Roadmap Discussion and HELIAS Perspectives

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Physics

W7-X optimization

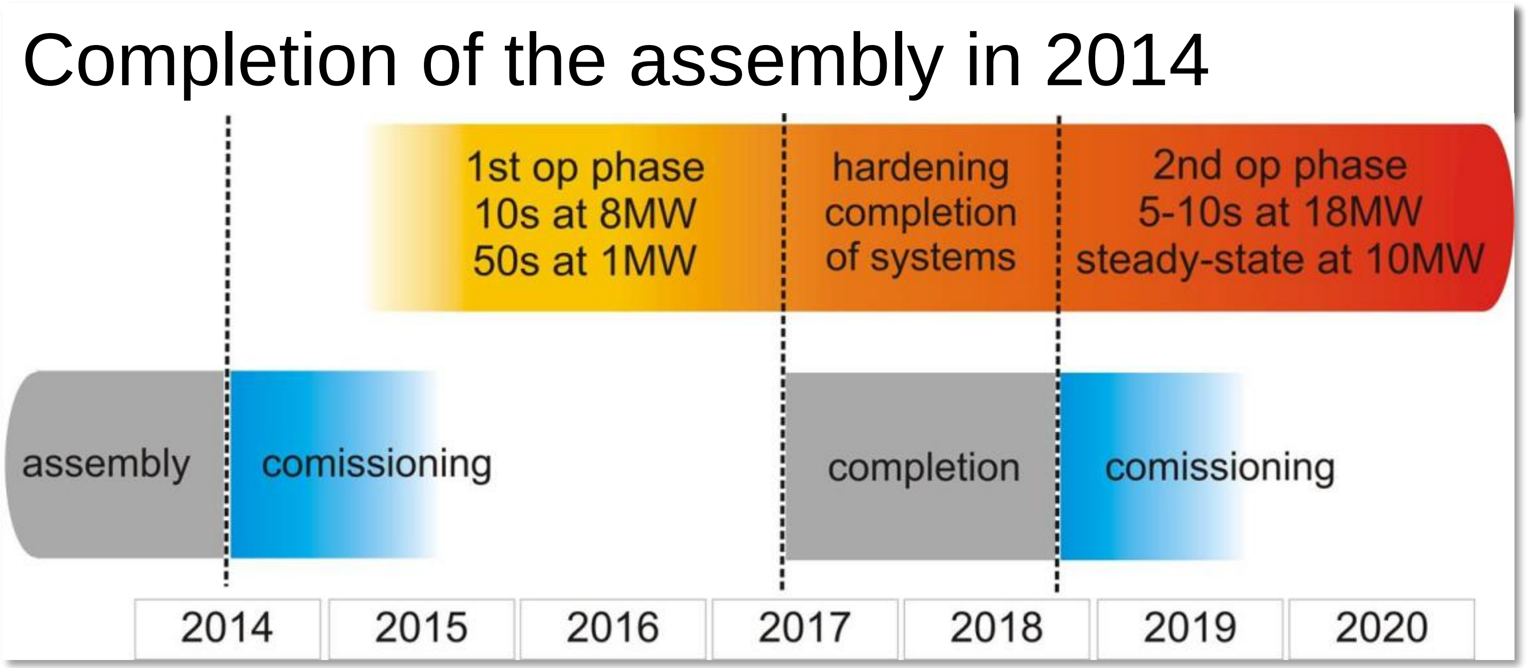
- High quality of vacuum field magnetic surfaces
- Good finite-β equilibrium properties
  - *small Pfirsch-Schlüter current reduces Shafranov shift and* $\Delta\iota$
- Good MHD stability properties
  - *allowing for* $\langle\beta\rangle \approx 0.05$ *operation*
- Reduced neoclassical transport in the 1/ν regime
  - $\chi_{1/\nu} \propto \varepsilon_{eff}^{3/2} T^{7/2}/(nR_0^2 B_0^2)$ *makes reduction of* ε_{eff} *mandatory*
- Small bootstrap current in the *lmfp regime*
  - *minimize effect of JBS on* ι*-profile and island divertor*
- Good collisionless fast-particle confinement
  - *diamagnetic effect produces minimum-B configuration at finite-*$\langle\beta\rangle$
- Good modular coil feasibility

Issues not directly covered by optimization

For extrapolation to a HELIAS W7-X has to demonstrate reactor relevant scenario with simultaneous achievement of

- true steady-state operation (discharges of several minutes)
- at high density ($n_e > 10^{20}\ m^{-3}$) and temperature
- with full density control (continuous pellet refueling)
- with viable divertor performance (high radiated-power fraction with at least partial detachment)
- without impurity accumulation

W7-X operation



OP1 ——————————→ OP2 —— ^v—→ *Upgrades*

W7-X operation, OP1

OP 1

- Pulsed operation: 10 MW / 10 s to 1 MW / 1 minute
  - *Effects on the L/R time cannot be observed*
- Limitations power (limitations of power supplies): 8 gyrotrons or 4 gyrotrons & 4 NBI PINIs
  - *Verification of neoclassical confinement optimization*
  - *High-β operation at 2.5 T unlikely, at reduced field also reduced power*
  - *Verification of fast ion confinement requires $\langle\beta\rangle \approx 4\%$*
- Test divertor unit without cryo-pumps, limited pumping capability
  - *Limitations of divertor operation*
  - *Divertor operation determined by conditioning of PFCs*

Extrapolation to next step

Background

- All physics laws must be dimensionally correct[1]
- Dimensional analysis leads to the space of dimensionless variables describing physics laws[2]
- For plasma physics employing B_t, B_p, T, n, R, a, m_i, m_e, e, c there are four dimensionless quantities (e.g. ρ^*,ν^*,β, N_D)[3] but N_D is orders of magnitude off and can be neglected[4]
- Representation in arbitrary bases employing ρ^*,ν^*,β, N_D is equivalent

[1] goes back to J.B. Fourier

[2] E. Buckingham, Phys. Rev. 4, 345 (1914)

[3] B.B. Kadontsev, Sov. J. Plasma Phys. 1, 295 (1975)

[4] C.C. Petty, Phys. Plasmas 15, 080501 (2008)

Extrapolation to next step

Specific choices[5]

$$\rho^* \propto T^{1/2} a^{-1} B_t^{-1}$$

$$\nu^* \propto aq\lambda_{mfp}^{-1} \propto an_e q T^{-2}$$

$$\beta_t \propto n_e T B_t^{-2}$$

are transformed to

$$B^* \propto B_t\, a^{5/4}$$

$$P^* \propto P_{heat} a^{3/4}$$

$$n^* \propto n a^{3/4} B_t^{-1}$$

assuming the identity of trivially dimensionless parameters (geometry κ, δ, ε, q, …).
To eliminate the temperature, the scaling of the energy confinement is employed:

$$\tau \propto W / P \propto nTa^2 RP^{-1} \propto nTa^3 \varepsilon^{-1} P^{-1} \propto \underbrace{\frac{a^2 B}{T}}_{\tau_B} \underbrace{\rho^{*\alpha_\rho} \beta^{\alpha_\beta} \nu^{*\alpha_\nu}}_{\text{scaling}}$$

and inserted into the expressions for ρ*,ν*,β these parameters can be represented by B*,P*, n*

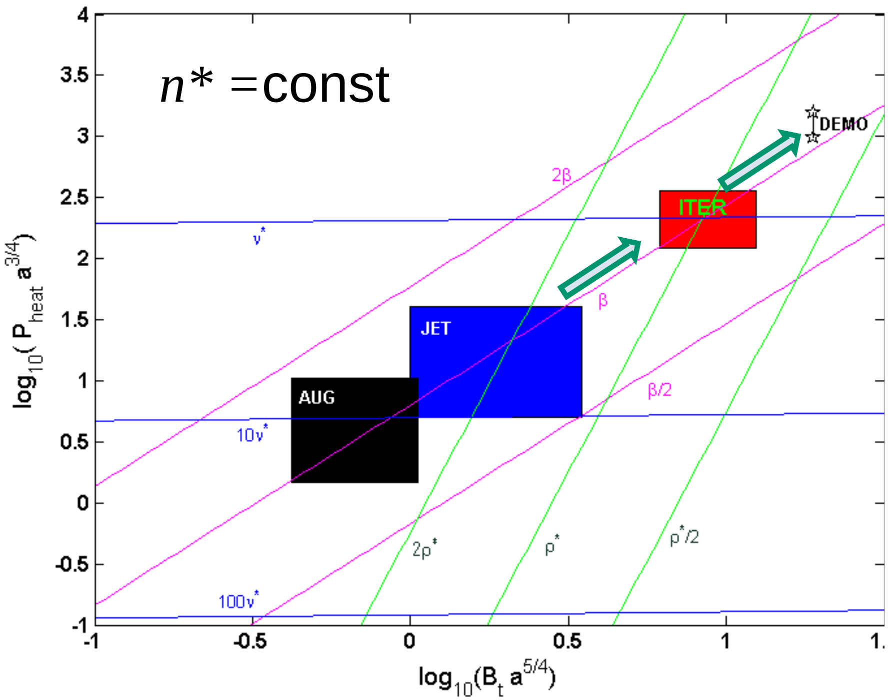
The transformation for the ISS scaling is
rho* ~ B*^{-0.8104} P*^{0.1934} n*^{-0.2302}
nu* ~ B*^{0.2418} P*^{-0.7737} n*^{1.9207}
beta ~ B*^{-0.6209} P*^{0.3868} n*^{0.5397}

[5] K. Lackner, Fusion Sci. Technol. 54, 989 (2008)

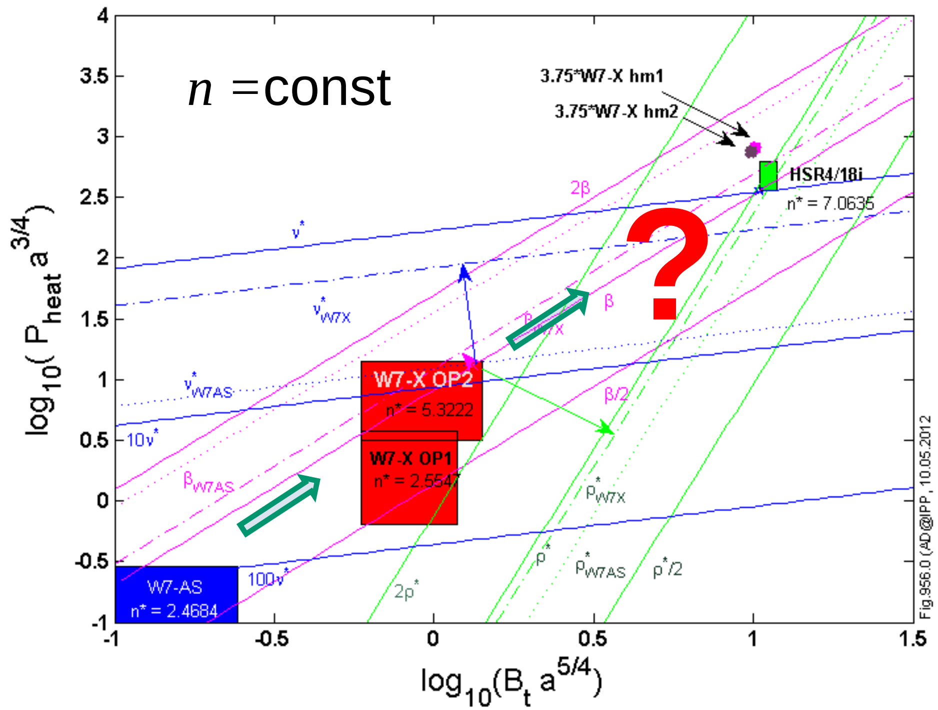
Extrapolation to next step: Step ladder to a FPP

How far away is the next step?

Tokamak



Stellarator



The boxes are from design values or roughly represent the achieved parameters

K. Lackner, Fusion Sci. Technol. 54, 989 (2008)

A. Dinklage (priv. comm.)

Input assumptions

Tokamak

	$B_{t,\min}$ (T)	$B_{t,\max}$ (T)	$P_{heat,\min}$ (MW)	$P_{heat,\max}$ (MW)	a (m)
DEMO	6	6	500	500	2.5
ITER	2.6	5.2	73	150	2
JET	1	3.5	5	40	1
ASDEX Upgrade	1	2.5	2.5	25	0.5
DIII-D	0.8	2.2	2.5	25	0.67
Alcator C-Mod	3	9	1.5	6	0.22

K. Lackner, Fusion Sci. Technol. 54, 989 (2008)

Stellarator

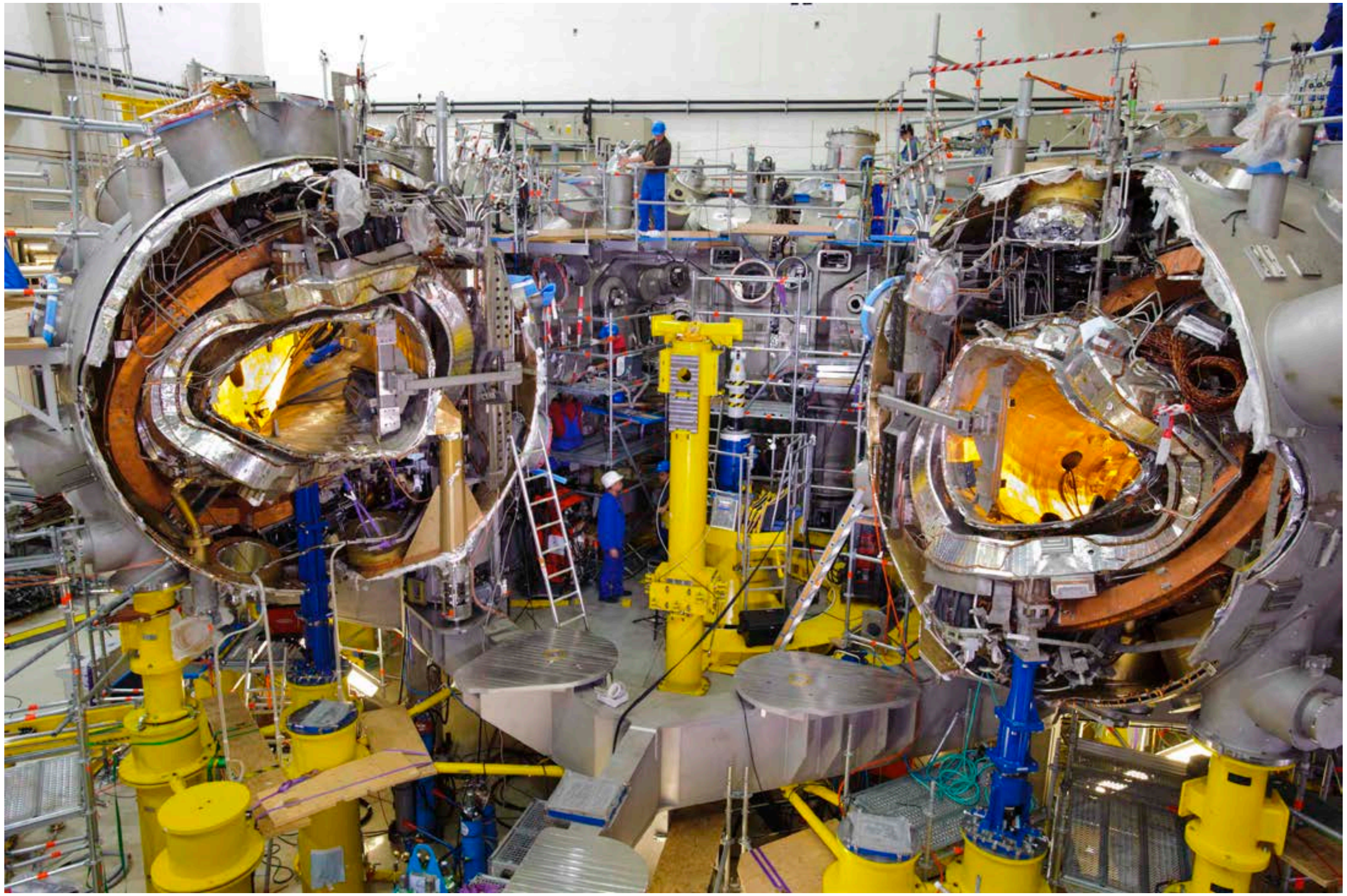
	Bmin	Bmax	Pmin	Pmax	a	R	R/a	nmin	nmax
W7-AS	0.75	2.5	0.2	3	0.155	2	11.8	15	21
W7-X OP1	1.25	2.5	1	10	0.55	5.5	10	5	10
W7-X OP2	2.5	3.0	5	30	0.55	5.5	10	10	25
HSR4/18i*	4.4	5	214	260	2	18	9	15	21
3.75W7-X hm 1	4.5		487		1.92	20.7	10.8	19	
3.75W7-X hm 2	5.5		523		1.6	20.7	12.9	23	

*Wobig et al, Concept of a HELIAS ignition experiment, Nucl. Fusion 43, 889 (2003) – cf. Tab. 4

Technology

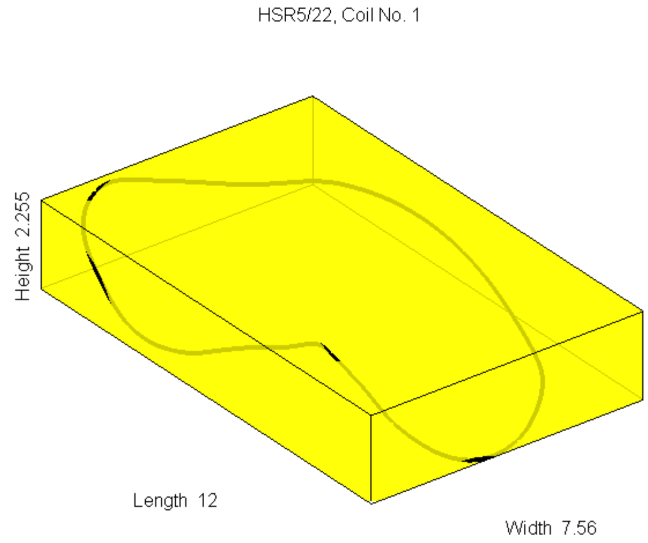
Starting point

- Wendelstein 7-X will show / has shown that a large stellarator with modular coils can be built



HELIAS 5-B

Vergleich Reaktortypen								
13.04.2012								
HSR5/22								
	I (MA)	13,65						
	$B_{pl\text{-}axis}$ (T)	5,9						
	Min. distance plasma - current sheet (m):			1,74				
	Coil Nr.	Min. height	Max. length	Width	Circumfer.	Curv.Rad.	Max. force	Force/circ.
	-	(m)	(m)	(m)	(m)	(m)	(MN)	(MN/m)
	1	2,26	11,98	7,56	34,24	1,73	207,8	6,1
	2	2,59	11,79	7,35	33,90	1,68	217,6	6,4
	3	3,11	11,59	7,37	34,02	1,63	220,7	6,5
	4	3,33	11,29	7,53	34,06	1,74	221,1	6,5
	5	3,82	11,27	7,06	34,73	1,70	126,3	3,6
HSR4/18								
	I (MA)	13,90						
	$B_{pl\text{-}axis}$ (T)	6,0						
	Min. distance plasma - current sheet (m):			1,74				
	Coil Nr.	Min. height	Max. length	Width	Circumfer.	Curv.Rad.	Max. force	Force/circ.
	-	(m)	(m)	(m)	(m)	(m)	(MN)	(MN/m)
	1	2,85	13,09	7,79	37,36	1,28	291,8	7,8
	2	3,23	12,78	7,78	36,58	1,24	278,4	7,6
	3	3,10	12,19	7,43	34,94	1,45	289,1	8,3
	4	3,55	11,34	7,52	34,68	1,32	197,9	5,7
	5	4,45	11,03	7,53	36,98	1,22	90,9	2,5
W7-X * 4								
	Coil Nr.	Min. height	Max. length	Width	Circumfer.	Curv.Rad.		
	-	(m)	(m)	(m)	(m)	(m)		
	1	2,20	12,05	7,48	34,22	1,67		
	2	2,53	11,87	7,48	33,93	1,66		
	3	2,93	11,65	7,39	33,89	1,58		
	4	3,34	11,45	7,51	34,23	1,63		
	5	3,43	11,22	7,27	34,04	1,76		
ITER								
	TFC	–	≈ 12,5	≈ 8	34.5	≈2		



F Schauer (private comm.)

Steps to a DEMO (reference is *HELIAS 5-B*)

The following target parameters are from HELIAS 5-B which is a DEMO reactor (the next step stellarator could be designed smaller)

- Higher magnetic field: 3 $\rightarrow$ 5 / 6T (ITER value, lower than in tokamak DEMO)
- Larger plasma volume: 30 $\rightarrow$ 1500 m^3
- Fusion power 10 – 20 MW (external) $\rightarrow$ 600 MW (α-power) / 3GW (fusion power)
- Average heat flux 0.1 MW/m^2 $\rightarrow$ 0.4 MW/m^2
- Divertor heat flux similar to tokamak (P/R scaling?) $\rightarrow$ large radiation fraction required
- Average neutron flux 1 MW/m^2, peak neutron flux 1.6 MW/m^2 (typically below values of tokamaks)

Stellarator specific issues

Additional issues as consequence of 3D geometry, not necessarily more complicated solutions than in tokamak

- **Divertor**
  - Depending on the divertor solution (helical, island, etc.) direct transfer of tokamak design to a stellarator not possible
  - Helical and island divertors follow a helical path inside the plasma vessel (also on top of plasma vessel!)
  - Complex magnetic field line topology affecting the shape of the plasma-wall interface and its alignment requirements
- **Blanket and Shield**
  - Compared to tokamak larger aspect ratio of the Helias leads to relaxation of most technical constraints
  - Adaptation to the specific plasma vessel geometry
  - Current design such that space of ~1.3 m is available
  - In Helias 5-B a larger magnetic field as in the original HSR5/22 is employed and thus more space is possible

Stellarator specific issues (cont'd)

- **Coil System**
  - Original Helias studies use NbTi superconductor (field below 4.5 T or superfluid He-cooling)
  - Nb_3Sn (ITER TF): technology challenges similar to those faced in ITER (higher field, better confinement, e.g. more space for blanket)
  - Helias coils are basically modified ITER TF coils
- **Plasma Vessel and Cryostat**
  - Basically same as for a tokamak with the exception of the shape
  - W7-X experience
- **Remote Maintenance**
  - Some of the envisaged tokamak solutions for divertor and blanket replacement are not applicable in a stellarator
  - Special effort is required to accommodate the specific requirements of the stellarator
  - Helias port size 6 × 2 m^2 / torus separation conceivable (?); currently under investigation

Roadmap

Roadmap

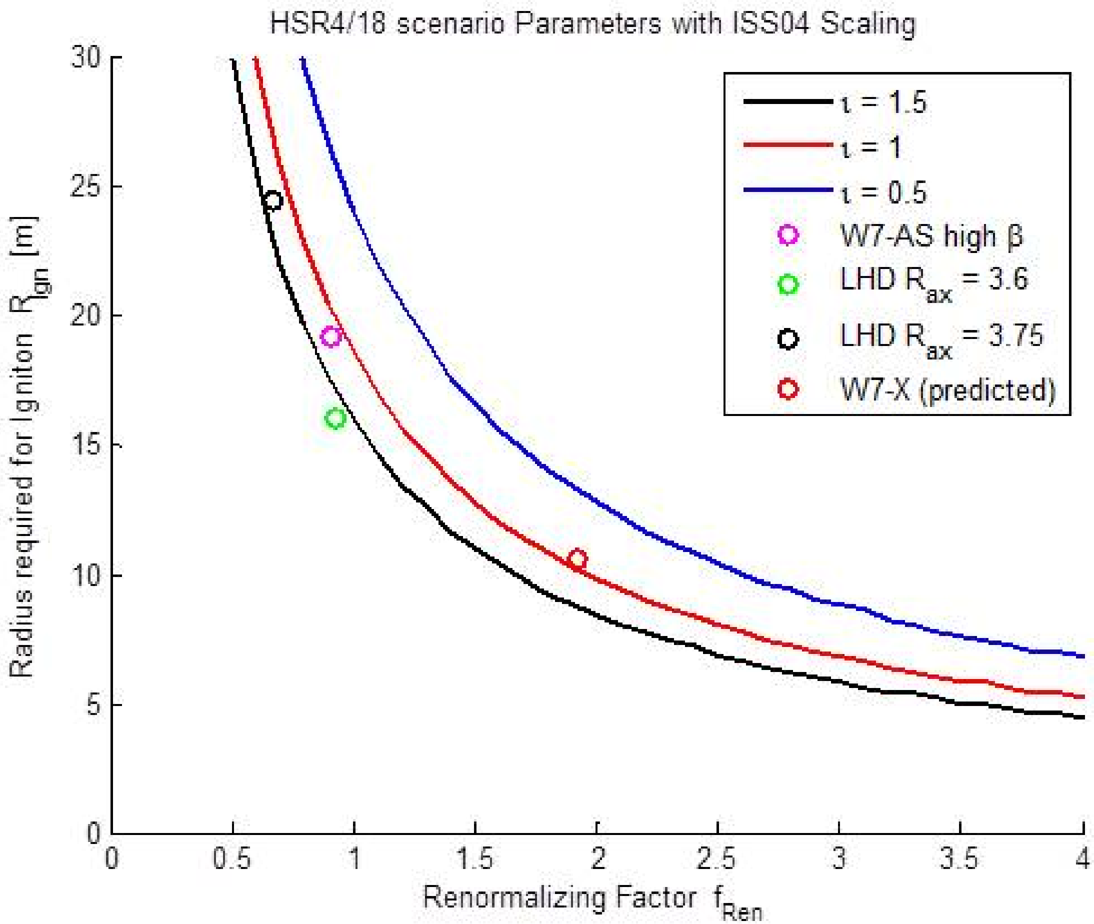
AUG → JET → ITER
1st burning fusion plasma
test of DEMO technologies → DEMO
pulsed?
T-self sufficiency
$Q_{eng} \sim 1$ → Commercial fusion

By 2025 decision
on how to proceed

W7-X
verify physics of concept → Stellarator-ITER
burning fusion plasma / ignited?
technology tests (e.g. TBM)
< 5 dpa → Use DEMO technology → Commercial fusion

W7-X → Stellarator DEMO
HELIAS 5B → Commercial fusion

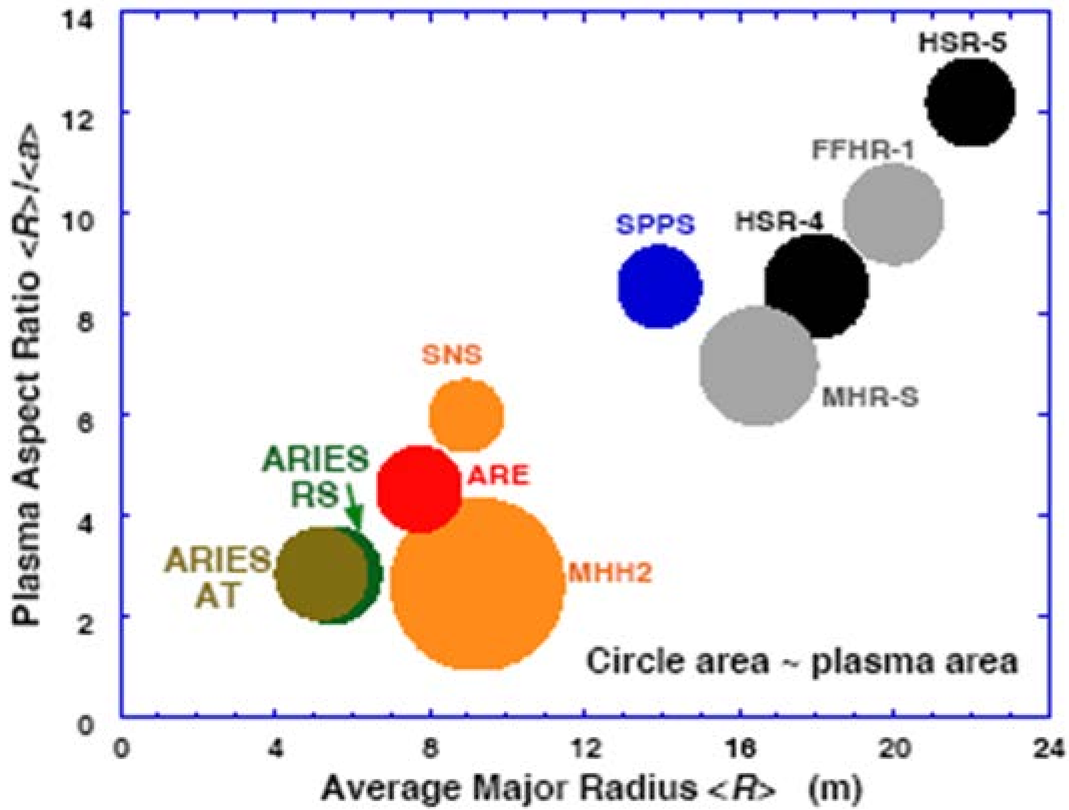
“Reduced” HELIAS ?



- Scaled-up W7-X
- How optimistic is the underlying confinement scaling ?
- W7-X will find out
- No full T-breeding
- No space reservation for a full T-blanket (neutron shield and test blanket modules instead)

$\rightarrow$ presentation by F. Warmer

Roadmap



HSR-5
FFHR-1
SPPS
HSR-4
MHR-S
SNS
ARIES RS
ARE
ARIES AT
MHH2
Circle area ~ plasma area
Plasma Aspect Ratio <R>/<a>
Average Major Radius <R> (m)
0
2
4
6
8
10
12
14
0
4
8
12
16
20
24