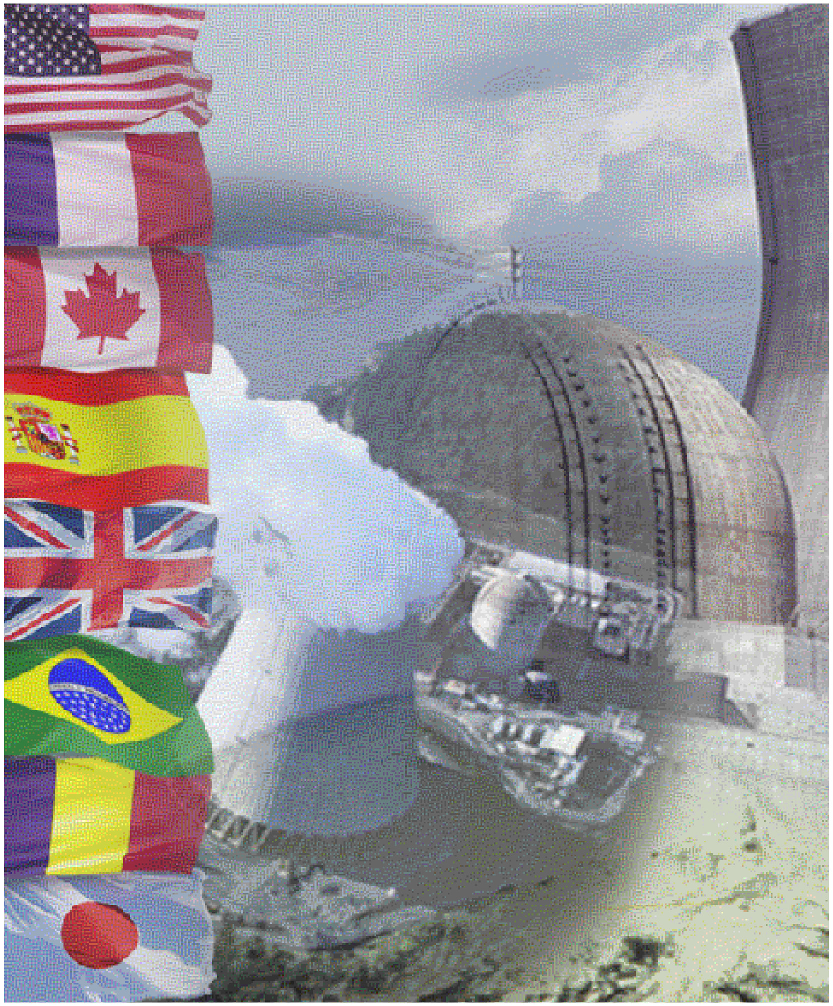


MRP RI-ITG Program Results and Status

Jeff Gilreath
Duke Energy
Rege Shogan
Westinghouse

H. T. Tang
EPRI

NRC Meeting
October 23, 2003
NRC Headquarters
Rockville, MD

Agenda

Time	Item	Presenter
9:00AM	Introductions Purpose of Meeting	J.D. Gilreath, Duke Energy Corp.
9:05 AM	MRP RI-ITG Goals, Objectives and Scope	J.D. Gilreath, Duke Energy Corp.
9:20 AM	MRP RI-ITG Programs and Status	H. T. Tang, EPRI
10:30 AM	Recent Hot Cell Test Results	Rege Shogan, Westinghouse
11:15 AM	Open Discussion/NRC Comments	All
11:30 AM	Adjourn	

Purpose of Meeting

Provide the status of the Material Reliability Program (MRP) Reactor Internals (RI) Issues Task Group (ITG), which also supports License Renewal Aging Management Programs referenced by some owners groups and utilities.

RV Internals Programs Background

- PWR Owners Groups
  - Have addressed rv internals for many years
    - A286 & X750 bolting, X750 split pin, thermal shields, upflow mod.
  - In early 1990s began evaluating significance of baffle bolt cracking
    - Plant categorization, operability evaluation, industry follow
  - In mid to late 1990s initiated more comprehensive programs addressing potential of baffle bolt cracking
    - Identified bolt design Information and susceptibility
    - Performed some baffle bolt inspections in U.S.
    - Removed and replaced some baffle bolts
    - Performed hot cell evaluations on failed baffle bolts
    - Prepared Safety Evaluation
  - Formed the JOBB
  - In late 1990s evaluated all susceptible rv internal components for potential aging effects - license renewal topicals
  - Industry-wide PWR Materials Reliability Program (MRP) developed (1998)
  - Created an Issue Task Group (ITG) to manage RV Internals aging issues

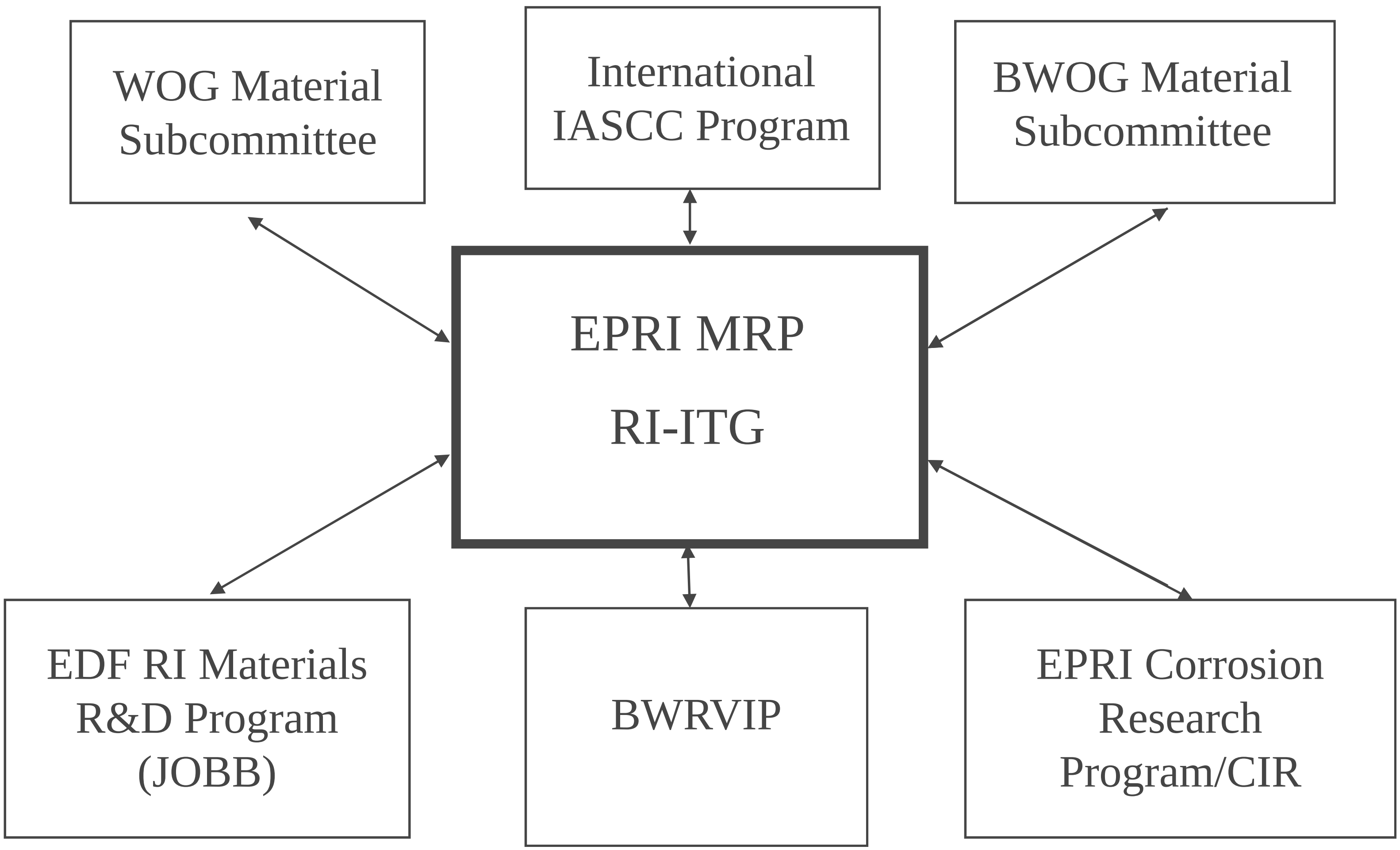
RI-ITG Program Mission/Vision

- To support the establishment of overall technically correct programs, which will **assure reactor vessel internals performing its design function through plant life** (60+ years of operation).
- This function is supported by:
  - serving as an industry focal point for resolution of issues related to reactor internals materials degradation,
  - **implementing needed programs to bring resolution of potential aging effects,**
  - **providing research on the effects of identified aging mechanisms on reactor internals,**
  - providing a focal point to support communication with the NRC when addressing aging of PWR reactor internals components.

RI-ITG Program Formation and Funding

- The RI-ITG program is formed to support EPRI/MRP member utilities to manage aging of reactor internals components
- The program cooperates with international partners to achieve cost effectiveness
- MRP/RI-ITG is funded by:
  - All US nuclear utilities
  - Foreign members of the EPRI Nuclear Power Sector, e.g., EDF, TEPCO
  - Foreign members of the MRP program, e.g., Kansai Electric Power Corp. and Japan Atomic Power Corp.

RI-ITG Program Coordination



RI-ITG Program Scope and Future Products

- Define and Quantify Material Degradation Mechanism
- Demonstrate Component Functionality
- Develop Flaw Tolerance Technical Basis and Acceptance Criteria
- Develop Screening Criteria and Inspection Guidelines

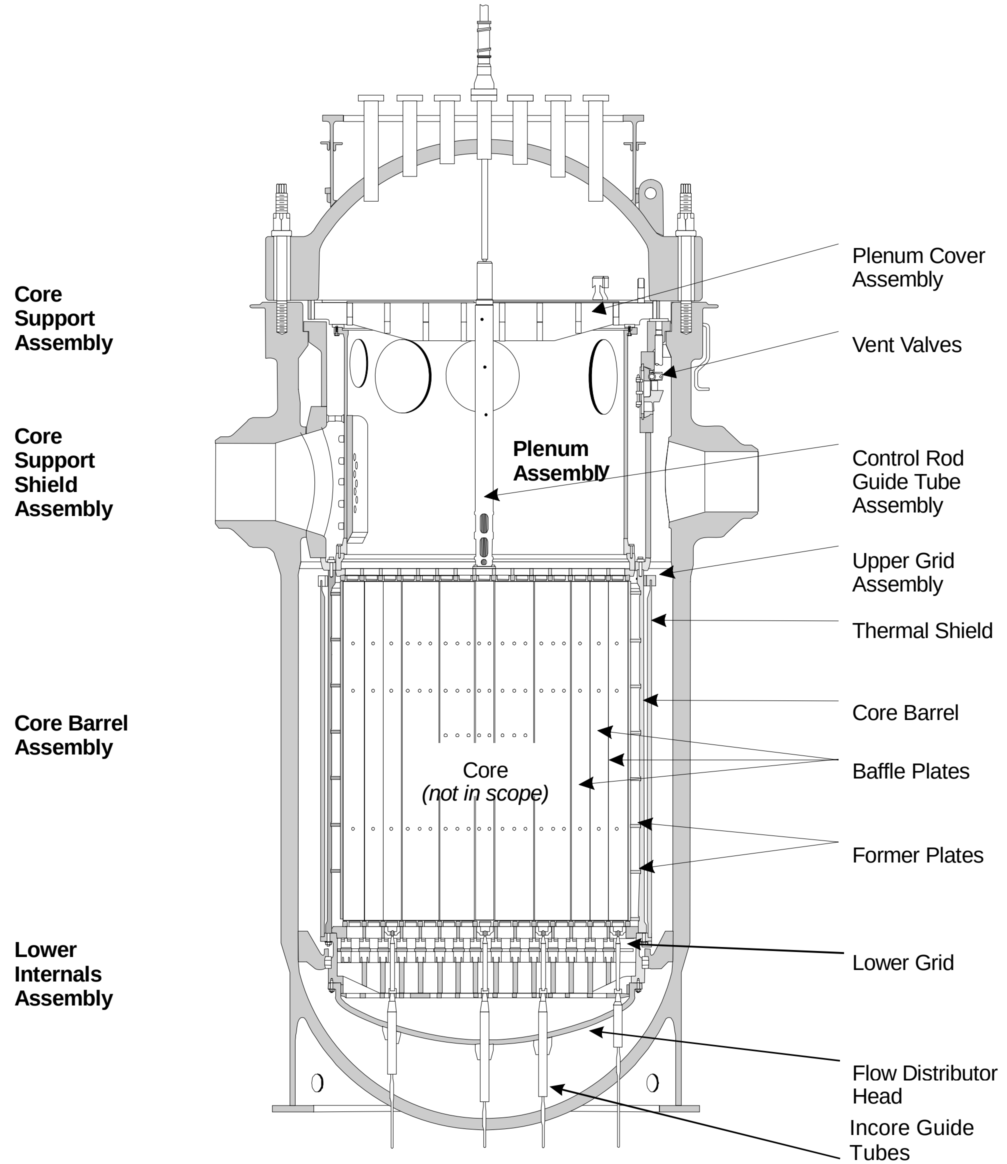
R-ITG Approach to Address Aging

- Identify potential aging effects to RV internals
- Screened RV Internal's components for potential susceptibility
- Develop research programs to address identified needs
- Utilize experimental data and operating experience to further screen components where aging effects are negligible and determining lead components for inspections
- Perform functionality assessment on susceptible components
- Continue self assessments to assure end products are meeting need of industry and future products are highly affective addressing identified issues

PWR Reactor Internals Materials (Examples)

Material Type	Product Form	Components	Material Specifications
304 / 304L 316 / 316L 347	Bar	lugs, pads, pins, shims, plugs, retainers	A276 TP-304 A479 TP-304
	Plate	core barrel, plenum cylinder, former and baffle plates, guide tubes	A240 TP-304L A240 TP-304
	Forgings	flanges, nozzles, radial keys, lugs	A473 TP-304 A182 Gr F304
	Pipe/Tubing	pipe, support posts, column sleeves, column extensions	A312 TP-304 A316 TP-304 A213 TP-304
	Fastener	bolts, cap screws	A193 Gr B8 SA193 Gr B8M
308 / 308L	Welding Rod and Filler Metal	welds	A298 TE-308 A371 TER-308
Alloy A286	Bar / Fastener	bolts, cap screws	SA-453 Gr 660
CF8 / CF3M	Casting	vent valve body, lower support columns	A351 Gr 660 A351 Gr CF8

Reactor Vessel Internals Description



RI-ITG Program Elements to Address Reactor Internals Aging

State of Knowledge Assessment

- Radiation embrittlement
- IASCC
- Void swelling
- Creep/Stress relaxation
- License renewal SERs

Operating Conditions Analysis

- Temperature
- Fluence
- Stress

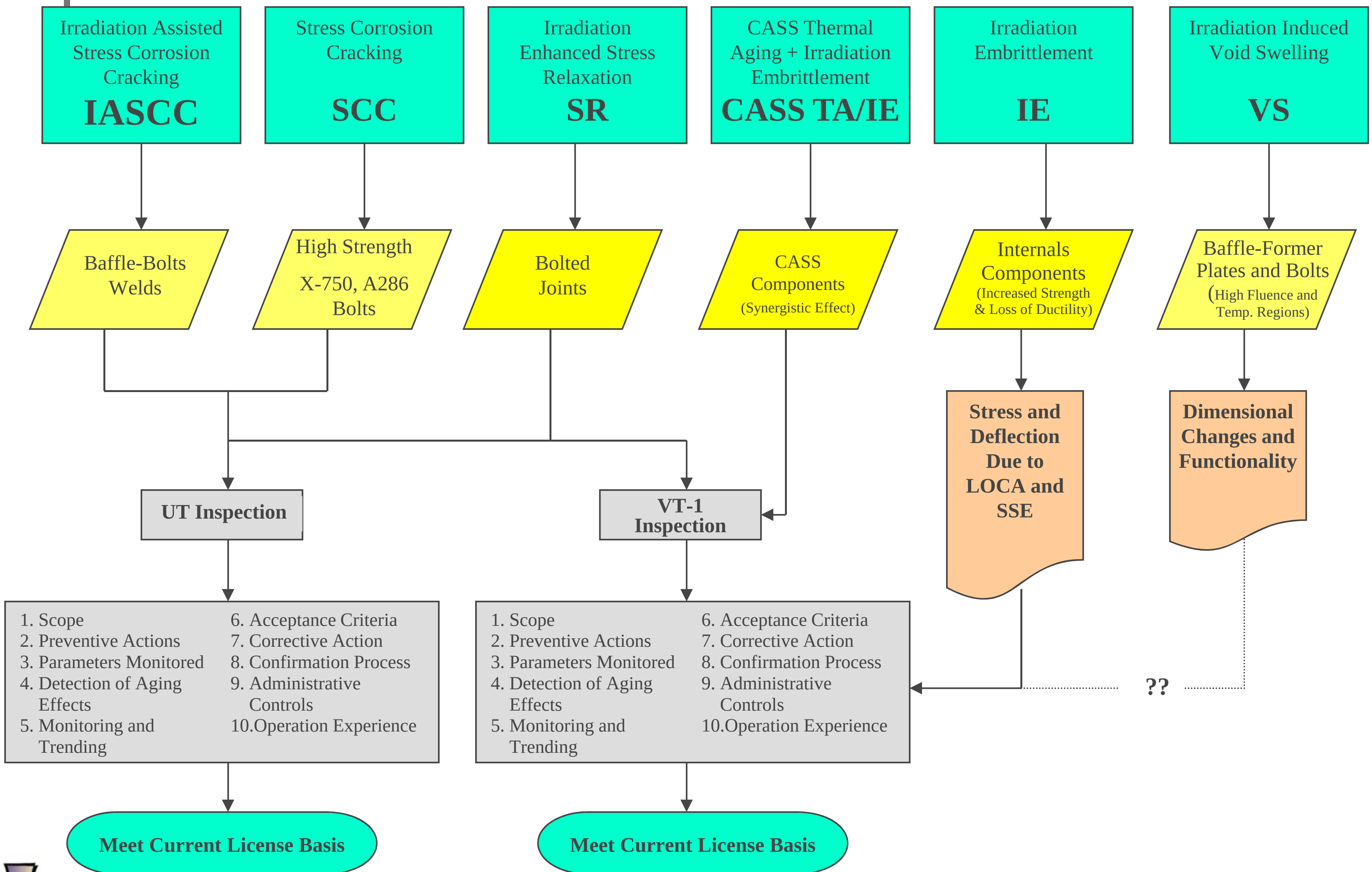
Hot Cell Testing

- Mechanical Properties
- Fracture toughness
- Crack initiation/growth
- Void swelling
- Creep/Stress Relaxation

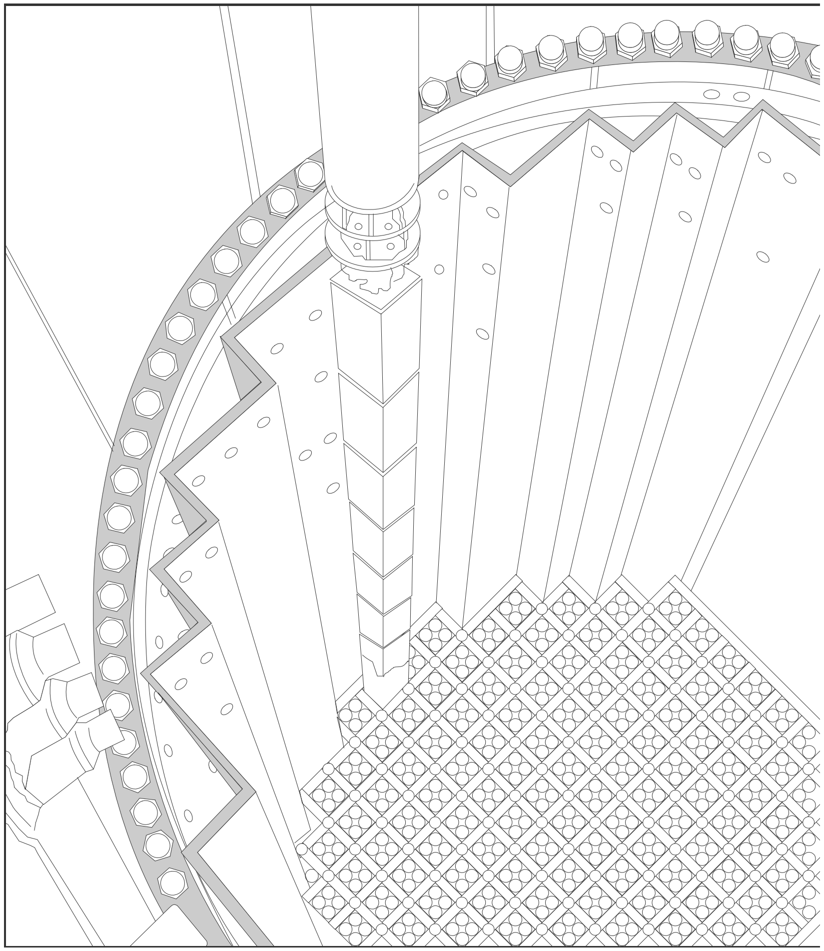
Aging Management

- Screening/Inspection
- Operating conditions
- Improved materials
- Functionality
- Repair/replacement

Reactor Internal Inspections for License Renewal

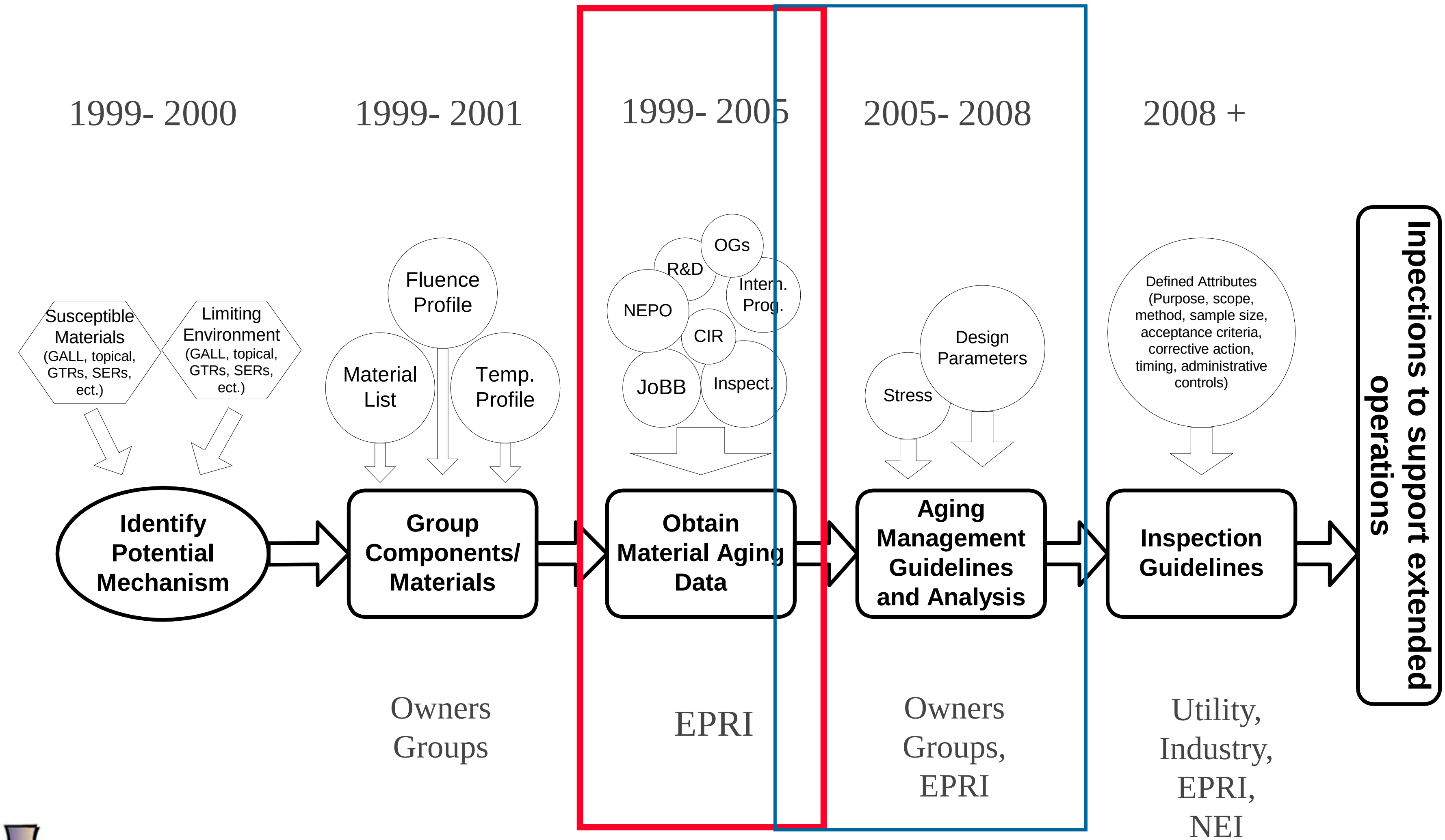


Inspection Attributes



1. Scope
2. Preventive actions
3. Parameters Monitored
4. Detection of aging effects
5. Monitoring and Trending
6. Acceptance Criteria
7. Corrective Action
8. Confirmation Process
9. Administrative Controls
10. Operating Experience

Reactor Internals Aging Management Coordination and Time Line



MRP

EPRI

RI-ITG 1999 – 2000 Products Delivered

- Analysis of Baffle Former Bolt Cracking in EDF CPO Plants (MRP-03), TR-112209, 6/1999
- JOBB – CD Version 00.06, AP-114929-CD
- JOBB - CD Version 00.12, 1000777, 11/2000
- Interim Report on Hot Cell Testing of Lead Plants Bolts (MRP-28), 100971, 11/2000
- Inspection and Replacement of Baffle to Former Bolts at Point Beach-2 and Ginna, TR114779, 2/2000
- EPRI Baffle Bolt Project Summary, AP-114779-CD, 4/2000

RI-ITG 2001 Products Delivered

- Hot Cell Testing of Baffle/Former Bolts Removed From Two Lead Plants (MRP-51), 1003069, 11/2001
- Determination of Operating Parameters of Extracted Bolts (MRP-52), 1006075, 10/2001
- JOBB – CD Version 01.06, 1001360, June 2001
- *In-Situ* NDT Measurements of Irradiation-Induced Swelling in PWR Core Internal Components -- Phase 1: Testing of Unirradiated Surrogate Material, 100658, 10/2001
- Technical Basis Document Concerning Irradiation-Induced Stress Relaxation and Void Swelling in Pressurized Water Reactor Vessel Internals Components (MRP-50), 1000970, 11/2001

RI ITG 2002 Products Delivered

- Strategies for Management of Aging Effects in PWR Reactor Vessel Internals (MRP-62), 1006582, 2/2002
- JOBB CD 01.12, 1002858, 3/2002
- Characterization of Type 316 Cold-Worked Stainless Steel Highly Irradiated under PWR Operating Conditions (MRP-73), 1003525, 8/2002
- A Review of Radiation Embrittlement of Stainless Steels for PWRs (MRP-79), 1003524, 11/2002
- JOBB CD 02.12, 1002810, 12/2002

RI-ITG 2003 Studies

- Hot cell testing of decommissioned PWR baffle plate, former plate and core barrel samples (cofund with NEPO 051577)
  - Tensile, crack initiation, crack growth, fracture toughness, microstructure (void swelling, …..)
- In-pile PWR crack growth testing at Halden (Cofund with CIR and ROBUST Fuel Program)
- Hot Cell testing of Boris 5 and 6 materials
  - Tensile, crack initiation
- Boris 6 and 7 irradiation to achieve 40, 60 and 80 dpa
  - US materials
- Integration of CIR (Cooperative IASCC Research) Program data and understanding
- Inspection and flaw evaluation strategy
- Integrated component functionality evaluation strategy
  - Embrittlement, void swelling, stress relaxation/creep, fracture toughness, crack growth

Research Program and Results

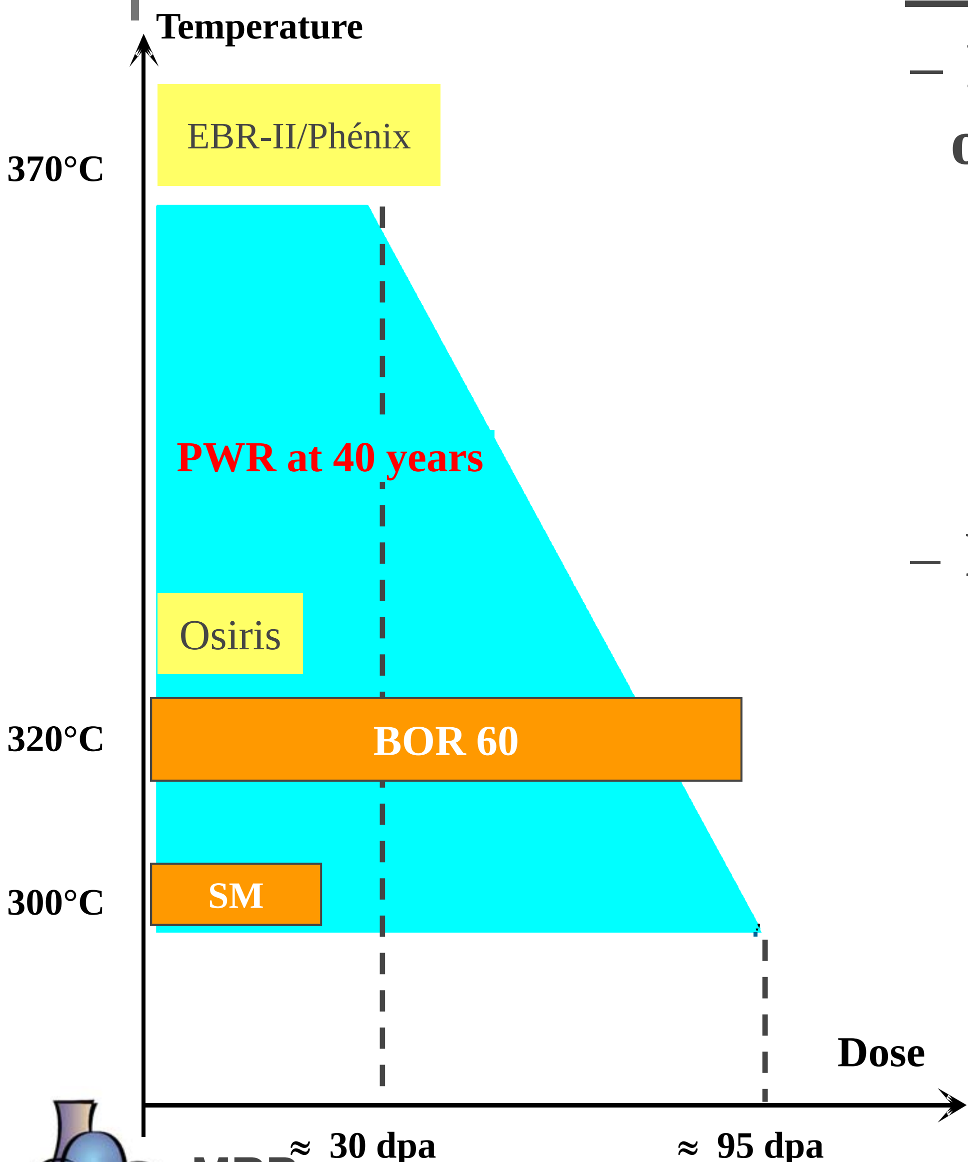
RI-ITG Program Results and Planned Studies

- Characterize Irradiated Materials Properties
  - Programs
    - JOBB program
    - Baffle/former bolts and high strength bolts test program
    - Decommissioned PWR internals materials test program
    - International IASCC program
    - Halden crack growth program
  - Areas
    - Tensile tests – stress, strain, fracture toughness
    - Creep-stress relaxation tests
    - Corrosion tests – crack initiation, crack propagation
    - TEM investigation – microstructure, void swelling
- Inspection and flaw evaluation strategy (future reporting)
- Integrated component functionality evaluation strategy (future reporting)

JOBB Program

- EDF experience on service induced cracking of baffle/former bolts
- EDF R&D programs on effects of irradiation on current baffle bolting and vessel internal and possible replacement materials
- Performance of current and possible replacement materials
- Irradiation in BOR 60 fast reactor
  - EDF materials
  - US materials

JOBB Irradiation



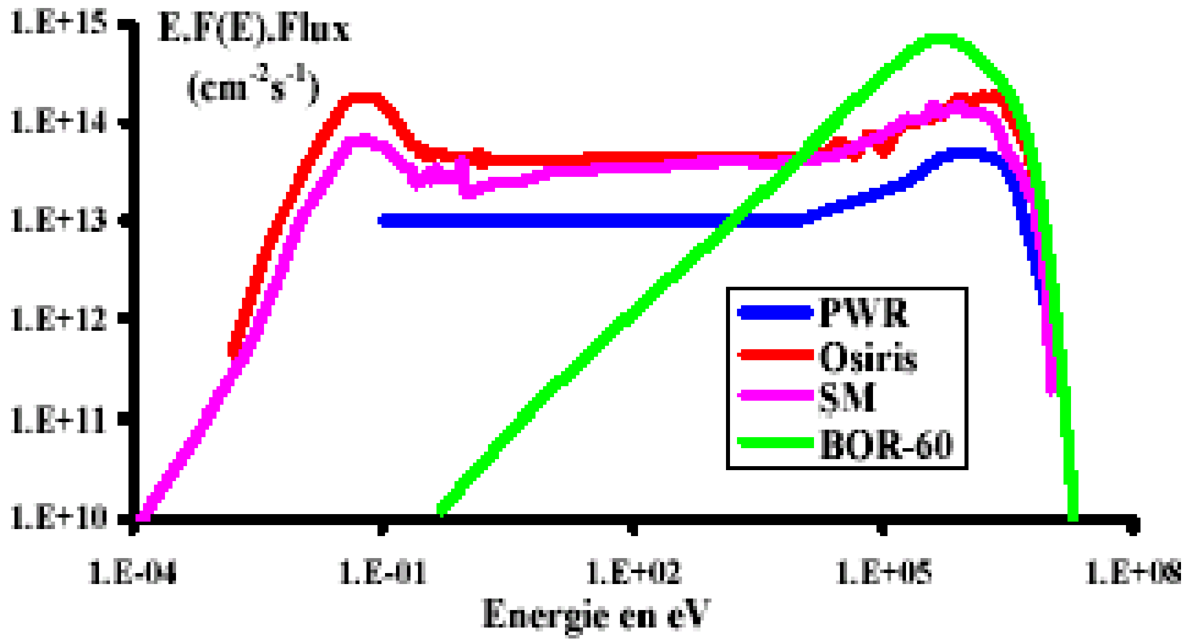
Materials

- **Representative of Core Internals of PWRs**
  - **SA 304L Baffle plates, Formers, Core barrel**
  - **CW 316, 347 Baffle bolts**
  - **308 Welds, 304 HAZ**
  - **CASS**
- **Possible "replacement" materials**
  - Density,
  - Irradiation creep,
  - Tensile Properties,
    - High doses,
    - Neutron spectrum effect,
  - Fracture toughness,
  - Microstructural investigations

Neutron Spectrum Profiles

Irradiation : Reactor thermal/fast mixed spectrum ≡ PWR

Production of gaz atoms and point defects



OSIRIS : Opencore pool-type research thermal reactor 70 MW

1 dpa ~ 7 x 10^{20} n/cm^2 (E > 1.0 MeV)

Database of JOBB Materials Irradiated in the Boris Experiments

Materials representative of core internals:

- Type 316 CW: 4 bolt materials (F) + 3 bolt materials (US)
- Type 304L SA: 2 materials (F) + 2 materials (US) + 2 HAZ (F+US)
- Type 308: 2 materials (F) + 2 material (US)
- Type 347 SA: 1 material (D) + 1 material (US)
- Type 321 SA: 1 material (D) + 1 material (R) + 1 material (F)
- Type CF8 CASS: 1 material (US), 3 different TT
- Two 316 SA and one 304 CW

Other industrial materials:

- N9 (12Cr-25Ni-Si-Ti) Solution Annealed, Cold Worked and Thermal Aged
- Inconel 690, Incolloy 800
- Nitronic 50 et 60
- Uranus, NMF18 SA and CW

“Tailored" materials type 316:

- Type 316 small grains, large grains, monocristal
- Type 316 High Purity, High Purity + Si, with adding of Zr, Hf or Fe
- 316 Titanium (CEA) : (0.25 Ti-0.35 Si), (0.25 Ti-0.8 Si), (1.1 Ti-0.35 Si), SA and CW
- 316 SPh SA and CW 20%
- 316Nb

“Tailored" materials type 304:

- 304 with adding of Zr, Hf et Fe

JOBB Phase 1 US Materials Irradiated in BOR 60 (tensile specimens only)

Material *No. of heats (Code)* (Supplier)	Fluence 20dpa (Boris 4)	Fluence 40dpa (Boris 5)	Fluence* 60dpa (Boris 6)	Fluence* 80dpa (Boris 7)
Type 347 *1 heat (EC)* (W)	4	2		3
Type 316CW *2 heats (EA & EB)* (W)	5	2		5
Type 316SA *1 heat (ED)* (CE)		3	4	
Type 304SA *2 heats (FD & EH)* (FTI & CE)	4	4	4	3
Type 308 *1 heat (FE)* (FTI)	5			4

* Irradiations planned or in progress

JOBB Phase 2 US Materials in Bor 60 - Boris 6 & 7

Materials	Specimen type	5 dpa	10 dpa	20 dpa	40 dpa
308 TIG/MIG weld and 304 HAZ (CE)	Tensile O-ring CT 3mm disc	12 2 1 16	9 2 2 16	15 4 2 16	4 0 0 16
304 SA and CW (CE)	Tensile O-ring CT 3mm disc	6 1 0 8	12 2 1 16	14 3 2 16	5 2 2 16
316 two heats (W, EDF)	Tensile O-ring CT 3mm disc	0 0 0 0	12 3 0 16	7 3 0 16	5 1 0 16
Cast austenitic – as-received & two levels of thermal aging* (W)	Tensile O-ring CT 3mm disc	19 5 6 24	13 3 6 24	0 0 0 0	0 0 0 0

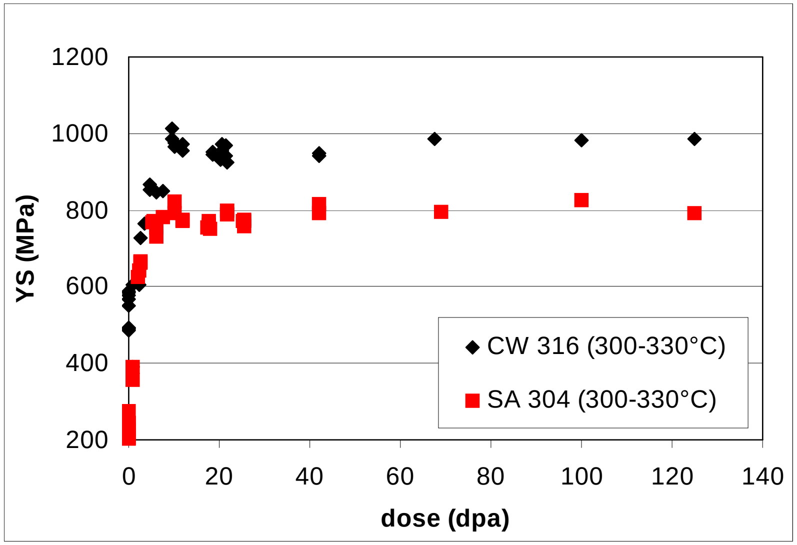
*400 ^{o}C for 100 and 1000 hours

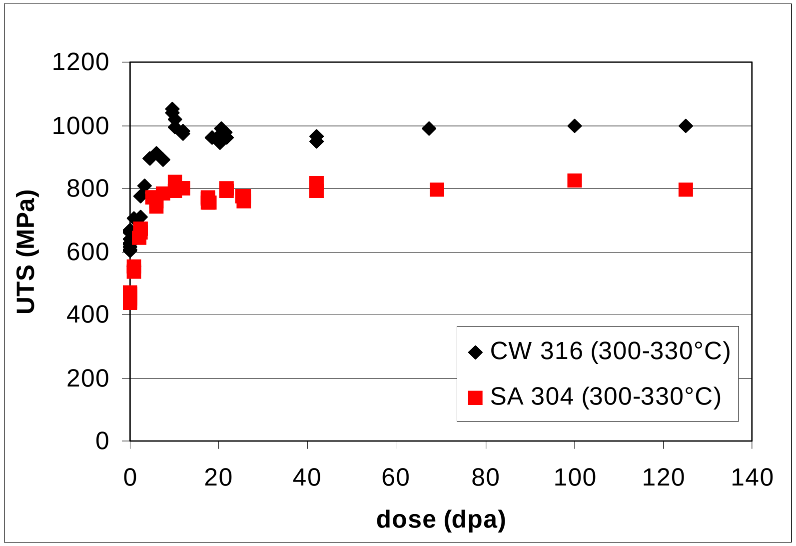
JOBB Tensile Tests

Synthesis of the Tensile Tests Results: JOBB Boris Irradiation Representative Materials

- Saturation of the tensile characteristics between 5 and 10 dpa; higher for 316CW than for 304SA
- No significant change noticed between 10 and 125 dpa (at 330°C)
- Residual ductility at saturation is significant at ~10% total elongation while uniform elongation is often <1%
- No heat to heat variations of tensile properties after irradiation for 316CW nor for 304SA
- 308 welds and CASS have roughly the same behaviour as SA 304

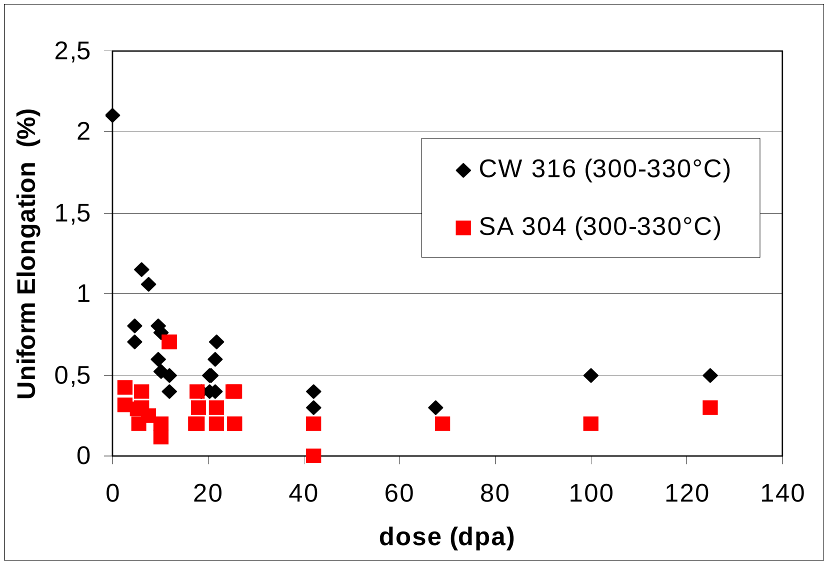
Synthesis of the Tensile Tests Results (up to 125 dpa): Tensile Characteristics (YS and UTS) at 330°C

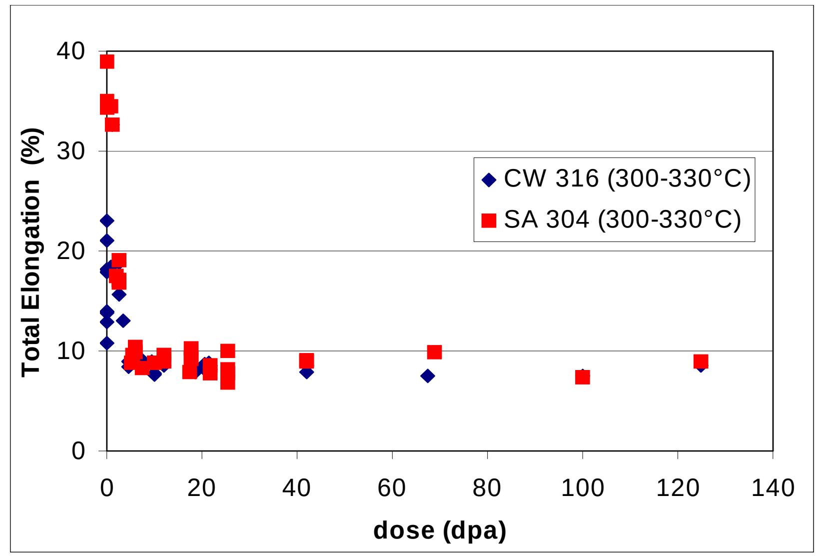




- Saturation of the hardening between 5 and 10 dpa; earlier for SA304 than for CW316
- Saturation hardening higher for 316CW (1000 MPa) than for 304SA (800 MPa)
- No significant change between 10 and 125 dpa (at 330°C)

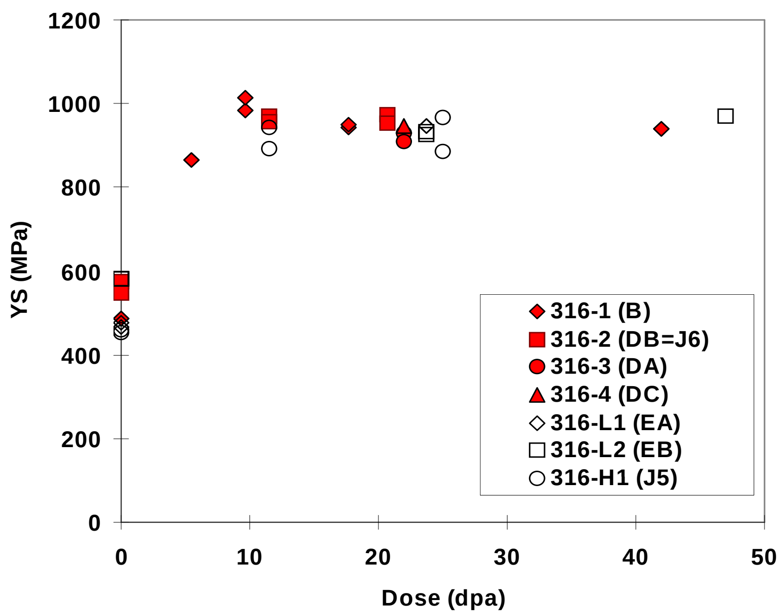
Synthesis of the Tensile Tests Results (up to 125 dpa): Tensile Characteristics (UE and TE) at 330°C

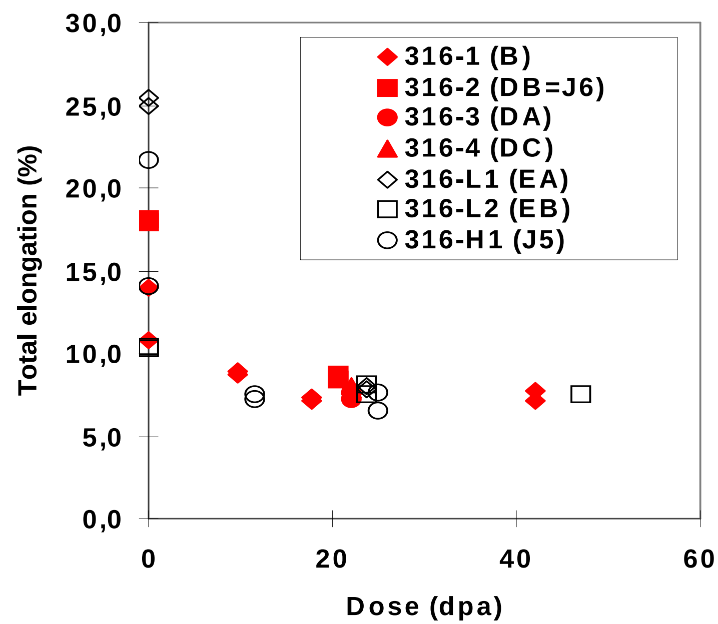




- UE saturation level higher for CW316 than for SA304, TE similar (8-10%)
- No significant change between 10 and 125 dpa (at 330°C)

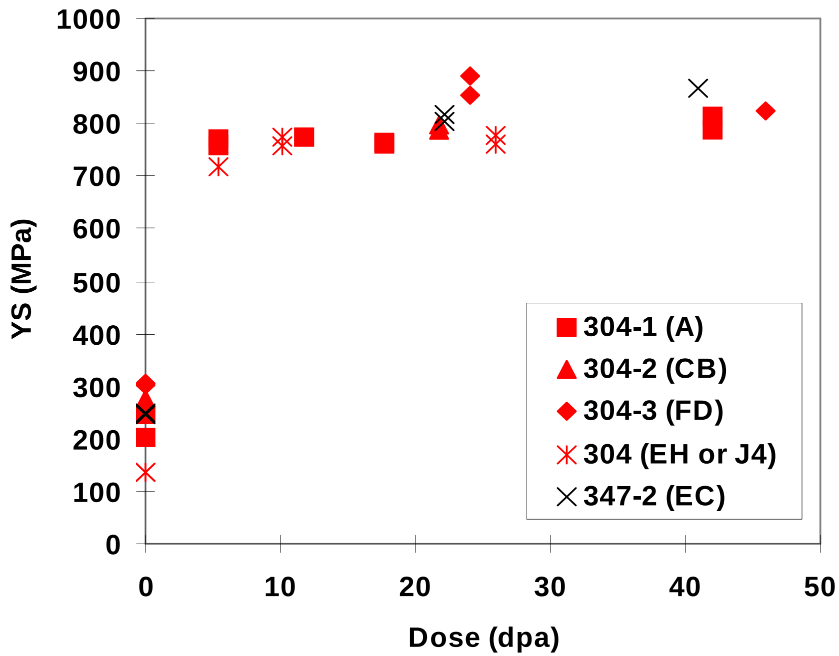
Synthesis of the Tensile Tests Results
Heat to Heat Variations: CW 316

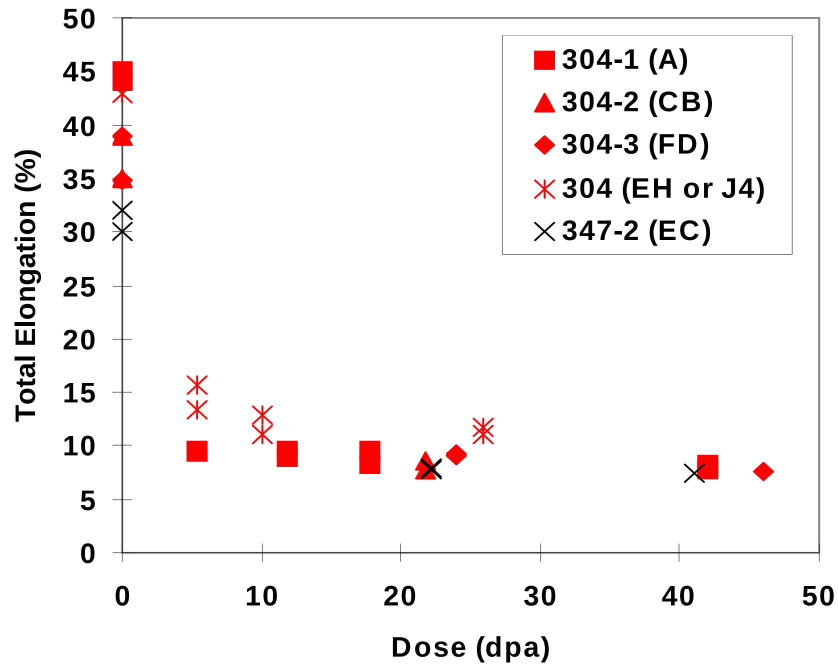




- 7 different CW 316 materials ; good homogeneity of mechanical characteristics
- No heat to heat variations after irradiation in terms of tensile properties

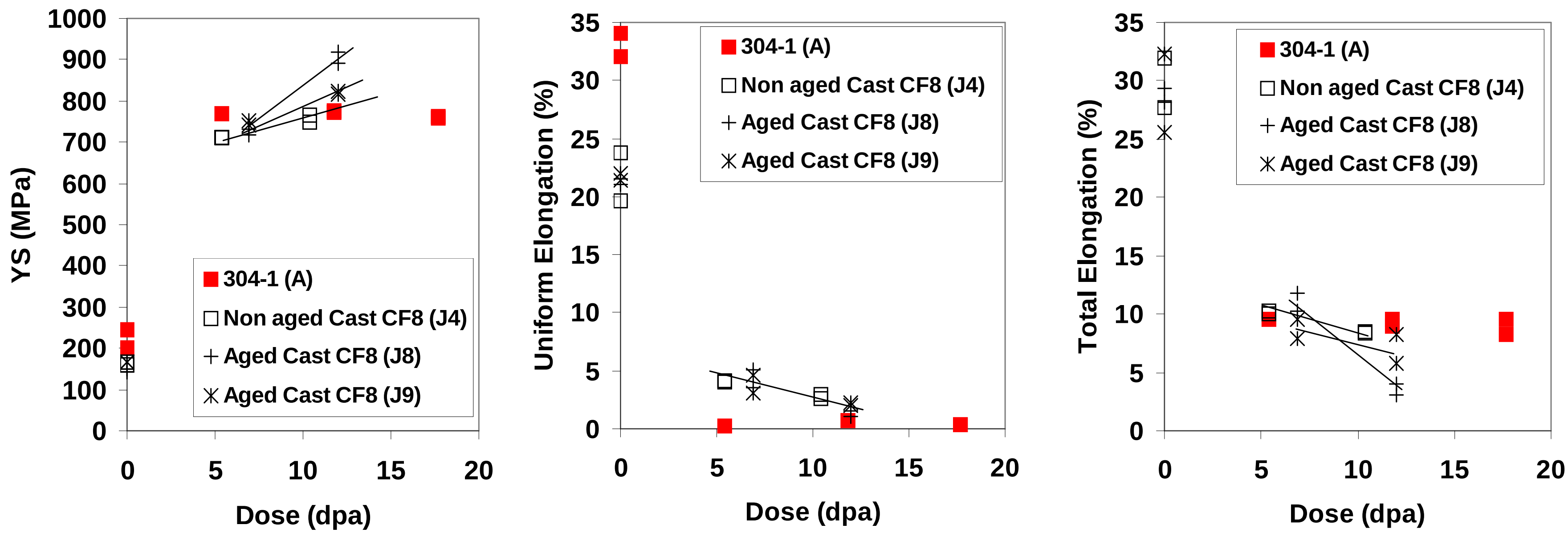
Synthesis of the Tensile Tests Results
Heat to Heat Variations: SA 304





- 4 different SA 304 materials (also SA 347) ; good homogeneity of mechanical characteristics
- No heat to heat variations after irradiation in terms of tensile properties

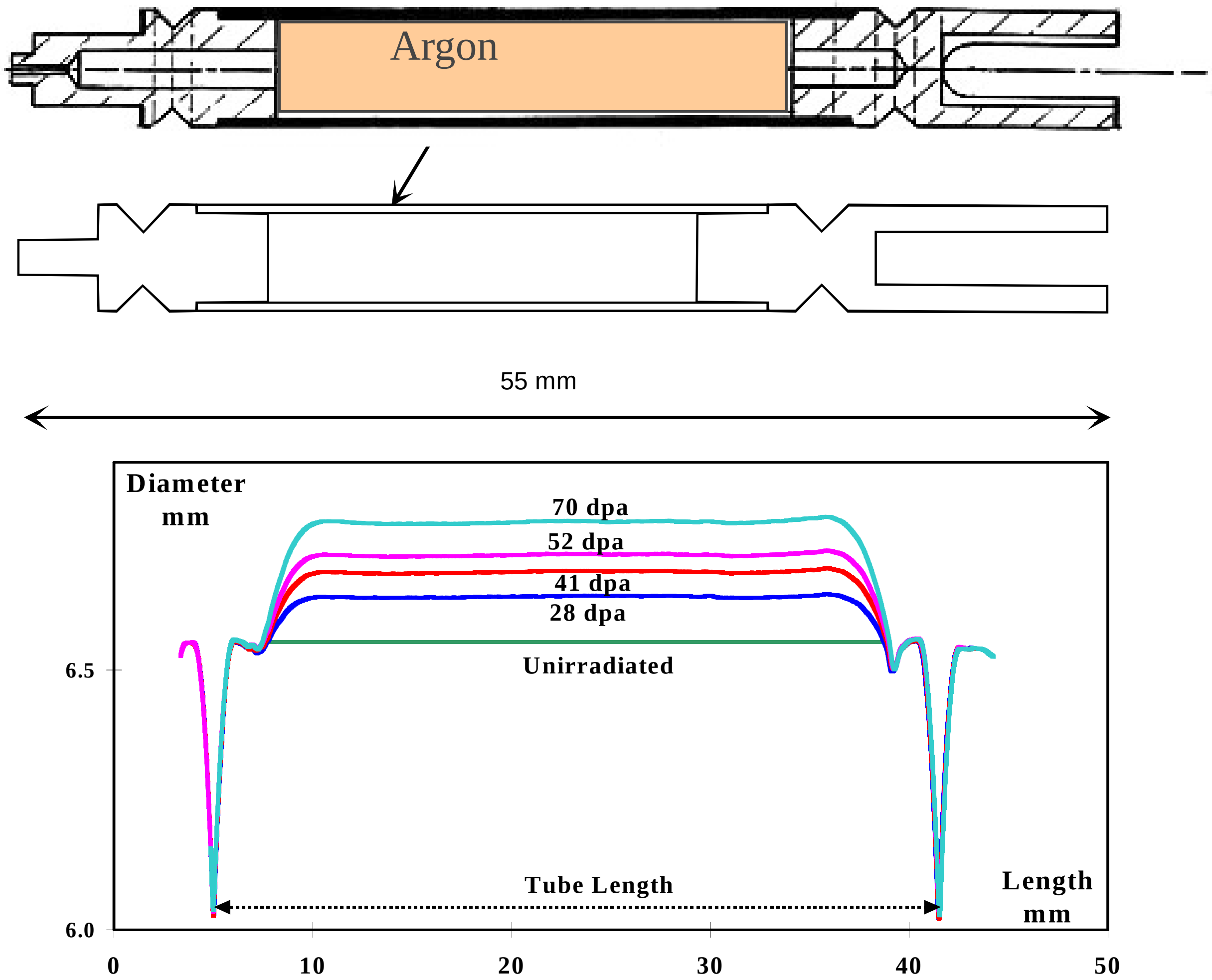
Synthesis of the Tensile Tests Results: Cast Stainless Steels



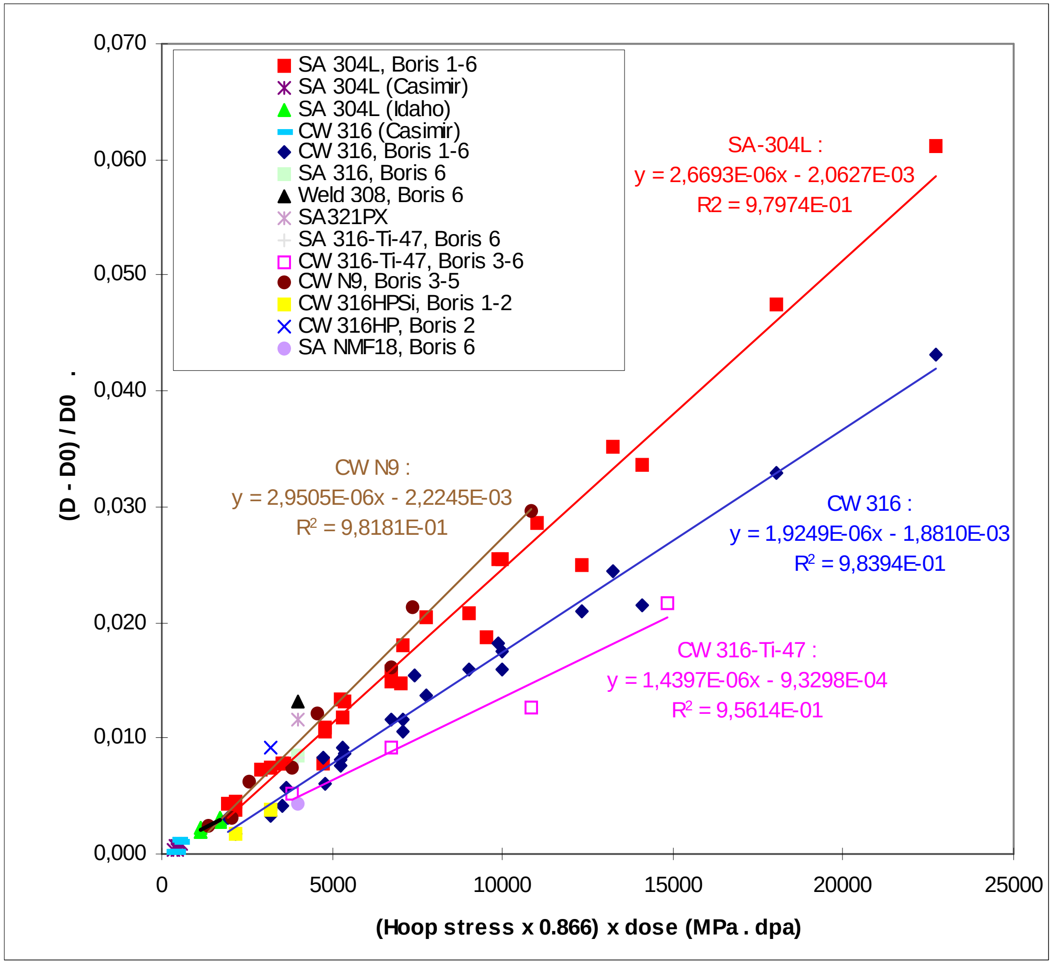
- Before irradiation, minimum thermal aging effect.
- Thermally aged hardening is higher than non-aged.
- CASS shows higher uniform elongation than the reference SA-304L after 5 dpa. After 10 dpa, the total elongation of the thermally aged CASS is lower than the non-aged CASS and the reference SA-304L.

JOBB Creep Tests

Synthesis of the Creep-Irradiation Results (up to 120 dpa)

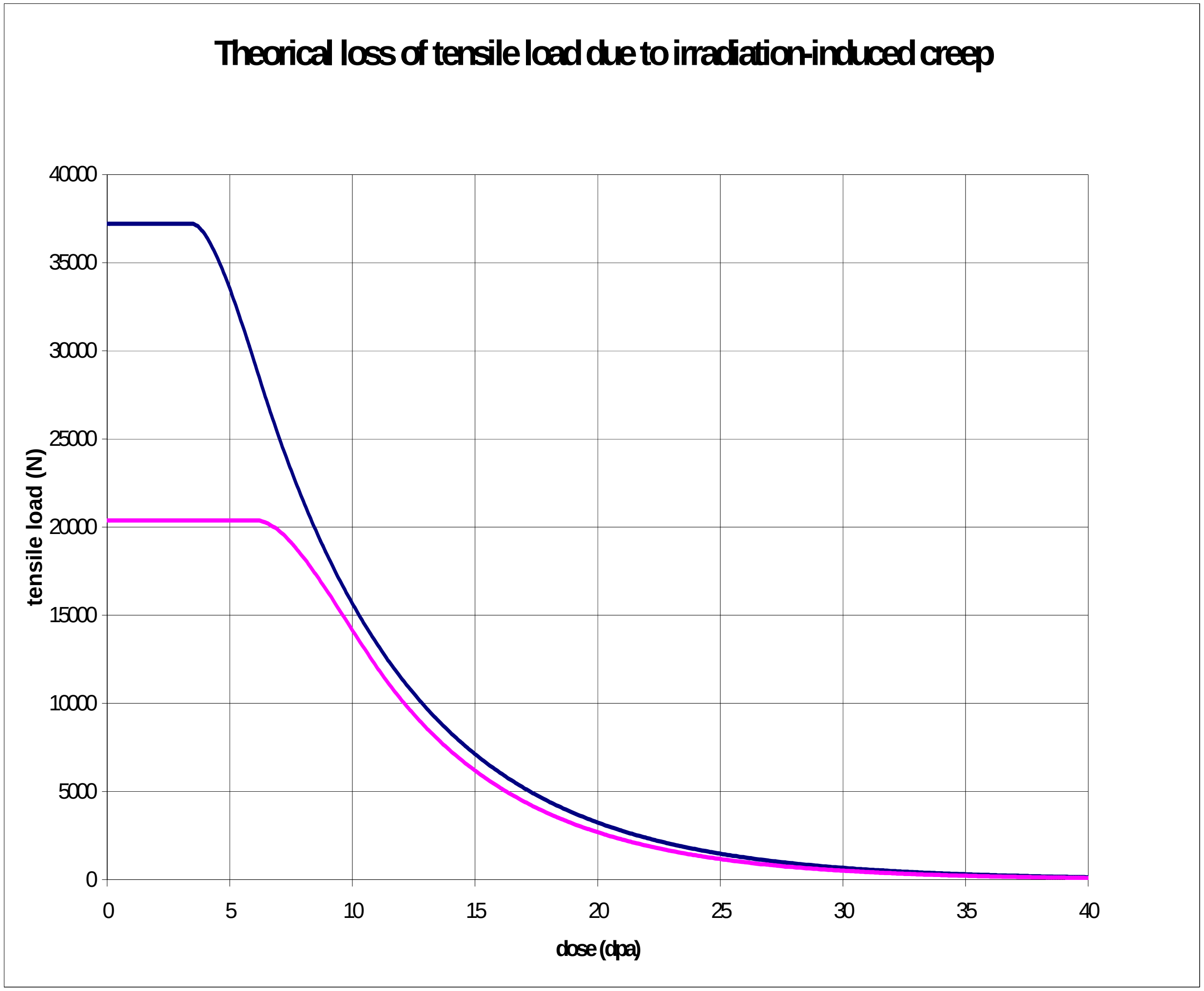


Synthesis of the Creep-Irradiation Results



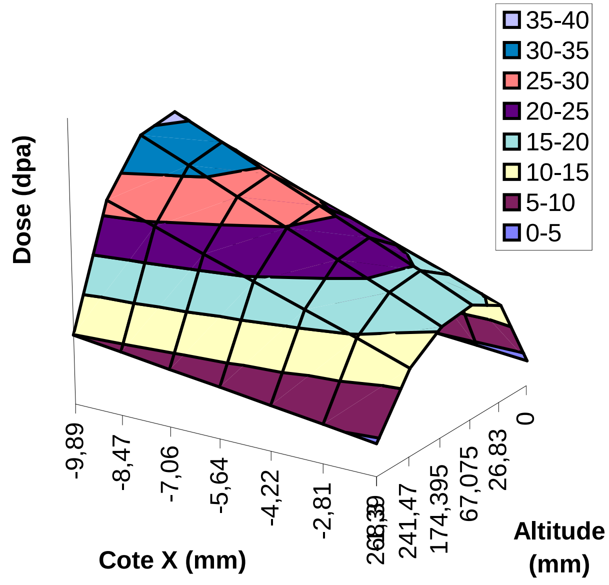
A creep law of the type: $\varepsilon = B_0 \cdot \text{stress} \times \text{dose} - B_1$, where B_0 is the creep rate, B_1 relates to incubation dose and stress is the uniaxial equivalent stress

Stress Relaxation Based on Irradiation Induced Creep (Qualitative)

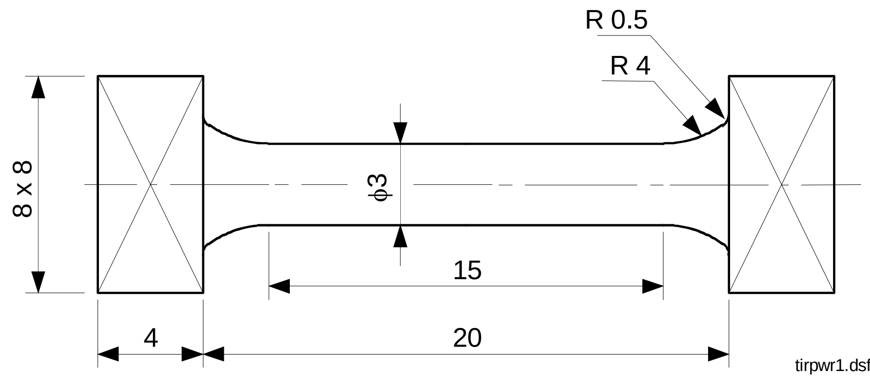


JOBB Constant Load test

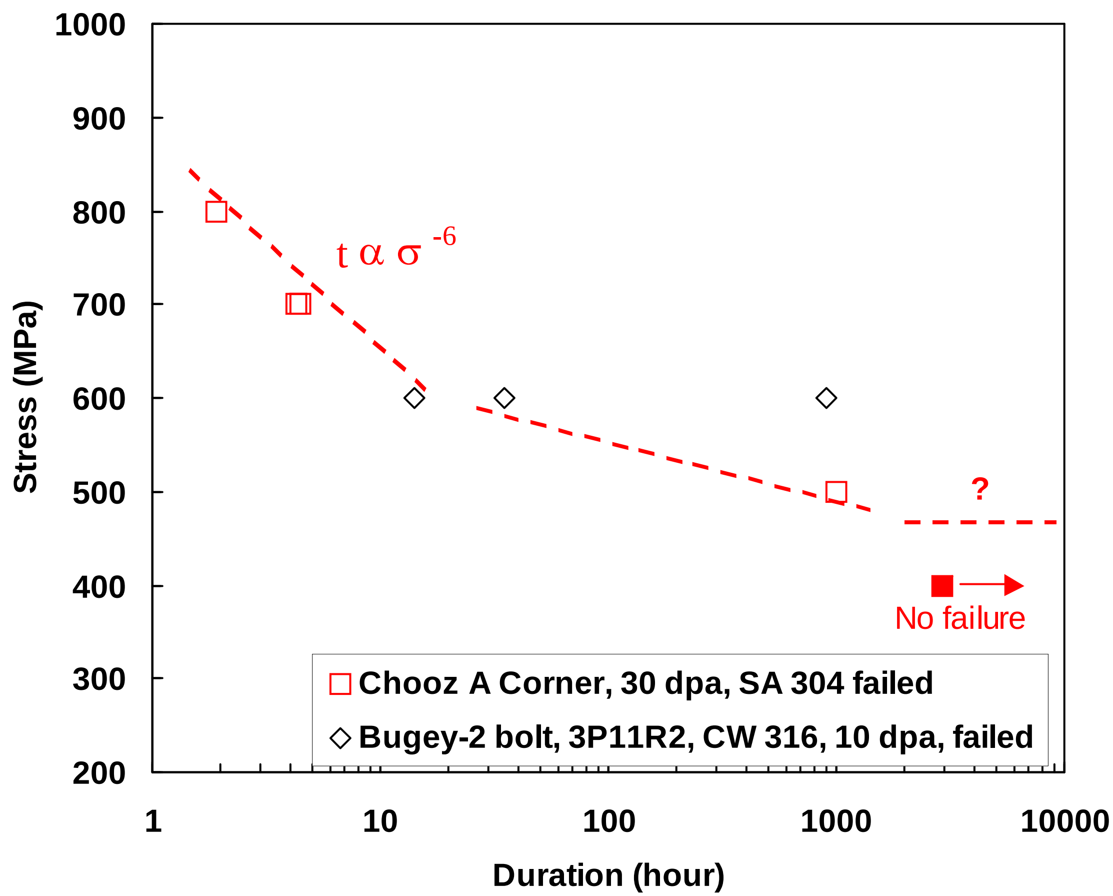
Constant Load Tests on CHOOZ A Material in VTT LAB



Material :
- Chooz A baffle plate
- SA 304 stainless steel
- dose : 26-32 dpa,
- temperature : ~ 310°C,
- YS : 895 MPa ; UTS : 900 MPa



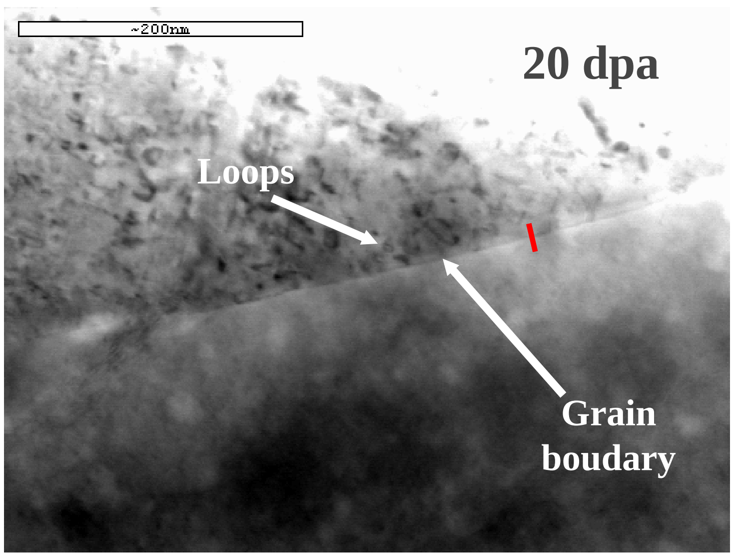
Constant Load Tests on CHOOZ A Material - Stress Versus Time-to-Failure Curve



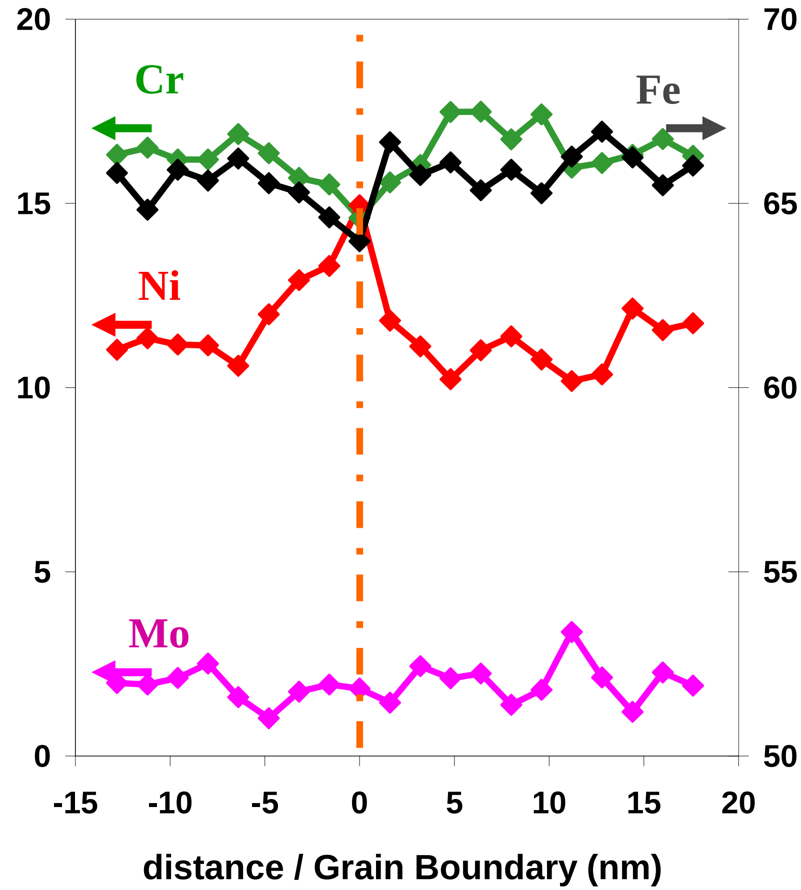
JOBB Microstructure Investigation

Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

Intergranular segregation

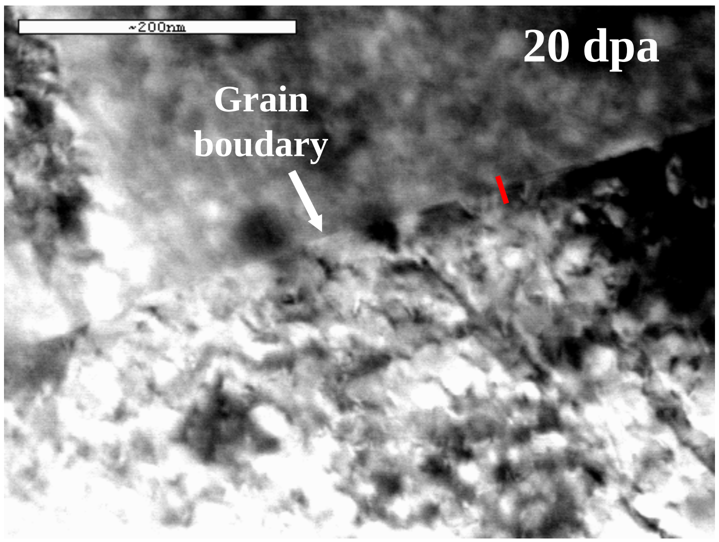


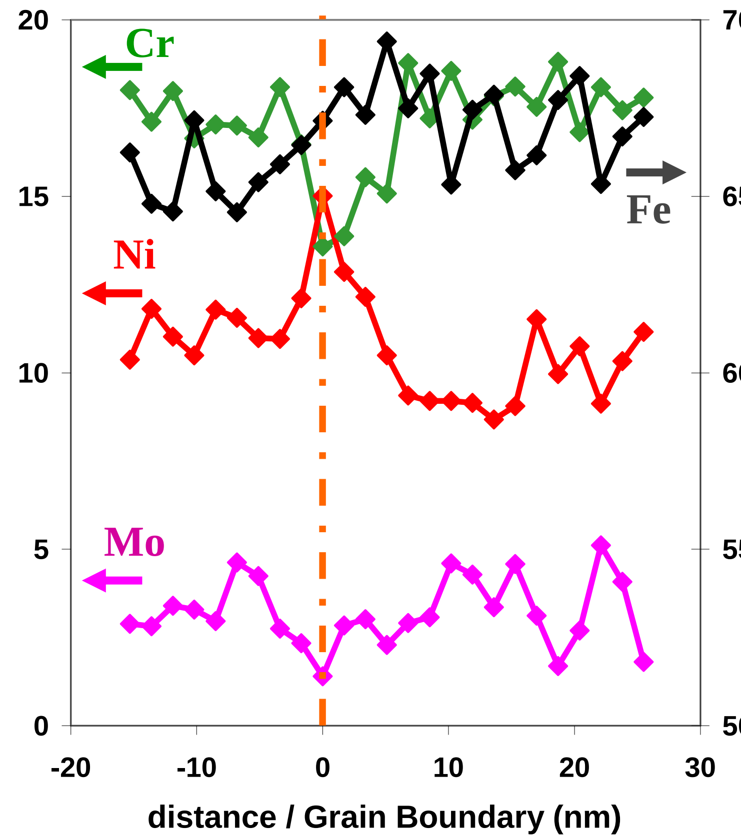
Slight increase in Ni at Grain Boundary
11 ➔ 15 %
Slight decrease in Fe, Cr



Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

Intergranular segregation



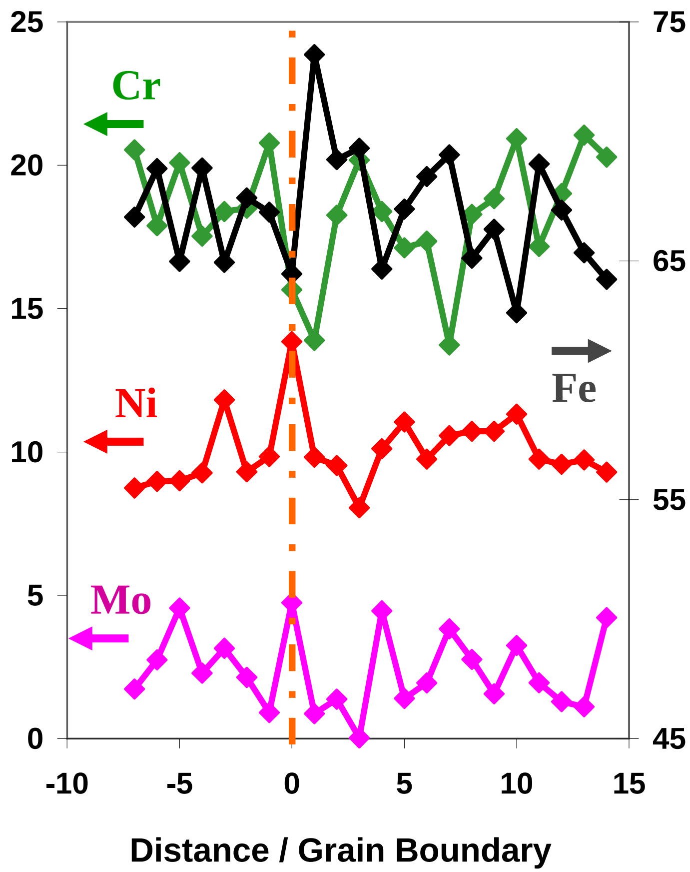


Slight increase in Ni at Grain Boundary
11 ➔ 15 %

Slight decrease in Cr
19 ➔ 14%

Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

Intergranular segregation



10 dpa

Study in progress

Slight increase in Ni 14%

Radiation Induced Segregation in CW 316 Steel Irradiated in BORIS

- Intergranular segregation
  - Slight increase in Ni content
  - Slight decrease in Cr and Fe
  - Slight difference between 10 and 20 dpa
  - Mo shows no change

  To be confirmed with future data at 10 and 40 dpa

- Same tendency observed in CW 316 Bugey Bolt and SA 304 Chooz A Corner

- Baffle/former bolts test program
- Decommissioned PWR internals materials test program
- International IASCC program
- JOBB corrosion test program

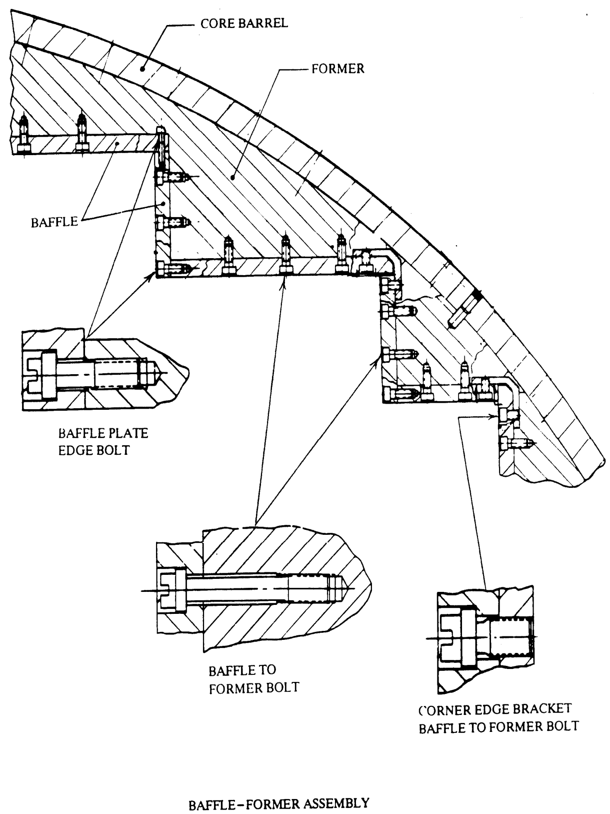
Types of Materials Being Studied

Program	Alloy	Component	Irradiation Source	Irradiation environment
US Baffle-Former bolts	**CW316 SS**	**Baffle-former bolts**	**PWR**	**Water 8-15 dpa**
	347SA SS	**Baffle-former bolts**	**PWR**	**Water 2-21 dpa**
	CW304 SS	**Lock bars & washers**	**PWR**	**Water 20 dpa**
International IASCC Adv. Com.	**CW316 SS**	**BMI thimble**	**PWR**	**Water 01-65 dpa**
	CW 316 SS	**Bar**	**BOR60**	**Sodium 0, 20, 40 dpa**
Decommissioned 304 SS	**304SA SS**	**Baffle**	**PWR**	**Water 0-23 dpa**
	304SA SS	**Former**	**PWR**	**Water 0-18 dpa**
	304SA SS	**Barrel**	**PWR**	**Water 0-0.07 dpa**
	304CW SS	**Baffle-former bolts**	**PWR**	**Water 0-23 dpa**
JOBB/MRP	**CW316 SS-W**	**PWR bolting**	**BOR60**	**Sodium 0, 20 dpa**
	304SA SS- FTI	**Baffle-former bolts**	**BOR60**	**Sodium 0, 20 dpa**
	308 SS Weld-FTI	**Core barrel weld**	**BOR60**	**Sodium 0, 20 dpa**
	347SA SS-W	**PWR bolting**	**BOR60**	**Sodium 0, 20 dpa**

1 dpa~7×10^{20} n/cm^2, E>1MeV

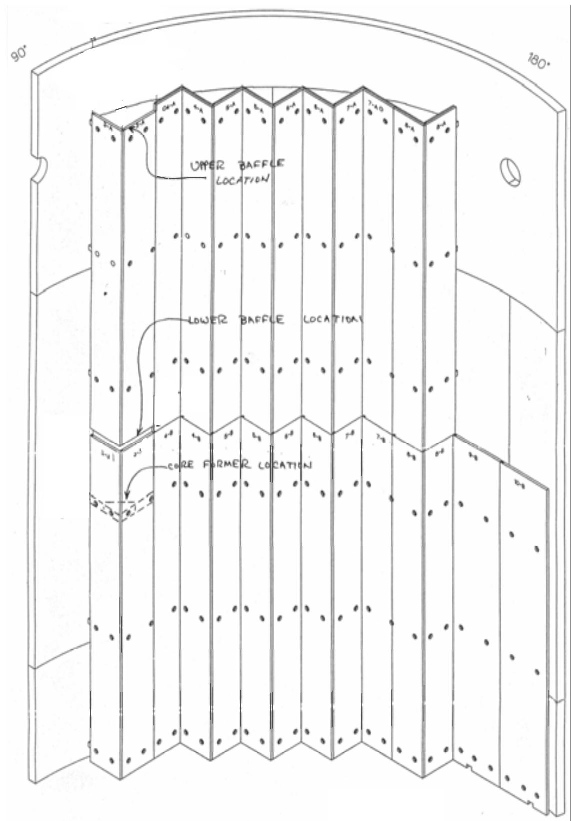
BOR60 is a fast neutron spectrum reactor

Baffle-Former Bolt Configuration



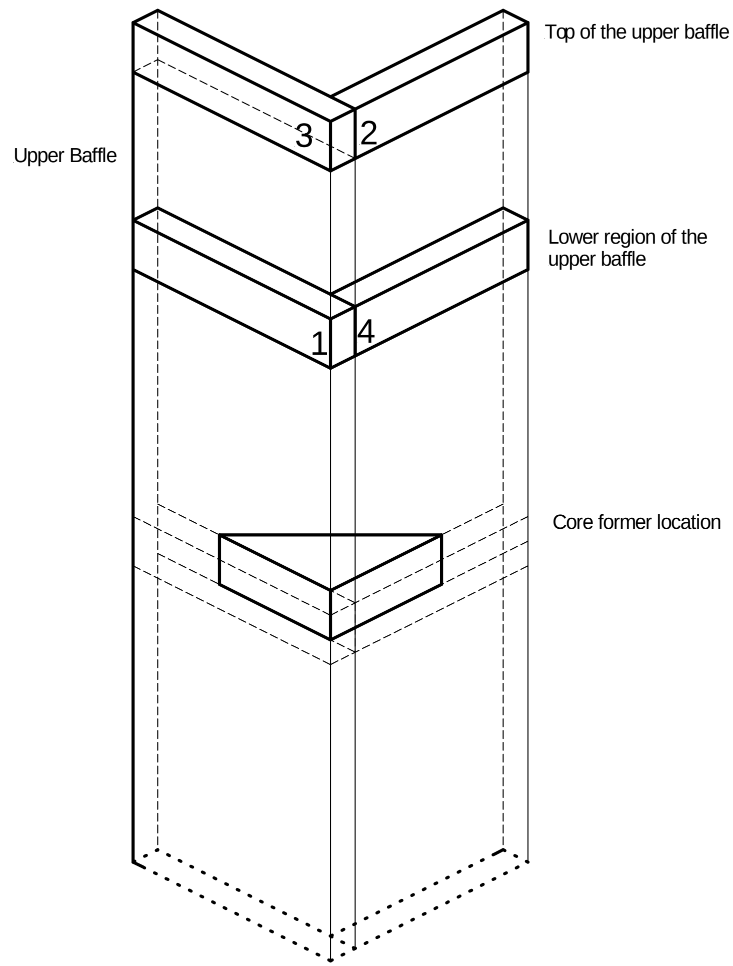
CE reactor design has a welded rather that a bolted assembly

304SS Removed from Decommissioned Plant

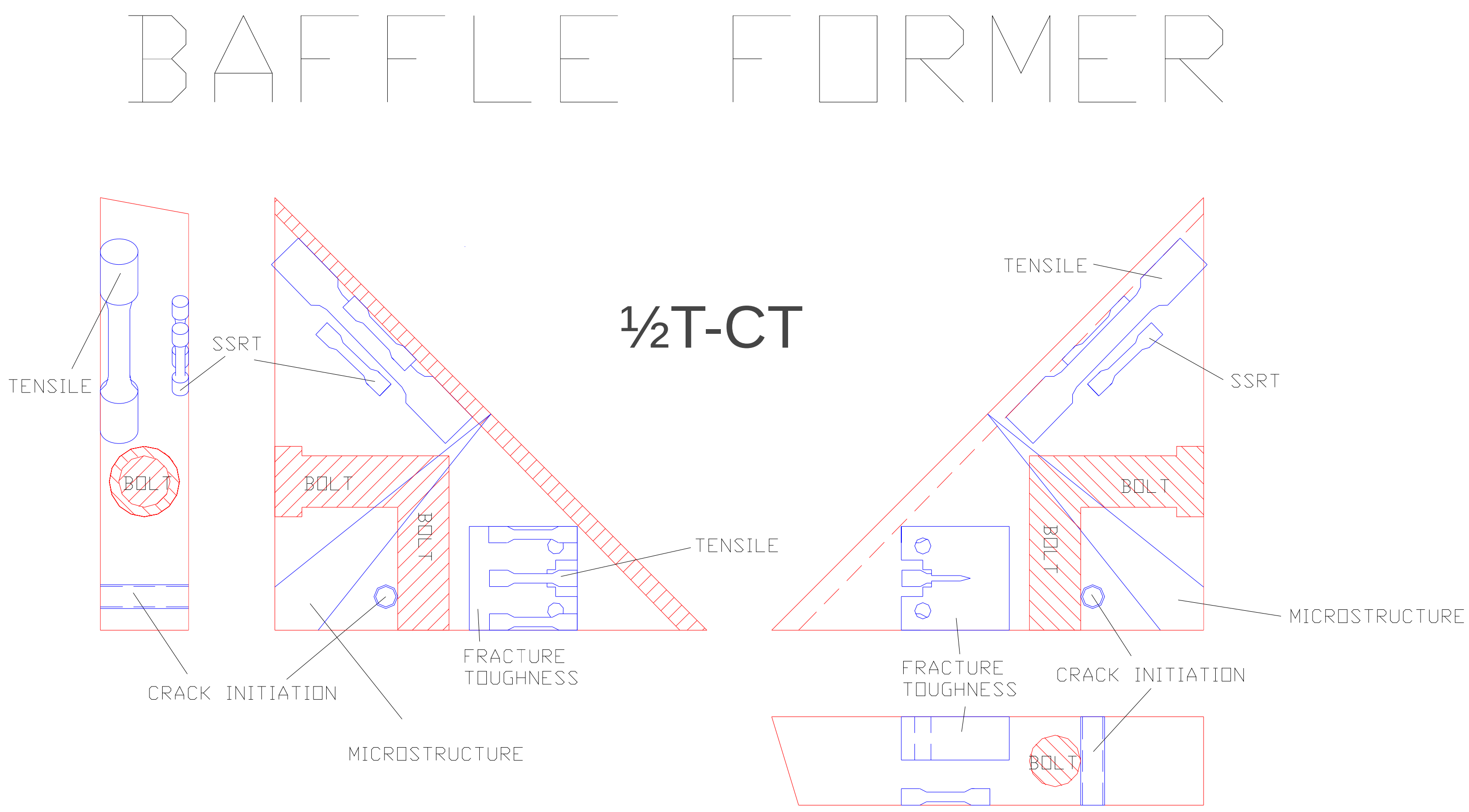


NORTHWEST QUADRANT

Locations of Baffle and Former Plate Samples



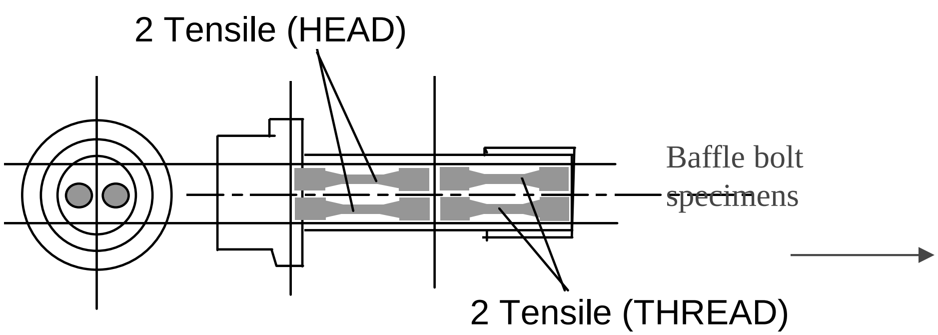
Specimen Cutting

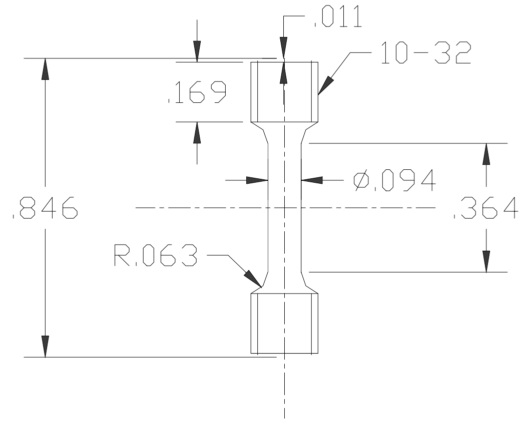


Types of Tests

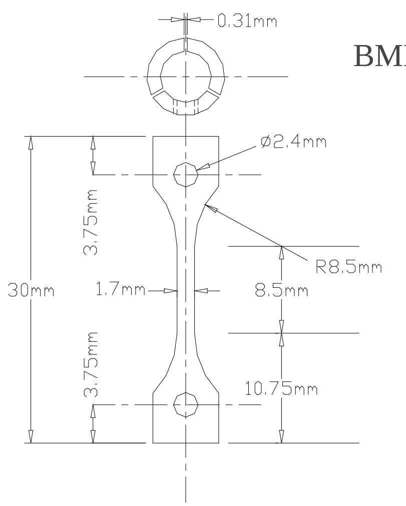
Program	Alloy	Component	Tensile	Fracture Toughness	SSRT	Crack initiation	Crack growth rate	Microstructure /swelling
US B-F bolts	CW316 SS	Baffle-former bolts	x	x	x	x	x	x
	347SA SS	Baffle-former bolts	x	x	x	x	x	x
	CW304 SS	Lock bars	x		x			x
International IASCC Adv. Com.	CW316 SS	BMI thimble	x		x			x
	CW 316 SS	Bar	x		x			x
Decommissioned 304 SS	304SA SS	Baffle	x	x	x	x	x	x
	304SA SS	Former	x	x	x	x	x	x
	304SA SS	Barrel	x	x	x	x	x	x
	304SA SS	Baffle-former bolts	x		x			x
JOBB/MRP	CW316 SS-W	Baffle-former bolts	x		x			x
	304SA SS- FTI	Baffle-former bolts	x		x			x
	308 SS Weld-FTI	Core barrel weld	x		x			x
	347SA SS-W	Baffle-former bolts	x		x			x

Tensile Testing

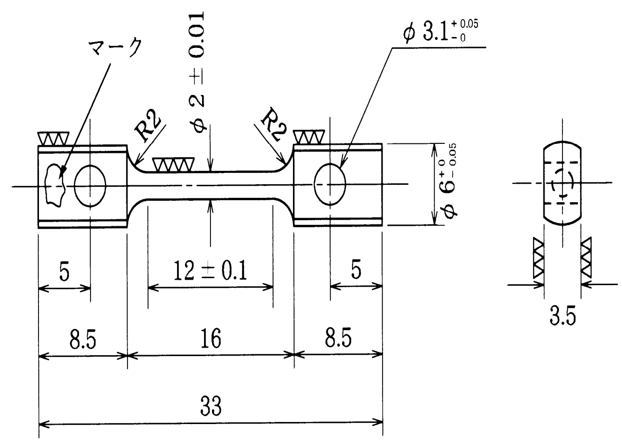




1-0.25 dia 304SS from former



BMI Tubing

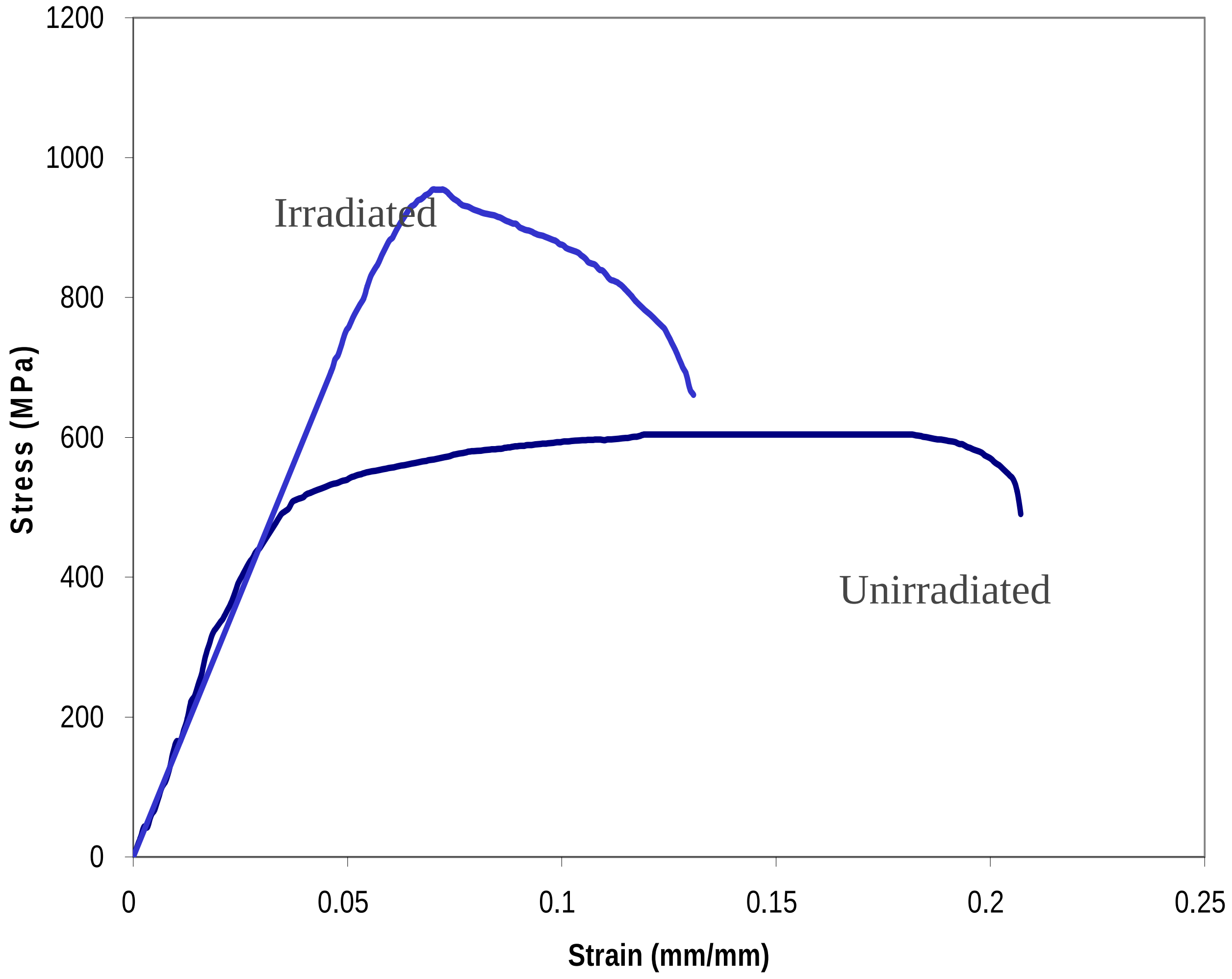


BOR reactor irradiated specimens

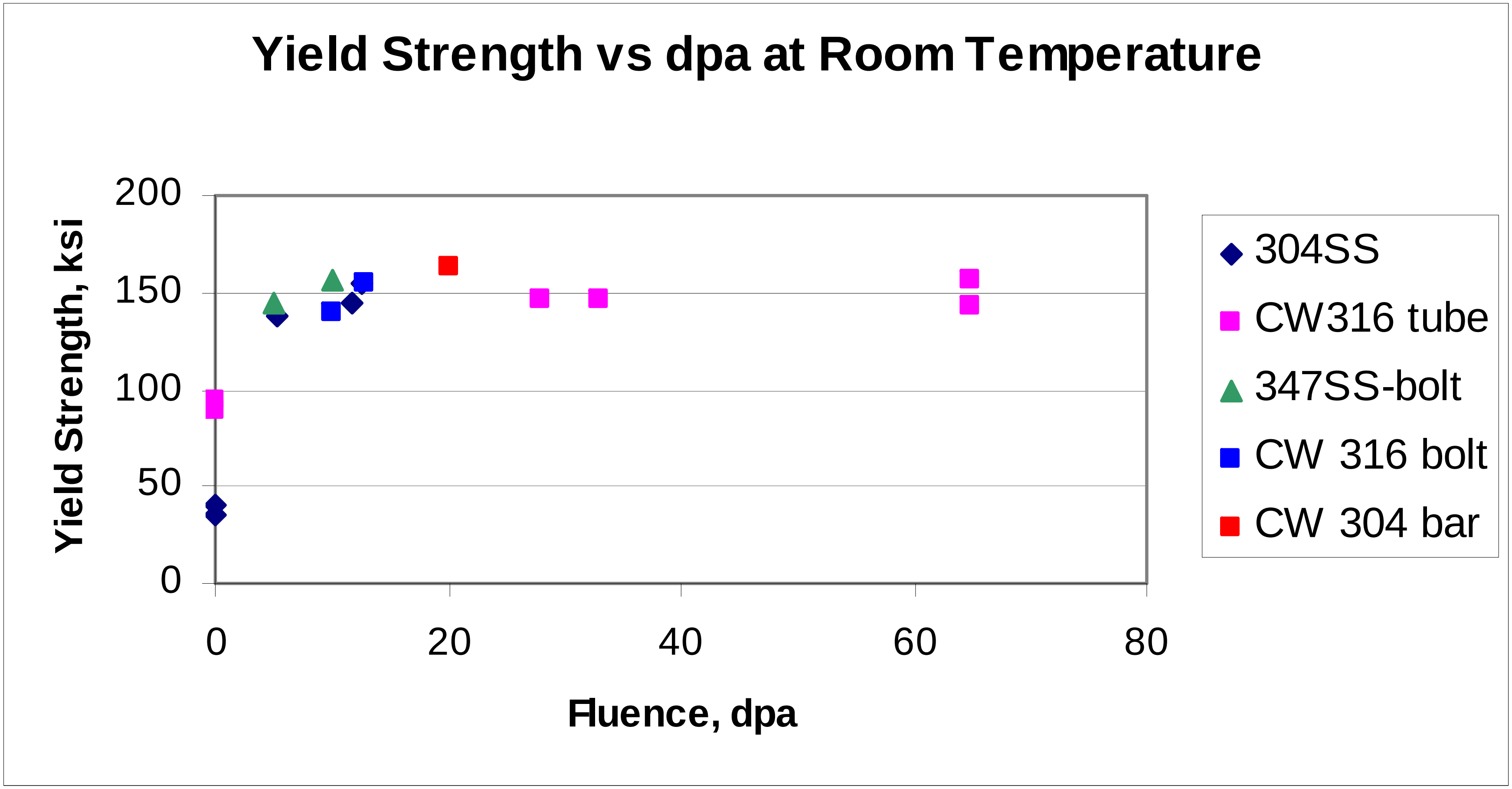
Tensile Test Parameters

- Generally subsized specimens sized relative to the available component size
- ASTM E8 & 21 as applicable
- Air environment
- Primarily 608^{o}F (320^{o}C), + room temperature

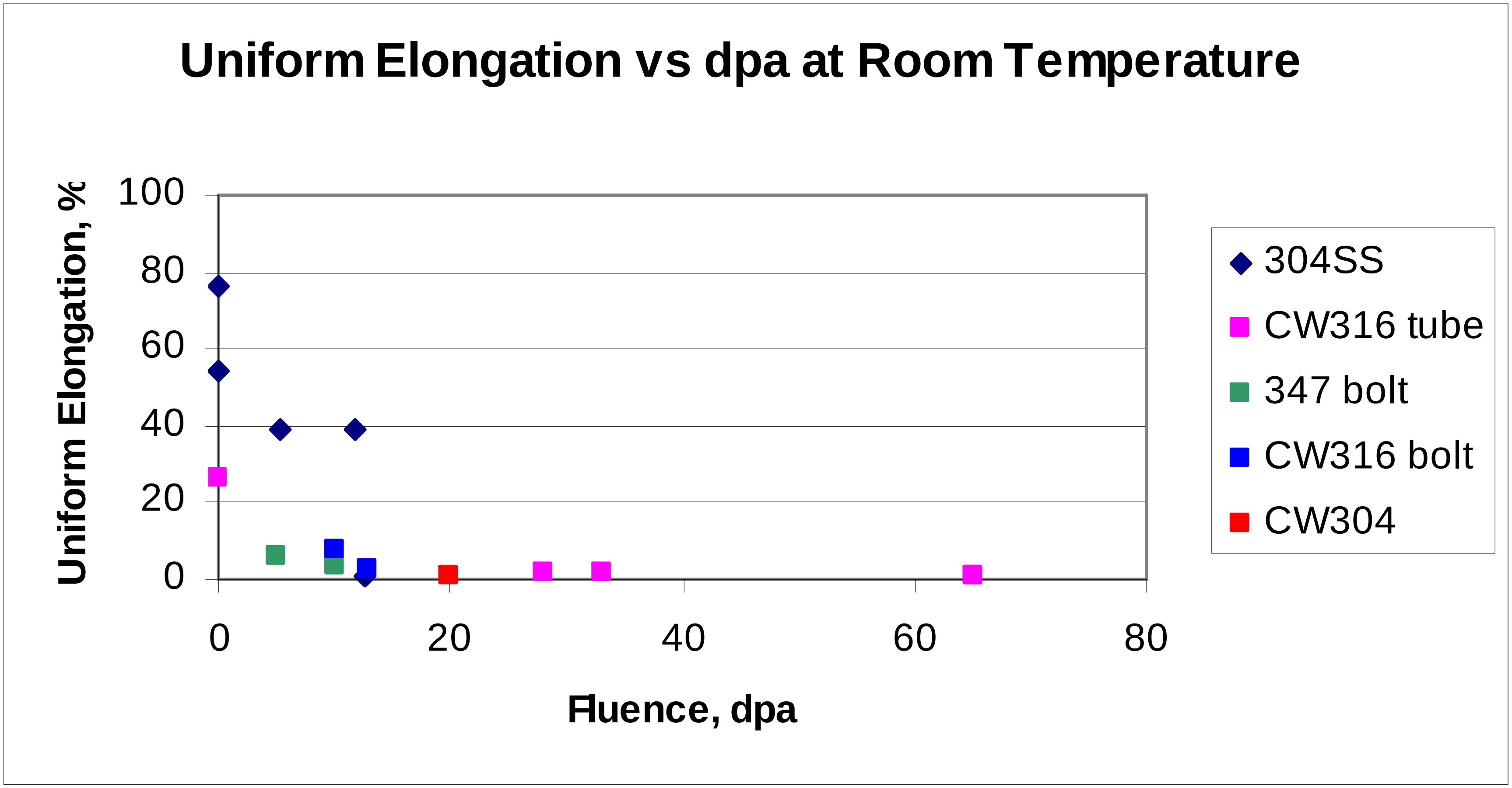
Effects of Irradiation on the Stress-Strain Curve



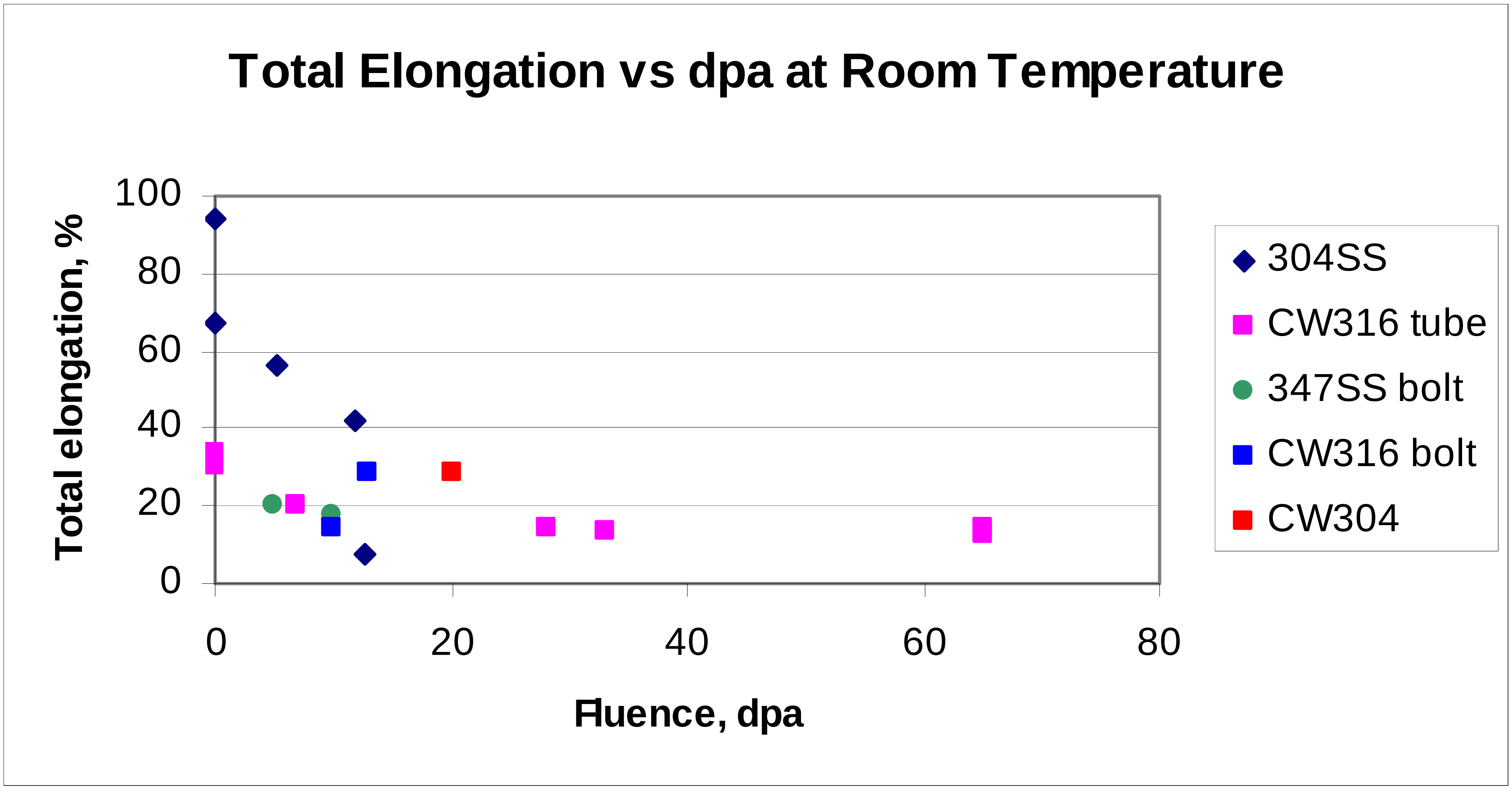
Tensile Test Results – Yield Stress at RT



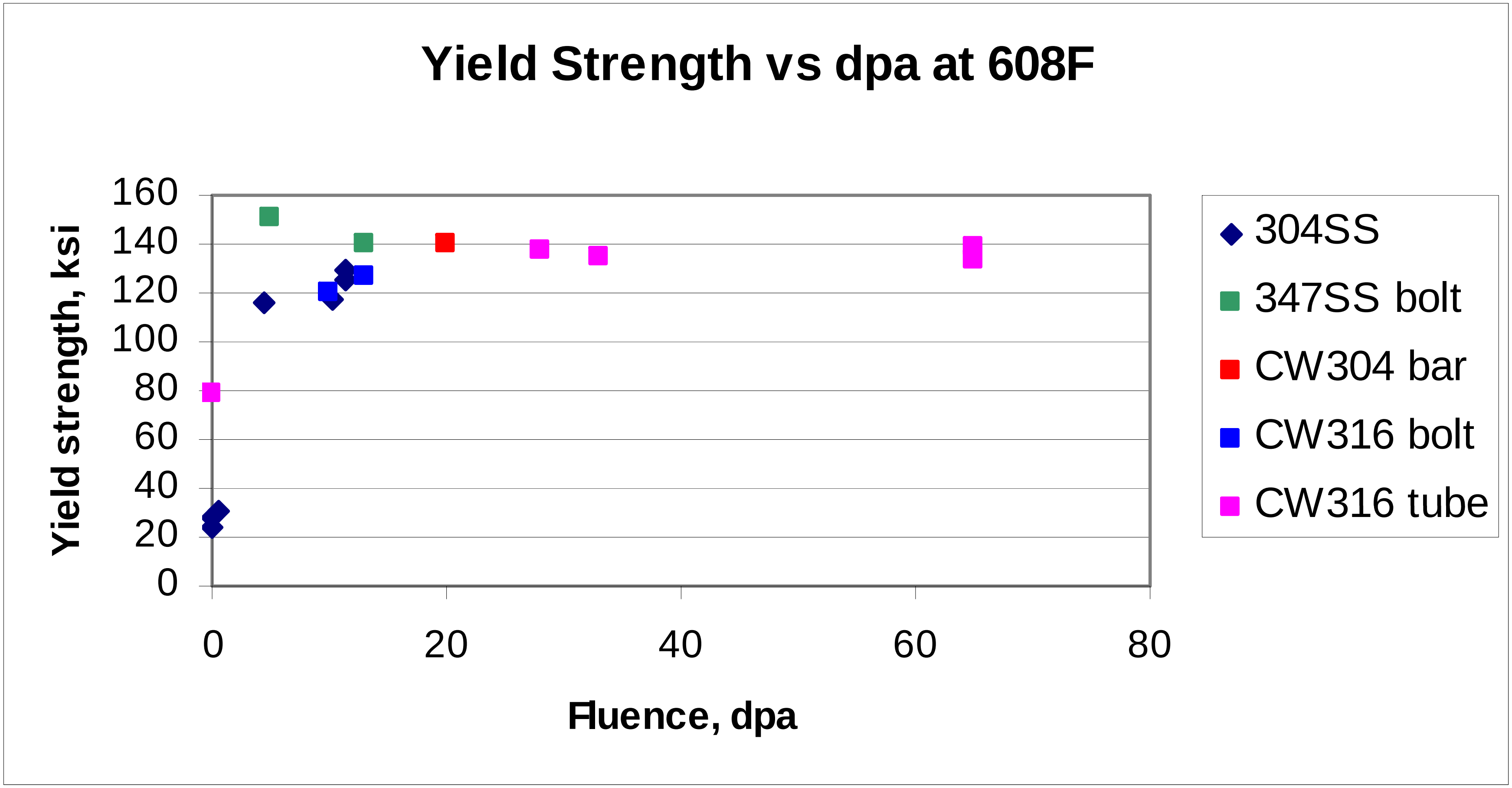
Tensile Test Results – Uniform Elongation at RT



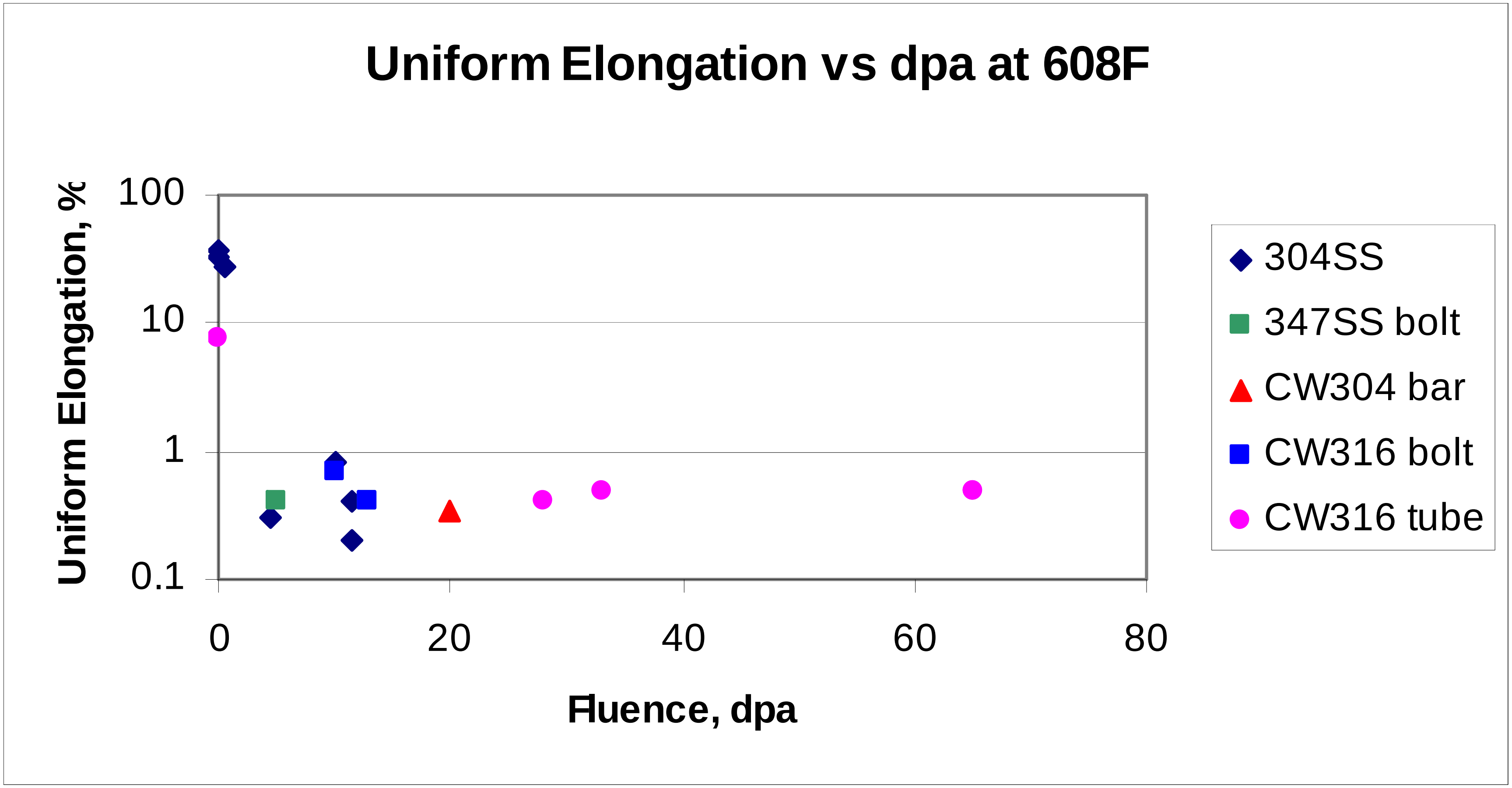
Tensile Test Results – Total Elongation at RT



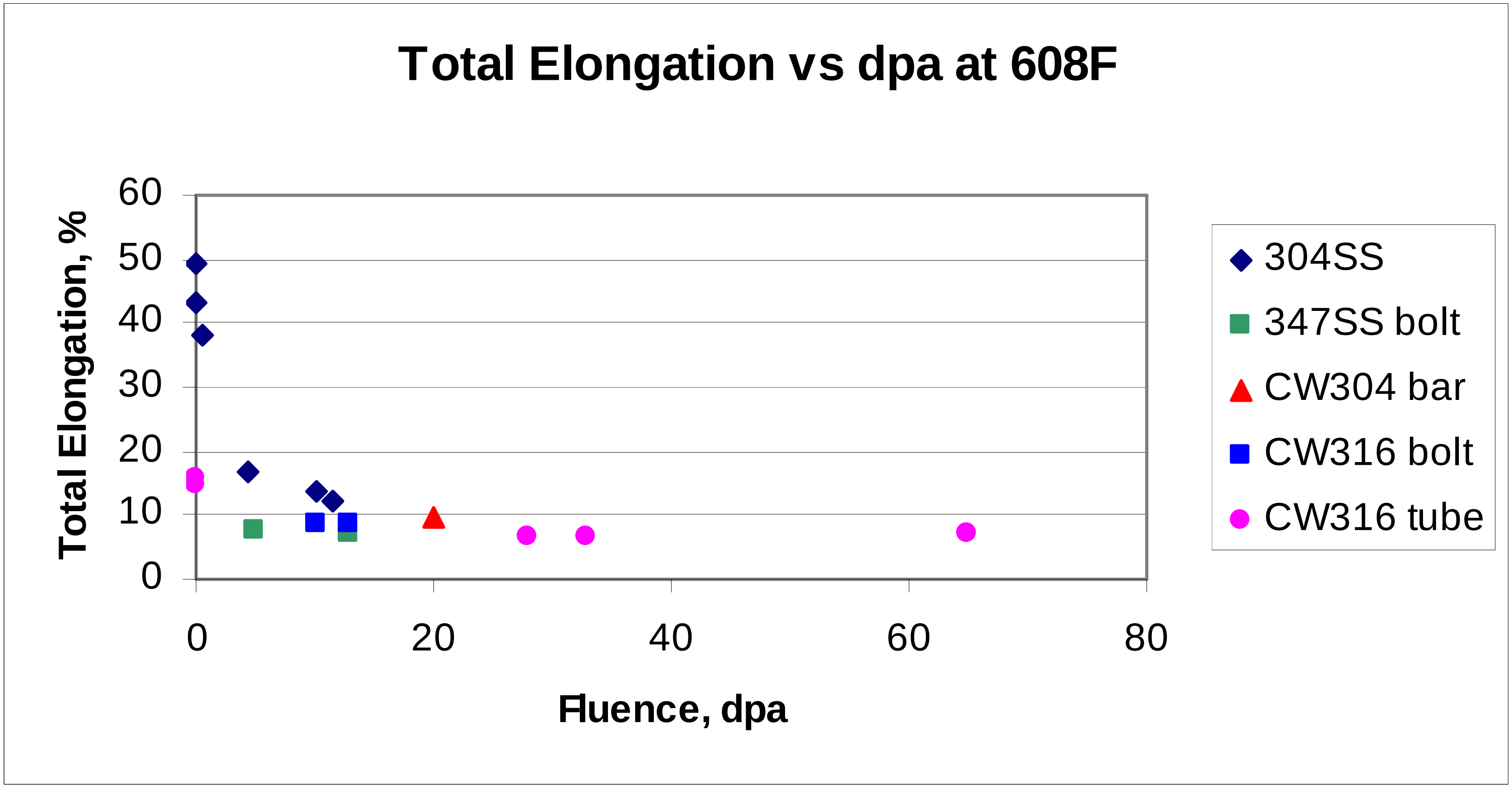
Tensile Test Results – Yield Strength at 608F



Tensile Test Results – Uniform Elongation at 608F

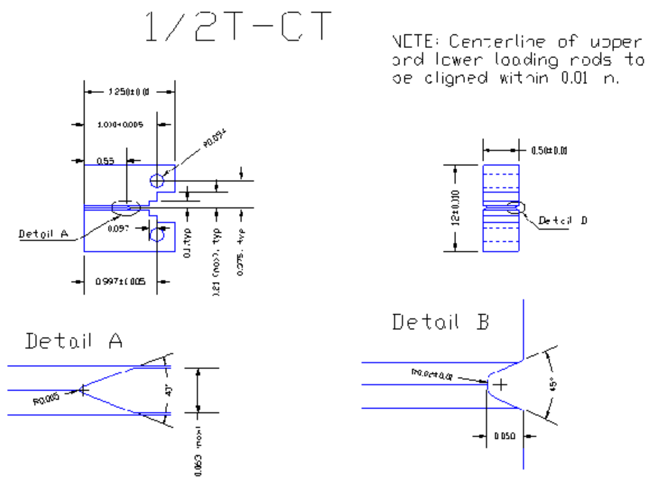


Tensile Test Results – Total Elongation at 608F

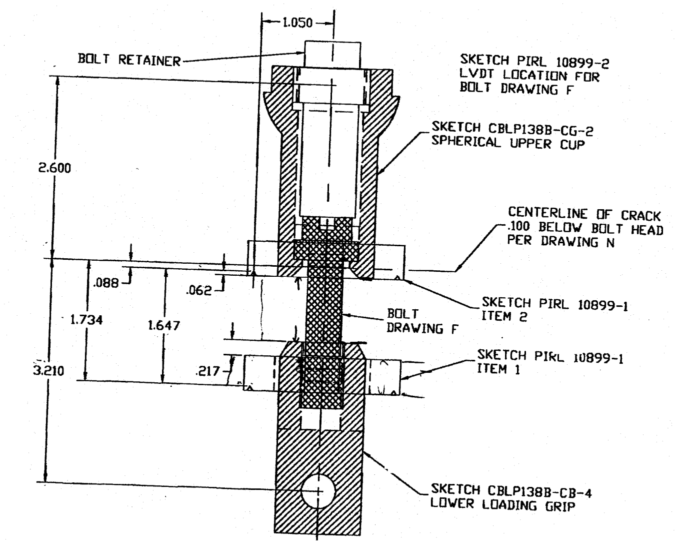


Fracture Toughness Testing Specimens

Decommissioned 304 Material Program – Standard 1/2T and 1T-CT Specimens



Baffle bolt Program used actual bolts with ½ through crack under head

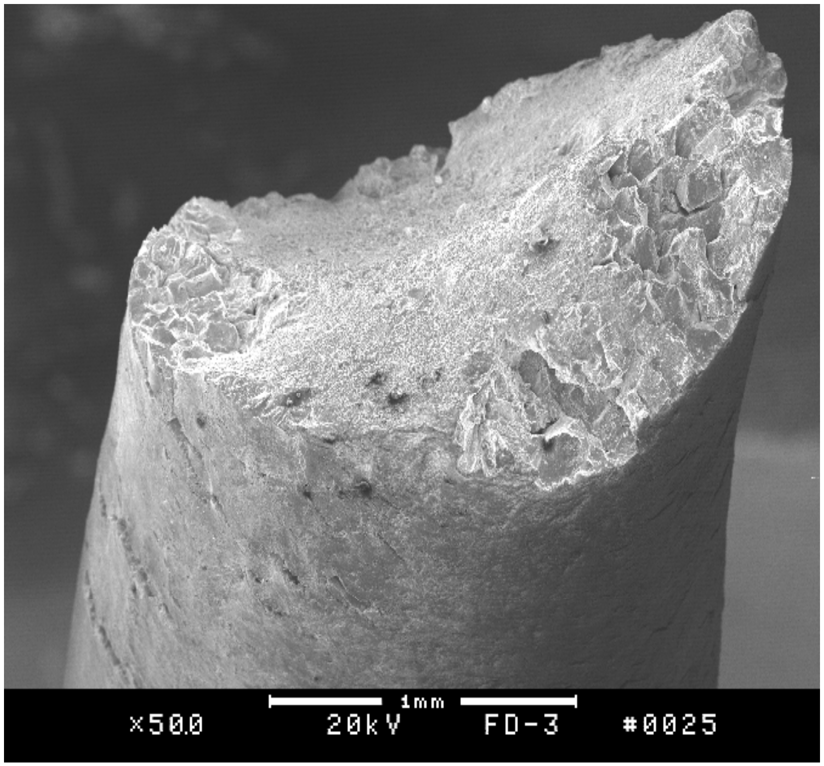


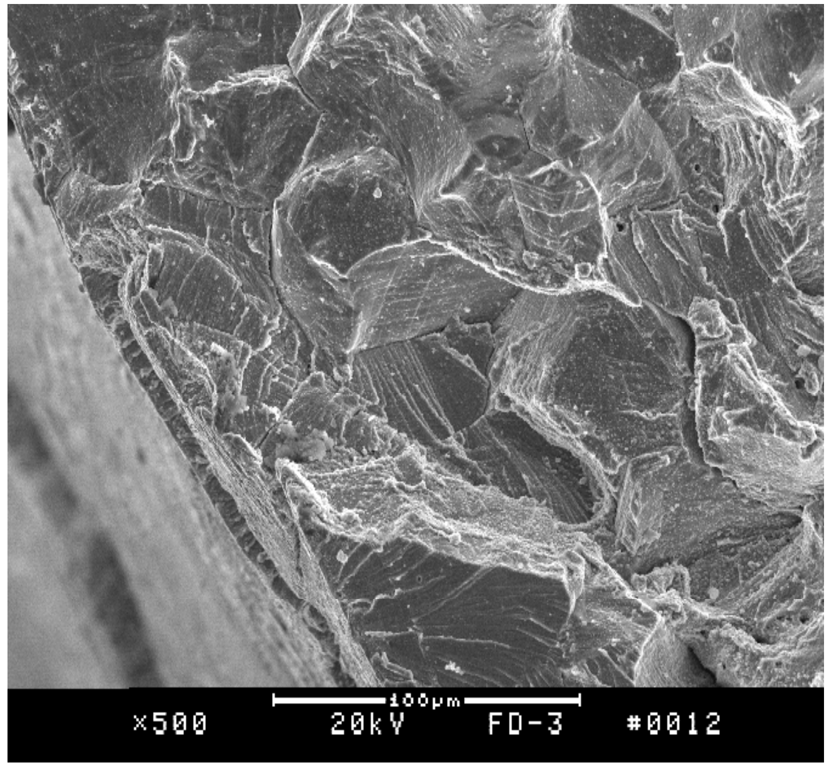
IASCC Susceptibility - SSRT Test Parameters

- Same specimen designs as for tensile testing
- Tensile test at ~10^{-7} s^{-1}
- Simulated PWR water

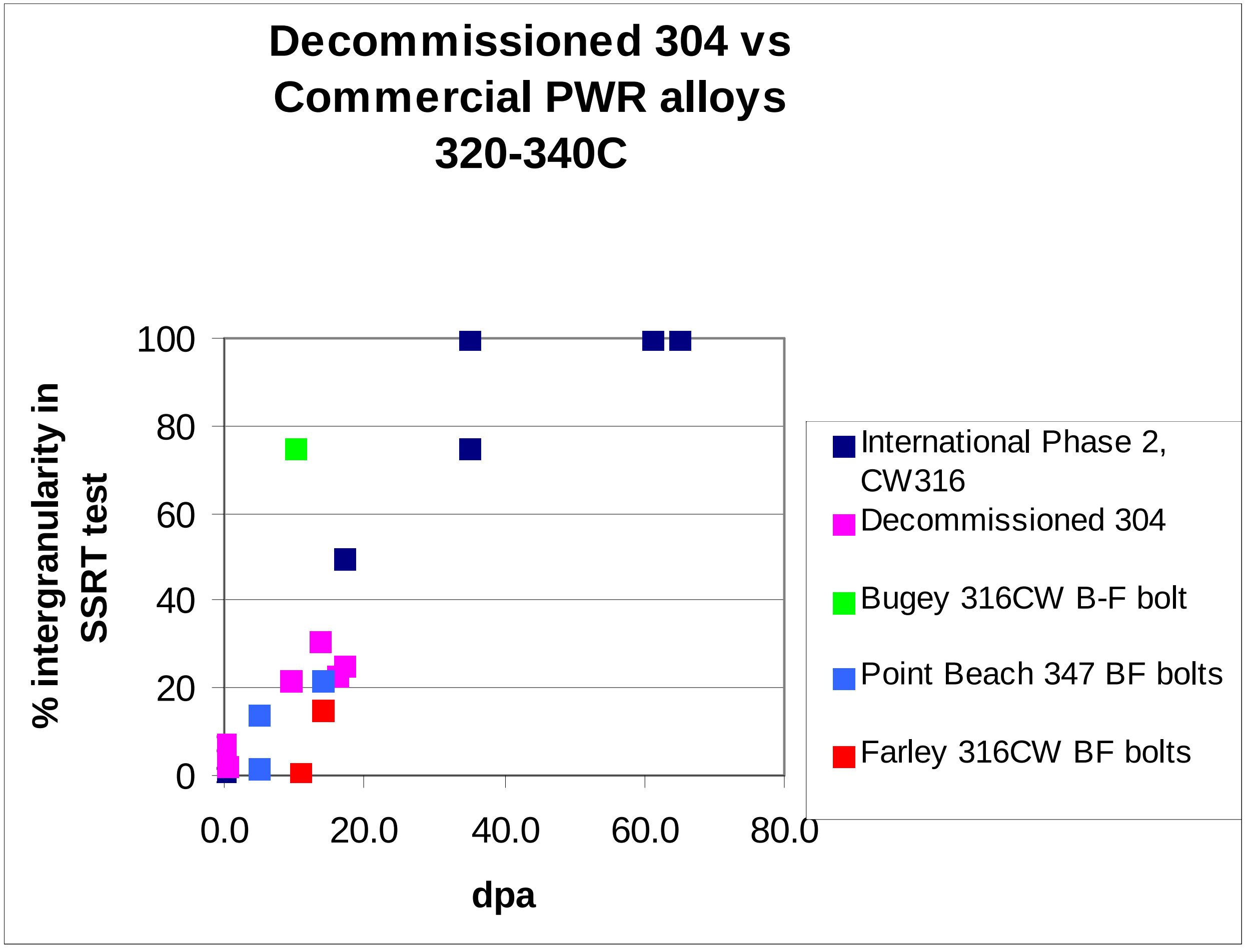
  H_3BO_3 1000 ppm as B
  LiOH 2 ppm as Li
  Dissolved oxygen< 5 ppb
  Dissolved hydrogen 30 cc/kg
  Chloride < 30 ppb
  Fluoride < 30 ppb
  340^{o}C

SSRT Fractography



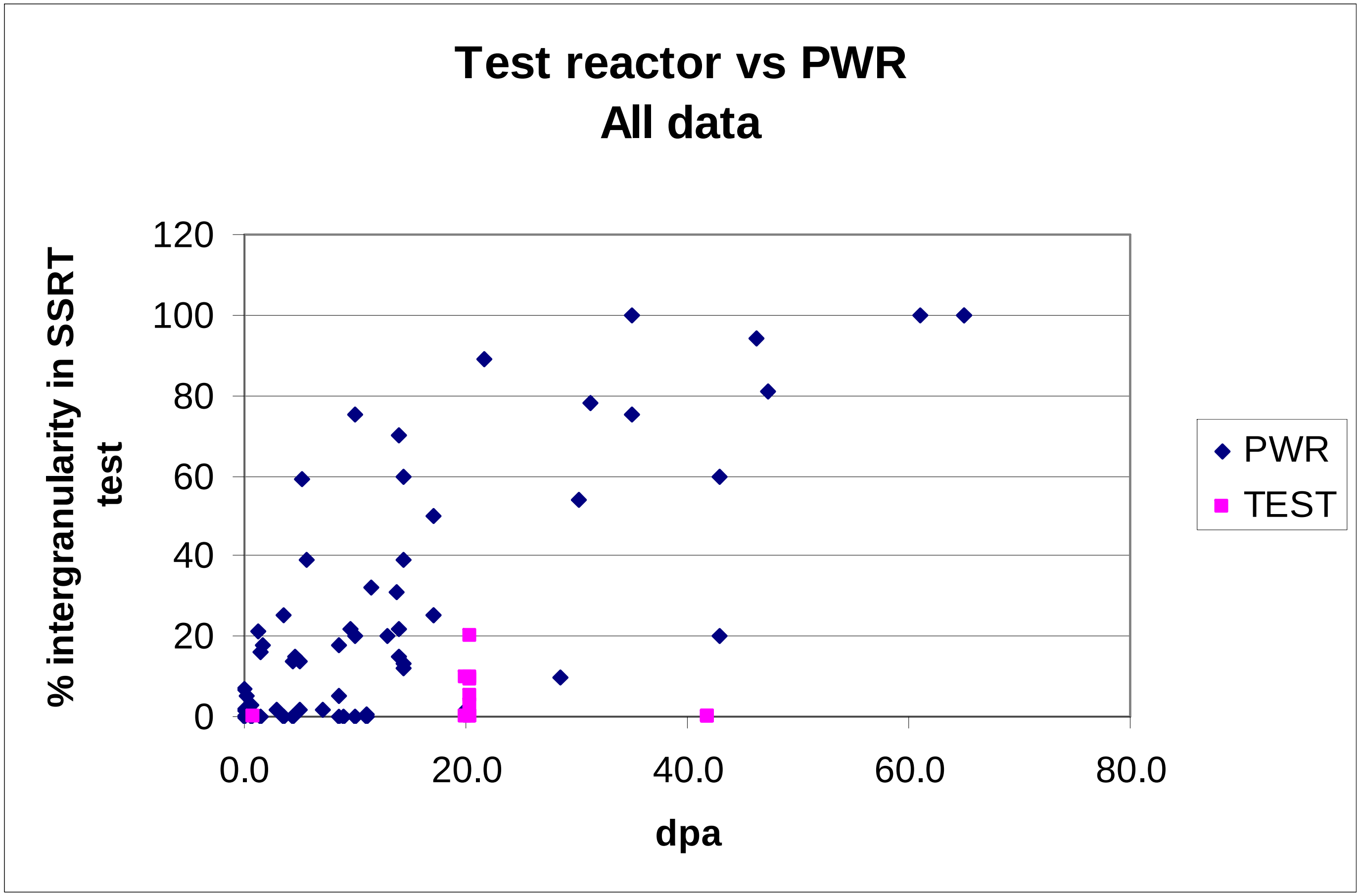


IASCC Test Results

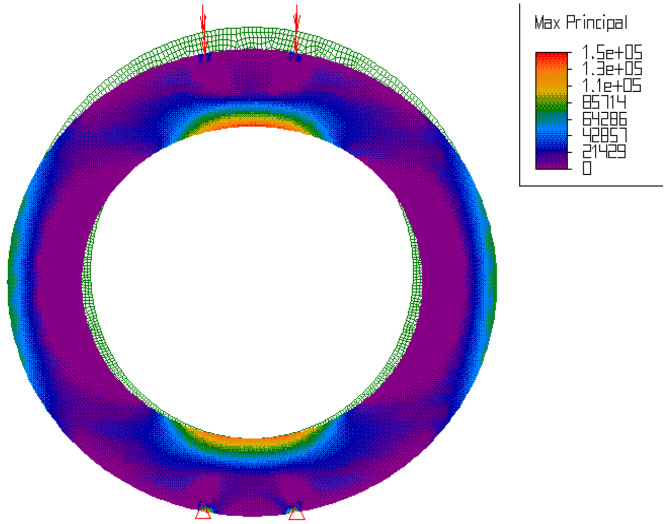


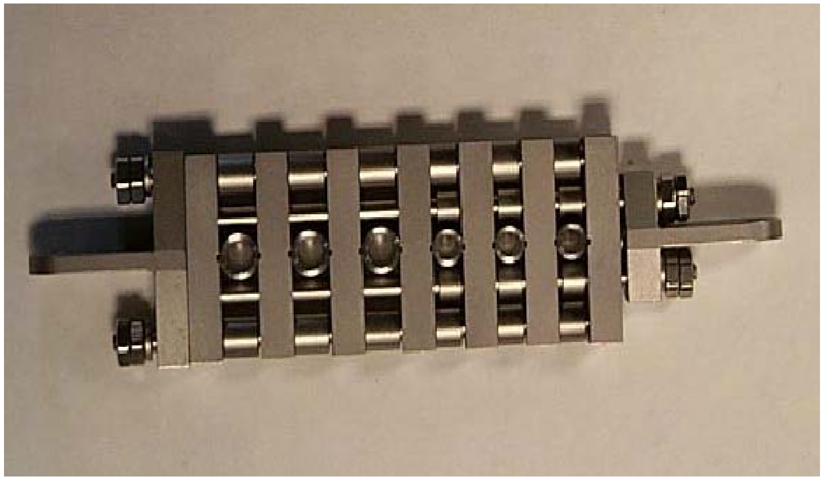
SSRT Test Results

Effects of Reactor Source



Crack Initiation Testing





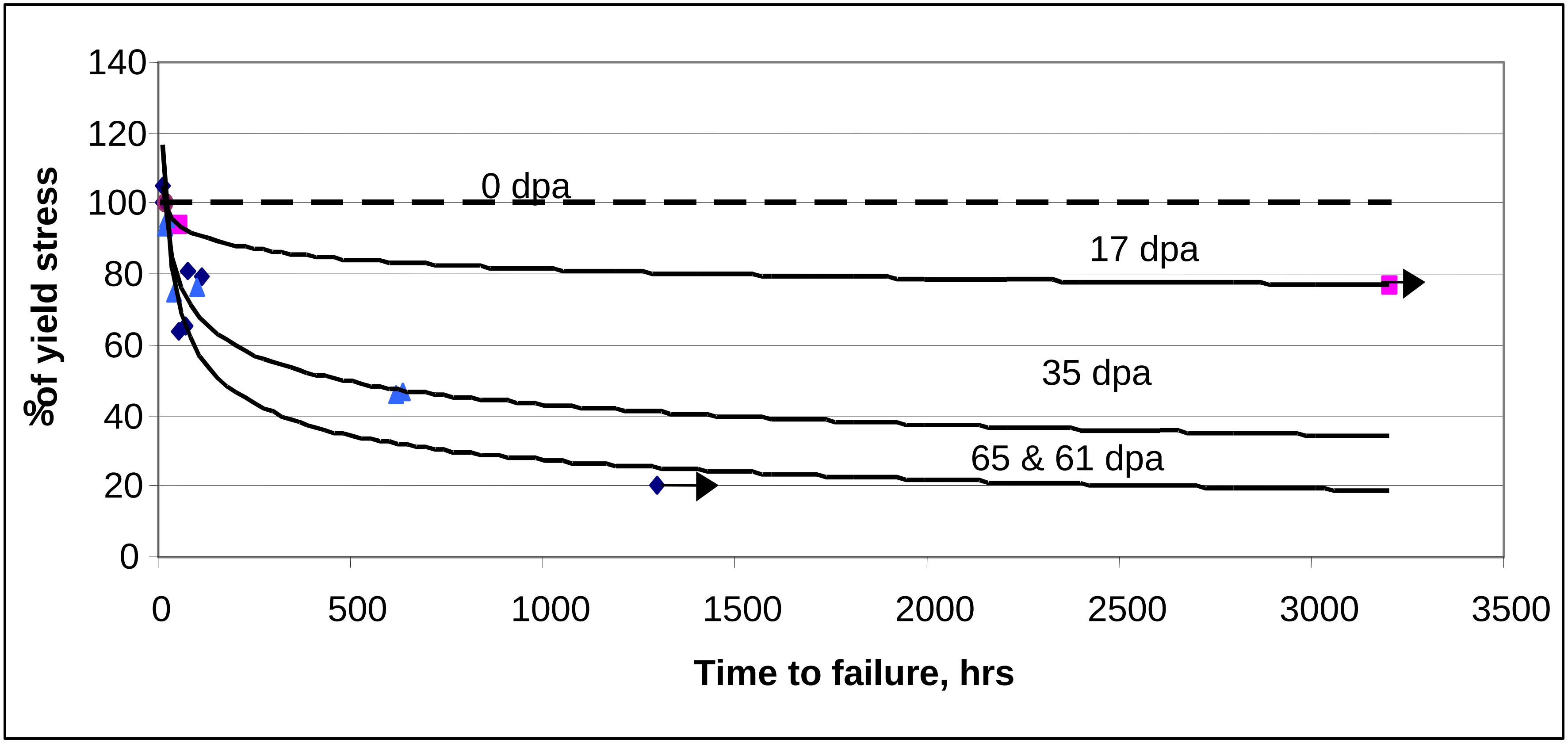
Crack Initiation Test Parameters

- Stressed “O” rings at constant load
- 316 CW thimble tube specimens
- Time to failure (~time to crack initiation)
- Simulated PWR water

  H_3BO_3 1000 ppm as B
  LiOH 2 ppm as Li
  Dissolved oxygen< 5 ppb
  Dissolved hydrogen 30 cc/kg
  Chloride < 30 ppb
  Fluoride < 30 ppb

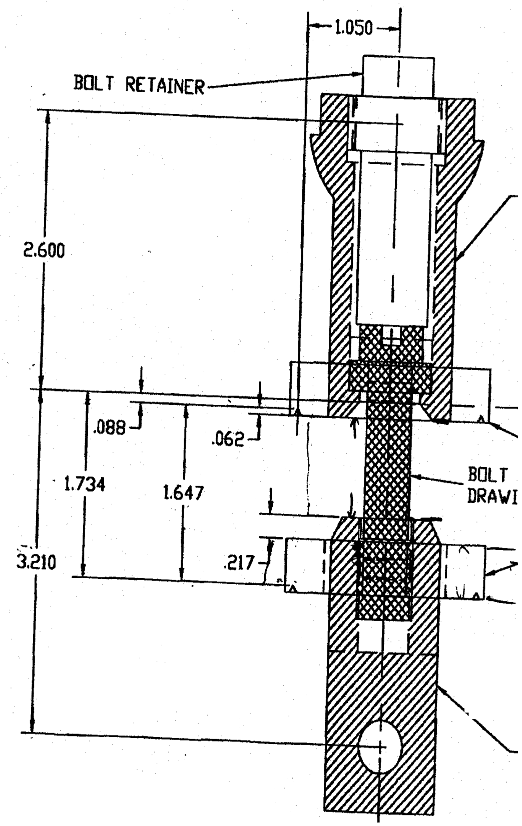
- 340^oC

Crack Initiation Results



*No failures in BOR60 irradiated specimens at up to 40 dpa and loads equal to 120% of the yield strength

Crack Growth Rate Measurements

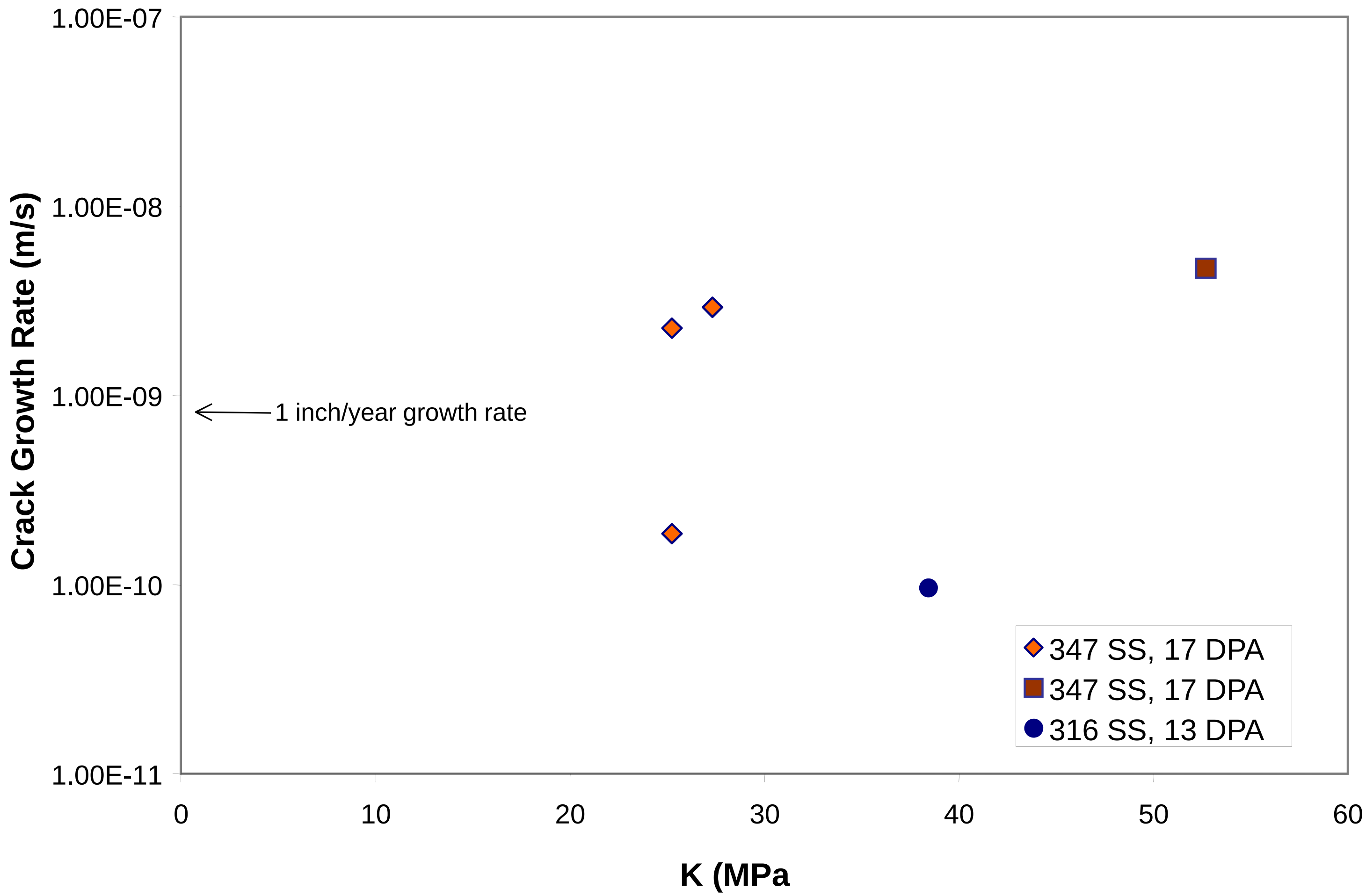


- Constant K
- Simulated PWR water

H_3BO_3	1000 ppm as B
LiOH	2 ppm as Li
Dissolved oxygen< 5 ppb	
Dissolved hydrogen 30 cc/kg	
Chloride	< 30 ppb
Fluoride	< 30 ppb

- 320^{o}C

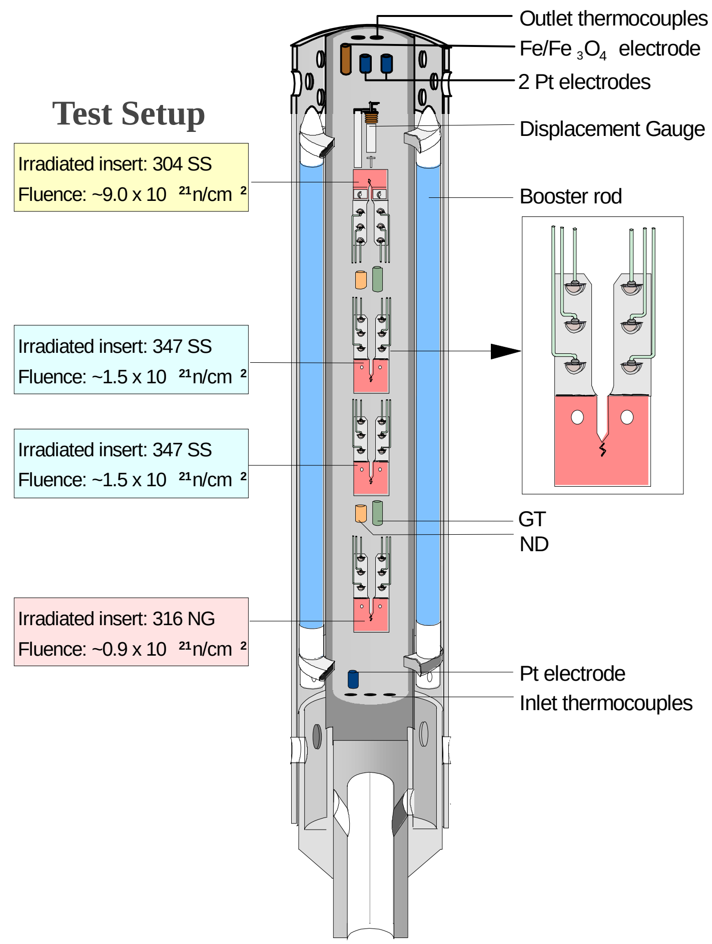
Crack Growth Rate Measurement Results



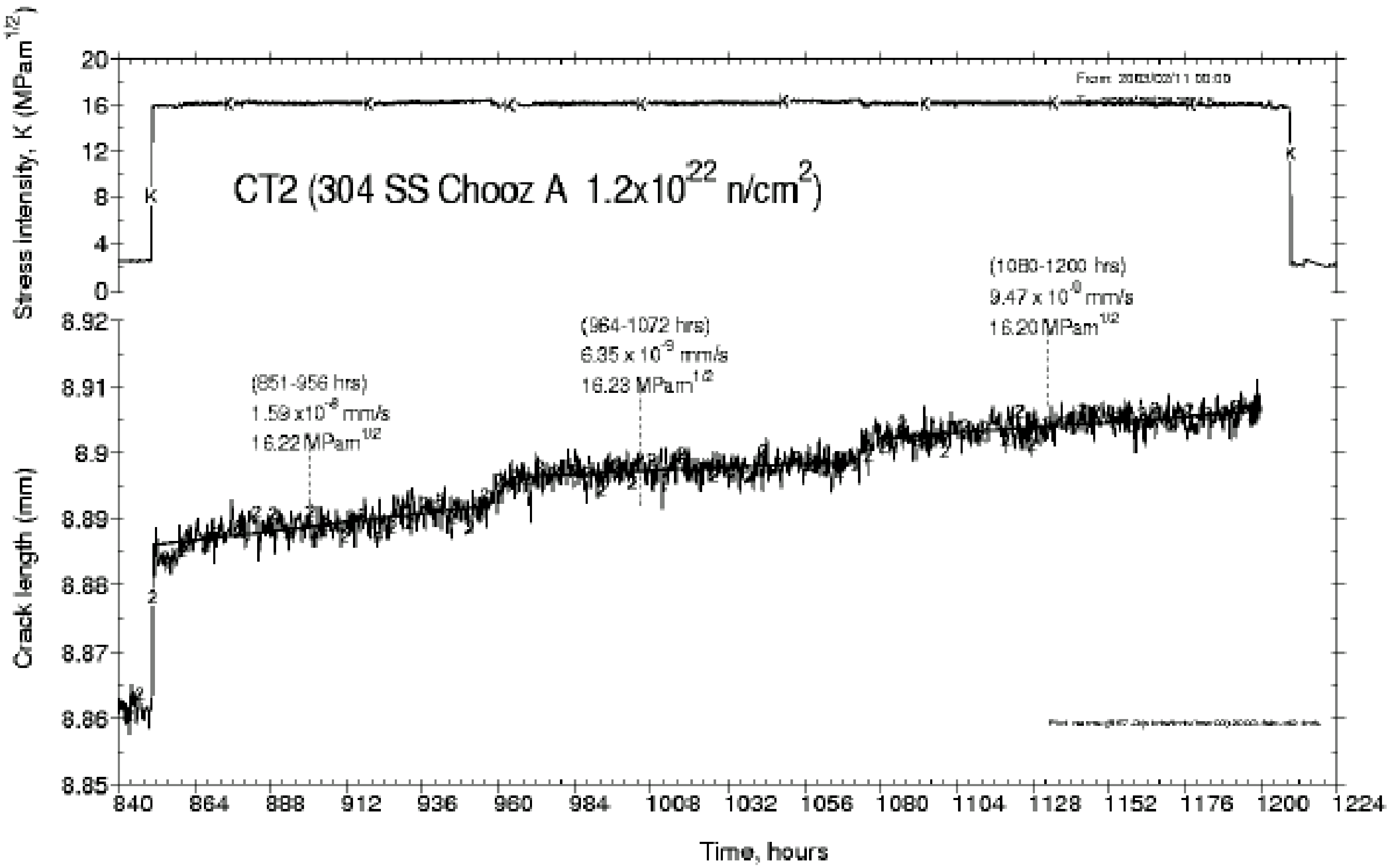
Halden Crack Growth Test Program

Halden Crack Growth Test in PWR Environment

- In-core, long term, crack growth data under PWR conditions of 2 ppm Li, 1200 ppm B and 2-4 ppm hydrogen at 340°C
- Four 304 SS CT specimens with fluence of 2.5 x 10^{22}, 2.5 x 10^{22}, 1.2 x 10^{22} and 9.0 x 10^{21} n/cm^2 (>1 MeV)

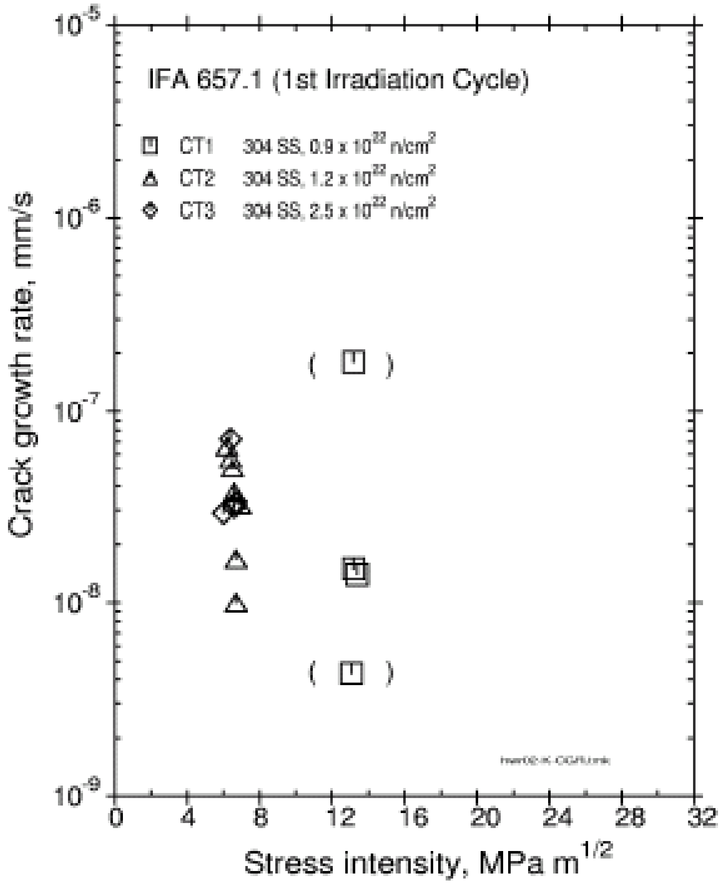


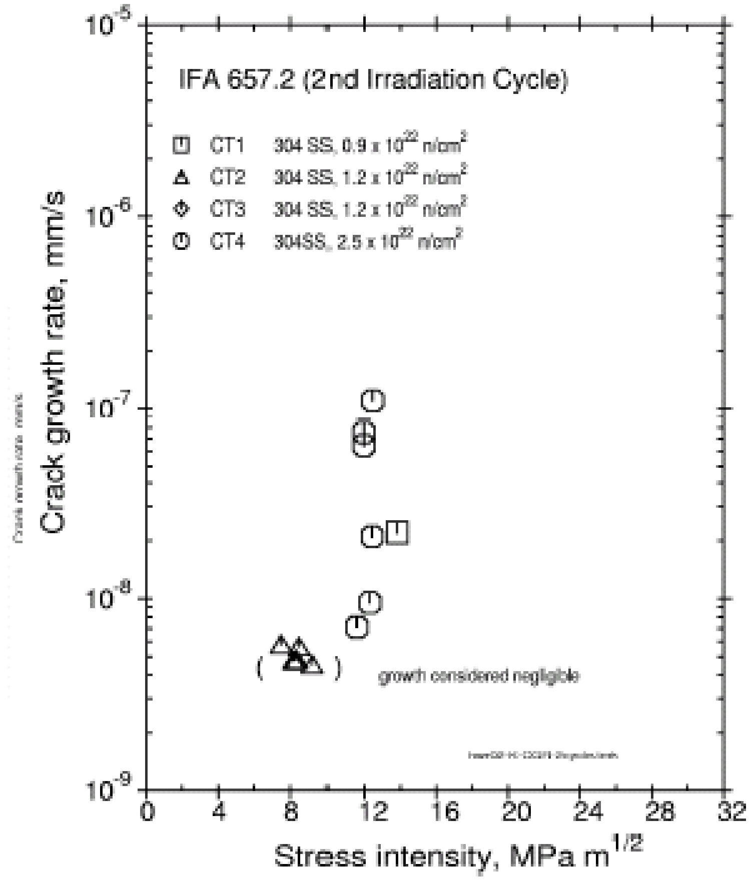
Halden 304 SS Crack Growth Test Results – An Example



Halden PWR Crack Growth Test

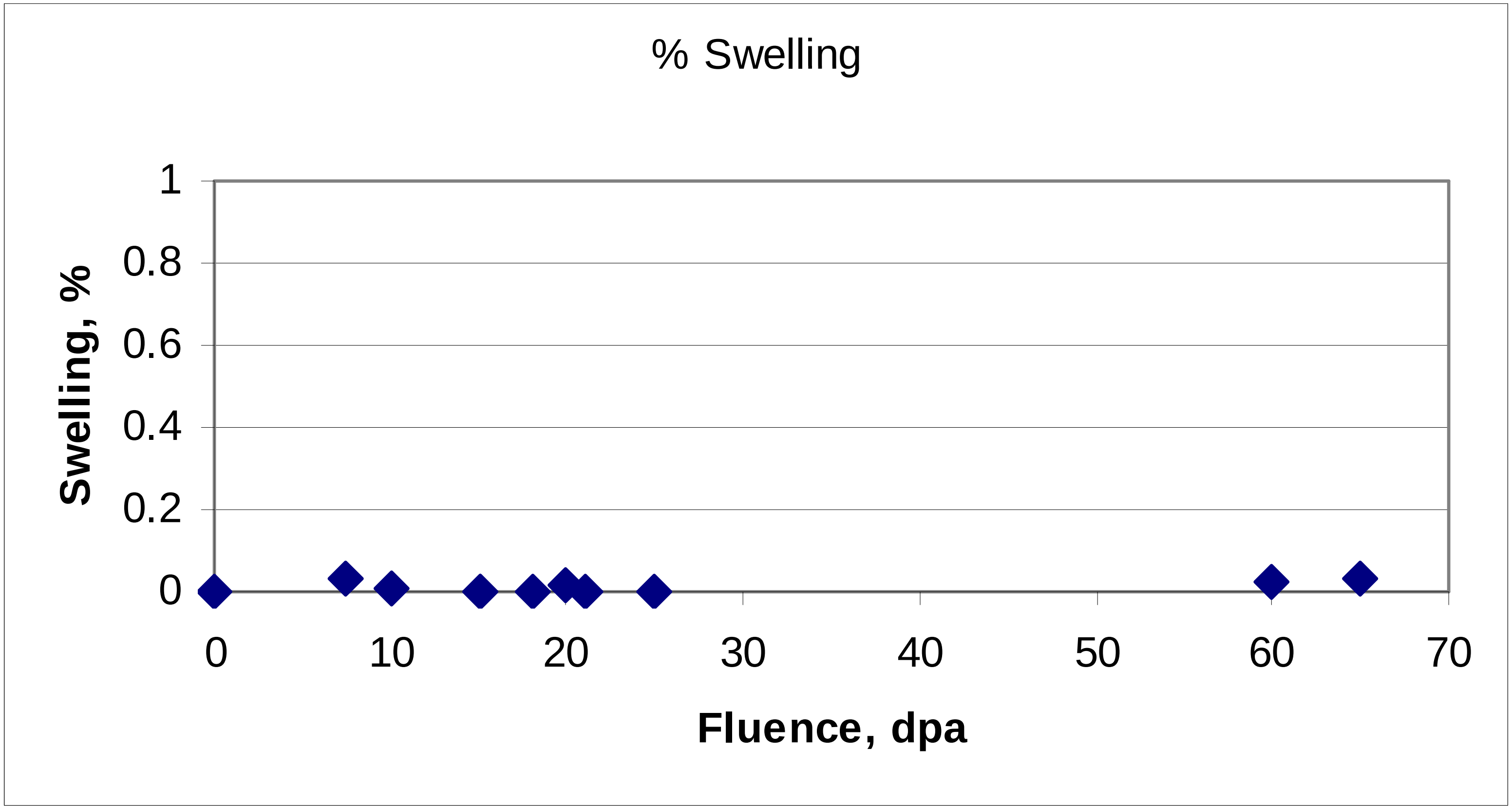
IFA 657: PWR conditions (temp. 335 °C, 2-3 ppm H_2)





Microstructure Investigation

Swelling (290 to 350C)

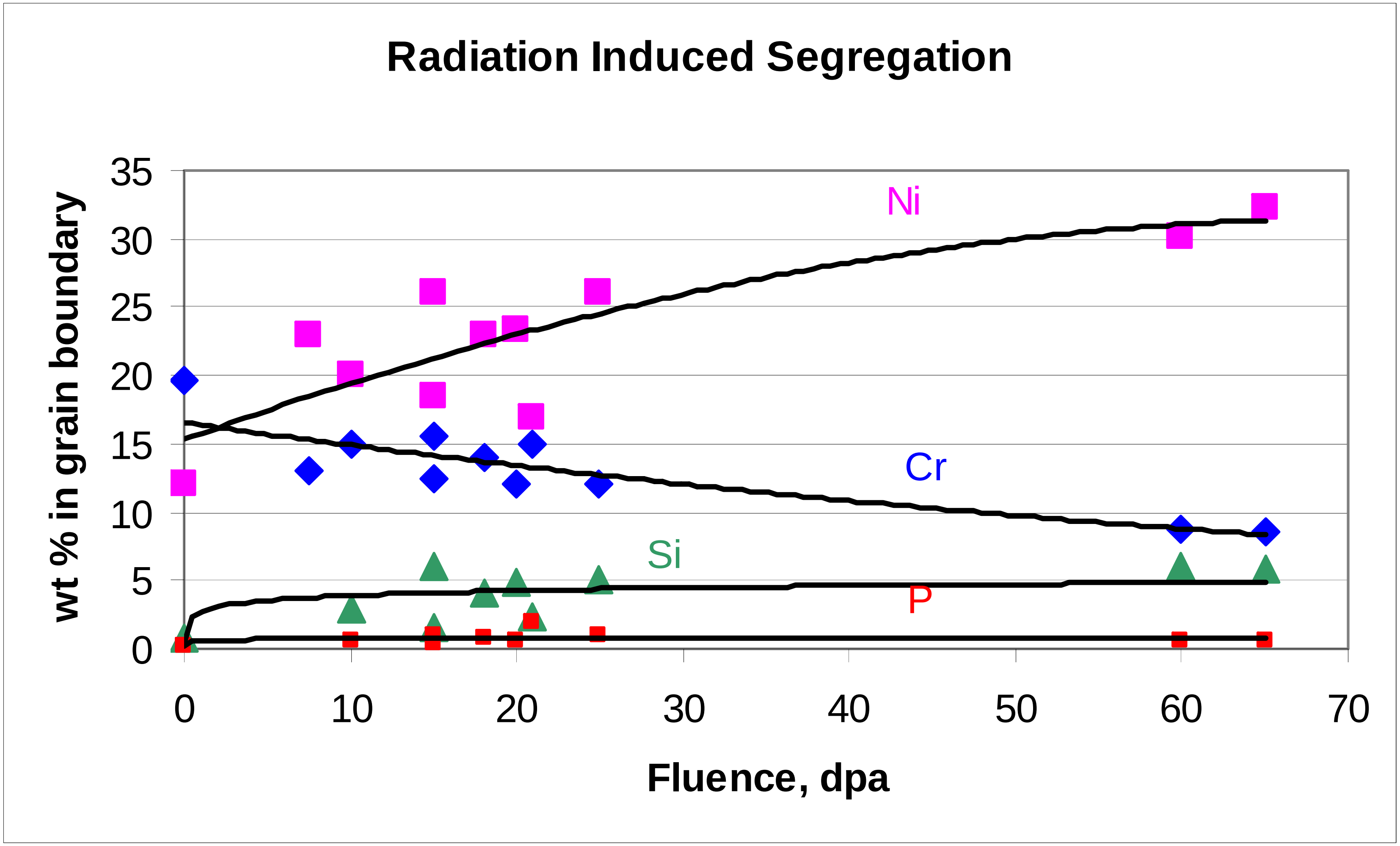


Data from CW316SS & 347SS B-F bolts & CW316SS tube

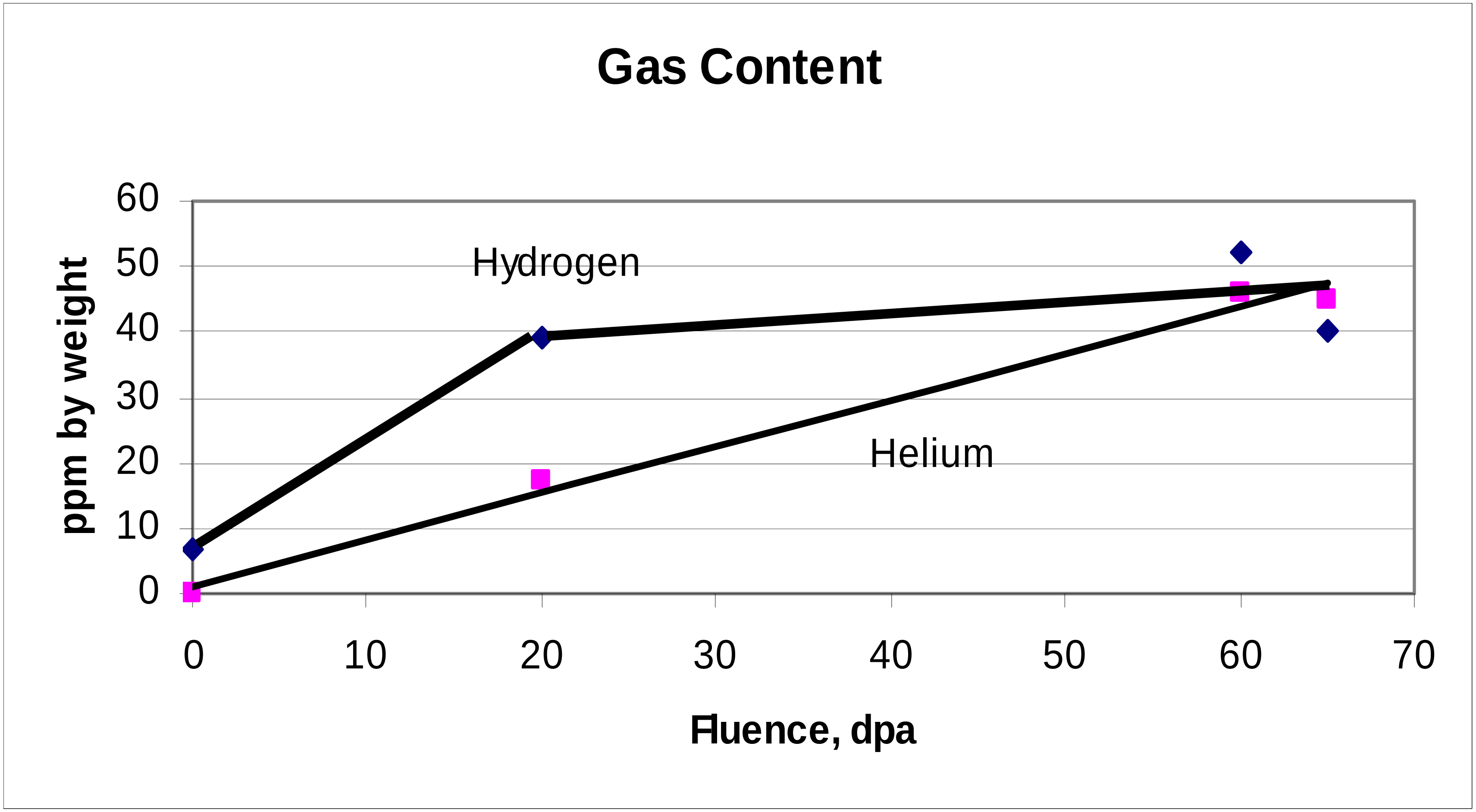
PWR Void Swelling Results to Date

Samples	Swelling (%)	Temperature Range (°C)	Exposure (dpa)
Point Beach Bolt SA 347	0.029	359 to 401	7 to 8
Farley Bolt CW 316	0.011	359 to 401	9 to 11
Tihange Bolt A CW 316L (EDF)	0.12	363 to 373	12.05
Tihange Bolt B CW 316L S1 (PNNL)	<0.010	320	19.5
Tihange Bolt B CW 316L S2 (PNNL)	0.2	342	11.9
Tihange Bolt B CW 316 L S3 (PNNL)	0.24	330	7
High Flux Thimble CW 316 S1	0.0294	320	65
High Flux Thimble CW 316 S2	0.0274	295	61
High Flux Thimble CW 316 S3	0.014	325	17

Radiation Induced Segregation



Gas Content



Gas Content
60
50
40
30
20
10
0
ppm by weight
Hydrogen
Helium
0
10
20
30
40
50
60
70
Fluence, dpa

Radiation and Temperature Analysis

Baffle Plate Results - Midplane

Macroscopic Sample Data from Decomissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence and Stainless Steel 304 dpa

Baffle Plate - Left

Core Side Surface			Middle of Plate			Back Side Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	2.9E+21	4.2	**1**	2.5E+21	3.7	**1**	2.2E+21	3.2
2	1.6E+22	23.2	**2**	1.5E+22	22.2	**2**	1.5E+22	21.4
3	2.9E+21	4.2	**3**	2.5E+21	3.7	**3**	2.2E+21	3.2
4	1.6E+22	23.2	**4**	1.5E+22	22.2	**4**	1.5E+22	21.4

Looking at Sample from Core Side of Sample

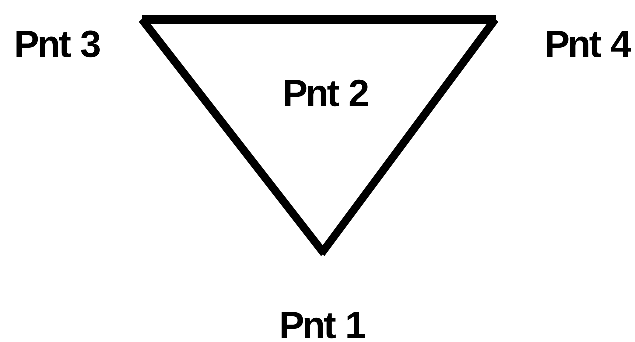
Former Plate Results - Midplane

Macroscopic Sample Data from Decomissioned PWR Core Midplane XYZ TORT
Fast (E > 1.0 MeV) Neutron Fluence and Stainless Steel 304 dpa

Former Plate 3 - Inner Corner Sample

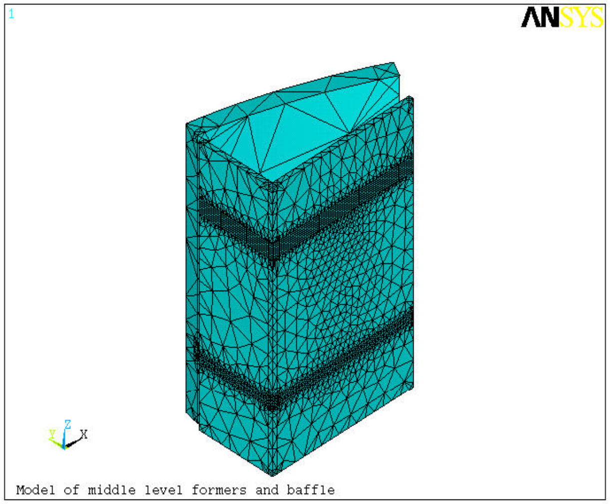
Bottom Surface			Middle of Plate			Top Surface		
Point	**Phi**	**dpa**	**Point**	**Phi**	**dpa**	**Point**	**Phi**	**dpa**
1	1.2E+22	18.1	**1**	1.2E+22	18.4	**1**	1.2E+22	18.2
2	3.5E+21	5.3	**2**	3.6E+21	5.5	**2**	3.5E+21	5.3
3	4.2E+21	6.3	**3**	4.4E+21	6.4	**3**	4.2E+21	6.3
4	7.5E+21	11.1	**4**	7.7E+21	11.4	**4**	7.5E+21	11.1

Looking Down at Sample from Above

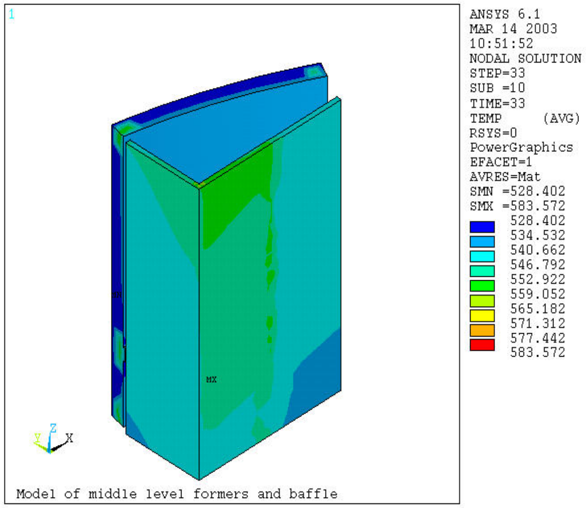


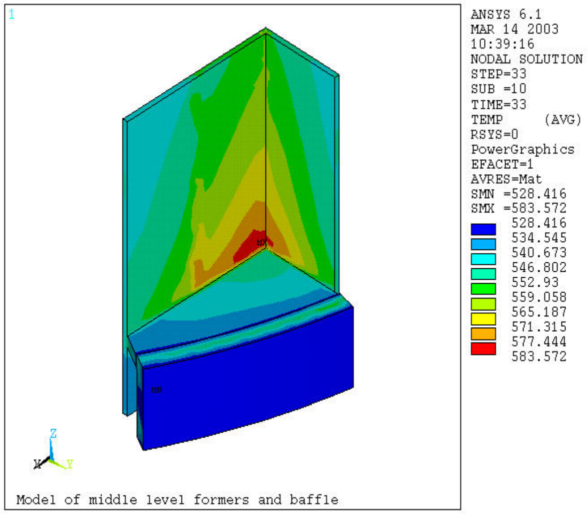
Temperature Calculations

Finite element Mesh



Temperature Contours at EOC, Cycle 11

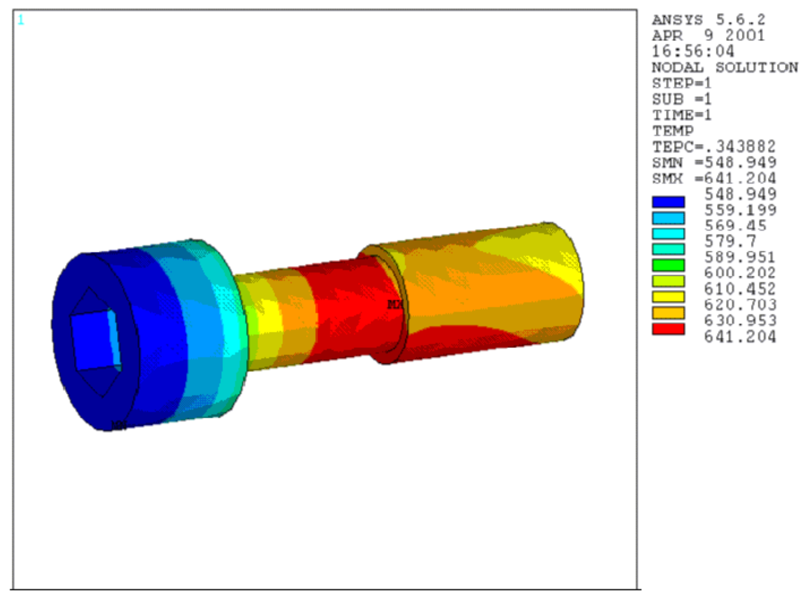




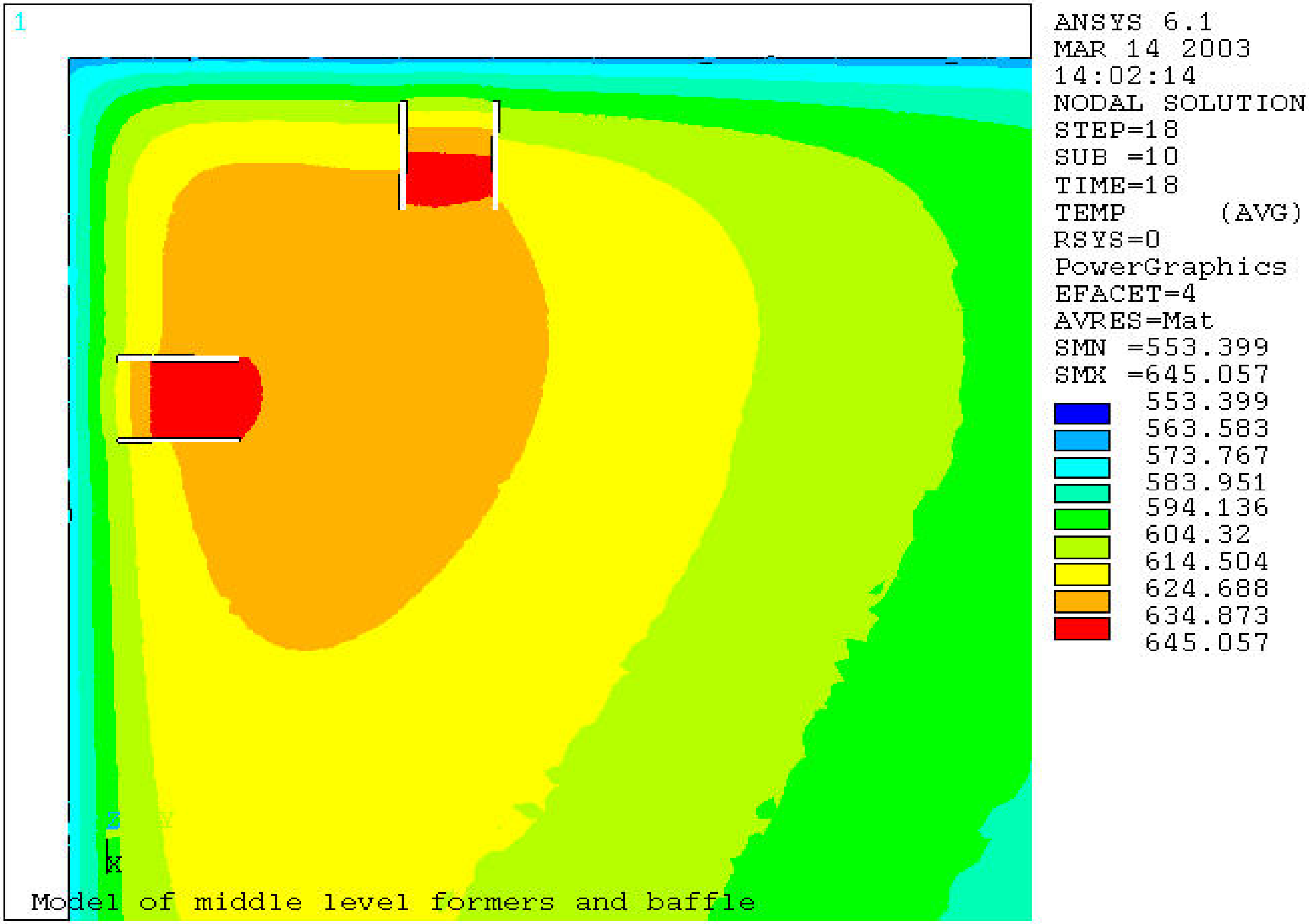
Bolt Centerline Temperature

Removed Bolt ID	Former Level	Centerline Bolt Max. Temperature (Deg.F)	(Deg.C)	HGR State Point	Fast (E > 1.0 MeV) Neutron Fluence	Stainless Steel dpa	Stress Ranking (1=Highest)
3321	1	596	313	EOC-05	4.51E+21	7.0	6
4821	1	577	303	EOC-06	2.37E+21	3.6	1
1312	2	657	347	EOC-05	6.80E+21	10.5	2
4122	2	657	347	EOC-05	6.80E+21	10.5	2
2922	2	601	316	EOC-06	8.26E+21	12.4	8
2522	2	621	327	EOC-02	1.06E+22	16.4	5
3812	2	621	327	EOC-02	1.06E+22	16.4	5
4522	2	621	327	EOC-02	1.06E+22	16.4	5
3722	2	602	317	EOC-22	1.15E+22	17.7	9
4722	2	602	317	EOC-22	1.15E+22	17.7	9
3133	3	662	350	BOC-01	6.65E+21	10.3	7
4313	3	662	350	BOC-01	6.65E+21	10.3	7
3324	4	640	338	BOC-01	1.26E+22	19.6	10
4126	6	615	324	EOC-02	5.09E+21	7.9	4
4116	6	634	334	EOC-02	9.05E+21	14.1	11
4326	6	634	334	EOC-02	9.05E+21	14.1	11

Broken bolts are in bold



Temperature Map of the Decommissioned PWR Plant Baffle/Former Assembly

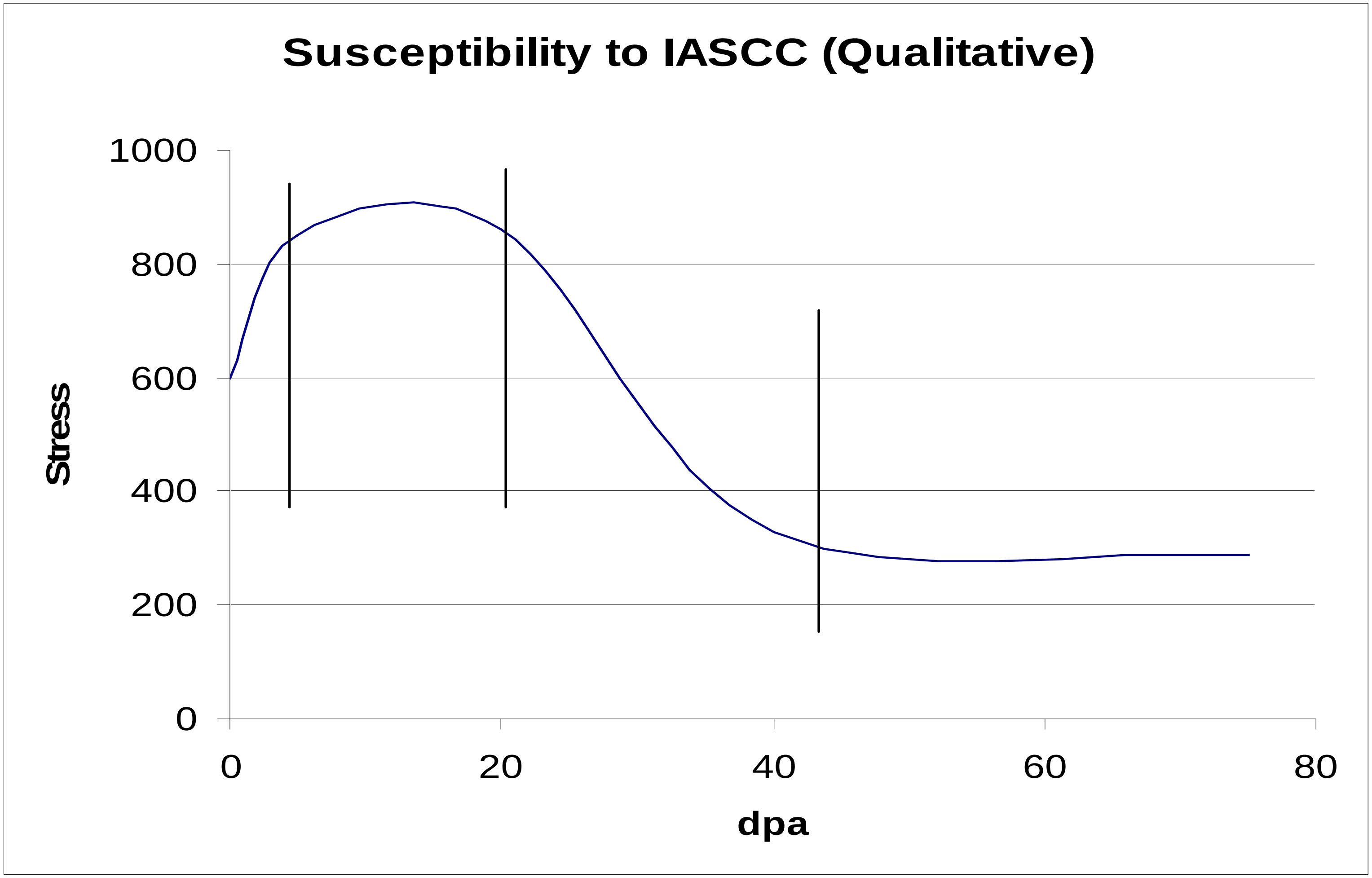


Overall Summary

IASCC Observations

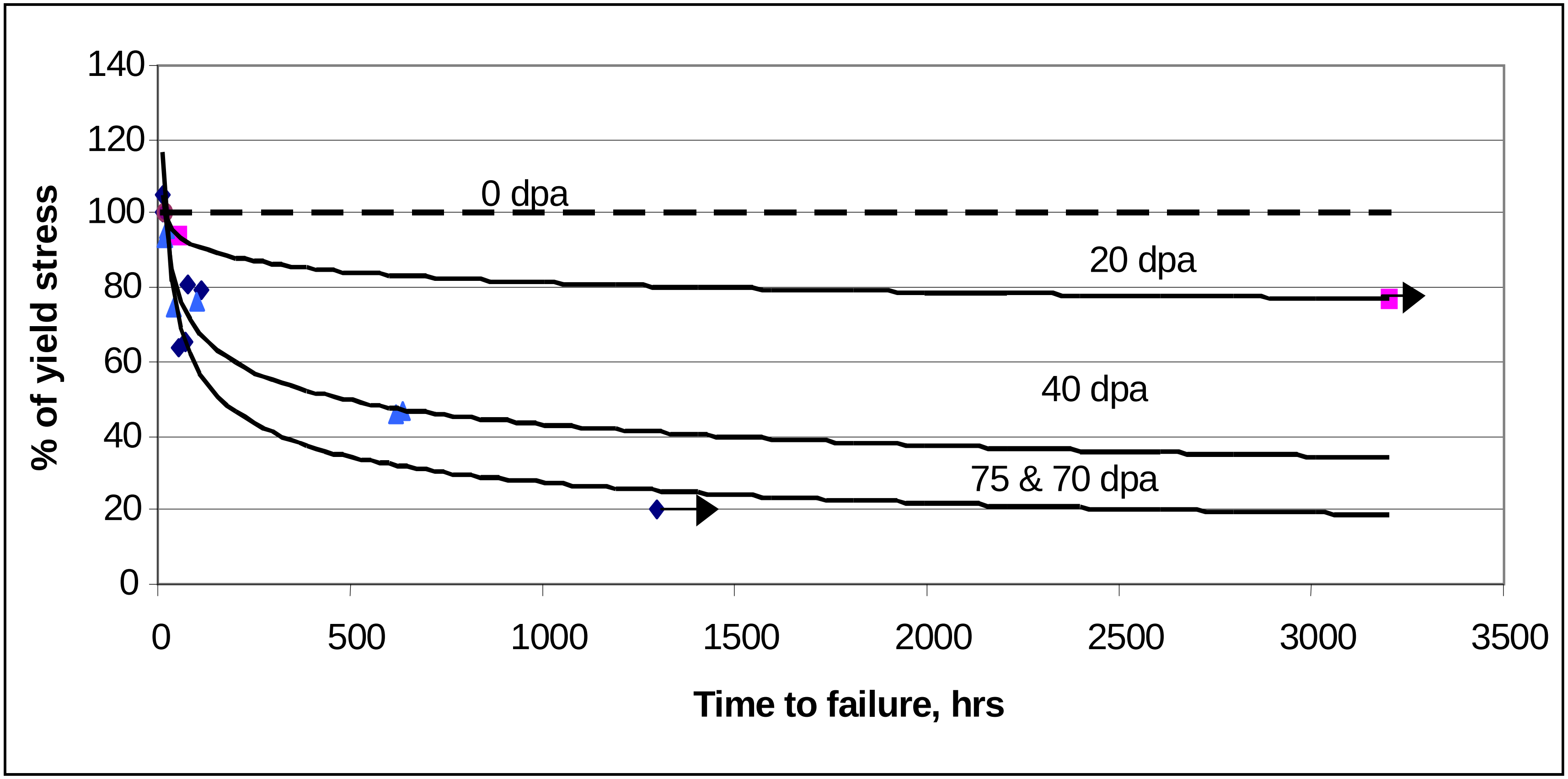
- No evidence for significant electrochemical polarization due to radiolysis or of chemical concentration in super-heated crevices in PWRs
- IASCC can occur in highly irradiated (generally >5dpa) 300 series stainless steels in hydrogenated PWR water (consistent with results in hydrogen water chemistry for BWRs)
- No evidence that grain boundary segregations, helium or hydrogen embrittlement play a significant role in IASCC
- High strength from irradiation hardening does seem to be an important factor and may explain heat to heat variability
- Quantitative correlation of PWR and Fast Reactor Data will be performed

SSRT Test Results – Stress/dpa Effect



O-ring Tests - Crack Initiation

Need quantification



Fracture Toughness

- Reductions in the toughness of the austenitic stainless steel internals components are expected during PWR operation
- The materials are expected to retain sufficient ductility and toughness
- PWR irradiated data currently available is ~ 15 dpa using non-standard bolt specimens
- Faster reactor irradiated data available is ~ 10 to 20 dpa
- More data are being and will be generated, decommissioned PWR materials, Bor irradiated materials
- Bor materials will also include thermally aged CASS materials

Crack Growth

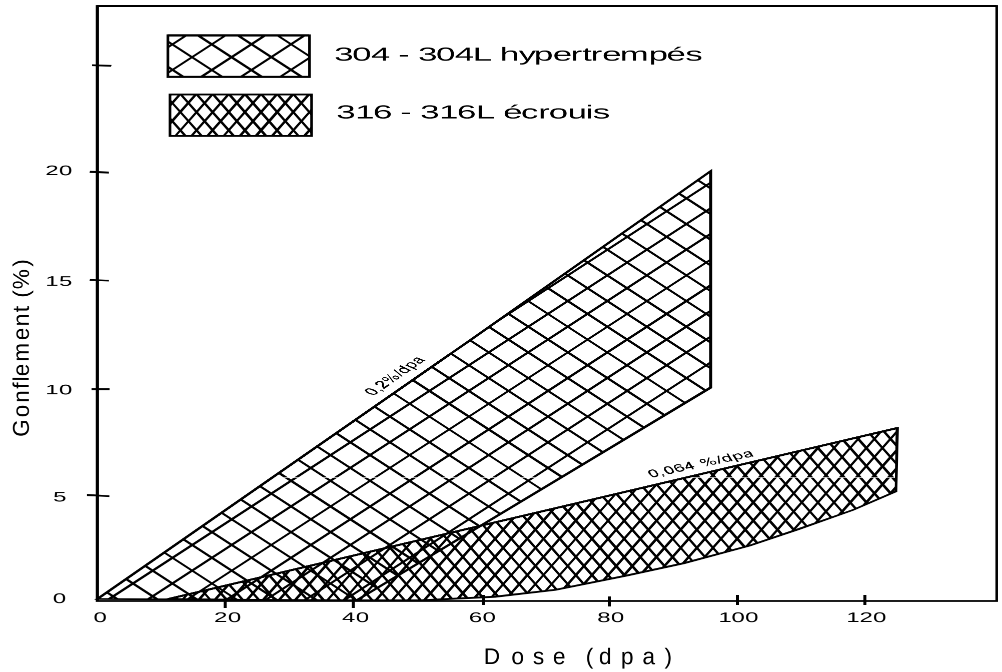
- Non-standard bolt specimens of ~ 15 dpa gave some qualitative data of crack growth, however not satisfying validity rule
- CT specimen data are gradually coming out for fluence >10 dpa – Halden in-pile testing, decommissioned PWR materials, CIR program
- Future data will include both fast and PWR irradiated materials

Void Swelling

- Limited PWR data show very small swelling (low dose CW316 and SA347)
- Fast reactor data, many from non-PWR type of materials, fitted equations such as the Foster-Flynn equation can not and should not be applied to PWRs
- 304 appears to swell more that 316
- Swelling is coupled with state of creep and stress relaxation
- Need PWR high dose and high temperature swelling data to develop swelling prediction applicable to PWRs

SA 304 and CW 316 Swelling

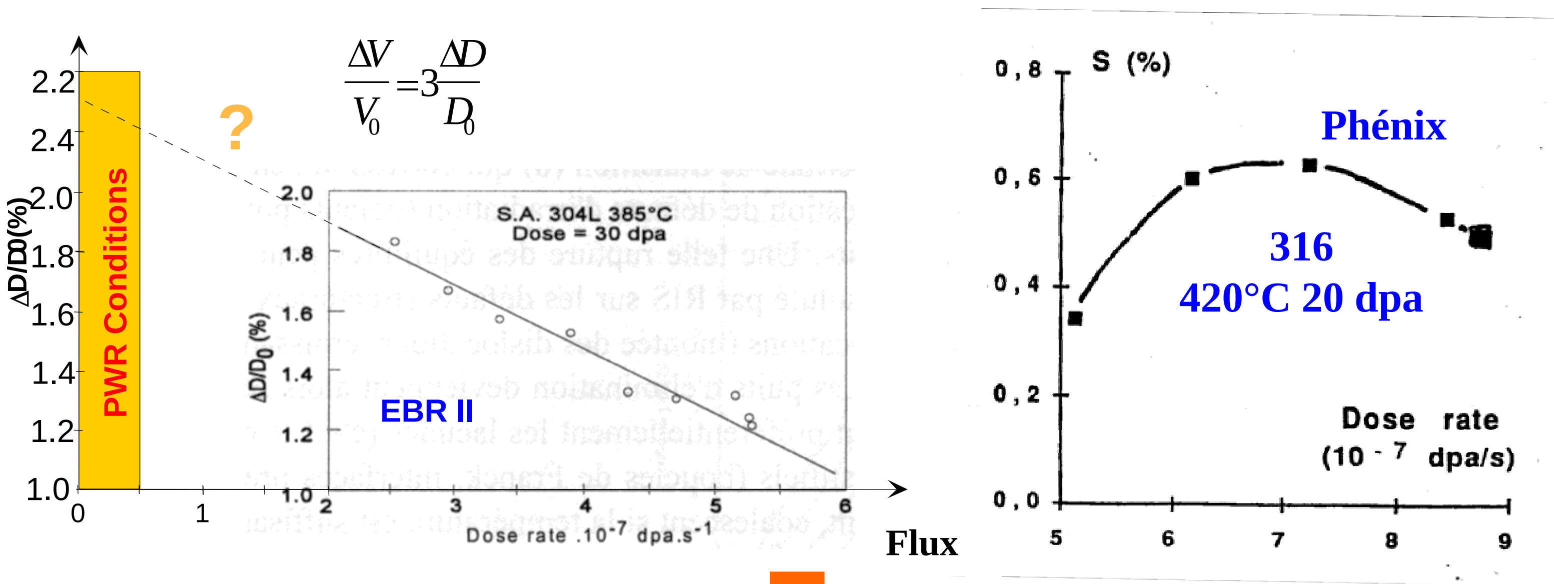
Irradiation 375°C – 450°C

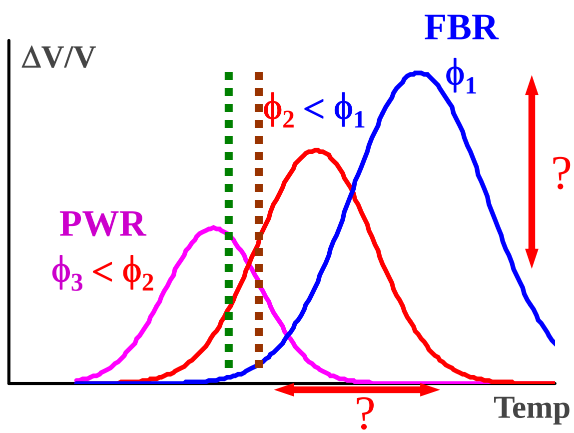


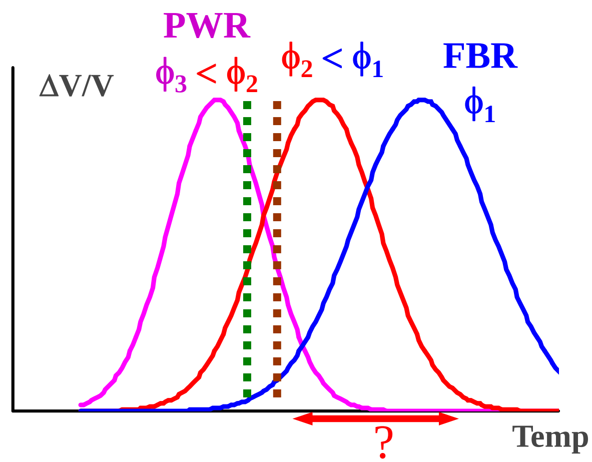
	Incubation dose	Swelling rate
SA 304-304L	10-15 dpa	0.2% dpa^{-1}
CW 316-316L	30-40 dpa	0.06% dpa^{-1}

Parameters fonction of Temperature

Flux Effect on Swelling







Summary

- Reactor internals material degradation mechanisms have been extensively studied
- Data obtained and to be obtained will support the development of degradation threshold (behavior model) as a function of material, fluence, temperature and stress
- The degradation threshold values (behavior models) will support
  - the development of screening criteria and flaw tolerance technical basis
  - the evaluation of component functionality
  - the disposition of inspection findings
  - the development of inspection guidelines for aging management