IRRADIATION ASSISTED STRESS CORROSION CRACKING

(IASCC)

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Outline

1. What is IASCC
2. Field cases - examples
3. Present understanding
4. IASCC key factors
   - Threshold dose
   - Localized deformation
   - Threshold stress
   - Helium/Hydrogen effects
   - Oxidation effects
5. Summary

What is IASCC

Irradiation assisted stress corrosion cracking

- **Sensitivity to cracking** of structure materials **acquired after irradiation** above threshold dose (1-3 dpa)
- An **intergranular crack initiation and growth** in irradiated materials, subject to load in contact with water coolants

- Issue of **Austenite stainless steels** and **Ni based alloys** of BWR and PWR reactor vessel internals (RVI)

IASCC – Field Experience

~1980: First field failures in **BWR** and **PWR**
→ easy replaceable components

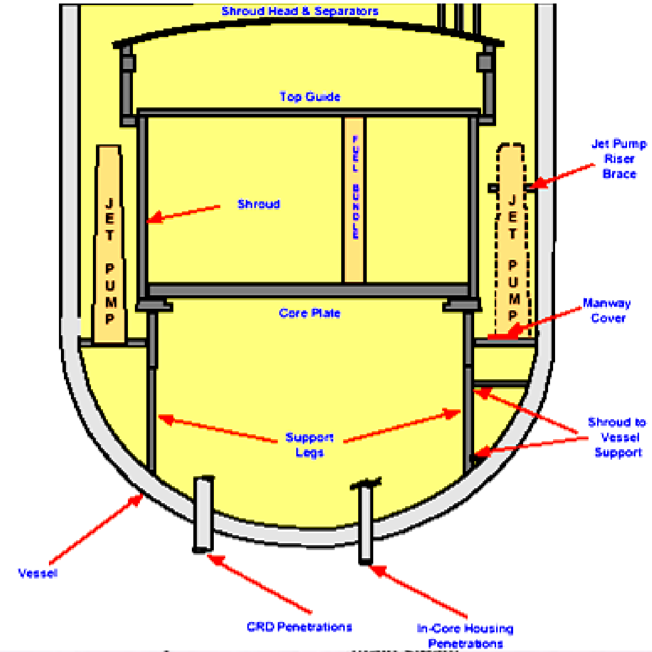
~1990: IASCC in **BWR core shrouds**, mainly around welds,
→ repairs / replacement

~1990: IASCC in **PWR - baffle bolts** failures
→ replacement

BWR: Core shroud

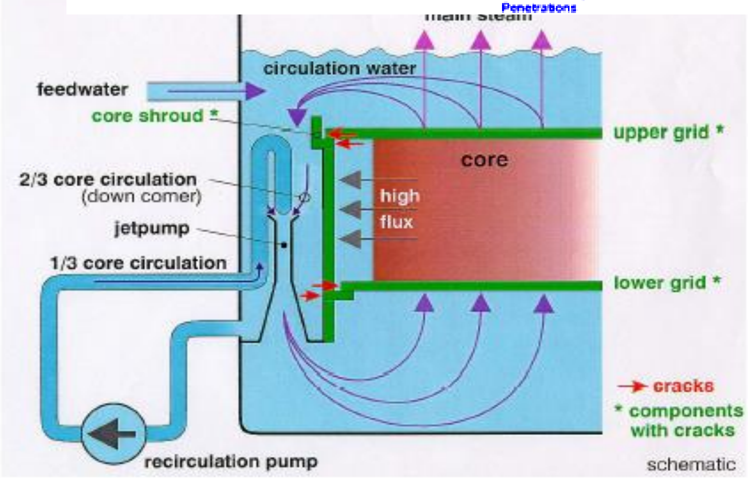
Materials: 304, 304L, 347 SS

Cracking indicated mainly around welds after plant operation for 17-23 years



Actions

- Operation continued (crack < limit)
- Repair if longer cracks
- Replacement
  - Japan (Fokushima-Daiichi),
  - Sweden (Oskarshamn, Forsmark)



BWR with recirculation line

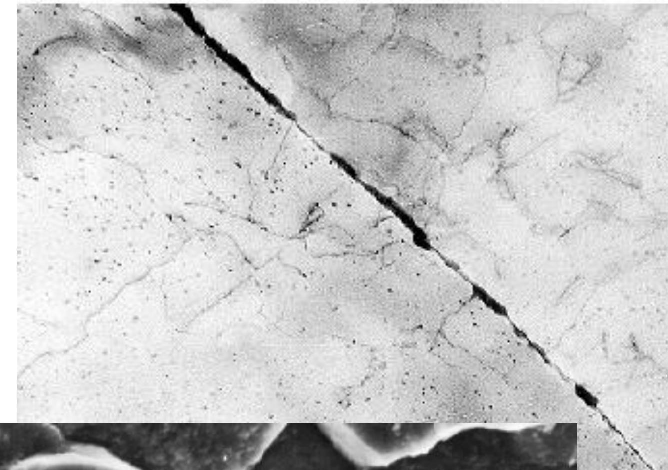
Intergranular fracture in field failures

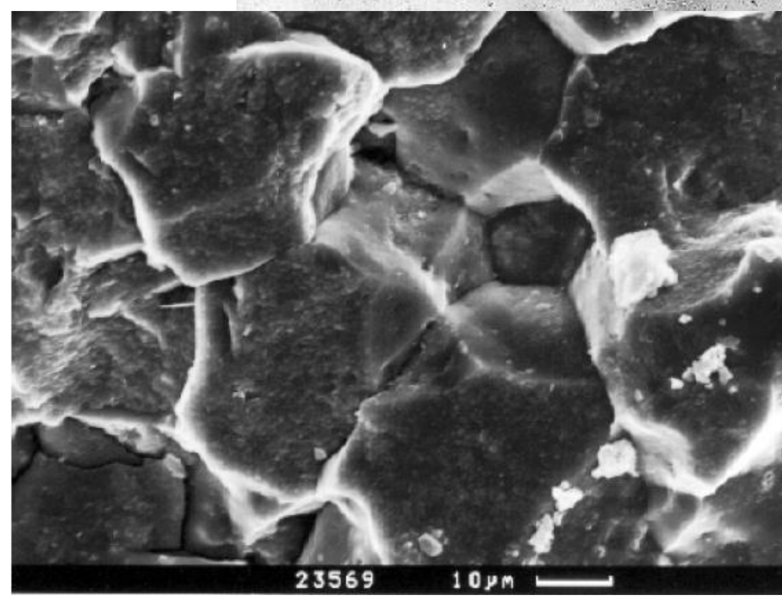
BWR (Germany, KWW, 1994) : **IGSCC**

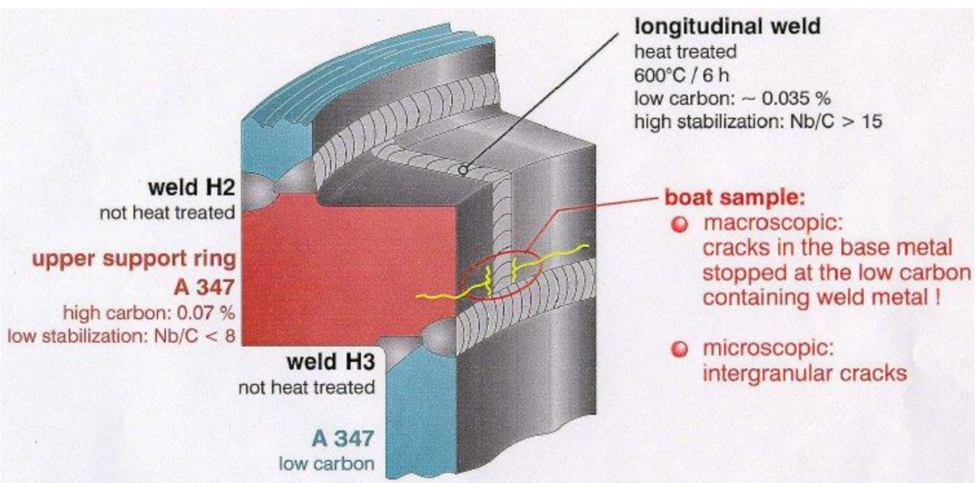
Core shroud / **347** (Nb stabilized) SS

→ coarse grain boundary carbides arising from thermal sensitization

→ Intergranular (IG) crack surface







A. Roth et. al, , 2004,
IAEA Technical Meeting on Reactor Core Internal Behavior and Technology of Repair and Replacement in NPPs, Erlangen.

PWR: Core baffle

Materials: 304, 316L, 321(WWER) SS

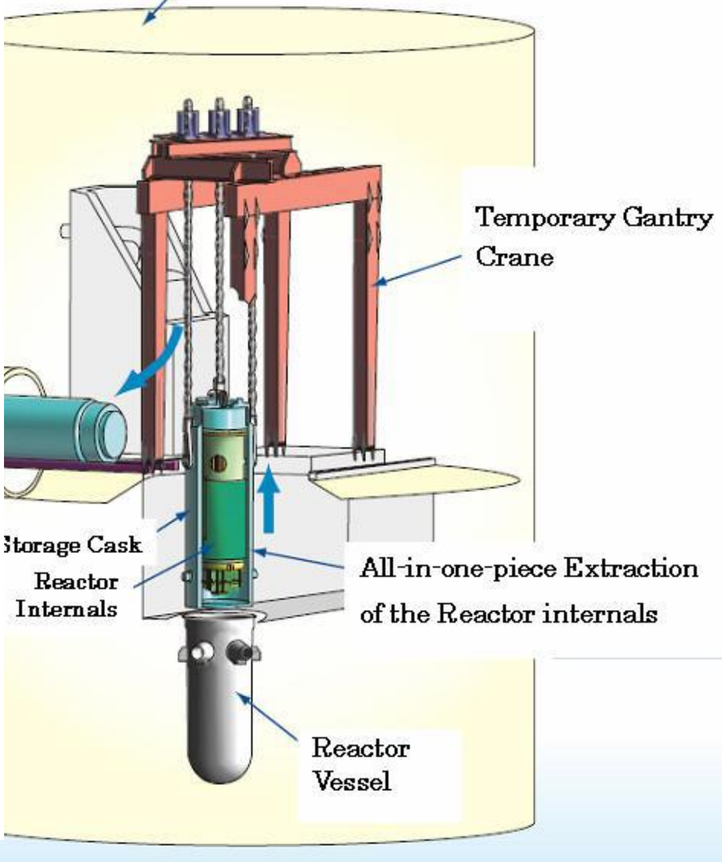
No IASCC reported

no cracking detected in Chooze A baffle 10-36 dpa

Actions

Preventive replacement

2005 – 2006 in 3 plants in Japan



The all-in-one-extraction Method

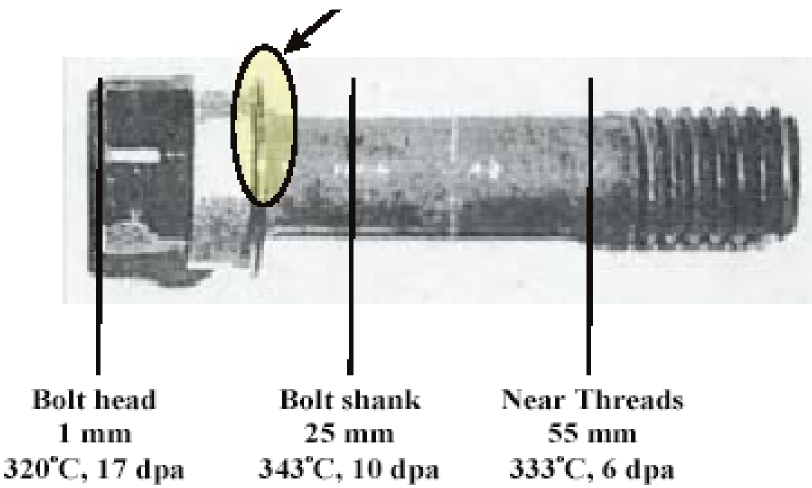
S. Yaguchi, J. Uchiyama, 2006, Fontevraud 6, International Symposium

PWR: Baffle Bolts

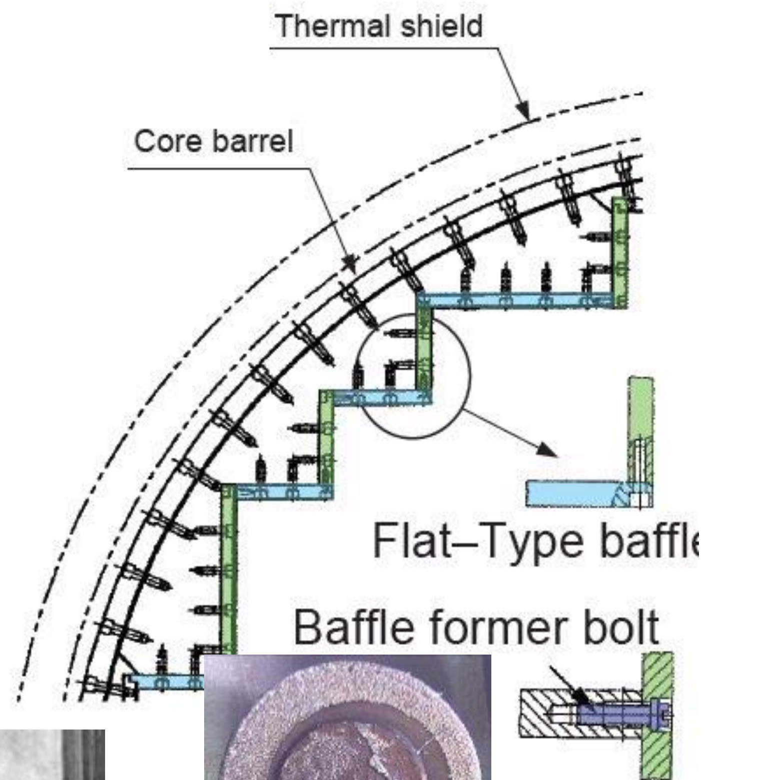
Materials: **316CW, 347, 321(WWER) SS**
Crack detected by UT, VT inspections

Actions: Replacement

The baffle bolt of NPP Tihange used for detailed laboratory study



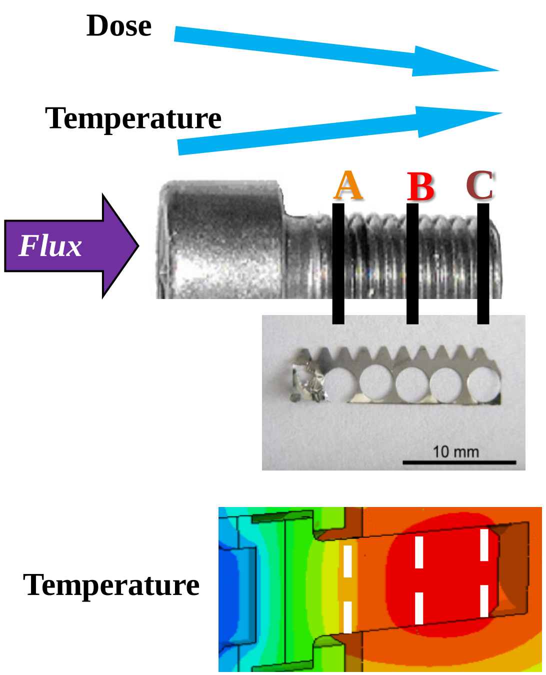
R. S. Pathania, K. Gott, 2002 , Fontevraud 5

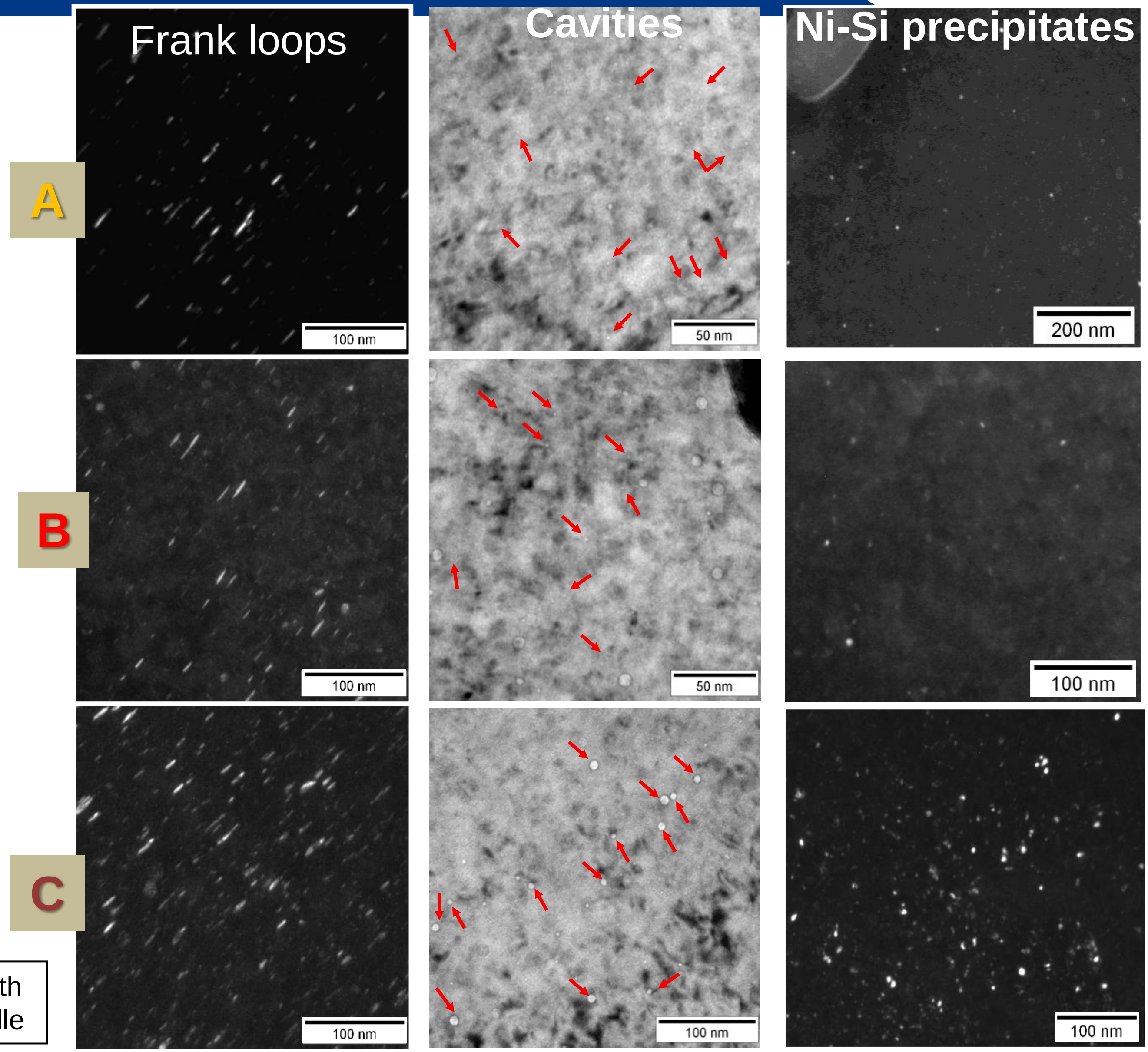


H. T. Tang, 2006 , Fontevraud 6

Bolts – complex radiation damage

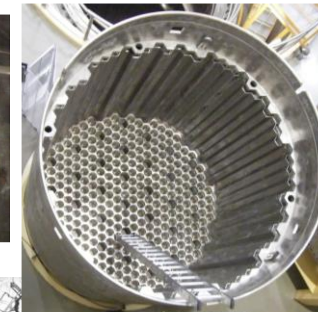
08Ch18N10T (~321CW):
- 12 dpa
- WWER
- 260 – 400(?) ˚C
- 5-40 ×10^{-9} dpa/s





J.Michalička et al., 2013, 16th EnvDeg conference, Asheville

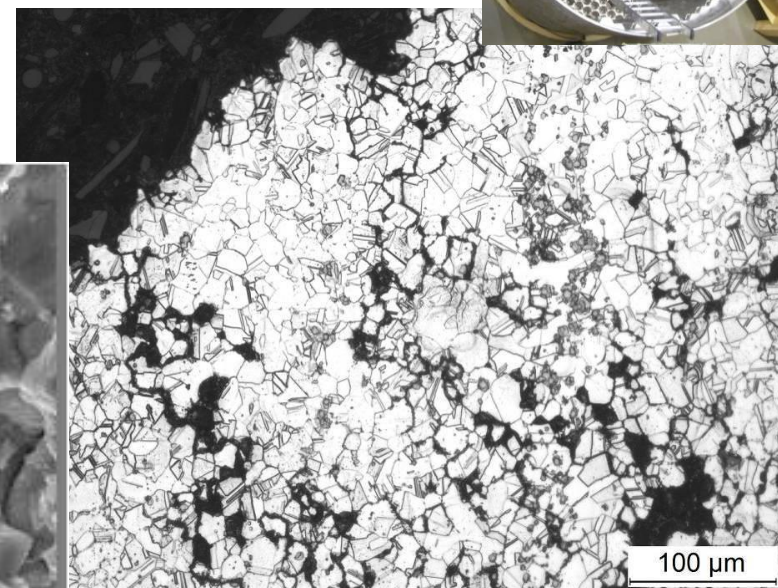
Intergranular fracture in field failures



PWR / WWER (Finland, Loviisa, 2004): **IASCC**

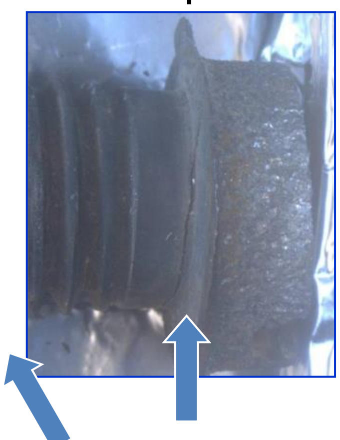
Baffle bolt / 08Ch18N10T(Ti stabilized) SS ≈ 321 SS

→ Bolt section - extensive secondary IG

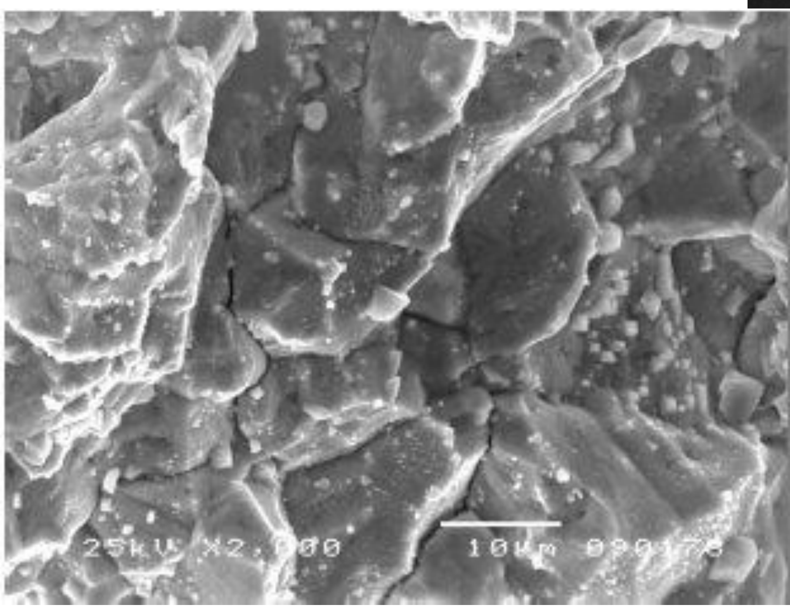


↓ Intergranular (IG) crack surface

2.9 dpa



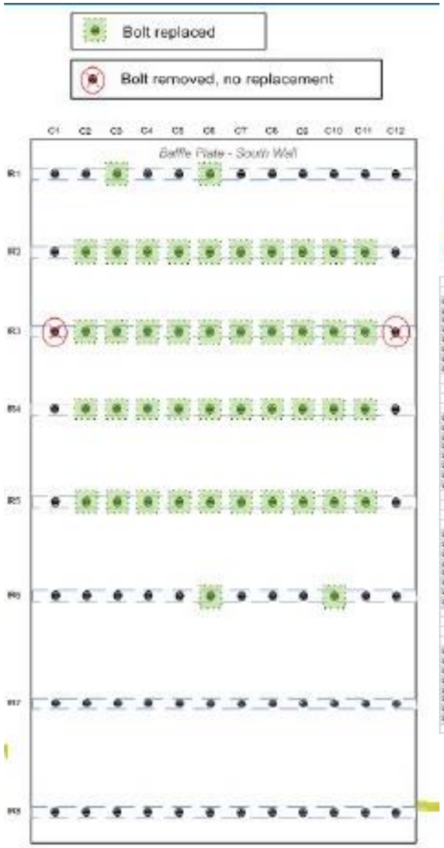
2 cracks

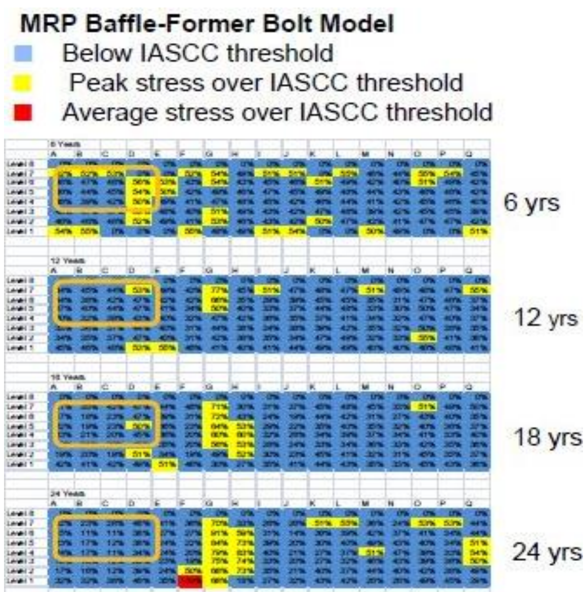


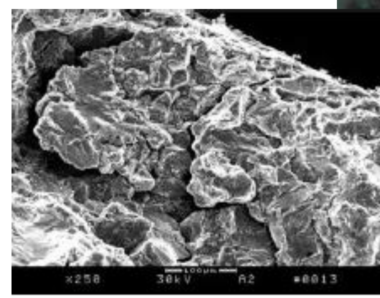
U. Ehrnsten et al. , 2011, 15th Environmental Degradation of Materials in Nuclear Power System, Colorado Springs

Issue of IASCC prediction

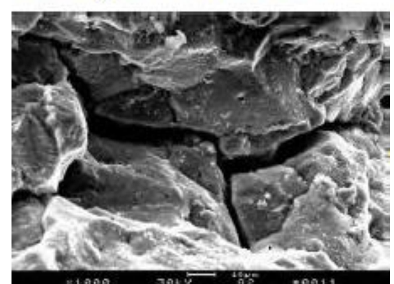
- DC Cook 2 exchanged at once 44 bolts of one wall (2.-5. rows of 7)







Intergranular IASCC



J.K. McKinley et al. , 2012, Int. BWR &PWR Materials Reliability Conference &Expo, Washington

Present understanding of IASCC

- Synergistic effect of many interrelated factors

Threshold stress for crack initiation + growth is f(dose), at highest ≈ 50%YS

Crack growth rate CGR ~ $d\varepsilon_{CT}/dt$

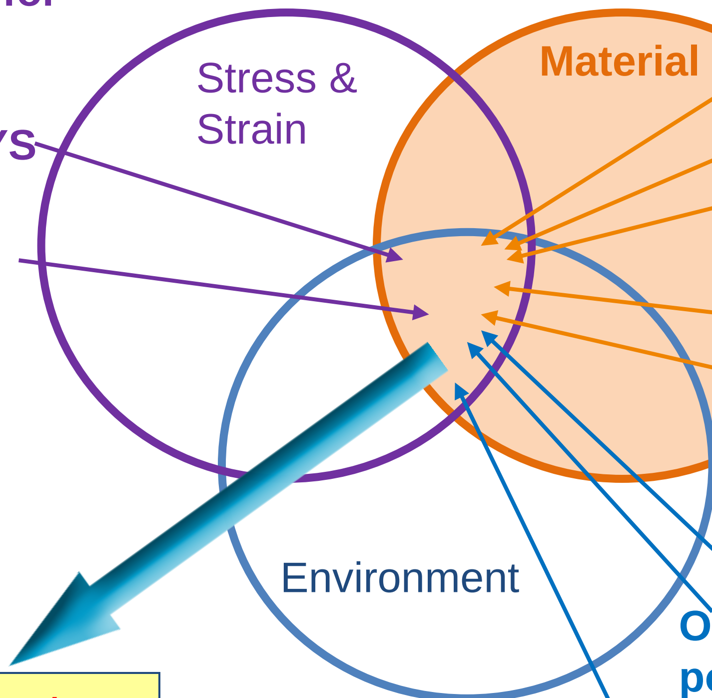
1-3 dpa threshold dose

↑ΔYS → ↑ %IG

RIS of grain boundaries
- Cr, Ni, Mo, P not dominant
- Si shows correlation

Voids and cavities presence

Localized deformation - strong correlation



Radiolysis: H_2O_2 production

Oxidation: oxide layer is more porous and thinner under irradiation

Typical IG fracture

%IG in BWR, PWR > %IG in inert gas

Unified mechanism of crack growth in BWR & PWR

Hydrogen effects

H in steels **~3 ppm, → ↑ ~30 ppm** due irradiation in water

BWR HWC: for 5-10 dpa H retards crack growth

PWR: H content ↑ %IG IASCC

H increases strain localization

IASCC – known facts

1. **Threshold dose 1-3 dpa** (BWR < PWR)

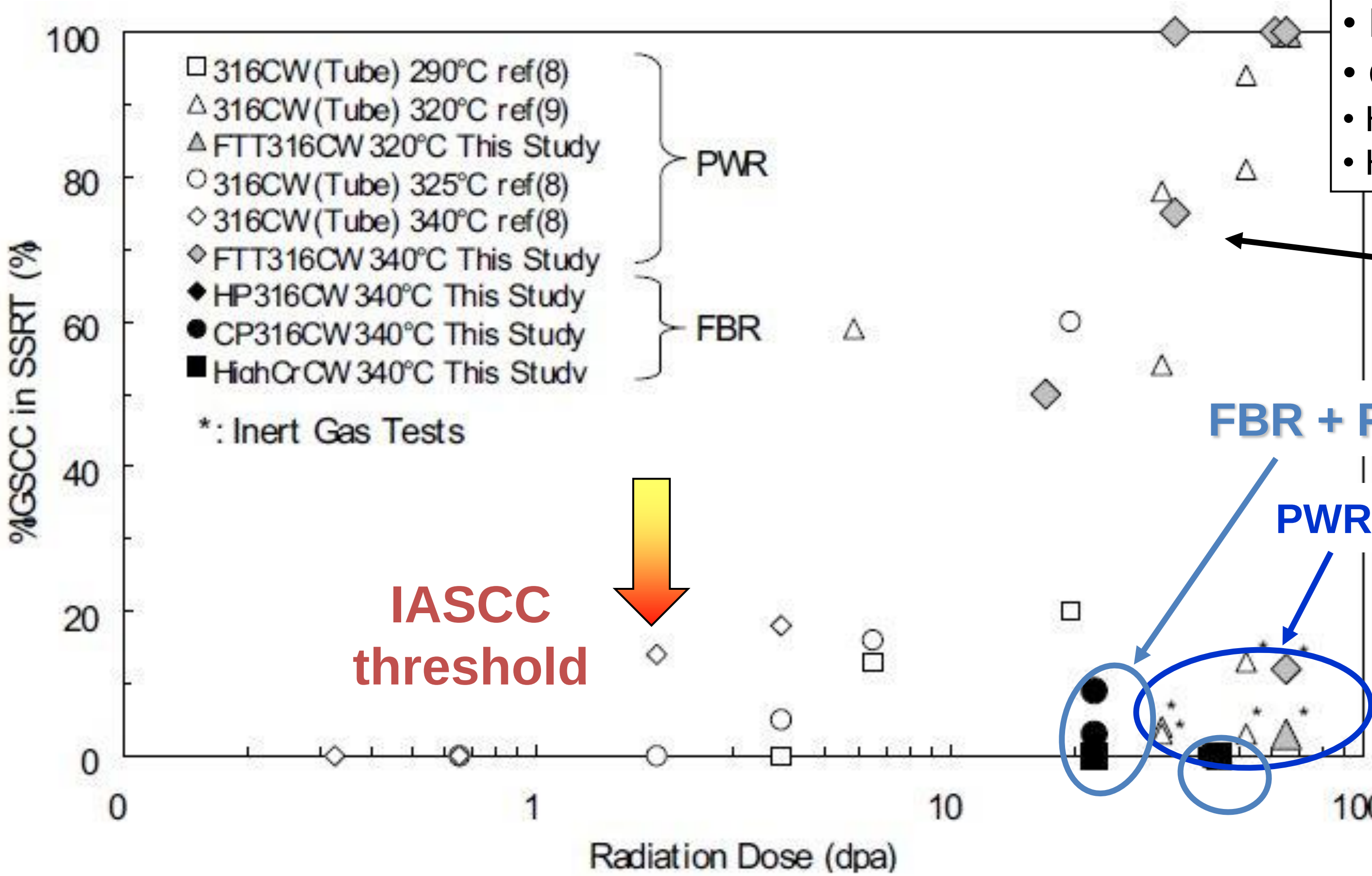
2. Radiation hardening & reduced plasticity ↑ IASCC

3. **IG fracture**: %IG in BWR, PWR >> %IG in Ar

4. **Strain rate** effect

IASCC threshold dose

316CW SS:
- Flux Thimble Tube (FTT)
- Commercial Purity (CP)
- High Purity (HP)
- High Cr

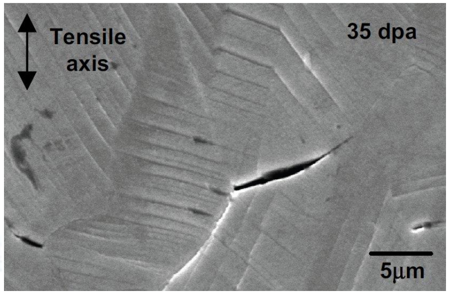


PWR + PWR test

FBR + PWR test

PWR + Inert Gas

IG in Ar!



K. Fukuya et al., 2005, 12th EnvDeg., Salt Lake City

"This Study" K. Fujimoto et al., 2005, 12th EnvDeg., Salt Lake City;
ref(8) I. Suzuki et al., 1996, ICONE, ASME, 5, p.205
ref(9) K. Fukuya et al., 2001, 10th EnvDeg., Stevenson

Strain rate effect in air & water

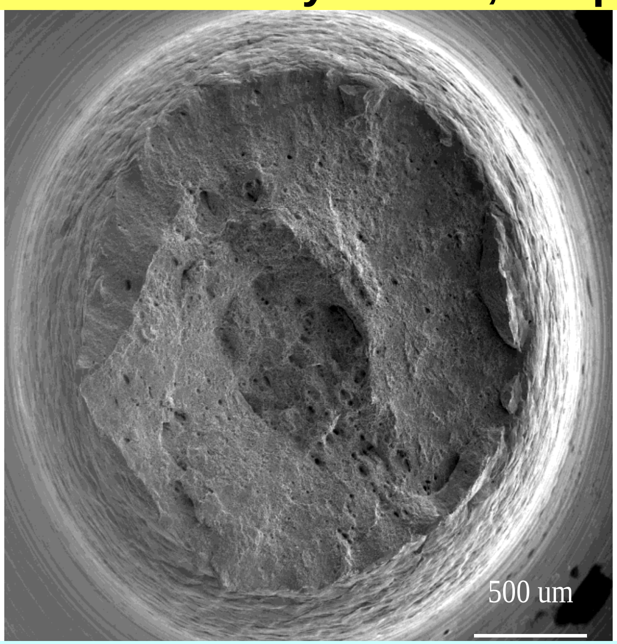
- 08Ch18N10T (~321 SS) irradiated in WWER plant to 5 dpa

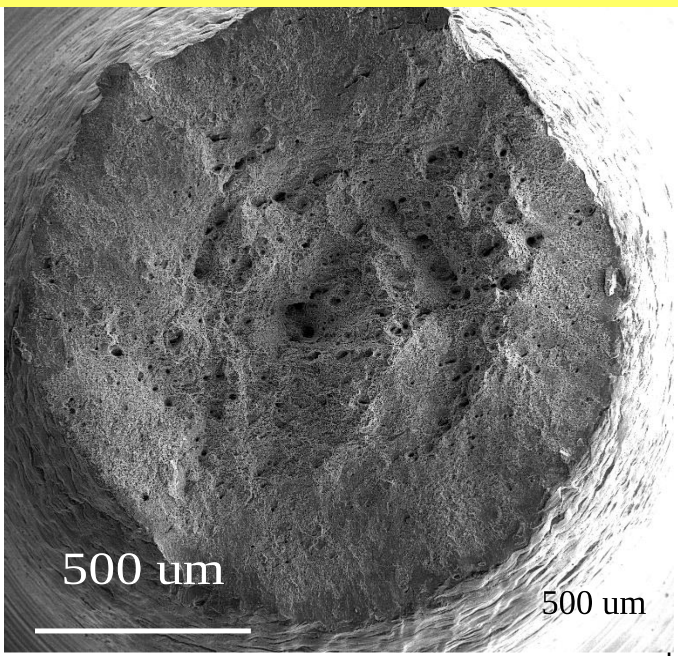
In air, radiation hardening &/or slow straining did not cause IG

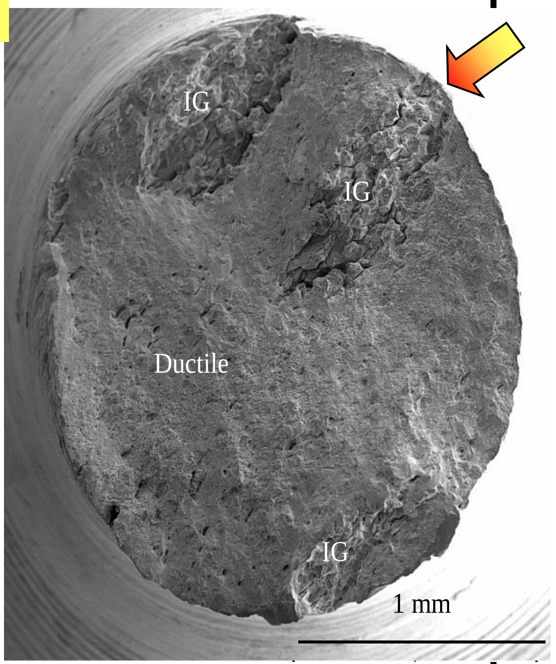
IG+TG = IASCC only in water

Tensile Tests in Air 320^{o}C	**SSRT water, 320^{o}C**

Fully ductile, dimpled and shear fracture





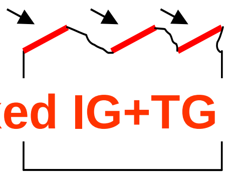


Normal test rate, 4*10^{-5} 1/s

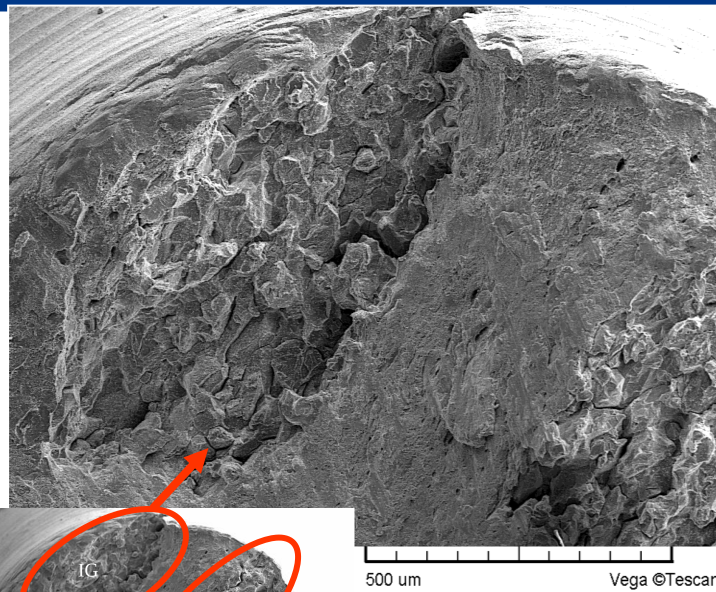
Slow test rate, 3*10^{-7} 1/s

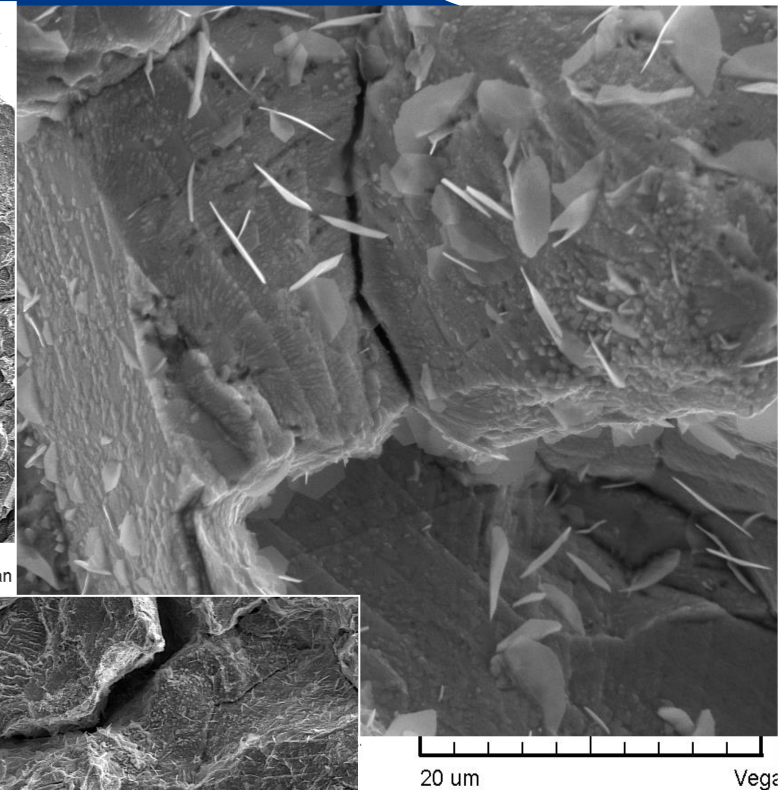
IASCC fracture character

5.2 dpa
WWER
SSRT
3×10-7 s-1

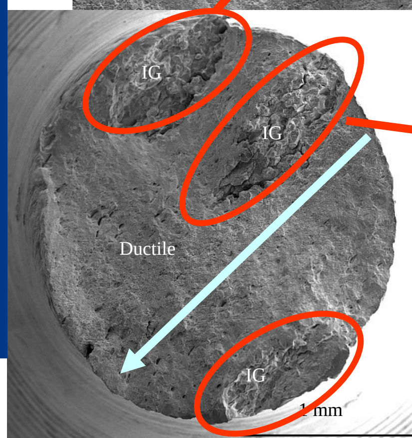


Mixed IG+TG

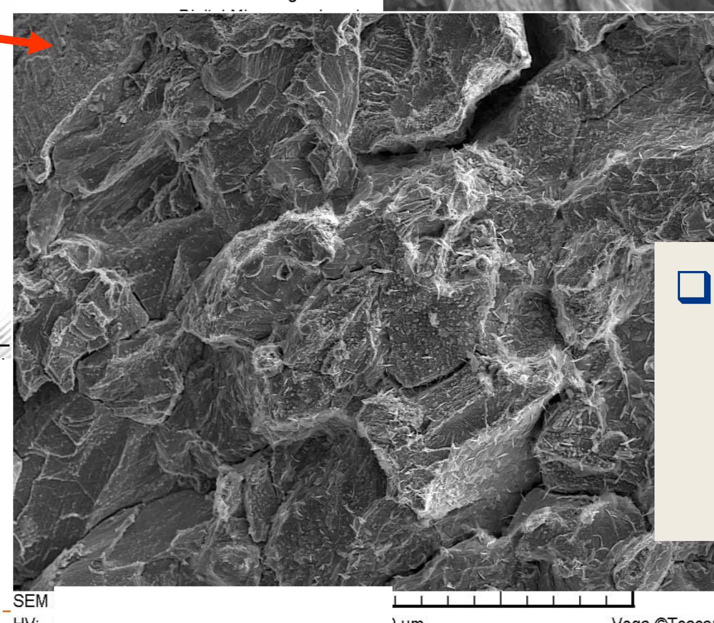




Fracture path inclined ~45° to load axis → **strain controlled crack growth**



Crack growth parallel to δ ferrite texture



- IASCC has features of locally ductile fracture mode

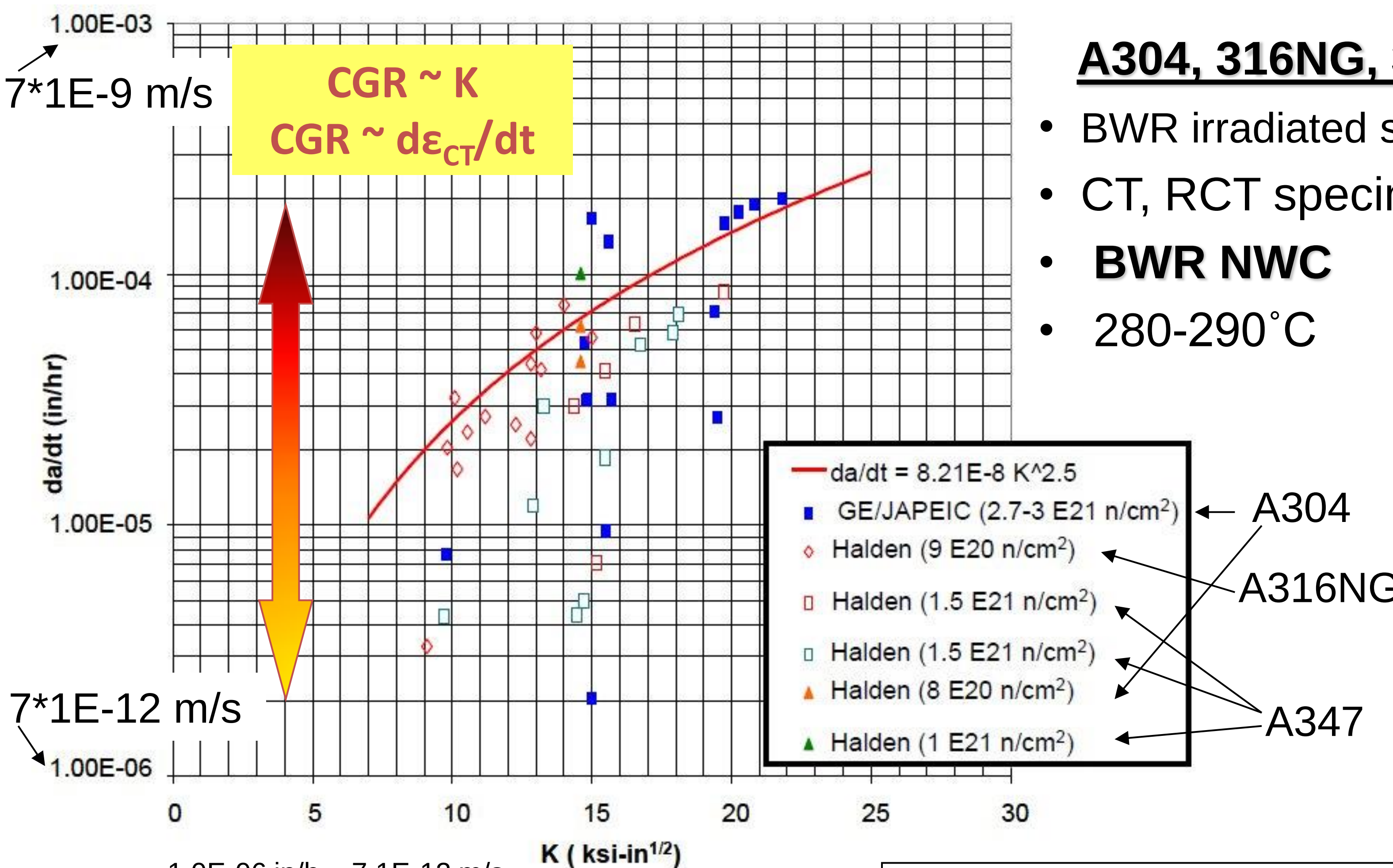
IASCC – further known facts

5. **Crack growth rate** (CGR)
   - CGR ↑ dose ↔ CGR ↑ YS
   - Modeled as CGR ~ $d\varepsilon_{CT}/dt$
   - Single CGR mechanism in BWR and PWR

6. **Localized deformation** in channels correlates with crack initiation

7. **RIS** radiation induced segregation, low correlation

CGR: screened quality data in BWR NWC



1.0E-06 in/h = 7.1E-12 m/s
1 ksi √in = 1.1 MPa√m

A304, 316NG, 347:

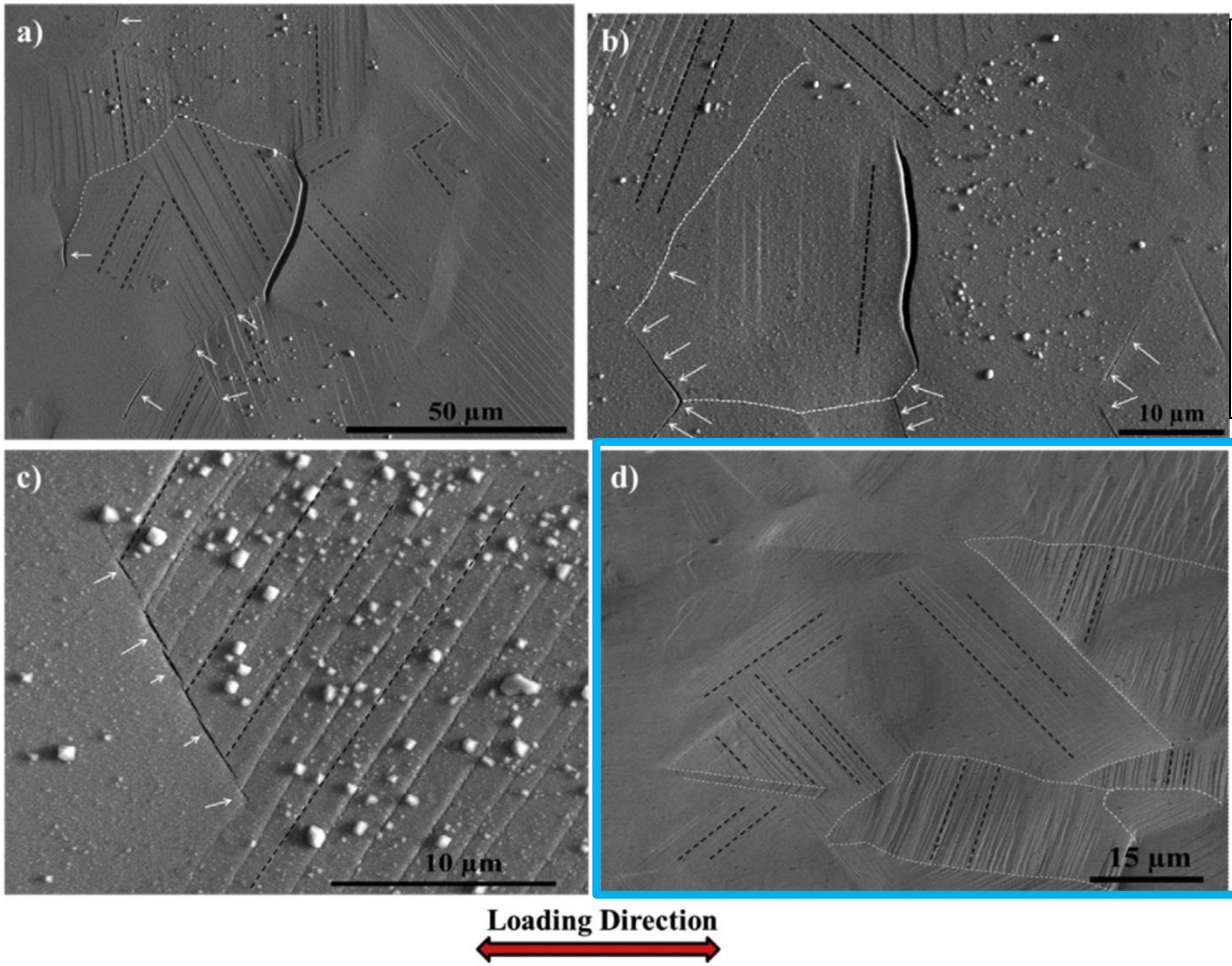
- BWR irradiated steels 1-5 dpa
- CT, RCT specimen
- **BWR NWC**
- 280-290˚C

Scatter due to material variance ?

R. Pathania et al. , 2009, 14th EnvDeg., Virginia Beach.

Role of localized deformation

❑ Crack initiation



304L SS:

- 10 MeV Fe ion irradiation (~10^{-4} dpa/s) at 450°C
- up to 5 dpa
- Tensile specimens 2×2×18 mm
- Strained to 4% in a **PWR** water (340°C, 5×10^{-8} s^{-1})

Air

❑ Localized strain is not in itself capable of initiating IG

J. Gupta et al., 2016, Journal of Nuclear Materials 476

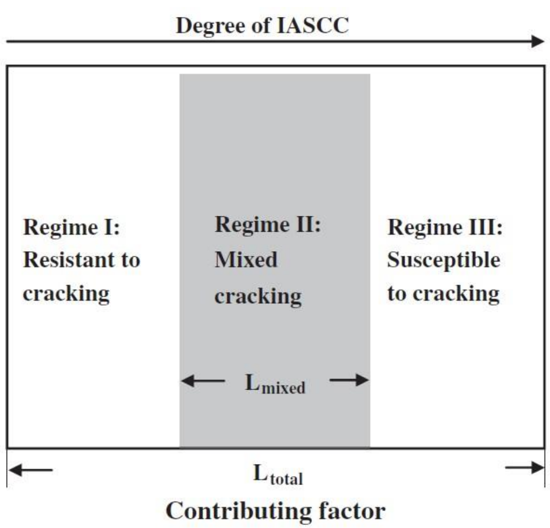
IASCC: Role of localized deformation

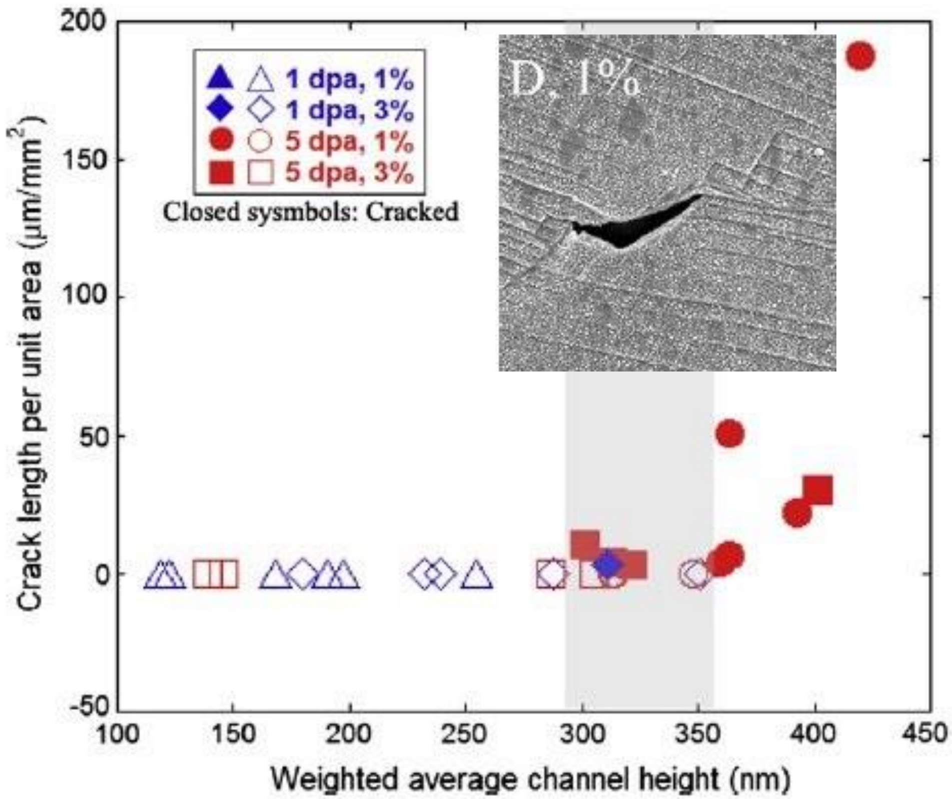
- Localized deformation measured by weighted average channel height

$$\bar{h} = \frac{\sum_{i=1}^{n} h_i^2}{\sum_{i=1}^{n} h_i}$$

- Correlation strength test

$$S_{IASCC} = \frac{L_{total} - L_{mixed}}{L_{total}} = \mathbf{0.88}$$





A304 + others 15-22Cr12-32Ni:

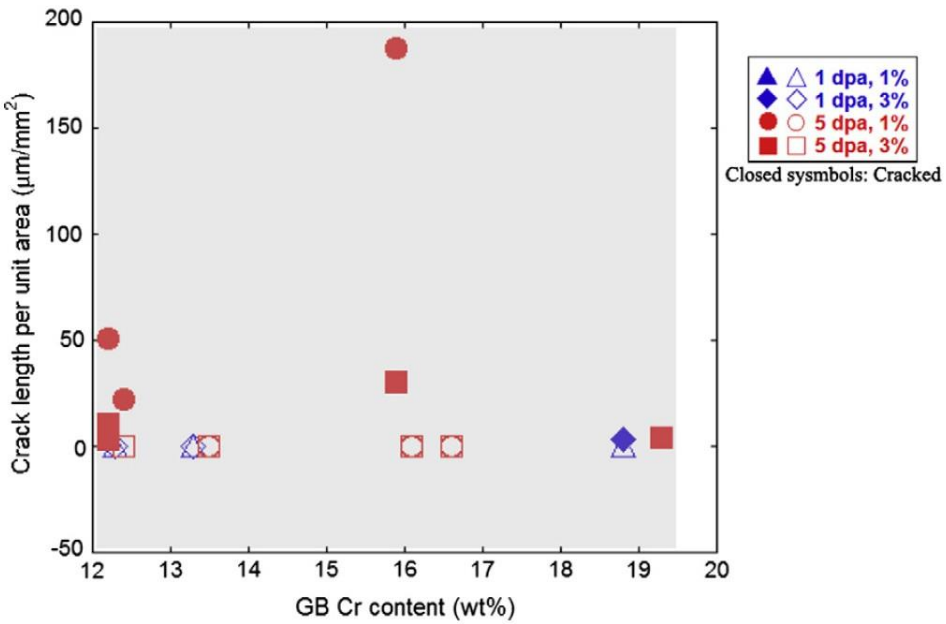
- proton irradiated, 1+5 dpa, 360 ˚C
- **BWR NWC,** 288˚C, 2 ppm DO
- SSRT, $4*10^{-7}s^{-1}$ up to strain 1+3%

Z. Jiao, G.S. Was, 2011, Journal of Nuclear Materials 408

Role of Radiation Induced Segregation

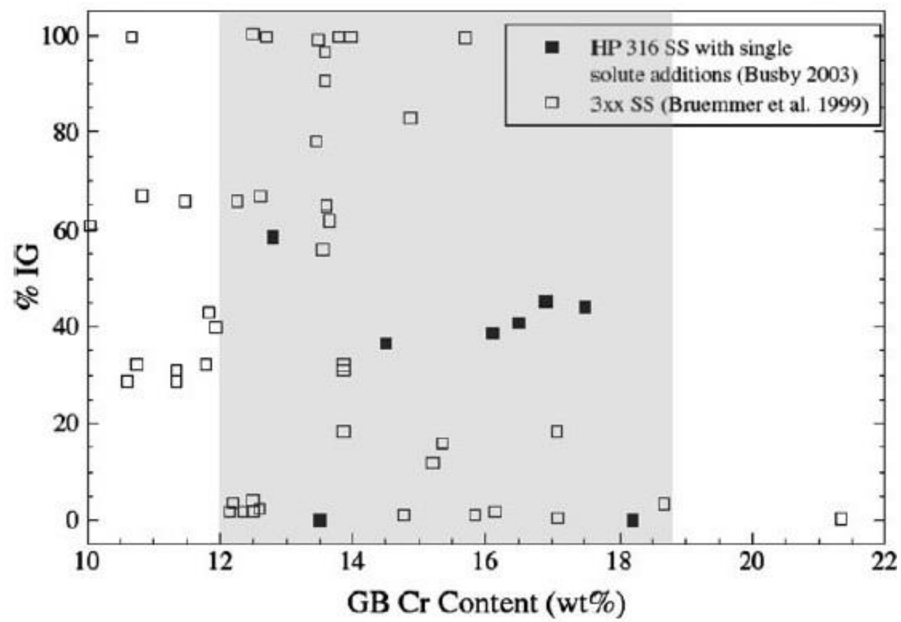
- RIS is not the only dominant factor of IASCC

Z. Jiao, G.S. Was / Journal of Nuclear Materials 408 (2011) 246–256



IG crack length vs. RIS → **S_IASCC** = 0

%IG vs. RIS → **S_IASCC** = 0.4



Z. Jiao, G.S. Was, 2011, Journal of Nuclear Materials 408

IASCC facts to need further attention

8. **Hydrogen/Helium** effects
   - BWR HWC: dose < 5-10 dpa → H decreases CGR
   - PWR: H content ↑ increases %IG

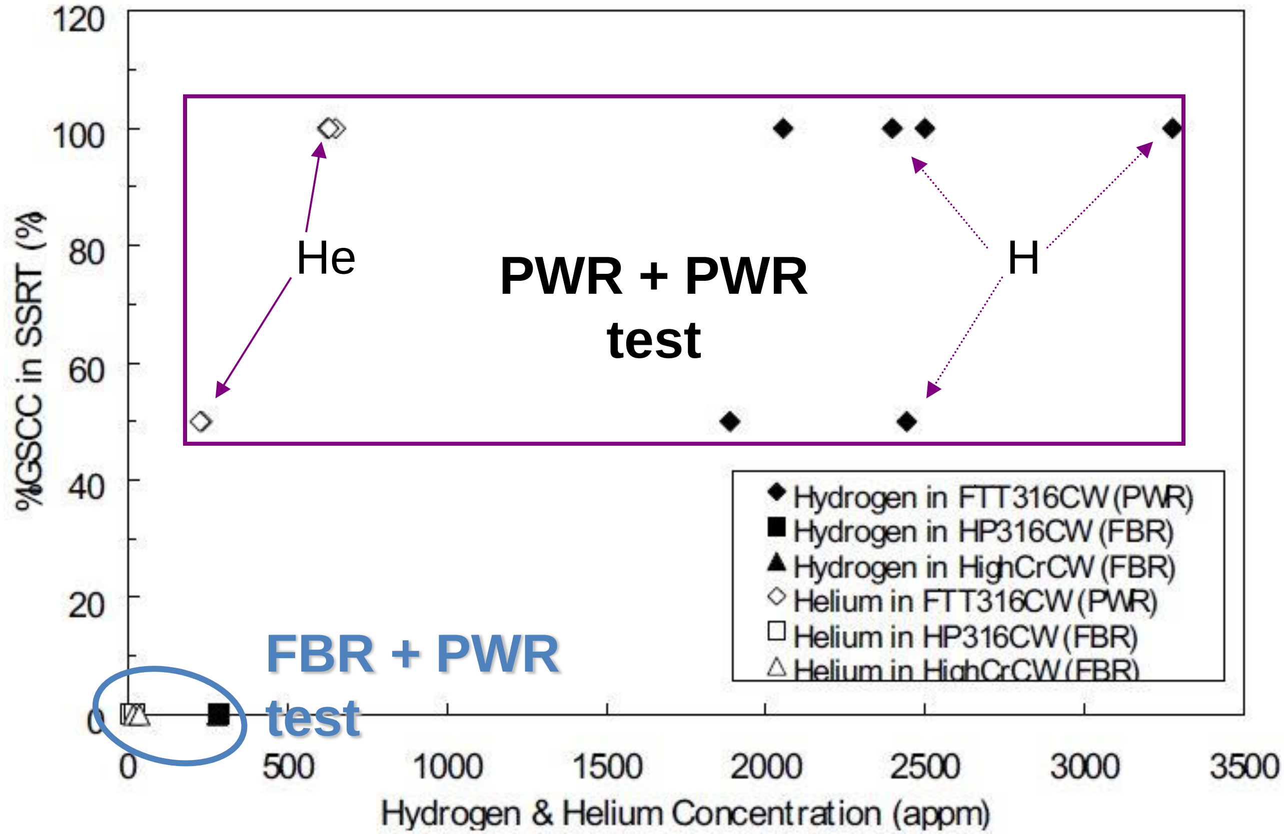
9. **Spectrum**: Fast vs. thermal reactor irradiation

10. **Threshold stress** for initiation (≈ 50% YS)

11. **Oxidation** & in-pile

IASCC vs. He and hydrogen content

SSs irradiated in a FBR revealed less H/He gas and less IASCC susceptibility compared to PWR irradiated.

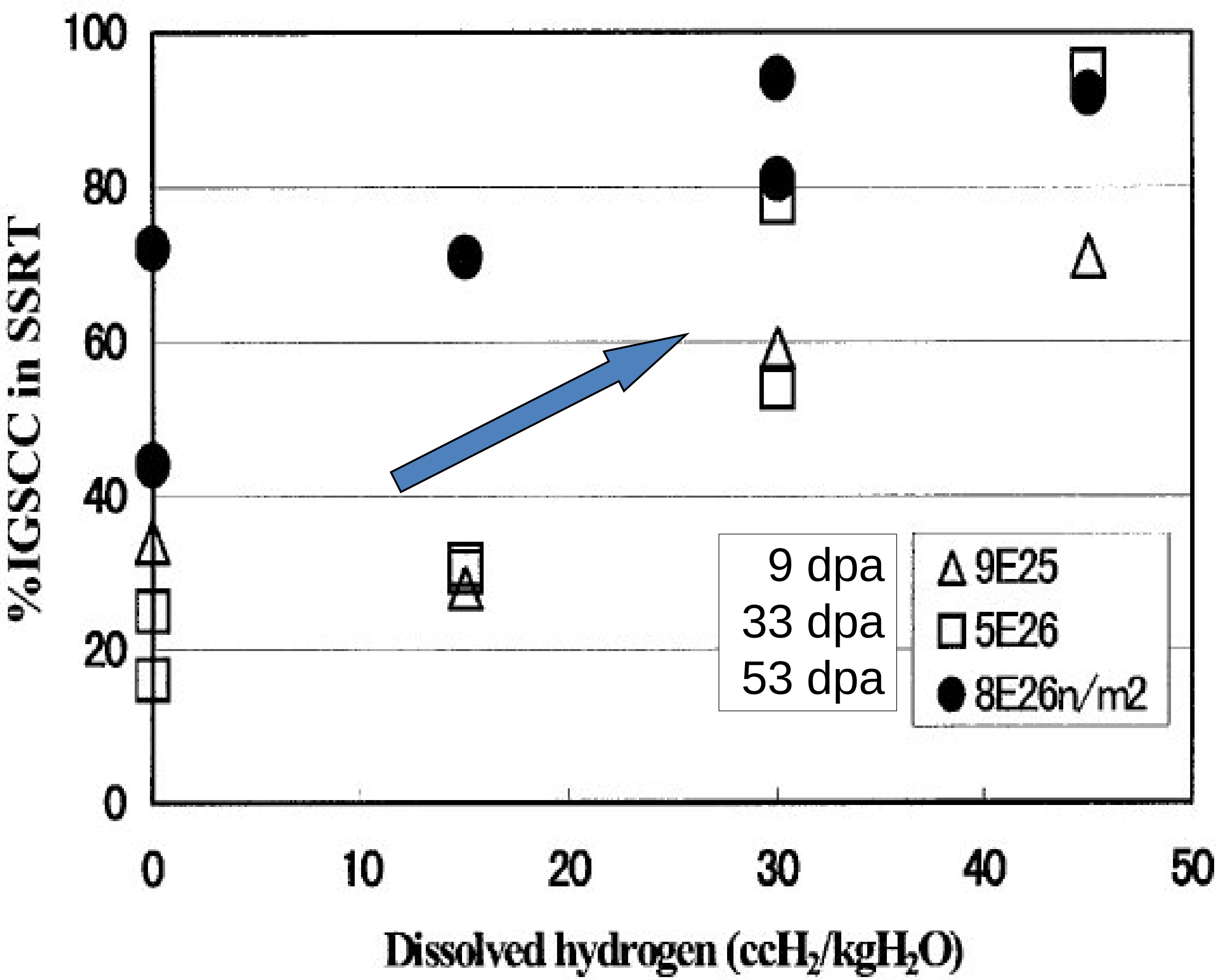


K. Fujimoto et al., 2005, 12th EnvDeg., Salt Lake City.

316CW:

- fast (FBR) reactor irradiation
- thermal (PWR) reactor irradiation - FTT
- SSRT tested
- 320 – 340˚C
- $1x10^{-7}$/s

Effect of **hydrogen in PWR**



A316CW:

- Flux Thimble Tube

  7, 33, 53 dpa
- PWR
- 340˚C
- 6.7*10^{-8} s^{-1}

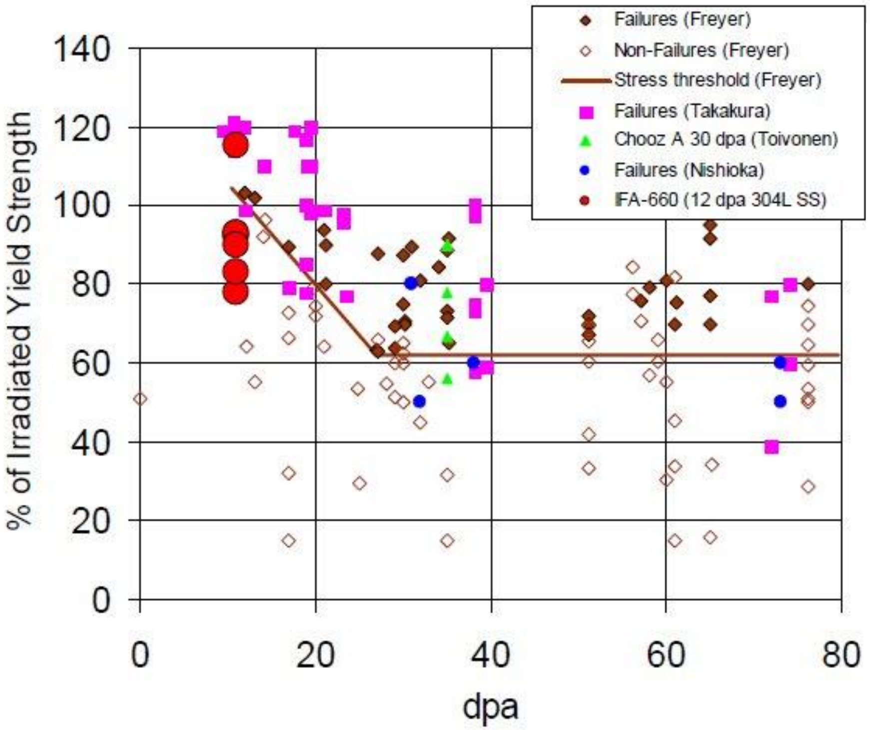
K. Fukuya et al., 2001, 10th EnvDeg.,Stevenson

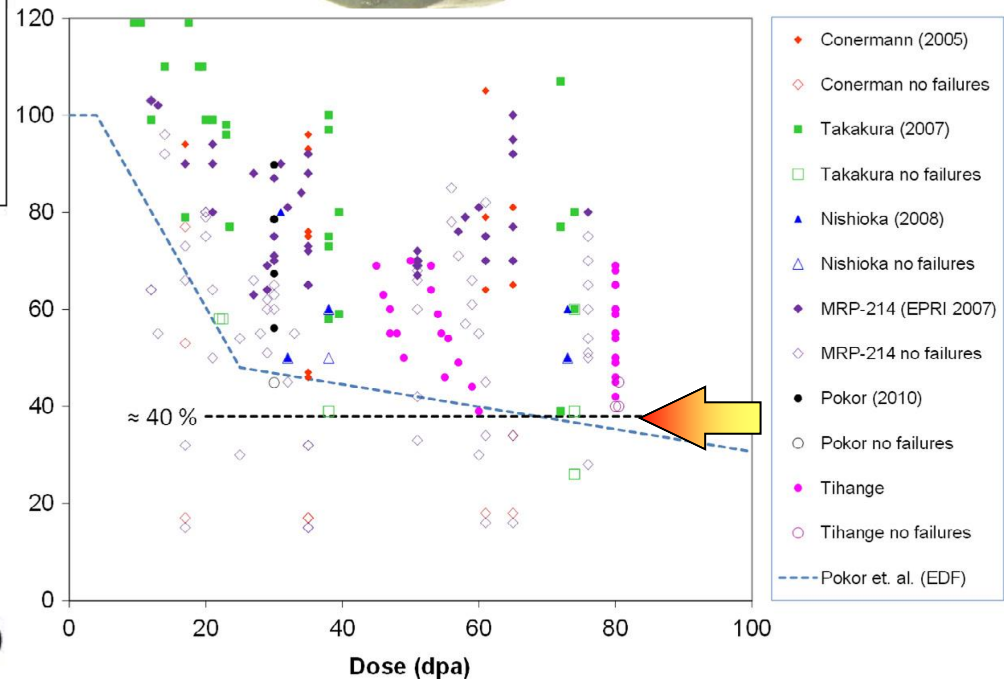
IASCC threshold stress

304L + 316CW SS

- In-service irradiated FTT&BFB;
- O&C ring, tensile; **PWR**, 320-340˚C

IASCC threshold stress 40 - 62% YS





O.K. Chopra, A.S. Rao, 2011, Journal of Nuclear Materials 409

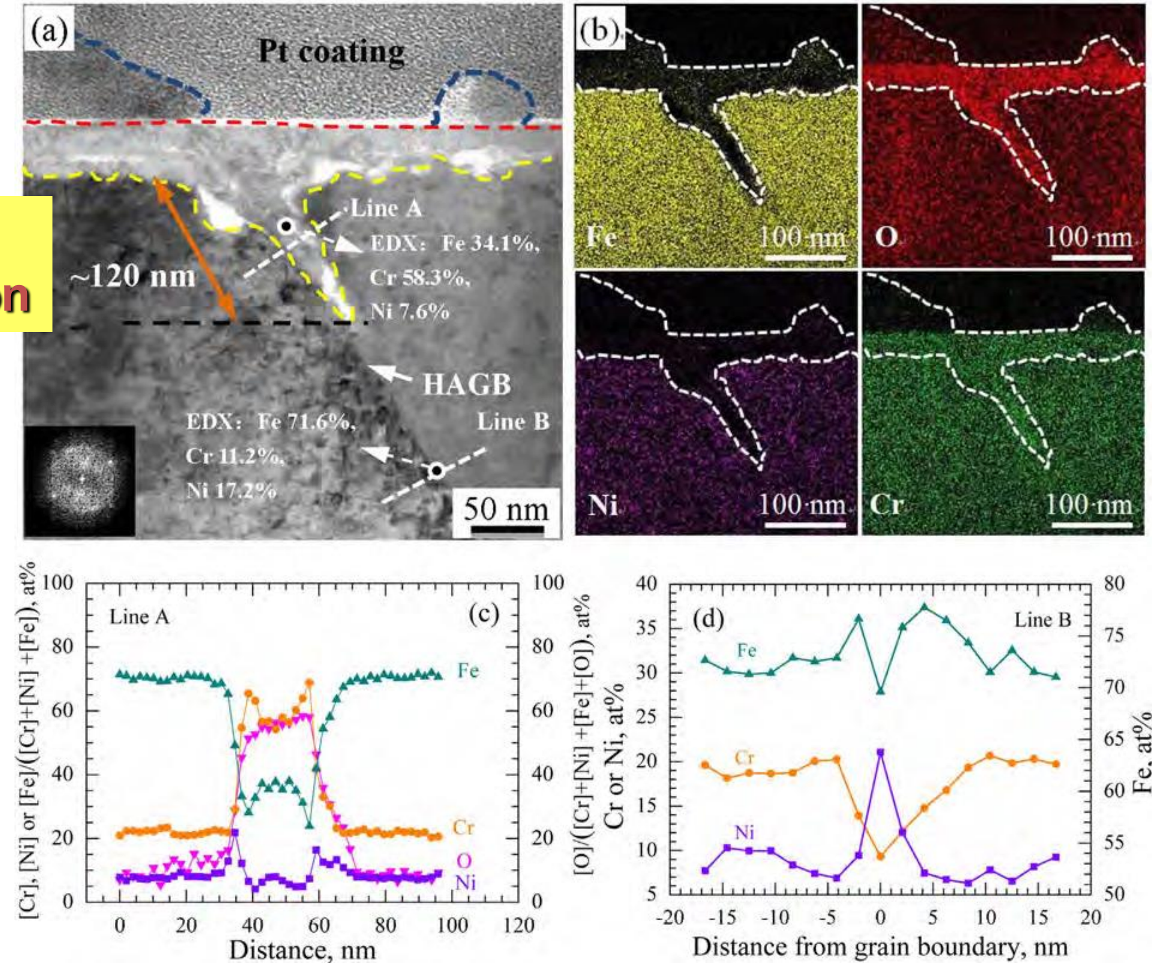
R.-W. Bosch et al, 2015, Journal of Nuclear Materials 461

P.D. Freyer et al. , 2007, 13th Environmental Degradation of Materials in Nuclear Power Systems, Whistler

Oxidation

- **304NG SS: localized corrosion at a GB** in 3-dpa protons irradiated(2 MeV, 360 °C, 6×10^{-6} s^{-1}); PWR, 320°C, 500 h

Gb oxidation

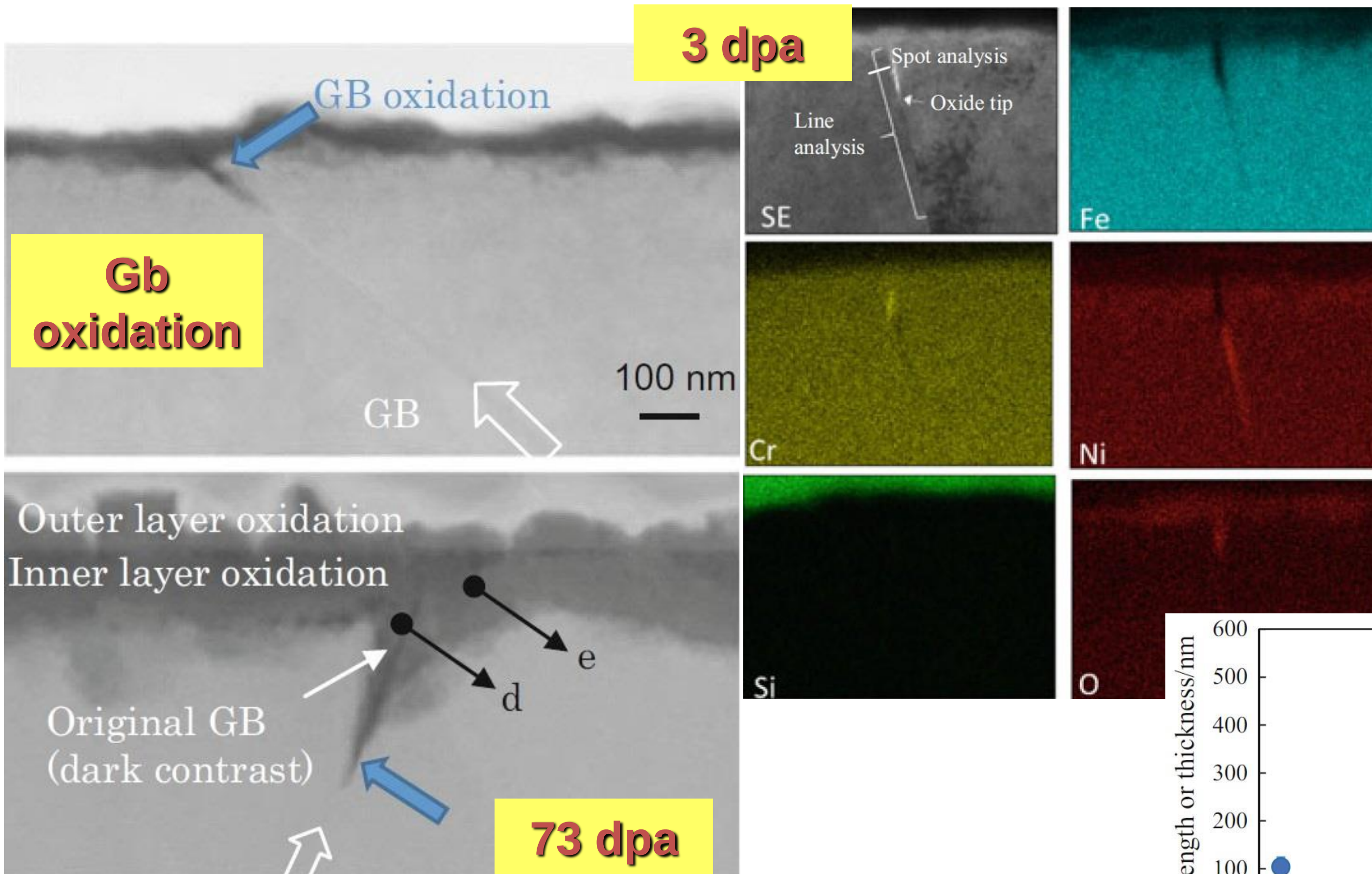


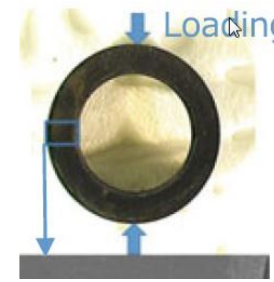
RIS

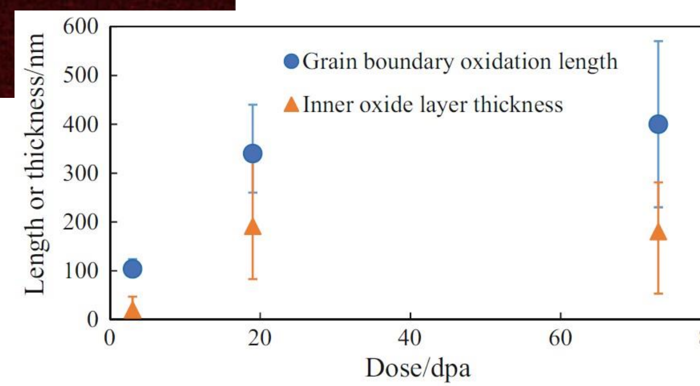
P. Deng et al., 2017, Corrosion Science 127

Oxidation

- **CW316 SS: localized GB oxidation** in 3-, 19- & 73-dpa neutron irradiated(in PWR); loaded in PWR, 320°C, 1150 h

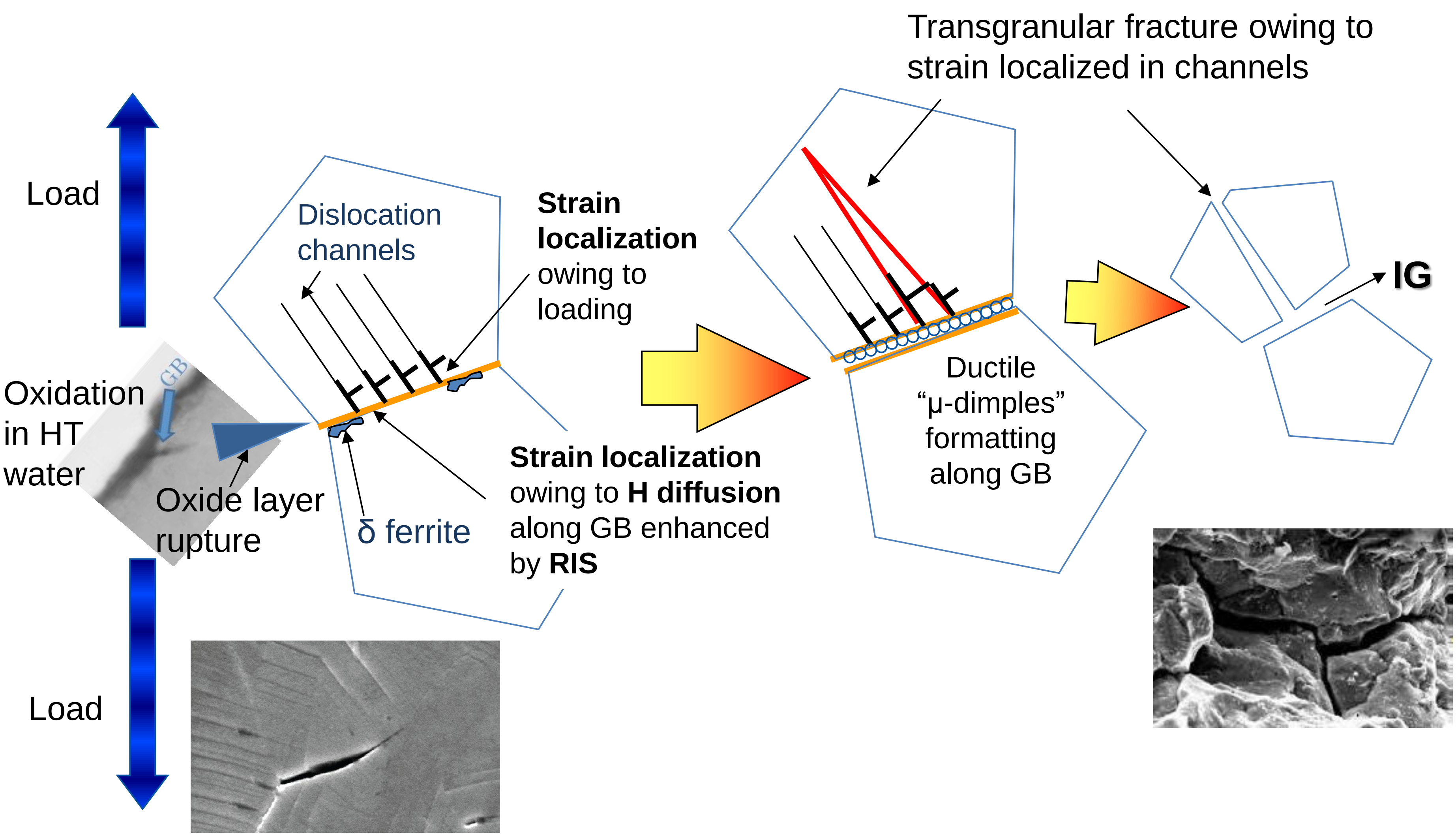






T. Fukumura et al., 2017, 18th EnvDeg., Portland

Concept of IASCC mechanism



Summary

- Mechanism is not fully understood yet as well as mitigation

- Evaluation of parameters influencing IASCC is needed for prediction and modeling

- Potential concern of life extension of Gen II+III plants
  - IASCC mitigation
    - Water chemistry
    - RVI annealing
    - RVI replacement with new components made from an alternative material

- Potential risk of damage of RVI of new built plants
  - Extensive research of alternative materials performance under irradiation/high T water effects
    - RAFM steels
    - Ferritic ODS steels
    - Titanium alloys
    - High strength Ni base alloys (Inconel 718, etc.)

Acknowledgment

This work was supported by the **SOTERIA** Project.

Thank you for your attention!

http://cvrez.cz
anna.hojna@cvrez.cz

IASCC - Laboratory Testing

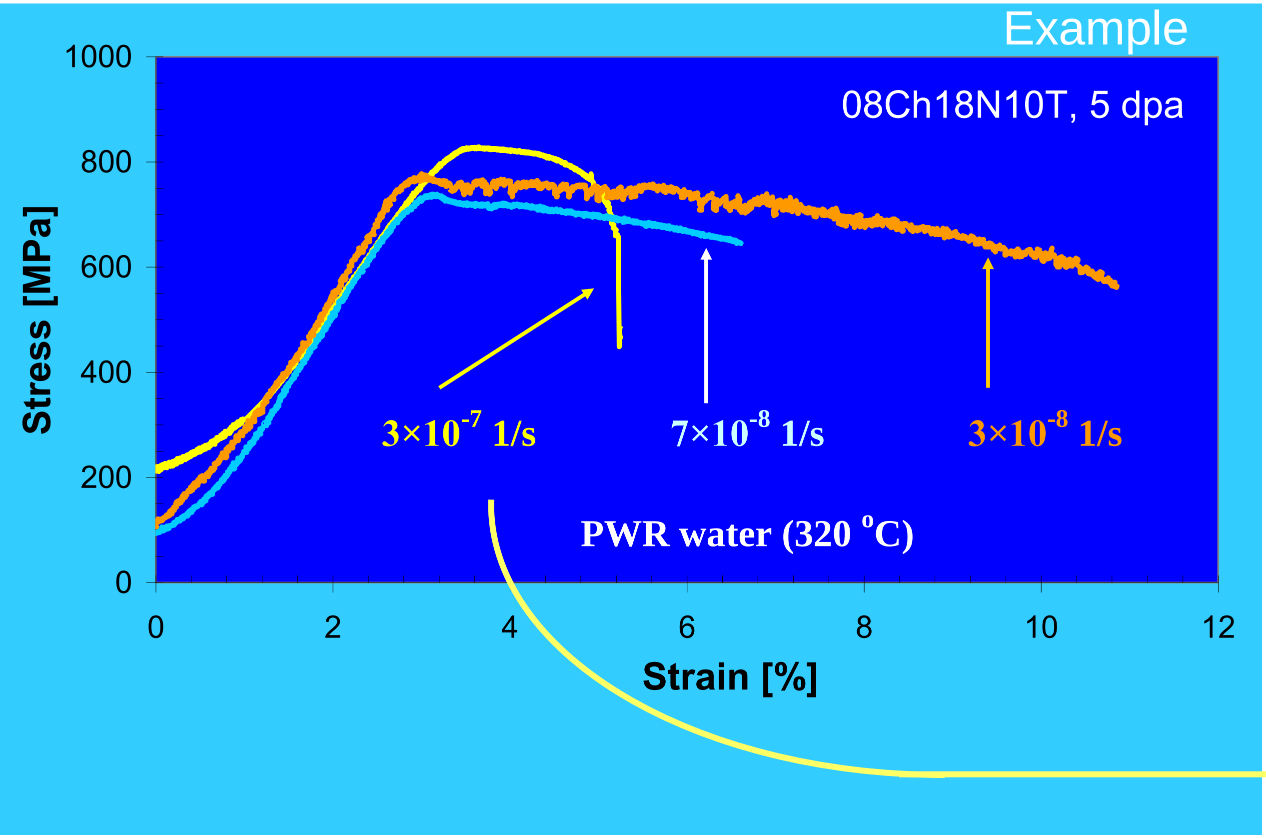
Three basic types of tests are used to obtain information about IASCC susceptibility and evolution of cracking in simulated BWR/PWR environment:

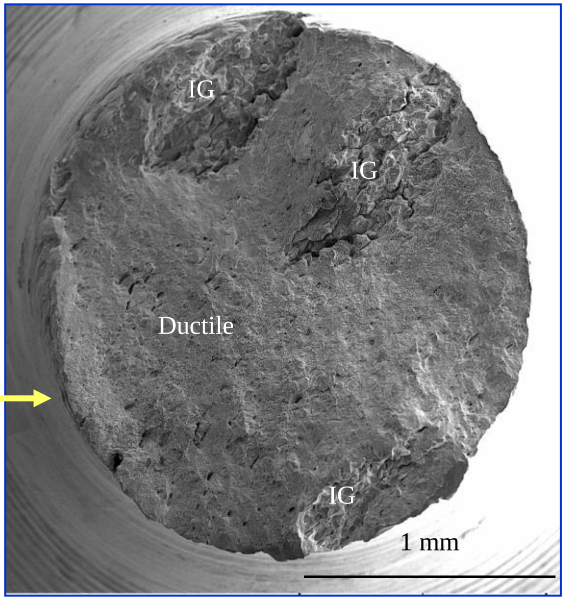
- **SSRT** Slow Strain Rate Test
- **CLT** Constant Load Test
- **CGR** Crack Growth Rate test

Materials can be irradiated with

- Neutrons (fast or PWR&BWR reactors' spectrum),
- Protons and
- Ions

Slow Strain Rate Test → susceptibility to **IASCC**



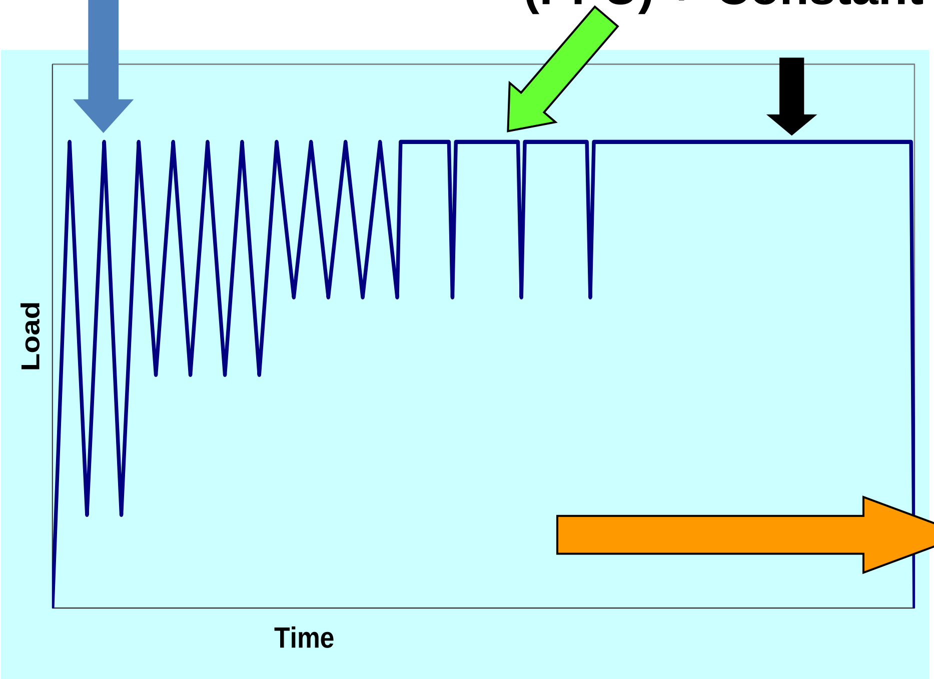


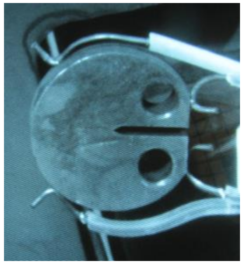
Crack Growth Rate test: evolution of existing crack

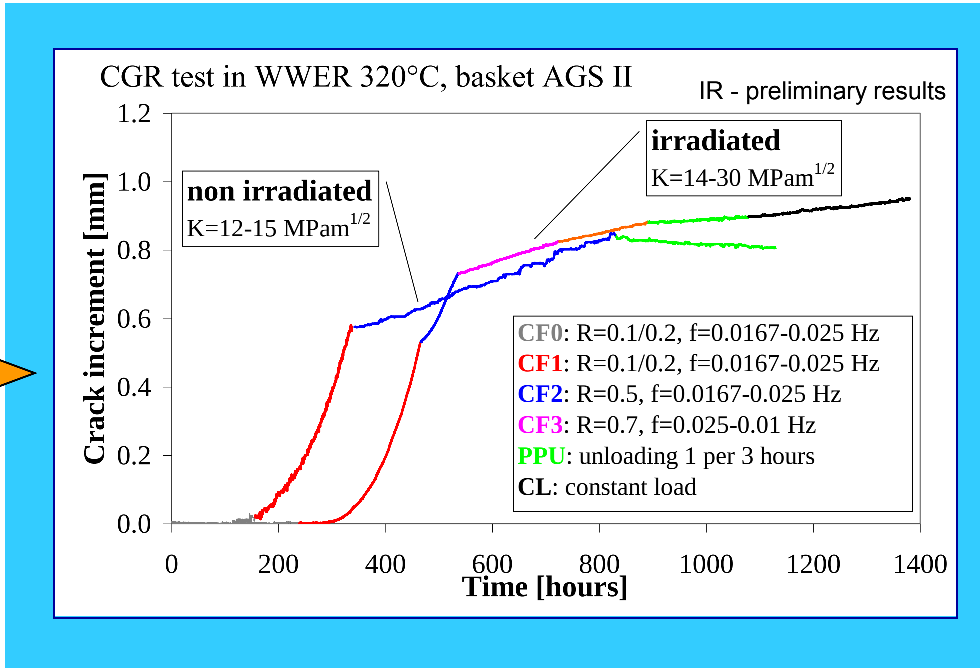
One specimen - two test phases: cyclic (CF) and constant load (CL)

CF: Pre-cracking

CL: Periodical Partial Unloading (PPU) + Constant Loading



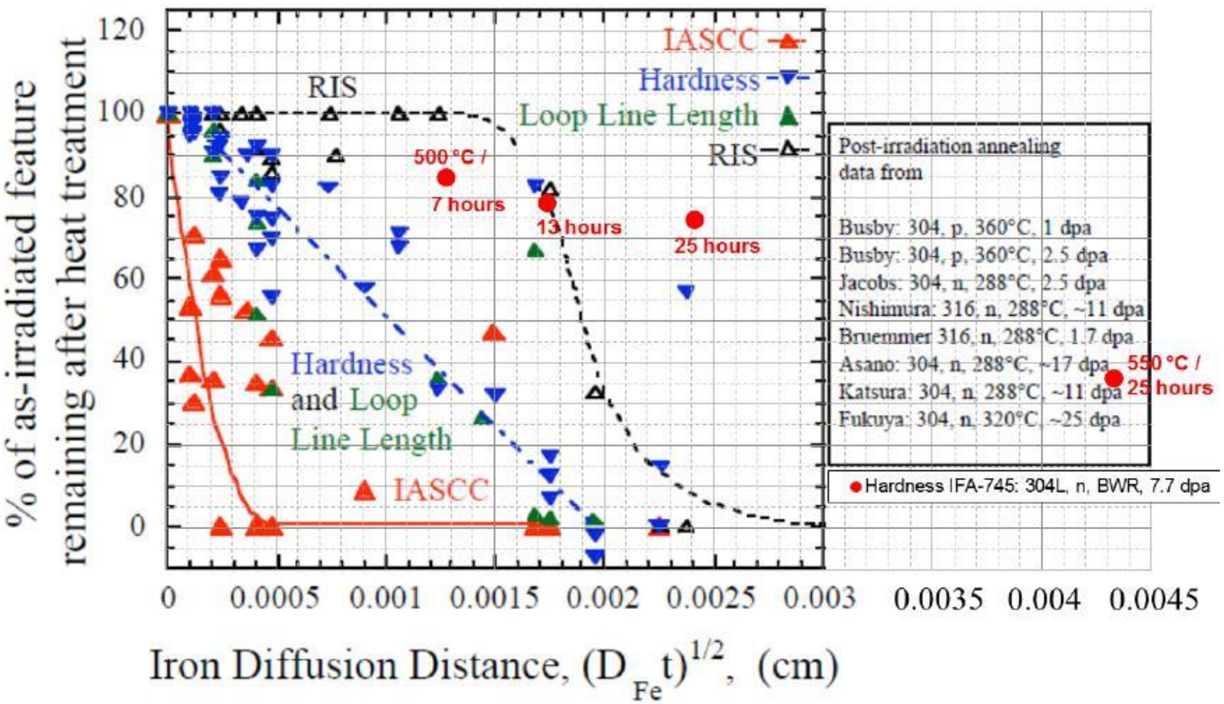




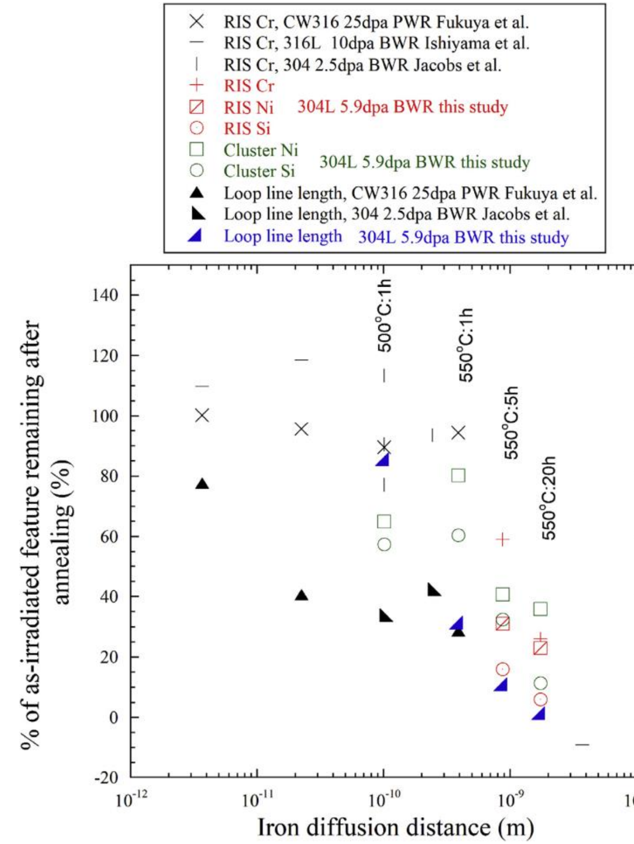
Several chemical regimes can be applied in one test

Post-irradiation annealing (PIA)

- Is PIA mitigating IASCC?



T. Karlsen, 2014 , IASCC review meeting, Halden



Z. Jiao, J. Hesterberg, G.S. Was, 2018, Journal of Nuclear Materials 500