Multi-recycling of plutonium and MA in PWR using hydride fuels

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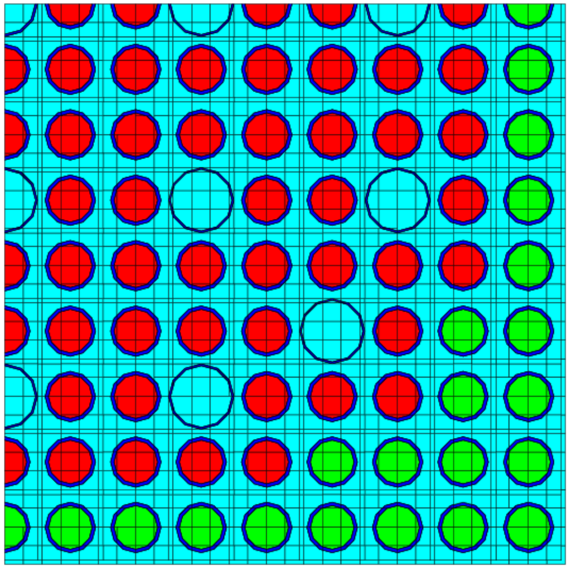
The 5th Joint Reactor Seminar
GoNERI, The University of Tokyo
and
Nuclear Engineering Department, UC Berkeley

Is it possible to multi-recycle Pu in PWR?

MOX allows multi-recyclings only up to about $12^{w}/_{o}$ Pu: after that the large void reactivity coefficient becomes positive. This happens after 2-3 recycles.



Solutions proposed up to now:

CORAIL, MOX-UE, CONFU (all TRU).

All use ^{235}U to reduce plutonium mass, either homogeneously or heterogeneously mixed.

Drawbacks

- -) reach Pu (or TRU) stabilization, not net destruction;
- -) large power peak for heterogeneous configurations;
- -) no substantial U or SWU saving over conventional UO_2.

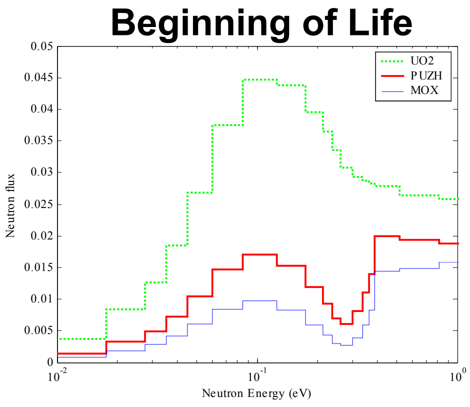
Our solution:

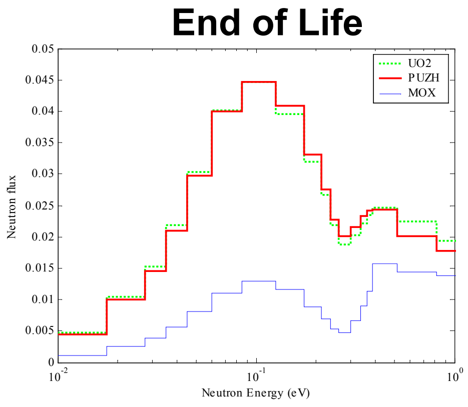
Use hydride fuels, of the type:

$U_x\text{-}Zr_nH_{1.6n}\text{-}Th_mH_{2m}\text{-}Pu_pH_{2p}\ (\text{-}MA_qH_{2q})$

The rationale:

The extra H in the fuel (H_{fuel}~H_{water}) will increase plutonium (and MA) destruction through a softer spectrum and will counteract the effects of large voiding.


Beginning of Life


End of Life

Design variables

- -) Fuel cycle scheme (total or partial recycling of U, Pu, Np etc…)
- -) Fuel composition: mixture of $ZrH_{1.6}$, ThH_2, U in various amounts
- -) Possible use of burnable poisons
- -) Possible use of different U enrichment

Assumptions

- -) Instantaneous reprocessing
- -) 0.1% of U and Pu is lost during reprocessing
- -) 0% losses during fabrication and irradiation
- -) Feed Pu vector is fixed (from [1])
- -) Feed MA vector is fixed (from [2])

[1] Youinou, G. and Vasile, A., 2005. Plutonium Multirecycling in Standard PWRs Loaded with Evolutionary Fuels. Nuclear Science and Engineering: 151, 25-45.

[2] ANL ICONE 10-22575, Table 1 col. b, based on “extended PWR benchmark with 10y cooling”

Constraints

-) Negative core-average burnup-dependent reactivity coefficients (fuel temperature, coolant temperature, small void, large void)

-) Similar cycle time (1430 EFPD) and control system of standard UO_2

-) Same fuel D and P/D of standard UO_2

Unit Cell Geometry and Specific Power

	Hydride Fuels	**Oxide Fuels**
Clad outside diameter	**0.95 cm**	**0.95 cm**
P/D	**1.3261**	**1.3261**
Fuel diameter	**0.8192 cm**	**0.8205 cm**
Clad inside diameter	**0.8357 cm**	**0.8357 cm**
Pitch	**1.26 cm**	**1.26 cm**
Specific power	**76.715 W/giHM**	**36.138 W/giHM**

Methodology

TRITON/NEWT code is used for depletion analysis, extensively tested for degraded plutonium and MA–bearing fuels, both in numerical benchmark and in measured samples (long experience with SAS2H).

BONAMI/NITAWL: XSEC pre-processing
NEWT: transport, 3-groups XSEC collapsing
ORIGEN-S: 0-D depletion and decay

Transport calculations performed with 238 energy group, directly including 40 actinides, 187 fission products.
More nuclei are followed using ORIGEN-S default XS.

Pu amount is adjusted exactly to match desired cycle length
Two steps approach for greater precision:

1. Two runs that over and under estimate cycle length;
2. Final run to estimate discharged and cooled composition.

Methodology – types of fuels analyzed for Pu recycling

PuH_2-$ZrH_{1.6}$-U (uranium varies from 0 g/cm^3 to 3.72 g/cm^3)
PuO_2-ZrO_2-UO_2 (uranium varies from 0 g/cm^3 to 8.28 g/cm^3)

In both fuel types the U amount is fixed*, the plutonium is adjusted to reach the desired cycle length, the remaining space is taken by zirconium in the appropriate chemical form (hydride or oxide). If the zirconium is zero and more space is needed, uranium is taken out in the desired amount.

For Th-based fuels, ThH_2 replaces U. Same methodology.

* The depleted uranium amount is initially fixed at a pre-assigned value, after depletion the entire uranium amount is recycled, adding new depleted uranium if:
-) needed to make up for the consumption
-) needed to reduce enrichment below 20% ^{235}U or 12% ^{233}U

Methodology

hydride fuels for Pu + Np or TRU recycling

PuH_2-NpH_2-(MAH_2)-$ZrH_{1.6}$-U   (uranium varies from 0 g/cm^3 to 3.72 g/cm^3)

Same methodology as before to fill the fuel volume.

PuH_2 has a density of 10.4 g/cm^3, not much is known of hydrides of MA, so it is assumed that the density will be the same* as for PuH_2.

The density is not adjusted to account for varying isotopic composition*.

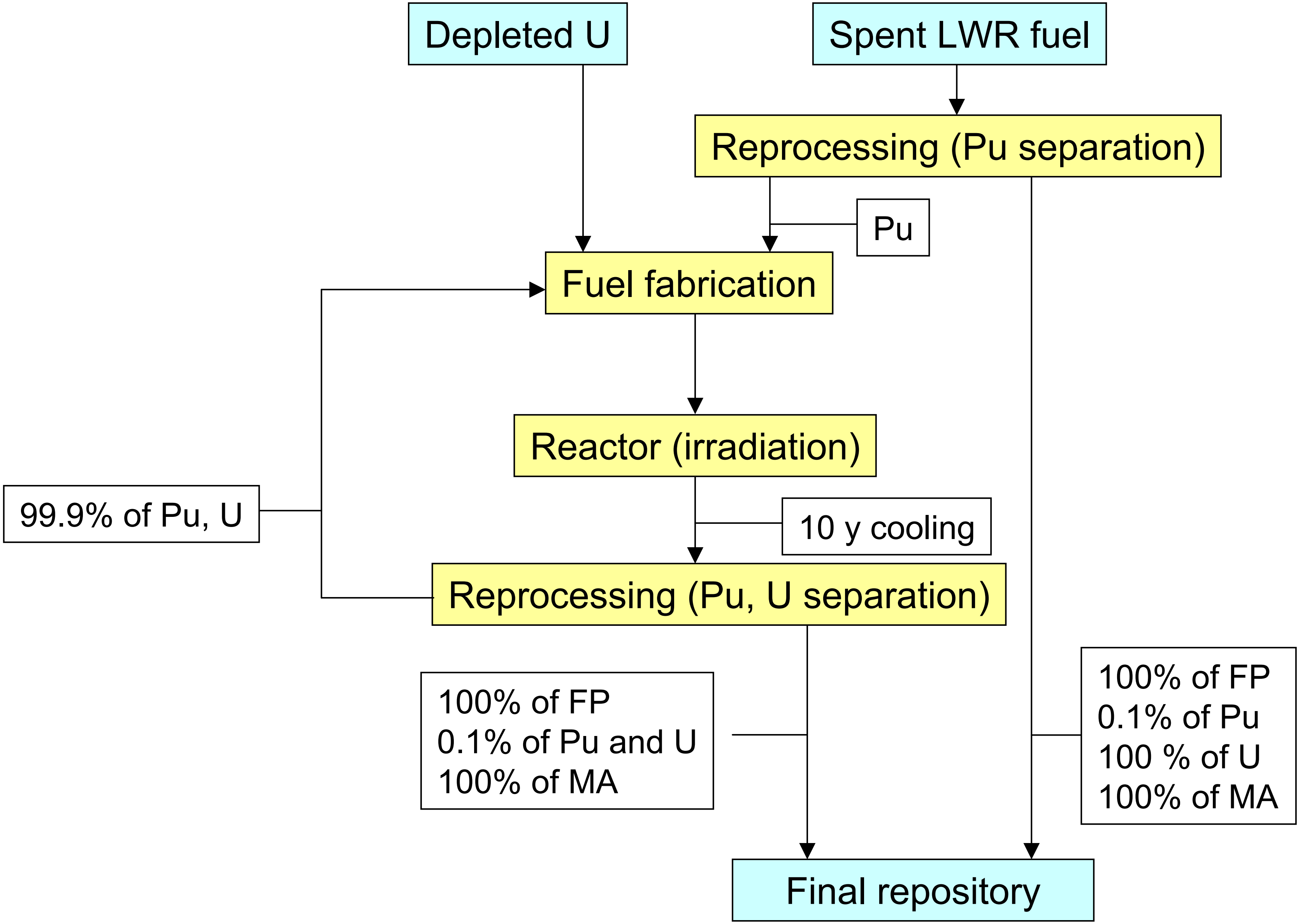
When a given amount of Pu is taken from LWR spent fuel, a proportional amount of Np (or MA) is taken as well, according to the following vector**:

```
            U,         Np,       Pu,       Am,       Cm
MA_frac=[0.003810517, 0.07691684, 1, 0.07117211, 0.006312254];
```

*recommended by prof. Olander

**from ANL ICONE 10-22575, Table 1 col. b, based on “extended PWR benchmark with 10y cooling”

Fuel cycle scheme with plutonium recycle



Depleted U

Spent LWR fuel

Reprocessing (Pu separation)

Pu

Fuel fabrication

Reactor (irradiation)

99.9% of Pu, U

10 y cooling

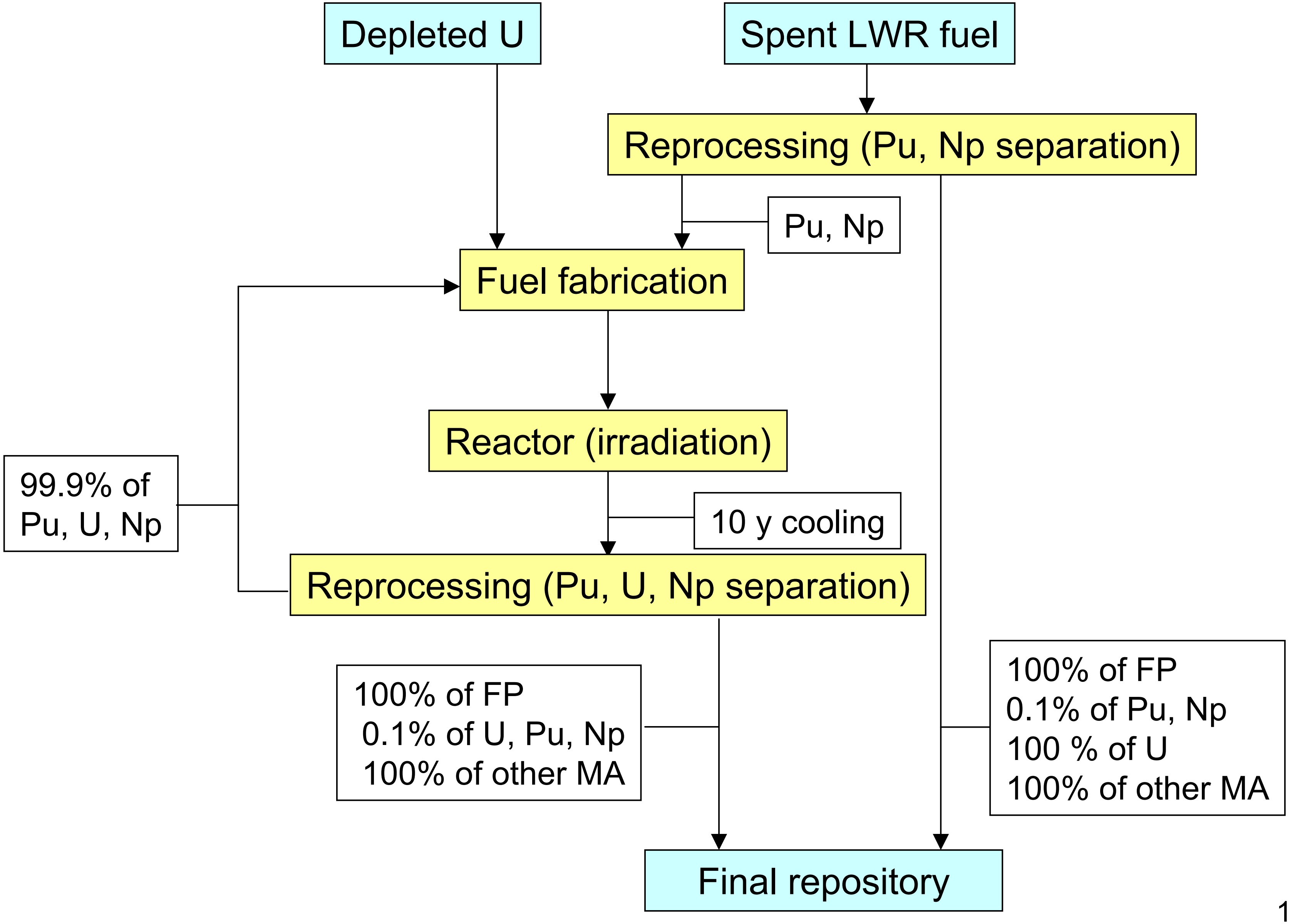
Reprocessing (Pu, U separation)

100% of FP
0.1% of Pu and U
100% of MA

100% of FP
0.1% of Pu
100 % of U
100% of MA

Final repository

Fuel cycle scheme with plutonium and neptunium recycle



Depleted U

Spent LWR fuel

Reprocessing (Pu, Np separation)

Pu, Np

Fuel fabrication

Reactor (irradiation)

10 y cooling

99.9% of
Pu, U, Np

Reprocessing (Pu, U, Np separation)

100% of FP
0.1% of U, Pu, Np
100% of other MA

100% of FP
0.1% of Pu, Np
100 % of U
100% of other MA

Final repository

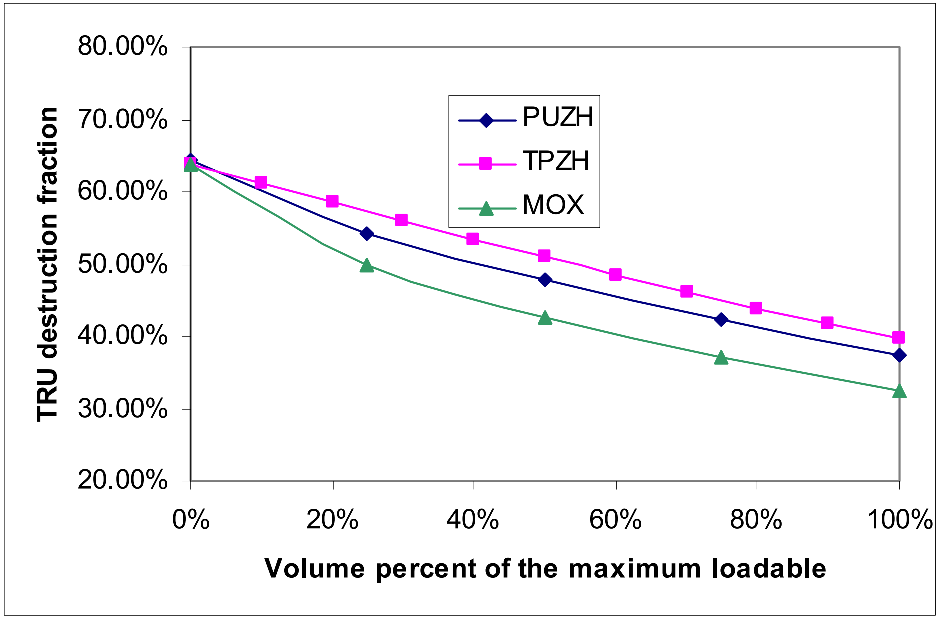
Results: selected characteristics of PUZH and MOX with variable uranium loadings at First Recycle

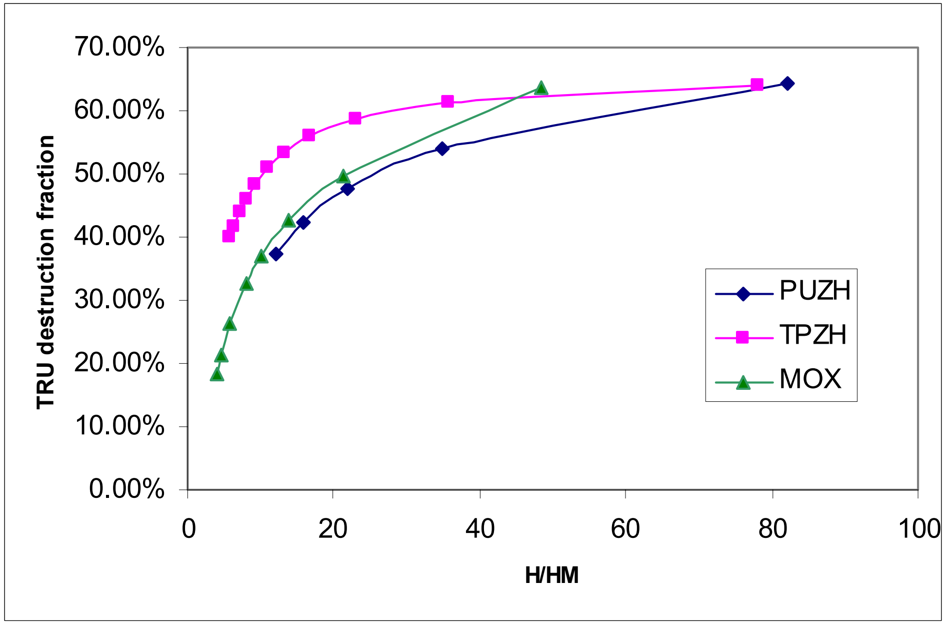
Characteristic	Fraction of maximum uranium in PUZH*					Standard MOX
	0%	25%	50%	75%	100%	--
Rho U (g/cm3)	0	0.9301	1.8601	2.7902	3.7203	**8.4612**
Rho Pu (g/cm3)	0.7337	0.7564	0.7791	0.8018	0.8245	**0.8038**
Rho HM (g/cm3)	0.7338	1.6865	2.6392	3.592	4.5448	**9.265**
Rho fuel (g/cm3)	5.9415	6.6087	7.2759	7.9432	8.6104	**10.465**
H/HM	82.04	34.89	21.85	15.73	12.18	**3.9**
Burnup (GWD/MTiHM)	628.4	284.2	182.1	132.6	102.9	**54.66**
EFPD	1376.0	1430.1	1434.3	1420.9	1395.8	**1512.45**

U % of total in PUZH*	0%	25%	50%	75%	100%	Standard MOX
PUZH TRU destruction fraction	64.36%	54.13%	47.83%	42.39%	37.48%	**--**
MOX TRU destruction fraction	63.65%	49.76%	42.65%	37.11%	32.56%	**23.7%**
PUZH Fissile fraction at EOL	21.63%	32.87%	38.52%	42.74%	46.13%	**--**
MOX Fissile fraction at EOL	24.50%	38.05%	43.39%	46.99%	49.65%	**60.3%**

* maximum uranium in PUZH: 3.7203 g/cm^3

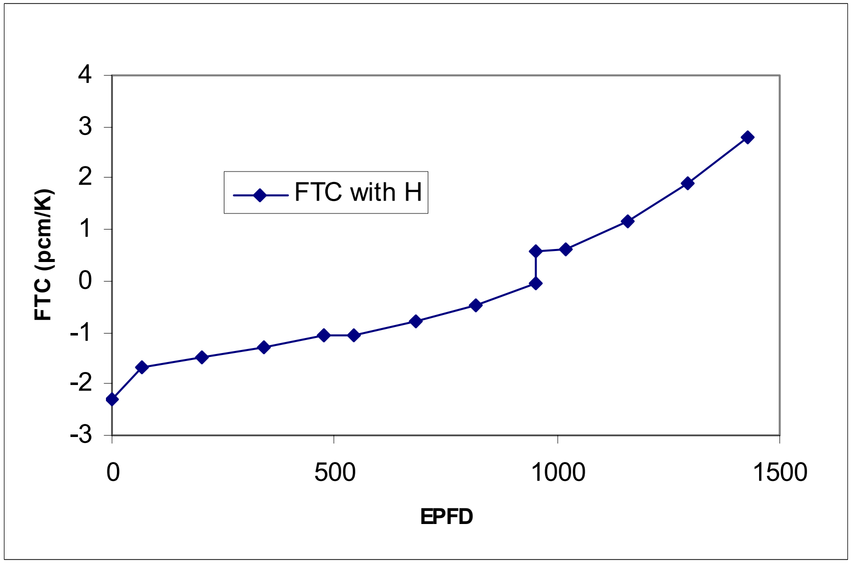
Results:
TRU destruction fraction

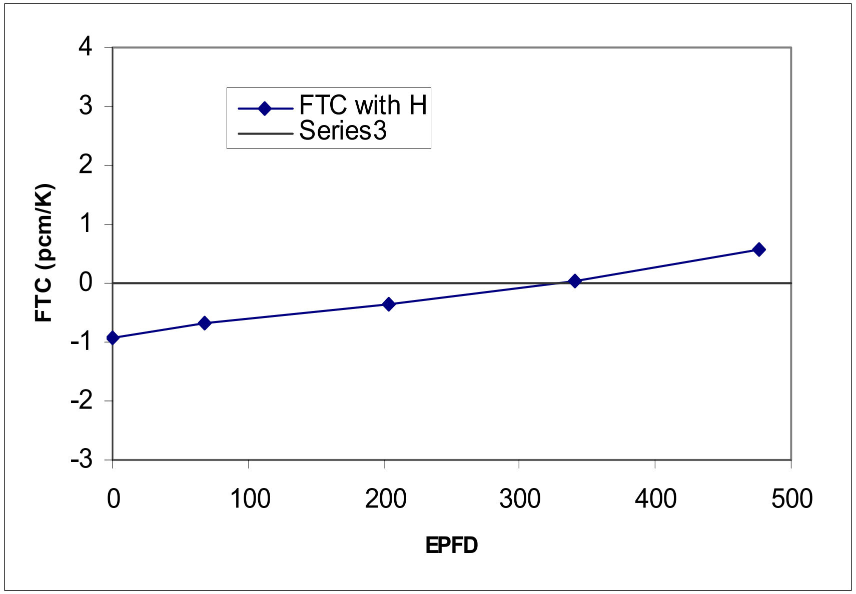




- TPZH = ThH_2-PuH_2-$ZrH_{1.6}$;
- PUZH = PuH_2-U−$ZrH_{1.6}$;
- Maximum loading is Th for TPZH and U for PUZH and MOX,
- 100% is for PUZH, not MOX (which can load higher U amounts)

Results:
Fuel temperature coefficient of reactivity (FTC)





Results:
Fuel temperature coefficient of reactivity (FTC)

FTC of PUZH

Burnup-dependent FTC<0 for U>25%.

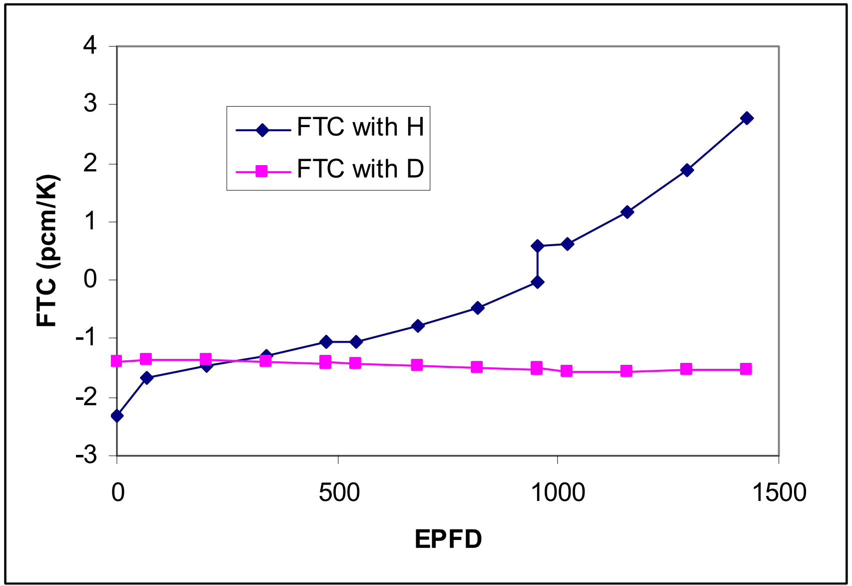
But

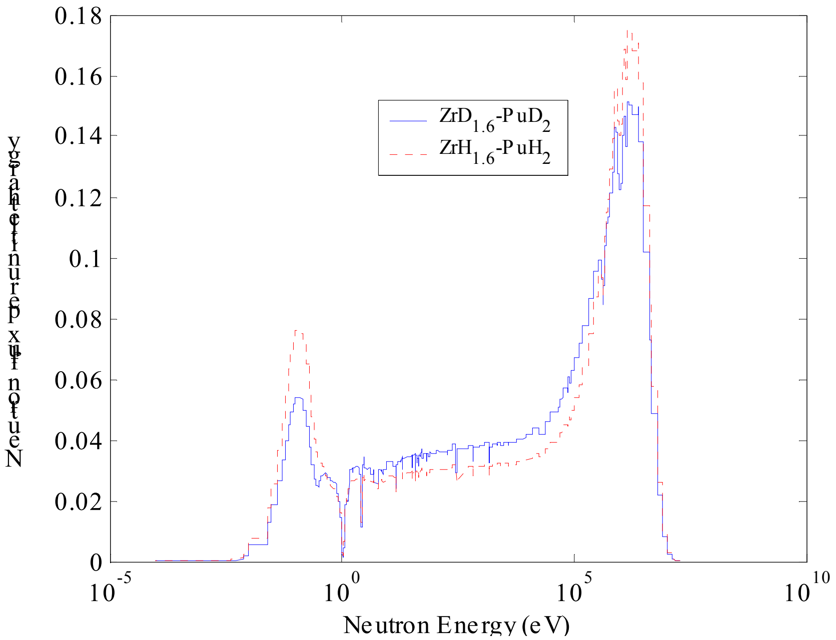
FTC>0 for U=0 during the 3rd batch.

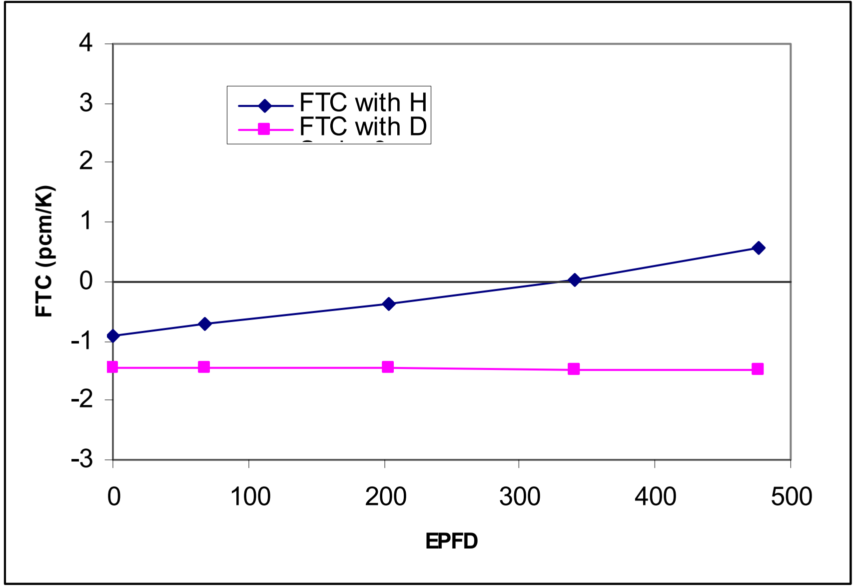
Solutions

-) burn only up to 2nd batch

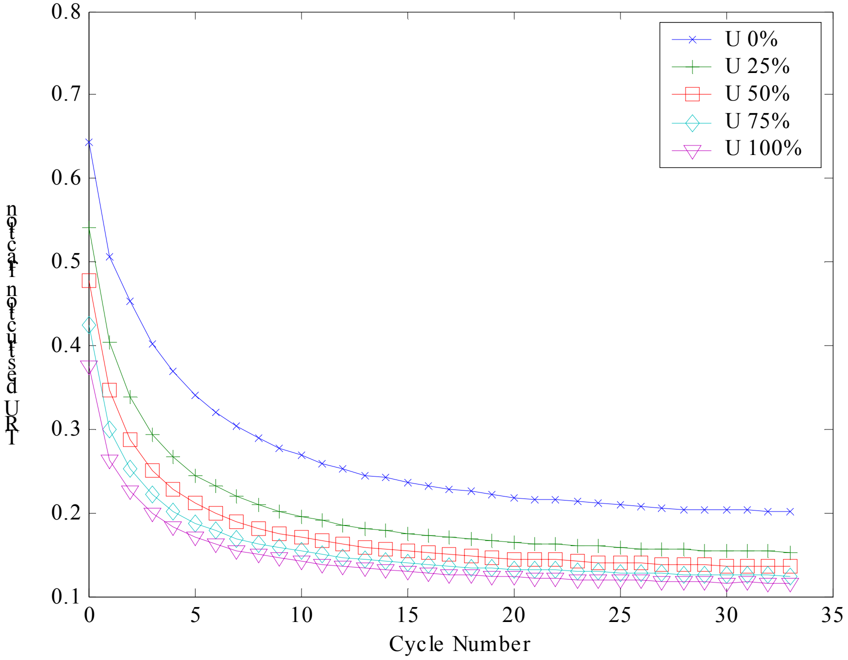
-) <u>Use D instead of H</u>

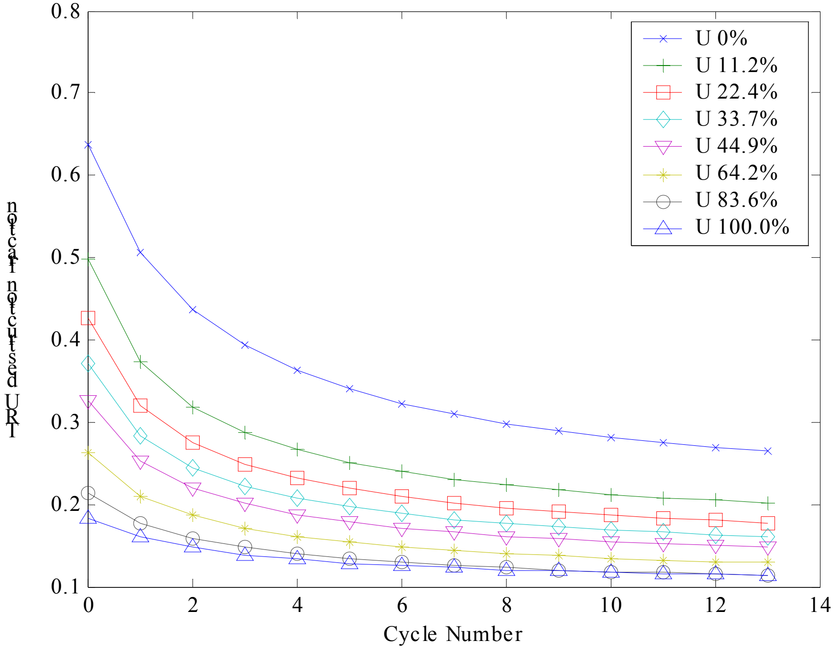






Selected results:
TRU destruction fraction as a function of recycle number
PUZH (left) MOX (right)

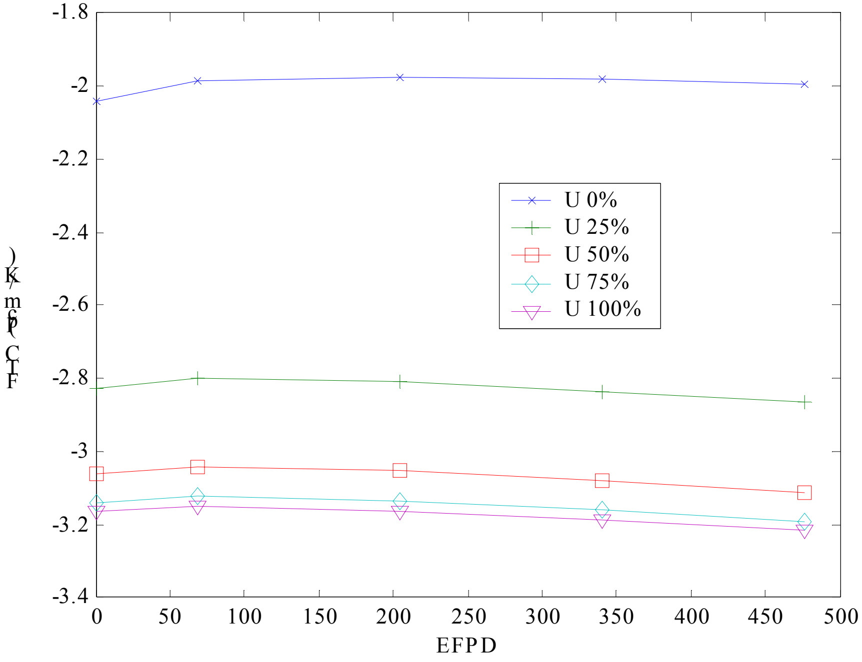


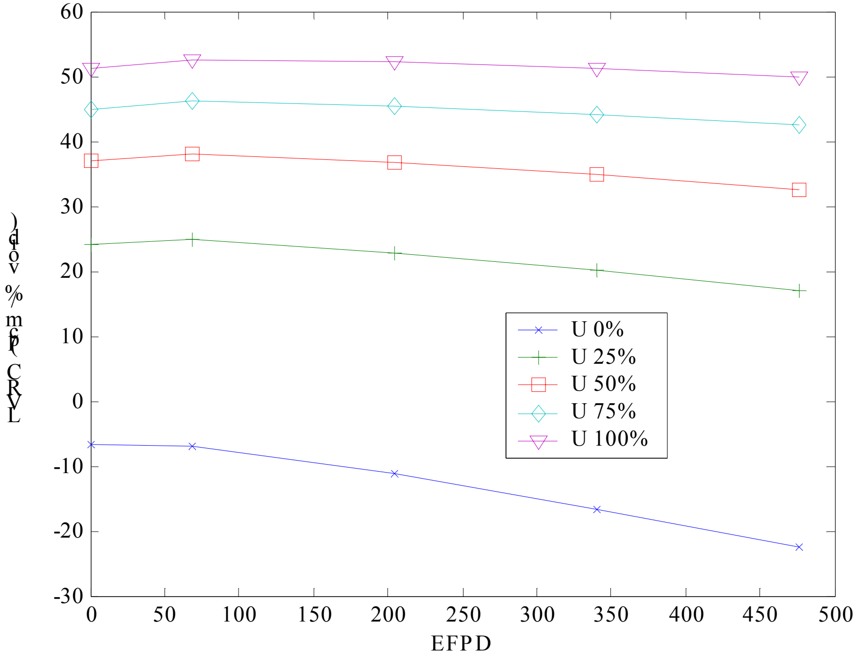


Results:
PUZH, 33rd recycle: core-average FTC, and LVRC (90%) without soluble boron

Major finding:

The LVRC of PuH_2-$ZrH_{1.6}$ remains negative throughout the recyclings!



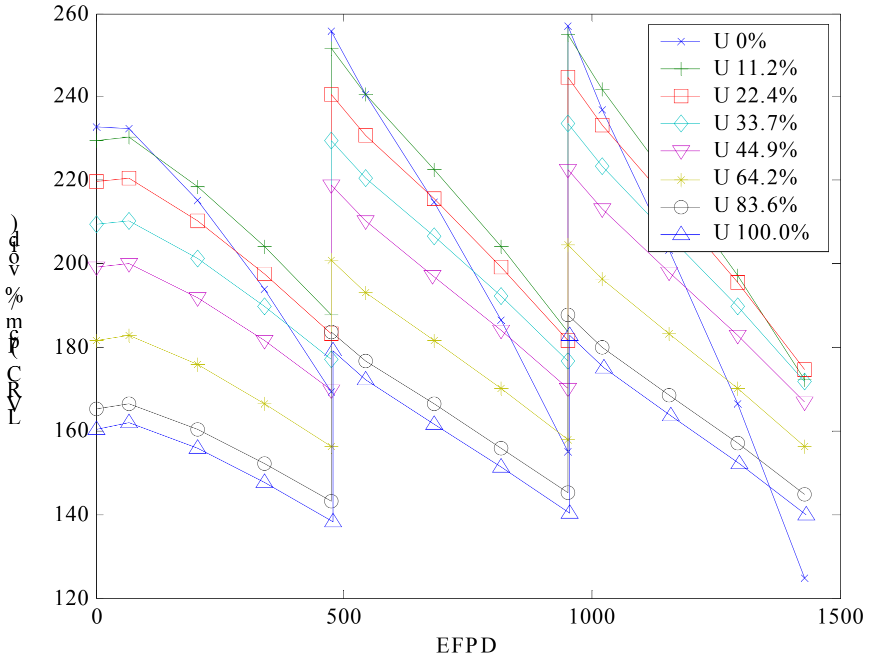


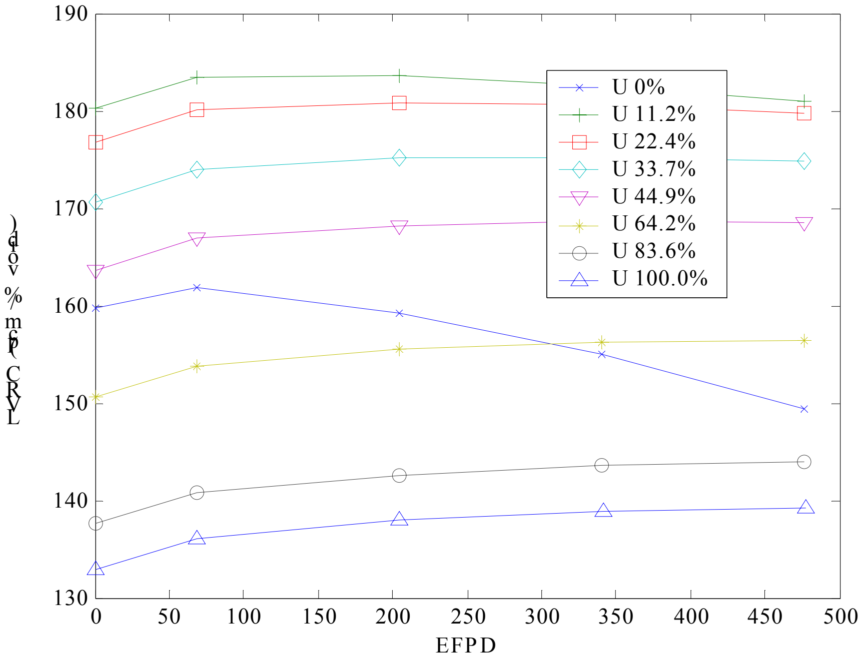
NOTE: with 100% void LVRC becomes positive, but core leakage compensates

Results:
MOX, 13rd recycle: LVRC (90%)
with soluble boron (left) and core-average without (right)

while

The LVRC of PuO_2-ZrO_2-(U) becomes positive at the 13th recycling

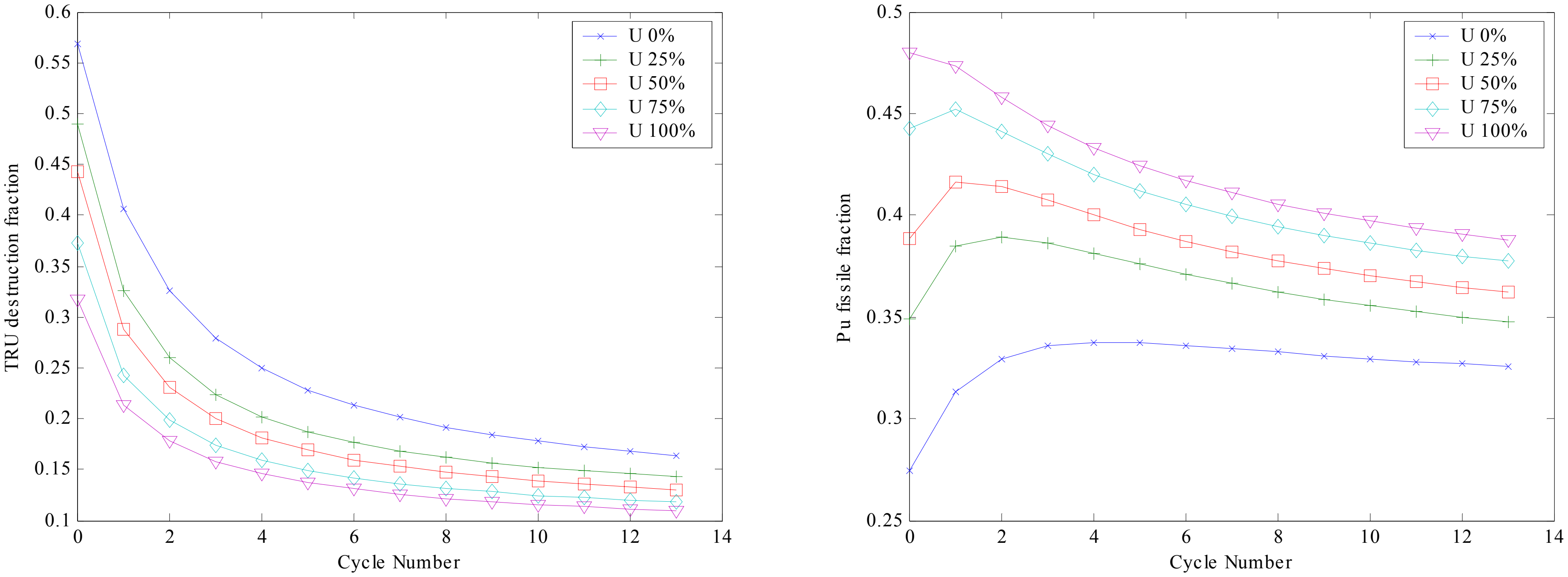




Conclusion:

Possible To Indefinite Recycle Pu In PWR Using Hydride But Not Oxides

Results: PuH_2-NpH_2-U-$ZrH_{1.6}$ TRU destruction fraction (left) and fissile fraction at discharge (right) as a function of recycle number



Conclusions

-) **It is found possible to infinitely recycle Pu in PWR using as feed only depleted uranium and Pu coming from LWR spent fuel, provided that most of the excess reactivity will be compensated by means other than soluble boron.**

-) MOX, on the other hand, only allows recycling up to 9-10 times.

-) The TRU destruction fraction, >60% for first recycle, stabilizes at ~20% at equilibrium.

-) It appears possible to multi-recycle up to 8-12 times Pu and Np in PUZH fuel (corresponding to 110-170 years).

-) All-TRU recycle in U-based hydrides appears feasible only up to few recycles (2-3).

Thank you

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Results: TRU-H_2-U-$ZrH_{1.6}$ TRU destruction fraction (left) and fissile fraction at discharge (right) as a function of recycle number