



# Phenomenon of (irradiation assisted) stress corrosion cracking for internals of PWR & BWR systems

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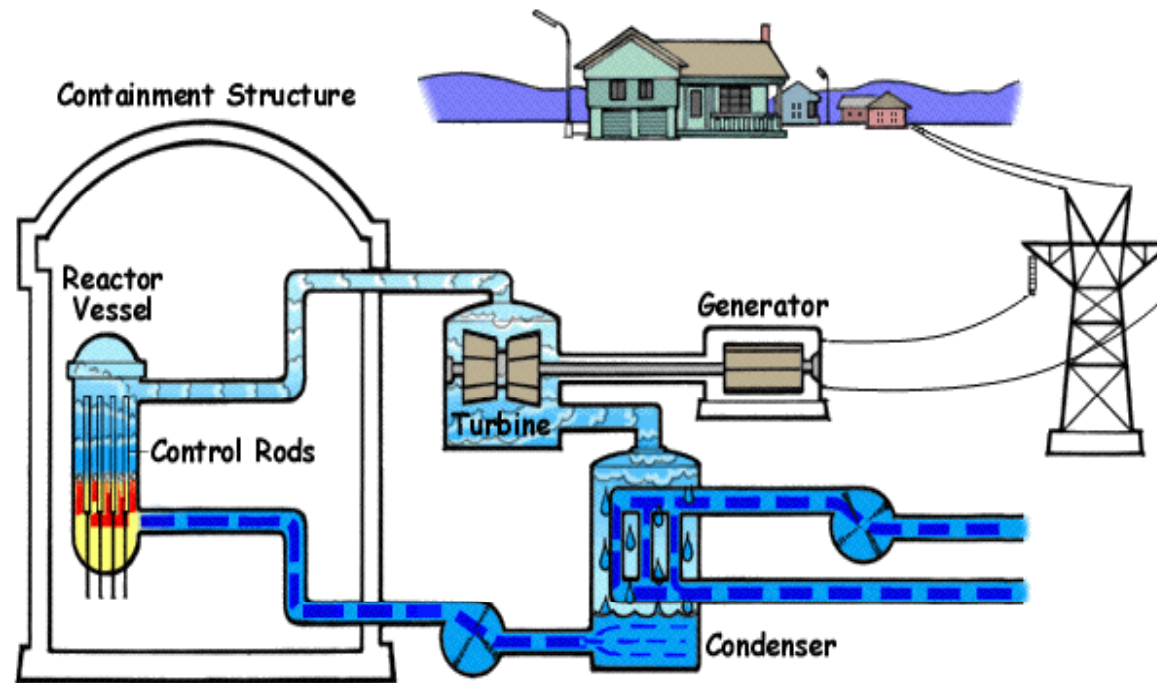
<http://www.jrc.ec.europa.eu>

Trieste, April 2009

# CONTENT

- Ageing
- Corrosion
- SCC – PW SCC
- Radiation effects
- IASCC
- CONCLUSIONS

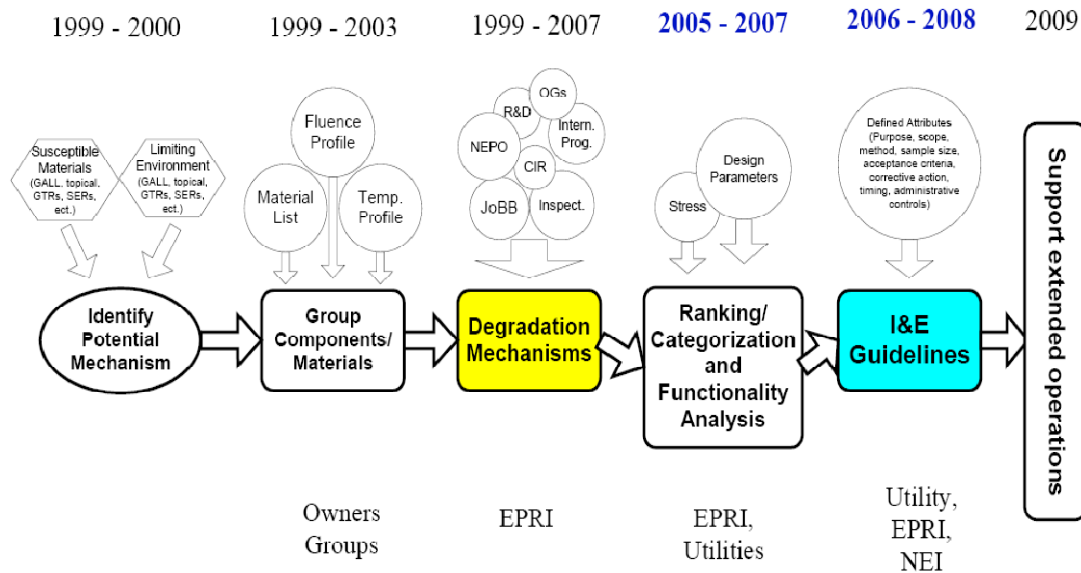
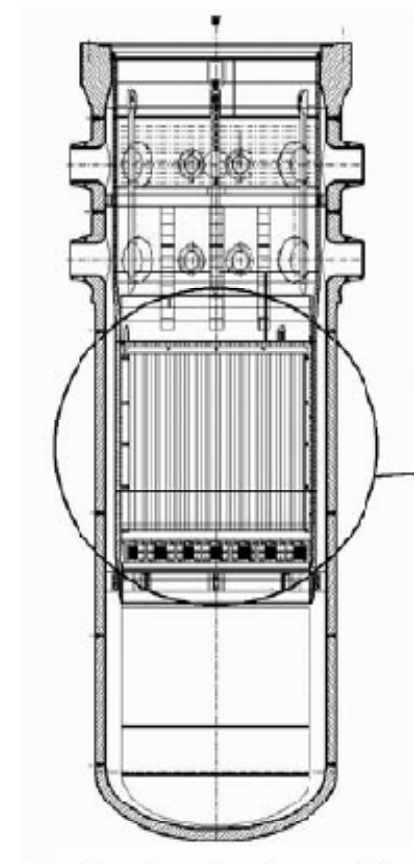




- Effort to plan life management of ageing NPPs
- Reliability of in-core in-vessel structural materials
- Control of the degradation of the structural materials
- Guiding replacement campaigns
- **Rising issues for alloys in LWR conditions:**
  - **Corrosion and**
  - **Stress Corrosion Cracking (SCC PW SCC)**
  - **IASCC**

# Industry's Top Ten R&D Priorities (from MRP-205)

1. Inspection & Evaluation (I&E) Guidelines: **Reactor Internals**
2. NDE Technology: Dissimilar Metal (DM) Butt Welds
3. **PWSCC Mitigation: Environmental Controls**
4. I&E Guidelines: Bottom Mounted **Nozzles**
5. Vibration Fatigue: Small Bore Piping
6. **Environmental Fatigue Issues: Press. Boundary Components**
7. NDE Qualification Program: **Ni-Alloy Penetrations**
8. NDE Accessibility: **Reactor Internals**
9. **PWSCC Mitigation: Stress Improvement (SI) of Butt Welds**
10. Thermal & Irradiation Embrittlement: Synergistic Effects on CASS & SS Welds - **Internals**



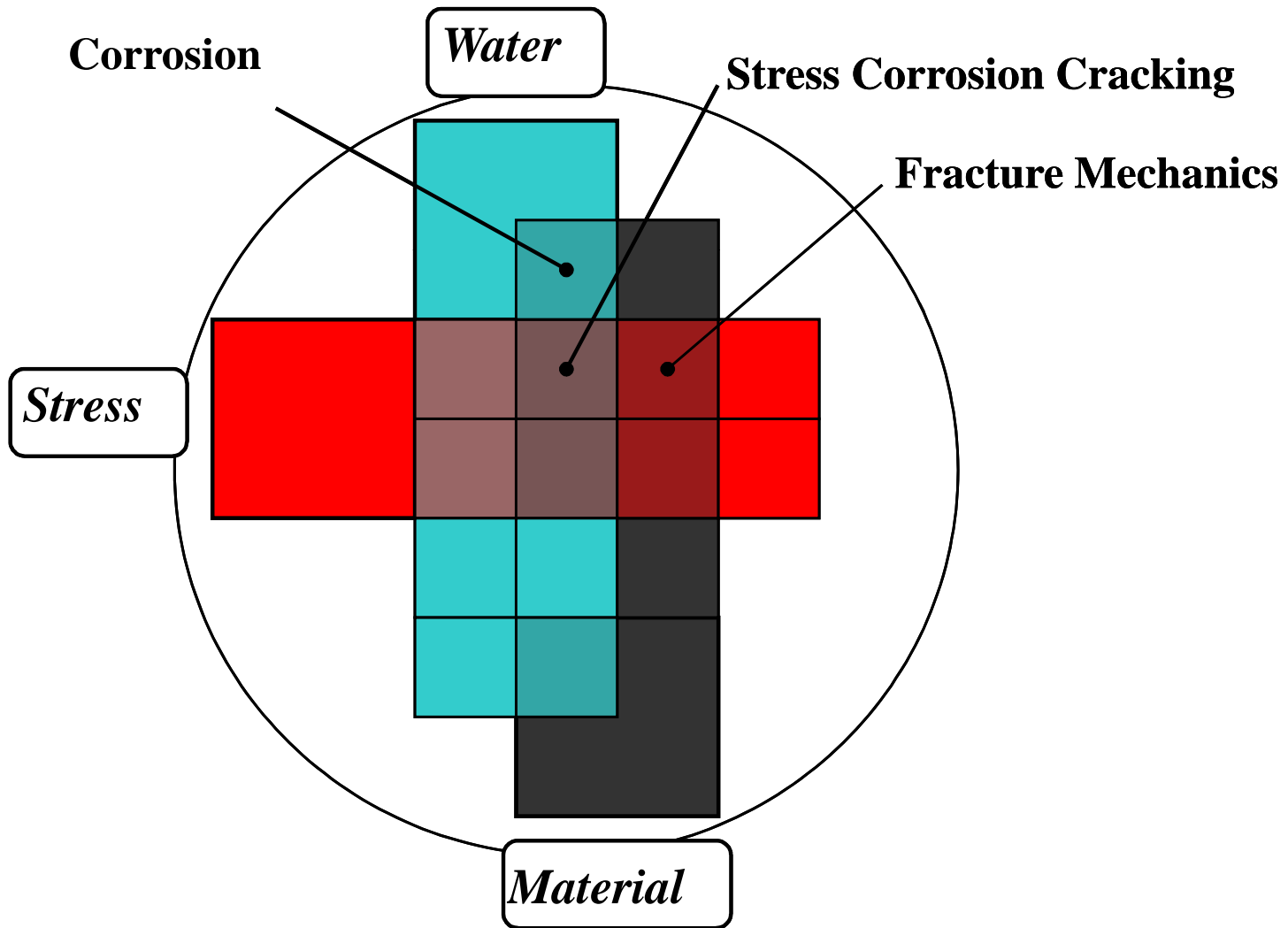
# Essential service water system pipe degradation



# Stress Corrosion Cracking

- **Attachment welds:** SCC significant at welded pad/bracket locations in the vessel shell
- **Nozzles:** SCC of nozzles is a significant issue
- **Closure studs**
- **Penetrations:** SCC of CRD stub tubes (high residual stresses in sensitized weld material)
- **Safe ends:** Observed at several plants. SCC is a potentially significant degradation mechanism for safe ends





# Main Degradation Mechanisms

Component	Irr. Emb.	Fatigue	Thermal Ageing	Wear
Reactor pressure vessel	○	○	...	
Control rod drive mechanisms		○		○
Internals structures	○	○	○	
Reactor coolant pump casing		○	○	
Piping and safe ends		○	○	
Pressurizer		○		
Surge and Spray lines		○		
Steam generator tubing		○		○
Steam generator shell and nozzles		○	○	



# Loviisa



# PWSCC Experience in Vessel and Other Components

## Plants with leaks:

- ❑ V.C. Summer – axial, reactor vessel nozzle (2000)
- ❑ Tsuruga 2 – axial, pressurizer valve nozzle (2003)

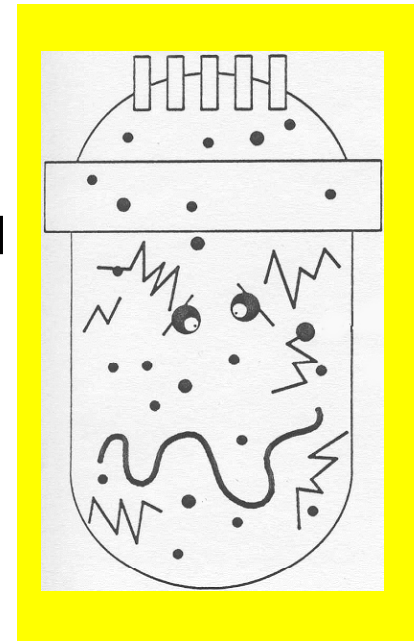
## Plants with cracks/indications:

- ❑ Ringhals 3, 4 – axial, reactor vessel nozzle (2000)
- ❑ V.C. Summer – circ. y axial, reactor vessel nozzles (2000)
- ❑ Tihange 2 – axial, pressurizer nozzle (2003)
- ❑ Calvert Cliffs 2 (2005), DC. Cook 1 (2005), Calvert Cliffs 1 (2006), Davis Besse (2006).....

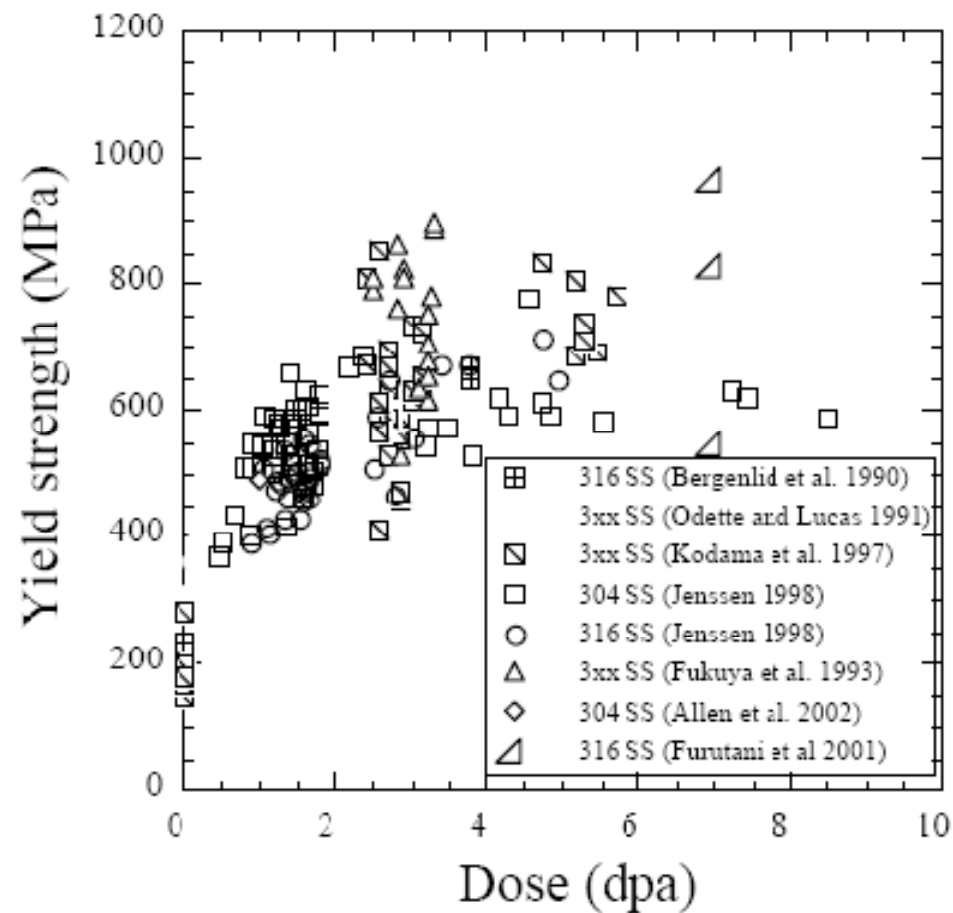


# Corrosion & radiation

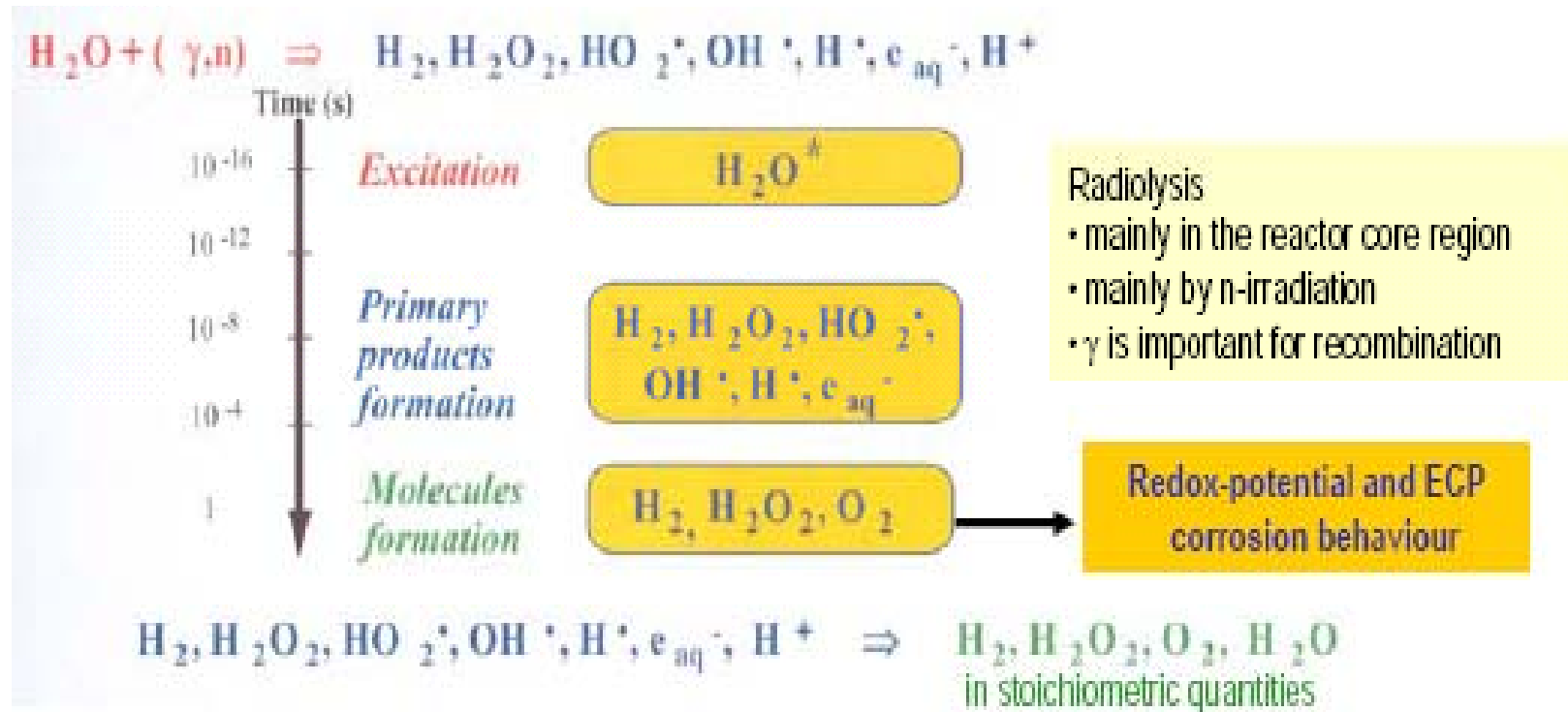
- **Effects of irradiation on materials are well investigated**
  - Defects due to irradiation
  - Mechanical properties evolution with dose
- **For LWRs, passive materials used**
- **Corrosion enhanced by water radiolysis**
  - oxidant species ( $\text{OH}^-$ ,  $\text{H}_2\text{O}_2$ , etc.)
  - together with reducing species (H, etc)
  - influence free corrosion potential
- **Radiation affects also semi-conductive properties of the oxides, particularly the behaviour of the passive layer**



## Irradiation hardening increases with (dose)<sup>1/2</sup>

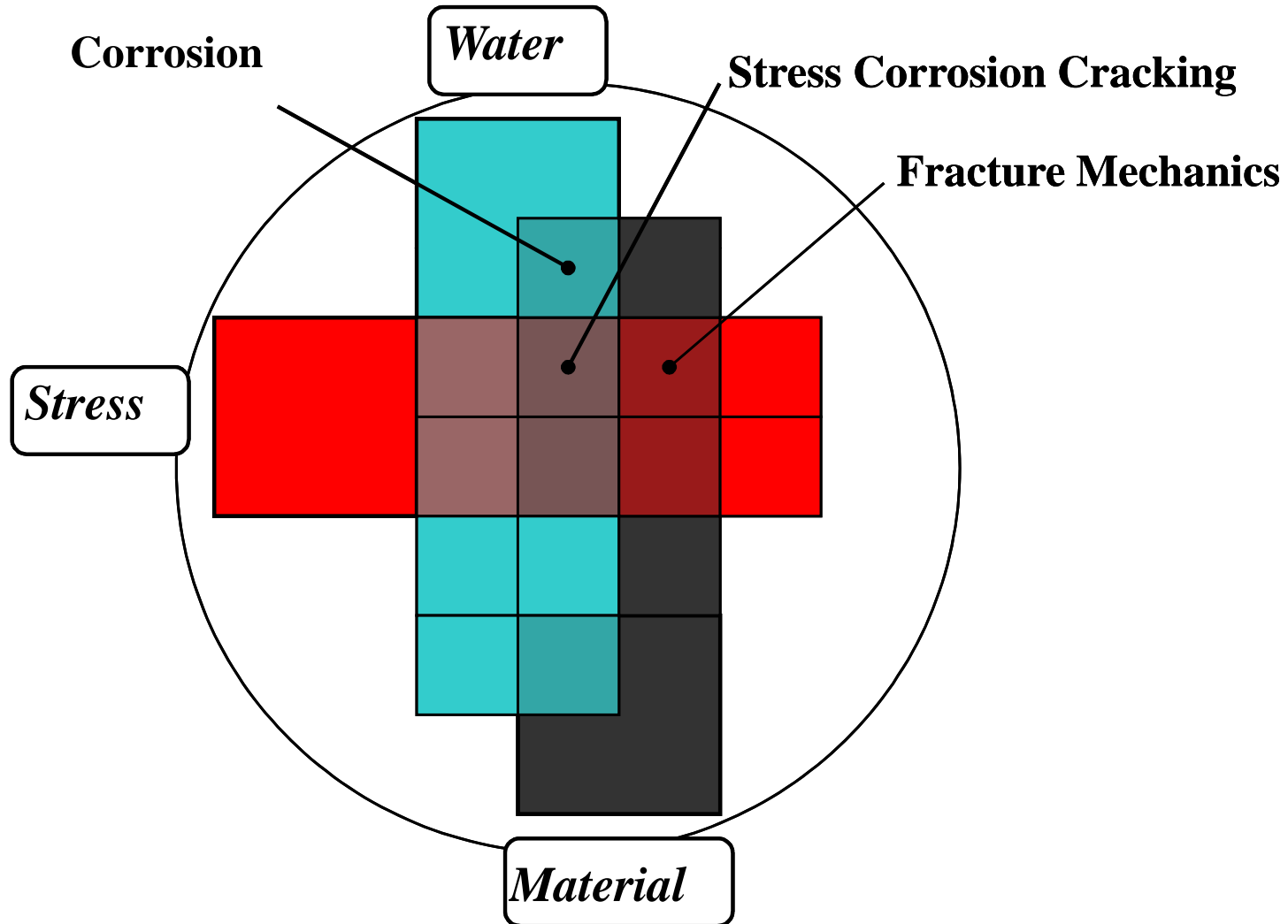


## Radiolysis of Water by n- and $\gamma$ Radiation

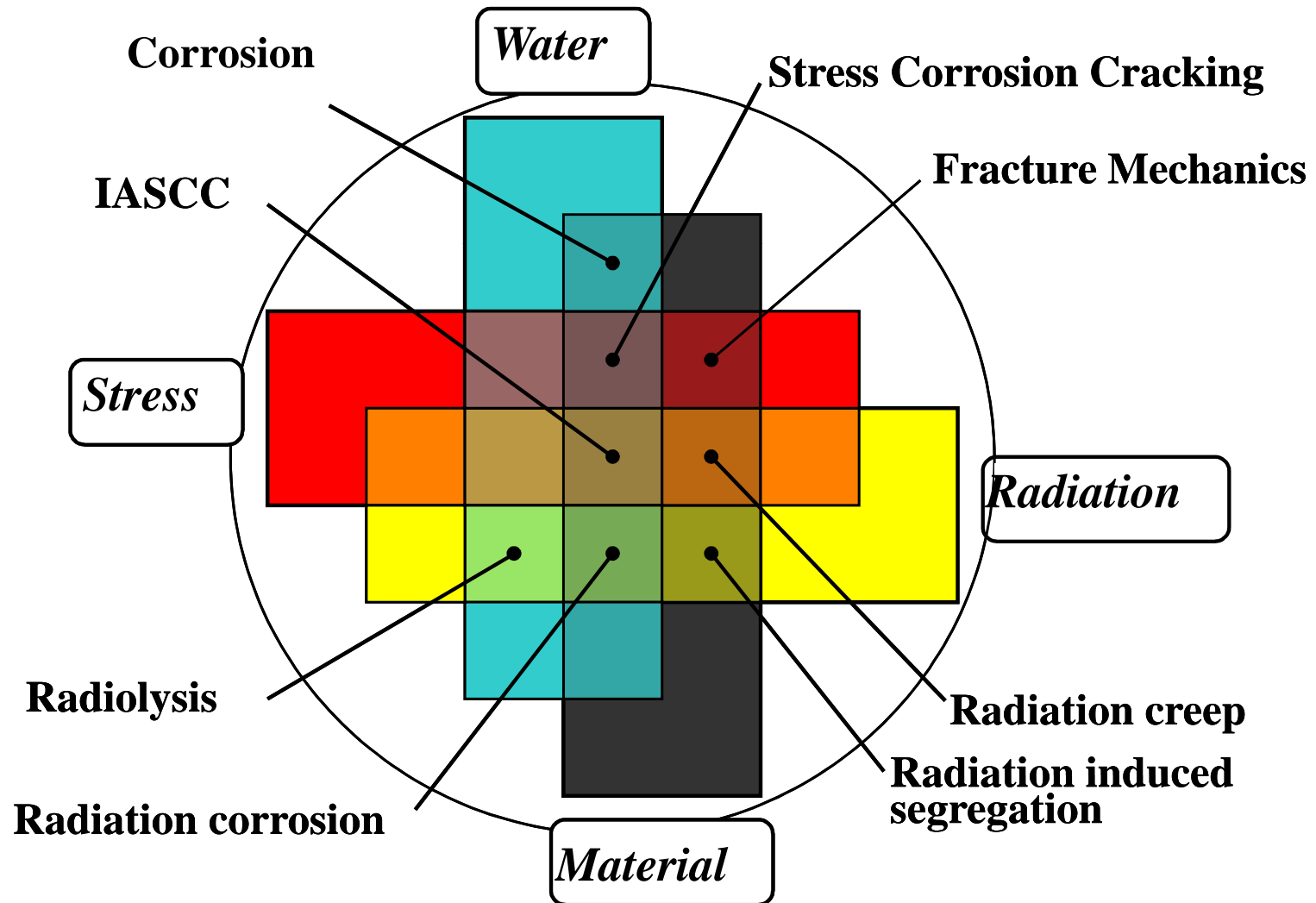


- **PWR:**  $\text{H}_2$ -addition (in VCT)  $\rightarrow$  recombination with  $\text{O}_2 + \text{H}_2\text{O}_2$  to  $\text{H}_2\text{O} \rightarrow \text{H}_2 \gg \text{O}_2 + \text{H}_2\text{O}_2 < 5 \text{ ppb}$   
 $\rightarrow$  low redox-potential  $\rightarrow$  ECP = -800 to -500 mV<sub>SHE</sub>
- **BWR/NWC:**  $\text{H}_2\text{O}_2$  is non-volatile  $\rightarrow \text{O}_2 + \frac{1}{2} \text{H}_2\text{O}_2 > \text{H}_2 \rightarrow$  oxidizing  $\rightarrow$  ECP = -50 to +200 mV<sub>SHE</sub>
- **BWR/HWC or NMCA:**  $\text{H}_2$ -addition to feedwater  $\rightarrow$  recombination with  $\text{O}_2 + \text{H}_2\text{O}_2$  to  $\text{H}_2\text{O} \rightarrow$   
 $\text{H}_2 \gg \text{O}_2 + \text{H}_2\text{O}_2 < 5 \text{ ppb} \rightarrow$  low redox-potential  $\rightarrow$  ECP = -500 to -200 mV<sub>SHE</sub>

- The integrity of fuel elements (zircaloy) may also be affected
- Localised corrosion phenomena are also affected by the free corrosion potential changes, particularly SCC phenomena



# + Radiation



# Present generation



**LWR** → **BWR, PWR, VVER**

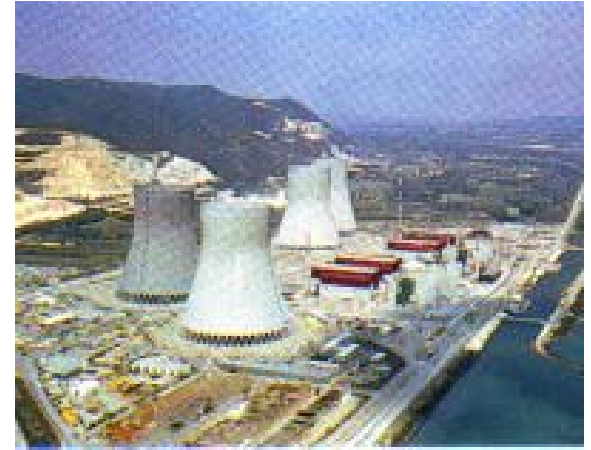
**HW** → **CANDU**

**GCR** → **MAGNOX, AGR**

**LM** → **FR**



# BWR - PWR



- Interactions corrosion behaviour and radiolysis
- Slightly diversified for the two main LWRs
  - Boiling water reactors (BWR) and
  - Pressurised water reactor (PWR)

## Water Chemistry Conditions in BWR and PWR

### BWR/NWC

- $T = 274 - 288 \text{ }^\circ\text{C}$
- $\text{pH}_T = 5.65$
- Neutral
- High-purity water
- Impurities  $< 1 \text{ ppb}$
- $\text{O}_2 \sim 200 \text{ ppb}$
- $\text{H}_2 \sim 15 \text{ ppb}$
- $\text{ECP} = -50 \text{ to } +200 \text{ mV}_{\text{SHE}}$

### BWR/HWC

- $T = 274 - 288 \text{ }^\circ\text{C}$
- $\text{pH}_T = 5.65$
- Neutral
- High-purity water
- Impurities  $< 1 \text{ ppb}$
- $\text{O}_2 < 5 \text{ ppb}$
- $\text{H}_2 \sim 50 \text{ to } 300 \text{ ppb}$
- $\text{ECP} = -500 \text{ to } -200 \text{ mV}_{\text{SHE}}$

### PWR

- $T = 286 - 323 \text{ }^\circ\text{C}$
- $\text{pH}_T = 6.8 \text{ to } 7.4$
- Slightly alkaline
- Pure water
- $\text{H}_3\text{BO}_3/\text{LiOH}$
- $\text{O}_2 < 5 \text{ ppb}$
- $\text{H}_2 \sim 2000 \text{ to } 5000 \text{ ppb}$
- $\text{ECP} = -800 \text{ to } -600 \text{ mV}_{\text{SHE}}$

# BWR - PWR



**BWRs:** stainless steels (304 type) mainly used for core components

- exposed to water often up to 288 °C
- SCC is controlled by hydrogen conditioning (HWC: H water chemistry)
  - free corrosion potential (ECP) at low values

## PWRs

- primary water chemistry based on B (added as boric acid)
- for neutronic purposes with Li addition (added as LiOH)
- increase pH to limit the general corrosion and activation of components
- large hydrogen concentration used to limit radiolysis effects
  - 25 to 35 ml of hydrogen per Kg water!

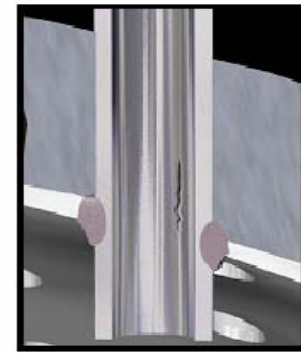
# Boric Acid Corrosion

- Boric acid leakage is a consequence of Alloy 600 cracking
- This leakage can lead to boric acid corrosion of low-alloy steel
- Davis-Besse, March 2002
- Order EA-03-009.  
Inspection requirements according to the parameter EDY



Davis-Besse

# Davis-Besse

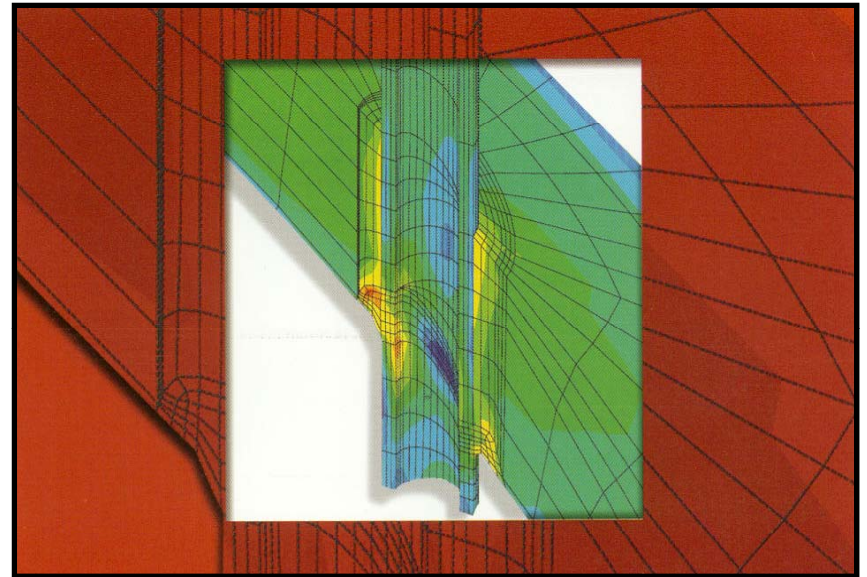




# Plants with Replaced RPV Upper Head - USA

- 25 plants replaced RPV upper heads
  - 24 with Alloy 690 penetrations
  - 1 with Alloy 600 (Davis-Besse)
- 13 of the 21 remaining high and moderate susceptibility plants

## *PWSCC. Vessel Head*



- The *Alloy 600/82/182/* has been changed by the *Alloy 690/52/152* in the penetrations of the new vessel heads
- The inspection practices vary from country to country, reducing the inspection intervals for the vessel heads with Alloy 600

# Stress Corrosion Cracking

- ❑ BWR internals susceptible to two forms of SCC:
  - ❑ Intergranular stress corrosion cracking (IGSCC)
  - ❑ Irradiation assisted stress corrosion cracking (IASCC)
- ❑ Degradation via IGSCC is potentially significant. Programmes to effectively manage this degradation mechanisms are required
- ❑ IASCC is a concern in BWR core internal components such as a portion of the core shroud and the top guide.



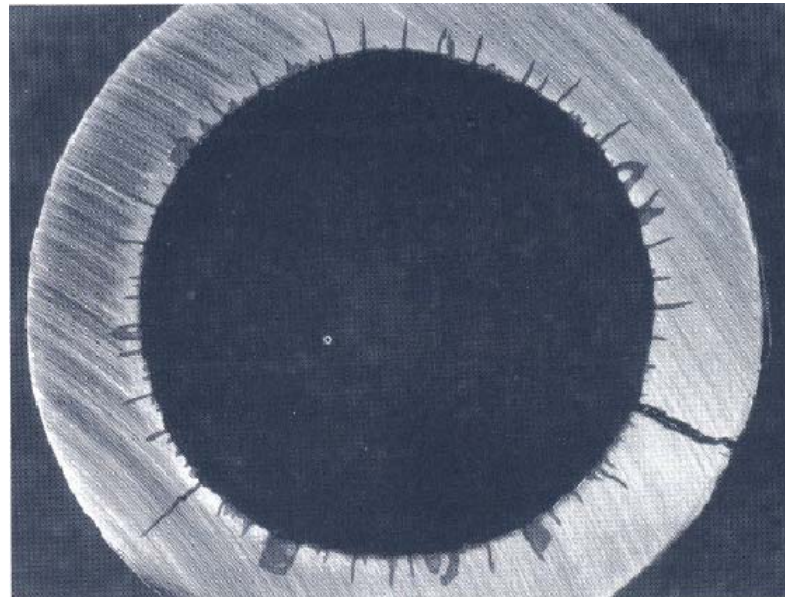
# Degradation Incidents of RPVIs Safety Relevant

Component	Degradation Mechanism
1. Core Plate	IGSCC
2. Core Spray Internal Piping	IGSCC
3. Core Spray Sparger	IGSCC
4. CRD Guide Tube	No incidents of cracking reported
5. CRD Housing	No incidents of cracking reported
6. In-Core Housing	IGSCC
7. Jet Pump	
– Diffuser	IGSCC
– Hold-down beam	IGSCC
– Inlet mixer	Fatigue due to improper installation
– Riser	IGSCC
8. LPCI Coupling	No incidents of cracking reported
9. Orificed Fuel Support	No incidents of cracking reported
10. Core shroud	IGSCC/IASCC
11. Shroud Support	IGSCC
12. Top Guide	IGSCC/IASCC

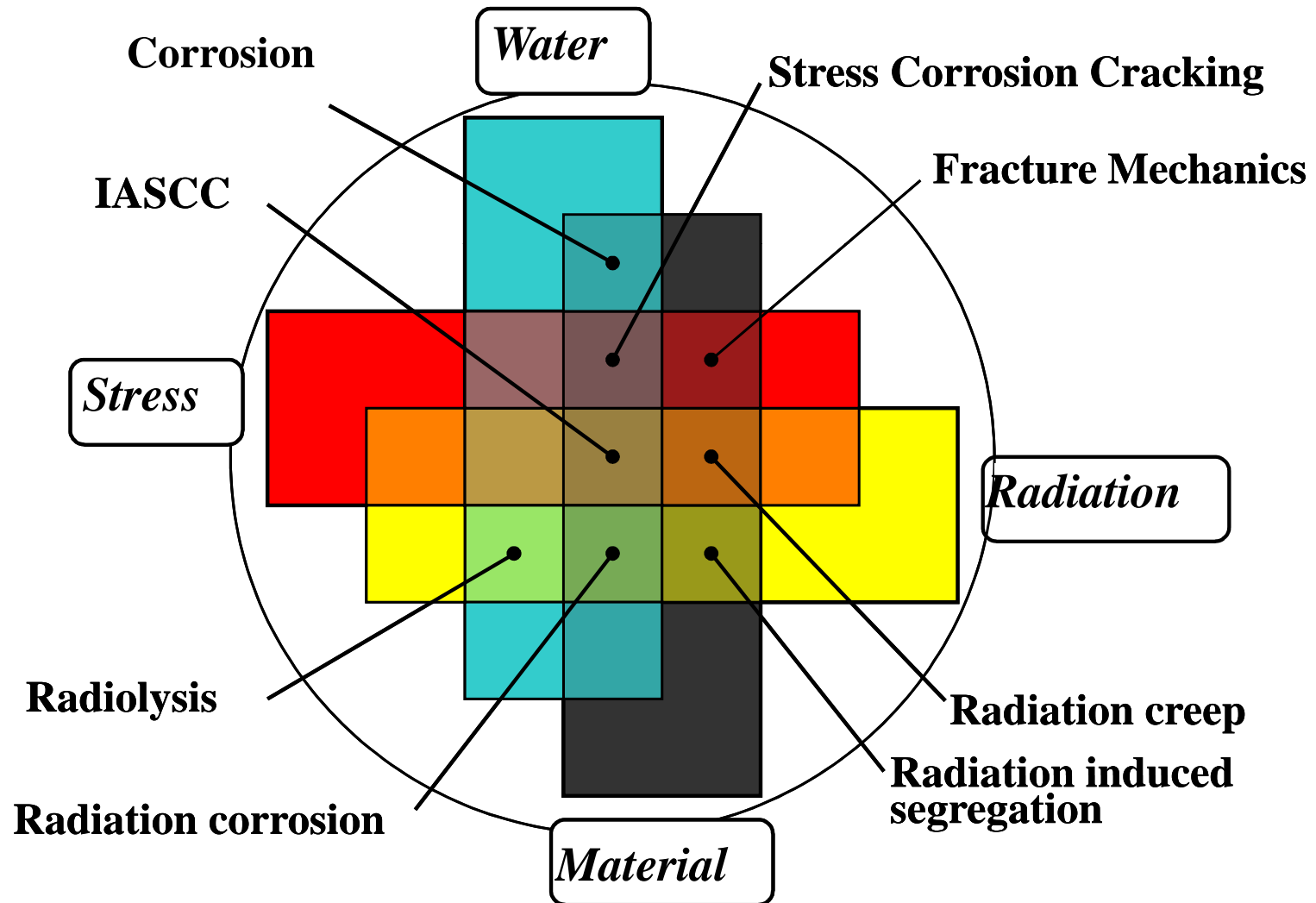
# Mitigation Technologies for SCC

## Water chemistry control or surface treatment

- Hydrogen water chemistry
- Noble Metal Chemical Application (NMCA)
- Deposition of noble metals by plasma spray
- Surface melting/Solution annealing



# + Radiation



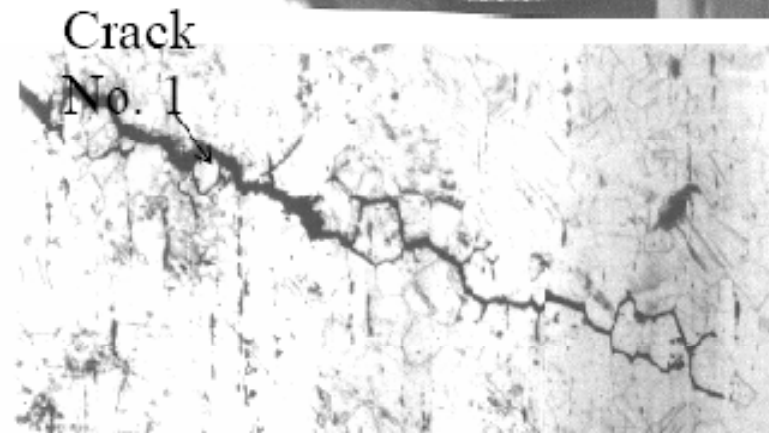
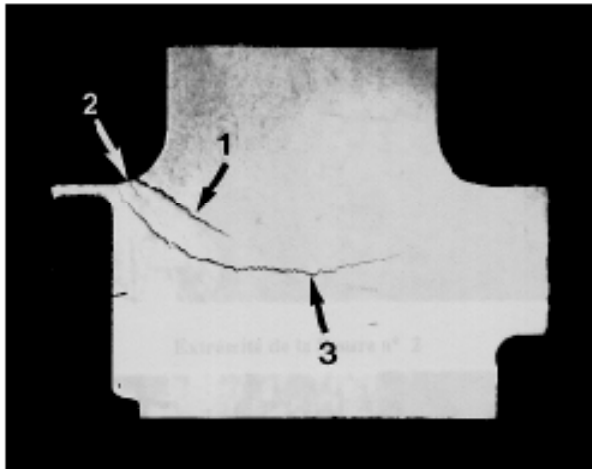
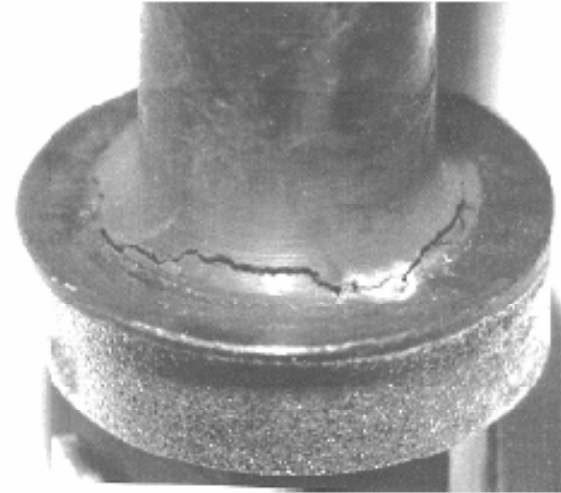
## Some Effects of Irradiation

- **Radiolysis of reactor coolant by n- and  $\gamma$ -irradiation**
  - change of oxidizing capacity of environment → corrosion, EAC
- **Change of microstructure of reactor internals and RPV by n-irradiation**
  - displacement damage
  - irradiation-induced hardening, embrittlement, segregation, creep, void swelling, ...
  - IASCC of reactor internals, n-embrittlement of RPVs
- **Transmutation and activation**
  - activation of corrosion products, activity transport and built-up in piping system
  - dry well dose rates (Co-60, which is strong  $\gamma$ -emitter, is created by thermal neutron activation of natural Cobalt 59 (from wear-resistant Co-alloys))
- **$\gamma$ -heating**

# IASCC Service Experience

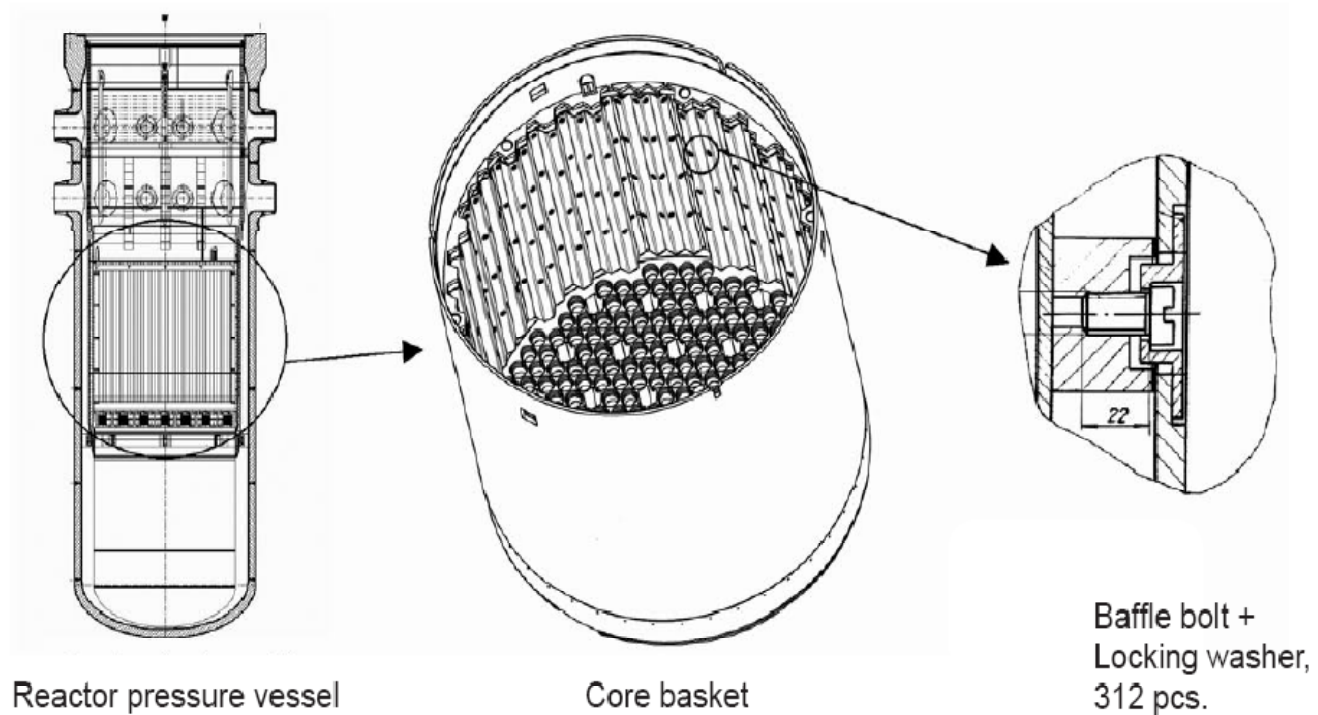
Component	Material	Reactor Type	Possible Sources of Stress
Fuel Cladding	304 SS	BWR	Fuel Swelling
Fuel Cladding	304 SS	PWR	Fuel Swelling
Fuel Cladding *	20%Cr/25%Ni/Nb	AGR	Fuel Swelling
Fuel Cladding Ferrules	20%Cr/25%Ni/Nb	SGHWR	Fabrication
Neutron Source Holders	304 SS	BWR	Welding & Be Swelling
Instrument Dry Tubes	304 SS	BWR	Fabrication
Control Rod Absorber Tubes	304/304L/316L SS	BWR	B <sub>4</sub> C swelling
Fuel Bundle Cap Screws	304 SS	BWR	Fabrication
Control Rod Follower Rivets	304 SS	BWR	Fabrication
Control Blade Handle	304 SS	BWR	Low stress
Control Blade Sheath	304 SS	BWR	Low stress
Control Blades	304 SS	PWR	Low stress
Plate Type Control Blade	304 SS	BWR	Low stress
Various Bolts **	A-286	PWR & BWR	Service
Steam Separator Dryer Bolts **	A-286	BWR	Service
Shroud Head Bolts **	600	BWR	Service
Various Bolts	X-750	BWR & PWR	Service
Guide Tube Support Pins	X-750	PWR	Service
Jet Pump Beams	X-750	BWR	Service
Various Springs	X-750	BWR & PWR	Service
Various Springs	718	PWR	Service
Baffle Former Bolts	316 SS Cold Work	PWR	Torque, differential swelling
Core Shroud	304/316/347 /L SS	BWR	Weld residual stress
Top Guide	304 SS	BWR	Low stress (bending)

## Examples of IASCC: Cracked Baffle Former Bolts in a PWR

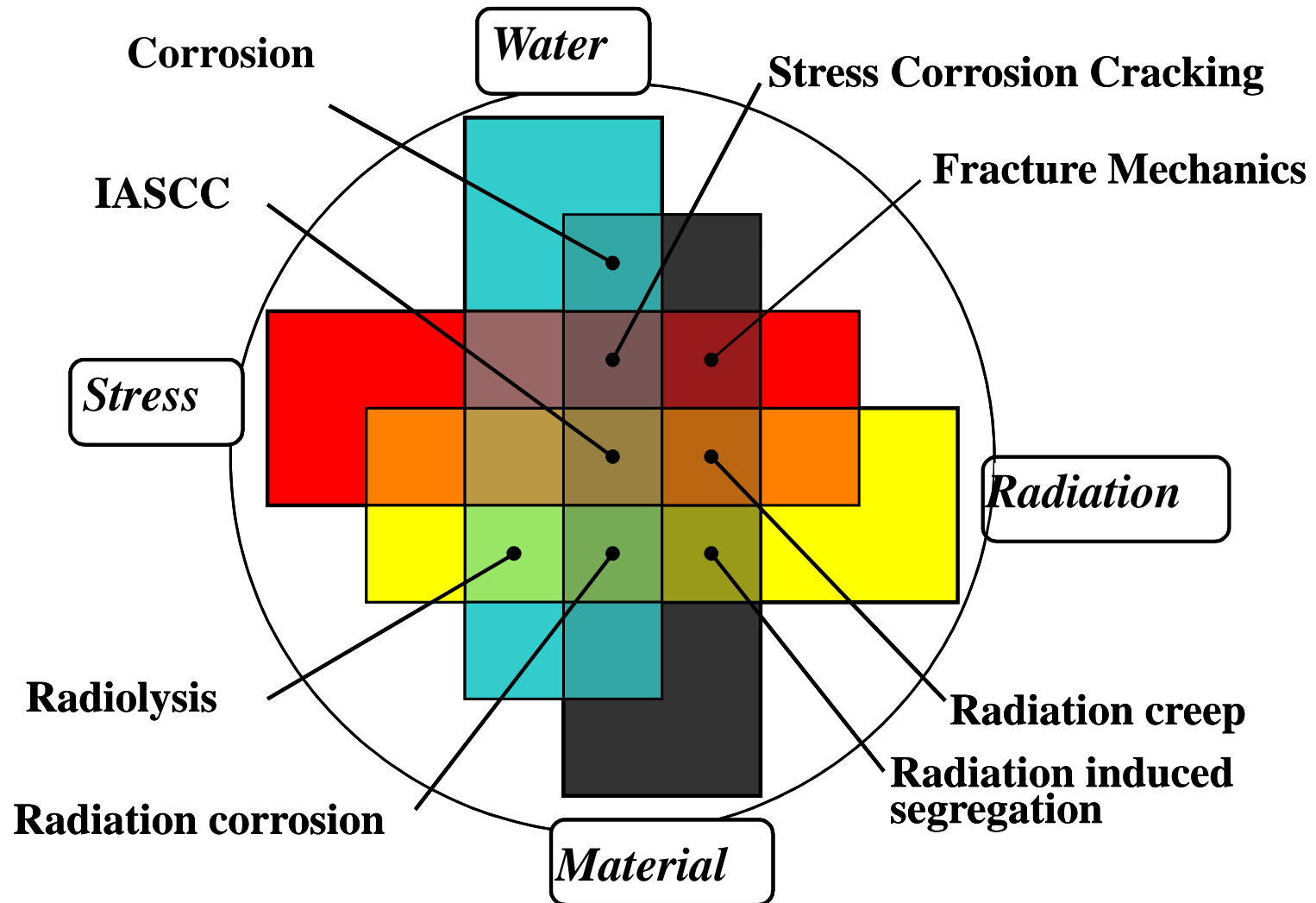


# *Loviisa core basket*

- Visual and ultrasonic inspection of all 312 locking bolts (Tecnatom)
- Removal of defective locking bolts (Westinghouse)
- Assembly of the new fixing system (Westinghouse)
- Internals TV-inspection of the core basket (Tecnatom)

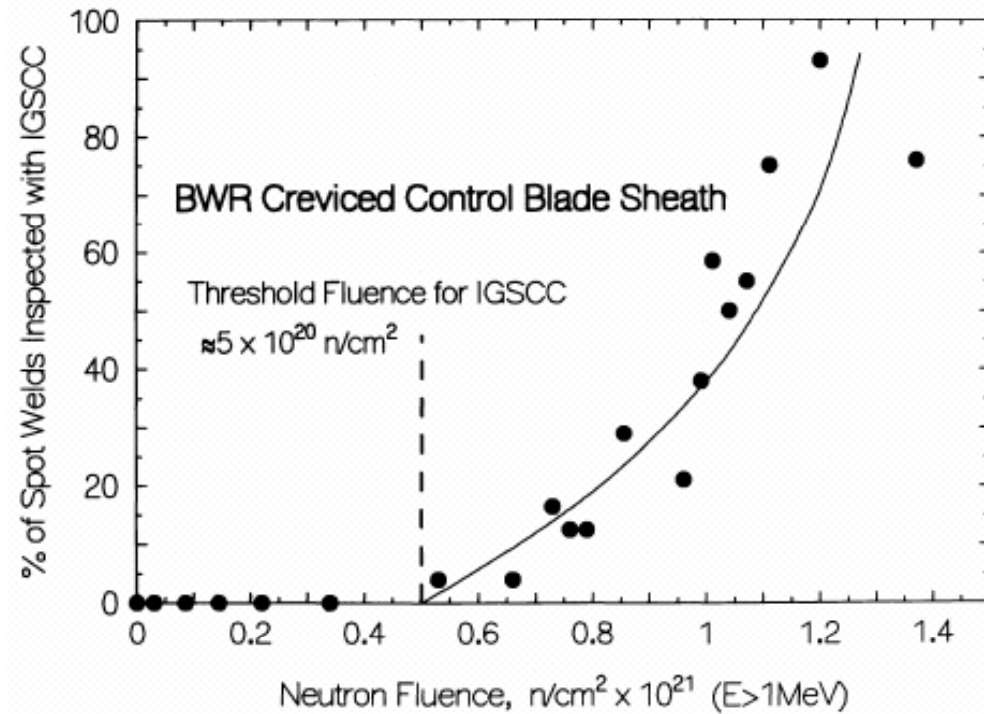
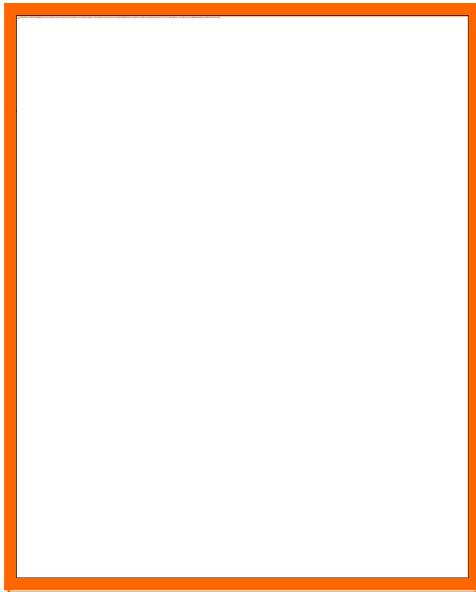


# + Radiation

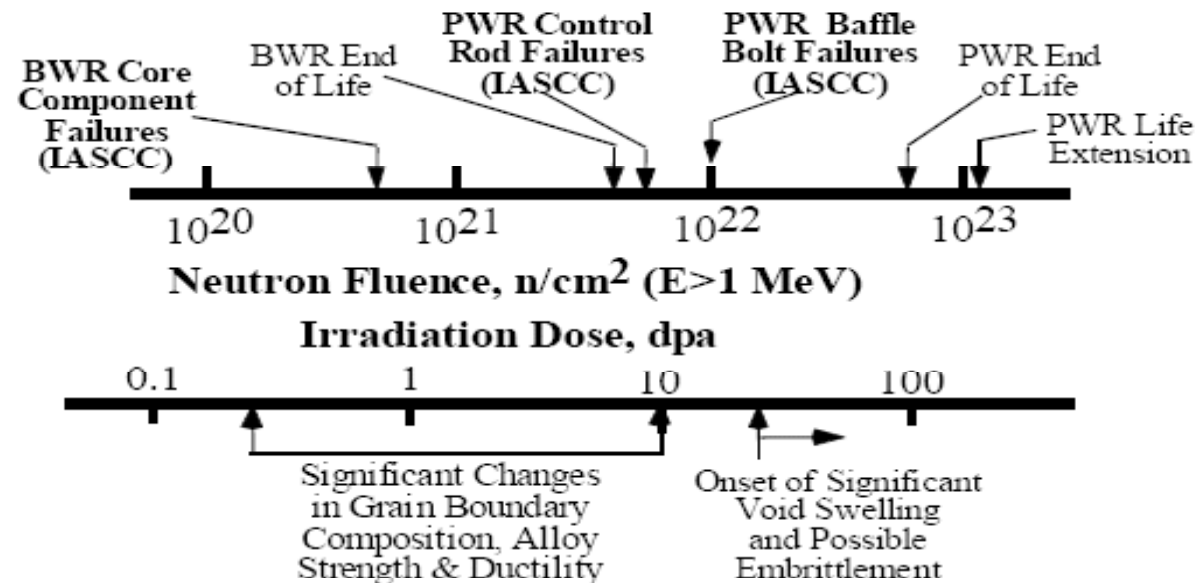
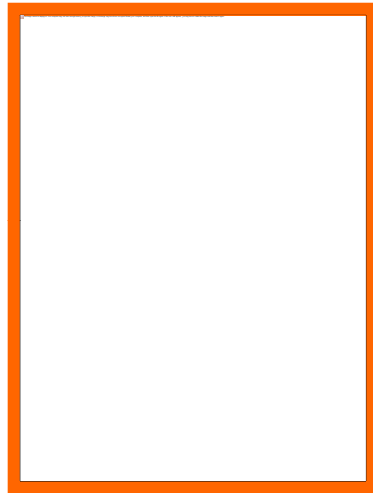




# Strong influence of radiation



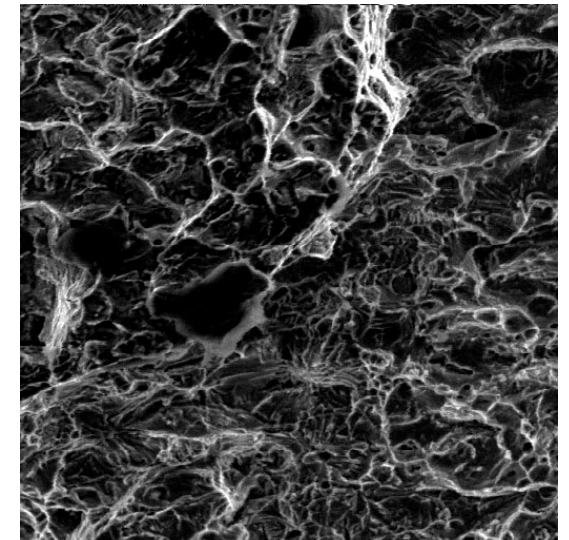
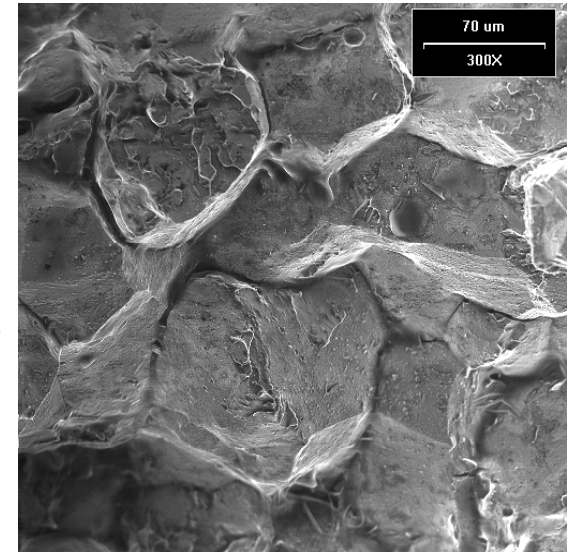
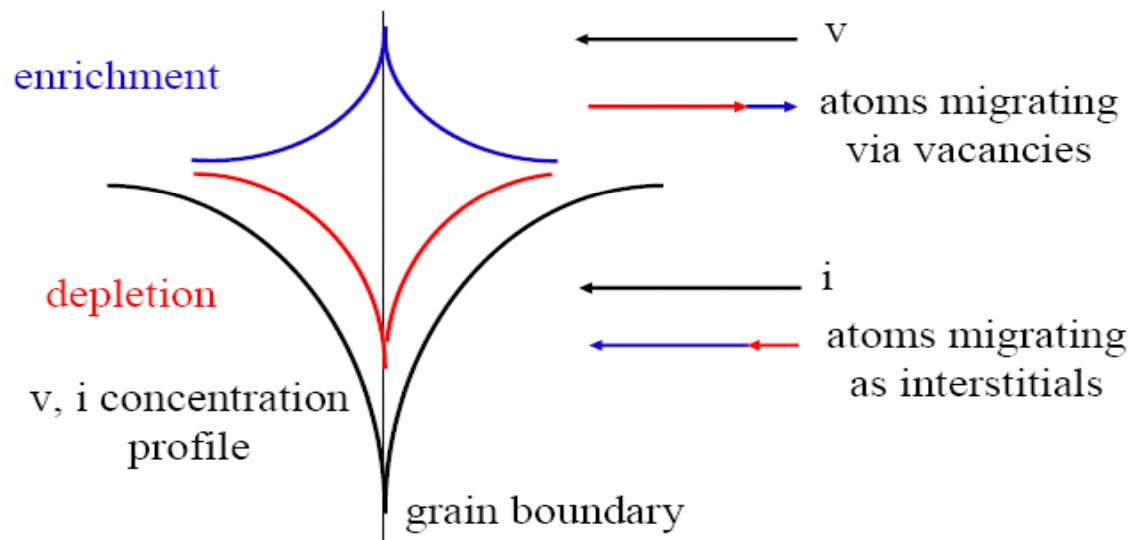
# Microstructure Changes can be Correlated to Irradiation Dose/Fluence



**Note:** ~15 dpa =  $10^{22}$  n/cm<sup>2</sup>  $E \geq 1$  MeV (for PWR and BWR neutron spectra)  
~7 dpa =  $10^{22}$  n/cm<sup>2</sup>  $E \geq 0.1$  MeV (for PWR and BWR neutron spectra)

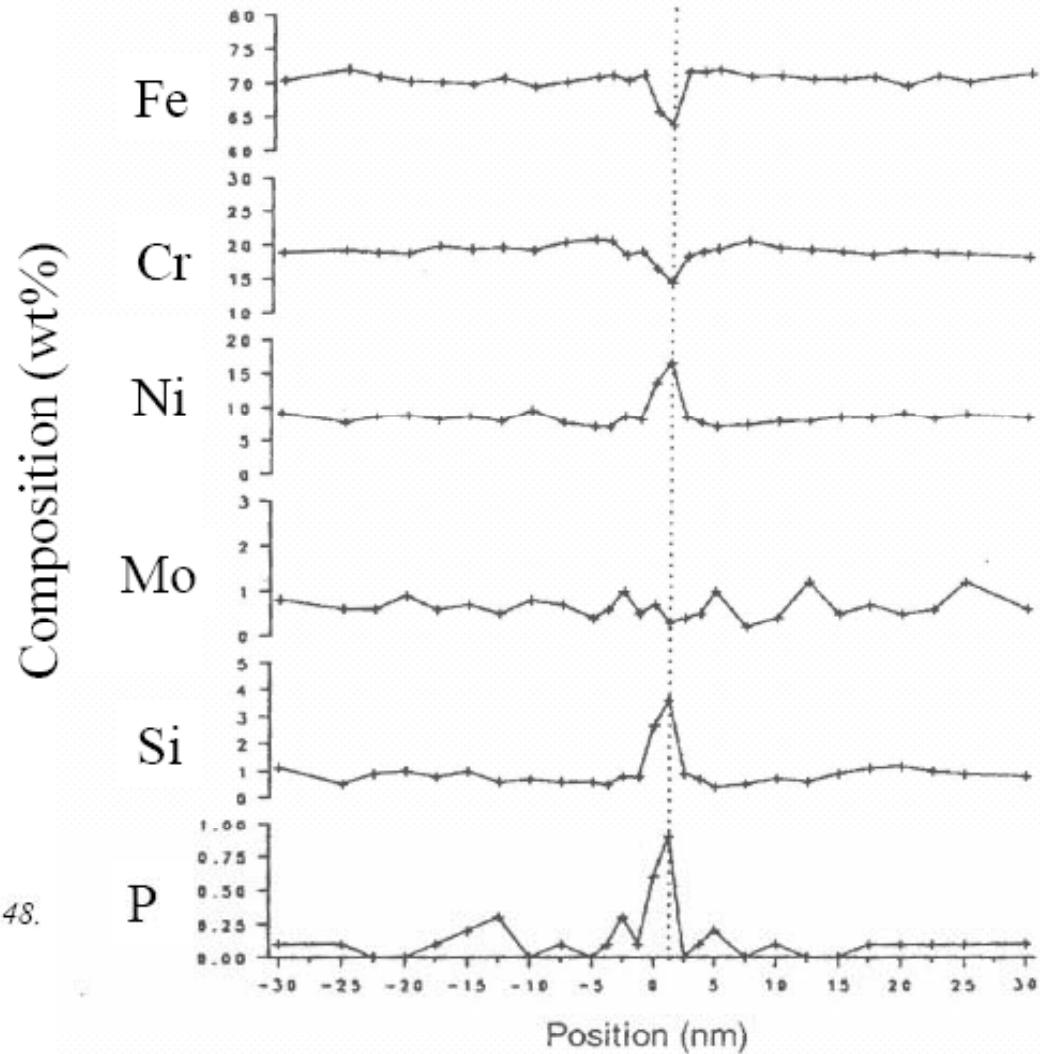
# Radiation Induced Segregation - RIS

Radiation Induced Segregation - RIS - occurs by preferential association between solute atoms and point defects



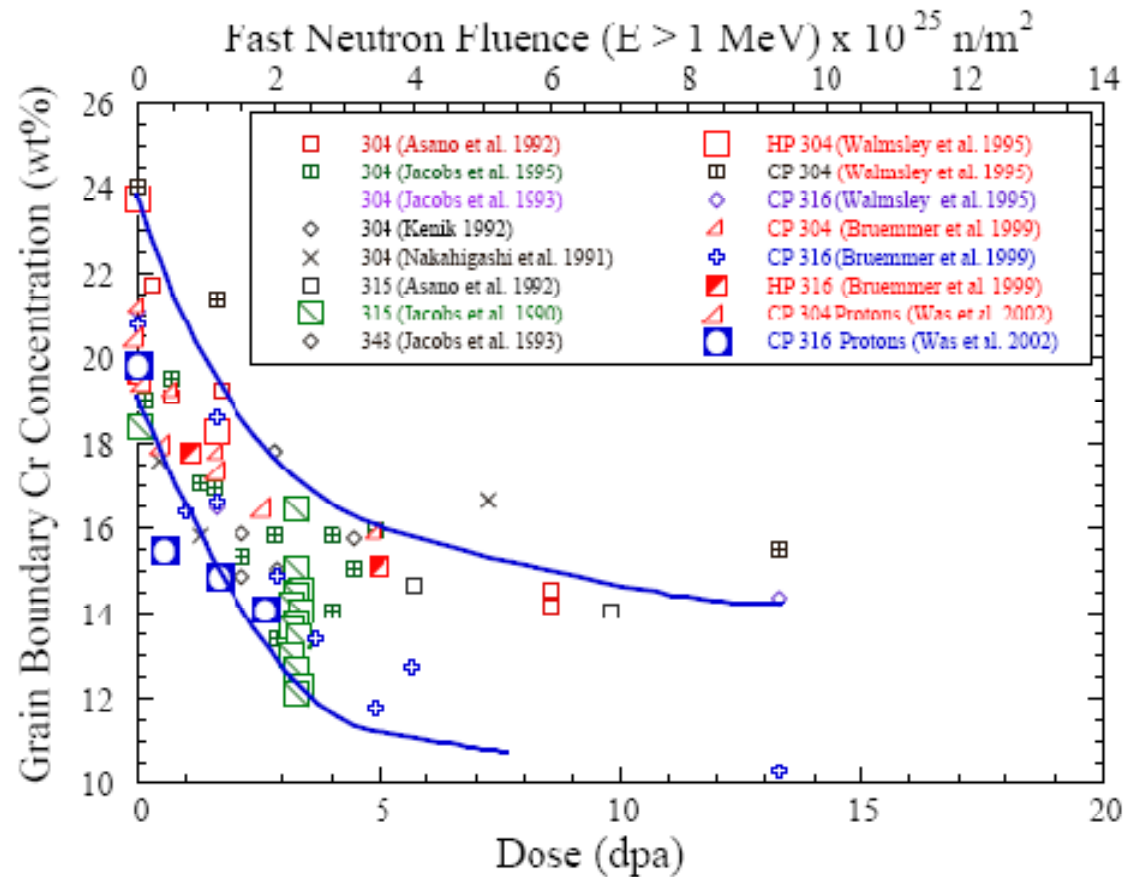
## Concentration Profiles for CP 304 SS after after neutron irradiation at 288°C to a fluence of $10^{22}$ n/cm<sup>2</sup> (E > 1 MeV)

- Cr, Mo deplete
- Ni, Si, P enrich
- Fe depends on alloy composition

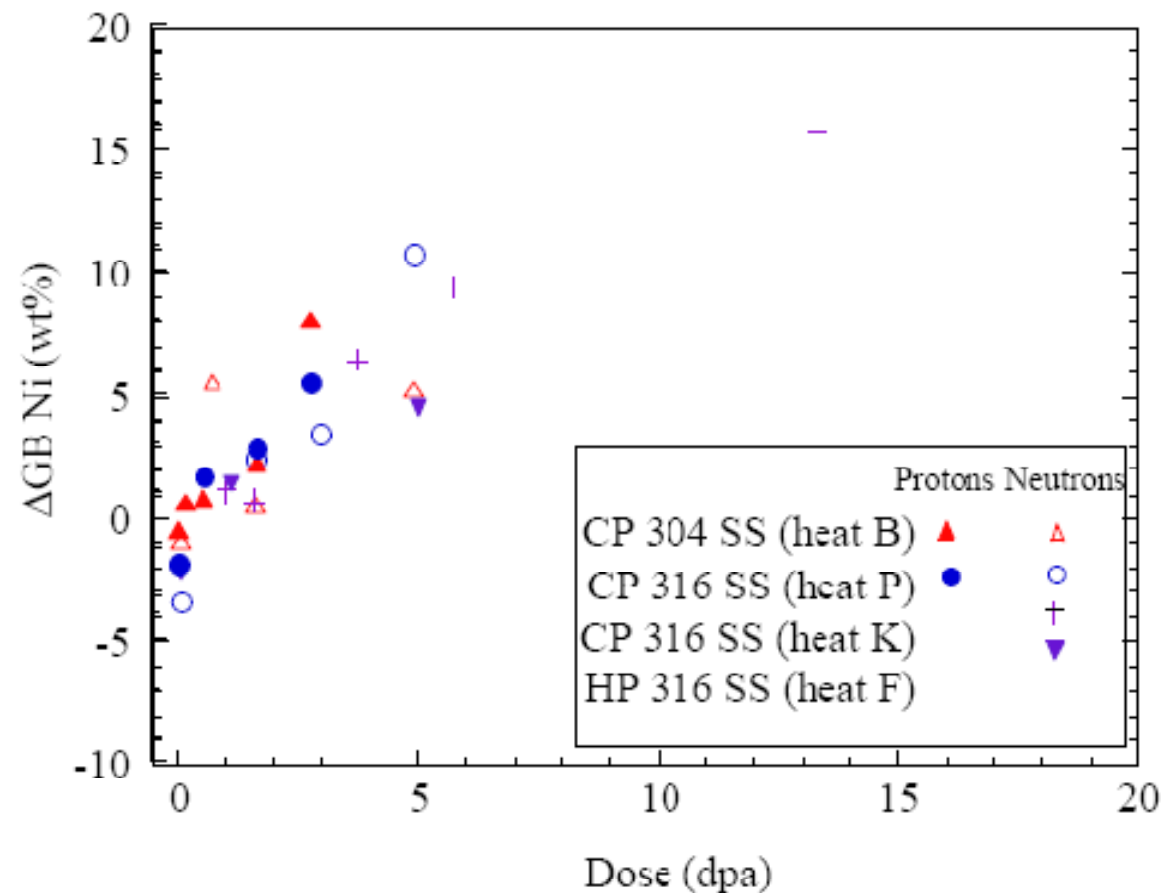


*A. Jenssen et al, Corrosion 54 1 (1998) 48.*

# Grain boundary chromium depletion in irradiated austenitic iron base alloys



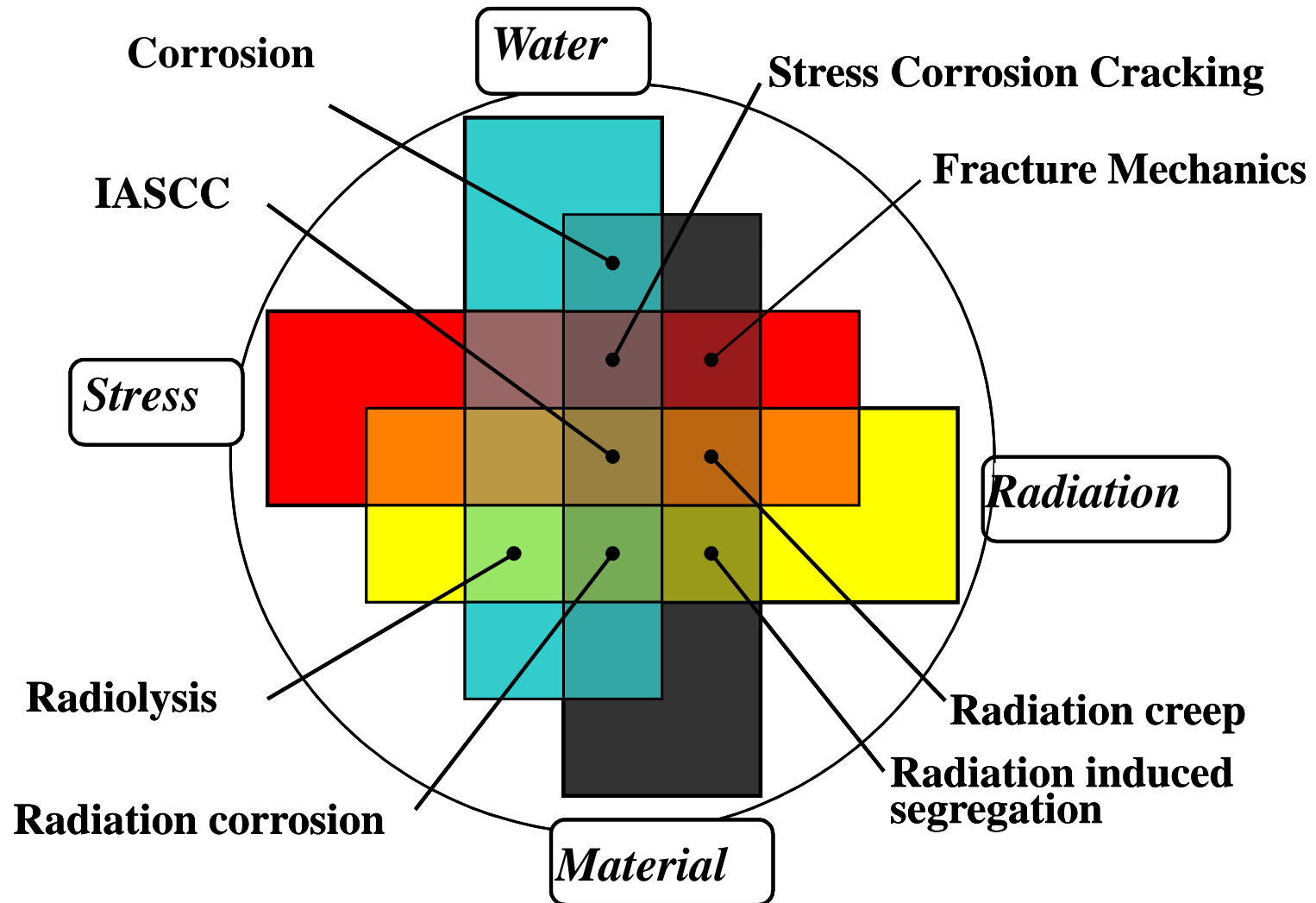
## Grain boundary nickel enrichment in irradiated austenitic iron base alloys



## Segregation of Minor Elements and Impurities

- Mo and Mn - strongly deplete with dose
- Si- very strong grain boundary segregation- as much as 10x. Segregation can lead to the formation of  $\gamma'$  ( $\text{Ni}_3\text{Si}$ ) in matrix or at grain boundary.
- P- thermally segregates and irradiation provides a small additional amount of segregation
- C, N, B- should also segregate, but there is little evidence due to difficulty in measurement

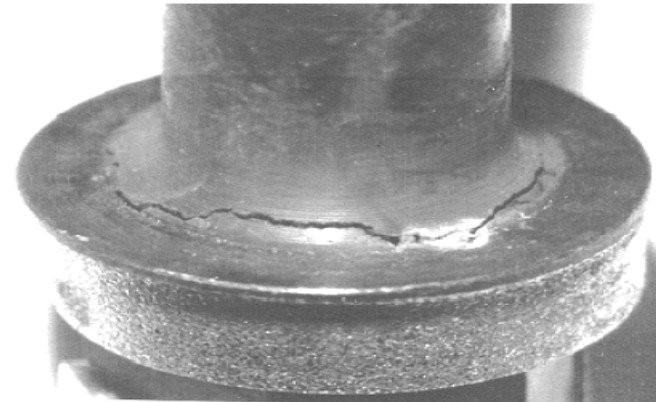
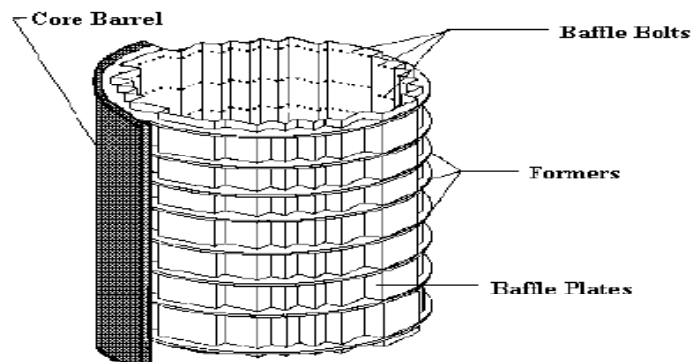
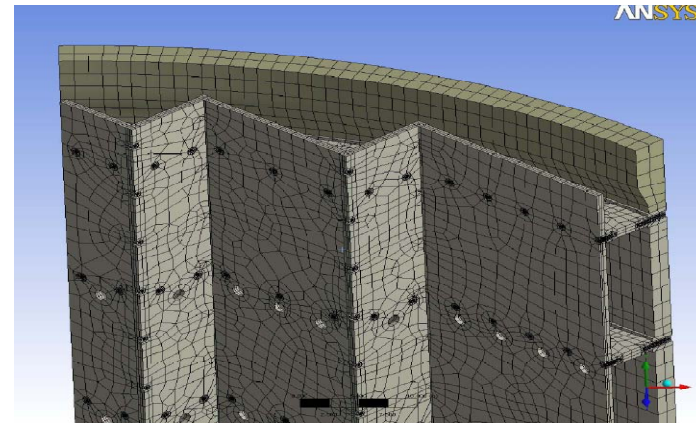
# + Radiation





# IASCC

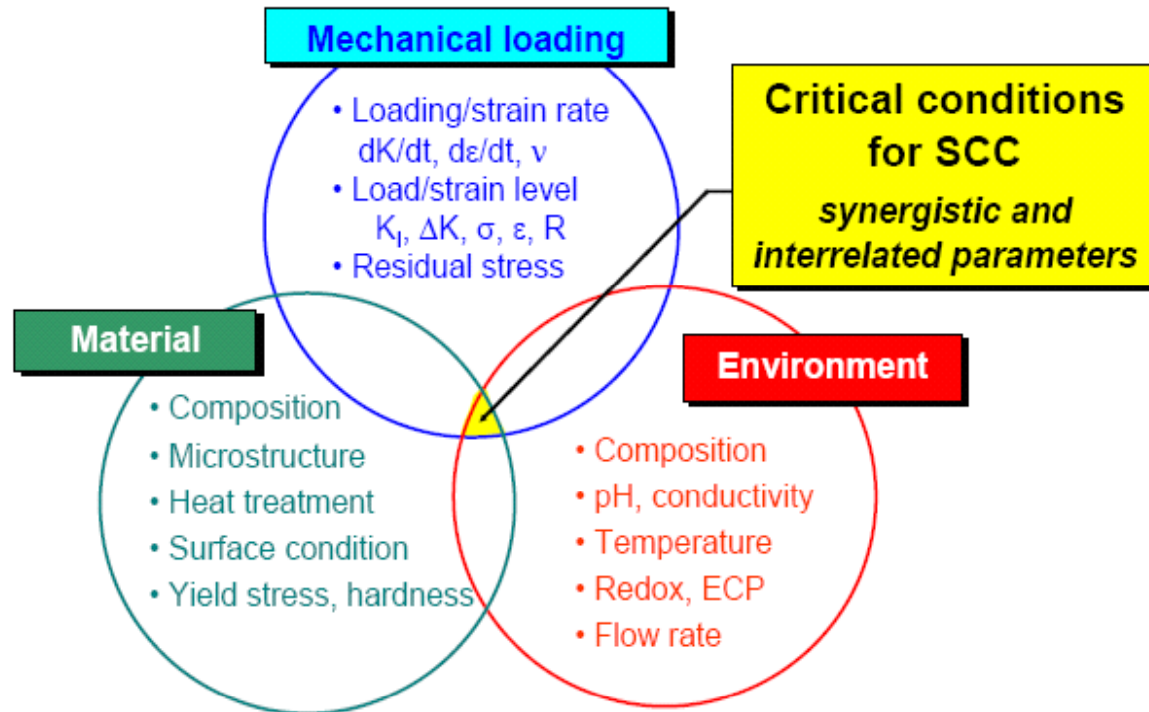
- Radiation
- Stress
- Time
- Temperature
- Environment



# IASCC – Irradiation Assisted Stress Corrosion Cracking

- Terminology used to describe cracking of materials
- Exposed to nuclear reactor coolant and ionizing radiation
- Like all Stress Corrosion Cracking phenomena it requires critical combinations of applied stress or strain, environmental chemistry & metallurgical structure to occur

## Major factors influencing Environmentally Assisted Cracking (EAC)



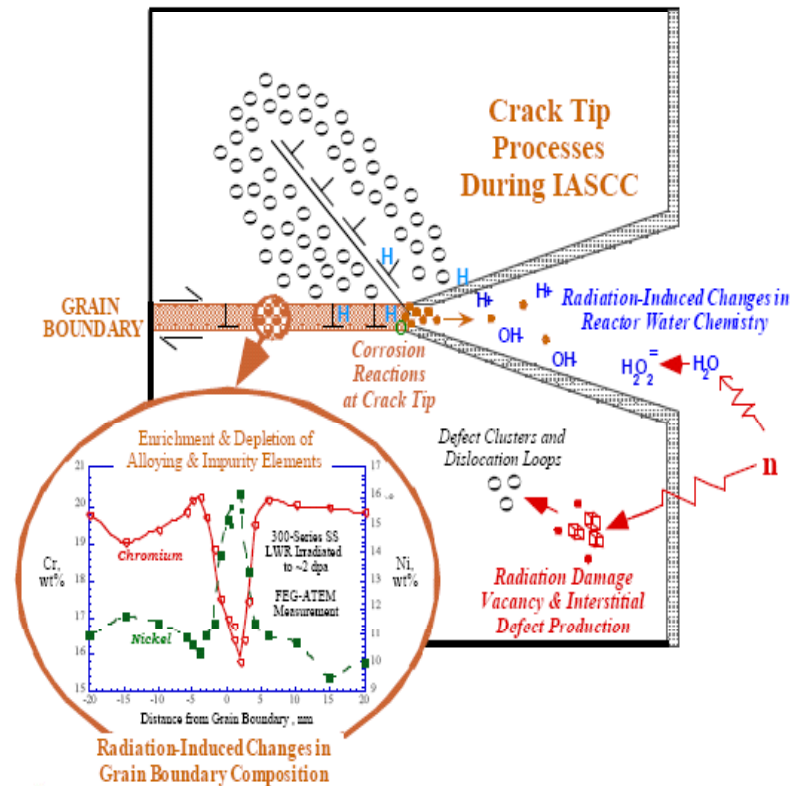
# Modelling

## IASCC added feature to EAC:

- atomic displacements
- neutron irradiation significantly alters metallurgical microstructure
- ionizing ( $\alpha, \beta$  and  $\gamma$ ) radiation modify the

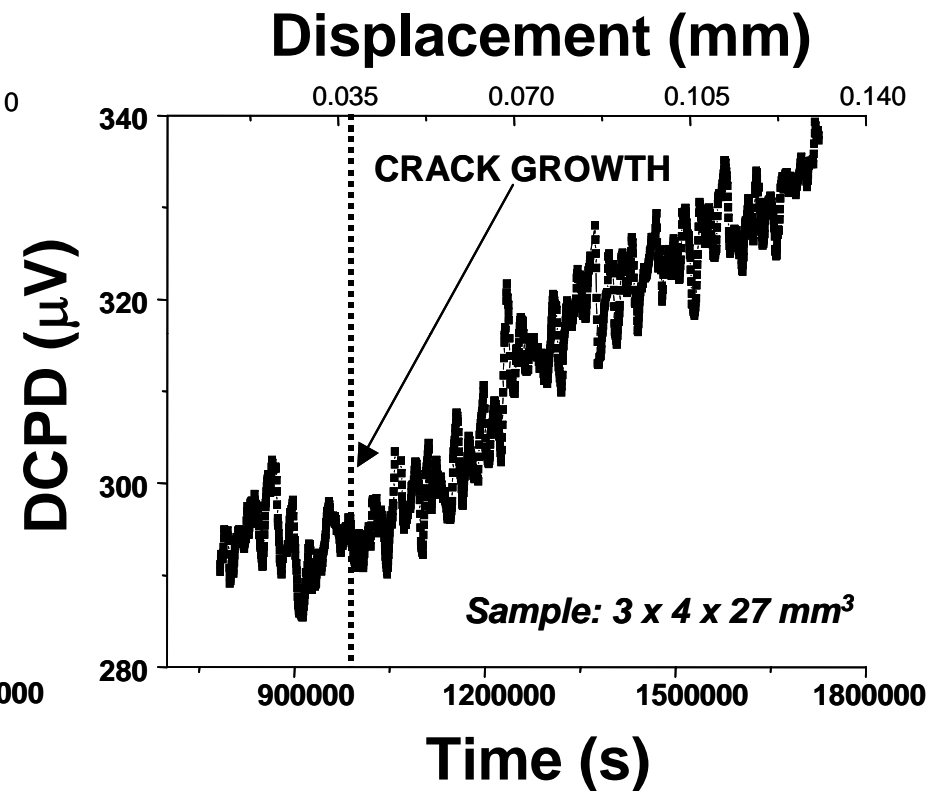
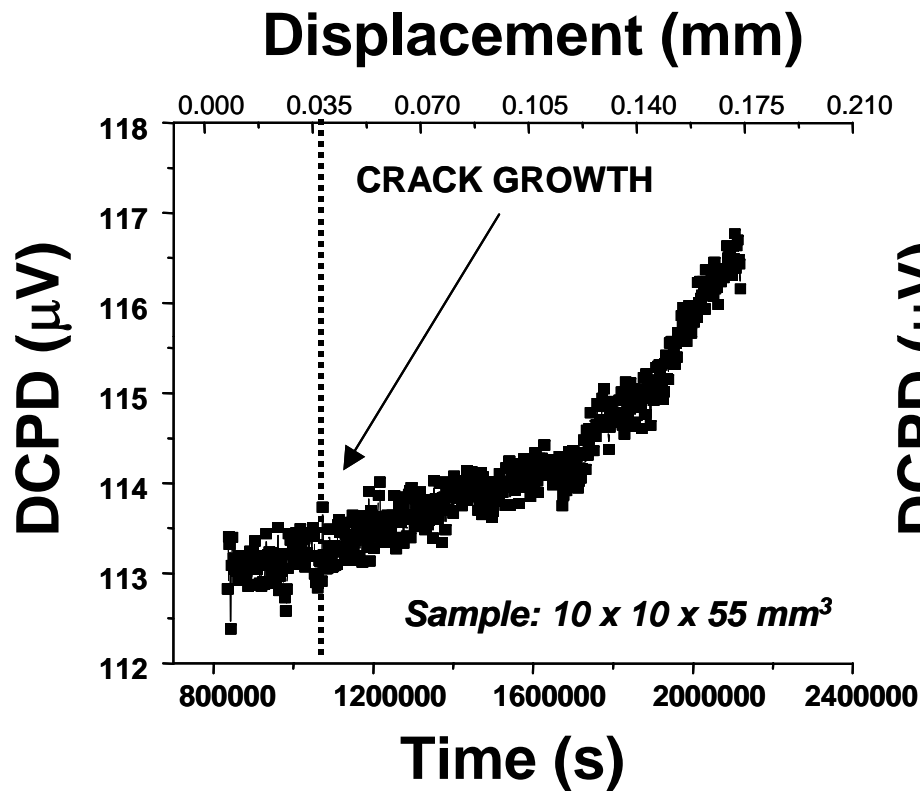
## Effects of irradiation on SCC:

- primary defects
- defects segregation
- dislocation interaction
- grain boundaries
- localized stress and strain
- environment
- stress relaxation by irradiation
- creep (beneficial factor for IASCC)





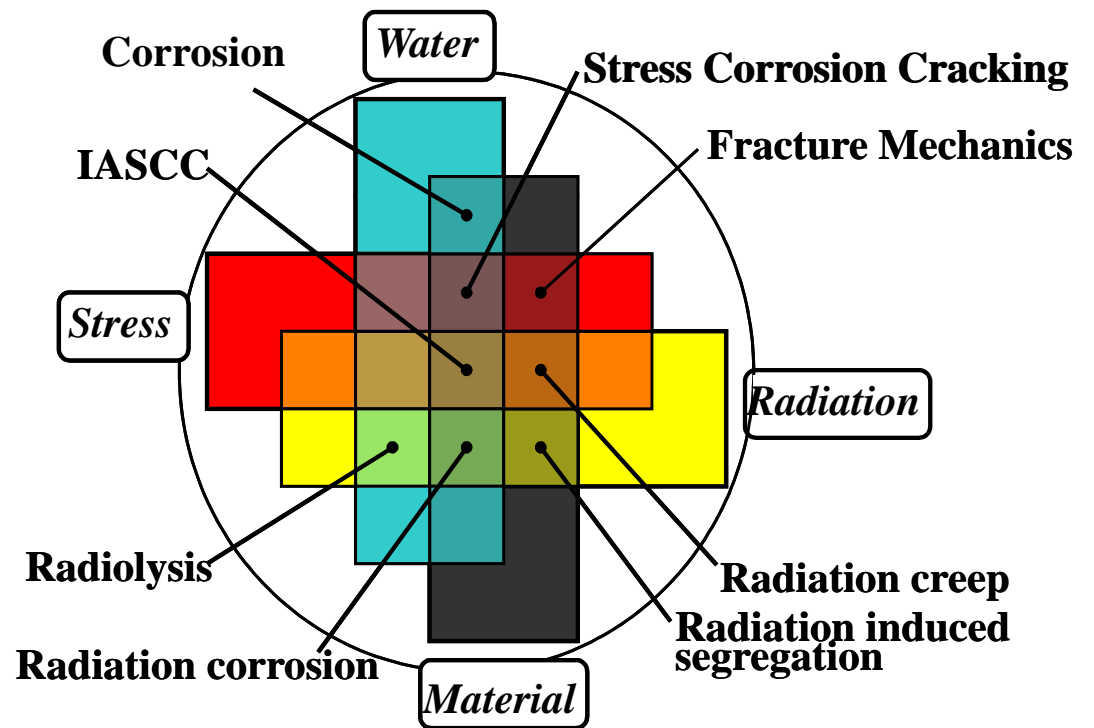
# SCC - RPV internals



Testing Environmentally Assisted Cracking (EAC) of Reactor Materials using Pneumatic Servo Controlled Fracture Mechanics (PSCFM), sub. International Journal of Pressure Vessel and Piping, 2006  
R. Novotny, F. Sevini, L. Debarberis, S. Petr, M. Kytka,

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