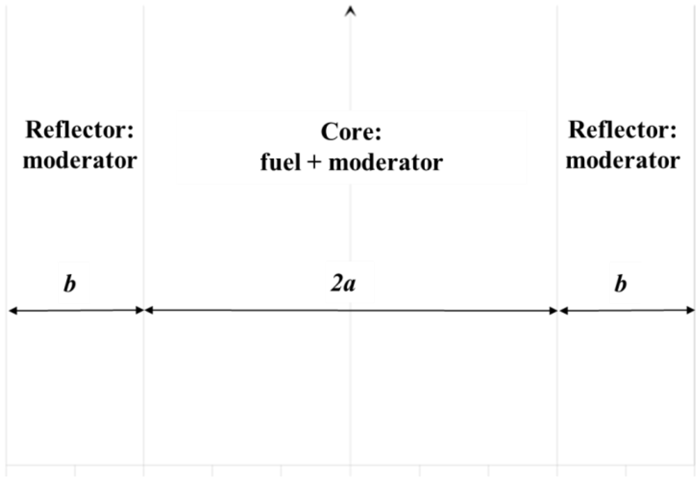
- 6 distinct exercises need to be carried out by each student to produce a portfolio of scripts (and associated write-up < 5 pages each) solving standard reactor physics problems
  1. ~~Attenuation of neutrons from a 1D planar source~~
  2. ~~1D/1G reactor slab~~
  3. ~~1D/2G reactor slab~~
  4. ~~1D/2G reactor slab with a loading pattern~~
  5. ~~Fuel Evolution with exposure in a homogeneous media~~
  6. Reactor evolution during a cycle (space and time)

Week	Date
9	15-Nov
10	23-Nov
11	30-Nov
12	06-Dec
13	13-Dec
14	20-Dec

Reactor evolution during a cycle (space and time)

- Using the 1D reactor model of a three batch core approach (Problem #4)
- Each assembly is 20cm wide, there are 4 fresh (ID 1), 4 once burned (ID 2) and 4 twice burned (ID 3) fuel assemblies
- The reflector assembly has the ID 4
- The loading pattern over a half space is [2 1 2 1 3 3 4]
  - $a = 120\ cm$
  - $b = 20\ cm$
- The reactor produces a constant power of 240 MW
- The homogenized fuel assembly has the following characteristics:
  - 264 fuel rods per assembly
  - Fuel rod is 400cm high
  - Outer pellet radius is 0.4cm
  - Fuel density (UO_2) is 10.4 gcm^{-3}
- The concentrations of Pu-239, U-235, U-238 and Y are tracked in every region of the core.
  - All isotopes are stable
  - All fissionable isotopes produce only Y with a FY of 2
  - The decay of U-239 into Pu-239 happens instantaneously



Initial Fuel Composition

State	U-238	U-235	Pu-239	Y
Fresh (1)	8.47E-03	2.62E-04	0.00E+00	0.00E+00
Once Burnt (2)	8.38E-03	1.39E-04	3.31E-05	3.31E-04
Twice Burnt (3)	8.28E-03	6.76E-05	4.15E-05	4.15E-04

Reactor evolution during a cycle (space and time)

- The microscopic XS are not affected by the change of composition in the medium
- The sets of 2G macroscopic XS are build from microscopic cross sections
  - The macroscopic scattering matrix are assumed fixed irrespective of the fuel composition.

Macro XS	Reflector	
Grp. ID	1	2
D	1.20000E+00	4.00000E-01
Σ_a	1.00000E-03	2.00000E-02
$\upsilon\Sigma_f$	0.00000E+00	0.00000E+00
$\kappa\Sigma_f$	0.00000E+00	0.00000E+00
Scattering Matrix		
to grp	1	2
from 1	2.51780E-01	2.50000E-02
from 2	0.00000E+00	8.13330E-01

Microscopic Cross Sections

Isotope	U-238		U-235		Pu-239		Y	
Grp. ID	1	2	1	2	1	2	1	2
Kappa	3.3169E-11	3.3169E-11	3.2407E-11	3.2407E-11	3.3305E-11	3.3305E-11	0.0000E+00	0.0000E+00
Nu	2.8156E+00	2.4921E+00	2.4511E+00	2.4367E+00	2.8962E+00	2.8665E+00	0.0000E+00	0.0000E+00
sig_f	1.2620E-01	8.8950E-06	8.1513E+00	2.7936E+02	9.0170E+00	7.8388E+02	0.0000E+00	0.0000E+00
sig_c	7.2612E-01	1.4287E+00	3.8226E+00	4.9061E+01	5.2792E+00	4.3977E+02	0.0000E+00	7.0000E+01

Macroscopic Scattering Cross Sections

to grp	1	2
from 1	0.4	0.02
from 2	0.0	1.0

Learning outcomes

1. Merge diffusion and depletion solver; modeling of a reactor cycle
2. Evolution of the non-uniformity with exposure
3. Light water reactors produce a part of their energetic output by fission of plutonium
4. Spectral change within a reactor and effect on the macroscopic cross sections

Coupling neutron transport and depletion calculations

Algorithm:

- For each time step do:
  - ➢ Compute macroscopic cross sections for each mesh of the problem
  - ➢ Determine the matrix A, F
  - ➢ Determine keff and phi using power iterations
  - ➢ Normalize the flux level to the reactor power
  - ➢ For each spatial mesh
    - ✓ Setup the Bateman matrix A
    - ✓ Determine the isotopic composition for the next time step using the Matrix Exponential method
  - ➢ Go to the next time step

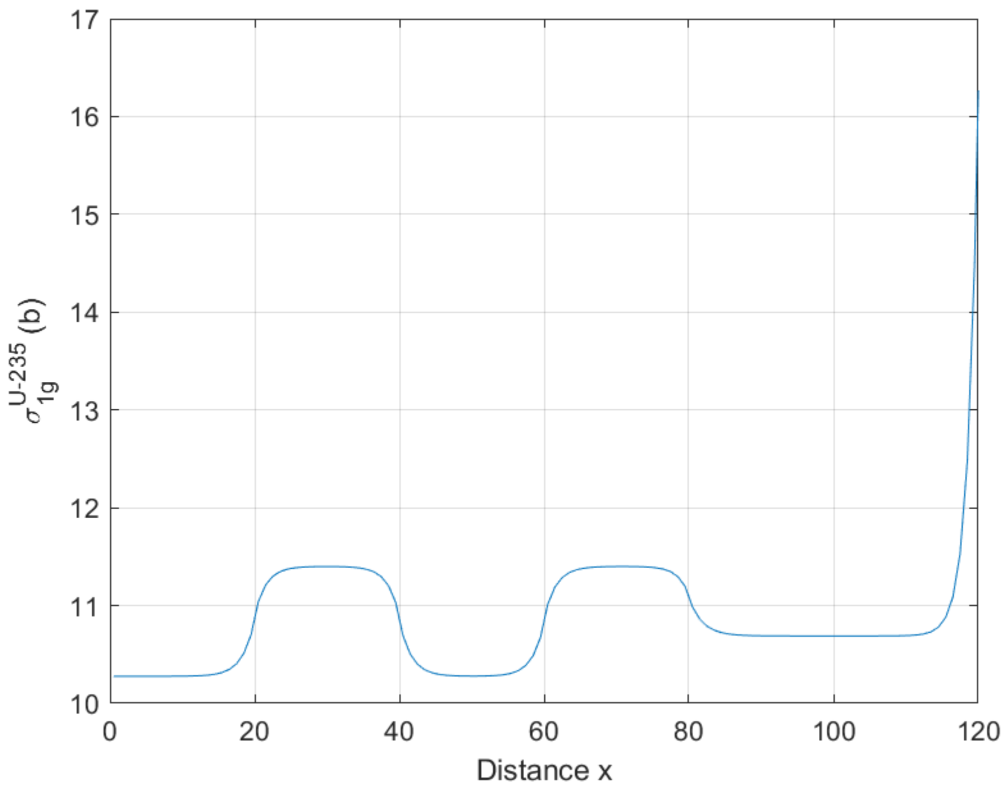
- Express the total macroscopic XS for a group *g* of a given spatial mesh *i* considering:
  - ➢ Microscopic capture and fission XS
  - ➢ Isotopic compositions
  - ➢ Fixed macroscopic scattering matrix

$$\Sigma_{t,i}^{g} = \sum_{j} \left(\sigma_{c,i}^{g} + \sigma_{f,i}^{g}\right) N_i^j + \sum_{g'} \Sigma_{s,i}^{gg'}$$

- Produce a table of 2G macroscopic XS for each assembly at the Beginning of Cycle
  - ➢ Report D, sig_t, sig_a, sig_f, kappa sig_f, nu sig_f and the scattering matrix

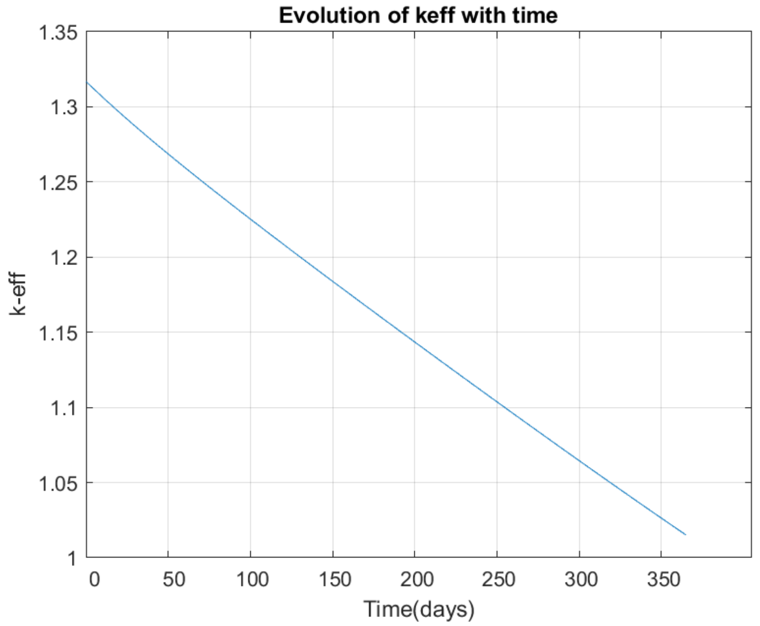
Ass. ID	1		2		3		4	
Grp. ID	1	2	1	2	1	2	1	2
D	7.74550E-01	3.03540E-01	7.76490E-01	2.97280E-01	7.77980E-01	2.99260E-01	1.20000E+00	4.00000E-01
sig_t	4.30360E-01	1.09810E+00	4.29280E-01	1.12130E+00	4.28460E-01	1.11390E+00	2.77780E-01	8.33330E-01
sig_a	1.03560E-02	9.81480E-02	9.28000E-03	1.21300E-01	8.46000E-03	1.13860E-01	1.00000E-03	2.00000E-02
nusig_f	8.24440E-03	1.78350E-01	6.61930E-03	1.69000E-01	5.37650E-03	1.39270E-01	0.00000E+00	0.00000E+00
ksig_f	1.04670E-13	2.37200E-12	8.17370E-14	2.12260E-12	6.49800E-14	1.69550E-12	0.00000E+00	0.00000E+00
scat mat								
to grp	1	2	1	2	1	2	1	2
from 1	4.00000E-01	2.00000E-02	4.00000E-01	2.00000E-02	4.00000E-01	2.00000E-02	2.51780E-01	2.50000E-02
from 2	0.00000E+00	1.00000E+00	0.00000E+00	1.00000E+00	0.00000E+00	1.00000E+00	0.00000E+00	8.13330E-01

- Report the flux normalization factor at BOC (4 sig digs)
  - 4.3612e+13
- Plot the 1group microscopic capture XS of U-235 vs position at BOC in the fuel assemblies



Solve the problem numerical using the matrix exponential method, a time step of 1 day. The spatial resolution for the core is 1cm. The cycle length is 365 days. The convergence criteria for keff is set to 10^{-6}. Predictor-corrector algorithm is not used initially.

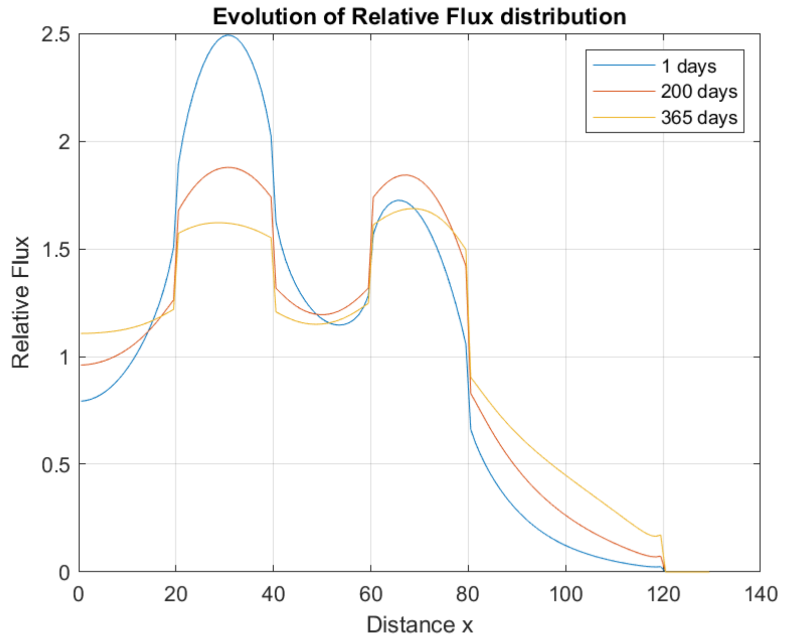
- Plot k-eff as a function of time;



- Report the core exposure at EOC in MWd/kgU (initial heavy metal mass) as well as the initial U mass (in kg, 4 sig digs)
  - Initial U mass 3.3128e+03 kg
  - Burnup at EOC 13.2213 MWd/kgU

Fuel Evolution with exposure

- Plot the spatial distribution of the relative power after 1day, 200 days and 365 days.



- Plot the spatial distribution of U-235, Pu-239 and Y after 1day, 200 days and 365 days.