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Engineering Design of a Proof-of-Principle ADS Facility

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CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE

MYRRHA – Draft 2
Sub-critical Core Neutronics Design Calculations

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On behalf of MYRRHA team and MYRRHA support

http://www.sckcen.be/myrrha

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1. Core configuration
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7. Reactivity effects and Operational sub-criticality margins
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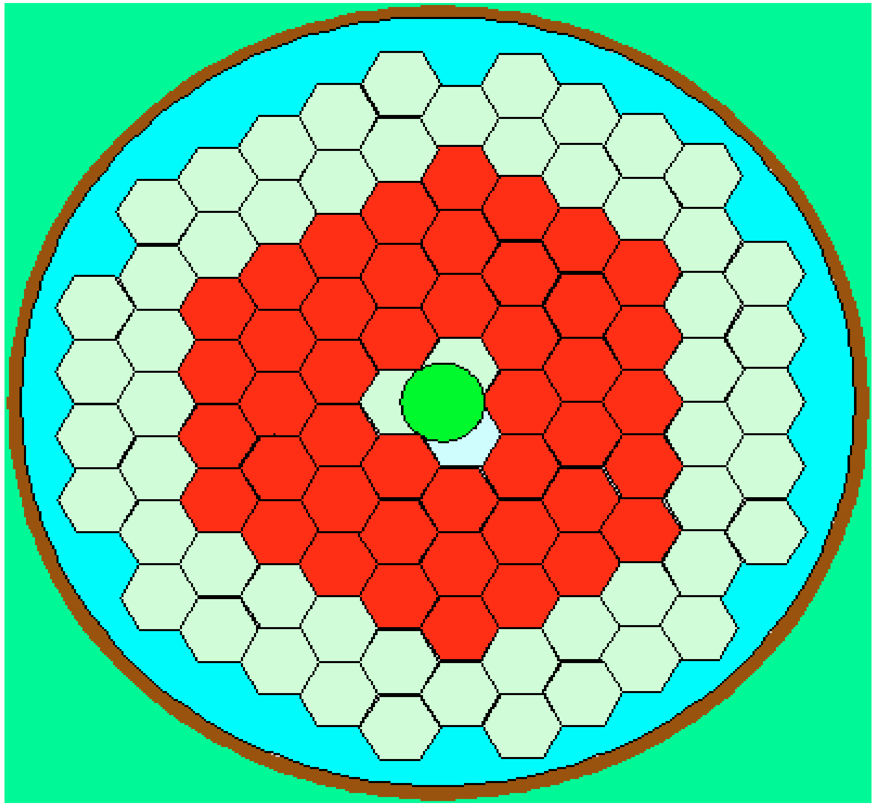
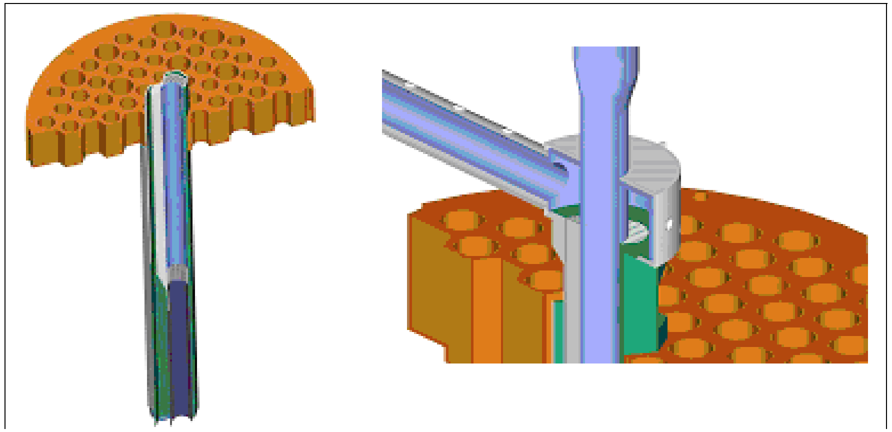
Sub-critical core design requirements and constraints

- Proton Beam:
  - 350 MeV-5 mA
  - Spot size (FWHM)=15 mm (gaussian spatial shape assumed)
- The initial $k_{eff} \sim 0.95$
- Nominal power ~*50 MWth*
- Fast neutron flux: $\sim 10^{15}$ n/cm²s
- Thermal neutron flux (inside IPS-like loop): $1.0 - 2.0\ 10^{15}$ n/cm²s

MYRRHA ADS: Typical Core Configuration





- 102 channels
- Target-block hole fitted out within the 3 central channels
- Surrounding active zone loaded with 45 fuel SA (30wt% Pu/HM; 91 pins/SA)
- Outer reflector zone composed of 54 "reflector" assemblies

Core Analysis tools

- Nuclear data (within table range; INC-model beyond):
  - Neutrons: JEF2.2 (MCB-package) combined to LA150n(Pb, Bi and steel elements);
  - LA150h or physical models for protons.
- MCNPX 2.5.e beta version used:
  - Enables one to "mix-and-match" data tables having different upper energy boundaries and table data with INC models
- ALEPH (home-made)code, coupling MCNPX and ORIGEN2.2 in a more efficient way, to carry out core burn-up calculations
  - Nuclear data: JEF2.2 processed using NJOY99.90

Geometrical model

- Updated and Completed
  - Fuel pin and assembly design revised
  - Assembly extension parts from the inlet nozzle through the outlet nozzle
  - Assembly and fuel-pin bundle grids
  - Core barrel and core suspension tube
  - Top lid and radial shielding concrete
  - Top (pool) gas plenum
  - Spallation target loop (inner part)

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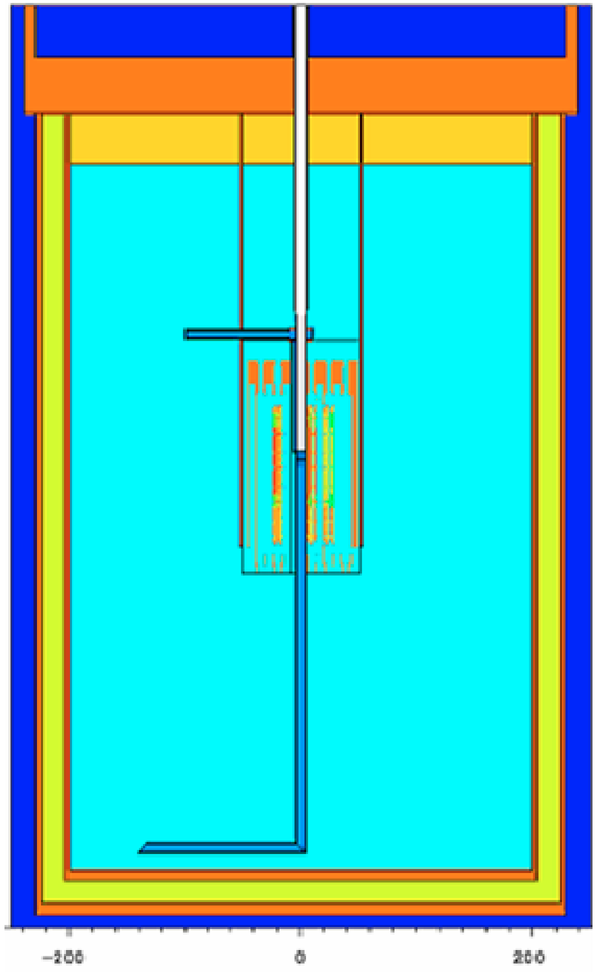
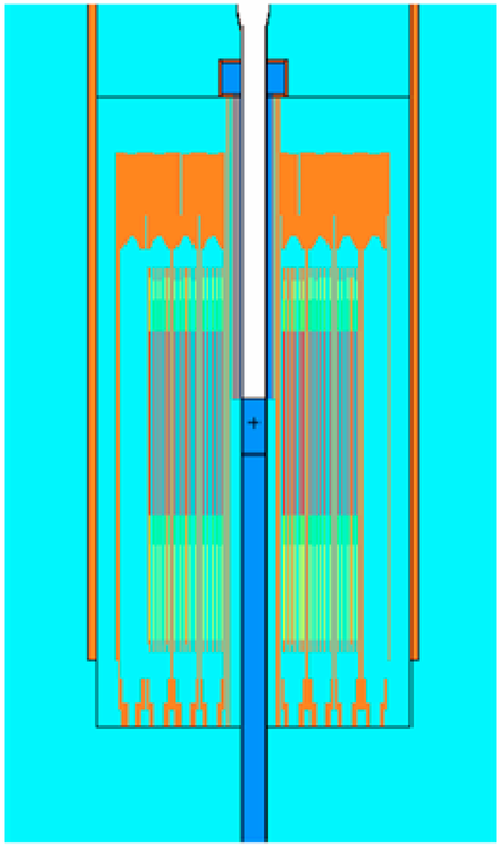
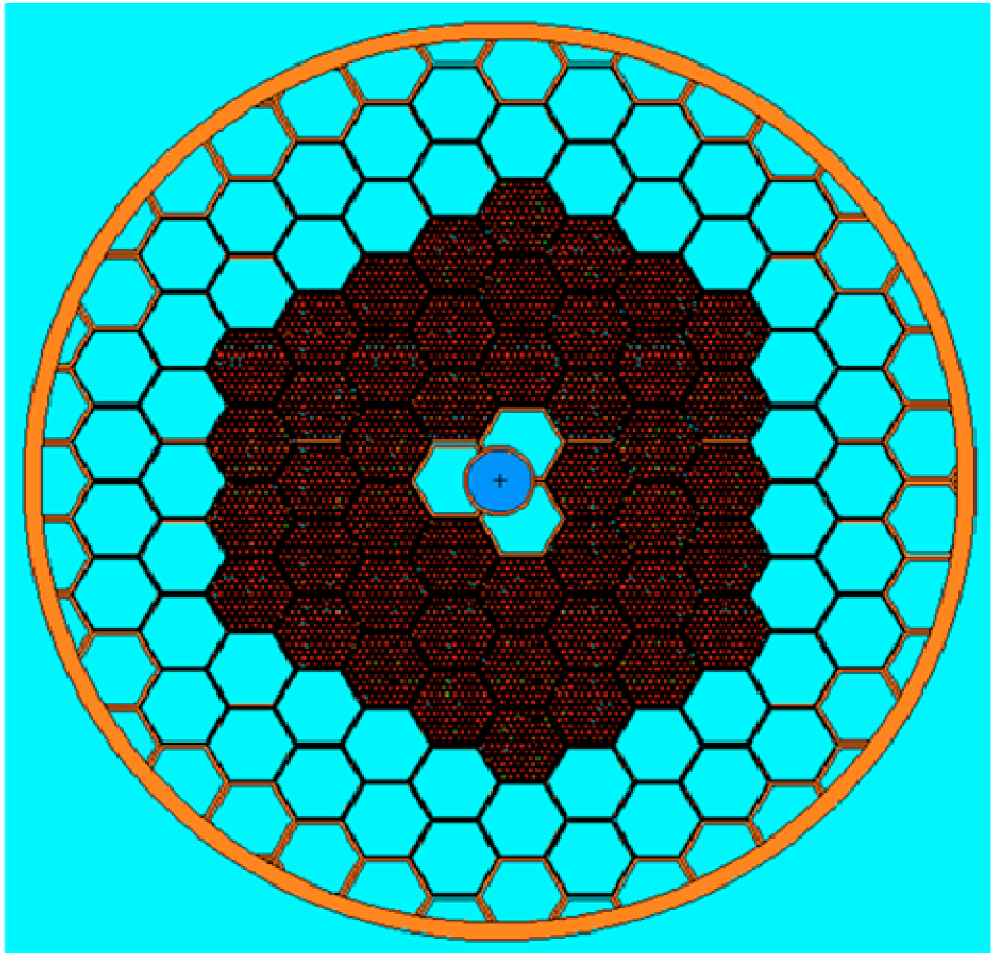
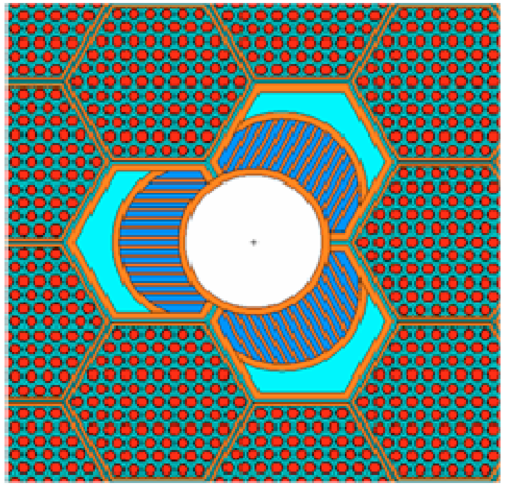
MYRRHA MODEL for MCNPX calculations

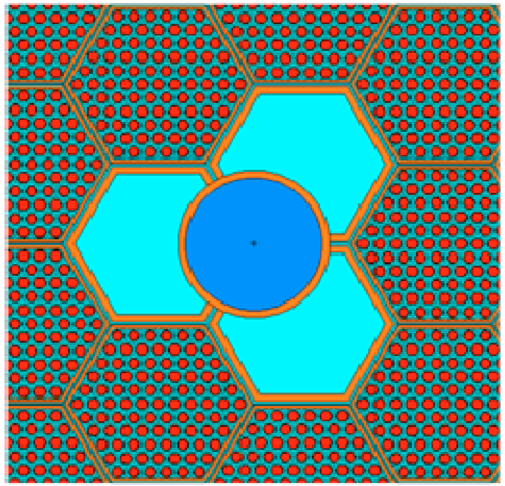




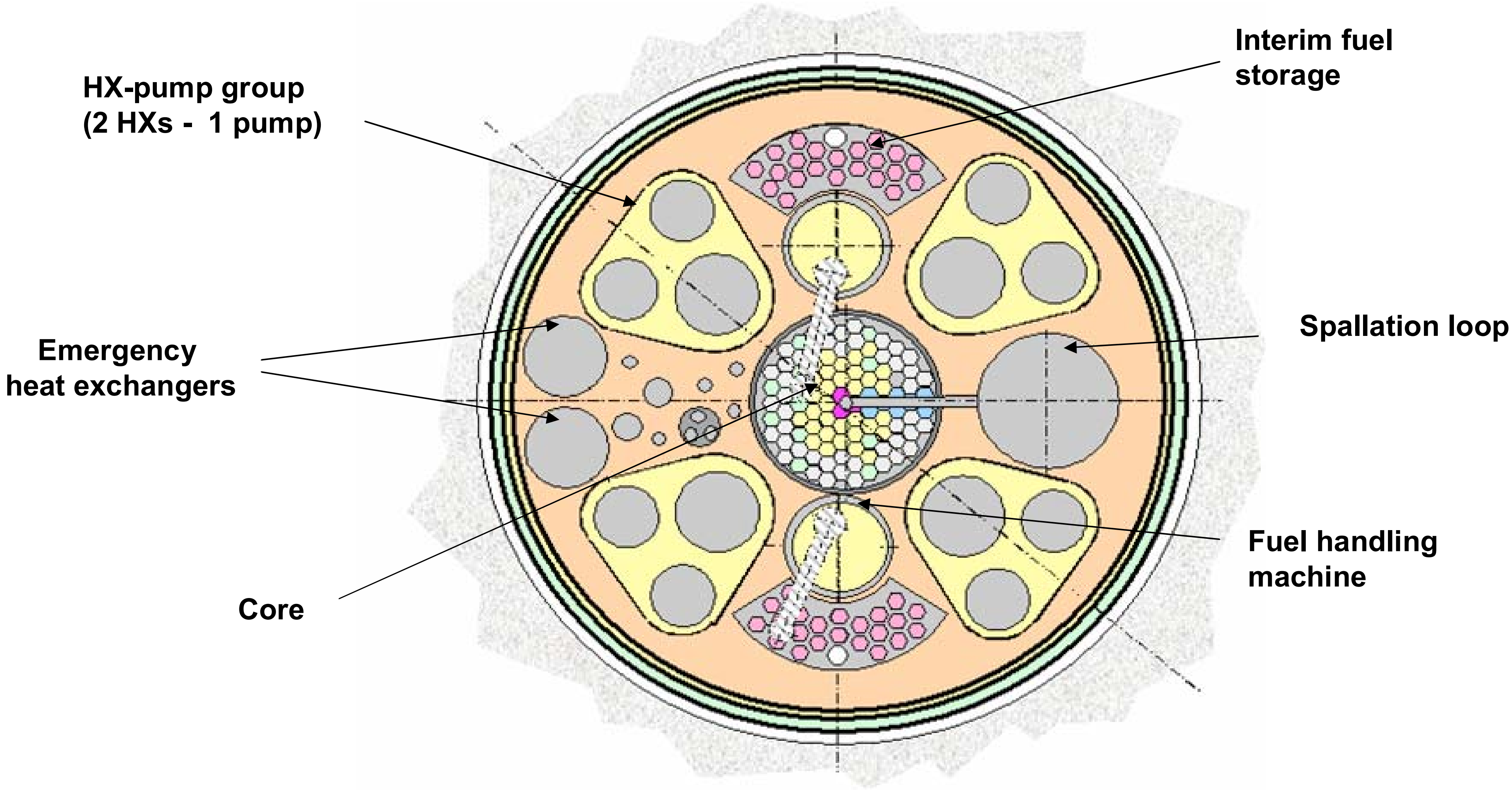
MYRRHA MODEL for MCNPX calculations (cont'd)







MYRRHA: general sketch



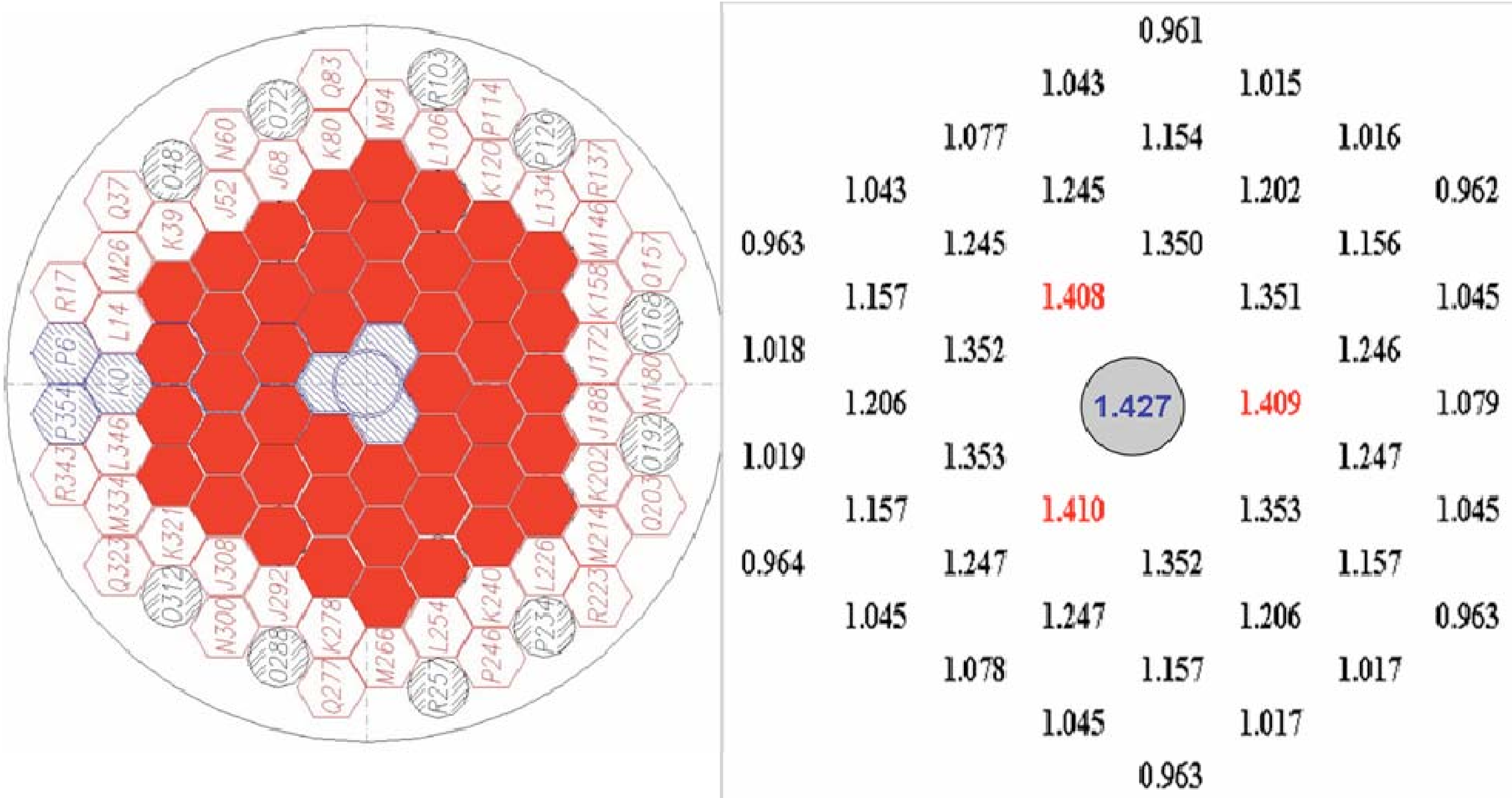
Results

- Comprehensive and reliable set of results provided:
  - Reactivity effects
  - Nuclear data sensitivity analysis
  - Operational sub-criticality margins
  - Consistent Power and Flux maps
  - Irradiation-induced damage parameters (DPA, gas-production)
  - MA and LLFP transmutation performances
- Improved quality of document

Overview of the MYRRHA core characteristics (BOL)

Neutronics Parameters	Units	MYRRHA ADS values
Proton beam energy	MeV	350
Accelerator current	mA	5
Proton beam heating	MW	1.43
Spallation neutron yield		6.03
neutron source Intensity	10^{17} n/s	1.88
Initial fuel mixture	MOX	$(U\text{-}Pu)O_2$
Initial (HM) fuel mass (m_{fuel})	Kg	514
Initial Pu-enrichment (Pu/U+Pu)	wt%	30
Initial Pu isotopic vector $^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$	wt%	1.27/61.88/23.50/8.95/4.40
K_{eff}		0.9552
K_s		0.9601
MF = 1 / (1 - K_s)		25.04
Source importance: φ*		1.127
Thermal Power (‡) (P_{th})	MW	51.75
Av. Fuel power density (P_{th}/V_{fuel})	W/cm^3	937
Specific power	kW/kgHM	101
Peak linear Power (hottest pin)	W/cm	352
Av. Linear Power (hottest pin)	W/cm	272
Max Φ_{total} in the fast core (near the hottest pin)		4.1
Max $\Phi_{>1\ MeV}$ in fast core (near the hottest pin)	10^{15} n/cm^{2}s	0.8
Max $\Phi_{>0.75\ MeV}$ in fast core (near the hottest pin)		1.0
(‡) E_f = 210 MeV/fission		

Sub-critical Core:
Assembly Power map



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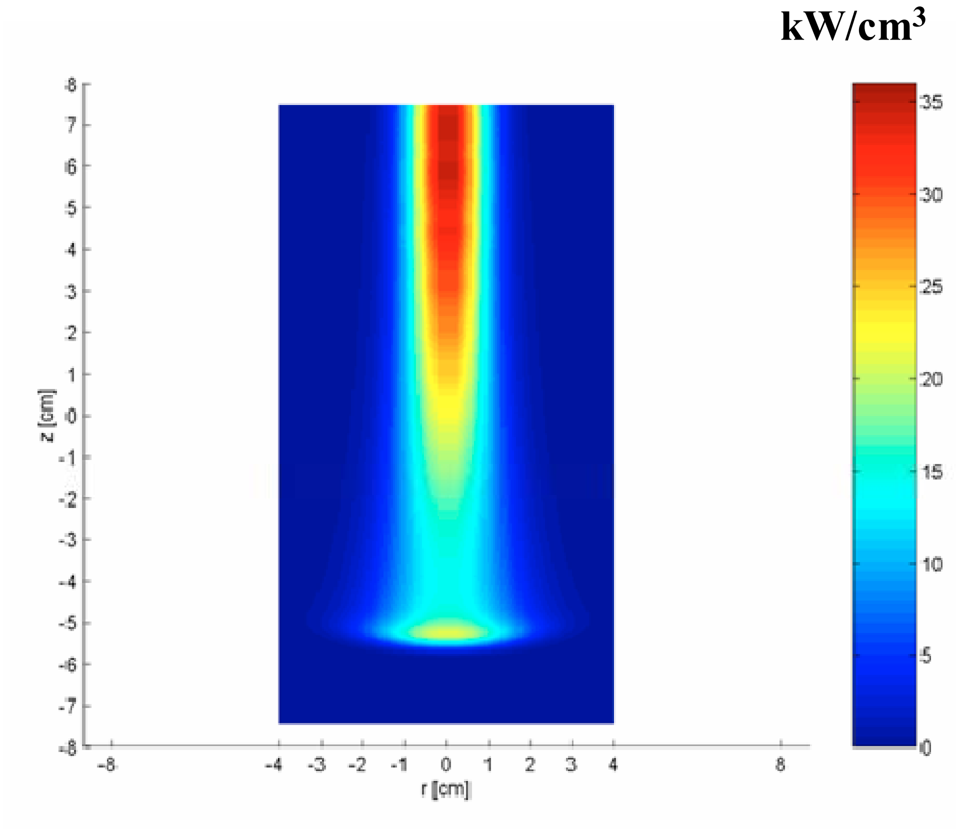
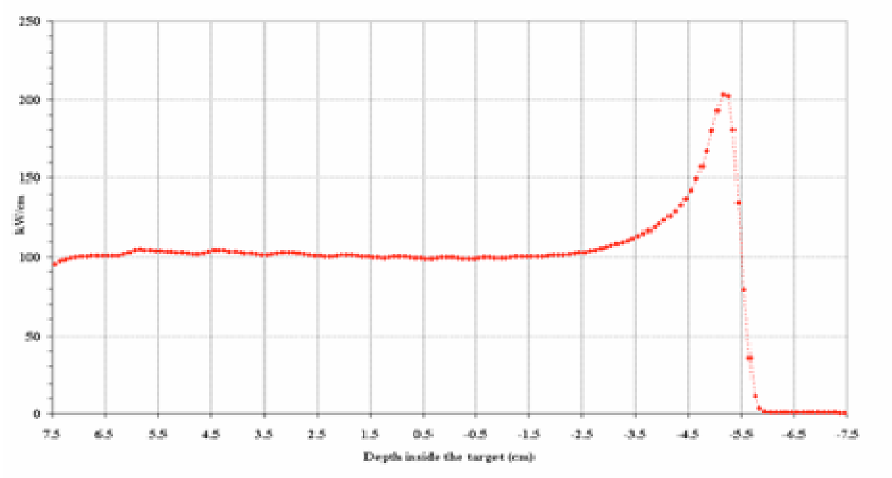
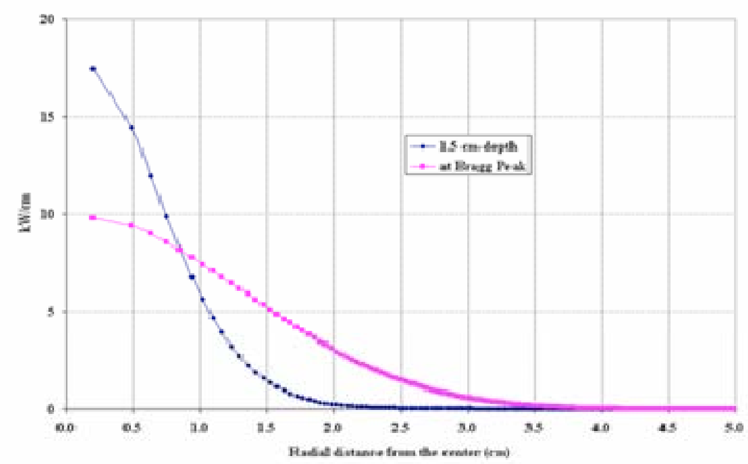
Spallation target Heating

MYRRHA







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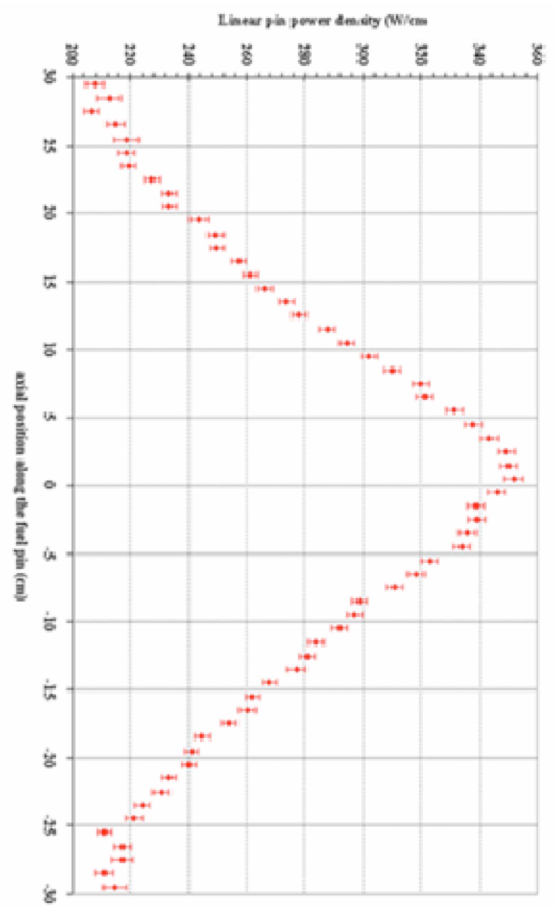
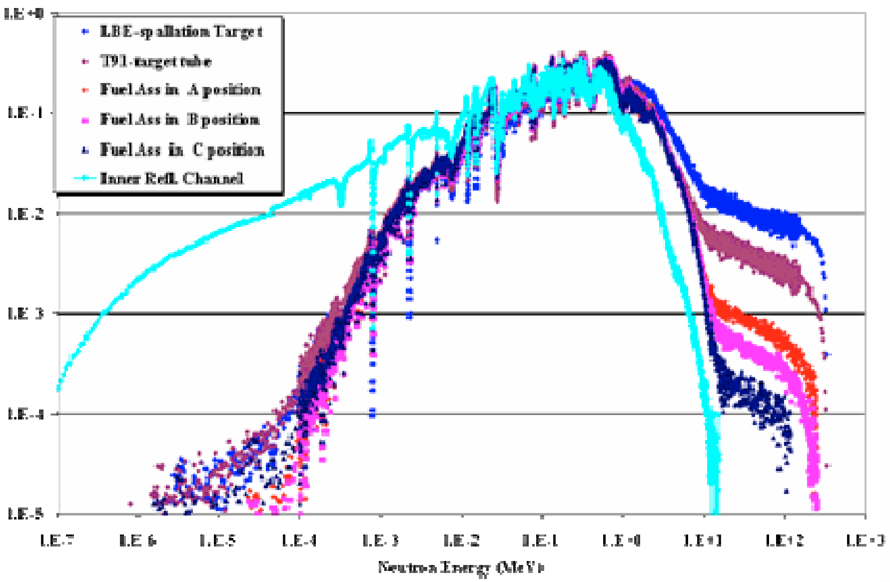
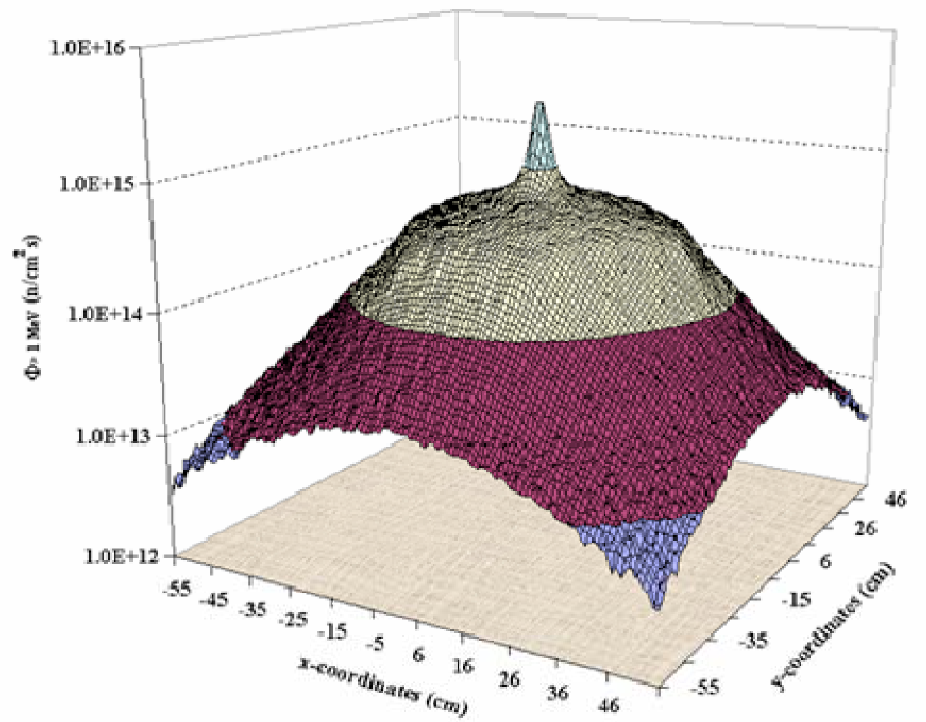
Pin-by-pin power map (hot assembly) and linear power density curve (hot pin)

15.72	15.58	15.50	15.30	15.15	14.98					
15.84	15.74	15.58	15.47	15.31	15.11	14.97				
15.97	15.83	15.72	15.61	15.47	15.30	15.11	14.93			
16.09	15.97	15.82	15.70	15.56	15.41	15.27	15.10	14.85		
16.18	16.06	15.93	15.79	15.67	15.52	15.37	15.19	14.98	14.76	
16.30	16.11	16.00	15.85	15.75	15.62	15.42	15.28	15.03	14.85	14.68
16.16	16.05	15.91	15.80	15.62	15.44	15.32	15.16	14.93	14.78	
16.11	15.97	15.84	15.71	15.59	15.38	15.21	15.04	14.84		
15.99	15.84	15.73	15.59	15.45	15.29	15.08	14.91			
15.83	15.75	15.61	15.47	15.33	15.11	14.94				
15.68	15.60	15.46	15.33	15.11	14.96					

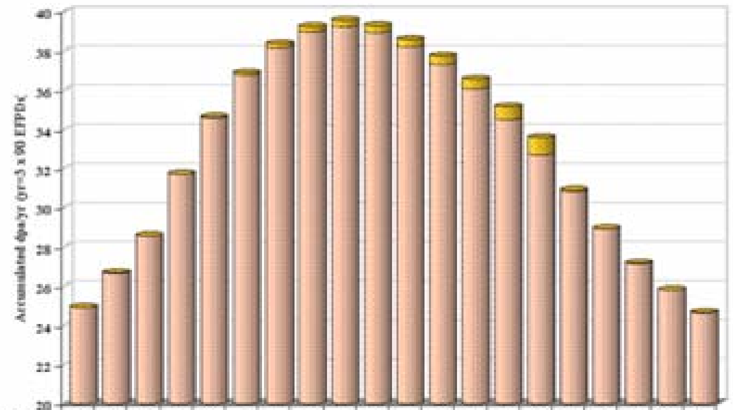


Spectra and Flux

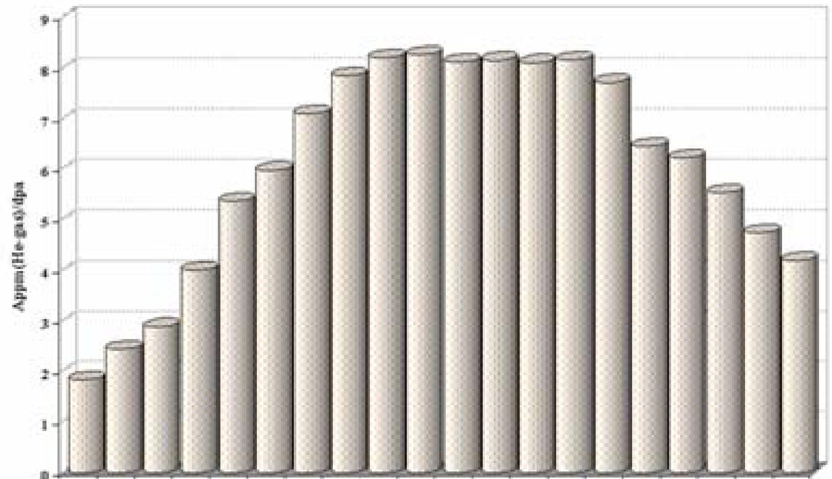




DPA-damage and Helium-gas production in T-91steel pipe

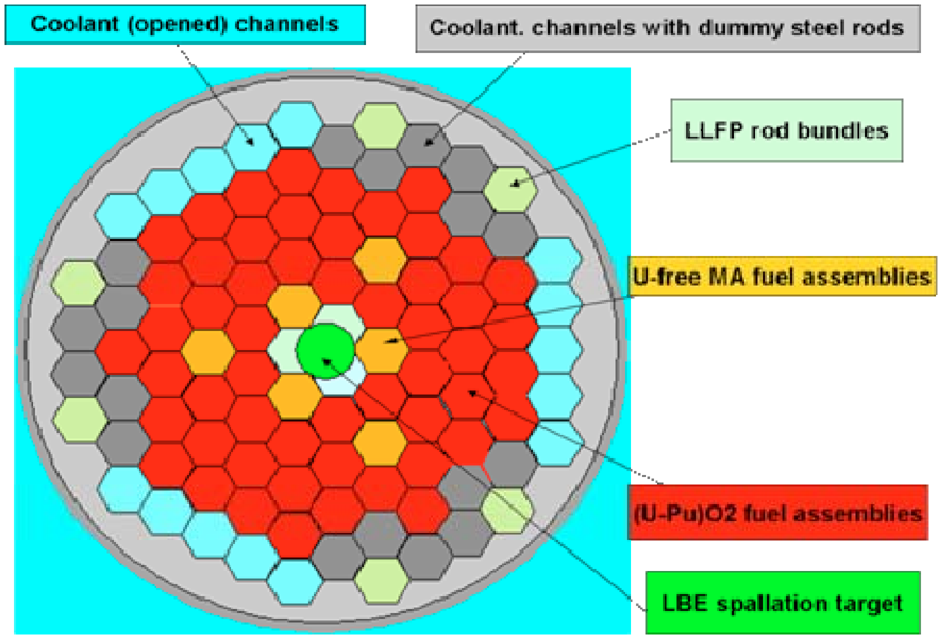


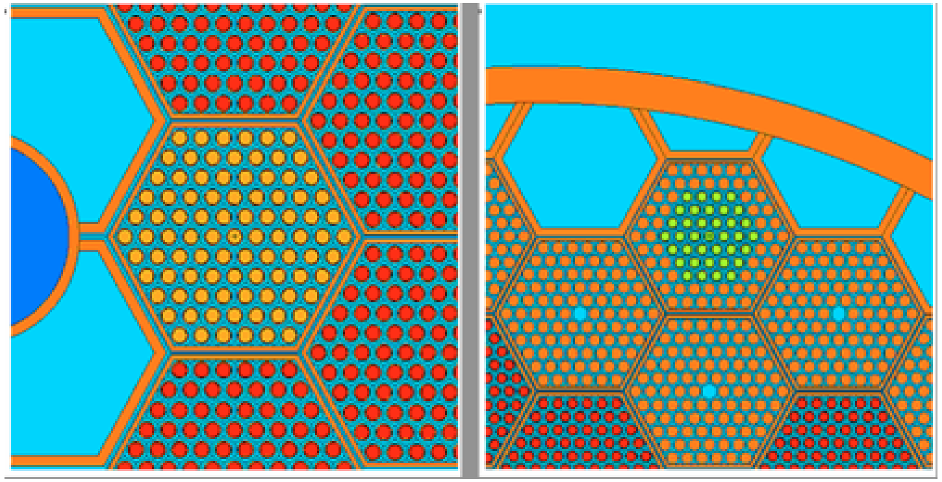
mid-z -->	9.5	8.5	7.5	6.5	5.5	4.5	3.5	2.5	1.5	0.5	-0.5	-1.5	-2.5	-3.5	-4.5	-5.5	-6.5	-7.5	-8.5	-9.5
p-dpa/yr	0.0	0.0	0.0	0.0	0.1	0.1	0.2	0.3	0.3	0.4	0.4	0.4	0.5	0.7	0.9	0.1	0.0	0.0	0.0	0.0
n-dpa/yr	24.9	26.7	28.6	31.7	34.6	36.8	38.2	39.0	39.2	38.9	38.2	37.3	36.1	34.5	32.7	30.8	28.9	27.2	25.8	24.7



mid-z -->	9.5	8.5	7.5	6.5	5.5	4.5	3.5	2.5	1.5	0.5	-0.5	-1.5	-2.5	-3.5	-4.5	-5.5	-6.5	-7.5	-8.5	-9.5
appm(He)/dpa	1.8	2.4	2.9	4.0	5.4	6.0	7.1	7.9	8.2	8.3	8.1	8.2	8.1	8.2	7.7	6.5	6.2	5.5	4.7	4.2

MA and LLFP transmutation: Core loading





MA pellet vector	wt% fraction
Pu/Am/Cm/Mg/O	23.25/30.32/6.06/19.18/20.19
$^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$	5.06/37.91/30.31/13.21/13.51
$^{241}Am/^{243}Am$	66.67/33.33
$^{244}Cm/^{245}Cm$	90/10

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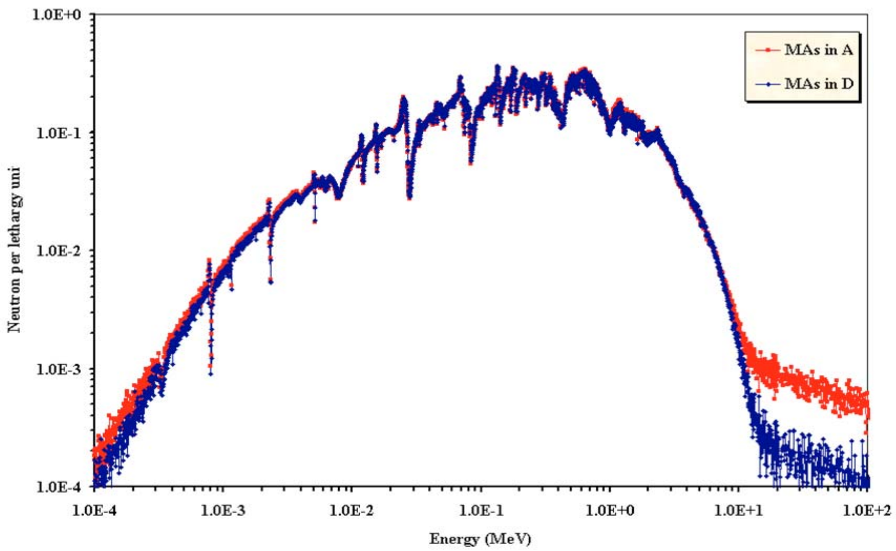
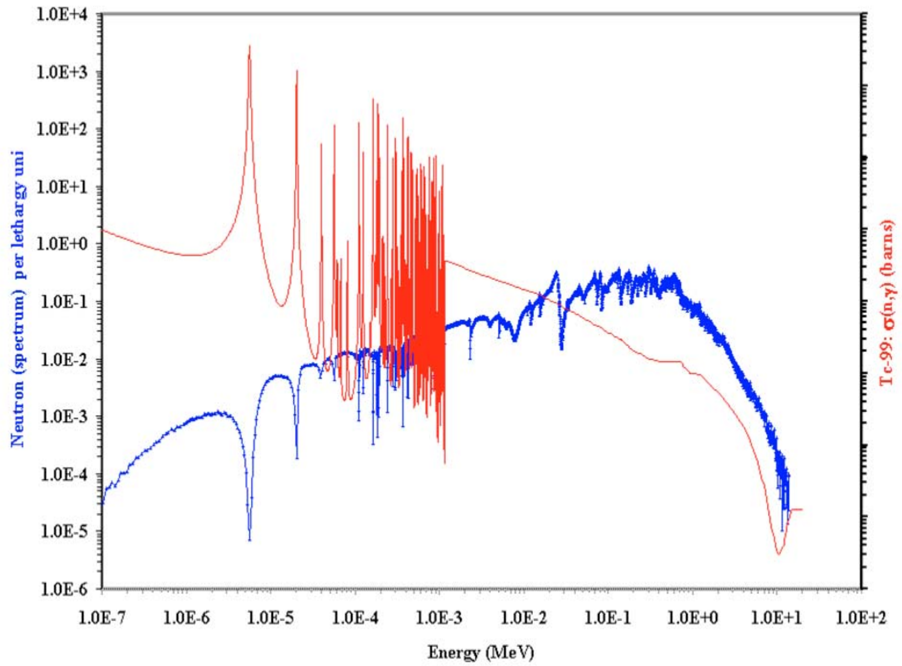
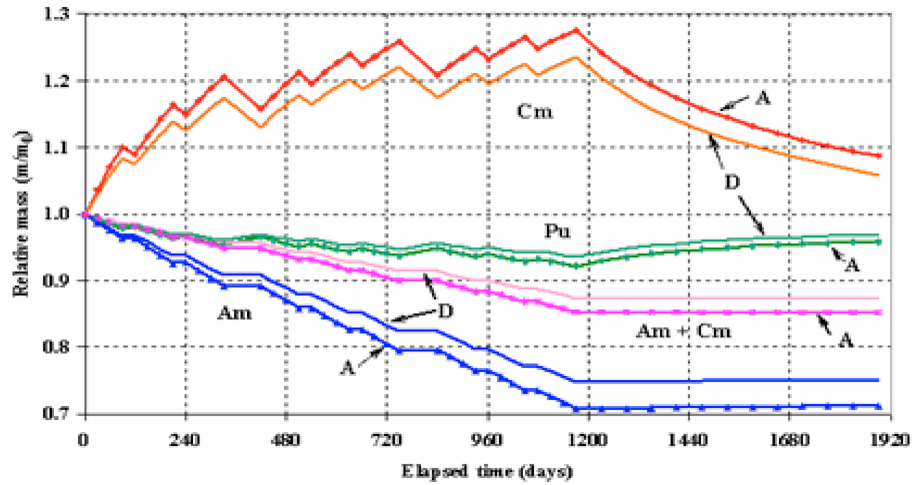
Neutron spectra in MA and LLFP samples



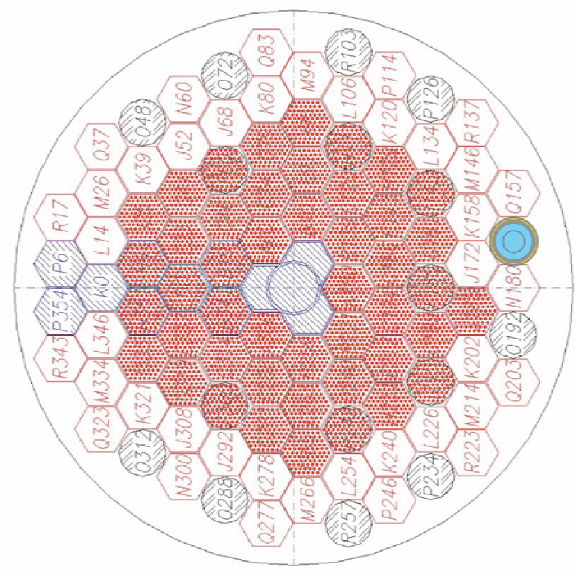


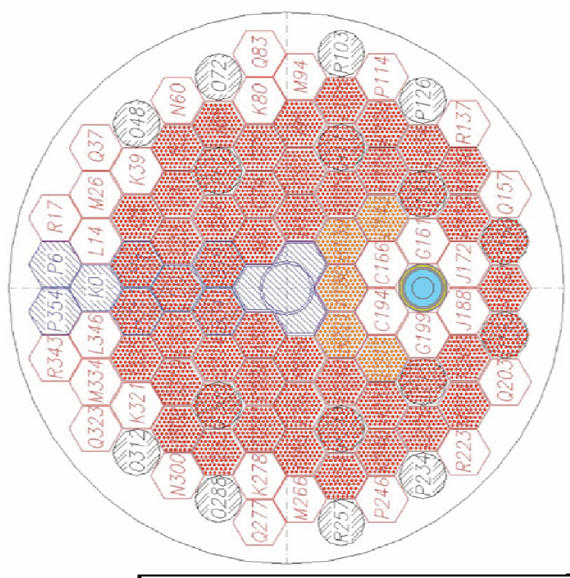
MA and LLFP (amounts in gram)

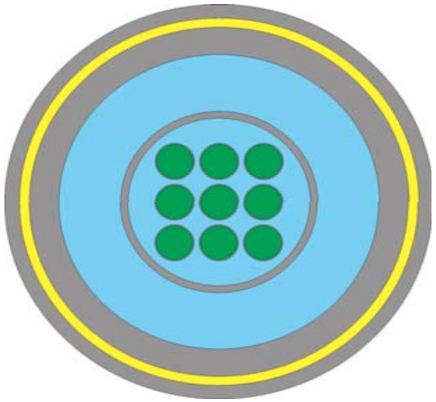
	A or D	A	D	A	D
Pu-238	183	326	294		
Pu-239	1372	-493	-434		
Pu-240	1097	-15	3		
Pu-241	478	-179	-167		
Pu-242	489	79	72		
Pu	3619	-282	-232	-7.8%	-6.4%
Am-241	3015	-1025	-892		
Am-242		124	115		
Am-243	1507	-419	-366		
Am	4522	-1319	-1143	-29.2%	-25.3%
Cm-242	0	107	97		
Cm-243	0	6	5		
Cm-244	813	104	84		
Cm-245	90	26	22		
Cm-246	0	5.8	4.9		
Cm	903	249	212	27.5%	23.4%
All (Z>88) Actinides	9044	-1333	-1143	-14.7%	-12.6%
ℵtot (n/cm^2s)		$3.15\ 10^{15}$	$2.71\ 10^{15}$		



Core configuration with IPS-like water-moderated loop

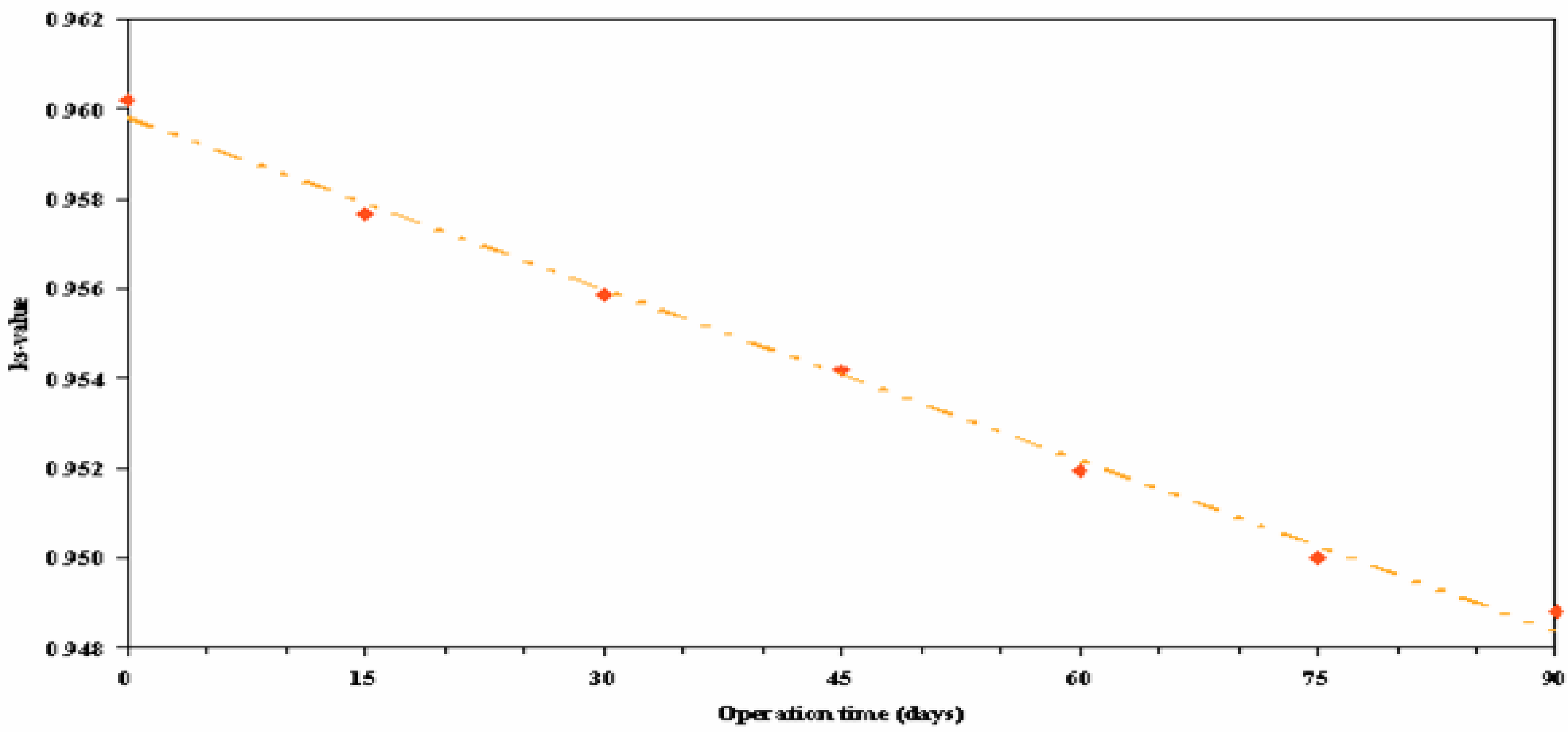






	inner position	outer position
ks	0.959	0.957
Ptot (MW)	50	48
Peak linear Power (W/cm)	327	311
Total flux (10^{15} n/cm^{2}s)	2.5	1.7
¬<0.5 eV (10^{15} n/cm^{2}s)	1.1	0.8

K_s swing

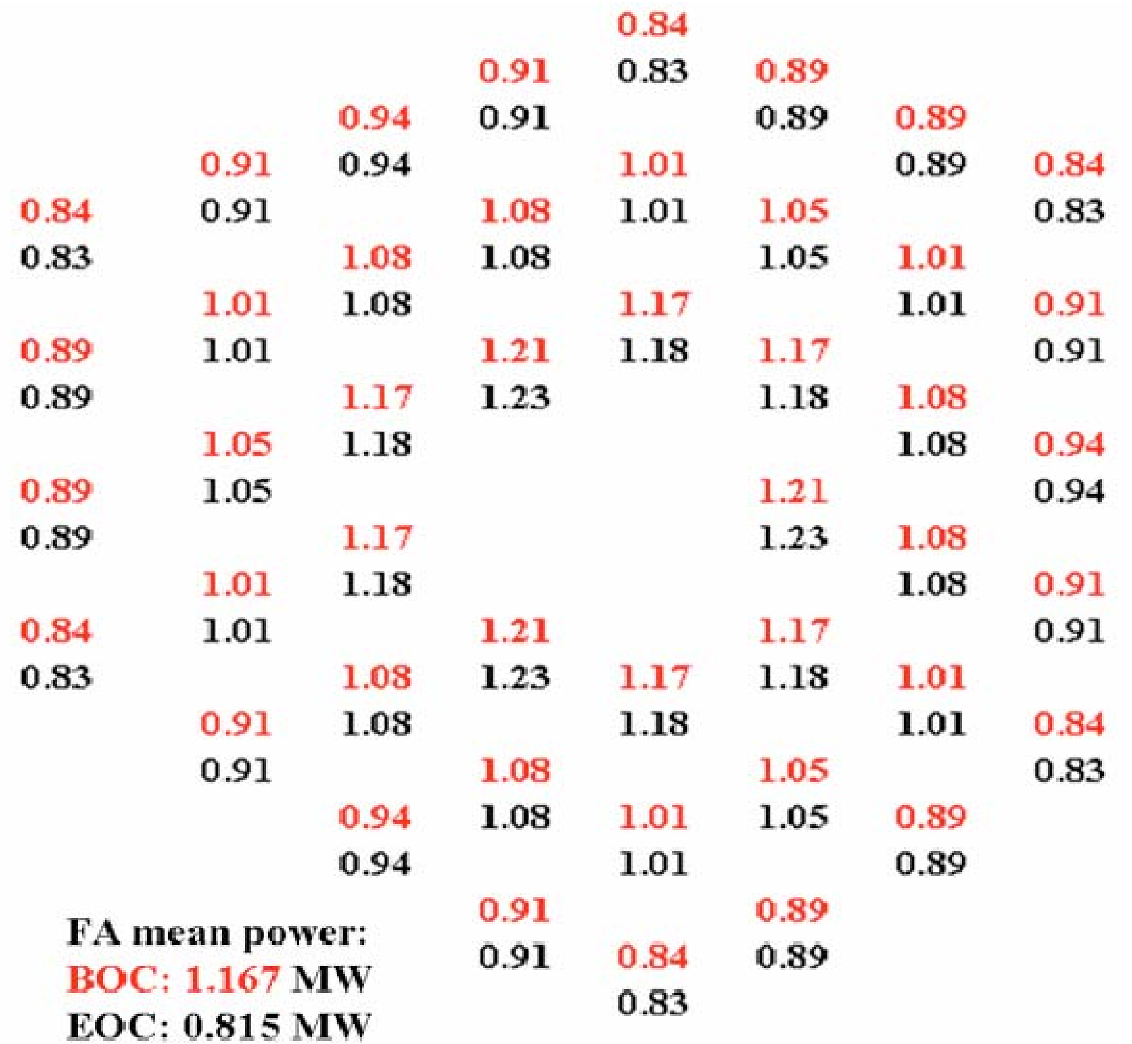


- $\Delta\rho$=-1667 pcm pcm/cycle (1cycle=90 EFPDs) (i.e., -19 pcm/EFPD).

Fuel burn up after 90 EFPDs in MWd/kgHM)

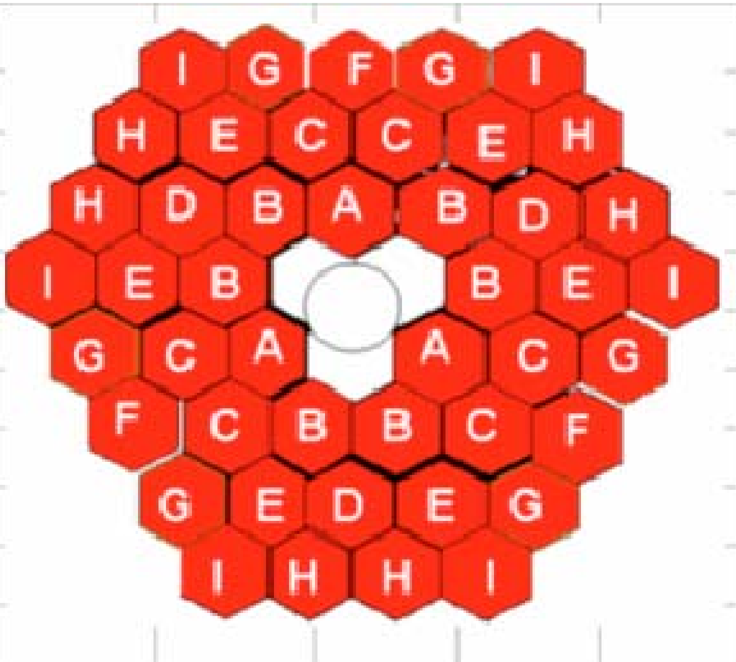
				6.57			
			7.13		7.19		
		7.36		7.88		7.19	
	7.13		8.48		8.20		6.57
6.57		8.48		9.18		7.88	
	7.88		9.56		9.18		7.13
7.19		9.18				8.48	
	8.20				9.56		7.36
7.19		9.18				8.48	
	7.88		9.56		9.18		7.13
6.57		8.48		9.18		7.88	
	7.13		8.48		8.20		6.57
		7.36		7.88		7.19	
			7.13		7.19		
				6.57			

Assembly relative power at BOC and at EOC

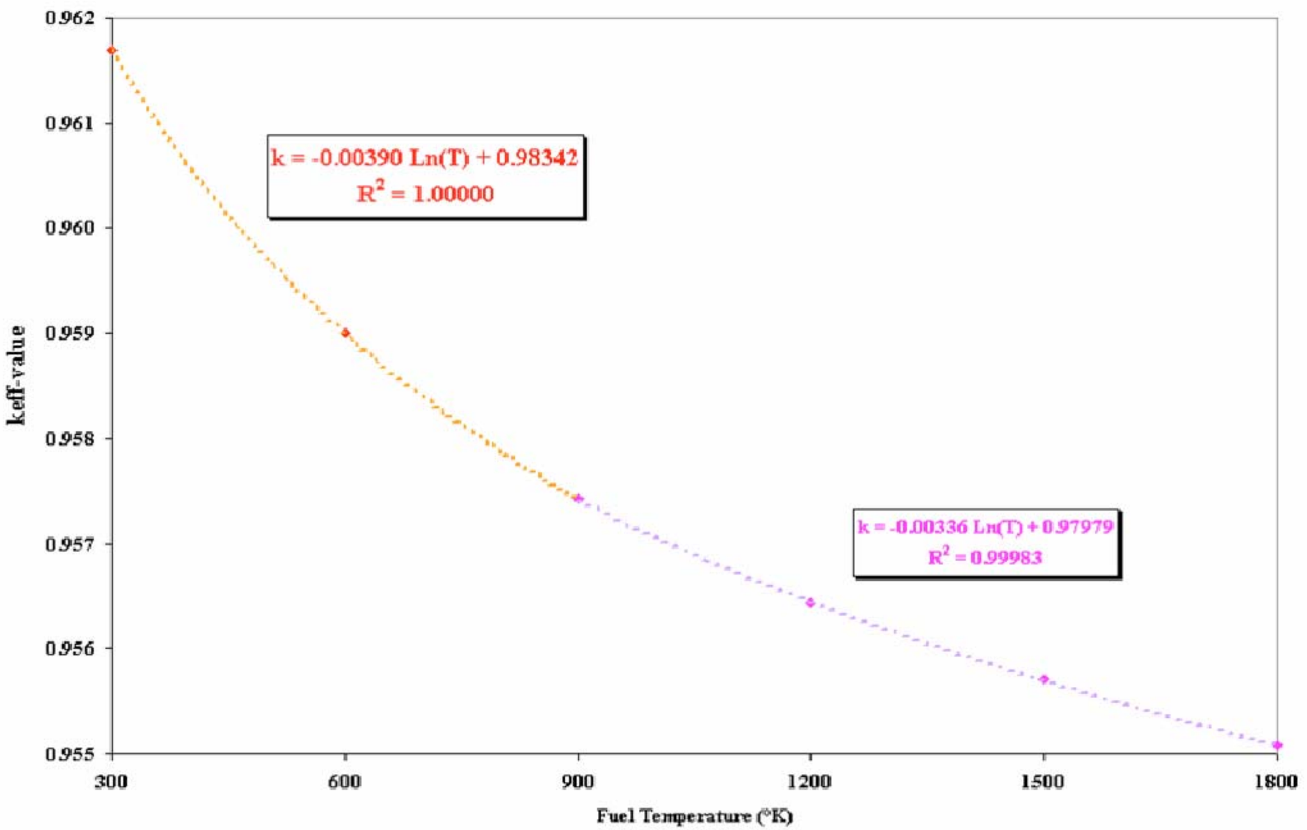


Fuel Assembly reactivity worth map

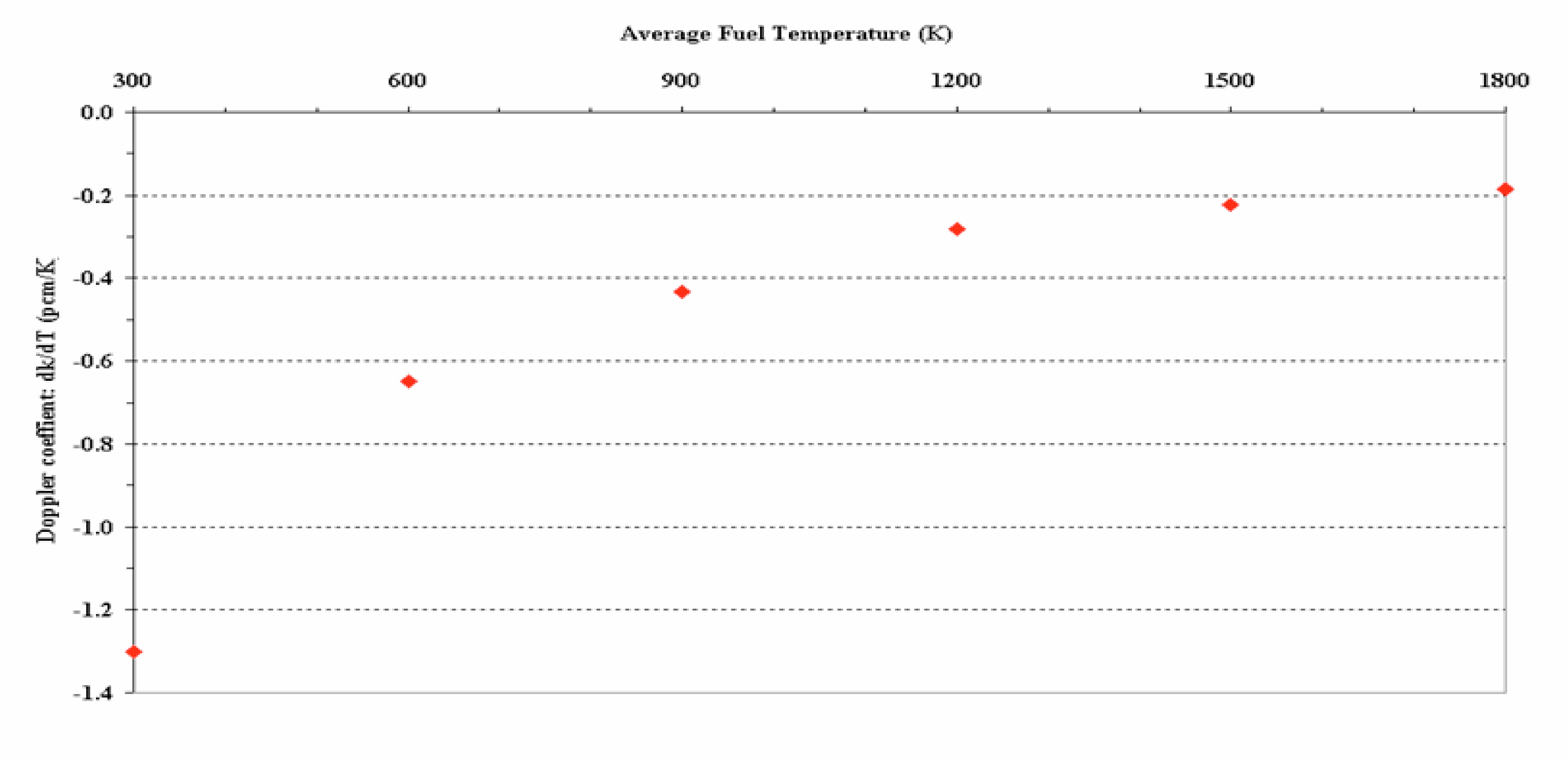
	$\Delta\rho$ (pcm)		$\Delta\rho_s$ (pcm)	
FA position	BOC	EOC	BOC	EOC
A	1406	1403	1260	1239
B	1320	1368	1237	1330
C	1195	1227	1048	1126
D	1128	1215	948	976
E	1030	1104	896	927
F	827	878	694	796
G	756	807	708	675
H	754	742	632	675
I	612	677	568	516



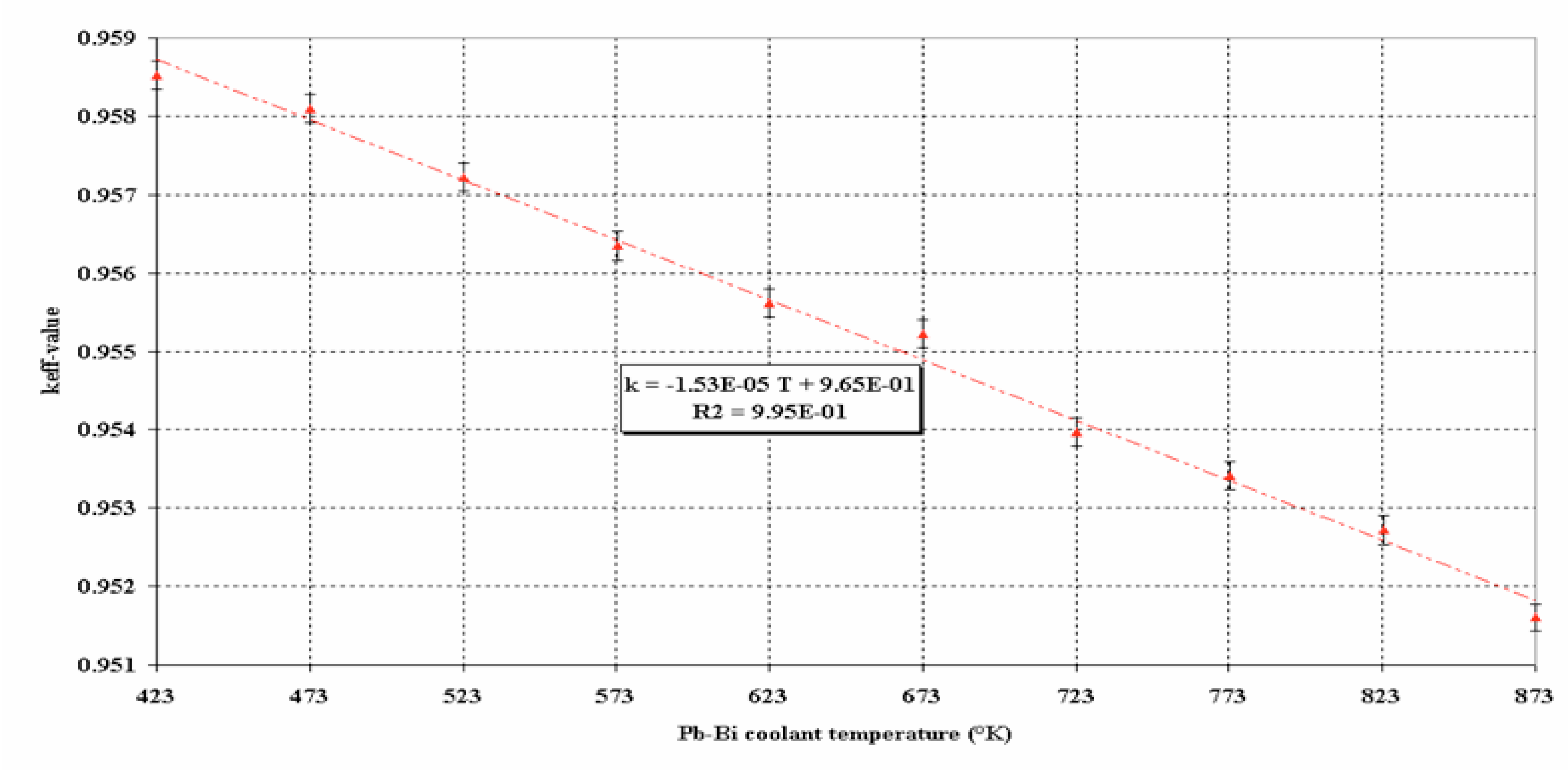
Fuel Temperature (Doppler) effect Doppler constant ($K_D = Tdk/dT$)



Fuel Temperature (Doppler effect) Doppler coefficient (dk/dT)



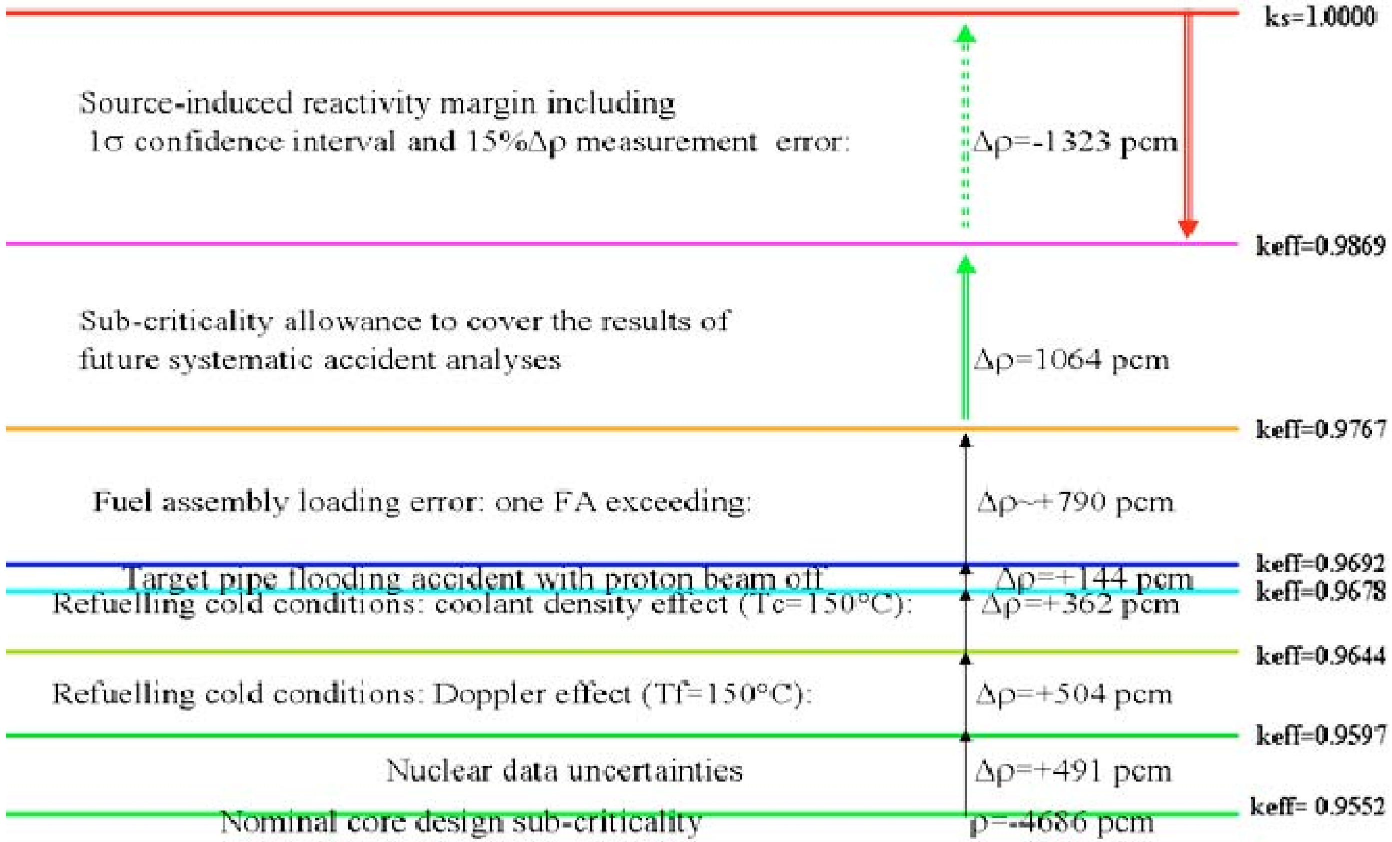
Coolant Temperature (density) reactivity effect (dk/dT)



Sensitivity to neutron cross-section libraries

Nuclear data		err	errmax	errint	Ptable	ks	keff	Δρ	φ*
MCB	**JEF 2.2**	0.002	0.04	5.0E-7	yes	0.95961	0.95506	**496**	**1.12**
		0.002	0.04	5.0E-7	no	0.95979	0.95578	**437**	**1.10**
	ENDF 6.8	0.002	0.04	5.0E-7	yes	0.96881	0.95895	**1061**	**1.33**
SCK•CEN	**JEF 2.2**	0.001	**Optimal accuracy**		yes	0.96470	0.95479	**1076**	**1.29**
		0.001	0.01	5.0E-8	yes	0.96423	0.95435	**1074**	**1.29**
		0.001	0.01	5.0E-8	no	0.96457	0.95568	**964**	**1.26**
		0.002	0.02	1.0E-7	yes	0.96437	0.95509	**1008**	**1.27**
		0.002	0.04	5.0E-7	yes	0.96464	0.95480	**1068**	**1.29**
	ENDF 6.8	0.001	0.01	5.0E-8	yes	0.96898	0.95971	**997**	**1.31**
	JEFF 3.0	0.001	0.01	5.0E-8	yes	0.96511	0.95533	**1061**	**1.29**

Estimated operational sub-criticality margins



Concluding remarks

- The sub-critical core achieves a primary source neutron multiplication factor, ks, of 0.9600 (the keff-eigenvalue being 0.9552). The adopted sub-criticality level, -4686 pcm, is larger enough to keep the MYRRHA core far away from criticality.
- The reactivity swing induced by core burn-up amounts to about -19 pcm/EFPD starting from a fresh core
- At 5 mA beam intensity, the sub-critical core delivers a thermal power of 51.75 MW. An additional 1.43 MW is deposited by the proton beam mainly inside the liquid metal spallation target.
- The average linear power density over the hottest pin is 272 W/cm with the peak power limited to 352 W/cm.

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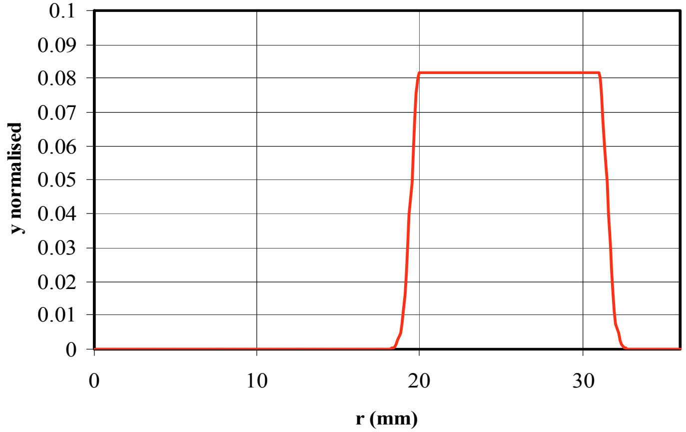
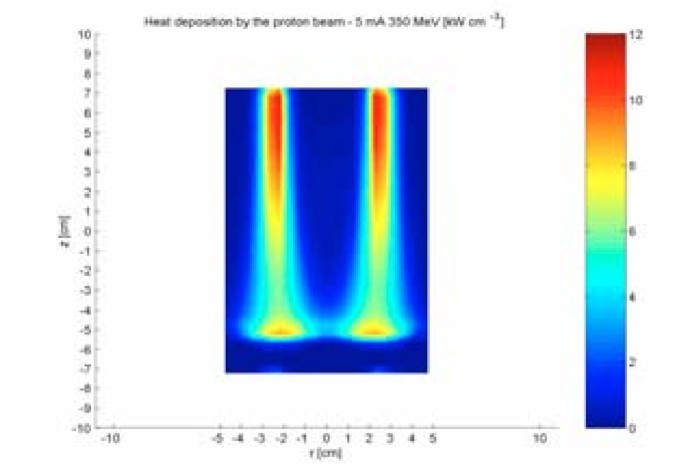
Concluding remarks

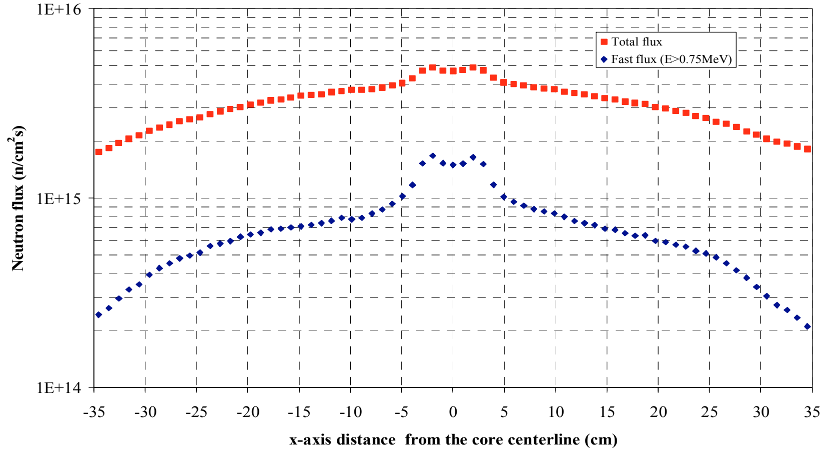
- The targeted order of magnitude in fast flux, viz. 10^{15} n/cm^{2}s, is achieved in the near of the hottest fuel pin.
- An accumulated dpa-damage dose up to 39 dpa over a 3x90 EFPDs irradiation period may be expected along the spallation target pipe with appm(He)-to-dpa ratios up to 8.
- MA transmutation has been investigated by considering six IMF-target assemblies, containing 7.24 kg of low graded plutonium, 9.04 kg of americium and 1.81 kg of curium, irradiated in fast spectrum channels during a 3-years campaign (810 EFPDs in total). The calculations yield a net decrease of 2.48 kg in the actinide mass, mainly due to the removal of americium (-2.46 kg). There is net mass increase of 0.46 kg for curium. The burned-out mass of plutonium is 0.51 kg

Effect of proton beam spatial shape





	Gaussian spatial beam profile	Real Beam Profile
k_S	0.9601	0.9597
M	25.04	24.82
n/p	6	6
P (MW)	51.75	50.9

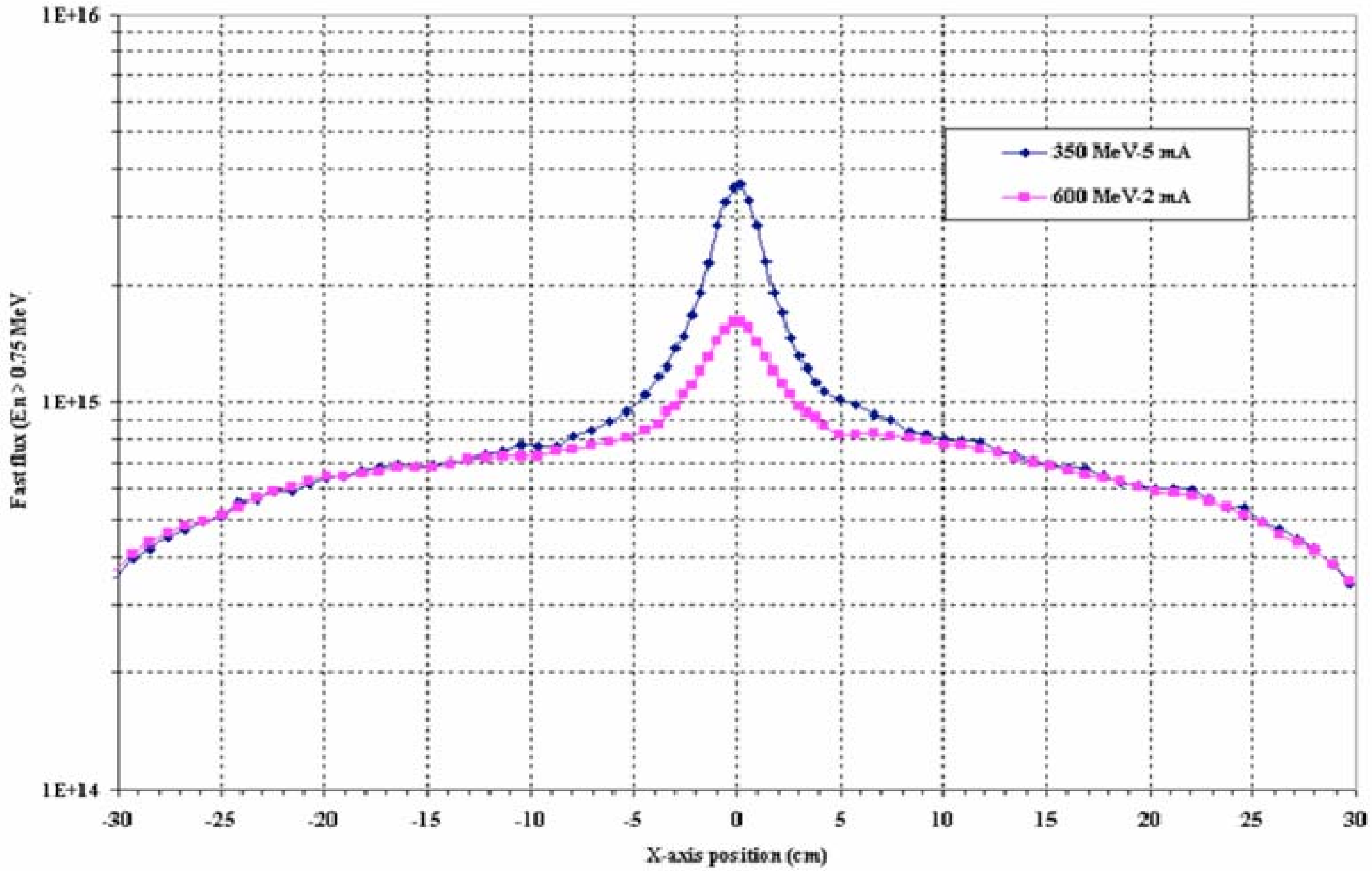


Proton beam option:
350 MeV-5 mA Vs. 600 MeV-2 mA

Neutronics Parameters	Units	MYRRHA 350 MeV	Vs 600 MeV
Proton beam energy	MeV	350	600
Accelerator current	mA	5	2
Proton beam energy	MW	1.75	1.20
Proton beam heating		1.43	0.74
Deposited fraction of beam energy	%	81.5	61.4
In-depth p-beam penetration (~Bragg peak)	mm	126	290
Free surface z-position	mm	+75	+150
Source neutron yield per incident proton	n/p	6.0	15.6
neutron source Intensity	10^{17} n/s	1.9	1.9
Initial fuel mixture	MOX	$(U\text{-}Pu)O_2$	$(U\text{-}Pu)O_2$
Initial (HM) fuel mass (m_{fuel})	Kg	514	514
Initial Pu-enrichment (Pu/HM)	wt%	30	30
K_{eff}		0.95521	0.95522
K_s		0.96007	0.95847
MF = 1 / (1 - K_s)		25.04	24.08
Source importance: φ*		1.127	1.082
Thermal Power (†) (P_{th})	MW	51.75	51.27
Specific power	kW/kgHM	101	100
Peak linear Power (hottest pin)	W/cm	352	324
Av. Linear Power (hottest pin)		272	268
Φ_{total} (at the hottest pin position)	10^{15} n/cm²s	4.04	3.86
$\Phi_{>1\,MeV}$ (at the hottest pin position)		0.74	0.64
$\Phi_{>0.75\,MeV}$ (at the hottest pin position)		0.98	0.85

(†) E_f = 210 MeV/fission

350 MeV-5 mA Vs. 600 MeV-2 mA: Fast flux



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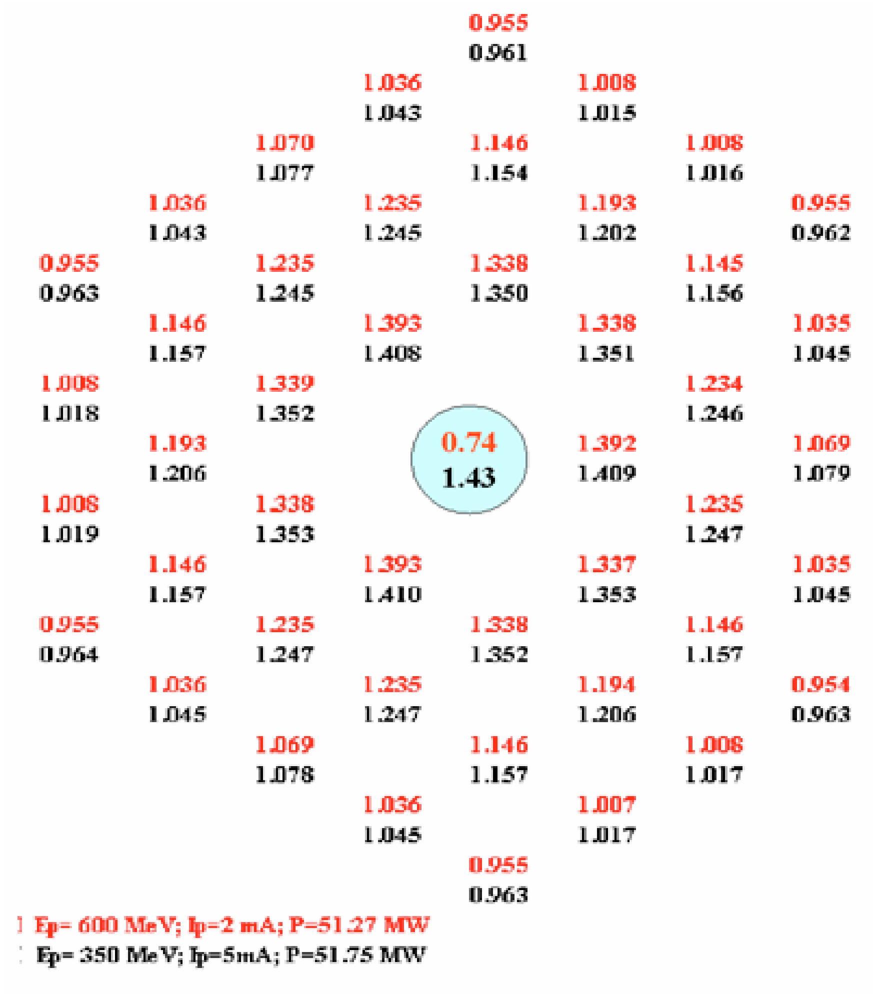
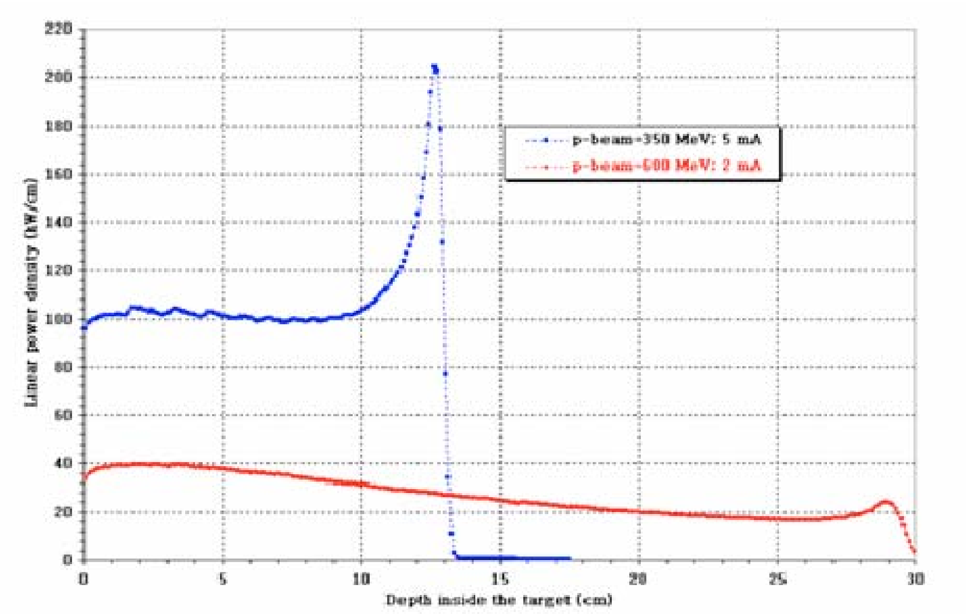
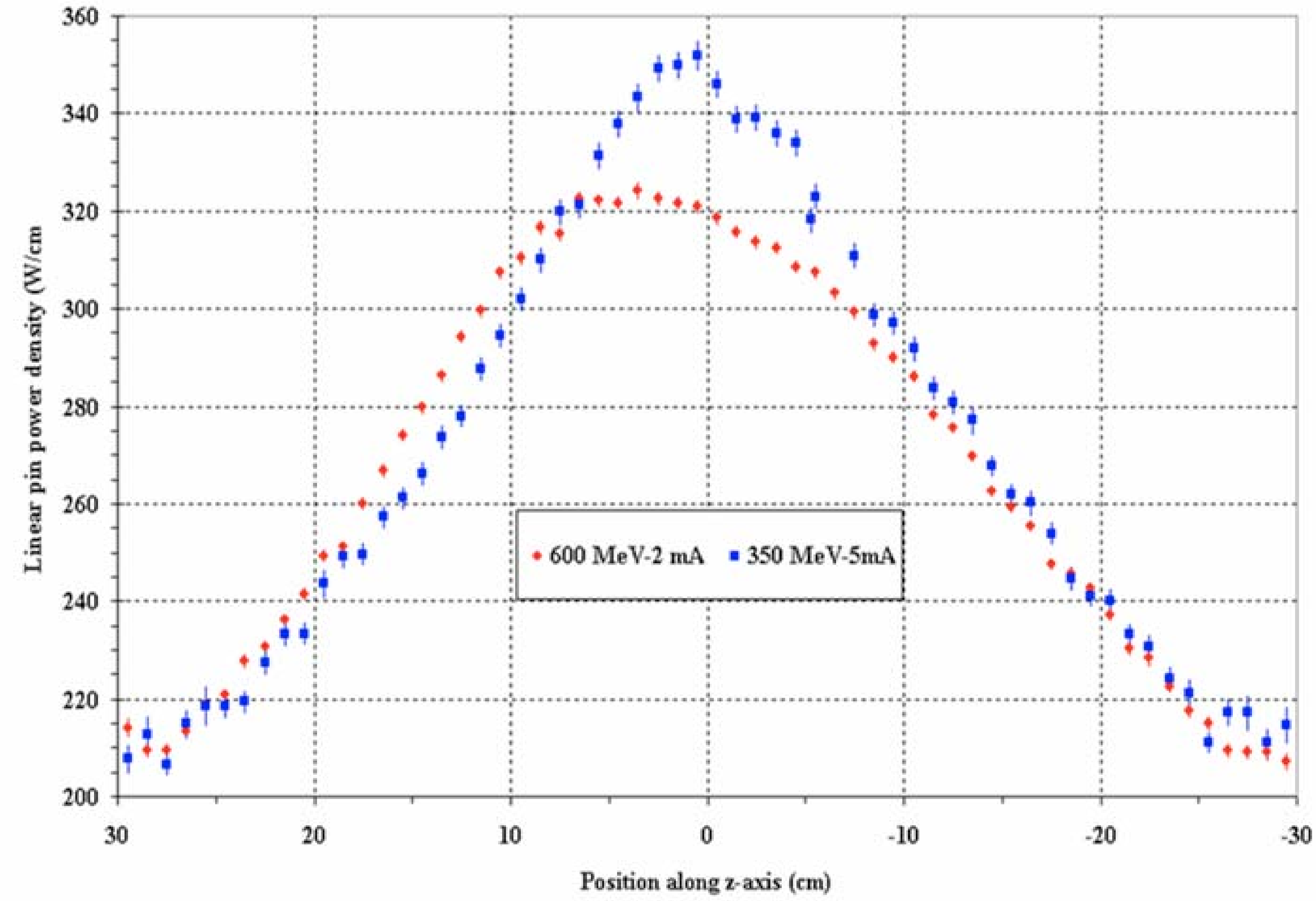
350 MeV-5 mA Vs. 600 MeV-2 mA: Fast flux





350 MeV-5 mA Vs. 600 MeV-2 mA: Linear power density along the hottest pin



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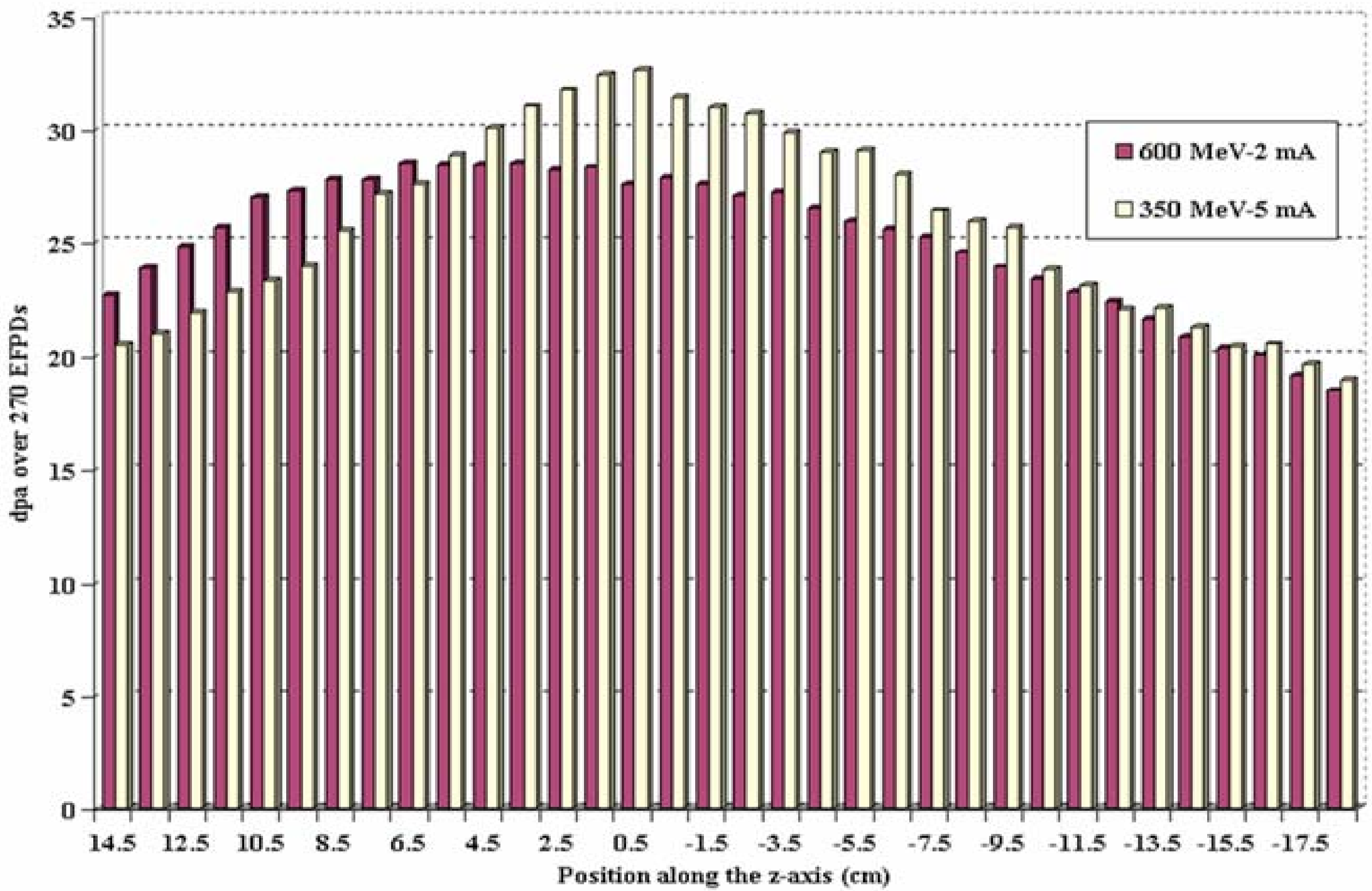
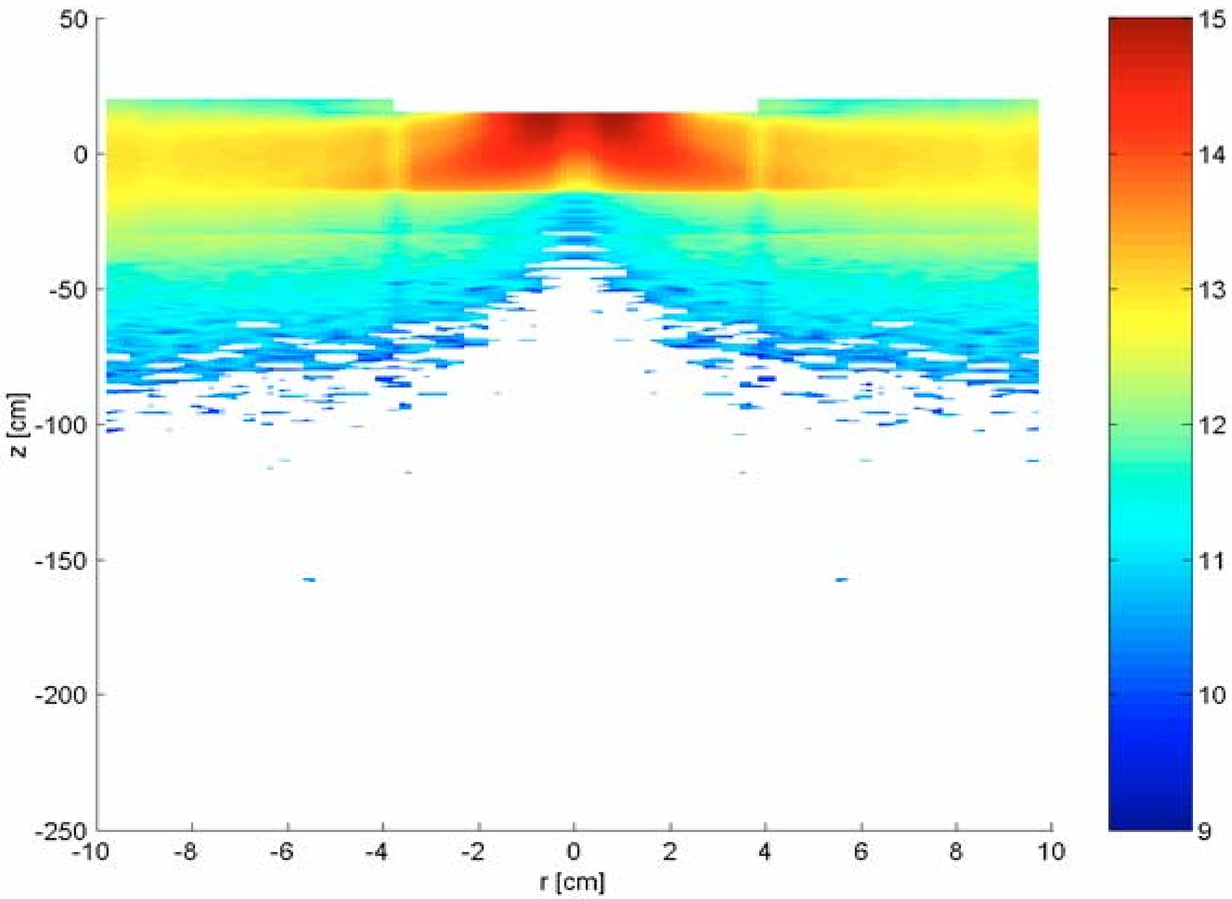
350 MeV-5 mA Vs. 600 MeV-2 mA: DPA/270EFPDs along the target duct

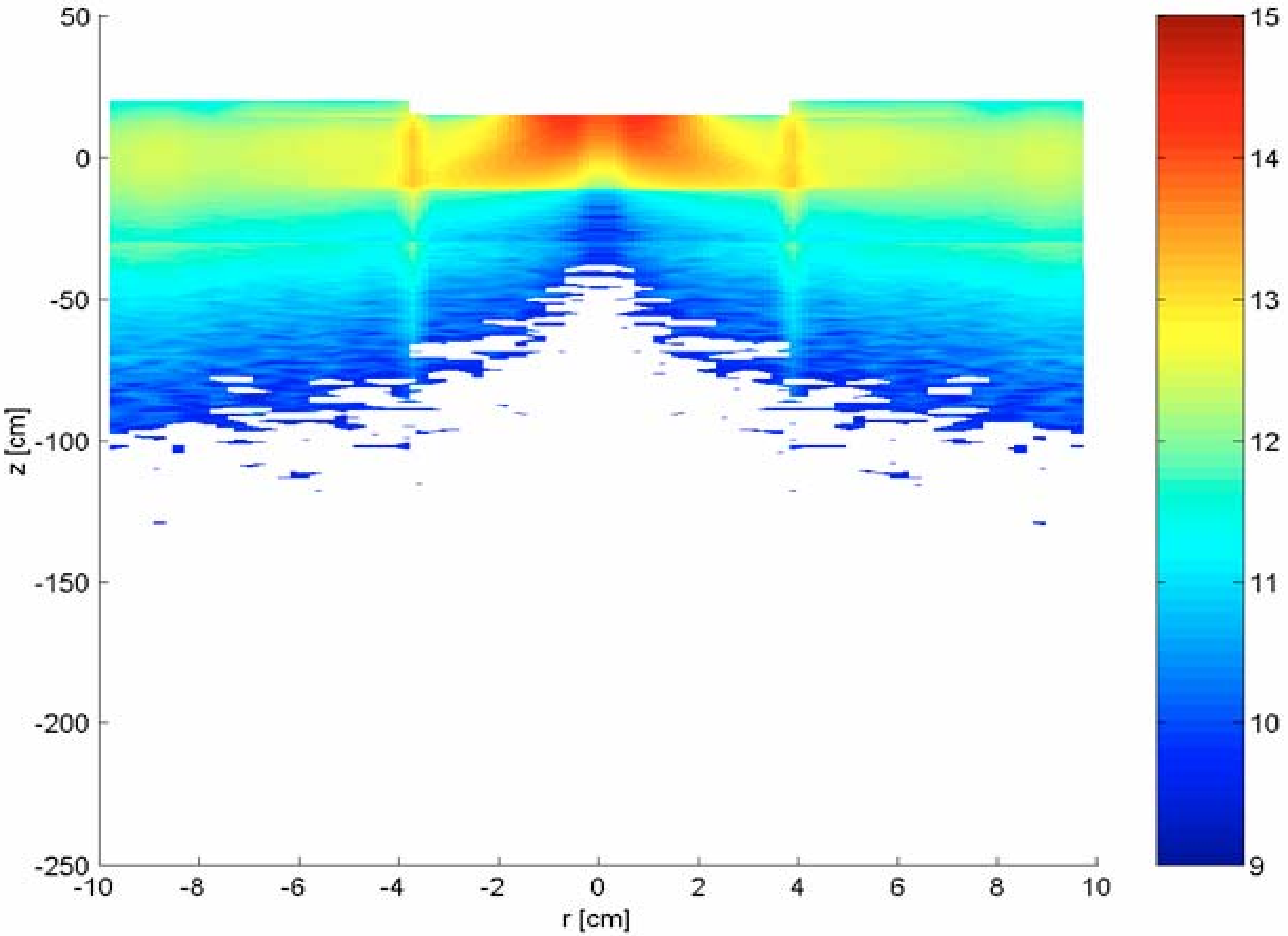
350 MeV-5 mA Vs. 600 MeV-2 mA: DPA/270EFPDs along the hottest pin clad



Non-fission (external) neutron source distribution



Proton particle distribution



Concluding remarks

- A 600 MeV-2 mA driving proton beam is shown to yield similar core characteristics as a 350 MeV-5mA proton beam
- In the case of 600 MeV the flux and hottest fuel pin power are less peaked and the peak dpa-damage is also lower.
- Moreover the target heating rate inside the liquid-metal spallation target is lower by a factor of two.
- Calculations show that neither neutrons nor protons will reach the bottom of the tank. No specific shielding is therefore required.

Still to carry out or to validate/set up the appropriate methodology

- Kinetic parameters
  - Neutron generation time (1.5 μs after B. Verboomen and W. Haeck)
  - Effective delayed neutron fraction
- Sensitivity studies with respect to nuclear data (covariance matrix) and physical properties (fuel density and composition) uncertainties
- Include other components (HX-loops, internal storage) in the geometrical model

MYRRHA – Draft 2
Fuel Pins & Fuel Assembly Pre-Design

Vitaly Sobolev, Hamid Aït Abderrahim

On behalf of MYRRHA team and MYRRHA support

http://www.sckcen.be/myrrha

CONTENTS

1. General approach to fuel design (1)

Needed input information:

- core (spectrum, total power, power density or neutron flux);
- fuel type (oxide, metal, cermet, …);
- initial fuel enrichment, composition and density;
- aimed fuel burn-up;
- coolant type (liquid metal, gas, …);
- allowed coolant temperature and flow velocity;
- cladding material;
- allowed cladding temperature, corrosion; stresses and strains.

1. General approach to fuel design (2)

Core parameters choice :

- Neutron spectrum -> fast
- k_{eff} -> ~ 0.95
- Total power -> ~ 50 MW(th)
- Fast neutron flux -> ~ 10^{15} cm^{-2} s^{-1}

(fuel power density -> ~ 1.5 kW cm^{-3})

1. General approach to fuel design (3)

Fuel choice:

- Fuel type -> oxide -> MOX;
- Composition and density -> $(Pu,U)O_2$ of 95 % TD
- Initial enrichment -> 20-30 % Pu (PWR) in HM
- Aimed (peak) burn-up -> ~ 100 MWd/kg iHM
- Maximum allowed temperature -> 0.9 T_m ~ 2100-2400 °C

1. General approach to fuel design (4)

Fuel choice:

- Pu isotopic vector:

Isotope	Content, wt.%
^{238}Pu	1.27
^{239}Pu	**61.88**
^{240}Pu	23.50
^{241}Pu	**8.95**
^{242}Pu	4.40

… however, the MOX is in disagreement with RERTR program ?

1. General approach to fuel design (5)

Coolant parameters choice:

- coolant type -> LBE (T_m=124 °C)
- allowed temperatures -> from 200 °C up to 450 °C
- allowed flow velocity -> 2 m s^{-1}

… however, the lower temperature limit should (may be) increased because of the clad embrittlement problems ..

1. General approach to fuel design (6)

Clad material requirements:

1. Keeping the adequate mechanical performances (strength, ductility, swelling, creep) at high doses and operation temperatures.

2. Resistance to corrosion-erosion attack of LBE flow

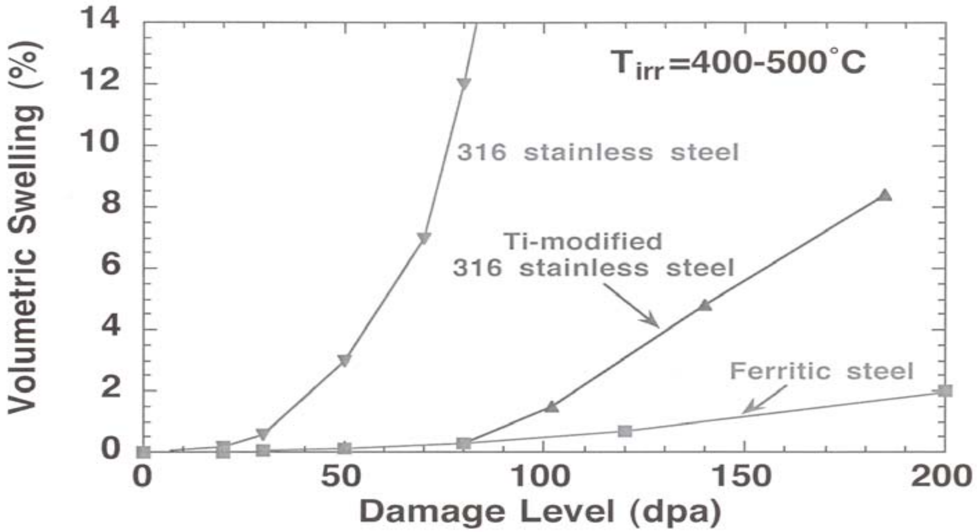
3. Resistance to cycling stresses caused by the trips and restarts of the proton beam.

→ *ferrite-martensitic or austenitic steels ?*

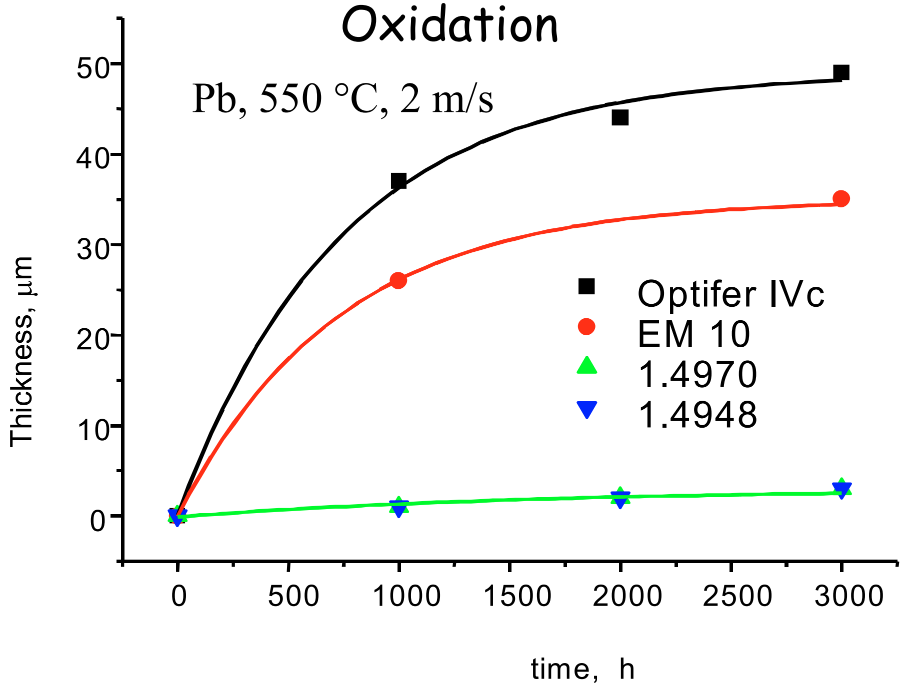
1. General approach to fuel design (6)

Comparison of austenitic and ferrite-martensitic steels

Swelling

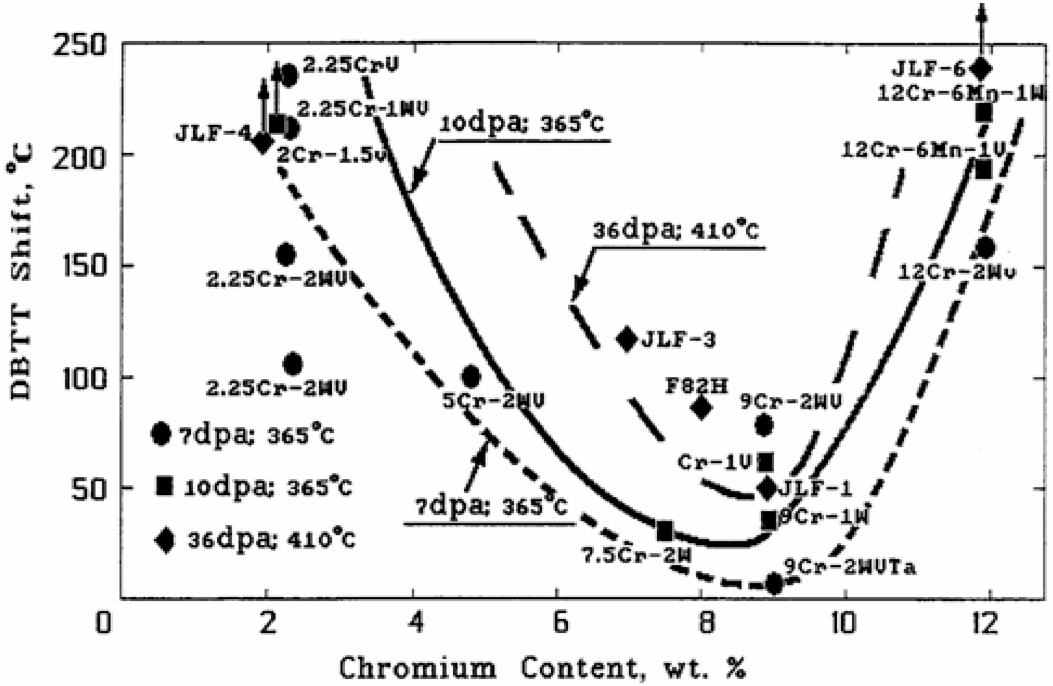


Oxidation



1. General approach to fuel design (7)

Embrittlement of Cr-steels



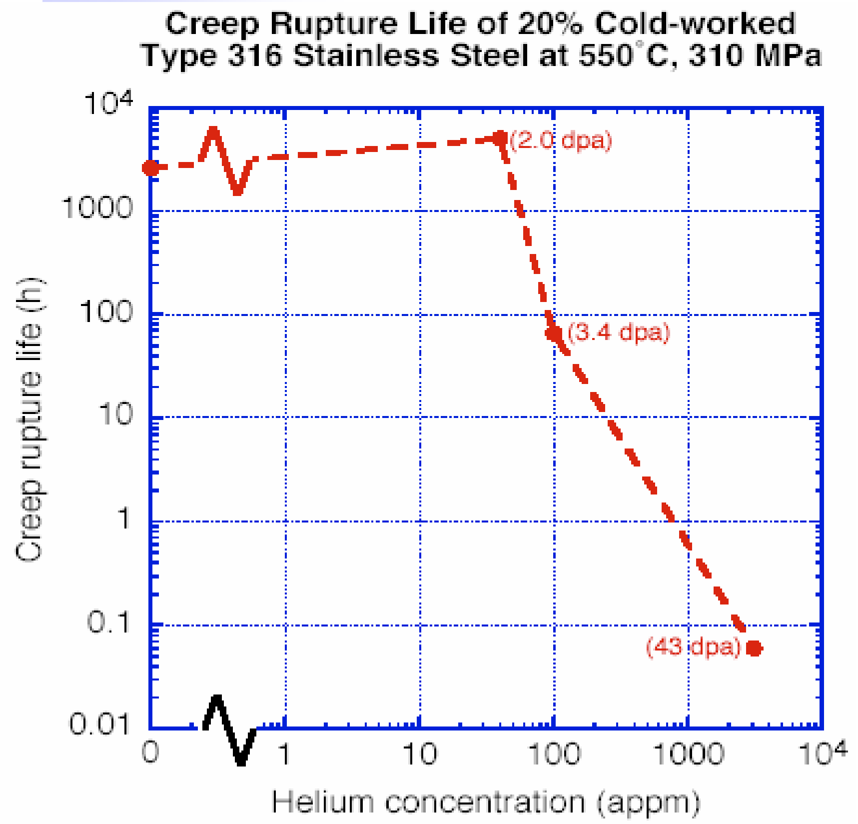
1. General approach to fuel design (8)

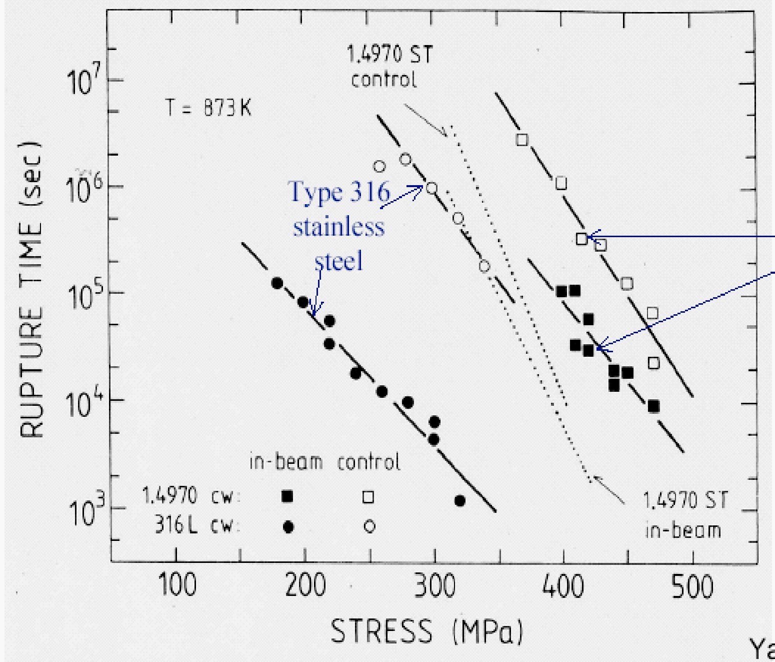
Cladding parameters choice:

- Material -> T91 MS (oxygen protection)
- Allowed temperature -> 500 °C (normal operation)
  600 °C (transients)
- Allowed radiation damage -> ~100 dpa
- Allowed swelling -> ~ 5 %
- Allowed corrosion -> ~ 10 %

SS 316 Ti (corrosion protected) is still kept as back-up solution.
…however, helium induced embrittlement can be a problem…

1. General approach to fuel design (8a)





Yamamoto & Schroeder
J.Nucl. Mat. 155-157 (1988)

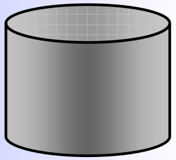
SS 316 Ti (corrosion protected) is still kept as back-up solution.
…however, helium induced embrittlement can be a problem…

1. General approach to fuel design (9)

Main steps in the fuel rod pre-design:

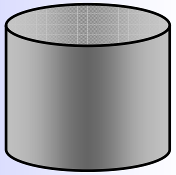
- Fuel pellet sizes
- Clad diameter and thickness
- Fuel column and gas plenum
- Preliminary design of a whole rod
- Design test with fuel performance codes
- Design optimisation

2. Determination of fuel pellet sizes (1)

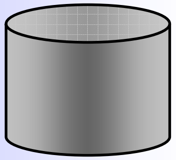


- Pellets without hole to simplify fabrication.
- Pellet diameter to satisfy the fuel non-melting conditions:

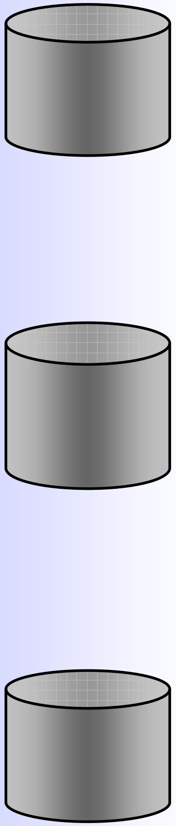
$$T_{melt} < T_{fuel\,\max} = T_{cool} + \frac{\pi \cdot d^2_{pellet} \cdot q_{v\max}}{4} \cdot \left(\Re_{cool} + \Re_{clad} + \Re_{gap} + \Re_{pellet} \right)$$



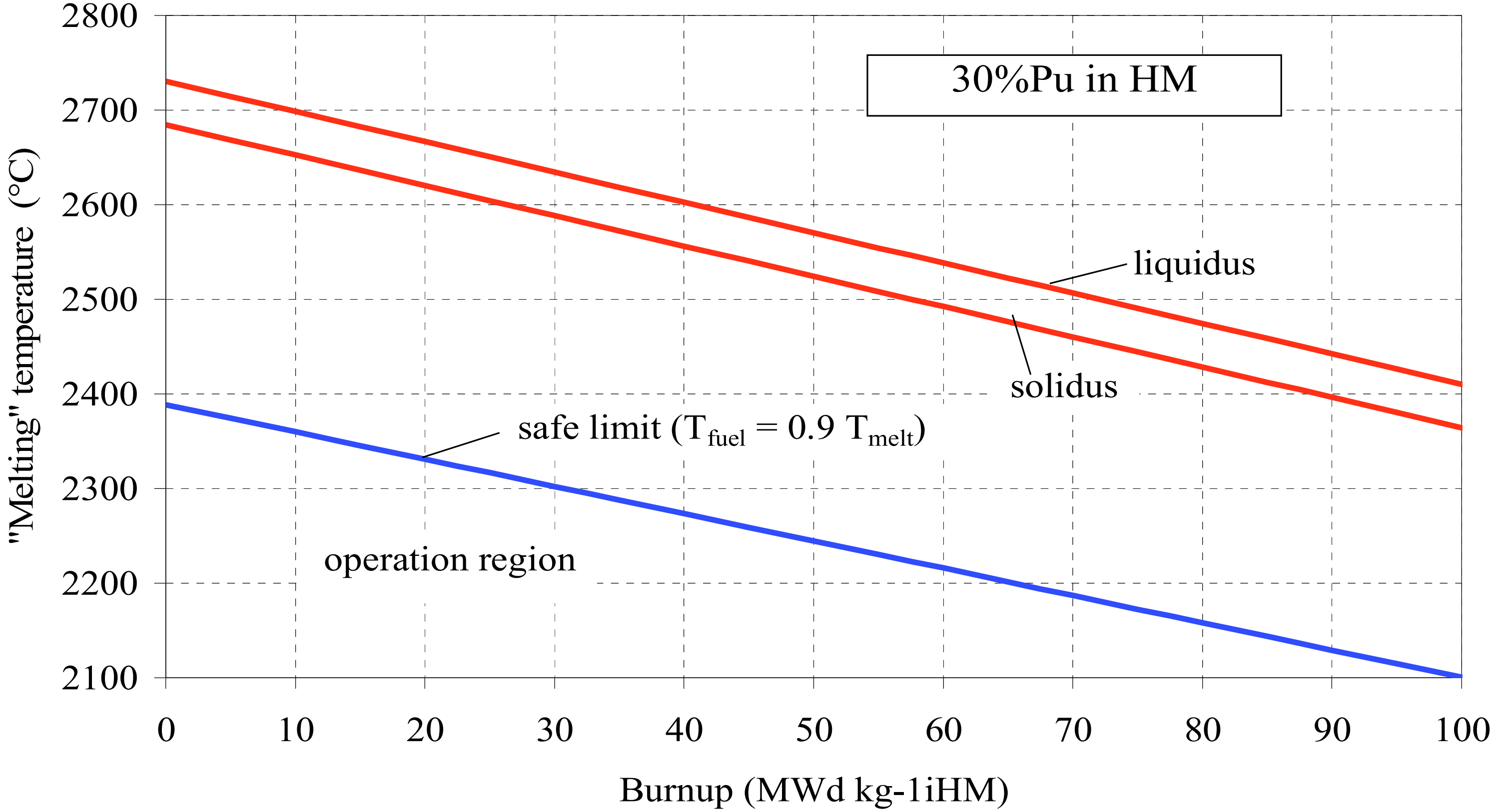
- $q_{v\,max} \sim 1.5\ kW/cm^3$ to obtain $\Phi_{fast} \sim 10^{15}\ cm^{-2}\ s^{-1}$
- Safety margin $T_{fuel\,max} = 0.9\ T_{melt}$



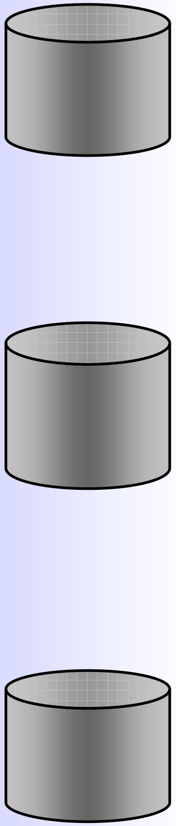
2. Determination of fuel pellet sizes (2)



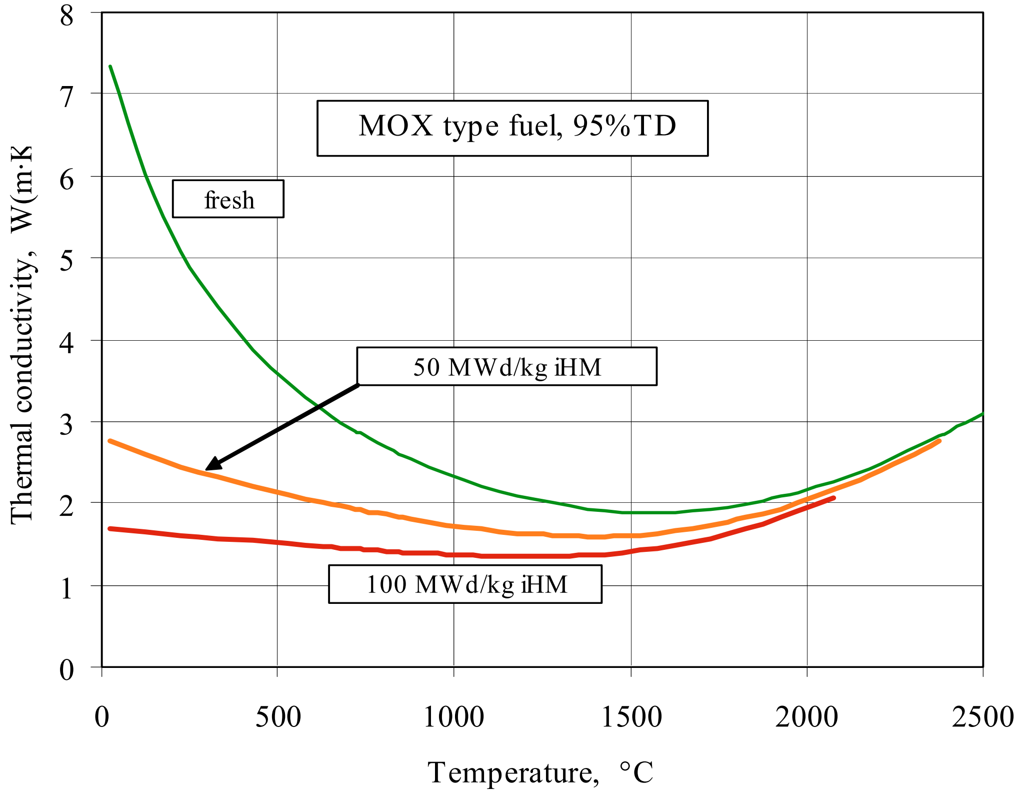
Safety margins for fuel temperature



2. Determination of fuel pellet sizes (3)



Fuel thermal conductivity degradation

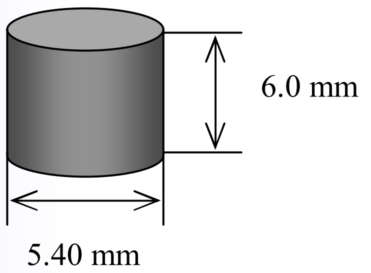


2. Determination of fuel pellet sizes (4)

Radial thermal resistivity of fuel rod

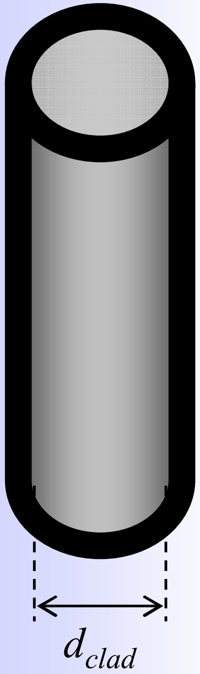
$$T_{melt} < T_{fuel\,\max} = T_{cool} + \frac{\pi \cdot d^2_{pellet} \cdot q_{v\max}}{4} \cdot \left(\Re_{cool} + \Re_{clad} + \Re_{gap} + \Re_{pellet} \right)$$

Pellet type	***Time***	$\Re_{cool}$	$\Re_{clad}$	$\Re_{gap}$	$\Re_{pellet}$	$q_{v\ max}$	T_{max}	T_{cool}	$D_{pellet\ max}$
		K·m/kW	K·m/kW	K·m/kW	K·m/kW	kW/cm³	°C	°C	mm
Solid	*BOL*	2.38	1.13	22.4	35.0	1.5	2390	300	5.40
Solid	*EOL*	2.38	1.18	0.25	53.0	1.2	2100	300	5.80



- *The chosen pellet : Ø 5.40 x 6.0 mm (q_v = 1.5 W/cm^{-3} ~ q_l = 350 W cm^{-1}).*
- *…however, it would be better to use the pellets with the same sizes as in the developed LMFR (SNR, Phenix, …)*

3. Cladding sizes

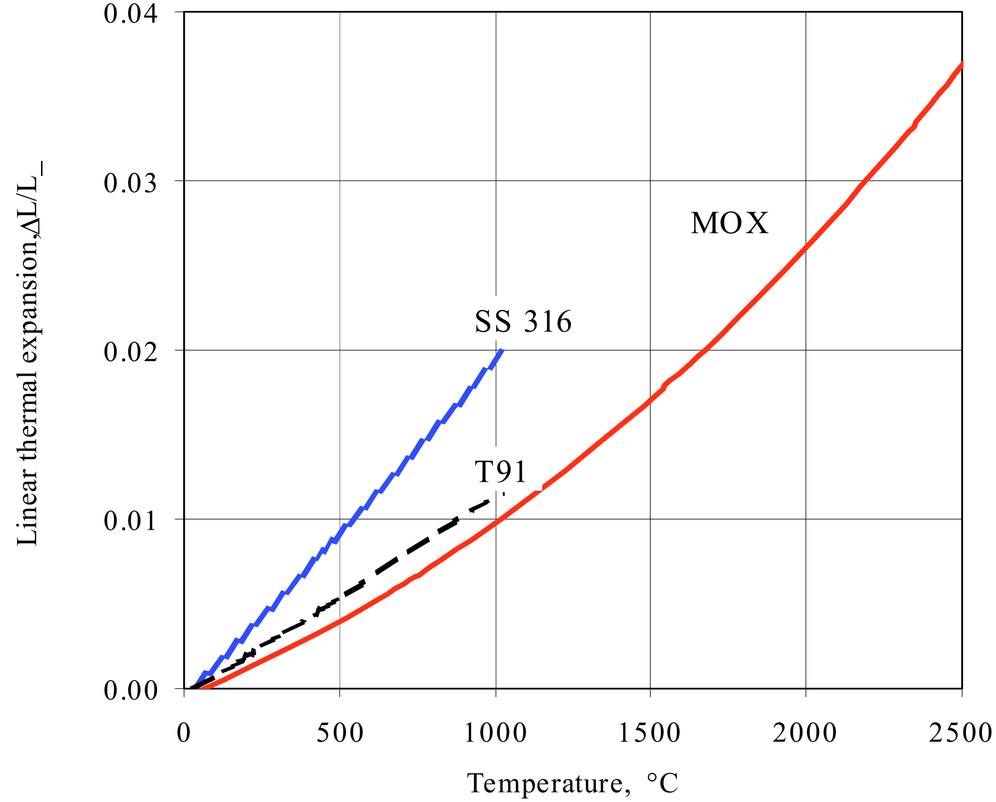


- Clad inner diameter:
  - Gap is to avoid or reduce PCMI
  - Gap thickness δ_{gap} should compensate:
    - fuel thermal expansion
    - fuel irradiation induced swelling (~1.6vol.% per 10 MWd kg^{-1}iHM)
  - Inner clad diameter: $d_{clad} = D_{pellet} + \delta_{gap}$

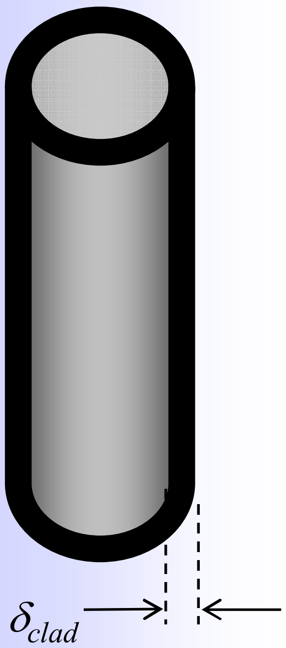
δ_{gap} (radial)= 75 microns and d_{clad} = 5.55 mm have been obtained as the first estimate.

3. Cladding sizes

Liner thermal dilatation of MOX, SS 316 and FMS T91



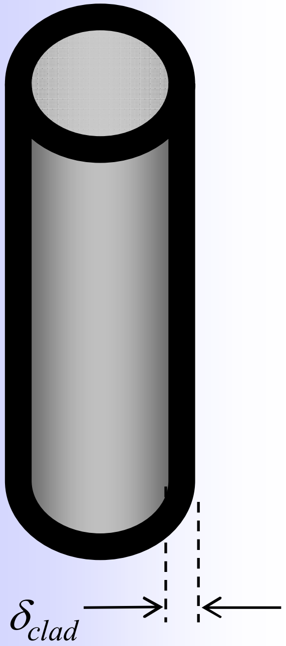
3. Cladding sizes

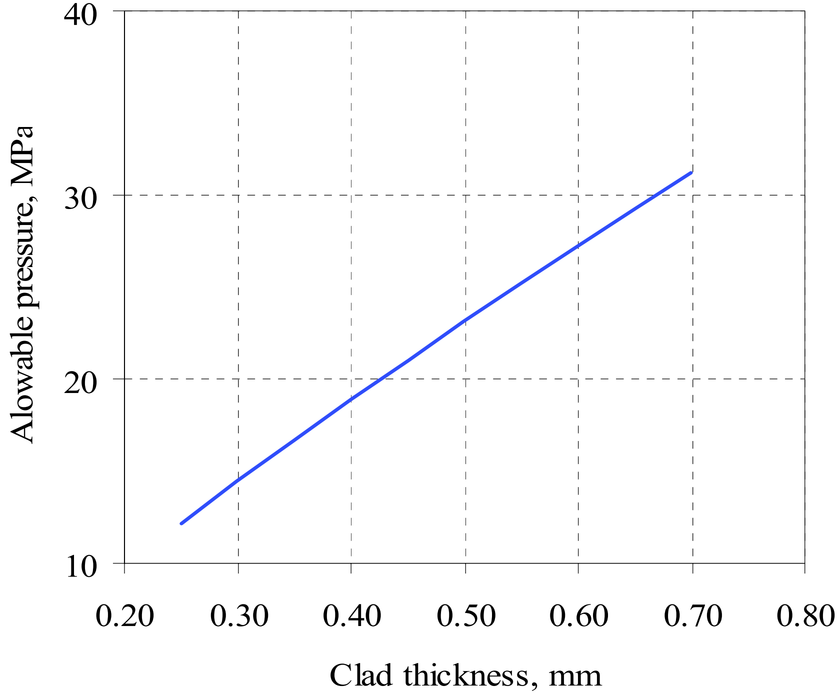


Clad thickness is chosen to withstand:

- intrinsic thermal expansion stresses
- pressure of inside gazes
- pressure of outside coolant
- inside corrosion attack of fission products
- outside corrosion attack of LBE coolant
- fatigue initiated by power changes caused by the proton beam trips and restarts.
- PCMI -> for ASS < (0.4-1)% plastic deformation
  for FMS …?

3. Cladding sizes



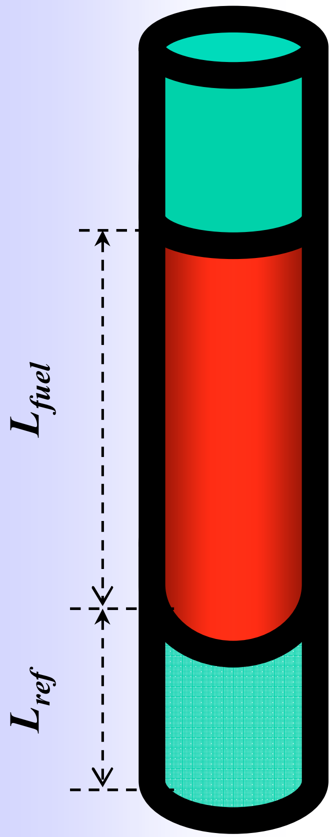


$$p_{\max} = \frac{[\sigma] \cdot \delta_{clad}}{0.5 \cdot D_{clad\ inside} + 0.6 \cdot \delta_{clad}}$$

$$[\sigma] = \min\{\sigma_{uts}/2.6;\ \sigma_{0.2}/1.5\}$$

δ_{clad}= 0.5 mm obtained as the first estimate (p_{max} = 23 MPa).
...however cladding sizes should still be optimised after determination of T91 properties at representative irradiation conditions.

4. Whole rod pre-design (1)



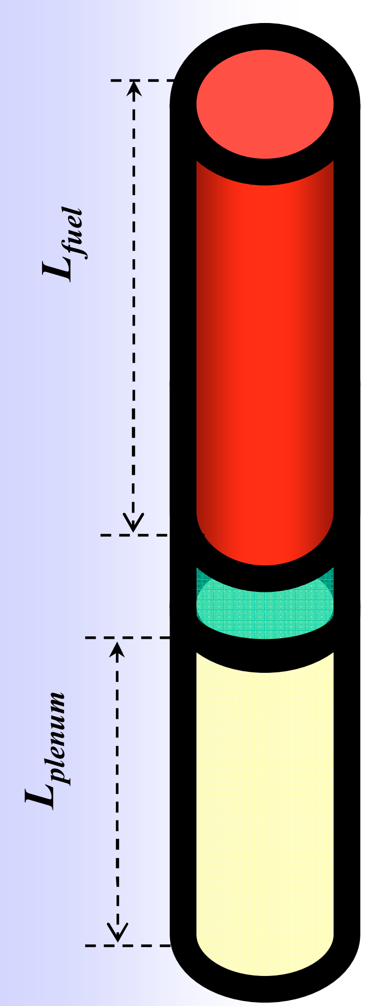
Fuel column length

- Compact core -> $L_{active\ zone} \sim D_{active\ zone}$
- Limited axial form factor -> 1.2-1.3
- Neutronic estimates -> $L_{fuel} \approx 600$ mm

Reflector segments

- Neutronic estimates - > I_{ref} = 50-100 mm
- Material -> YSZ

4. Whole rod pre-design (2)



Gas plenum volume is determined:

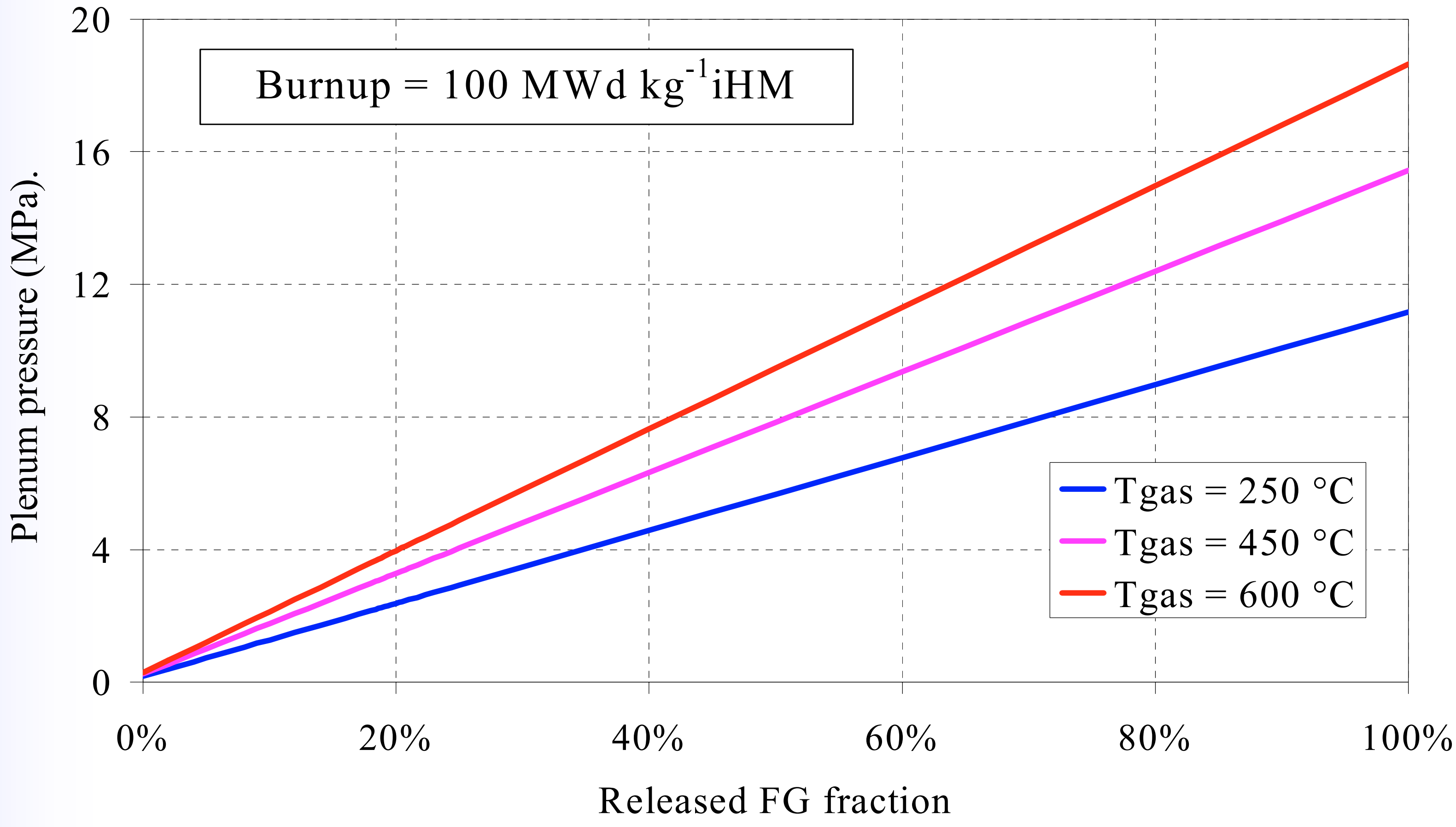
- by the released amount of fission gas (production rate ~ 115 mole m^{-3} per 10 MWd kg^{-1}iHM)
- by the gas temperature in plenum
- by the cladding mechanical resistance

$$p_{tot} = \frac{p_{0\,He} \cdot T_{gas}}{T_0} + \frac{R \cdot \eta_{FG} \cdot Bu \cdot \rho_{fuel} \cdot V_{fuel} \cdot T_{gas}}{\rho_{TD\ fuel} \cdot V_{plenum}} < p_{\max}$$

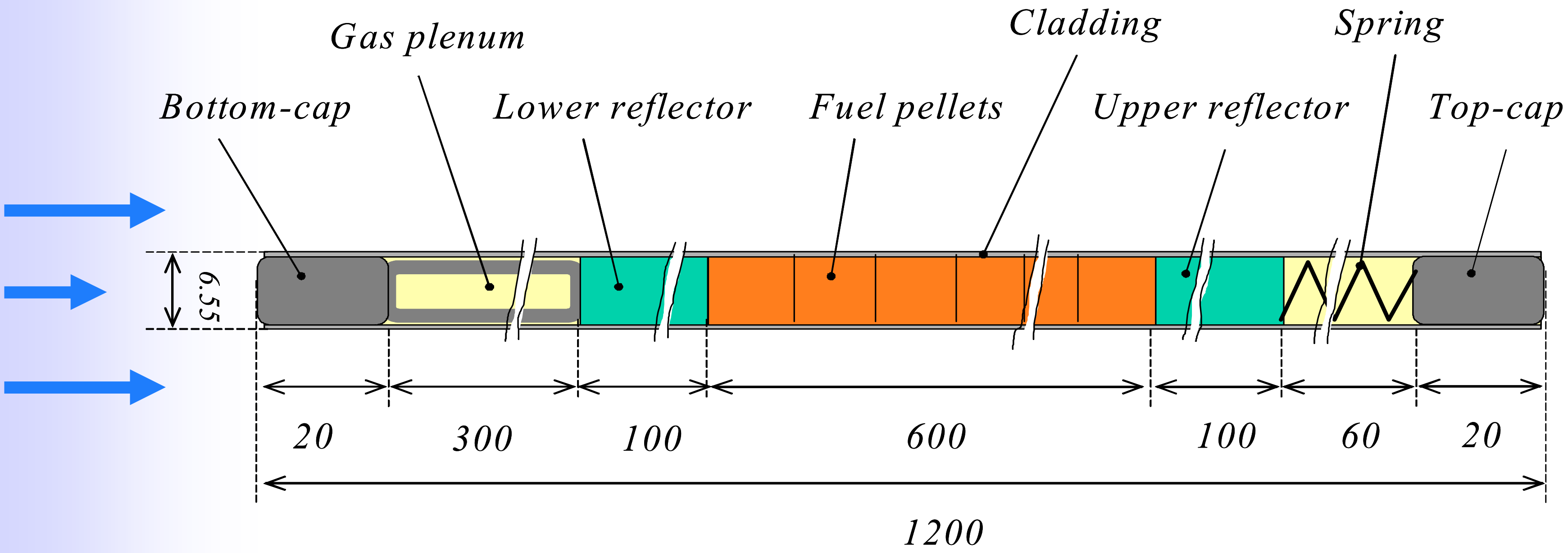
Temperature of gas is a critical parameter which is difficult to determine, especially, in the case of a rapid (burst) gas release.

L_{plenum} = 60+300 = 360 mm was chosen

4. Whole rod pre-design (3)



4.Whole rod pre-design (4)
Axial schematics



A typical design of LMFR rod has been adapted to the MYRRHA specific conditions.

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4. Whole rod pre-design (5)

MYRRHA

Table 5. Main geometrical parameters (*in mm*) of the fuel pins of some fast neutron reactors and of ADS MYRRHA.

	SPX	Phenix	SNR-300	BN-600*	EFR	**MYRRHA**
Diameter	8.50	6.55	6.00	6.90	8.65	6.55
Total length	2700	1793	2475	1100	3600	1200
upper gas-plenum	162	93	50	20	1700	60
upper breeder/reflector	300	(0)**	400	50	250	100
active part	850	900	950	500	1400	600
lower breeder/reflector	300	(300)**	400	50	150	100
lower gas-plenum	852	442	650	421	545	300

* experimental fuel rod with the holed pellets;
** special design.

5. Pre-design of a fuel assembly (1)

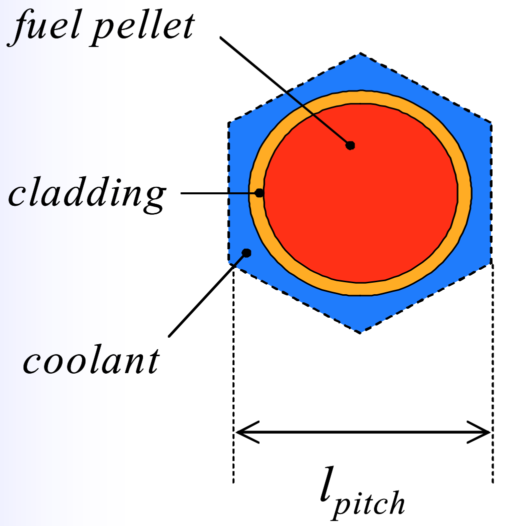
Main steps in the fuel assembly design:

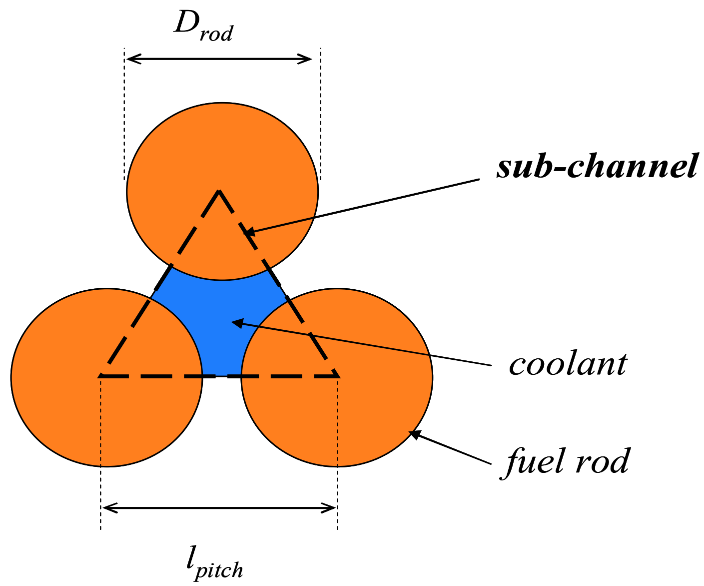
1) Fuel micro-cell (type and pitch)
2) Assembly radial cross-section
3) Assembly axial schematics
4) Preliminary design of a whole assembly
5) Modelling with suitable thermohydraulic and thermomechanical codes
6) Optimisation

5. Pre-design of a fuel assembly (2)

Microcell = fuel rod + coolant + spacer

Microcell type → hexagon or triangle





Pitch (l_{pitch}) ? -> heat balance + pressure drop + fuel fraction

5. Pre-design of a fuel assembly (3)

Heat balance → pitch = f (Q_{rod}, ΔT < 200°C, v_{cool} < 2 m/s)

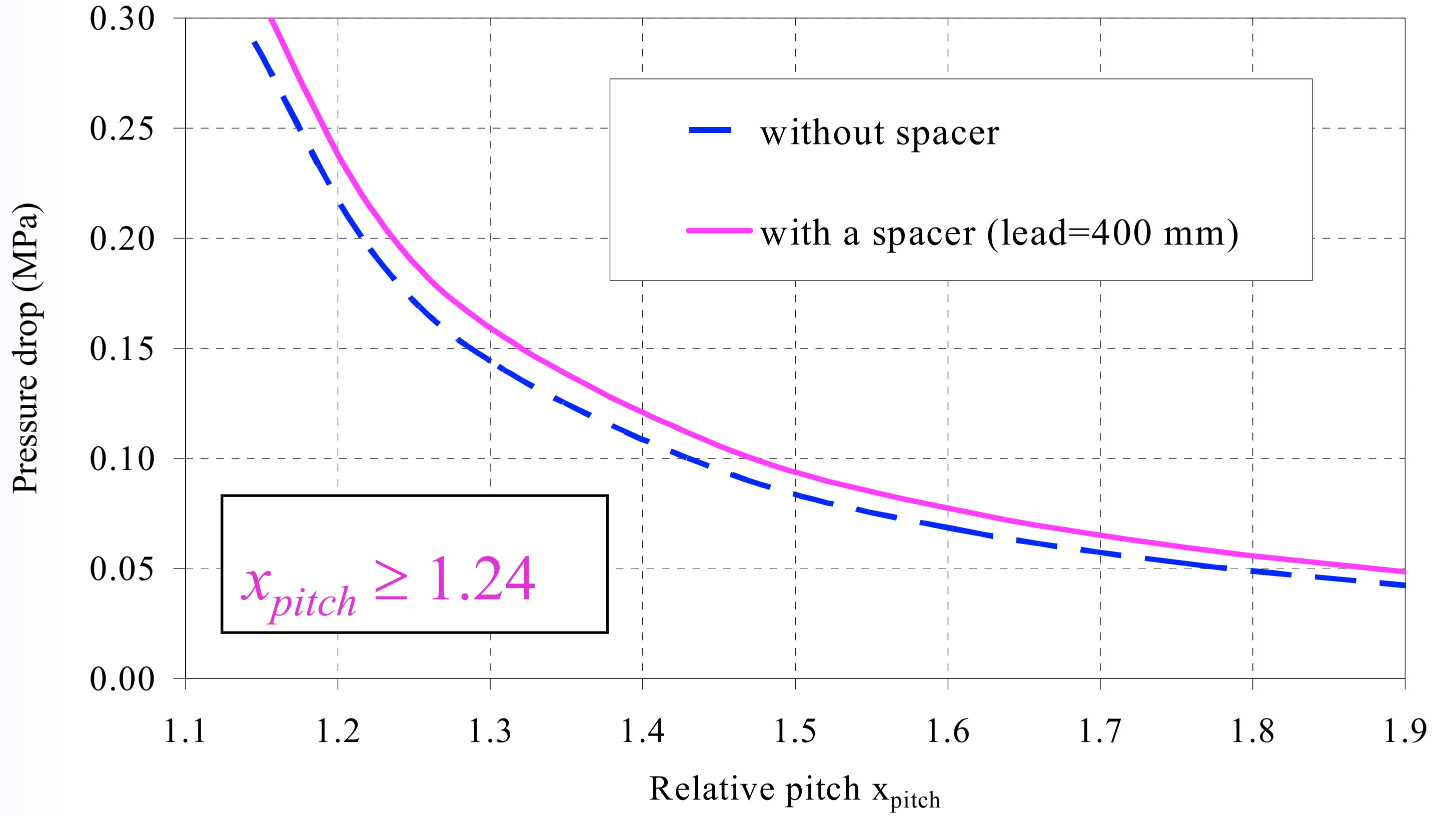
$$Q_{sch} = \langle \rho_{cool} \cdot c_{p\,cool} \rangle \cdot \Delta T_{cool} \cdot \mathrm{v}_{cool} \cdot S_{cool}$$

$$x_{pitch} \equiv \frac{l_{pitch}}{D_{clad}} \geq \sqrt{\frac{\pi}{2\sqrt{3}} \cdot \left(1 + \frac{4 \cdot \langle q_{l\,rod} \rangle \cdot l_{fuel}}{\pi \cdot D_{clad}^2 \cdot \rho_{cool} \cdot \mathrm{v}_{cool} \cdot \langle c_{p\,cool} \rangle \cdot \Delta T_{cool}} \right)}$$

$x_{pitch} \geq 1.224$ -> $l_{pitch} \geq 8.02$ mm

5. Pre-design of a fuel assembly (4)

Pressure drop → pitch = f (Δp < 2 bar, v_{cool} < 2 m/s)



5. Pre-design of a fuel assembly (5)

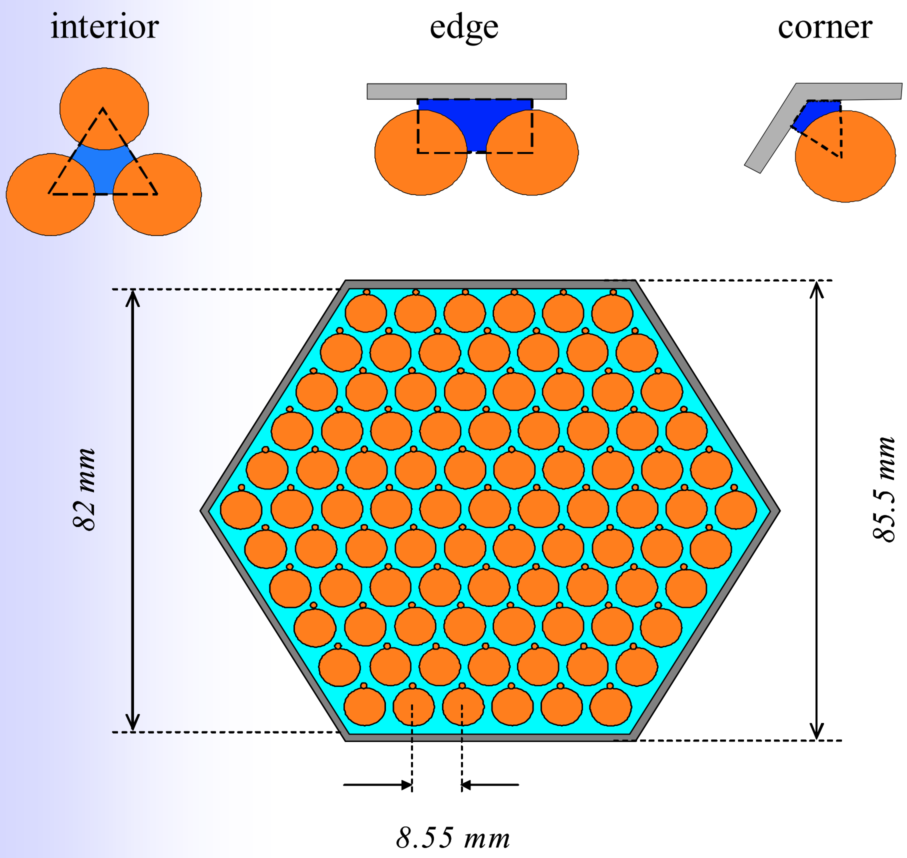
Fuel fraction →

pitch = f (minimum coolant fraction to obtain k_{eff} ~ 0.95)

$x_{pitch} = 1.305$ -> $l_{pitch} = 8.55$ mm was chosen at this stage of the pre-design.

…however, a large value is preferable for natural circulation build-up in the case of a pump trip.

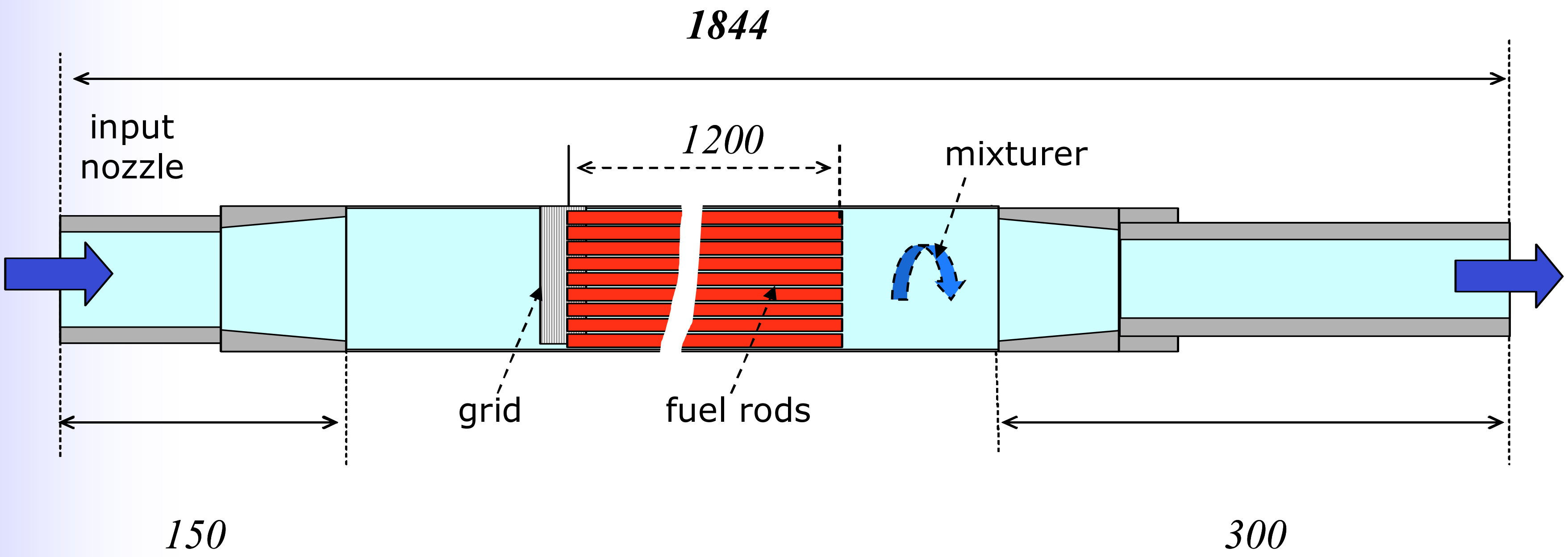
5. Pre-design of a fuel assembly (6)



Radial cross-section design:

1. Edge and corner sub-channels optimisation
2. Determination of a number of rods in FA →
   Radial gradient limits.
4. Shroud thickness → "bowing", "deflection" ?
5. Bundle grids and other elements.
6. Thermohydraulic and thermomechanical modelling
7. Optimisation

5. Pre-design of a fuel assembly (7)



A typical design of LMFR sub-assembly has been adapted to the MYRRHA specific conditions.

...however the estimates have been performed only at start conditions.

5. Pre-design of a fuel assembly (8)

Main geometrical parameters (*in mm*) of the hexagonal sub-assemblies of some LMFR and of ADS MYRRHA.

	SPX	Phenix	SNR-300	BN-600	EFR(II)	**MYRRHA**
Number of pins	271	217	166+3	127	331	91
Pin diameter	8.50	6.55	6.00	6.9	8.2	6.55
SA Width	173	124	110.25	96	183	85.5
Total length	5400	4300	3700	3500	5300	1844
Fuel pin length	2700	1793	2475	2400	3600	1200
SA Pitch	179	127	115	98	188	87

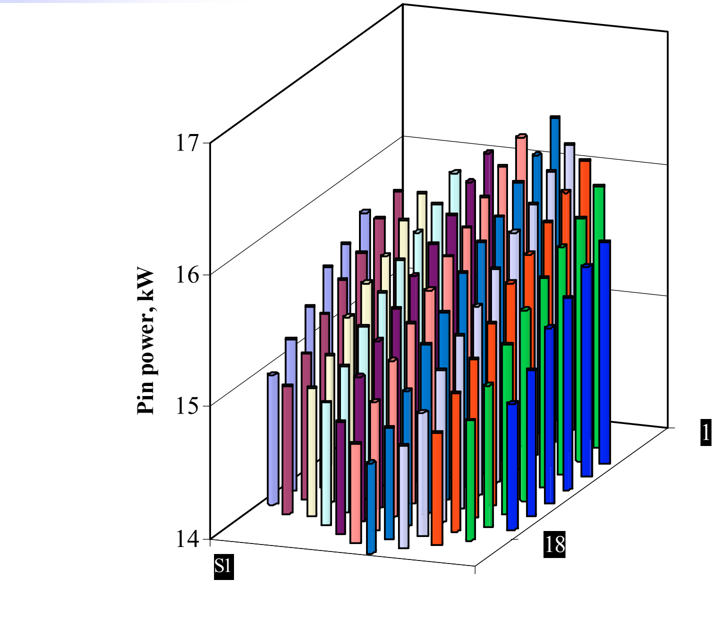
6. Preliminary estimations of fuel operation parameters (1)

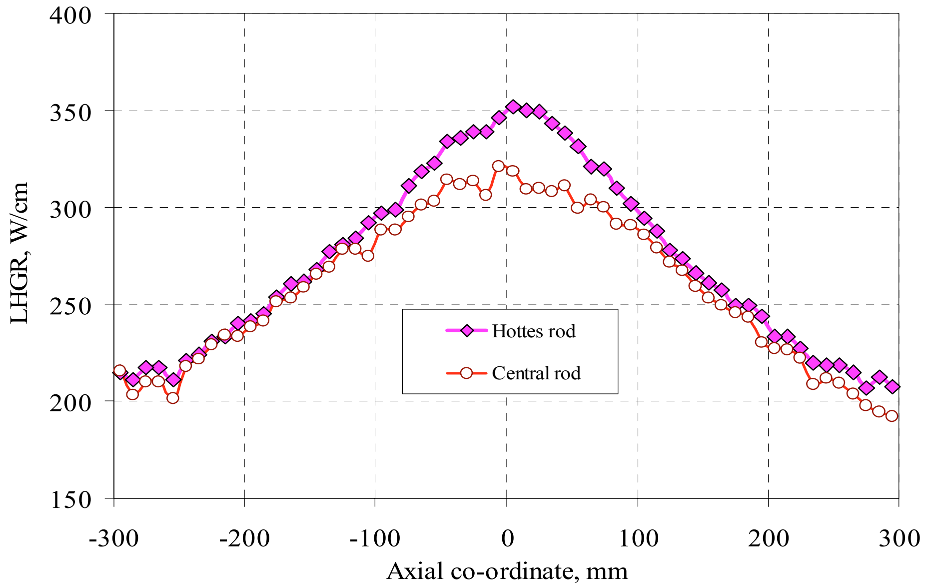
Input from neutronic modelling: Power and flux in the hottest rod

Neutron flux (near the hottest rod): total En > 0.75 MeV En > 1 MeV	10^{15} n/cm^{2}s	4.0 1.0 0.8
Core thermal power	MW	51.8
Peak power density (fuel)	kW/cm^3	1.54
Average power density (fuel)	kW/cm^3	0.937
Radial power form-factor	(max/aver rod)	1.29
Peak liner power (hottest rod)	W/cm	352
Average liner power (hottest rod)	W/cm	272
Axial power form-factor (hottest rod)	(max/aver)	1.30

6. Preliminary estimations of fuel operation parameters (2)

Input from neutronic modelling: Power distribution in the hottest assembly and in the hottest rod





6. Preliminary estimations of fuel operation parameters (3)

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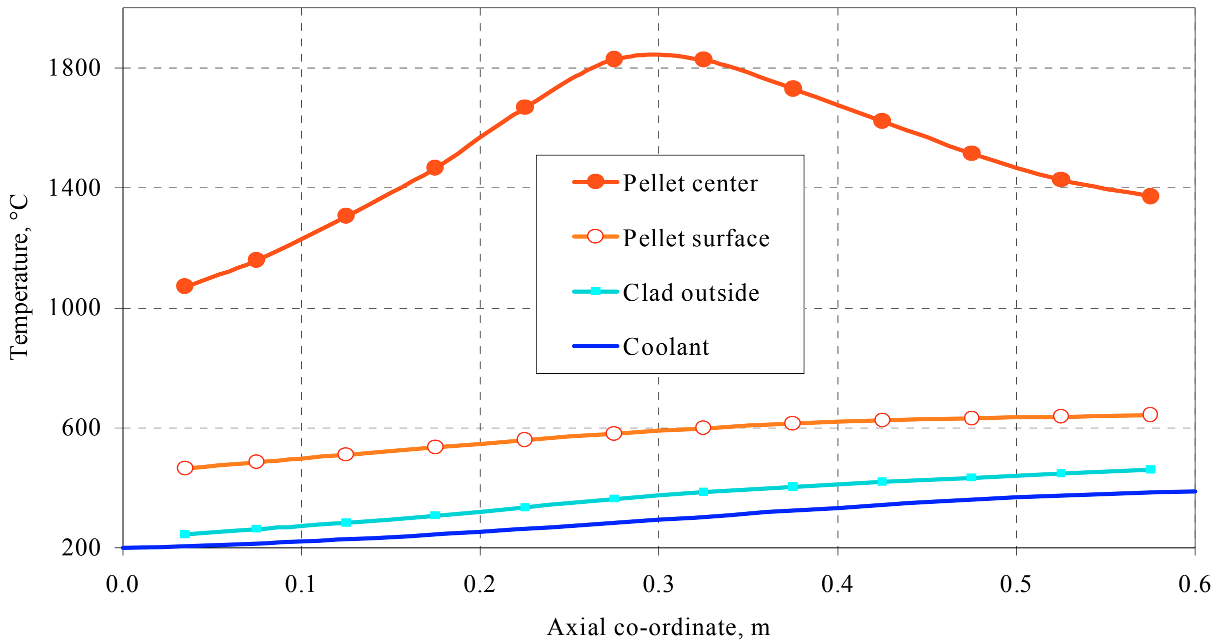
Initial axial temperature distribution in the hottest rod

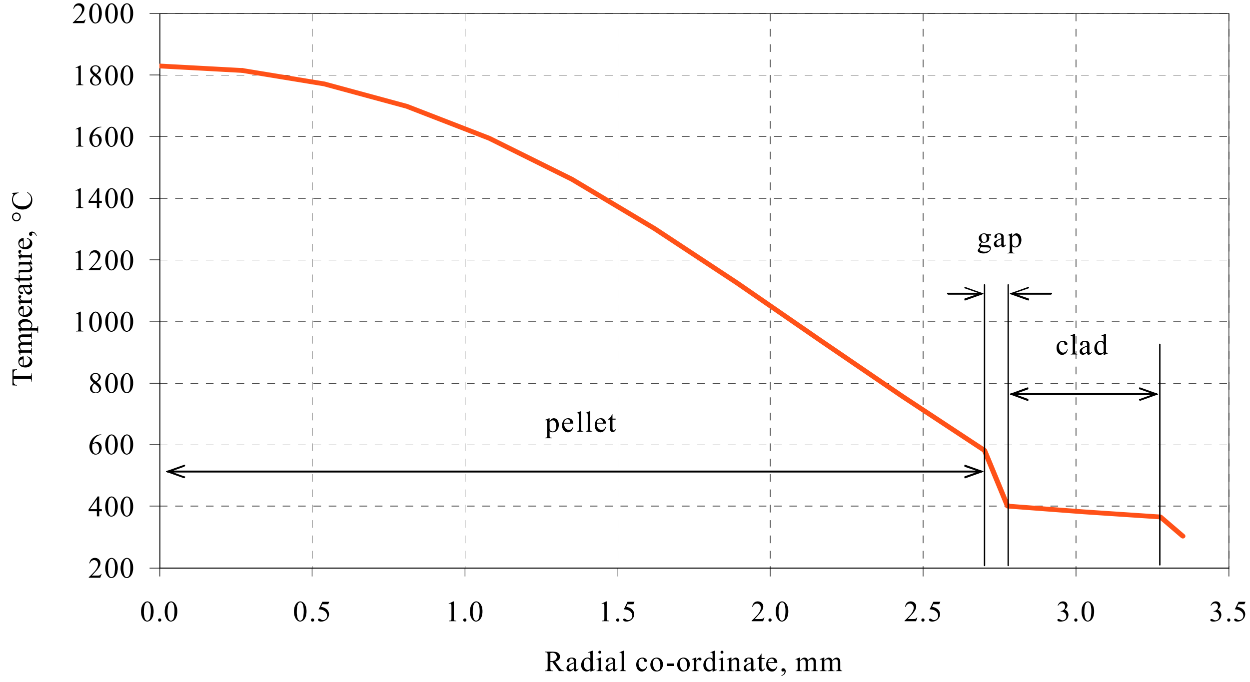
$T_{melt} = 2685\ °C$



6. Preliminary estimations of fuel operation parameters (4)

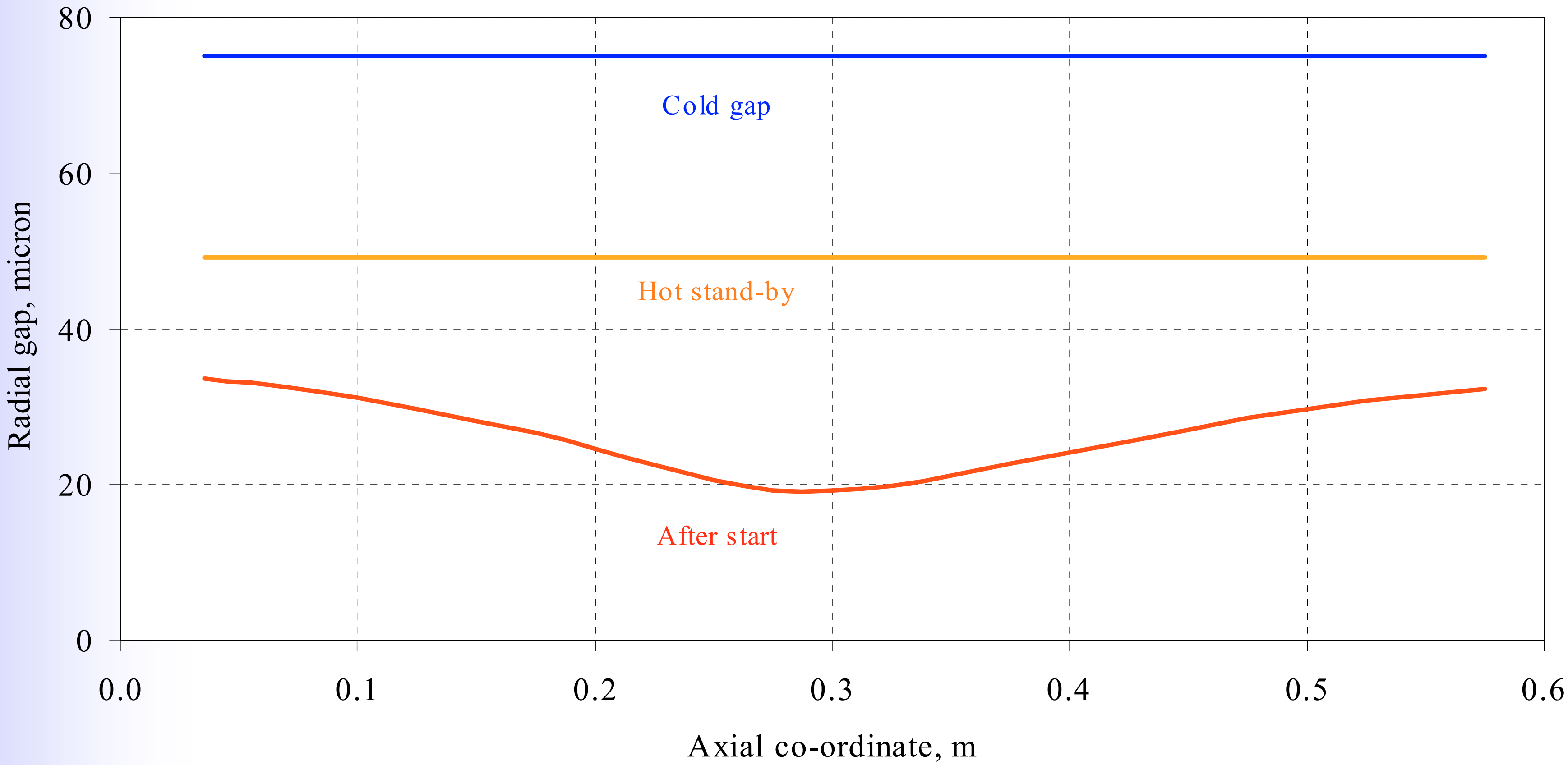
Initial radial temperature distribution in the hottest rod

T_{melt} = 2685 °C



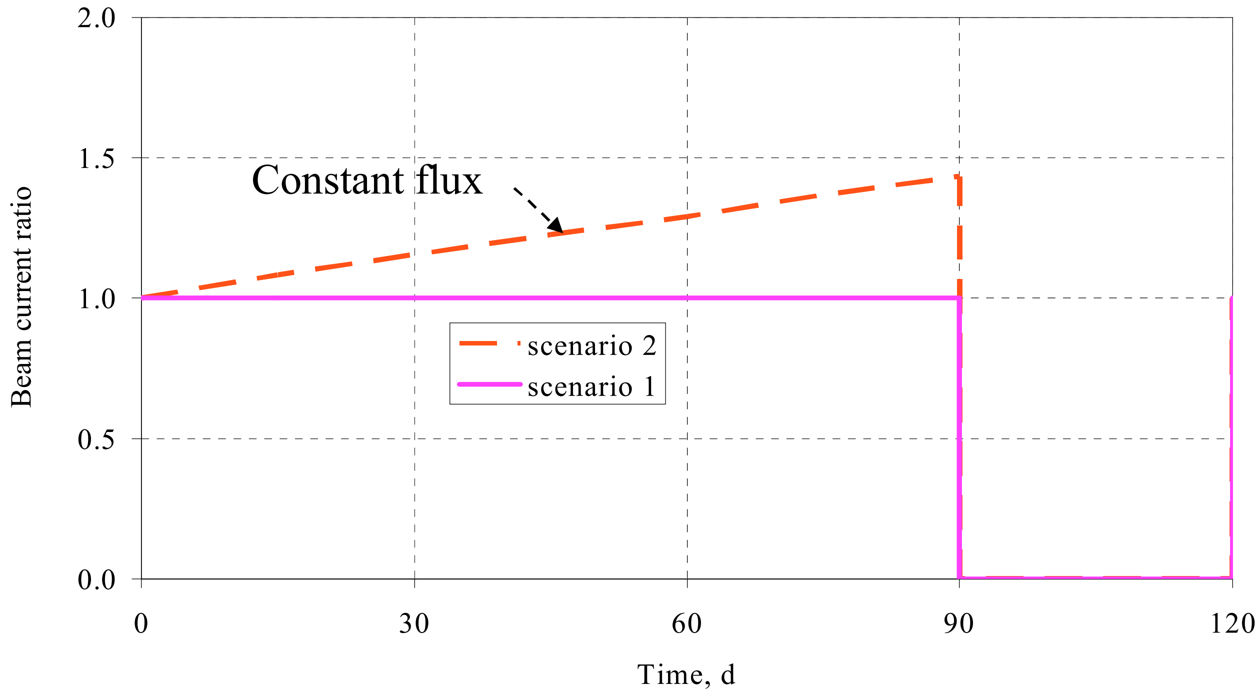
6. Preliminary estimations of fuel operation parameters (5)

"Pellet-clad" gap at start within the hottest rod



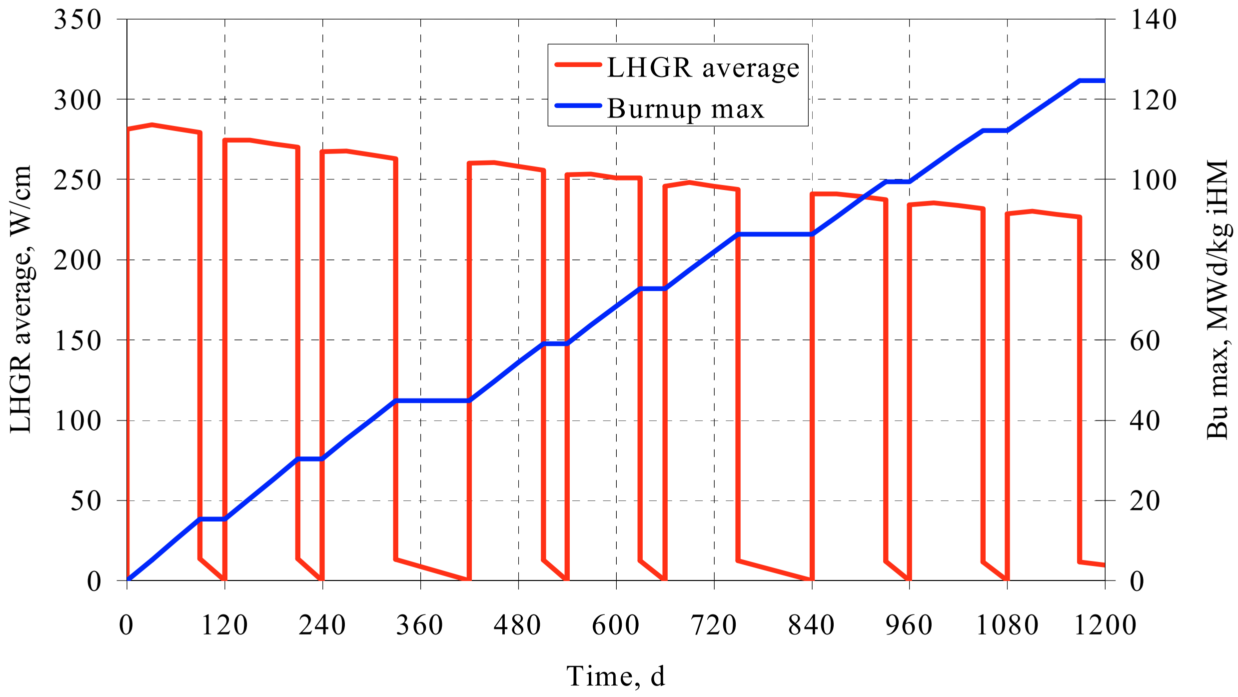
6. Preliminary estimations of fuel operation parameters (6)

Two scenario's for the proton beam operation in a cycle



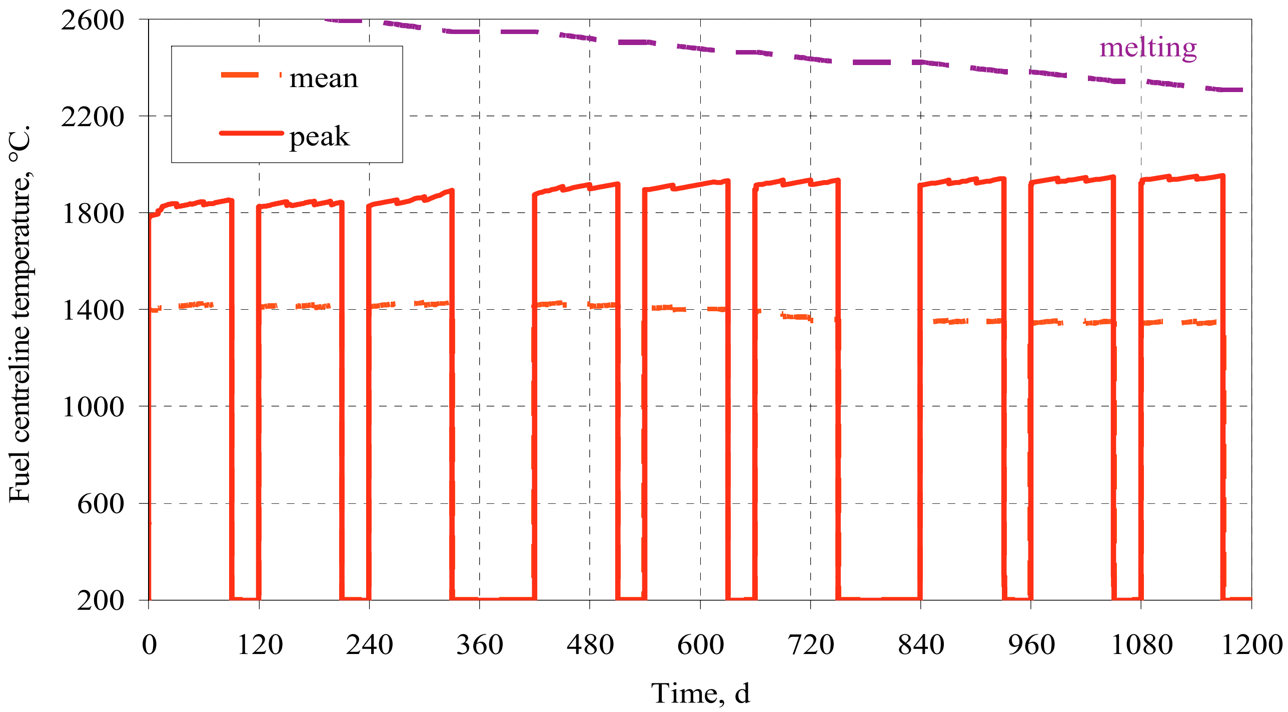
6. Preliminary estimations of fuel operation parameters (7)

Power history and peak burnup evolution in the hottest rod
(constant flux regime)



6. Preliminary estimations of fuel operation parameters (8)

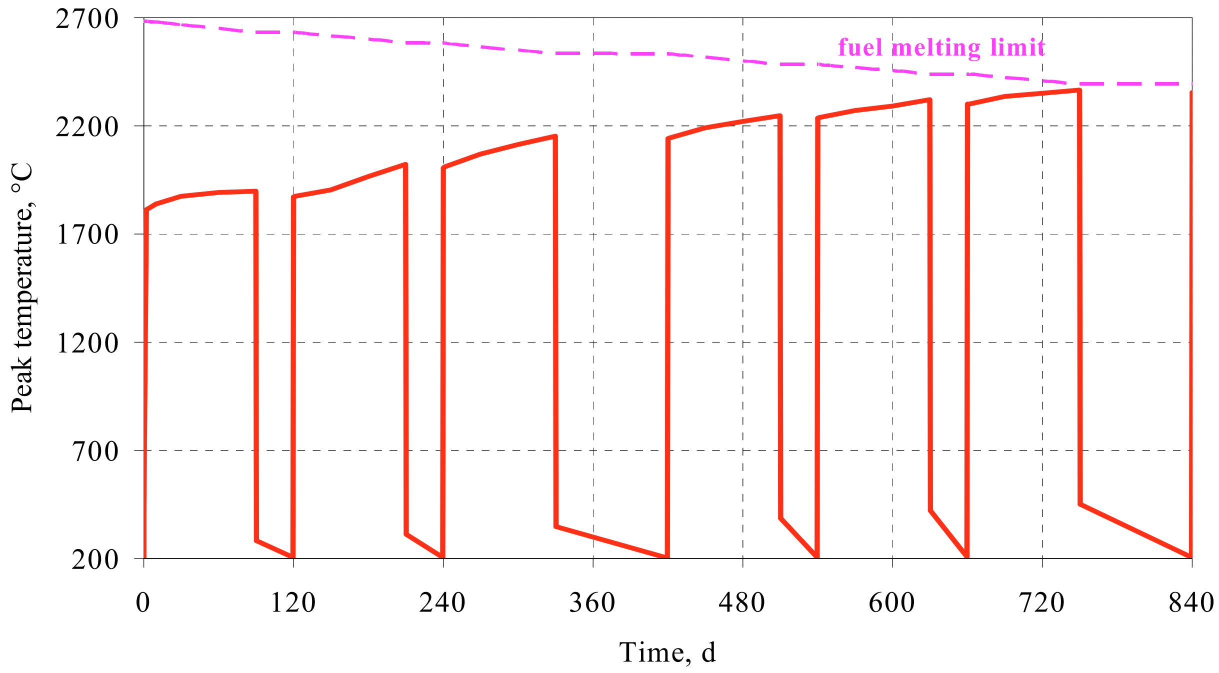
Peak temperature evolution in the hottest rod
(modelling with MACROS)



6. Preliminary estimations of fuel operation parameters (8a)

Peak temperature evolution in the hottest rod

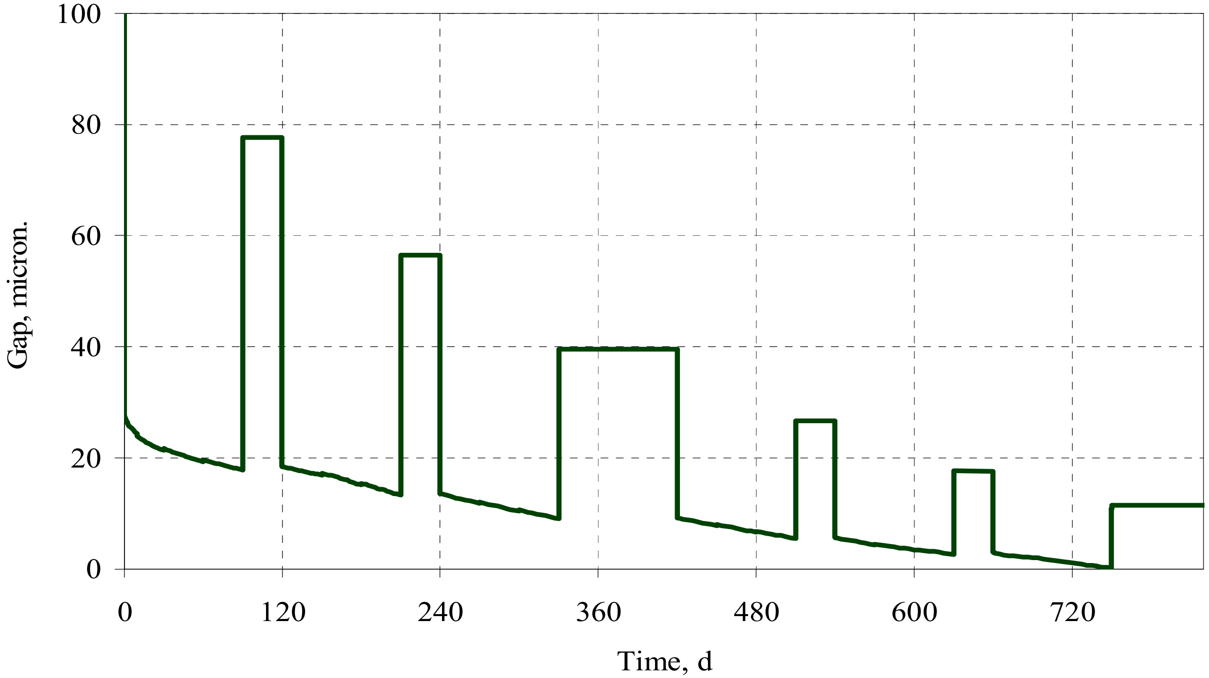
(modelling with FEMAXI – conservative case)



6. Preliminary estimations of fuel operation parameters (9)

Evolution of the mid-plane pellet-clad gap in the hottest rod

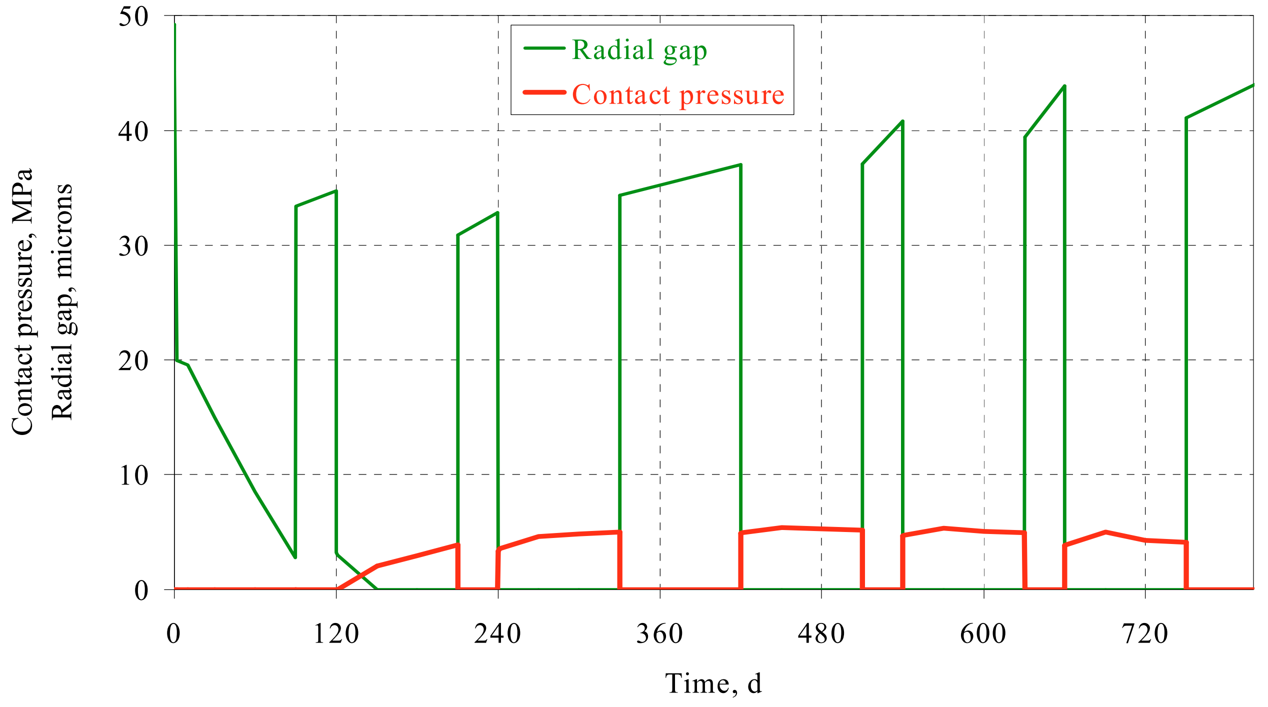
(modelling with MACROS)



6. Preliminary estimations of fuel operation parameters (9a)

Evolution of the mid-plane pellet-clad gap in the hottest rod

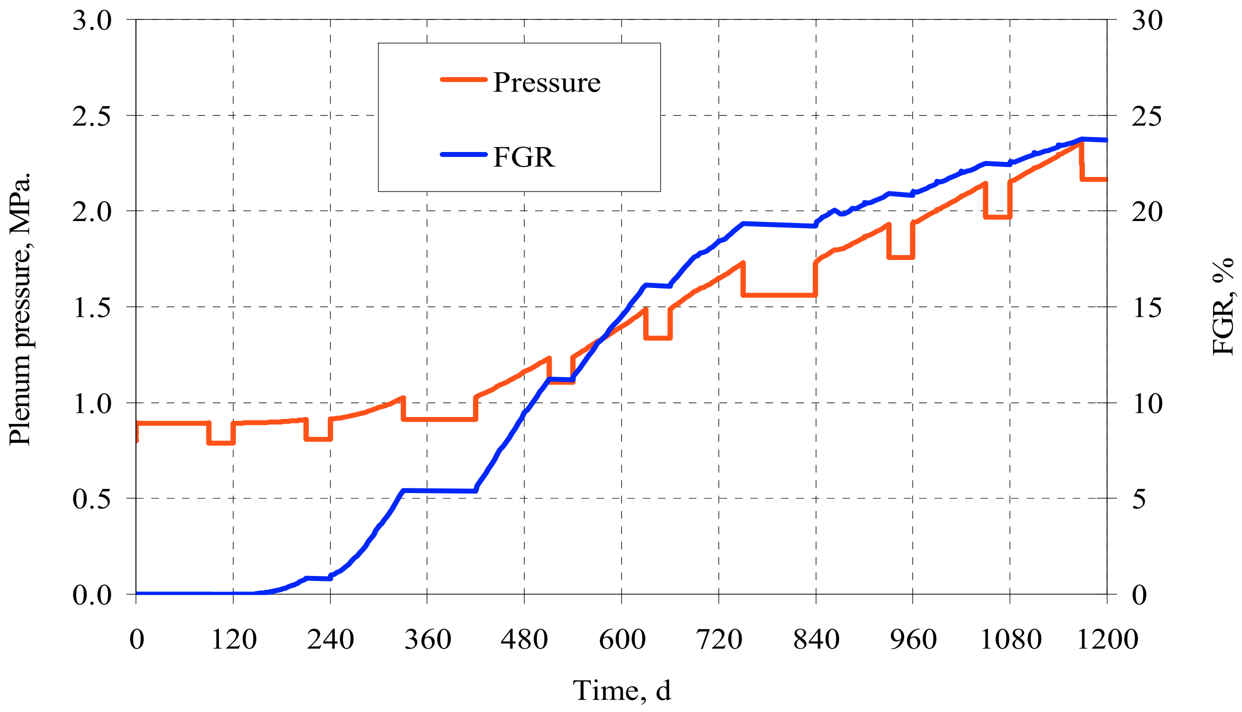
(modelling with FEMAXI – conservative case)



6. Preliminary estimations of fuel operation parameters (10)

FGR and pressure build-up in the hottest rod

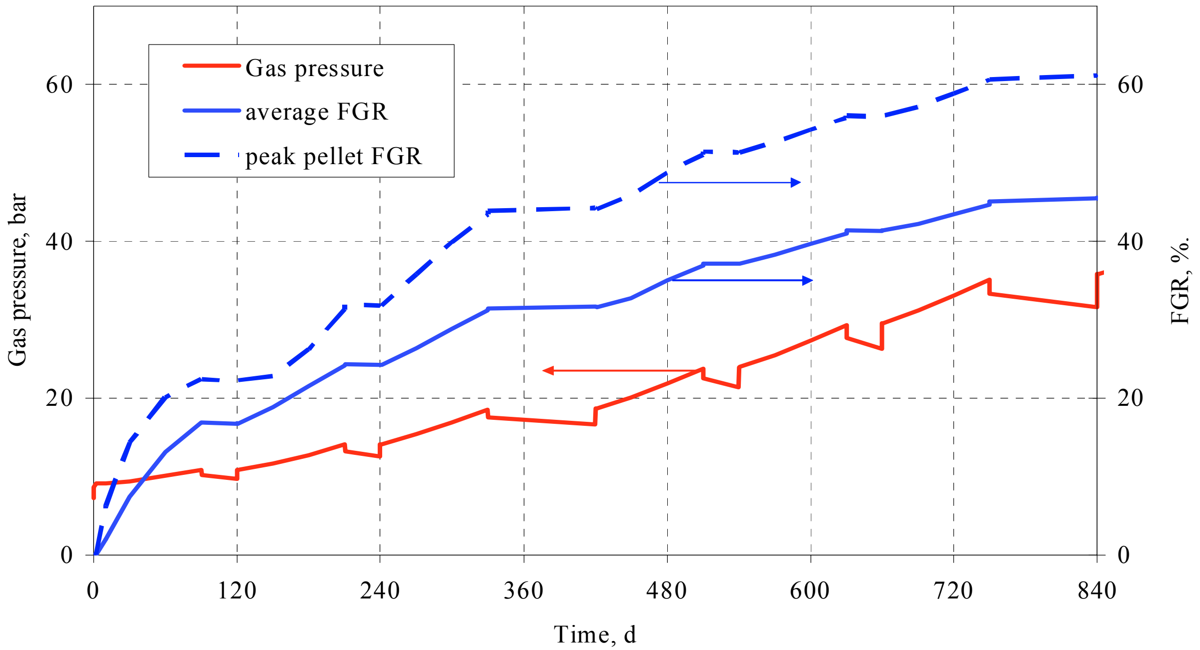
(modelling with MACROS)



6. Preliminary estimations of fuel operation parameters (10a)

FGR and pressure build-up in the hottest rod

(modelling with FEMAXI – conservative case)



6. Preliminary estimations of fuel operation parameters (11)

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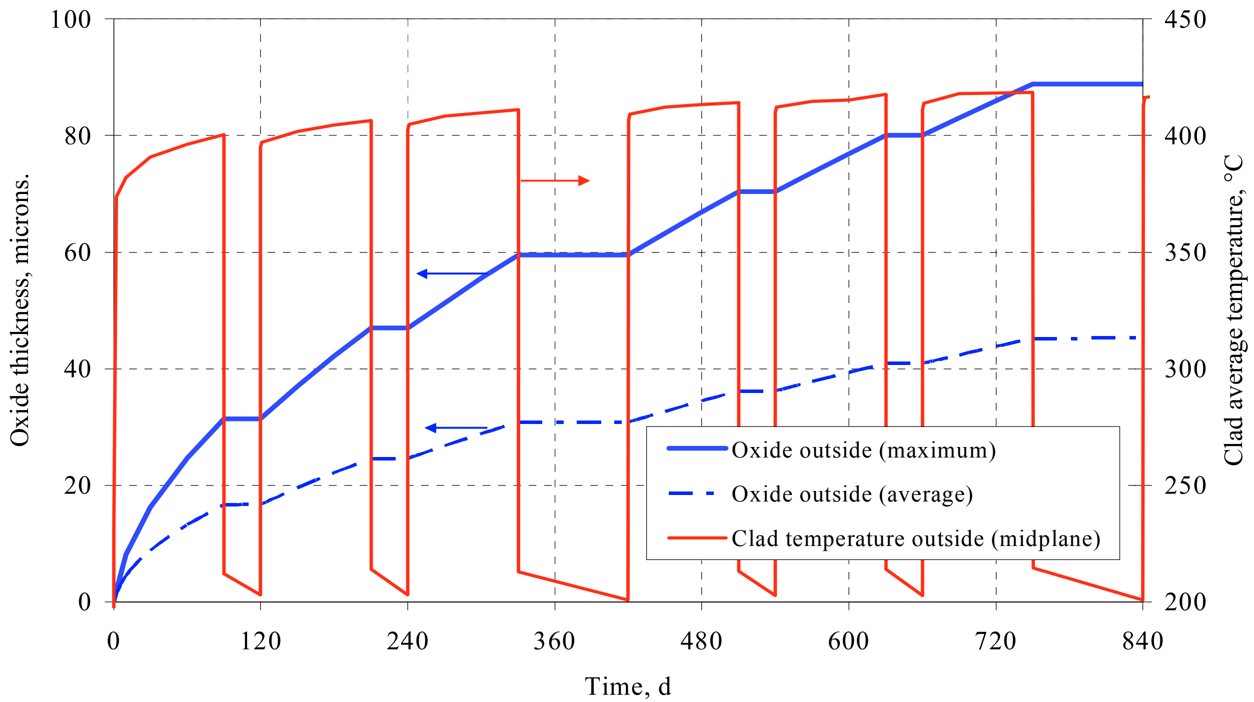
MYRRHA

Clad oxidation and temperature rise in the hottest rod



A better protection of the T91 cladding is needed or a lower temperature after 2-3 cycles of operation

6. Preliminary estimations of fuel operation parameters (12)

Pressure drop in assembly (<T> = 300 °C, G = 55.5 kg s^{-1})

$$\Delta p_{assembly} = \sum_i \Delta p_i = \frac{G^2}{2 \cdot \rho} \cdot \sum_i \frac{\xi_i^{(friction)} + \xi_i^{(contr/\exp an)}}{S_{i\ flow}^2}$$

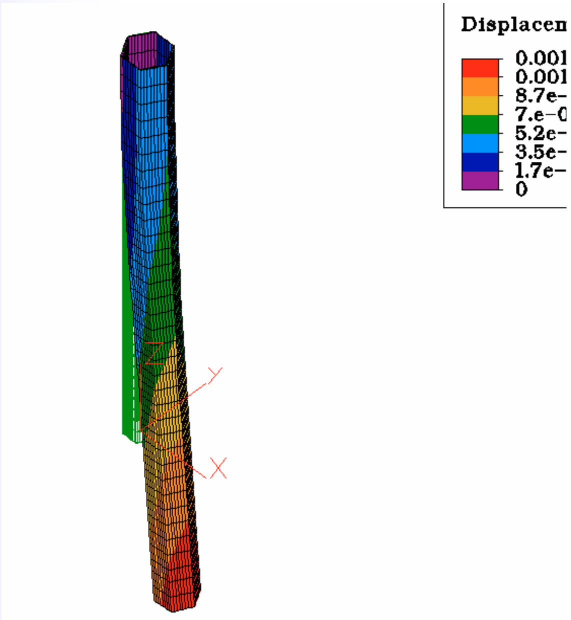
No.	Element	Δp	
		bar	%
1	Inlet tube, nozzle, hex-duct	0.05	3.1
2	Fuel rod bundle (free part)	1.37	85.6
3	Upper grid	0.12	7.5
4	Upper hex-duct, matching cone, outlet tube	0.06	3.8
	TOTAL	**1.6**	

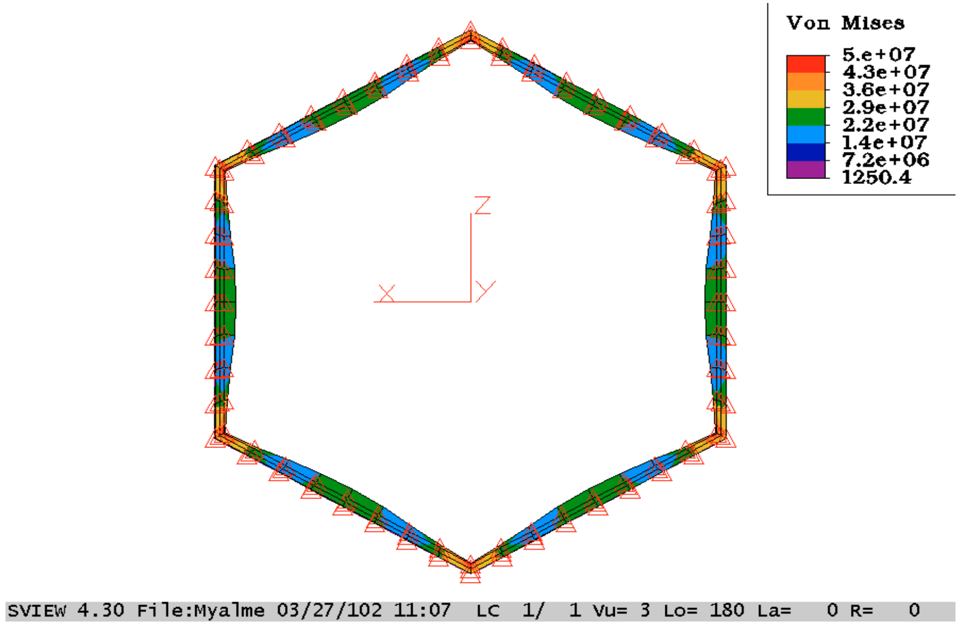
A more detailed thermal hydraulic modelling of assembly was performed with RELAP5 by SH,

… but the results at the normal operation have not yet been included in Draft-2

6. Preliminary estimations of fuel operation parameters (13)

Thermomechanical modelling of assembly
(with STRAW by BELGONUCLEAIRE, but old variant from Draft-1)





Thermomechanical modelling of assembly has still to be performed .

… but with which code?

7. Items still under consideration

- To fix the fuel Pu enrichment and the Pu isotopic vector.
- To establish a highly enriched MOX (30% Pu) properties database up to burn-up of 100 MWd/kg iHM.
- To establish the irradiated cladding properties database (T91 and others).
- To define realistic core management scenarios (k_{eff} swing compensation with meeting the requested performance).
- To perform thermomechanical modelling of fuel assembly.
- To optimise the current designs of fuel rod and fuel assembly.

8. Conclusions (I)

- Preliminary design of the MYRRHA fuel rod, fuel assembly and core has been updated to meet 50 MW(th) power.
- Modelling of the thermomechanical behaviour of the fuel rod under conservative (constant flux) irradiation conditions shows that the initial safety margins are sufficient for about three (two) years of the normal operation up to the aimed maximum burnup of ~100 MWd/kg iHM.
- The clad damage limit of 100 dpa are estimated to be within the achievable range taking into account the clad operating temperature range of 250-480 °C and the moderate He production rate (maximum 8 appm He/dpa).

8. Conclusions (II)

- The designed hexagonal fuel assemblies with medium pitch ratio of 1.3 can provide the adequate heat removal at normal operation with the maximum LBE local velocity of 2 m s^{-1} (and at protected DBC transients ?).

- The following progress in the optimisation of the designs of the fuel pin and the fuel assembly will be made after solving urgent problems existing in the fuel and cladding database properties and redefining a realistic core management scenarios.

- A validation and qualification programme for fuel is highly recommended to start ASAP, taking into account that at least 2-3 (up to 5) years are needed to fulfil this kind of irradiation programme.

Acknowledgements

This presentation was prepared
with the contributions of

LEMEHOV Sergei,
Al MAZOUZI Abderrahim,
MALAMBU Edouard,
HAECK Wim

ANNEX

What we had in DRAFT-1

- Only two pages on the fuel pin and assembly design (pp. 20-22, three figures included) were presented in the Draft-1 Document.
- **Three different fuel designs were analysed: SPX, BN-600 and SNR-300.**
- **The existing SPX fuel design (but with HT-9, T91 or AISI 316L cladding) was used as reference** in order to keep the shortest pre-design and expected deployment schedules.
- A **high flux** of the **fast neutrons**: ***~ 10^{15}* n/cm²s** in the hottest experimental channels at the initial ***k_{eff} ~ 0.95***. A small core thermal power - few tens of MW
- **Fuel performance calculations only at start.**

Choice of the driver fuel

Options:

1. What actinides ?
2. → enriched U, **Pu-U**, Pu-Th, U-Th.
3. Enrichment level ?
4. → *how to deal with "20 % U-235 equivalent limit"?*
5. What chemical form ?
6. → metal, **oxide**, carbide, nitride.
7. Physical state ?
8. → solid solution, **mixture**, CERMET, …

$(Pu,U)O_2$ MOX with 30 wt.% RG Pu in HM has been chosen in MYRRHA, however, it would be useful to revisit other options.

Cladding choice

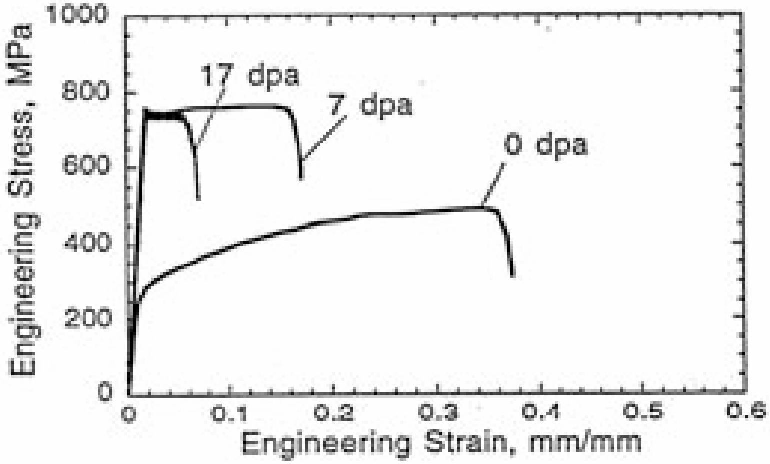
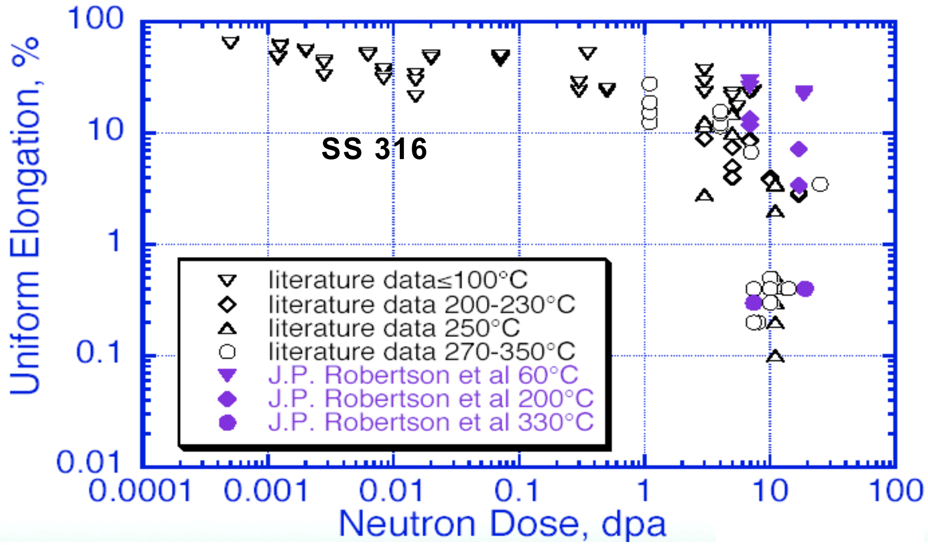


Figure 3-44. Effect of irradiation on the stress-strain curve of 316 SS at 473 K



Determination of fuel pellet sizes

Radial thermal resistivity of fuel rod

$$\Re_{pellet} = \frac{1}{4 \cdot \pi \cdot \langle \lambda_{pellet} \rangle}$$

$$< \lambda_{pellet} > \equiv \frac{1}{\Delta T_{fuel}} \cdot \int_{T_{fuel\ surface}}^{T_{fuel\ centre}} \lambda_{pellet}(T) \cdot dT$$

$$\Re_{cool} = \frac{1}{\pi \cdot D_{clad} \cdot h_{heat}}$$

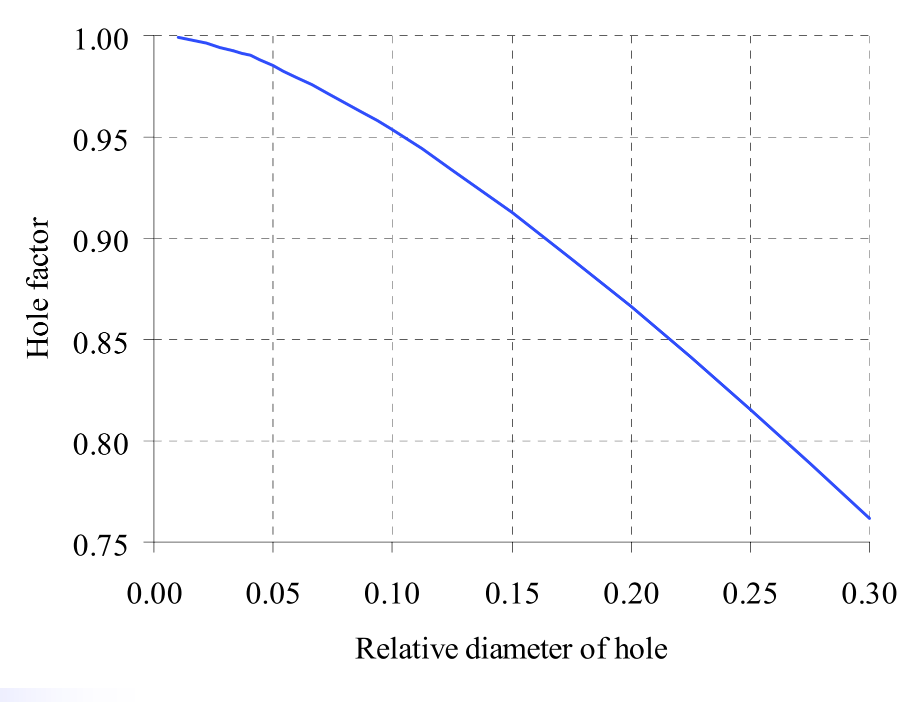
$$h_{heat} = 0.58 \cdot \left(1.1 \cdot x_{pitch}^2 - 1\right)^{0.55} \cdot \left(\frac{\mathrm{v}_{cool} \cdot D_{clad}}{a_{p\ cool}} \right)^{0.45} \cdot \frac{\lambda_{cool}}{D_{clad}}$$

$$\Re_{clad} \approx \frac{\delta_{clad}}{\pi \cdot \langle D_{clad} \rangle \cdot \langle \lambda_{clad} \rangle}$$

$$\Re_{gap} \approx \frac{\delta_{gap}}{\pi \cdot \langle D_{gap} \rangle \cdot \langle \lambda_{gap} \rangle}$$

Determination of fuel pellet sizes

Radial thermal resistivity of the holed pellet

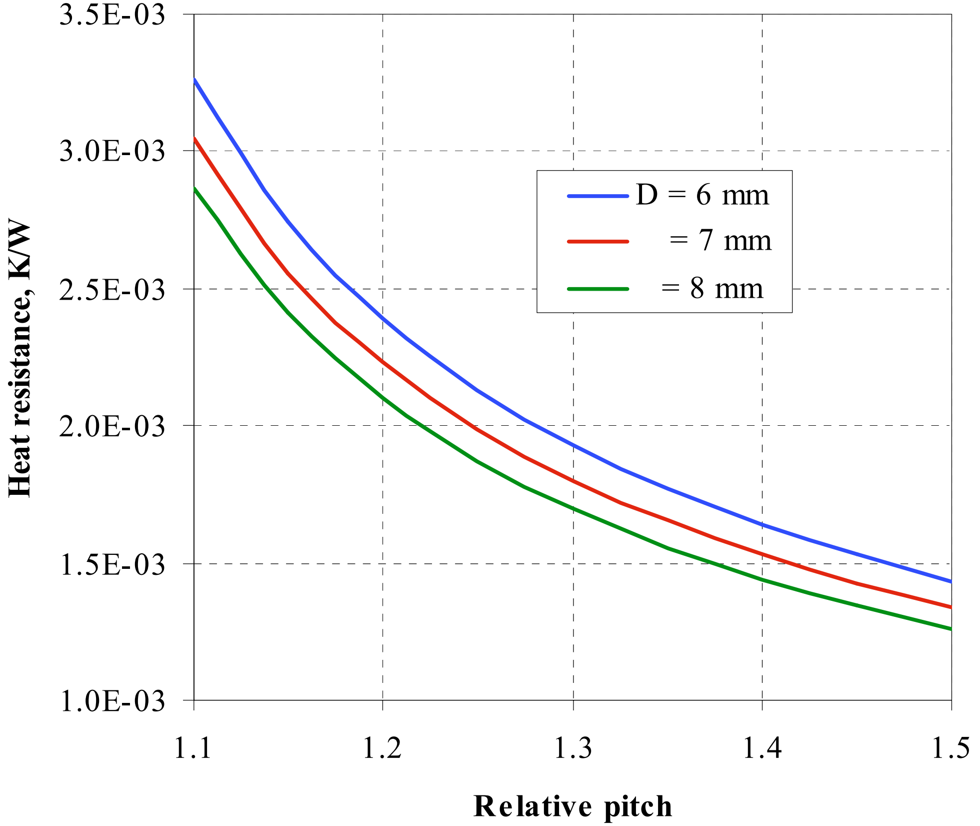


$$\Re_{pellet} = \frac{1}{4 \cdot \pi \cdot \langle \lambda_{pellet} \rangle} \cdot \left(1 + \frac{\left(\frac{d_{hole}}{D_{pellet}} \right)^2 \cdot \ln \left(\frac{d_{hole}}{D_{pellet}} \right)^2}{1 - \left(\frac{d_{hole}}{D_{pellet}} \right)^2} \right)$$

$$< \lambda_{pellet} > \equiv \frac{1}{\Delta T_{fuel}} \cdot \int_{T_{fuel\ surface}}^{T_{fuel\ centre}} \lambda_{pellet} (T) \cdot dT$$

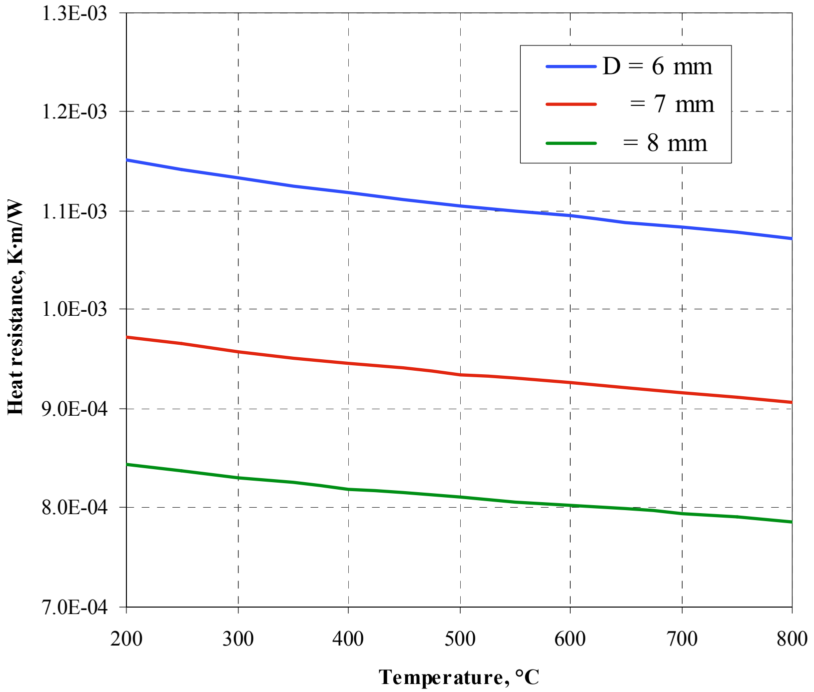
Determination of fuel pellet sizes

Radial thermal resistivity of the coolant boundary layer: v = 2 m/s, <T> = 300 °C



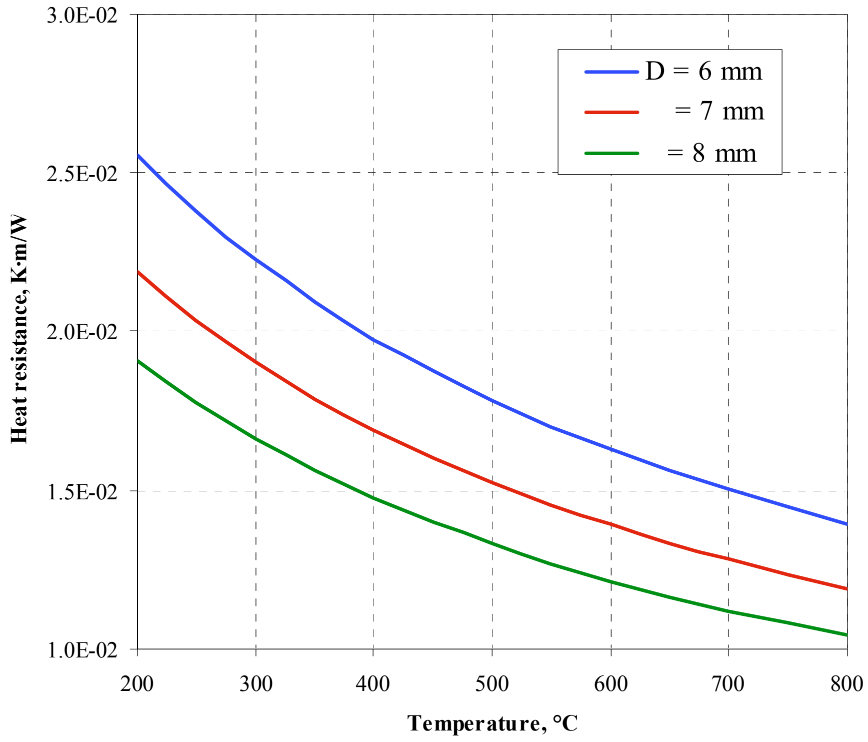
Determination of fuel pellet sizes

Radial thermal resistivity of 0.5 mm T91 cladding



Determination of fuel pellet sizes

Radial thermal resistivity of 0.1 mm gap filled with He-gas at 0.5 MPa (STP)



MYRRHA - Draft 2
Primary System Design

D. Maes, H. Aït Abderrahim

On behalf of MYRRHA team and MYRRHA Support

http://www.sckcen.be/myrrha

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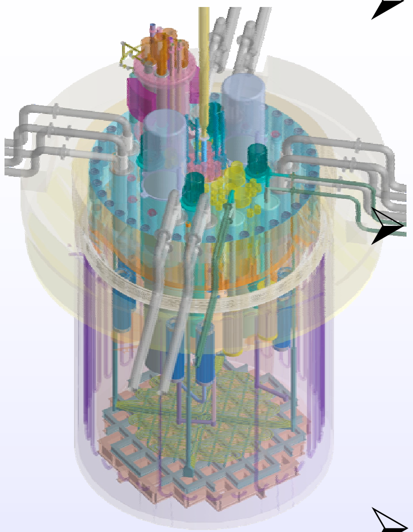
Design of the small scale eXperimental ADS: MYRRHA

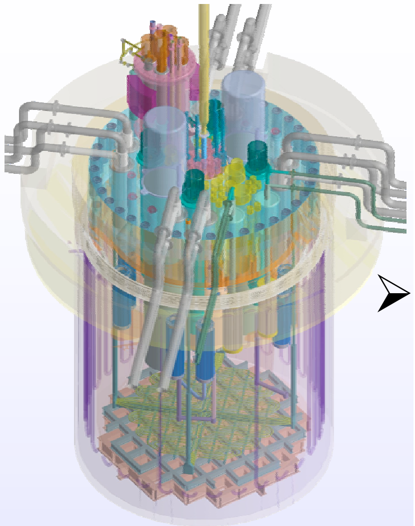
- Introduction
- Design requirements
- Design description
  - overall configuration and general characteristics
  - spallation loop and core interference
  - primary cooling system
  - diaphragm
  - in-vessel fuel manipulators
  - emergency cooling system
  - vessel and reactor cover
  - remote handling
- MYRRHA in the European frame

Design of MYRRHA
Introduction



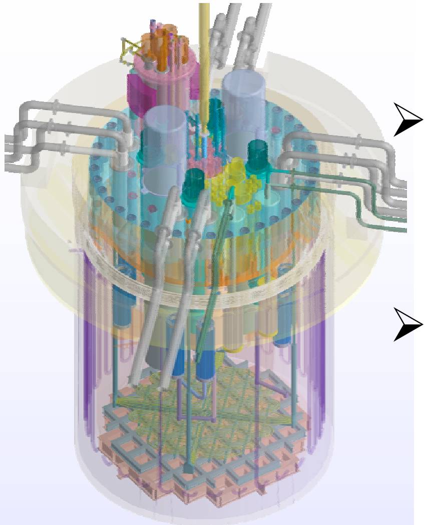
- Since 1998, SCK•CEN, in partnership with many European research laboratories, is designing a multipurpose ADS.
- In a first stage, the project focuses mainly on
  - *demonstration* of the ADS concept;
  - *safety* research of sub-critical systems;
  - nuclear waste *transmutation* studies.
- Subsequently, MYRRHA will be used as
  - a *fast* spectrum irradiation *test* facility (research on structural materials, nuclear fuel, liquid metal technology);
  - a *radio-isotope* production facility.

Design of MYRRHA
Design requirements



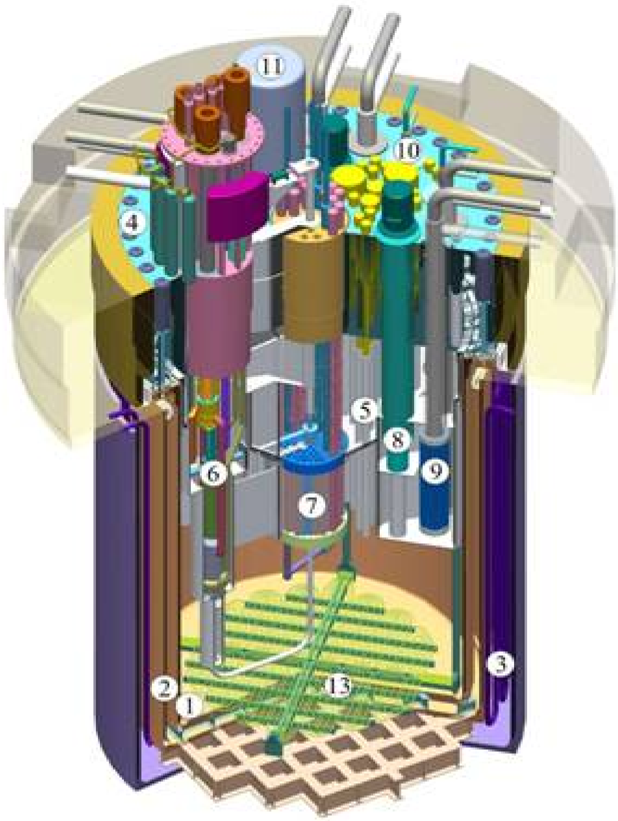
- As an irradiation test facility, MYRRHA must have
  - the capability to host experimental *irradiation rigs* in the core and in positions out of the core;
  - *flexible core management* for the fuel assemblies and for the experimental irradiation devices.
- The demonstration of transmutation requires a fast and high n-flux (~10^{15} n/cm².s, >0.75 MeV), that in turn implies:
  - a *compact* core;
  - this flux almost mandates *HLM cooling* (LBE);
  - the structure must be sufficiently *resistant* against irradiation, corrosion/erosion in the LBE.
- Direct access in the MYRRHA hall for personnel is highly improbable → ISIR and O&M are performed by *remote handling.*

Design of MYRRHA
Design requirements (cont'd)



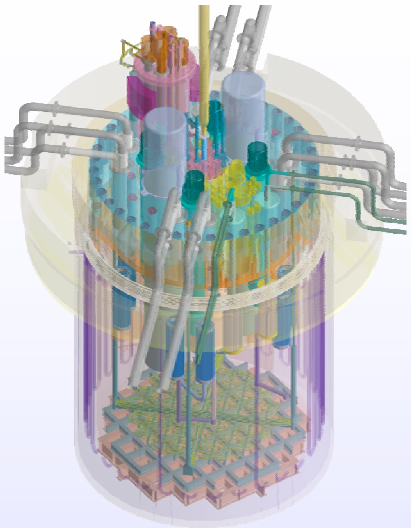
- Core *cooling* has to be guaranteed in all condi-tions in order to prevent damage to the system.
- All in-vessel components can be *removed* and *exchanged* during lifetime of the installation for maintenance.
- A *pool-type* reactor was chosen:
  - for safety reasons (large thermal inertia of several hundreds of tons of LBE);
  - the LBE pool serves as primary coolant for the spallation target and the core;
  - the LBE pool serves as reflector/shielding for the fast neutrons and gamma rays;
  - it provides an extremely flexible core management for the fuel assemblies and the experimental irradiation devices.

Design of MYRRHA
Overall configuration



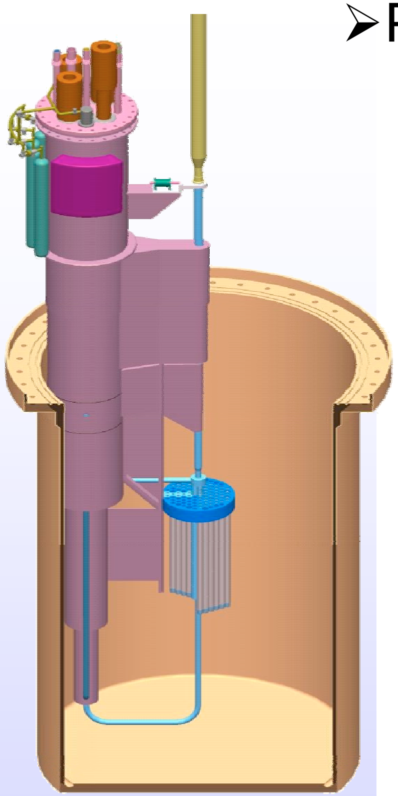
1. inner vessel
2. guard vessel
3. cooling tubes
4. cover
5. diaphragm
6. spallation loop
7. sub-critical core
8. primary pumps
9. primary heat exchangers
10. emergency heat exchangers
11. in-vessel fuel transfer machine
12. in-vessel fuel storage
13. coolant conditioning system

Design of MYRRHA
General characteristics



GENERAL CHARACTERISTICS	
Core external diameter	1,000 mm
Core height	1,800 mm
Fuel length	600 mm
Vessel inner diameter	4,400 mm
Vessel total height (cover not included)	7,000 mm
Vessel cover thickness	abt. 2 m
Gas plenum height above the coolant	< 500 mm
Nominal power	50 MW_{th}
Primary coolant	LBE
Coolant pressure	hydrostatic / +5 bar
Core inlet temperature	200 °C
Core outlet temperature	337 °C
Coolant velocity in the core	2.0 m/s
Primary coolant core flow rate (nominal)	2,500 Kg/s
Secondary coolant	water or steam

Design of MYRRHA
Spallation loop



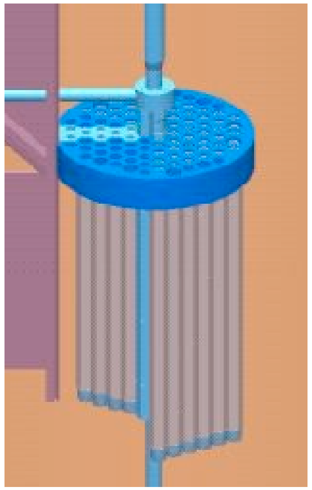
➢Reasons for the off-centre arrangement:

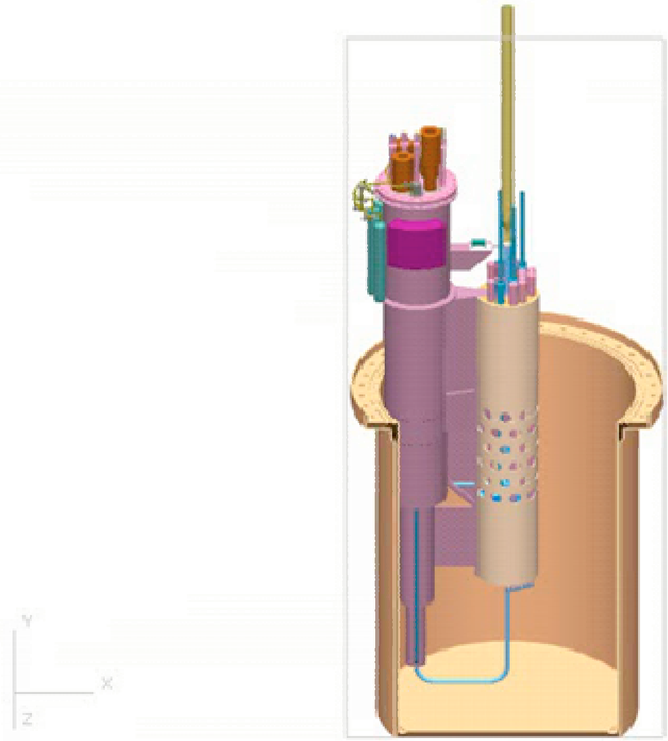
- The *small central hole* in the very compact core, which is mandatory to achieve the required neutron flux, offers only space for the LBE to follow in one direction (top-down path);
- The *circulation pumps* of the SL are located *under* the level of the target free surface (windowless target!) and there is clearly no possibility to do that in the small central channel;
- Locating the large SL confinement vessel centrally above the core would close the door for *easy access* to the core for the experimental rigs, jeopardising the flexibility of MYRRHA as a research irradiation facility.
- Off-centre arrangement *limits* the radiation *damage* of all sensitive components of the SL.

Design of MYRRHA
Spallation loop

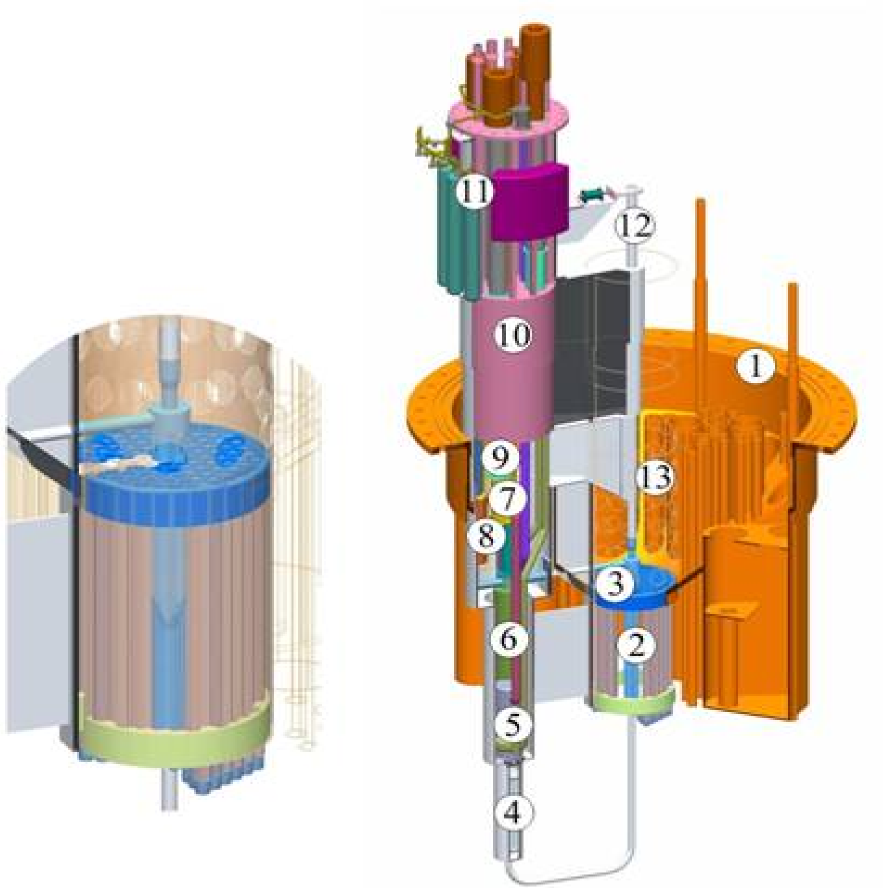
- For *periodic maintenance*, the SL can be *extracted* from the reactor. A *slot* in the core barrel is there-fore foreseen.

- Necessary "*patch*" to fill the special slot in the core support plate.





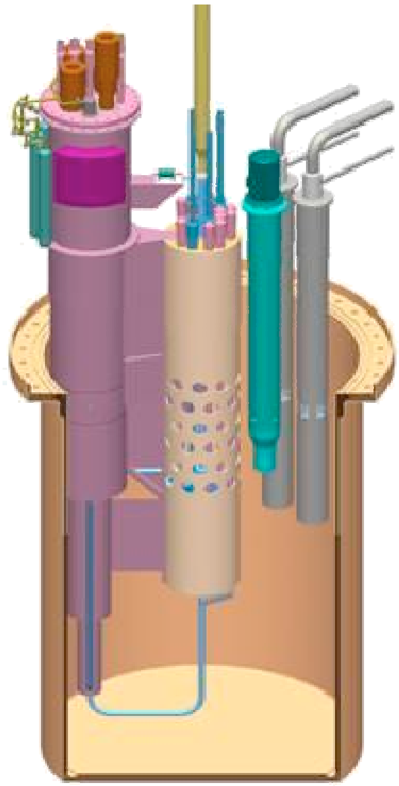
Design of MYRRHA
Spallation loop



1. diaphragm
2. spallation target
3. core support plate slot
4. heat exchanger
5. turbine & pump
6. electromagnetic pump
7. hydraulic drive
8. Pb-Bi conditioning system
9. vacuum system with cryopumps
10. shielding bloc
11. regeneration circuit with absorber pumps
12. proton beam line
13. core barrel

Design of MYRRHA
Primary cooling system

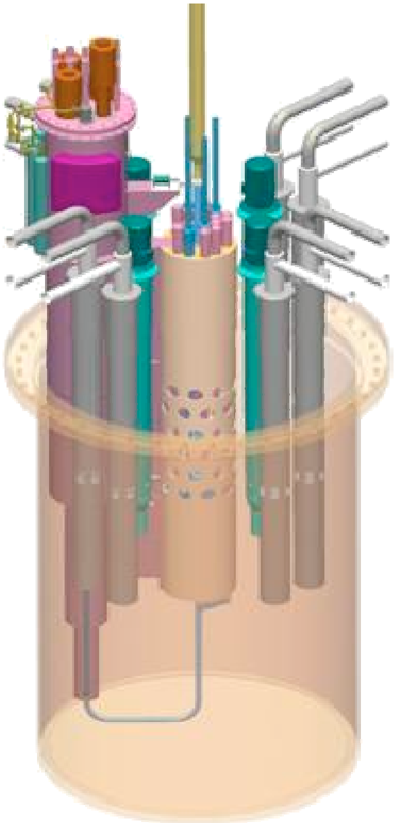
- The primary cooling system uses *water* as secondary coolant to evacuate the heat pro-duced in the vessel.
- The eight *heat exchangers* (HX) have the straight tubes, are single pass and counter-current.



- The four primary pumps are vertical units with an *impeller* at the bottom end of a long shaft. A one-way *valve* is fitted on the discharge pipe of the pump to avoid a reverse flow when the pump is shut down.

Design of MYRRHA
Primary cooling system

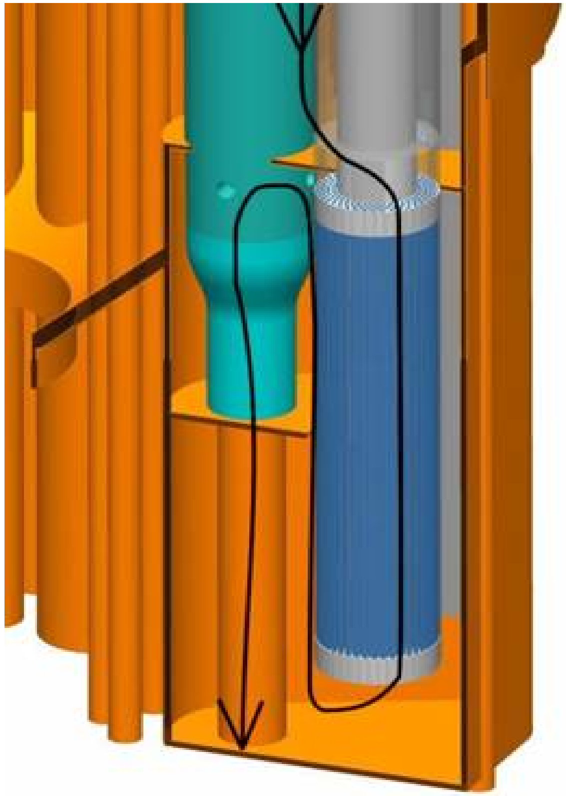
- The cooling system is *designed* for *60 MW_{th}*
- The total heat production in the vessel is the sum of the nominal *core* heat production (50 MW_{th}) and *other* heat sources (1.8 MW)



- *Four groups* with each one pump and two (secondary) water heat exchangers are installed at the periphery of the vessel = 4 pumps and 8 heat exchangers.
- The system is capable to evacuate the total heat production even in the case of the *failure of one pump*

Design of MYRRHA
Primary cooling system

- Each HX/PP group is placed in its casing in such a way that the flow path describes a vertical *chicane* which should help to *avoid water ingress* in the core by providing the separation of water/ vapour and Pb-Bi in case of a tube rupture.
- A *leak detection system* on each HX/pump casing is foreseen. It detects the presence of steam or water at the high point of the chicane.



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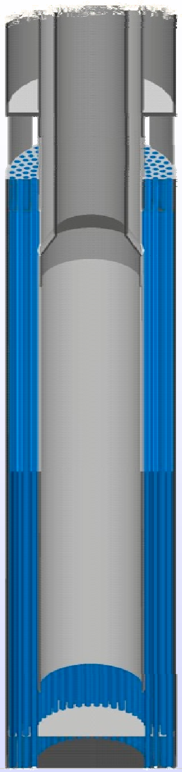
Design of MYRRHA
Primary cooling system

MYRRHA

Sizing heat exchangers

- Calculations in Mathcad (several files)
- Calculation notes :
    - 300_DM_Calcnote_Water-HEX_1.0.0.doc
    - 300_DM_Calcnote_Thermal-Stress-HEX_1.0.0.doc
    - 300_DM_Calcnote_Boiling-Water-HEX_1.0.0.doc
- LBE properties : Database of thermal properties for melted Lead_Bismut Eutectic – V. Sobolev, Internal report (IR-32-B043-...)
- Water/steam properties : steam tables integrated in Mathcad
- Material properties : "Standards of the Tubular Exchanger Manufacturers Association" Tables D-10, D-11 and D-12, which refer to the ASME codes, Sect. VIII, Div. 2
  Cr2¼, T91, A316L

Design of MYRRHA
Primary cooling system



Not boiling water HX :

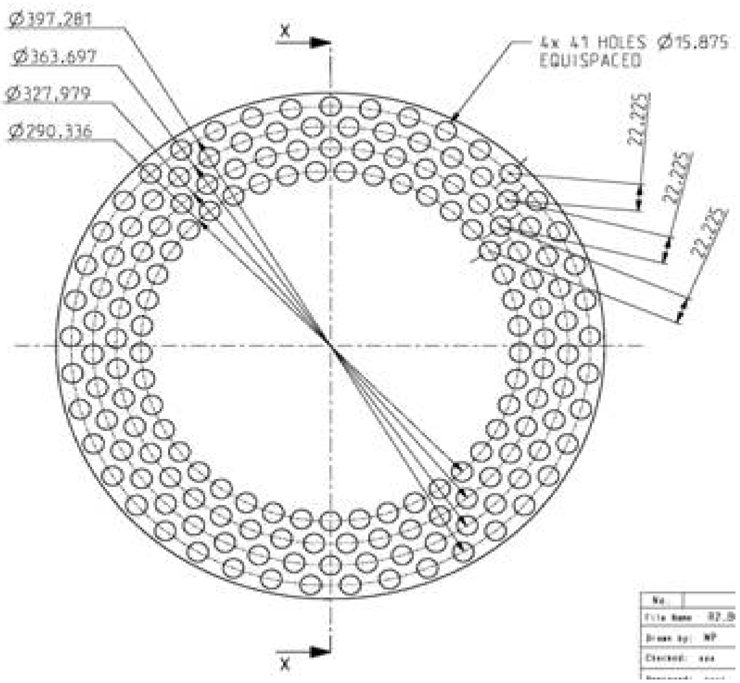
- water outside tubes;
- LBE inside tubes;
- tubes in radial lattice.

Heat transfer correlations :

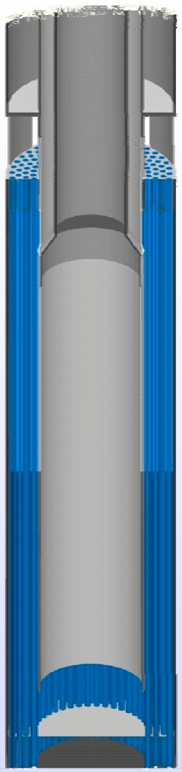
1) LBE side:

$$Nu = 5 + 0.025 \cdot Pe^{0.8}$$

$$Pe = Re \cdot Pr \qquad Re = \frac{v_i \cdot \phi_i}{\nu_L}$$



Design of MYRRHA
Primary cooling system



2) WATER side :

$$Nu = \left[0.0165 + 0.02 \cdot \left[1 - 0.91 \cdot \left(\frac{l_{pe}}{\phi}\right)^{-2}\right] \cdot \left(\frac{l_{pe}}{\phi}\right)^{0.15}\right] \cdot Re^{0.8} \cdot Pr_w^{0.4}$$ (Kirrilov)

$$Re = \frac{v_u \cdot \phi_e}{\nu_w} \qquad \phi_e = \frac{4 \cdot \left[\frac{\pi}{4}\left[\varphi(NN)^2 - \varphi(1)^2 - \left(n_b - n\right) \cdot \phi^2\right]\right]}{(NN - 1) \cdot n \cdot \pi \cdot \phi}$$

$$l_{pe} = \phi \cdot \sqrt{\left(2 \cdot \frac{\phi_e}{\phi} + 2\right) \cdot \frac{\pi}{4 \cdot \sqrt{3}}}$$

$$R_e \neq Re_H = \frac{v_u \cdot D_H}{\nu_w} \qquad D_H = 4 \cdot \frac{\left[\frac{\pi}{4} \cdot \left(D_i^2 - d_u^2 - n_b \cdot \phi^2\right)\right]}{\left(\pi \cdot D_i + \pi \cdot d_u + n_b \cdot \pi \cdot \phi\right)}$$

Design of MYRRHA
Primary cooling system

PRIMARY HEAT EXCHANGER CHARACTERISTICS		
capacity for one heat exchanger	nominal	6.25 MW
	design	7.50 MW
	(if 1 pump fails) maximum	10.00 MW
total heat capacity	nominal	50 MW
	design	60 MW
O.D. tubes		5/8 inch
thickness tubes		0.042 inch
tube pitch		1.4 x 5/8 inch
number of tubes		164
tube length		1,383 m
tube material		T91
I.D. shroud		420 mm
O.D. shroud		431 mm
shroud material		A316 L
primary / secondary fluid	PbBi eutectic	water
design conditions (60 MW)	primary	secondary
inlet temperature	337 °C	140 °C
outlet temperature	200 °C	160 °C
flow rate per heat exchanger	375 kg/s	85 kg/s
pressure	hydrostatic	25 bar

Design of MYRRHA
Primary cooling system

Thermal stresses

$$\sigma_z(z,r,F) = \frac{F}{\pi\cdot\left(r_1^2 - r_0^2\right)} + \frac{E_t\cdot\alpha_t\cdot(C + D\cdot z)}{2\cdot(1-\nu)}\cdot\left(\frac{1 - 2\cdot\ln\left(\frac{r_1}{r}\right)}{\ln\left(\frac{r_1}{r_0}\right)} - \frac{2}{\frac{r_1^2}{r_0^2} - 1}\right)$$

$$\sigma_\varphi(z,r) = -\frac{E_t}{2\cdot(1-\nu)}\cdot\alpha_t\cdot(C + D\cdot z)\cdot\left(\frac{\frac{r_1^2}{r^2} + 1}{\frac{r_1^2}{r_0^2} - 1} + \frac{\ln\left(\frac{r_1}{r}\right) - 1}{\ln\left(\frac{r_1}{r_0}\right)}\right) - \frac{\frac{r_1^2}{r_0^2} + \frac{r_1^2}{r^2}}{\frac{r_1^2}{r_0^2} - 1}\cdot\left(p_1 - p_0\right) - p_0$$

$$\sigma_r(z,r) = \frac{E_t}{2\cdot(1-\nu)}\cdot\alpha_t\cdot(C + D\cdot z)\cdot\left(\frac{\frac{r_1^2}{r^2} - 1}{\frac{r_1^2}{r_0^2} - 1} - \frac{\ln\left(\frac{r_1}{r}\right)}{\ln\left(\frac{r_1}{r_0}\right)}\right) - \frac{\left(\frac{r_1^2}{r_0^2} - \frac{r_1^2}{r^2}\right)}{\left(\frac{r_1^2}{r_0^2} - 1\right)}\cdot\left(p_1 - p_0\right) - p_0$$

F depends on differential expansion between shroud and tubes.

Design of MYRRHA
Primary cooling system

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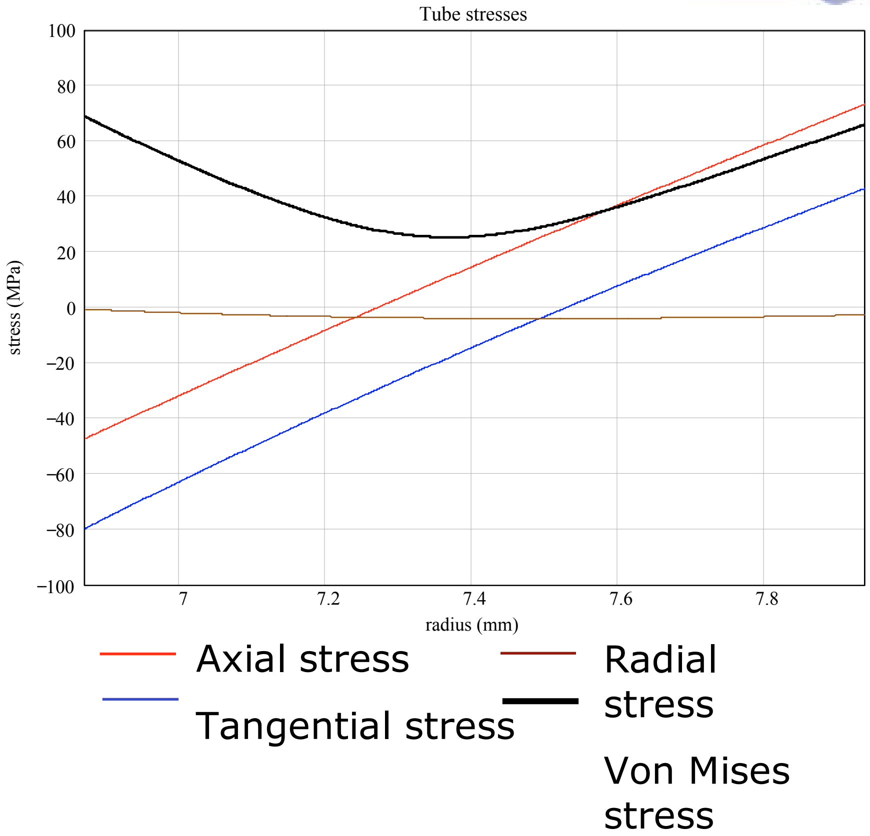
- *Differential expansion* between the tubes (T91 or Cr2¼) and the shroud causes additional axial stress in the tubes. This can totally be compensated if the PbBi flows inside the tubes and A316 is used for the shroud, which has larger thermal expansion than T91.

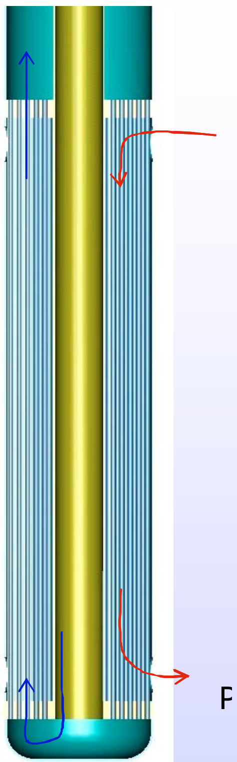


$$\sigma_{VM}(z,r,F) := \sqrt{\frac{\left(\sigma_{\varphi}(z,r) - \sigma_{z}(z,r,F)\right)^2 + \left(\sigma_{\varphi}(z,r) - \sigma_{r}(z,r)\right)^2 + \left(\sigma_{z}(z,r,F) - \sigma_{r}(z,r)\right)^2}{2}}$$

Design of MYRRHA
Primary cooling system

- Consequences of a tube rupture can be diminished by decreasing the *water flow rate* and *pressure*.
- Therefore *boiling water* heat exchangers were investigated

Design of MYRRHA
Primary cooling system



Boiling water HX :

- water/steam inside tubes;
- LBE outside tubes;
- tubes in hexagonal lattice.

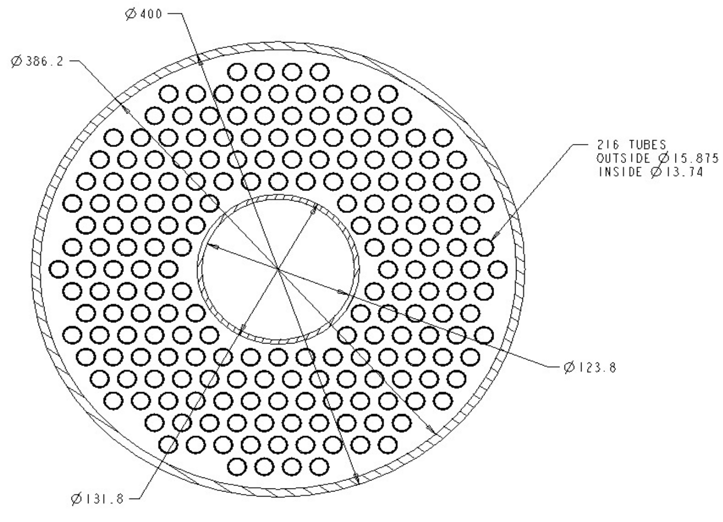
Heat transfer correlations :

1) LBE side:

$$Nu = 0.58 \cdot \left[1.1 \cdot \left(\frac{l_p}{\phi} \right)^2 - 1 \right]^{0.55} \cdot Pe^{0.45}$$

$$h = \frac{Nu \cdot k}{\phi_e} \qquad Pe = Re \cdot Pr \qquad Re = \frac{v_i \cdot \phi_e}{\nu_L}$$

$$\phi_e = \frac{\phi}{2} \cdot \left[\frac{4 \cdot \sqrt{3}}{\pi} \cdot \left(\frac{l_p}{\phi} \right)^2 - 2 \right]$$



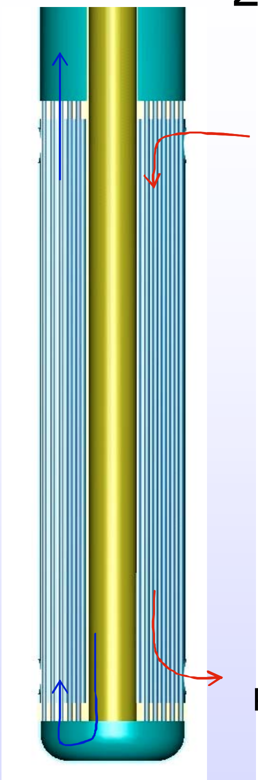
STEAM HEAT EXCHANGER
HORIZONTAL X-SECTION

$$\phi_e \neq D_H = 4 \cdot \frac{\left[\frac{\pi}{4} \cdot \left(D_i^2 - d_u^2 - n_b \cdot \phi^2 \right) \right]}{\left(\pi \cdot D_i + \pi \cdot d_u + n_b \cdot \pi \cdot \phi \right)}$$

!!

$$Re_L = \frac{v_i \cdot D_H}{\nu_L}$$

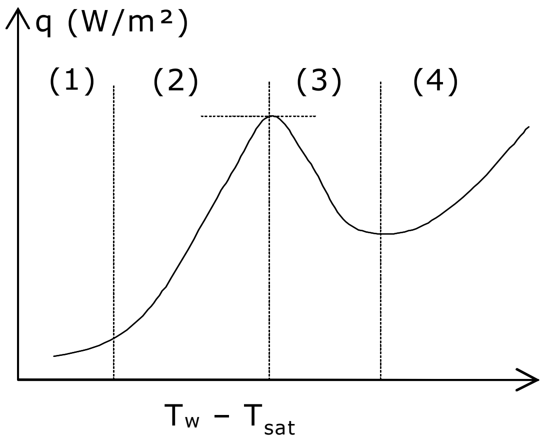
Design of MYRRHA
Primary cooling system



2) WATER side:

$$(1) \quad h = \frac{Nu \cdot k}{\phi_i} \qquad Nu = 0.023 \cdot Re^{0.4} \cdot Pr^{0.8}$$

$$(2) \quad h = \frac{Nu \cdot k}{\phi_i} + 2.54 \cdot \frac{(T_W - T_{sat})^4}{T_W - T} \cdot \exp\left(\frac{\frac{p}{MPa}}{1.551}\right)$$ (Jacob)



(1) single phase forced convection

(2) nucleate boiling

(3) transition boiling

(4) stable film boiling

CHF : critical heat flux

Tw : tube wall temperature

Tsat : saturation temperature

q : heat flux

Design of MYRRHA
Primary cooling system

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MYRRHA

Critical heat flux :

modified Zuber and Biasi correlations

$$CHF = 0.131 \cdot h_{fg} \cdot \rho_s \cdot \left[\frac{\sigma_w \cdot g \cdot (\rho_w - \rho_s)}{\rho_s^2} \right]^{\frac{1}{4}} \cdot \left(\frac{\rho_w}{\rho_w + \rho_s} \right)^{\frac{1}{2}} \cdot \left[1 - \frac{x}{\left[\chi + \frac{\rho_s}{\rho_w} \cdot (1 - \chi) \right]} \right] \quad (1)$$

$$CHF = \frac{1.883 \cdot 10^7}{\phi_i^{0.4} \cdot \left[\frac{M_W}{\left(n_b \cdot \frac{\pi}{4} \cdot \phi_i^2 \right)} \right]^{\frac{1}{6}}} \cdot \left[\frac{0.7249 + 0.099 \cdot (p_{sat}) \cdot \exp(-0.032 \cdot p_{sat})}{\left[\frac{M_W}{\left(n_b \cdot \frac{\pi}{4} \cdot \phi_i^2 \right)} \right]^{\frac{1}{6}}} - \chi \right] \quad (2)$$

$$CHF = \frac{3.78 \cdot 10^7 \cdot \left[-1.159 + 0.149 \cdot p_{sat} \cdot \exp(-0.019 \cdot p_{sat}) + \frac{8.99 \cdot p_{sat}}{\left(10 + p_{sat}^2 \right)} \right]}{\phi_i^{0.4} \cdot \left[\frac{M_W}{\left(n_b \cdot \frac{\pi}{4} \cdot \phi_i^2 \right)} \right]^{0.6}} \cdot (1 - \chi) \quad (3)$$

(1) $G_W < 100 \frac{kg}{m^2 \cdot s}$

(1)-(3) $100 \frac{kg}{m^2 \cdot s} < G_W < 200 \frac{kg}{m^2 \cdot s}$

(3) $200 \frac{kg}{m^2 \cdot s} < G_W < 300 \frac{kg}{m^2 \cdot s}$

max(2 , 3) $300 \frac{kg}{m^2 \cdot s} < G_W$

Design of MYRRHA
Primary cooling system

Two phase flow:

slip factor $$K_s = \frac{v_s}{v_l} = \frac{1-\alpha}{C - \alpha + (1-C)\cdot\alpha^r}$$ Bankoff correlation

$$C = 0.71 + \left(\frac{0.29}{0.32062}\right)\cdot\left(\frac{p_{sat}}{10000\cdot psi}\right)$$

$$r = 3.53125 - 0.1875\cdot\left(\frac{p_{sat}}{1000\,psi}\right) + 0.58594\cdot\left(\frac{p_{sat}}{1000\,psi}\right)^2$$

$$\alpha = \text{if}\left[\chi \neq 0, \text{root}\left[\alpha\cdot\left(\frac{1}{\chi} + \frac{\rho_w}{\rho_s} - 1\right) - (1-C)\cdot\alpha^r\cdot\frac{\rho_w}{\rho_s} - C\cdot\frac{\rho_w}{\rho_s}, \alpha, 0, 1\right], 0\right]$$

Design of MYRRHA
Primary cooling system

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Comparison with WALEBI

		Math_ Model	Walebi	
P_{HX}	MW	7,50	7,50	Power per HX
p	bar	7,00	7,00	Water pressure
M_W	kg/s	20,00	20,00	Water masss flow
M_L	kg/s	375	375	PbBi mass flow
T_{Ci}	°C	140,0	140,0	Water temperature at inlet
T_{Cu}	°C	165,0	164,0	Water temperature at outlet
∪	-	0,129	0,129	Flow quality at outlet
x	-	0,025	0,025	Thermodynamical quality
∪	-	0,863	0,864	Void fraction
T_{Hi}	°C	337,0	327,6	PbBi temperature at inlet
T_{Hu}	°C	200,0	189,8	PbBi temperature at outlet
v_u	m/s	0,68	0,68	Water velocity in tubes at inlet
v_i	m/s	0,60	0,73	PbBi velocity around the tubes
h_i	kW/m².°C	10,30	12,05	Heat transfer rate at tube PbBi-side
q_{max}	MW/m²	1,34	1,38	Max. heat flux through the tubes
q_{cr}	MW/m²	4,27	4,21	Critical heat flux
$∪_{VMmax}$	N/mm²	91	-	Max. Von Mises stress in tubes

Design of MYRRHA
Primary cooling system

	Boiling HX	Not-boiling HX
Liquid inside the tubes	**water**	**PbBi**
Number of tubes	216	164
O.D. of tubes	5/8"	5/8"
Pitch of the tubes	1.4 x 5/8"	1.4 x 5/8"
O.D. of HX	**400 mm**	**432 mm**
Effective length	1355 mm	1383 mm
Pressure	**7 bar**	**25 bar**
Water mass flow rate	**20 kg/s**	**85 kg/s**
Inlet/outlet water temp.	140 / 165°C	140 / 160°C
Flow steam quality out	12,9%	-
Vapor void fraction	86%	-
PbBi flow rate out	375 kg/s	375 kg/s
Inlet/outlet PbBi temp.	337 / 200°C	337 / 200°C
Water velocity inlet/outlet	0.68 / 4.4 m/s	3.66 m/s
Steam velocity tubes outlet	26 m/s	-
Von Mises stress	**91 MPa**	**69 MPa**

Design of MYRRHA
Primary cooling system

NOTE

Corrosion/erosion considerations:

Preliminary calculations for XT-ADS (Eurotrans) boiling water HX show that max. 40 μm Fe_2O_3-equivalent on both tube sides is allowed:

in such case, water pressure has to be reduced from 25 bar to 9 bar to preserve the cooling capacity.

Design of MYRRHA
Primary cooling system

Sizing primary pumps

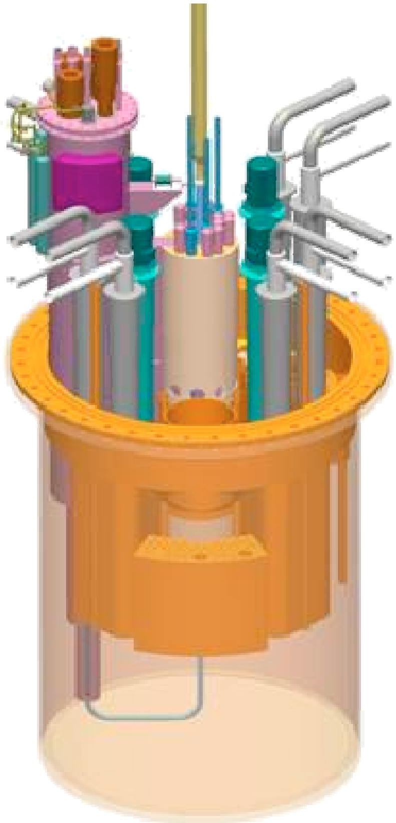
- Calculations in Mathcad (several files)
- Calculation notes are underway (Eurotrans)
- Based on "PUMP HANDBOOK", Igor J. Karassik, Third Edition, 2001
- Similitude theory:
    - rotor geometry is optimum in terms of the specific speed at best efficiency point (BEP).
    - experimentally determined during decades of years
    - for WATER !
    - small influence of Re (for Re >>)
    - But : influence of density ??

Design of MYRRHA
Primary cooling system

PRIMARY PUMPS CHARACTERISTICS		
number of pumps		4 (1 for 2 HX)
shaft length	min.	3.5 m
	max.	6 m
suction head	min.	0.5 bar
discharge head	approx.	5 bar
flow rate per pump	nominal	625 kg/s
	design	750 kg/s
	maximum	1000 kg/s
temperature at impeller		200 °C
temperature along shaft	max.	350 °C

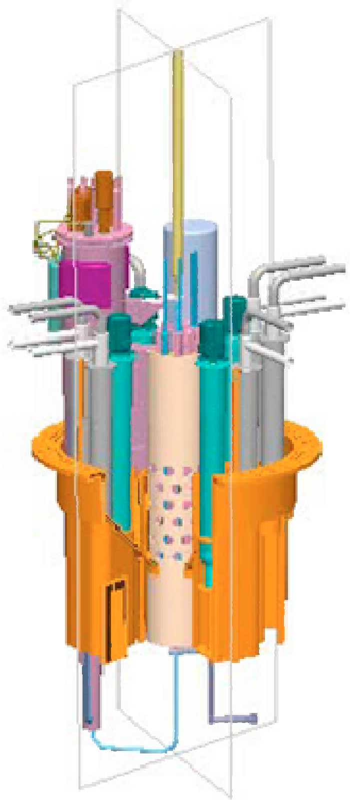
Design of MYRRHA
Diaphragm

- forces the coolant *flow path* through the core, separating the lower part (200°C, high pressure) of the Pb-Bi from the upper part (337°C, low pressure);
- supports the two in-vessel *fuel storages* (which are foreseen to avoid excessive delay between operation cycles);



- has *4 casings* containing the pumps and heat exchangers;
- has numerous *penetrations* for the large components (spallation loop, core, pumps, heat exchan-gers, handling machines) and for the smaller irradiation devices.

Design of MYRRHA
Diaphragm



Design of MYRRHA Diaphragm

Stresses in diaphragm

- Assessed with Pro/Mechanica
- Calculation note:
  - 300_DM_Calcnote_Diaphragm-stresses_1.0.0.doc
- the mechanical stress caused by the (Pb-Bi and pump) pressure is acceptable;
- the thermal (=secondary) stresses (in case of a wall temperature difference of 137°C) is rather high (~320 MPa);
- it's necessarily to re-assess the stresses with more realistic boundary conditions by taking into account the convective heat transfer from the liquid metal to the diaphragm rather than the prescribed temperature difference of 137°C over the diaphragm wall, which is a very conservative constraint.

Design of MYRRHA Diaphragm

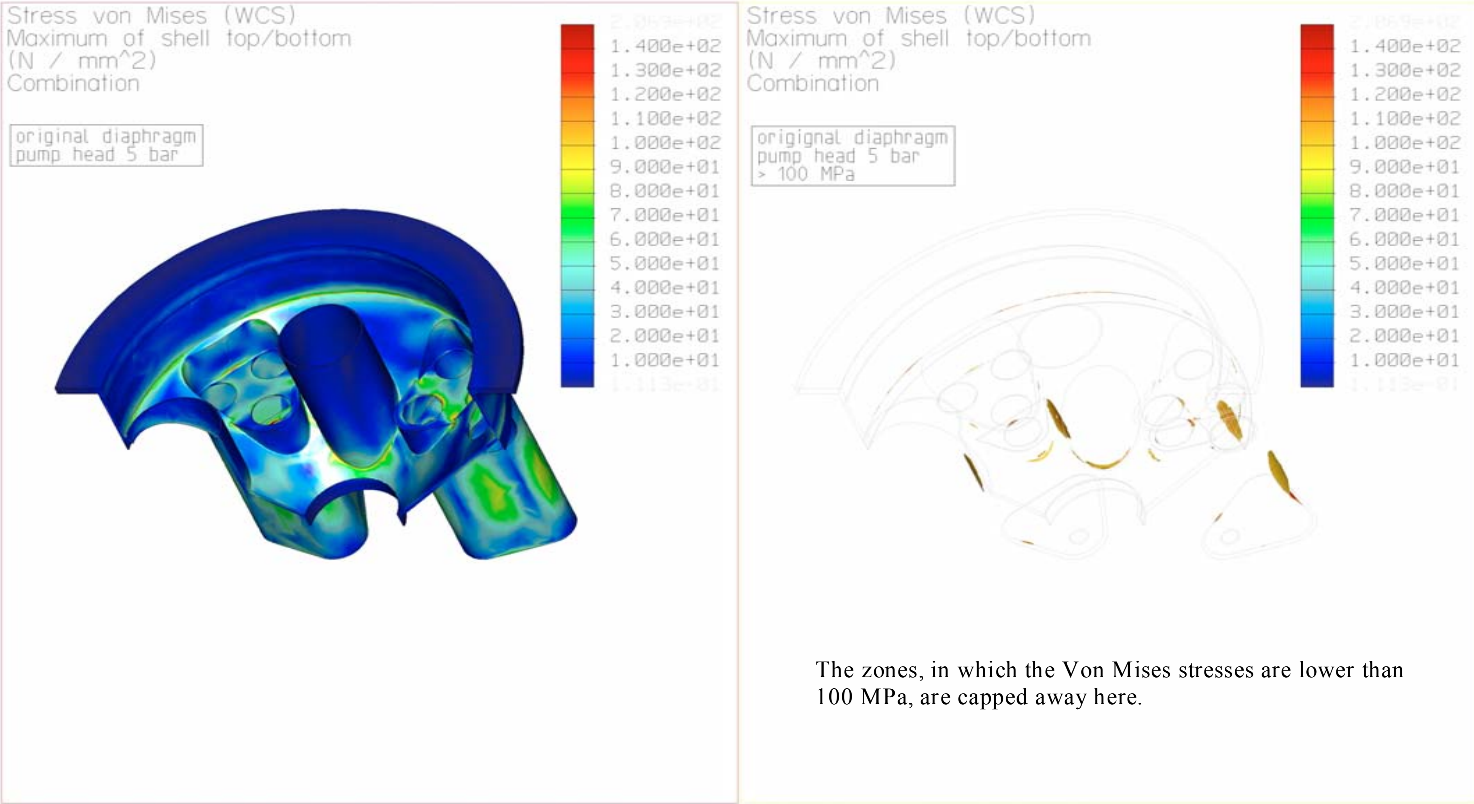


Figure 4. Von Mises stress for pump head 5 bars

Design of MYRRHA
Diaphragm

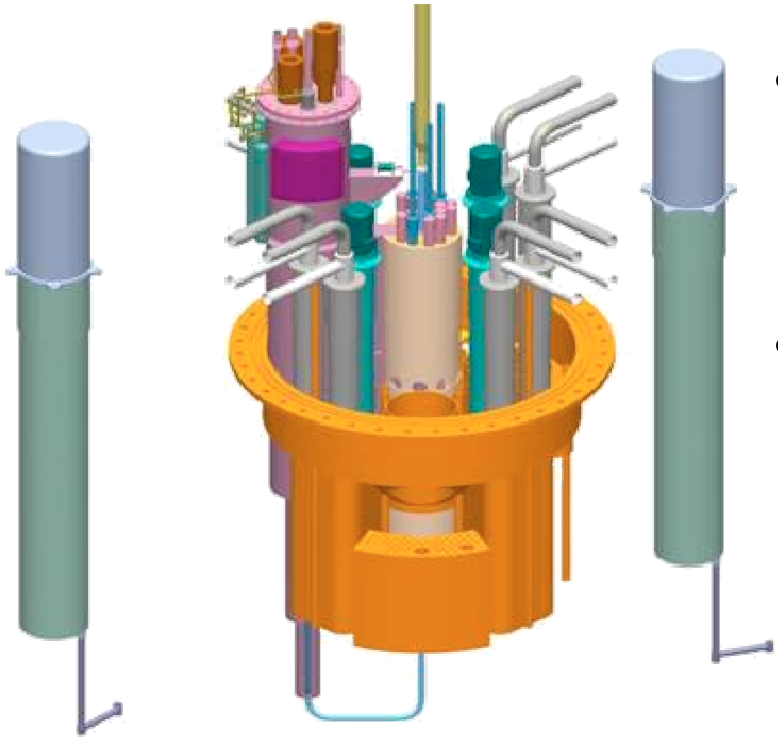
Figure 10. Von Mises stress for thermal load.

Design of MYRRHA
In-vessel fuel manipulators

- The fuel handling is performed *underneath* the core:
  - the room situated directly above the compact core will be *occupied* by instrumentation of the IPS, the beam tube and partially by the spallation loop, with which the fuel handling would interfere if performed from the top of the core,
  - the *interlinking* of the spallation loop with the core makes some fuel assembly positions inaccessible from above,
- The fuel assemblies rest by *buoyancy* force under the support plate.

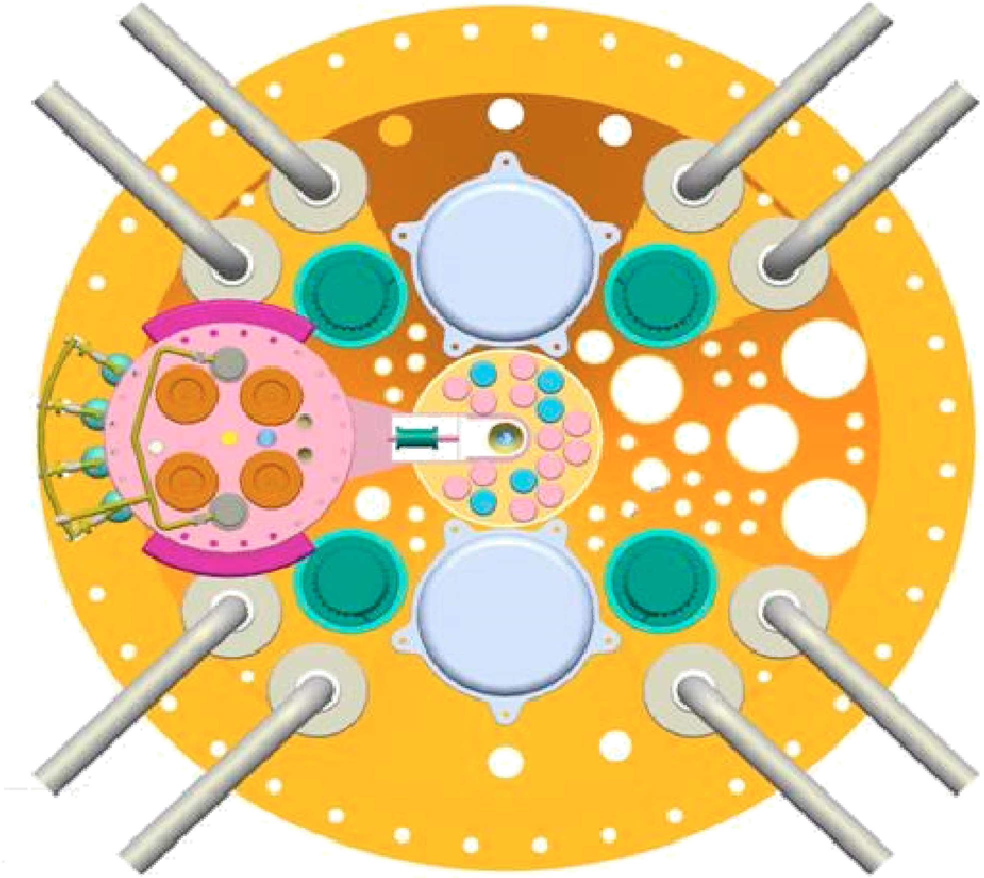
Design of MYRRHA
In-vessel fuel manipulators

- *Two* handling systems are inserted in a penetration of the reactor cover on opposite sides of the core.
- Each system has a *rotating plug*, with an *offset arm*.

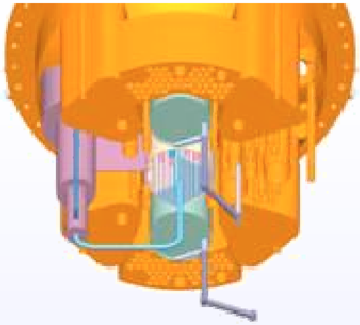


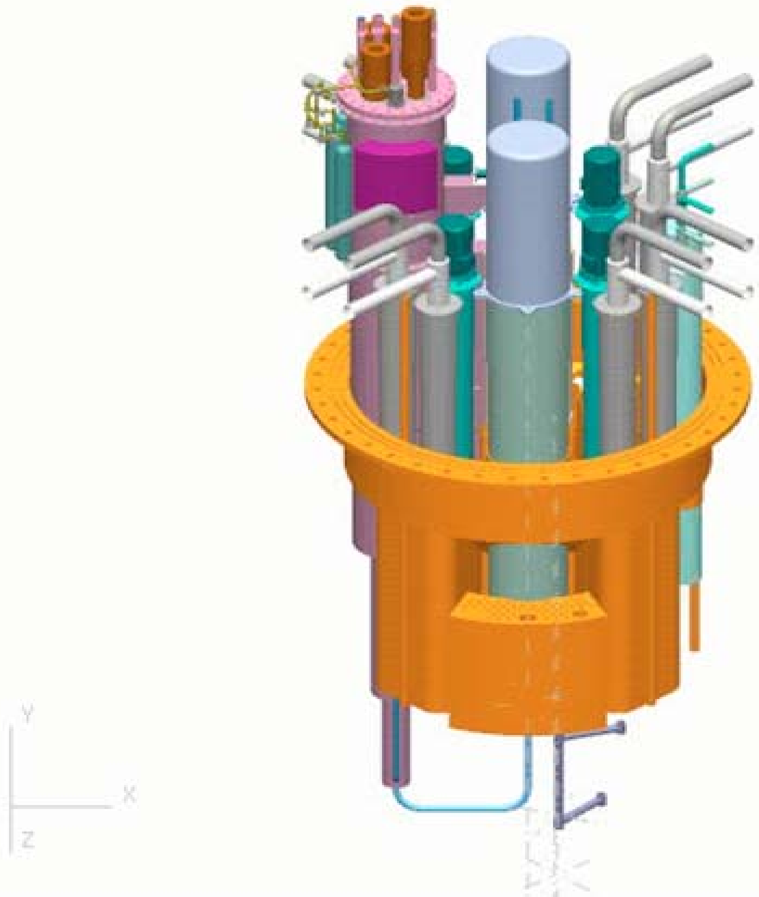
- The arm can rotate in the rotating plug, and so has *access to half of the core*.
- The arm can move up and down by about 2 m to extract the assemblies from the core.

Design of MYRRHA
In-vessel fuel manipulators



Design of MYRRHA
In-vessel fuel manipulators





Design of MYRRHA
Emergency cooling system

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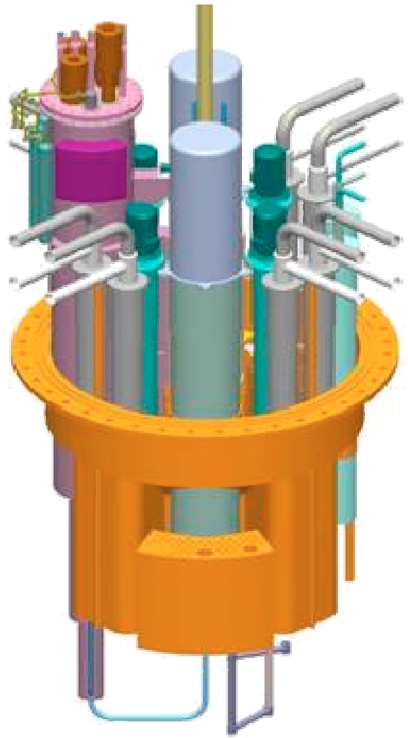
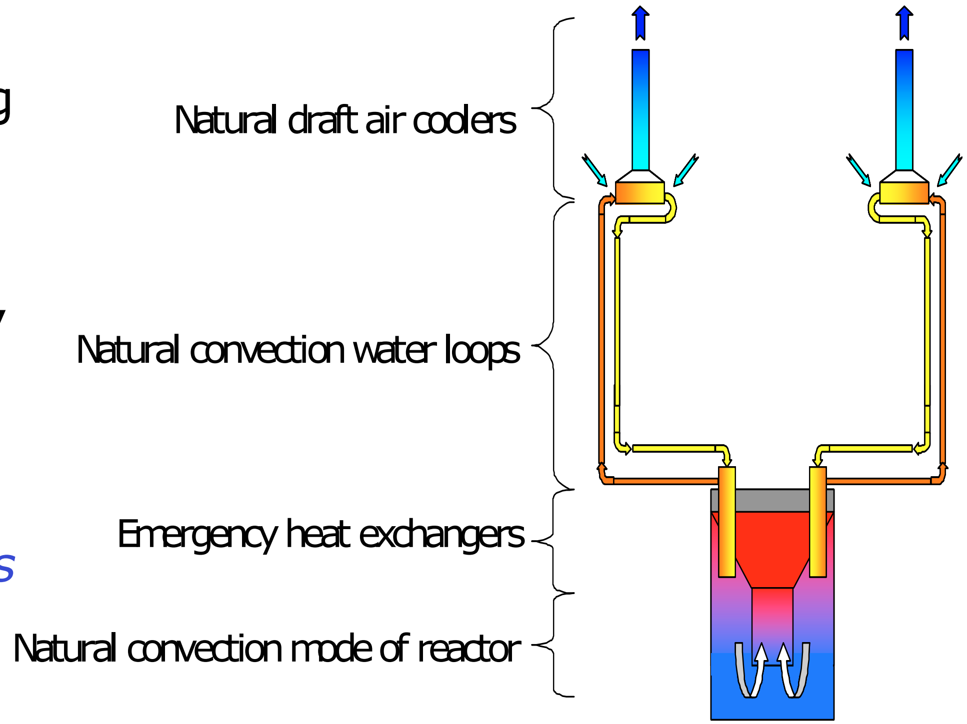
- The *4* MW_{th} heat production after loss of flow and beam shut-off consists of :
  - the *core decay* heat: max. 7% of nominal power after 3 months of operation;
  - the decay heat in the in-vessel *fuel storage*: max. 0.5% after 1 month of maintenance;
  - ^{210}Po decay heat : 0.1% after 3 years of operation.
- There is no other way than the primary coolant for evacuating this heat. Most favourable way = natural convection.

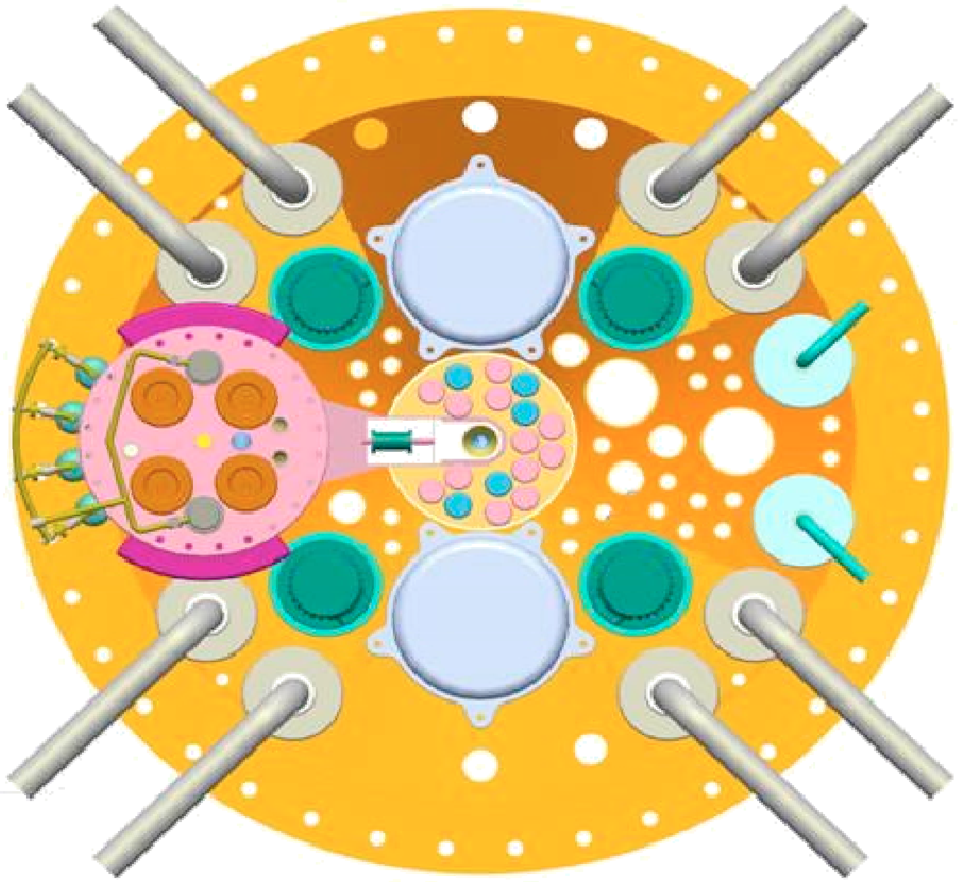


Design of MYRRHA
Emergency cooling system

- *redundant*: two completely independent loops, each one consisting of 3 circuits operating in natural convection mode
- a *passive* system: there are no pumps, no human intervention is required and there are no power operated valves,
- maintain the reactor temperature within safe limits at all times, after *loss of heat sink*.
- Sizing :
  - water/air HX : TE
  - Chimneys : TE
  - LBE/water HX : SCK•CEN



Design of MYRRHA
Emergency cooling system

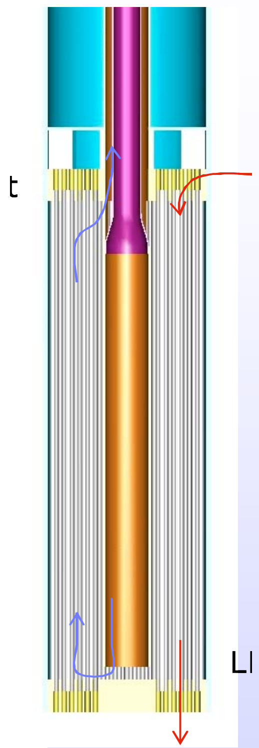


Design of MYRRHA
Emergency cooling system

Sizing LBE/water emergency heat exchangers

- Calculations in Mathcad (several files)
- Compatible with geometries/sizes of aircoolers (TE)
- Calculation note :
  - 300_DM_Calcnote_Emergency-HEX_1.0.0.doc
- LBE properties : Database of thermal properties for melted Lead_Bismut Eutectic – V. Sobolev, Internal report (IR-32-B043-...)
- Water/steam properties : steam tables integrated in Mathcad
- Material properties : "Standards of the Tubular Exchanger Manufacturers Association" Tables D-10, D-11 and D-12, which refer to the ASME codes, Sect. VIII, Div. 2

Design of MYRRHA
Emergency cooling system



Not boiling water EHX :

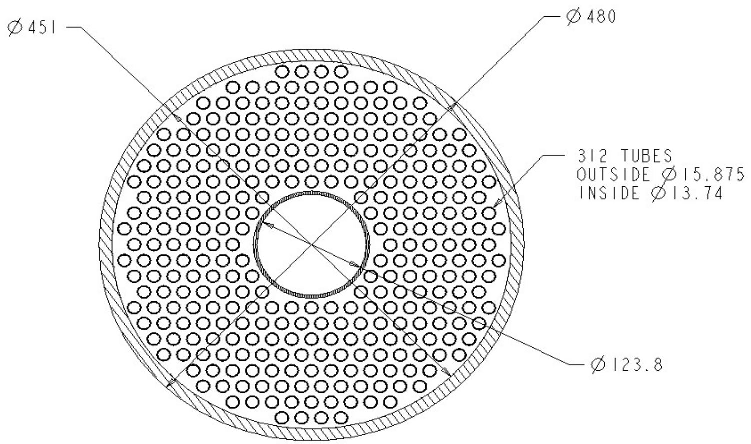
- water outside tubes;
- LBE inside tubes;
- tubes in hexagonal lattice.

Heat transfer correlations :

1) LBE side:

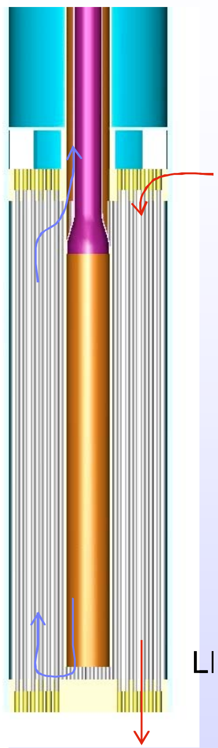
$$Nu = 5 + 0.025 \cdot Pe^{0.8}$$

$$Pe = Re \cdot Pr \qquad Re = \frac{v_i \cdot \phi_i}{\nu_L}$$



EMERGENCY HEAT EXCHANGER
HORIZONTAL X-SECTION

Design of MYRRHA
Primary cooling system



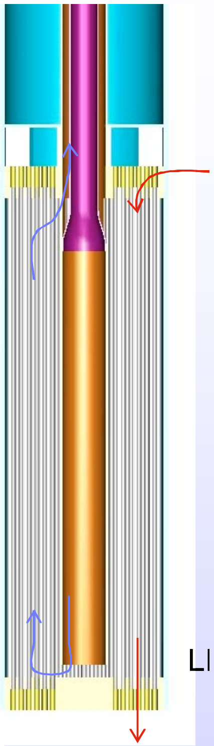
2) WATER side :

$$Nu = \left[0.0165 + 0.02 \cdot \left[1 - 0.91 \cdot \left(\frac{l_p}{\phi} \right)^{-2} \right] \cdot \left(\frac{l_p}{\phi} \right)^{0.15} \right] \cdot Re^{0.8} \cdot Pr^{0.4}$$

$$\phi_e = \frac{\phi}{2} \cdot \left[\frac{4 \cdot \sqrt{3}}{\pi} \cdot \left(\frac{l_p}{\phi} \right)^{2} - 2 \right]$$

Design of MYRRHA Emergency cooling system

Natural convection in LBE and water loop:



$$\left(\rho_{LBE}(T_{Hu}) - \rho_{LBE}(T_{Hi})\right)\cdot g\cdot\left(\Delta H - \frac{L}{2}\right) = \frac{1}{2}\cdot\rho_L\cdot\left[0.6 + \frac{0.15}{(0.4)^2}\right]\cdot\left(\frac{4\cdot M_L}{\rho_L\cdot\pi\cdot D_i^2}\right)^2 \ldots$$

Idel'cik, pag.347
Diagramme 9.15, with
$D_o = D_i$ and $\frac{h}{D_o} = 0.4$

$$+\frac{1}{2}\cdot\lambda_L\cdot\frac{L}{\phi_i}\cdot\rho_L\cdot\left(\frac{M_L}{\frac{\pi}{4}\cdot\phi_i^2\cdot\rho_L\cdot n_b}\right)^2 \ldots$$

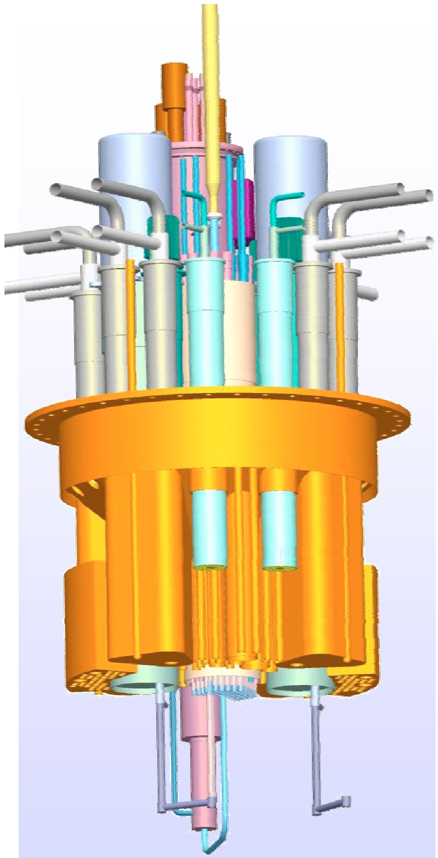
Em.HX pressure drop

$$+1.25\cdot\lambda_c\cdot\frac{1}{2}\cdot\frac{L_f}{D_c}\cdot\frac{M_L^2}{\rho_L\cdot\left(N_a\cdot\Omega_c\right)^2}\cdot(1+\xi)$$

core pressure drop

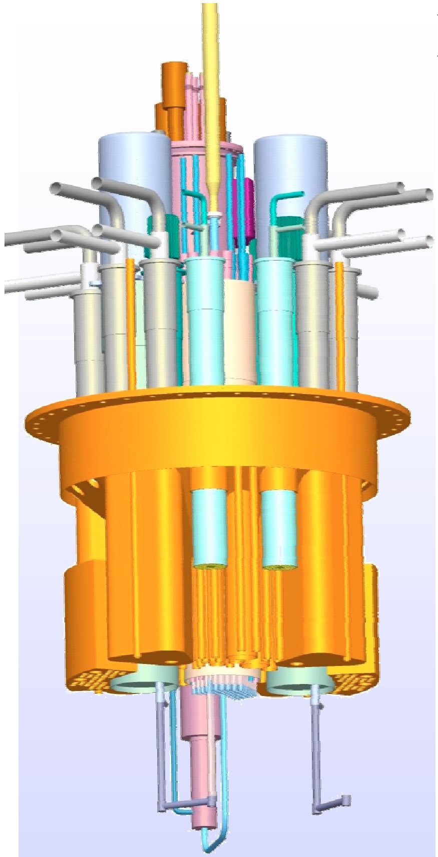
$$\left(\rho_{sw}(p_{min}, T_{Ci}) - \rho_{sw}(p_{min}, T_{Cu})\right)\cdot g\cdot\Delta H_{E.V} = \lambda_{\Delta p}\cdot M_W^2 + \Delta p_{wHX} + \frac{1}{2}\cdot\lambda_{pp}\cdot\frac{L_{pp}}{\phi_{pp}}\cdot\rho_w\cdot\left[\frac{M_W}{\rho_w\cdot\left(\frac{\pi}{4}\cdot\phi_{pp}^2\right)}\right]^2$$

Design of MYRRHA
Emergency cooling system



EMERGENCY HEAT EXCHANGER		
nominal power		4 MW
O.D. shroud		5/8 inch
thickness tubes		0.042 inch
tube pitch		1.4 x 5/8 inch
number of tubes		312
tube length		1.2 m
tube material		T91
I.D. shroud		451 mm
O.D. shroud		480 mm
primary / secondary fluid	PbBi eutectic	water
design conditions (4 MW)	primary	secondary
flow rate	156 kg/s	17.5 kg/s
inlet temperature	385°C	157°C
outlet temperature	209°C	209°C

Design of MYRRHA
Emergency cooling system



- The transients of emergency situations were calculated by RELAP code; some important conclusions are:
  - Protected loss of heat sink and loss of flow (PLOHLOF):
    - No peak in T of fuel and cladding
    - The EHXs are sufficient to cool the fuel with natural convection
  - Unprotected loss of heat sink (UPLOH):
    - Sufficient delay (~1300s) to stop the beam and take appropriate actions;
    - T_{clad} rises to 597°C (allowed 650°C)
    - T_{fuel} rises to 2107°C (allowed 2650°C)

Design of MYRRHA
Inner vessel

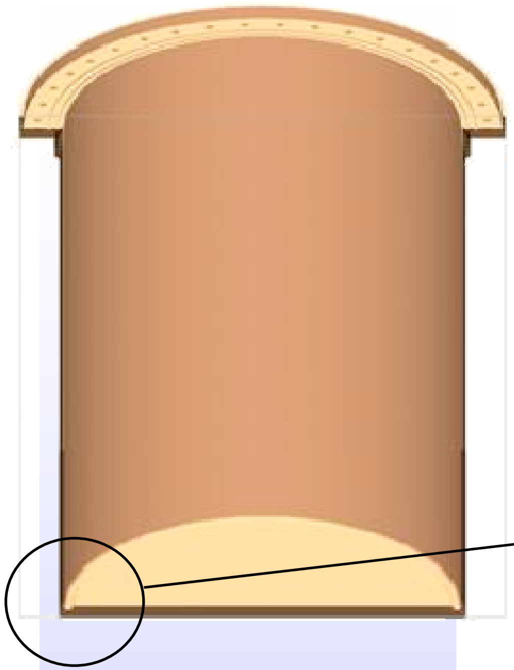
Sizing inner vessel

- Analytical calculations in Mathcad, based on elasticity theory
- FEM calculations (Pro/Mechanica)
- Calculation note :
    - 300_DM_Calcnote_Sizing-Vessel_1.0.0.doc

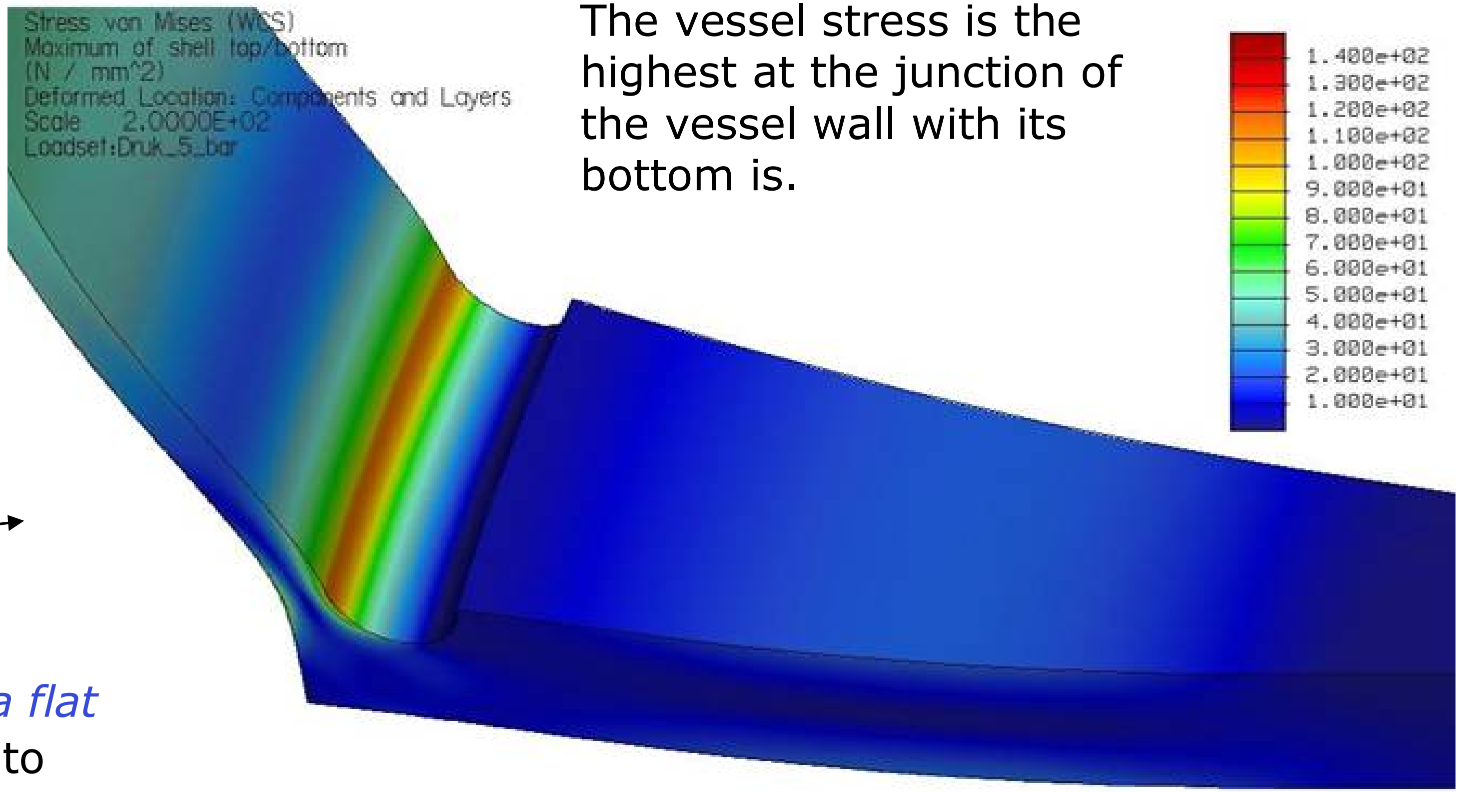
Analytical assessment

- 22 elasticity equations
- together with 22 boundary conditions at junction of vessel wall and its bottom
- ...all stresses in all directions can be assessed.

Design of MYRRHA
Inner vessel



The inner vessel has a flat bottom, which allows to minimize the volume of Pb-Bi in the vessel.



The vessel stress is the highest at the junction of the vessel wall with its bottom is.

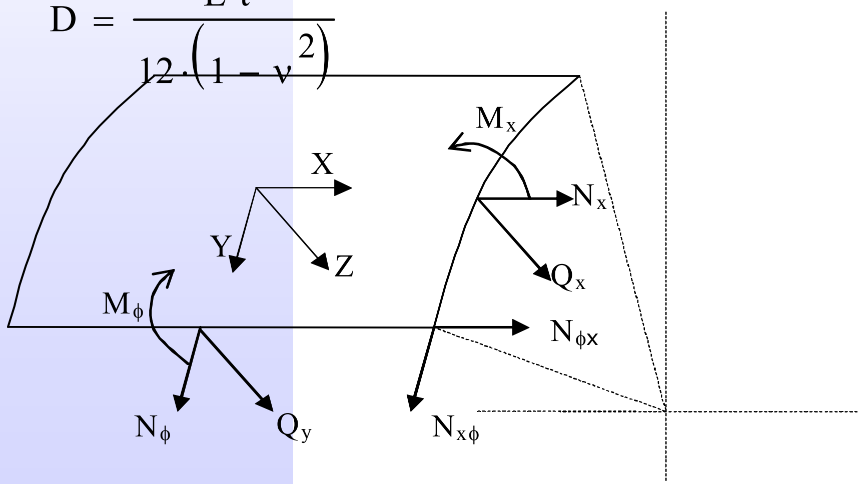
Design of MYRRHA
Inner vessel

Vessel wall

$$\frac{d^4}{dx^4}w + 4\cdot\beta^4\cdot w = \frac{p}{D} + \frac{\nu\cdot N_x - \alpha_t\cdot E\cdot t\cdot T}{R\cdot D} \quad \Rightarrow$$

$$\beta = \sqrt[4]{\frac{3\cdot\left(1-\nu^2\right)}{R^2\cdot t^2}}$$

$$D = \frac{E\cdot t^3}{12\cdot\left(1-\nu^2\right)}$$



$$N_x = \frac{E\cdot t}{1-\nu^2}\cdot\left(\frac{du}{dx} - \nu\cdot\frac{w}{R}\right) - \frac{E\cdot\alpha\cdot T\cdot t}{1-\nu}$$

$$N_\phi = \frac{E\cdot t}{1-\nu^2}\cdot\left(-\frac{w}{R} + \nu\cdot\frac{du}{dx}\right) - \frac{E\cdot\alpha\cdot T\cdot t}{1-\nu}$$

$$M_x = -D\cdot\frac{d^2 w}{dx^2}$$

$$M_\phi = -\nu\cdot D\cdot\frac{d^2 w}{dx^2} = \nu\cdot M_x$$

$$Q_x = -D\cdot\frac{d^3}{dx^3}w$$

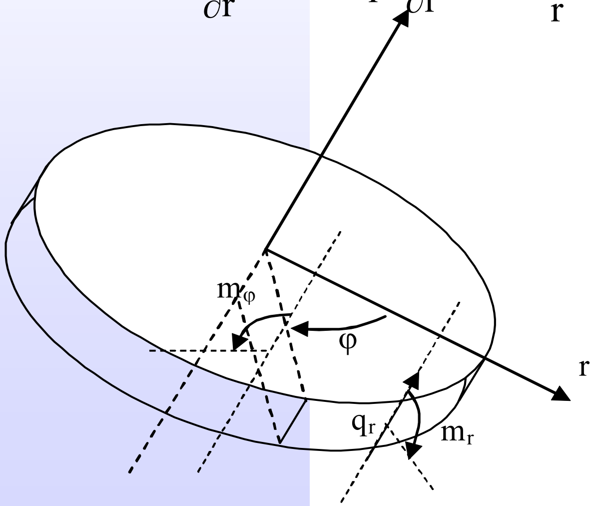
Design of MYRRHA
Inner vessel

Vessel bottom

$$\Delta \cdot \Delta w = \frac{p}{K} \quad \Rightarrow$$

$$\Delta w = \frac{\partial^2}{\partial r^2} w + \frac{1}{r} \cdot \frac{\partial}{\partial r} w + \frac{1}{r^2} \cdot \frac{\partial^2}{\partial \varphi^2} w$$

$$\Delta \cdot \Delta w = \frac{\partial^2}{\partial r^2} \Delta w + \frac{1}{r} \cdot \frac{\partial}{\partial r} \Delta w + \frac{1}{r^2} \cdot \frac{\partial^2}{\partial \varphi^2} \Delta w$$



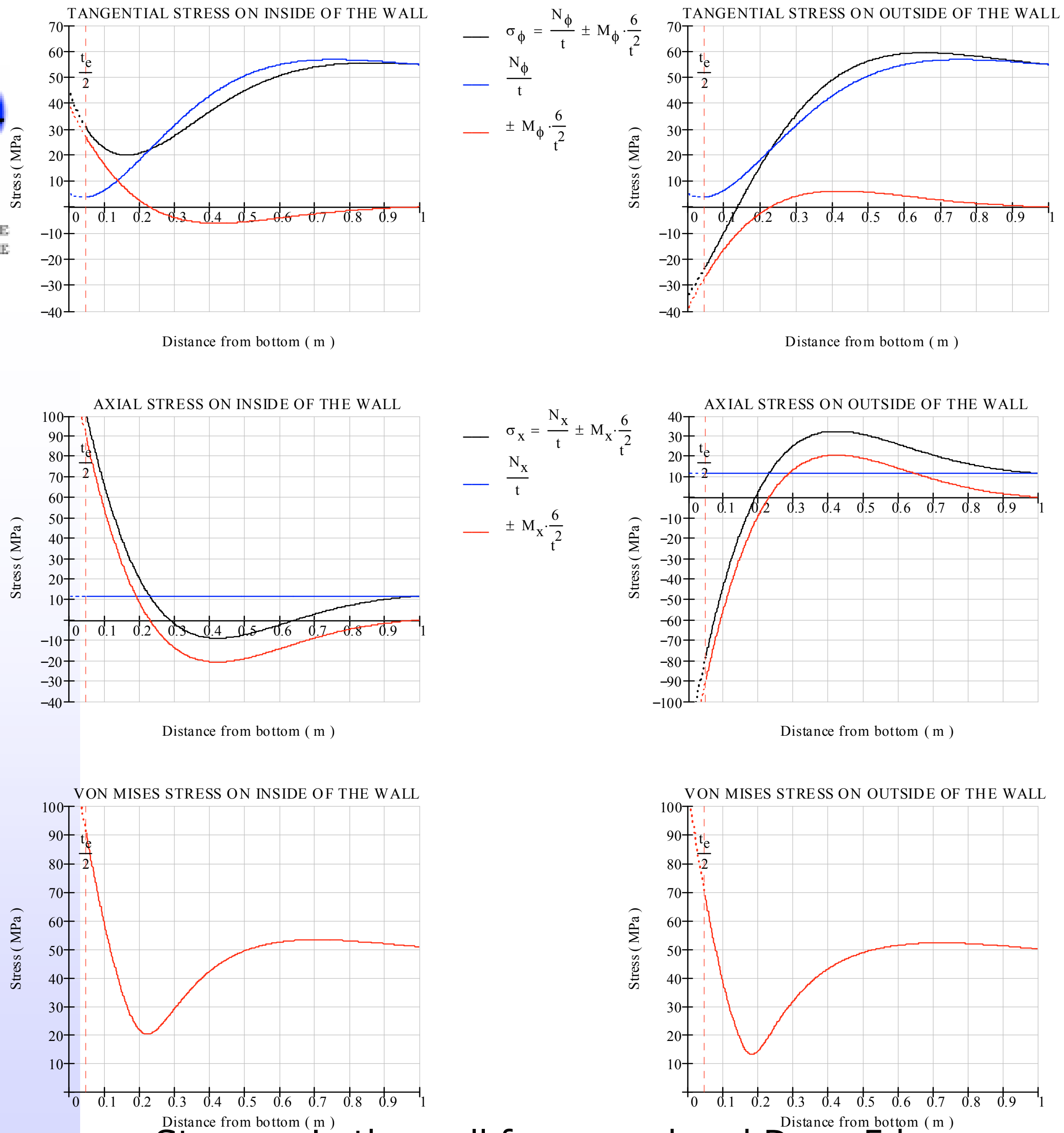
$$m_r(r,\varphi) = -K \cdot \left(\frac{\partial^2}{\partial r^2} w + \nu \cdot \frac{1}{r} \cdot \frac{\partial}{\partial r} w + \nu \cdot \frac{1}{r^2} \cdot \frac{\partial^2}{\partial \varphi^2} w \right)$$

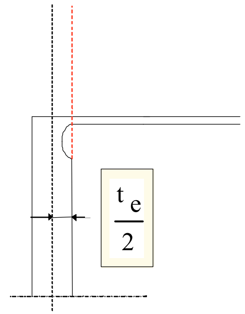
$$m_\varphi(r,\varphi) = -K \cdot \left(\frac{1}{r} \cdot \frac{\partial}{\partial r} w + \frac{1}{r^2} \cdot \frac{\partial^2}{\partial \varphi^2} w + \nu \cdot \frac{\partial^2}{\partial r^2} w \right)$$

$$m_{r\varphi}(r,\varphi) = -(1 - \nu) \cdot K \cdot \frac{\partial}{\partial r} \left(\frac{1}{r} \cdot \frac{\partial}{\partial \varphi} w \right)$$

$$q_\varphi(r,\varphi) = -K \cdot \frac{1}{r} \cdot \frac{\partial}{\partial \varphi} \Delta w$$

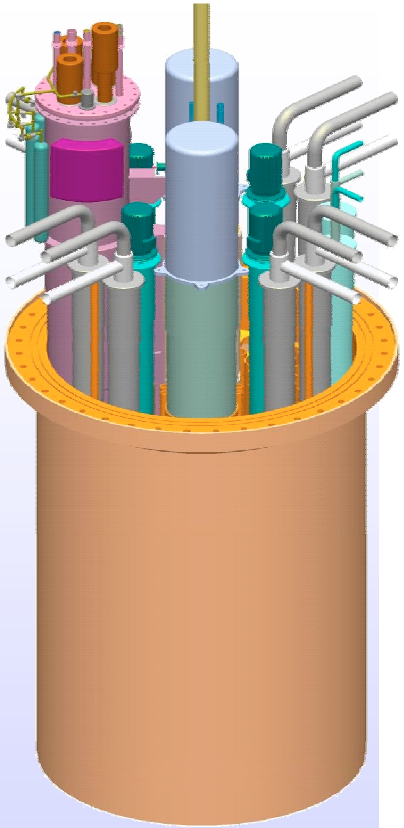
$$q'_r(r,\varphi) = -K \cdot \frac{\partial}{\partial r} \Delta w + \frac{1}{r} \cdot \frac{\partial}{\partial \varphi} m_{r\varphi}(r,\varphi)$$





Stresses in the wall for pump head Dp = 5 bar

Design of MYRRHA
Inner vessel

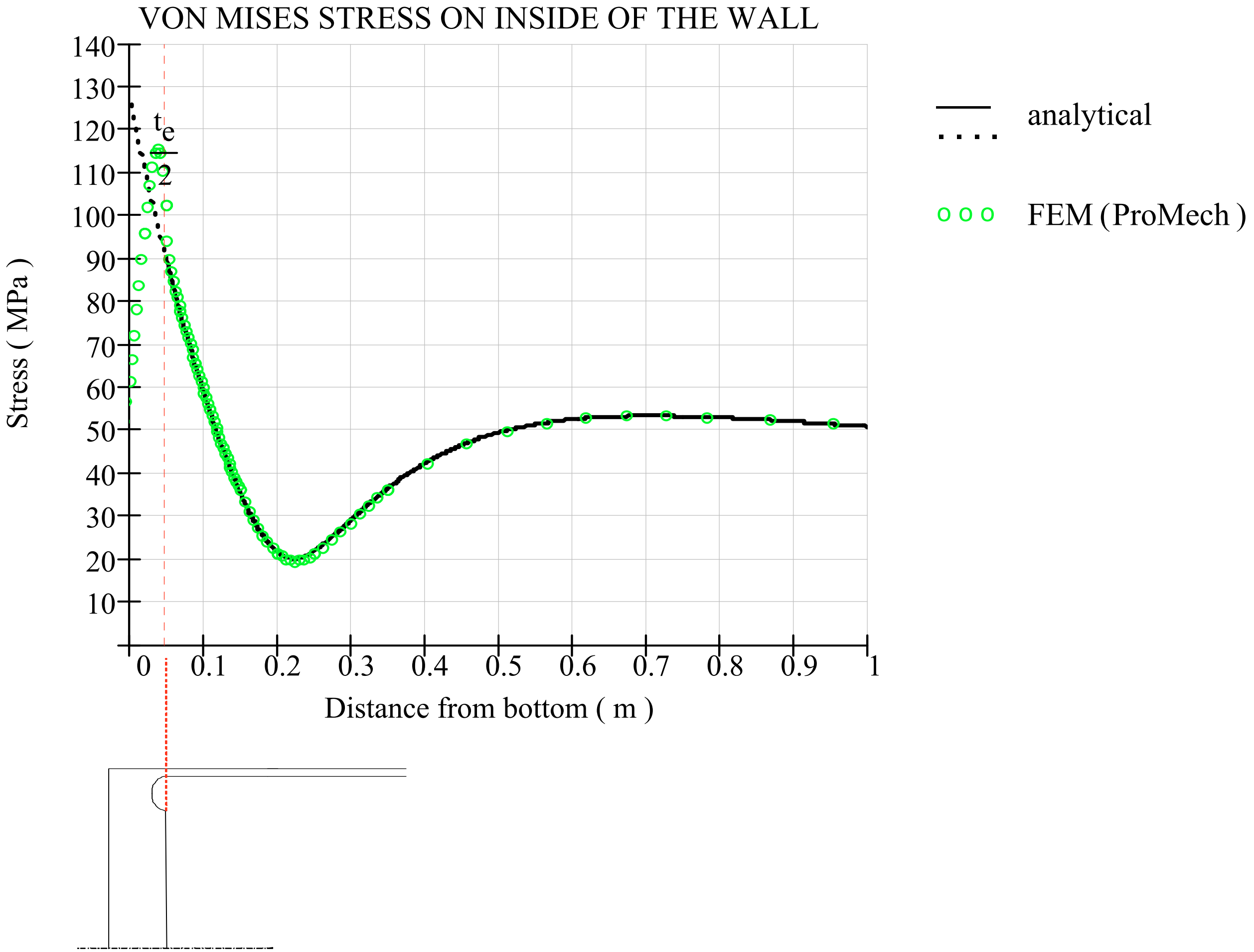


INNER VESSEL		
material		A316 L
inner diameter		4400 mm
outer diameter		4490 mm
thickness bottom plate		150 mm
radius joint wall with bottom		55 mm
flange	outer diameter	5375 mm
	thickness	150 mm
overall height		7000 mm

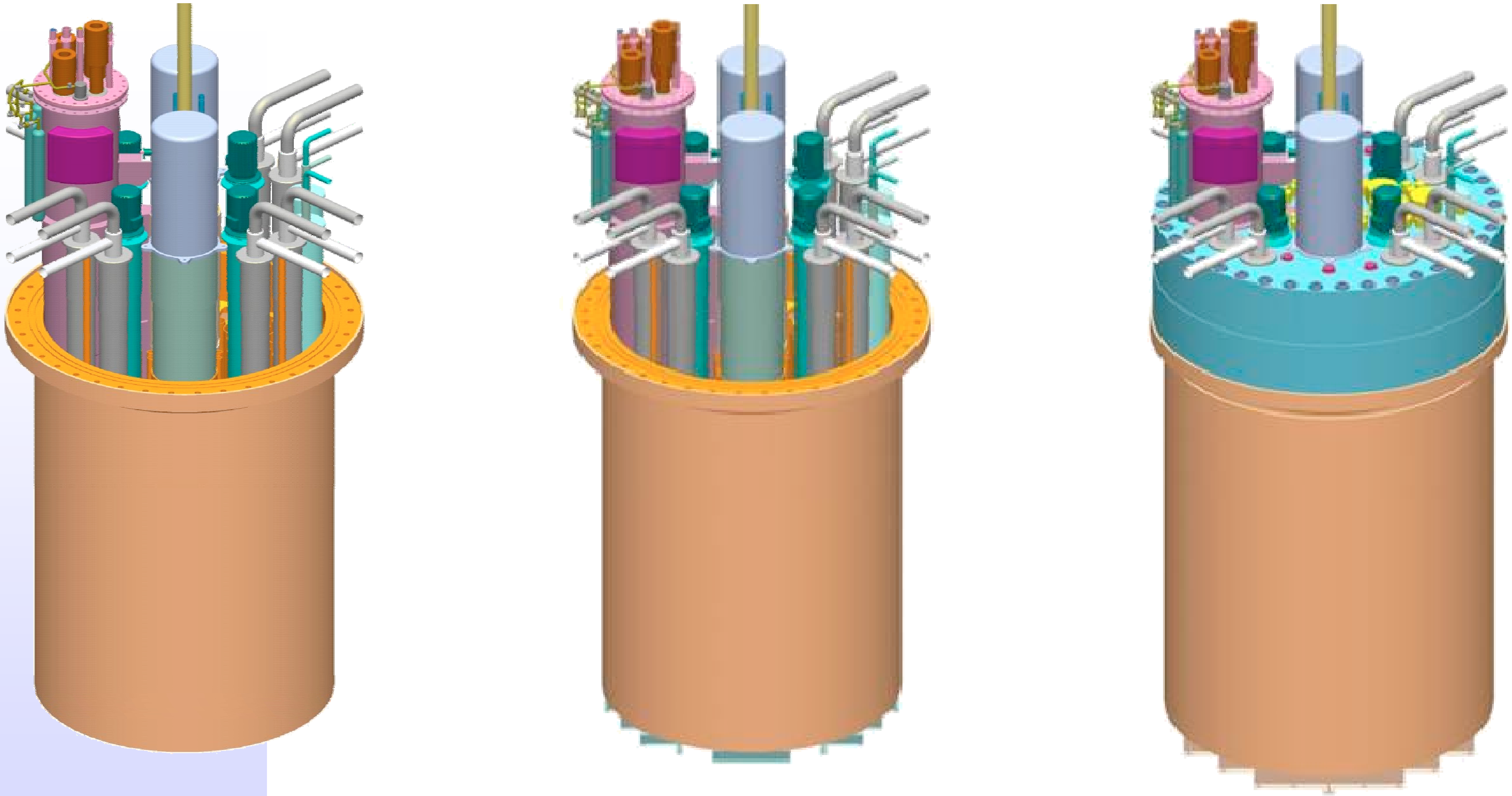
Design of MYRRHA
Inner vessel

Von Mises stress : *analytical* versus *FEM*

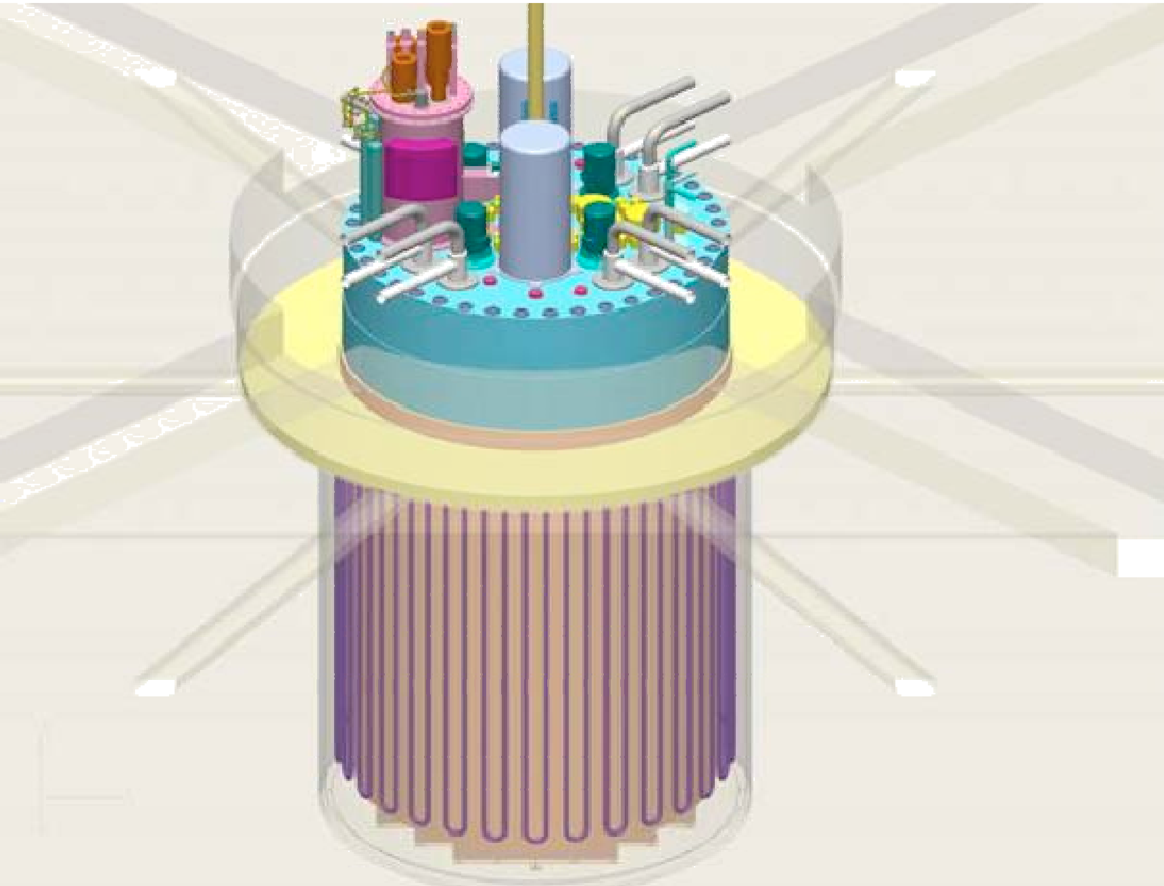
both methods match perfectly.



Design of MYRRHA
Guard vessel and reactor cover



Design of MYRRHA Guard vessel and support structure



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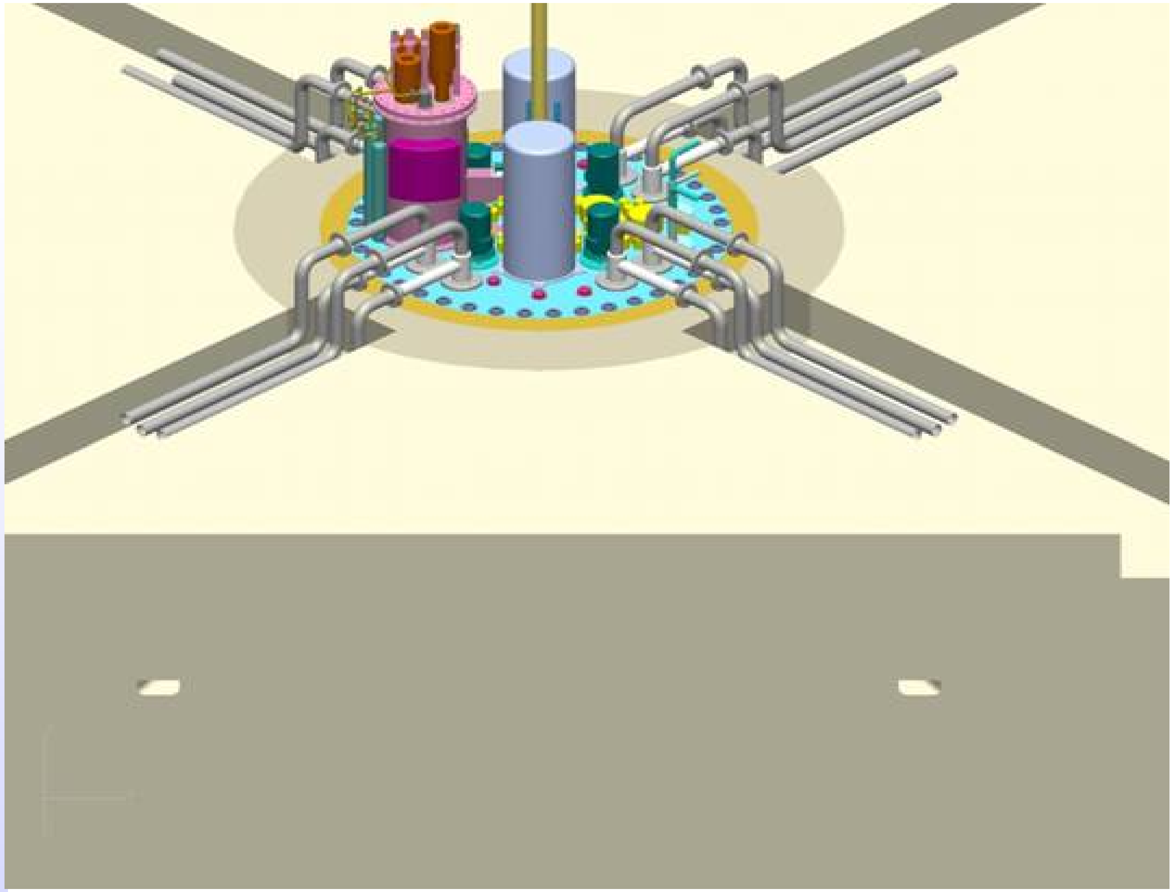
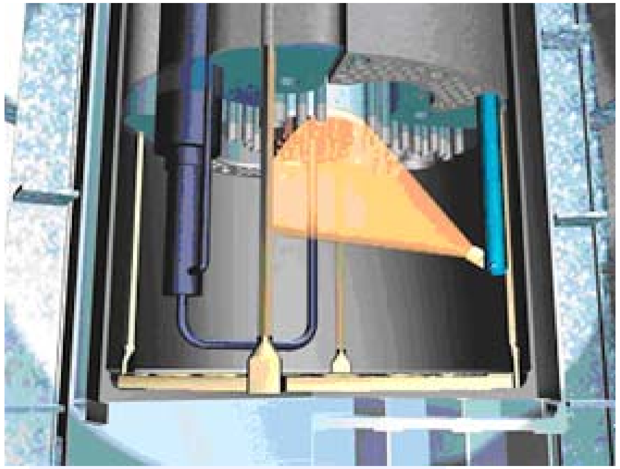
Design of MYRRHA Guard vessel and support structure

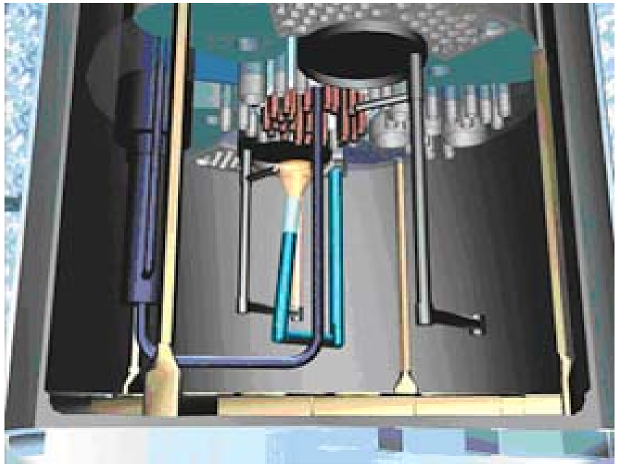


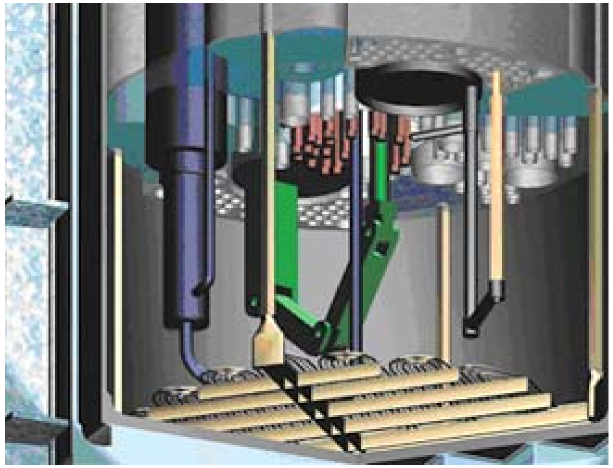
Design of MYRRHA
Remote handling

- Direct access for personnel is highly improbable:
  - *High activation* on top of the reactor due to neutrons streaming through the beam;
  - *^{210}Po (α) contamination* in open times when extracting components during maintenance;
  - The choice of an *oxygen-deficient atmosphere* in the MYRRHA hall limiting the LBE contamination during maintenance.
- Therefore, all in-service inspection & repair (ISIR) and maintenance operations are performed by remote handling, reducing the personnel exposure.
- The MYRRHA remote handling approach has been evaluated by *O.T.L.* on basis of existing and demonstrated technology in the fusion facility JET.

Design of MYRRHA In-Service Inspection & Repair







Two *permanently* installed In-Vessel Inspection Manipulators (*IVIM*) with US camera to provide a *general overview*. (periscope type device with three degrees of freedom)

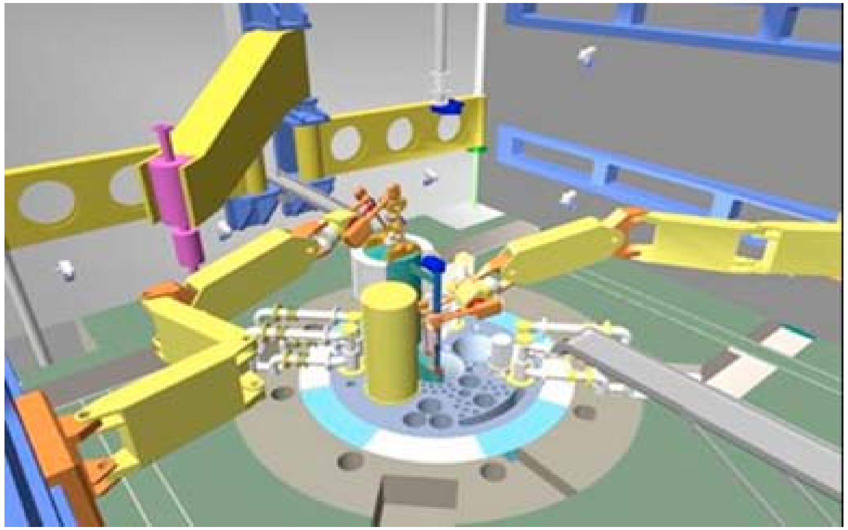
Another *IVIM* positions the camera close to critical components for *detailed* inspection. (anthropomorphic type device with five degrees of freedom)

The *repair* manipulator recovers debris or deploys specialised tooling for repair. (anthropomorphic type device with eight degrees of freedom)

O.T.L. concludes positive on the feasibility of the proposed RH approach.

Design of MYRRHA
Ex-vessel remote handling

- All ex-vessel maintenance operations are performed by a remote handling system, which is based on the *Man-In-The-Loop principle*:
  - force reflecting servo-manipulators
  - Master-Slave mode: the slave servo-manipulators are com-manded by remote operators using kinematically identical master manipulators
  - supported with closed-cycle TV (CCTV) feedback
- The arms are equipped with dedicated *tooling* for completing all classical maintenance and repair (cutting, welding, screwing, ...)



O.T.L. concludes positive on the feasibility of the proposed RH approach.

MYRRHA in the European frame

- MYRRHA design is opened to a larger European community in order to better meet the objectives of the *eXperimental ADS for Transmutation (XT-ADS)* in the frame of the integrated FP6 project EUROTRANS.
- **2005-2008** : potential obstacles towards realisation of an XT-ADS and of an industrial ADS later on the *European Facility for Industrial Transmutation (EFIT)* should be overcome.
  (selection of appropriate *materials* in contact with LBE, solve LBE *conditioning* problems in pool conditions, development of *instrumentation* (O_2 meters, US sensors for visualisation, free surface monitoring, LBE velocity measurement), demonstration of *accelerator reliability*, complete spallation loop design, coupling of ADS components (accelerator, SL, core) should be realised, design of XT-ADS progressed for addressing licensing authorities).
- **2008**: a dedicated project should be initiated for the construction of the XT-ADS that would last 10 years for bringing the project to the full power operation.
- Feedback experience will be used to design the industrial ADS prototype EFIT that would be constructed during the period 2025-2030. After ten years of operation the deployment at industrial scale would be possible.

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Design of MYRRHA Conclusions

MYRRHA

- The MYRRHA design provides an extremely *flexible core management* for the fuel assemblies and for the experimental irradiation devices due to
  - the fuel handling from underneath the core;
  - the off-centre positioned spallation confinement vessel;
  - the pool-type reactor.
- The design of the primary & emergency cooling circuits assures a *safe and adequate cooling* of the sub-critical core.
- Sufficient *resistance against corrosion/erosion* is obtained by limiting the LBE velocity below 2.0 m/s.
- All in-vessel components can be removed and exchanged by *remote handling*, which reduces the personnel exposure.
- MYRRHA is now open to a larger European community in the frame of the integrated FP6 project *EUROTRANS* (XT-ADS).

Design of MYRRHA Conclusions (cont'd)

- The studies so far definitely have shown no insuperable difficulties and it is demonstrated that the main components can be sized within allowable stress limits to fulfil their task within safety limits.
- A visualisation system based on ultrasonic technology is proposed for the in-vessel, under-LBE inspection. A support R&D programme has been launched.

Design of MYRRHA
Draft 2 Conclusions

- The conceptual design of MYRRHA primary system and associated equipment remains similar to the "Draft – 1" version (2002) but differs in:
  - the omission of the water tank around the outer vessel (according to recommendations of the ITGC in 2002);
  - the additional cooling system surrounding the outer vessel;
  - many dimensional adjustments following the increased power level (50 MWth instead of 30 MWth);
  - and the introduction of a passive emergency cooling system.
- Some components of the primary system (vessel, primary heat exchangers, emergency heat exchangers, diaphragm) were assessed more profoundly and some alternatives were suggested (heat exchangers with boiling water or other liquids, various vessel constructions). The studies definitely show no insuperable difficulties and it is demonstrated that these components can be sized within allowable stress limits to fulfil their task within safety limits.
- Nevertheless, many questions remain (e.g. liquid Pb–Bi metal related problems as corrosion/erosion) and they must be answered by the R&D findings. The knowledge of the structural materials behaviour in flowing liquid Pb–Bi will influence the size of the components (e.g. the heat exchangers) and they will have to be revisited in the future.

Design of MYRRHA
Draft 2 Conclusions (cont'd)

- Other primary system components (e.g. the primary pumps and the MHD pump for the spallation loop) are still under study and the dimensioning has been assessed so far in a more or less rough way because no much information of those particular components is available in the industry or in the literature.
- We mention here that the feasibility of the remote handling (RH) system (see also chapter "Operation, Inspection & Maintenance") has been investigated by O.T.L. (Oxford Technologies Limited). This system covers the In-Service Inspection & Repair and the ex-vessel manipulation. Important is that O.T.L. concludes positive on the feasibility on the proposed RH approach.
- Many other components or systems still remain under design considerations (e.g. the in-vessel and ex-vessel fuel transfer machines, components handling, maintenance, fuel hot cells, secondary and tertiary cooling system, instrumentation, Pb–Bi conditioning system, …) of which no further progress has been made, simply because of lack of manpower and time. There is still much work to do!

Design of MYRRHA
Draft 2 Conclusions (cont'd)

In the short term, it is suggested :

- to reassess the diaphragm with more realistic boundary conditions and/or cooling fins;
- to resize the reference heat exchangers with T91 for tube material instead of Croloy 2 ¼ because of possible corrosion problems;
- to resize the reference heat exchangers taking into account high thermal resistant oxide layers which are formed on the tube surfaces, plus taking into account fouling on the water side of the tubes;
- to design the diaphragm perforations for high hydraulic resistance to minimise the leakage; to assess those leakages;
- to reconsider the water tank as shielding around the outer vessel: it is very unlikely (even impossible) that water would enter the vessel that is filled with Pb–Bi which has a density 10 times more than water;
- to size the primary pumps by means of the similitude theory; to accurately assess the Pb–Bi pressure drops in the pool;
- to get rid of the turbine pump in the spallation loop: it will simply bring more complexity and failure possibilities; a long shaft with appropriate bearings is suggested;
- to design the MHD pump of the spallation loop;
- to design the in-vessel fuel manipulator;
- to design the core instrumentation integration & handling from the top of the core;

MYRRHA – Draft 2
System Operation, Inspection & Maintenance

H. Aït Abderrahim, D. De Bruyn, P. Baeten,
W. Haeck & D. Maes

On behalf of MYRRHA team and MYRRA support

http://www.sckcen.be/myrrha

Workshop on "Technology and Applications of Accelerator Driven Systems" Trieste, 17-28 October 2005

Summary

- Introduction
- Working Regime of MYRRHA
- Reactivity monitoring approach in MYRRHA
- I&C approach in MYRRHA
- ISI&R approach in MYRRHA
- Conclusions

Introduction

- MYRRHA is thought and designed as an experimental facility
- An experimental irradiation facility has a short operation cycle in order to allow loading and retrieval of irradiated devices on very regular and flexible manner
- an availability rate of 65% is targeted (3 MO +1 MSShD)*2 +3 MO + 3 MLShD)
- Advantages of such short cycle:
  - ➢ Preventive maint. On Accel. => improve Reliab.
  - ➢ Δk_{eff} small => afford I_p ct. during cycle => ease licensing

➢ ISI&R via RH & Robotics needed to achieve 65% avail.

Working Regime of MYRRHA (1)

- 3 MO +1 MSShD)*2 +3 MO + 3 MLShD:
  - 3 Months Operation:
    - ✓ Δk_{eff} small (-~1700 pcm) : can be compensated by core reshuffling, burnable absorber or combination
    - ✓ Allow good irradiation results for material damage (7 to 15 dpa/cycle), MA transmutation (>> than Chemical. Measurement), Fuel BU (10 GWd/t)
    - ✗ More challenging for short irradiations such as radioisotope production but feasible (need further design work)

Working Regime of MYRRHA (2)

- 3 MO +1 MSShD)*2 +3 MO + 3 MLShD:
  - 1 Month Short Shut Down :
    - ✓Partial Core reload (few fresh FA) and reshuffling
    - ✓Experimental devices handling
    - ✓Routine inspection
  - 3 Months Long Shut Down :
    - ✓larger Core reload and reshuffling
    - ✓Fuel transfer from the in-vessel storage
    - ✓Experimental devices handling
    - ✓Heavy maintenance such as Spallation Target extraction and parts replacement

Reference Core

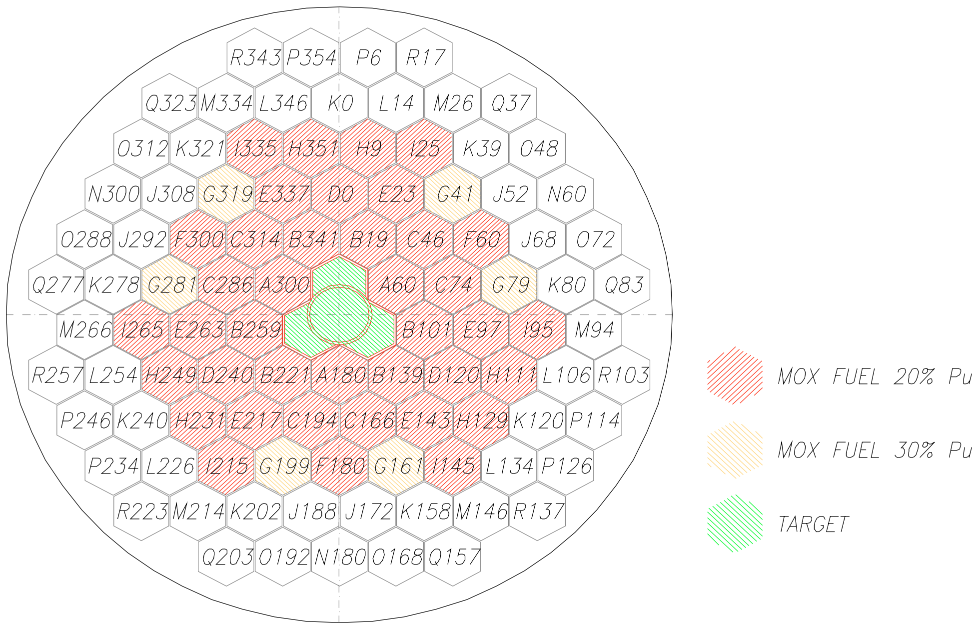
The reference core consists of 45 fuel assemblies:

- 39 assemblies with 30% MOX (positions A-F, H-I)
- 6 assemblies with 20% MOX (position G)

Targeted operating regime:

- 90 days of operation
- 30 days for maintenance, …
- 3 cycles a year

k_{eff}	0.94589
k_s	0.95236
ρ [pcm]	-5721
ρ_s [pcm]	-5002
P [MWth]	43



Calculations

Burn up calculations were performed using:

- MCNPX 2.5.d2
- ORIGEN 2.2
- ℵ SPECTRUM 1.0a

Every assembly is divided into 5 segments – every segment has a different burn up library calculated by ℵ SPECTRUM

A cycle of 90 days is subdivided into intervals of 15 days – MCNPX calculates the total flux for depletion calculation in ORIGEN 2.2

JEF2.2 nuclear data is used in all calculations

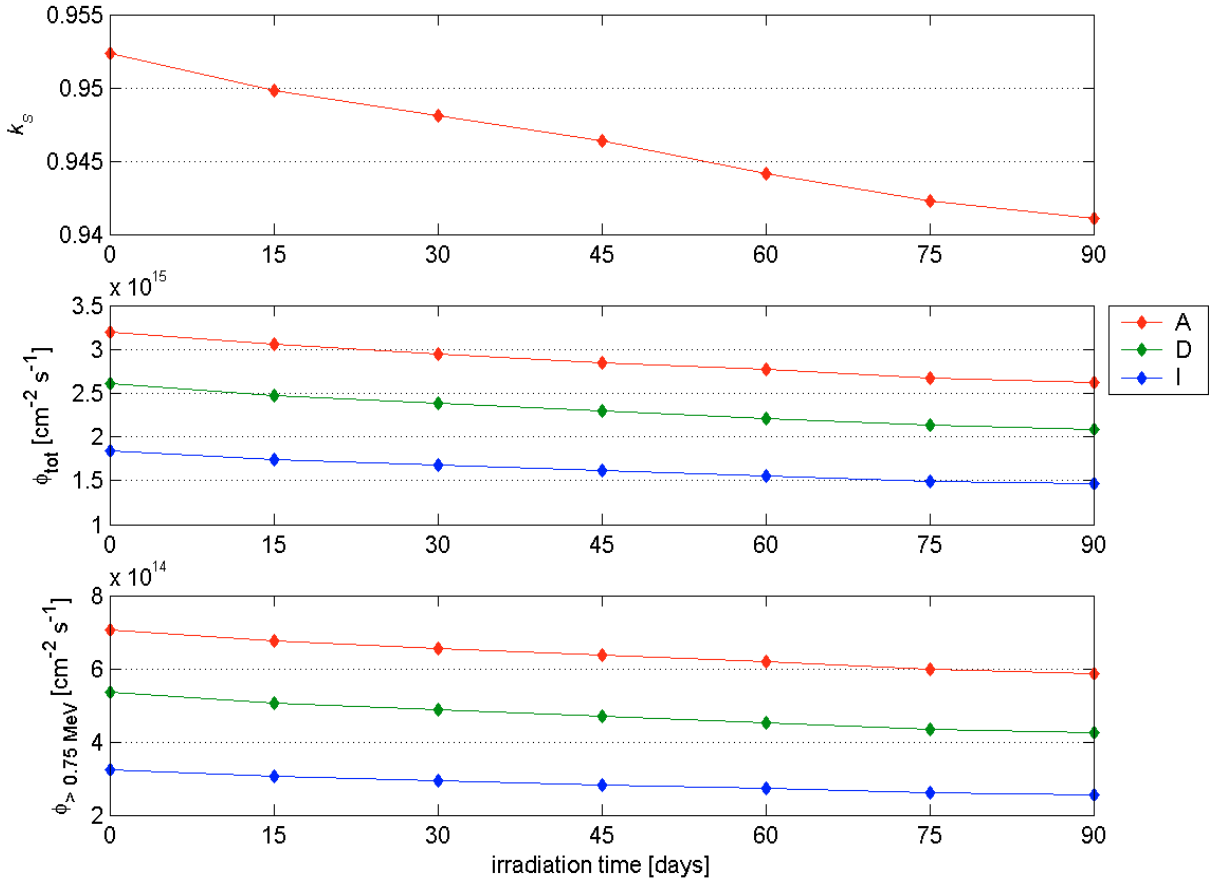
Burn up results

	BOC	EOC
k_s	0.95236	0.94105
k_{eff}	0.94589	0.93279
ρ_s [pcm]	-5002	-6265
ρ [pcm]	-5721	-7205
P [MWth]	43	34

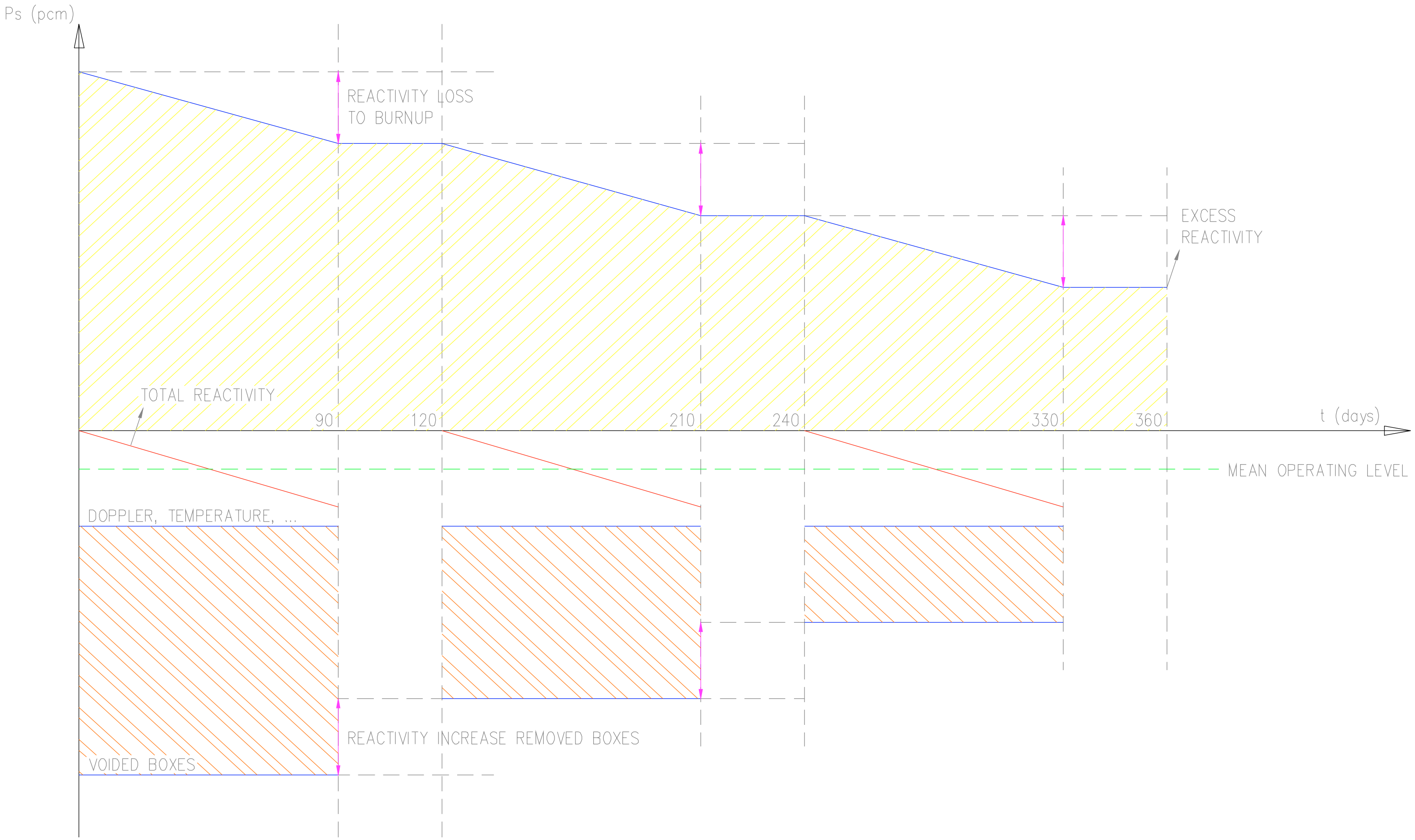
Reactivity loss:

$$\Delta\rho_s = -1263 \text{ pcm}$$

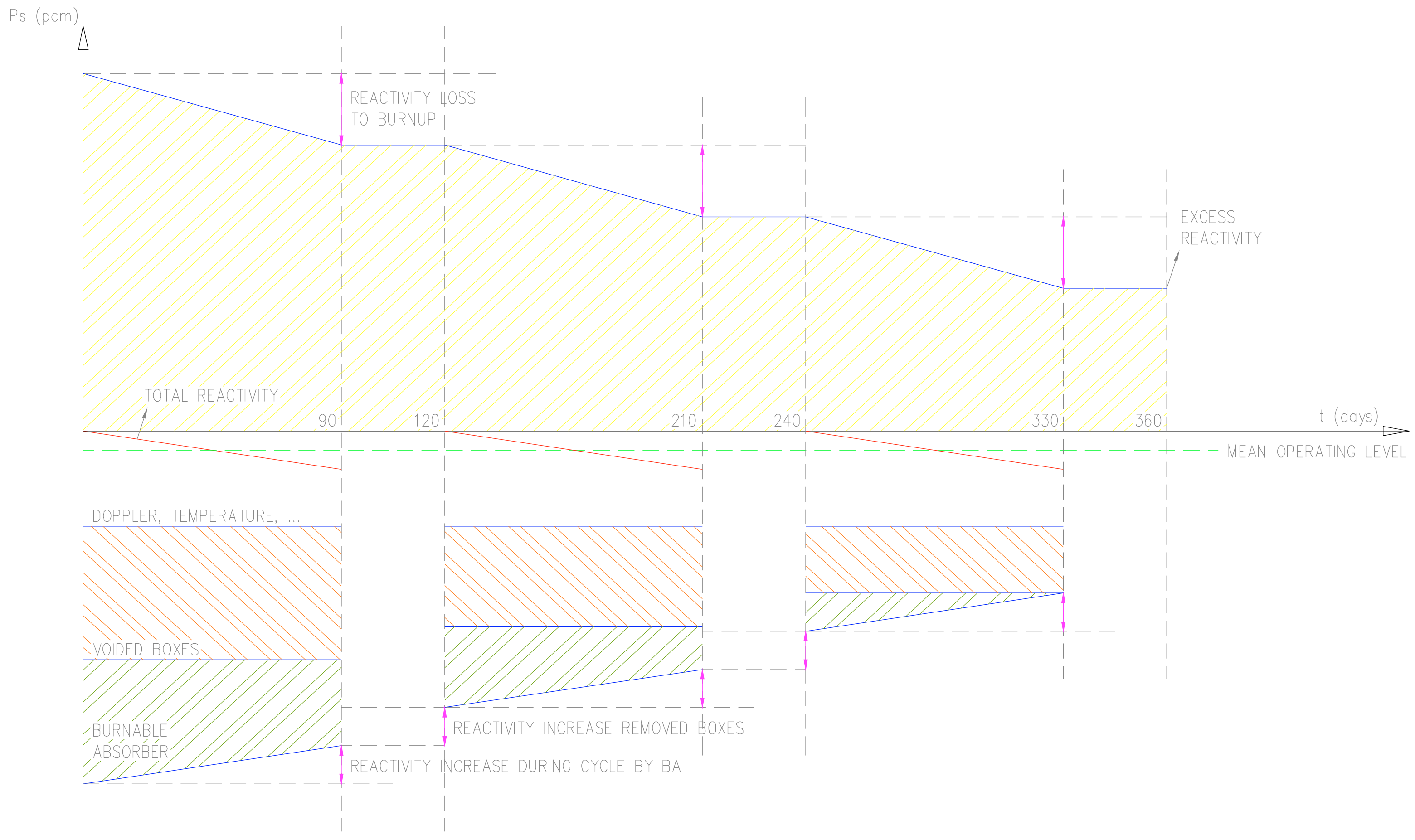
$$\Delta\rho = -1484 \text{ pcm}$$



Reactivity Compensation Using Voided Boxes



Reactivity Compensation Using Voided Boxes and Burnable Absorber

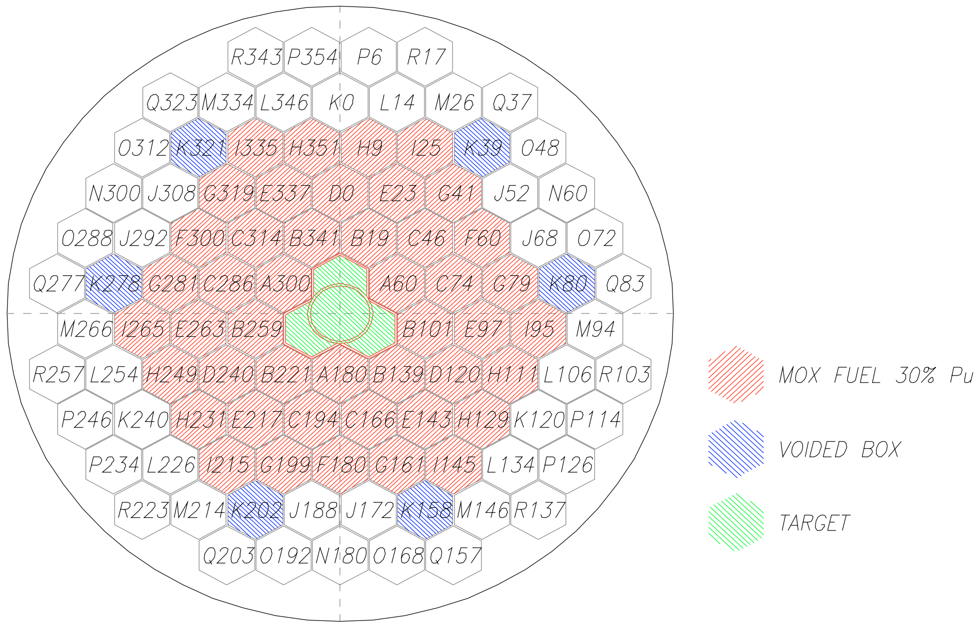


Modified Core

To demonstrate the previous concepts, we performed burn up calculations on a modified core:

- Replaced 20% MOX by 30 % MOX: $\Delta\rho_s = +1783$ pcm
- Added 6 voided box assemblies: $\Delta\rho_s = -1421$ pcm

	Reference core	Full 30% MOX core	Adding voided boxes
k_{eff}	0.94589	0.96614	0.94969
k_s	0.95236	0.96881	0.95565
ρ [pcm]	-5721	3505	-5298
ρ_s [pcm]	-5002	-3219	-4640
P [MWth]	43	69	46



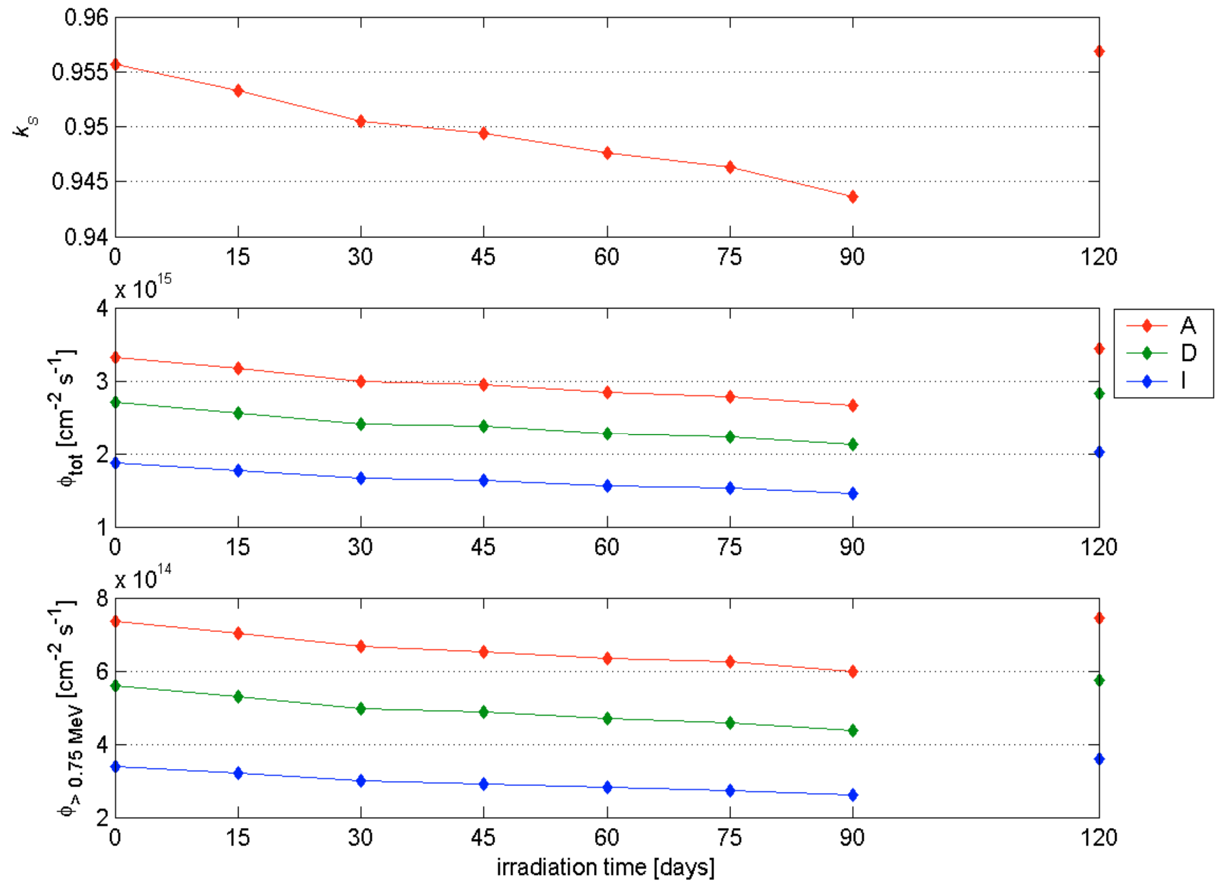
Burn up results

	BOC1	EOC1	BOC2
k_s	0.95565	0.94367	0.95682
k_{eff}	0.94969	0.93611	0.95118
ρ_s [pcm]	-4640	-5969	-4513
ρ [pcm]	-5298	-6825	-5133
P [MWth]	46	35	48

Reactivity loss cycle 1:

$$\Delta\rho_s = -1329 \text{ pcm}$$

$$\Delta\rho = -1527 \text{ pcm}$$



Conclusions For BU swing Mgt

During a cycle of 90 days, the power and flux in MYRRHA drop by 20% on average

The first burn up results demonstrate that:

- the proposed operational cycles are realistic
- in the case studied, no new fuel assemblies are needed in the second cyle to obtain the same operational level of the first cycle

Further study:

- introducing BA into the core – combined with voided box assemblies
- burn up calculations of multiple cycles

Reactivity Monitoring Techniques

- "Reference" method in critical reactors
  - Rod-drop → MSA/MSM
- Continuous monitoring techniques
  - Current-to-flux indicator
  - Harmonic source oscillation
- Pulsed Neutron Source methods
  - Fitting method
    - 1, 2, 3 exponentials
    - Kp-method
  - Area method
- "Source Jerk" type techniques
  - Standard Source Jerk technique
  - Source Modulation method
  - ADS Source Jerk technique (Beam trips)
- Noise Techniques
  - Rossi-alpha
  - Feynman-alpha
  - APSD, CPSD
  - Cf source driven method
- Reactivity monitoring philosophy in ADS

Reactivity values obtained with different experimental techniques in MUSE

	Rod drop + MSA	Rod drop +MSM	PNS fitting			Kp method	PNS Area		Source Jerk	Source mod.	Rossi-α (Area)
			1 exp	*2 exp*	*3 exp*		*SCK*	*CIEMAT*			
SC0	-1.9	**-1,86**	**-1,93**	**-1.92**		**-2,2**	**-2,00**	**-1.96**	**-1,92**	**-2,18**	**-2.04**
SC2 - 1006 cells	-9.1	**-8,7**			**-8.7**		**-8,9**	**-8.5**			**-8.8**
SC2 - 1004 cells	-9.7	**-9,1**				**-9,7**				**-9,7**	
SC3	-14.1	**-13,6**		**-15.6**	**-11,7**	**-14,1**	**-13,7**	**-12,3**	**-14.6**		**-13,4**

Current-to-flux reactivity indicator (1)

- Subcritical multiplication

$$\varphi = c\frac{I}{1-k_{eff}} = c\frac{I\,(1-\rho_{PI})}{-\rho_{PI}}$$

$$\Rightarrow \quad \rho_{PI} = -\frac{c\dfrac{I}{\varphi}}{1-\dfrac{cI}{\varphi}} \approx -c\frac{I}{\varphi}$$

- Characteristics
  - On-line measurement
  - Current- and flux measurement
    - Well-known technology
    - Very good relative accuracy < 1% ?
    - Simplicity of the current & flux measurement:
      - No data-treatment
      - No additional time constants are introduced during the measurement
  - Close to critical reactor instrumentation

Current-to-power reactivity indicator (2)

- Characteristics (continued)
  - Sensitivity to the actual source multiplication and not to the effective multiplication factor
    - Conversion to effective neutron multiplication by source importance factor via interim calibration

$$\frac{\rho_2}{\rho_1} = \frac{\varepsilon_2}{\varepsilon_1}\frac{\varphi_2^*}{\varphi_1^*}\frac{C_1}{C_2} = \frac{\varepsilon_2}{\varepsilon_1}\frac{\overline{\Psi_{f_1}}}{\overline{\Psi_{f_2}}}\frac{C_1}{C_2} = \frac{\langle \sigma_D \varphi_s \rangle_2}{\langle \sigma_D \varphi_s \rangle_1}\frac{\langle \varphi_0^+, \nu F \varphi_s \rangle_1}{\langle \varphi_0^+, \nu F \varphi_s \rangle_2}\frac{C_1}{C_2} = f_{MSM}\frac{C_1}{C_2}$$

  - Absolute reactivity determination: accuracy: <10%
    - Depends on the calibration method
      - Interim cross-checking: response to reactor trip
      - “zero”-power calibration: PNS, Noise techniques
  - Relative reactivity determination:
    - Precision on current & flux measurement: <1% ?
    - 10% change in reactivity gives a 10% change in reactivity indicator
    - Precision on reactivity indicator: 1%
    - Accuracy: depends on the monitoring of the stability of parameters influencing the source importance
      - ♥ Spallation source position: axial and horizontal
      - ♥ Proton energy

Needs for ADS criticality and reactivity monitoring

- On-line and continuous sub-criticality monitoring
- Low uncertainty between detection and real effects
- Robust absolute reactivity assessment

Methodology for on-line reactivity monitoring and Absolute reactivity assess

- Step-wise approach
- **Current-to-power indicator as an on-line indicator**
  - Uncertainty on relative deviations of about 1%.
  - Proportionality constant is checked regularly by interim cross-check
- **Interim cross-check**
  - verification of the proportionality constant of the current-to-power indicator:
    - Proton beam tripping
    - Slope fitting technique (Kp ?, exponential Fitting : 2 or 3 ?)
    - Frequency: at every beam trip or fixed repetition frequency
- **In-depth calibration to determine kinetic parameters**
  - No rod-drop/MSM techniques and noise techniques with intrinsic source are applicable
  - PNS area method techniques with a pulsed source : YES
  - Noise Analysis to be checked with CW beam excitation before deciding

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Achievements in MUSE

MYRRHA

- **No on-line method yet established**
- **Regular proportionality check I_p/ϕ_n established** : Response to beam trip in fast system,
- **Absolute reactivity assessment established: PNS Area method for instance**, Other promising techniques should be re-assessed in CW beam conditions

Recommendations for the future:

- **On-line techniques in** CW conditions (**YALINA (BELARUS) in FP6**) I_p/ϕ_n **should be** established and **demonstrated in**:
  - **Start up conditions**
  - **Nominal conditions at various K_{eff} values**
  - **Shutting down conditions**
- **Absolute reactivity assessment** : a priori **non promising techniques (in MUSE** conditions pulsed mode) **should be revisited in** (CW conditions) (**Noise techniques**)
- Complementarities with RACE (USA) and SAD (Russia) should not be forgotten.

I&C Approach in MYRRHA

- The I&C has not been worked out yet in MYRRHA and need to be addressed urgently
- A diagram principal scheme has been established based on FBR approach with instrumentation foreseen at the outcome of (each) FA to monitor:
  - Temperature
  - Neutron Flux
  - Coolant velocity
  - Pressure
- O_2 Control in the reactor pool is needed

ISI&R Approach in MYRRHA

- To achieve the 65% availability for MYRRHA, Remote Handling approach is mandatory due to:
  - High activation level in the MYRRHA Hall (neutron streaming through the Beam line)
  - Potential a-contamination in the MYRRHA Hall due to ^{210}Po
  - Inert gas environment in the MYRRHA Hall to avoid the Pb-Bi contamination by O2 (PbO sludge formation)
- No real experience within the team => contracted a feasibility study by OTL Ltd (JET, UK)

1. PROJECT OVERVIEW

- Define the plant
- Define the working environment
- Define the task requirements
- Define the remote handling system requirements
- Decide a remote handling approach
- *Derive a remote handling concept & plant layout*
- *Validate the remote handling concept*
- *Estimate costs of implementing and running the systems*
- *Establish any technological areas requiring further development*
- *Deliver written report and a VR model of the proposed concept*

2. PLANT DEFINITION

- **Spallation Loop**
- **Core Support Tube**
- **Heat Exchangers**
- **Main Pumps**
- **Internal Robots**
- **Lid**
- **Diaphragm with chemical insert module**
- **Emergency Heat Exchanger**
- **Beam Line in MYRRHA Hall**
- **Experimental devices**
- **Pb-Bi vessel (for decommissioning)**

3. ENVIRONMENT DEFINITION

- **100% inert atmosphere**
- **0% humidity**
- **Particulate and gaseous contamination**
- **Max normal dose rate exposure of 7 Gy/hr**
- **Worst case dose rate exposure of 21.5 Gy/hr**
- **Total dose of 46500 Gy**

4. TASK REQUIREMENTS

- **Removal and replacement of plant items**
- **Plant maintenance (e.g Spallation zone replacement)**
- **Decontamination of plant items**
- **Packaging of waste items**
- **Recovery from failure during plant handling (e.g jamming)**
- **Recovery of a failed Ex-vessel Fuel Transfer machine**
- **Recovery of debris from PbBi**

5. REMOTE HANDLING SYSTEM REQUIREMENTS

- **Fully remote**
- **System Availability >95%**
- **Fail-safe system**
- **Recoverable after failure**
- **Perform replacement of Spallation loop within a 3 month shutdown**
- **Reach and examine all parts of the MYRRHA Hall**
- **Be easy to operate**
- **Be easy to support and maintain**
- **Be able to deal with unexpected tasks**
- **Minimise the secondary wastes**
- **Operate in the specified radiation environment for 30 yrs.**
- **Manipulate loads up to 60 tonne**
- **Perform specialist operations (e.g cutting, welding, 3-D metrology)**

6. REMOTE HANDLING APPROACH

- **Man-In-The-Loop using a Bi-lateral, force-reflecting, Master-Slave Servomanipulator.**
- **Robotic features to ease operation**
- **Cameras for visual feedback**
- **Independent craneage for lifting heavy loads**
- **Independent tool service system**
- **All remote handling work to be done within the same hall**
- **Remote equipment and tooling to be stored and maintained within the same hall**
- **Use of air-locks for transfers between areas**

7. PLANT LAYOUT AND INFRASTRUCTURE

- **MYRRHA Hall**
- **Contamination control**
- **Commissioning, Assembly, Test and Mock-up facilities**
- **Decontamination**
- **Waste Packaging**
- **Active workshop**
- **Remote handling control rooms**
- **Health Physics Laboratory**

Science Fiction or Reality?

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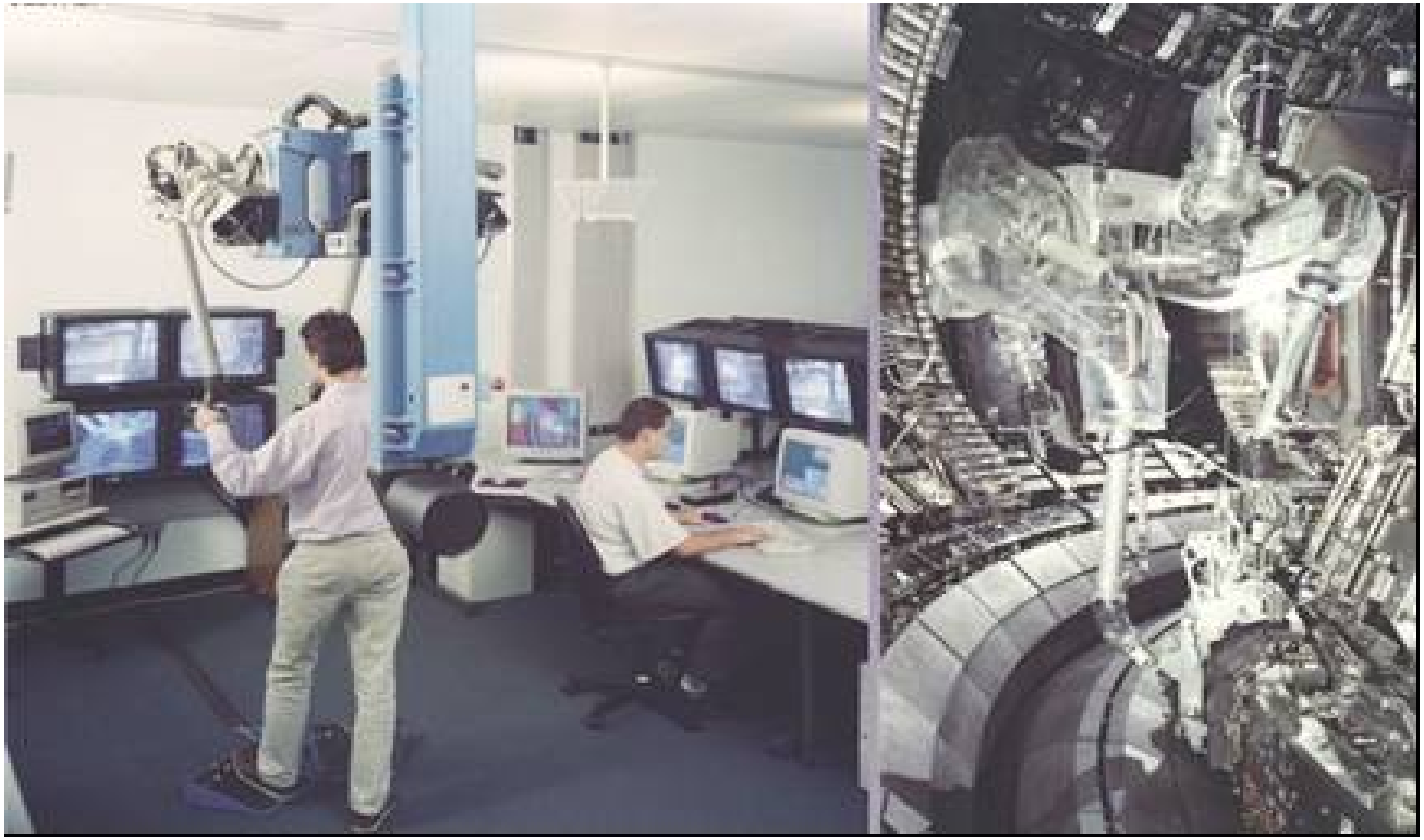
MYRRHA

Reality



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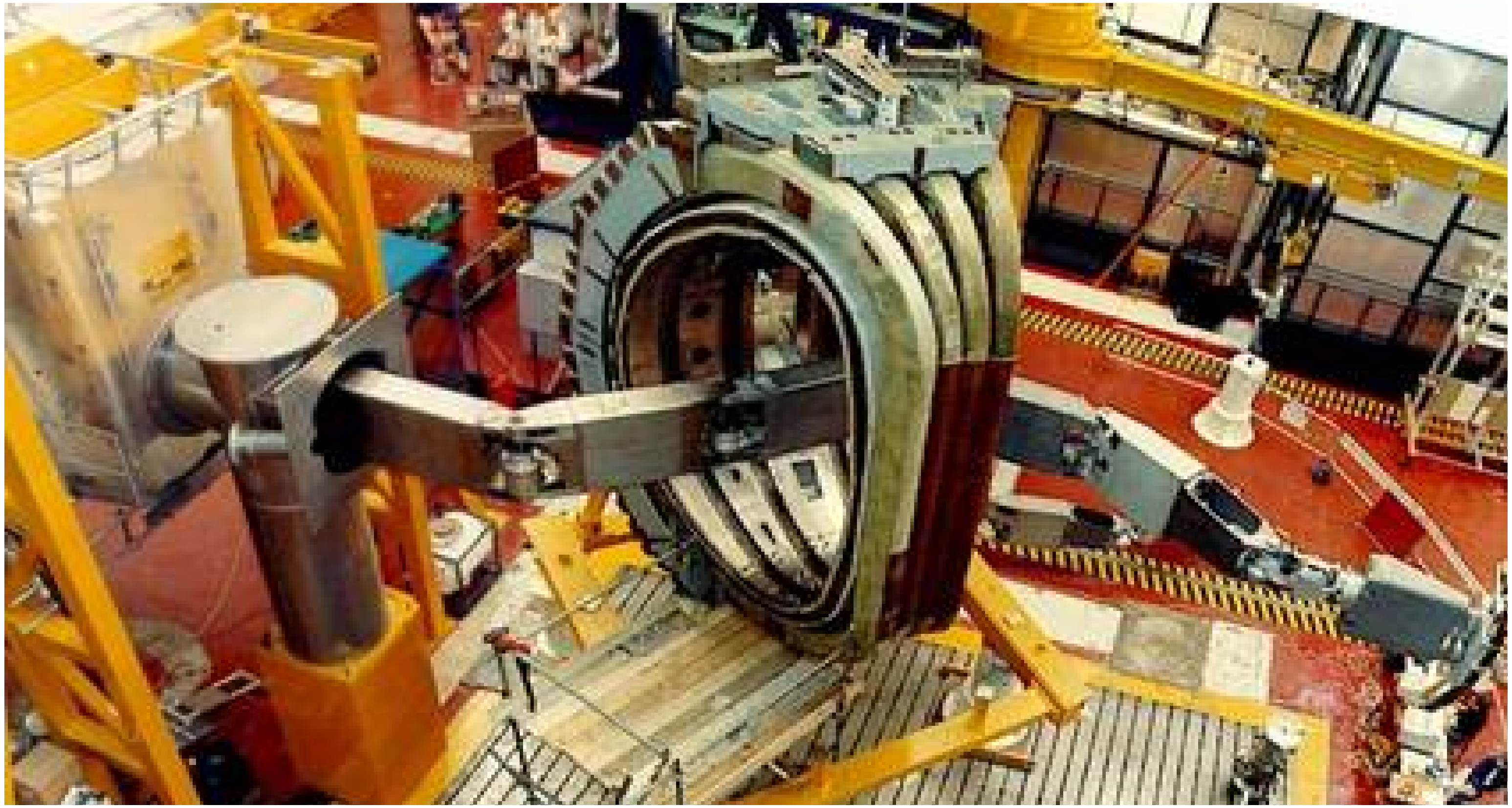
Reality



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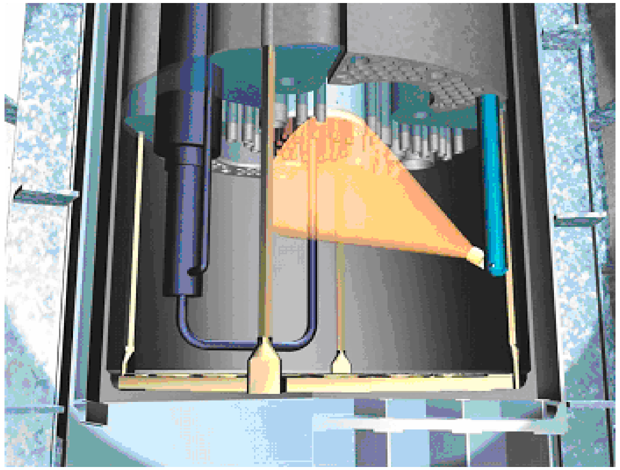
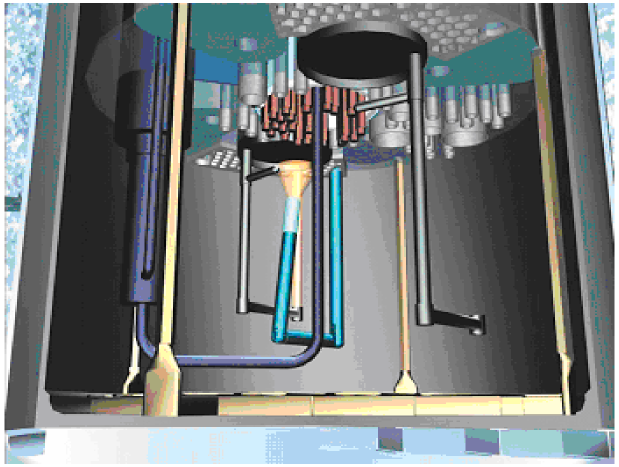
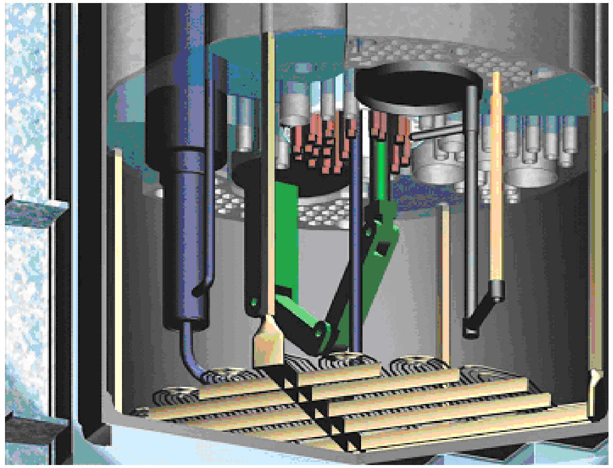
Design of MYRRHA
In-service inspection and repair



Two permanently installed *inspection* manipulators with US camera to provide a general *overview*. (periscope type device with three degrees of freedom)



The second *inspection* manipulator positions the camera close to critical components for *detailed* inspection. (anthropomorphic type device with five degrees of freedom)



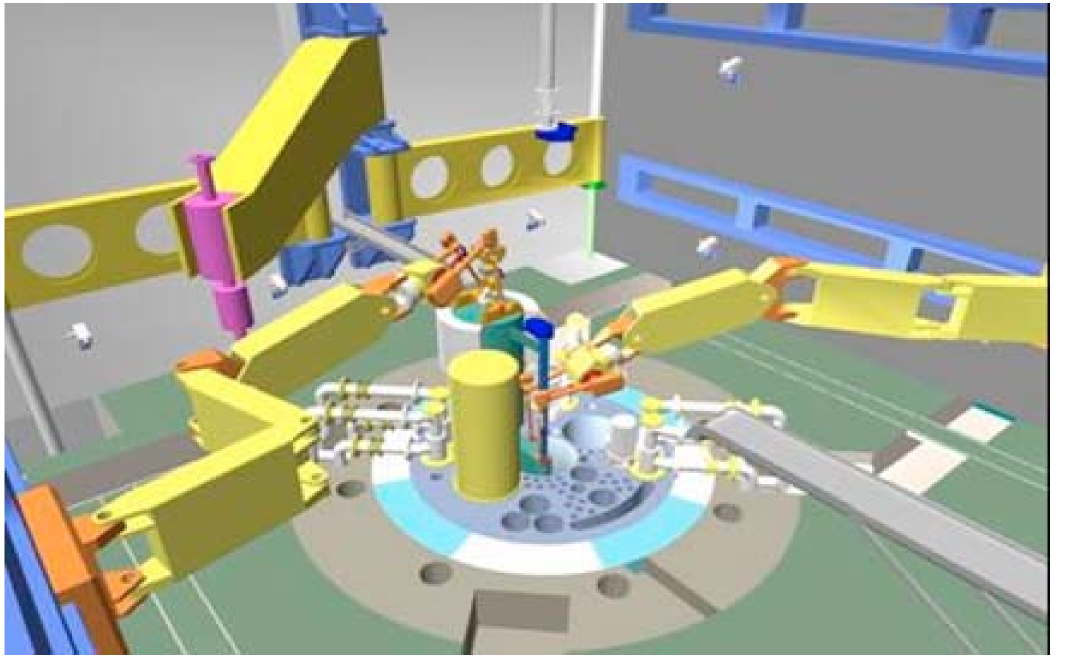
The *repair* manipulator recovers debris or deploys specialised tooling for repair. (anthropomorphic type device with eight degrees of freedom)

O.T.L. concludes positive on the feasibility of the proposed RH approach.

Design of MYRRHA
Remote handling

All MYRRHA maintenance opera-tions on the machine primary systems and associated equipment is performed by remote handling, which is based on the *Man-In-The-Loop principle*:

- force reflecting servomani-pulators
- Master-Slave mode: the slave servo-manipulators are commanded by remote operators using kinematically identical master manipulators
- supported with closed-cycle TV (CCTV) feedback



O.T.L. concludes positive on the feasibility of the proposed RH approach.

Conclusions

- MYRRHA being an irradiation facility has dictated the choices of the remote handling as a first choice for achieving an availability factor of 65% which compatible with operational cost that would be affordable.
- The fact of being a first-of-a-kind has also conditioned some design option in terms of operation cycle and the allowable beam trip mitigation via a preventive maintenance made repetitively during the shut down periods between cycles.
- The k_{eff} drop being limited per three months cycle make it manageable by a policy of fuel reshuffling supplemented by the use of burnable poisons or void boxes.
- The instrumentation and control is sketched and is not very different from the classical one of a classical reactor but need urgently further development.
- The sub-criticality monitoring is addressed and a promising route for the on-line monitoring is proposed.