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STUDIECENTRUM VOOR KERNENERGIE
CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE

MATERIAL ISSUES FOR ADS: MYRRHA-PROJECT

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On behalf of

MYRRHA-TEAM and MYRRHA-Support

EUROMAT 2005 – Prague, Czech Republic

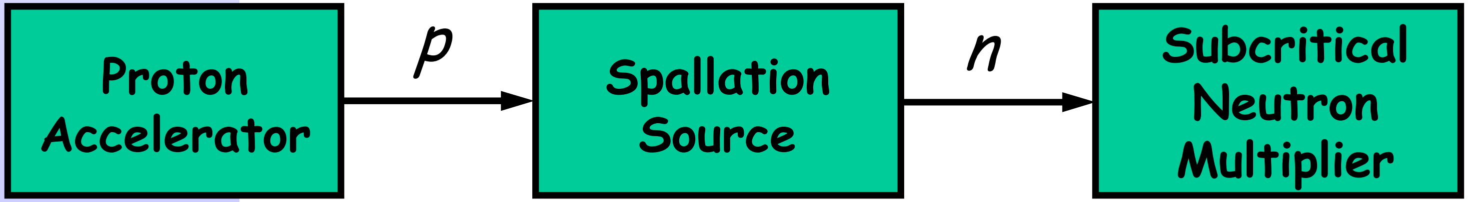
Sep 5-8, 2005

History: ADS Concept

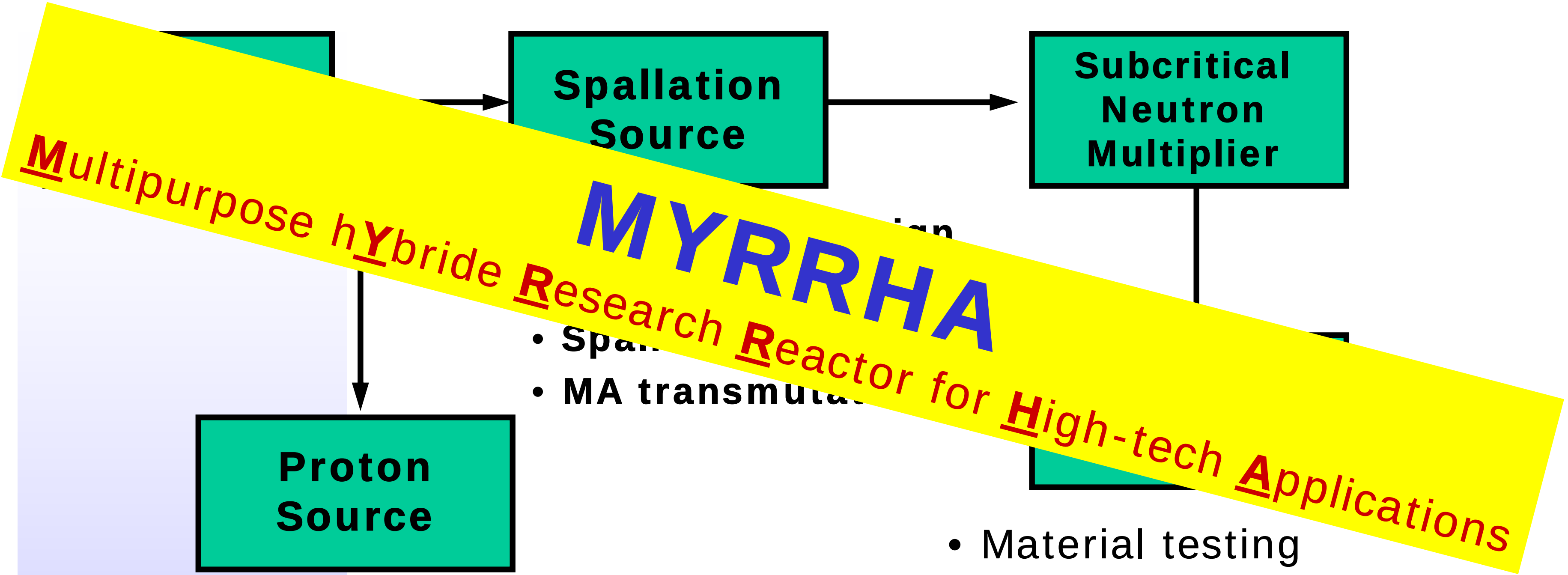
The idea of using accelerators to produce fissionable material was put forward by G.T. Seaborg in 1941. He produced the first human-made plutonium using an accelerator.

In the 90's, C. Bowman's group at LANL and C. Rubbia's group at CERN designed a transmutation facility using thermal neutrons (CB) and fast neutrons (CR), for burning both the actinides and long life fission products from spent LWR fuel.

The facility, called an ADS, combines high intensity proton accelerators with spallation targets and a subcritical core.



MYRRHA – concept: a multipurpose ADS



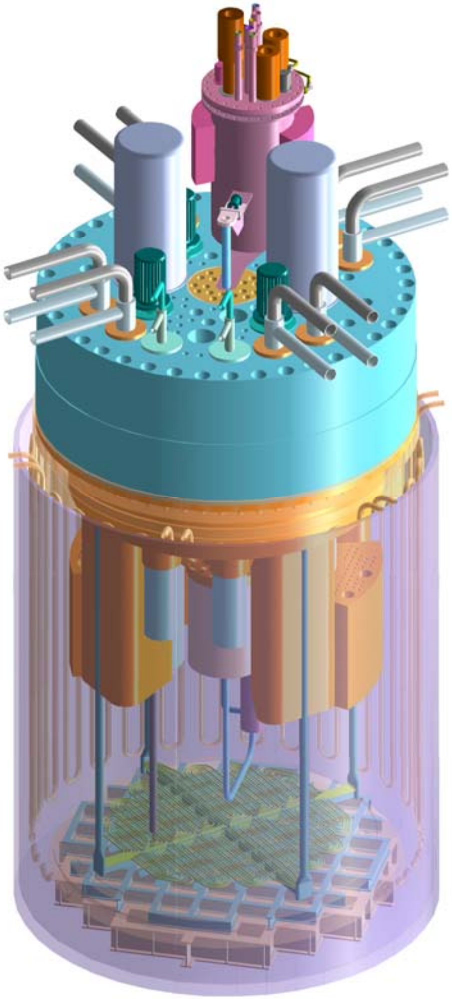
Multipurpose hYbride Research Reactor for High-tech Applications

MYRRHA

- Material testing
- Radioisotope production
- Proton therapy

- Material testing
- Fuel irradiation
- MA & LLFP transmutation
- Radioisotope production
- Neutron beams

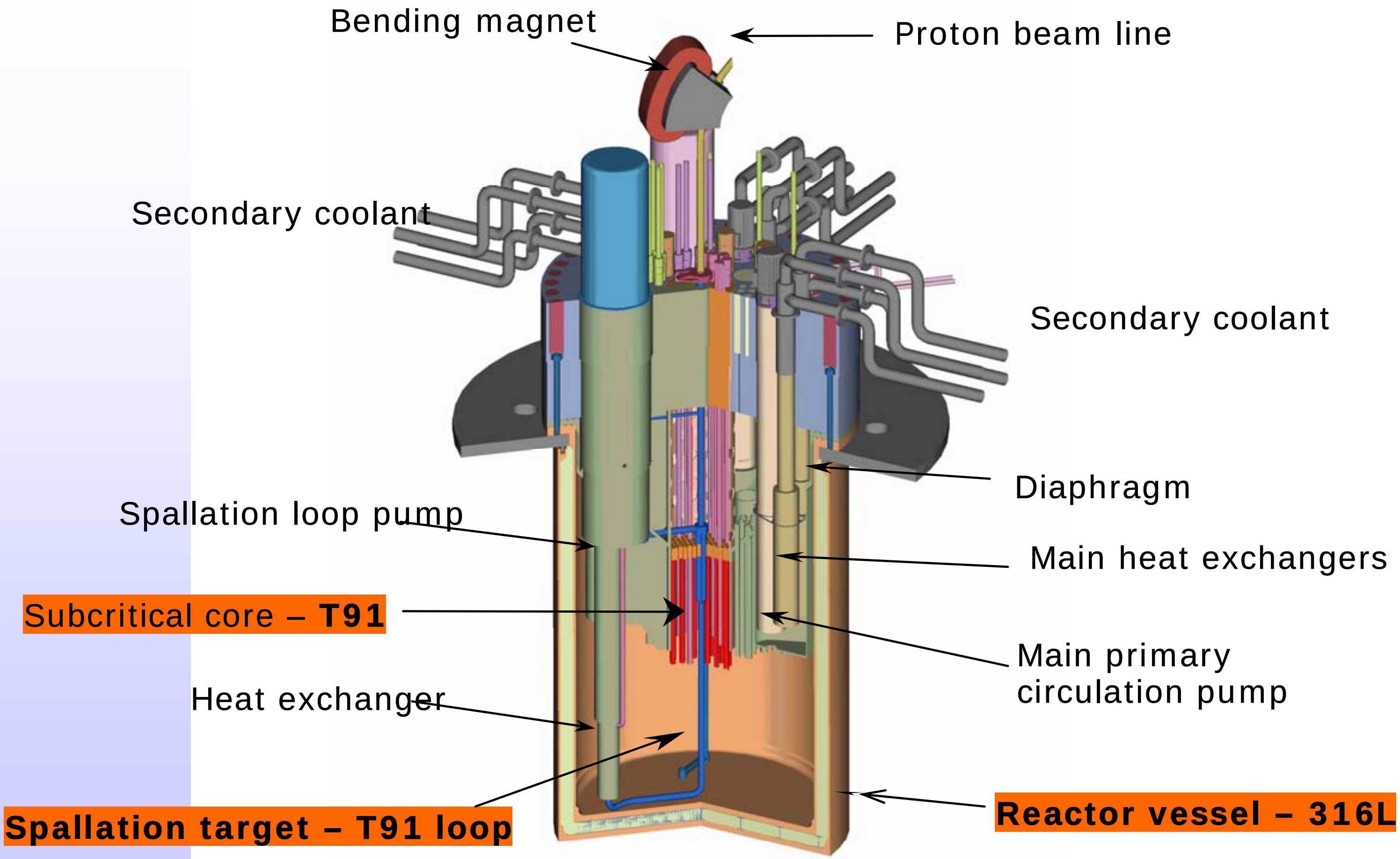
Purpose of Myrrha



MYRRHA is intended to be:

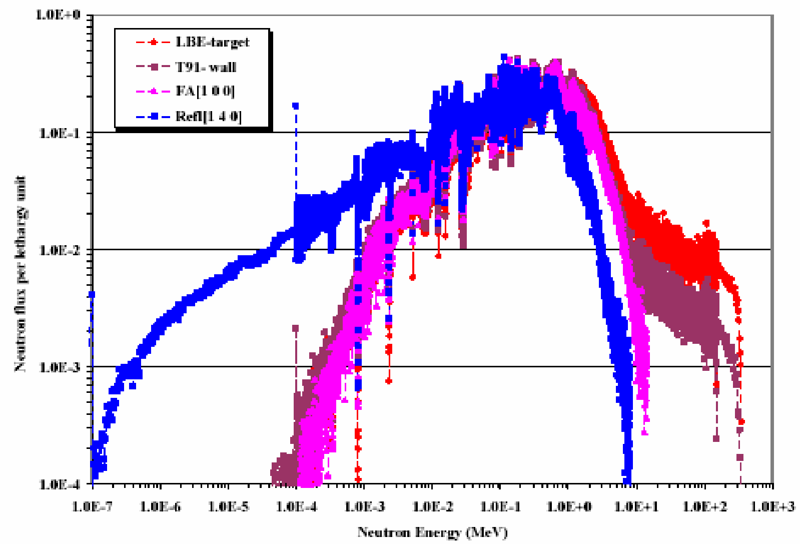
- A full step ADS demo facility
- A P&T testing facility
- A flexible irradiation testing facility in replacement of the SCK•CEN MTR BR2 (100 MW)
- An attractive fast spectrum testing facility in Europe
- An attractive tool for education and training of young scientists and engineers
- A medical radioisotope production facility

MYRRHA materials

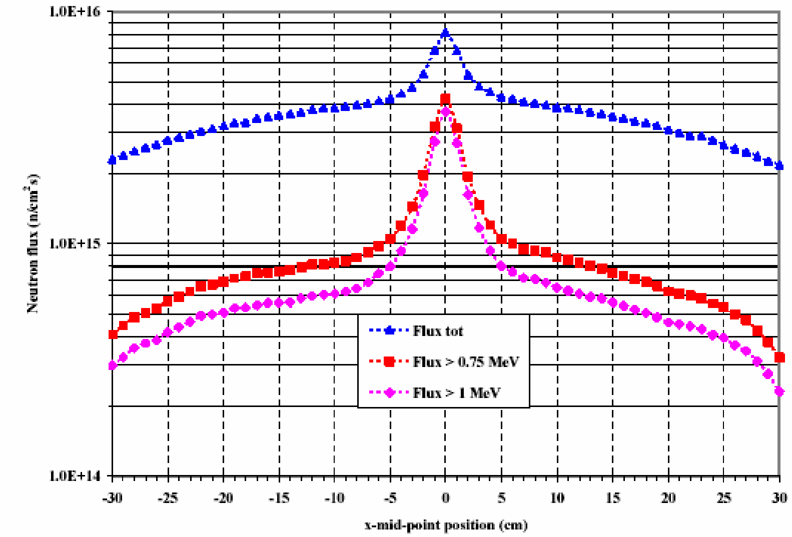


Neutronic Design constraints

Fast neutron spectra

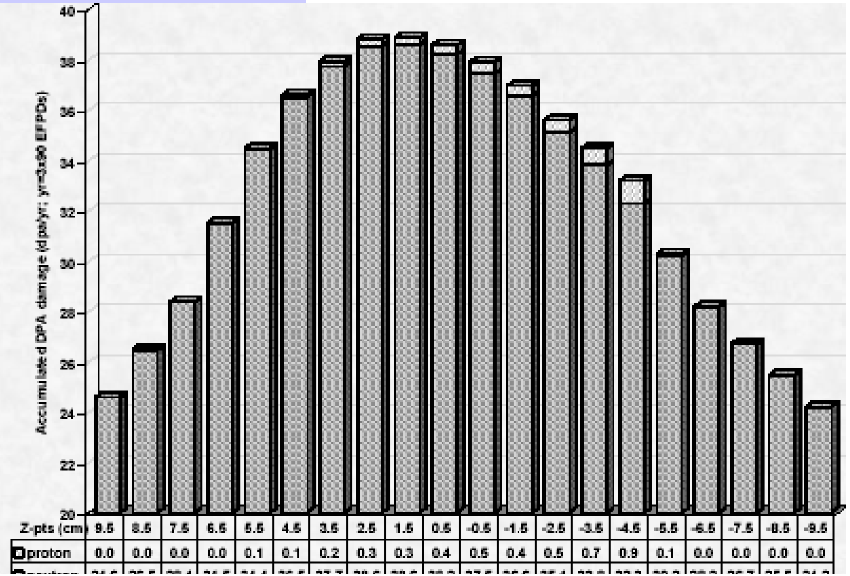


High energy flux

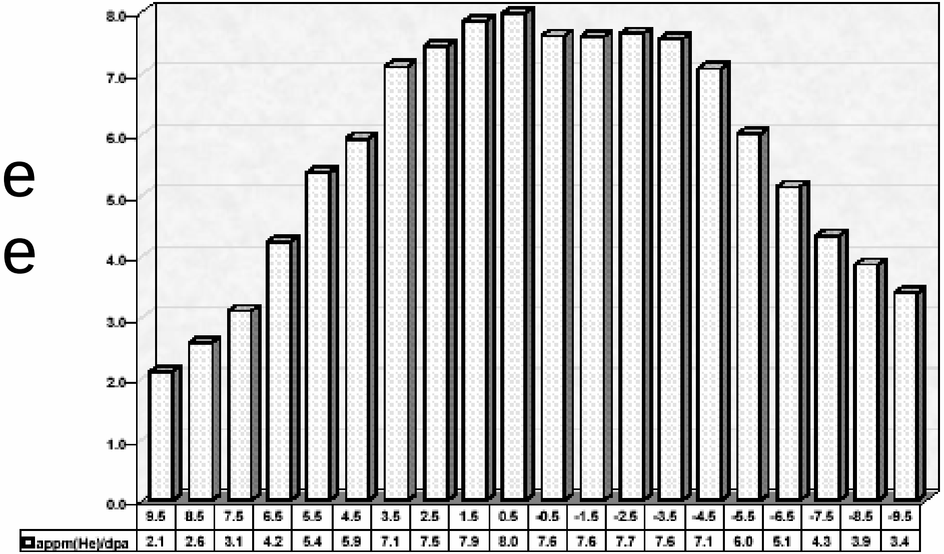


Design constraint on the spallation target

High Irradiation damage

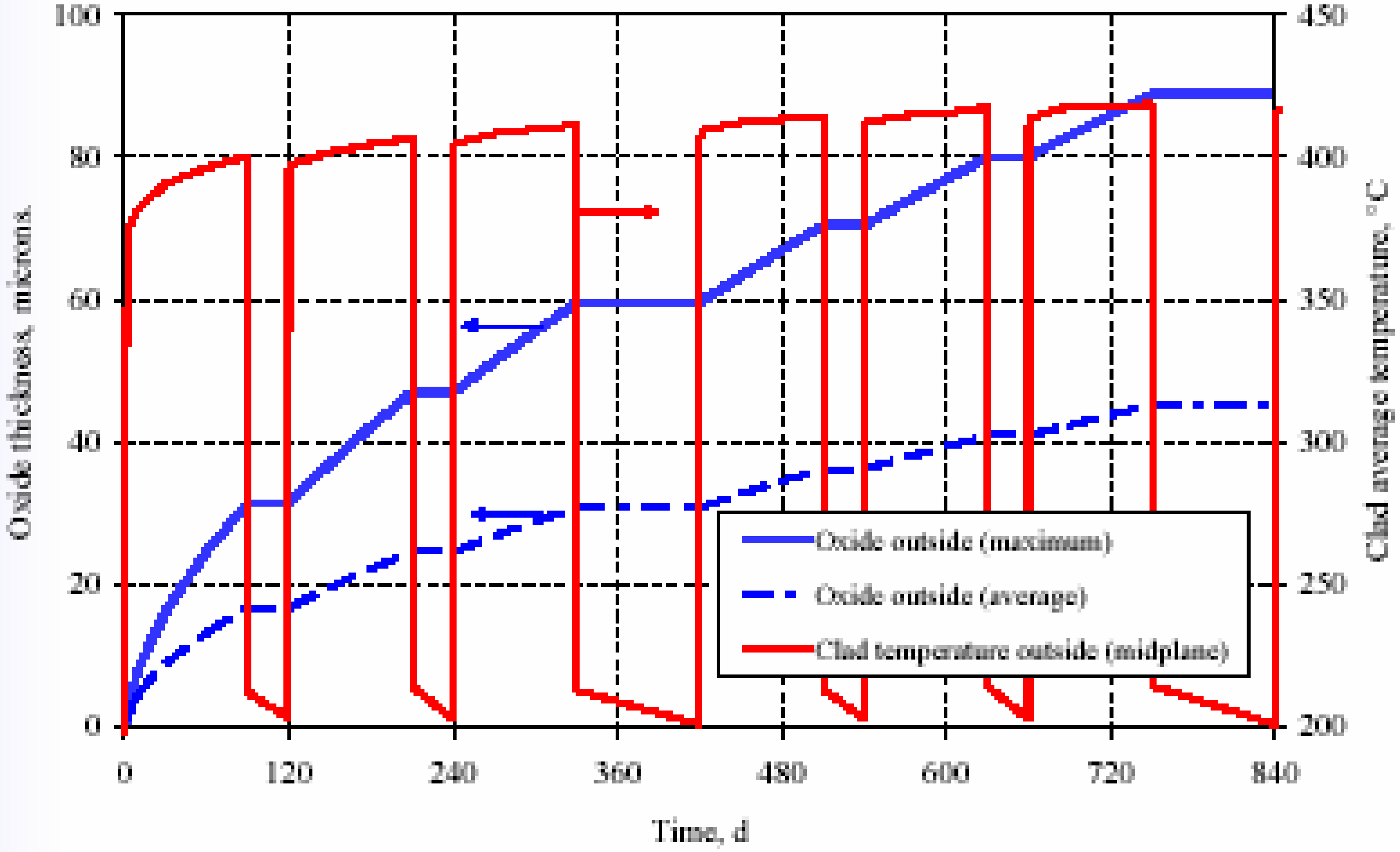


Considerable He production rate



Core design constraints

(in the hottest rod)



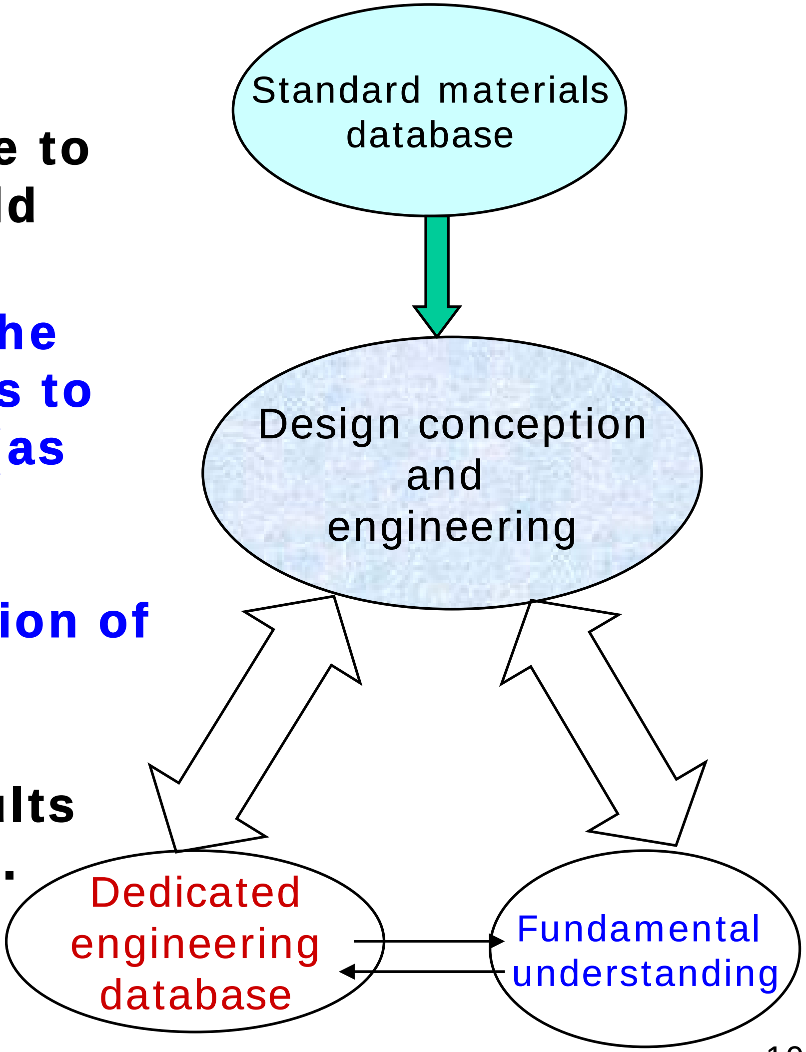
Design constraints

Operating conditions	Properties needed
•Temperature (200 to 450°C)	•High thermal conductivity, heat resistance, low thermal expansion
•High dose rate /high dose (5.10^{15}n/cm^{2}.s and up to 40dpa/year)	•Low DBTT shift, sufficient strength, limited loss of ductility and fracture toughness, low swelling rate
•High He/dpa production rate (3 to 8 appm)	•Adequate resistance to He and H embrittlement
•Beam trips and loading/reloading operations	•Resistance to fatigue in LBE
•High stress level (100 MPa), operational trips	•High creep and fatigue resistance
•Aggressive conditions (LBE)	•Corrosion and liquid metal embrittlement resistance

Materials for MYRRHA

The R&D program concerning the assessment of the materials suitable to sustain the design constraints should follow two routes:

- **Experiments and tests to support the engineering design: the material has to be tested under condition relevant (as close as possible) to the foreseen operational conditions to ensure an economically viable and safe operation of MYRRHA.**
- **Research to build the ability to interpolate and extrapolate the results from laboratory tests to the real life.**


Standard materials database

Design conception and engineering

Dedicated engineering database

Fundamental understanding

Material selection

- Al-alloys would not sustain high temperatures and also the irradiation conditions would induce severe embrittlement
- Ni-based alloys have a high affinity to dissolve in Pb-Bi besides their microstructural instability under irradiation
- Zr-alloys have shown drastic loss of strength and ductility under high temperature irradiation especially in the presence of H
- Austenitic steels are very susceptible to irradiation induced swelling and creep and poor resistance to corrosion in liquid Pb-alloys at high temperature.
- Ferritic-Martensitic steels (wrapper tubes in Phenix, candidate materials for Fast and Fusion reactors) for both fuel cladding and structures; they suffer from irradiation-induced embrittlement at low temperature

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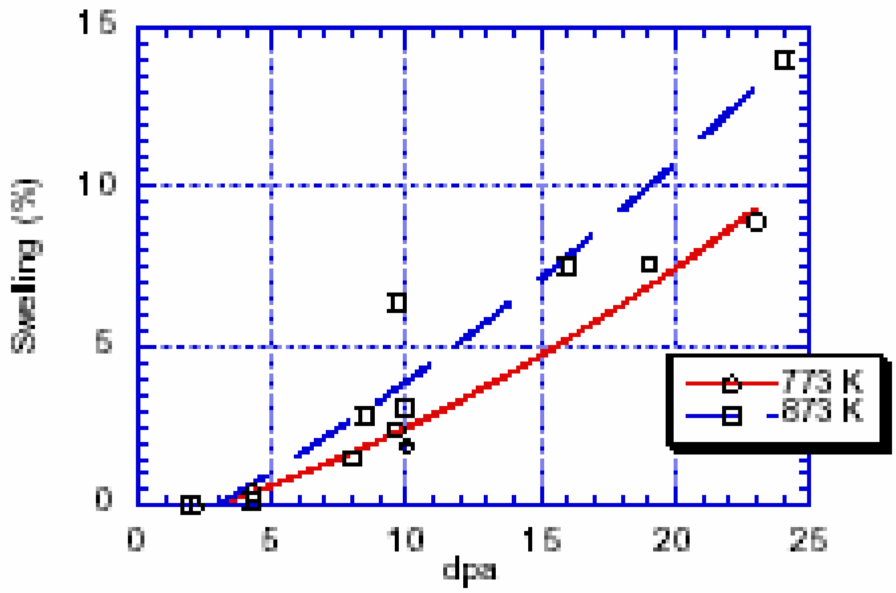
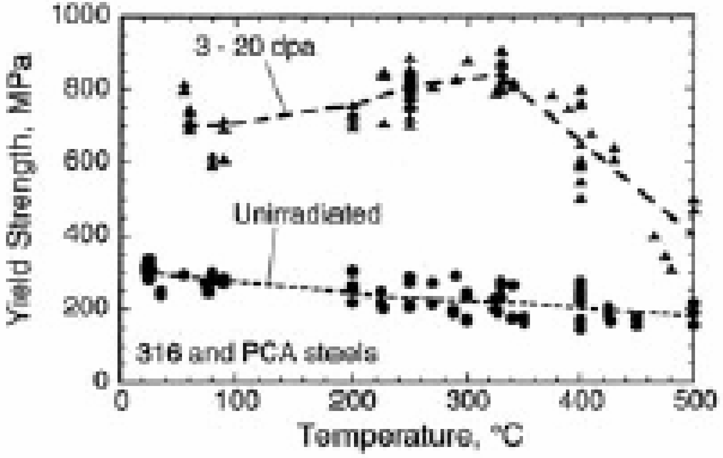
Stainless Steel (316L)

MYRRHA

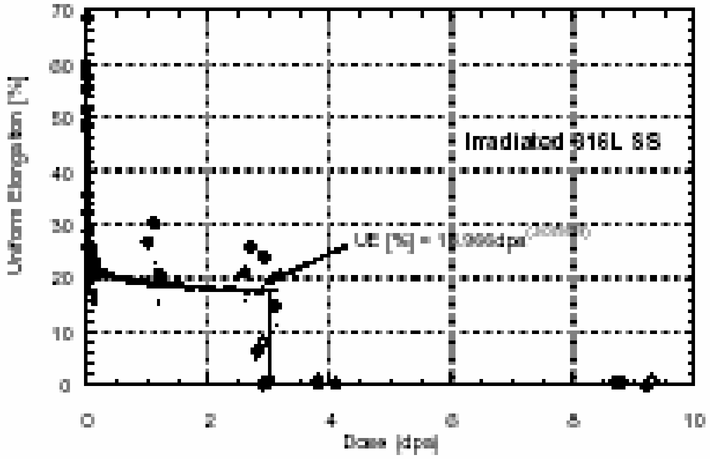
Swelling



Yield strength

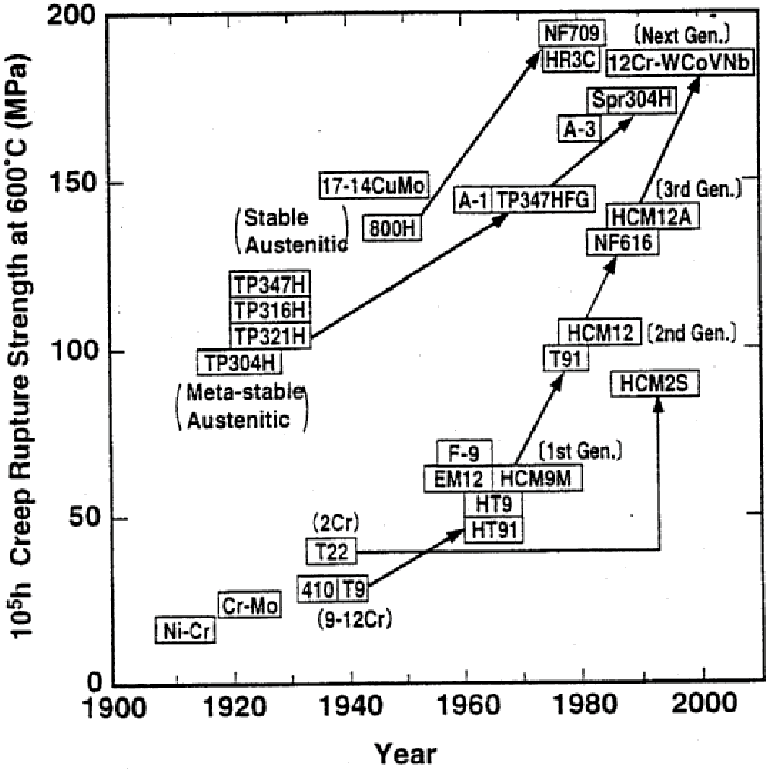


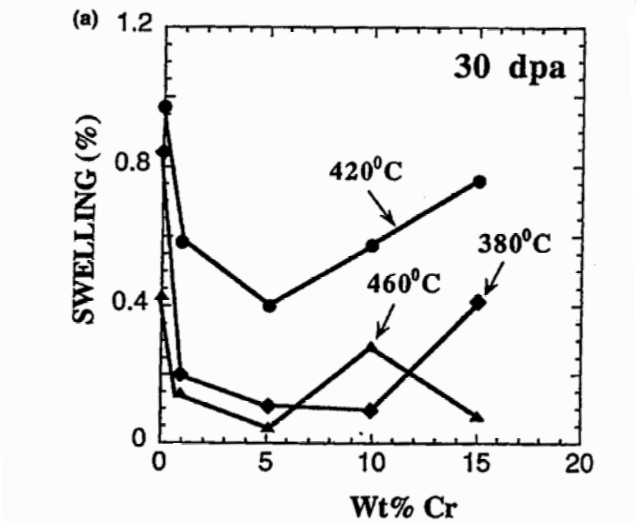
Uniform elongation

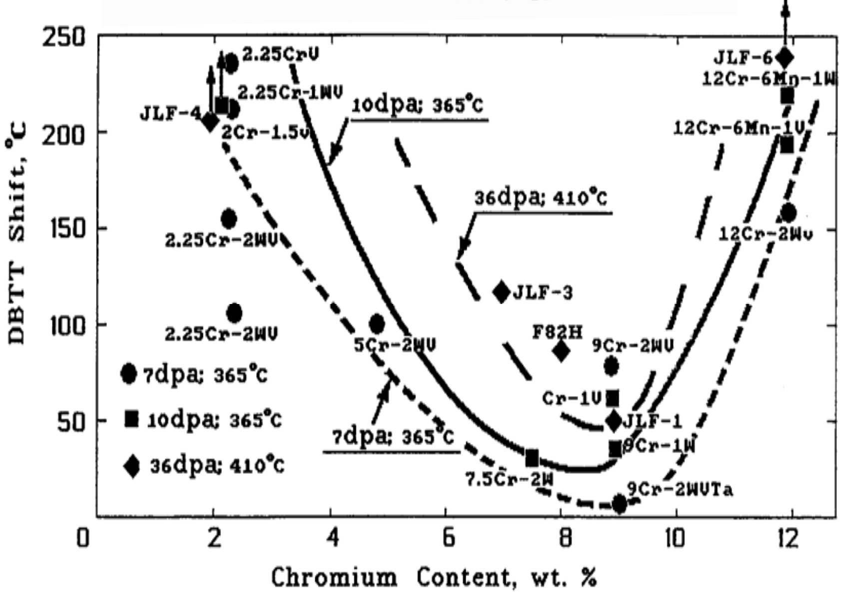


Material suited for relatively low doses and low temperatures (<300°C)

F/M Steels (why 9%Cr?)

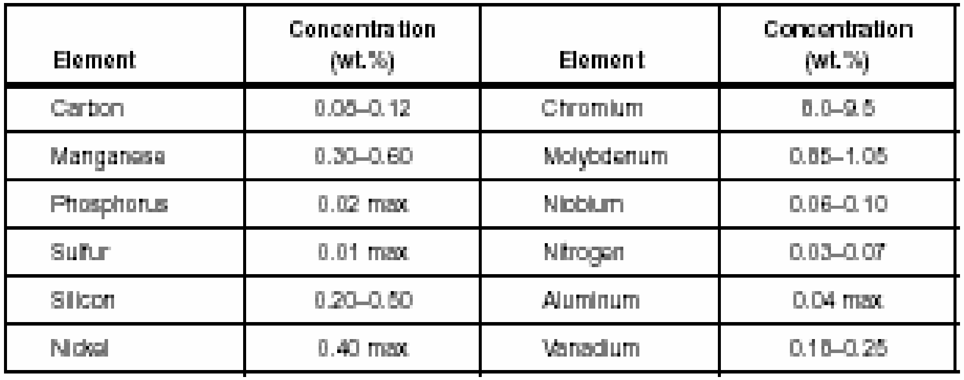




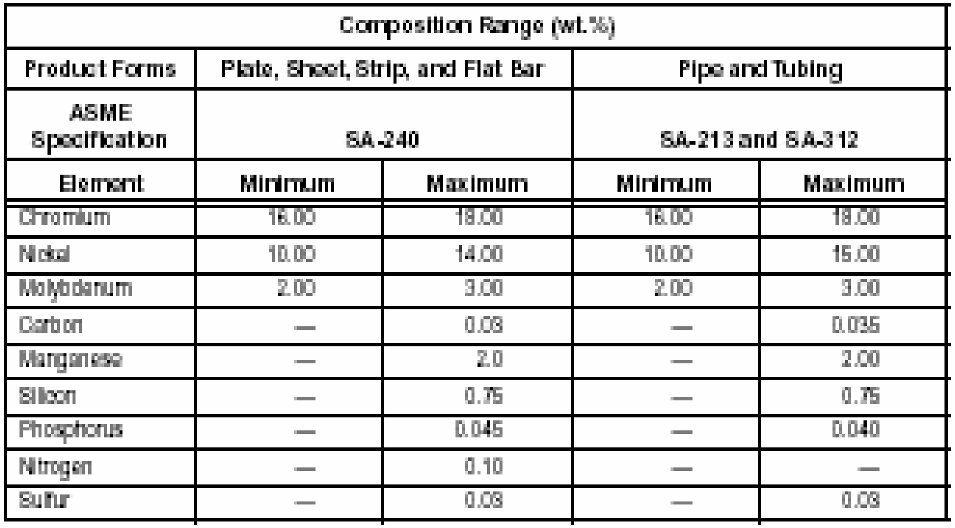


T91/T92 same family but T92 has less Mo, more W and B

Design properties of selected materials



Element	Concentration (wt.%)	Element	Concentration (wt.%)
Carbon	0.08–0.12	Chromium	8.0–9.5
Manganese	0.30–0.60	Molybdenum	0.85–1.05
Phosphorus	0.02 max	Niobium	0.06–0.10
Sulfur	0.01 max	Nitrogen	0.03–0.07
Silicon	0.20–0.50	Aluminum	0.04 max
Nickel	0.40 max	Vanadium	0.18–0.25



Composition Range (wt.%)				
Product Forms	Plate, Sheet, Strip, and Flat Bar		Pipe and Tubing	
ASME Specification	SA-240		SA-213 and SA-312	
Element	Minimum	Maximum	Minimum	Maximum
Chromium	16.00	18.00	16.00	18.00
Nickel	10.00	14.00	10.00	15.00
Molybdenum	2.00	3.00	2.00	3.00
Carbon	—	0.08	—	0.035
Manganese	—	2.0	—	2.00
Silicon	—	0.75	—	0.75
Phosphorus	—	0.045	—	0.040
Nitrogen	—	0.10	—	—
Sulfur	—	0.03	—	0.03

Parameter		**T91** *martensitic*	**AISI 316** *annealed*
Density	*kg/m³*	7756	7950
Young's Modulus	*Gpa*	215	200
Poisson's Ratio		0.3	0.30
Ultimate Strength (tensile)	*MPa*	670	520
Yield Strength	*MPa*	500	210
Elongation	*%*	22	30
Thermal expansion	*1/K*	$1.01 \cdot 10^{-5}$	$1.65 \cdot 10^{-5}$
Heat capacity	*J/kg-K*	450	500
Thermal conductivity	*W/m-K*	24.3	14
Neutron *(th)* **absorption section**	*barn*	2.61	2.76

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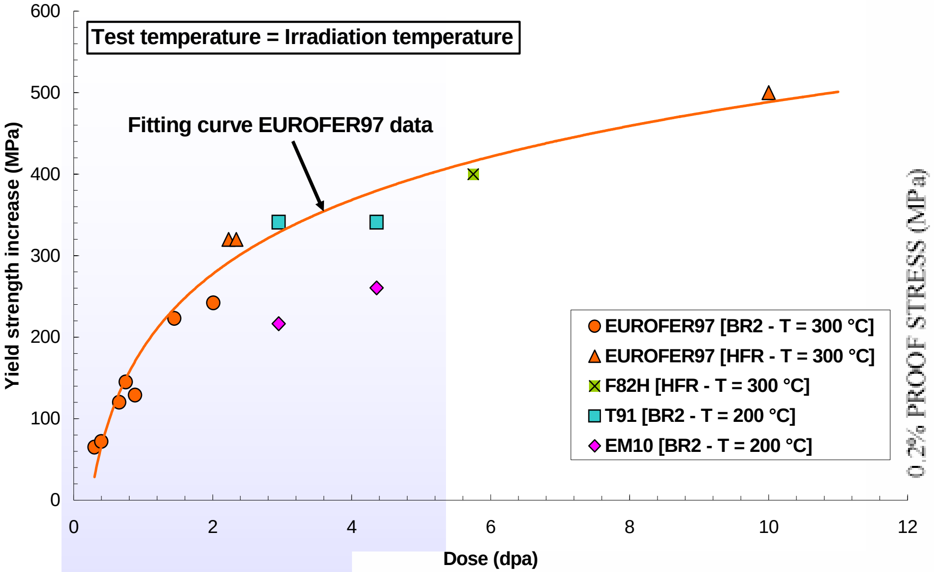
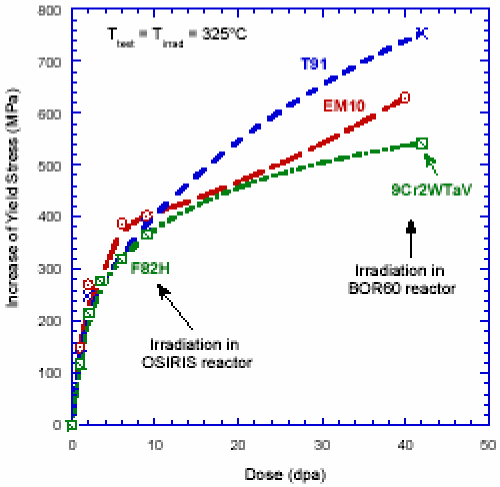
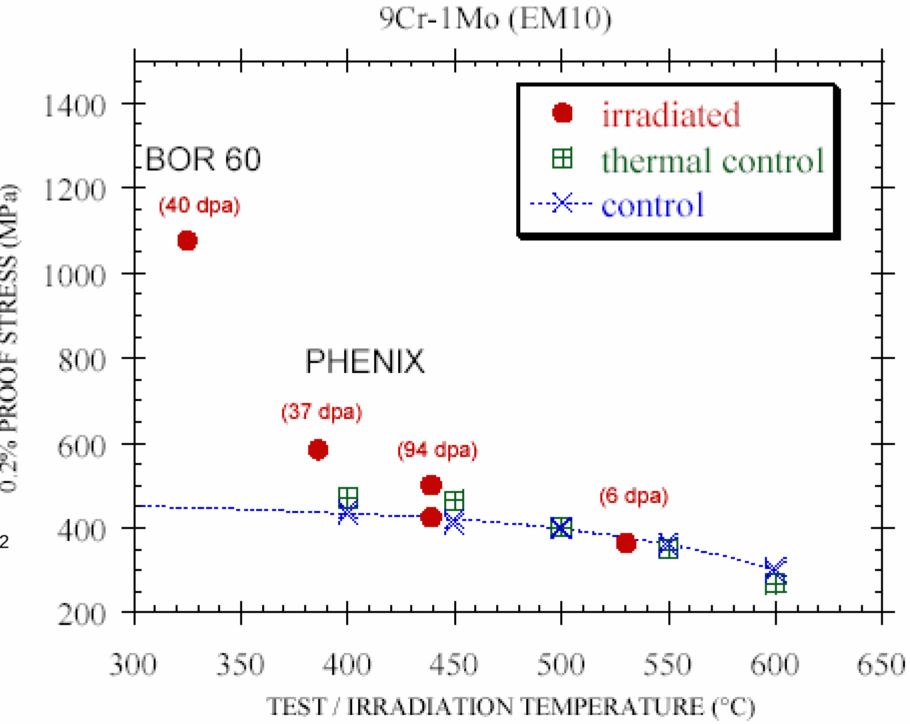
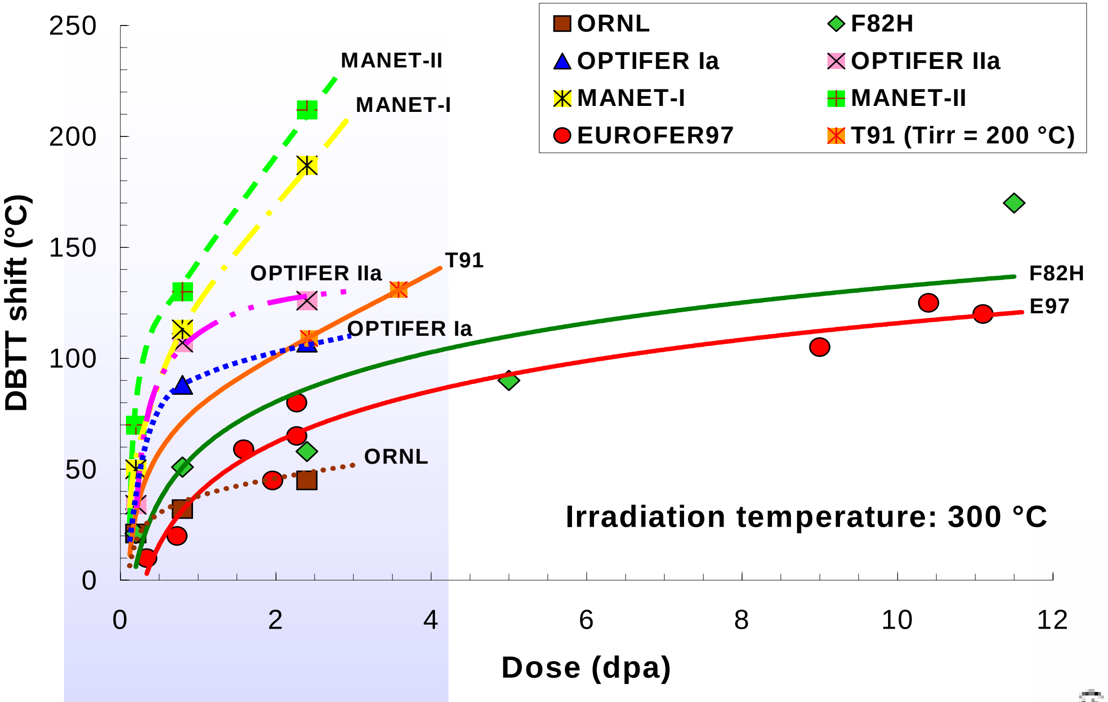
Irradiation effects on the yield stress (hardening)





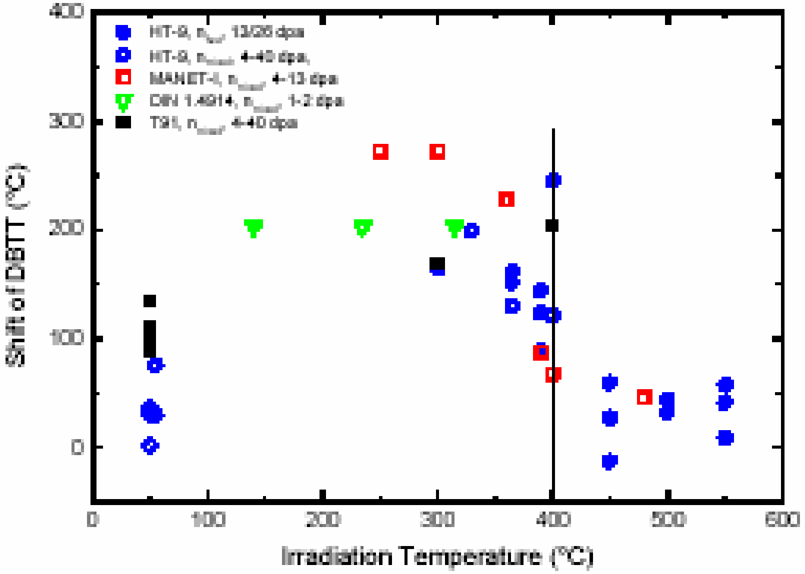


Irradiation effects on DBTT (Charpy tests)

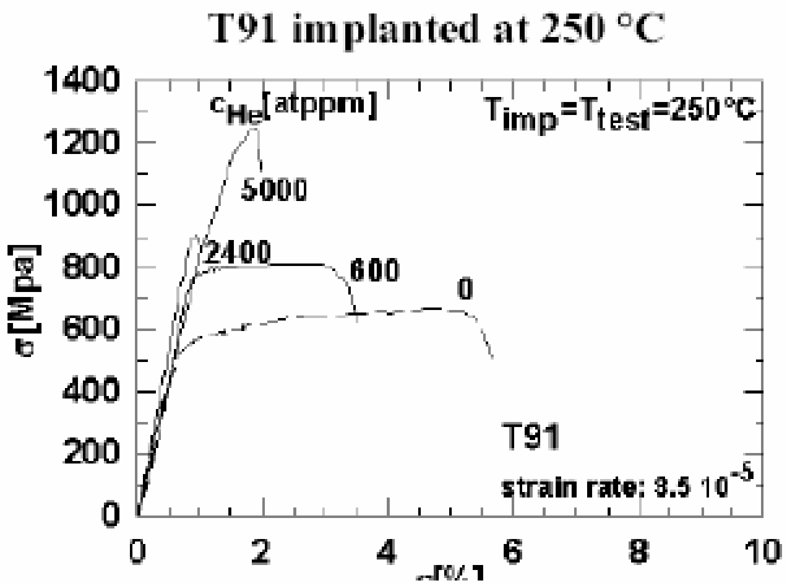


Dose effect

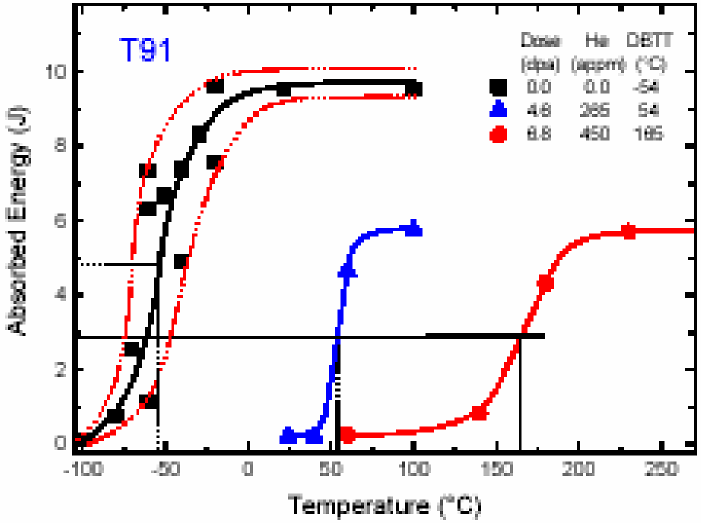
Irradiation temperature effect

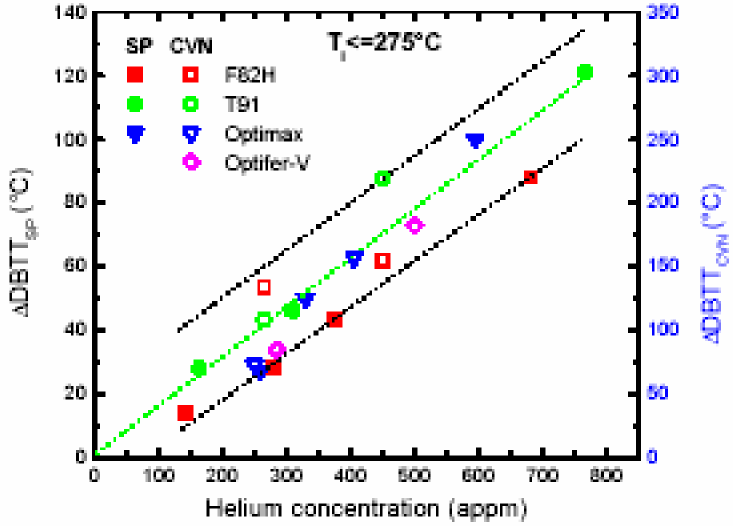


Irradiation under n&p: He-effects



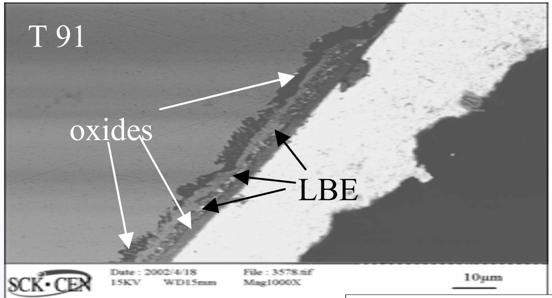
T91 implanted at 250 °C



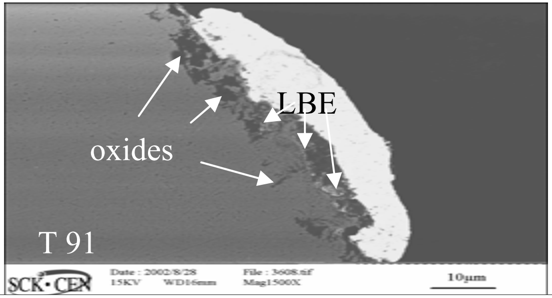


Corrosion problems in LBE

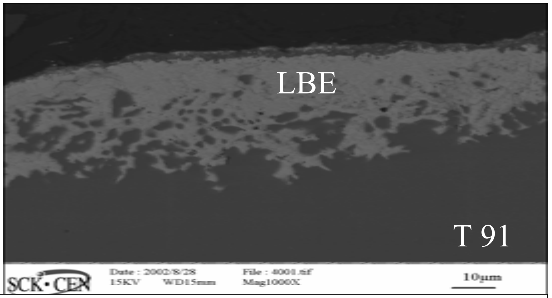
Stable oxide film

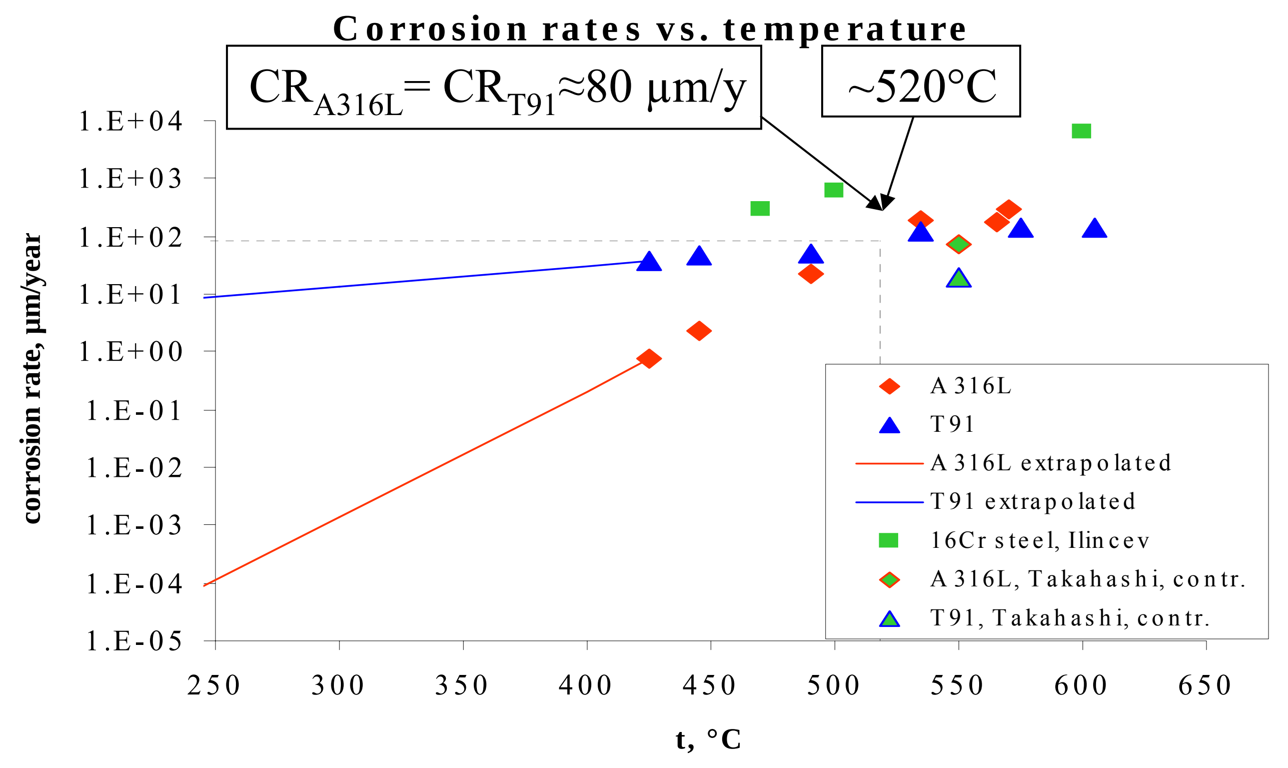


Transition mode



Dissolution mode





Corrosion rates vs. temperature

$CR_{A316L} = CR_{T91} \approx 80$ µm/y

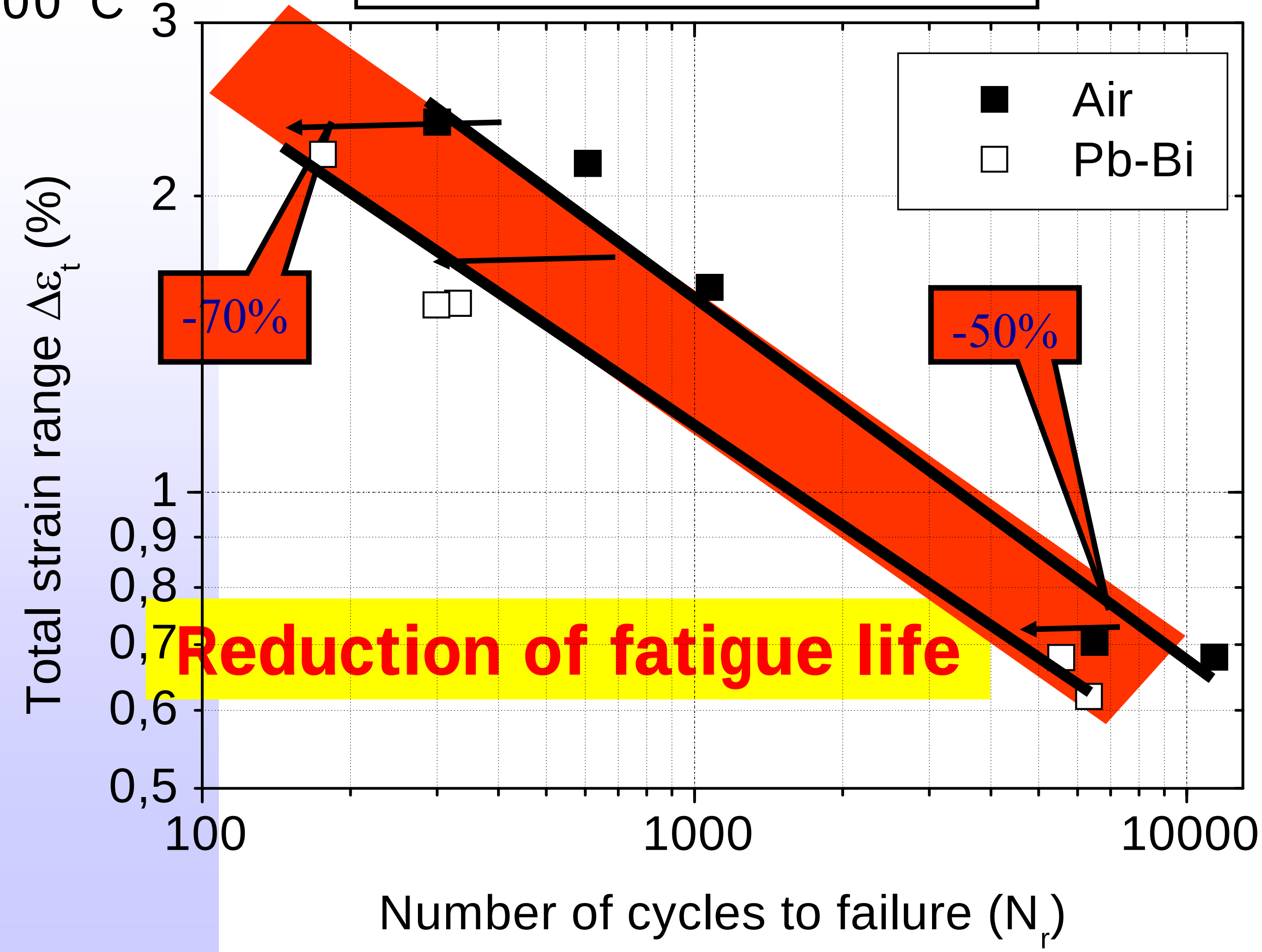
~520°C

*stagnant conditions, 5%H_2+Ar cover gas

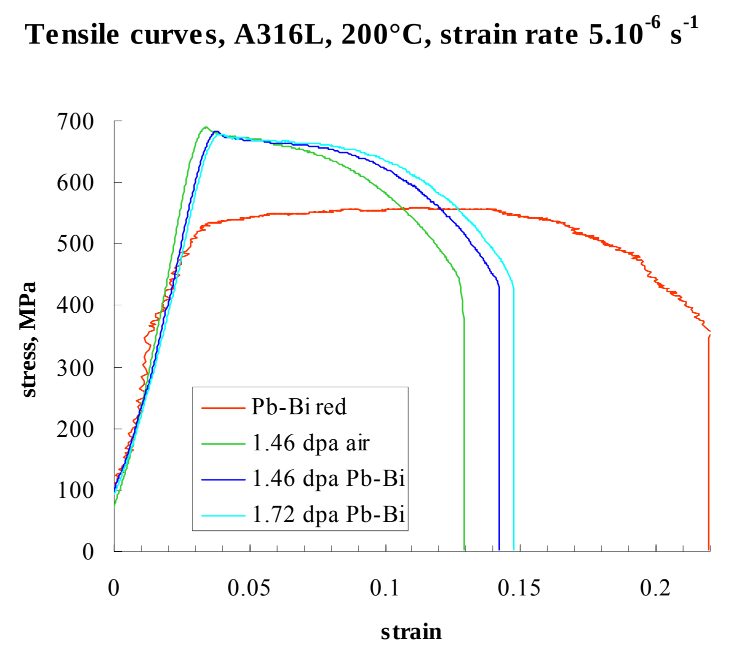
Fatigue of T91 in LBE (Vogt et al, JNM 2004)

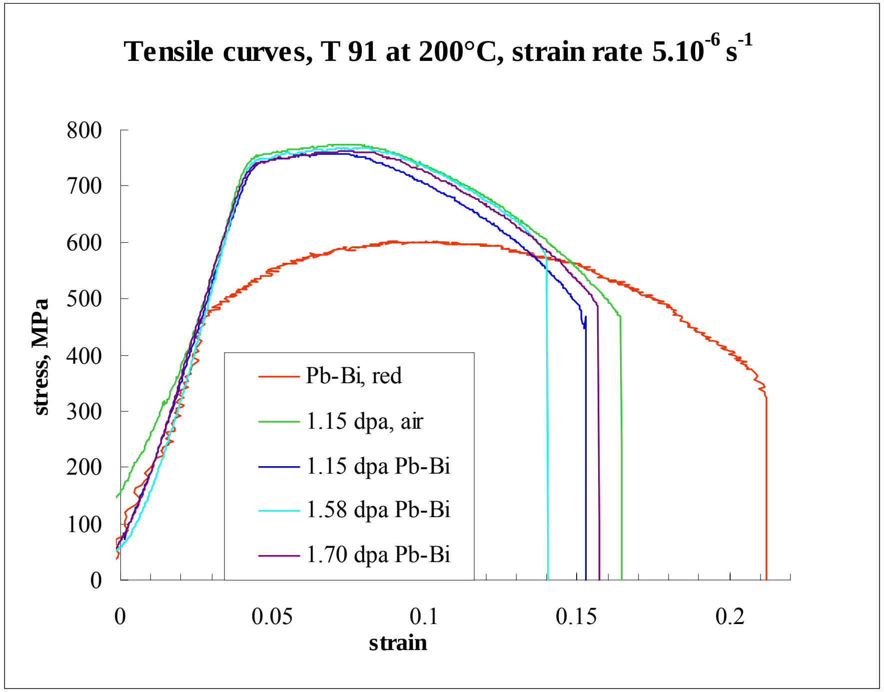
FATIGUE RESISTANCE

T = 300°C



Liquid Metal Embrittlement (LME) of irradiated materials





SSRT on irradiated 316L and T91 at 200 °C ➡ No evidence of LME!

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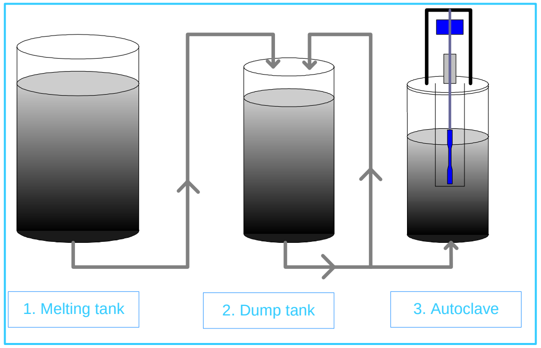
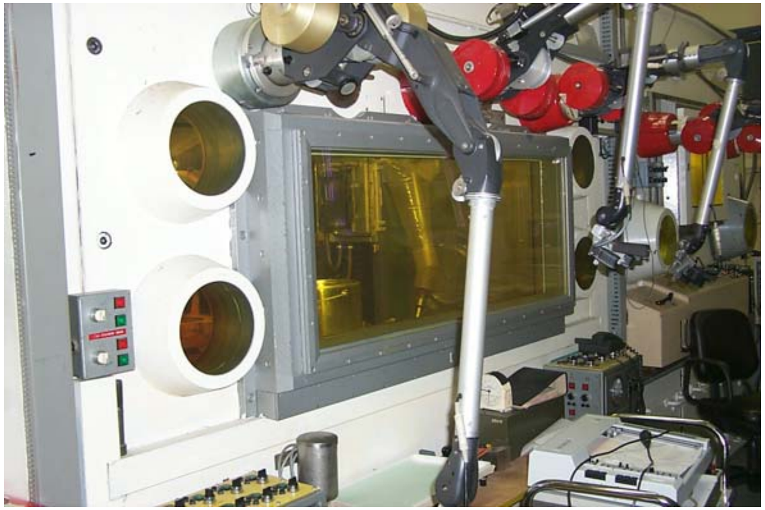
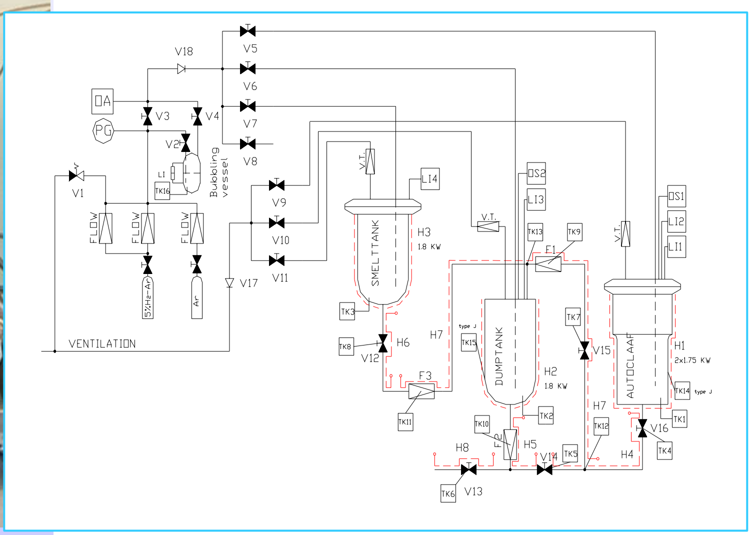
Dedicated facility for PIE in LBE available at SCK•CEN







- *Irradiated samples can be retrieved from the irradiation rig without any damage*
- *Tests can be performed under well controlled Liquid Metal chemistry*
- *SSRT, constant load and increasing load tests can be carried out at temperatures from 150°C to 500°C.*

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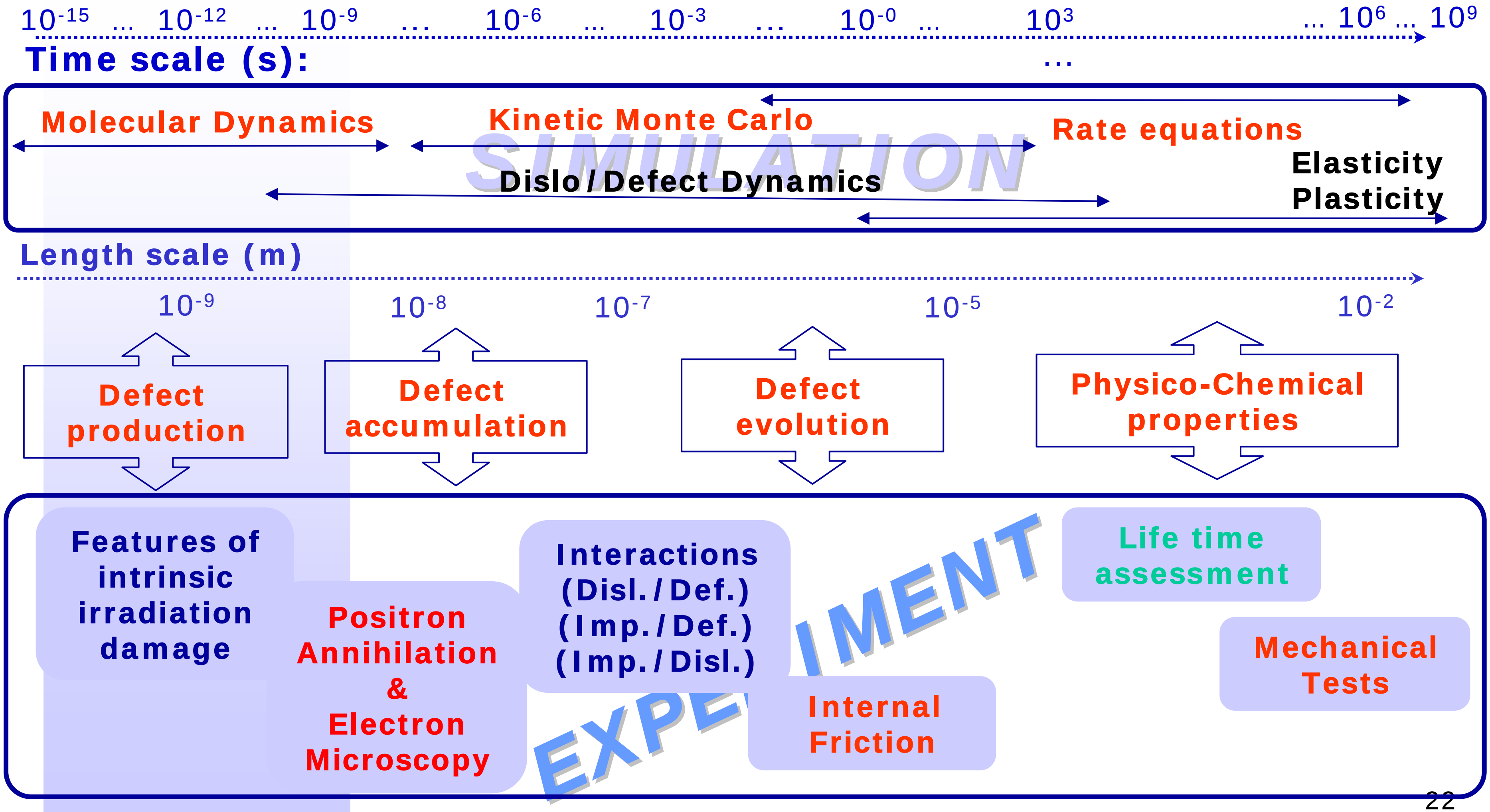
Multiscale computer simulation and experimental validation of irradiation damage in Fe-Cr based alloys



Time scale (s):

10^{-15} … 10^{-12} … 10^{-9} … 10^{-6} … 10^{-3} … 10^{-0} … 10^{3} … 10^{6} … 10^{9}

SIMULATION

Molecular Dynamics

Kinetic Monte Carlo

Rate equations

Dislo / Defect Dynamics

Elasticity
Plasticity

Length scale (m)

10^{-9} 10^{-8} 10^{-7} 10^{-5} 10^{-2}

Defect production

Defect accumulation

Defect evolution

Physico-Chemical properties

EXPERIMENT

Features of intrinsic irradiation damage

Positron Annihilation & Electron Microscopy

Interactions
(Disl. / Def.)
(Imp. / Def.)
(Imp. / Disl.)

Internal Friction

Life time assessment

Mechanical Tests

Conclusions

- In the last few years, a substantial effort has been made by SCK•CEN, in collaboration with partners inside and outside Europe (FP5:TECLA, SPIRE, MEGAPIE), on material issues for MYRRHA
- A R&D route is being set up in close collaboration with the design engineers
- With the aim of realising an XT-ADS at Mol, SCK•CEN is investing in:
  - ➢ Dedicated facilities (hot-cells, corrosion rigs, equipment for state-of-the-art material testing)
  - ➢ Special irradiation rigs in BR2 (higher temperatures)
  - ➢ Education of young researchers in material science (more than 10 PhDs, Post Docs or Engineers were hired during the last 2 years)