THE ENVIRONMENTAL IMPACTS OF KOREAN ADVANCED NUCLEAR FUEL CYCLE KIEP-21 AND DISPOSAL CONCEPTS

Jihae Yoon and Joonhong Ahn

Department of Nuclear Engineering

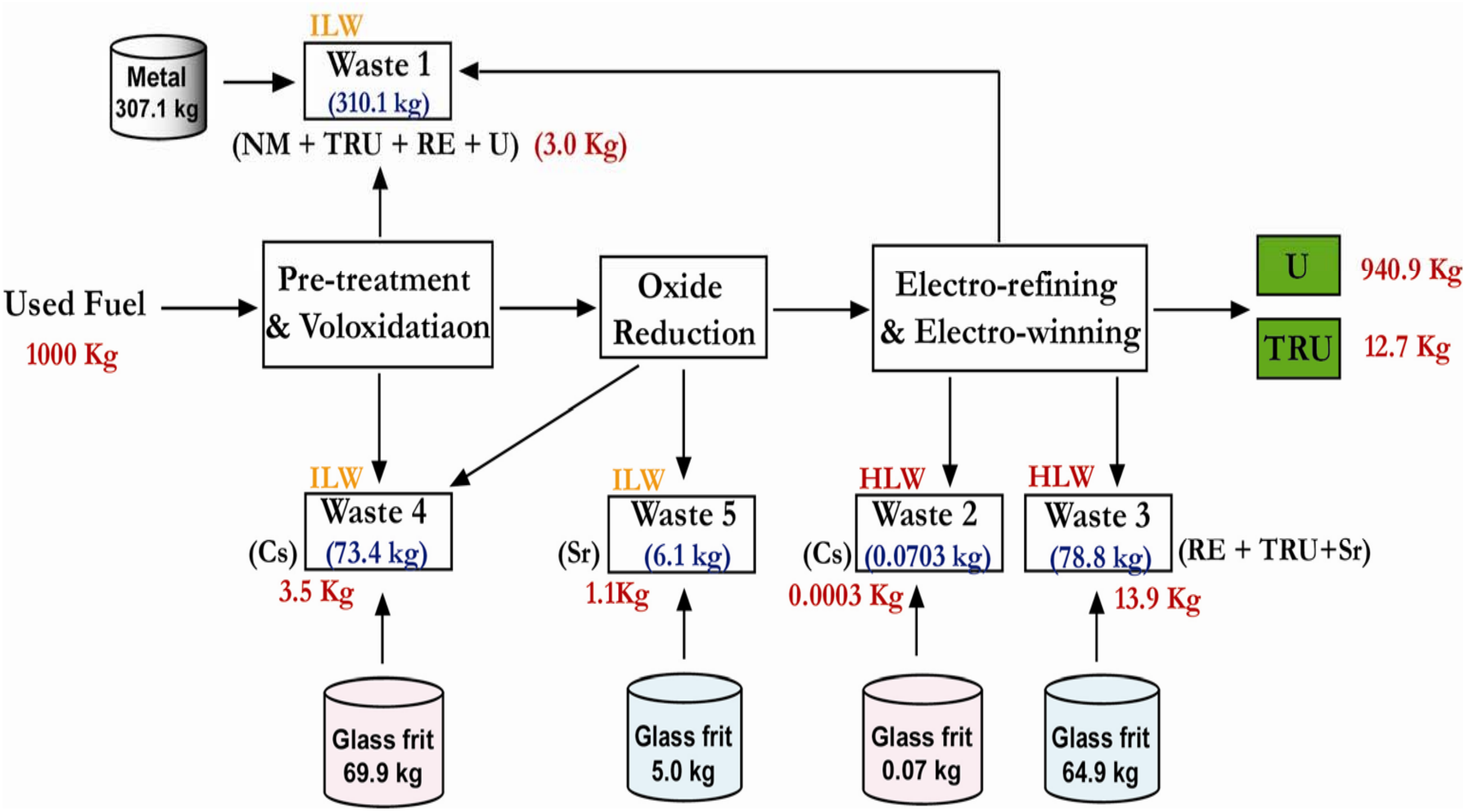
University of California, Berkeley

November 3, 2010

Contents

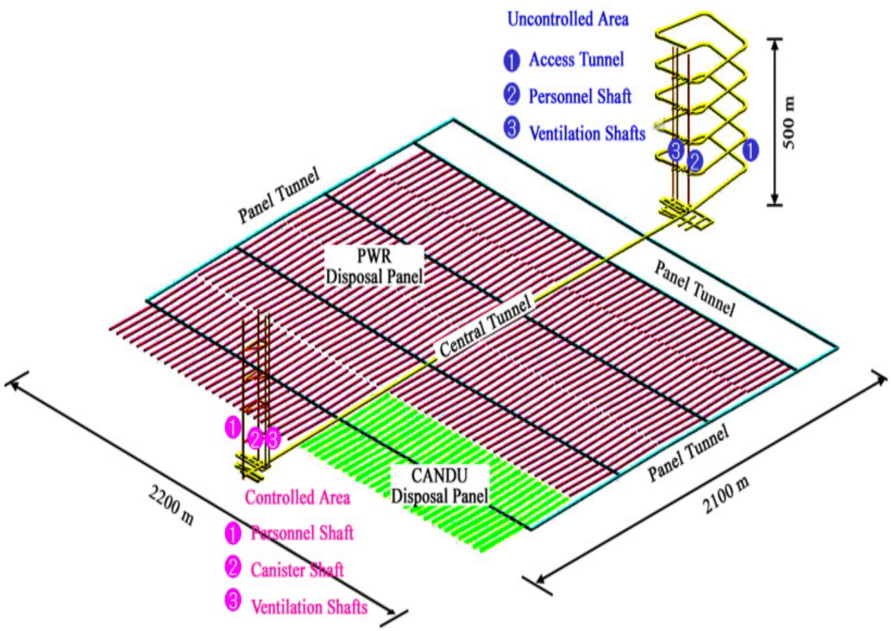
KIEP-21
: Waste streams and Repository Concept

Flowsheet for treatment of 1 MTHM of used PWR fuel with 4.5wt% U-235, 45000 MWD/MTU, and five provisional waste streams.



Repository concepts

Korean Reference Spent Fuel Disposal System (KRS)

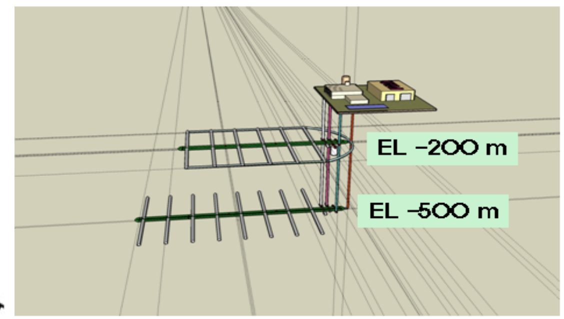


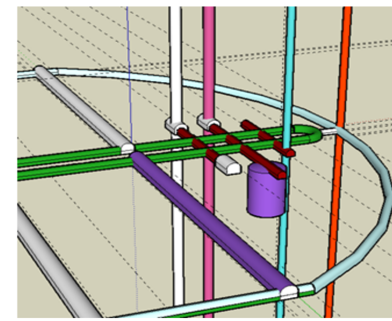
KAERI, High-Level Waste Long-term Management Technology Development (2010)

Advanced Korean Reference Disposal System (A-KRS)

Before pyro-process, PWR spent fuels will be stored at the level of 200 m depth

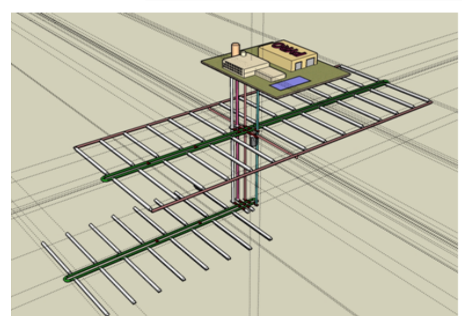
Waste 1 : Tunnel(or Silo) disposal at EL -200 m
Waste 2 3 : Storage for 100 yrs at EL - 500 and then final disposal (Closure)
Waste 4 5 : Storage for 300 yrs at EL -200 m and then final disposal (Closure)

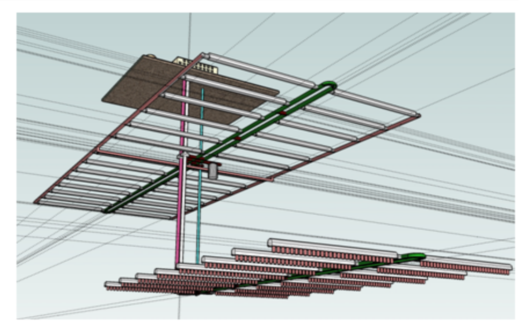




Before pyro-process, PWR spent fuels will be stored at EL -200 m depth

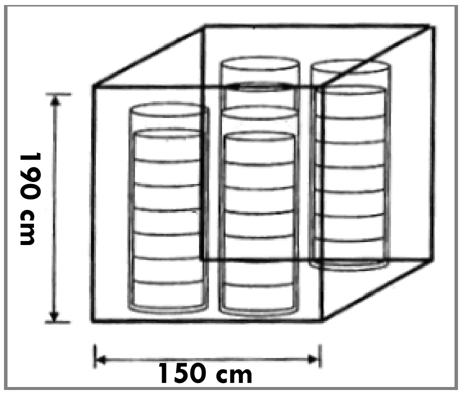
- Waste 2 3 will be stored at EL-200 level for 100 yrs, and then Transported to El -500m and disposed finally (vertical emplacement : KBS-3 type)
- Waste 4 5 will be stored at EL-200 level for 300 yrs, after then disposed finally
- Waste 1 : Tunnel(or Silo) disposal at EL -200 m





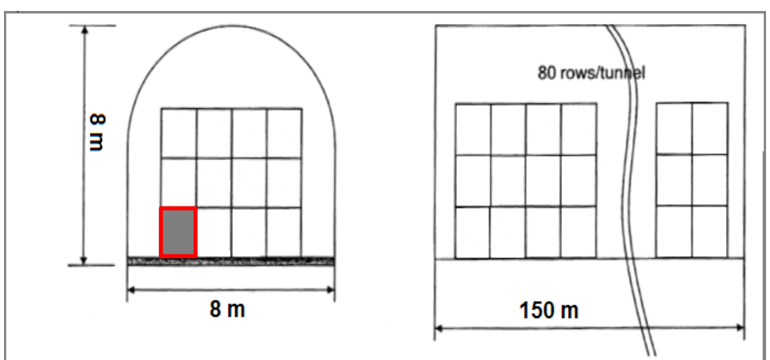
Waste Package/Disposal

(1) Metal waste-ILW

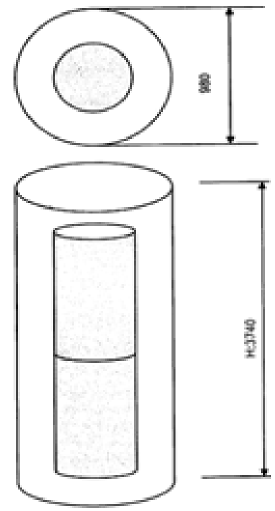


Compact-Metal Disposal Package (c-MDP)

0.92 ingot/1 MTU

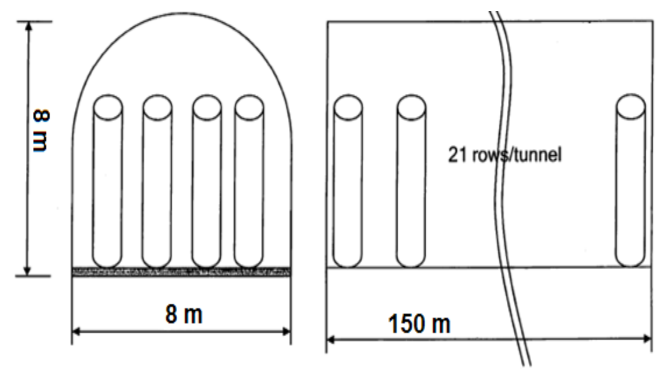


(3) Vitrified waste-HLW



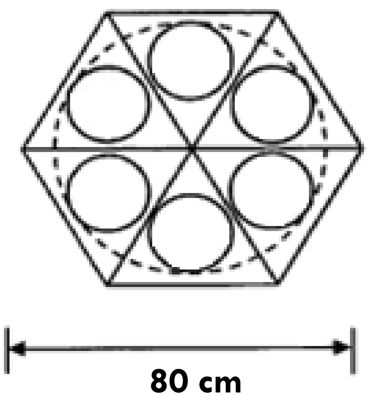
Storage & Disposal Container (SNDC-2)

0.1 block/1 MTU



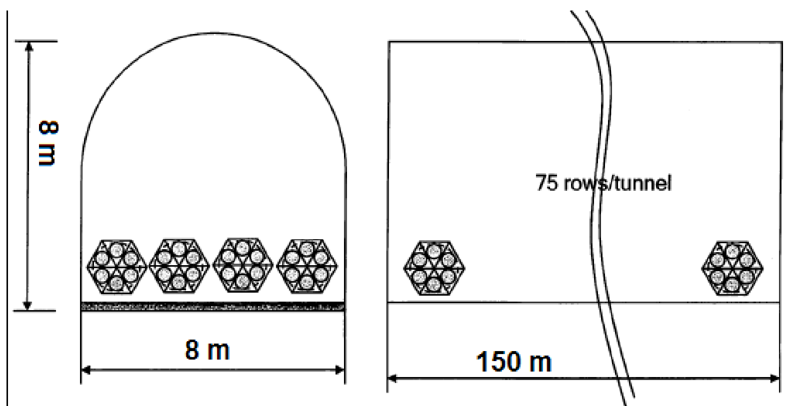
Waste Package/Disposal

(4) Ceramic waste-ILW

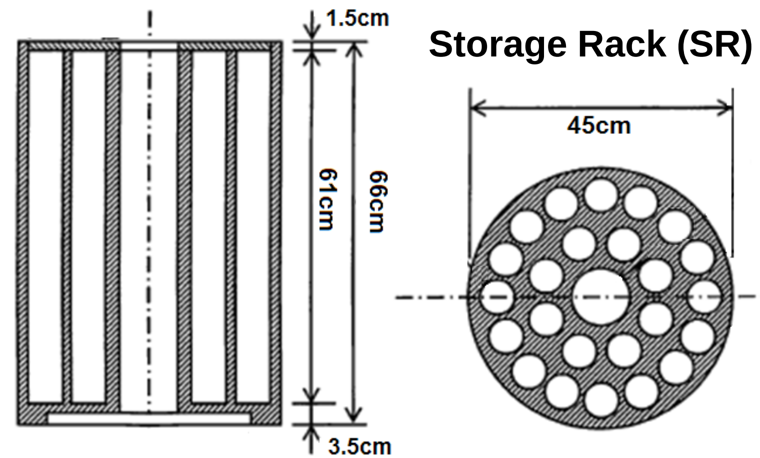


Storage Rack (SR)

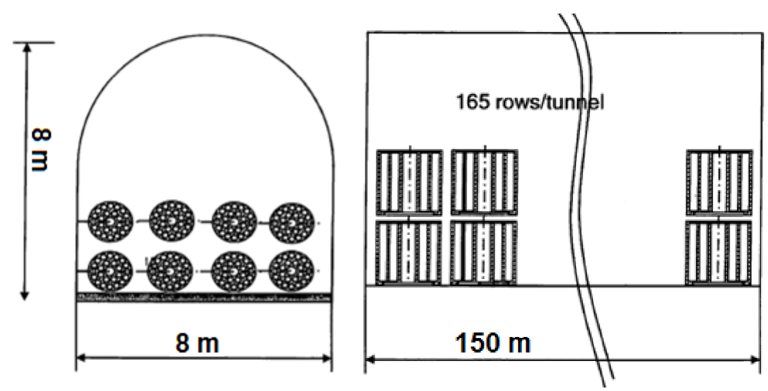
0.6 block/1 MTU



(5) Vitrified waste-ILW



2.4 blocks/1 MTU



Volume/Heat Emission

Volume

Pyroprocessing	
waste stream	Volume (unit: m^3/1 MT)
Waste 1	0.14
Waste 3	0.14
Waste 4	0.08
Waste 5	0.01
Total	0.37
Direct Disposal	3.06

→ 1/8 volume reduction

Heat emission

Concept	Heat emission (unit: watts/ton)
Pyroprocessing	2.00E+00
Direct Disposal	1.34E+02

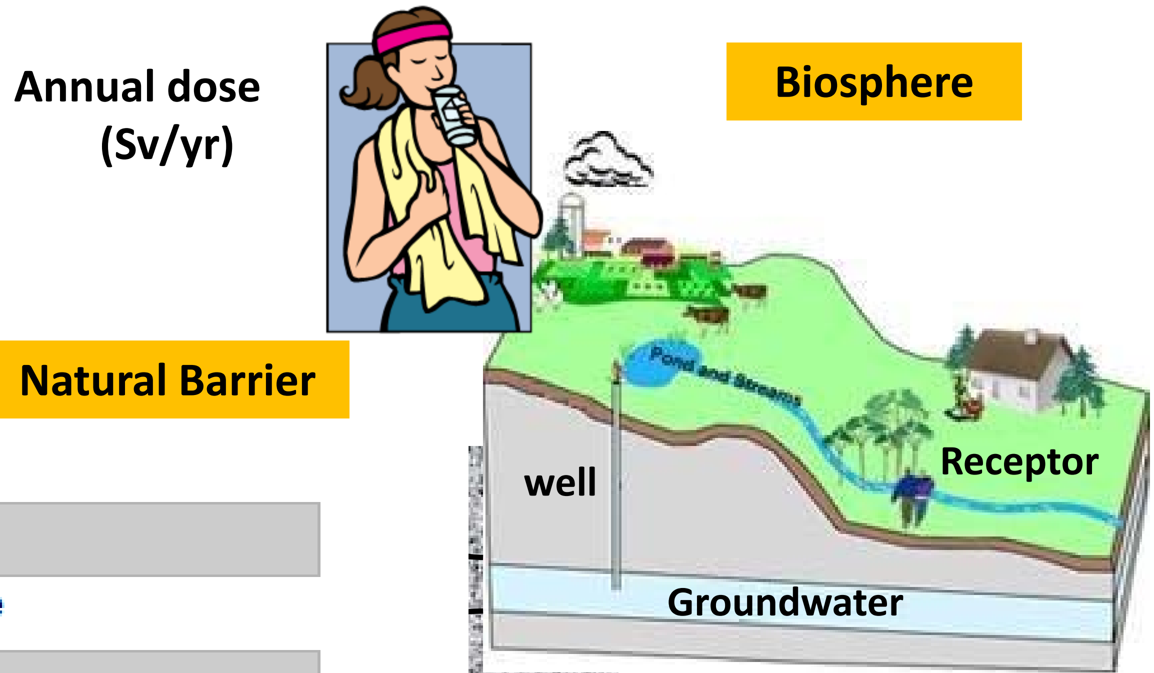
(after 300 yrs)

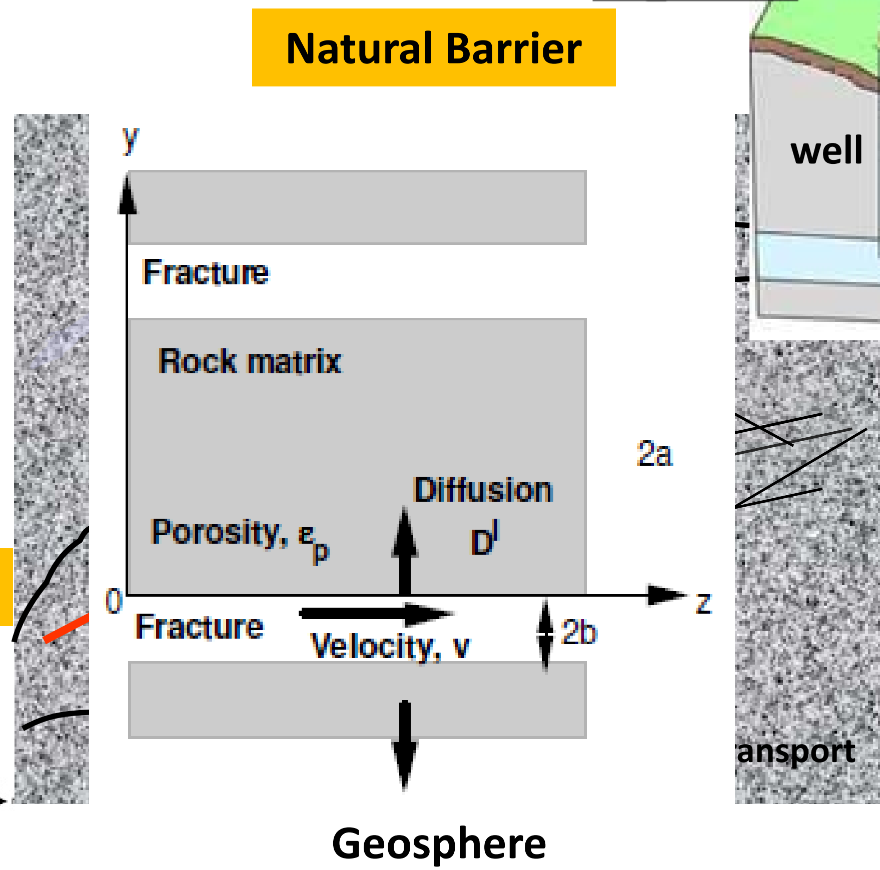
Radionuclides Transport Model

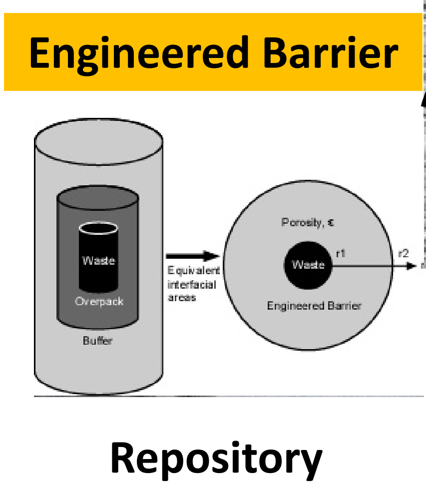
<Assumptions>

(1) no radionuclides are lost due to transport effect

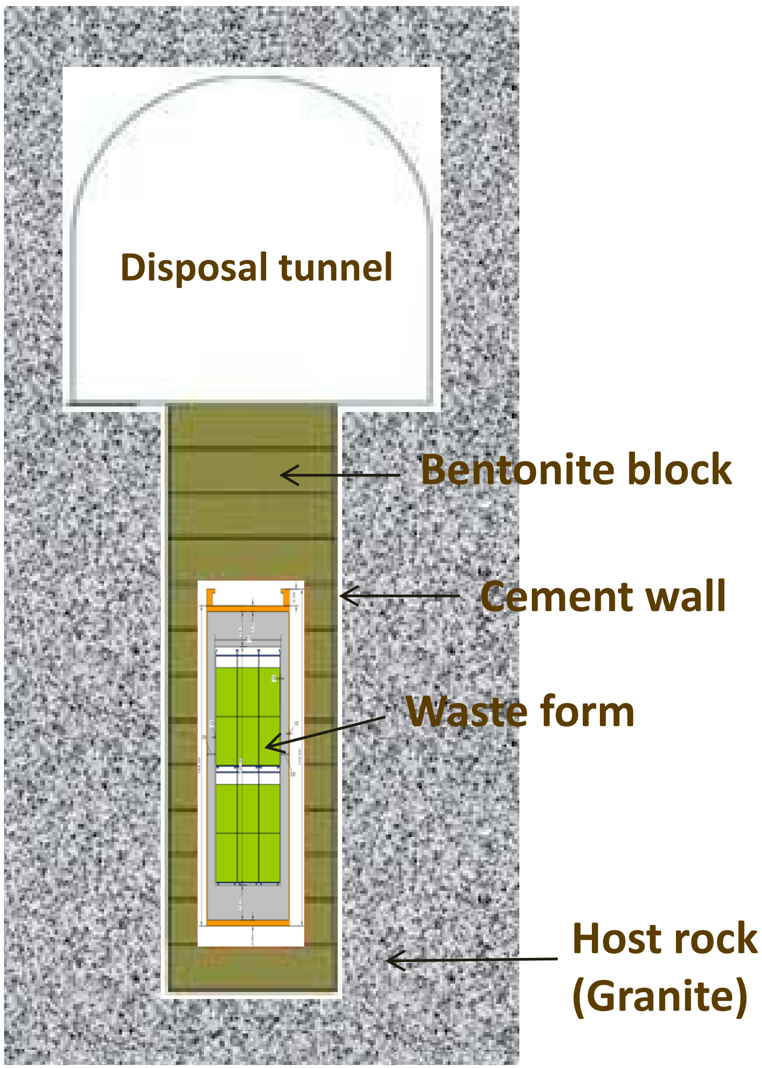
(2) all radionuclides reaching the biosphere would be ingested.







Geochemical modeling scheme



1. Granitic groundwater

2. Cement water
: Granitic groundwater with
Cement reaction

3. Bentonite porewater
: Cement water with MX-80 bentonite

4. Monazite porewater
: bentonite porewater with Monazite

* Neglect canister and overpack

* PHREEQC (V.2.17) code

Environmental Performance Analysis

- **Comparative and Parametric studies**

  - Effects of Back-end Fuel Cycle

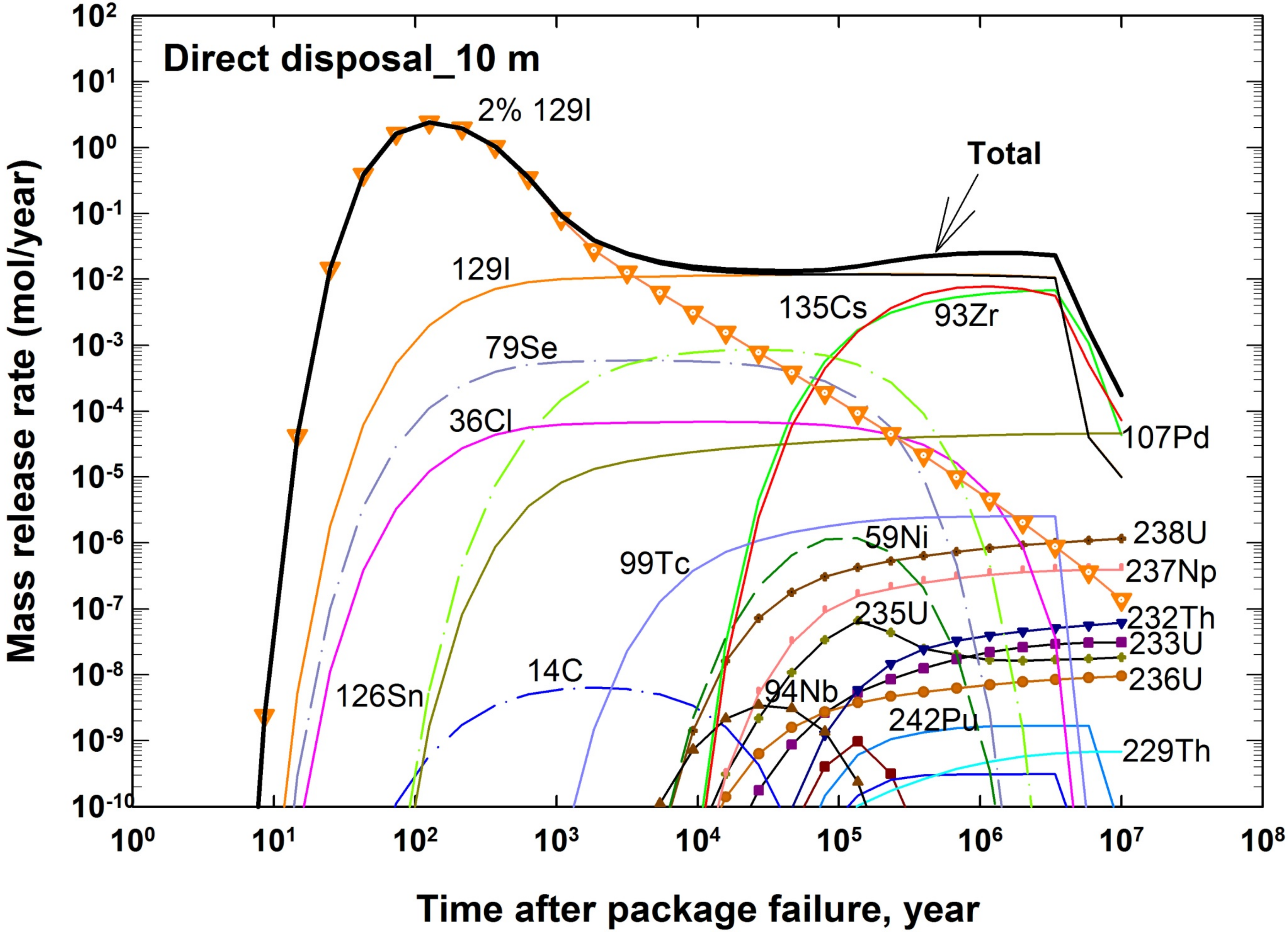
    : Direct disposal (KRS) and Pyroprocessing (A-KRS)

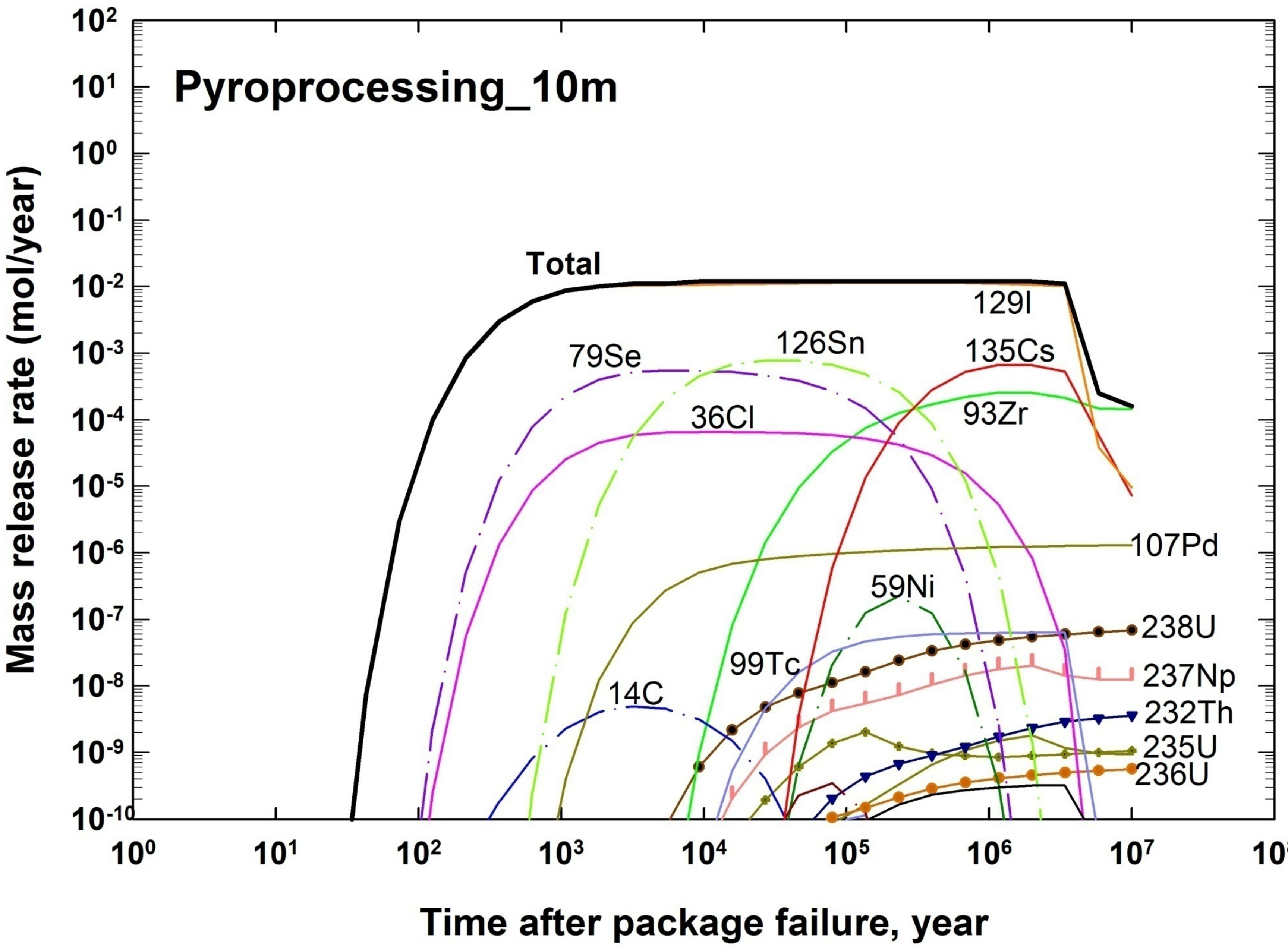
  - Waste Form Durability Effect

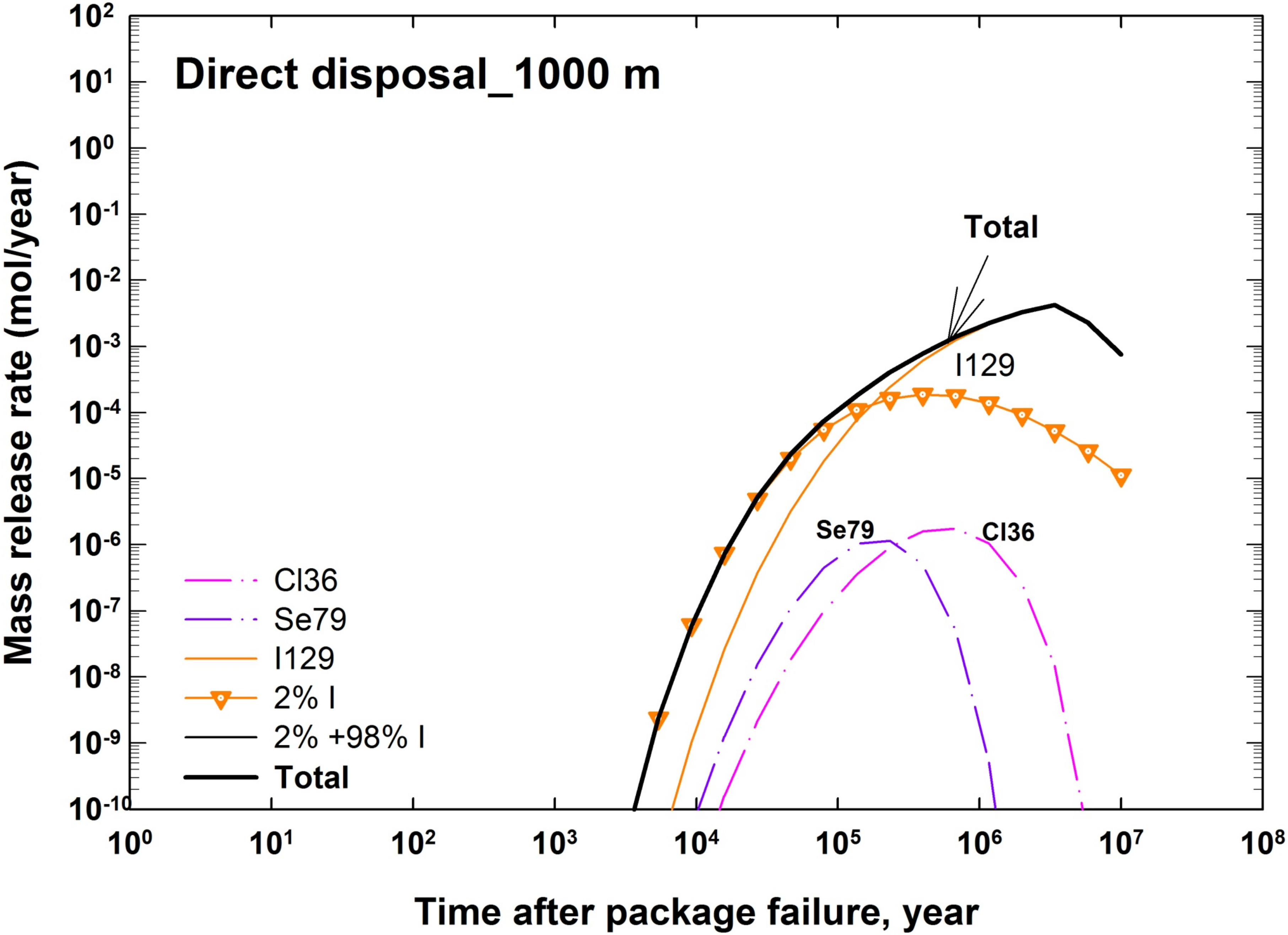
  - Solubility Effect

A. Effects of Back-end Fuel Cycle

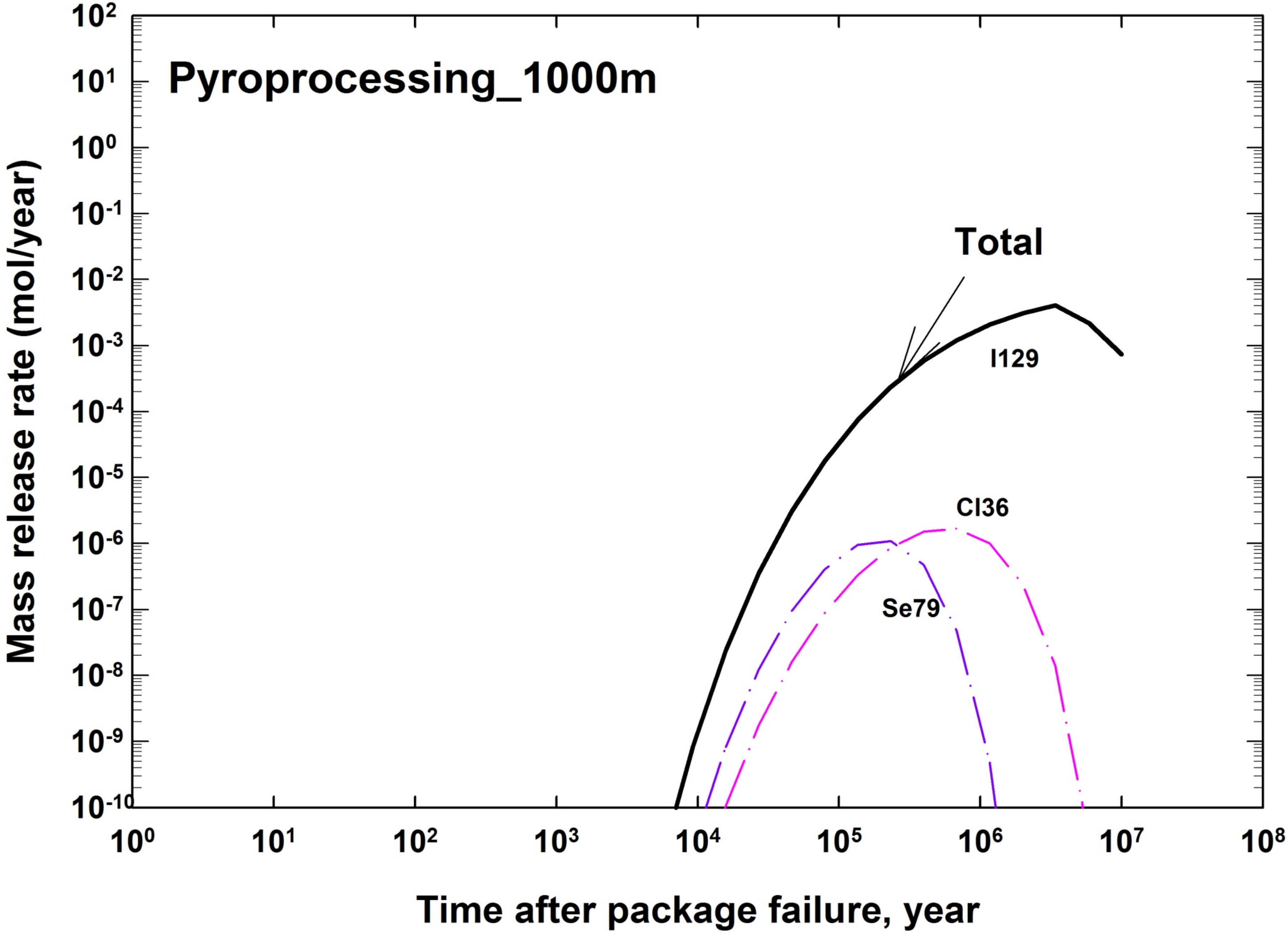
: Direct disposal (KRS) and Pyroprocessing (A-KRS)


Direct disposal_10 m
Mass release rate (mol/year)
Time after package failure, year
2% 129I
Total
129I
135Cs
93Zr
79Se
36Cl
107Pd
99Tc
59Ni
238U
237Np
235U
232Th
233U
236U
14C
126Sn
94Nb
242Pu
229Th
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}
10^{8}

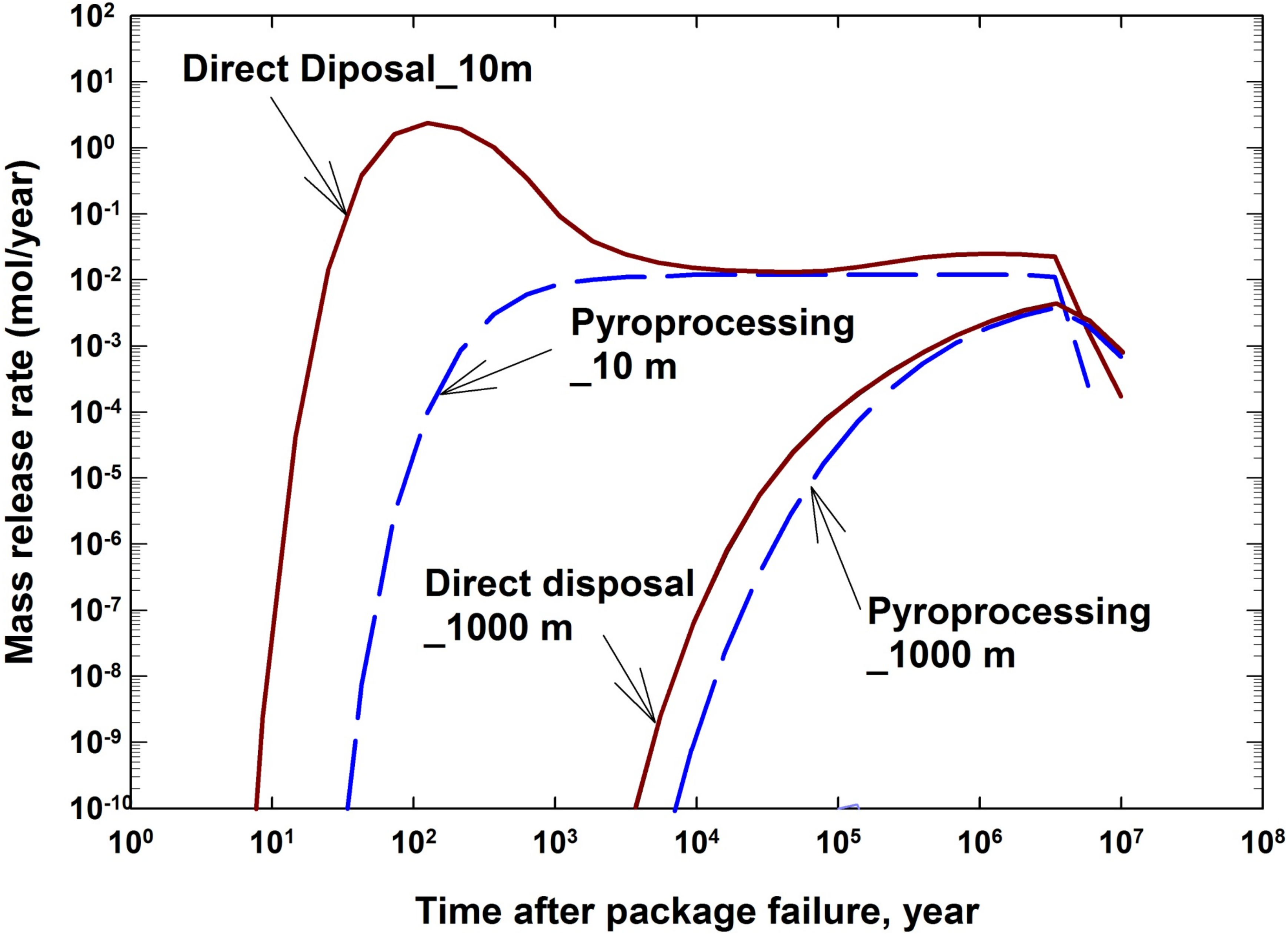

Pyroprocessing_10m
Mass release rate (mol/year)
Time after package failure, year
Total
129I
79Se
126Sn
135Cs
36Cl
93Zr
107Pd
59Ni
238U
99Tc
237Np
14C
232Th
235U
236U
10^2
10^1
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
10^-8
10^-9
10^-10
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8



Direct disposal_1000 m
Mass release rate (mol/year)
Time after package failure, year
Total
I129
Se79
Cl36
Cl36
Se79
I129
2% I
2% +98% I
Total



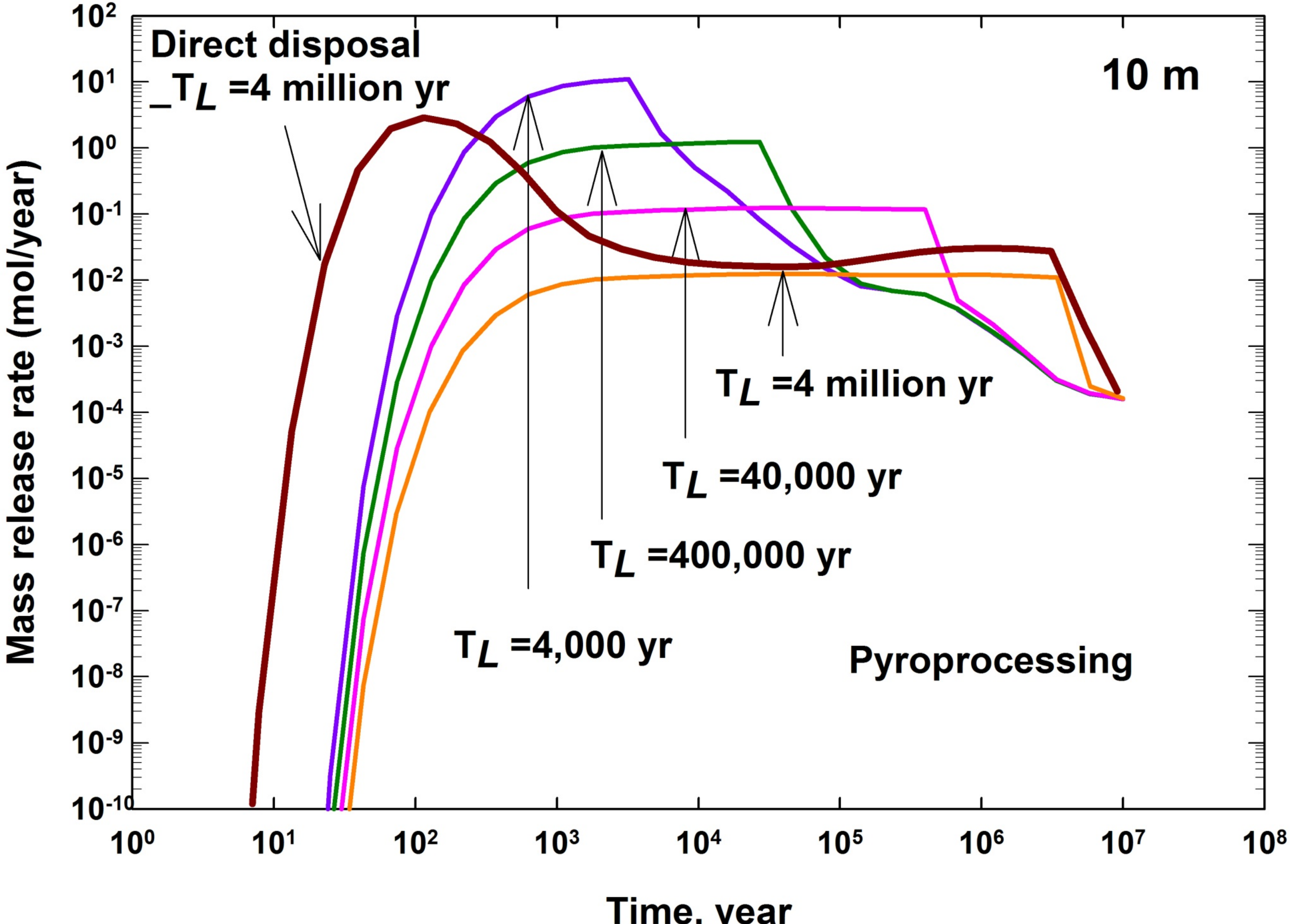
Pyroprocessing_1000m
Total
I129
Cl36
Se79
Mass release rate (mol/year)
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
10^{7}
10^{8}
Time after package failure, year



Direct Diposal_10m
Pyroprocessing
_10 m
Direct disposal
_1000 m
Pyroprocessing
_1000 m
Mass release rate (mol/year)
Time after package failure, year
10^2
10^1
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
10^-8
10^-9
10^-10
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8

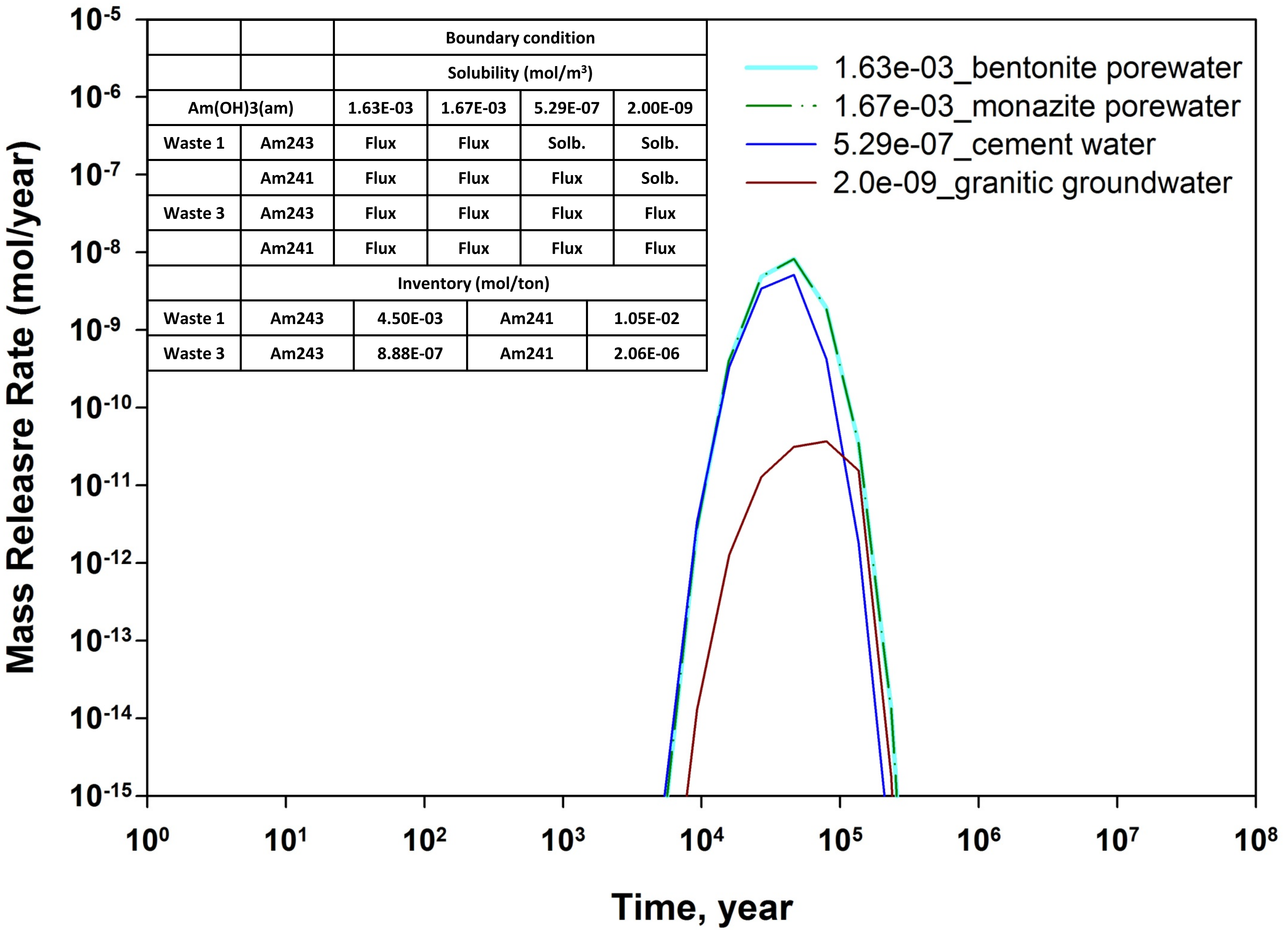
B. Waste Form Durability Effect

- From T_L = 4,000 year to T_L = 4 million year at 10 m location
- Direct disposal case (T_L = 4 million year)



Direct disposal
_T_L =4 million yr
10 m
Mass release rate (mol/year)
Time, year
T_L =4 million yr
T_L =40,000 yr
T_L =400,000 yr
T_L =4,000 yr
Pyroprocessing
10^2
10^1
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
10^-8
10^-9
10^-10
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7
10^8

C. Solubility Effect



		Boundary condition			
		Solubility (mol/m³)			
Am(OH)3(am)		1.63E-03	1.67E-03	5.29E-07	2.00E-09
Waste 1	Am243	Flux	Flux	Solb.	Solb.
	Am241	Flux	Flux	Flux	Solb.
Waste 3	Am243	Flux	Flux	Flux	Flux
	Am241	Flux	Flux	Flux	Flux
	Inventory (mol/ton)				
Waste 1	Am243	4.50E-03	Am241	1.05E-02	
Waste 3	Am243	8.88E-07	Am241	2.06E-06	

Discussions (1)

- Because most TRU isotopes are recovered by pyroprocessing for future use in the fast reactors, the heat emission from the waste is reduced by a factor of 67 at 300 years, compared with emission from PWR spent fuels.
  - Major heat emitters in the first 300 years, Cs and Sr, are included in the intermediate level wastes, and is assumed to be managed by active ventilation.

Discussions (2)

- In the near field, the peak radionuclide release rate is reduced by application of pyroprocessing.
  - This is not because of separation of TRUs from the PWR used fuels, but because of better performance of waste forms than the spent fuel, which contains Iodine-129 in the gap between fuel pellets and cladding.
  - Contributions of TRU elements are not significant.
- In the far field,
  - Those fission products that dominate the release rate in the near field also are main contributors.
  - Furthermore, difference in TRU inventories and waste form performance do not make significant difference, due to assumed mechanism of radionuclide retention and dispersal in the far field.
  - This needs further study to confirm.

Conclusion

- From the preliminary parametric study, it has been confirmed that waste-form durability and radionuclide solubility can have significant effects. For more meaningful comparison, we need to make a realistic assessment of repository performance, for which we need to achieve:
  - More detailed heat transfer analysis to determine repository configurations
  - More detailed geochemical analysis to determine waste-form dissolution, release of radionuclides from waste forms, and transport of radionuclides in the engineered barrier

Acknowledgement

- Grant from Korea Atomic Energy Research Institute (KAERI).
- Dr. Eung Ho Kim (KAERI)
- Dr. Yongsoo Hwang (KAERI)
- Dr. Jong Tae Jeong (KAERI)

Thank you for your attention!