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

Radek Novotny & Luigi Debarberis
Institute for Energy (IE)
Petten, The Netherlands

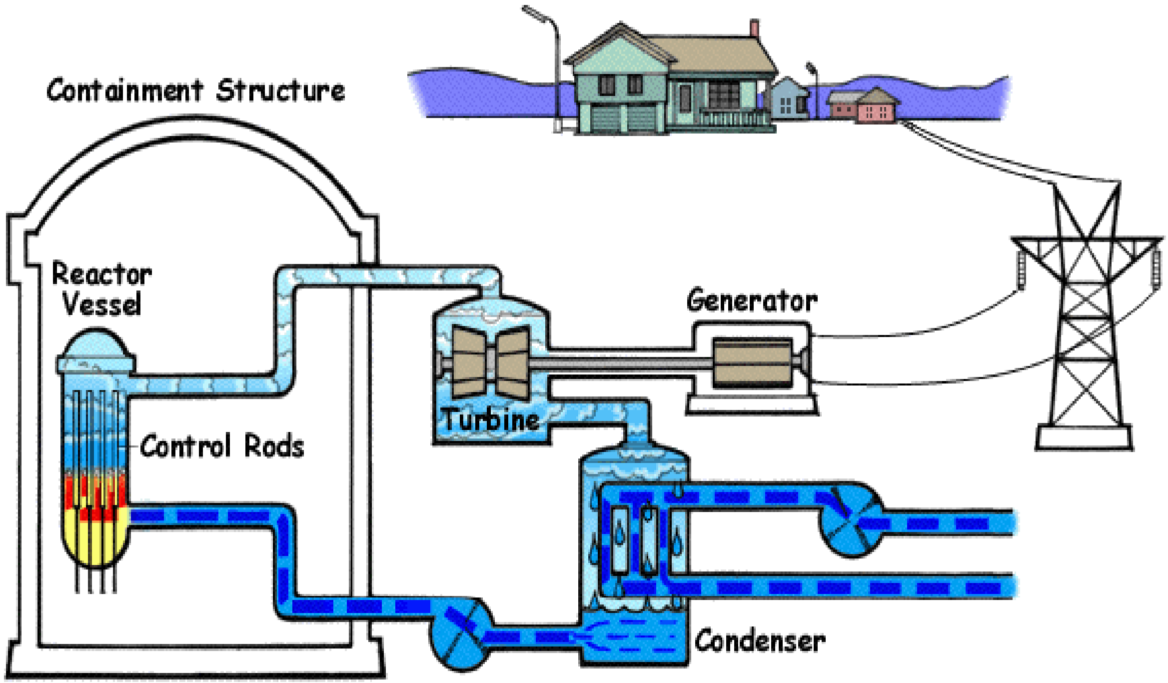
http://www.jrc.ec.europa.eu

Trieste, April 2009

CONTENT

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

Institute for Energy

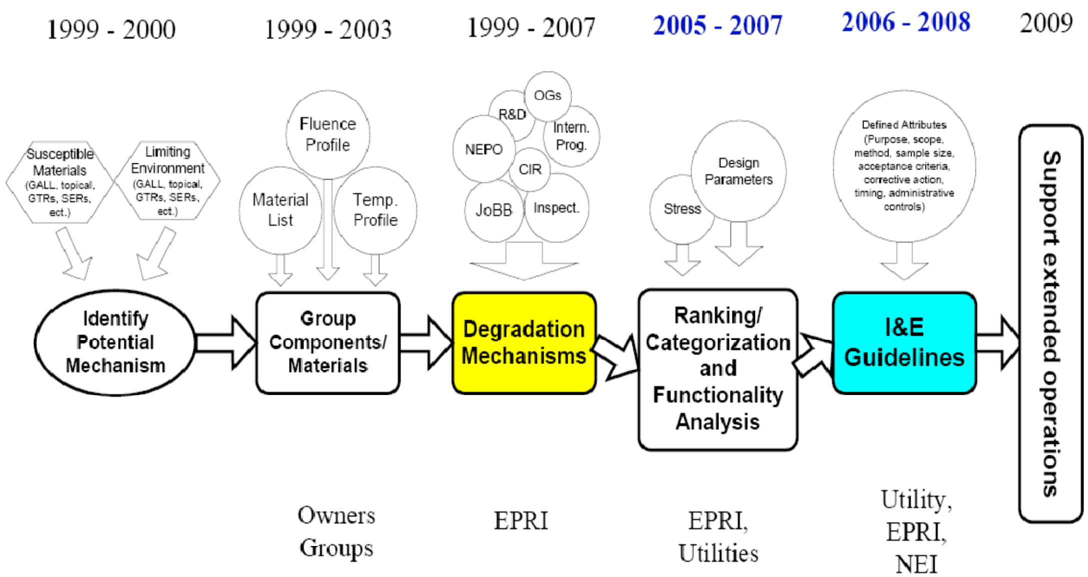


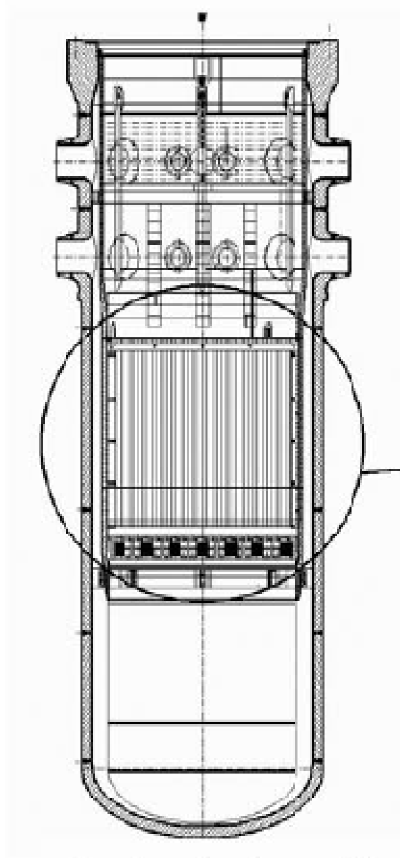
- Effort to plan life management of ageing NPPs
- Reliability of in-core in-vessel structural materials
- Control of the degradation of the structural materials
- Guiding replacement campaigns
- **Rising issues for alloys in LWR conditions:**
  - **Corrosion and**
  - **Stress Corrosion Cracking (SCC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

Institute for Energy

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





Institute for Energy

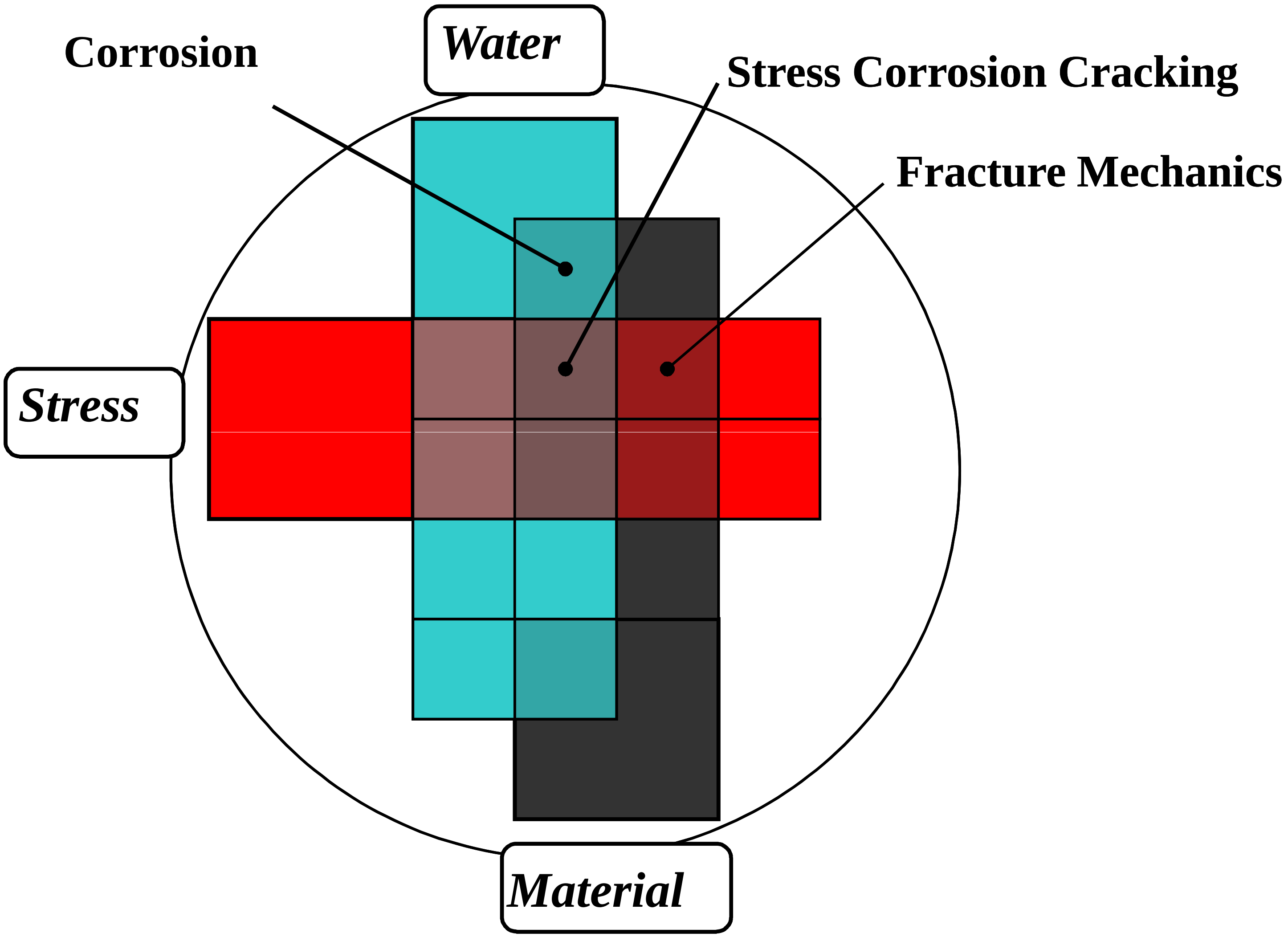
Essential service water system pipe degradation

Institute for Energy

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

Institute for Energy


Corrosion
Water
Stress Corrosion Cracking
Fracture Mechanics
Stress
Material

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		○	○	

Institute for Energy

Loviisa

PWSCC Experience in Vessel and Other Components

Plants with leaks:

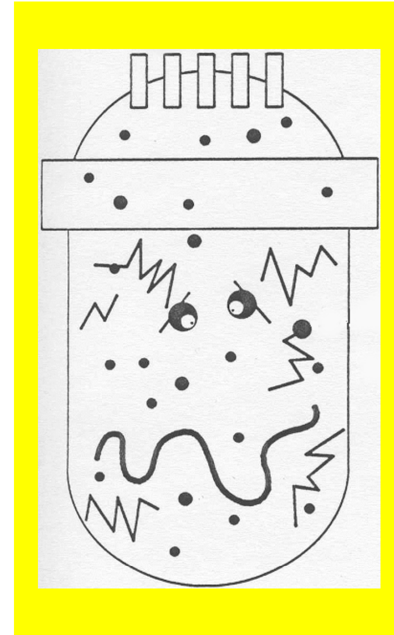
- ❑ V.C. Summer – axial, reactor vessel nozzle (2000)
- ❑ Tsurga 2 – axial, pressurizer valve nozzle (2003)

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).....

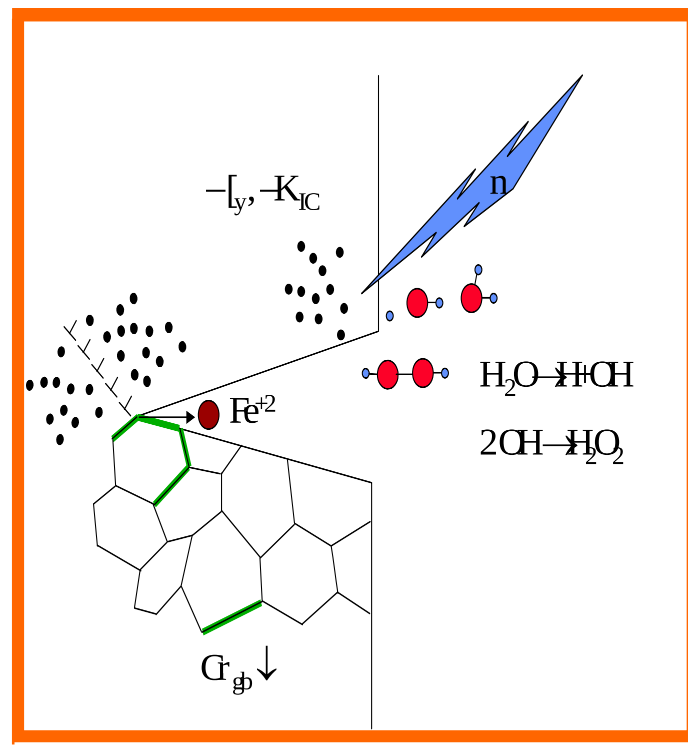
Institute for Energy

Corrosion & radiation

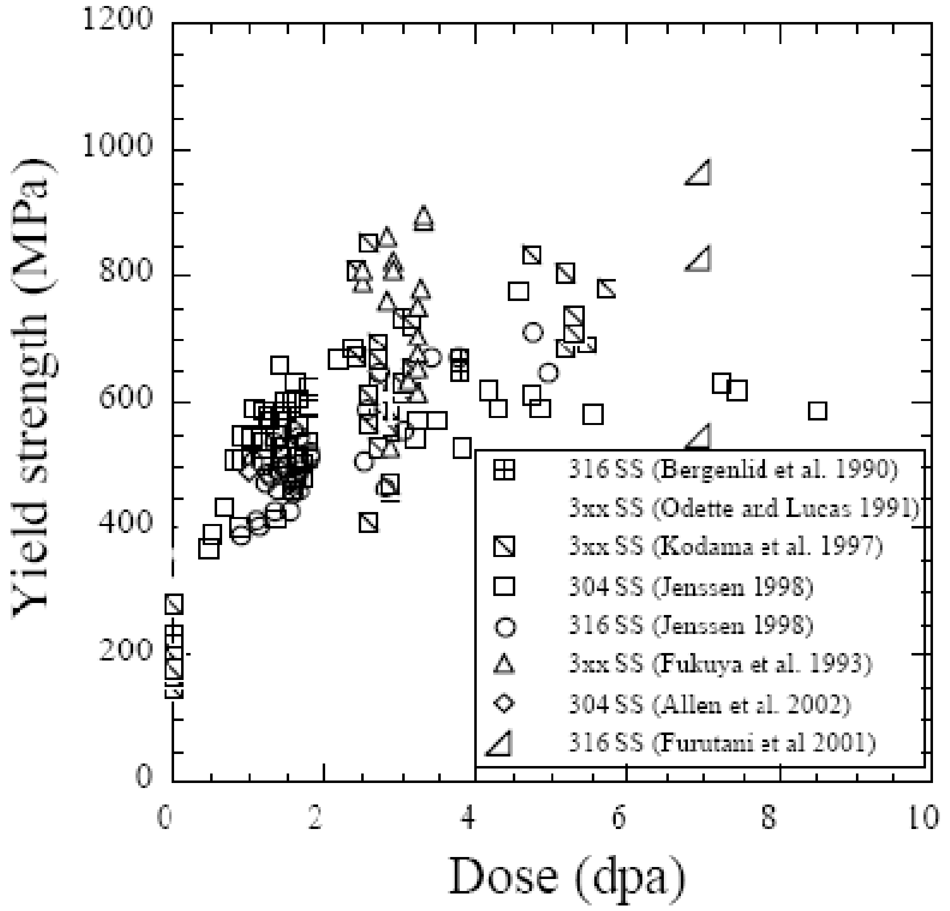


- **Effects of irradiation on materials are well investigated**
    - Defects due to irradiation
    - Mechanical properties evolution with dose

- **For LWRs, passive materials used**
- **Corrosion enhanced by water radiolysis**
    - oxidant species (OH^-, H_2O_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**



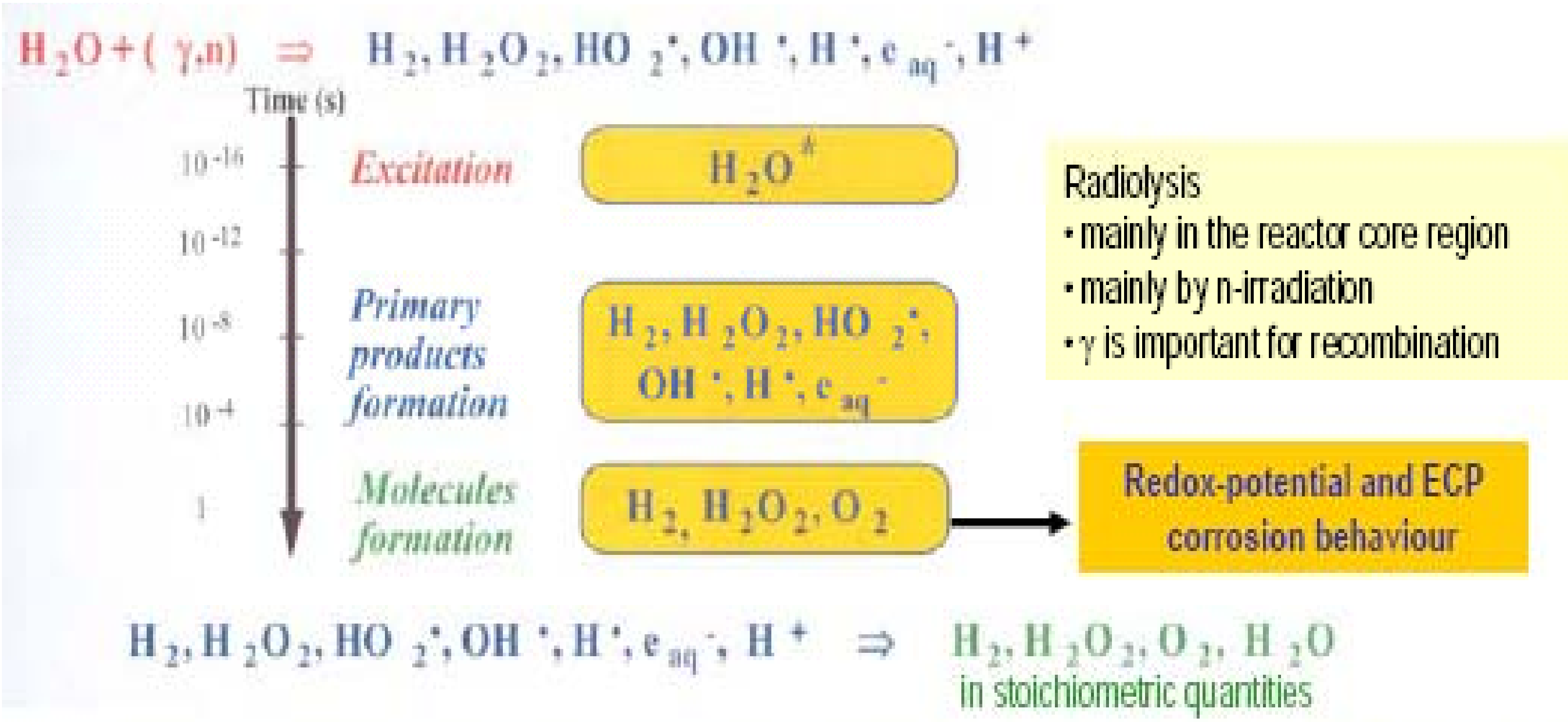
Irradiation hardening increases with $(dose)^{1/2}$



Michigan**Engineering**

Institute for Energy

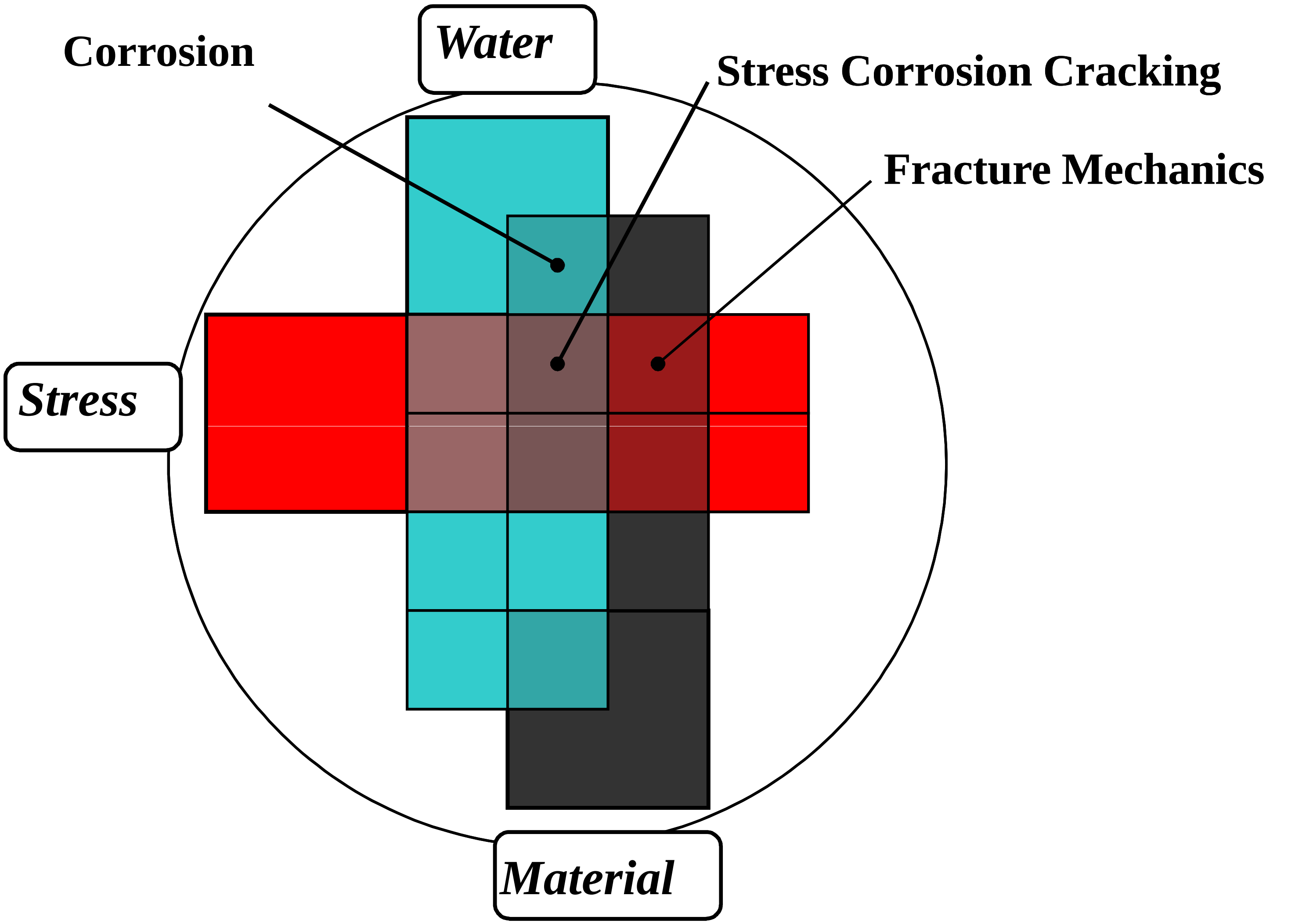
Radiolysis of Water by n- and γ Radiation



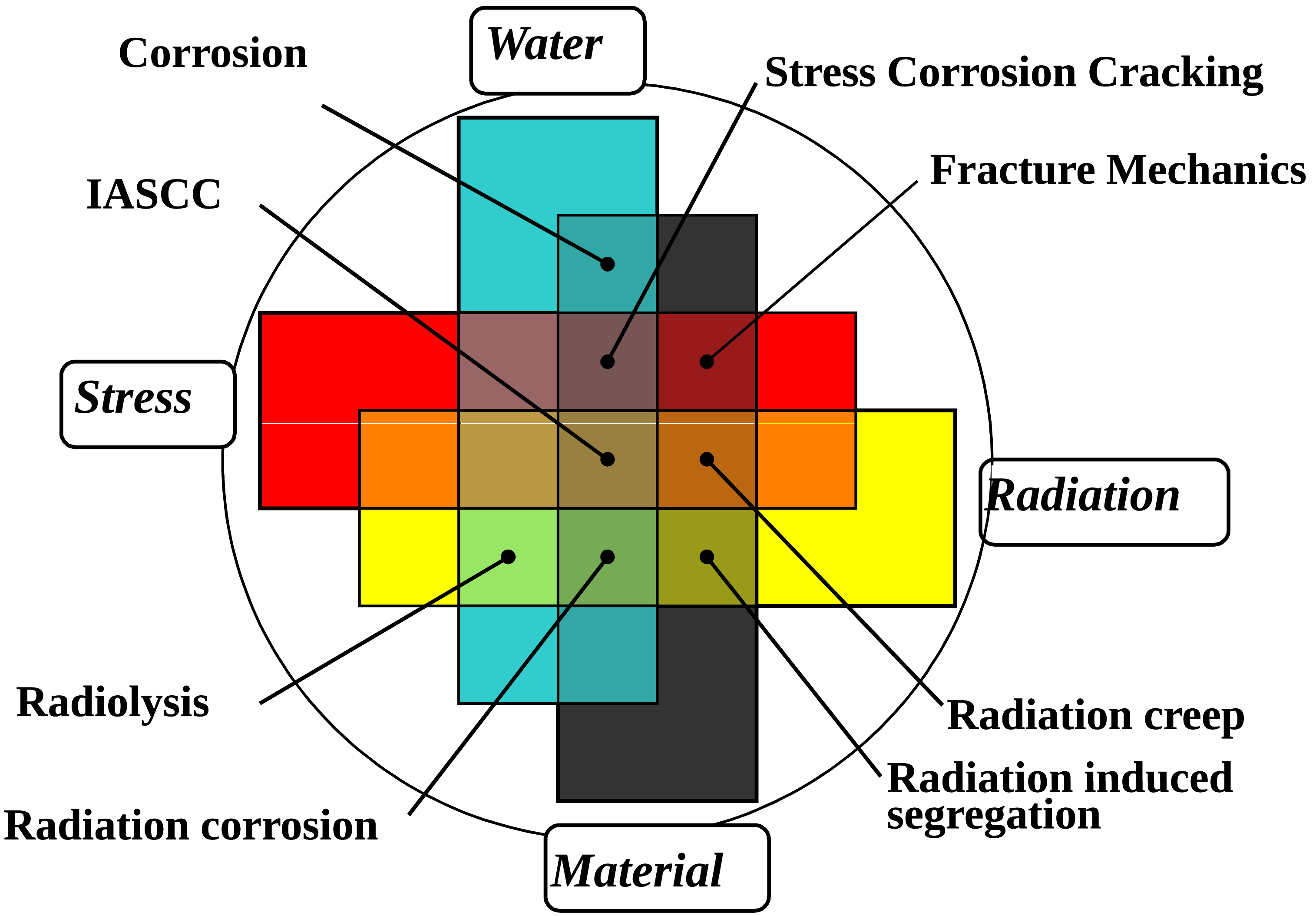
- **PWR:** H_2-addition (in VCT) $\rightarrow$ recombination with $O_2 + H_2O_2$ to $H_2O \rightarrow H_2 >> O_2 + H_2O_2 < 5$ ppb $\rightarrow$ low redox-potential $\rightarrow$ ECP = -800 to -500 mV_{SHE}
- **BWR/NWC:** H_2O_2 is non-volatile $\rightarrow O_2 + ½ H_2O_2 > H_2 \rightarrow$ oxidizing $\rightarrow$ ECP = -50 to +200 mV_{SHE}
- **BWR/HWC or NMCA:** H_2-addition to feedwater $\rightarrow$ recombination with $O_2 + H_2O_2$ to $H_2O \rightarrow$ $H_2 >> O_2 + H_2O_2 < 5$ ppb $\rightarrow$ low redox-potential $\rightarrow$ ECP = -500 to -200 mV_{SHE}

Institute for Energy

- 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

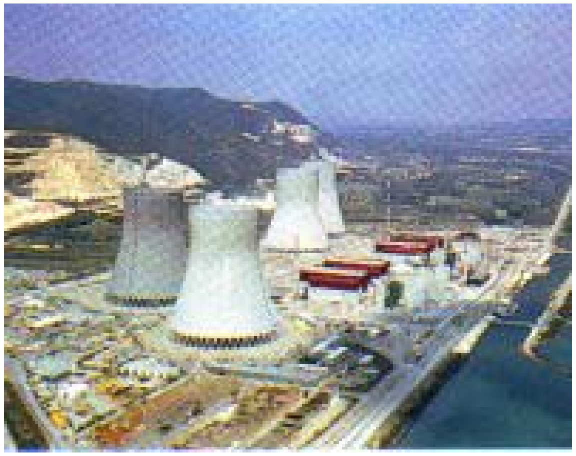

Corrosion
Water
Stress Corrosion Cracking
IASCC
Fracture Mechanics
Stress
Radiation
Radiolysis
Radiation creep
Radiation induced segregation
Radiation corrosion
Material

Present generation

LWR → BWR, PWR, VVER
HW → CANDU
GCR → MAGNOX, AGR
LM → FR

Institute for Energy

BWR - PWR



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

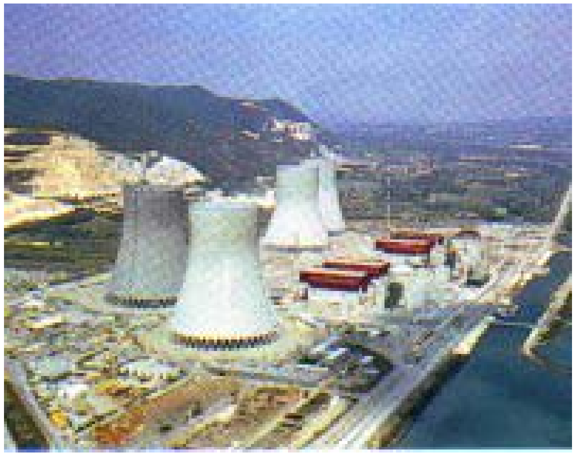
Institute for Energy

Water Chemistry Conditions in BWR and PWR

BWR/NWC	BWR/HWC	PWR
• T = 274 – 288 °C	• T = 274 – 288 °C	• T = 286 – 323 °C
• pH_T = 5.65	• pH_T = 5.65	• pH_T = 6.8 to 7.4
• Neutral	• Neutral	• Slightly alkalic
• High-purity water	• High-purity water	• Pure water
• Impurities < 1ppb	• Impurities < 1ppb	• H_3BO_3/LiOH
• O_2 ~ 200 ppb	• O_2 < 5 ppb	• O_2 < 5 ppb
• H_2 ~ 15 ppb	• H_2 ~ 50 to 300 ppb	• H_2 ~ 2000 to 5000 ppb
• ECP = -50 to +200 mV_{SHE}	• ECP = -500 to -200 mV_{SHE}	• ECP = -800 to -600 mV_{SHE}

Institute for Energy

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!**

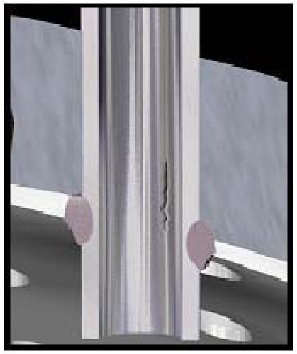
Boric Acid Corrosion

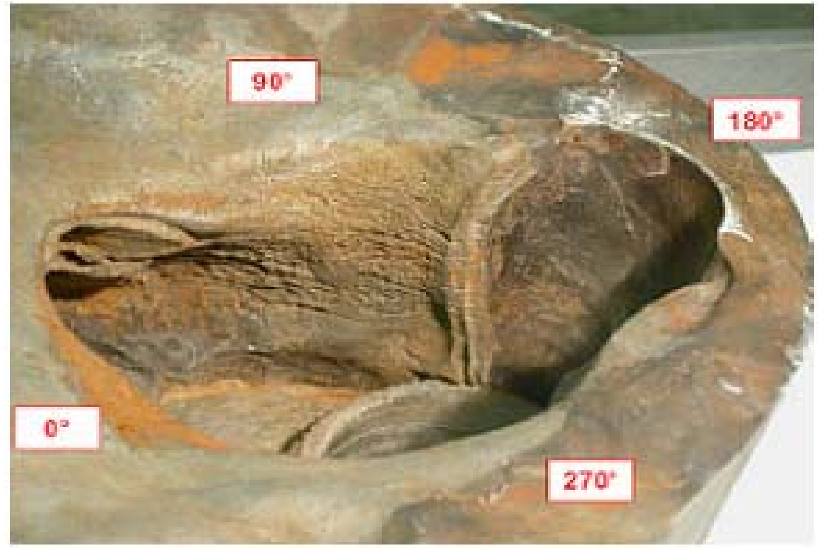
- Boric acid leakage is a consequence of Alloy 600 cracking
- This leakage can lead to boric acid corrosion of low-alloy steel
- Davis-Besse, March 2002
- Order EA-03-009.

Inspection requirements according to the parameter EDY

Davis-Besse

Davis-Besse

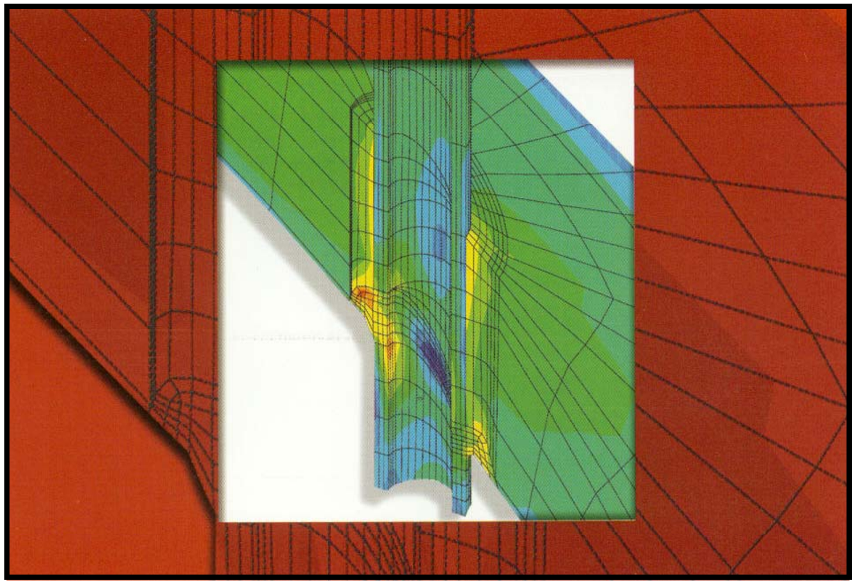




Institute for Energy

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

Institute for Energy

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.

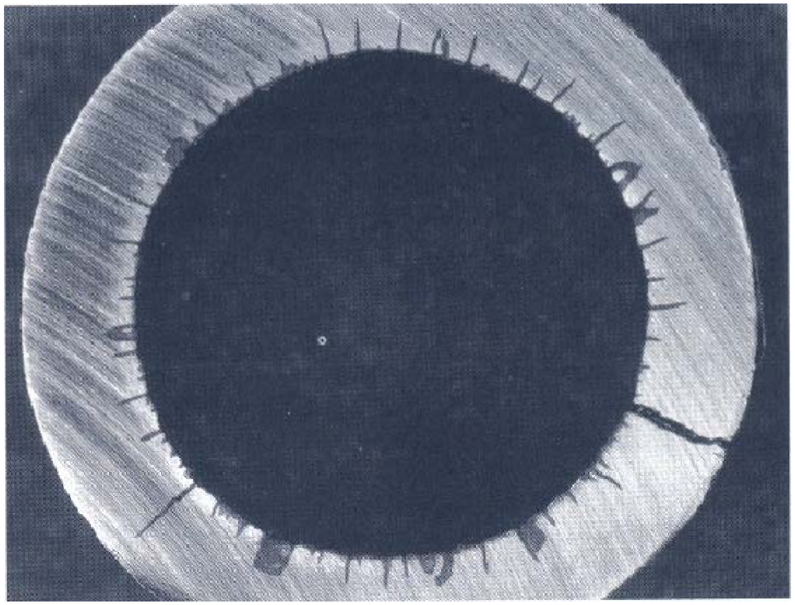
Degradation Incidents of RPVIs Safety Relevant

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

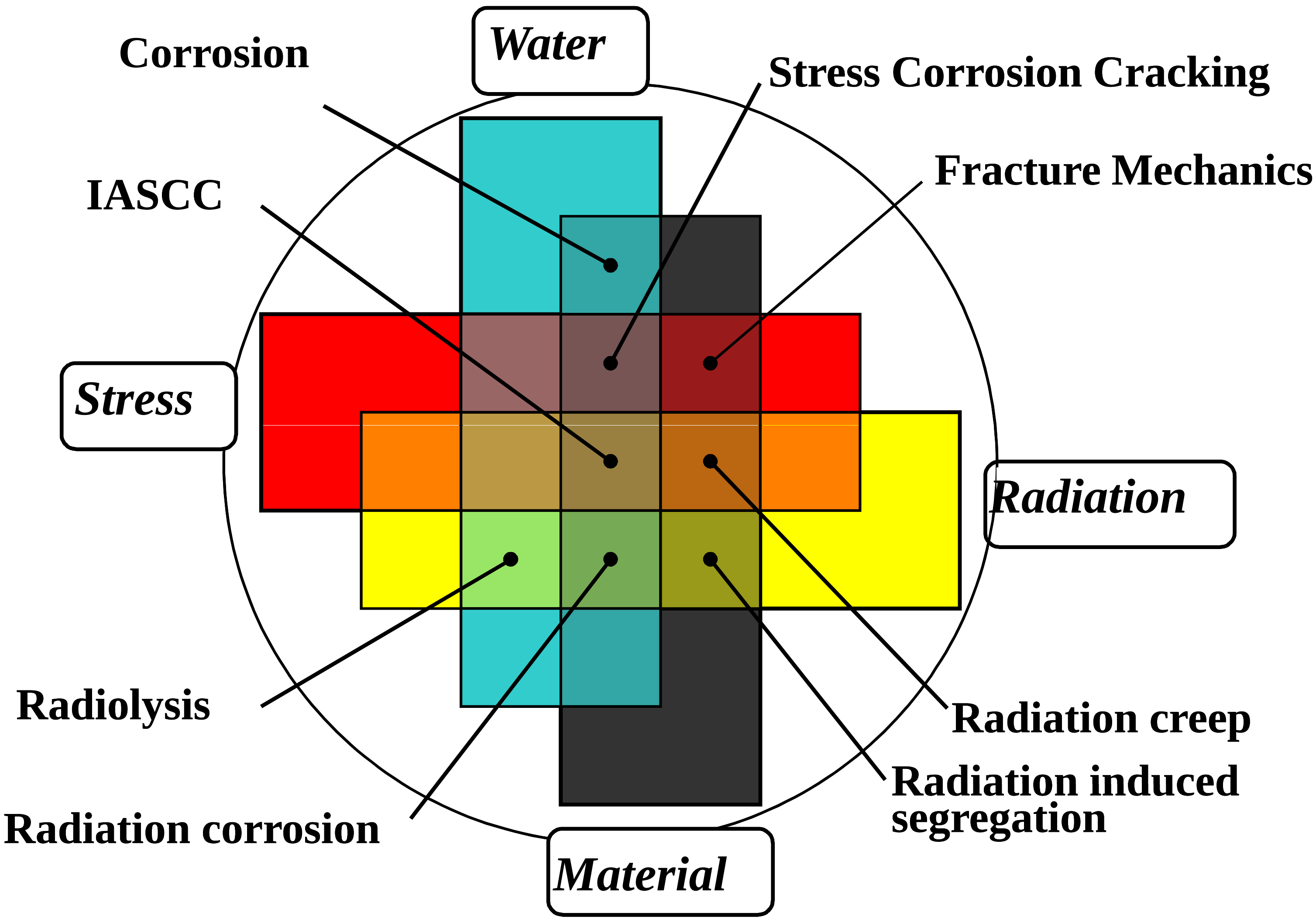
Mitigation Technologies for SCC

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



Water
Corrosion
Stress Corrosion Cracking
Fracture Mechanics
IASCC
Stress
Radiation
Radiolysis
Radiation creep
Radiation induced segregation
Radiation corrosion
Material

Some Effects of Irradiation

- **Radiolysis of reactor coolant by n- and γ-irradiation**

  → change of oxidizing capacity of environment → corrosion, EAC

- **Change of microstructure of reactor internals and RPV by n-irradiation**

  → displacement damage

  → irrad.-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 γ-emitter, is created by thermal neutron activation of natural Cobalt 59 (from wear-resistant Co-alloys))

- **γ-heating**

Institute for Energy

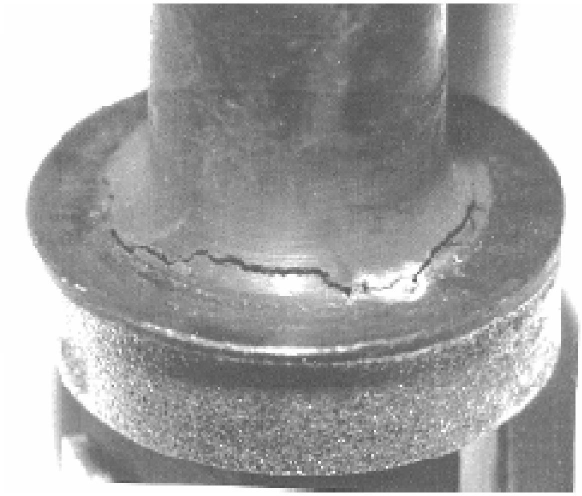
IASCC Service Experience

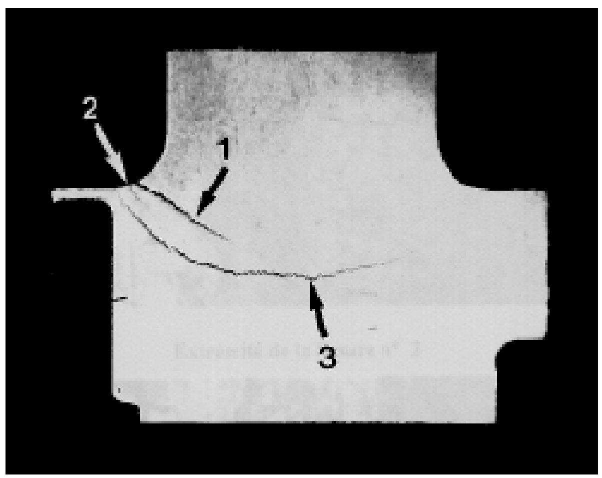
Component	Material	Reactor Type	Possible Sources of Stress
Fuel Cladding	304 SS	BWR	Fuel Swelling
Fuel Cladding	304 SS	PWR	Fuel Swelling
Fuel Cladding *	20%Cr/25%Ni/Nb	AGR	Fuel Swelling
Fuel Cladding Ferrules	20%Cr/25%Ni/Nb	SGHWR	Fabrication
Neutron Source Holders	304 SS	BWR	Welding & Be Swelling
Instrument Dry Tubes	304 SS	BWR	Fabrication
Control Rod Absorber Tubes	304/304L/316L SS	BWR	B_4C 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)

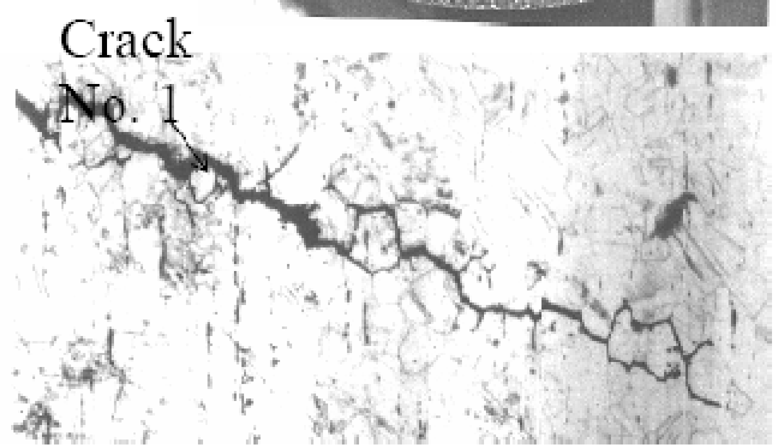
MichiganEngineering

Institute for Energy

Examples of IASCC: Cracked Baffle Former Bolts in a PWR

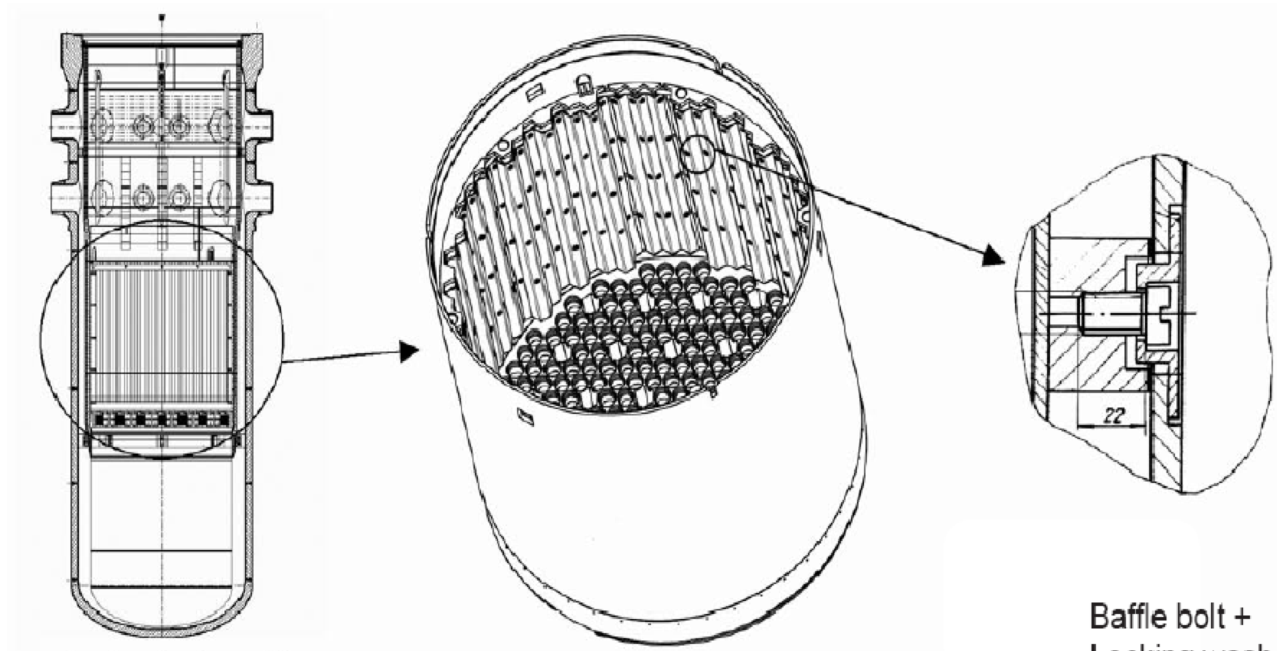






Loviisa core basket

- 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)



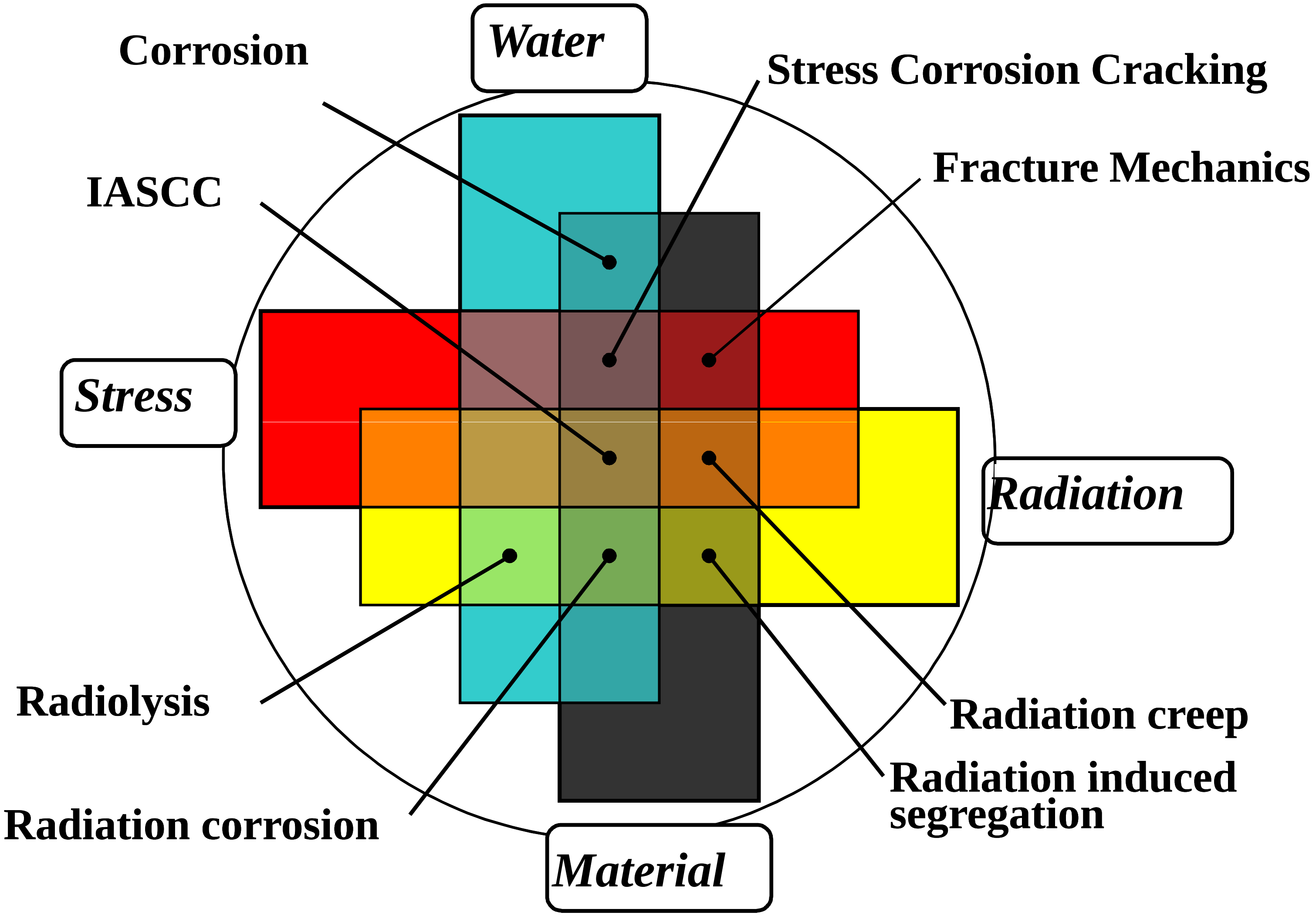
Reactor pressure vessel

Core basket

Baffle bolt +
Locking washer,
312 pcs.

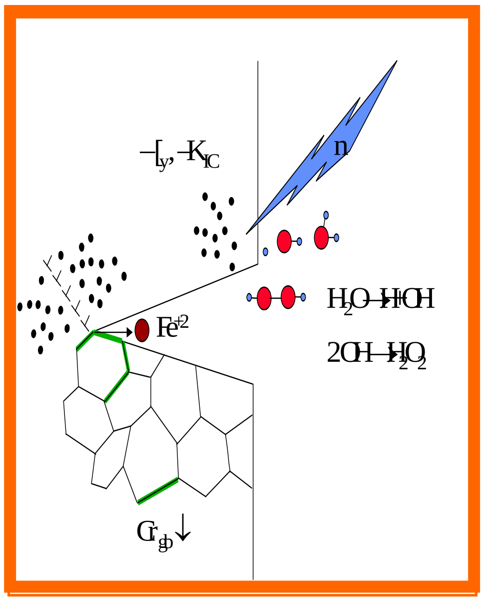
Institute for Energy

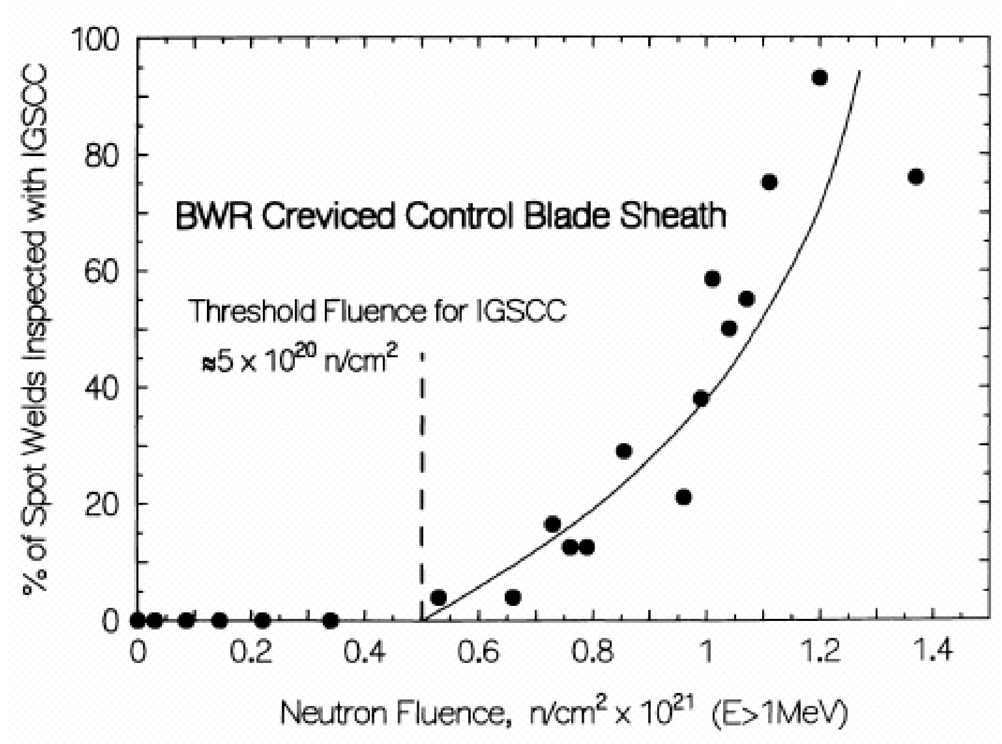
+ Radiation



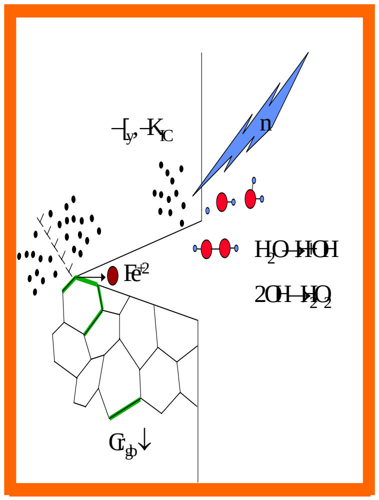
Institute for Energy

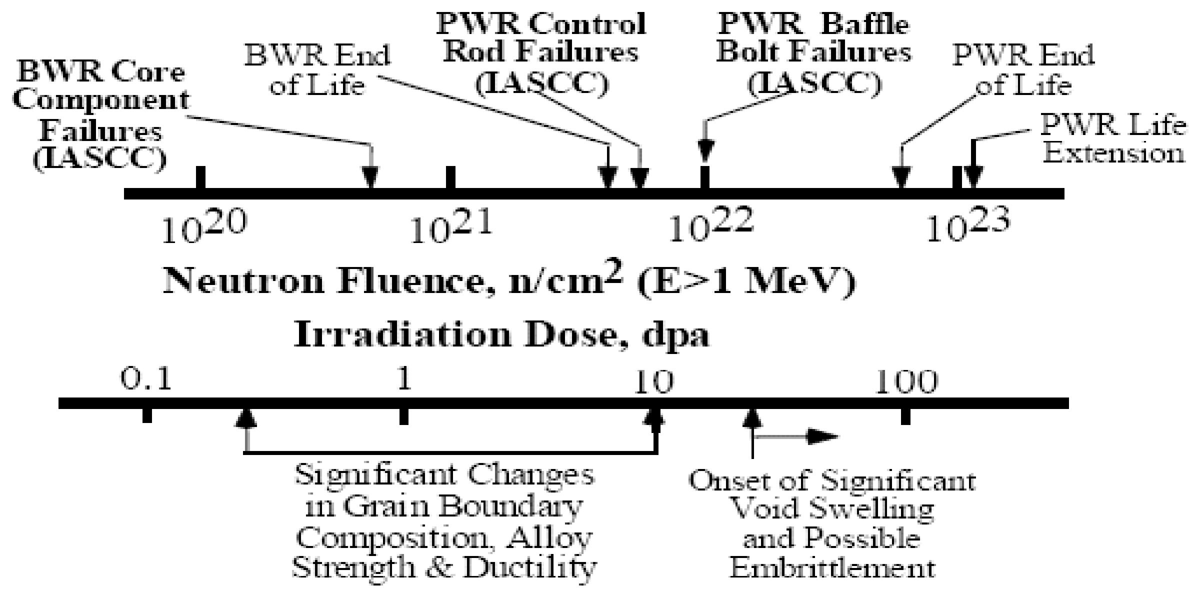
Strong influence of radiation





Microstructure Changes can be Correlated to Irradiation Dose/Fluence

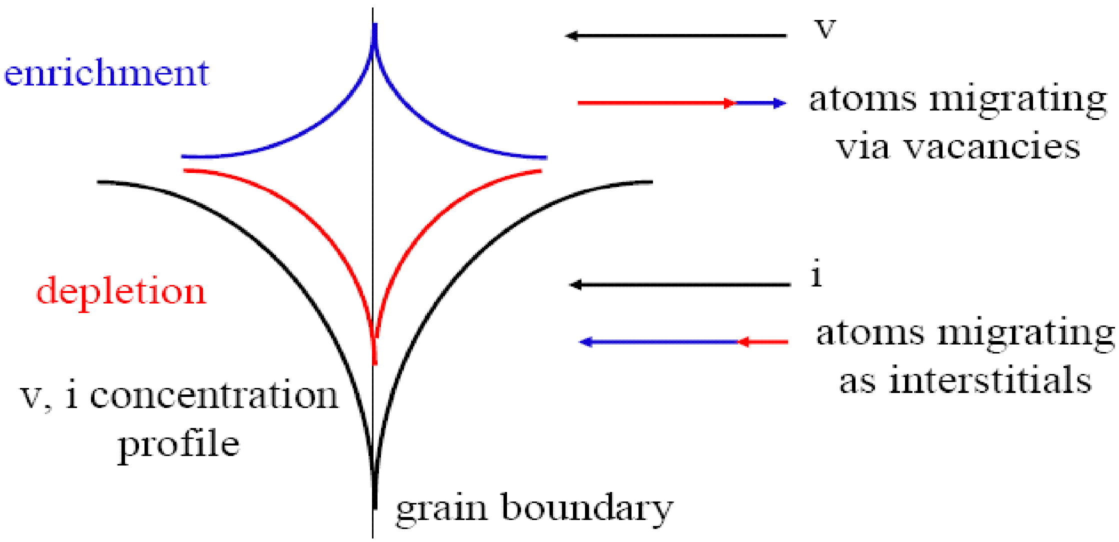


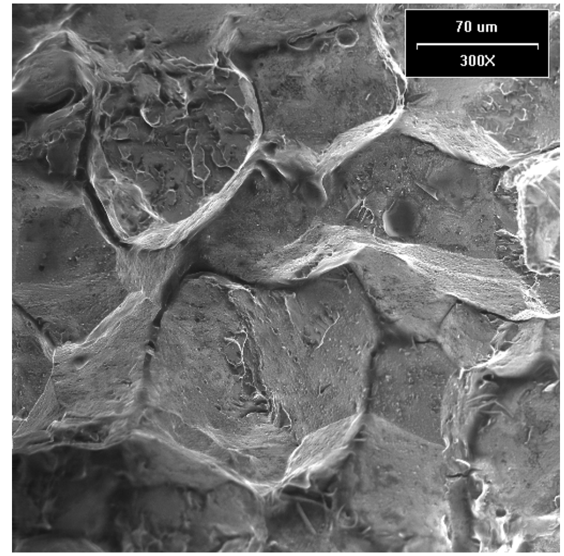


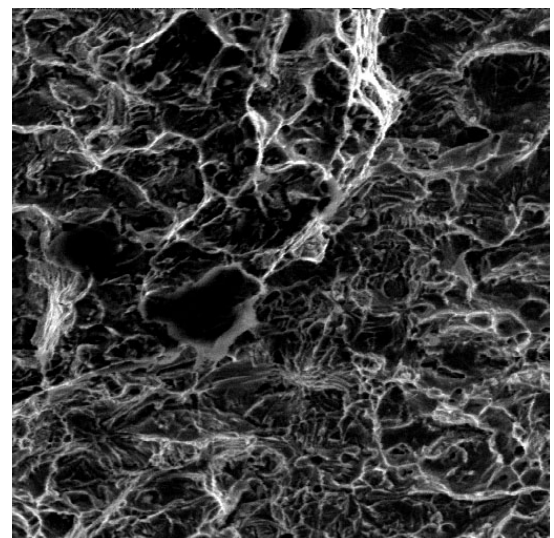
Note: ~15 dpa = 10^{22} n/cm^2 E $\geq$ 1 MeV (for PWR and BWR neutron spectra)
~7 dpa = 10^{22} n/cm^2 E $\geq$ 0.1 MeV (for PWR and BWR neutron spectra)

Radiation Induced Segregation - RIS

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



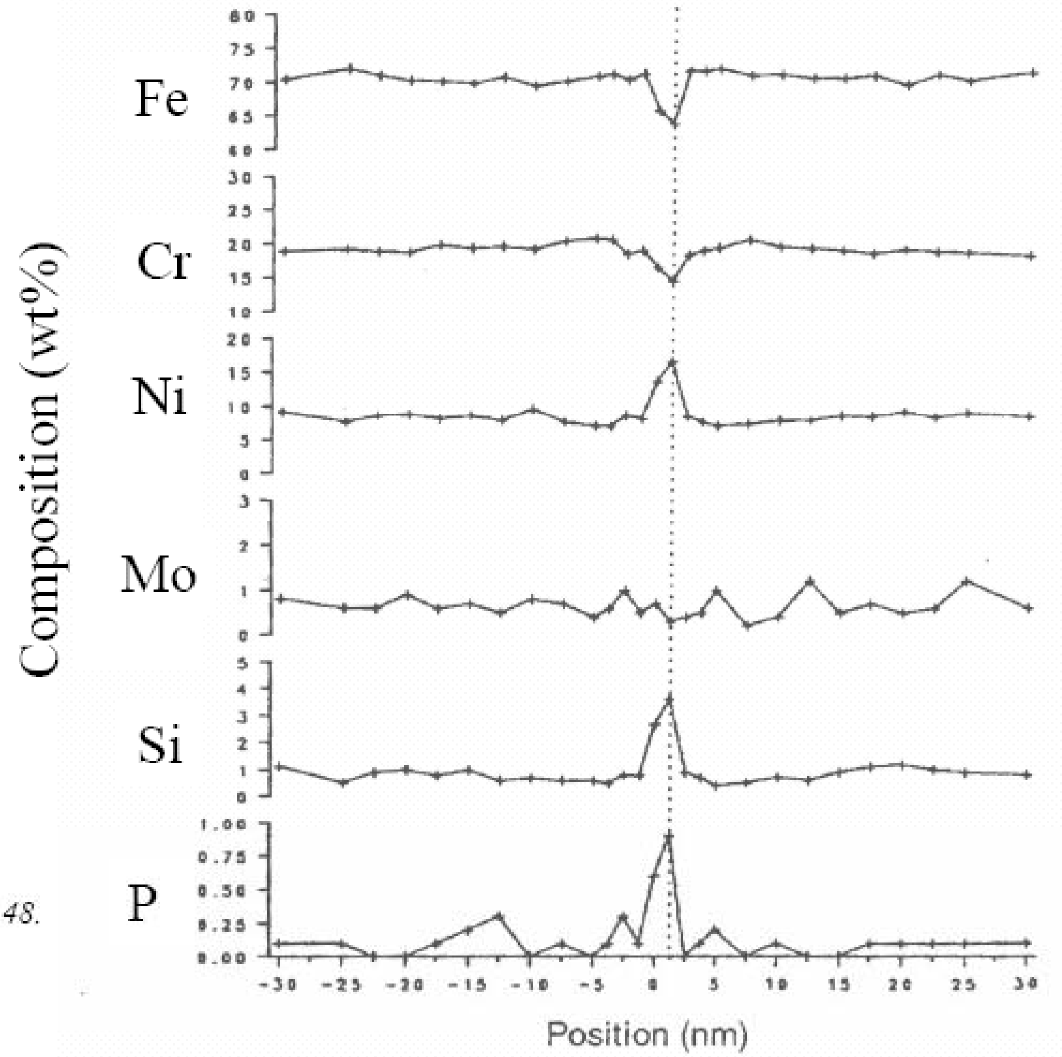




Institute for Energy

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

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

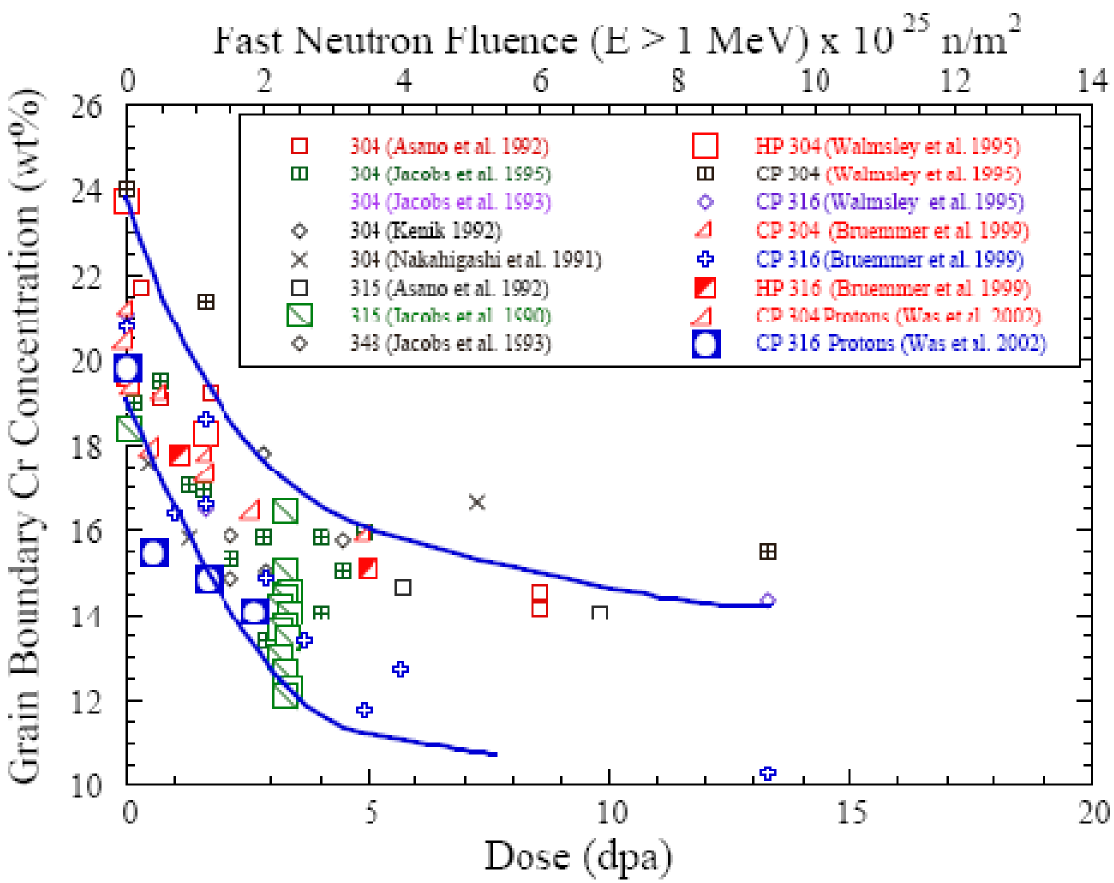


A. Jenssen et al, Corrosion 54 1 (1998) 48.

MichiganEngineering

Institute for Energy

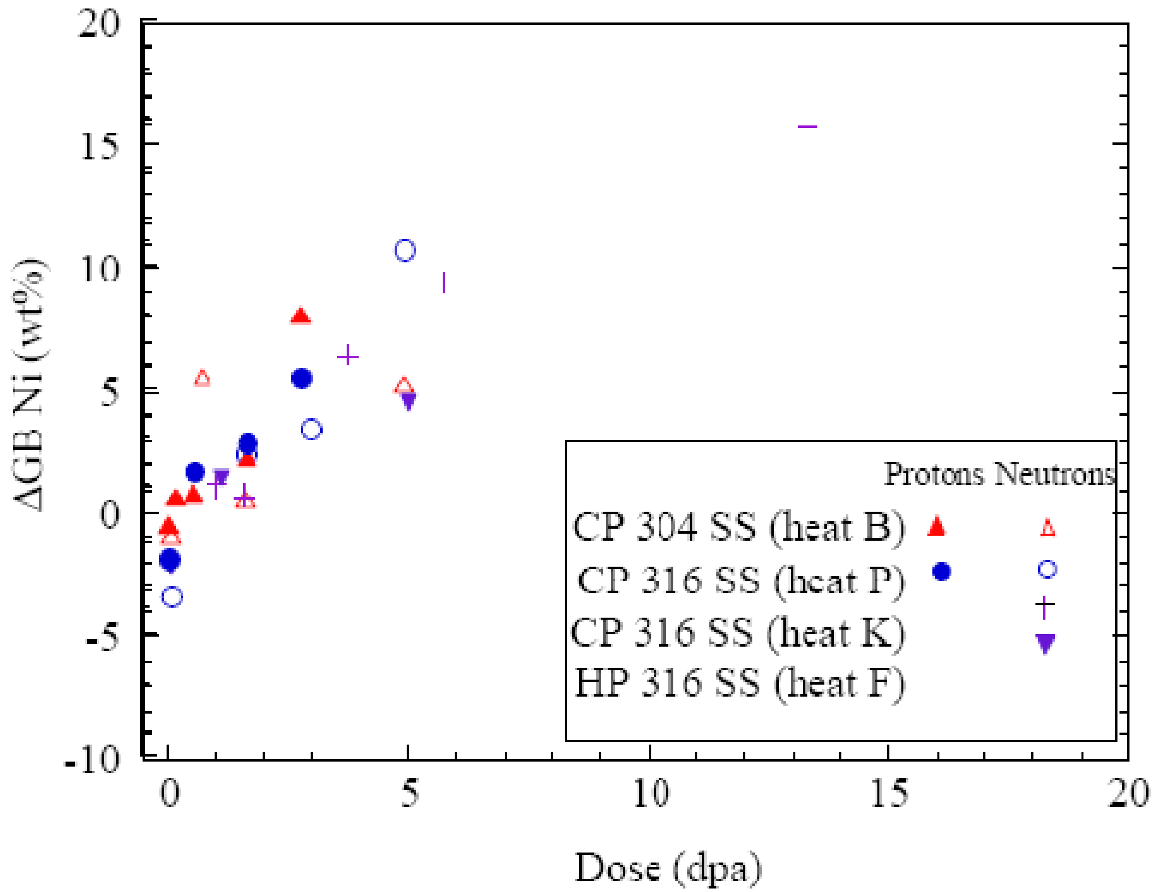
Grain boundary chromium depletion in irradiated austenitic iron base alloys



MichiganEngineering

Institute for Energy

Grain boundary nickel enrichment in irradiated austenitic iron base alloys



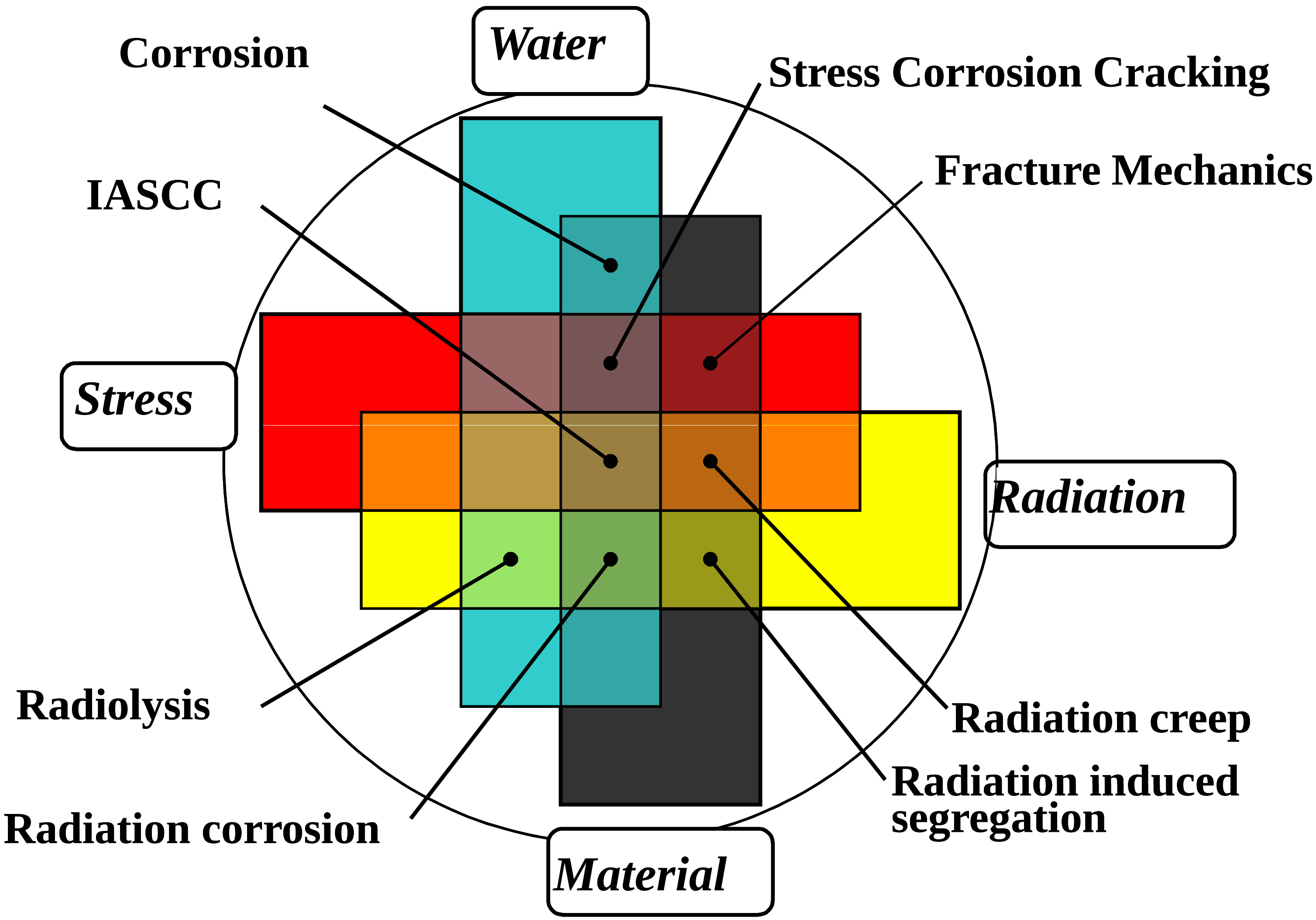
MichiganEngineering

Institute for Energy

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 γ' (Ni_3Si) 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

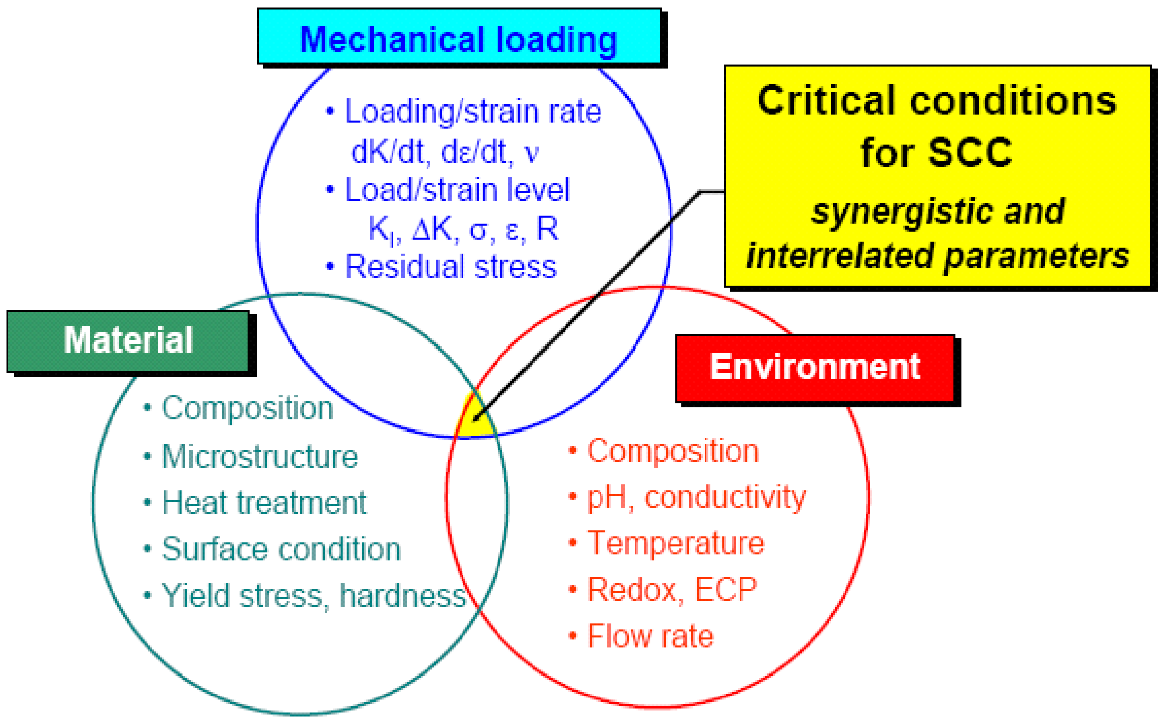


Institute for Energy

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)

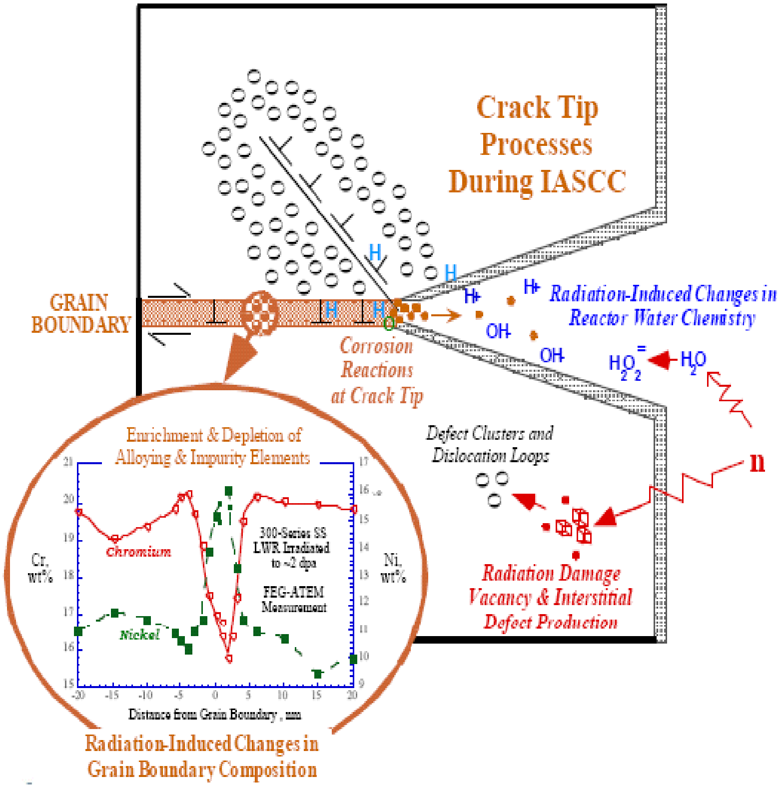


IASCC added feature to EAC:

- atomic displacements
- neutron irradiation significantly alters metallurgical microstructure
- ionizing (α,β and γ) radiation modify the

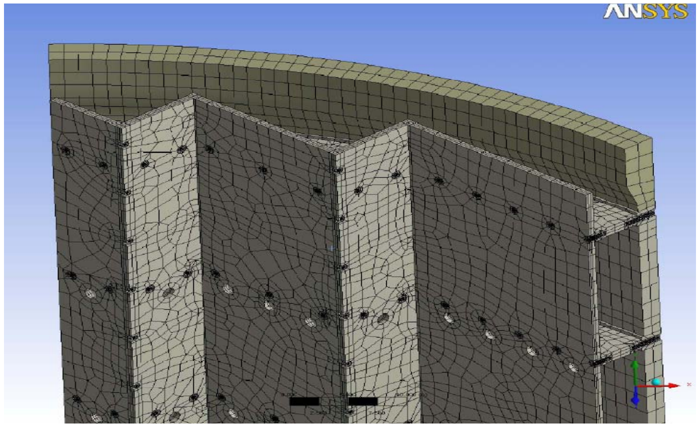
Effects of irradiation on SCC:

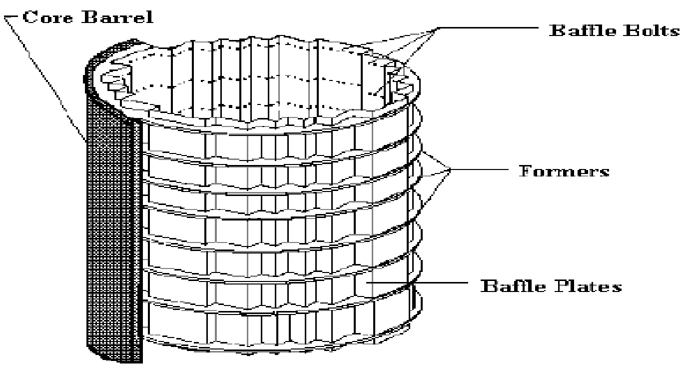
- primary defects
- defects segregation
- dislocation interaction
- grain boundaries
- localized stress and strain
- environment
- stress relaxation by irradiation
- creep (beneficial factor for IASCC)

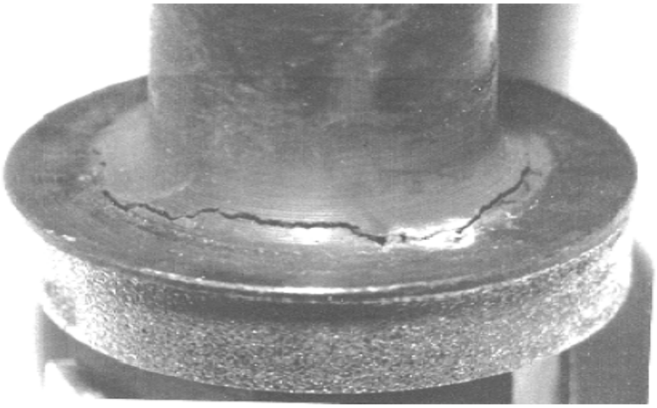


IASCC

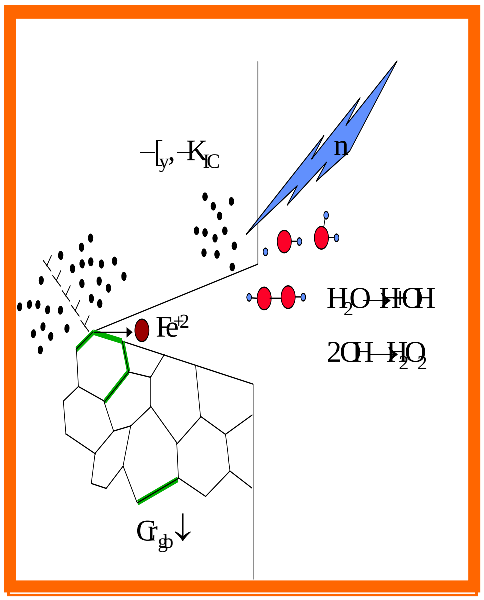
- Radiation
- Stress
- Time
- Temperature
- Environment

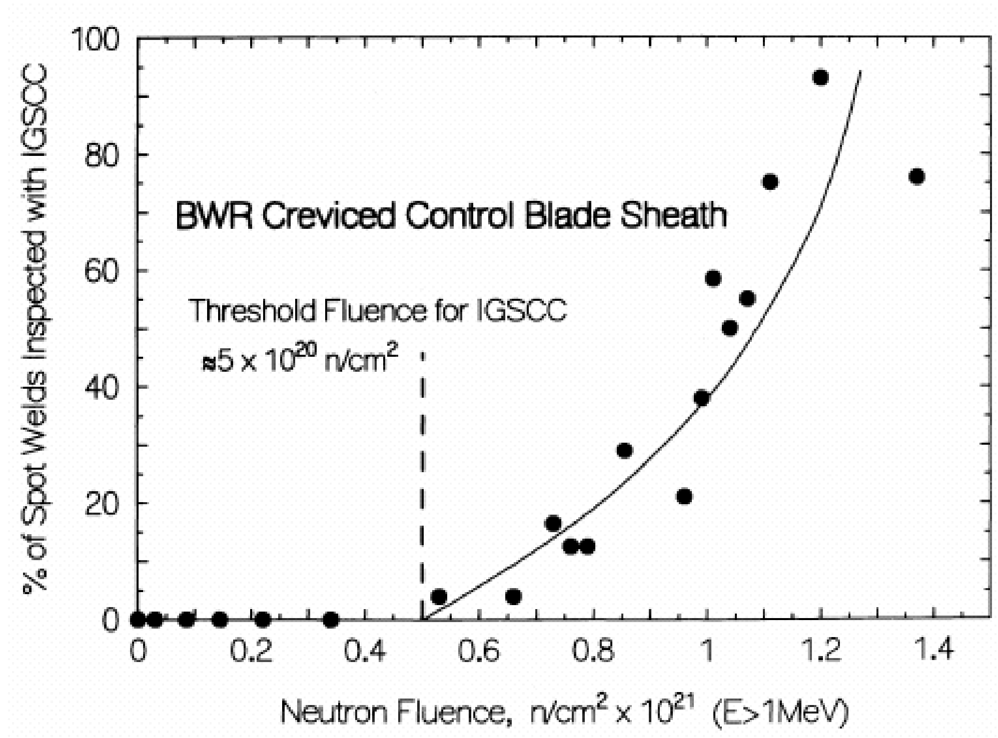






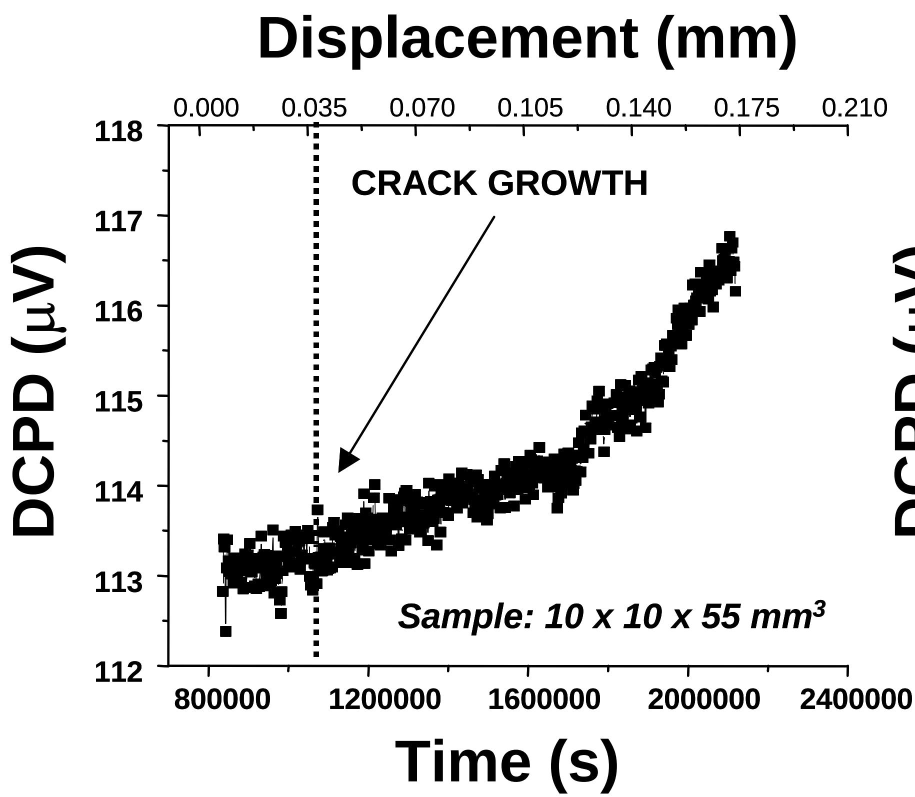
Strong influence of radiation

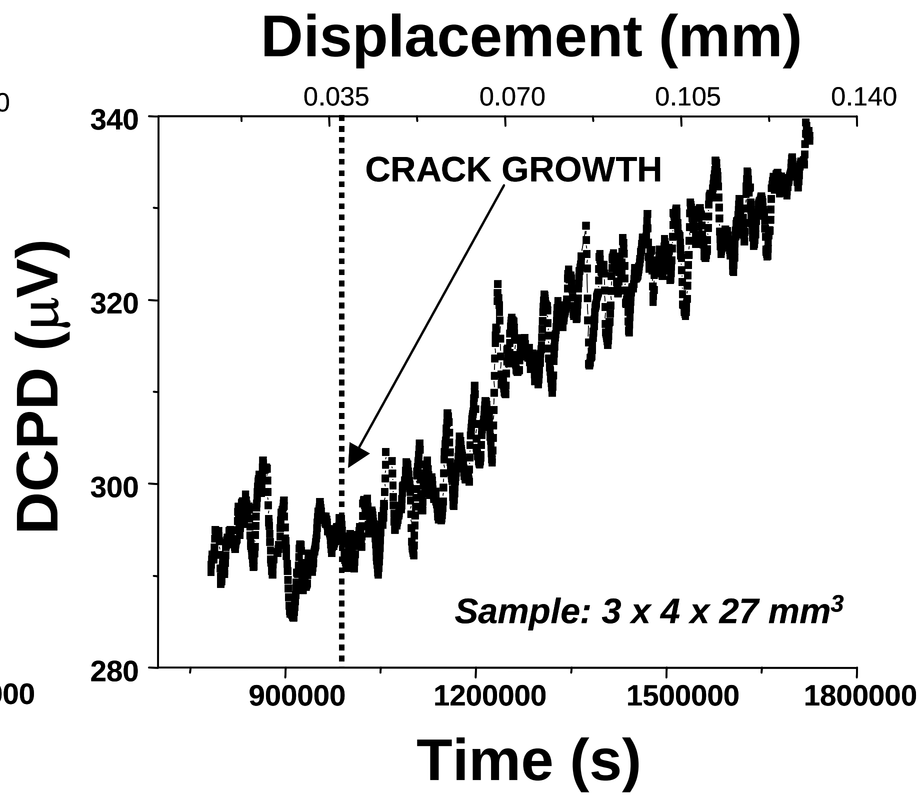




Institute for Energy

SCC - RPV internals



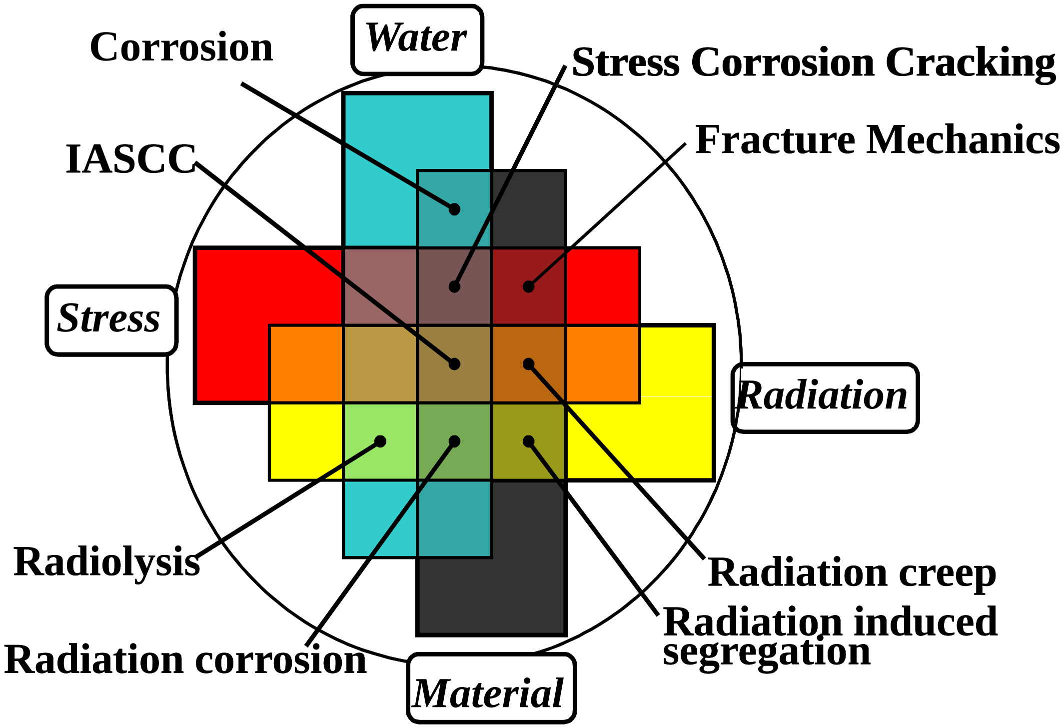


Testing Environmentally Assisted Cracking (EAC) of Reactor Materials using Pneumatic Servo Controlled Fracture Mechanics (PSCFM),
sub. International Journal of Pressure Vessel and Piping, 2006
R. Novotny, F. Sevini, L. Debarberis, S. Petr, M.Kytka,

Institute for Energy

CONTENT

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



Institute for Energy

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

Radek Novotny & Luigi Debarberis
Institute for Energy (IE)
Petten, The Netherlands

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

Trieste, April 2009