Objectives (Output) of Reactor Analysis CALCS

1. Fuel Management Info

   Cycle Length
   Batch Burnups
   Batch Isotopics

2. Operating Info

   Excess Reactivity vs Burnup
   Data to Perform Startup Tests
   Target Power Shapes (BWR)
   Control Rod Patterns
   Process Computer Input
   Plant Data Book

3. Licensing Data

   Expected and Limiting Power Dists.
   Shutdown Margin
   Reactivity Coeffs
   Ejected and Dropped Rod Effects
   Delayed Neutron Parameters
   Transient and Accident Evaluations
   CHF and LOCA Limits

Reactor Physics Parameters

- Power Distribution
- Reactivities

Reaction Rates
(or combinations)

Power Distribution

→ Fission Reaction Rate
→ $\Sigma_f \phi$

Reactivity

$$\frac{\text{Production} - \text{Destruction}}{\text{Production}}$$

$$\frac{\Sigma_f \phi - \left(\Sigma_a \phi + DB^2\right)}{\Sigma_f \phi}$$

Reaction Rates

$= N \sigma \phi$

$N \equiv$ Nuclide Concentration
$= N (x , B)$

$\sigma \equiv$ Cross Section
$= \sigma(x , E, B)$

$\phi \equiv$ Flux
$= \phi$ (x E, B, operating variables)

x → Position
E → Energy
B → Burnup of Fuel

N

-Initially, From Fuel Design

-Balance Equation:

$$\frac{dN}{dt} = -N\sigma\phi$$

$$\frac{dN(x,t)}{dt} = N\sigma\phi - \lambda N + \sum_i N^i \sigma^i \phi + \sum_j \lambda_j N_j$$

$\underline{\phi}$

Neutron Balance Equation

- Choice of Diffusion theory, transport theory, intermediates
- Choice of zero dimensions, one dimension, 2 – D, 3 – D or combinations
- Choice of spatial detail

PDQ – Diffusion Theory (1 – 3D)
Simulate – nodal (1 – 3D)
CASMO and other lattice codes (2 – D)

$\underline{\sigma}$

Divide Continuous Energy into a Few Energy <u>Groups</u>!

σ (x E, B) → σ_i (N operating variables)

$$\sigma_i(\text{N,Op Var}) = \frac{\int_{E_i} \sigma(E)\rho(N, E, \text{OpVr})}{\int_{E_i} \rho(N, E, \text{OpVar})}$$

<u>Unit Cell (Lattice Physics) Code</u>

Leopard – Unit Fuel (Rod) Cells
Casmo – Unit Fuel Assemblies
Color set (4 assemblies)

Unit Cell Codes

1. Since ϕ appears in top and bottom, only E dependance (spectrum), not absolute magnitude, matters

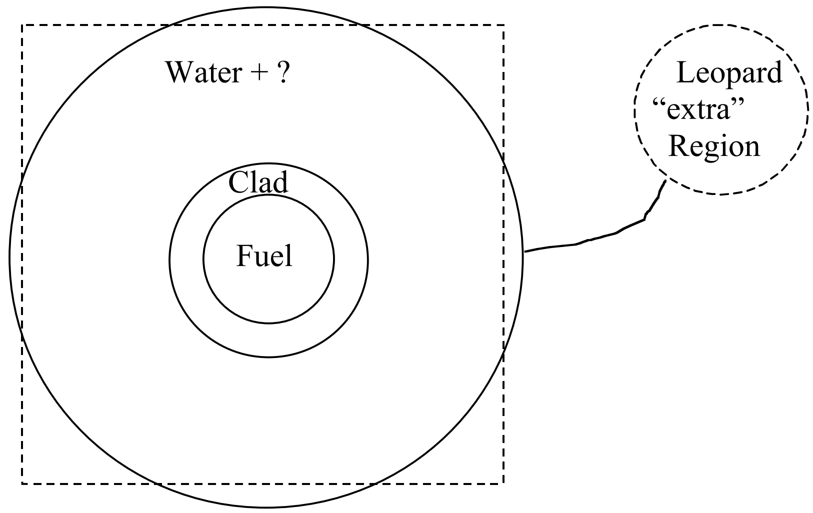
2. Therefore – a single unit cell calculation can represent many places in core (whew!)

3. Most include depletion ($\frac{dN}{dt}$)

4. Most come with multigroup of σ libe

5. Since approximations are used, code and libe are matched set (but not so much today)

Unit Cell Geometry



Unit Assy

Fuel Cells
Water Slots
Control Rods
Instrumentation Holes
Can
Poison PIN Cells

Effects to consider in Unit Cell Code

1. Energy
    - Fission Spectrum
    - Fast Fission
    - Slowing Down
    - Leakage During Slowing Down
    - Resonance Capture
    - Thermal Flux Spectrum
    - Self – Shielding

2. Space
    - Unit Pin Cells (= Lattice)
    - Non – Lattice Region
    - Effect on Resonance Capture
    - Fast Advantage Factor
    - Thermal Disadvantage Factor

Effects to Consider in Unit Cell Code Leopard

1. ENERGY 0 – 10 MEV 172 thermal (0 – 0.625 ev)
   Fast (0.625 ev – 10 Mev)
   - Fission Spectrum – U-235
   - Fast Fission Homogeneous
   - Slowing down B-1
   - Leakage during slowing down – buckling
   - Resonance capture – norm to hellstrand
   - Thermal flux spectrum – wigner wilkins
   - Self shielding – thermal → ABH by group
     Res → implicit in hellstrand
     Fast → none

2. SPACE
   - Unit Pin Cells (=Lattice)
   - Non – Lattice Region – Homogeneous
   - Effect on Resonance Capture - $\sqrt{\frac{S}{M}}$
   - Fast Advantage Factor = 1.0
   - Thermal Disadvantage Factor – ABH

Effects to Consider in Unit Cell Code CASMO

1. ENERGY 0 – 10 MEV 40 groups
   70 groups
   - Fission Spectrum U-253
     Pu-239
   - Fast Fission – Transp. Th.
   - Slowing Down – Space-Energy Together
   - Leakage During Slowing Down – Buckling
   - Resonance Capture- interp. On T and σ_p
   - Thermal Flux Spectrum – Numerical
   - Self – Shielding – Transport Theory

2. SPACE
   - Unit Pin Cells (= Lattice)
   - Non – Lattice Reion – Explicit assy
   - Effect on resonance Capture – Equiv. Th
   - Fast Advantage Factor – Transp. Th
   - Thermal Disadvantage Factor – Trans. Th.

Input to Unit Cell Codes

- Geometry
- Materials
- Temperatures
- Depletion Parameters
  - Power Level
  - Mechanism to obtain criticality
    (usually buckling or poison)

Output from Unit Cell Codes

Macroscopic few group cross sects.
Microscopic few group cross sects.
Nuclide number densities
Fluxes
Reaction rates
Neutron balance
K_∞
K_{EFF}

Cross – Sections

Macros

- Good only for conditions calculated (unless corrected)
  - Soluble born
  - Temperatures
- Can be used for Depletion

Micros

- Good for wide variety of conditions
- Can be used for depletion
- Can be used to estimate reactivity of absent capture nuclides

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LRM Proof of Efficacy; graph: Enrichment vs Core AVE EOFPL Burnup.

Please see pp. 4.3-83, 4.3-67, and 4.3-102 in
"Reference Safety Analysis Report -- RESAR-3."
Westinghouse Nuclear Energy Systems, April 1973.

Also see 15.2-1, 15.1-39 and 15.1-42 in "Amendment 4" of the above document, May 1974.

CONTROLLING CHARACTERISTICS IN POSTULATED INCIDENTS

POSTULATED INCIDENT	CONTROLLING CORE CHARACTERISTICS
Rod withdrawal, misoperation	Control rod worth
	MTC
	Doppler
Boron Dilution	Boron Worth
Incidents causing a change in ıt temperature	MTC
Steam line break	Rod worth
	MTC
Rod ejection	Rod worth
	Delayed neutron fraction
	Prompt neutron lifetime
	Doppler

MTC = Moderator Temperature Coefficient

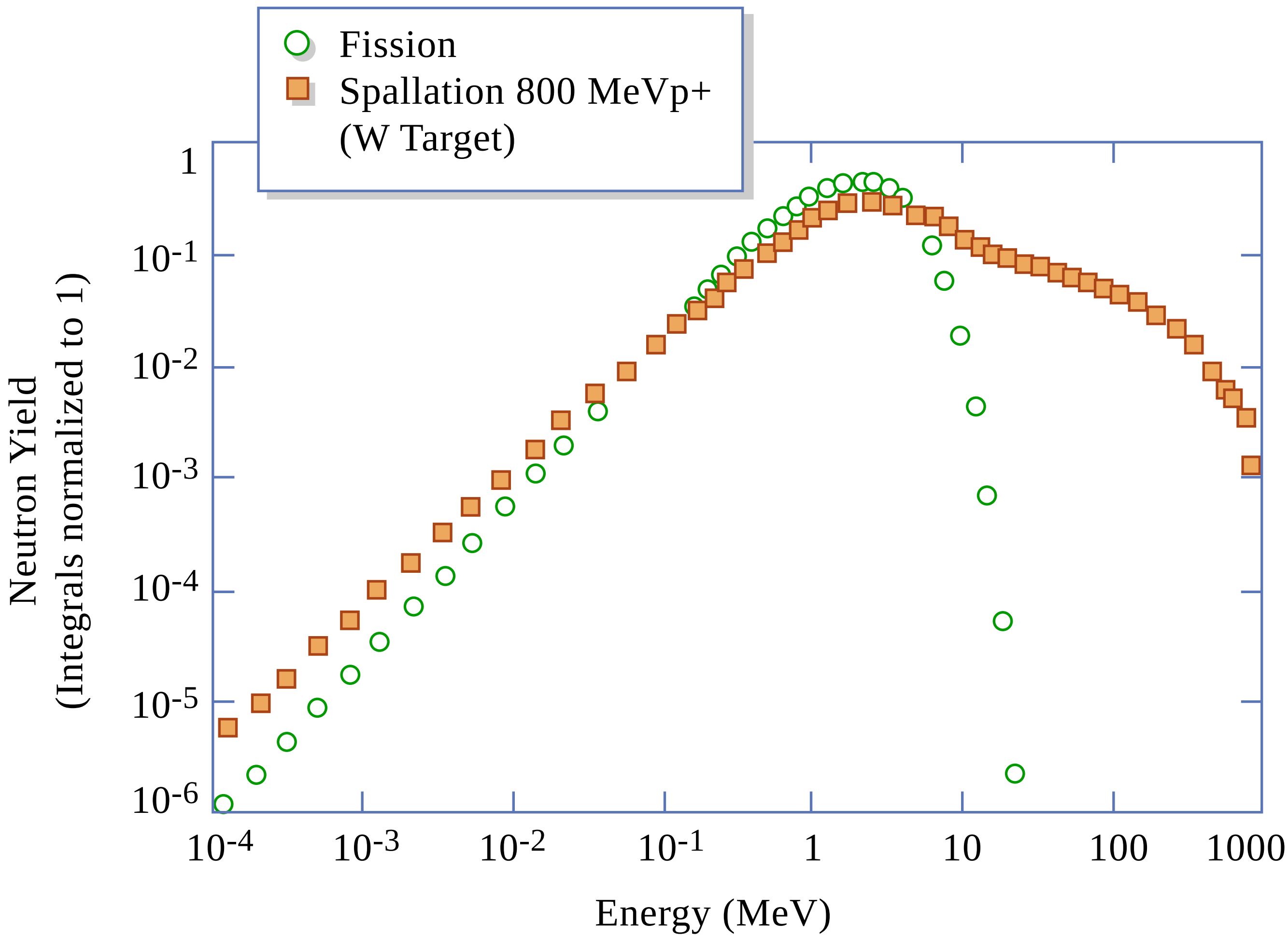
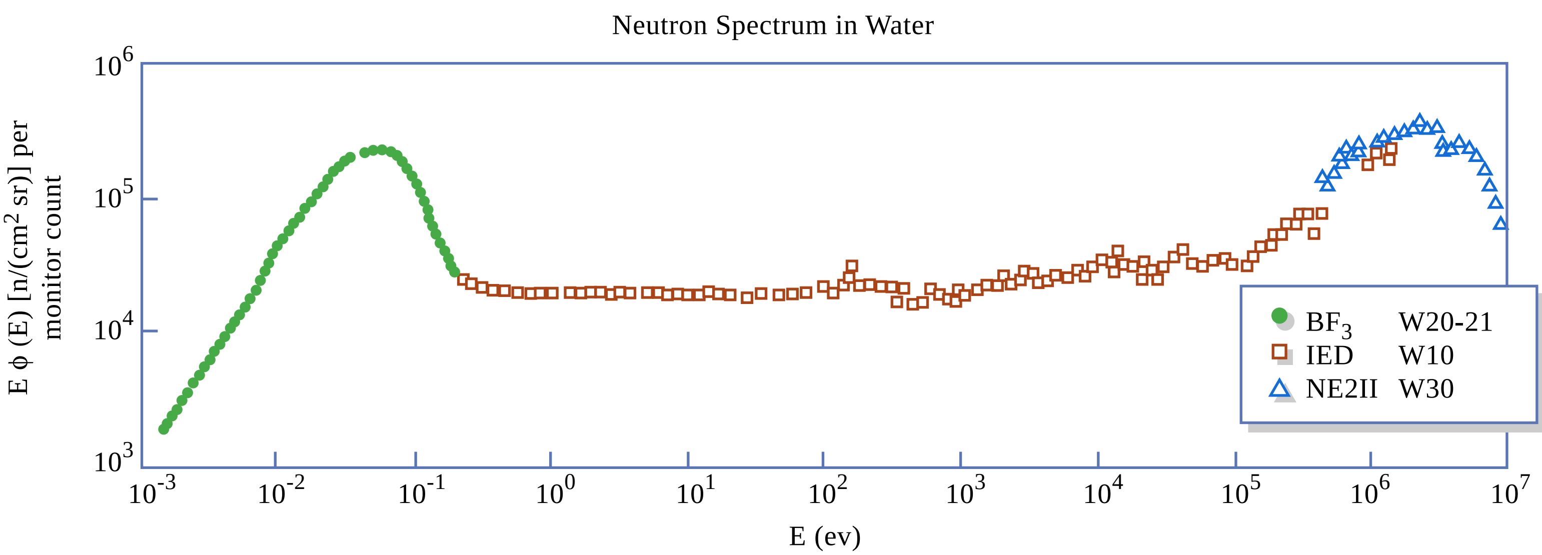
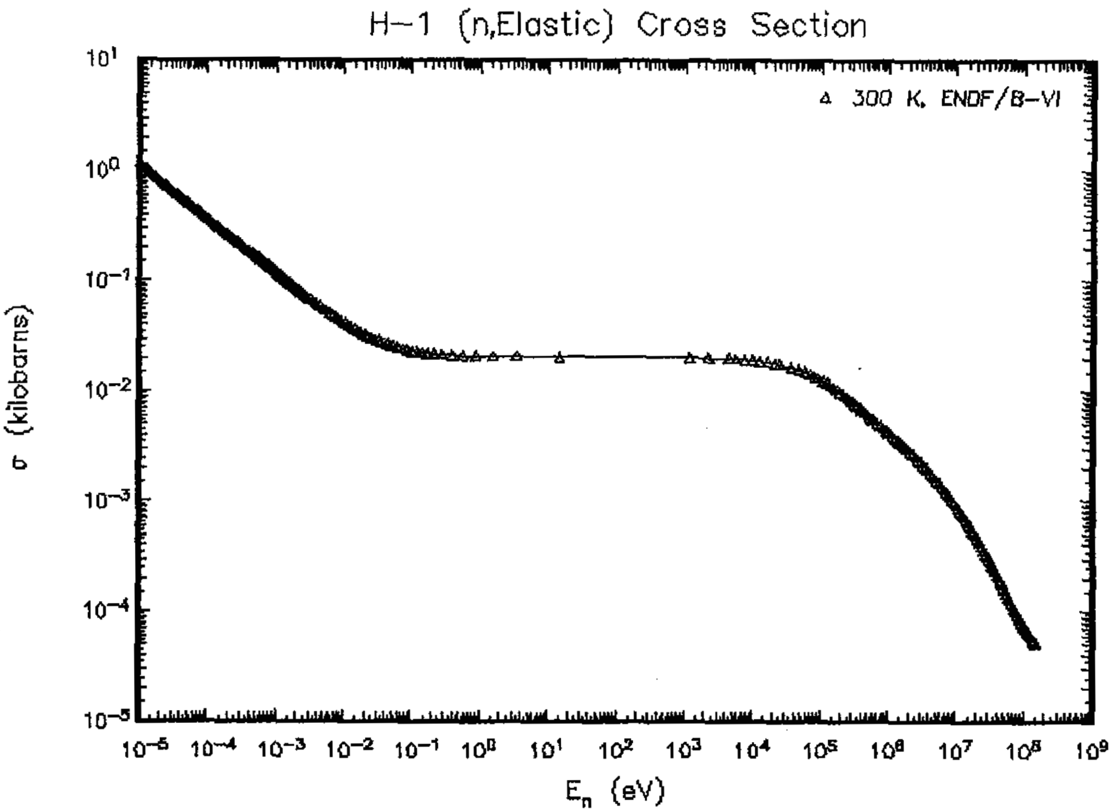


Figure by MIT OCW.

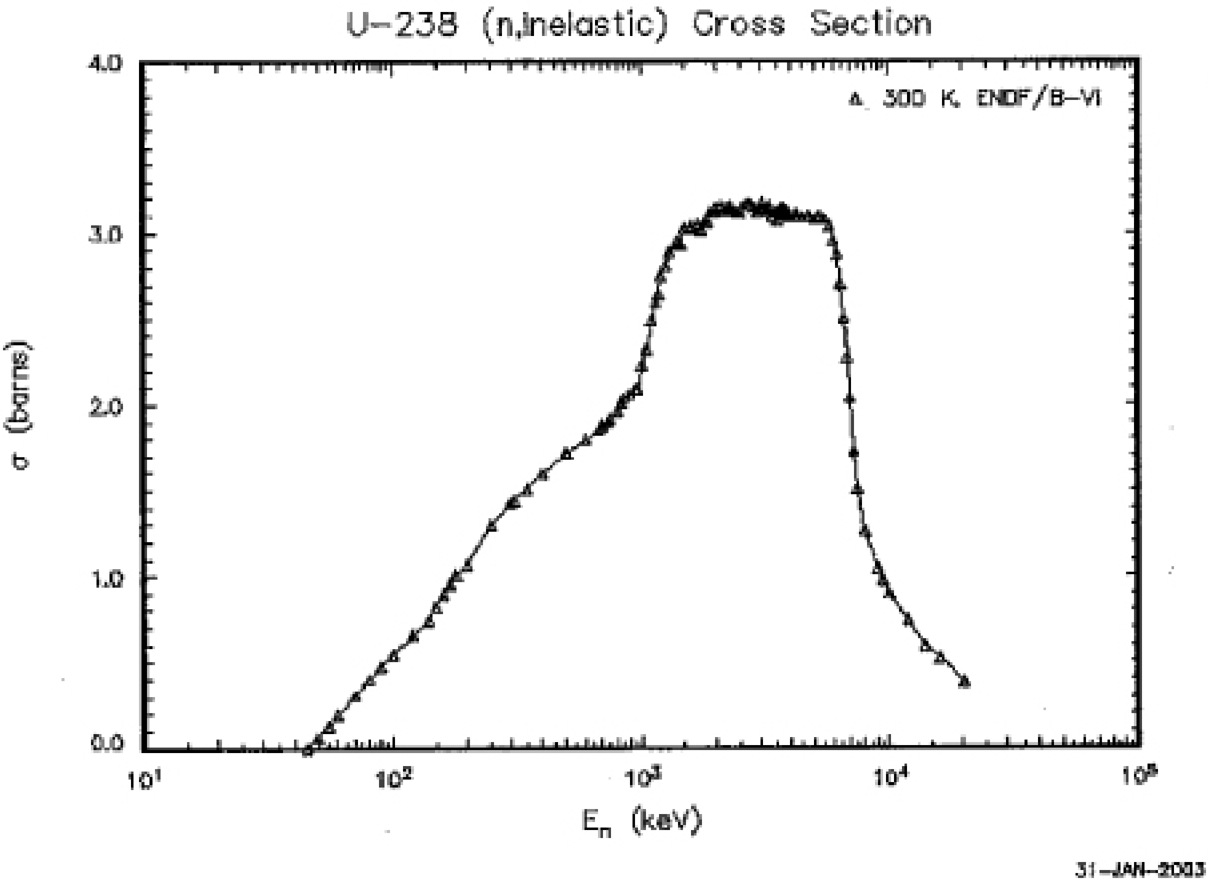


Complete Measured Spectrum at 0 deg, 9.35-cm Radius

Figure by MIT OCW.


H−1 (n,Elastic) Cross Section
△ 300 K, ENDF/B−VI
σ (kilobarns)
E_n (eV)
10^1
10^0
10^−1
10^−2
10^−3
10^−4
10^−5
10^−5 10^−4 10^−3 10^−2 10^−1 10^0 10^1 10^2 10^3 10^4 10^5 10^6 10^7 10^8 10^9

31−JAN−2003



U-238 (n,inelastic) Cross Section
△ 300 K, ENDF/B-VI
σ (barns)
4.0
3.0
2.0
1.0
0.0
10^1
10^2
10^3
10^4
10^5
E_n (keV)
31-JAN-2003

Classic Naval Reactors
4 Group Structure

Classic Commercial
2 Group Structure

ENERGY GROUP STRUCTURES

MICRO	MACRO	EDIT-A	EDIT-B	2D	PDQ		UPPER (BOUNDS)	LOWER (BOUNDS)	WIDTH (BOUNDS)	
1- 1	1->	1	1	1	1		10.00000	6.06550	3.93450	MeV
2- 2	2					*FISSION SPECTRUM & FAST FISSION*	6.06550	3.67900	2.38650	
3- 3	3						3.67900	2.23100	1.44800	
4- 4	4						2.23100	1.35300	0.87800	
5- 5	5						1.35300	0.82100	0.53200	
6- 6	6->		2	2			821.00000	500.00000	321.00000	keV
7- 9	7					*SLOWING DOWN & A LITTLE FISSION PECTRUN*	500.00000	111.00000	389.00000	
10- 14	8						111.00000	9.11800	101.88200	
15- 15	9						9.11800	5.53000	3.58800	
16- 21	10->		3	3			5.53000	0.14873	5.38127	
22- 23	11						148.73000	48.05200	100.67799	eV
24- 24	12						48.05200	27.70000	20.35200	
25- 25	13						27.70000	15.96800	11.73200	
26- 26	14						15.96800	9.87700	6.09100	
27- 27	15						9.87700	4.00000	5.87700	
28- 28	16->			4			4.00000	3.30000	0.70000	
29- 29	17						3.30000	2.60000	0.70000	
30- 30	18						2.60000	2.10000	0.50000	
31- 31	19					*RESONANCE CAPTURE & EPITHERMAL FISSION*	2.10000	1.85500	0.24500	
32- 32	20						1.85500	1.50000	0.35500	
33- 33	21						1.50000	1.30000	0.20000	
34- 34	22						1.30000	1.15000	0.15000	
35- 36	23						1.15000	1.09700	0.05300	
37- 39	24						1.09700	1.02000	0.07700	
40- 41	25						1.02000	0.97200	0.04800	
42- 42	26						0.97200	0.95000	0.02200	
43- 44	27						0.95000	0.85000	0.10000	
45- 46	28						0.85000	0.62500	0.22500	
47- 49	29->	2	4	5	2		0.62500	0.35000	0.27500	
50- 52	30						0.35000	0.28000	0.07000	
53- 54	31->			6			0.28000	0.22000	0.06000	
55- 55	32					*THERMAL REGION INCLUDING UPSCATTERING*	0.22000	0.18000	0.04000	
56- 56	33						0.18000	0.14000	0.04000	
57- 57	34->			7			0.14000	0.10000	0.04000	
58- 58	35						0.10000	0.08000	0.02000	
59- 60	36						0.08000	0.05800	0.02200	
61- 62	37->			8			0.05800	0.04200	0.01600	
63- 64	38						0.04200	0.03000	0.01200	
65- 67	39						0.03000	0.01500	0.01500	
68- 70	40						0.01500	0.00000	0.01500	

Spontaneous Fission and (α,n) Neutron Yields of Selected Isotopes

Isotope	Total Half Life (yr)	Spontaneous SF Half-Life (yr)	Fission Neutrons per SF	SF Neutrons Yield (n/g-s)	(α,n) in α-decay Half-Life (yr)	Reaction Oxide Neutron Yield (n/g-s)
^{232}Th	1.41×10^{10}	$>1 \times 10^{21}$	2.14	$>6 \times 10^{-5}$	1.41×10^{10}	2.2×10^{-5}
^{232}U	71.7	8×10^{13}	1.71	1.3	71.7	1.49×10^{4}
^{233}U	1.59×10^{5}	1.2×10^{17}	1.76	8.6×10^{-4}	1.59×10^{5}	4.8
^{234}U	2.45×10^{5}	2.1×10^{16}	1.81	5.02×10^{-3}	2.45×10^{5}	3.0
^{235}U	7.04×10^{8}	3.5×10^{17}	1.86	2.99×10^{-4}	7.04×10^{6}	7.1×10^{-4}
^{236}U	2.34×10^{7}	1.95×10^{16}	1.91	5.49×10^{-3}	2.34×10^{7}	2.4×10^{-2}
^{238}U	4.47×10^{9}	8.2×10^{15}	2.01	1.36×10^{-3}	4.47×10^{9}	8.3×10^{-5}
^{237}Np	2.14×10^{6}	1.0×10^{18}	2.05	1.14×10^{-4}	2.14×10^{6}	0.34
^{238}Pu	87.74	4.77×10^{10}	2.22	2.59×10^{3}	87.74	1.34×10^{4}
^{239}Pu	2.41×10^{4}	5.48×10^{15}	2.16	2.18×10^{-2}	2.41×10^{4}	38.1
^{240}Pu	6.56×10^{3}	1.16×10^{11}	2.16	1.02×10^{3}	6.56×10^{3}	1.41×10^{2}
^{241}Pu	14.35	(2.5×10^{15})	2.25	(4.94×10^{-2})	5.90×10^{5}	1.3
^{242}Pu	3.76×10^{5}	6.84×10^{10}	2.15	1.72×10^{3}	3.76×10^{5}	2.0
^{241}Am	433.6	1.05×10^{14}	2.27	1.18	433.6	2.69×10^{3}
^{242m}Am	152	9.5×10^{11}	2.34	1.35×10^{2}	152	33.1
^{243}Am	7.38×10^{3}	3.35×10^{13}	2.42	3.93	7.38×10^{3}	1.34×10^{2}
^{240}Cm	26.8 days	1.9×10^{6}	2.39	6.93×10^{2}	26.8 days	2.53×10^{7}
^{241}Cm	32.4 days	(1.6×10^{12})	(2.50)	(8.57×10^{1})	32.4 days	1.72×10^{5}
^{242}Cm	163 days	6.56×10^{6}	2.52	2.1×10^{7}	163 days	3.76×10^{6}
^{243}Cm	28.5	(1.2×10^{11})	(2.69)	(1.22×10^{3})	28.5	5.00×10^{4}
^{244}Cm	18.1	1.35×10^{7}	2.69	1.08×10^{7}	18.1	7.73×10^{4}
^{245}Cm	8.48×10^{3}	(4.0×10^{12})	(2.87)	(3.87×10^{1})	8.48×10^{3}	1.24×10^{2}
^{246}Cm	4.73×10^{3}	1.81×10^{7}	3.18	9.45×10^{6}	4.73×10^{3}	2.24×10^{2}
^{252}Cf	2.646	85.5	3.757	2.34×10^{12}	2.731	6.0×10^{5}

Figure by MIT OCW.

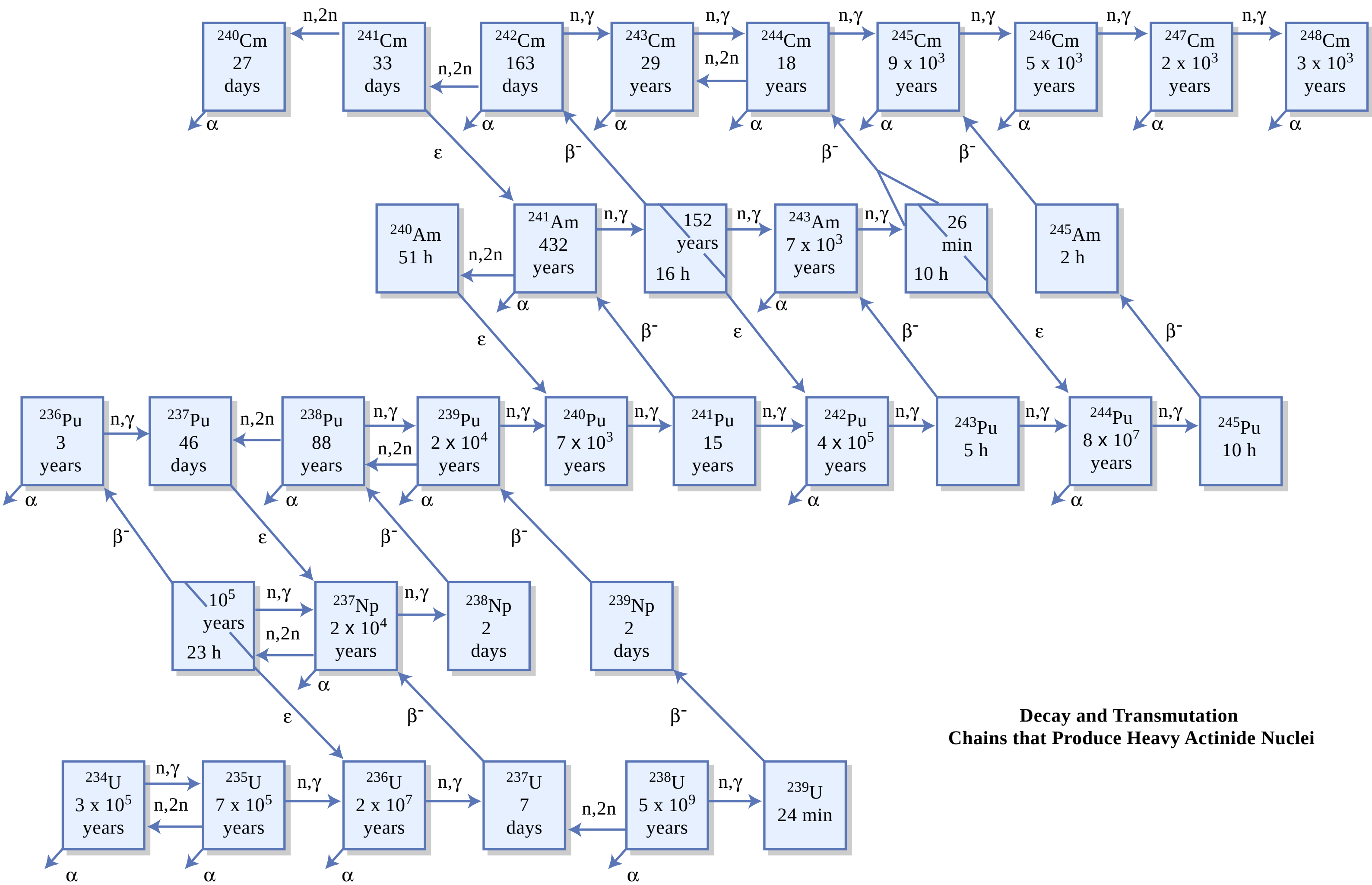


Figure by MIT OCW.

POSSIBLE PATHS TO ^{238}Pu

(1) $^{235}U(n, \gamma) \rightarrow {}^{236}U(n, \gamma) \rightarrow {}^{237}U \xrightarrow{\beta\ 6.75d} {}^{237}Np(n, \gamma) \rightarrow {}^{238}Np \xrightarrow{\beta\ 2.12d} {}^{238}Pu$

(2) $^{238}U(n, 2n) \rightarrow\rightarrow\rightarrow\rightarrow\rightarrow\rightarrow {}^{237}U \xrightarrow{\beta\ 6.75d} {}^{237}Np(n, \gamma) \rightarrow {}^{238}Np \xrightarrow{\beta\ 2.12d} {}^{238}Pu$

(3) $^{242}Cm \xrightarrow{\alpha\ 162.8d} {}^{238}Pu$

(4) $^{239}Pu(n, 2n) \rightarrow {}^{238}Pu$

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Please see: *Nuclear Technology* 91 (1990): 323, 324, and 326.