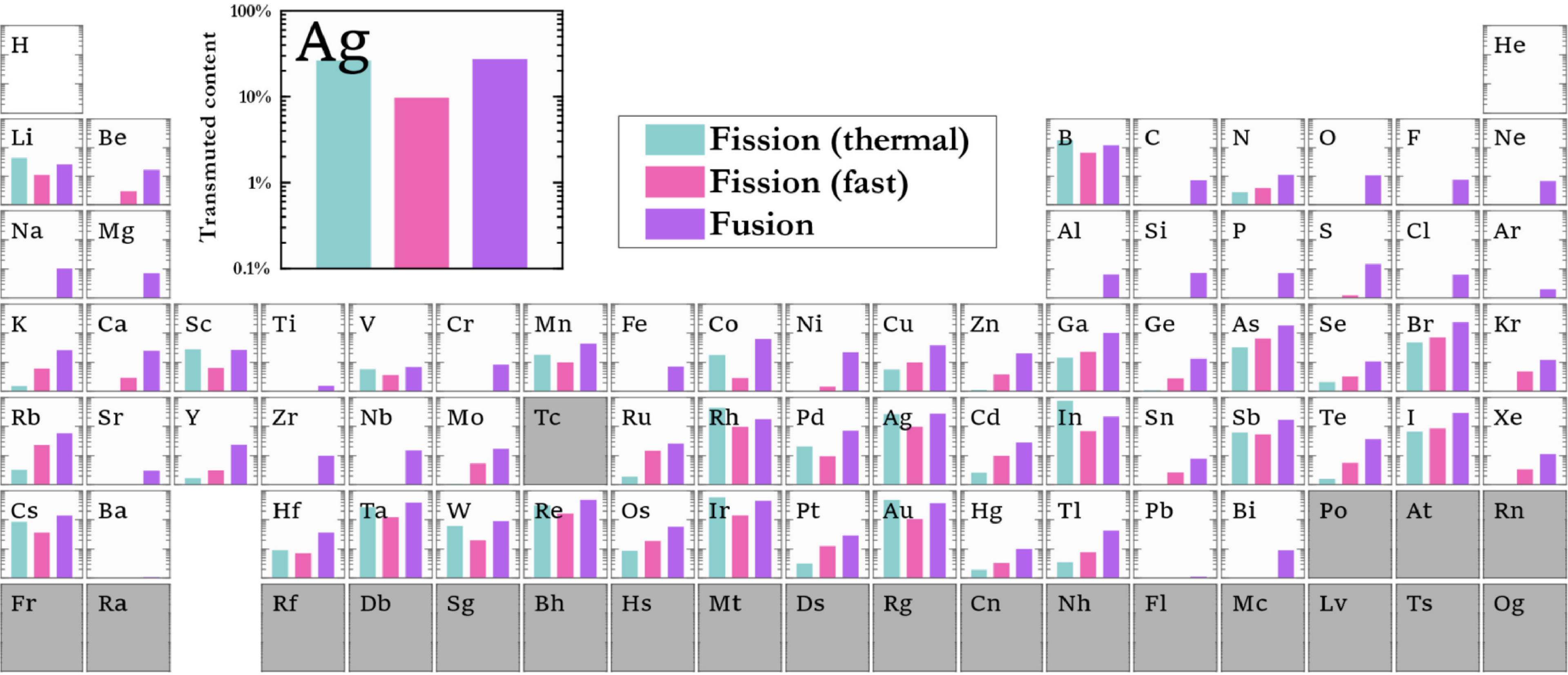
# Volumetric dose rate (irradiation under thermal fission conditions )

M. Moschetti, et al. JNM 2022

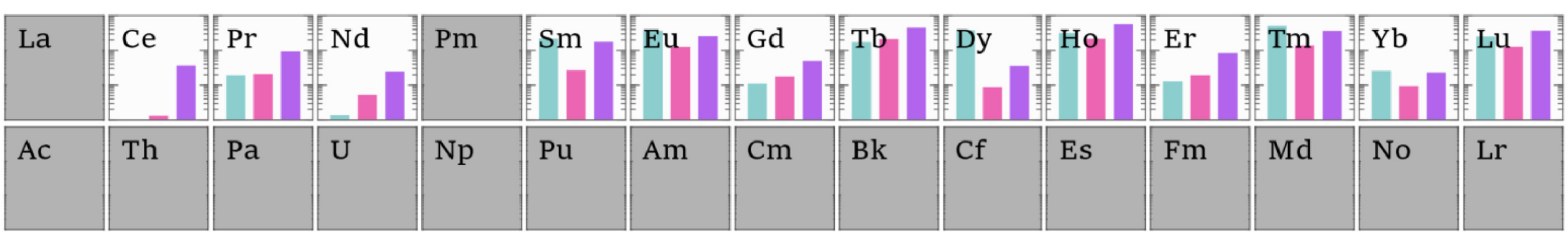


# Compositional stability of irradiated materials

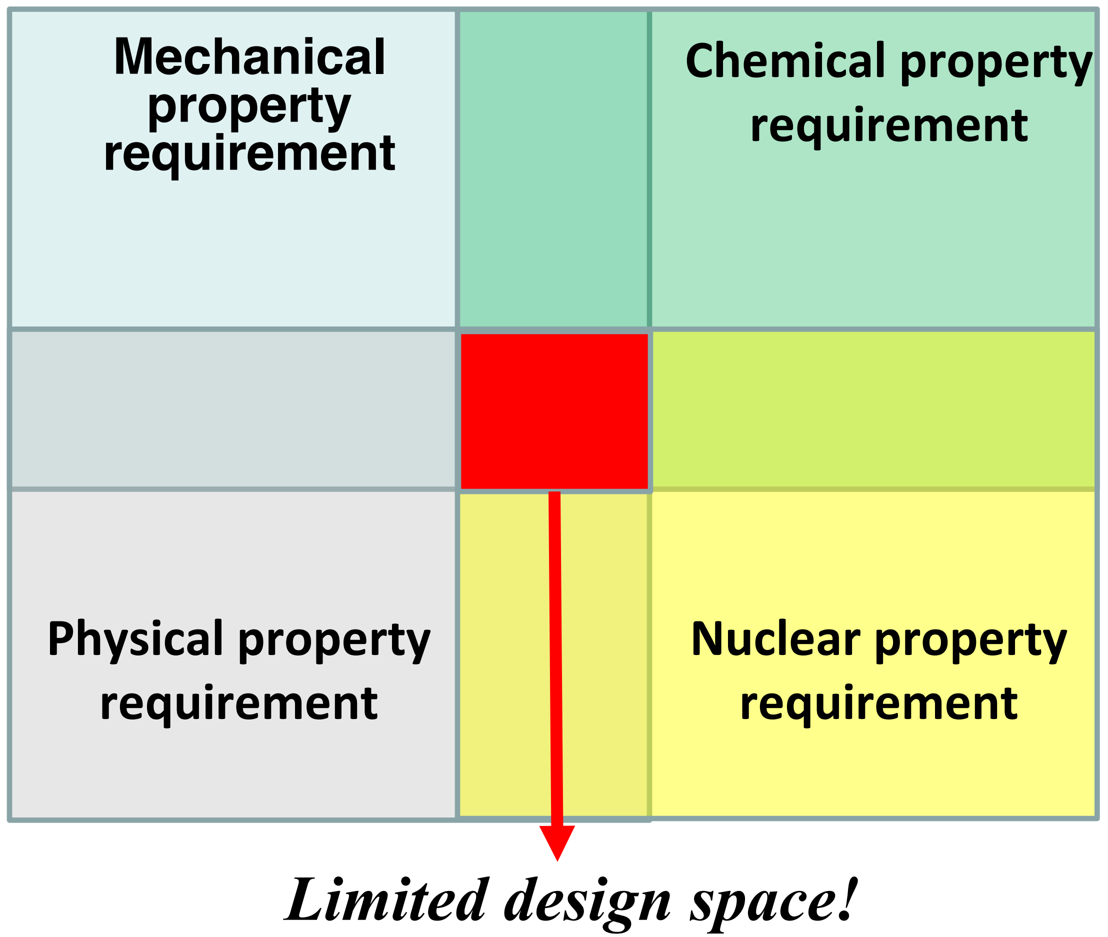
M. Moschetti, et al. JNM 2022




PWR and SFR, fusion first-wall (5 years of service)



# Materials selection for nuclear applications



## *Crystals and defects why are they important?*

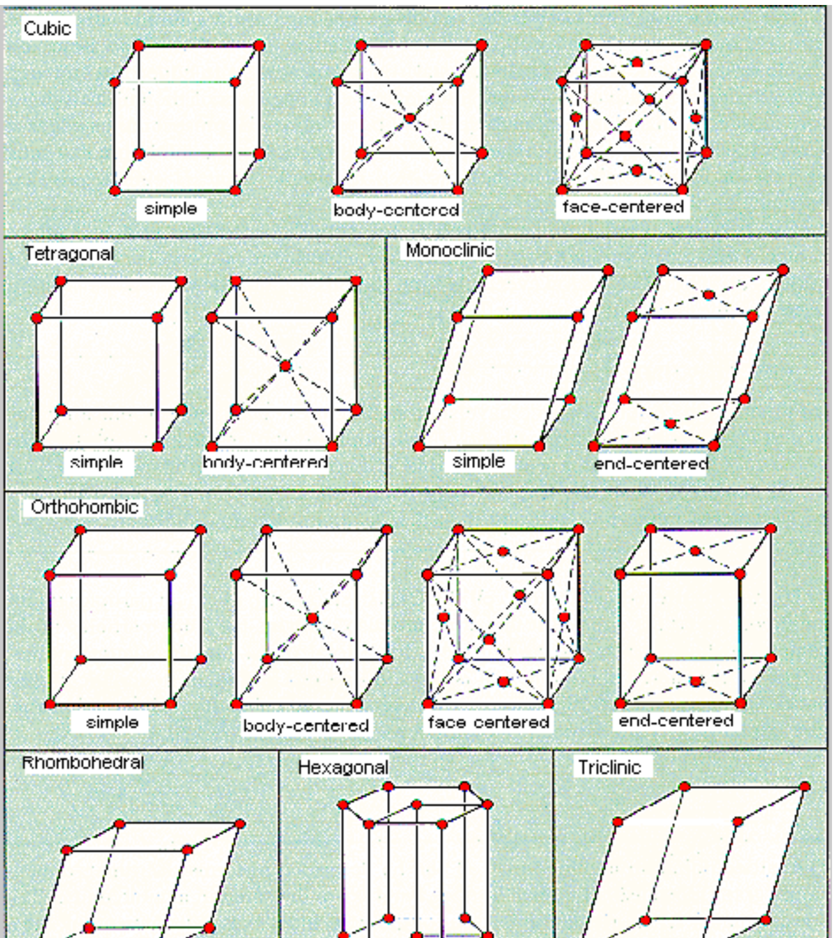
"Crystals are like people: it is the defects in them that make them interesting",

*Colin Humphreys Introduction to Analytical electron microscopy*

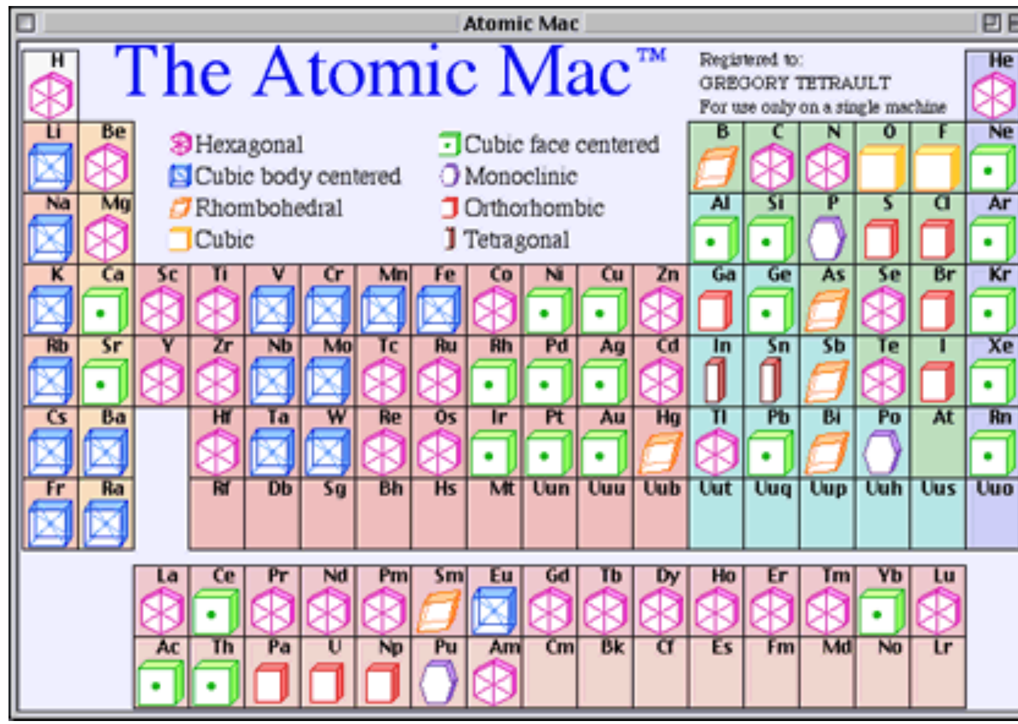
# Definition of crystal structures

**<u>Unit Cell</u>:**

- **Smallest unit translation of “space lattice” that reproduces a macroscopic crystal**




- **Crystalline solids: Many types of crystal arrangements, but all reduce to 14 unique lattice structures (Bravais lattice)**
- **Most common structural materials are of**




* *Ref:* http://chemed.chem.purdue.edu/genchem/topicreview/bp/ch13/unitcellframe.html

# Defects in crystals

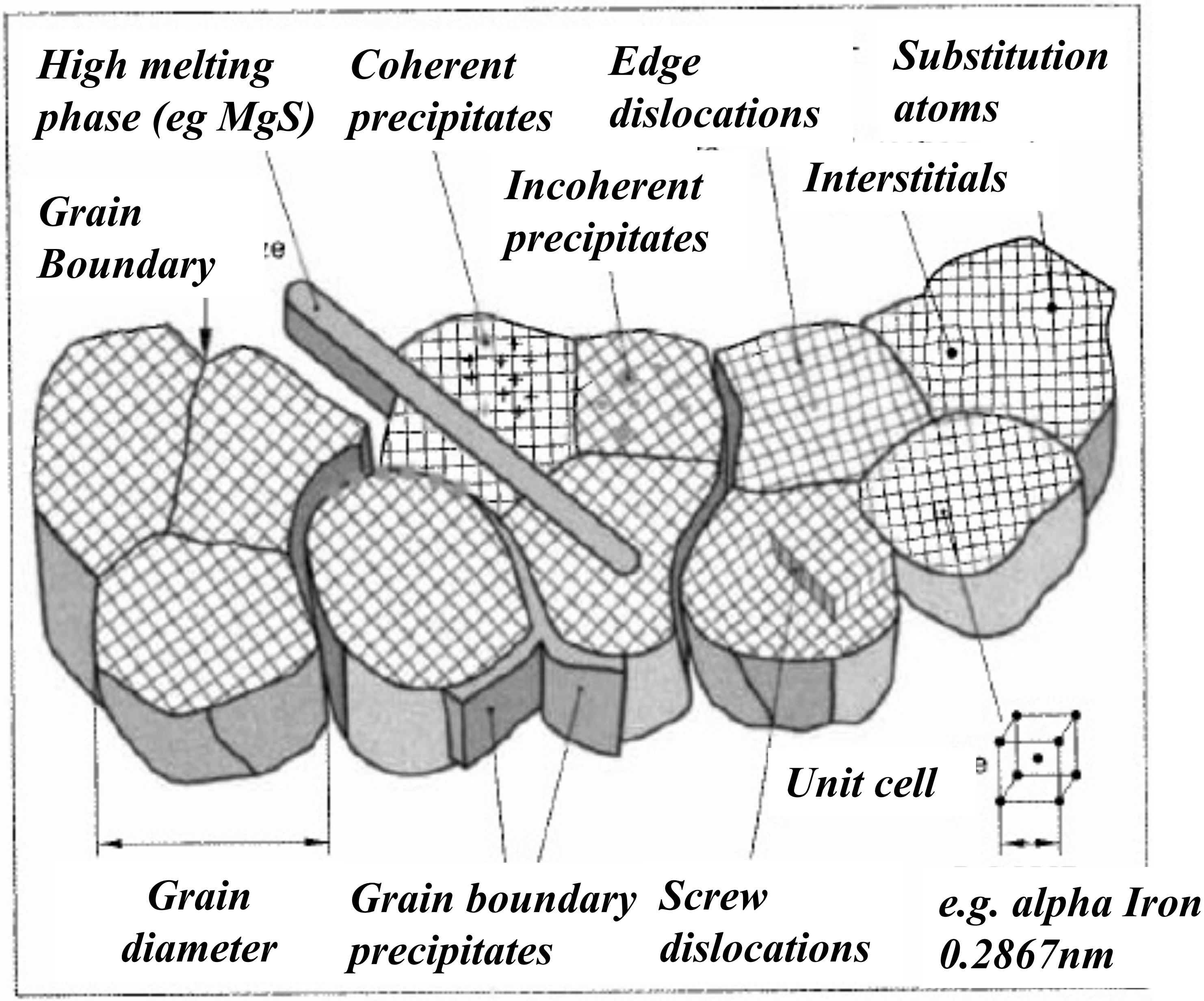
***0-dimensional defects:***

***1 dimensional defects:***

***2 dimensional defects:***

***3 dimensional defect:***

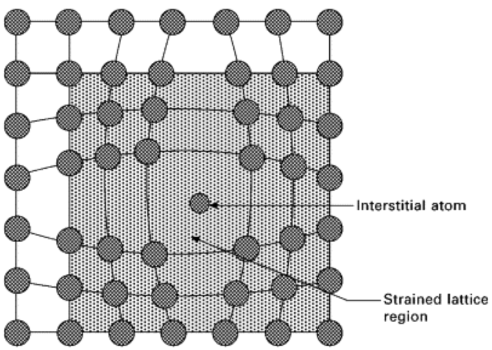
# Real materials always have defects.

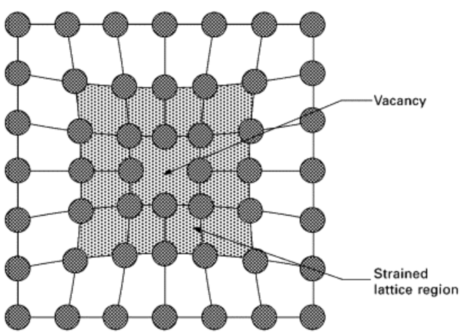


# Important note for later understanding! Each defect has a property

*Significant higher strain field*
*→ Strong attraction towards loops*

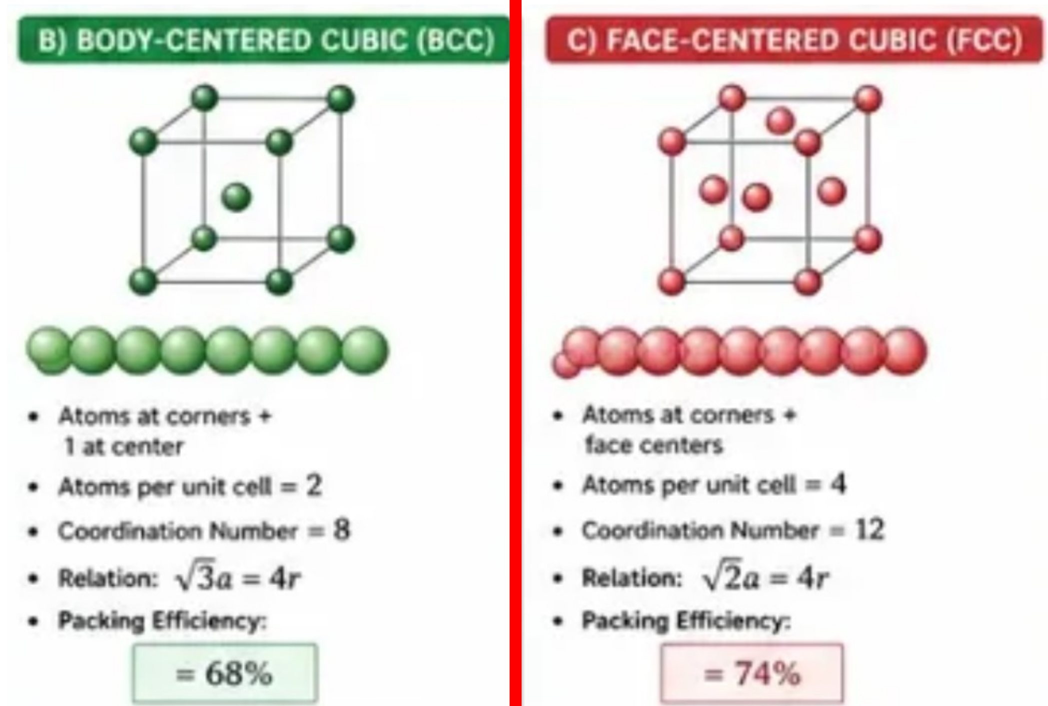
*Much faster diffusion*



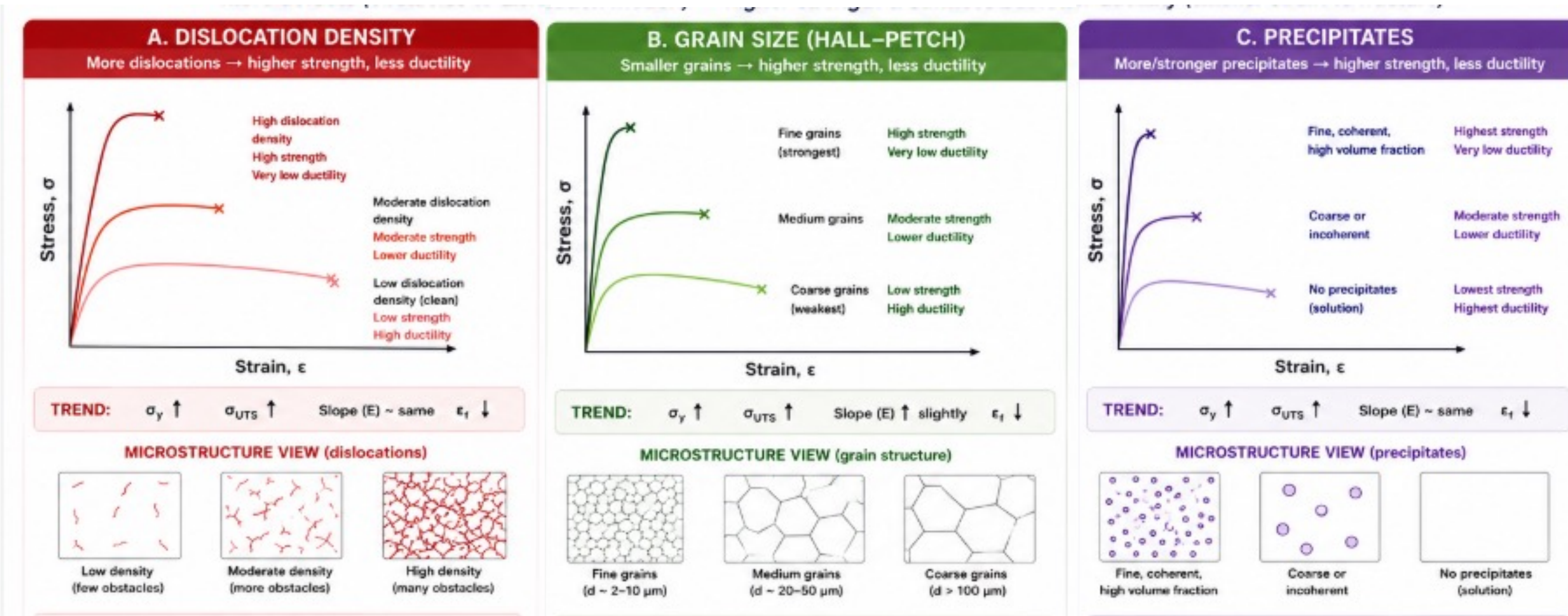



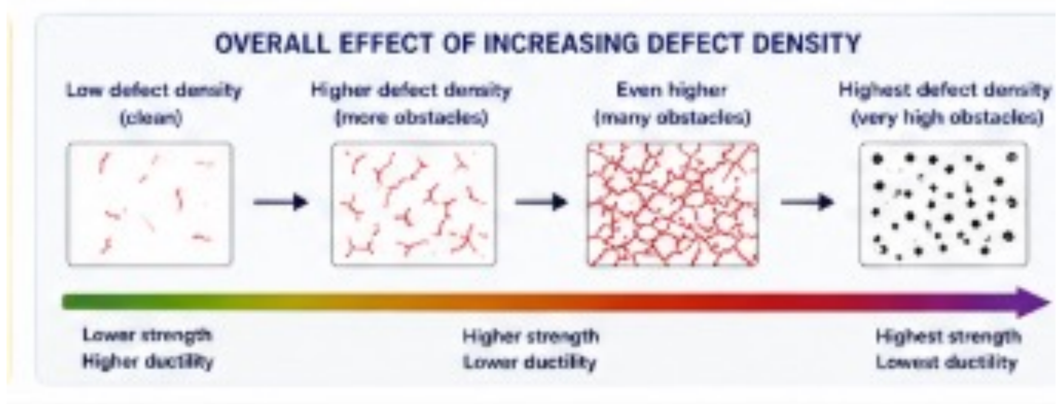

*Relative lower strain field*
*→weaker attraction towards loops*

*Much slower diffusion*



# How do defects change mechanical properties?

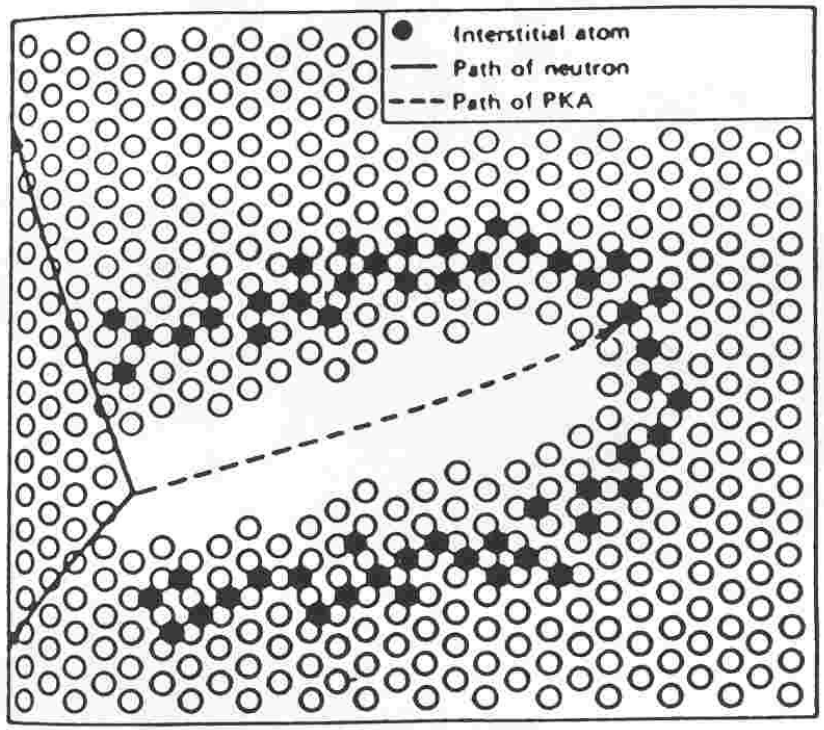


## *How does the radiation damage turn into these defects?*

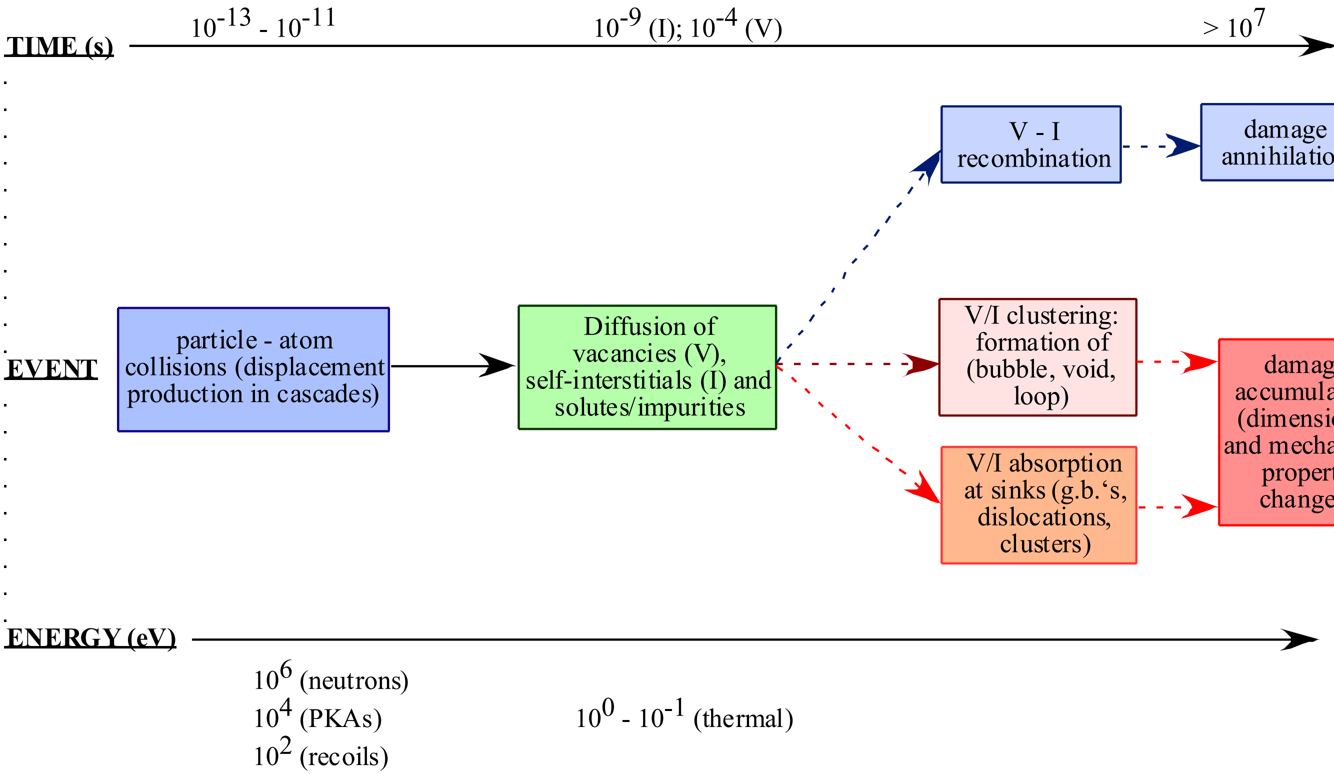
## *V-I interaction rate theory*

# Radiation damage creates point defects

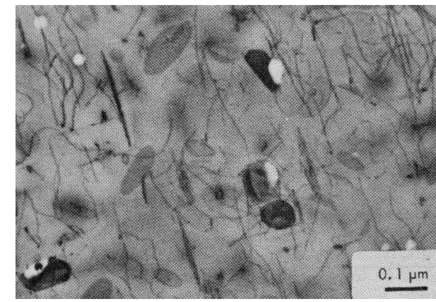



**Original model of the displacement spike,**
**J.A. Brinkman, *Amer. J. Phys* 24 (1956) 24.**

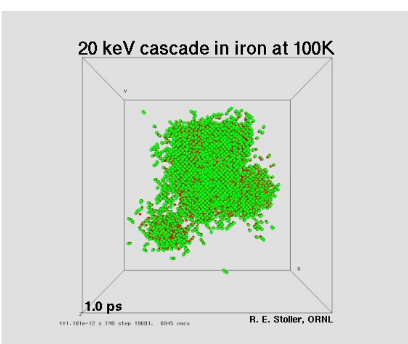
# *Radiation effects are governed by the ultimate fate of point defects (& transmutants)*



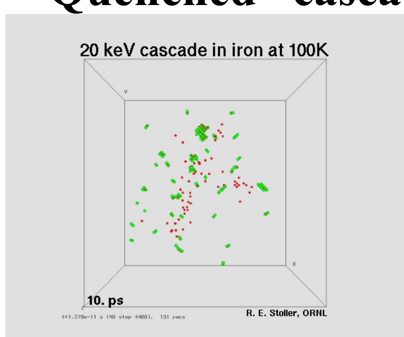

Voids, I-loops & precipitates in stainless steel




**Nascent cascade**




**"Quenched" cascade**




**Number of defects:**

**Nascent cascade: $N_i = N_v = \nu(T)$**

**"Quenched" cascade: $N_i' = N_v' < \nu(T)$**

# Point Defect Balance Equations

**Objective: determine the concentration of vacancies and self-interstitials in an irradiated solid**

**Assumes infinite medium (no long-range spatial gradients of $C_v$ or $C_I$)**
**Assumes that only V and I are mobile (no mobility of self-interstitial or vac-clusters)**

$$\boxed{\frac{\partial C_{i/v}}{\partial t} = Rate\ of\ change\ of\ C_{i/v} = \mathrm{Pr}oduction\ -\ Loss \qquad (9.1)}$$

# Point Defect Balance Equations

**Loss Mechanisms:**

***Recombination (V + SIA = null)***

$$\frac{recombination\ rate}{unit\ volume} = k_{iv} C_i C_v \qquad (9.4)$$

$$k_{iv} \cong \frac{500\Omega}{a_o^2}(D_i + D_v) \approx \frac{500\Omega}{a_o^2} D_i \qquad (9.5)$$

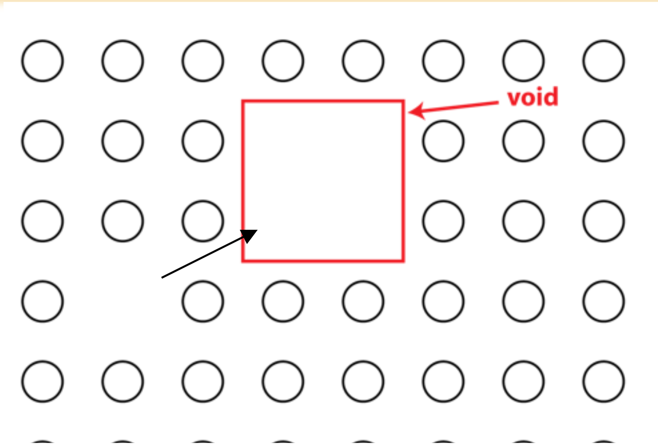
***kiv= recombination rate constant***

***500… combinatorial number describing the number of sites the recombination can occur (~250 atoms/nm$^3$ in bcc Fe)***

***$a_0$=lattice constant***

***interstitials are mobile***

# *Point Defect Balance Equations*




***Absorption at voids for vacancies***

**Rate of absorption # vacancies (or self-interstitials) per unit time**

$$\textbf{\textit{FYI:}} \quad J_v^{void} = 4\pi R^2 D_v \frac{\partial C}{\partial r}\bigg|_{r=R} = 4\pi R D_v \left[ C_v^{bulk} - C_v^{eq} ) \right]$$

$$J_v^C = 4\pi R D_v (C_{vb} - C_R) \qquad (9.6)$$

**$C_R$ = concentration of solute** species at cavity surface

**Volumetric absorption rate = $N_v$*$J_v^C$:**

$$= D_v [4\pi R N_v](C_{vb} - C_R) \qquad (9.7a)$$

$$S_v^C = 4\pi R N_v \quad (Cavity\ \sin k\ strength) \quad (9.7b)$$

**$C_b$ = concentration of solute** species (vacancies) in solid far from cavity in the bulk

R= cavity radius

# Point Defect Balance Equations

## Absorption at voids

**Self-interstitials:**

$$C_{ib} = C_i$$
$$C_R \approx C_i^{eq} \approx 0$$
$$\text{Volumetric absorption rate} = D_i S_i^C C_i$$

***No interstitials present in equilibrium***

**Vacancies:**

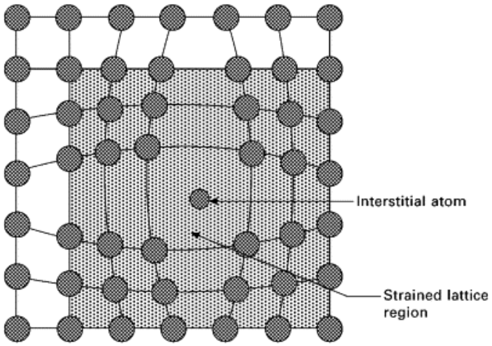
$$C_{vb} = C_v$$
$$C_R \approx C_v^{eq} * \exp((\Omega/kT(2\gamma/R-p))$$

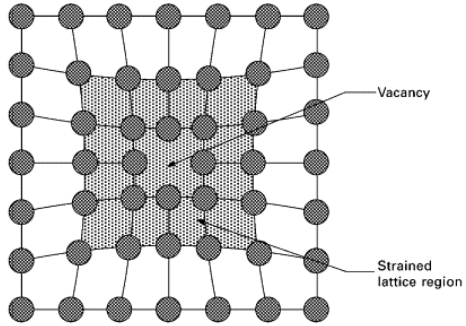
***Vacancies present in equilibrium***

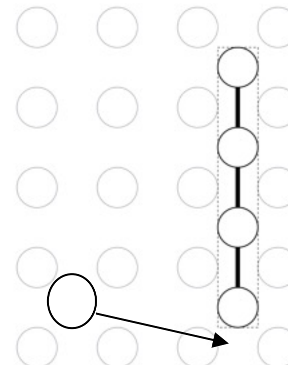
$$\text{Volumetric absorption rate}$$
$$= D_v S_v^C C_v - D_v S_v^C C_v^{eq} \exp[(\Omega/kT(2\gamma/R-p)]$$

**where, 2nd term represents the production of vacancies by thermal emission from cavities**

# Point Defect Balance Equations



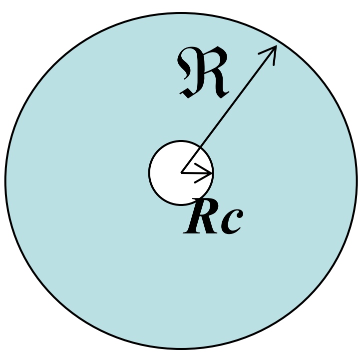
**Loss Mechanisms:**

***Absorption at dislocations***

$$Volumetric\ absorption\ rate = D_{iv} S_{iv}^{\perp} C_{iv} \qquad (9.8)$$

*$\rho$ = dislocation density*

$$S_i^{\perp} = \frac{2\pi\rho}{\ln(\Re / R_{ci})} = Z_i \rho \quad S_v^{\perp} = \frac{2\pi\rho}{\ln(\Re / R_{cv})} = Z_v \rho \quad (9.9)$$




$R_{cv} < R_{ci}$ (strain field interactions) $\therefore\ S_v^{\perp} < S_i^{\perp}$

**Dislocations are biased sink for self-interstitials!!**

# Summary equation

**Point Defect Balance intersttials:**

$$\frac{dC_i}{dt} = \underbrace{Q_i}_{\textit{production}} - \underbrace{k_{iv}C_iC_v}_{\textit{recombination}} - \underbrace{D_iC_iS_i^t}_{\textit{absorption}} \qquad S_i^t = S_i^C + S_i^{\perp}$$

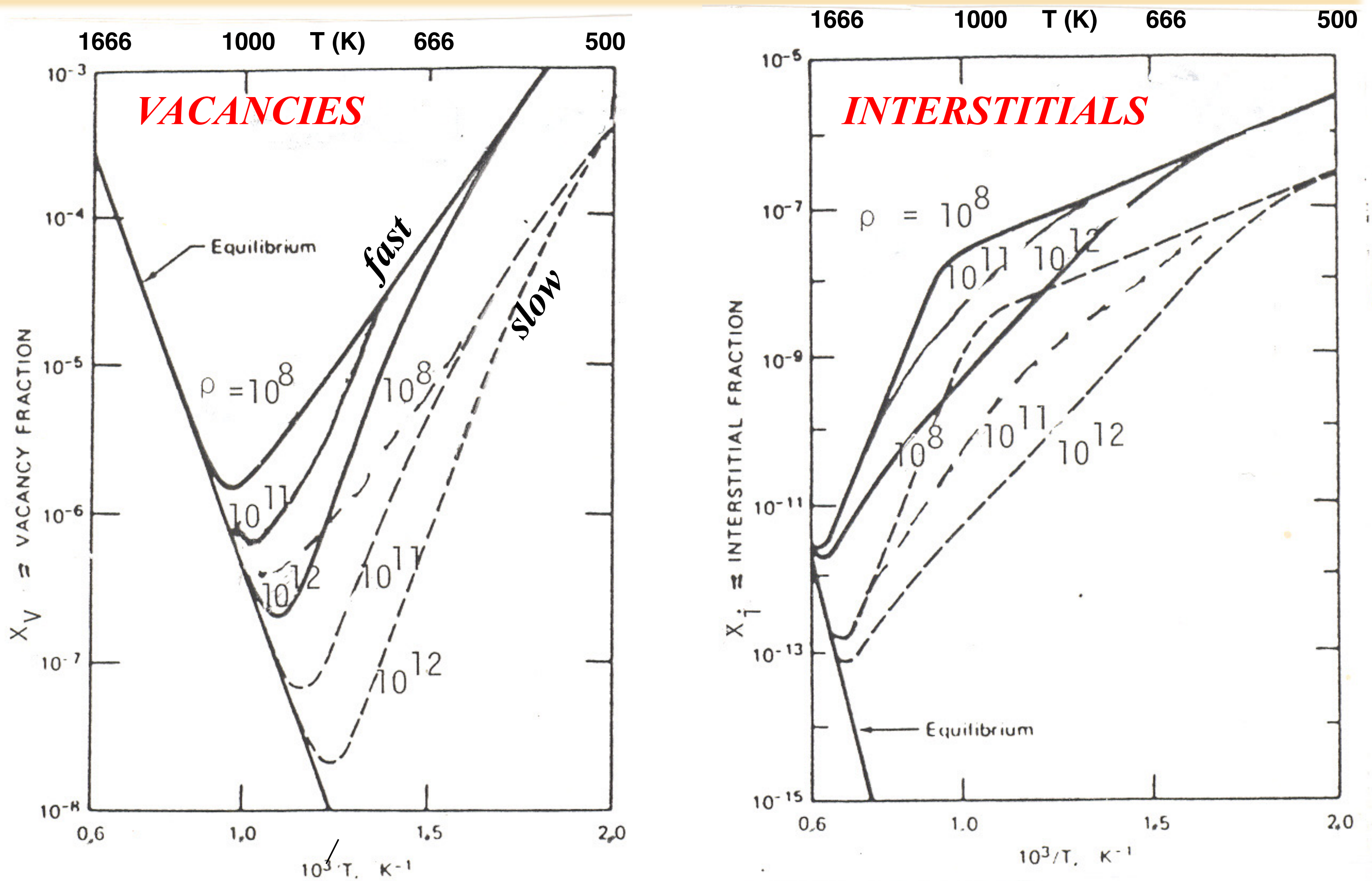
**Point Defect Balance vacancies**

*Thermal vacancies*

$$\frac{dC_v}{dt} = Q_v - k_{iv}C_iC_v - D_vC_vS_v^t + Q_v^{th} \qquad S_v^t = S_v^C + S_v^{\perp}$$

$$Q_v^{th} = D_vS_v^CC_v^{eq}\exp\left[\frac{\Omega}{kT}\left(\frac{2\gamma}{R} - p\right)\right]$$

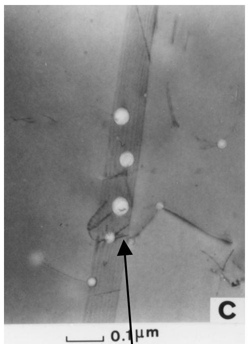
# Solutions to the Point Defect Balance Equations




**K = $10^{-6}$ dpa/s (dashed) and $10^{-4}$ dpa/s (solid)**

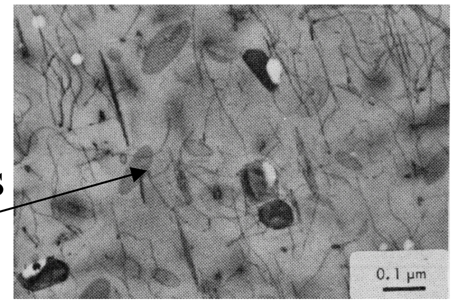
# Observed microstructural changes as a result

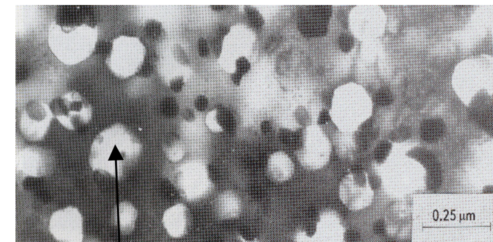
- **Cause: long-range diffusion and clustering of displacement damage (defects) produces microstructural changes in the form of dislocation loops, stacking fault tetrahedra, precipitates, bubbles, voids and solute segregation**




GB He bubbles in SS

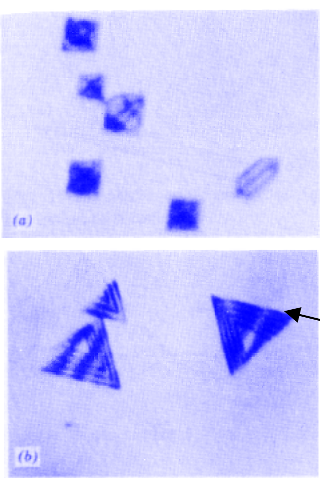
Voids, I-loops & precipitates in SS



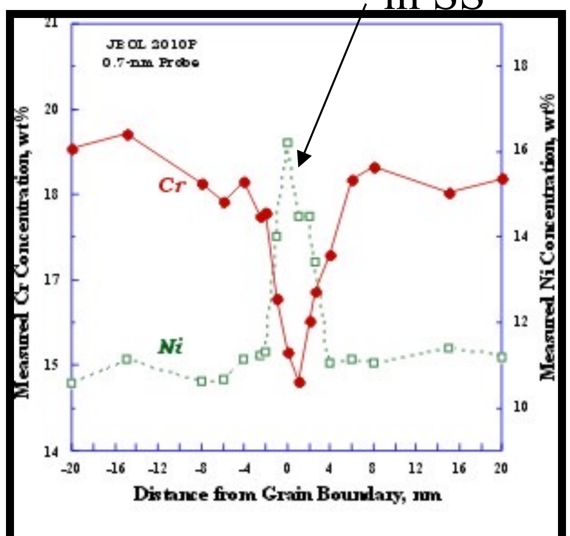



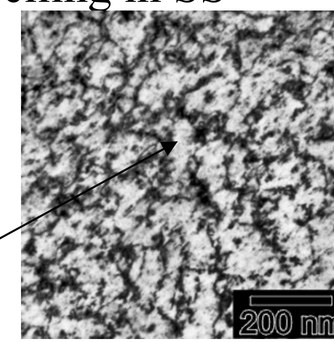

Void swelling in SS

GB solute segregation in SS

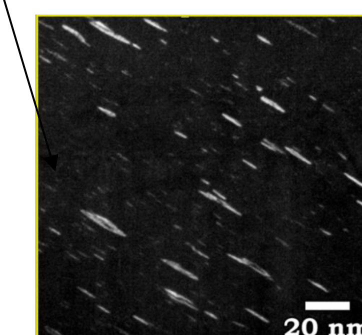



SFT in Au



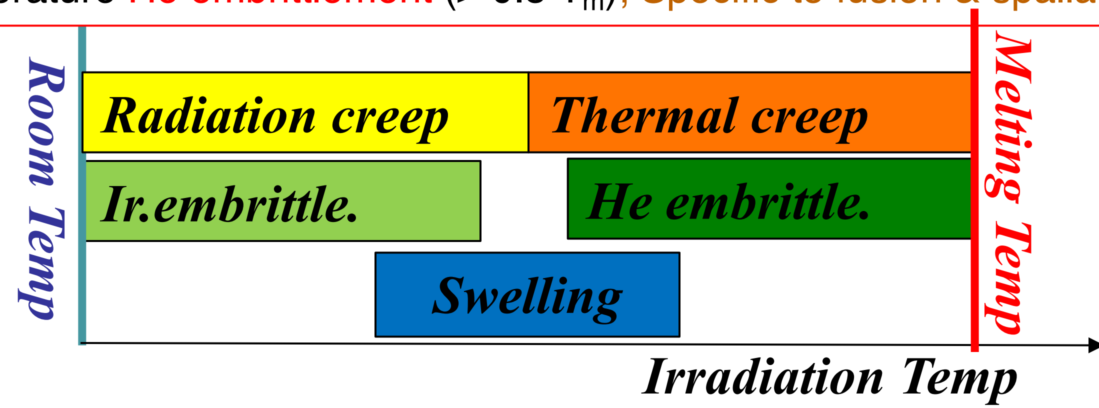



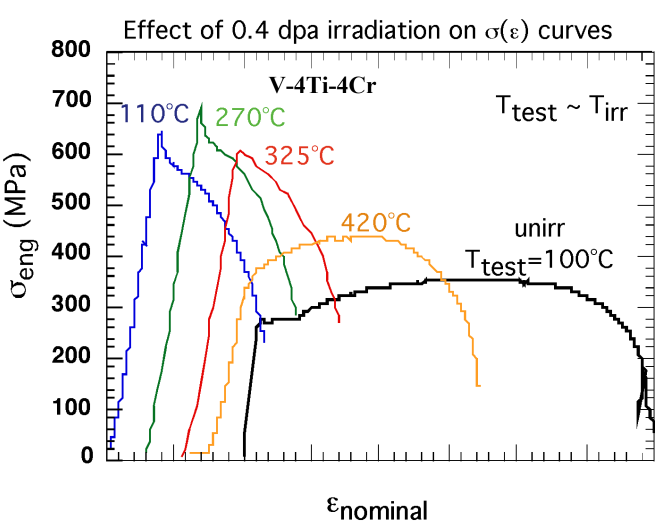

Dislocation-loops & network dislocations



# *Irradiation effects on structural materials*

- Exposure to neutrons degrades the mechanical performance of structural materials and impacts the economics and safety of current & future fission power plants:
  - Irradiation hardening and embrittlement/decreased uniform elongation (< 0.4 $T_m$)
  - Irradiation (<0.45 $T_m$) and thermal (>~0.45 $T_m$) creep
  - Volumetric swelling (0.3 - 0.6 $T_m$)
  - High temperature He embrittlement (> 0.5 $T_m$); Specific to fusion & spallation accelerators




**Variables**

- Materials (Fe-based steels, Vanadium and Ni-based alloys, Refractory metals & alloys, SiC, ODS alloys)
- Initial microstructure (cold-worked, annealed)
- Irradiation temperature
- Chemical environment & thermal- mechanical loading
- Neutron flux, fluence and energy spectrum
  - materials test reactor irradiations typically at accelerations of $10^2$ - $10^4$

**Synergistic Interactions between all of them!**

# *Extreme environments in Advanced Nuclear Energy Systems*

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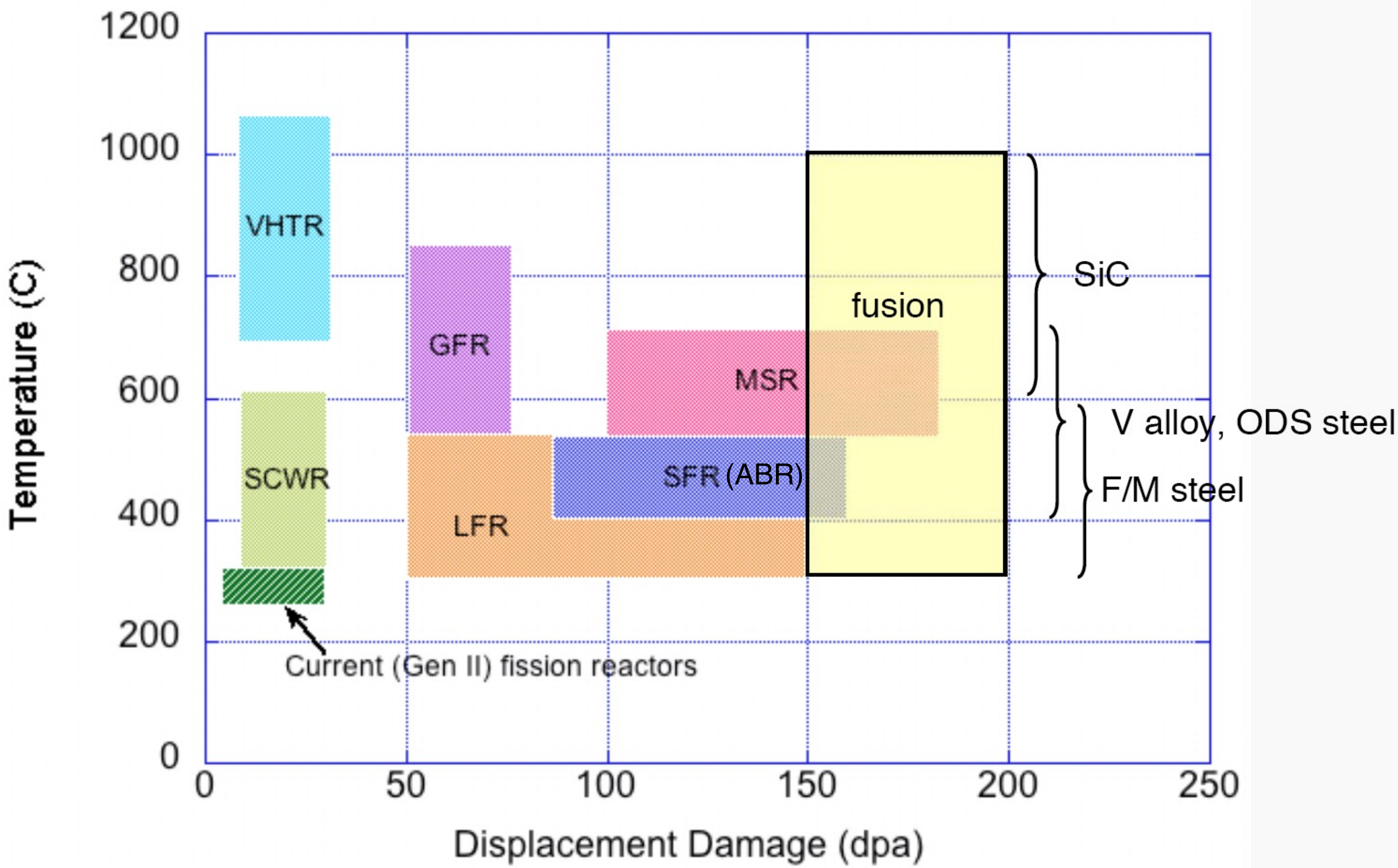


Very High Temperature Reactor
Super-Critical Water Reactor
Gas-cooled Fast Reactor

Lead-cooled Fast Reactor
Sodium-cooled Fast Reactor
Molten Salt Reactor

* S.J. Zinkle, ORNL

# *Why is it so complicated to pick the right materials?*

***So many parameters!***

***Dose***

***Dose rate***

***Temperature***

***Environment (corrosion)***

***Operation time***

***cycling of temperature or pressure***

***Elements in the material***

# *Radiation effects in materials:*

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## *Swelling*

## *Design for Swelling resistance*

# What makes a materials Swelling resistant?

- **Low interaction with neutrons (low X-section)**
  - **Choose your alloying elements wisely.**
- **bcc metal (less bias)**
- **Lots of interfaces as defect sinks**
  - **Nanoparticles (Oxide Dispersed Strengthened)**
  - **High dislocations density -> cold work, martensite**
  - **High interface density→ martensite**

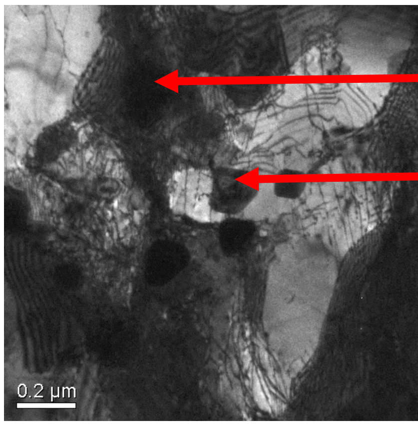
***Ferritic/ Martensitic Materials***

***ODS Materials***

# *Ferritic-martensitic steels: typical microstructure (T91)**

| Fe | Cr | Mo | Mn | V | Nb | Ni | Si | Cu | C | P | Al | S | N |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Bal | 8.37 | 0.90 | 0.45 | 0.216 | 0.076 | 0.21 | 0.28 | 0.17 | 0.10 | 0.009 | 0.022 | 0.003 | 0.048 |

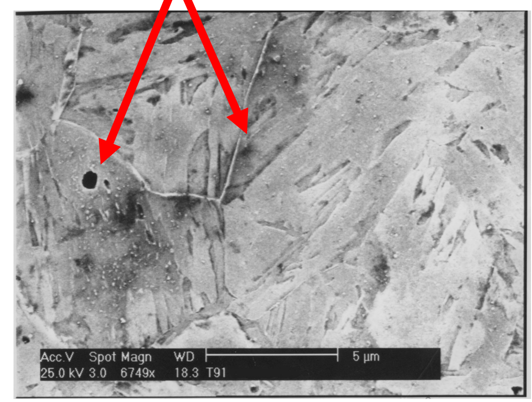
***Solution Anneal: 1066° C: 46mins air cool, Tempering: 790° C: 42mins air cool***

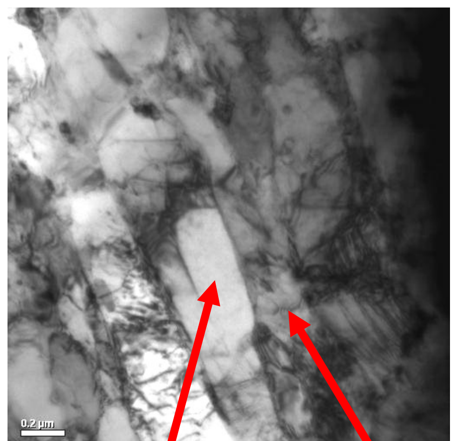


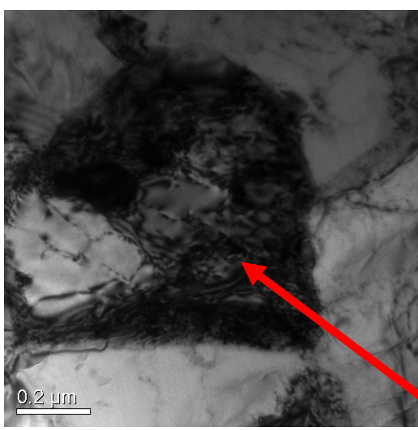

Dislocation tangles

precipitate

Prior austenite grain boundary (PAGB)



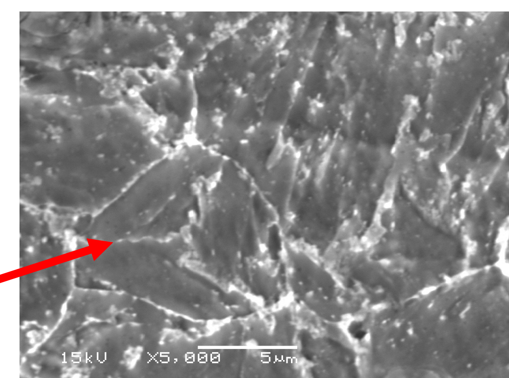




Lath

Subgrain

Dislocation cell

Packet boundary



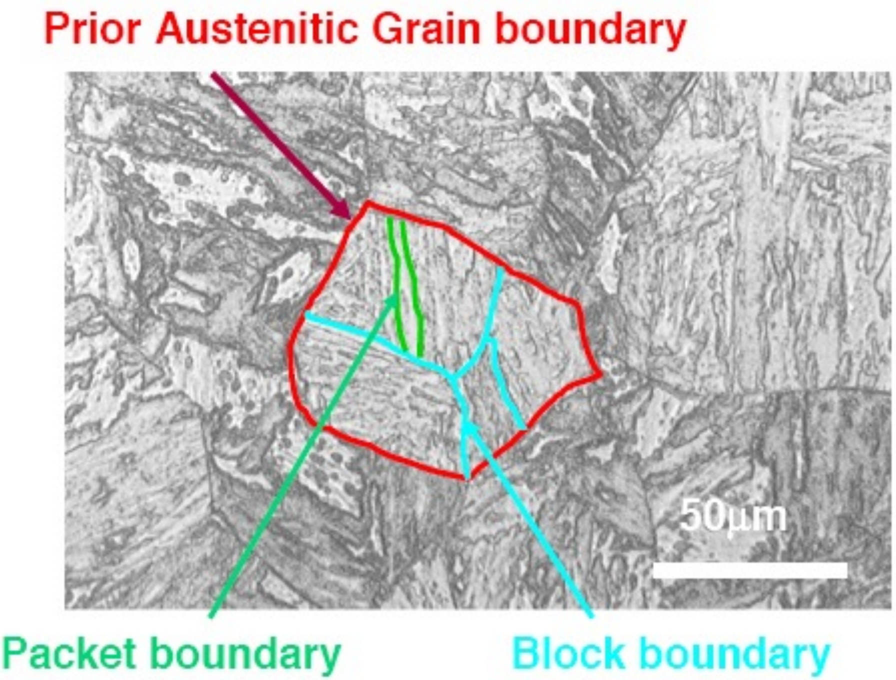

** From G. Gupta, UMich*

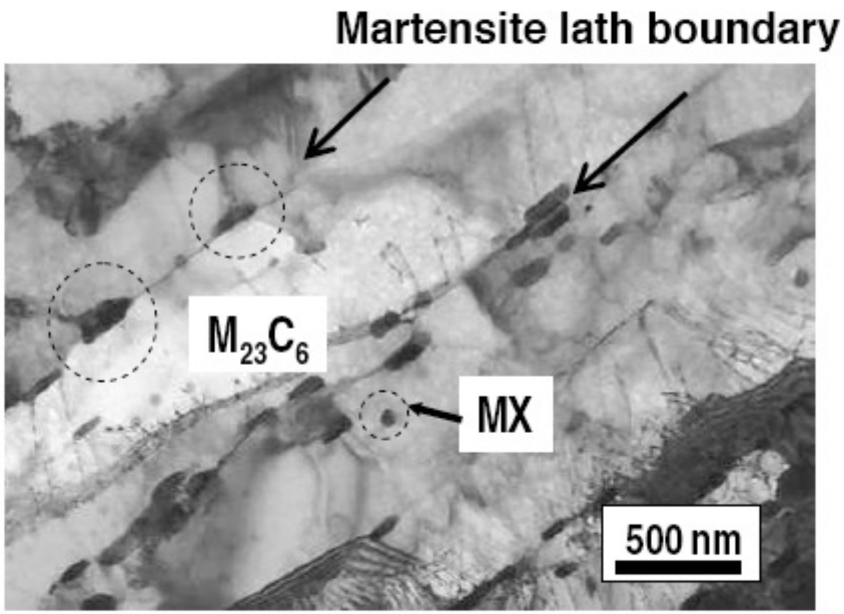
# *Ferritic-martensitic steels: typical microstructure (F82H, Eurofer97)**

Fe – 8 ~ 9 Cr - 1~2 W - V,Ta (F82H, JLF-1, EUROFER 97, etc.)

Based on Mod9Cr (T91: Fe 9Cr-1Mo V,Nb)

Mo replaced by W, Nb replaced by Ta, and reduced N, to reduce activation



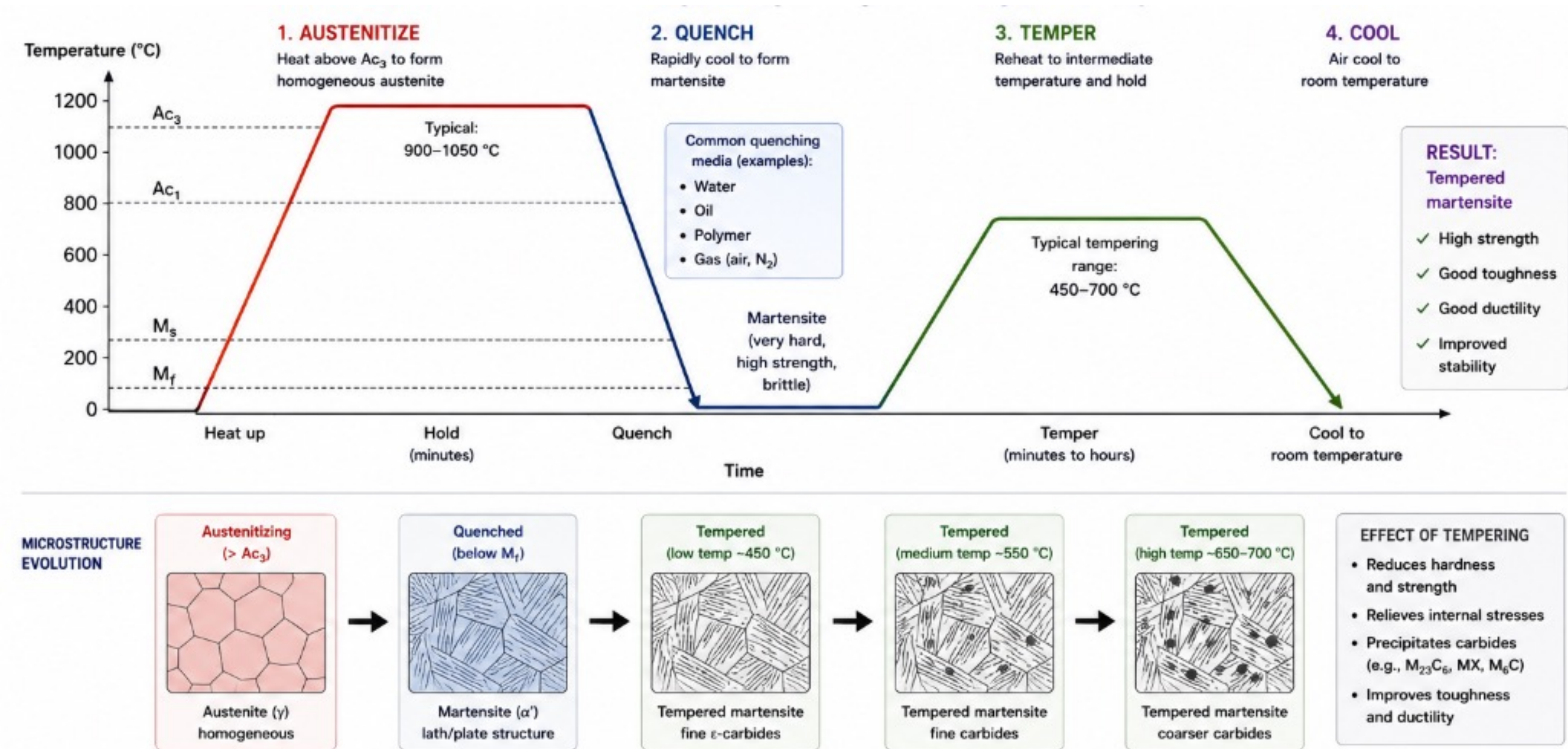

$M_{23}C_6$ : Major precipitate, Cr rich.

MX :Ta,V rich precipitates

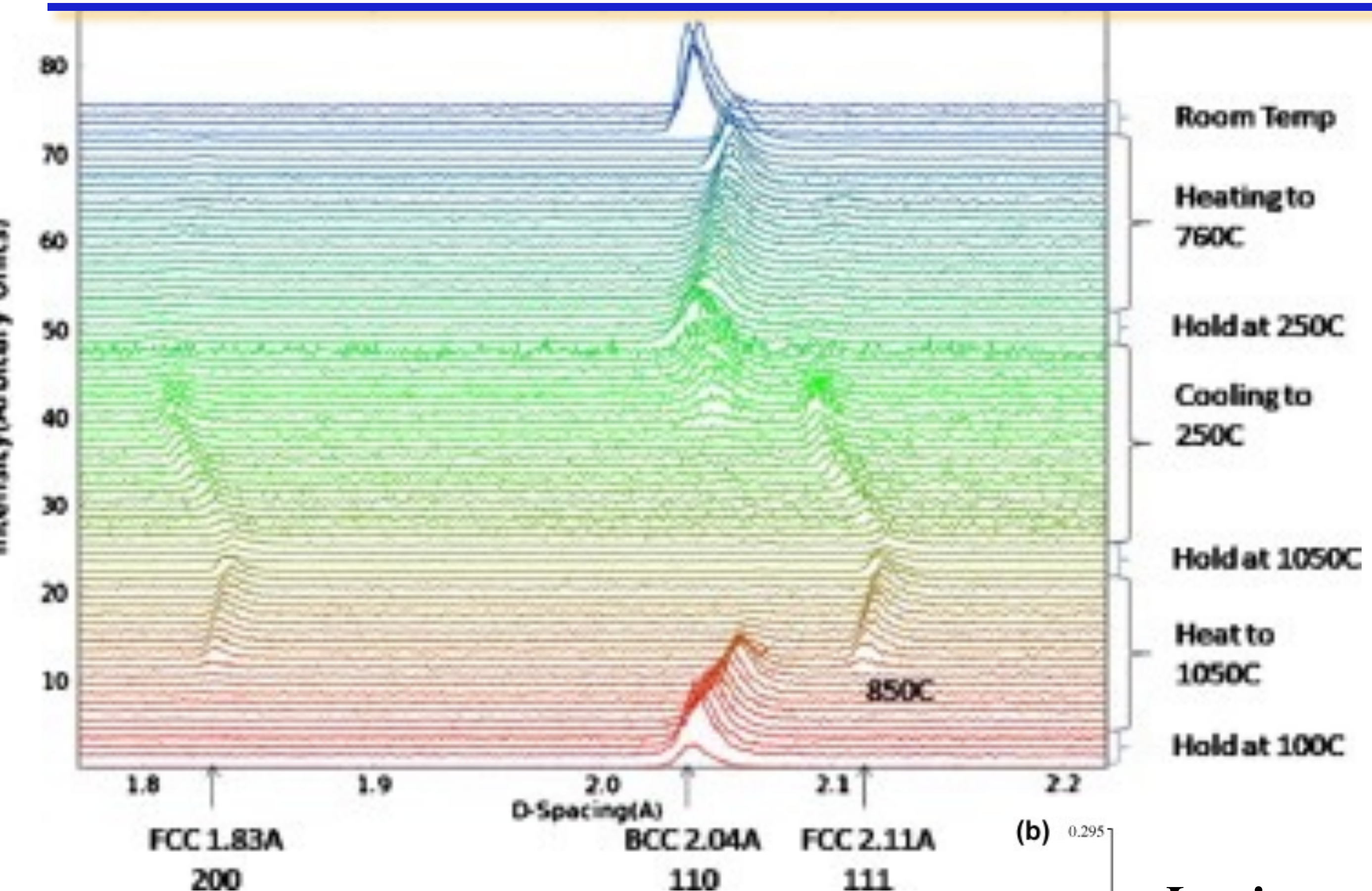
- Fully tempered Martensitic steels
- Fine boundaries and precipitates make the steel highly irradiation and heat resistant.
- Ferrite phase formation should be avoided to have good performance.

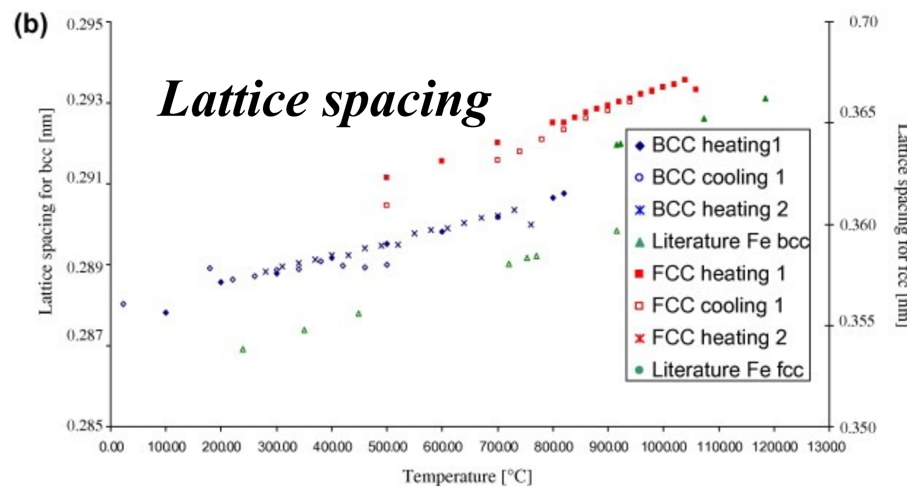
** From H. Tanigawa, JAEA*

# *Ferritic-martensitic steels how is it formed?*



# *In-situ diffraction experiment*

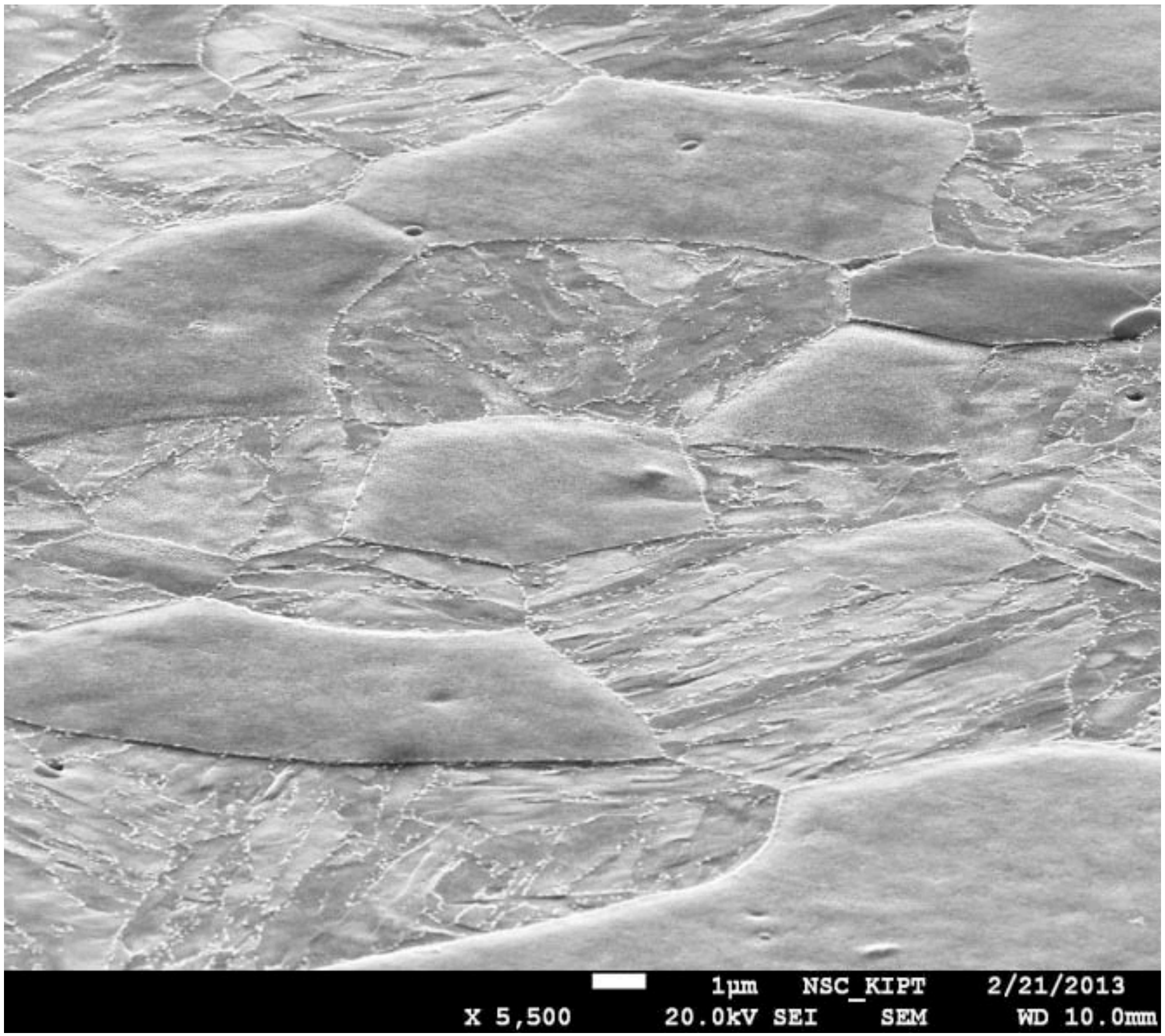



***Lattice spacing***

***T91 steel P. Hosemann et al 2010***

# *Irradiation induced swelling effect of microstructure*

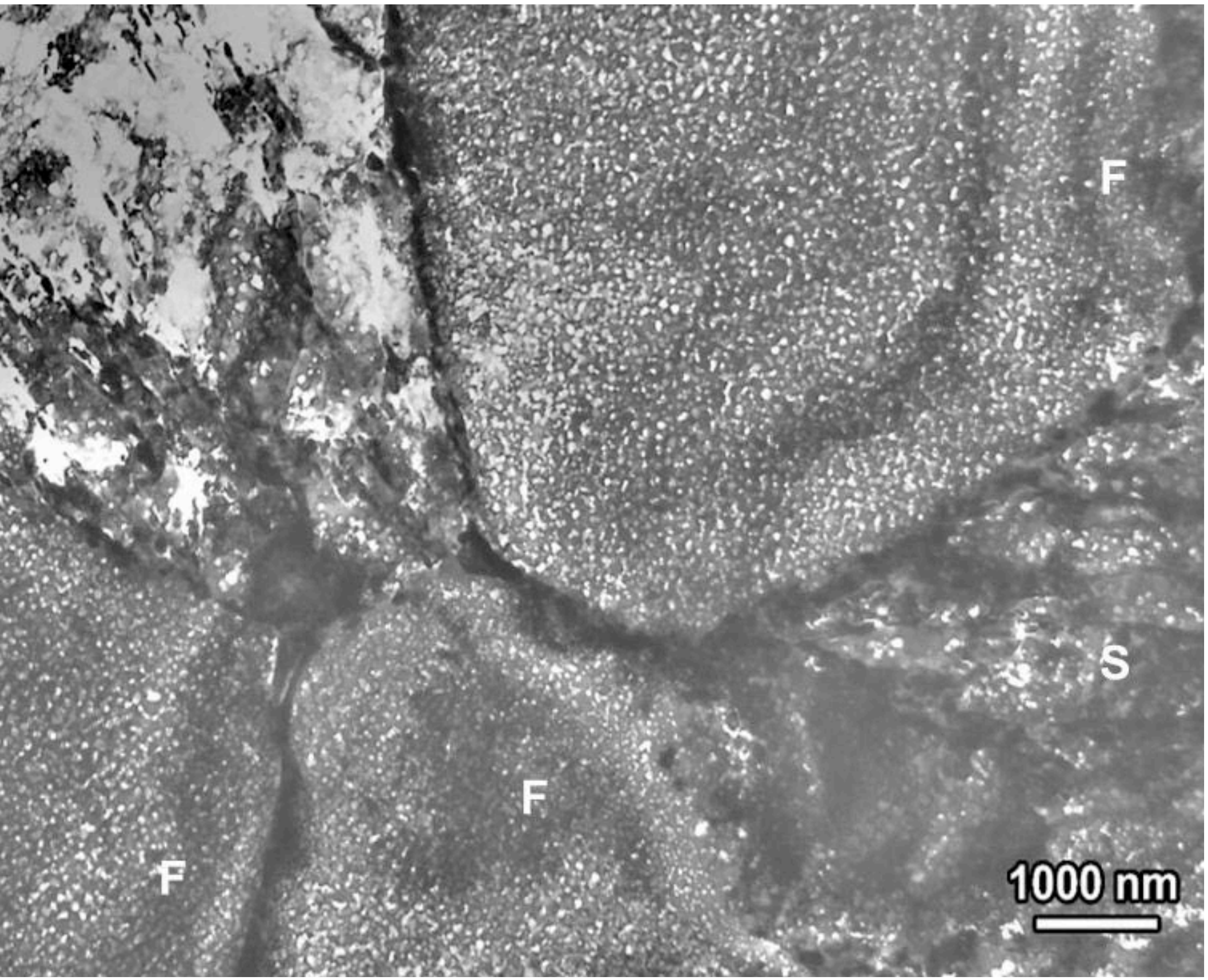



***F. Garner 2015:***

***Ferritic grains swell (grow) more than martensitic grains!***

***Ferritic (bcc) have a smaller capture radius for vacancies See NE220***

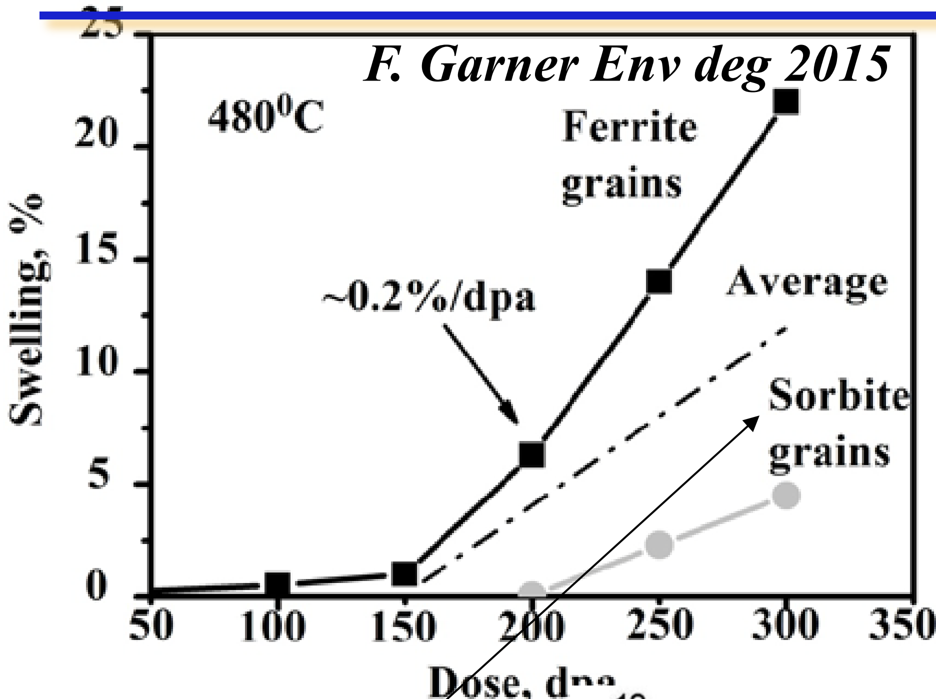
# *Irradiation induced swelling effect of microstructure*



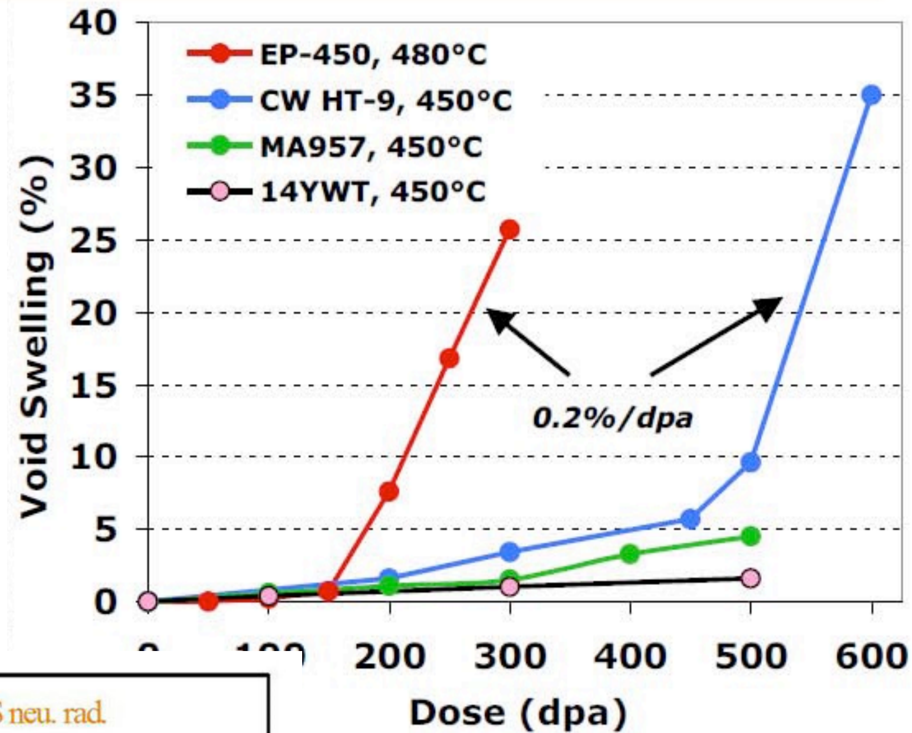

*F. Garner 2015:*

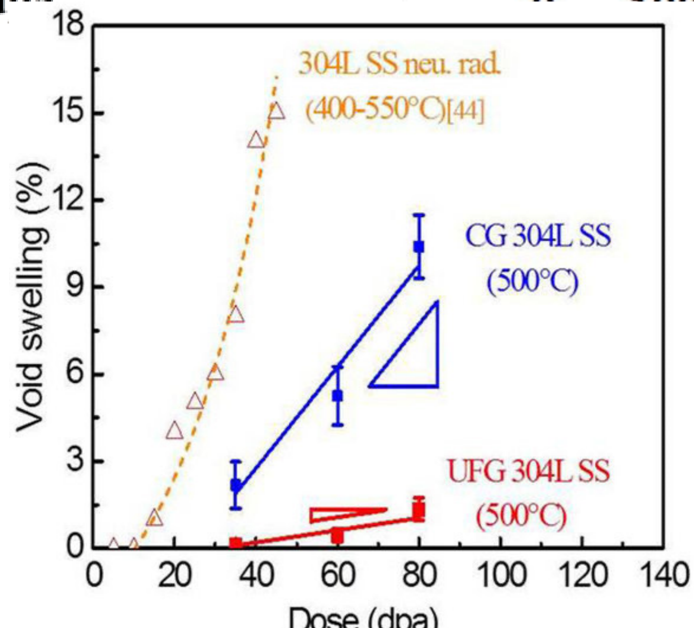
*Ferritic grains swell (grow) more than martensitic grains!*

# *Irradiation induced swelling effect of microstructure*




*Tempered Martensite*



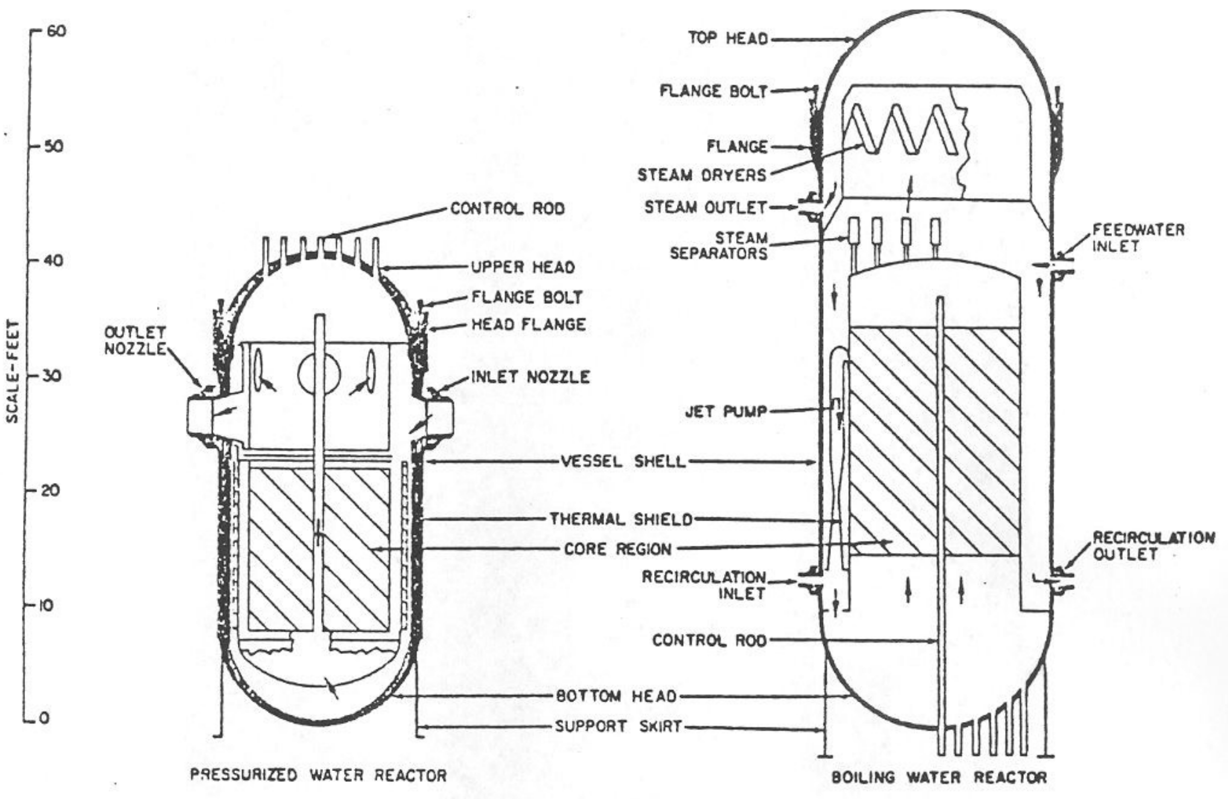



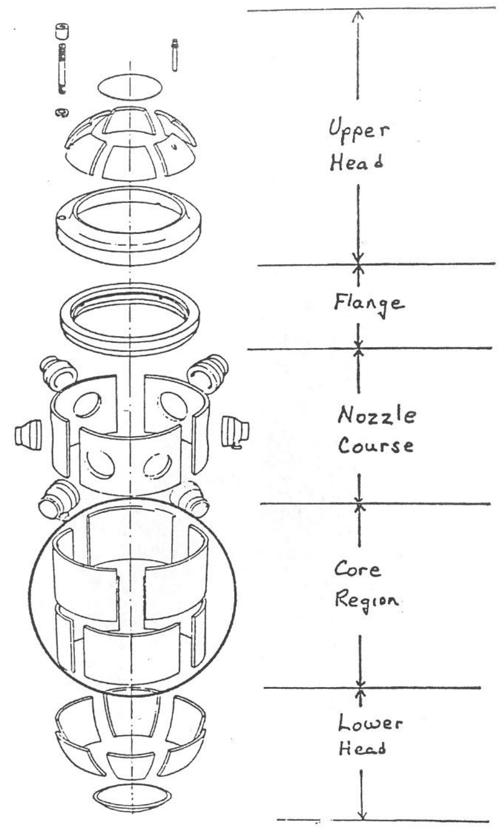

*C. Sun Nature reports 2015*

# Rector pressure vessel

# Reactor Pressure Vessel (RPV) embrittlement

- The RPV is perhaps the most critical safety component in a nuclear power plant. RPVs are fabricated from low alloy carbon steel (typically A302, A533B and A508-type). Early technology: plates rolled into appropriate shapes and welded together into an integral vessel. The head is bolted to the vessel and can be periodically removed (e.g., for re-fueling). More recently, rings have been fabricated from single ingot forgings, which eliminates the longitudinal weld.



# Forging of large vessels

***Older RPV's have been extensively welded.***

***However, today forging is preferred.***

# Forging of large vessels

***Reactor pressure vessel components are among the largest forged components made today!***

***Plants today: Japan, China, France Russia Korea***

***The larges forgings can be 650t with a 14000t press.***

***Second larges press 13.000t at Seah Besteel.***

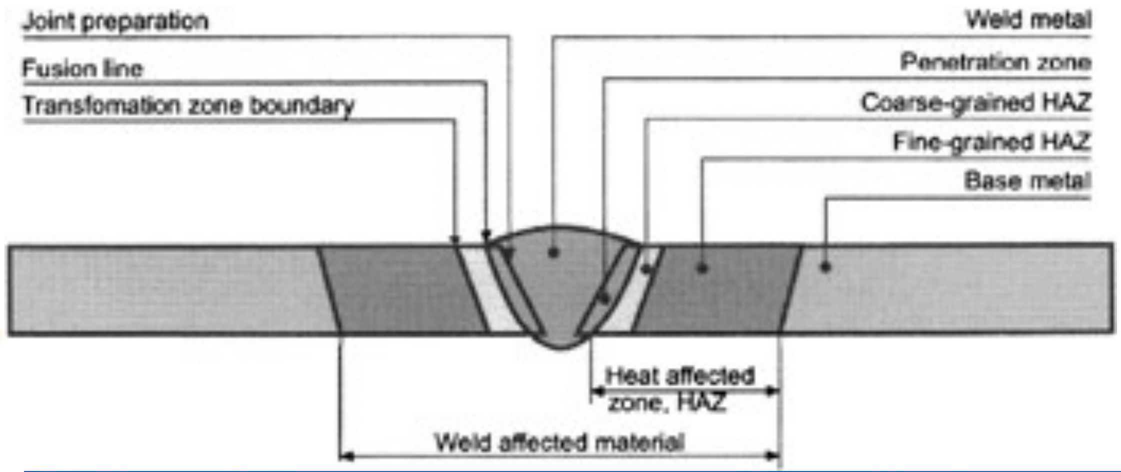
# RPV steels and alloys

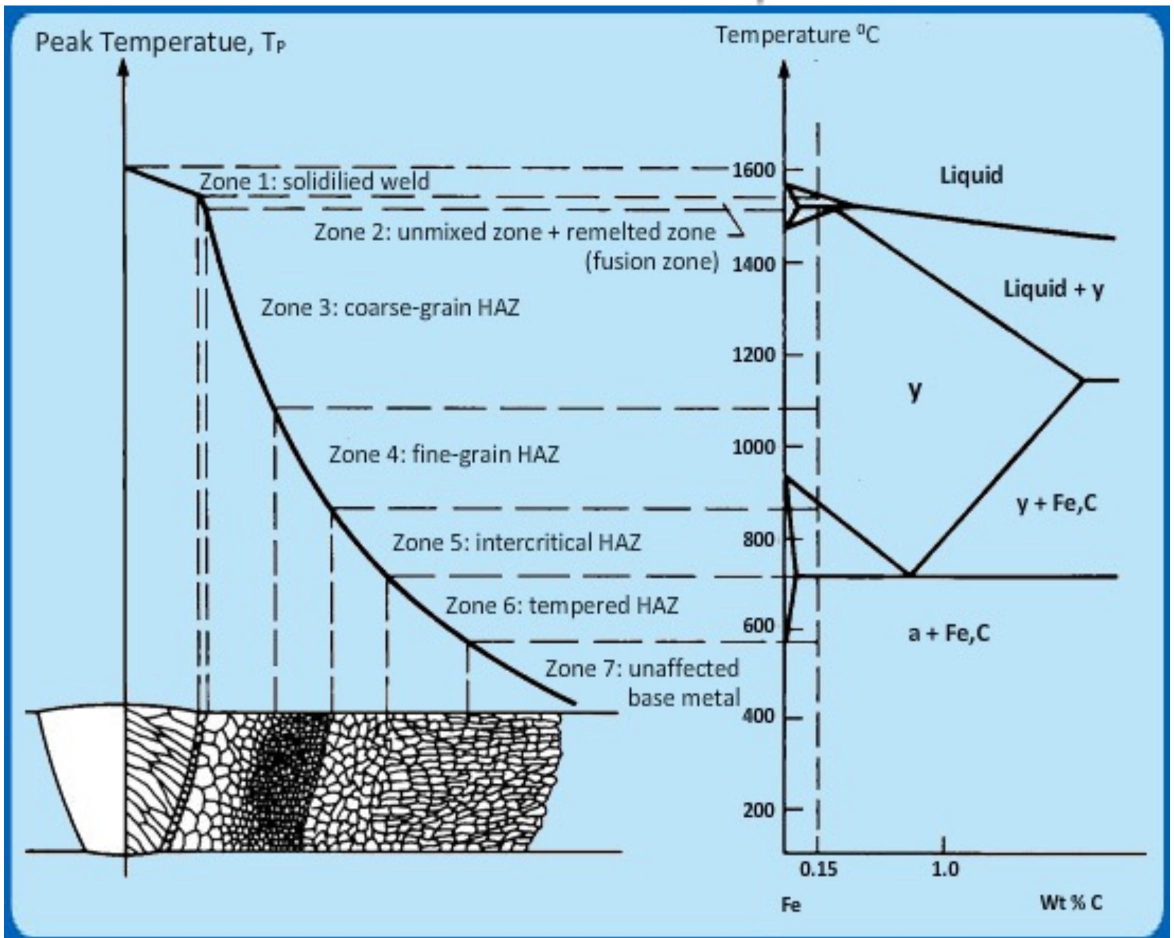
| Steel identification | Composition (wt.%)[a] | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | C max. | Mn | P max. | S max. | Si | Mo | Ni | Cu | Other |
| PLATE | | | | | | | | | |
| SA302 Grade B Class 1 | 0.25 | 1.10/1.55 | 0.035 | 0.040 | 0.13/0.32 | 0.41/0.64 | - | | |
| SA533 Grade B Class 1 | 0.25 | 1.15/1.50 | 0.035 | 0.040 | 0.15/0.30 | 0.45/0.60 | 0.40/0.70 | | |
| SA533 Grade B Class 1 (Low Copper) | (Same but for P, Cu) | | 0.017[b] 0.012[c] | | | | | 0.12[b] 0.10[c] | |
| SA537 Grade B Class 1 | 0.24 | 0.95/1.65 | 0.035 | 0.040 | 0.13/0.55 | 0.08 | 0.25[d] | 0.35 | Cr-0.25 |
| FORGING | | | | | | | | | |
| SA508 Class 2 | 0.27 | 0.50/0.90 | 0.025 | 0.025 | 0.15/0.35 | 0.45/0.60 | 0.50/0.90 | | Cr-0.25/0.45 V-0.05 max. |
| WELD – MANUAL | | | | | | | | | |
| E-7018 A1 | 0.12 | 0.90 | 0.03 | 0.04 | 0.80 | | 0.40/0.65 | - | |
| E-8018 C3 | 0.12 | 0.45/1.50 | 0.03 | 0.03 | 0.80 | 0.35 | 0.80/1.10 | - | V-0.05 |
| WELD SUBMERGED ARC | | | | | | | | | |
| High MnMoNiC | 0.13 | 2.25 | 0.020 | 0.020 | 0.10 | 0.40 | 0.55 | - | |

[a] Product analysis unless noted; maximum allowable analysis unless range given.
[b] Maximum allowable product analysis.
[c] Maximum allowable heat analysis.
[d] Typical filler analysis, usually specified by vessel fabricator.

These steels have specified yield strengths in the range of 220 - 350 MPa, ultimate tensile strengths in the range of 350 - 450 MPa and elongations in the range of 18-25%.

# Welding an important aspect

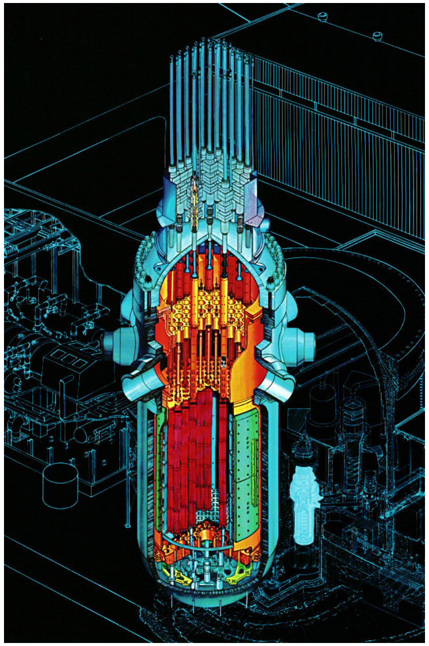


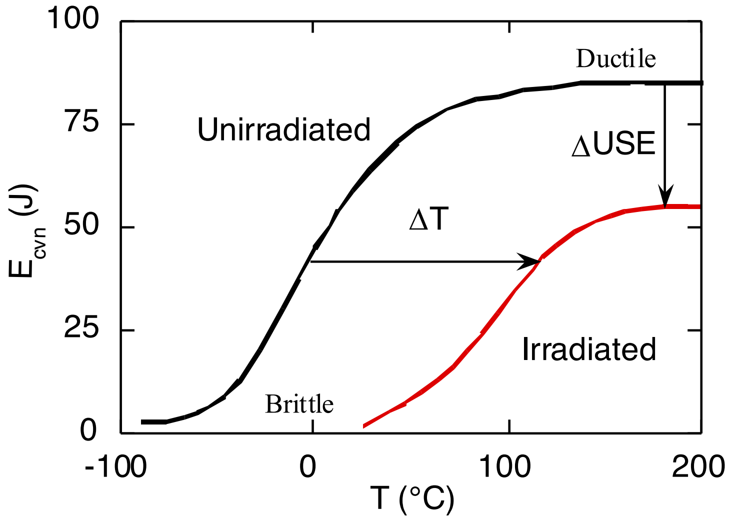



# RPV embrittlement

# *RPV embrittlement*

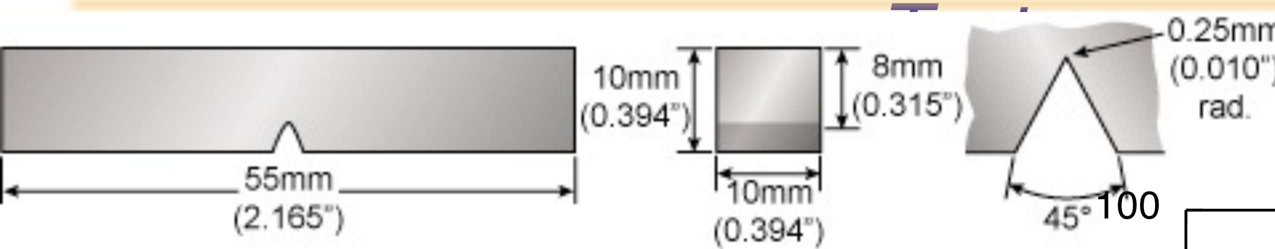
- **Reactor pressure vessel (RPV) embrittled by exposure to neutrons, manifested by transition temperature increases (ΔT) and upper shelf energy decreases (ΔUSE)**

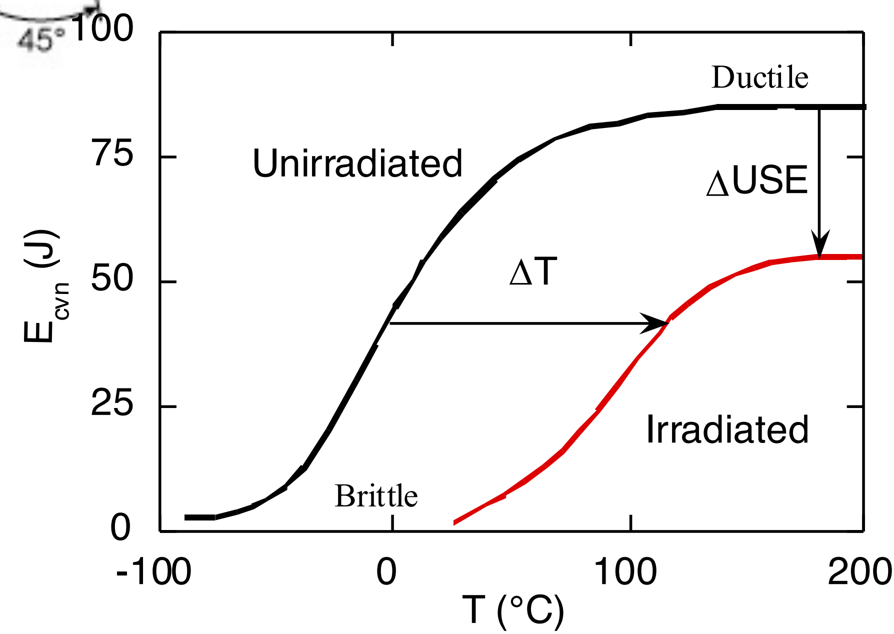


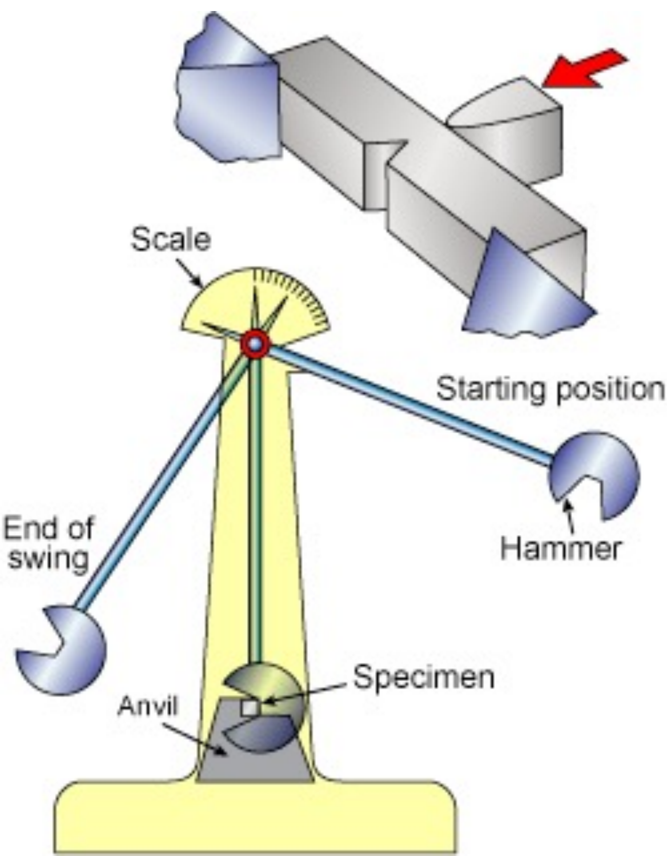



- **Early recognition of possible material degradation by regulators and industry led to RPV surveillance programs. Most reactors include capsules with representative RPV steels located on inside of RPV (higher dose rate) to provide early embrittlement estimates and database.**

# How to test for embrittlement: Charpy V-Notch

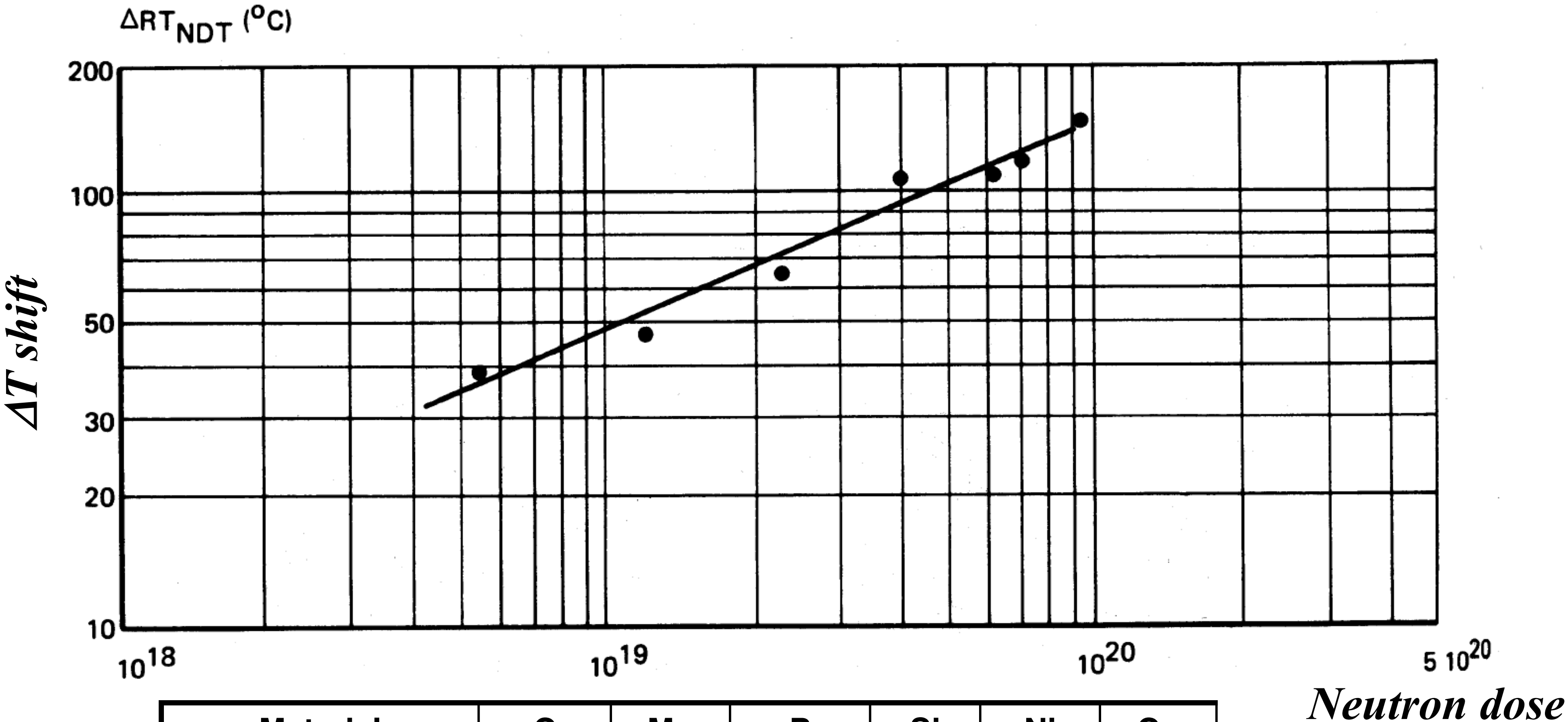


# Measured embrittlement

***Surveillance data for RPV embrittlement of the CHOOZ A PWR in France* – Surveillance material is from the Chooz pressure vessel C material***

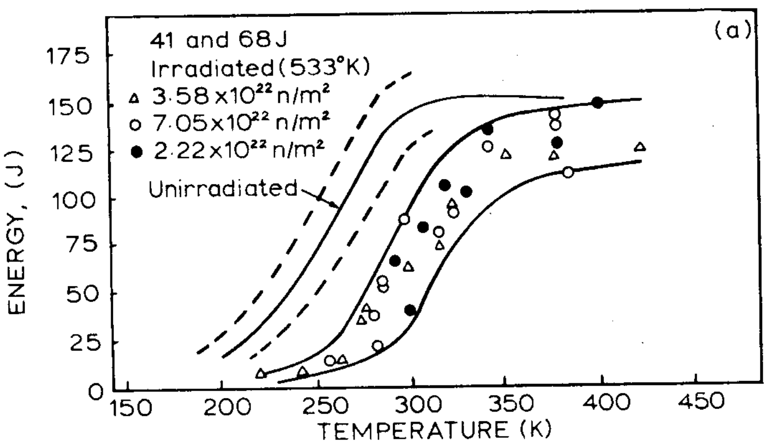


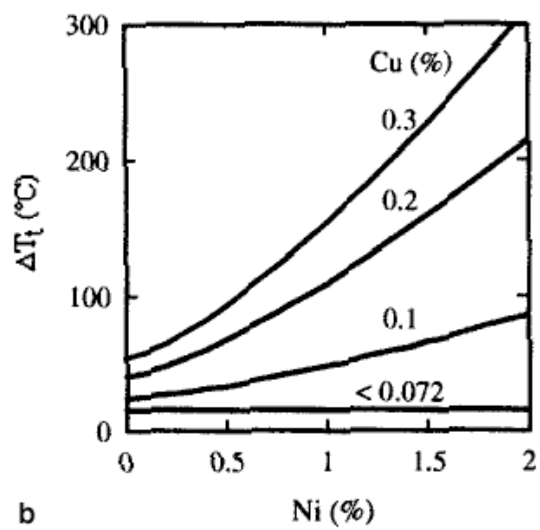

| Material | C | Mn | P | Si | Ni | Cu |
|---|---|---|---|---|---|---|
| Chooz C | 0.16 | 1.26 | 0.012 | 0.32 | 0.57 | 0.09 |

What is the underlying cause? What will happen at higher fluence?

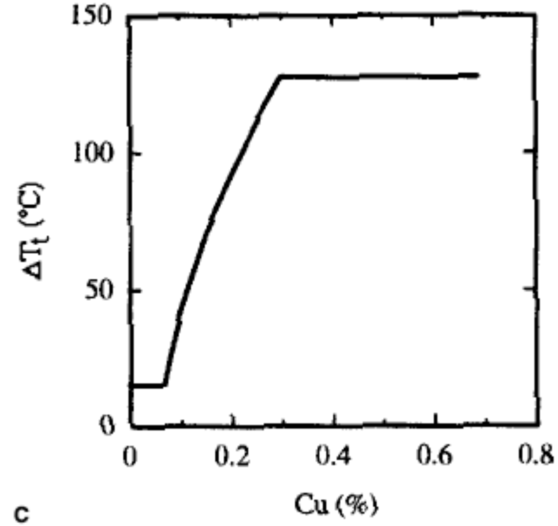
* J.C. Van Duysen, J. Bourgoin, C. Janot, and J.M. Pennisson, **ASTM STP** 1125 (1992) 117-130.

# Embrittlement Variable Trends



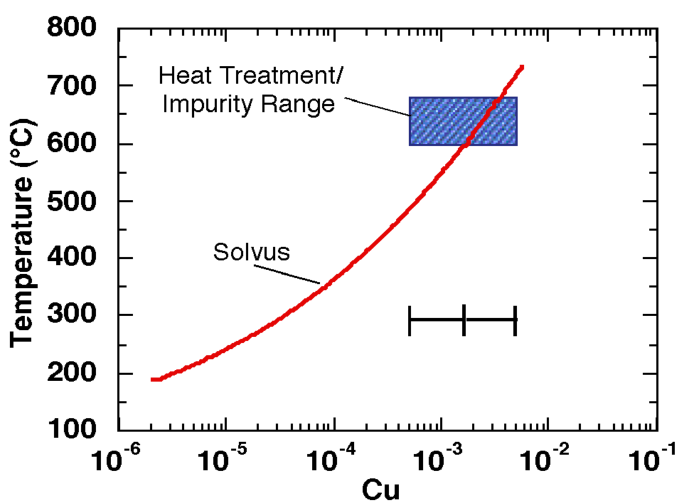



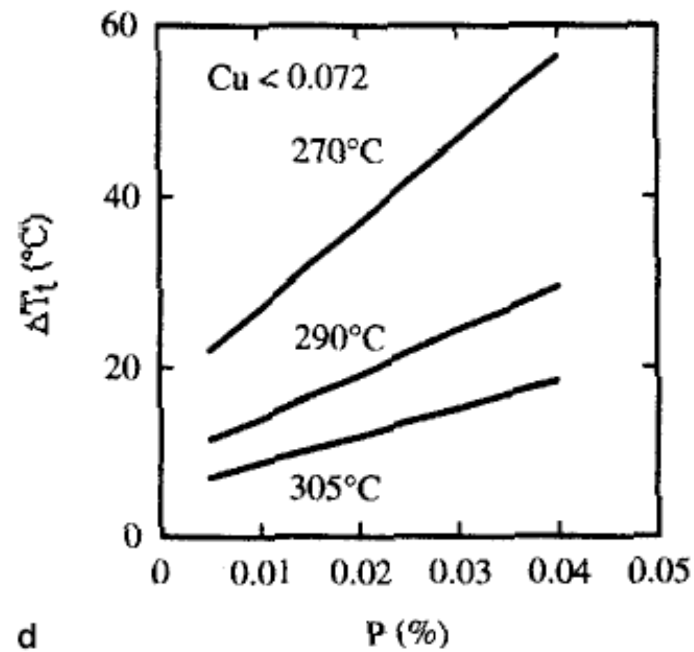

b




c

*Minor alloying impurities have a large effect on the DBTT*



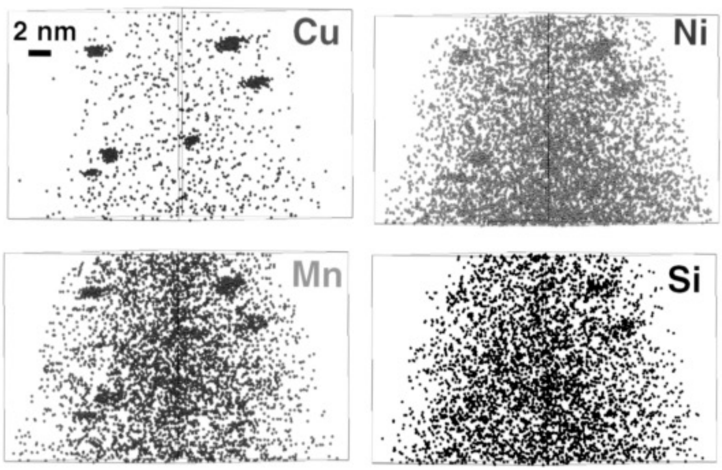



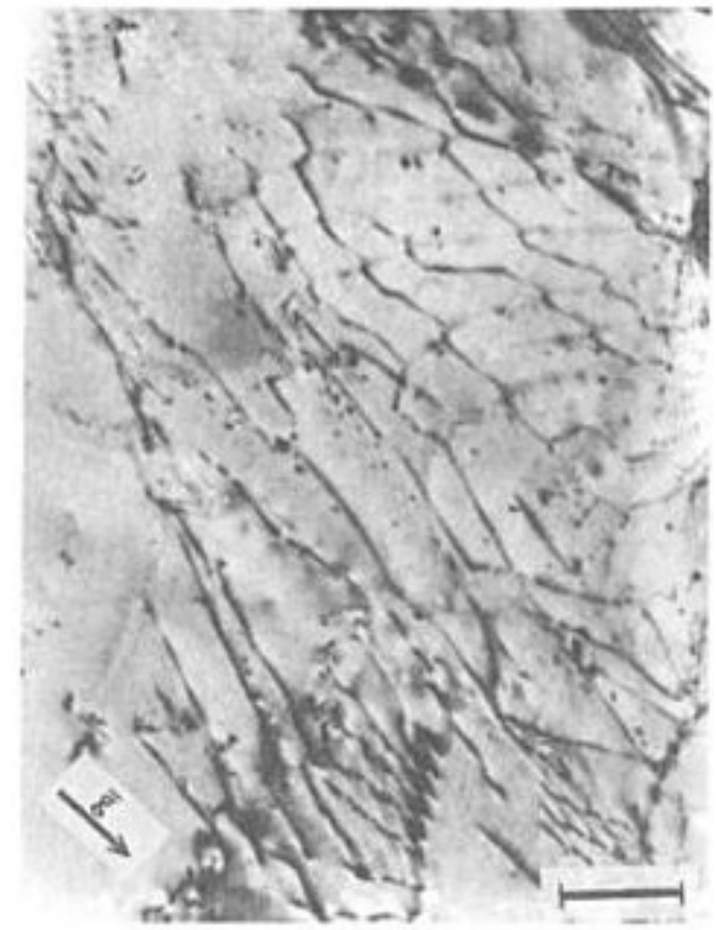

d

# Defects in RPV materials

| Material | C | Mn | P | Si | Ni | Cu |
|---|---|---|---|---|---|---|
| Chooz C | 0.16 | 1.26 | 0.012 | 0.32 | 0.57 | 0.09 |

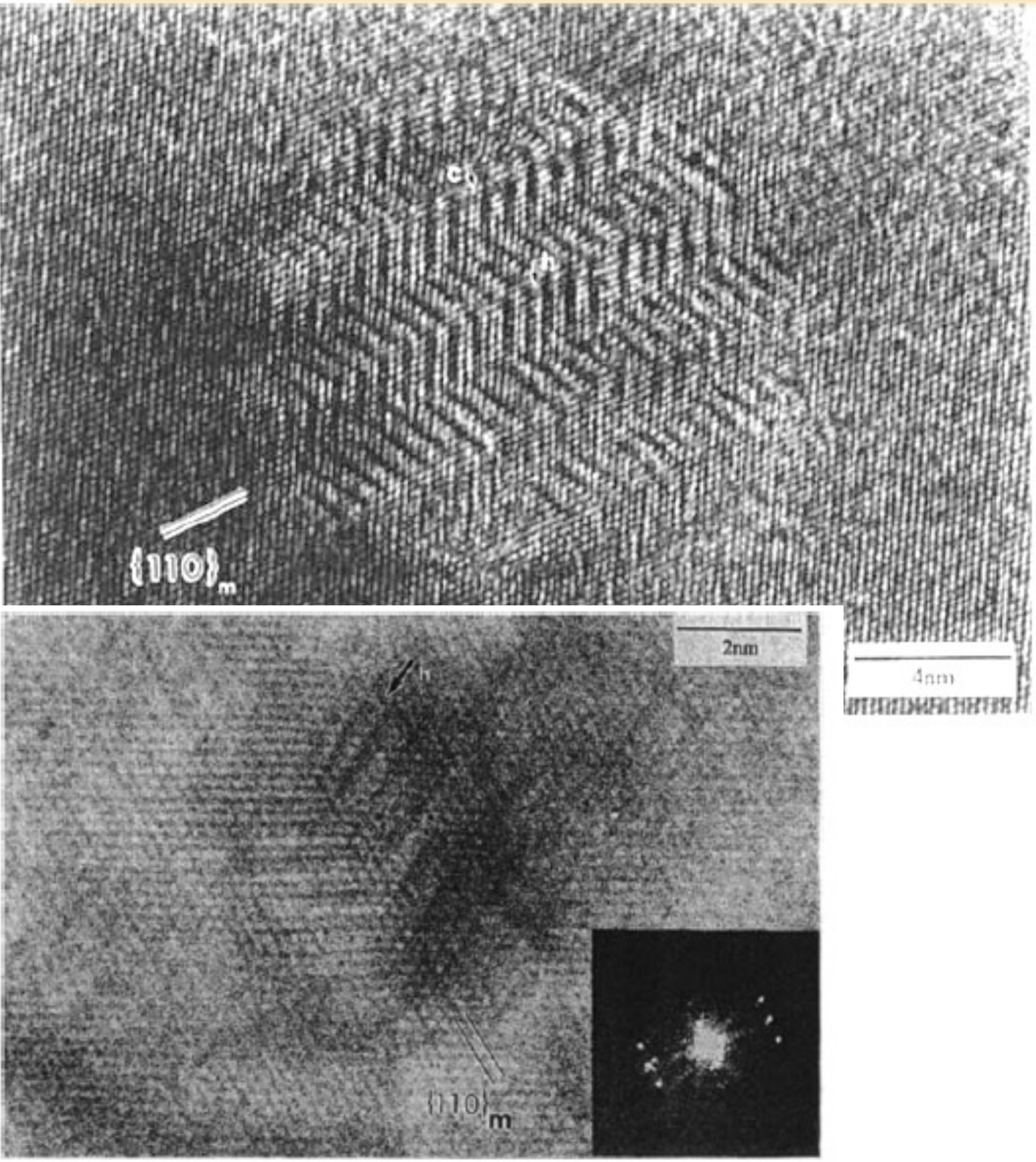
irradiated microstructure observed in the TEM*



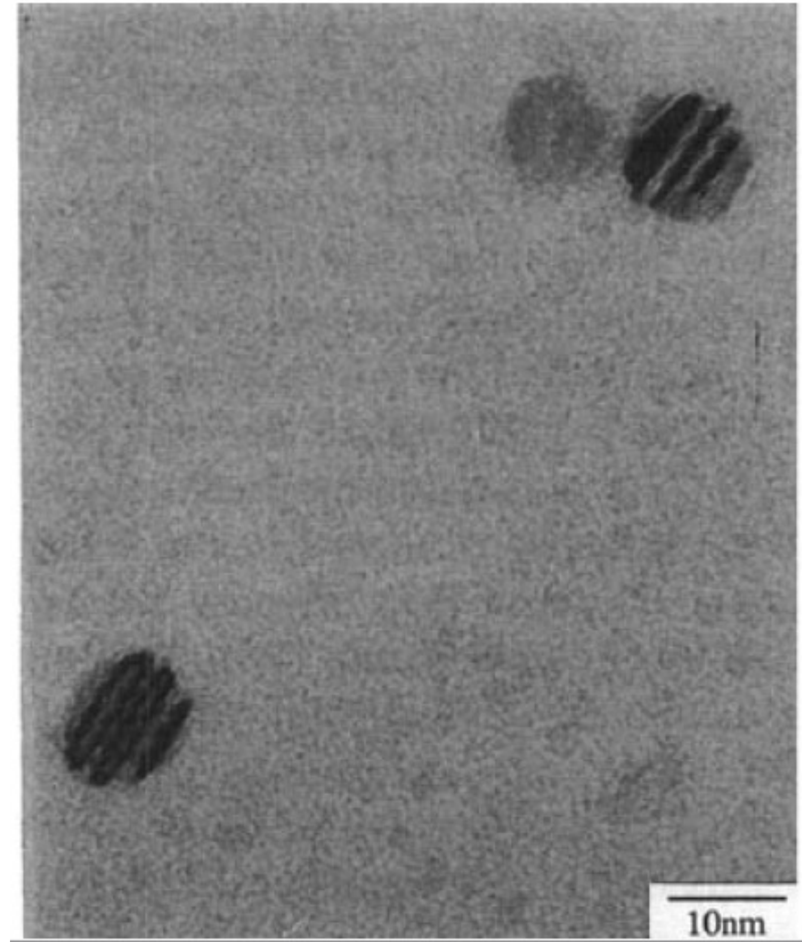



R.L. Steele Radiation embrittlement of nuclear reactor pressure vessel steels

# HR images of Cu in Fe




(a)




P. J. Othen, M. L. Jenkins & G. D. W. Smith (1994): High-resolution electron microscopy studies of the structure of Cu precipitates in α-Fe, Philosophical Magazine A, 70:1, 1-24

# *Lifetime of RPV assessment*

# *Pressurized thermal shock*

- Brittle fracture is not a problem during normal operation because the temperature is high (~300° C) and the main stress on the vessel is due only to residual stresses and the membrane (hoop) stress due to the internal pressure in the vessel.
- If cold water from the emergency core cooling system (ECCS) should be injected into the system (because of a LOCA) near the end of the vessel lifetime when $K_{Ic}$ is lowest due to irradiation embrittlement, brittle failure is possible.
- The rapid reduction in the inner wall temperature during ECCS creates tensile thermal stresses which add to existing stresses.
- Small flaws (cracks) on the inner wall of the vessel may have grown throughout the reactor lifetime due to fatigue during shutdown and startups.
- If the increased crack size and the corresponding increase in the applied stress intensity factor ($K_I$) exceeds $K_{Ic}$, the crack will grow rapidly.
- The crack may not penetrate the entire vessel wall because temperature, stress and irradiation dose change through the wall thickness. The crack will stop when the applied $K_I$ reaches the crack arrest toughness $K_{Ia}$.

# Remedies for RPV embrittlement and PTS

- "**New" pressure vessel:** eliminate impurities (Cu, P, high Ni/Mn) which enhance embrittlement, especially in weld material

- **Old vessels:**

  a) anneal critical zones to remove radiation embrittlement; nozzle welds and beltline welds - annealing at ~450° C for 1 week (extensively in Russia, not in US)

  b) change fuel management strategy (go to low leakage core):

  c) replace steel reflector elements at core periphery with materials which can more effectively absorb fast neutrons (tungsten and titanium hydride)

  d) heat the ECCS water to avoid/reduce thermal shock

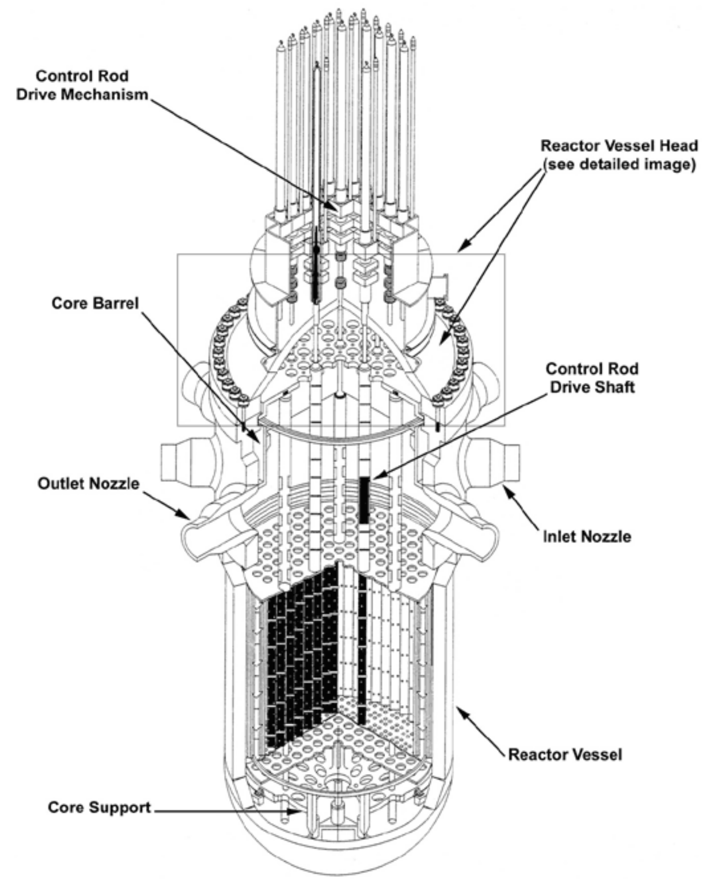
  e) demonstrate conservatism in the LEFM analysis

# *Davis Besse incident*

# Reactor vessel head corrosion

- ***Based on experience from French reactors, it was learned that axial cracks could form (likely stress corrosion cracking) in Control Rod Drive Mechanism penetration at the top (& bottom) of RPV***
- ***U.S. NRC issued Bulletin 2001-01 "Circumferential cracking of reactor pressure vessel head penetration nozzles" (Aug 2001).***
  - ***Inspections in fall 2001 revealed vessel head penetration nozzle cracks at Three Mile Island Unit 1, Crystal River Unit 3, Nort Anna Unit 1, and Oconee Unit 3.***
- ***August 2002, the NRC issued Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs"***
- ***Inspections at several PWRs in 2002, including Davis-Besse, found leakage and cracks that have required repairs or replacement***
- ***NRC Order on Feb 11, 2003 requires specific inspections***

**Typical Pressurized Water Reactor**

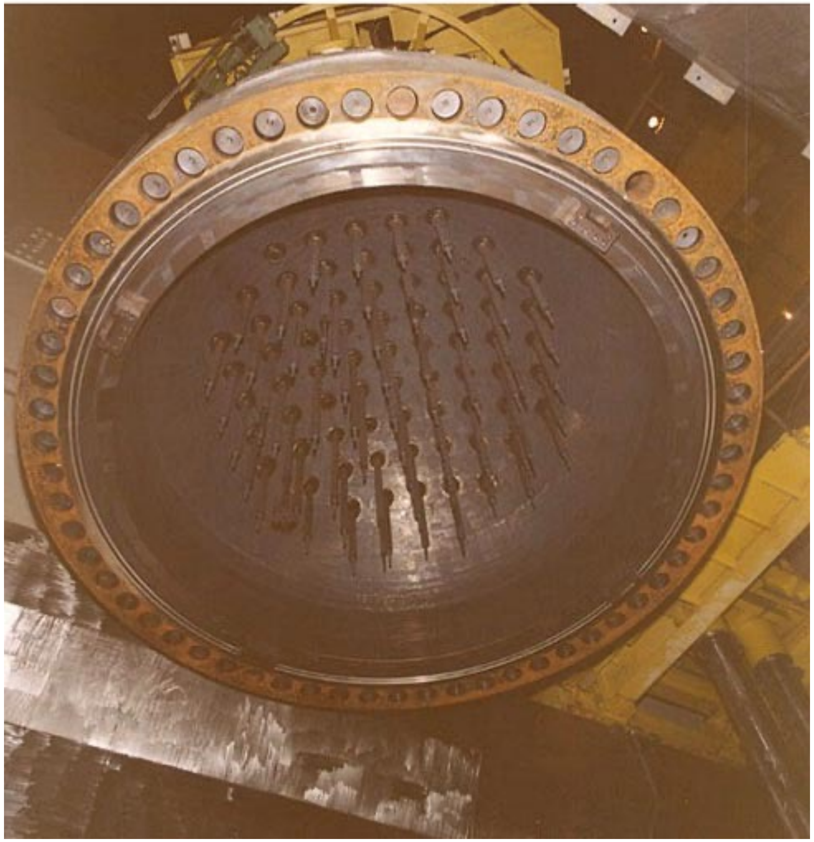



*Source: US NRC website:http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html*

# Reactor vessel head corrosion (Davis-Besse)

- Leaking of primary coolant and evidence of boric acid ($H_3BO_3$) outside Control Rod Drive Mechanisms (Reactor Pressure Vessel Head) at Davis-Besse indicated a cracking problem as early as fall 2001
- Davis-Besse management (First Energy Nuclear Corporation) continued to operate until forced by NRC order in Feb 2002 to specifically investigate the vessel head and associated penetrations.

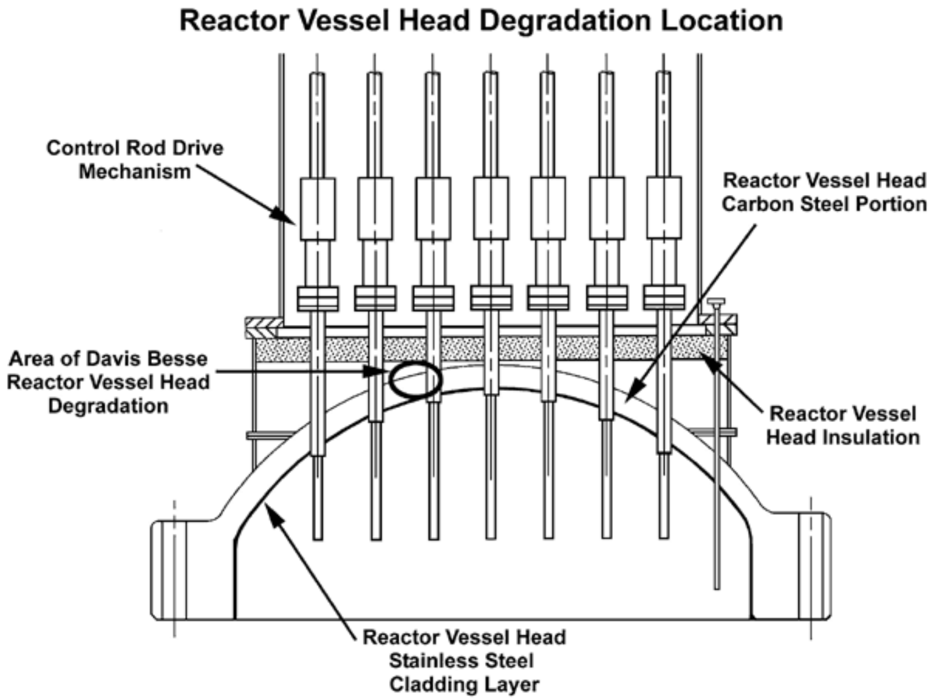
Davis-Besse Reactor Vessel Head



Source: US NRC website:http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html

# *Reactor vessel head corrosion (Davis-Besse, 3/02)*

- ***Leaking of primary coolant and evidence of boric acid outside Control Rod Drive Mechanisms** **(Reactor Pressure Vessel Head) at Davis-Besse indicated a** **cracking problem as early as fall 2001***
- ***Davis-Besse management (First Energy Nuclear Corporation) continued to operate until forced by NRC order in Feb 2002 to specifically investigate the vessel head and associated penetrations.***
- ***Inspection at Davis-Besse revealed cracking in CRDM Nozzles 1, 2 and 3 - and, upon further investigation - significant corrosion of RPV head***



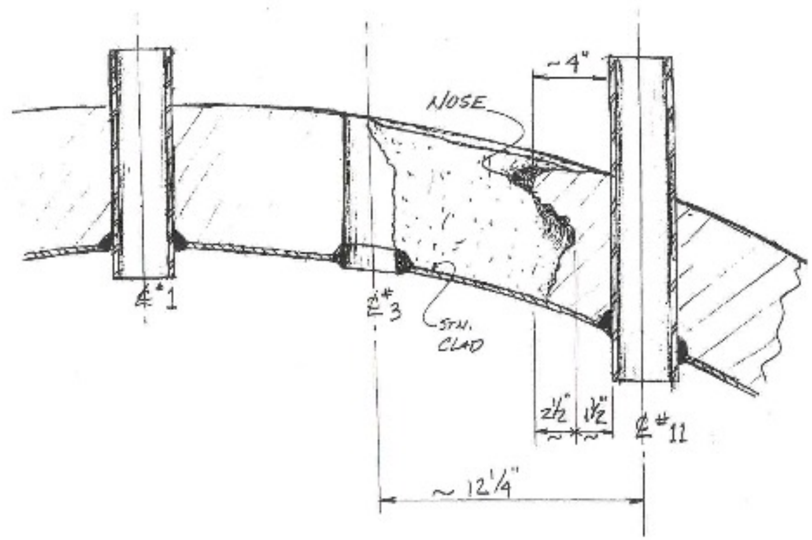

*Top-view of Vessel Head, region around CRDM 3 (cutout)*


*Source: US NRC website:http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html*

# *Reactor vessel head corrosion (Davis-Besse, 3/02)*

***"The wastage area was found to extend approximately 5 inches downhill on the RPV head from the penetration for CRDM nozzle 3 and was approximately 4 to 5 inches at its widest part. The minimum remaining thickness of the RPV head in the wastage area was found to be approximately 3/8 inch. This thickness was attributed to the thickness of the stainless steel cladding on the inside surface of the RPV head, which is nominally 3/8 inch thick."***


Davis Besse Reactor Vessel Head Degradation Head Cutaway View




The above figure shows the Davis Besse reactor vessel head degradation between nozzle #3 and nozzle #11. This sketch was provided to the NRC by the Licensee.

*Source: US NRC website:http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html*

# *Reactor vessel head corrosion (Davis-Besse, 3/02)*

***Further, the stainless steel cladding had significantly bowed upward and cracked, indicating substantial plastic deformation - the cladding, not designed as a pressure containment structure is all that prevented a severe loss of coolant accident***

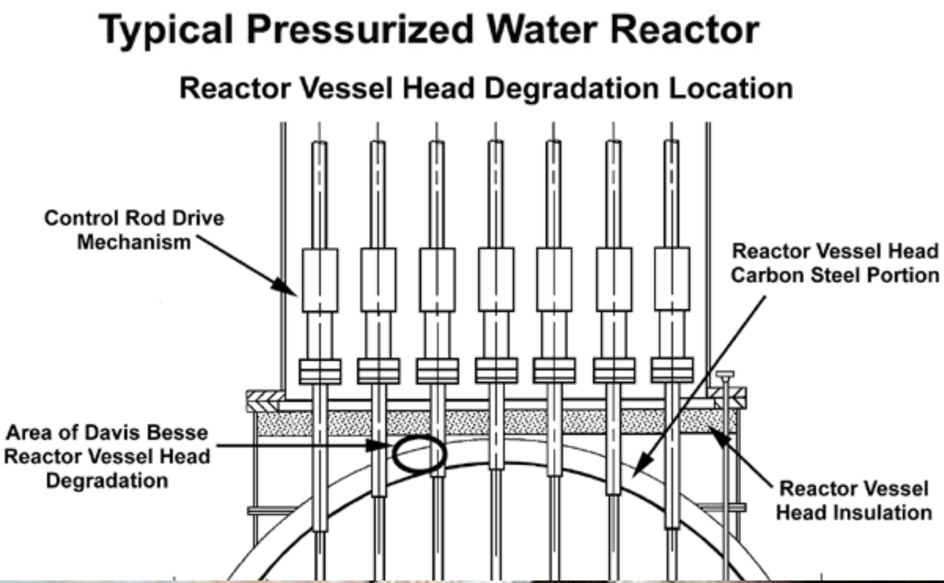


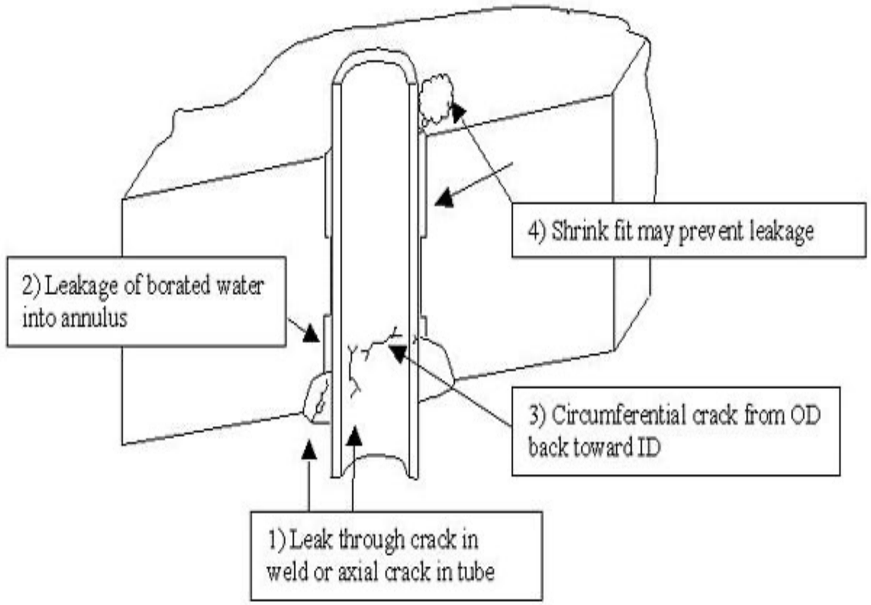
*"The investigation of the causative conditions surrounding the degradation of the RPV head at Davis-Besse is continuing. Boric acid or other contaminants could be contributing factors. Other factors contributing to the degradation might include the environment of the RPV head during both operating and shutdown conditions (e.g., wet/dry), the duration for which the RPV head Is exposed to boric acid, and the source of the boric acid (e.g., leakage from the CRDM nozzle or from sources above the RPV head such as CRDM flanges)."*

***Source: US NRC website:http://www.nrc.gov/reactors/operating/ops-experience/vessel-head-degradation.html***


2002. 5. 14

# What seems to have happened?





Top View - Nozzle #3 Removed



RM-69


***Outcome:***
***First Energy Corporation pay fines of $23.7 million, with an additional $4.3 million to be contributed to environmental agencies.***
***Andrew Siemaszko 5 years in prison and 250.000$ fine. Additional worker was convicted with three years probation***