Preliminary Assessment of Possibilities for Improving Performance of SCWR Using PuH_2-ThH_2 Fuel

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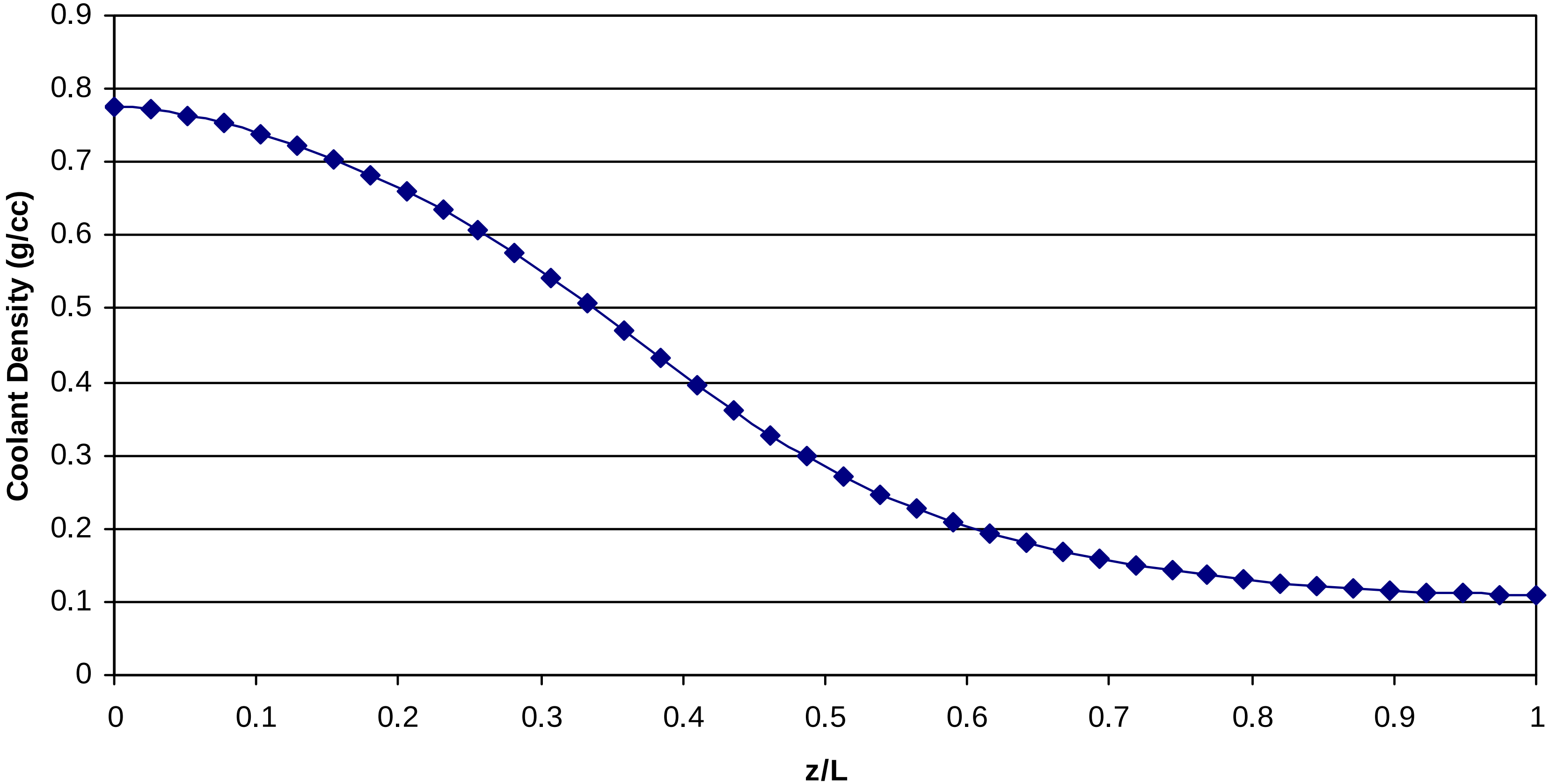
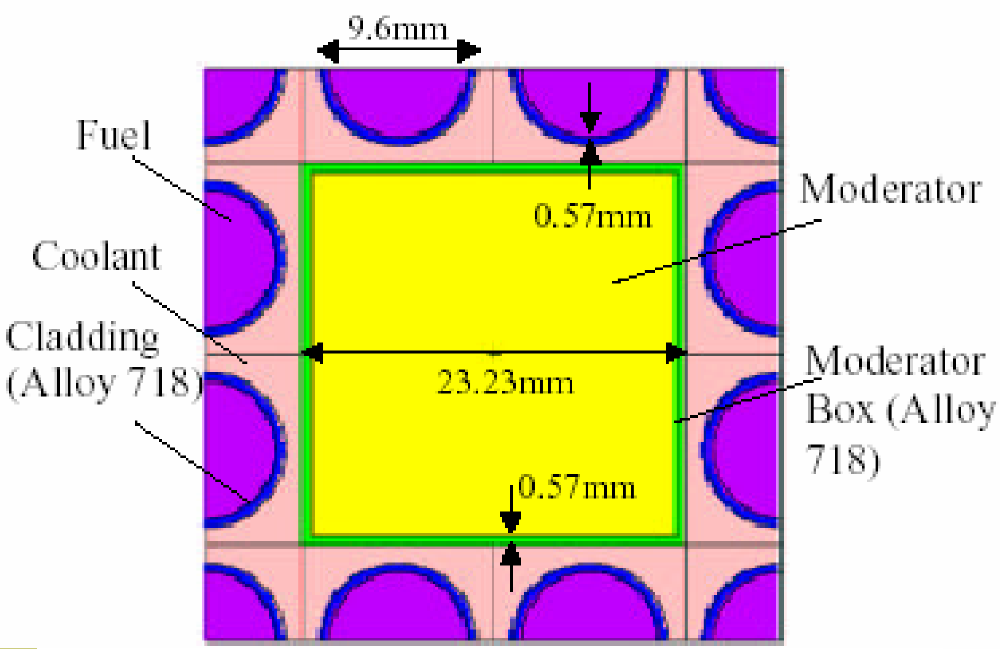
Difficulties in SCWR core design

Coolant density is very low at upper part of core which is very under-moderated

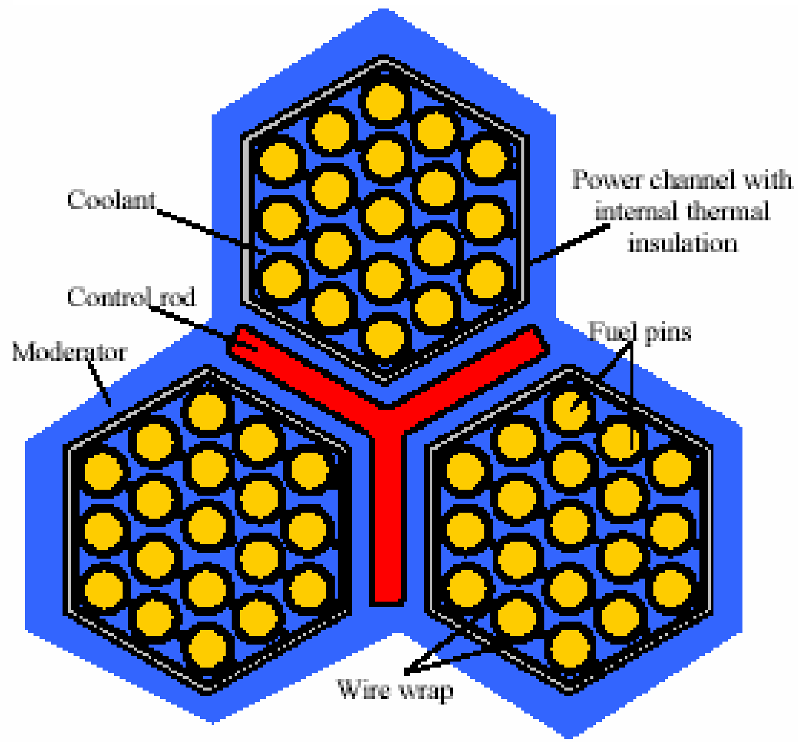


Design solutions considered

A. Solid moderator prisms



B. Low temperature water channels



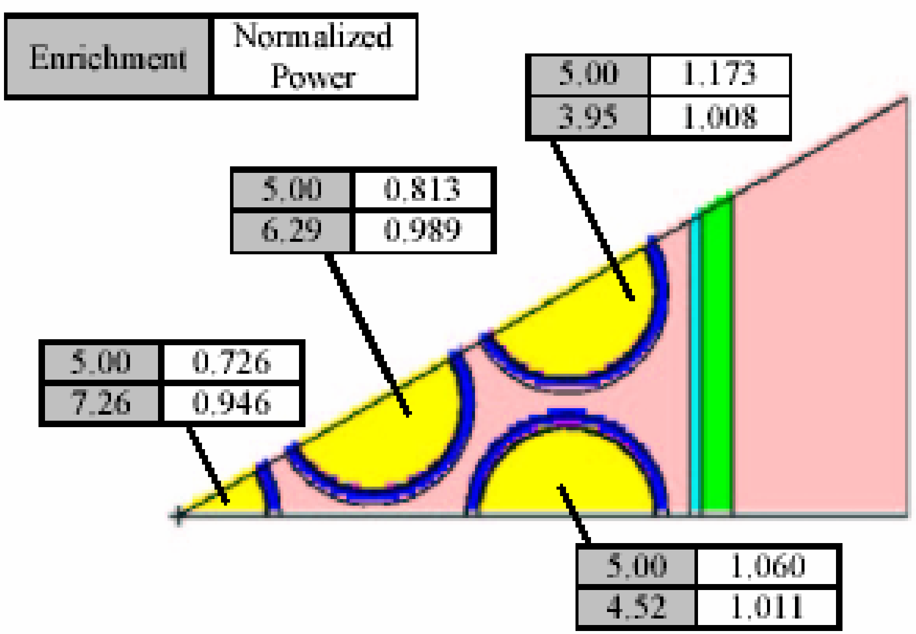
<u>Drawbacks</u>

- *~45% of core volume for moderator*
- *Significant parasitic n absorption*

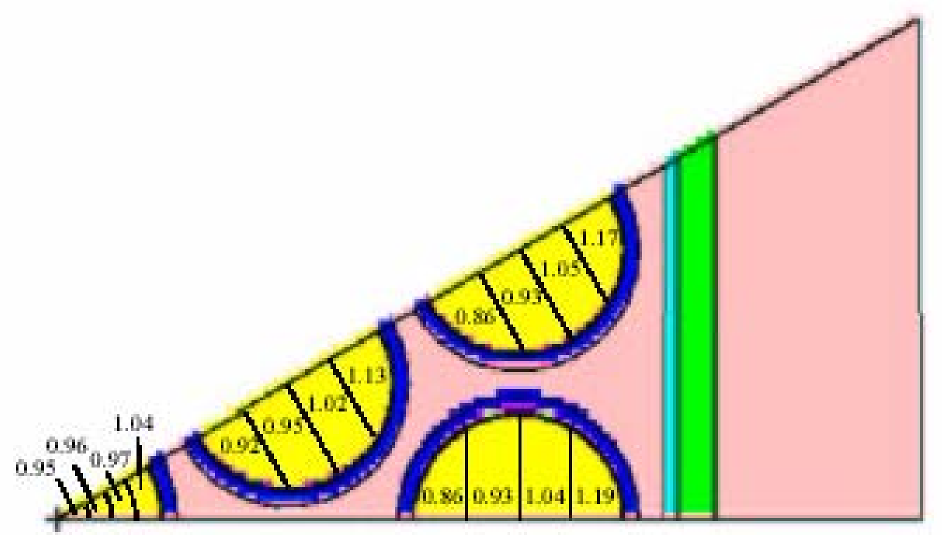
- *~45% of core volume for moderator*
- *Coolant outlet temp is penalized*

Drawbacks of proposed designs (cont)

- Heterogeneity
- Asymmetry



Variation of pin powers at BOL for uniform (top) & variable enrichment



Asymmetry in power distribution in fuel rods (Normalized to pin power)

Contents

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- Approach
- Methodology
- Results
- Conclusions

Study goal

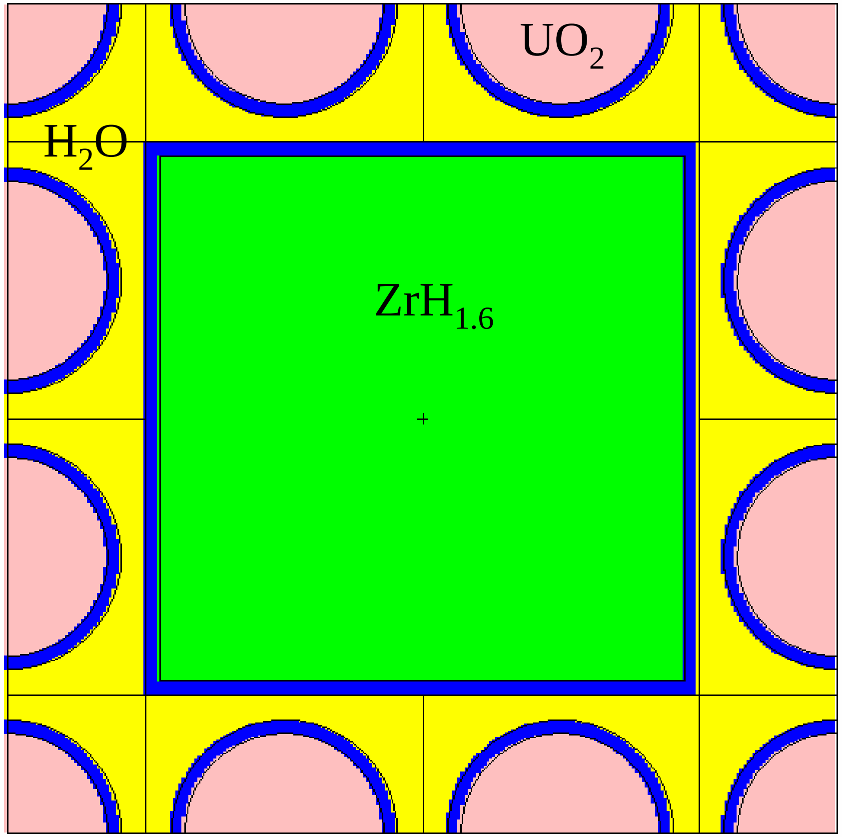
- A VERY preliminary assessment of the feasibility of improving the design of a SCWR with a thermal neutron spectrum

Design approach

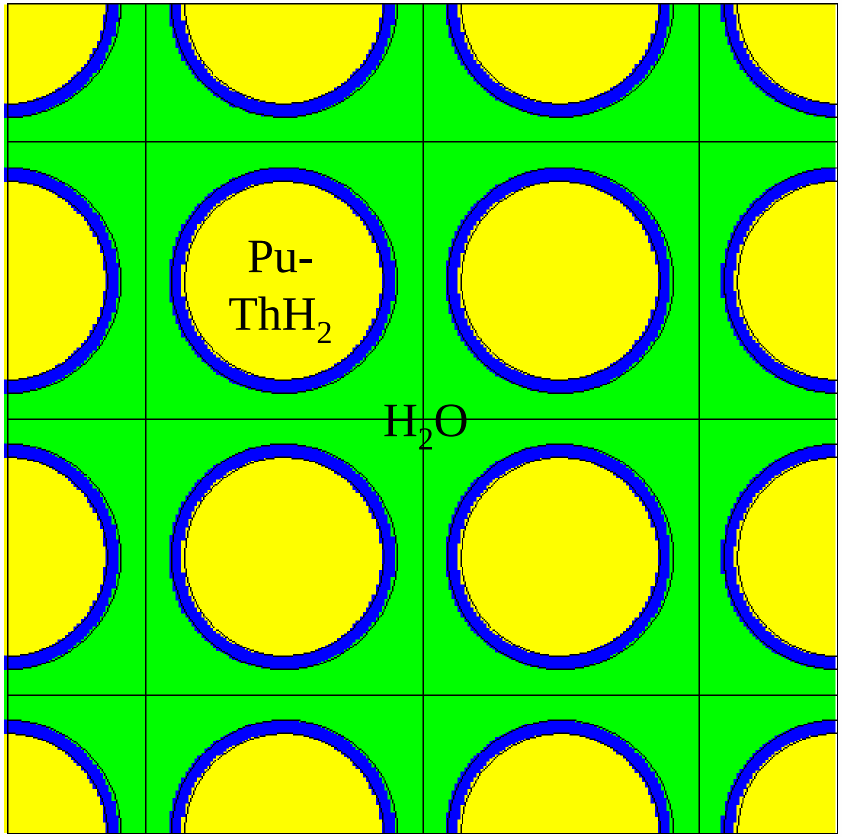
- Replace oxide fuel and $ZrH_{1.6}$ solid hydride by solid hydride fuel
- Solid hydride fuel considered is PuH_2-ThH_2
- Make use of the volume released by the solid moderator to load more HM into a given volume core or to get better moderation and, hence, higher discharge burnup

SCWR design approach

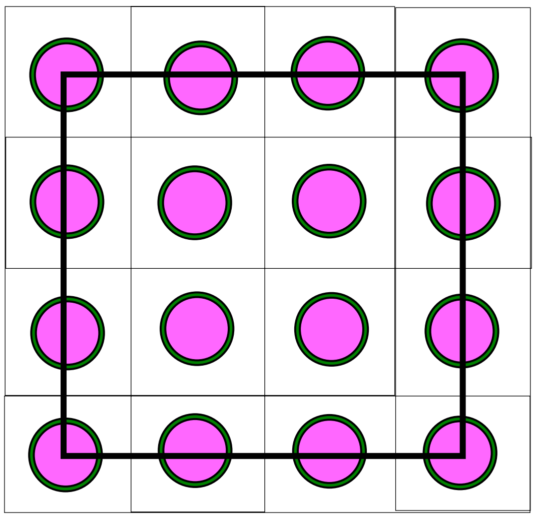
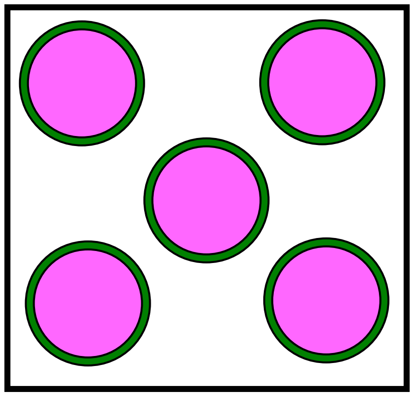
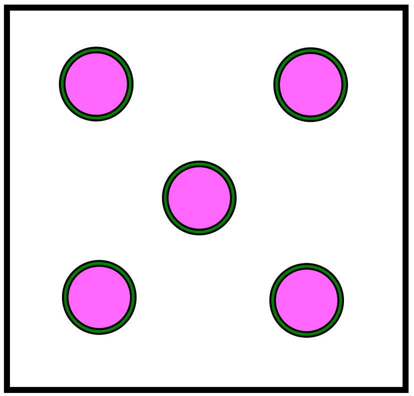
Reference (INEEL)



Design A (colors are inconsistent)



Pu-ThH$_2$ fuel cases considered

	Case A	Case B	Case C
			
# fuel rods:	9	5	5
Coolant area: (per rod)	same	same	larger

Design approach (4)

Region	**Material**	**Temperature**
Moderator and Coolant	Water Axial Distribution (0.1 – 0.78 g/cc) Average=0.3713 g/cc (over the axial volume)	583 K (Typical PWR)
Fuel Clad	Alloy 718 (8.19 g/cc) Ni 52.90; Cr 19.08; Fe 18.122; Nb 5.05; Mo 3.01; Ti 0.91; Al 0.49; Mn 0.20; Si0.19, C=0.038	623 K (Typical PWR)
Fuel	10w/o Pu + 90w/o ThH_2 ρ=10.0223 g/cc (Power Grade: Pu238 1.0; Pu239 62.0; Pu240 22.0; Pu241 12.0; Pu242 3.0)[1]	978 K (Typical PWR)

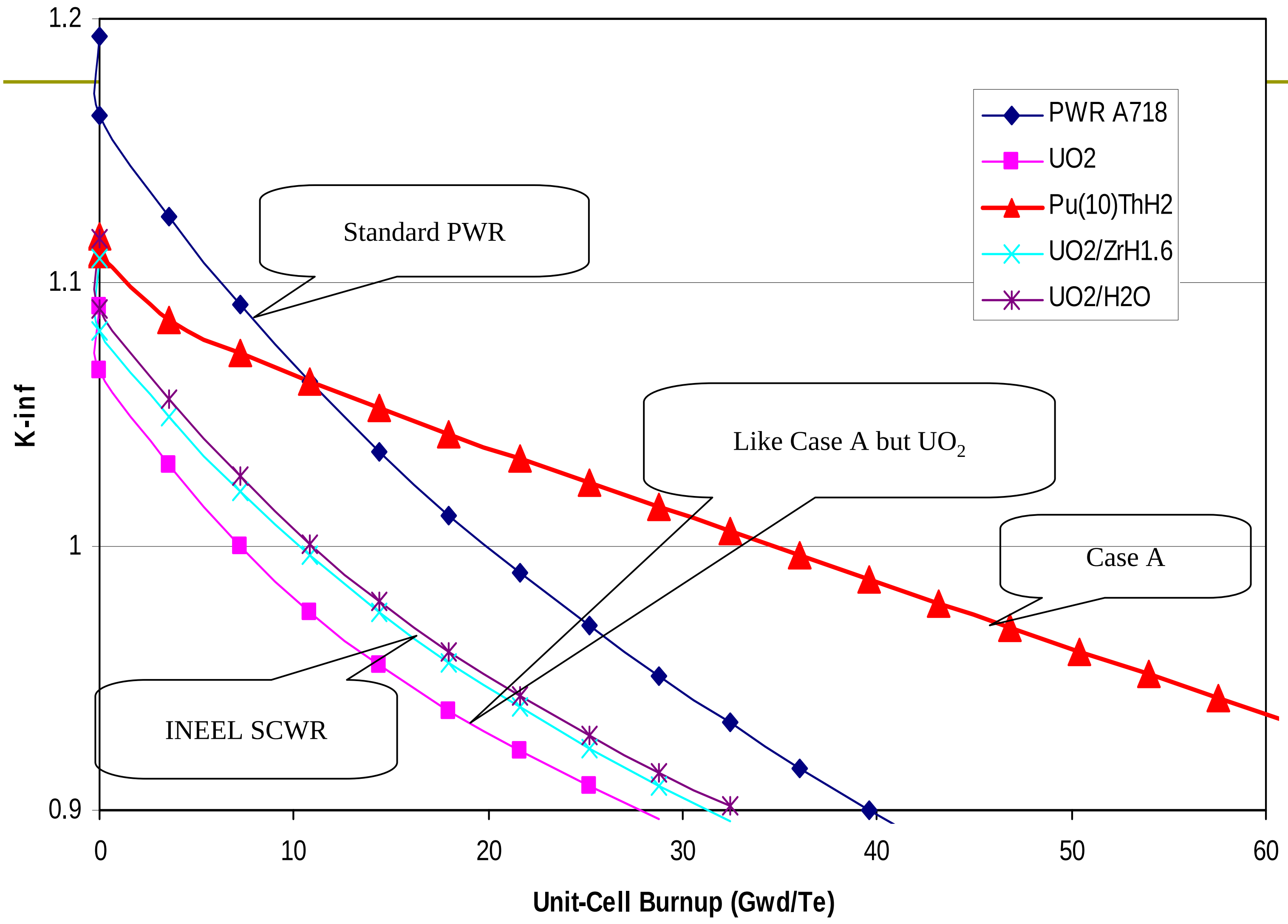
Methodology/benchmarking

comparing BOL k_∞

Fuel type/ Moderator	Computer Code			Relative Error	
	MCNP4 B (INEEL)	MCNP4B2 (UCB)	WIMSD5B (UCB) ρ_{water}=0.6g/cc	INEEL/UCB	MCNP4B2/ WIMSD5B
UO_2/H_2O 5 rod core cell	1.151	1.14180	1.11702 (Cluster)[a]	1.008	1.022
$UO_2/ZrH_{1.6}$ 5 rod core cell	1.132	1.11743	1.10903 (Cluster)	1.013	1.007
PWR with Alloy 718 4.1 % U235	1.20		1.19308 (Unit-cell)		1.0058

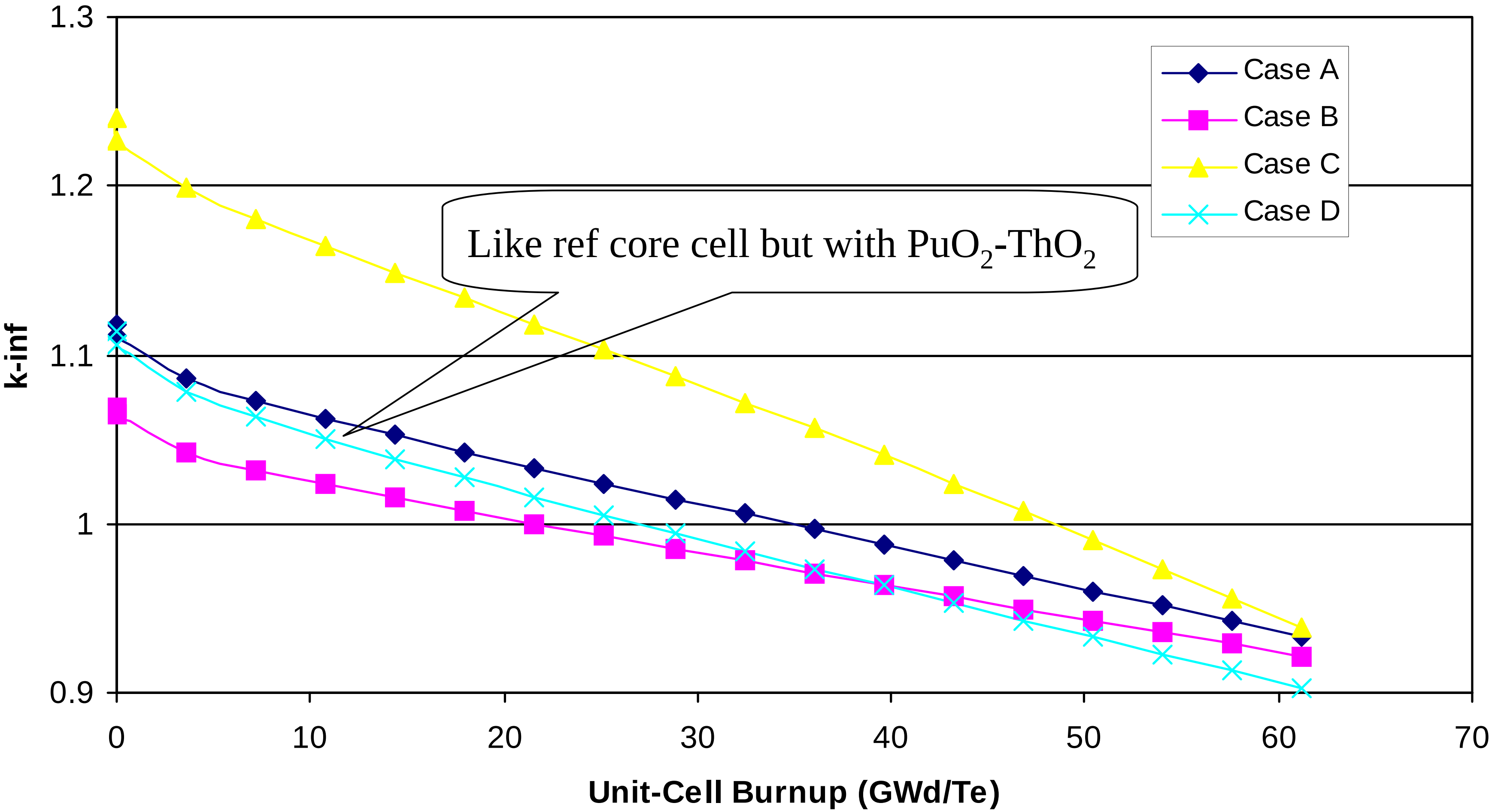
[a] The water density inside the water box is 0.778 g/cm^3.

Results



1.2
1.1
1
0.9
K-inf
0
10
20
30
40
50
60
Unit-Cell Burnup (Gwd/Te)
PWR A718
UO2
Pu(10)ThH2
UO2/ZrH1.6
UO2/H2O
Standard PWR
Like Case A but UO_2
Case A
INEEL SCWR

Results – using Pu



SCWR summary of results

Characteristic	Design				
	Reference	PuO_2 -ThO_2	A	B	C
HM per core cell (relative)	1.0	0.909	1.949	2.108	1.083
Thermal power possible (relative), if same q’	1.0	1.0	1.8	1.0	1.0
BOL k_∞	1.11	1.11	1.12	1.07	1.24
Discharge BU (GWD/tHM) for k_∞=1.03 and 3 batches	10.5	24	33	10.9	62.3
Cycle duration at nominal P (d)	130	296	407	135	769
Energy per core (relative)	1.0	2.1	6.1	2.2	6.4

SCWR: Conclusions

Relative to the PuO_2-ThO_2 fueled core cell used as a reference, use of Pu-ThH_2 fuel can offer one of the following benefits:

- 80% more power per given volume core (Design A)
- 2.5 times higher average discharge burnup (Design C)
- 3 times more energy generation per core loading (Design A and C)
- 2.5 times cycle duration

All above without Fuel/clad compatibility problem !

A much more thorough study/design optimizations need to be performed in order to establish sound quantitative conclusions

Can hydride fuel operate at LHR (w/cm) of oxide fuel?

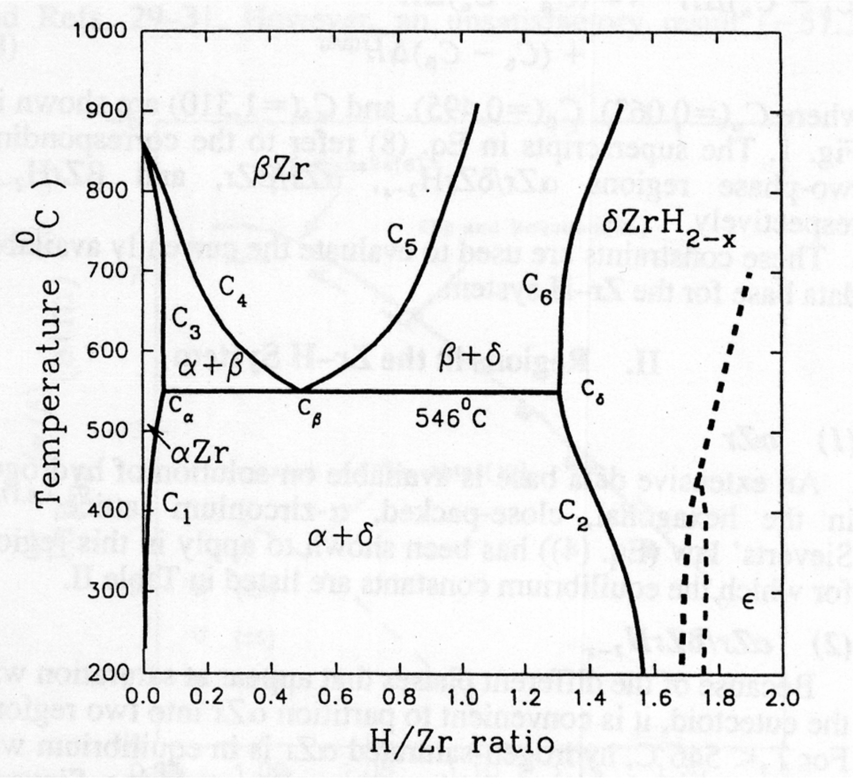
Characteristics	ROMANIAN TRIGA	SCWR
Fuel pin O.D. (cm)	1.294	0.95
Cladding		
Material	SS	MA956[a]
Thickness (mm)	0.40	0.57
Fuel loading (kg U/m)	0.489	0.485
Avg. linear-heat-rate (kW/m)	37	19
Specific power (W/g-HM)	75.7	36.4
Power density (W/cm^3)	138.7	90.5
Discharge burnup (MWd/kgHM)	120	62[b]
Energy extracted from fuel (MWd/m)	59.2	17.9
Peak fuel temperature (^{o}C)	550	2000
Coolant exit temperature (^{o}C)	~70	~500

[a] One possibility. [b] Assumed as of PWR lead rod average

In SCWR, q' ~ [(750-500)/(550-70)] 37 = ~ 19.2 kW/m

Zr-H Phase Diagram

α(alpha-metal): low-temp. solid solution of H in hexagonal closed-packed α - Zr
β(beta-metal): high-temp. solid solution of H in body-centered cubic β - Zr
δ(delta-hydride): face-centered cubic hydride
ε(epsilon-hydride):face-centered tetragonal hydride with c/a < 1, extending to ZrH_2.



Uranium added is rejected from solution during hydriding and forms a fine uniform dispersion but shifts all phase boundaries lower by a few degrees.

Properties of hydride fuel: ZrH_x phase diagram

- 1atm H partial pressure ~ 800°C
- Our steady-state peak T design limit is 700°C
- Our transient peak T design limit is 1050°C

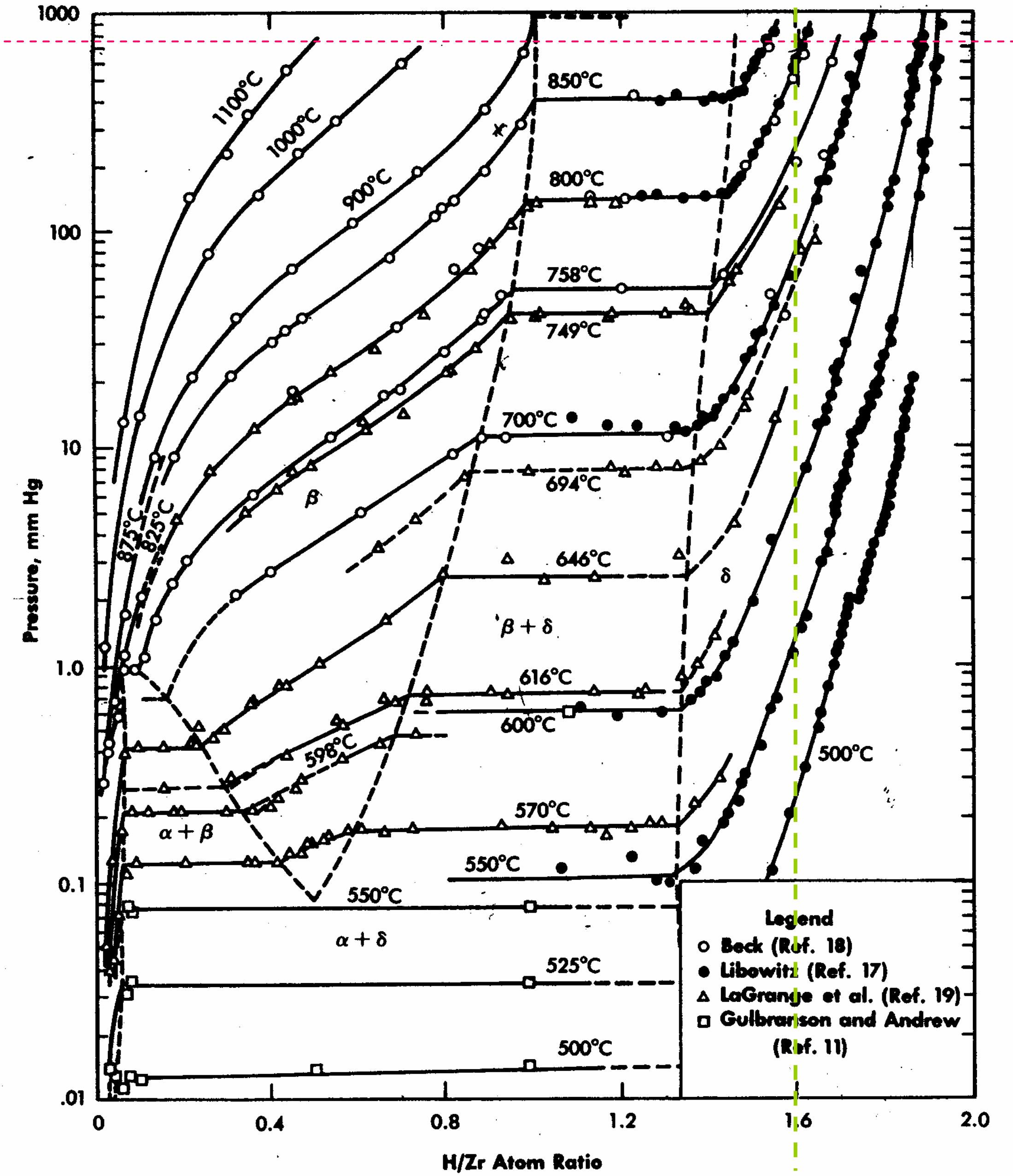
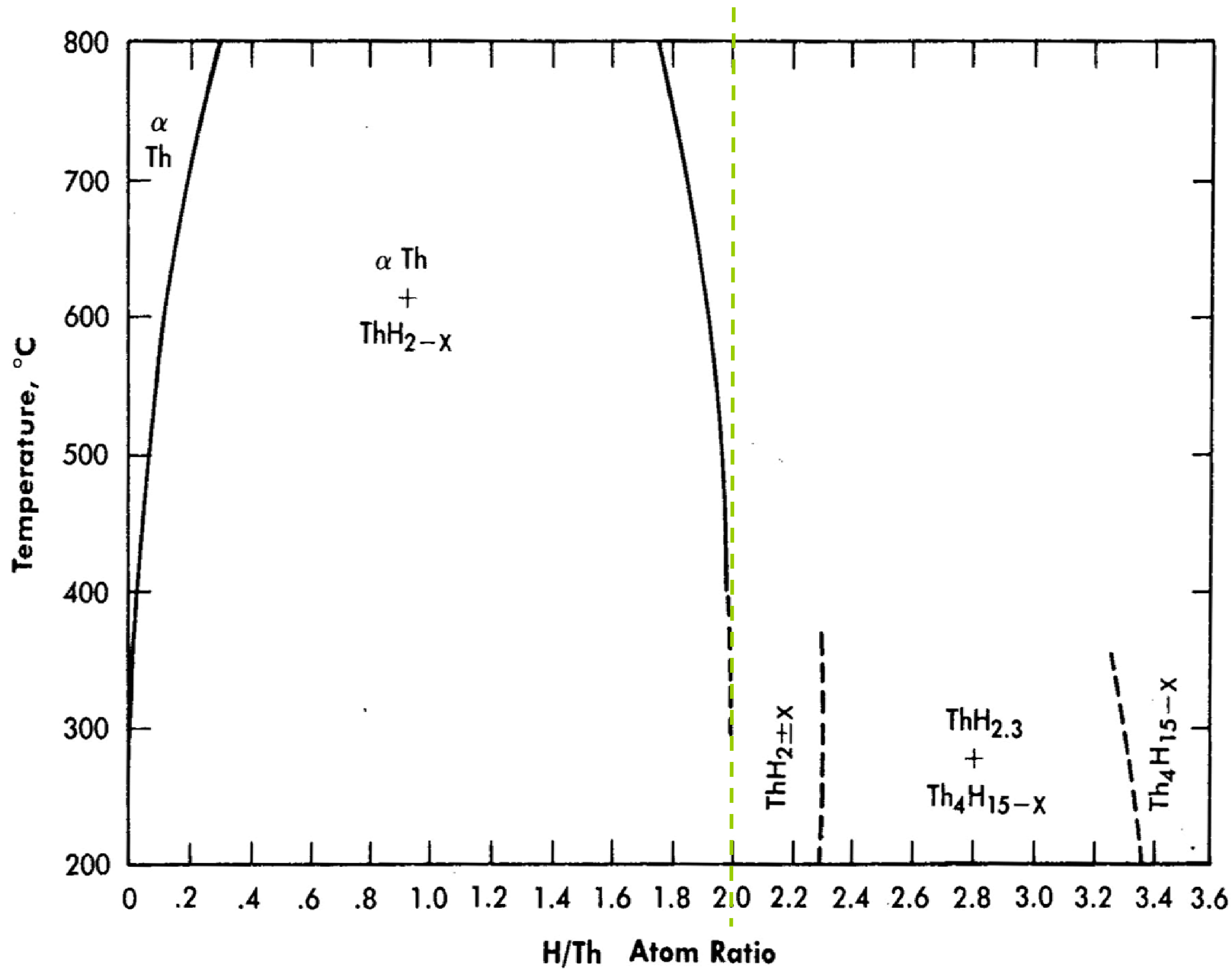


FIG. 7.2 Pressure–composition isotherms (composite data).

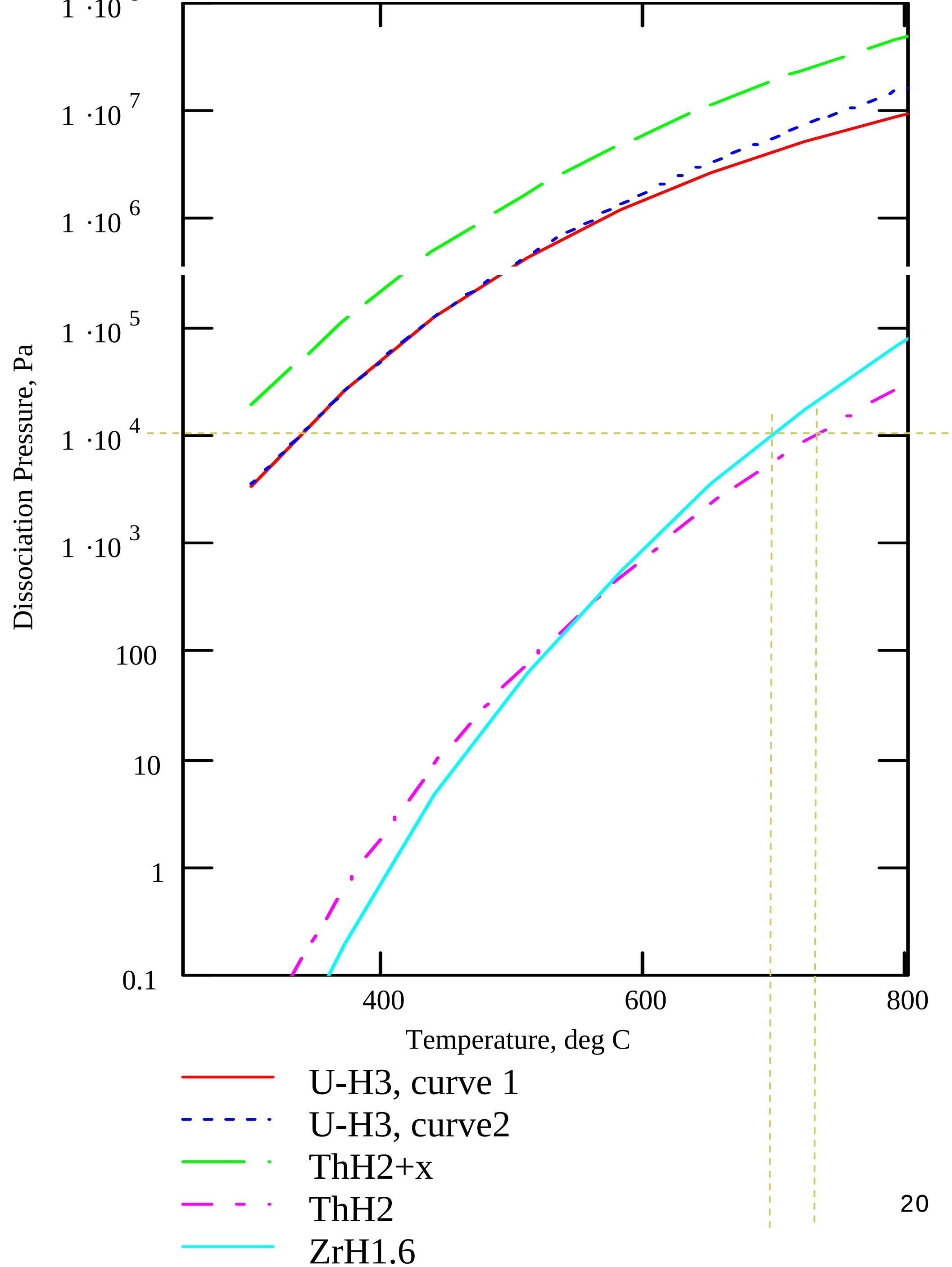
Properties of hydride fuel (2) – ThH_x phase diagram



Properties of hydride fuel(3) – dissociation pressure

- ThH_2 has a somewhat lower dissociation pressure than $ZrH_{1.6}$

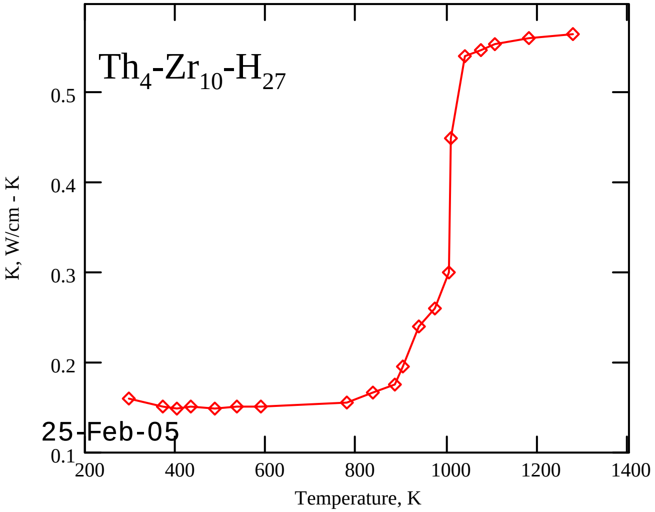
Can operate at somewhat higher temperatures

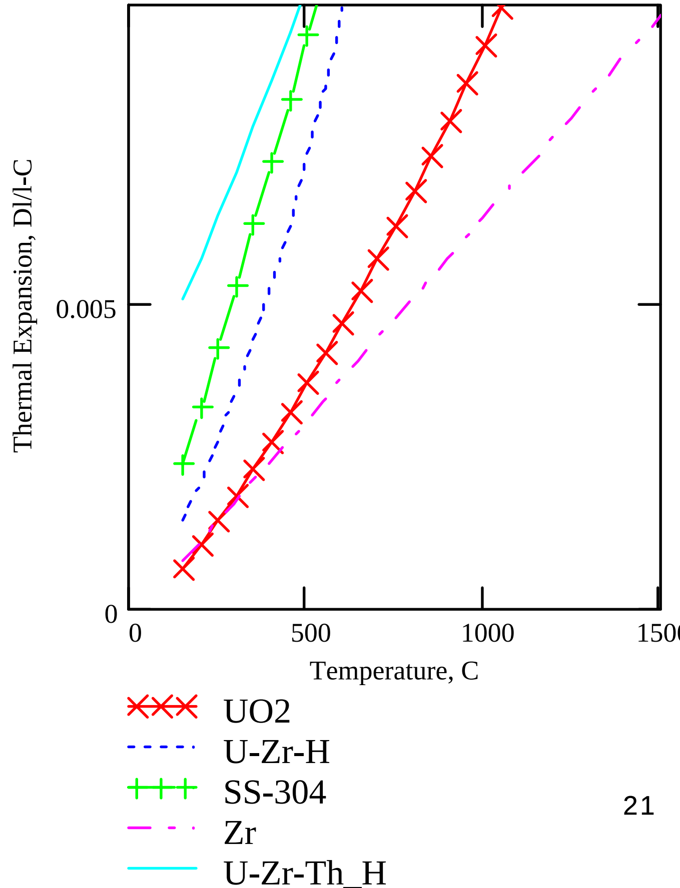


Properties of hydride fuel – thermal conduct

	$UZrH_{1.6}$	$UThH_2$	UO_2
Thermal conductivity (W/cm-C)	0.18	> 0.15	0.035
U density (g/cc)	3.715	10.26	9.225

Thermal expansion coef. (1/C)



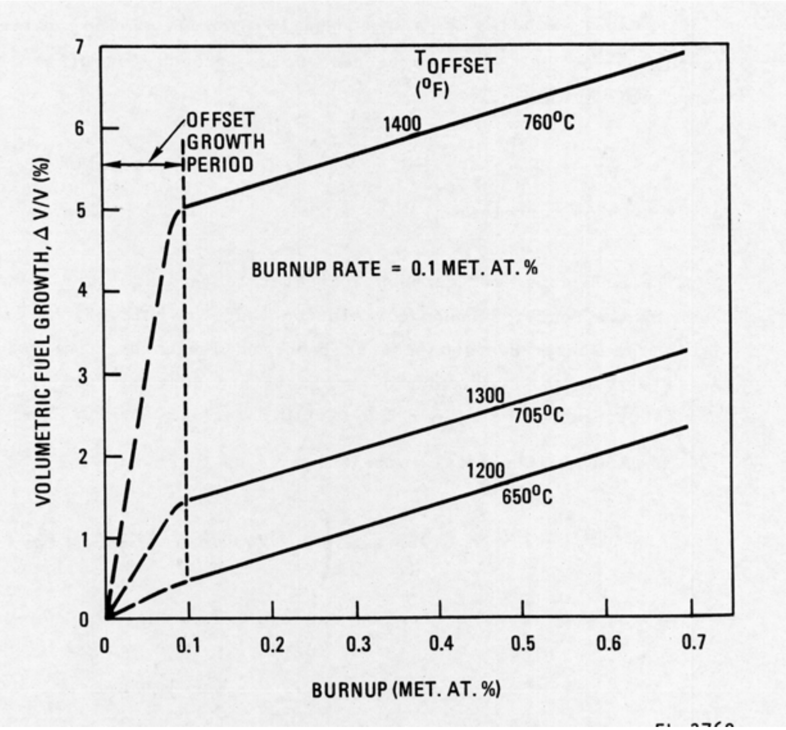


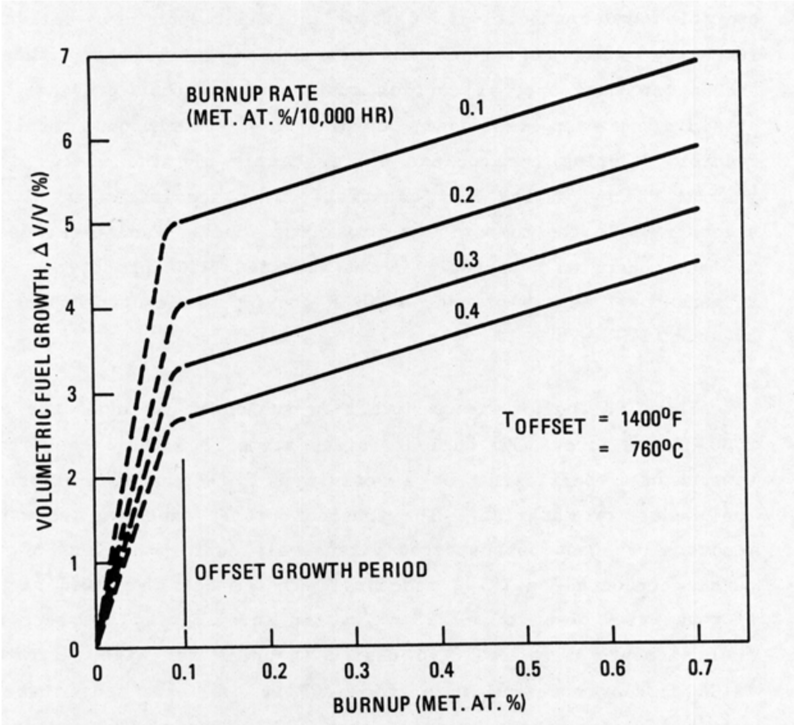
In-Pile Tests of Hydride Fuel

GA Technologies of San Diego has extensive database of in-pile experiments with TRIGA reactors (U-ZrH fuels).

- 63 TRIGA reactors in 23 countries
- 800 reactor-years of operation (1957-1985)
- 6000 fuel elements - 7 different types used
- Fuel types
  - 8.5 wt% U- 20% enrichment, pulsing,low power, [1957]
  - 12 wt% U- 3 wt% Er (HEU)-20% enrichment 1-14 MW, [1975]
  - 30-45 wt % U - 20% enrichment (LEU) commercial reactors, [1978]
- Oakridge Research Reactor (ORR) tests [1979-1984]
  - 16-rod cluster
  - 20,30, and 45 wt% fuels
  - 25-53 KW/element
  - 901 full power days, burnup to 65% U235 (**~100GWD/tHM**)
  - Destructive and non-destructive testing

Swelling of Hydride Fuel





- Available data indicate larger swelling than UO_2.
- Yet TRIGA fuel has ~1/2 the gap width of UO_2 and has been tested up to BU of ~100 GWt/tHM without signs of deformation
- LM gap fill (33.3% each Pb,Sn, Bi) proposed will alleviate problem