Water-Cooled Thorium Breeder Reactors
水冷却トリウム増殖炉

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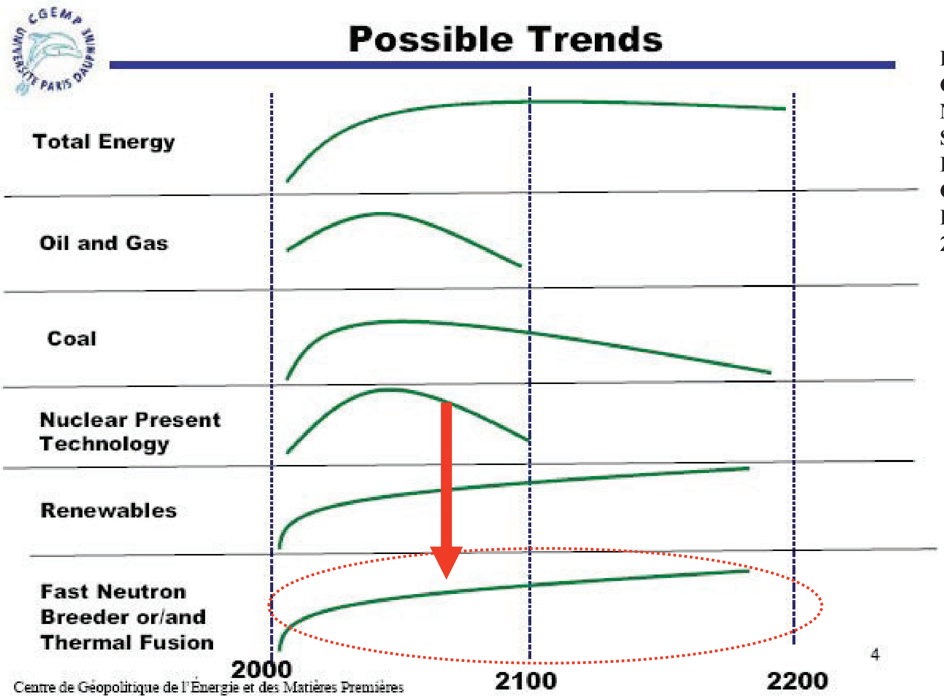
The 3rd Working Group on Thorium Fuel Utilization
in Light Water and Fast Reactors
第三回軽水炉・高速炉におけるトリウム燃料の利用WG
東京大学平成22年10月15日（金）

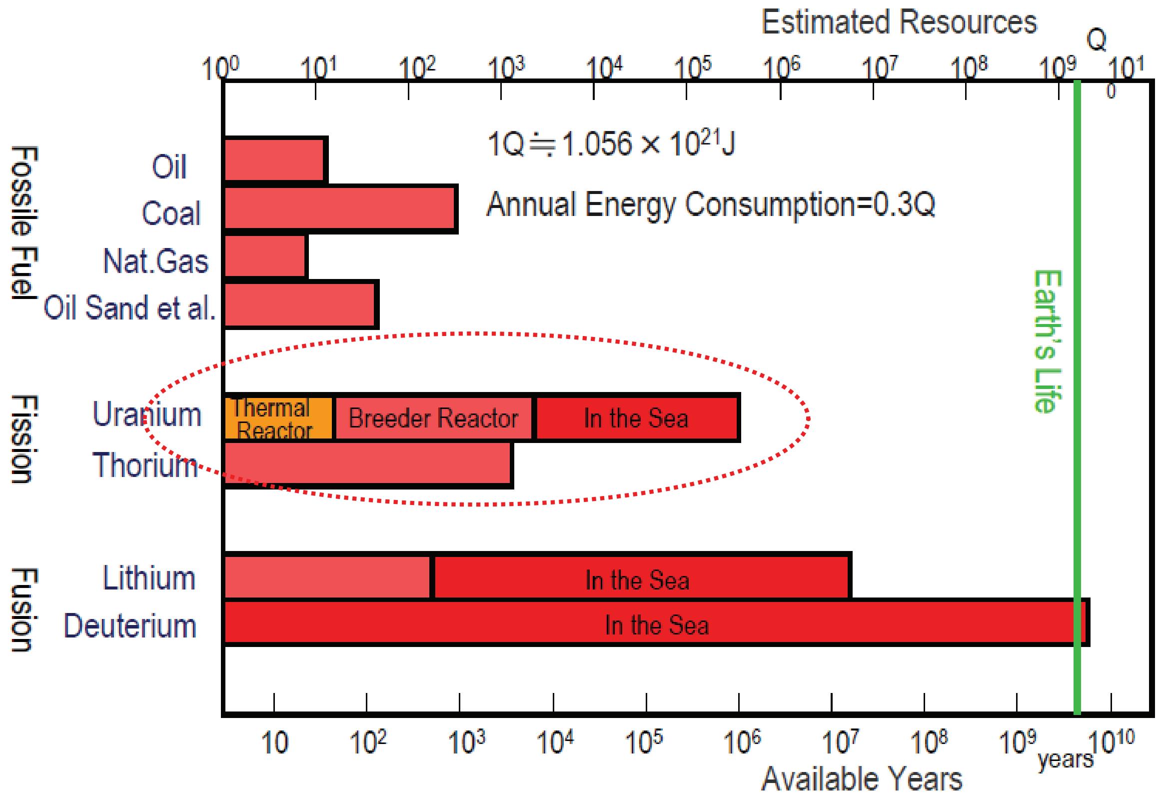
Estimated Energy Sources



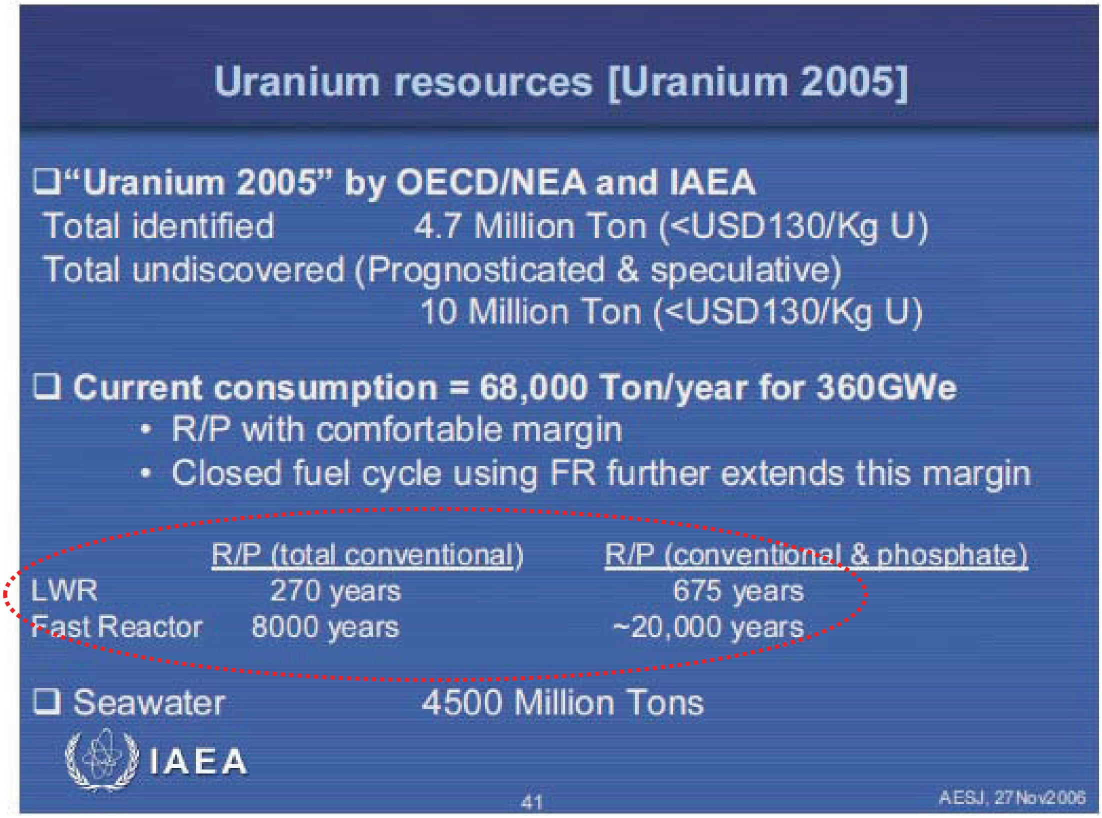
Reference : Presented paper, Contribution of Coal and Nuclear to Sustainable Energy Supply: Perspectives and Problems, President's meeting of G8 countries, Brazil, China, India and South Africa, Moscow 2006

Possible trends for the contribution of different sources to total energy supply in the next centuries

Estimated Energy Sources



Estimated Energy Sources



Uranium resources [Uranium 2005]

❑ **"Uranium 2005" by OECD/NEA and IAEA**

Total identified 4.7 Million Ton (<USD130/Kg U)
Total undiscovered (Prognosticated & speculative)
10 Million Ton (<USD130/Kg U)

❑ **Current consumption = 68,000 Ton/year for 360GWe**

- R/P with comfortable margin
- Closed fuel cycle using FR further extends this margin

	R/P (total conventional)	R/P (conventional & phosphate)
LWR	270 years	675 years
Fast Reactor	8000 years	~20,000 years

❑ Seawater 4500 Million Tons

IAEA

41

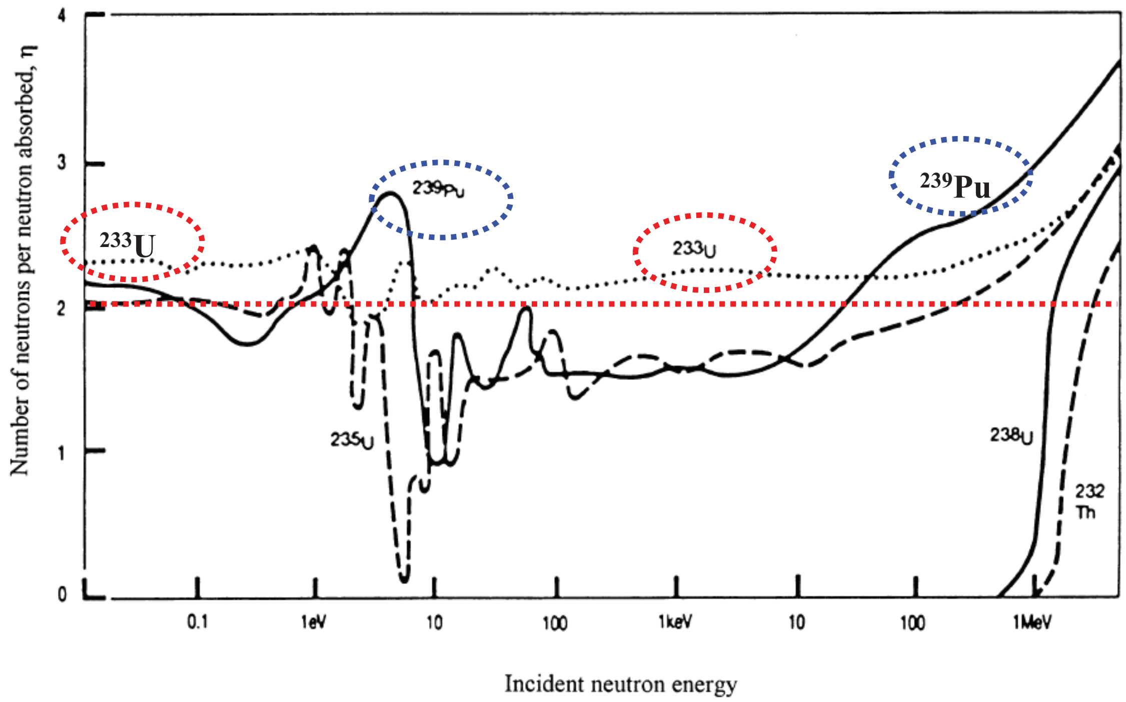
AESJ, 27Nov2006

Resources of Nuclear Fuel

Reasonably Assured Reserves (RAR) and Estimated Additional Reserves (EAR) of thorium comes from OECD/NEA, Nuclear Energy, "Trends in Nuclear Fuel Cycle", Paris, France (2001):

Country	RAR Th (tonnes)	EAR Th (tonnes)
Brazil	606,000	700,000
Turkey	380,000	500,000
India	319,000	—
United States	137,000	295,000
Norway	132,000	132,000
Greenland	54,000	32,000
Canada	45,000	128,000
Australia	19,000	—
South Africa	18,000	—
Egypt	15,000	309,000
Other Countries	505,000	—
World Total	**2,230,000**	**2,130,000**

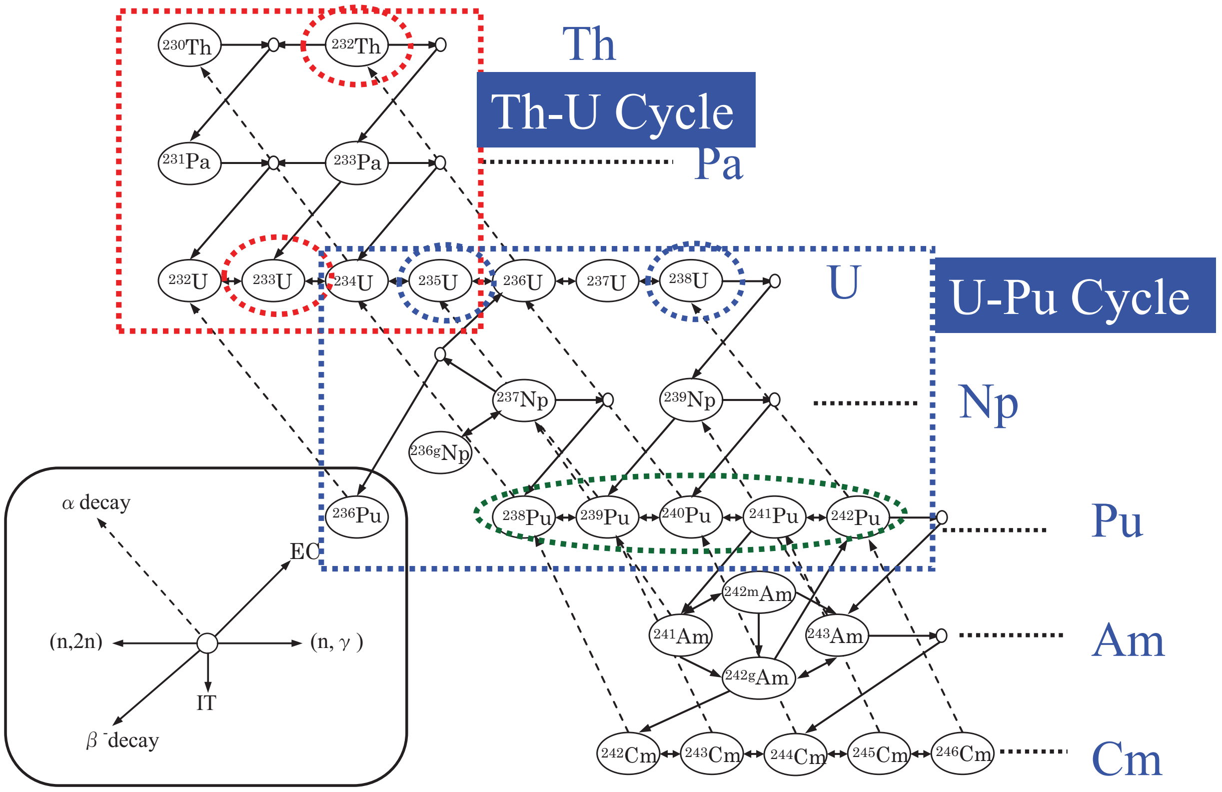
Possible Breeding of Each Fissile Material



Neutron regeneration ratio of each nuclide as a function of neutron energy

Refference : L. Michael and G. Otto, 1998

Nuclide Chain Mechanism



Water-Cooled Thorium Breeder Reactors
水冷却トリウム増殖炉

Content of Presentation

1. MOX fuel behavior on Water coolant reactors
2. Comparative analysis on physical properties of water coolant reactor for different fuel
3. Feasibility analysis on water-cooled breeder reactor
4. Feasibility analysis on water-cooled breeder reactor with MA doping as supply fuel
5. Core design analysis on water-cooled breeder reactor

Water-Cooled Thorium Breeder Reactors
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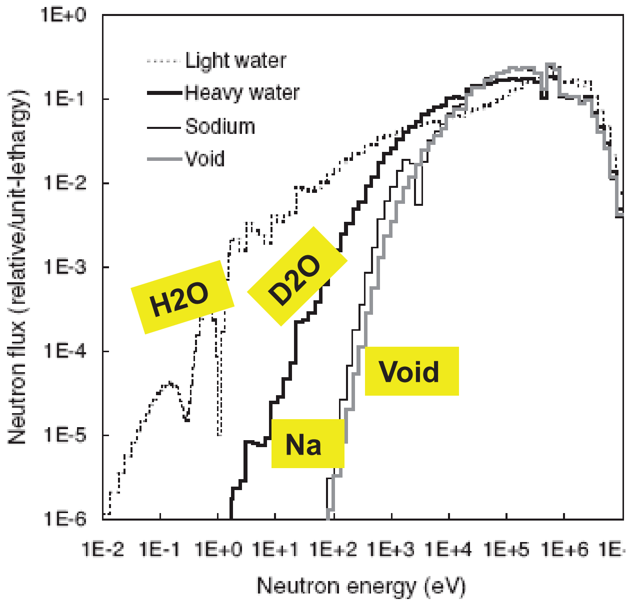
MOX fuel properties of Water Cooled Reactors

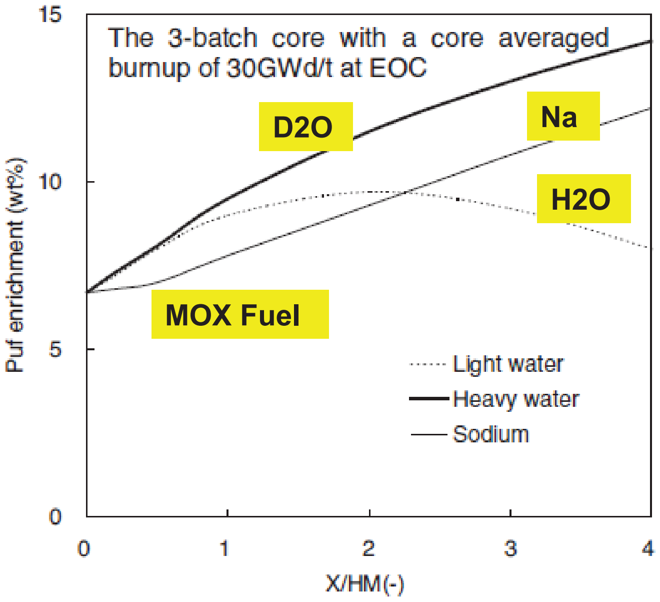
MOX Fuel Behavior [1]

Neutron Spectrum for diff. coolants



Hard Spectrum : Void>Na>D2O>H2O

Plutonium Composition [wt %]

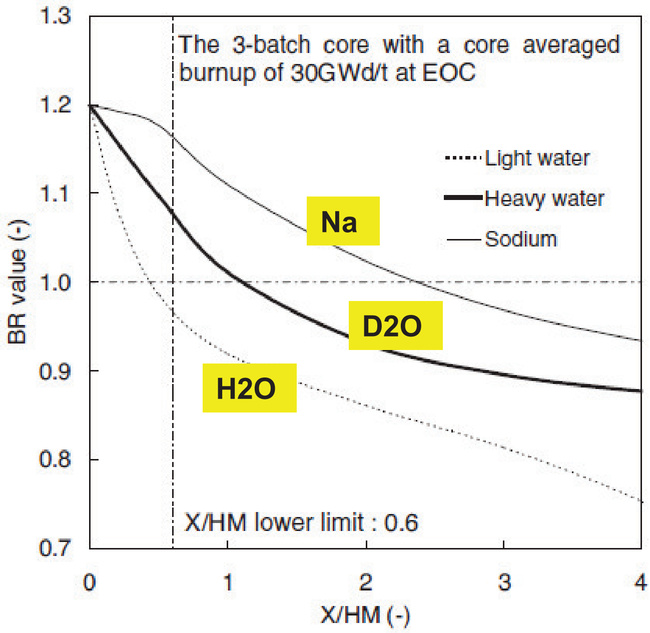


Fissile Content (X/HM<2):Na<H2O<D2O

Ref : - Hibi and Sekimoto / Journal of nucl. Science and technol, Vol. 42, No. 2, p. 153–160 (2005)

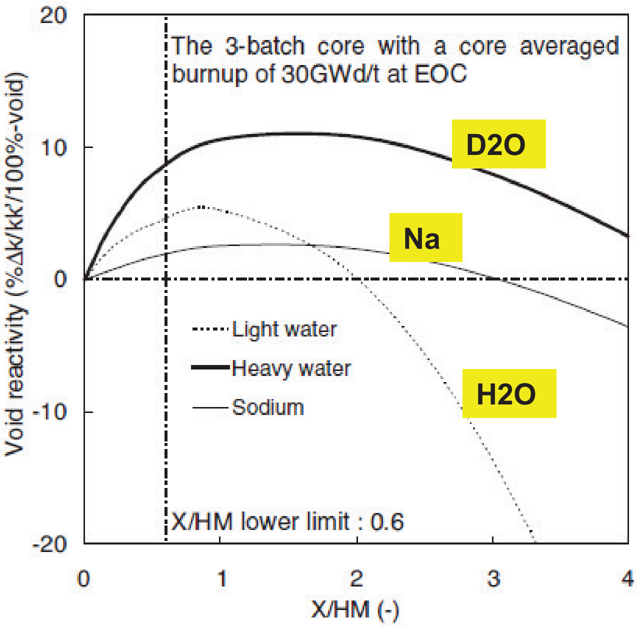
MOX Fuel Behavior [2]

Breeding Ratio Profile



High BR : Na>D2O>H2O

Void Reactivity Coefficient



Less Void (X/HM<2):Na<H2O<D2O

Ref : - Hibi and Sekimoto / Journal of nucl. Science and technol, Vol. 42, No. 2, p. 153–160 (2005)

Water-Cooled Thorium Breeder Reactors
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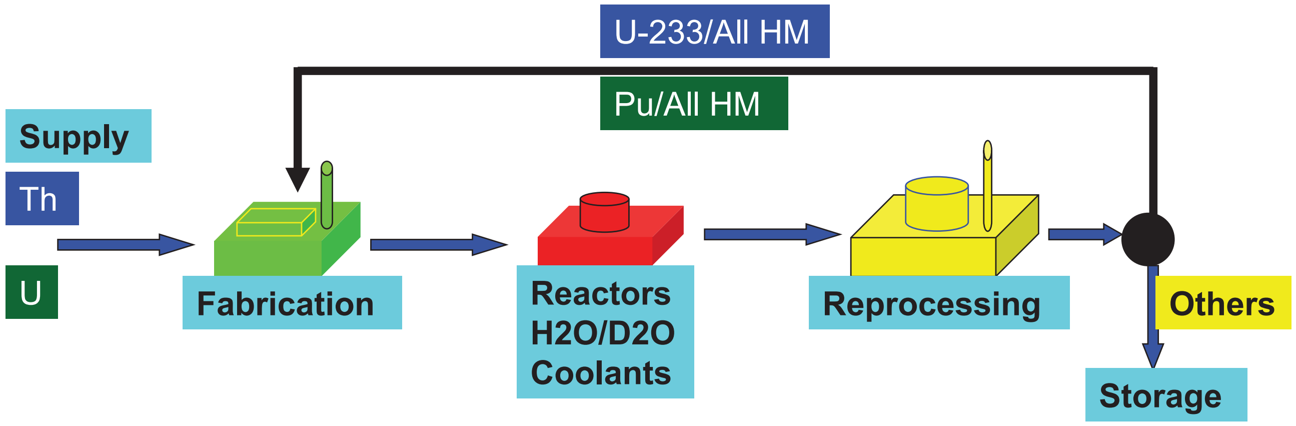
Comparative Analysis on Physical Properties of Water Cooled Reactors

Fuel Cycle Options

Supply Fuel : Natural Uranium or Thorium

Recycled Fuel : Plutonium or U-233 or All Heavy Metals (HM)



Physical Parameters : Neutron Spectrum, Eta-value

Investigated Parameters : Required Enrichment, Breeding and void reactivity coefficient

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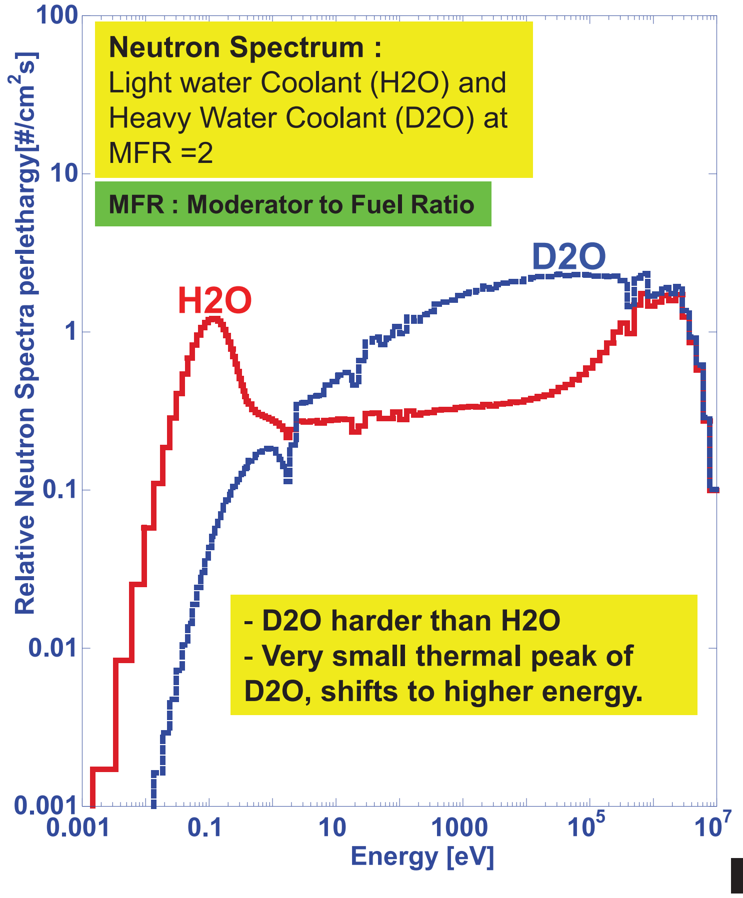
Neutron Spectrum[2]

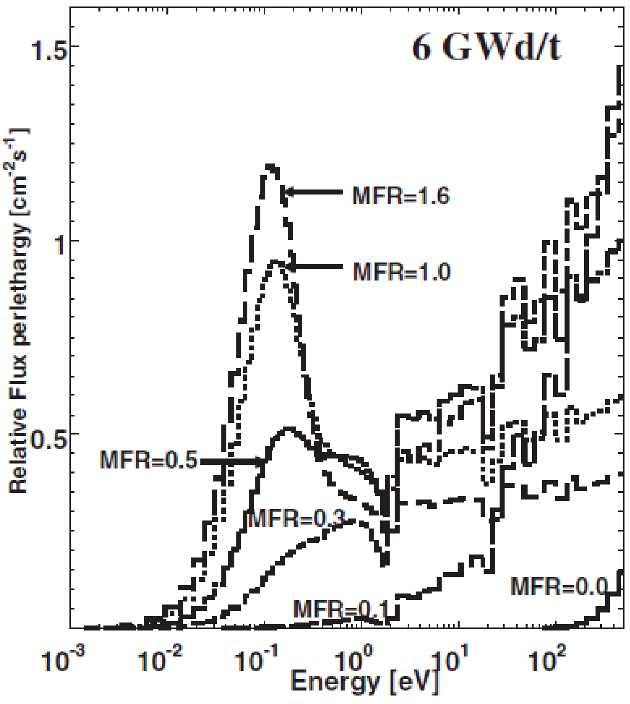


Neutron Spectrum :
Light water Coolant (H2O) and Heavy Water Coolant (D2O) at MFR =2

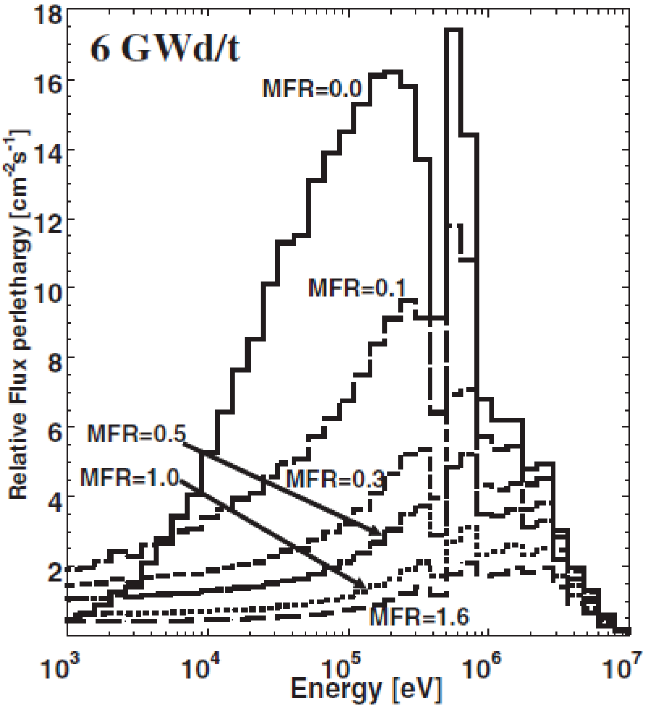
MFR : Moderator to Fuel Ratio

- D2O harder than H2O
- Very small thermal peak of D2O, shifts to higher energy.

Neutron Spectrum[4]



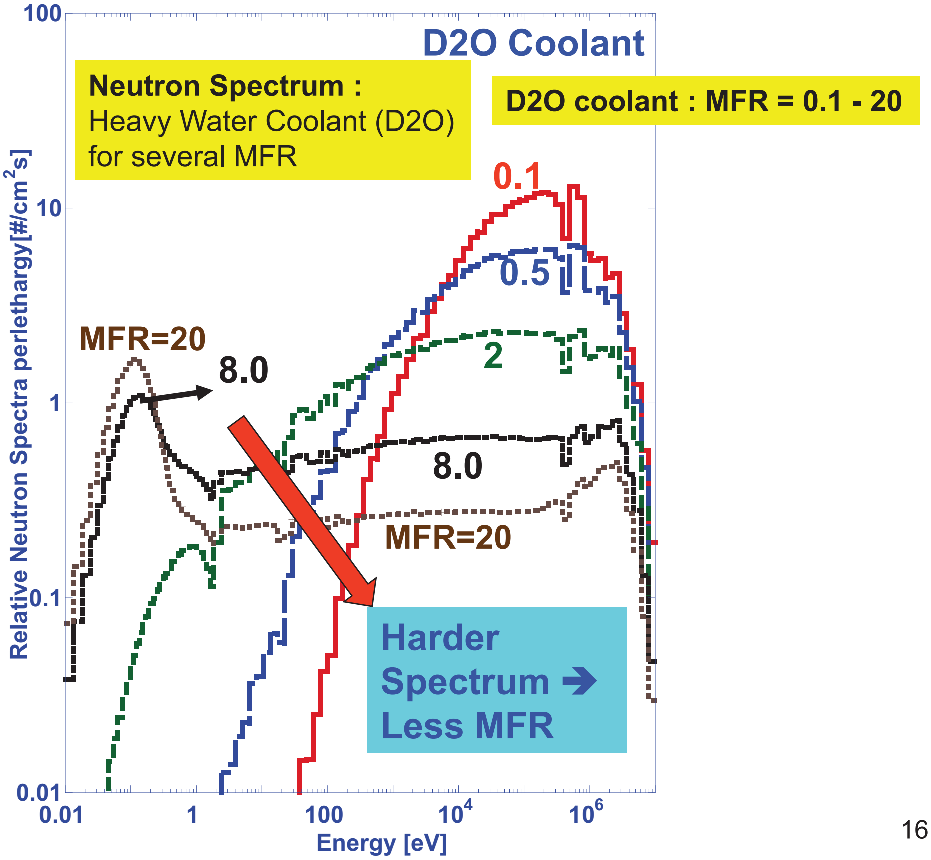
H2O coolant , MFR: 0.1 – 1.6
→ Low energy region < 1 keV.



H2O coolant
→ High energy region > 1 keV.

Ref : - S. Permana et al. / Journal of nucl. Science and technol, Vol. 44, No. 7, p. 946–957 (2007)

Neutron Spectrum[4]

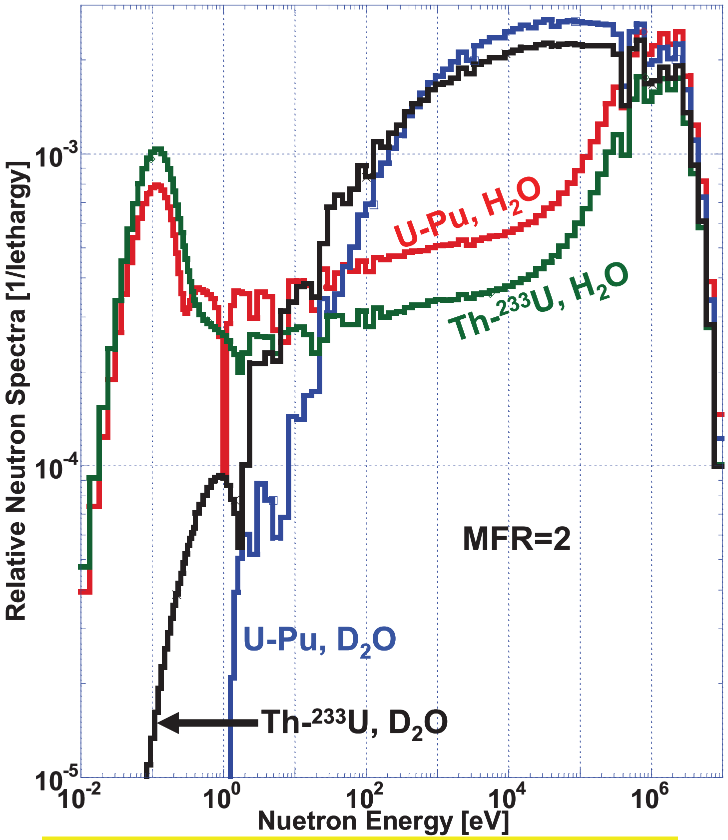


Neutron Spectrum :
Heavy Water Coolant (D2O)
for several MFR

D2O coolant : MFR = 0.1 - 20

Harder Spectrum → Less MFR

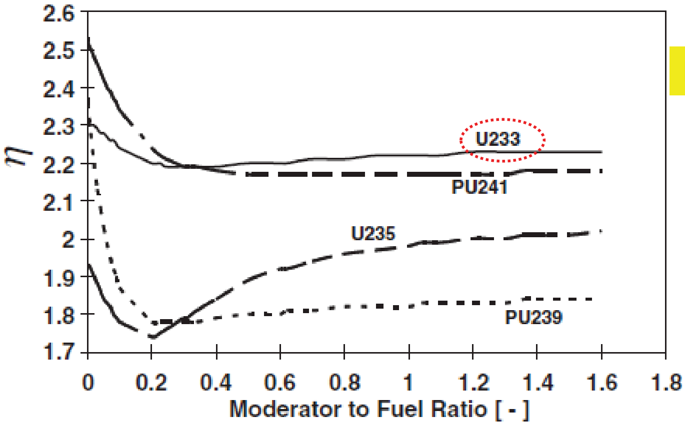
Neutron Spectrum[3]



Thorium fuel shows softer than U-Pu fuel for both water coolants

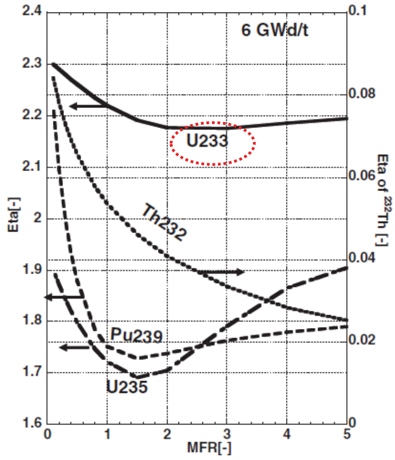
H2O and D2O Coolants for different fuels

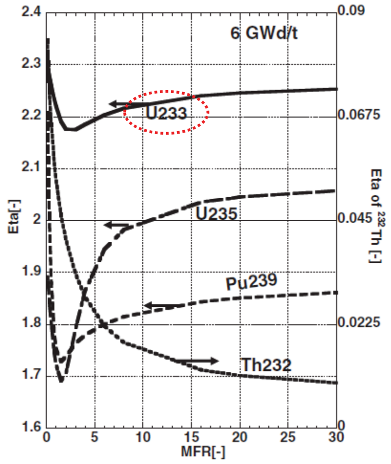
Eta Value [1]



H2O coolant, MFR : 0 – 1.6

Eta value of U-233 → superior than other fissile materials



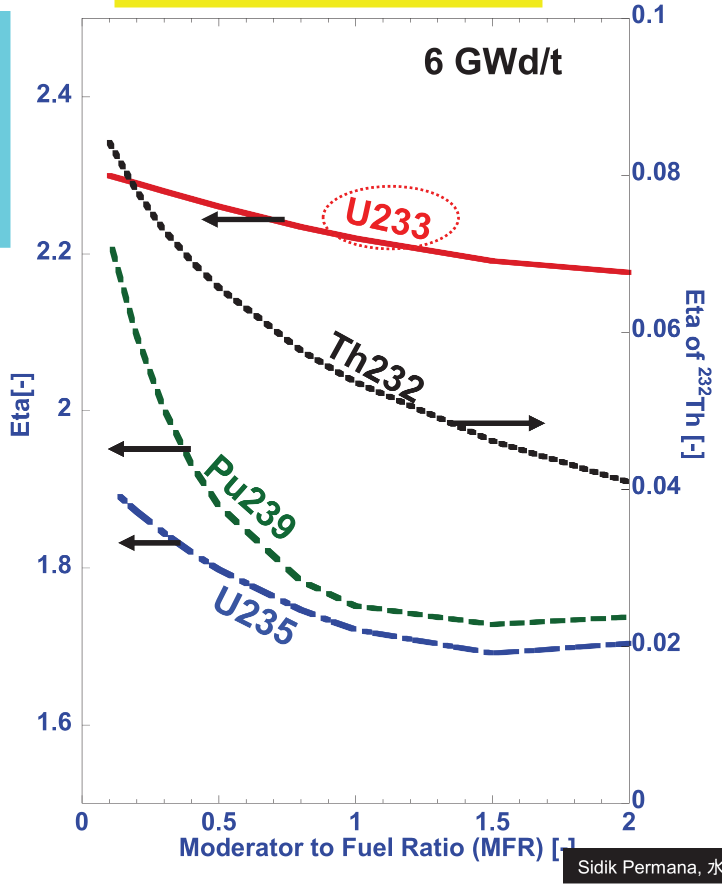


D2O coolant, MFR : 0 – 30

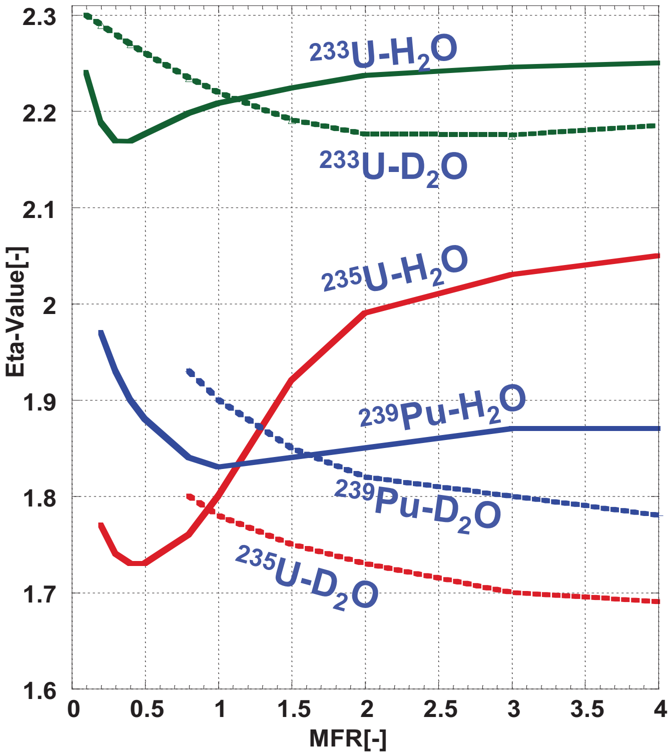
Eta Value [2]

Eta value of U-233 → superior than other fissile materials and almost constant along the MFR

D2O coolant, MFR : 0.1 – 2



Eta Value [3]

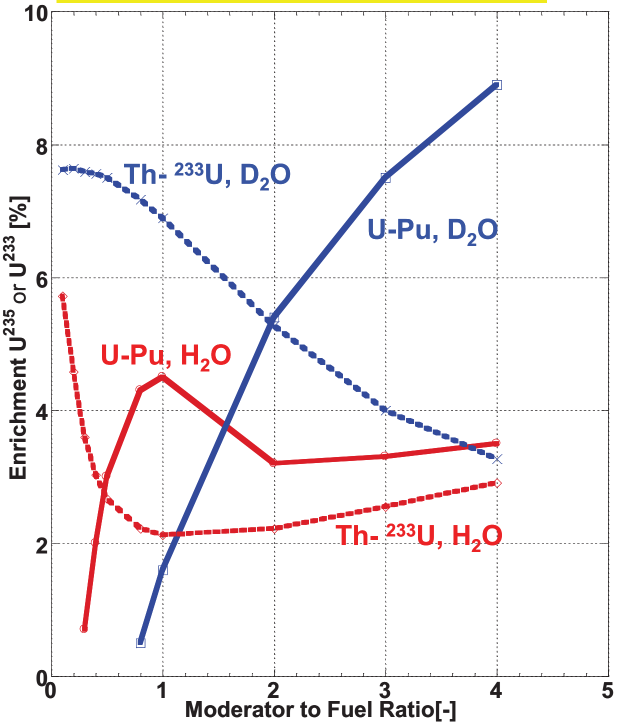


H2O and D2O Coolants for different fuels

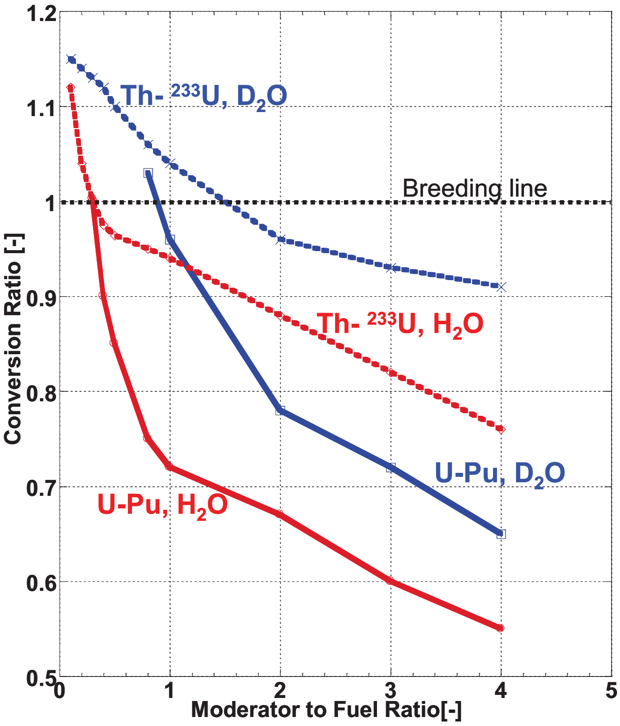
Eta value of U-233 → superior than other fissile materials along the MFR

Fissile Content and Conversion Ratio

Required Fissile Content

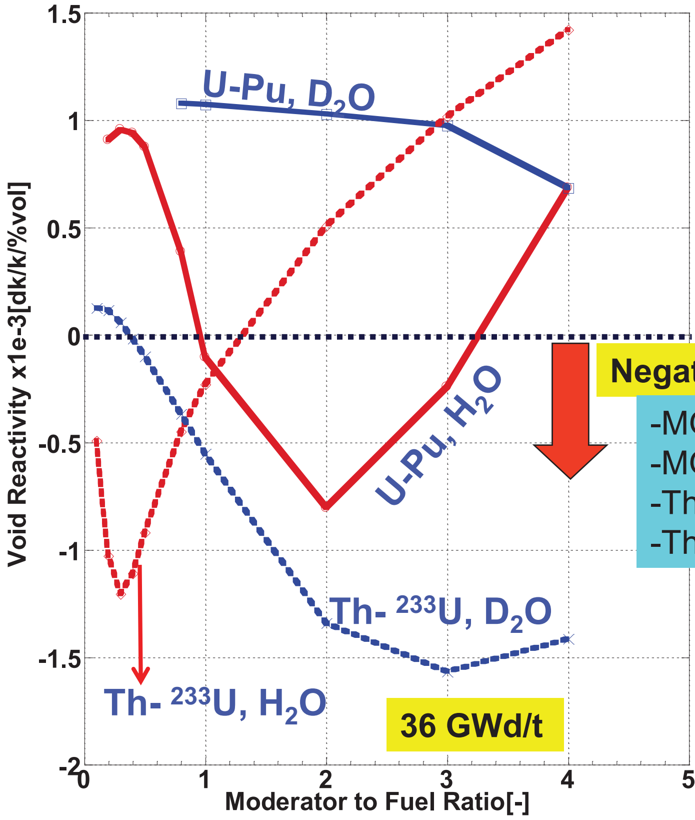


Conversion Ratio



H2O and D2O Coolants for different fuels

Void Reactivity Coefficient



Negative Void Coefficient

- -MOX_D2O : Always Positive
- -MOX_H2O : Negative 1<MFR<3.4)
- -Th_U233_D2O : Negative for MFR>0.4
- -Th_U233_H2O : Negative for MFR<1.4

Water-Cooled Thorium Breeder Reactors
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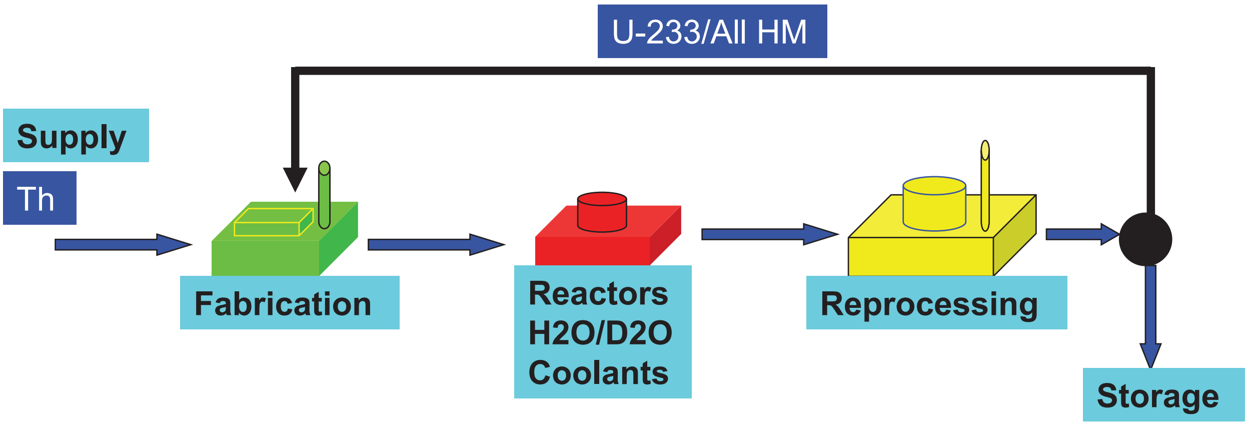
Feasibility Analysis on Water-Cooled Breeder Reactor

Fuel Cycle Options

Supply Fuel : Thorium

Recycled Fuel : U-233 or All Heavy Metals (HM)



U-233/All HM

Supply

Th

Fabrication

Reactors
H2O/D2O
Coolants

Reprocessing

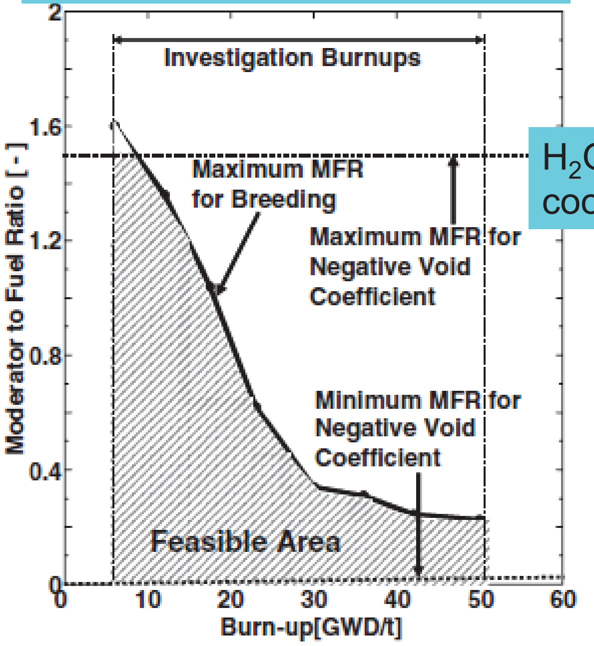
Storage

Physical Parameters : Neutron Spectrum, Eta-value

Investigated Parameters : Required Enrichment, Breeding and void reactivity coefficient

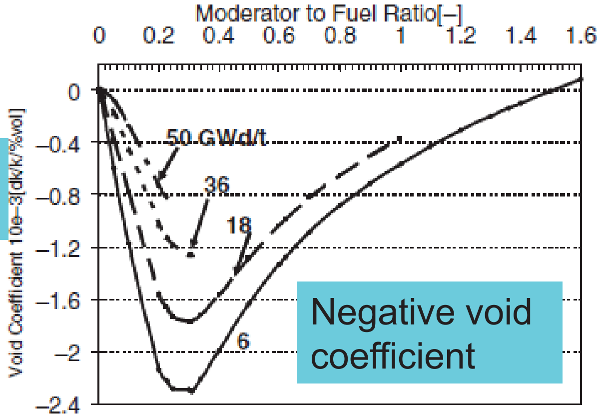
Feasible Breeding and Negative Void Reac.

Feasible area of breeding and negative void coefficient



H_2O coolant

Void coefficient profile



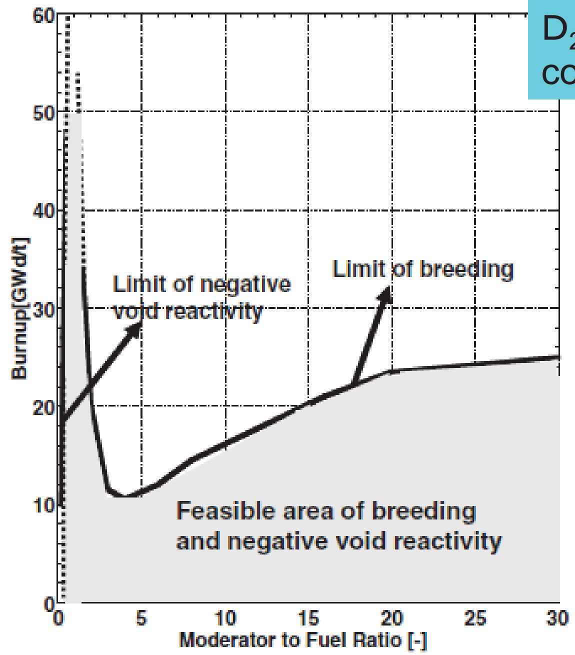
Negative void coefficient

Light water coolant →
Shows A feasible design area for breeding and negative void reactivity coefficient

Ref : - S. Permana et al. / Journal of nucl. Science and technol, Vol. 44, No. 7, p. 946–957 (2007)
- Global, 2007

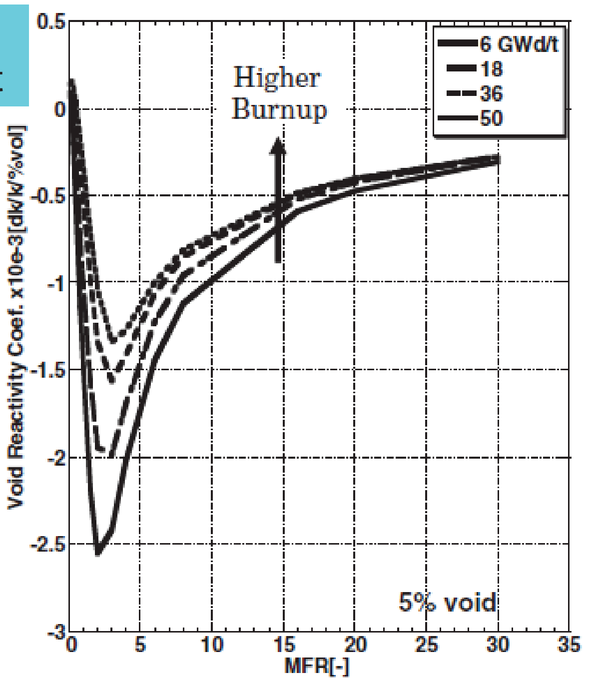
Feasible Breeding and Negative Void Reac.

A feasible design of breeding and negative void coefficient



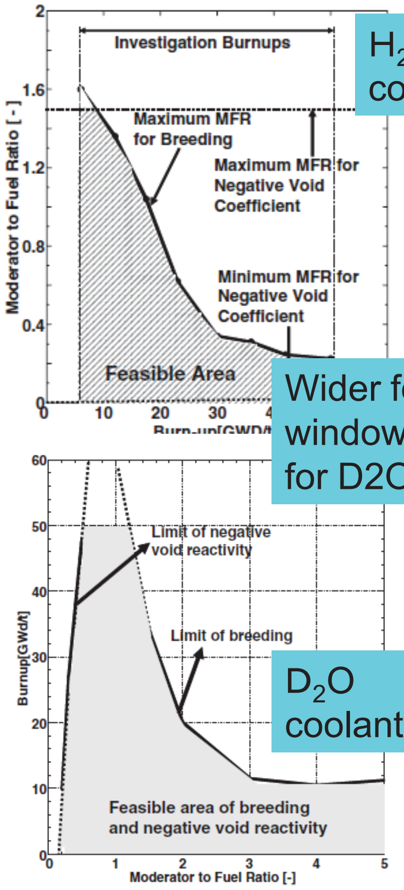
D_2O coolant

Void coefficient Profile



Ref : - S. Permana et al. / Journal of nucl. Sciece and technol, Vol. 45, No. 7, p. 589–600 (2008)
- Global, 2007

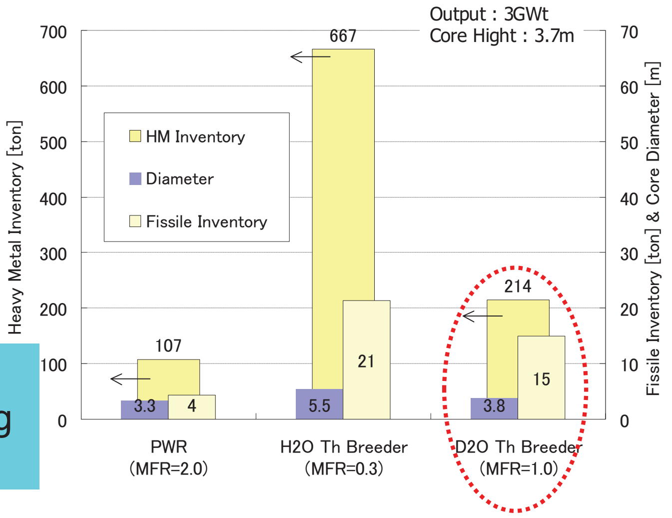
Comparative H2O and D2O Coolants



H_2O coolant

Wider feasible window of breeding for D2O-cooled

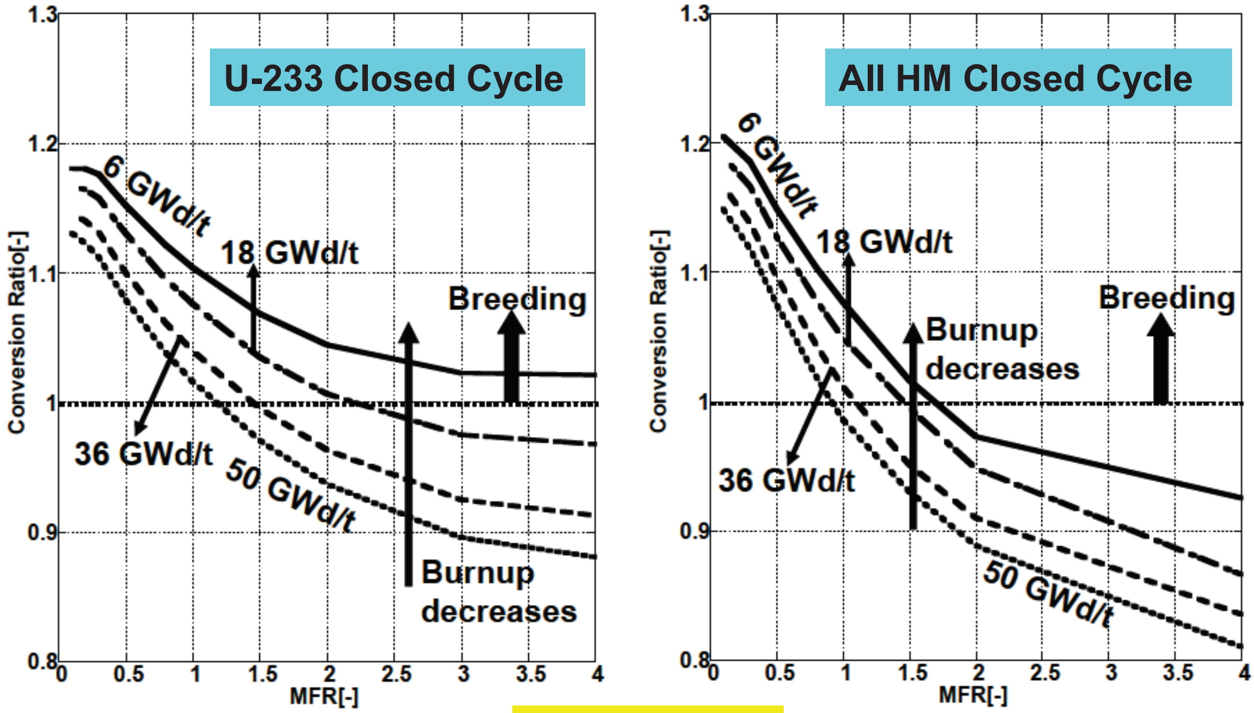
D_2O coolant



D2O-cooled breeder reactor
- MFR=1
- Pellet power density of 140W/cc
- Burn-up of 50 GWd/t.

Ref :
- S. Permana et al. / Journal of nucl. Sciece and technol, Vol. 45, No. 7, p. 589–600 (2008)
- S. Permana et al. / Journal of nucl. Sciece and technol, Vol. 44, No. 7, p. 946–957 (2007)
- Global, 2007

Comparative U-233 and All HM closed Cycle

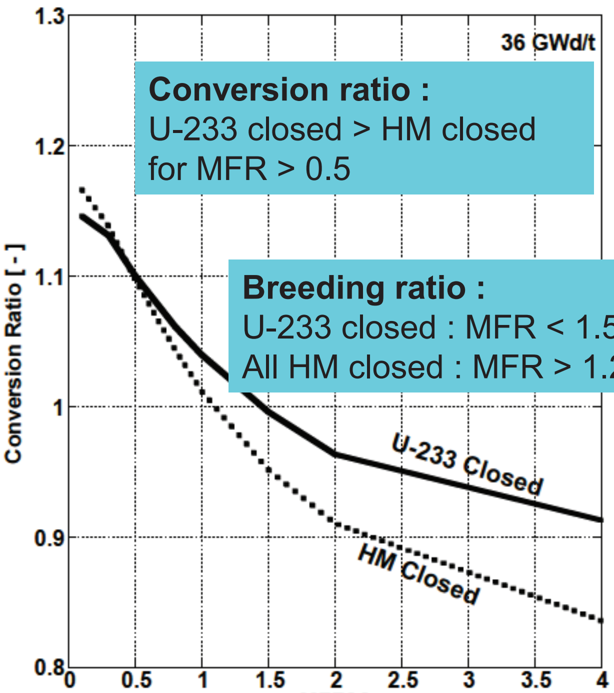

U-233 Closed Cycle

All HM Closed Cycle

D2O Coolant

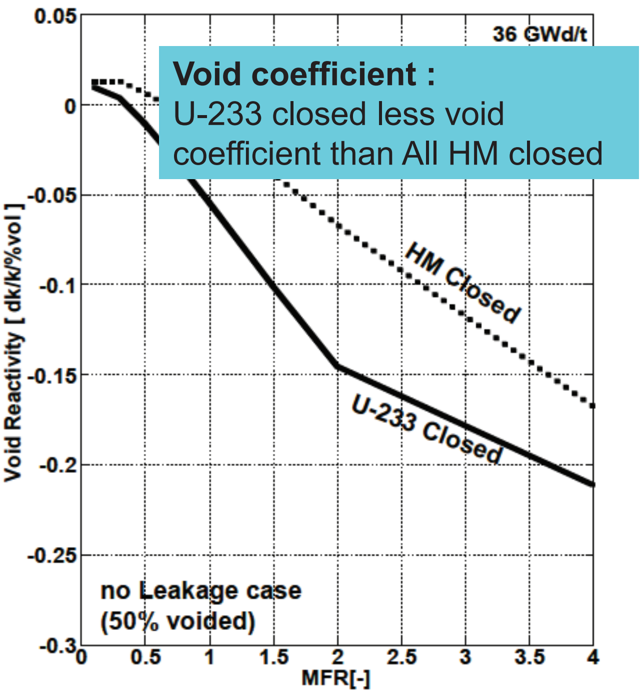
Ref :
- S. Permana et al. / Annals of Nuclear Energy, Accepted, 2010

Comparative U-233 and All HM closed Cycle



Conversion ratio :
U-233 closed > HM closed
for MFR > 0.5

Breeding ratio :
U-233 closed : MFR < 1.5
All HM closed : MFR > 1.2

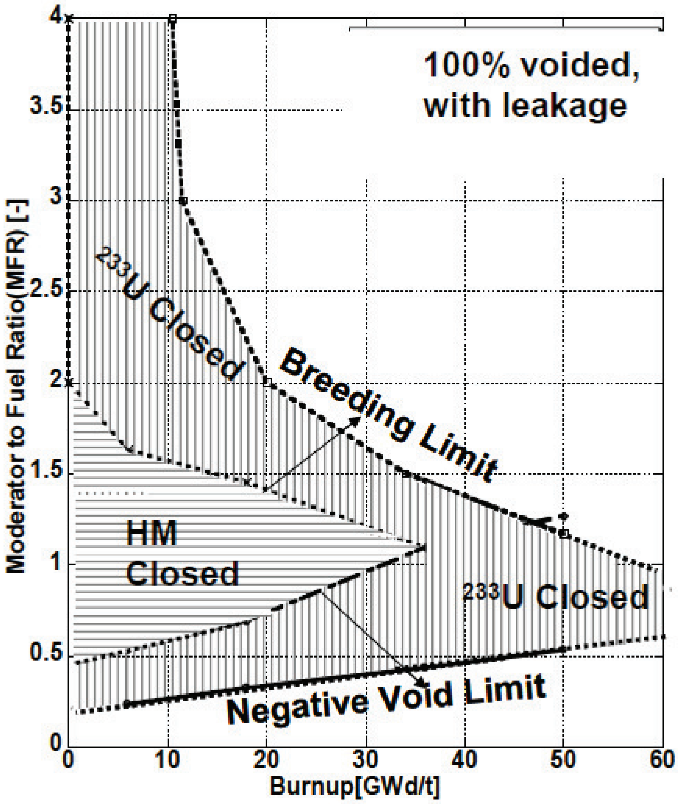


Void coefficient :
U-233 closed less void
coefficient than All HM closed

D2O Coolant

Ref :
- S. Permana et al. / Annals of Nuclear Energy, Accepted, 2010

Comparative U-233 and All HM closed Cycle



Heavy water coolant for both U-233 only recycling and All HM recycling options → Shows A feasible design area for breeding and negative void reactivity coefficient

U-233 only recycling case obtains wider feasible design area for breeding and negative void reactivity coefficient than All HM closed cycle.

Ref :
- S. Permana et al. / Annals of Nuclear Energy, Accepted, 2010

Water-Cooled Thorium Breeder Reactors
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Feasibility Analysis on Water-Cooled Breeder Reactor with MA doping

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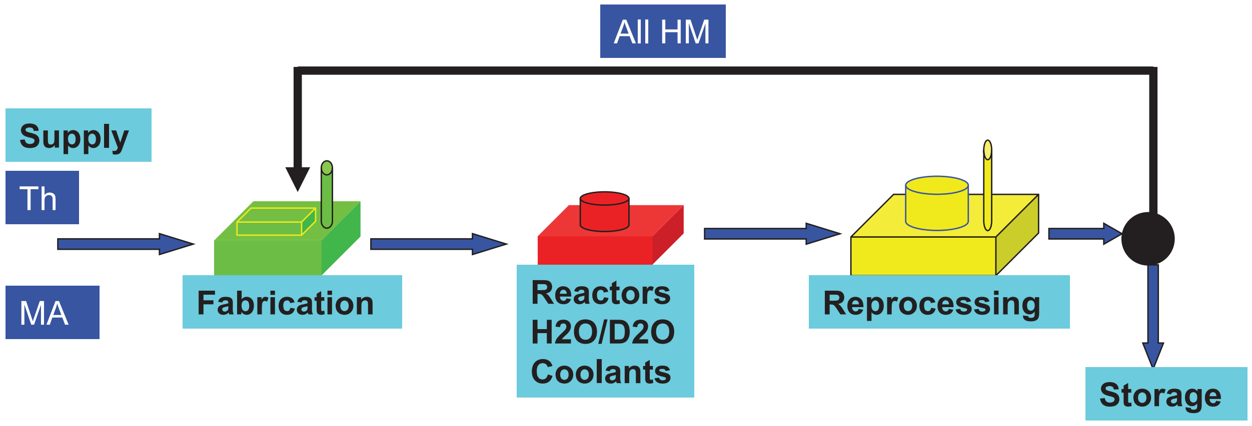
Fuel Cycle Options

Supply Fuel : Thorium

Recycled Fuel : U-233 or All Heavy Metals (HM)



All HM

Supply

Th

MA

Fabrication

Reactors
H2O/D2O
Coolants

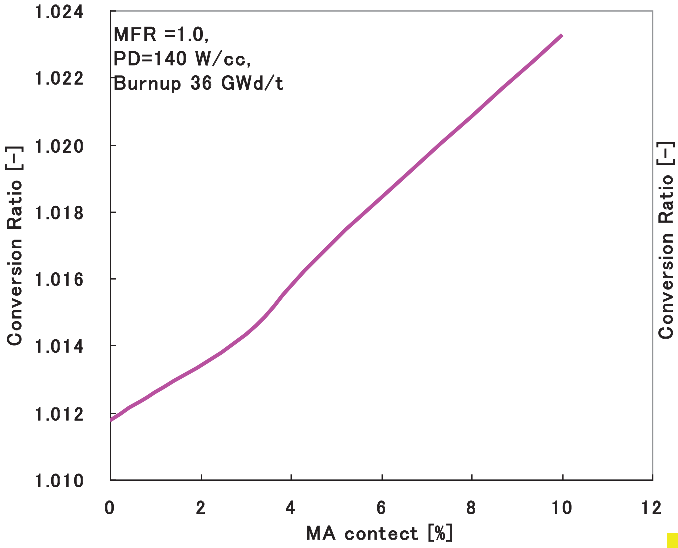
Reprocessing

Storage

Investigated Parameters : Required Enrichment, Breeding and void reactivity coefficient

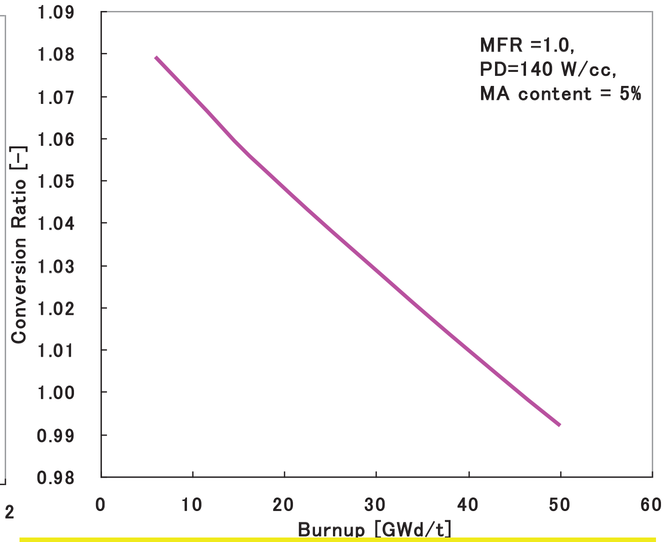
Conversion Ratio

Effect of MA content (%)



By doped MA, the breeding capabilities are improved. Higher breeding for Higher MA doped.

Effect of Burnup

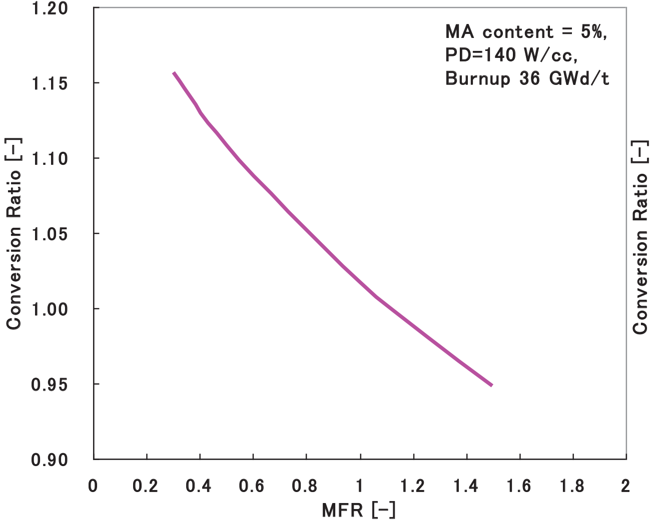


- Conversion ratio monotonically decreases with increasing burnup.

- Breeding can be achieved for burnup < 45 GWd/t.

Conversion Ratio

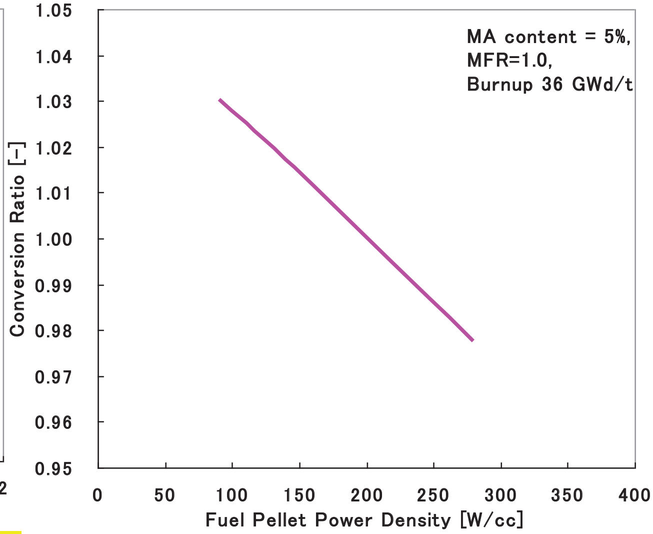
Effect of MFR



- Breeding capability monotonically decreases with increasing MFR.

- Breeding can be achieved for MFR < 1.1

Effect of Power Density



- Breeding capability monotonically increases with decreasing power density (PD) of fuel pellet.

-Breeding can be achieved for fuel pellet PD ≤ 200 W/cc.

Water-Cooled Thorium Breeder Reactors
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Core Design Analysis on Water-Cooled Breeder Reactor

Objectives and Evaluation

Evaluate the reactor core performances and fuel management by using core burnup of SRAC COREBN calculations which adopted 2-dimensional hexagonal model as the core fuel configuration.

- Basic Reactor : Heavy water cooled thorium reactors
- Core configuration : Refers to optimum result of equilibrium cycle iterative calculation systems (ECICS)

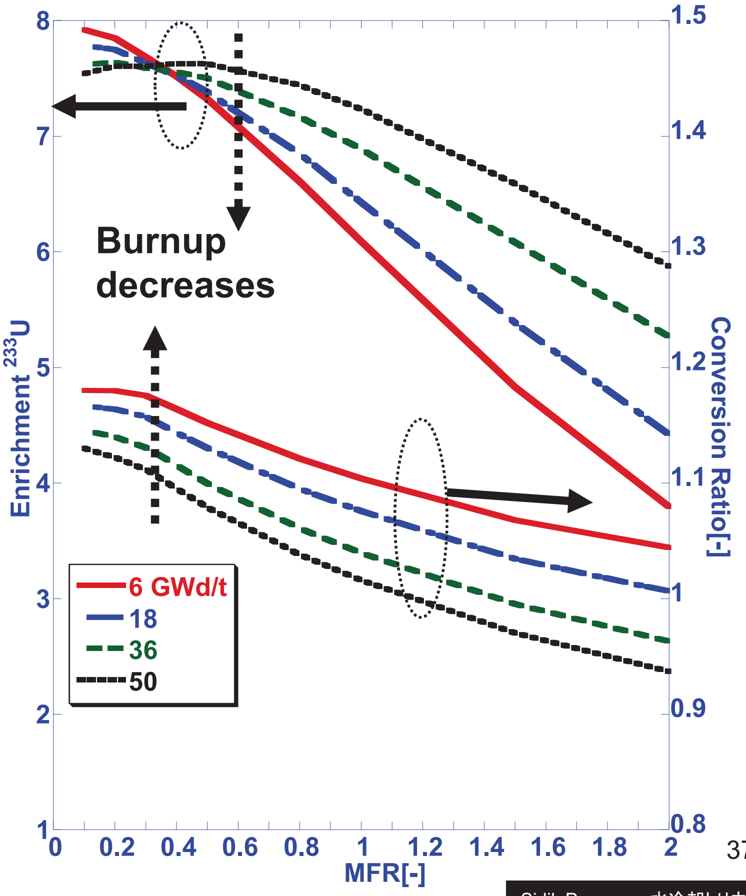
-Refueling modes : Once Through methods.
- Three batches core configuration systems.
- Preliminary Thermal Hydraulic Analysis.

- Fuel Breeding Capability and Reactor Criticality
- Negative void reactivity coefficient. -
Thermal Hydraulic Properties comparable to PWR.

Required enrichment and Conversion ratio

High burnup → High required enrichment or fissile content, except for very low MFR with less moderator

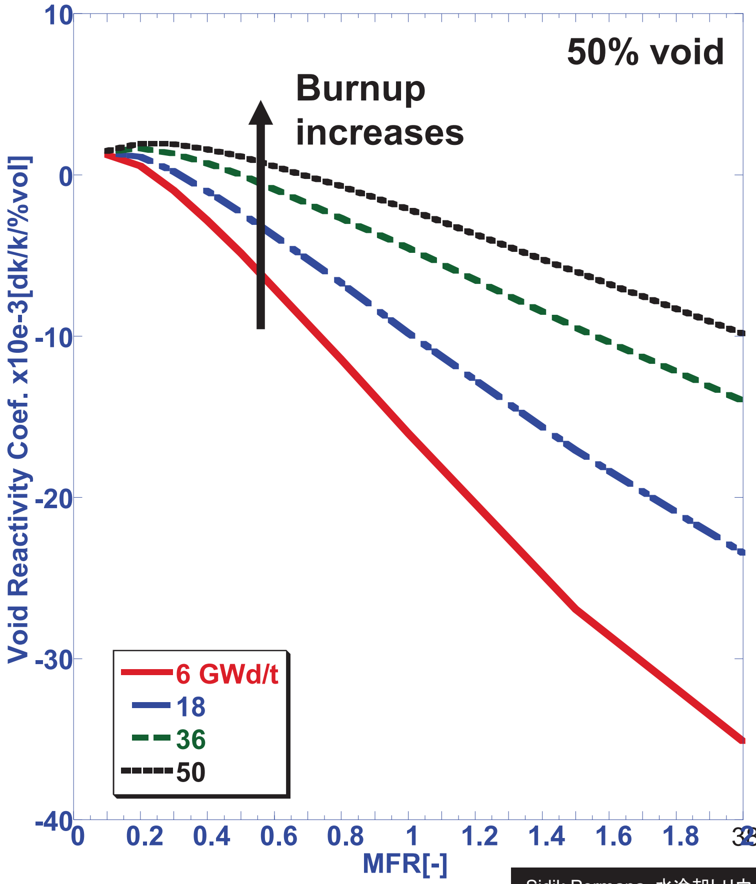
High burnup → less conversion ratio (higher consumption ratio of fissile)



Void Reactivity Coefficient

High burnup → Less void reactivity coefficient or becomes positive void.

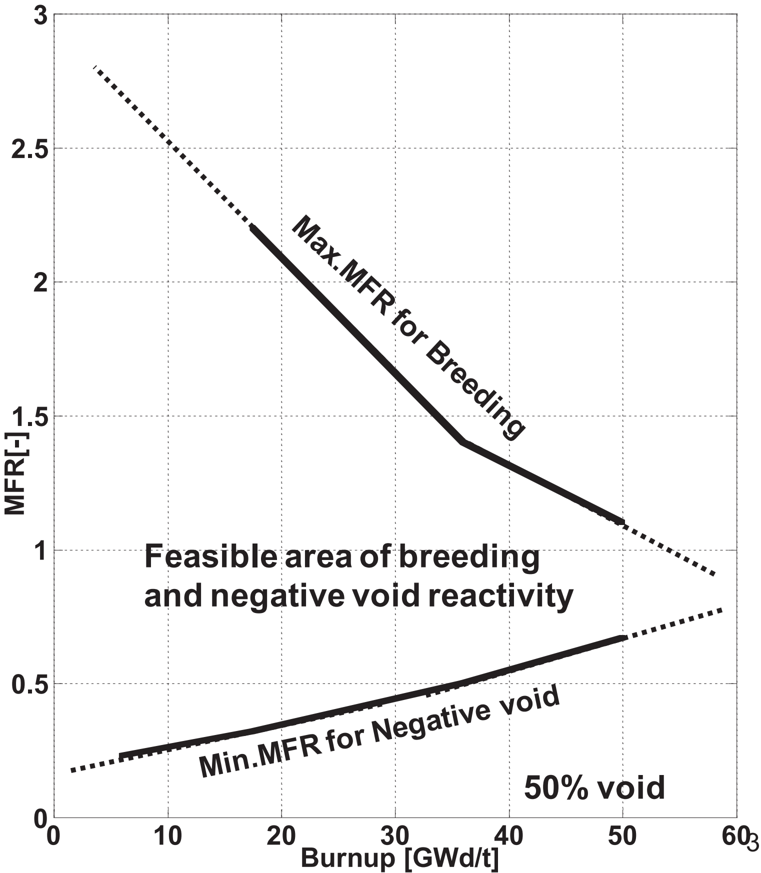
High MFR → High Void reactivity coefficient



Feasible Breeding Area

High burnup → Narrow feasible area of breeding and negative void coefficient

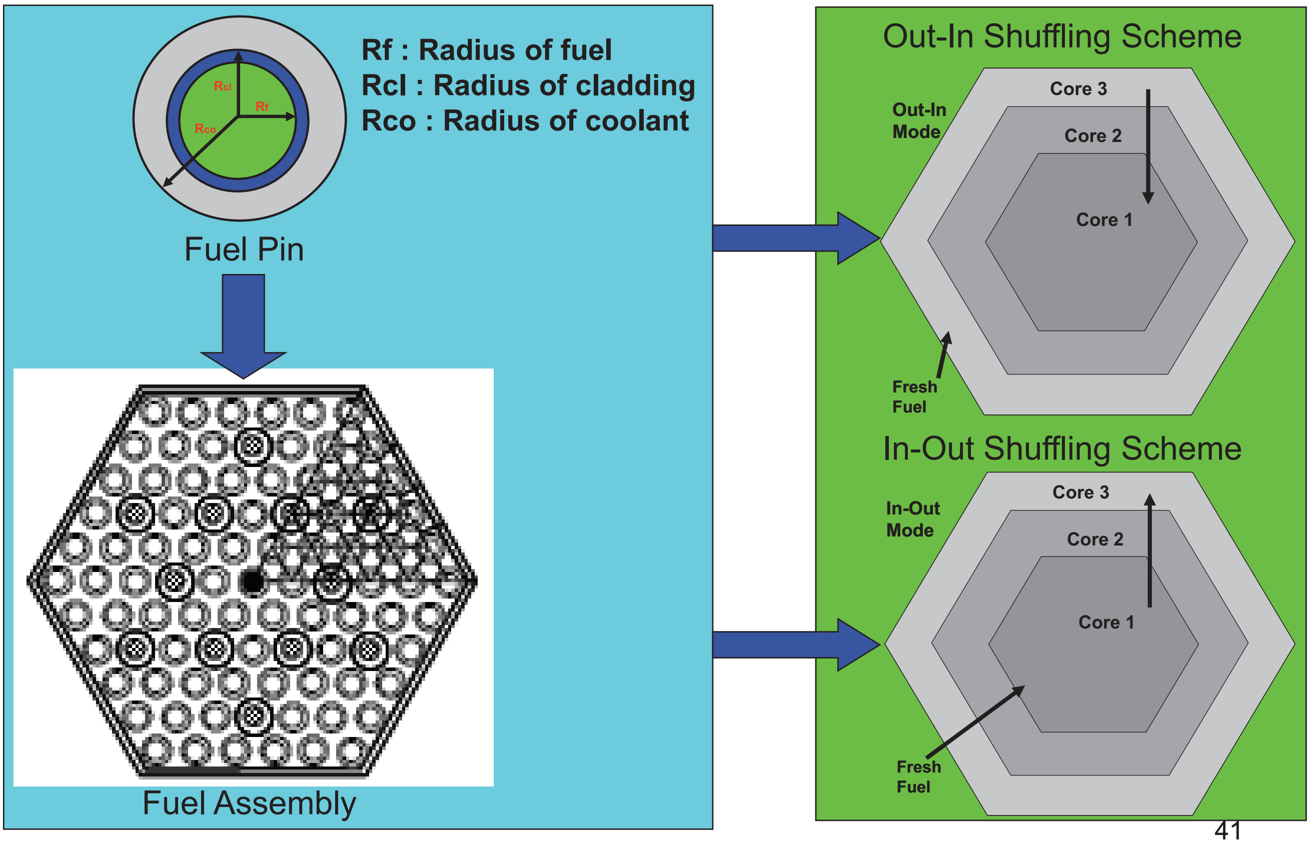
Less MFR → preferable to have better breeding, however, its limited by negative void reactivity limitation



Basic Parameters

Parameters		Unit
Power	3411	MWt
Core Height (no reflector)	370	cm
Core Radius	179	cm
Reflector width	24	cm
Fuel pellet Diameter	1.31	cm
Fuel Pin Diameter	1.45	cm
MFR (without cladding)	1	-
Pin Pitch Gap	0.4	cm
Pin Pitch	1.86	cm
P/D	1.282	-
U-233 Enrichment	6.8	%
Cycle Length	700	Days
Achievable burnup	38	GWd/t
Refueling Scheme	3	batch

Calculation Schemes

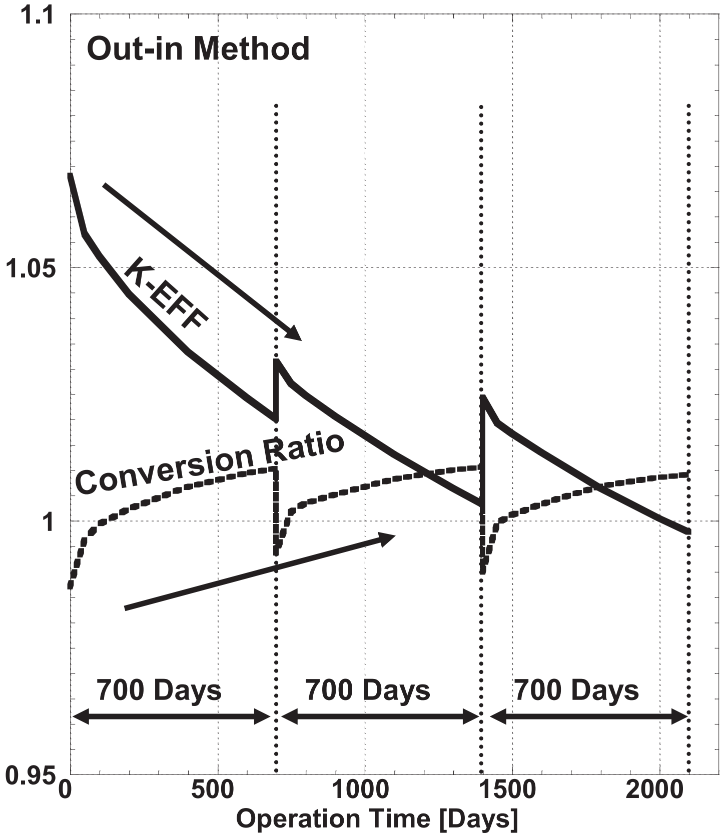


Rf : Radius of fuel
Rcl : Radius of cladding
Rco : Radius of coolant

Fuel Pin

Fuel Assembly

Criticality and Breeding Profiles



Cycle Length : 700 Days
Fuel Core Batches : 3 Batches

Criticality (K-Eff) : Decreases with Reactor Operation Time

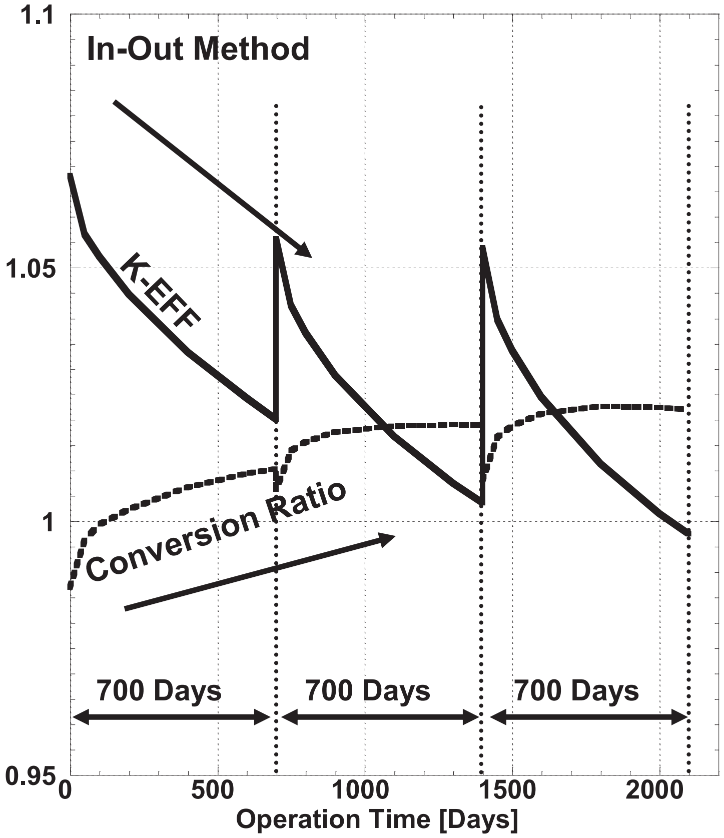
Breeding (Conversion Ratio): Increases with Operation Time

At BOC : Breeding → less than unity
At EOC : Breeding → High than unity

Achievable Discharged Fuel Burnup: More than 33 GWd/t

Out In Method :
→Less criticality for next recycling step
→Conversion ratio starts from less than unity at BOC and reaches higher than unity at EOC.
→It confirmed breeding can be achieved

Criticality and Breeding Profiles



Cycle Length : 700 Days
Fuel Core Batches : 3 Batches

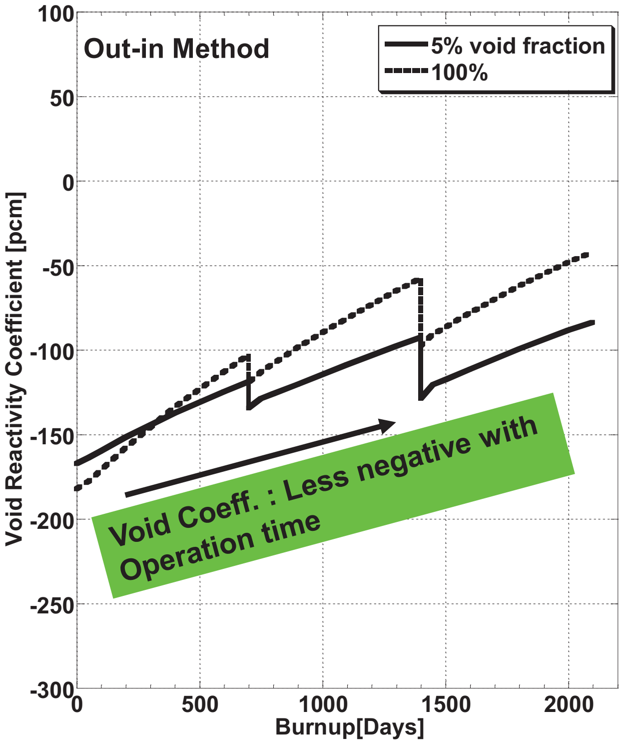
Criticality (K-Eff) : Decreases with Reactor Operation Time

Breeding (Conversion Ratio): Increases with Operation Time

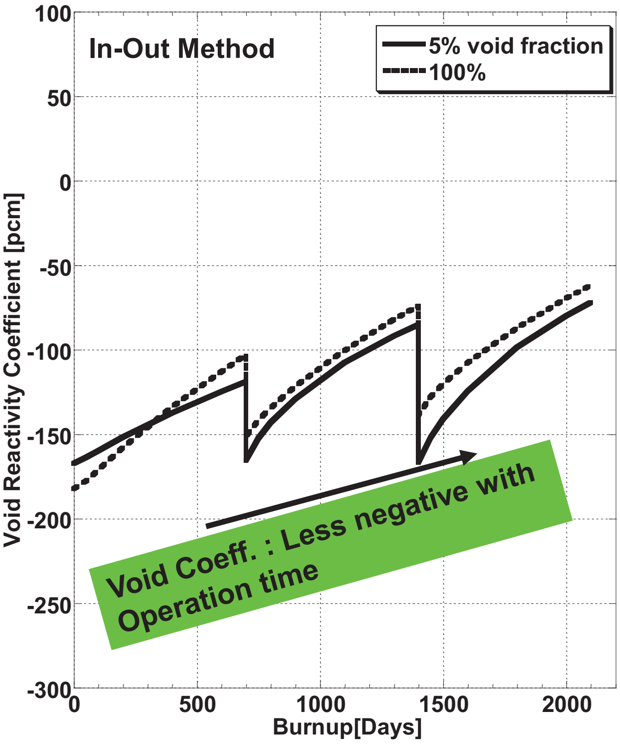
Out In Method :
→It confirmed breeding can be achieved

Breeding (Conversion Ratio):
In-Out Method : 1.01
Out-In Method : 1.02

Void Reactivity Coefficient

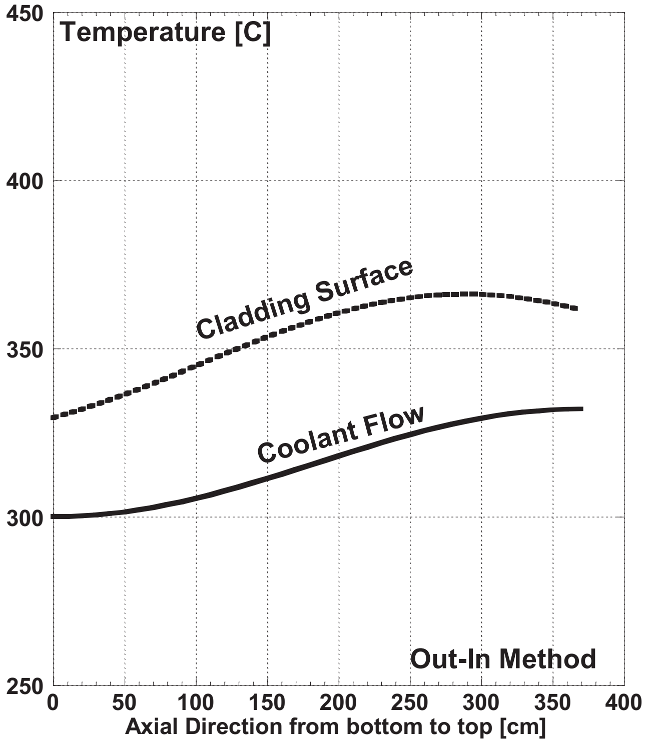


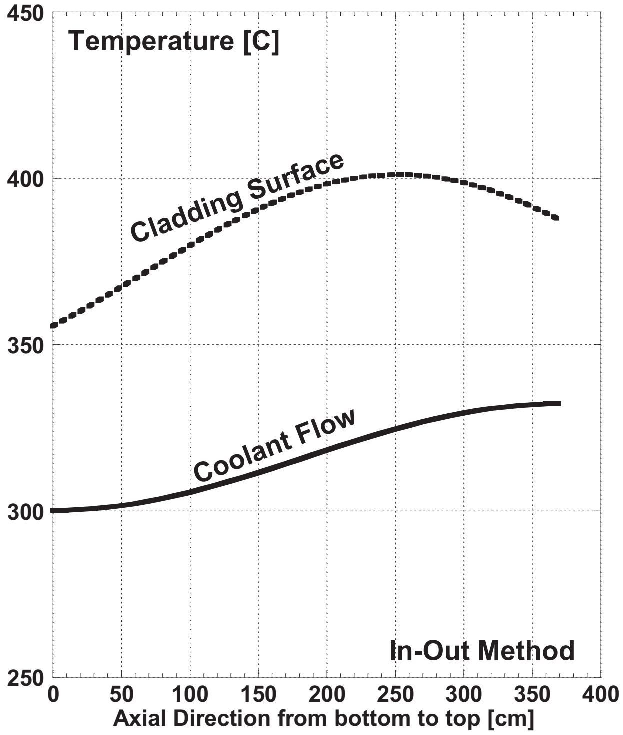
Void reactivity coefficient :
→ Always negative during reactor operation
→ Higher void fraction : less negative at EOC



Void reactivity coefficient :
→ Voided fraction effect : low for out-in method and higher for in-out method

Temperature Distribution





Temperature :
→ Maximum Temp. cladding surface : Out-In : Less 400 C, In-out : reaches 400 C

Temperature:
→ Out-in method : Relatively Higher temperature than In-out method

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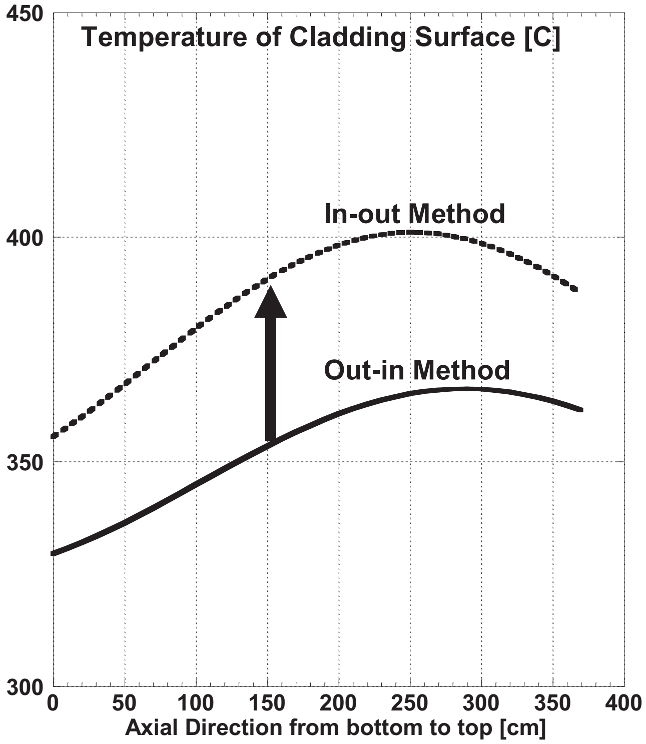
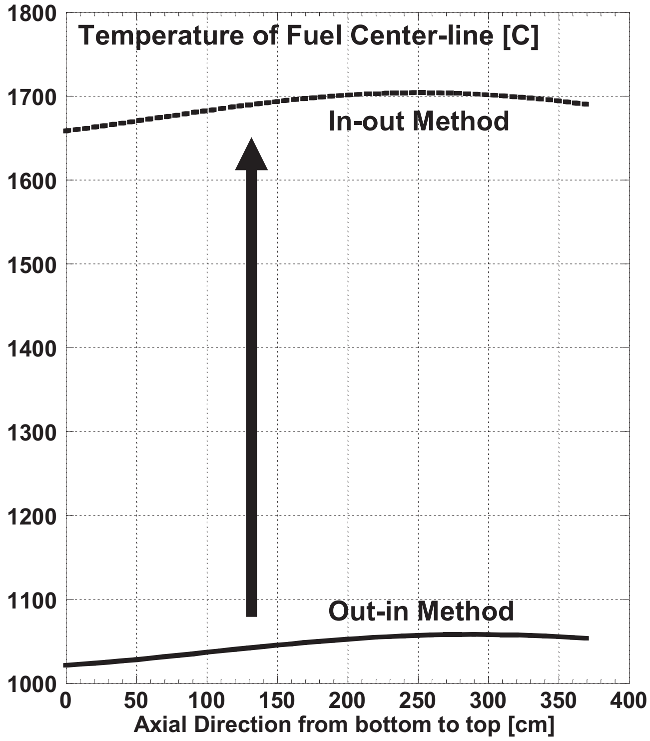
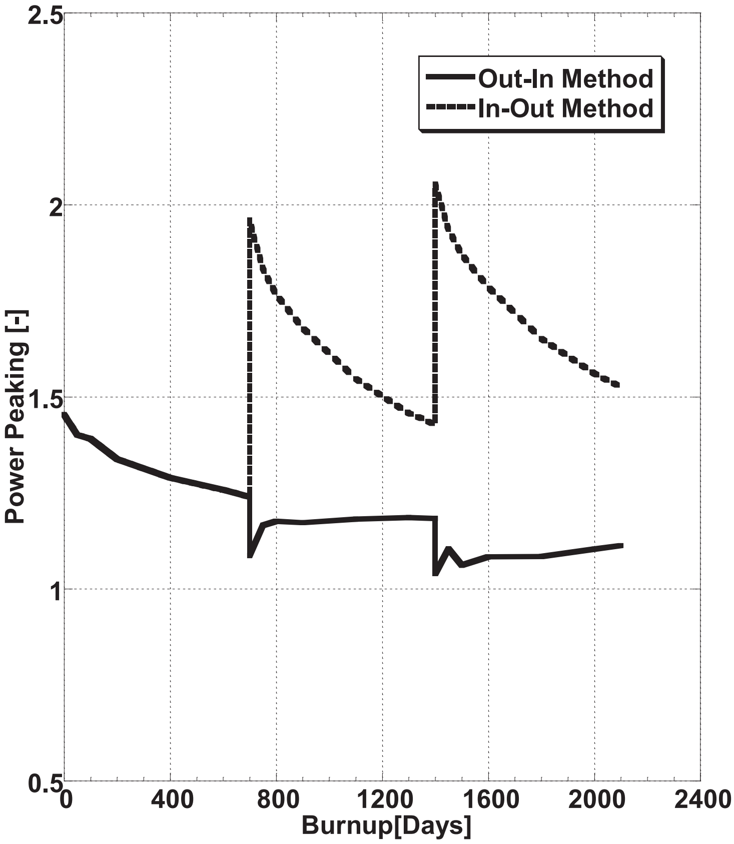
Temperature Distribution





Power Peaking Profile



Power peaking : Ratio of maximum power density to the average power density
- Out-In Method : decreases for the next recycling step
- In-Out Method : increases for nex recycling step

- Out-in method : less than 1.5
- In-Out Method : reaches more than 2

Power peaking profile shows the maximum different of power at a certain location to the average total power distribution.

Thermal Hydraulic Properties

Parameters		Unit
Tinlet	300	°C
Toutlet	332	°C
Average Thermal Conductivity of TH	2.75	W/m K
Average Thermal Conductivity of Zr-	10.7	W/m K
Themal Conductivity of Heavy water	0.483	W/m K
Specific heat capacity of Heavy water	4228	J/Kg K
Fuel Area	1.66E-04	m2
Volume of Fuel	6.12E-04	m3
Coolant Area	1.08E-04	m2
Max Power density of Core	1.10E+08	W/m3
Fluid Density	720	Kg/m3
P/D	1.28	-
Hydraulic Diameter	0.0118	m
Heat Flux clad surface	4.07E+05	W/m2

Thermal hydraulic parameter		Unit
Mean flow velocity	4.14	m/s
Renoylds number	2.80E+04	
Prandtl number	10.94	-
Heat Trannsfer	2.85E+04	W/m2 K
Fanin Friction	6.11E-03	
Friction Pressure drop	0.47	bar
Fuel Temperature Drop	526	C
Total Temperature drop	679	C
Maximum Coolant Temperature	332	C
Maximum Fuel Temperature	1058	C

Out-in Method

Thermal Hydraulic Properties

Parameters		Unit
Tinlet	300	°C
Toutlet	332	°C
Average Thermal Conductivity of TH	2.75	W/m K
Average Thermal Conductivity of Zr-	10.7	W/m K
Themal Conductivity of Heavy water	0.483	W/m K
Specific heat capacity of Heavy water	4228	J/Kg K
Fuel Area	1.66E-04	m2
Volume of Fuel	6.12E-04	m3
Coolant Area	1.08E-04	m2
Max Power density of Core	2.11E+08	W/m3
Fluid Density	720	Kg/m3
P/D	1.28	-
Hydraulic Diameter	0.0118	m
Heat Flux clad surface	7.66E+05	W/m2

Thermal hydraulic parameter		Unit
Mean flow velocity	7.79	m/s
Renoylds number	5.28E+04	
Prandtl number	10.94	-
Heat Trannsfer	5.38E+04	W/m2 K
Fanin Friction	5.22E-03	
Friction Pressure drop	1.43	bar
Fuel Temperature Drop	1008	C
Total Temperature drop	1303	C
Maximum Coolant Temperature	332	C
Maximum Fuel Temperature	1704	C

In-Out Method

Conclusion

1. Core burnup calculations have confirmed that breeding is feasible for water cooled thorium reactor system. It also confirmed that negative void reactivity coefficients are obtained during reactor operation.
2. Fuel breeding capabilities have been shown 1.01 (Out-In method) and 1.02 (In-Out method) at the end of cycle.
3. Thermal hydraulic parameters show the comparable result with conventional reactors and have the large margin to the limitation of thermal hydraulic properties.
4. Reactor core optimization for neutronic and thermal hydraulic aspects should be done for future investigation

END OF PRESENTATION

Thank You

Questions or comments?