Influence of Fuel Design and Reactor Operation on Spent Fuel Management

International Conference on
The Management of Spent Fuel from Nuclear Power Reactors

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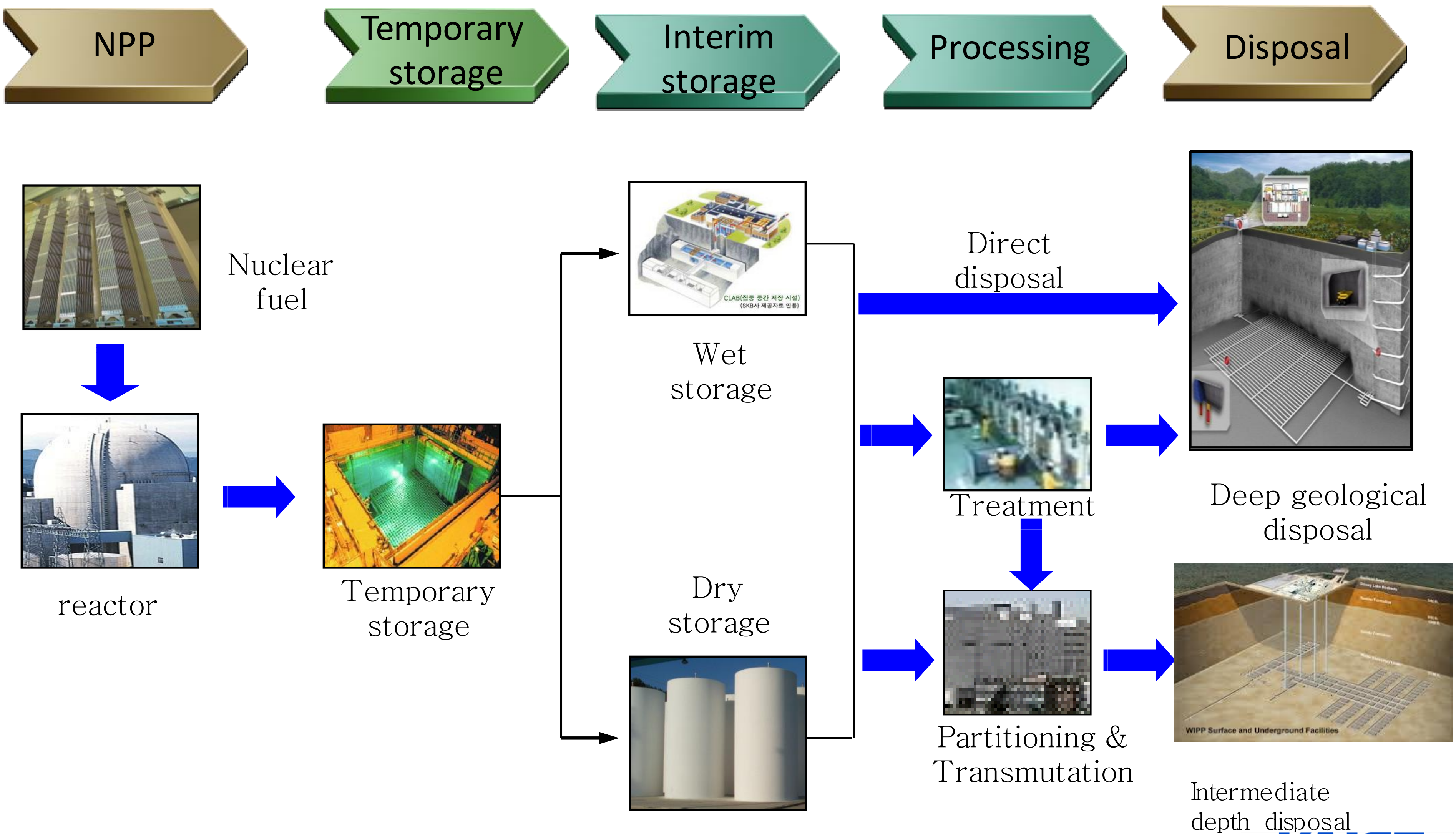
Man-Sung Yim
Department of Nuclear & Quantum Engineering
KAIST

Spent Fuel Management



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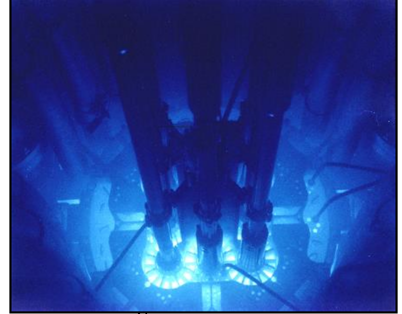
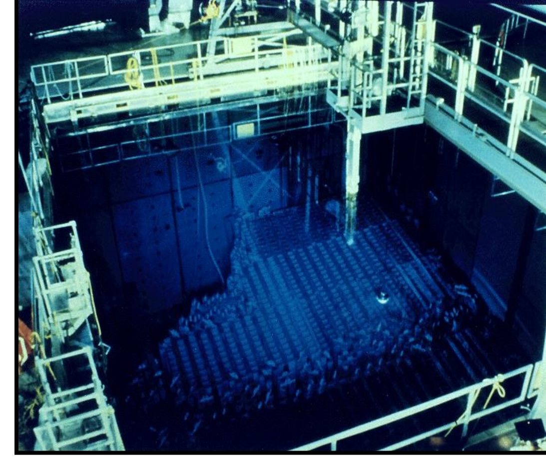
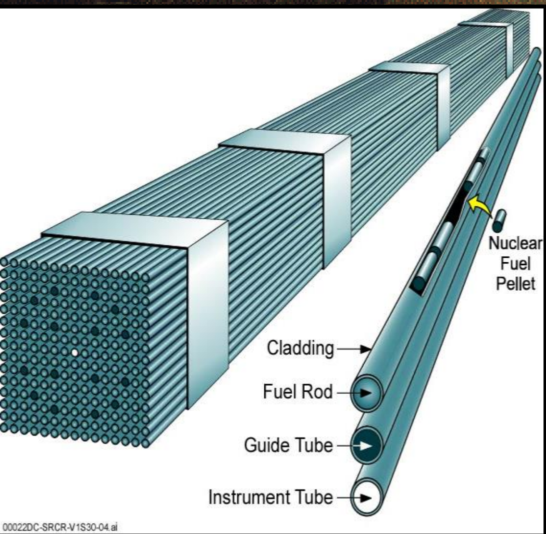
Spent Fuel Functions

- Confinement/Containment
- Criticality Control
- Retrievability to facilitate safe retrieval of the fuel
- Maintain radiological dose within the prescribed safety envelope
- Decay heat removal







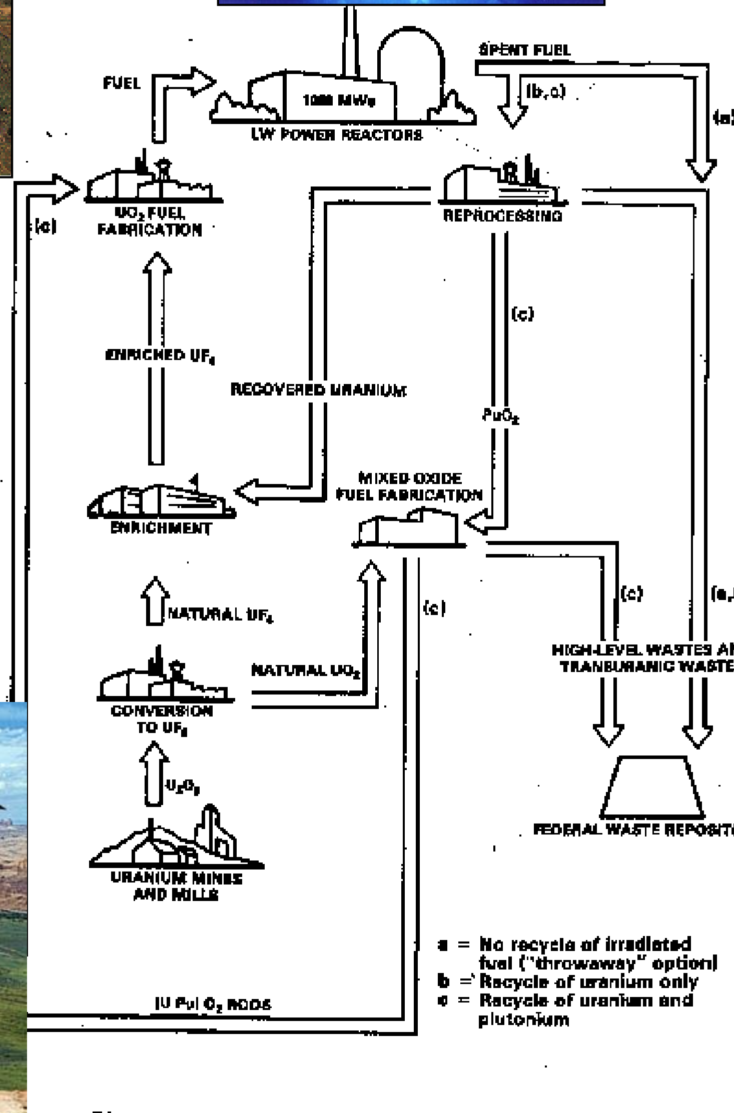
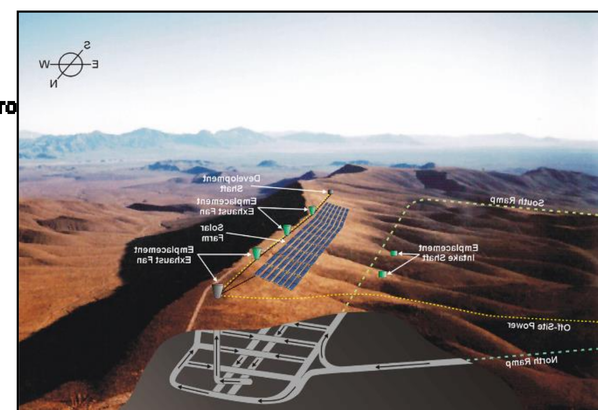
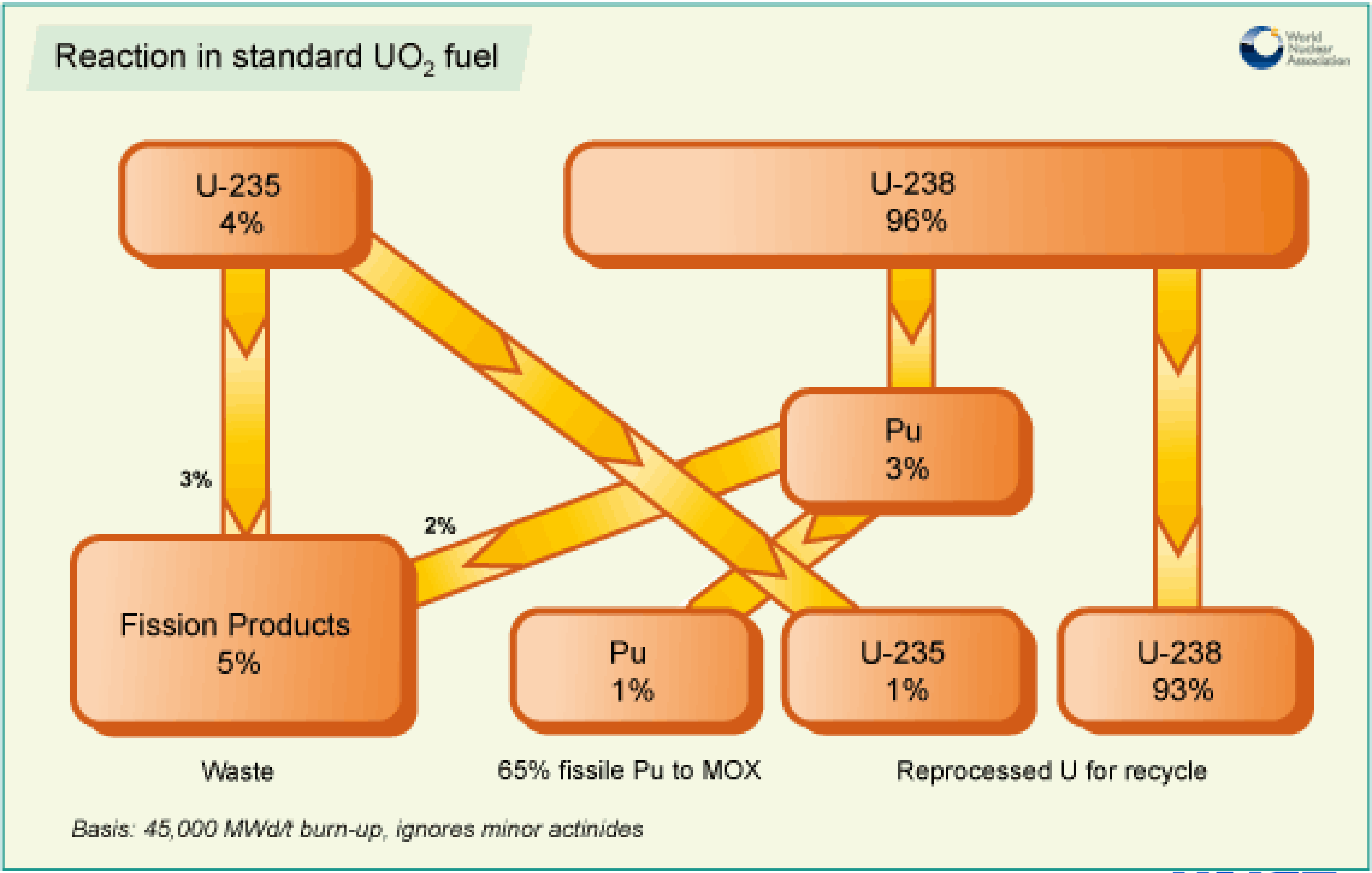


Figure 1 Current options for the nuclear fuel cycle





Reaction in standard UO_2 fuel
World Nuclear Association
U-235
4%
U-238
96%
3%
2%
Pu
3%
Fission Products
5%
Pu
1%
U-235
1%
U-238
93%
Waste
65% fissile Pu to MOX
Reprocessed U for recycle
Basis: 45,000 MWd/t burn-up, ignores minor actinides

Reactor Irradiation
(Example)

	Before	After	Difference
U-235	33	7.9	-25.1
U-236	0	4	+4
U-238	967	942.9	-24.1
Np-237	0	0.75	+0.75
Am+Cm	0	0.2	+0.2
Pu	0	9.05	+9.05
F.P.	0	35.1	+35.1
Total	1000	999.1	0.1

The radionuclides contained in spent fuel

- Fission products:
  - ^{90}Sr, ^{137}Cs, ^{3}H, ^{129}I, ^{99}Tc, ^{95}Nb, ^{133}Xe, ^{144}Ce, ^{135}Cs, etc.
- Actinides:
  - ^{237}Np, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, ^{244}Cm, etc.

    (most of these are also activation products)
- Activation products:
  - $^{14}N(n,p)^{14}C$, $^{17}O(n,\alpha)^{14}C$, $^{59}Co(n,\gamma)^{60}Co$, $^{62}Ni(n,\gamma)^{63}Ni$, $^{54}Fe(n,\gamma)^{55}Fe$, etc.

The material balance

$$\frac{dX_{ij}}{dt} = \sum_{k=1}^{N_i} l_{k\to j}\lambda_k X_{ik} + \sum_{k=1}^{N_i} f_{k\to j} X_{ik}\int \sigma_k(E)\phi_i(E)dE - \lambda_j X_{ij} - X_{ij}\int \sigma_j(E)\phi_i(E)dE$$

$$i = 1, 2, \cdots, n; \quad j = 1, 2, \cdots, N_i$$

where X_{ij} = atom number density of nuclide j in cell i,

N_i = number of nuclides in cell i,

$l_{k\to j}$ = fraction of radioactive disintegrations by k that lead to formation of j,

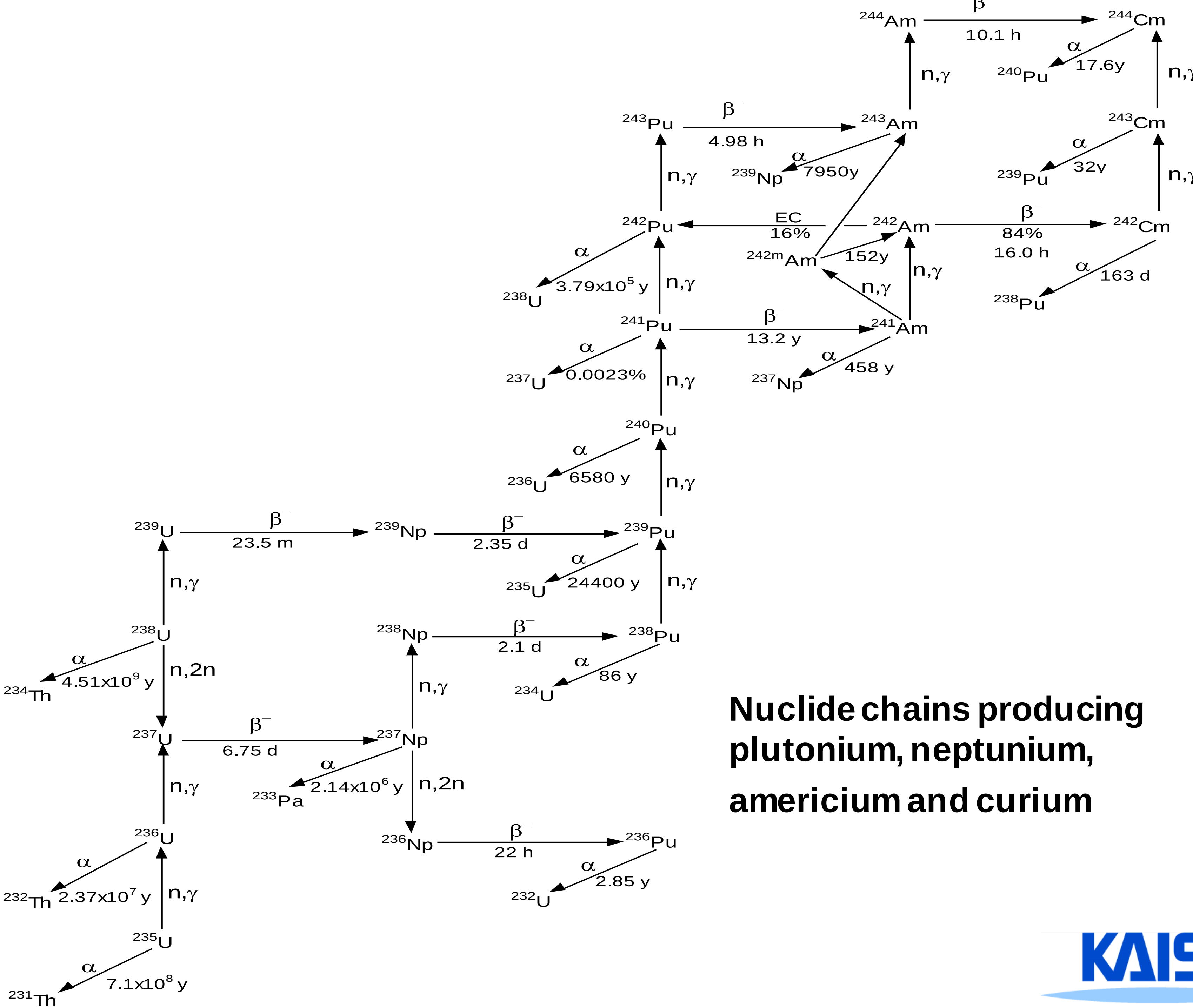
λ_k = radioactive decay constant of nuclide k,

$f_{k\to j}$ = fraction of neutron absorptions by k that lead to formation of nuclide j,

$\phi_i(E)$ = spatial-average neutron energy spectrum in cell i,

$\sigma_k(E)$ = neutron absorption cross section of nuclide k.

A. G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," *Nuclear Technology*, **62**, 335-352, September 1983.

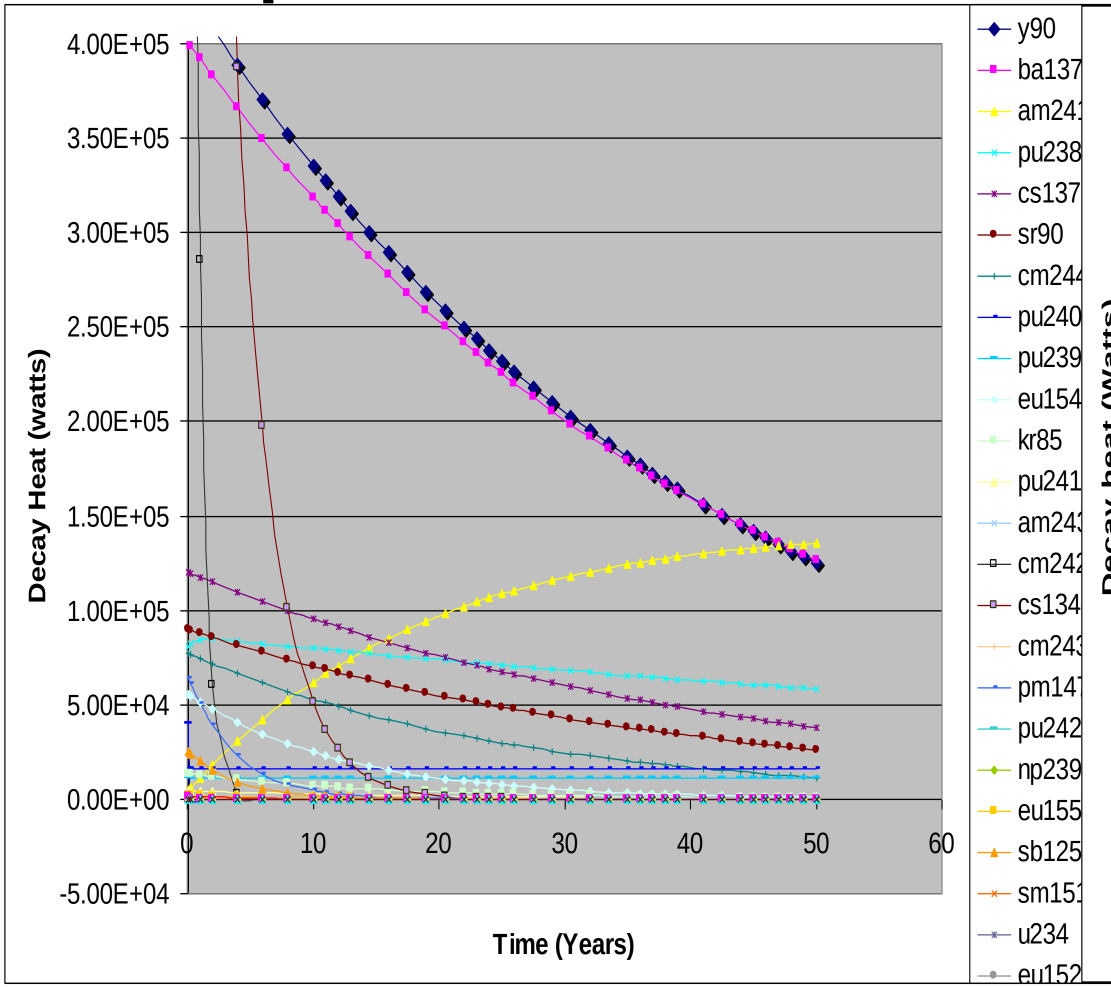


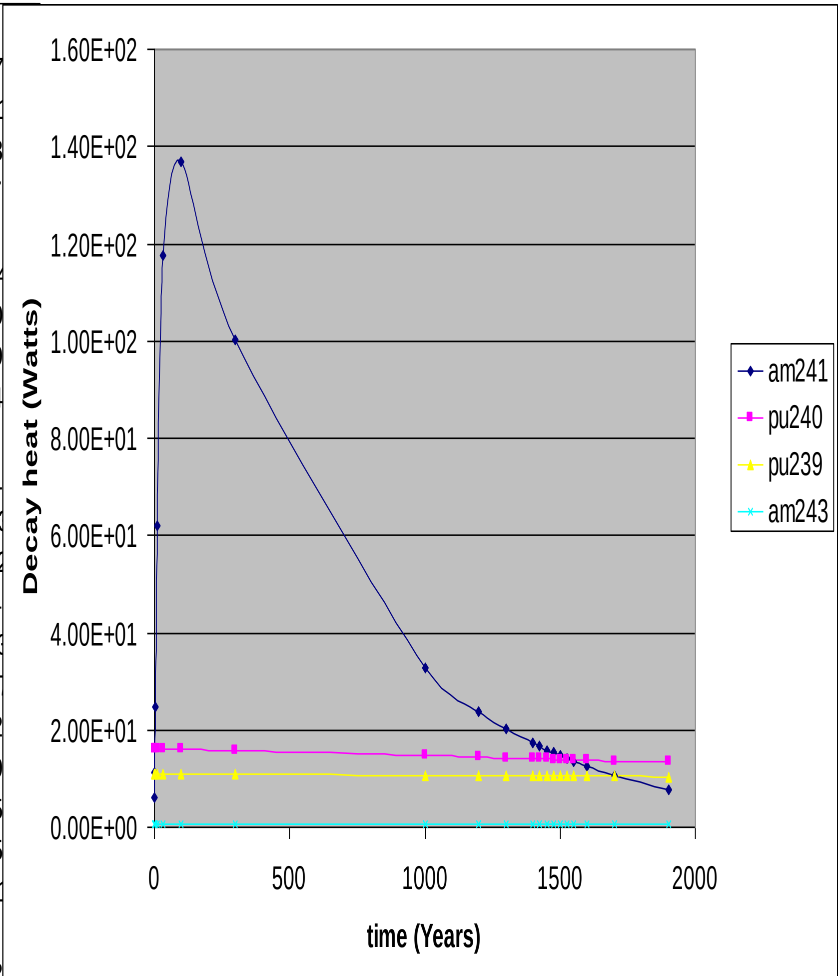
Nuclide chains producing plutonium, neptunium, americium and curium

Key Radionuclides

- Decay Heat: Cs-137, Sr-90, Am-241, Cm-244, Pu-238
- Radiation Dose:
  - Gamma dose: Co-60, Cs-134, Cs-137/Ba-137m, Sr-90/Y-90, Eu-154, Ce-144/Pr-144, Rh-106
  - Neutron dose: Cm-244, Cm-246, Pu-238, Pu-240, Pu-242, Cm-245, Am-241
- Criticality: U, Pu, Am, Np
- Repository Dose: I-129, Tc-99, Np-237, Cs-135, Sn-126, Pu-239, Nb-94

Changes in Decay Heat in LWR Spent Fuel (Burnup @44 GWD/MTU)



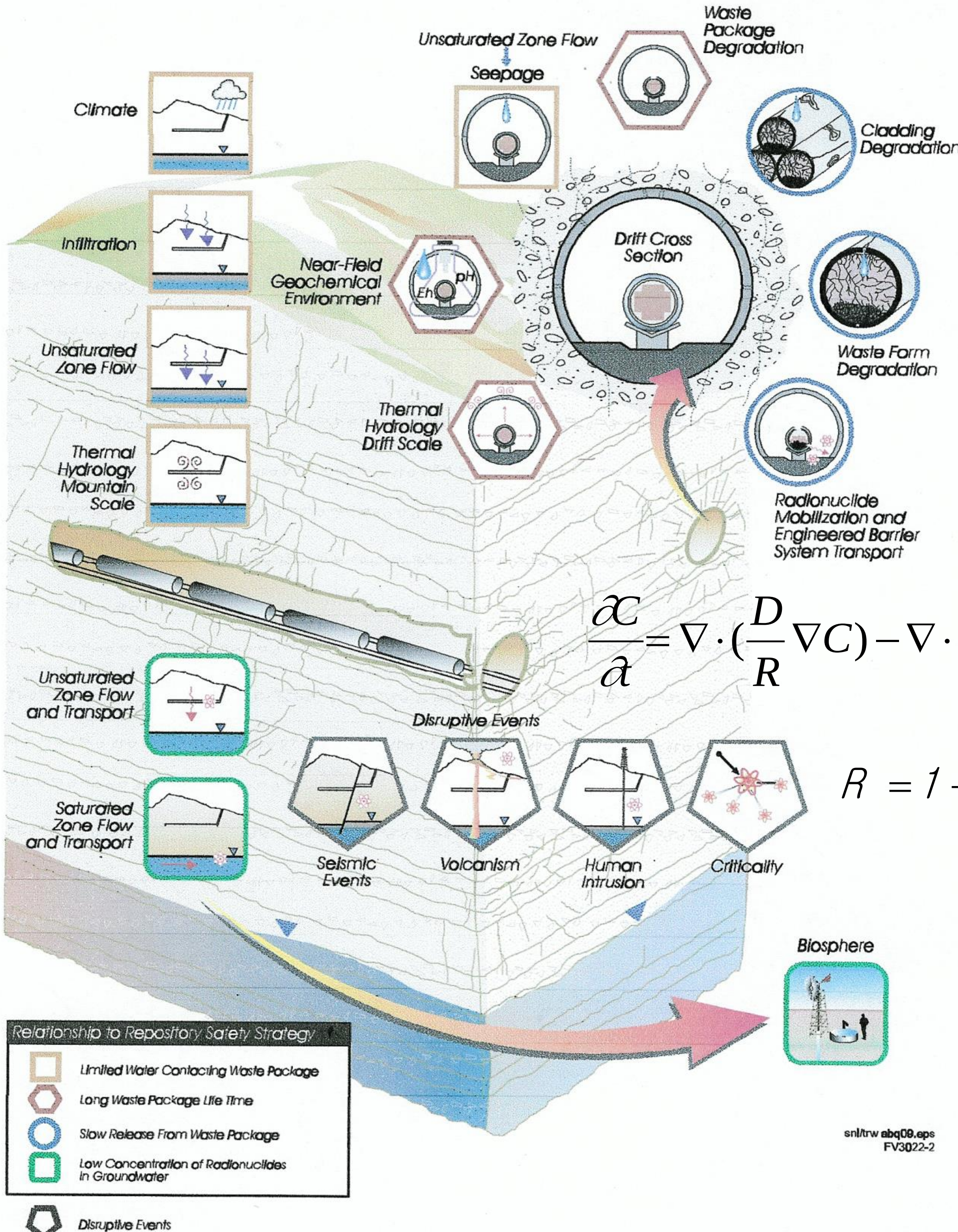


$$^{90}_{38}Sr \xrightarrow{29.12\,year} {}^{90}_{39}Y \xrightarrow{64.0\,hr} {}^{90}_{40}Zr\ (stable)$$

$$^{137}_{55}Cs \xrightarrow{30.17\,year} {}^{137m}_{56}Ba \xrightarrow{2.552\,min} {}^{137}_{56}Ba\,(stable)$$

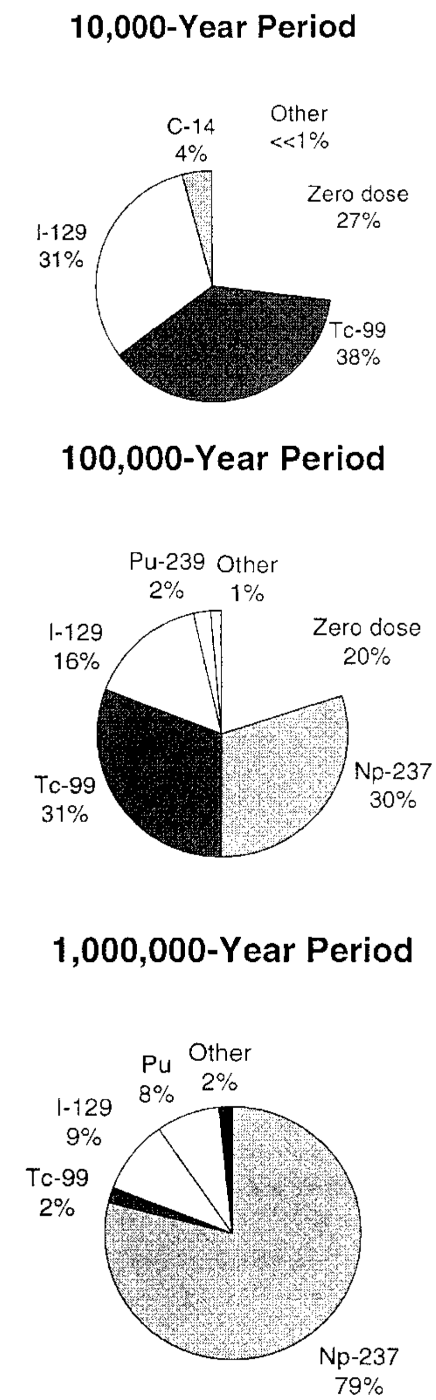
Nuclear Properties of Fissile and Fertile Materials

Isotope	Half-life (y)	Neutrons/sec-kg	Watts/kg	Critical mass (kg) (bare sphere)	Comment
Pa-231	32.8×10^{3}	Nil	1.3	162	
Th-232	14.1×10^{9}	Nil	Nil	Infinite	
U-233	159×10^{3}	1.23	0.281	16.4	
U-235	700×10^{6}	0.364	6×10^{-5}	47.9	
U-238	4.5×10^{9}	0.11	8×10^{-6}	Infinite	
Np-237	2.1×10^{6}	0.139	0.021	59	
Pu-238	88	2.67×10^{6}	560	10	Heat
Pu-239	24×10^{3}	21.8	2.0	10.2	
Pu-240	6.54×10^{3}	1.03×10^{6}	7.0	36.8	Neutrons
Pu-241	14.7	49.3	6.4	12.9	
Pu-242	376×10^{3}	1.73×10^{6}	0.12	89	Neutrons
Am-241	433	1540	115	57	Heat
Am-243	7.38×10^{3}	900	6.4	155	
Cm-244	18.1	11×10^{9}	2.8×10^{3}	28	Neutrons & Heat
Cm-245	8.5×10^{3}	147×10^{3}	5.7	13	
Cm-246	4.7×10^{3}	9×10^{9}	10	84	
Bk-247	1.4×10^{3}	Nil	36	10	
Cf-251	**898**	**Nil**	**56**	**9**	



$$\frac{\partial C}{\partial t}=\nabla\cdot\left(\frac{D}{R}\nabla C\right)-\nabla\cdot\left(\frac{\upsilon}{R}C\right)-\lambda C$$

$$R = 1+\frac{K_d\,\rho_b}{\varepsilon}$$



Top Dose Contributors, US DOE (1998)

Major Factors Affecting Nuclide Inventory

- Neutron spectrum
  - Fuel temperature
  - Moderator temperature
  - Moderator density
    - Hydrogen-to-heavy metal ratio
    - Lattice geometry
  - Soluble boron
  - Burnable poisons
- Operating history
  - Specific power
    - Power level variations
    - Load-following
    - Power uprating
  - Cycle length
    - Burnup
  - Recycling (Fuel cycle options)
- Cooling
  - Decay of fission products
  - Buildup of actinides

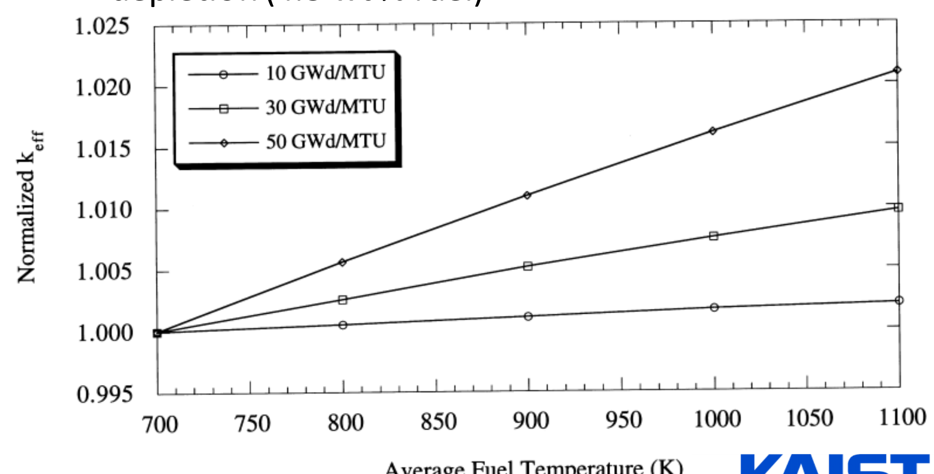
Neutron Spectrum Effects

- Fuel temperature
  - At higher fuel temperature, resonance absorption in ^{238}U is increased due to Doppler broadening.
  - This results in spectral hardening and increased plutonium (e.g., Pu-239, Pu-241) production.
- Moderator temperature
  - As the moderator temperature increases, the moderator density decreases.
  - This leads into spectral hardening and increased plutonium production.
- Moderator density
  - Hydrogen-to-heavy metal ratio
  - Lattice geometry
- Soluble boron
  - Spectral hardening results from the absorption of thermal neutrons in the moderator by the soluble poison.
- Burnable poisons
  - Localized spectral hardening

Trends in k_∞ with varying fuel temperature during depletion

Ave. fuel temperature (K) GWd/MTU	Infinite lattice neutron multiplication factor, k_∞ 3.0 wt % 10 GWd/MTU	3.0 wt % 30 GWd/MTU	3.0 wt % 50 GWd/MTU	4.5 wt %	4.5 wt % 10 GWd/MTU	4.5 wt % 30 GWd/MTU 50
700	0.86070	0.72494	0.63645	1.00368	0.86387	0.74521
800	0.86179	0.72848	0.64139	1.00421	0.86610	0.74940
900	0.86282	0.73184	0.64626	1.00476	0.86832	0.75336
1000	0.86383	0.73499	0.65094	1.00534	0.87034	0.75719
1100	0.86471	0.73781	0.65515	1.00580	0.87224	0.76077

Trends in k_∞ with varying fuel temperature during depletion (4.5 wt % fuel)



M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," ORNL/TM-12973, 1996.

Trends in k_∞ with varying moderator temperature during depletion

Ave. moderator temperature (K)	Infinite lattice neutron multiplication factor, k_∞					
	3.0 wt % 10 GWd/MTU	3.0 wt % 30 GWd/MTU	3.0 wt % 50 GWd/MTU	4.5 wt % 10 GWd/MTU	4.5 wt % 30 GWd/MTU	4.5 wt % 50 GWd/MTU
500	0.84784	0.67865	0.56747	0.99761	0.83523	0.68731
520	0.84977	0.68588	0.57757	0.99852	0.83964	0.69622
540	0.85212	0.69457	0.59033	0.99967	0.84505	0.70729
560	0.85506	0.70484	0.60561	1.00109	0.85155	0.72031
580	0.85867	0.71706	0.62403	1.00276	0.85912	0.73539
600	0.86282	0.73184	0.64626	1.00476	0.86832	0.75336

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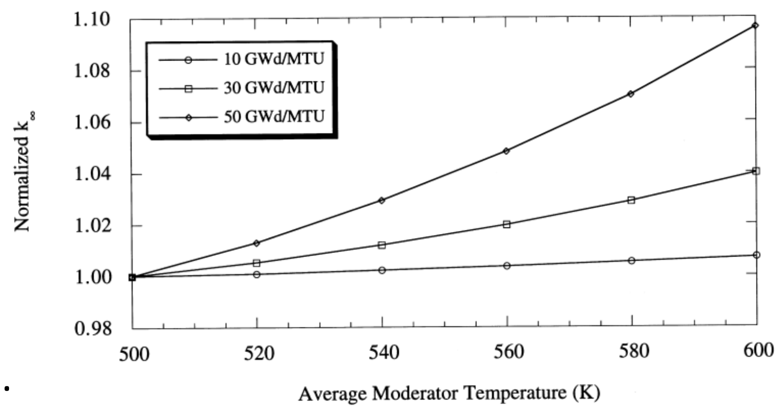


Trends in k_∞ with varying moderator temperature during depletion (4.5 wt % fuel).

Hydrogen-Heavy Metal Ratio (H/HM) Variations in Different Fuel Designs

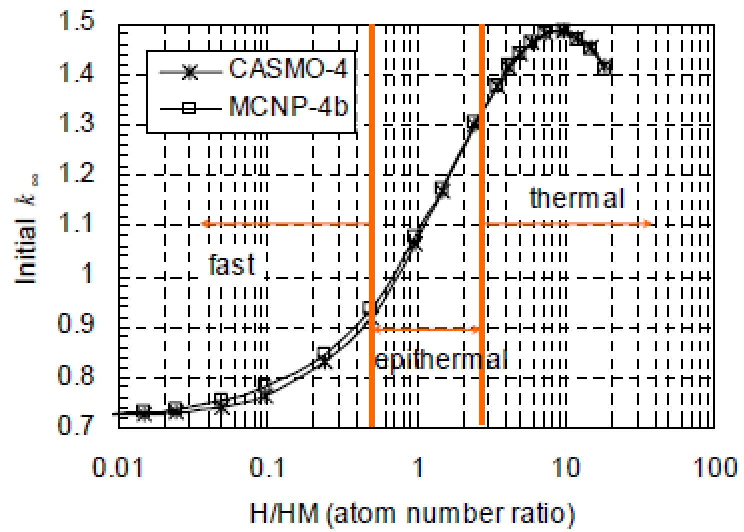
Designer	Pitch P (cm)	Rod Dia. D (cm)	Pellet Dia. d (cm)	Pellet Den. ρ (g/cm^3)	V_M/V_{PIN}	V_M/V_F	H/HM†
ABB Atom							
16×16	1.435	1.075	0.911	10.50	1.27	1.77	3.56
17×17	1.26	0.95	0.819	10.50	1.24	1.67	3.36
18×18	1.275	0.95	0.805	10.50	1.29	1.80	3.63
ABB CENP							
14×14	1.47	1.118	0.968	10.5	1.20	1.60	3.23
16×16	1.29	0.970	0.826	10.5	1.25	1.73	3.48
BNFL/Westinghouse							
17×17	1.26	0.914	0.784	10.5	1.42	1.93	3.89
European Fuel Group (EFG)							
14×14	1.407	1.072	0.929	10.47	1.19	1.59	3.21
15×15	1.43	1.072	0.929	10.5	1.27	1.69	3.39
17×17	1.26	0.95	0.819	10.47	1.24	1.67	3.37
Framatome Cogema Fuels (FCF)							
15×15	1.434	1.06	0.918	10.52	1.33	1.77	3.57
17×17	1.26	0.95	0.819	10.52	1.24	1.67	3.35
Fragema							
14×14	1.408	1.072	0.929	10.5	1.20	1.59	3.21
15×15	1.43	1.072	0.929	10.5	1.27	1.69	3.39
16×16	1.434	1.075	0.912	10.5	1.27	1.76	3.54
17×17	1.26	0.95	0.819	10.5	1.24	1.67	3.36
18×18	1.275	0.95	0.806	10.5	1.29	1.80	3.62
Korea Nuclear Fuels							
14×14	1.407	1.016	0.8748	10.43	1.44	1.94	3.94
16×16	1.29	0.97	0.826	10.46	1.25	1.73	3.49
17×17	1.26	0.95	0.8192	10.43	1.24	1.67	3.38
Mitsubishi							
14×14	1.407	1.072	0.929	10.41	1.19	1.59	3.23
15×15	1.43	1.072	0.929	10.41	1.27	1.69	3.42
17×17	1.26	0.95	0.819	10.41	1.24	1.67	3.39
VVER-1000							
Hex	1.275	0.91	0.755*	10.5	1.16	1.89	3.80

† H/HM uses the estimated value of the H atom concentration at 300°C for a water density of 0.705 g/cm^3.

* annular pellet, 0.24 cm diameter central hole.

** All dimensions are cold (room temperature): pitch increases by ~ 0.2% when hot.

*** After considering the presence of water-filled guide tubes and the gap between assemblies, the whole core H/HM value can be obtained through multiplying unit cell H/HM by ~ 1.16.



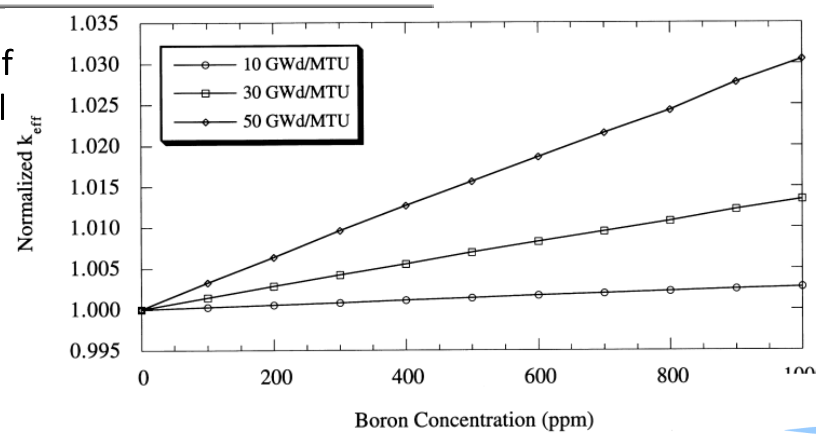
Initial $k\infty$ as a function of H/HM ratio, 4.5% U-235 enrichment

Xu, Zhiwen, "Design Strategies for Optimizing High Burnup Fuel in Pressurized Water Reactors", MIT doctoral thesis, January 2003.

Effect of moderator boron concentrations on k_∞

Boron concentration (ppm)	Infinite lattice neutron multiplication factor, k_∞ 3.0 wt % 10 GWd/MTU	3.0 wt % 30 GWd/MTU	3.0 wt % 50 GWd/MTU	4.5 wt % 10 GWd/MTU	4.5 wt % 30 GWd/MTU	4.5 wt % 50 GWd/MTU
0	0.85904	0.71966	0.62853	1.00340	0.86233	0.74181
100	0.85981	0.72215	0.63221	1.00367	0.86353	0.74424
200	0.86055	0.72464	0.63583	1.00395	0.86475	0.74651
300	0.86132	0.72696	0.63943	1.00421	0.86593	0.74894
400	0.86212	0.72943	0.64295	1.00449	0.86705	0.75117
500	0.86282	0.73184	0.64626	1.00476	0.86832	0.75336
600	0.86359	0.73403	0.64971	1.00509	0.86941	0.75557
700	0.86428	0.73622	0.65292	1.00536	0.87053	0.75773
800	0.86503	0.73855	0.65608	1.00562	0.87163	0.75983
900	0.86575	0.74069	0.65924	1.00588	0.87281	0.76235
1000	0.86644	0.74282	0.66231	1.00615	0.87391	0.76446

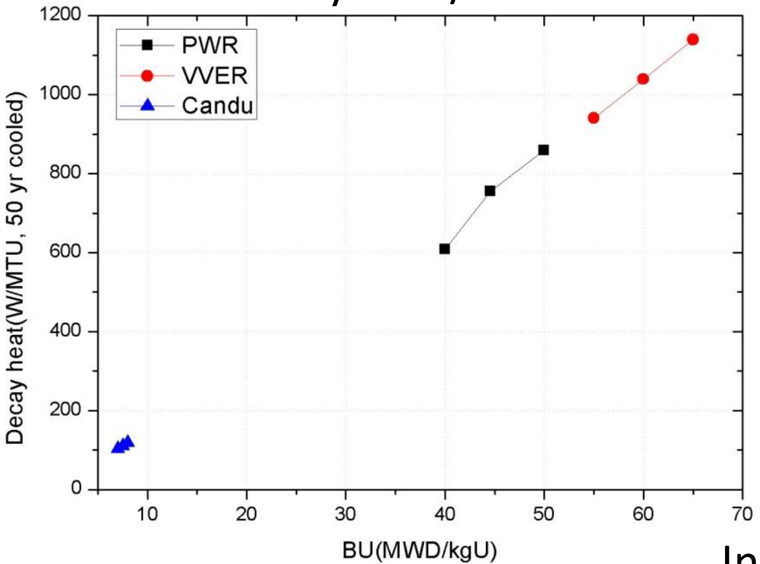
M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," ORNL/TM-12973, 1996.



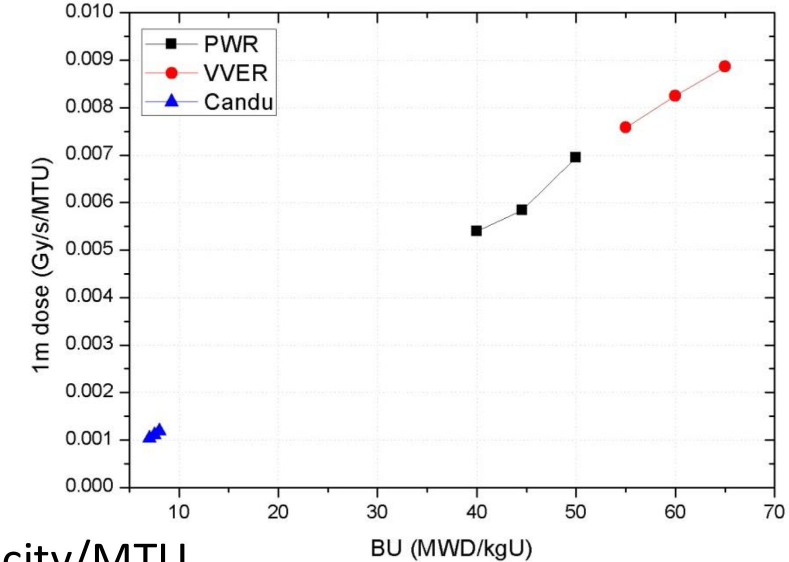
Trends in k_∞ with varying boron concentration during depletion (4.5 wt % fuel).

Comparison of PWR, VVER, and CANDU

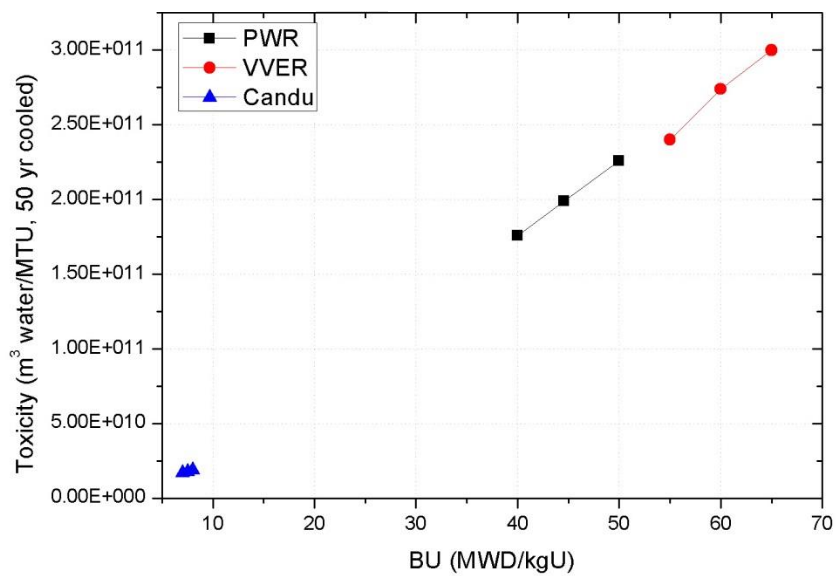
Decay Heat/MTU



1m Dose/MTU



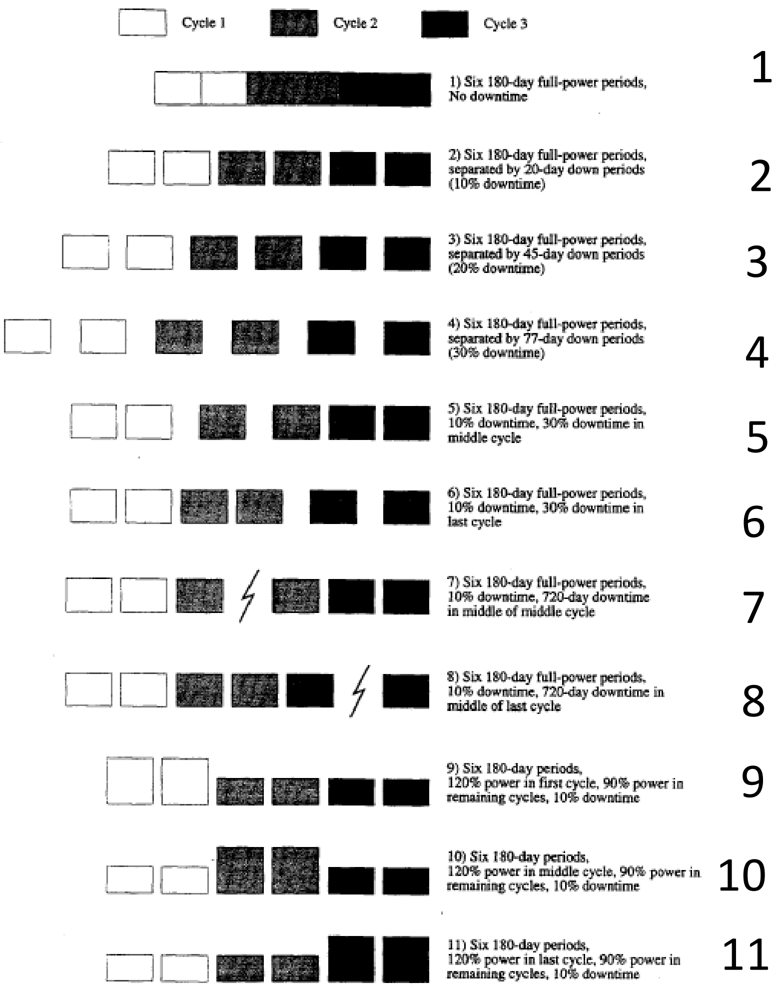
Ingestion Toxicity/MTU

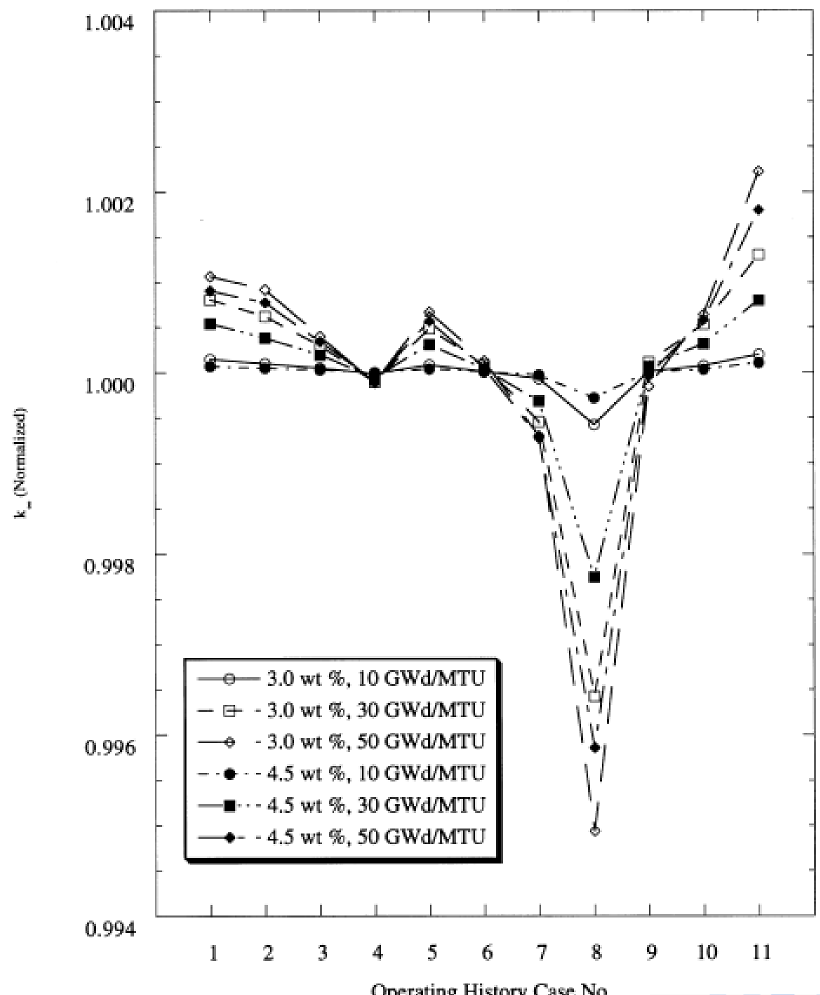


Operating History Effects

- Specific power
  - Power level
    - Changes in power level directly affect nuclide inventory.
    - Duration of down time between cycles or during the cycle has an impact.
  - Power uprating
    - Power uprating increase in depletion and radioactivity buildup.
  - Axial variations in flux
    - Axial variations in flux result in a non-uniform burnup distribution along the axial length of SNF.
  - Load-following
    - Axial burnup profile variations
- Cycle length
  - Burnup
- Reycling

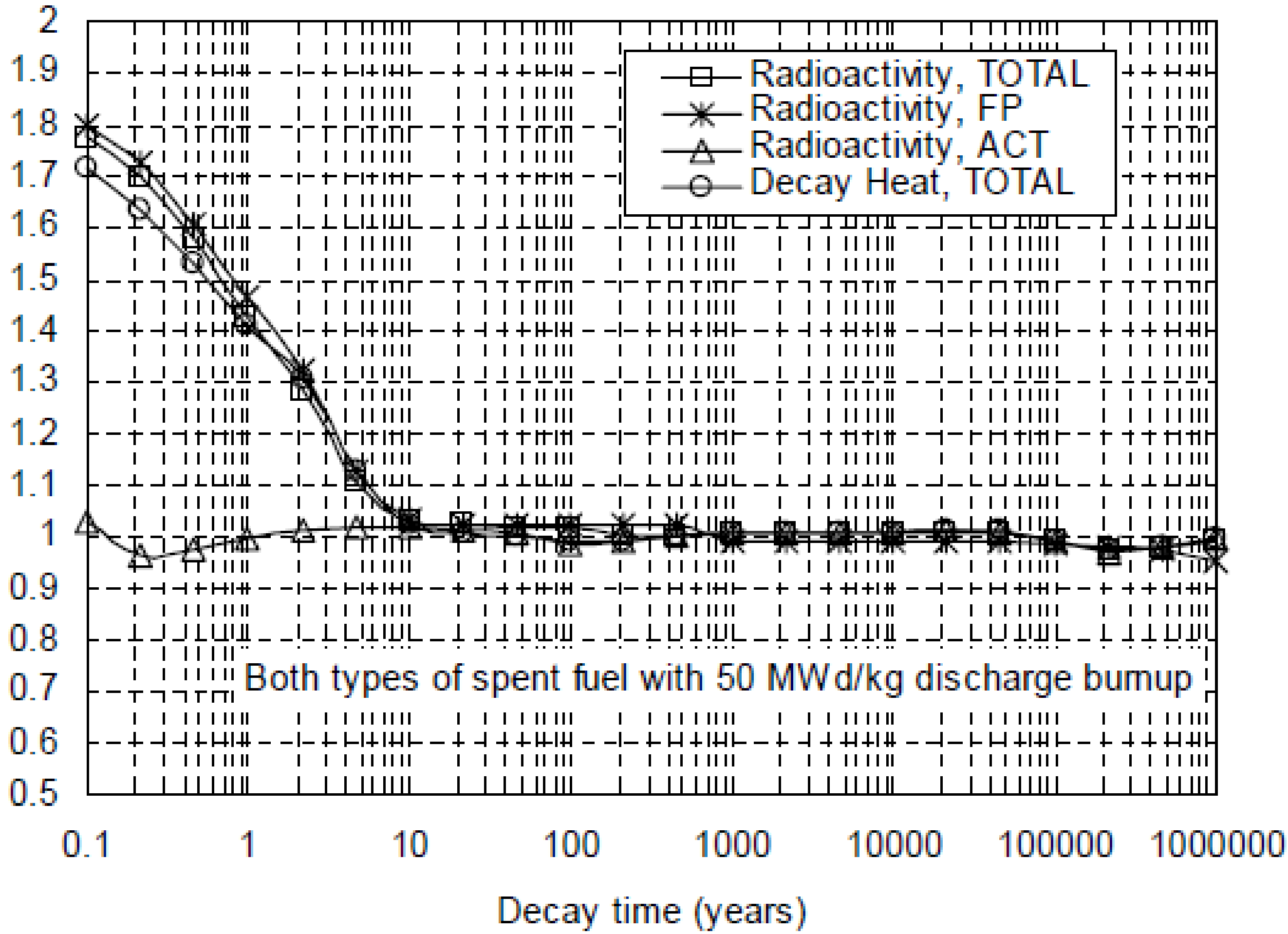
Effect of Operating History on k_∞ (Nuclide Inventory)





M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages," ORNL/TM-12973, 1996.

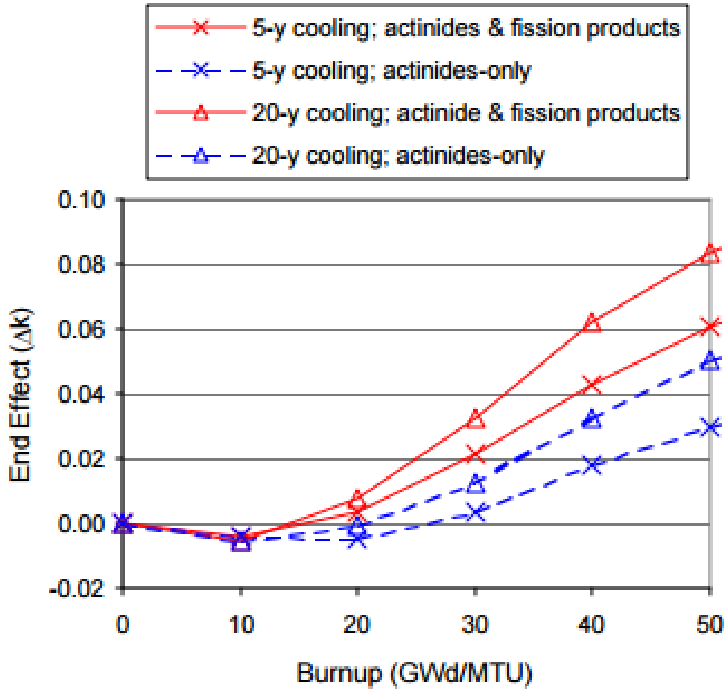
Effect of power uprating on radioactivity of spent fuel



Xu, Zhiwen,"Design Strategies for Optimizing High Burnup Fuel in Pressurized Water Reactors", MIT doctoral thesis, January 2003.

Axial variations in neutron flux

- The end effect
  - The erroneous prediction of the multiplication factor when assuming a uniform-burnup distribution
  - The top of spent fuel is less burned.



End effect in GBC-32 cask.

J. C. WAGNER, Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit, ORNL/TM-2000/306

Actinide compositions (wt%) in high-burnup spent fuels

B_d	33 MWd/kg	50 MWd/kg	100 MWd/kg	150 MWd/kg
10 years after discharge				
U-234	0.02	0.02	0.04	0.06
U-235	0.90	1.07	1.62	2.21
U-236	0.38	0.61	1.47	2.60
U-238	97.54	96.77	94.33	91.66
Np-237	0.04	0.08	0.20	0.34
Pu-238	0.01	0.03	0.14	0.29
Pu-239	0.65	0.78	1.12	1.42
Pu-240	0.24	0.31	0.46	0.58
Pu-241	0.10	0.13	0.22	0.29
Pu-242	0.05	0.08	0.15	0.19
Am-241	0.06	0.09	0.16	0.21
Am-242m	—	—	—	0.01
Am-243	0.01	0.02	0.05	0.08
Cm-244	—	0.01	0.03	0.05
Cm-245	—	—	0.01	0.01
Pu-239/(Pu_{total})	62%	57%	54%	51%

B_d	33 MWd/kg	50 MWd/kg	100 MWd/kg	150 MWd/kg
1000 years after discharge				
U-234	0.03	0.06	0.17	0.35
U-235	1.06	1.10	1.65	2.25
U-236	0.54	0.64	1.52	2.66
U-238	97.54	96.76	94.36	91.66
Np-237	0.20	0.25	0.49	0.74
Pu-238	—	—	—	—
Pu-239	0.49	0.76	1.09	1.39
Pu-240	0.09	0.28	0.44	0.57
Pu-241	—	—	—	—
Pu-242	0.05	0.08	0.15	0.19
Am-241	—	0.05	0.08	0.11
Am-242m	—	—	—	—
Am-243	—	0.02	0.05	0.07
Cm-244	—	—	—	—
Cm-245	—	—	—	0.01
Pu-239/Pu_{total}	78%	68%	65%	65%

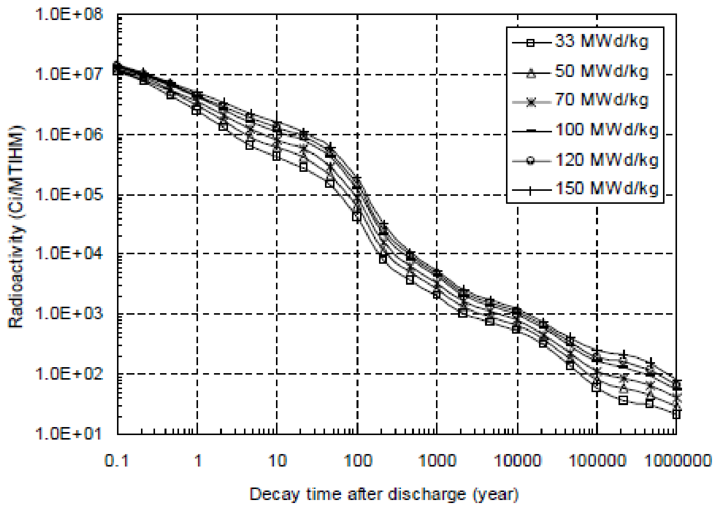
M. D. DeHart, “Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages,” ORNL/TM-12973, 1996.

Changes in activity and heat load of different spent fuel as a function of cooling time

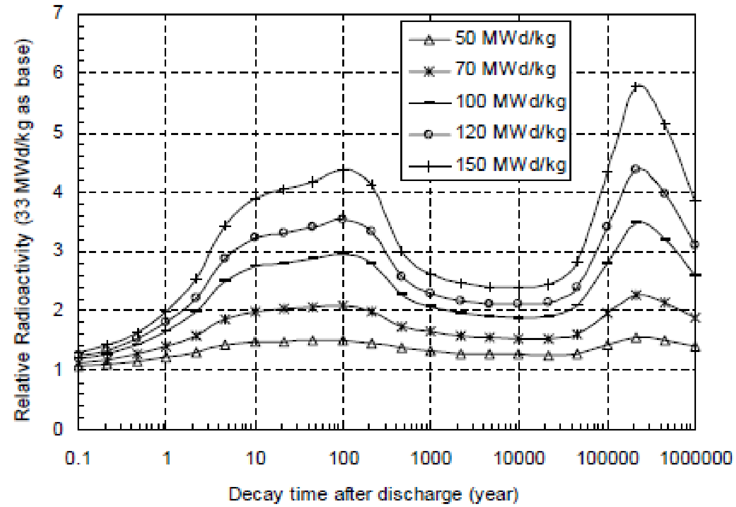
	Standard Burnup UO2 fuel		Extended burnup UO2 fuel (60 GWD/THM)		MOX fuel (50 GWD/THM)	
Cooling periods-yr	Actinides	Total	Actinides	Total	Actinides	Total
	Activity (Ci)					
1	5.75E+05	8.72E+06	8.06E+05	9.73E+06	4.11E+06	1.24E+07
3	4.85E+05	3.35E+06	6.66E+05	4.42E+06	3.47E+06	6.52E+06
5	4.42E+05	2.13E+06	6.07E+05	3.10E+06	3.18E+06	4.96E+06
7	4.04E+05	1.73E+06	5.56E+05	2.61E+06	2.91E+06	4.30E+06
10	3.54E+05	1.46E+06	4.89E+05	2.24E+06	1.92E+06	2.83E+06
	Heat load (W)					
1	2.84E+03	3.75E+04	6.04E+03	4.51E+04	2.92E+04	6.48E+04
3	1.20E+03	1.24E+04	3.01E+03	1.80E+04	1.66E+04	2.86E+04
5	1.13E+03	6.92E+03	2.80E+03	1.16E+04	1.56E+04	2.18E+04
7	1.12E+03	5.25E+03	2.71E+03	9.28E+03	1.51E+04	1.94E+04
10	1.11E+03	4.36E+03	2.60E+03	7.84E+03	1.32E+04	1.57E+04

Cooling Effects

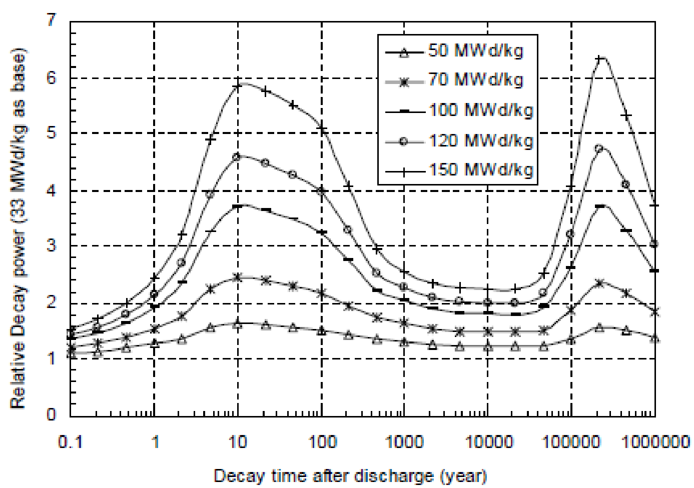
Radioactivity per MTIHM after discharge



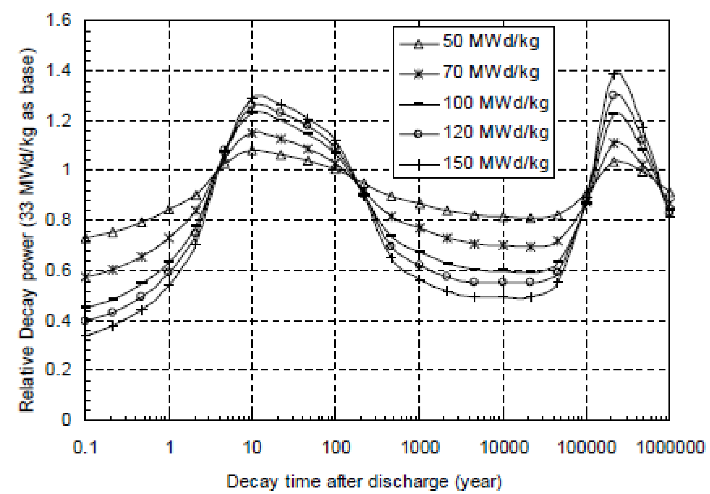
Radioactivity per GW-yr(e) after discharge



Decay power per MTIHM after discharge

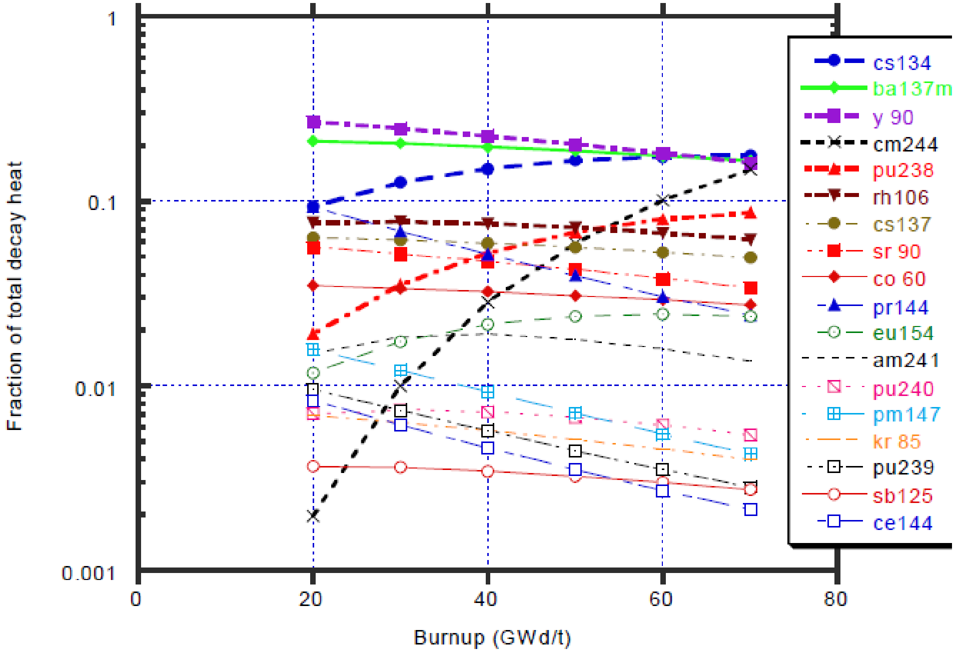


Decay power per GW-yr(e) after discharge

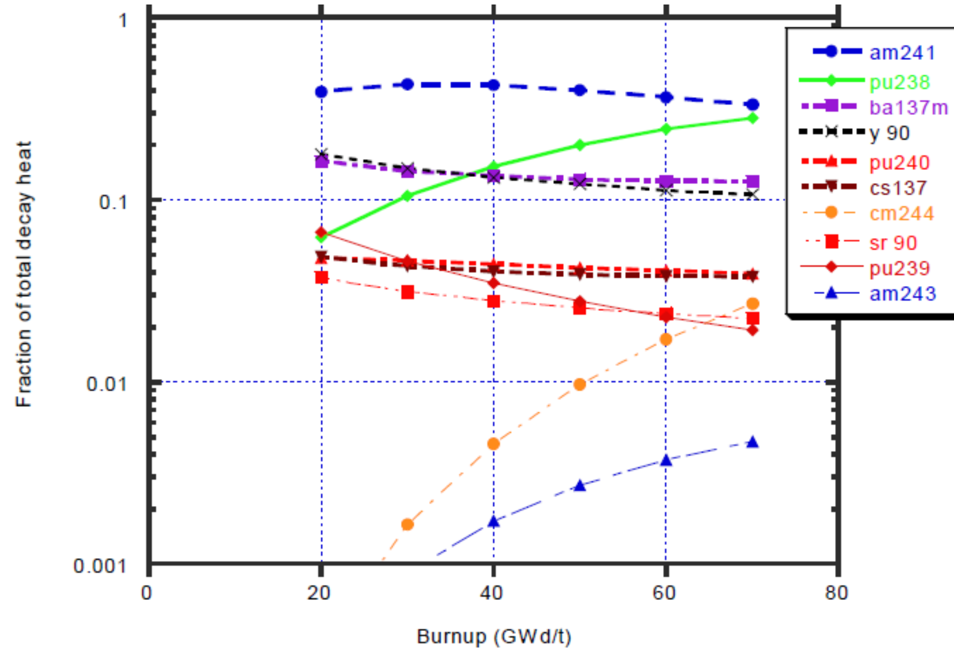


Xu, Zhiwen,“Design Strategies for Optimizing High Burnup Fuel in Pressurized Water Reactors”, MIT doctoral thesis, January 2003.

Fraction of decay-heat generation for 5-wt % 235U PWR fuel



5-year cooling



100-year cooling

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

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Radionuclides decay-heat rankings for PWR fuel

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	1.96	10	1.54	12	1.96	10	1.44	12	1.90	12	1.36	12
Ba-137m	20.10	2	16.81	2	20.01	2	16.68	2	19.75	2	16.51	2
Ce-144	0.75	15	0.29	18	0.58	17	0.25	18	0.46	17	0.21	18
Cm-244	0.69	16	13.25	4	1.66	12	14.10	4	2.84	10	14.91	4
Co-60	4.68	8	3.80	8	3.81	9	3.18	9	3.23	9	2.74	9
Cs-134	10.88	3	17.45	1	13.43	3	17.70	1	15.01	3	17.74	1
Cs-137	6.02	6	5.03	7	5.99	6	4.99	7	5.92	5	4.95	7
Eu-154	1.41	11	2.26	11	1.84	11	2.32	11	2.15	11	2.38	11
Kr-85	0.60	17	0.39	15	0.59	16	0.40	15	0.57	15	0.40	15
Pm-147	1.36	12	0.53	14	1.11	13	0.47	14	0.93	13	0.43	14
Pr-144	8.42	5	3.24	9	6.49	5	2.77	10	5.15	7	2.40	10
Pu-238	2.38	9	6.43	6	3.83	8	7.59	5	5.24	6	8.66	5
Pu-239	0.87	14	0.34	16	0.68	15	0.31	16	0.57	16	0.28	16
Pu-240	0.94	13	0.68	13	0.82	14	0.60	13	0.72	14	0.54	13
Pu-241	0.19	19	0.14	19	0.18	19	0.13	19	0.17	19	0.12	19
Rh-106	10.08	4	8.57	5	8.74	4	7.19	6	7.55	4	6.17	6
Sb-125	0.41	18	0.34	17	0.38	18	0.30	17	0.34	18	0.27	17
Sr-90	4.84	7	3.20	10	4.78	7	3.32	8	4.72	8	3.39	8
Y-90	23.08	1	15.27	3	22.82	1	15.83	3	22.48	1	16.15	3
Total (W/t)	1090		3170		1630		3790		2180		4430	
100-Year cooling												
Am-241	46.58	1	39.22	1	44.98	1	35.98	1	42.48	1	33.32	1
Am-243	0.08	10	0.50	10	0.13	10	0.49	10	0.17	10	0.47	10
Ba-137m	13.93	2	12.99	3	13.69	2	12.84	3	13.47	3	12.62	3
Cm-244	0.11	9	2.42	7	0.27	9	2.56	7	0.46	9	2.69	7
Cs-137	4.17	7	3.89	6	4.10	7	3.85	6	4.04	6	3.78	6
Pu-238	7.00	4	21.09	2	11.13	4	24.80	2	15.20	2	28.09	2
Pu-239	5.43	6	2.39	8	4.20	6	2.13	9	3.49	7	1.92	9
Pu-240	5.79	5	4.90	5	4.99	5	4.35	5	4.43	5	3.92	5
Sr-90	2.90	8	2.14	9	2.83	8	2.21	8	2.78	8	2.24	8
Y-90	13.84	3	10.22	4	13.52	3	10.55	4	13.28	4	10.68	4
Total (W/t)	175		456		265		549		356		645	

[a] Greater than 0.1% of total decay-heat-generation rate for any burnup/enrichment combination.

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

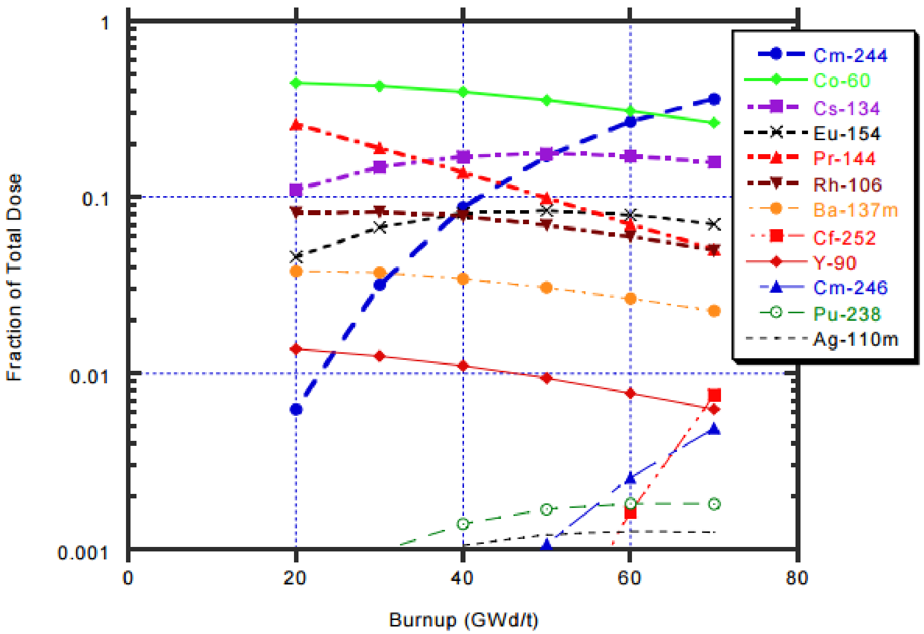
	Burnup / Enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	1.94	9	1.43	11	1.88	10	1.32	11	1.79	11	1.24	11
Ba-137m	20.72	2	17.18	2	20.48	2	17.04	2	20.18	2	16.87	2
Ce-144	0.78	15	0.30	18	0.60	17	0.25	18	0.47	17	0.22	18
Cm-244	0.78	16	13.94	4	1.80	11	14.68	4	2.99	9	15.38	4
Co-60	1.59	10	1.30	12	1.29	12	1.09	12	1.09	12	0.94	12
Cs-134	11.40	3	17.97	1	13.84	3	18.15	1	15.33	3	18.13	1
Cs-137	6.21	6	5.15	7	6.14	6	5.10	7	6.05	5	5.05	7
Eu-154	1.49	11	2.25	10	1.90	9	2.29	10	2.19	10	2.32	10
Kr-85	0.62	17	0.40	15	0.61	16	0.41	15	0.59	15	0.41	15
Pm-147	1.39	12	0.53	14	1.13	13	0.48	14	0.95	13	0.43	14
Pr-144	8.70	5	3.32	8	6.68	5	2.84	9	5.29	7	2.46	9
Pu-238	2.56	8	6.49	6	3.99	8	7.56	5	5.35	6	8.54	5
Pu-239	0.87	14	0.33	17	0.67	15	0.29	17	0.55	16	0.26	17
Pu-240	0.97	13	0.71	13	0.84	14	0.63	13	0.75	14	0.57	13
Pu-241	0.19	19	0.13	19	0.18	19	0.12	19	0.16	19	0.11	19
Rh-106	10.32	4	8.71	5	8.82	4	7.29	6	7.56	4	6.25	6
Sb-125	0.42	18	0.35	16	0.38	18	0.31	16	0.35	18	0.28	16
Sr-90	5.00	7	3.29	9	4.92	7	3.41	8	4.85	8	3.48	8
Y-90	23.83	1	15.70	3	23.47	1	16.28	3	23.12	1	16.62	3
Total (W/t)	1060		3090		1590		3710		2130		4330	
100-Year cooling												
Am-241	45.56	1	37.18	1	43.30	1	33.93	1	40.55	1	31.31	1
Am-243	0.08	10	0.51	10	0.13	10	0.49	10	0.17	10	0.47	10
Ba-137m	14.19	2	13.48	3	14.08	2	13.37	3	13.94	3	13.20	3
Cm-244	0.13	9	2.58	7	0.29	9	2.72	7	0.49	9	2.84	7
Cs-137	4.25	7	4.04	6	4.22	6	4.01	6	4.18	6	3.95	6
Pu-238	7.46	4	21.60	2	11.64	4	25.20	2	15.71	2	28.34	2
Pu-239	5.36	6	2.29	8	4.11	7	2.03	9	3.40	7	1.83	9
Pu-240	5.94	5	5.21	5	5.18	5	4.64	5	4.65	5	4.20	5
Sr-90	0.01	8	2.24	9	2.93	8	2.32	8	2.90	8	2.36	8
Y-90	14.13	3	10.66	4	13.96	3	11.06	4	13.82	4	11.25	4
Total (W/t)	172		439		257		526		343		616	

[a] Greater than 0.1 % of total decay-heat-generation rate for any burnup/enrichment combination.

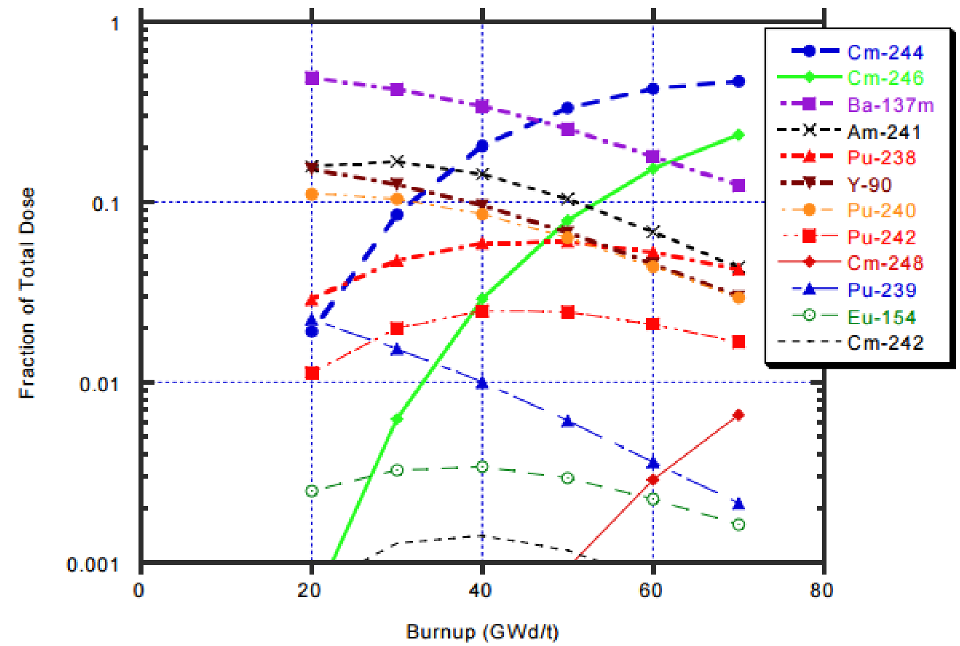
Radionuclides decay-heat rankings for BWR fuel

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

Fraction of total dose from dominant nuclides (5-wt % PWR fuel, steel transport cask)



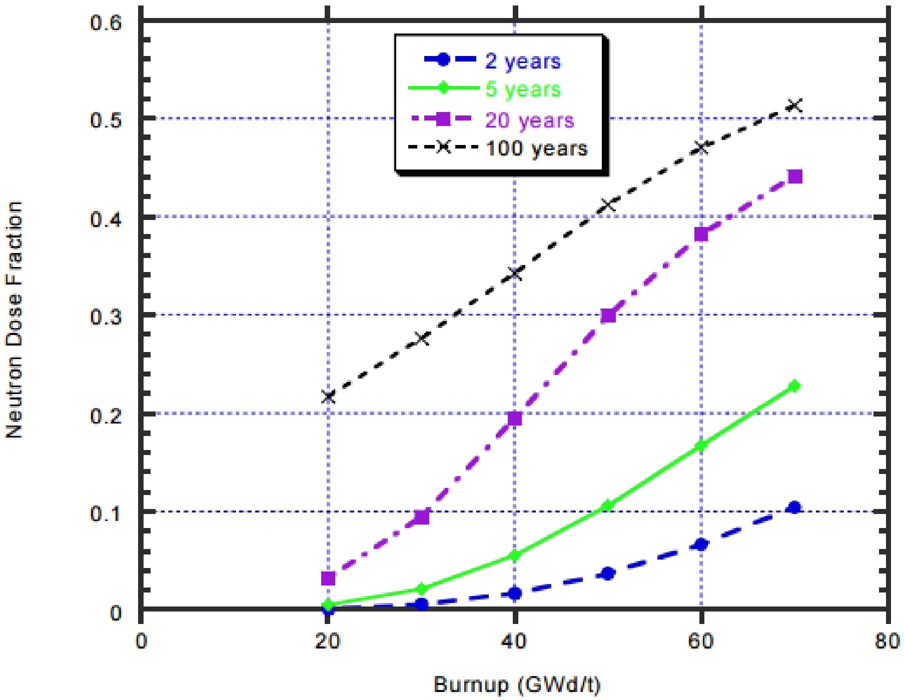
5 year cooling



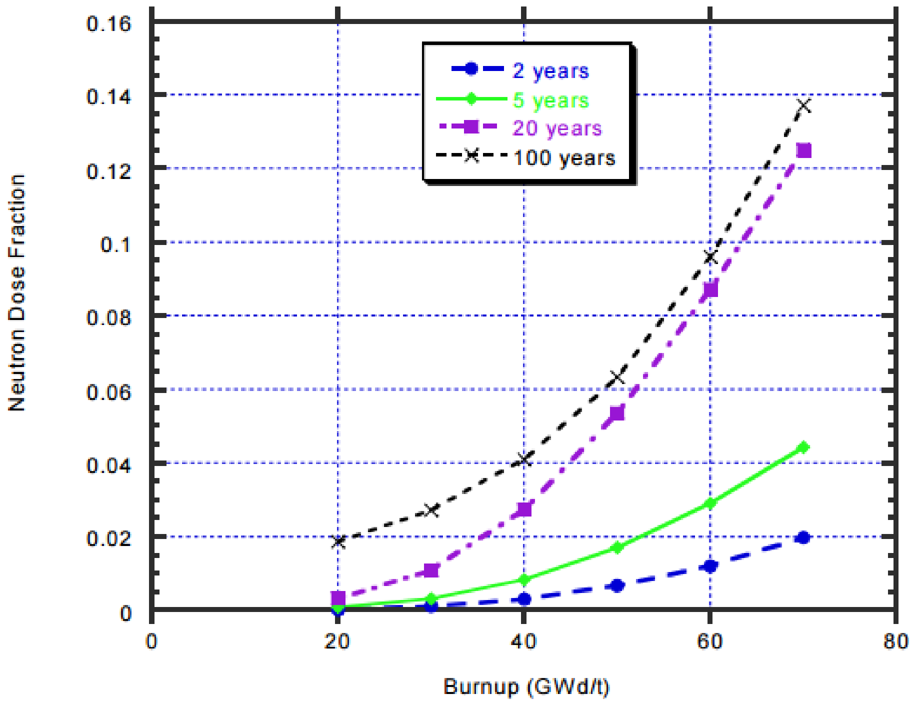
100 year cooling

I. C. Gauld and J. C. Ryman, “Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel,” NUREG/CR–6700, 2000.

Relative contribution of neutrons to the total dose rate (5-wt % PWR fuel)



Steel transport cask



concrete storage cask

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

Shielding rankings for dominant radionuclides in PWR fuel

	Steel cask				Concrete cask				Lead cask			
	20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %	
Nuclides[a]	Percent[b]	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Ba-137m	3.0	6	2.3	7	9.2	4	8.9	4	0.2	8	0.1	8
Cm-244	1.8	7	36.1	1	0.3	8	8.3	5	1.9	6	36.6	1
Co-60	49.8	1	26.4	2	50.3	1	34.6	1	56.4	1	29.9	2
Cs-134	10.7	3	15.8	3	14.0	2	26.7	2	10.2	3	15.0	3
Eu-154	4.6	5	7.0	4	5.5	6	10.9	3	4.5	5	6.9	4
Pr-144	19.6	2	5.0	5	12.4	3	4.2	7	17.3	2	4.4	6
Rh-106	9.0	4	5.0	6	6.9	5	5.0	6	8.2	4	4.5	5
Y-90	1.0	8	0.6	8	1.1	7	0.9	8	1.0	7	0.6	7
100-Year cooling												
Am-241	18.6	2	4.3	4	2.2	3	1.3	5	2.9	1	5.4	3
Ba-137m	42.0	1	12.4	3	85.6	1	66.9	1	3.6	6	0.7	8
Cm-244	6.0	5	46.6	1	0.7	5	14.8	2	9.5	4	52.3	1
Cm-246	0.3	9	23.7	2	0.0	9	7.5	3	0.5	9	26.6	2
Pu-238	3.2	6	4.2	5	0.4	6	1.3	6	5.6	5	5.2	4
Pu-239	1.8	8	0.2	9	0.2	8	0.1	9	3.2	8	0.3	9
Pu-240	13.3	3	2.9	7	1.6	4	0.9	7	21.5	2	3.3	6
Pu-242	2.2	7	1.7	8	0.3	7	0.5	8	3.5	7	1.9	7
Y-90	11.9	4	3.0	6	8.7	2	5.8	4	18.9	3	3.4	5

[a] Dominant nuclides that contribute more than 1% to the total dose rate.

[b] Indicates percentage contribution of the nuclide to the total dose rate for the cask configuration.

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

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Shielding rankings for dominant radionuclides in BWR fuel

	Steel cask				Concrete cask				Lead cask			
	20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %	
Nuclides[a]	Percent[b]	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Ba-137m	4.5	6	2.7	7	13.9	4	11.6	4	0.3	8	0.2	9
Cf-252	0.0	9	1.3	8	0.0	9	0.3	9	0.0	9	1.4	7
Cm-244	3.0	7	43.9	1	0.5	8	10.9	5	3.3	6	45.8	1
Co-60	24.5	2	10.7	3	24.9	1	15.1	2	29.7	1	12.5	3
Cs-134	16.2	3	19.1	2	21.3	2	34.9	1	16.5	3	18.7	2
Eu-154	7.0	5	8.1	4	8.4	6	13.5	3	7.4	5	8.2	4
Pr-144	29.3	1	6.1	5	18.7	3	5.4	7	27.6	2	5.5	6
Rh-106	13.4	4	6.0	6	10.3	5	6.4	6	13.0	4	5.6	5
Y-90	1.5	8	0.8	9	1.6	7	1.2	8	1.6	7	0.8	8
100-Year cooling												
Am-241	17.8	2	3.8	5	2.1	3	1.2	6	31.5	1	4.8	4
Ba-137m	42.0	1	12.2	3	85.6	1	66.7	1	3.6	6	0.7	8
Cm-244	6.6	5	46.2	1	0.8	5	14.8	2	10.4	4	51.7	1
Cm-246	0.4	9	25.0	2	0.0	9	8.1	3	0.6	9	28.0	2
Pu-238	3.4	6	4.0	4	0.4	6	1.3	5	5.8	5	4.9	3
Pu-239	1.8	8	0.2	9	0.2	8	0.1	9	3.2	8	0.2	9
Pu-240	13.4	3	3.0	6	1.6	4	0.9	7	21.6	2	3.4	5
Pu-242	2.2	7	1.6	8	0.3	7	0.5	8	3.4	7	1.8	7
Y-90	11.9	4	3.0	7	8.7	2	5.8	4	19.0	3	3.3	6

[a] Dominant nuclides that contribute more than 1% to the total dose rate.

[b] Indicates percentage contribution of the nuclide to the total dose rate for the cask configuration.

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	0.63	7	1.36	6	0.82	7	1.40	6	0.95	7	1.43	6
Am-243	0.03	12	0.50	10	0.08	12	0.56	10	0.13	12	0.61	11
Np-237	0.26	8	0.73	9	0.42	8	0.91	9	0.59	8	1.08	9
Pu-238	0.05	11	0.41	11	0.10	11	0.52	11	0.17	10	0.63	10
Pu-239	25.23	2	27.66	2	24.71	2	26.66	2	24.14	2	25.82	2
Pu-240	6.48	4	9.17	3	6.87	4	8.97	3	7.04	4	8.80	3
Pu-241	3.06	5	6.51	4	3.70	5	6.39	4	4.08	5	6.30	5
Pu-242	0.18	9	1.05	7	0.32	9	1.09	8	0.44	9	1.13	8
U-234	0.14	10	0.08	12	0.15	10	0.10	12	0.16	11	0.12	12
U-235	22.88	3	5.22	5	21.69	3	6.31	5	20.97	3	7.15	4
U-236	0.75	6	0.94	8	1.01	6	1.17	7	1.23	6	1.37	7
U-238	30.88	1	30.35	1	29.27	1	29.09	1	28.00	1	28.00	1
Total	90.59		84.12		89.17		83.34		87.94		82.62	
100-Year cooling												
Am-241	2.43	5	5.11	5	3.09	5	5.16	5	3.53	5	5.21	5
Am-243	0.03	10	0.51	9	0.08	10	0.57	9	0.13	10	0.62	9
Np-237	0.38	7	0.99	7	0.58	7	1.18	7	0.77	7	1.36	7
Pu-238	0.02	11	0.20	11	0.05	11	0.25	11	0.08	11	0.30	11
Pu-239	25.50	2	28.31	2	25.01	2	27.24	2	24.45	2	26.35	2
Pu-240	6.51	4	9.74	3	6.94	4	9.60	3	7.14	4	9.49	3
Pu-242	0.19	8	1.08	6	0.32	8	1.12	8	0.44	8	1.15	8
U-234	0.16	9	0.22	10	0.19	9	0.28	10	0.23	9	0.34	10
U-235	23.21	3	5.38	4	22.04	3	6.49	4	21.32	3	7.34	4
U-236	0.77	6	0.97	8	1.03	6	1.20	6	1.25	6	1.41	6
U-238	31.29	1	31.13	1	29.71	1	29.79	1	28.44	1	28.63	1
Total	90.52		83.78		89.08		83.04		87.85		82.36	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

Criticality safety rankings of actinides in PWR fuel

I. C. Gauld and J. C. Ryman, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel," NUREG/CR–6700, 2000.

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Criticality safety rankings of actinides in BWR fuel

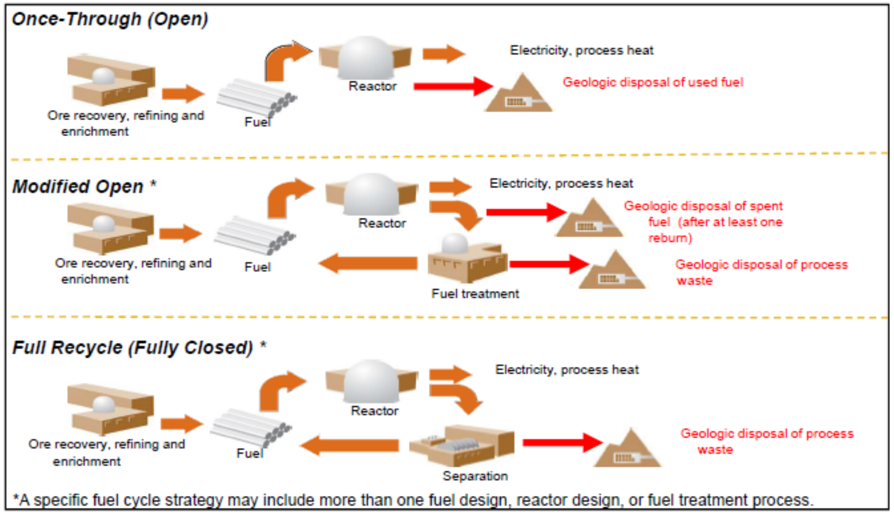
	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	0.61	7	1.31	6	0.79	7	1.34	6	0.91	7	1.36	6
Am-243	0.04	12	0.51	10	0.08	12	0.56	10	0.13	12	0.61	11
Np-23	0.27	8	0.75	9	0.43	8	0.92	9	0.59	8	1.09	9
Pu-238	0.06	11	0.45	11	0.11	11	0.56	11	0.18	10	0.67	10
Pu-239	4.69	2	27.21	2	24.14	2	26.21	2	23.56	2	25.36	2
Pu-240	6.25	4	8.95	3	6.57	4	8.75	3	6.72	4	8.59	3
Pu-241	2.99	5	6.41	4	3.59	5	6.30	4	3.95	5	6.21	5
Pu-242	0.19	9	1.04	7	0.32	9	1.07	8	0.43	9	1.10	8
U-234	0.14	10	0.09	12	0.15	10	0.10	12	0.16	11	0.12	12
U-235	3.42	3	5.11	5	22.35	3	6.15	5	21.65	3	6.95	4
U-236	0.76	6	0.93	8	1.00	6	1.15	7	1.20	6	1.34	7
U-238	30.05	1	29.74	1	28.39	1	28.48	1	27.12	1	27.38	1
Total	89.47		82.63		87.94		81.77		86.64		80.99	
100-Year cooling												
Am-241	2.38	5	4.94	5	2.98	5	4.98	5	3.38	5	5.01	5
Am-243	0.04	10	0.52	9	0.08	10	0.57	9	0.13	10	0.62	9
Np-237	0.39	7	1.00	7	0.58	7	1.18	7	0.77	7	1.35	7
Pu-238	0.03	11	0.22	11	0.05	11	0.27	11	0.09	11	0.33	11
Pu-239	24.94	2	27.87	2	24.43	2	26.81	2	23.86	2	25.92	2
Pu-240	6.28	4	9.52	3	6.64	4	9.38	3	6.82	4	9.27	3
Pu-242	0.19	8	1.06	6	0.33	8	1.10	8	0.44	8	1.13	8
U-234	0.16	9	0.22	10	0.19	9	0.28	10	0.23	9	0.34	10
U-235	23.75	3	5.27	4	22.70	3	6.33	4	22.01	3	7.14	4
U-236	0.77	6	0.97	8	1.02	6	1.19	6	1.23	6	1.38	6
U-238	30.44	1	30.52	1	28.81	1	29.20	1	27.54	1	28.04	1
Total	89.40		82.25		87.85		81.43		86.55		80.69	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

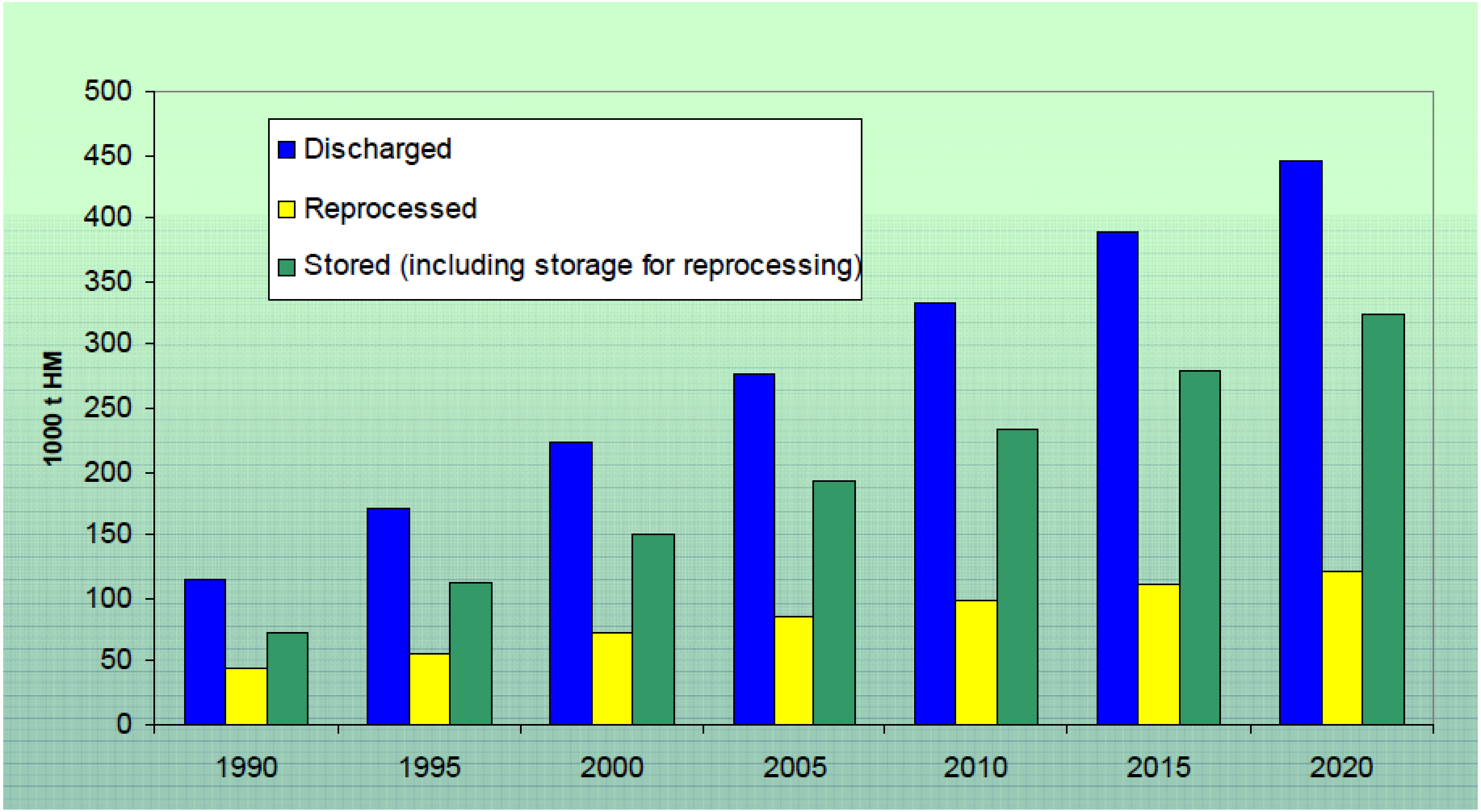
I. C. Gauld and J. C. Ryman, “Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel,” NUREG/CR–6700, 2000.

Fuel Cycle Concepts and Impacts

- Once-through: Open pass through reactor, used fuel directly disposed in a geologic repository
  - Low uranium utilization - Appropriate for a low price uranium future
  - Appropriate when repository capacity is not limited
- Modified Open: No or limited separation steps and processing applied to used fuel to extract more energy
  - Higher uranium utilization and burnup - Appropriate for a high price uranium future
  - Appropriate when repository capacity is a major constraint
- Closed (Full Recycle): Only elements considered to be waste are discarded and useful elements are recycled to more fully utilize resources
  - Multiple reprocessing and recycle steps resulting in transmutation of most actinides

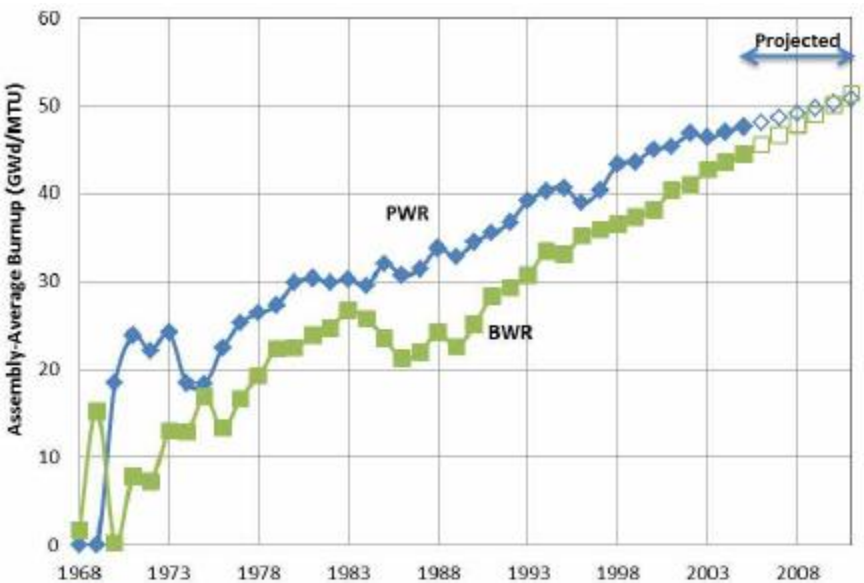


Global Spent Fuel Generation (IAEA)

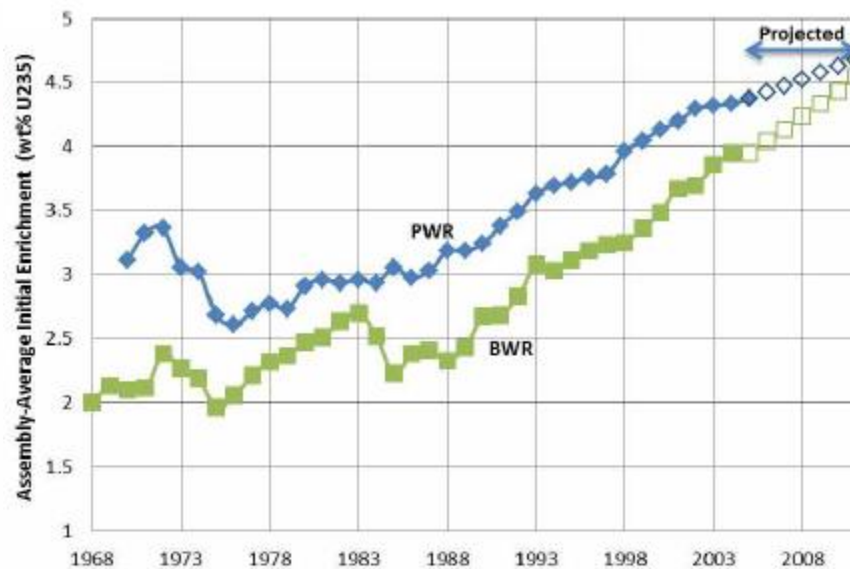


Spent Fuel Characteristics

Less diversity expected in the future

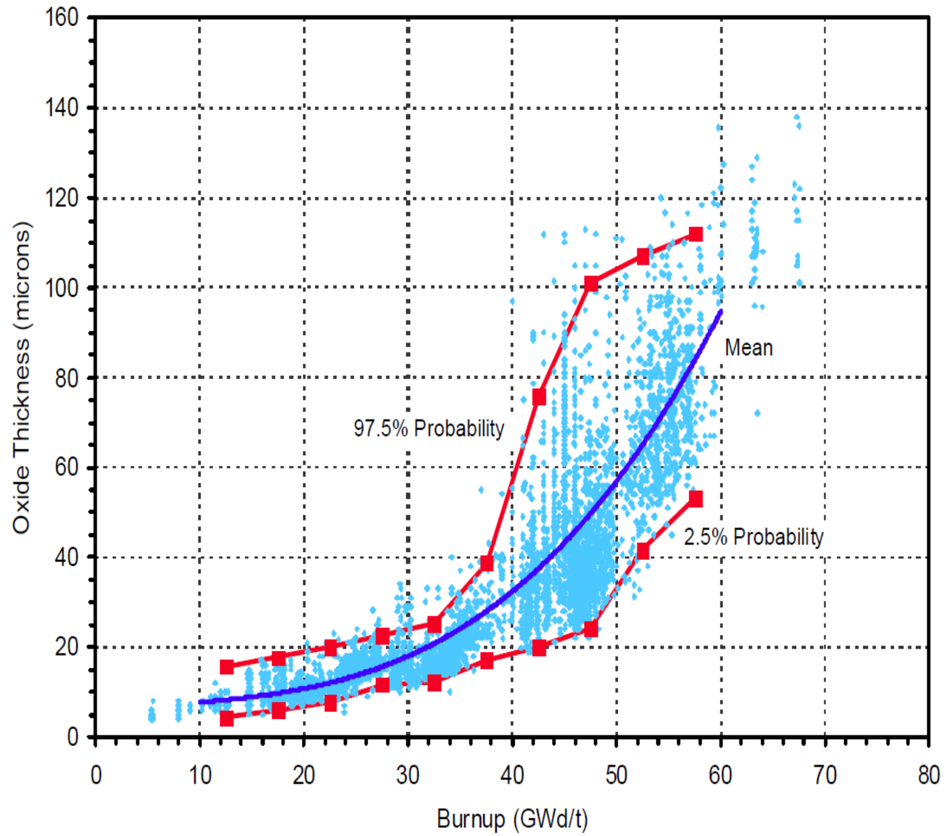


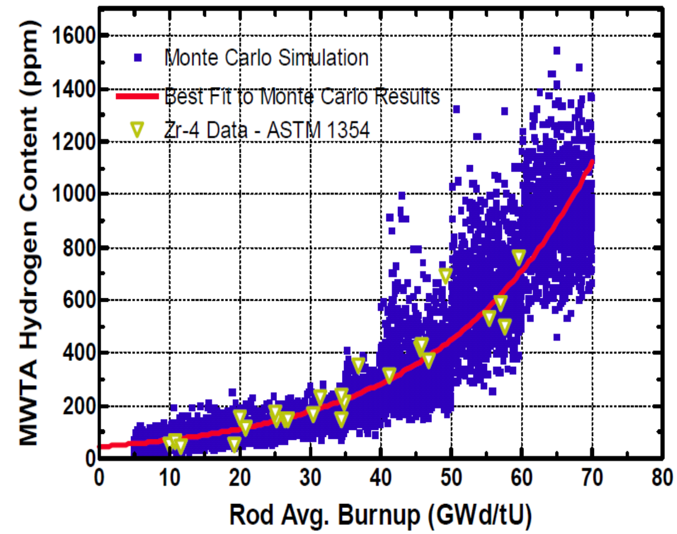
Average discharge burnup



Average initial enrichment

Spent Fuel Burnup and Integrity



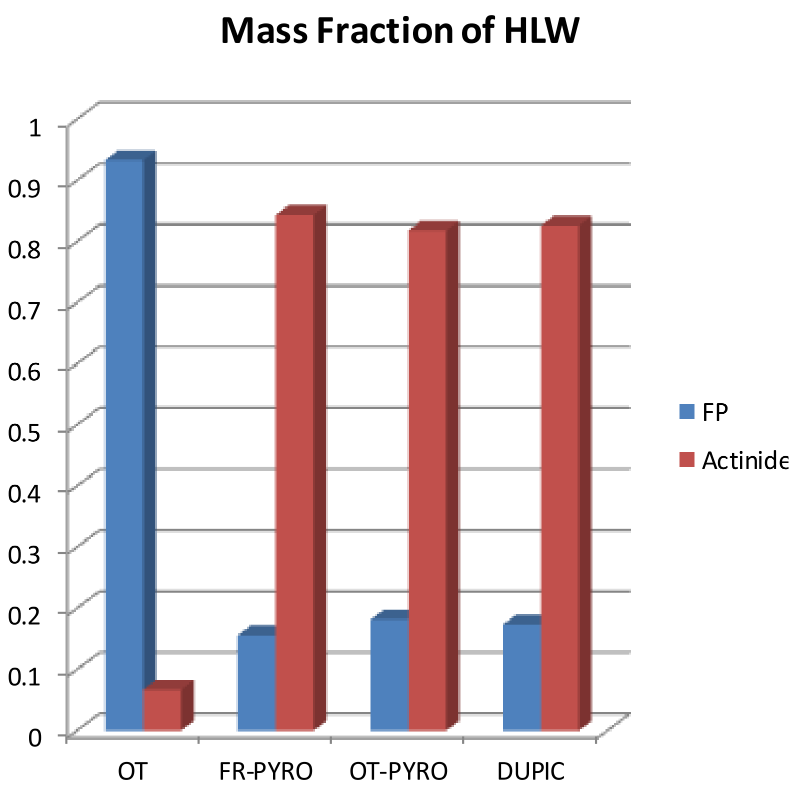


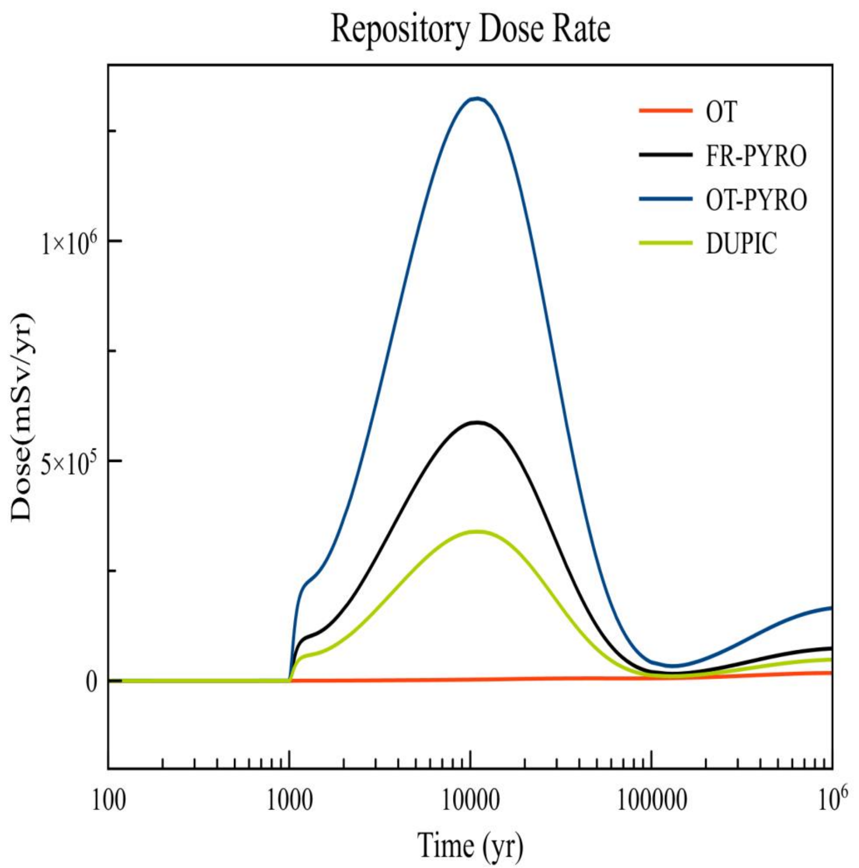
** MWTA(Maximum Wall Thickness Average)*

US Nuclear Waste Tech. Review Board, "Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel (2010)

- BU increase -> Oxide thickness increase -> Hydrogen content increase
  ➡ Reduced heat removal capability, Hydride reorientation effect

Comparisons of Fuel Cycle Options



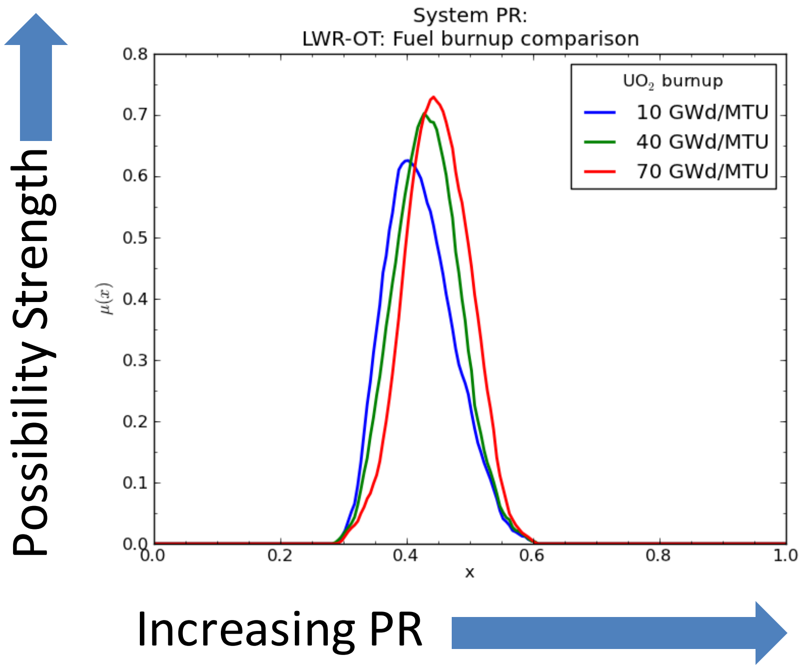


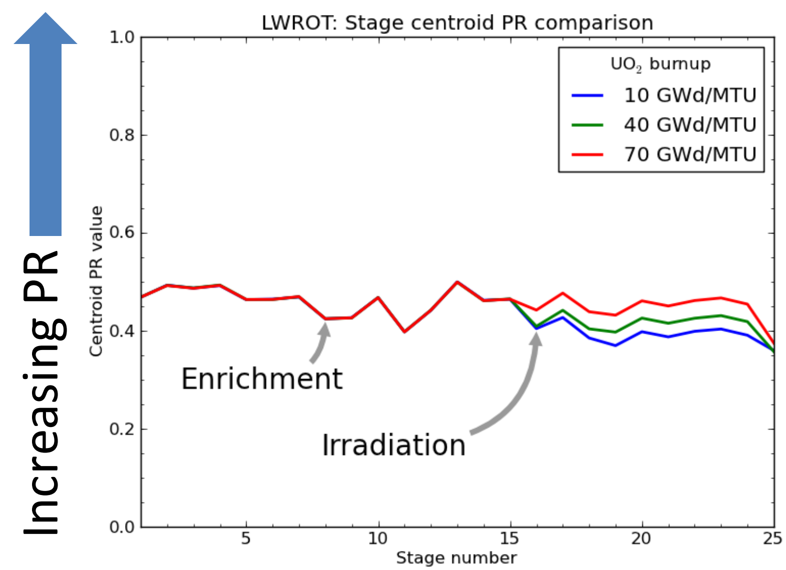
Inventory of HLW in Geologic Repository

Comparisons of Projected Repository Dose

Impact on proliferation resistance

LWR-OT: UO_2 burnup comparison

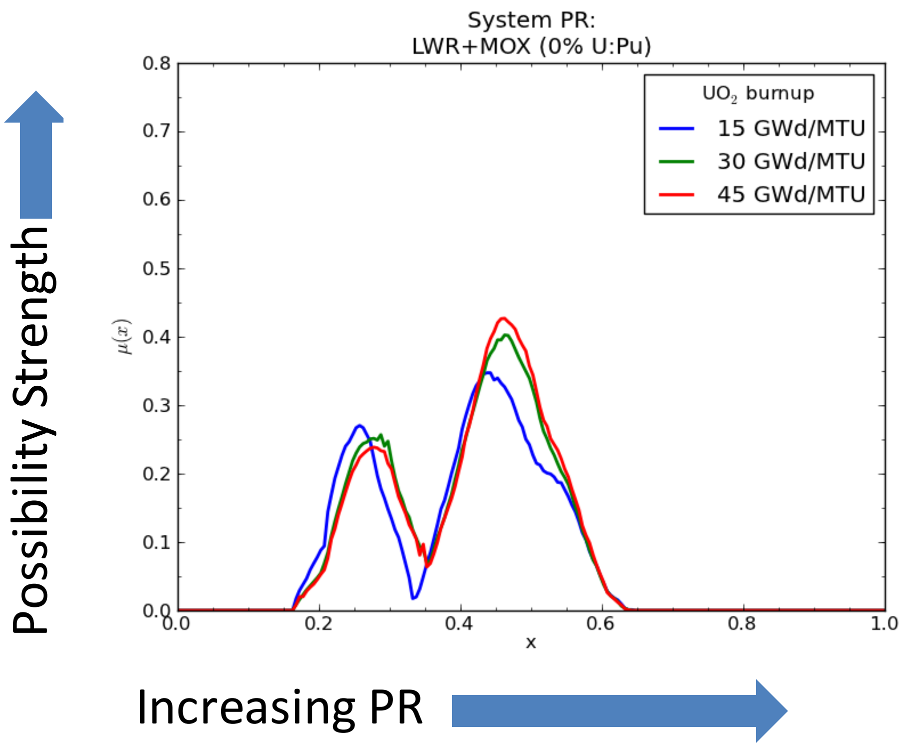


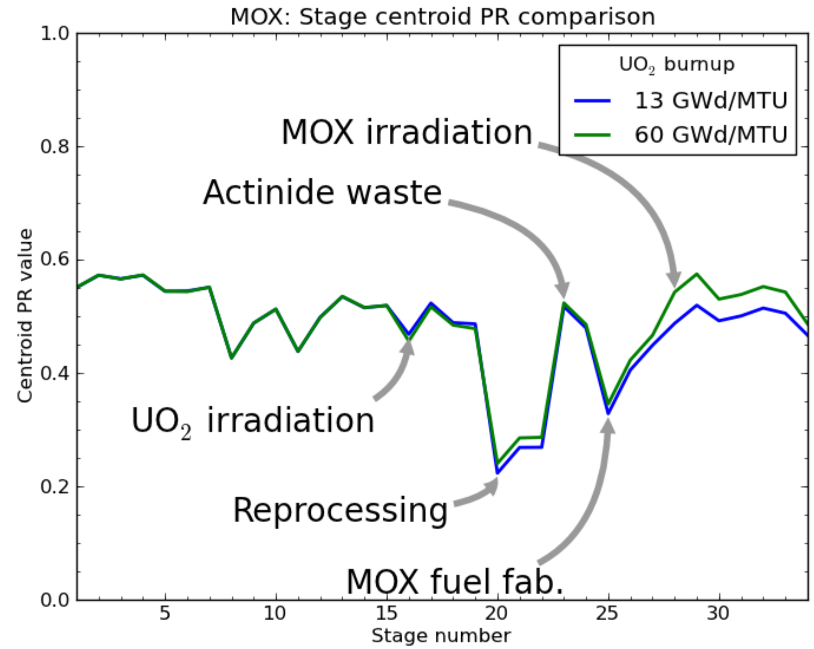


S. E. Skutnik and M.-S. Yim, "Assessment of Fuel Cycle Proliferation Resistance Dynamics Using Coupled Isotopic Characterization," Nuclear Engineering and Design, Vol. 241, no. 8, pp. 3270–3282, 2011

Impact on proliferation resistance

MOX: UO_2 burnup comparison





S. E. Skutnik and M.-S. Yim, "Assessment of Fuel Cycle Proliferation Resistance Dynamics Using Coupled Isotopic Characterization," Nuclear Engineering and Design, Vol. 241, no. 8, pp. 3270–3282, 2011

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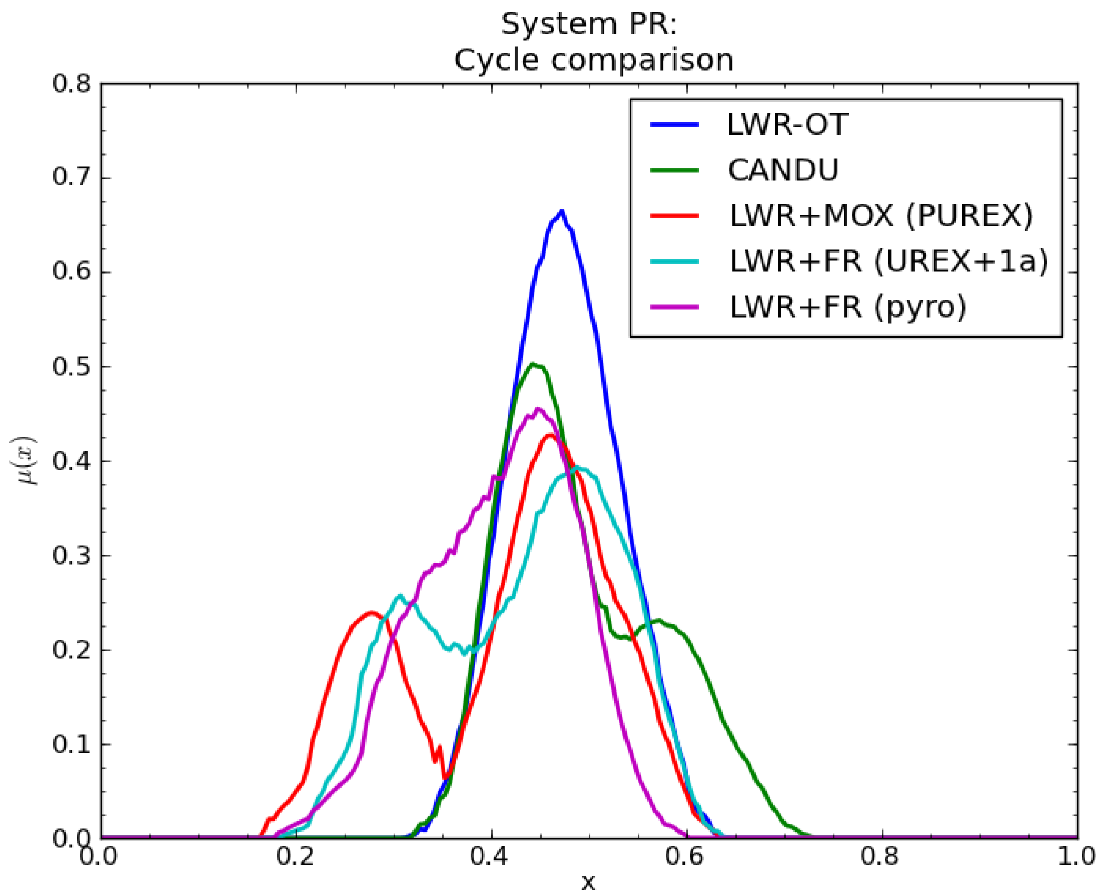
Fuel Cycle System Comparison

Possibility Strength



Increasing PR

Performance of Spent Fuel Management

(Comparison of Fuel Cycles)

	OT	FR-PYRO	OT-PYRO	OT-ER
HLW generation rate (kgHM/GWh)	1.824	0.112	0.142	1.814
U utilization efficiency (%)	0.84	1.54	0.84	0.84%
Natural uranium required (tU/TWh)	17.44	10.80	17.44	17.44
Maximum loading of HLW per repository (MTU)	35,361	42,433	46,035	58,499
Total energy produced per fully loaded HLW repository (GWh)	1.94E+7	3.80E+8	3.24E+8	3.22E+7
Ratio of total energy produced per one fully loaded HLW repository to the OT cycle case	1.0	19.6	16.7	1.67
Ratio of total cumulative dose to humans per fully loaded HLW repository to the OT cycle case	1.0	29.8	29.5	0.23
Proliferation resistance	0.537 (H)	0.523 (H)	0.501 (H)	0.503 (H)
Fuel cycle cost ($/MWh)	7.81	7.83	9.46	10.55
Total electricity generation cost ($/MWh)	34.9	41.7	36.6	37.7

M.S. Yim, An Analysis for Policy Development for Republic of Korea's National Spent Fuel Management Systems, KAIST, 2013

Summary

– Fuel design and reactor operation affect spent fuel management through their impact on the production of nuclide inventory, characteristics of the nuclides, and material integrity.
– With the select design features of nuclear fuels, the effect of fuel design on spent fuel management is relatively well characterized.
– The effect of reactor operation seems to have large variations in nuclide inventory and material integrity.
  - The final nuclide inventory is most sensitive to late-in-power variations.
  - Low-power operation near the end of cycle leads to decreased fission product inventory.
  - Long downtime between the cycles or during the cycle appears to have positive effect on spent fuel management
  - Recycling of spent fuel is shown to reduce the burden of final disposal of nuclear waste.
  - Burnup effect is dominant in controlling nuclide inventory in soent fuel.
– In general, achieving better uranium utilization efficiency appears to increase the burden in spent fuel management through increased production of the nuclides of concern.
– Overall performance of spent fuel management needs to be evaluated from a life cycle-based systems approach.