GENERAL INTRODUCTION: INTERNALS

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Contents

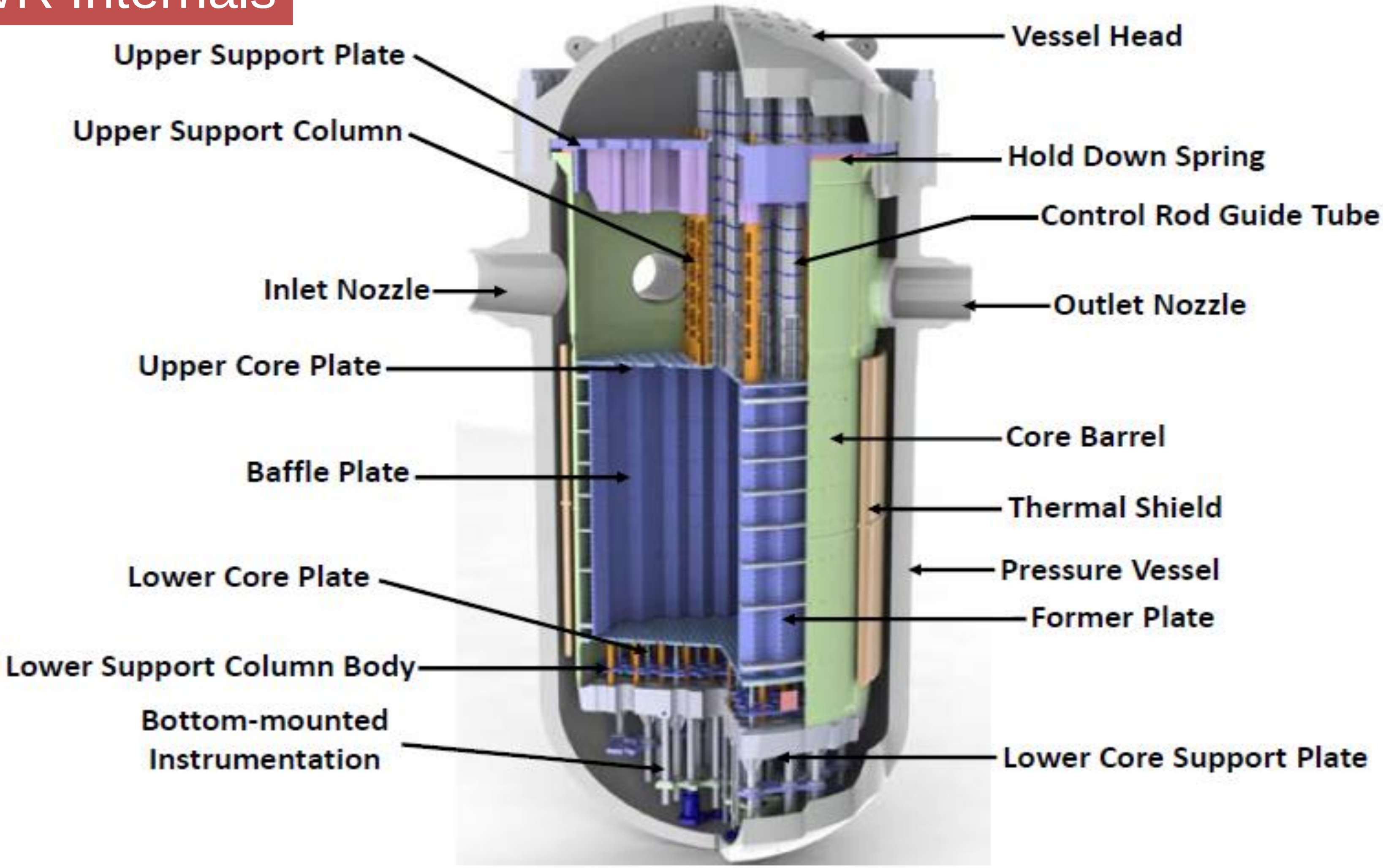
- Introduction
- Irradiated microstructure
- Ageing mechanisms

Introduction

- The function RPV Internals are to:
  - Provide support, guidance and protection for the reactor core;
  - Provide a passageway for the distribution of the reactor coolant flow to the reactor core;
  - Provide a passageway for support, guidance, and protection for control elements and in-vessel/core instrumentation; and
  - Provide gamma and neutron shielding for the reactor vessel
- This components are exposed to very harsh environments during their service.
  - High temperature water,
  - Stress,
  - Vibration, and
  - Intense neutron field.
- Degradation of materials in this environment can lead to reduced performance, and in some cases, unexpected early failure.

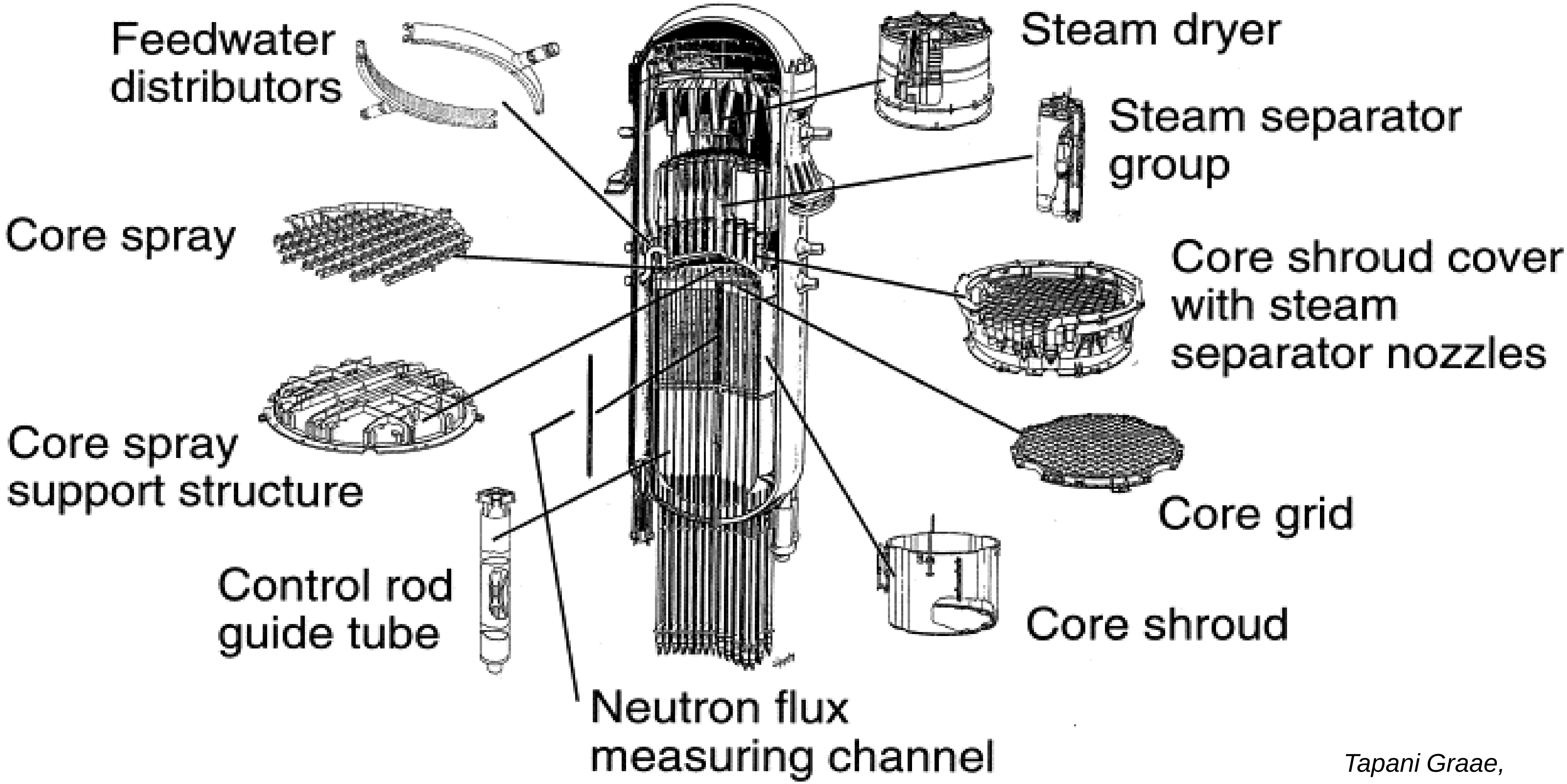
Introduction

PWR Internals



Introduction

BWR Internals



Tapani Graae,

RPVI - Materials

- Austenitic stainless steels and nickel (Ni)-base alloy steels are used for numerous core internal components in both BWRs and PWRs

Table 8.1. ASTM compositional specifications for austenitic stainless steel (304 SS, 304L SS, 316 SS, 316L SS, 316CW SS, 321 SS, 347 SS), A-286, and Ni-base alloys (600, 718, X-750) given in units of weight percent [1, 2]

Alloy	Carbon	Manganese	Chromium	Molybdenum	Nickel	Silicon	Phosphorus	Sulfur	Aluminum	Copper	Titanium	Iron	Other
304	0.08	2.0	18–20	-	8–10.5	1	0.45	0.03	-	-	-	Bal	-
304L	0.03	2.0	18–20	-	8–10.5	1	0.45	0.03	-	-	-	Bal	
316	0.08	2.0	16–18	2–3	10–14	1	0.45	0.03	-	-	-	Bal	
316L	0.08	2.0	16–18	2–3	10–14	1	0.45	0.03	-	-	-	Bal	
321	0.08	2.0	17–19	-	9–12	1	0.45	0.03	-	-	5× Cr min	Bal	-
347	0.08	2.0	17–19	0.75	9–13	1	0.45	0.03	-	0.5	-	Bal	
A-286	0.08	2.0	13–16	1–1.5	24–27	1	0.40	0.03	0.35	0.3	1.9–2.35	Bal	V, 0.1–0.5; boron (B), 0.001–0.1
600	0.15	1.0	14–17	-	72	0.5	-	0.015	-	0.5		6–10	
718	0.08	0.35	14–17	2.8–3.3	50–55	0.35	0.015	0.015	0.65–1.15	0.2–0.8	0.3	Bal	Niobium (Nb), 4.75–5.5; cobalt (Co), 1.0; B 0.006
X-750	0.08	1.0	14–17	-	70	0.5	-	0.01	0.4–1.0	.05	2.25–2.75	5–9	Co, <1; Nb+tantalum (Ta), 0.7–1.2

Expanded Materials Degradation Assessment Internals Piping ML14279A331

Introduction

- The core of a nuclear reactor is an extreme environment consisting of
  - High temperature water, ~270 °C (518 °F) to 340 °C (644 °F).
  - Imposed service stresses and strains,
  - Corrosive medium (span oxygenated to hydrogenated water) and
  - Intense radiation fields

Introduction

- Internals aging degradation mechanisms

- Irradiation Embrittlement
- IASCC
- Void Swelling
- Wear
- Fatigue & Corrosion fatigue
- Thermal ageing

Introduction

Irradiation embrittlement

Hardening

Loss of toughness

Irradiation creep

Void Swelling

P
W
R

RADIATION INDUCED MICROSTRUCTURE

IASCC

Introduction

Irradiation embrittlement

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R

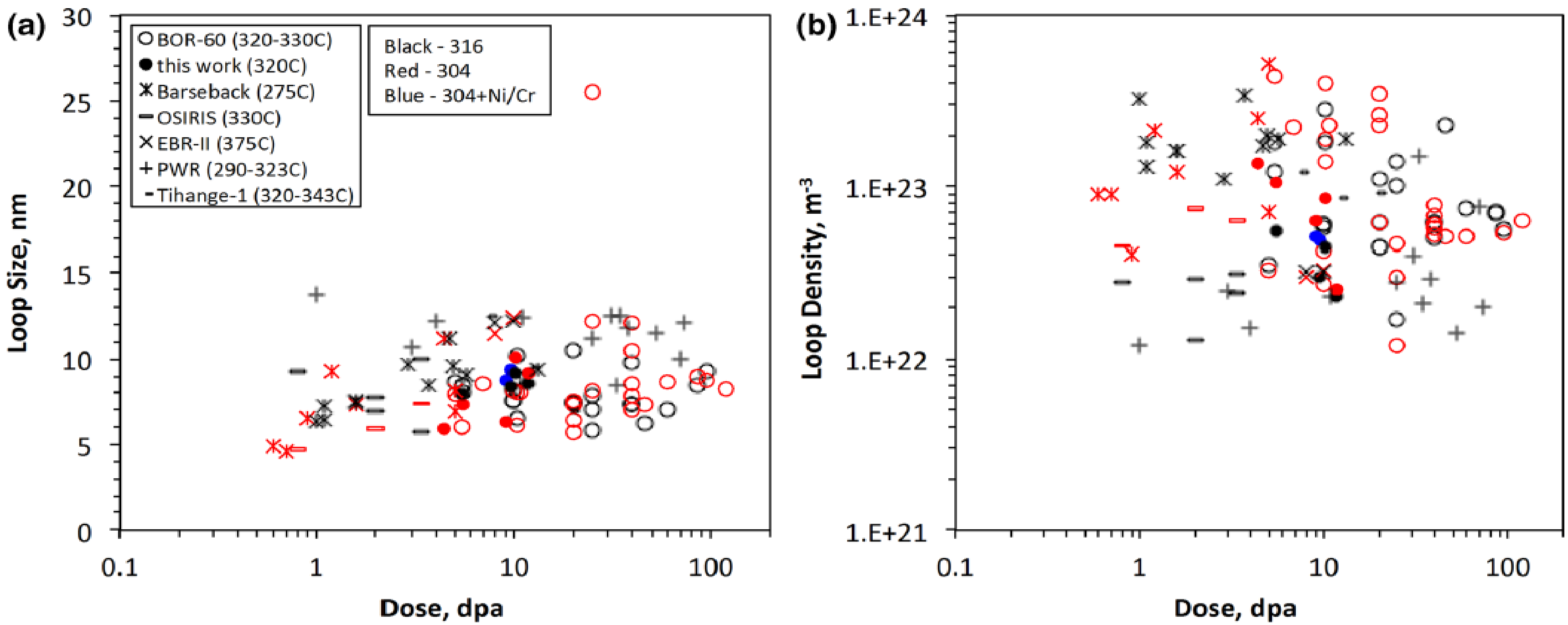
RADIATION INDUCED MICROSTRUCTURE

IASCC

Irradiated microstrucutre

❑ Frank loops have a primary contribution to radiation hardening

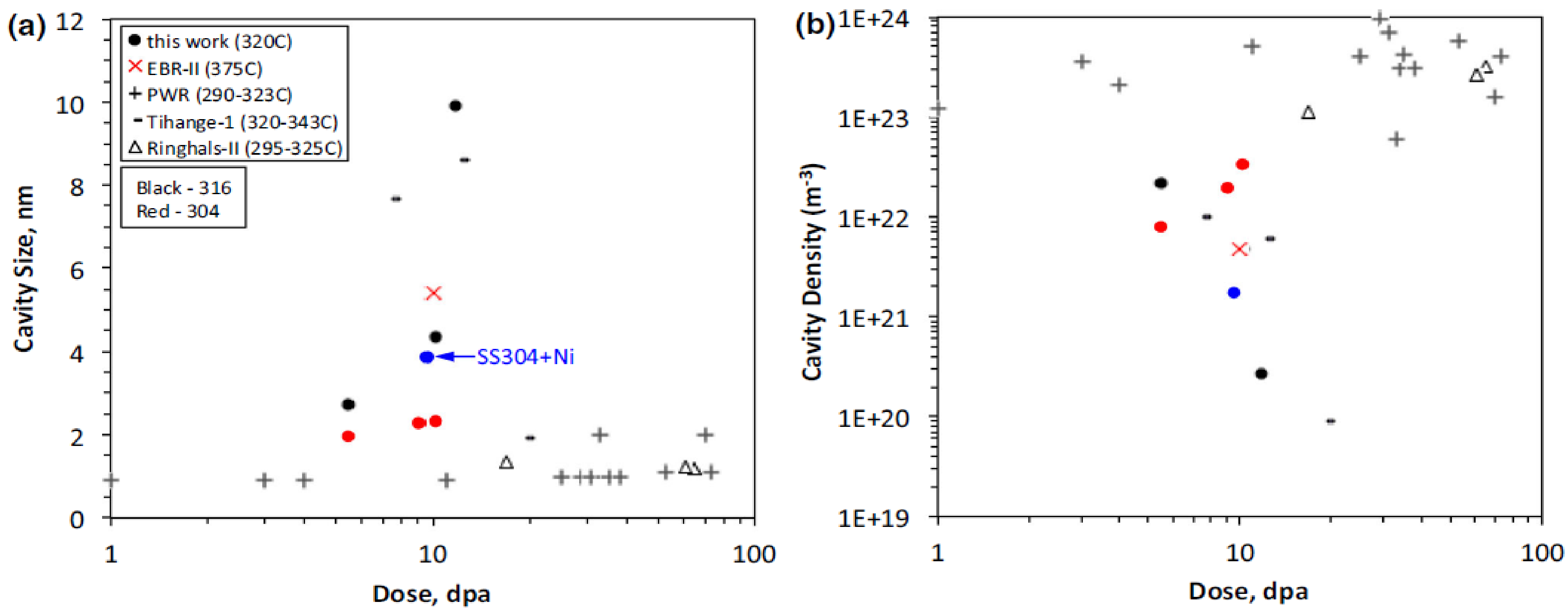
- data from LWRs generally exhibit loop size saturation at lower doses or larger loop size at the same dose than fast reactors
- The large data variations are likely attributable to the different irradiation temperatures (275–375C), dose rates, and alloy compositions.



Tan, Stoller, Field, Yang, Nam, Morgan, Wirth, Gussev, and Busby JOM, 2016

Irradiated microstrucutre

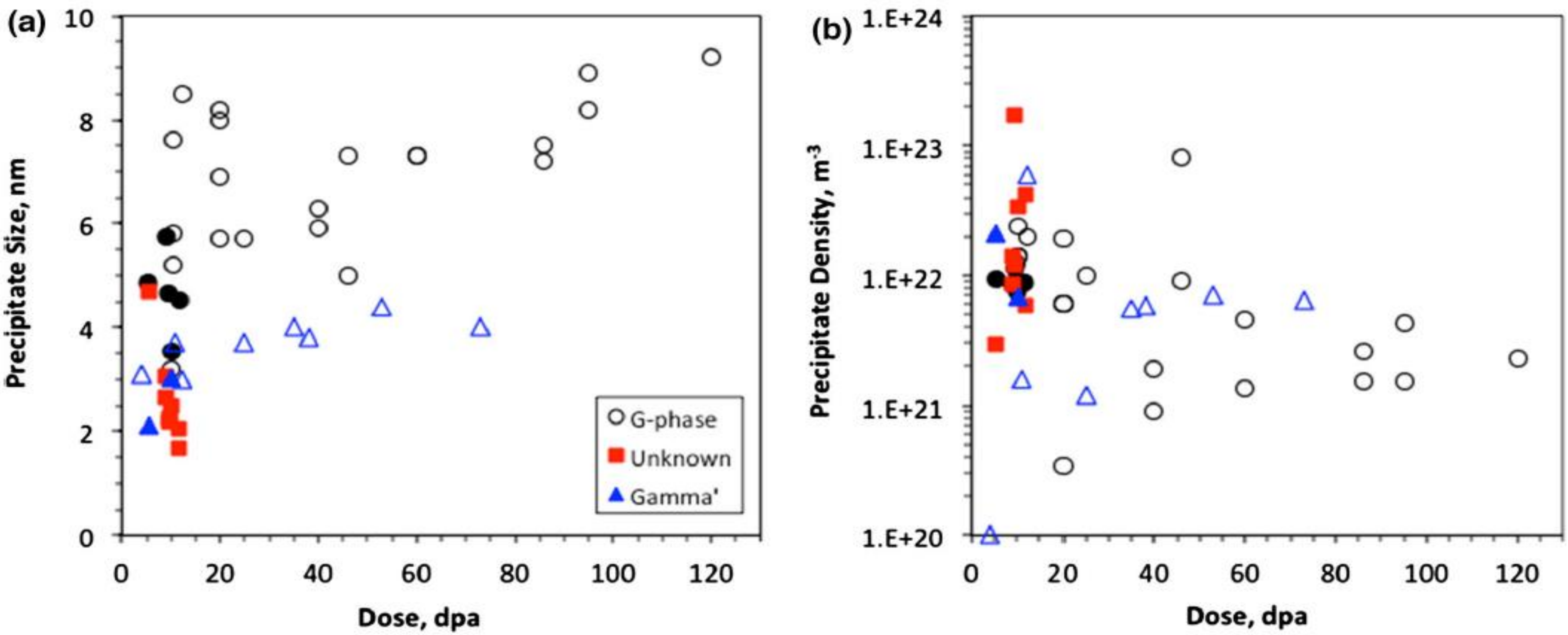
- Ultra-fine cavities were observed in some alloy samples.



Tan, Stoller, Field, Yang, Nam, Morgan, Wirth, Gussev, and Busby JOM, 2016

Irradiated microstrucutre

- Phase instability and formation of secondary phases (both radiation-induced and radiation-enhanced)



Tan, Stoller, Field, Yang, Nam, Morgan, Wirth, Gussev, and Busby JOM, 2016

Irradiated microstrucutre

- Irradiation induced clusters

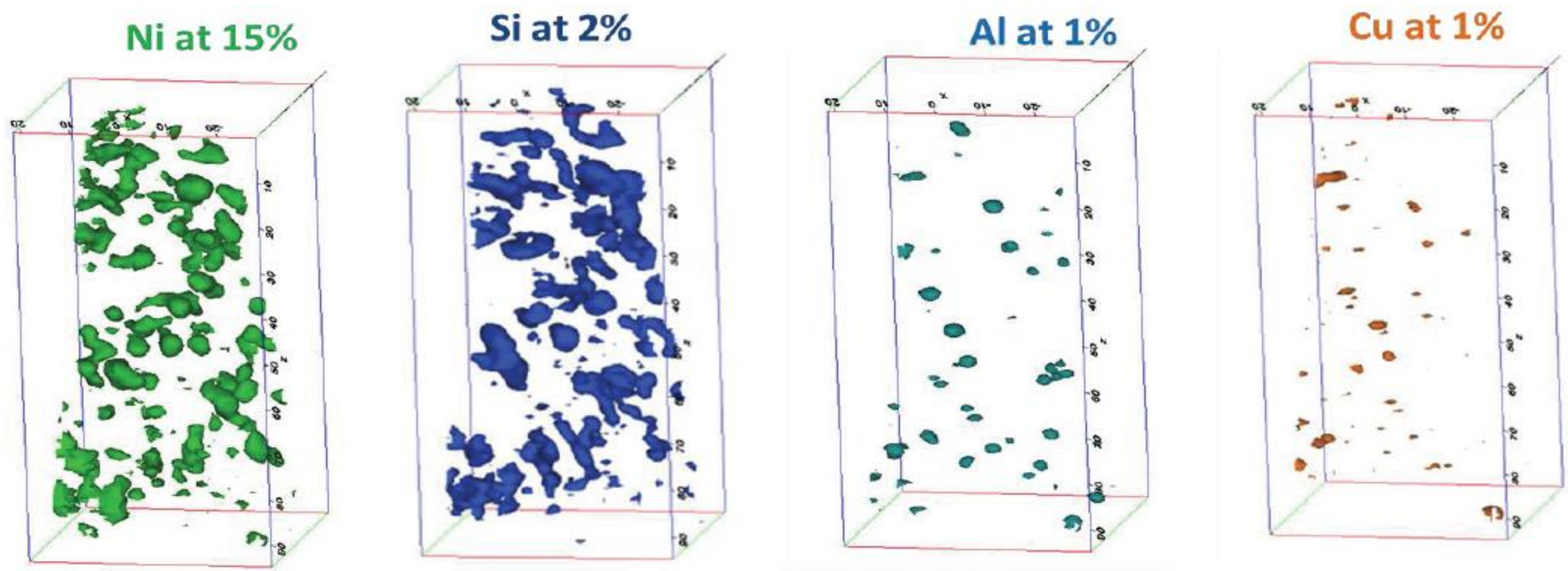
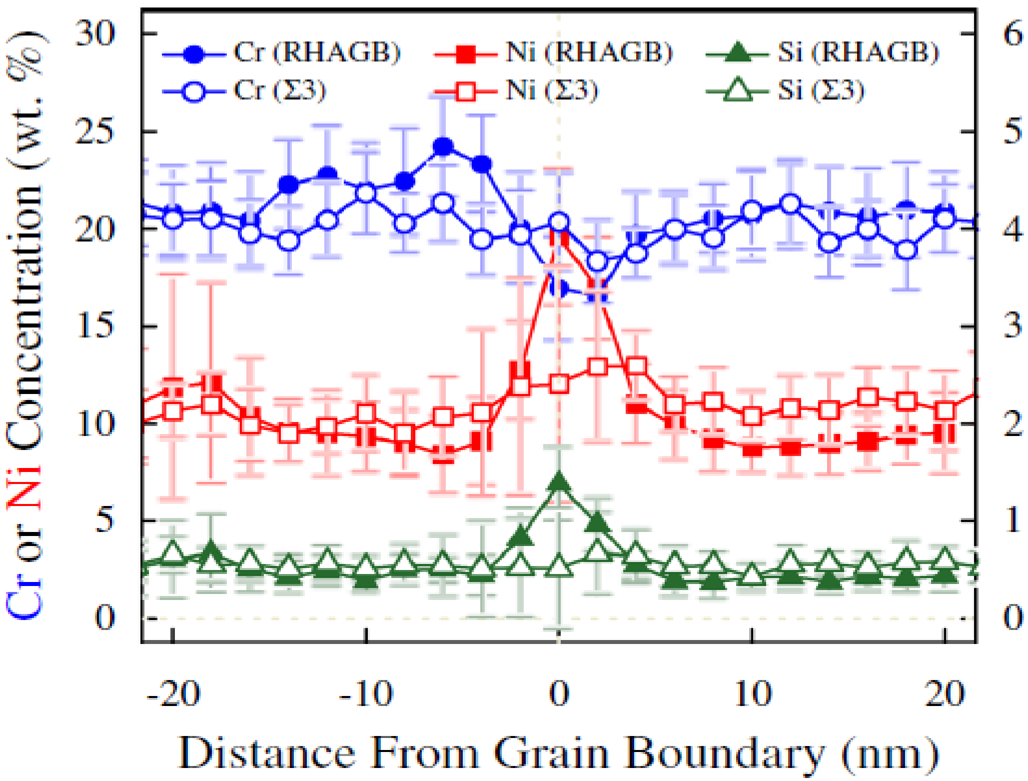


Figure 4-5. Ni-Si and Al-Cu clusters as observed in 304L SS irradiated to 5.9 dpa in BWR. Clusters are shown using isoconcentration surface plots from APT atom maps.

Gary S. Was LWRs

Irradiated microstructure

- Radiation induced segregation



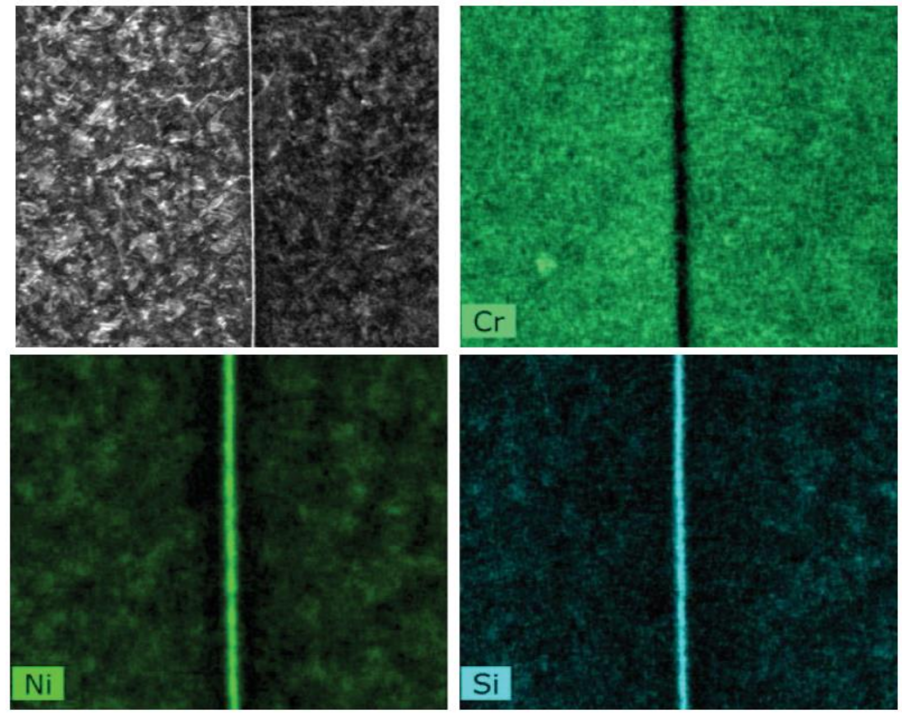


Figure 4-7: Radiation-induced segregation in 304L SS irradiated to 5.9 dpa in BWR as revealed by ChemiSTEM image. Depletion of Cr and enrichment of Ni and Si are evident.

Gary S. Was LWRs

Introduction

Irradiation embrittlement

Hardening

Loss of toughness

Irradiation creep

Void Swelling

PWR

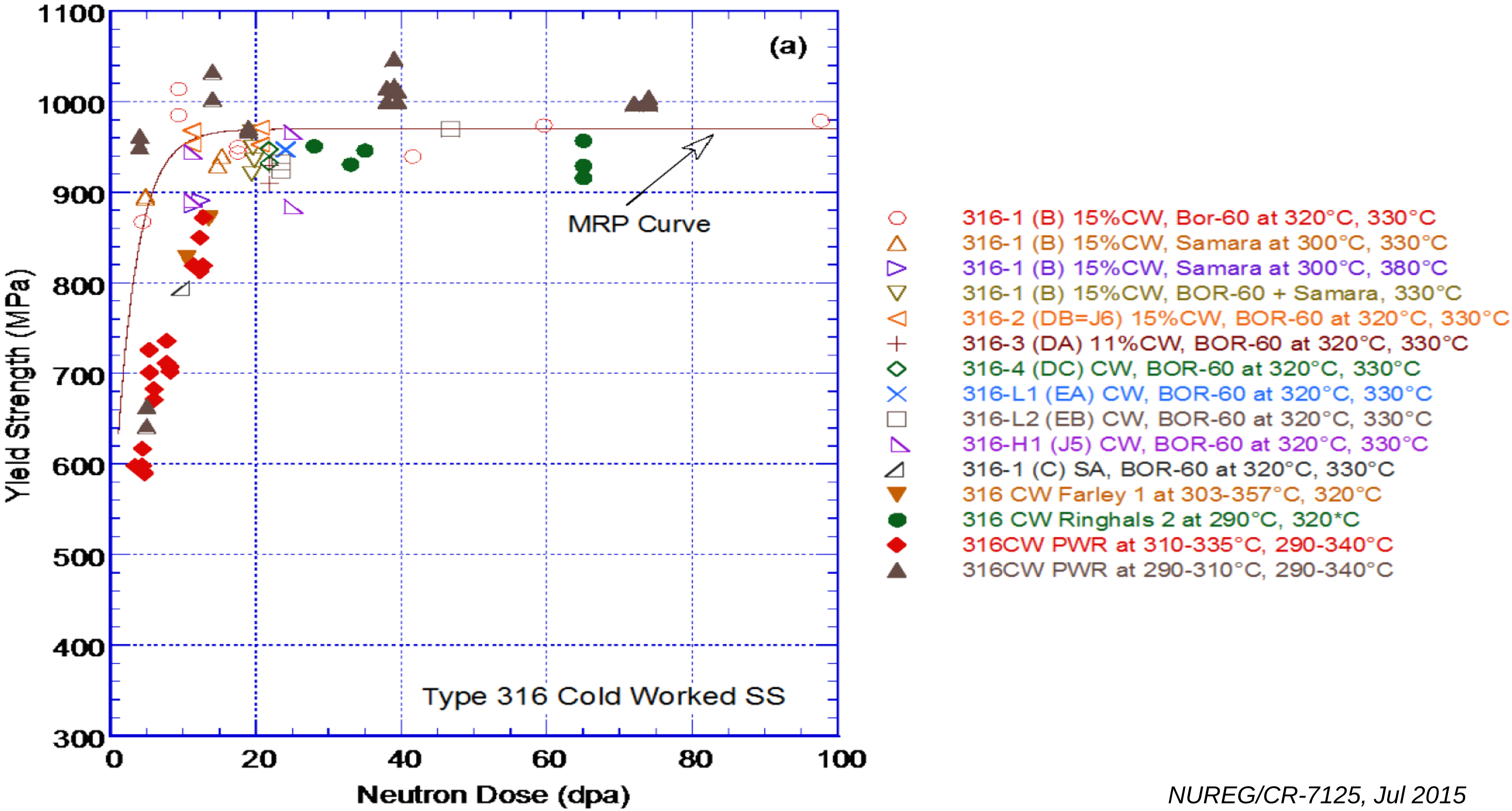
RADIATION INDUCED MICROSTRUCTURE

IASCC

Irradiation Embrittlement

- When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-based alloys can be changed.
- Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility and a loss of fracture toughness.
- The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence.
- While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

Irradiation Embrittlement



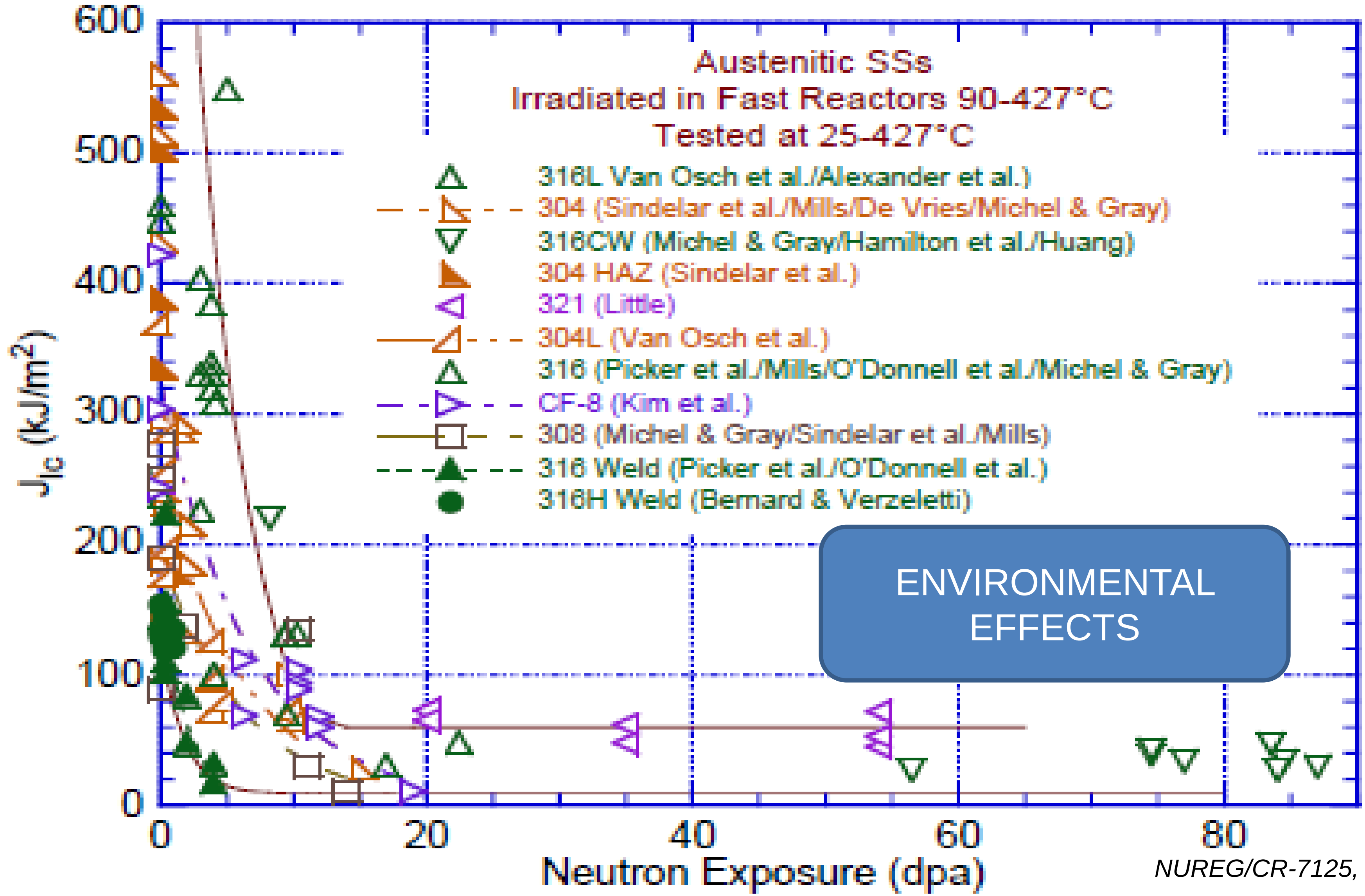
NUREG/CR-7125, Jul 2015

Reduction of fracture toughness

- Age related degradation mechanisms that may lead to a reduction of fracture toughness of internals:
  - Thermal ageing embrittlement
  - Irradiation embrittlement
  - Void swelling (>10%) PWR

- Environmental effects should be taken into account

Irradiation Embrittlement



ENVIRONMENTAL EFFECTS

NUREG/CR-7125, Jul 2015

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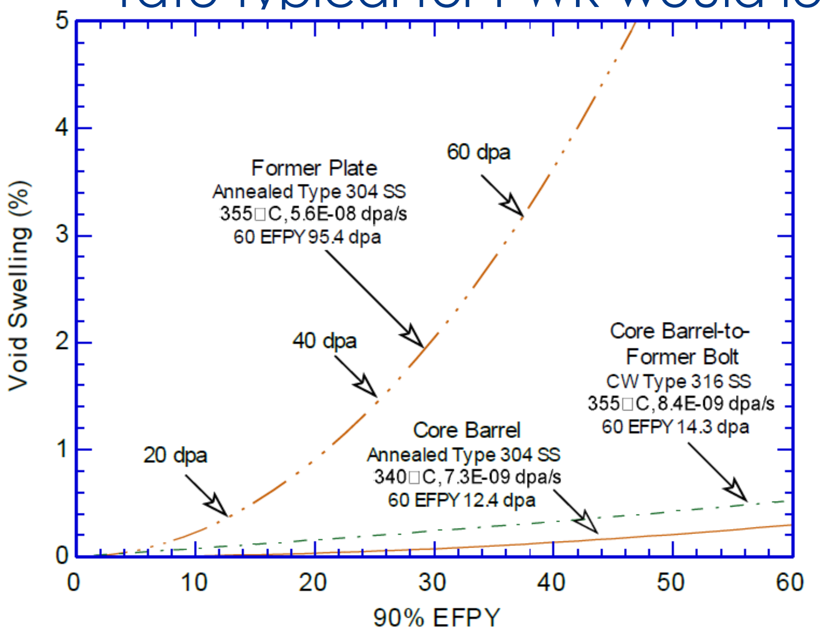
PWR

RADIATION INDUCED MICROSTRUCTURE

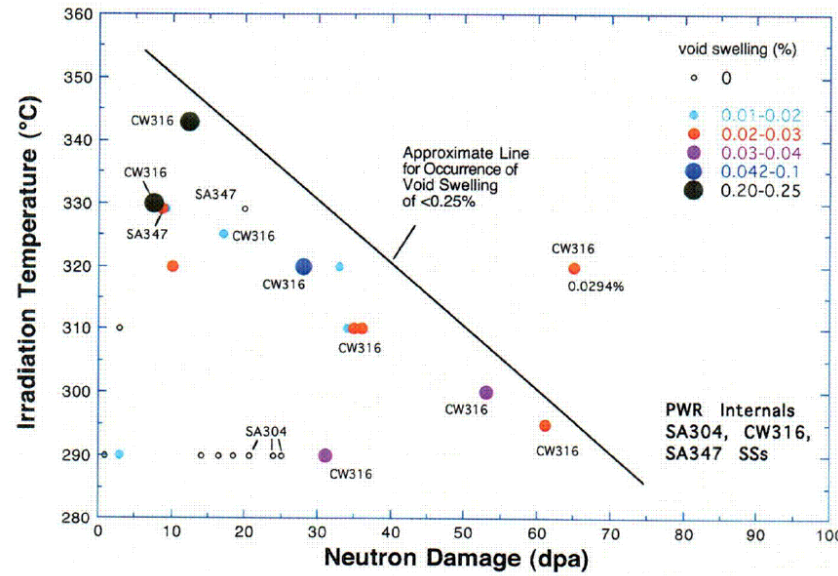
IASCC

VOID SWELLING

- Void swelling was not considered to be and important degradation mechanism for RPV internals in the design, mainly due to the lower temperature and neutron dose in comparison to fast reactor where swelling can be an important life limiting issue.
- Recent evidences point out that local gamma heating could increase the temperature of RPV internals up to 370°C and that the low dose rate typical for PWR would lead to a higher swelling level



Void swelling of PWR materials plotted as a function of effective full power years [NUREG/CR-7027]



Range of irradiation temperature and dose for which void swelling data (in color code) have been reported for PWR core internals There is no conclusive evidence that void swelling plays an important role in IASCC of PWR baffle bolts. NUREG/CR-6897 (2006)

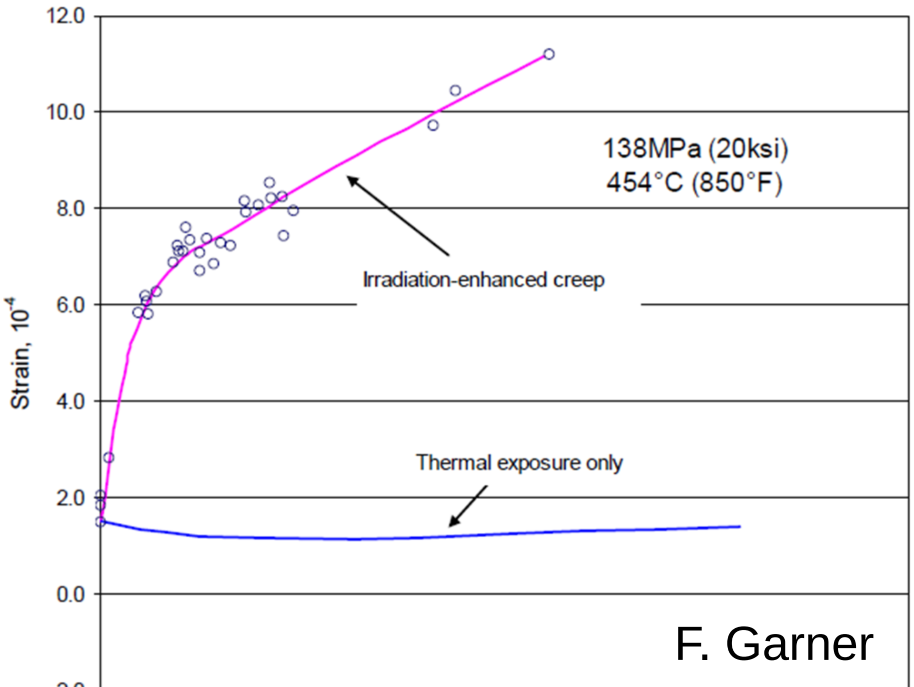
VOID SWELLING

- Changes in dimension due to void swelling could lead to loss of component function
  - Orientation, guidance and protection of the control element assemblies
  - Distribution of the coolant flow
  - Support, guidance and protection of in-vessel core instrumentation

EPRI MRP 175

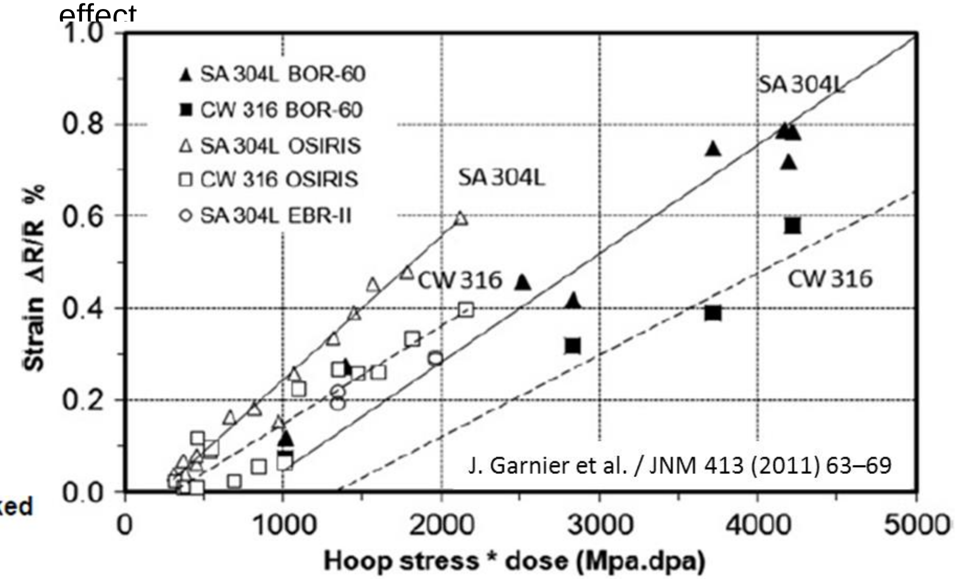
Irradiation creep

- Irradiation creep is important in the integrity and functionality assessment of internals, mainly in a synergy way with void swelling and the stresses generated by swelling.
- Bolted connections that are exposed to sufficient neutron fluence for irradiation creep / irradiation-enhanced stress relaxation to occur may become susceptible to fatigue



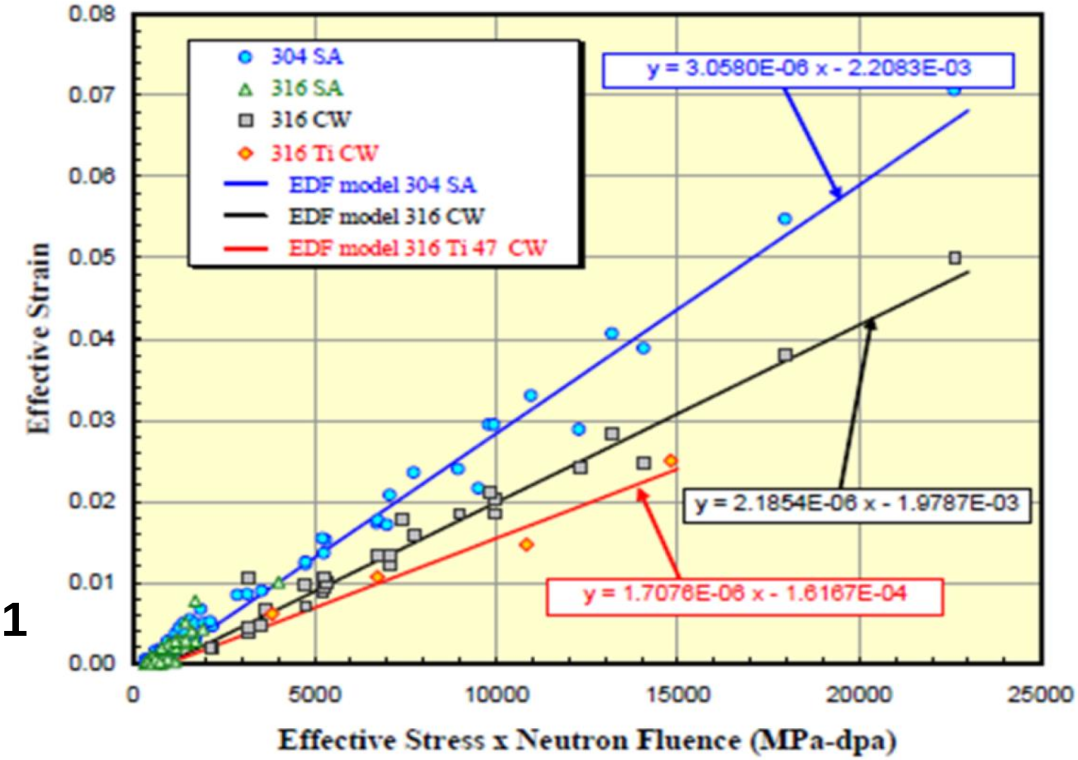
Comparison of Thermal Creep and Irradiation-Enhanced Creep of a 20% Cold-Worked Type 316 Stainless Steel Irradiated in EBR-II[H-5]

Creep in OSIRIS irradiation conditions is slightly faster than in BOR-60, and a large reduction in the incubation threshold is also noted. This difference can be due either to a spectrum or to a flux effect



Irradiation creep

- Thermal and irradiation creep/stress relaxation (ISR/IC)
- Neutron fluence, temperature and the degree of preloading are key parameters
- Correlations still indicate that a greater creep rate occurs for Type 304 SA material than for Type 316 CW material



MRP-211, Revision 1

Introduction

Irradiation embrittlement

Hardening

Loss of toughness

Irradiation creep

Void Swelling

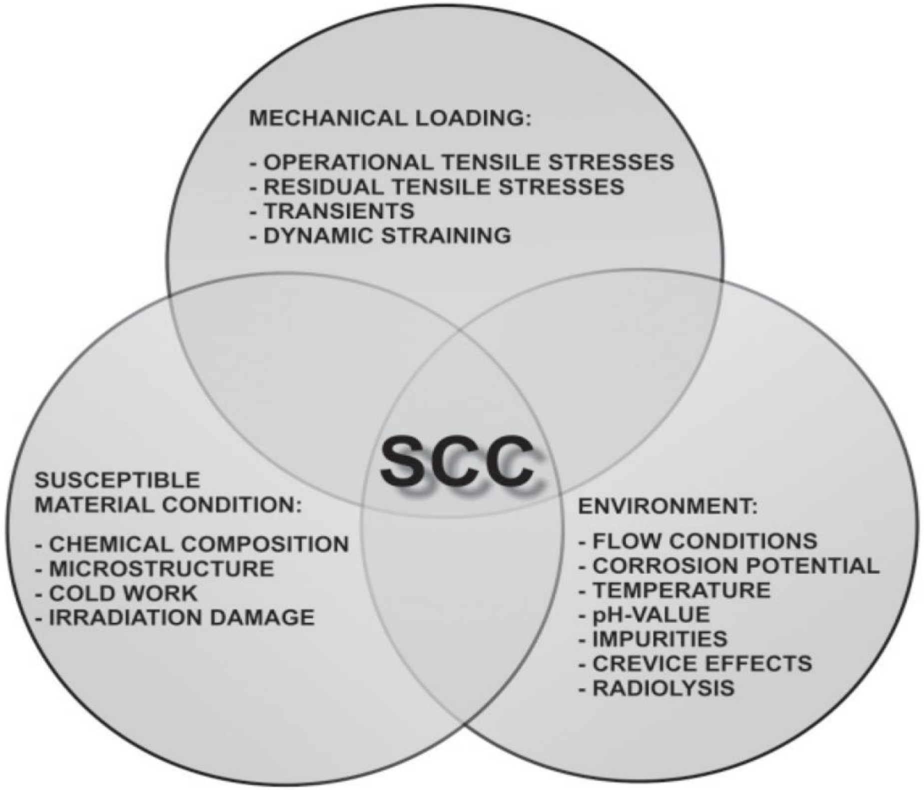
PWR

RADIATION INDUCED MICROSTRUCTURE

IASCC

IASCC

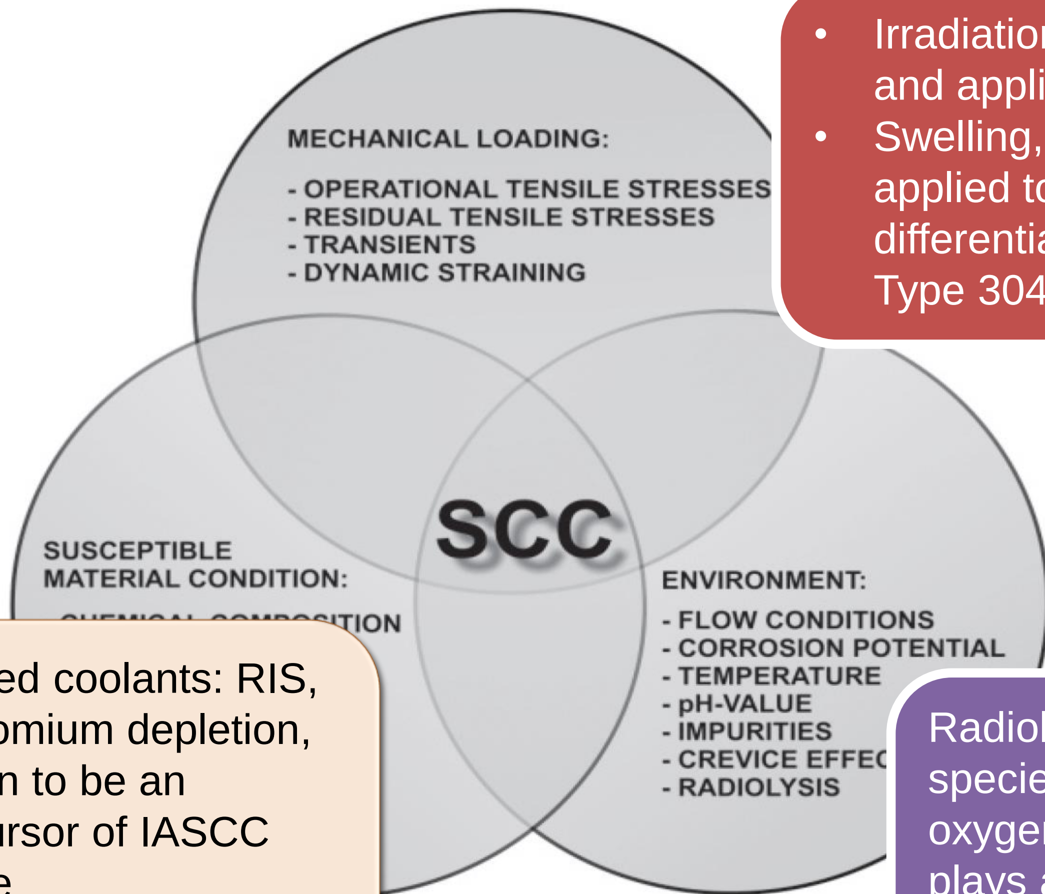
- Stress Corrosion Cracking (SCC)
  - Stress corrosion cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment and metallurgical properties.
  - The actual mechanism that causes SCC involves a complex interaction of loading, environmental and metallurgical factors.
  - The aging effect is cracking.



IASCC

- Intergranular Stress Corrosion Cracking (IGSCC)
  - Intergranular Stress Corrosion Cracking (IGSCC) in austenitic piping was a major issue for Boiling Water Reactors (BWRs) in the 1980s
  - Susceptibility of reactor internals to IGSCC was also recognized
  - Shroud cracking in 1993-1994 confirmed that IGSCC of internals is a significant issue forBWRs
- Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly irradiated components. The aging effect is cracking.
- IASCC is actually IGSCC combined with the presence of irradiation effects.

IASCC



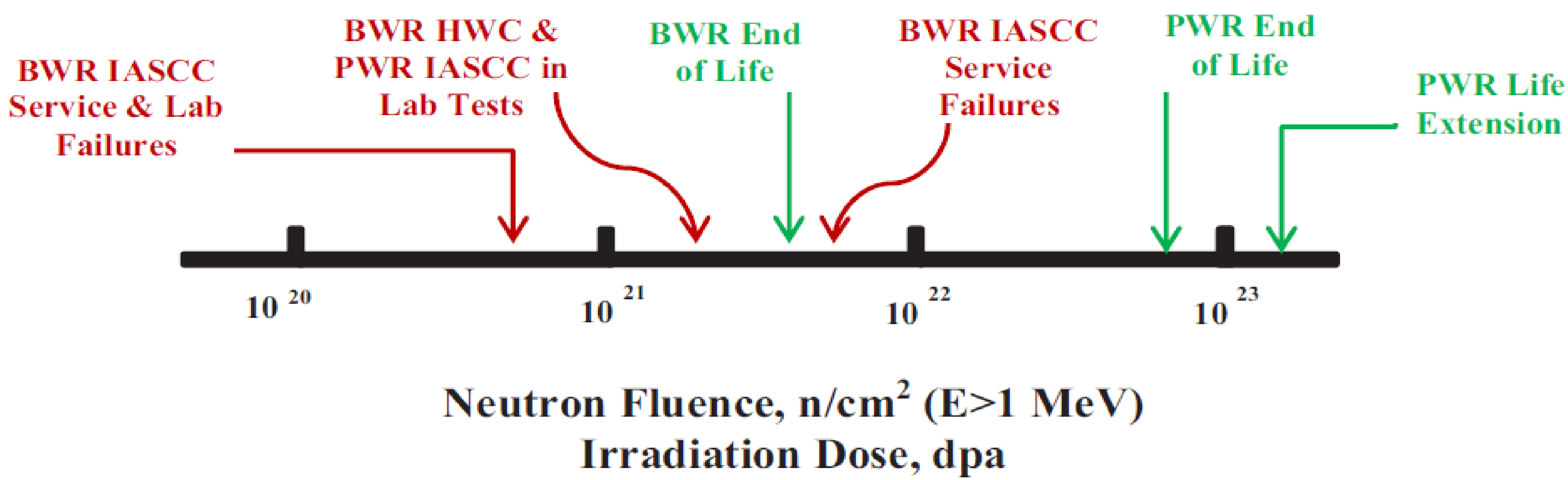
- Irradiation creep can relax residual and applied stresses
- Swelling, could affect the loads applied to components due to differential swelling rates between Type 304 and 316 stainless steels

- BWR oxygenated coolants: RIS, particularly chromium depletion, has been shown to be an important precursor of IASCC
- PWR High dose
    - hardening
    - helium bubbles in GB

Radiolysis: formation of oxidizing species,
oxygen and hydrogen peroxide plays an important role in this manifestation of IASCC in BWRs,

IASCC

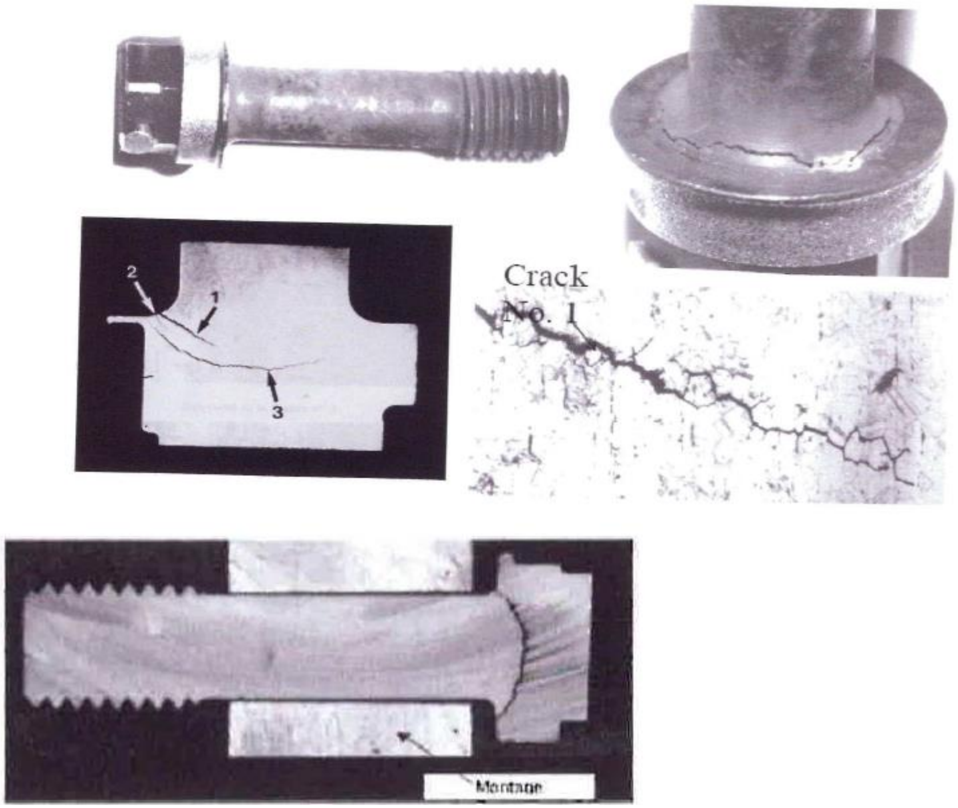
- Differences between BWR and PWR
  - Neutron fluence and temperature
  - Water chemistry
    - ECP in PWR primary water is well below the threshold level and far below the ECP of a BWR running with normal water chemistry.
    - The oxidizing environment used in BWRs operating with NWC has a stronger impact on the likelihood of SCC or IASCC than do the higher temperature and dose experienced by materials in PWRs.

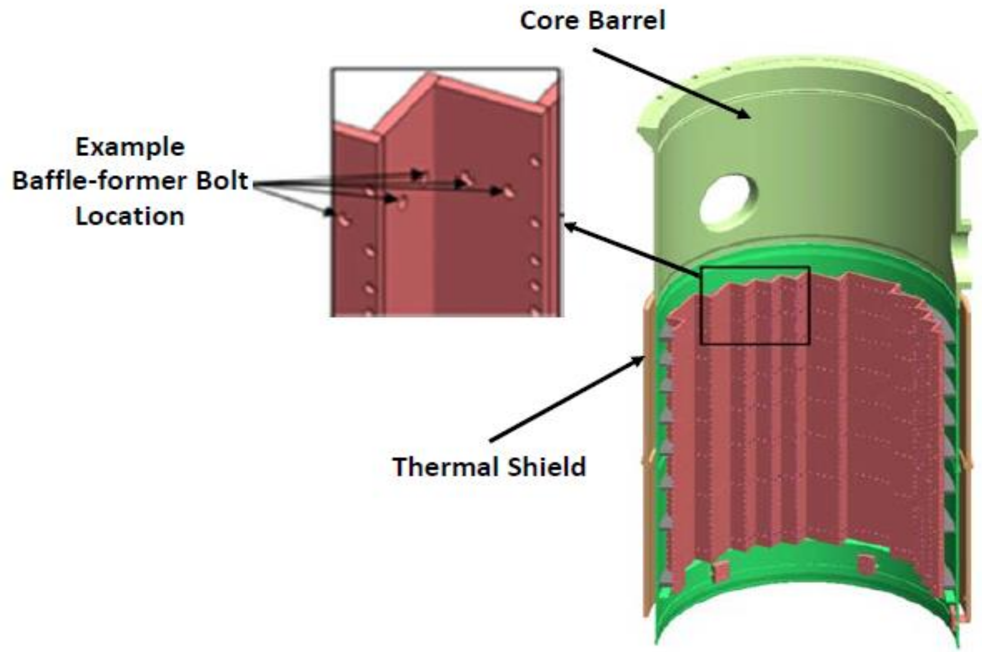


IASCC

- PWR experience - Issues of greatest concern have not been seen
  - SCC of austenitic welded components not observed o Cracking currently limited to high strength bolting (X-750) –
  - IASCC of SS welds not observed, indications limited to bolts –
  - Macroscopic effect of void swelling not observed
  - Baffle-to-former bolts are highly susceptible to Cracking – **Cracking has been observed in baffle-former bolts made of both Type 347 (in US) and Type 316 (in France) stainless steels.**

IASCC BAFFLE BOLTS



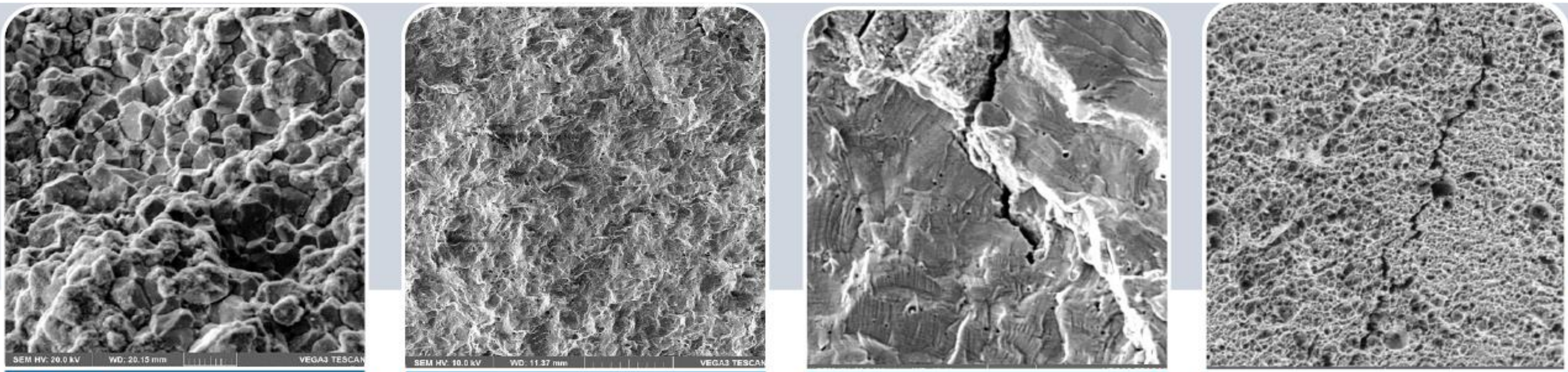


Examples of Baffle-Former Bolt Cracking,
Timothy J. Griesbach et al

IASCC BAFFLE BOLTS

- Which mechanisms contributed to the failure of the bolts?

BFB Hot Cell Testing –Crack Propagation



IASCC

- IASCC was seen on the fracture surface of bolts that were removed in two pieces
- IASCC was also seen on the fracture surface of an IP "anti-cluster" green bolt
- In some cases, IASCC is only a few grains deep

Transgranular Cracking

- TG and mixed-mode cracking is apparent on many fracture surfaces

Fatigue

- In some regions, fatigue striations can be seen; although, the striations may be attributed to transgranular stress corrosion cracking

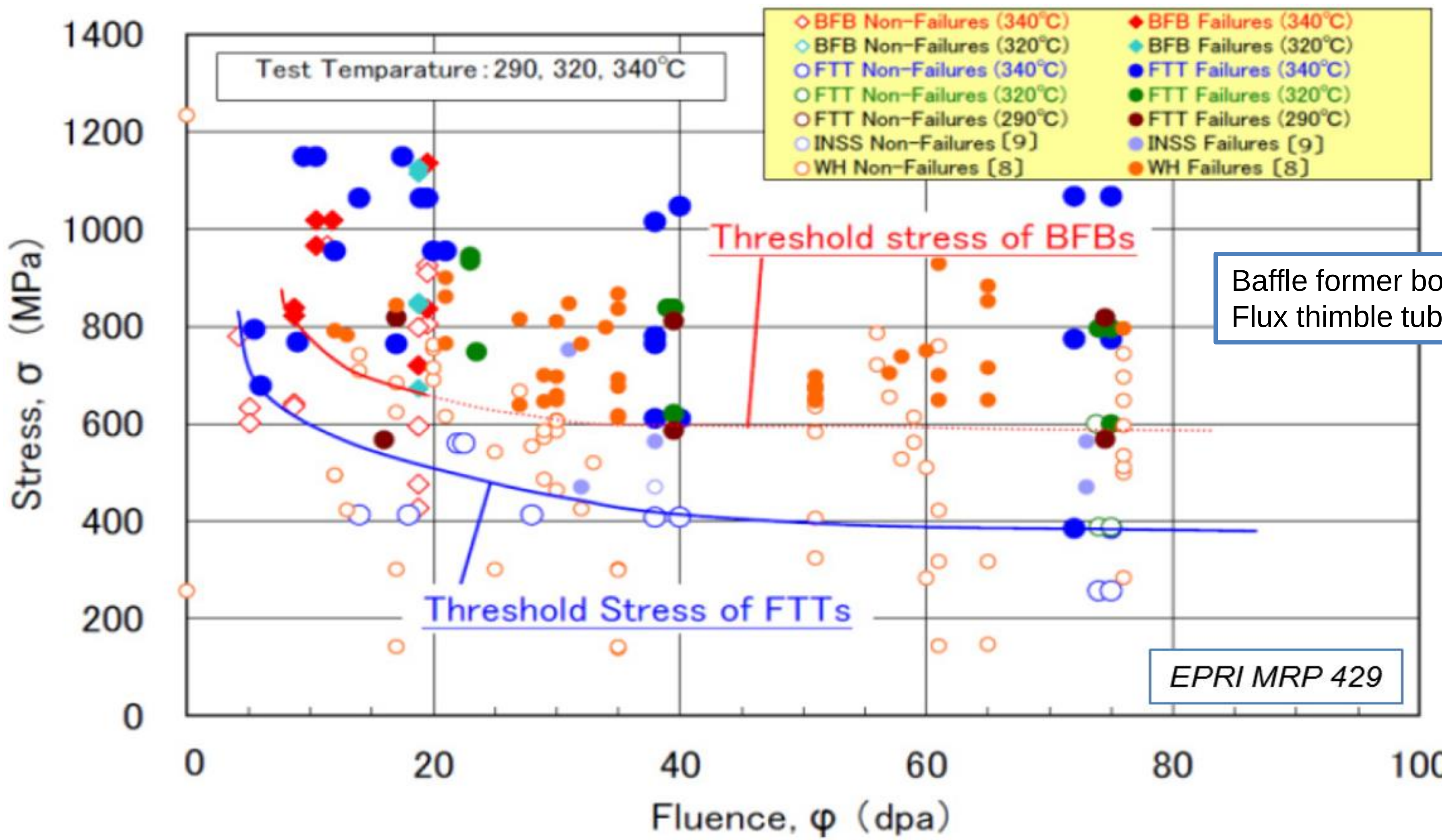
Ductile Overload

- The fracture of the remaining ligament shows retained ductility on most bolts consistent with tensile/pull test results

Joint EPRI MRP/PWR Owners Group Baffle-Former-Bolt Focus Group 2017

IASCC

IASCC Stress Threshold Curve Based on Fluence Level



Baffle former bolts (BFBs), Flux thimble tubes (FTTs)

EPRI MRP 429

Other degradation mechanisms

- Wear
- Thermal ageing
- Fatigue

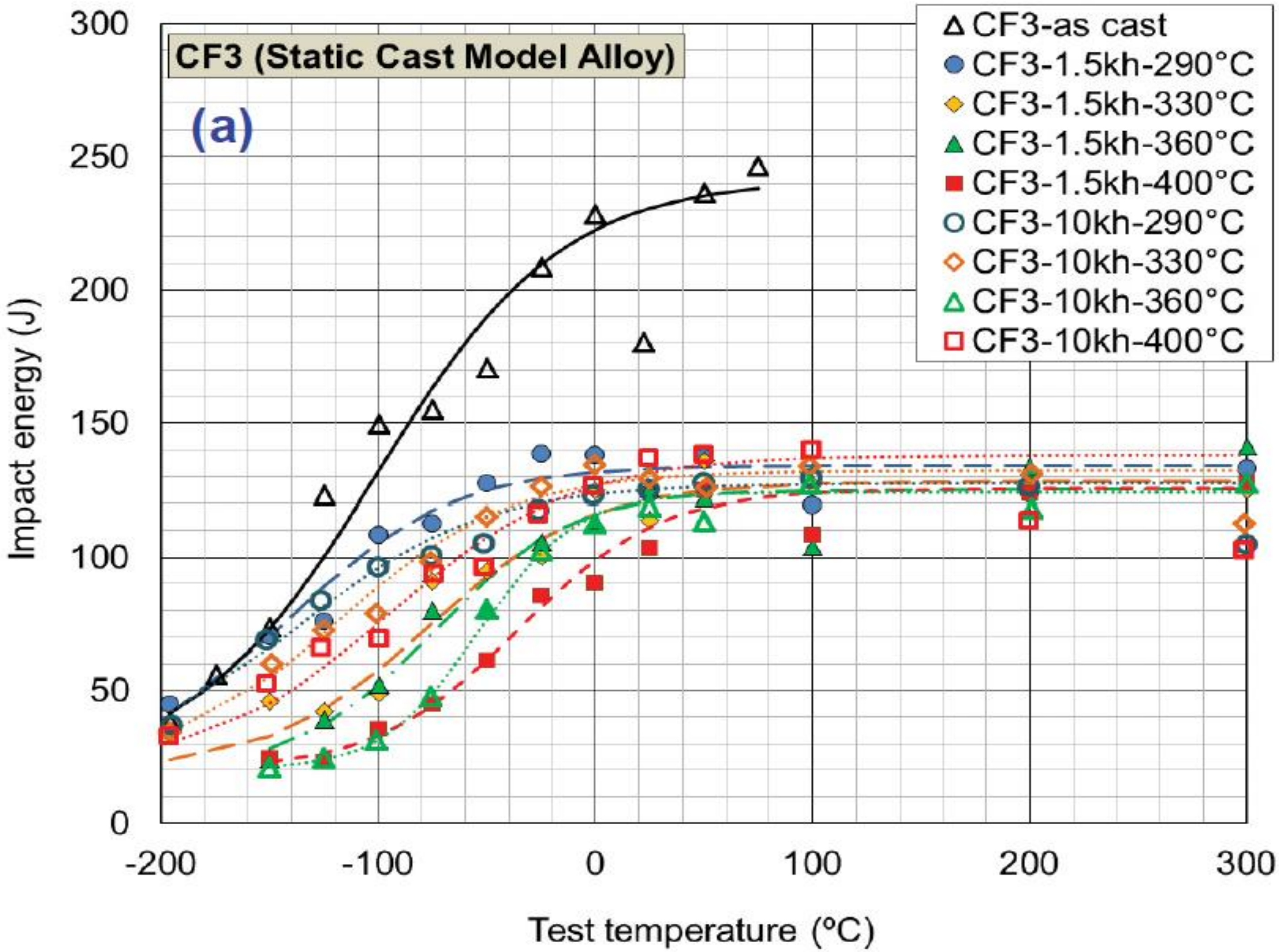
WEAR

- ❑ Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.
- ❑ PWR operating experience includes wear issues associated with flux thimbles, guide card wear by control rod cladding contact, and thermal sleeve flanges
  - For Alloy A-286 and Alloy X-750 reactor internals fasteners, wear may be a concern
  - Wear is a concern for the nickel-base alloy core guide lugs and clevis inserts (hardfaced with Stellite.)
- ❑ Wear concerns are related to experience with a number of BWR internal components.
  - Jet pump assembly (hardfaced with stellite), thereby contributing some additional 60Co to the reactor water.
  - Some steam dryer locations such as vessel lugs, feedwater end brackets and BWR/6 shroud head studs.

Thermal Aging Embrittlement

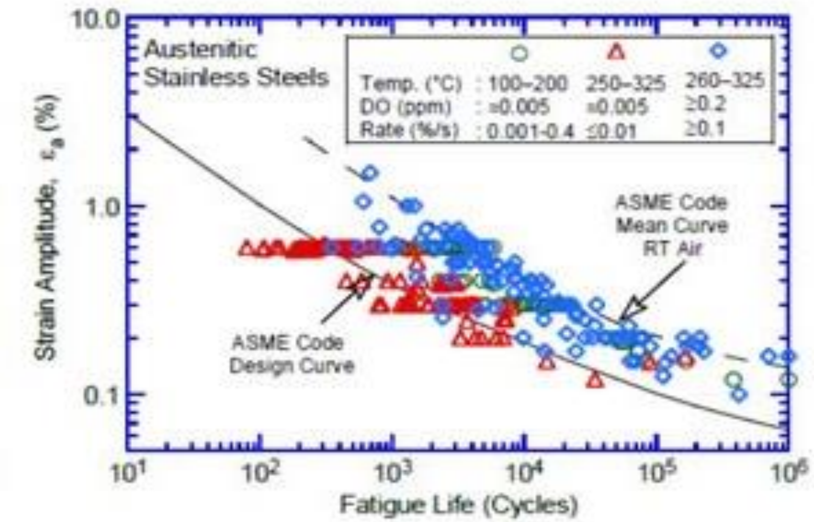
- Occurs for a limited number of internals materials
  - Cast austenitic stainless steel (CASS)
  - Austenitic stainless steel welds
  - Martensitic stainless steels
  - Martensitic precipitation-hardenable stainless steels
- CASS and austenitic SS welds depend on the Mo content and the ferrite content
- Concerns exist regarding the possible synergistic impact of thermal aging and irradiation embrittlement on SCC resistance of castings
  - CASS - Both the austenite and the ferrite are affected by irradiation Only the ferrite is affected by temperature

Thermal Aging Embrittlement



Fatigue

- Fatigue is a process involving the evolution of persistent slip bands at the surface of a material and subsequent crack formation and propagation during exposure to cyclic stresses
  - Low-cycle fatigue (LCF) is associated with plastic strains
  - High-cycle fatigue (HCF) occurs at stresses below the elastic limit
- For both cases, the environment can impact the final fatigue results and this is known as environmentally-assisted fatigue (EAF)



Environmental fatigue correction factor to account for the effects of the coolant environment
F_{en}, which is the ratio of fatigue life in air at room temperature to that in water under reactor operating conditions

Fatigue

- High-cycle fatigue is an issue of concern for selected BWR internals components, particularly steam dryers and jet pumps.
- For irradiated reactor internals, there are relatively limited data related to the effects of neutron fluence on fatigue life and crack growth rates.
  - Concerning fatigue crack propagation rates - deleterious effect of irradiation in the form of a superposition model with IASCC growth rates.

Summary

- Degradation of internals can lead to reduced performance, and in some cases, unexpected early failure.
- Main degradation mechanisms
  - Hardening and loss of fracture toughness
  - Swelling / Irradiation creep (PWR)
  - IASCC
  - Wear
  - Thermal ageing
  - Fatigue