OAK RIDGE National Laboratory

# Radiation emission calculations in ORIGEN



ORNL is managed by UT-Battelle, LLC for the US Department of Energy

U.S. DEPARTMENT OF ENERGY

# Learning goals



**Introduce** users to the radiation emission capabilities in ORIGEN (gamma, neutron, alpha, and beta) and their associated options



**Demonstrate** how OPUS can be used to visualize decay spectra calculated by ORIGEN

# Input component: Emission calculations

```
=origen
bounds{
    neutron=[20e6 1e3 1 1e-3] %eV
}
case{
    …
    print{
       neutron{
           spectra=yes
       }
    }
    neutron=yes
    %equivalent to
    %neutron{
    %   alphan_medium=CASE
    %   alphan_step=LAST
    %   alphan_bins=200
    %   alphan_cutoff=0.0
    %}
}
…
end
```

- Emission calculations have 3 parts (neutron as example)
  - define bounds in eV: bounds→neutron
  - enable calculation/set parameters: case→neutron
  - ask for printout: case→print→neutron
- Neutron and gamma emission includes a charged particle slowing down component
  - (alpha,n) component for neutron emission
  - β+, β- Bremsstrahlung for gamma emission
- Alpha and beta emissions are ***pure sources***, i.e., no slowing down!

# Input component: Neutron

```
=origen
bounds{
    neutron=[1e6 1e3 1] %2-group with 1MeV, 1keV, 1eV
    %read neutron bounds from file instead
    %neutron="scale.rev04.xn252v7.1"
}
case{
    …
    neutron{
        alphan_medium=CASE %0/UO2 for UO2
                           %1/BOROSILICATE
                           %  for Borosilicate glass
                           %2/CASE for case-specific
                           %(DEFAULT)
        alphan_bins=200    %use 200 bins for
                           %alpha slowing down
        alphan_cutoff=0.0  %cutoff for alpha,n
        alphan_step=LAST   %step index in this case
    } %alternatively neutron=yes
    print{
        neutron{
            summary=yes
            spectra=no
            detailed=no
        }
    }
}
end
```

- Neutron emission calculation has 3 components
  - (alpha,n) neutrons
  - spontaneous fission neutrons
  - delayed neutrons
- (alpha,n) requires a charged particle slowing down calculation→dependent on medium
  - `medium`: $UO_2$, Borosilicate glass, problem-dependent
  - Problem-dependent compositions taken from a single step (`alphan_step`)
  - Isotopes with atom fraction or an activity below cutoff are not considered

# Input component: Gamma

```
=origen
bounds{
    gamma=[1e6 1e3 1] %2-group with 1MeV, 1keV, 1eV
    %read gamma bounds from file instead
    %gamma="scale.rev10.xn27g19v7.0"
}
case{
    …
    gamma{
        sublib=ALL       %LT, FP, or AC - single sublib
                            %ALL - all sub-libraries
        brem_medium=UO2 %assume Bremsstrahlung in UO2
        immediate=yes    %load lines for immediate gamma/x-rays
        spont=yes        %include photons from spontaneous fission
                            %and alpha,n reactions
        continuum=yes %expand continuum data "lines" into proper continuua
        conserve_line_energy=no %conserve line energy instead of line intensity
        adjust_for_missing=no   %adjust photon intensity for missing data
        split_near_boundary=no  %lines within 3% of a group boundary are split
                                %into two bins
    }
    print{
        % default gamma options
        gamma{
            summary=yes
            spectra=no
            principal_step=NONE  %step index to calculate (NONE to suppress)
            principal_cutoff=2   %principal emitter cutoff in percent
            unbinned_warning=no  %print warning when line not binned
        }
    }
}
end
```

- Gamma emission calculation has 2 types of spectral data
  - line data
  - continuum data (e.g., bremsstrahlung)
- Bremsstrahlung requires charged particle slowing down
  - **Pre-tabulated** as a function of beta energy for $UO_2$ & $H_2O$ media
  - `brem_medium:` $UO_2$, $H_2O$ (no problem-dependent option like for neutron)
- Default calculation options generally fine except
  - `brem_medium=H2O/UO2/NONE`
  - If no option is appropriate, sensitivity can be assessed by trying each one

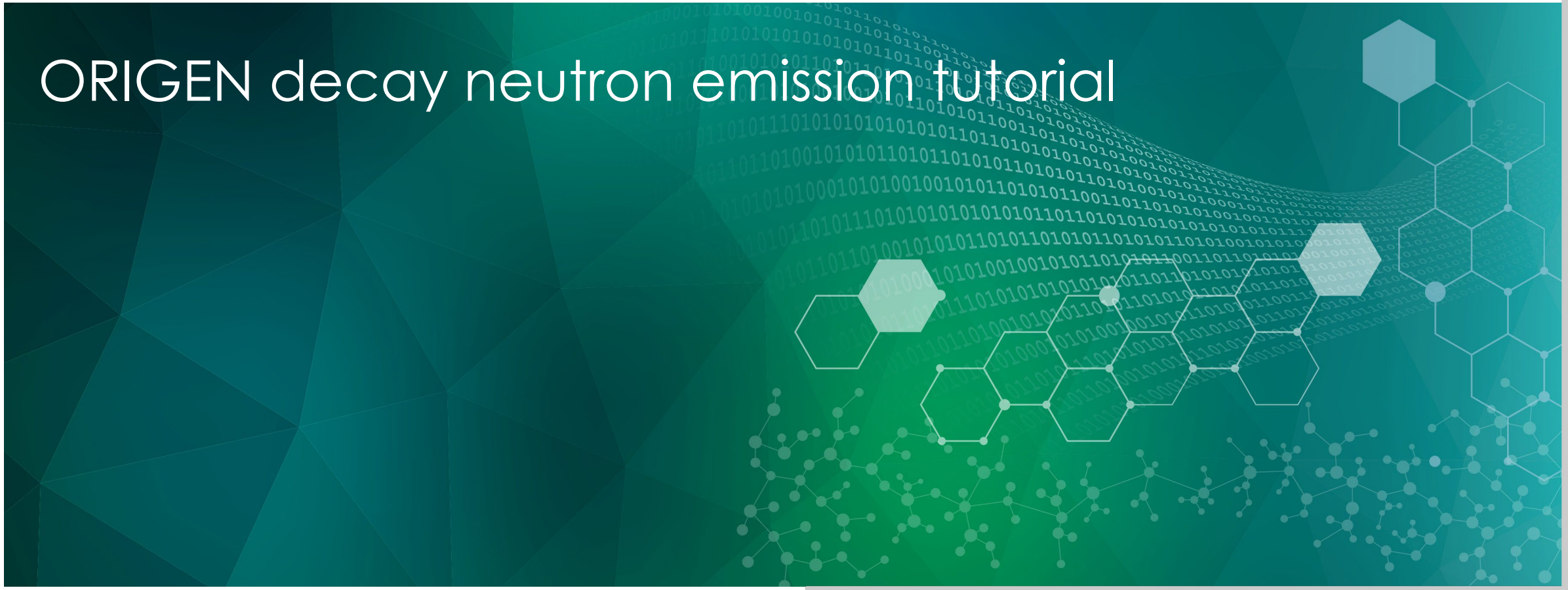
# Input component: Alpha/beta

```
=origen
bounds{
    alpha=[1e6 1e3 1]
    beta=[1e6 1e3 1]
}
case{
    …
     beta{
        sublibs=ALL   %NONE - no calculation
                      %LT, FP, or AC - single sublib
                      %ALL - all sub-libraries
    } %or beta=yes
    alpha=yes %or alpha{}
    print{
        alpha{
            summary=yes
            spectra=no
        }
        beta{
            summary=yes
            spectra=no
            principal_step=NONE %step index to calculate
                                %principal emitters
            principal_cutoff=2  %principal emitter cutoff
(%)
        }
    }
}
end
```

- Alpha/beta bounds
  - `alpha/beta="/path/to/library"` not allowed
  - only direct eV input
- Calculation blocks
  - alpha block has no options
  - beta block only sets sublibs
- Remember alpha/beta are **sources** with no slowing down physics

OAK RIDGE National Laboratory

# ORIGEN decay neutron emission tutorial



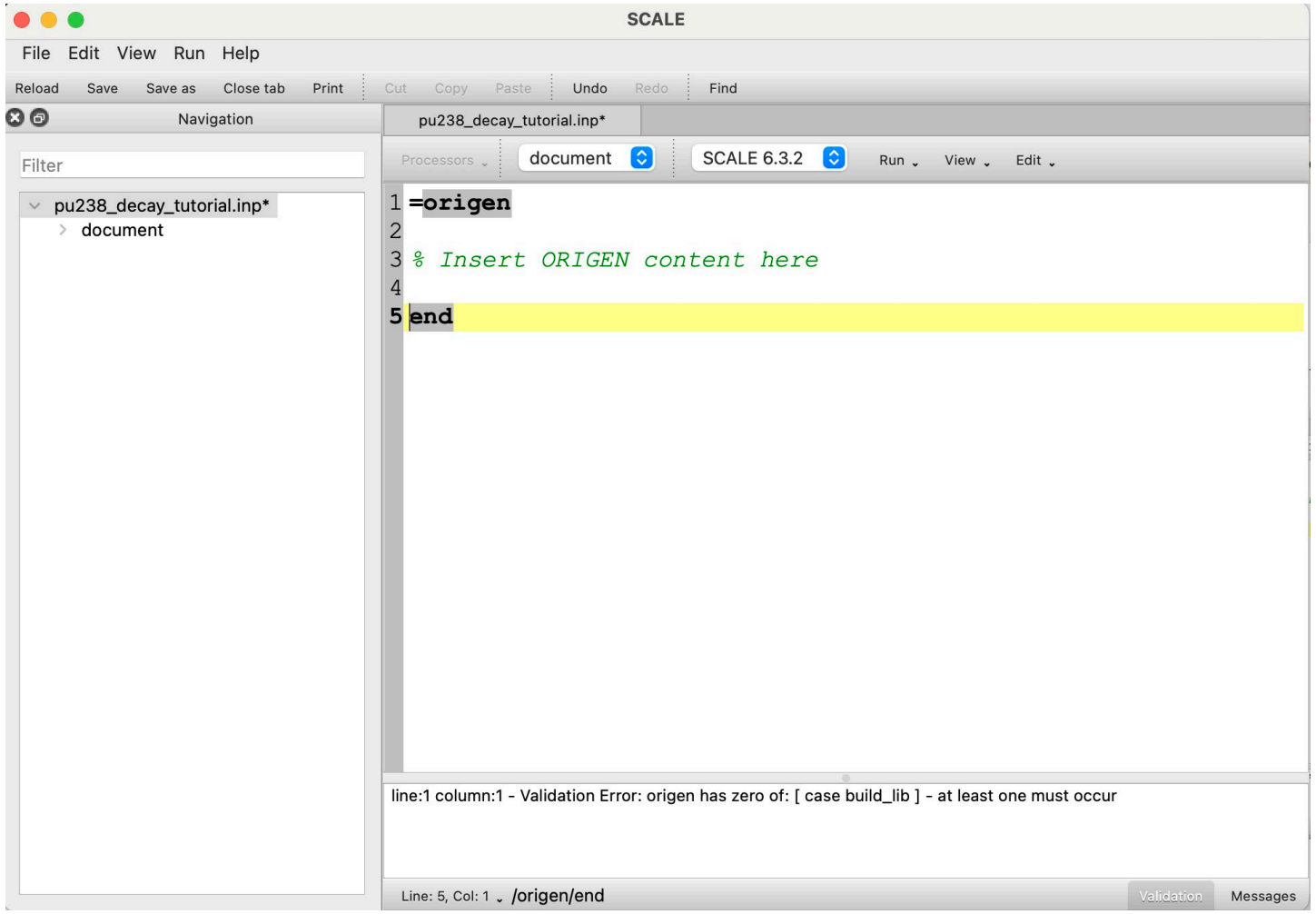
ORNL is managed by UT-Battelle, LLC for the US Department of Energy

U.S. DEPARTMENT OF ENERGY

# Overview

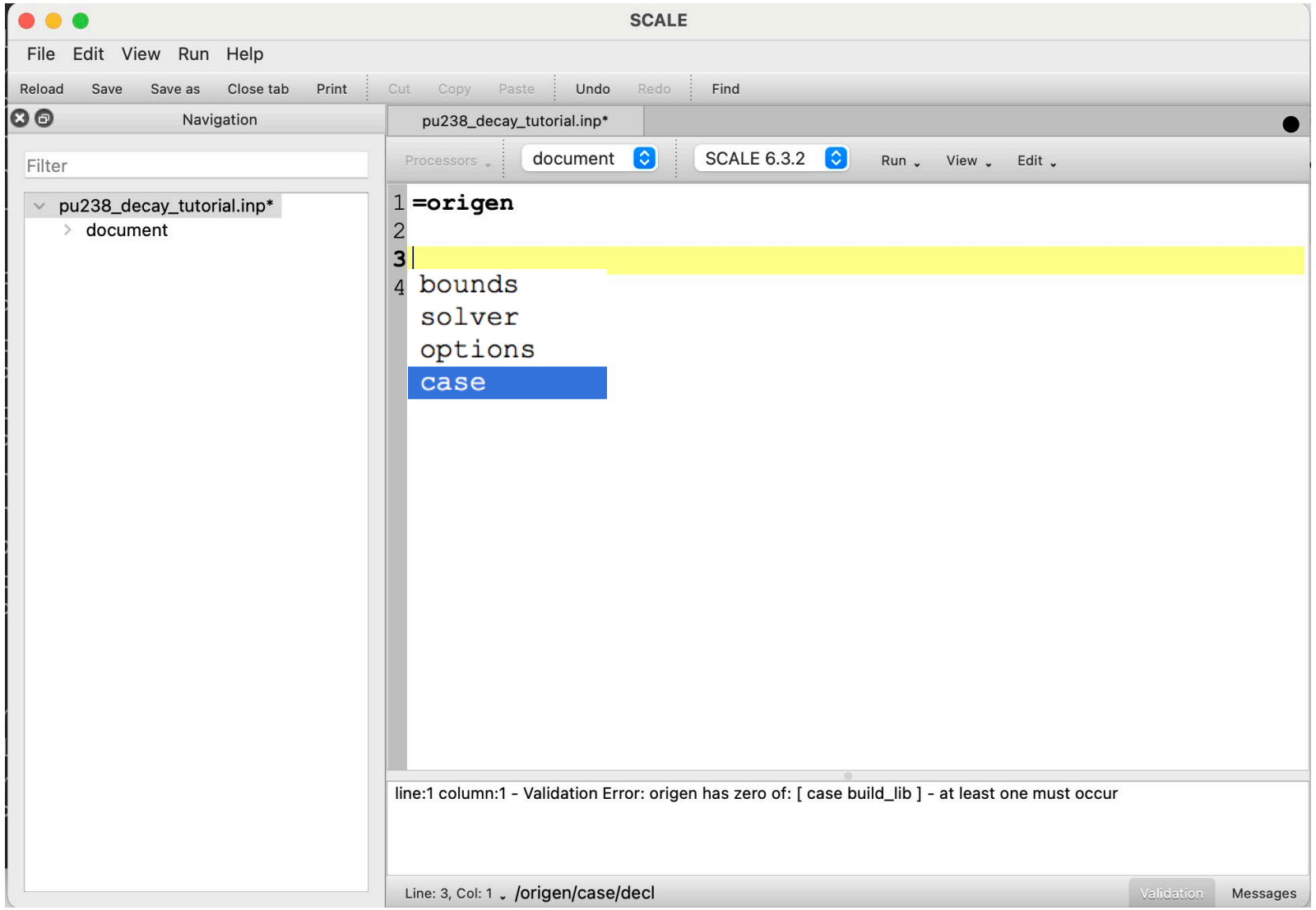
- Decay calculation with neutron emission
  - 1 g $^{238}Pu$
  - 5 years decay
  - enable neutron calculation
  - compare (a,n) options problem specific to oxide matrix

# Clean slate



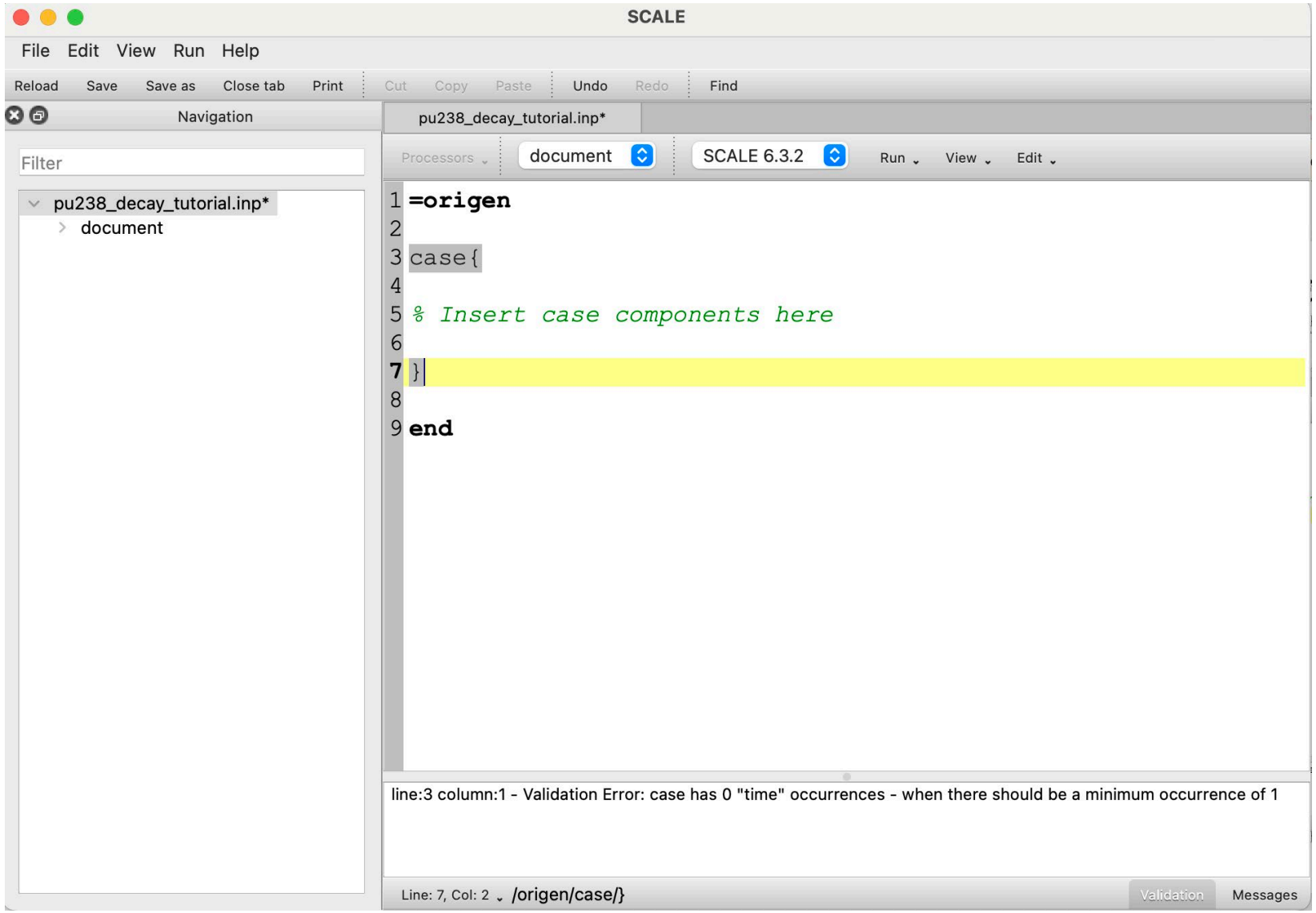

- Start with empty file
- For this tutorial, either
  - type commands as I do it
  - use CTRL+SPACE to autocomplete
- Click on "Validation" button to expand the validation window

# Start with a case




- Create a case
  - CTRL+SPACE
  - Down arrow to case

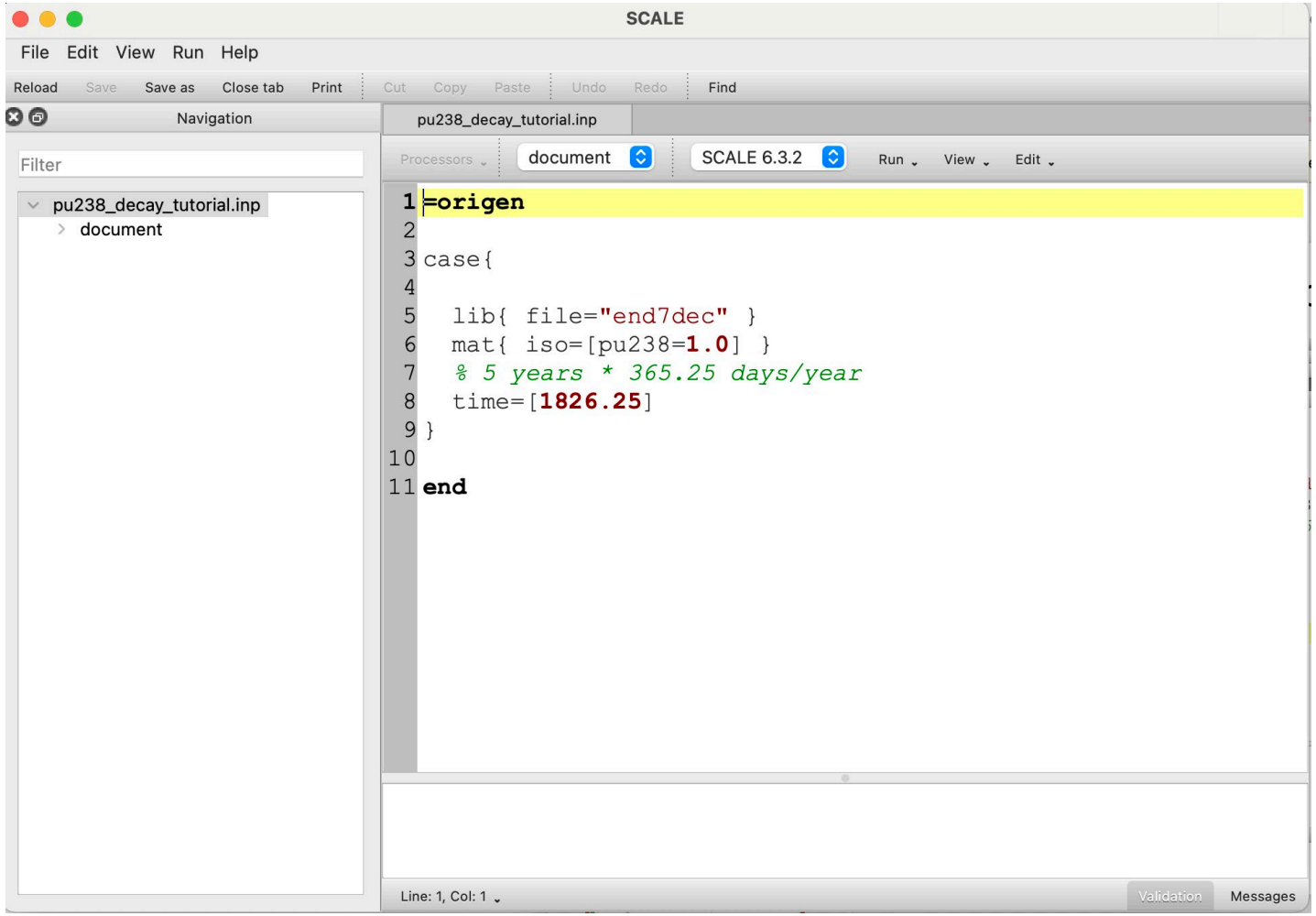
# 12 Possibilities in a Case




CTRL+SPACE inside case gives you

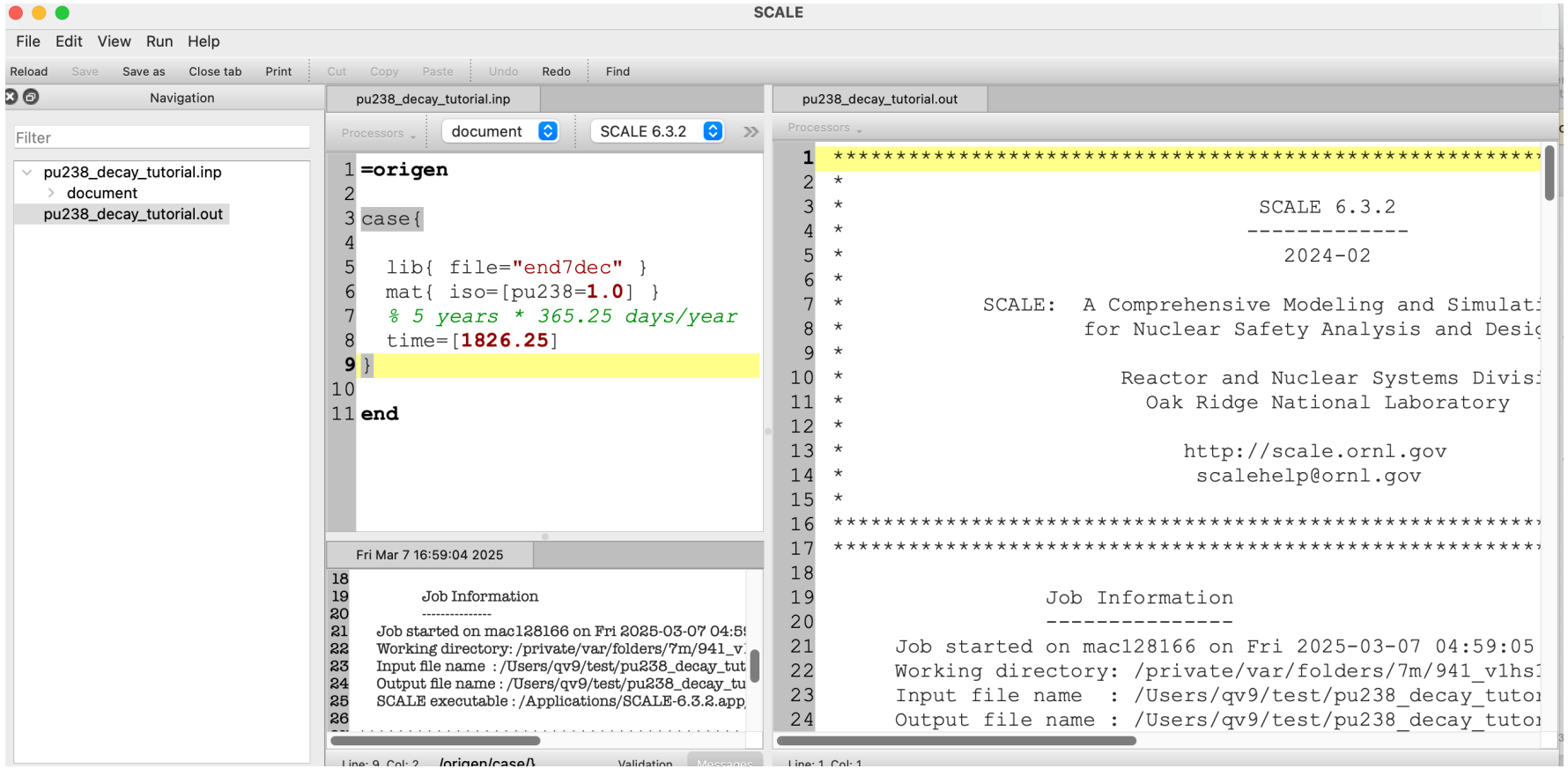
```
title
lib
time
flux
power
print
mat
save
neutron
gamma
beta
processing
```

# Three are used almost every time



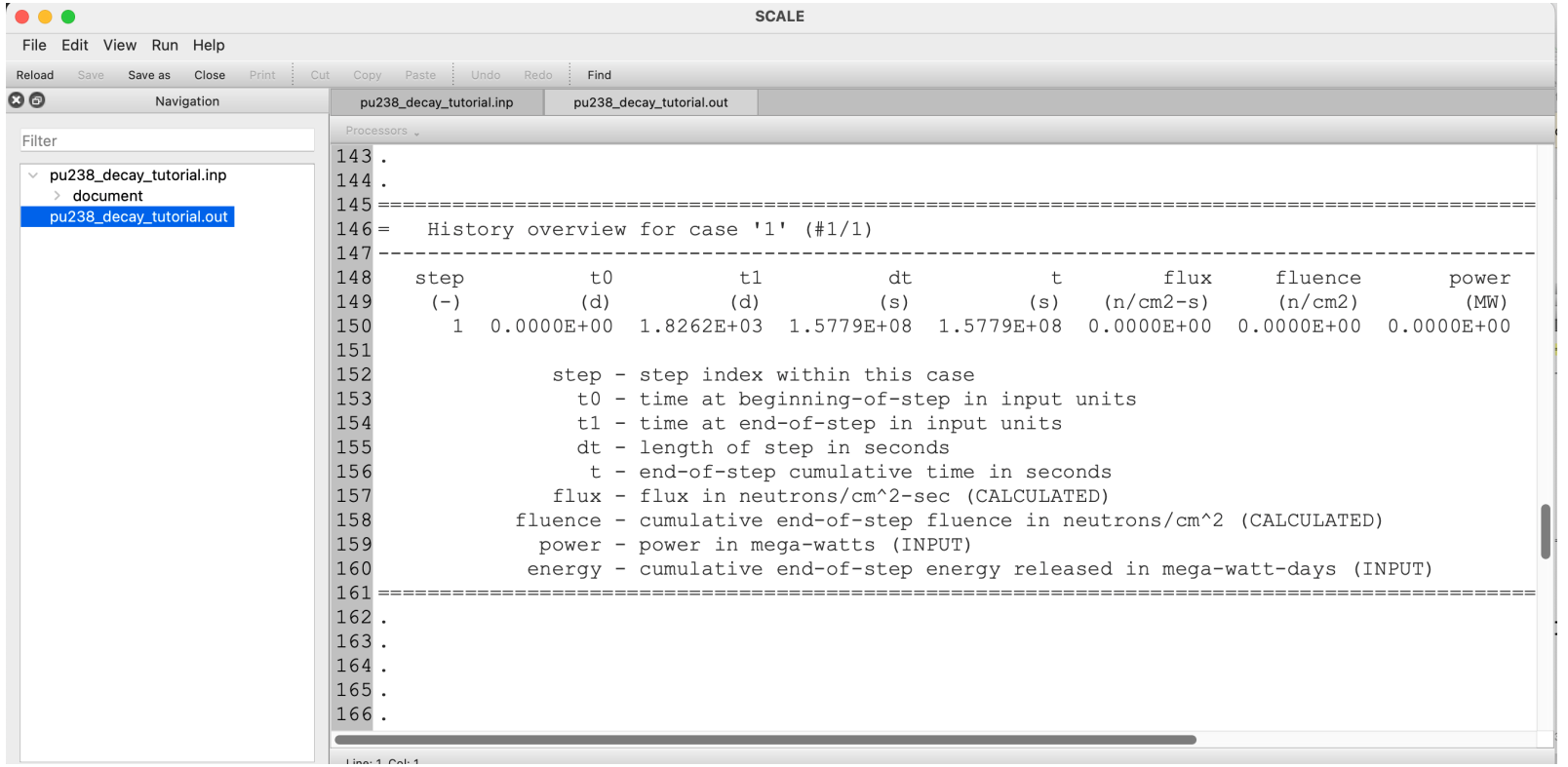

- Let's fill in the big 3
  - **lib**
    for decay, just use "end7dec"
  - **mat**
    iso array with default units
  - **time**
    time array with default units
- No validation errors!
- Click “Run”

# Side-by-side for input/output




- Right-click in **Navigation** on *.inp and open *.out & *.msg
- Drag *.out TAB to right side of Fulcrum to have split screen input/output

# Find the history table




- SCALE output is very wide! (~120 columns)
- History overview is very valuable for double checking operating history entry (time/power/flux)
  - Use “Find” and search for “History overview”

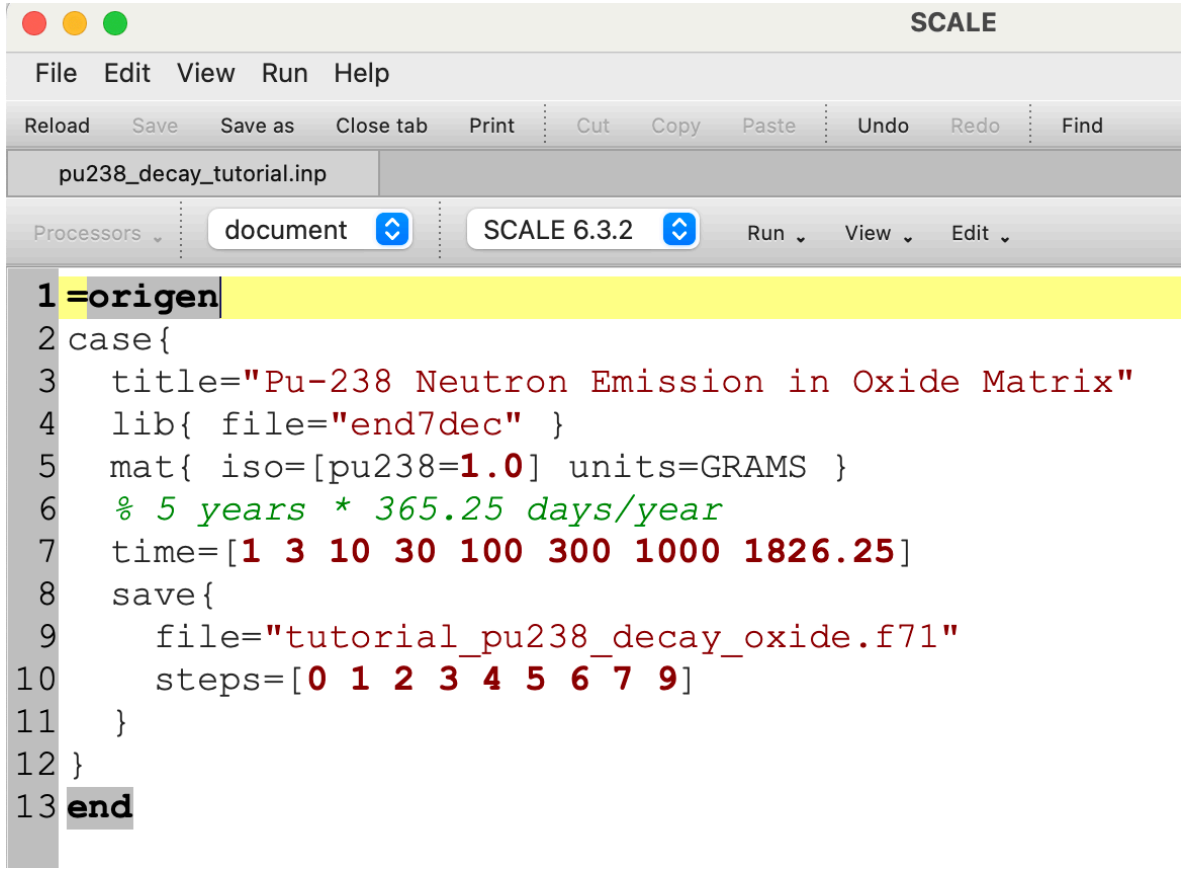
# Find the concentration table

```
.
**********************************************************************************************************************
**********************                                                                          ***************
**********************                                                                          ***************
**********************                        Concentration tables                              ***************
**********************                                                                          ***************
**********************                                                                          ***************
**********************************************************************************************************************
.
.
======================================================================================================================
=   Nuclide concentrations in grams for case '1' (#1/1)
----------------------------------------------------------------------------------------------------------------------
  (relative cutoff; integral of concentrations over time >   1.00E-06 % of integral of all concentrations over time)
.
                    0.000d   1826.250d
  he-4         0.0000E+00  1.5509E-01
  th-230       0.0000E+00  6.3330E-05
  u-234        0.0000E+00  9.0686E+00
  pu-238       2.3805E+02  2.2883E+02
------------
  totals       2.3805E+02  2.3805E+02
======================================================================================================================

```

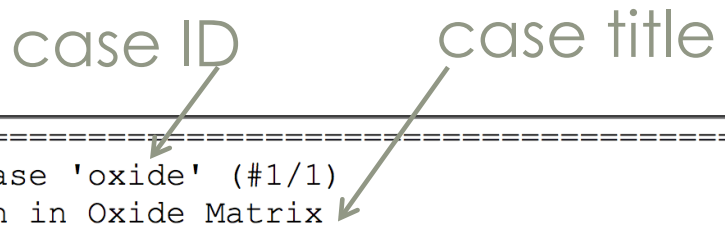
- Default output table is by nuclide in grams
- Verify 1 mole of Pu-238 = 238.05 grams
- Mass conserved, 0.15509 grams He generated

# Add more detail




- Let's go back and make the following changes
  - add a title for the case
  - add an ID for the case (oxide)
  - change amount to 1 gram of Pu-238 (units=GRAMS)
  - add more times explicitly (no array shortcuts) using "rule of 3s" and start with 1 day
  - save an f71 file with all steps explicitly (no array shortcuts) called **"tutorial_pu238_decay_oxide.f71"**
- Inspect new output file
  - history
  - default grams output
  - compare He with 1 time step vs. rule of 3 s
    - 1 step: 6.5152E-04 g
    - Rule of 3s: 6.5152E-04 g

# Verify in the output




```
146 ================================================================================================
147 =   History overview for case 'oxide' (#1/1)
148 =   Pu-238 Neutron Emission in Oxide Matrix
149 ------------------------------------------------------------------------------------------------
150    step          t0          t1          dt           t        flux     fluence       power      energy
151     (-)         (d)         (d)         (s)         (s)   (n/cm2-s)     (n/cm2)        (MW)       (MWd)
152       1  0.0000E+00  1.0000E+00  8.6400E+04  8.6400E+04  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
153       2  1.0000E+00  3.0000E+00  1.7280E+05  2.5920E+05  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
154       3  3.0000E+00  1.0000E+01  6.0480E+05  8.6400E+05  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
155       4  1.0000E+01  3.0000E+01  1.7280E+06  2.5920E+06  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
156       5  3.0000E+01  1.0000E+02  6.0480E+06  8.6400E+06  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
157       6  1.0000E+02  3.0000E+02  1.7280E+07  2.5920E+07  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
158       7  3.0000E+02  1.0000E+03  6.0480E+07  8.6400E+07  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
159       8  1.0000E+03  1.8262E+03  7.1388E+07  1.5779E+08  0.0000E+00  0.0000E+00  0.0000E+00  0.0000E+00
160
161              step - step index within this case
162                t0 - time at beginning-of-step in input units
163                t1 - time at end-of-step in input units
164                dt - length of step in seconds
165                 t - end-of-step cumulative time in seconds
166              flux - flux in neutrons/cm^2-sec (CALCULATED)
167           fluence - cumulative end-of-step fluence in neutrons/cm^2 (CALCULATED)
168             power - power in mega-watts (INPUT)
169            energy - cumulative end-of-step energy released in mega-watt-days (INPUT)
170 ================================================================================================
171 .
```

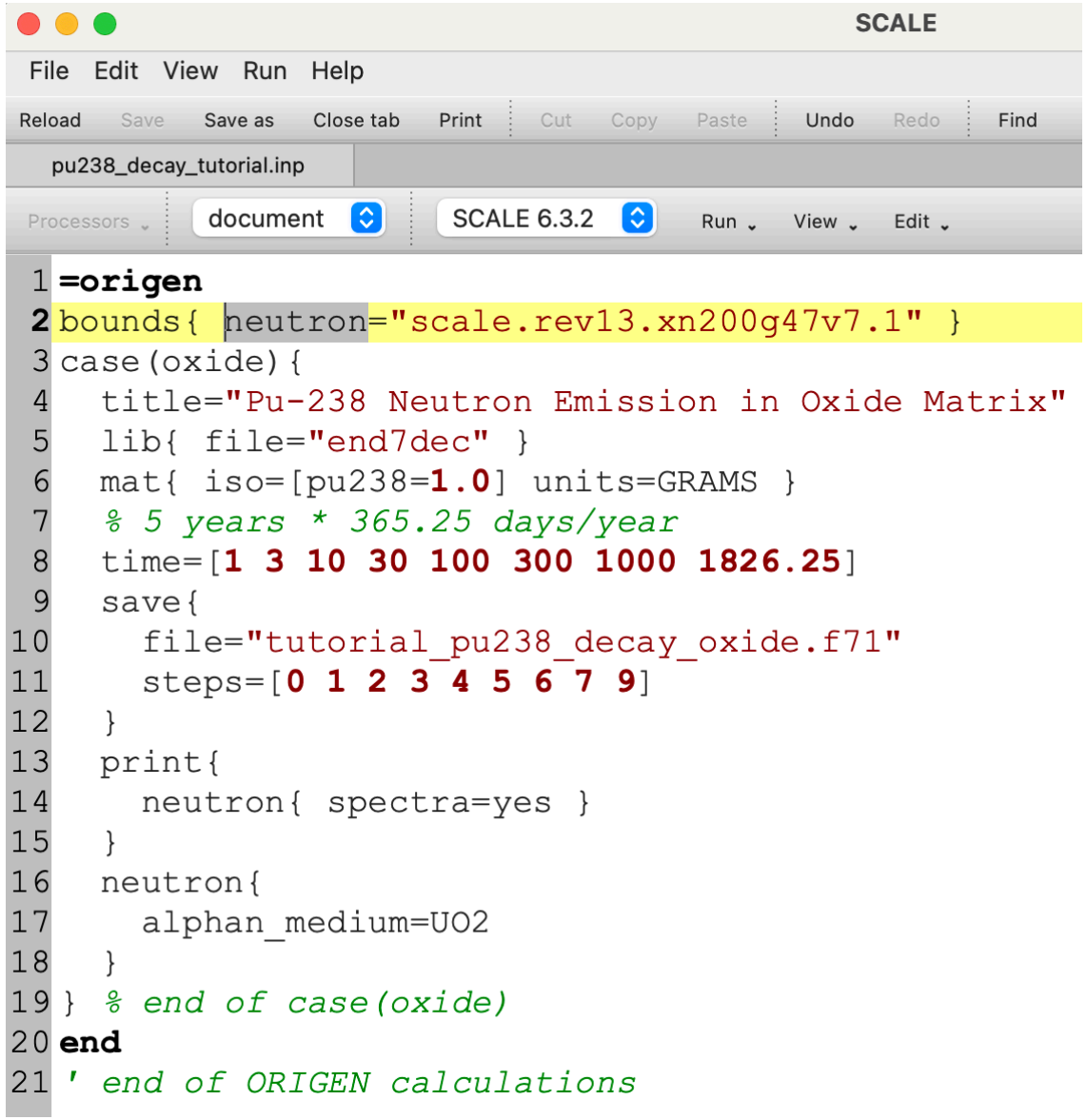
# Verify the F71 is being written

```
**********************************************************************************************************
**********************                                                                        ***********
**********************                                                                        ***********
**********************                    Written F71 File Information                        ***********
**********************                                                                        ***********
**********************                                                                        ***********
**********************************************************************************************************
.
.
 index        time        power          flux       fluence       burnup libpos    case    step DCGNAB
     1  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       0 DC----
     2  8.64000e+04  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       1 DC----
     3  2.59200e+05  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       2 DC----
     4  8.64000e+05  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       3 DC----
     5  2.59200e+06  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       4 DC----
     6  8.64000e+06  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       5 DC----
     7  2.59200e+07  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       6 DC----
     8  8.64000e+07  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       7 DC----
     9  1.57788e+08  0.00000e+00  0.00000e+00  0.00000e+00  0.00000e+00      1       1       8 DC----
    D - state definition present
    C - concentrations present
    G - gamma emission spectra present
    N - neutron emission spectra present
    A - alpha emission spectra present
    B - beta emission spectra present
```

available positions

content at each position

# Add the neutron calculation




```
=origen
bounds{ neutron="scale.rev13.xn200g47v7.1" }
case(oxide){
  title="Pu-238 Neutron Emission in Oxide Matrix"
  lib{ file="end7dec" }
  mat{ iso=[pu238=1.0] units=GRAMS }
  % 5 years * 365.25 days/year
  time=[1 3 10 30 100 300 1000 1826.25]
  save{
    file="tutorial_pu238_decay_oxide.f71"
    steps=[0 1 2 3 4 5 6 7 9]
  }
  print{
    neutron{ spectra=yes }
  }
  neutron{
    alphan_medium=UO2
  }
} % end of case(oxide)
end
' end of ORIGEN calculations
```

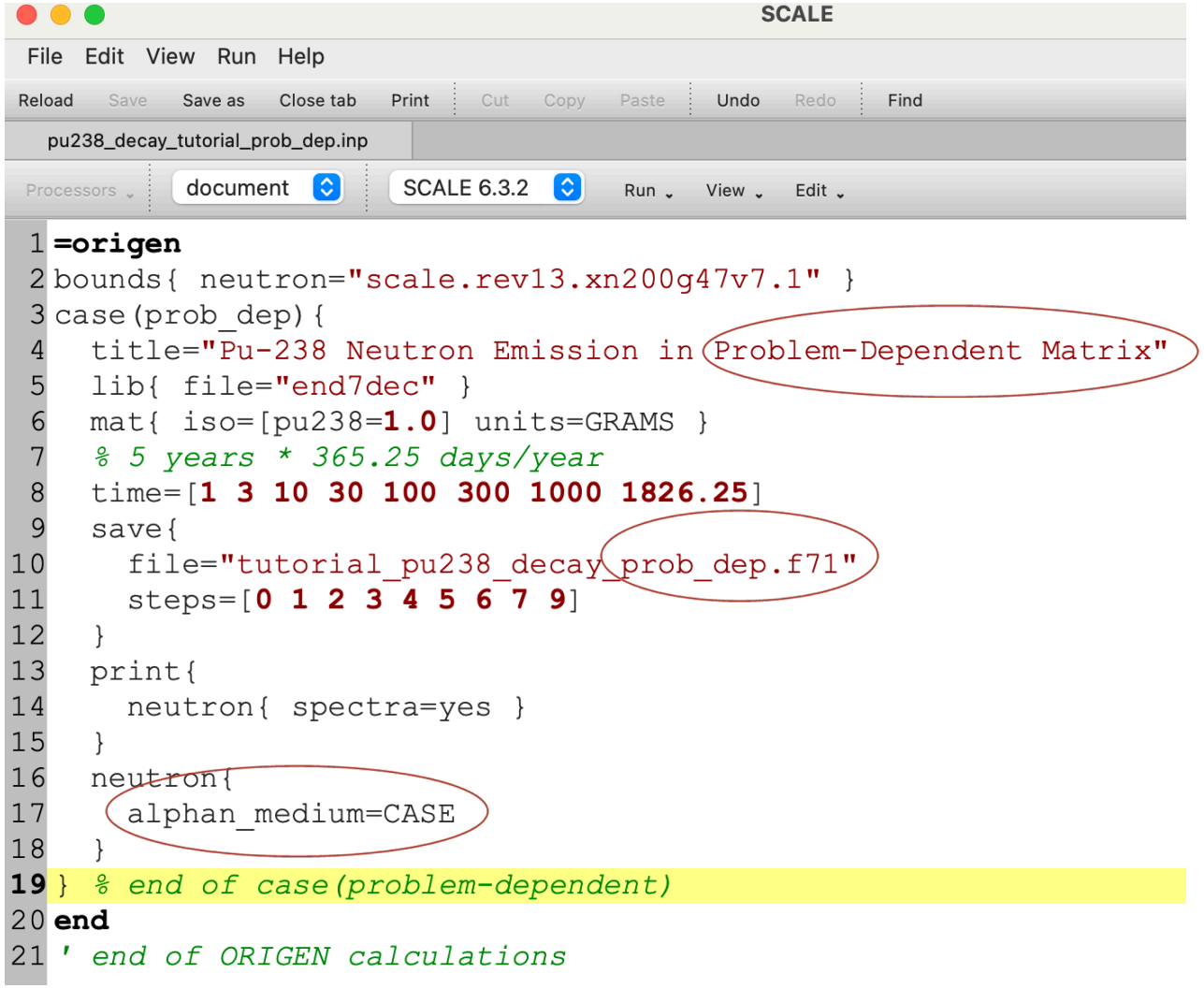
- Now let's add neutron calculation
  - case→neutron→alphan_medium=UO2
  - case→print→neutron→spectra=yes
  - bounds→neutron="**scale.rev13.xn200g47v7.1**"
- Inspect new output file
  - History
  - Default grams output
  - Compare He with 1 time step vs. rule of 3's
    - **1 step:** 6.5152E-04 g
    - **Rule of 3's:** 6.5152E-04 g
  - Verify F71 now has neutron data

# Verify neutron emission data on F71

```
1242 ********************************************************************************************************
1243 ********************
1244 ********************
1245 ********************                    Written F71 File Information
1246 ********************
1247 ********************
1248 ********************************************************************************************************
1249 .
1250 .
1251  index          time         power          flux       fluence        burnup libpos    case    step DCGNAB
1252      1   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       0 DC-N--
1253      2   8.64000e+04   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       1 DC-N--
1254      3   2.59200e+05   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       2 DC-N--
1255      4   8.64000e+05   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       3 DC-N--
1256      5   2.59200e+06   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       4 DC-N--
1257      6   8.64000e+06   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       5 DC-N--
1258      7   2.59200e+07   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       6 DC-N--
1259      8   8.64000e+07   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       7 DC-N--
1260      9   1.57788e+08   0.00000e+00   0.00000e+00   0.00000e+00   0.00000e+00      1       1       8 DC-N--
1261     D - state definition present
1262     C - concentrations present
1263     G - gamma emission spectra present
1264     N - neutron emission spectra present
1265     A - alpha emission spectra present
1266     B - beta emission spectra present
```

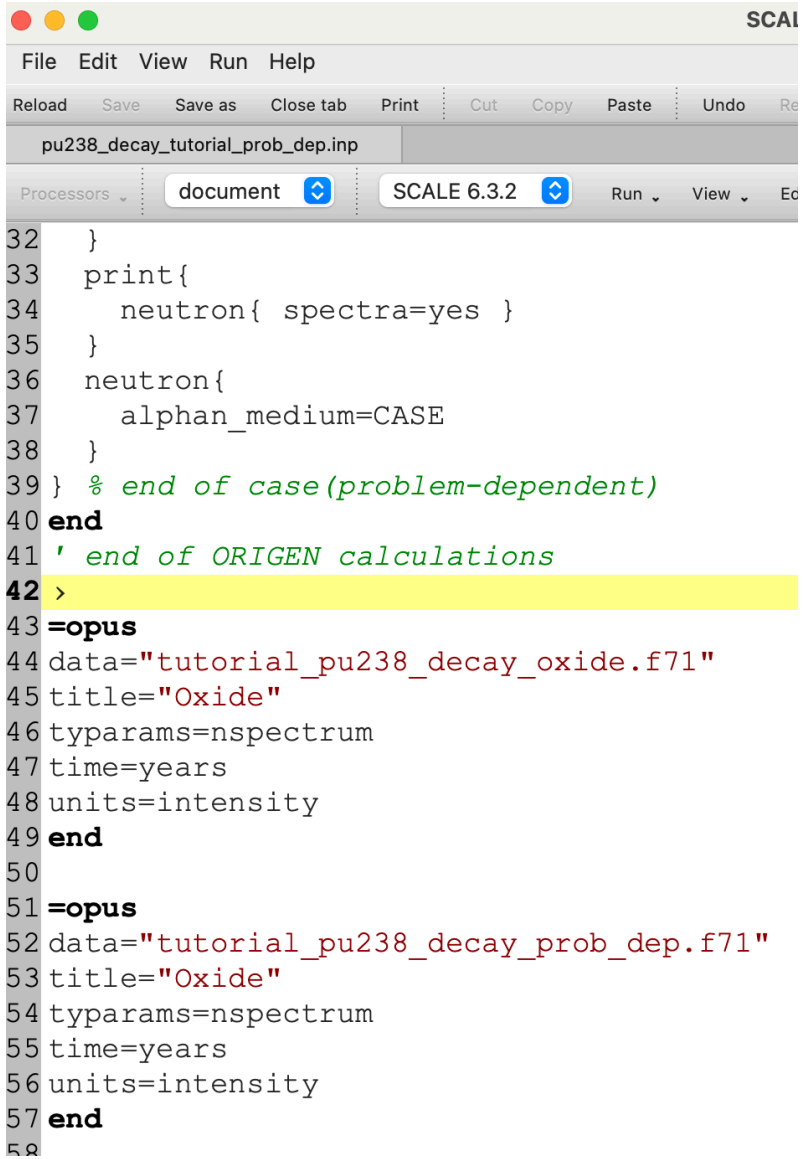
neutron data available

# Create a second case with different medium




- Now let's add a second case!
  - copy-paste (CTRL-C / CTRL-V)
- Create independent case "case(probdep)" with problem-dependent medium
  - case→title
  - case→neutron→alphan_medium=CASE
  - case→save→file= "tutorial_pu238_decay_probdep.f71"
- Run it, examine output
  - find output tables for each case
  - find F71 summary for each case

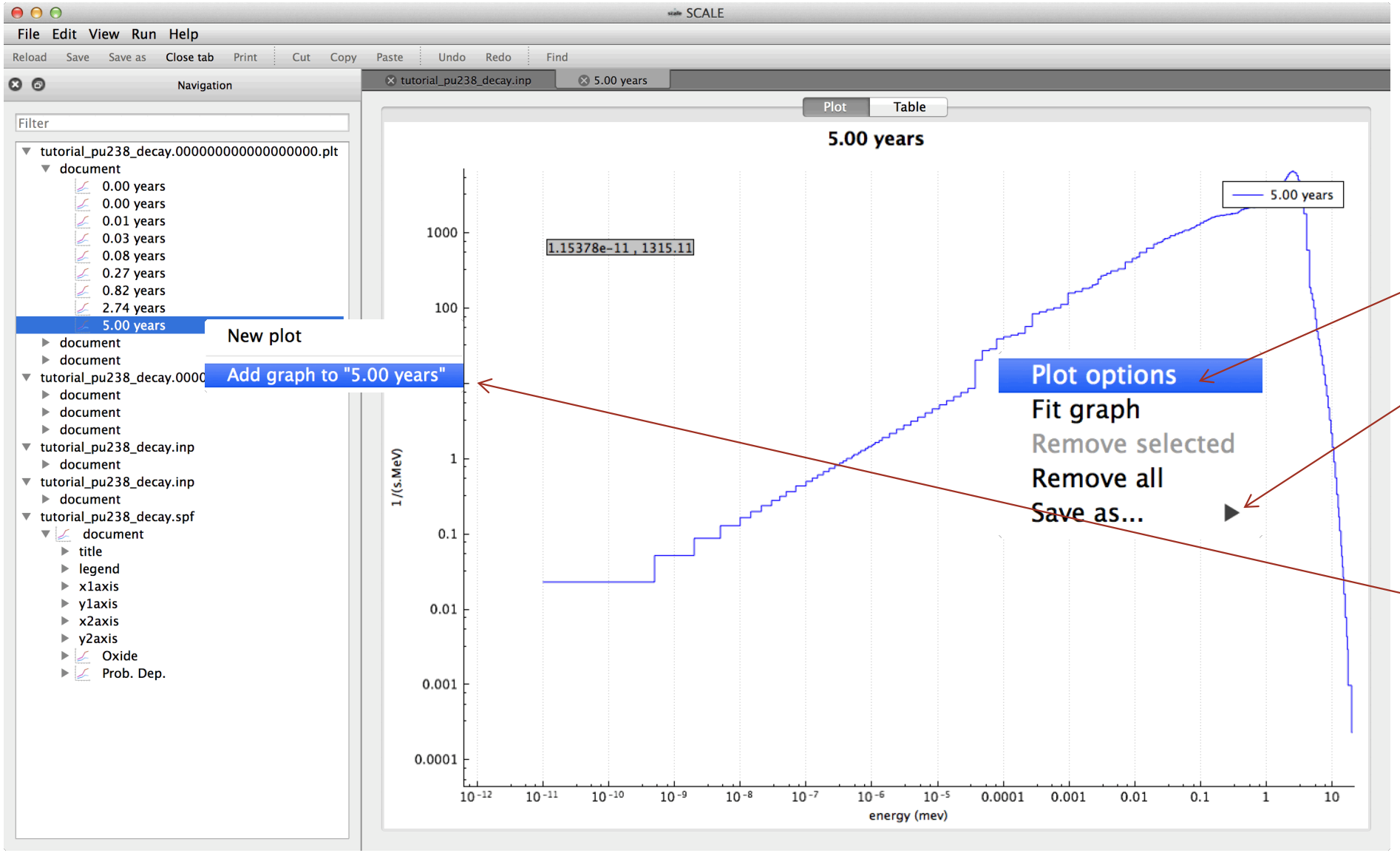
# Add OPUS




```
32    }
33    print{
34      neutron{ spectra=yes }
35    }
36    neutron{
37      alphan_medium=CASE
38    }
39  } % end of case(problem-dependent)
40  end
41  ' end of ORIGEN calculations
42  >
43  =opus
44  data="tutorial_pu238_decay_oxide.f71"
45  title="Oxide"
46  typarams=nspectrum
47  time=years
48  units=intensity
49  end
50
51  =opus
52  data="tutorial_pu238_decay_prob_dep.f71"
53  title="Oxide"
54  typarams=nspectrum
55  time=years
56  units=intensity
57  end
```

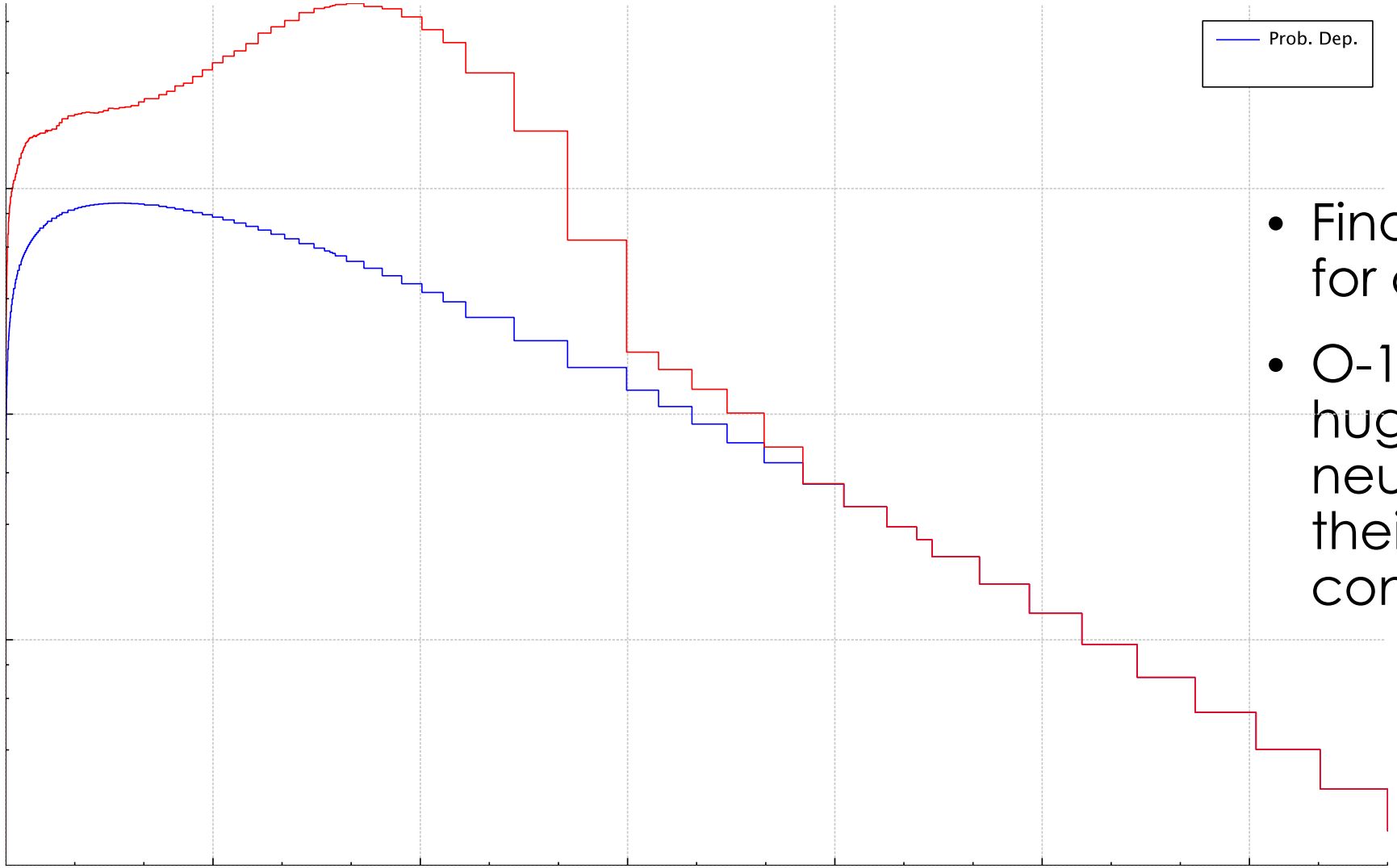
- Could copy/paste spectra from output file to Excel and make a comparison plot
- Here, we'll use OPUS and Fulcrum
  - Create one OPUS case for each f71 file
  - Rerun the calculation

# Plot manipulation




- Can right-click on plot window
  - Change axes, legends, etc.
  - Export to image file
- Can right-click on plottable in nav to add to another plot

# Final PDF




- Final PDF saved image for comparison
- O-17 and O-18 make huge difference in neutron source due to their (alpha,n) contribution

# Useful shortcuts

- `save=yes` instead of a save block, saves the f71 with all time steps
- `neutron=yes` instead of a neutron block, enables a default neutron calculation (still need bounds, though)
- `gamma=yes` instead of a gamma block, enables a default gamma calculation (still need bounds, though)

# Questions?



OAK RIDGE National Laboratory

# Workshop problems: Radiation emissions with ORIGEN



# Problem 1

- What is the shape of the $^{252}Cf$ spontaneous fission neutron spectrum?
  - Assume 1 gram of $^{252}Cf$
  - Decay for 1 day (1 step)
  - Calculate the $^{252}Cf$ neutron source in neutrons/s
- Strategy
  - Use neutron group structure (252-grp) from "**`xn252v7.1`**"
  - Save the ft71 file and use it with OPUS to plot neutron spectrum

# Problem 2

- What is the $^{60}$Co gamma (photon) spectrum?
  - Assume 40 mCi $^{60}$Co and decay for 1 day (1 step)
  - Use gamma group structure (19-gr) from "**xn28g19v7.1**"
  - Calculate the total $^{60}$Co gamma (photon) source in photons/s and plot the spectrum
- Strategy
  - Save ft71 file and use it with Opus to plot gamma spectra
    - use **linear scale on x axis** and **logarithmic scale on y axis**
    - use x range 0.1 – 4.0 and y range 1.0 – 1.0e12
  - Perform calculation twice to illustrate the impact of bremsstrahlung inclusion/exclusion on the spectrum in the low energy range
    - a) brem_medium=none
    - b) brem_medium=uo2

# Problem 3

- What are the $PuO_2$ neutron and gamma sources and spectra?
  - Reactor-grade $PuO_2$ fuel (1g Pu + 0.13g O)
  - Pu isotopic composition
    - $^{238}Pu$ 0.6 wt %
    - $^{239}Pu$ 70.4 wt %
    - $^{240}Pu$ 18.0 wt %
    - $^{241}Pu$ 9.0 wt %
    - $^{242}Pu$ 2.0 wt %
  - O natural isotopic composition
- Decay to 1 year (use 5 decay steps)

# Problem 3 (cont.)

- Strategy
  - Use neutron group structure (252-grp) from "xn252v7.1" and gamma group structure (19-grp) from "xn28g19v7.1"
  - Print neutron and gamma spectra information
    - Use "`brem_medium=none`" for gamma spectra
    - Use "`alphan_medium=case`" for (alpha,n) source term
  - Save ft71 file and use it with Opus to plot at 1 year decay
    - Total neutron spectrum
    - (alpha,n) neutron spectrum
    - Gamma spectrum
  - Use the following plot settings
    - Neutron spectra: **logarithmic scale on x axis** and **linear scale on y axis**, with x range 1.0e-4 – 10.0 and y range 1 – 125
    - Gamma spectra: **logarithmic scale on x axis** and **logarithmic scale on y axis**, with x range 0.01 – 15.0 and y range 1.0e-3 – 1.0e11

# Problem 4

- Compare PuC source spectra and $PuO_2$ neutron and gamma source spectra, in particles/s
  - Replace O in $PuO_2$ fuel from Problem 3 with 0.05 g of C (natural isotopic)
  - Perform the same type of calculation and plotting as for Problem 5
  - Compare PuC and $PuO_2$ source spectra
    - Compare (alpha,n) source (n/s/g Pu, from output file) and spectra
    - Compare total neutron source (n/s/g Pu, from output file) and spectra
    - Compare gamma source (photons/s/g Pu) and spectra
  - Use plot settings as for Problem 5

OAK RIDGE National Laboratory

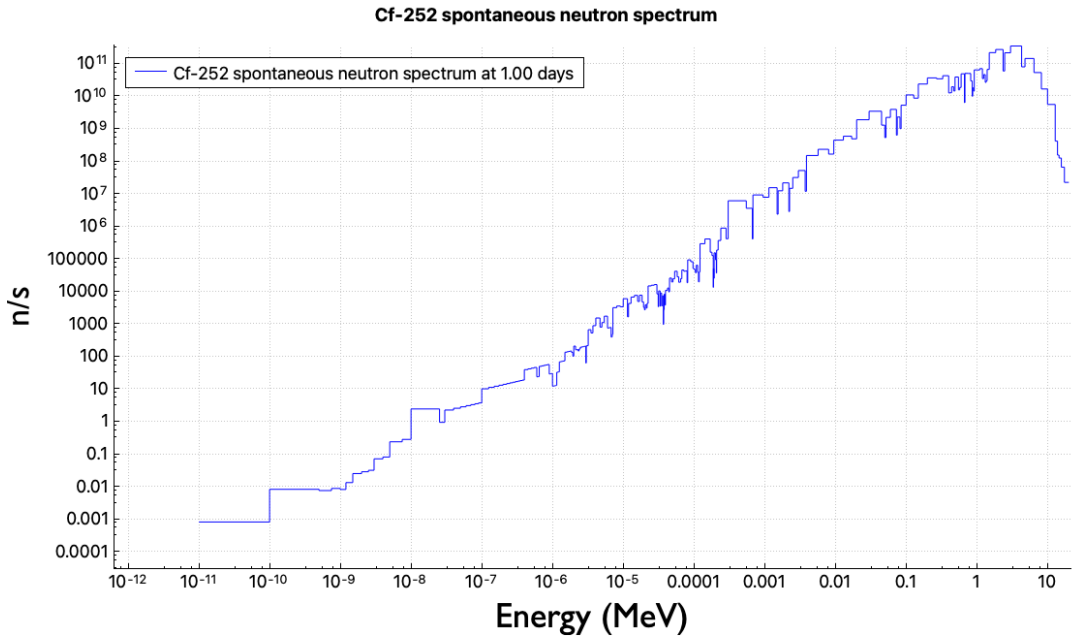
# Solutions



# Problem 1

What is the shape of the $^{252}Cf$ spontaneous fission neutron spectrum?

**ANSWER: 2.308E+12 neutrons/s**




```
=origen
% use the built-in 252-group neutron energy bin structure
bounds{ neutron="xn252v7.1" }
case{
    lib{ file="end7dec" } % use the ENDF7 decay library
    mat{
        units=grams
        iso=[cf252=1.0] % 1.0 grams natural Cf-252
    }
    time=[1] % Decay for 1 day (default time unit)
    neutron=yes % use default neutron calculation options
    save{ file="cf252.f71" steps=LAST }
}
end
=opus
  data=" cf252.f71"
  typarams=sfspectrum
  units=particles
  title="Cf-252 spontaneous neutron spectrum"
end
```

# Problem 2

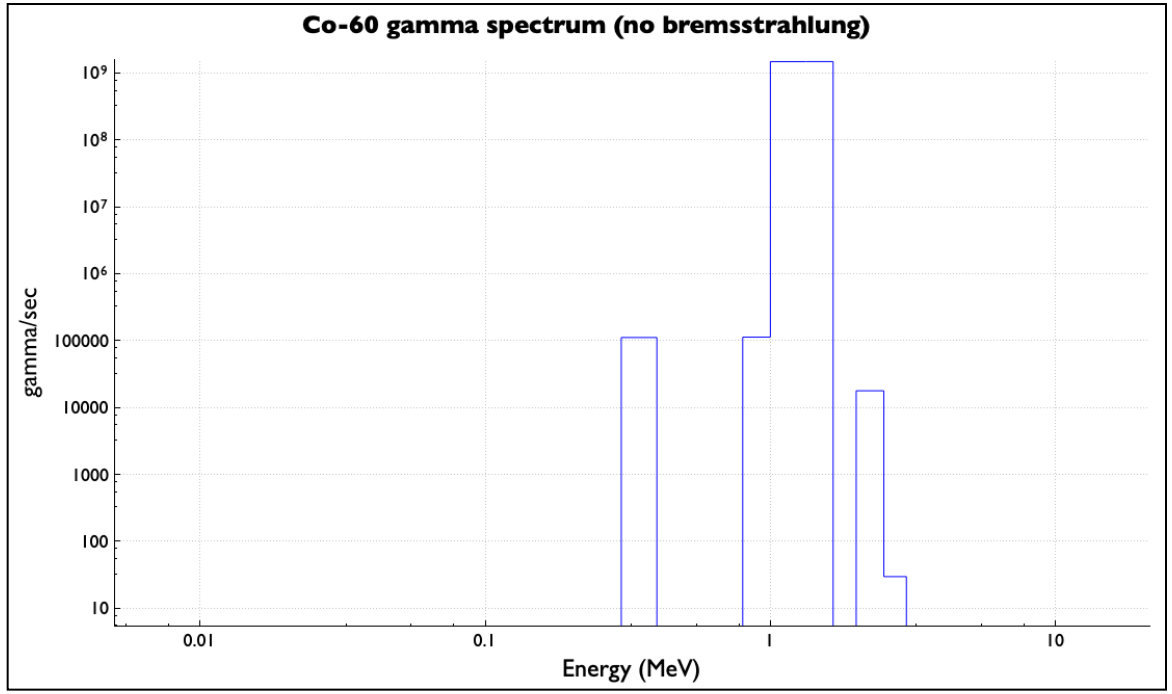
## What is the $^{60}$Co gamma (photon) spectrum?

```
=origen
% use the built-in 19-group gamma bin structure
bounds{ gamma="xn28g19v7.1" }
case{
  % use the ENDF7 decay library
  lib{ file="end7dec" }
  mat{
    units=curies
    iso=[co60=0.040] % 40 mCi Co-60
  }
  time=[1] % Decay for 1 day (default time unit)
  gamma{ brem_medium=uo2 } % UO2 bremsstrahlung medium
  save{ file="co60-uo2brem.f71" steps=LAST }
}
end
```

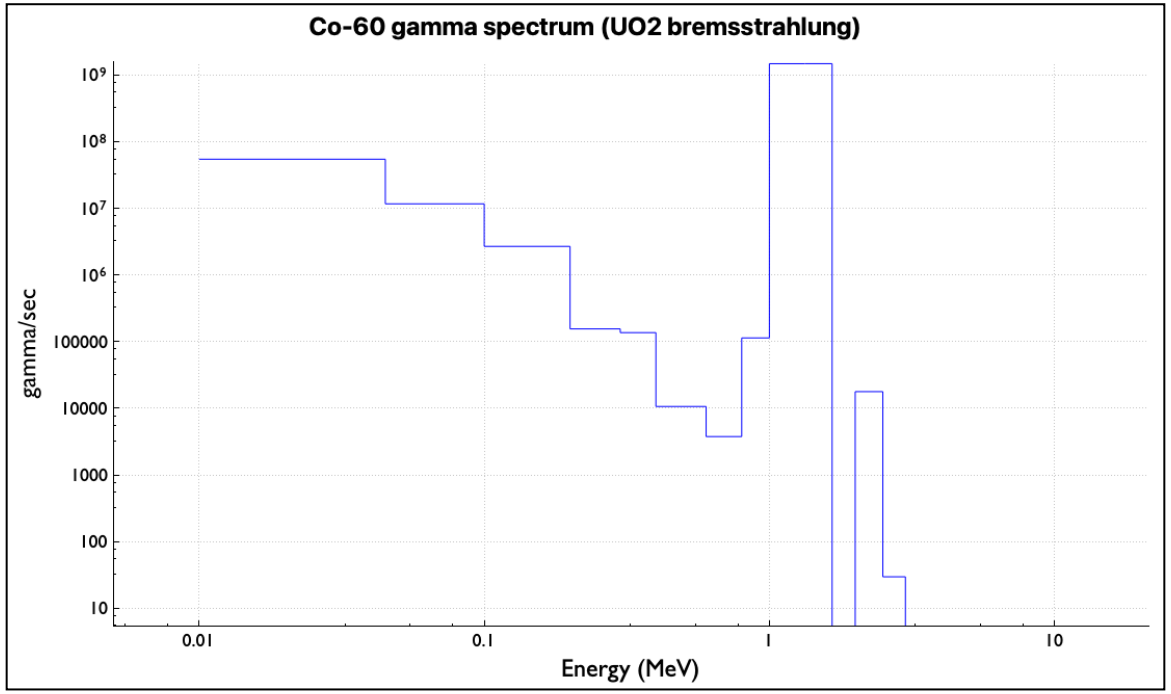
| brem_medium= | Intensity (photons/sec) |
|---|---|
| uo2 | 3.027E+09 |
| none | 2.958E+09 |

# Problem 2 (spectra)

## No bremsstrahlung




## $UO_2$ bremsstrahlung



# Problem 3

- What are the $PuO_2$ neutron source spectra?
  - Input

```
=origen
bounds{
  gamma="xn28g19v7.1" % built-in 19-group gamma bin
  neutron="xn252v7.1" % built-in 252-group neutron bins
}
case{
  lib{ file="end7dec" } % use the ENDF7 decay library
mat{
  units=grams
  % 1 gHM reactor-grade PuO2
  iso=[pu238=0.006 pu239=0.704 pu240=0.18 pu241=0.09 pu242=0.02 o=0.13]
}
time{
  t=[3L 0.1 1.0 ]
  units=years
}
gamma{ brem_medium=none } % no bremsstrahlung medium
neutron{
  % Use the problem-specific matrix for alpha-n source term
  alphan_medium=case
  % time step for problem-specific alpha,n source matrix
  alphan_step=LAST
}
save{ file="pu-oxide.f71" steps=LAST }
}
end
```

```
=opus
data="pu-oxide.f71"
typarams=nspectrum
units=particles
title="PuO2 total neutron spectrum"
end


=opus
data="pu-oxide.f71"
typarams=anspectrum
units=particles
title="PuO2 (alpha,n) neutron spectrum"
end

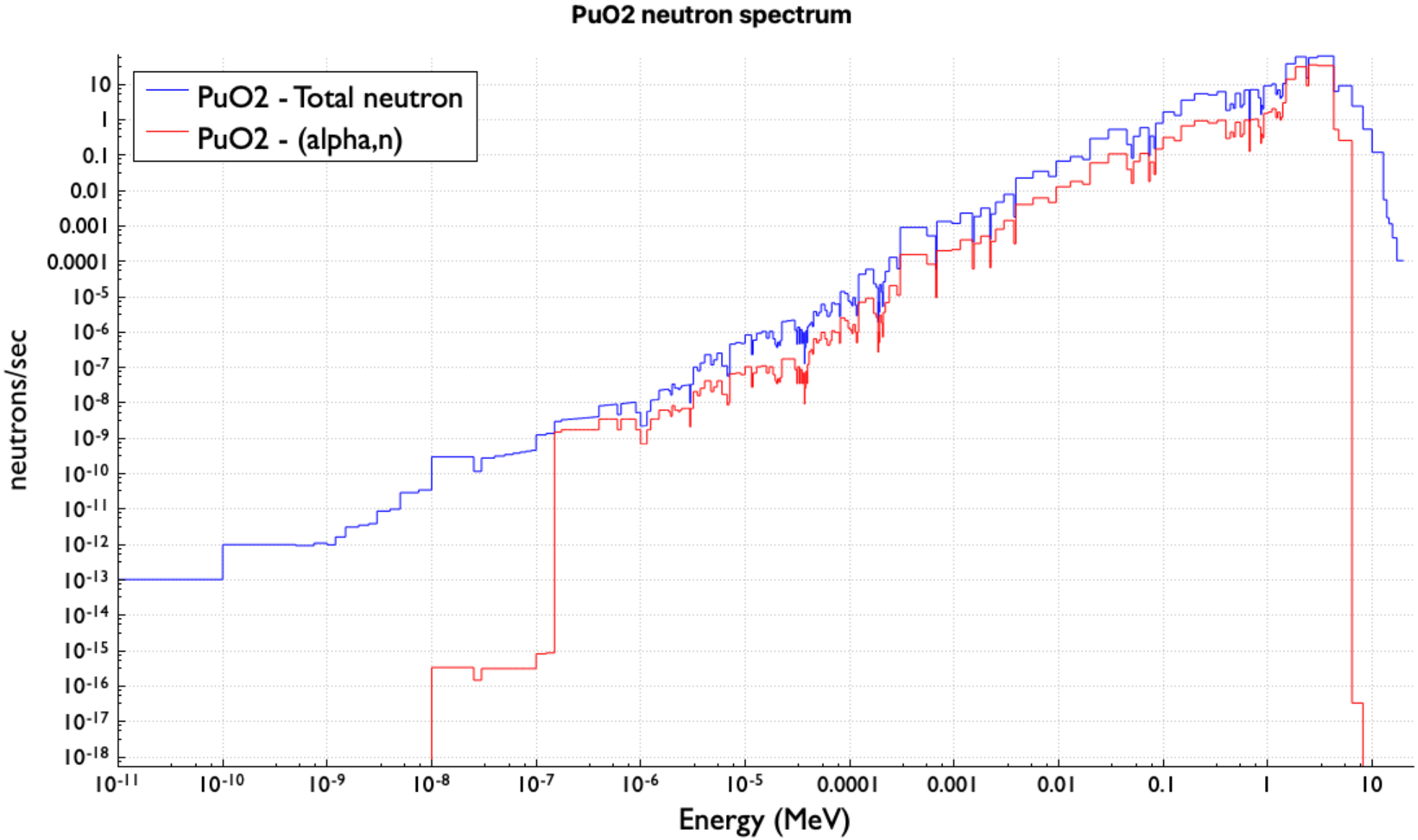

=opus
data="pu-oxide.f71"
typarams=gspectrum
units=particles
title="PuO2 gamma spectrum"
end
```

# Problem 3

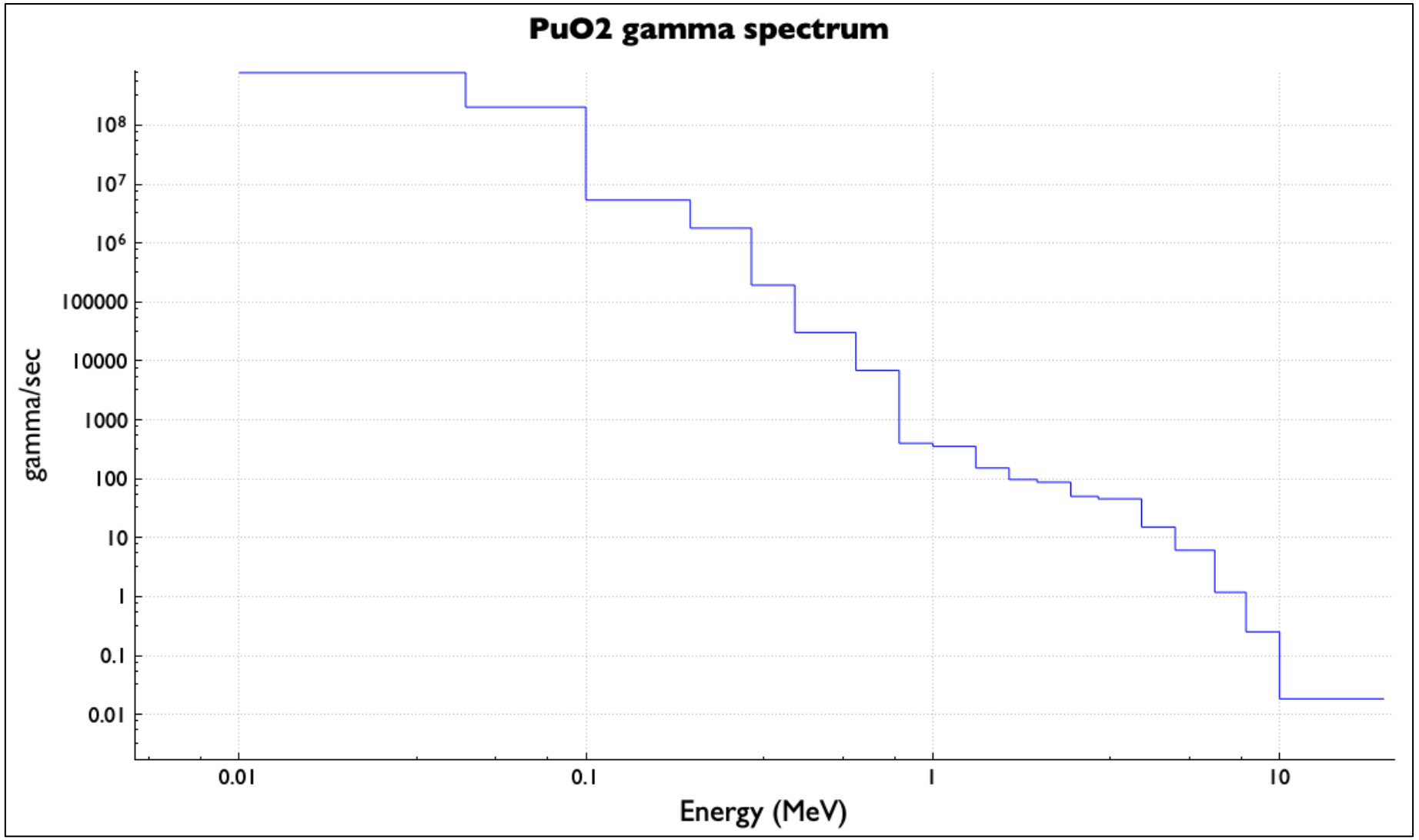
- What are the $PuO_2$ source spectra?
  - Sources at 1 year decay

| Source | Intensity (per gram Pu) |
|---|---|
| (alpha,n) | 1.491E+02 n/s/g Pu |
| Spontaneous fission neutron | 2.381E+02 n/s/g Pu |
| Total neutron | 3.872E+02 n/s/g Pu |
| Gamma | 1.104E+09 photons/s/g Pu |

# Problem 3 (neutron spectra)



# Problem 3 (gamma spectra)



# Problem 3 (comments on neutron spectra)

- Increase in neutron and gamma spectra due to buildup of Am-241 from Pu-241 decay
- Pu-238 and Pu-240 are dominant neutron sources;  Pu-238 has high specific SF neutron yield, but Pu-240 usually dominates because of its larger concentration
- The (alpha,n) reaction on oxygen can increase the neutron dose rate as much as 50% compared to same material in metal form

# Problem 3 (comments on gamma spectrum)

- For reactor grade Pu, the gamma dose rate can be expected to increase 50-100% in first year due to buildup of Am-241
- Most Pu photons are low energy (as seen in plot)
- Pu-238 and Pu-241 are dominant Pu gamma sources

# Problem 4

- What is the PuC source spectra?
  - Input

```
=origen
bounds{
  gamma="xn28g19v7.1" % built-in 19-group gamma bin
  neutron="xn252v7.1" % built-in 252-group neutron bins
}
case{
  lib{ file="end7dec" } % use the ENDF7 decay library
mat{
  units=grams
  % 1 gHM reactor-grade PuO2
  iso=[pu238=0.006 pu239=0.704 pu240=0.18 pu241=0.09 pu242=0.02 c=0.05]
}
time{
  t=[3L 0.1 1.0 ]
  units=years
}
gamma{ brem_medium=none } % no bremsstrahlung medium
neutron{
  % Use the problem-specific matrix for alpha-n source term
  alphan_medium=case
  % time step for problem-specific alpha,n source matrix
  alphan_step=LAST
}
save{ file="pu-carbide.f71" steps=LAST }
}
end
```

```
=opus
data="pu-carbide.f71"
typarams=nspectrum
units=particles
title="PuC total neutron spectrum"
end


=opus
data="pu-carbide.f71"
typarams=anspectrum
units=particles
title="PuC (alpha,n) neutron spectrum"
end


=opus
data="pu-carbide.f71"
typarams=gspectrum
units=particles
title="PuC gamma spectrum"
end
```

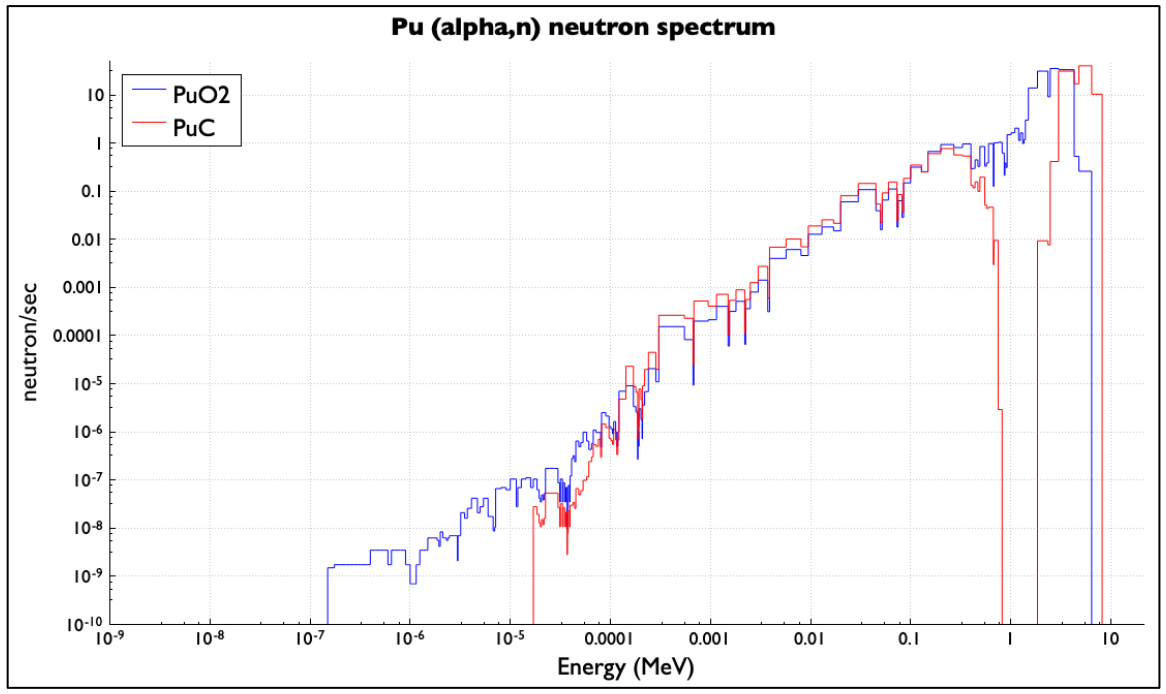
# Problem 4

- What is the PuC source spectra?
  - Sources at 1 year decay

| Source | Intensity (per gram Pu) | Change |
|---|---|---|
| (alpha,n) | 1.042E+02 n/s/g Pu | –26% |
| Spontaneous fission neutron | 2.381E+02 n/s/g Pu | (same) |
| Total neutron | 3. 423E+02 n/s/g Pu | –10% |
| Gamma | 9.872E+08 photons/s/g Pu | (same) |

# Problem 4 (neutron spectra)

## $PuC/PuO_2$ (alpha,n) spectrum




## $PuC/PuO_2$ total neutron spectrum

