

ORIGAMI for rapid fuel depletion calculations

SCALE 6.3.3

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



U.S. DEPARTMENT OF
ENERGY

Learning goals

- **Introduce** users to the ORIGAMI interface for ORIGIN
- **Demonstrate** how it can be used to perform easy spent fuel calculations for simple irradiation histories using the UO2 express form and SCALE's ORIGIN reactor libraries
- **Understand** the similarities and differences in the ORIGAMI workflow compared to using ARP+ORIGIN

Introduction to ORIGAMI



What is ORIGAMI?

More details:

Section 5.4 of the
SCALE manual

- **ORIG**en **AsseM**bly **I**sotopics
- Modern tool for easy depletion/decay calculations for LWR fuel assemblies, using ORIGEN with pre-generated ORIGEN reactor libraries
- Provides detailed nuclide inventories and radiation sources for LWR assemblies at desired complexity levels – 0D, 1D, 2D, 3D
- Calculations are performed based on the **ORIGEN-ARP methodology**
 - ORIGEN to solve the depletion/decay problem
 - ARP to create a specific library by interpolating on available ORIGEN reactor libraries

ORIGAMI enables rapid fuel inventory calculations

- The **ORIGAMI** sequence was released in SCALE 6.2 (2016) as a modern approach to using ORIGEN reactor libraries
- ORNL has developed the ORIGEN+ARP methodology with the ORIGEN code to rapidly generate nuclide inventories using **ORIGEN reactor libraries**
- SCALE's ORIGEN reactor libraries can be used with ORIGAMI by selecting the appropriate "Fuel Type"
- ORIGEN use of fundamental nuclear data allows the following metrics to be calculated from nuclide inventory
 - Mass
 - Decay heat
 - Activity
 - Gamma emission
 - Neutron emission

The screenshot shows the 'origami - UO2 express form (configurable)' window. It contains several input fields with dropdown menus for selection. The 'Fuel Type' field is circled in red and shows 'w17x17'. Below the input fields is a text area displaying the generated XML code for the ORIGAMI sequence. At the bottom right, there are buttons for 'Results', 'Log', 'Template', 'Cancel', and 'OK'.

Field	Value
Title	My Fuel
Fuel Type	w17x17
Uranium (MTU)	1.0
Enrichment (Wt%U235)	4.5
Burnup (MWd/MTU)	40000
Cycles	3
Number of Burnup Interpolations per Cycle	4
Cooling Time (days)	1825
Power History - Percent Up	95
Power History - Average Power (MW/MTU)	40

```
=origami
title="My Fuel"
options{ mtu=1.0 ft71=all}
libs=[ "w17x17" ]
fuelcomp{
  uox(fuel){ enrich=4.5 }
  mix(1){ comps=[ fuel=100 ] }
}
modz=[ 0.7332 ]
pz=[ 1.0 ]
hist{
  cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
```

Why use ORIGAMI?

- Get rapid answers to common questions, such as
What U/Cs/Pu content could I expect in a PWR fuel?
 - a. assuming constant power?
 - b. during a power maneuver?
 - c. after 1 cycle of irradiation?
 - d. after 100 days of cooling?
 - e. after 100 years of cooling?
 - f. at 50 GWd/MTU burnup?
 - g. at 3.55% enrichment?

Each answer requires a <10 second calc. on a single CPU

Why is speed important? All back-end analysis
can use this approach: dry storage casks, on-site storage, discharge inventory analysis, transportation packages

- Up-front work required
 - **Physics analysis** of the reactor system to understand the changes that impact neutron flux spectrum in the fuel (e.g., moderator density in BWR)
 - Run many CPU-hours of coupled transport + depletion cases to generate a database of 1-group cross sections σ which can be interpolated to a specific state of interest (ORIGEN reactor library)
 - These generated libraries can be used later (in ORIGAMI) to determine nuclide inventory and reaction rates:
$$RR(t) = \sigma(t) N(t) \phi(t)$$
 - **Why do it this way?**
If σ is insensitive to decay time, power level, then questions b through g can be answered based on a single TRITON depletion pre-calculation!

ORIGAMI capabilities highlights

With a narrower scope, many convenient features

- Short, concise interface to ORIGEN+ARP for easy spent fuel calculations
- Good default parameter settings for LWR UO_2
- Allows axial and radial variations to be investigated (e.g., axial decay heat in a BWR assembly)
- Produces F71 files automatically
 - Can be readily viewed in Fulcrum
 - *Used to seed an ORIGEN “restart” decay calculation
 - Can be post-processed with OPUS
- Can produce nuclide compositions
 - SCALE StdCmp format for nuclides important to burnup credit (e.g., for use in SCALE criticality models of transportation casks)
 - MCNP material cards format (e.g., for NDA detector models simulations)

ORIGAMI has two modes for user input

UO2 Express form

configurable

scale origami - UO2 express form (configurable)

Title: this-is-my-title

Fuel Type: w17x17

Uranium (MTU): 1.0

Enrichment (Wt%U235): 4.5

Burnup (MWd/MTU): 40000

Cycles: 3

Number of Burnup Interpolations per Cycle: 4

Cooling Time (days): 1825

Results Log Template

Cancel OK

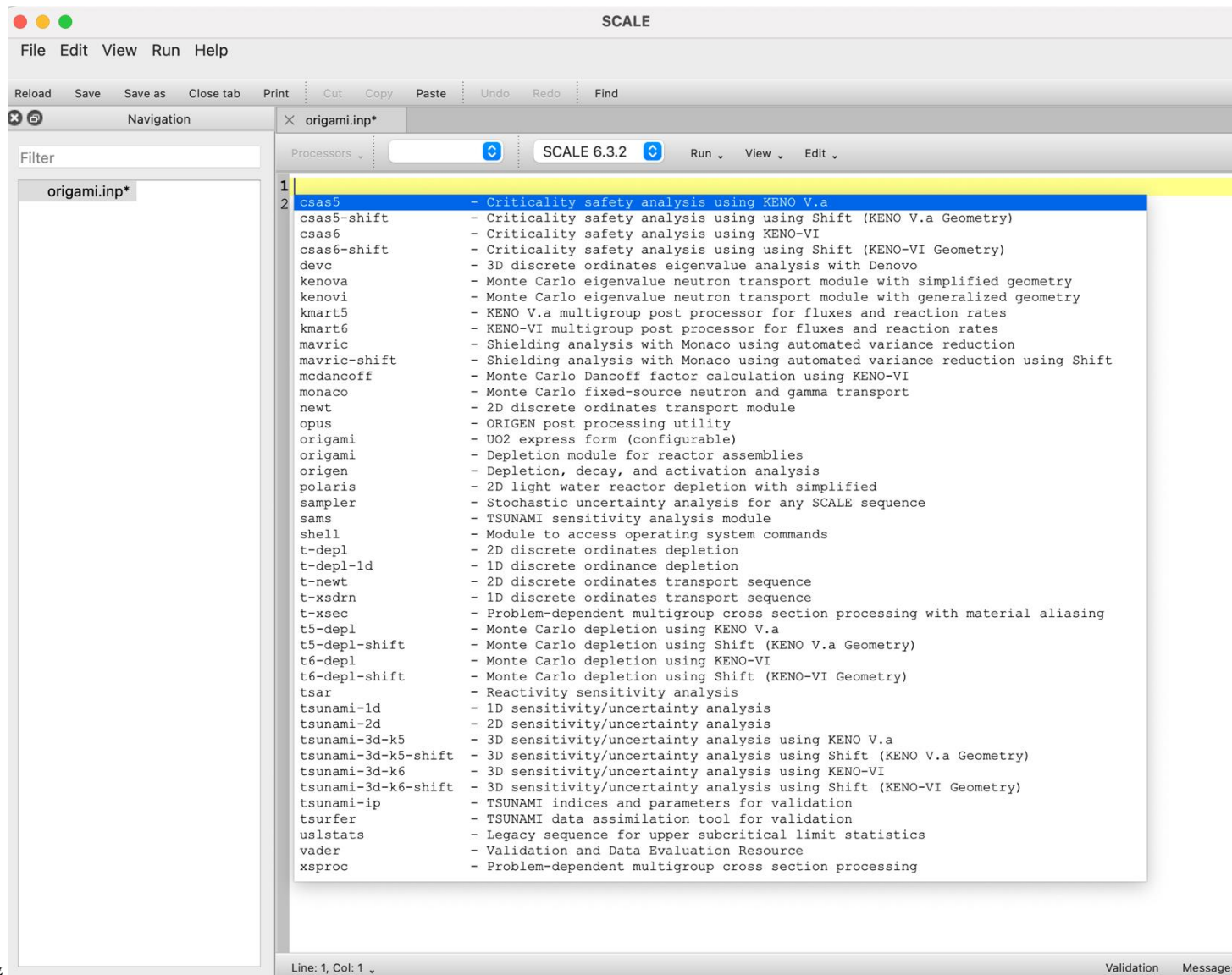
```
=origami
title="this-is-my-title"
options{ mtu=1.0 ft71=all}
libs=[ "w17x17" ]
fuelcomp{
  stdcomp(fuel){
    base=uo2 iso[92234=0.02848 92235=3.2
                 92236=0.01472 92238=96.7568] }
    mix(1){ comps[fuel=100] }
  }
nonfuel=[ cr=3.366 mn=0.1525 fe=6.309 co=0.0302
          ni=2.366 zr=516.3 sn=8.412 gd=2.860 ]
hist[
  cycle{ power=40 burn=284 nlib=4 down=54 }
  cycle{ power=38.6 burn=300 nlib=4 down=28 }
  cycle{ power=25.2 burn=250 nlib=3 down=30 }
]
print{
  nuc {
    sublibs=[ac fp]
    units=[grams moles]
    total=no }
}
```

Detailed input

*keyword-input with the SCALE
Object Notation (SON) syntax*

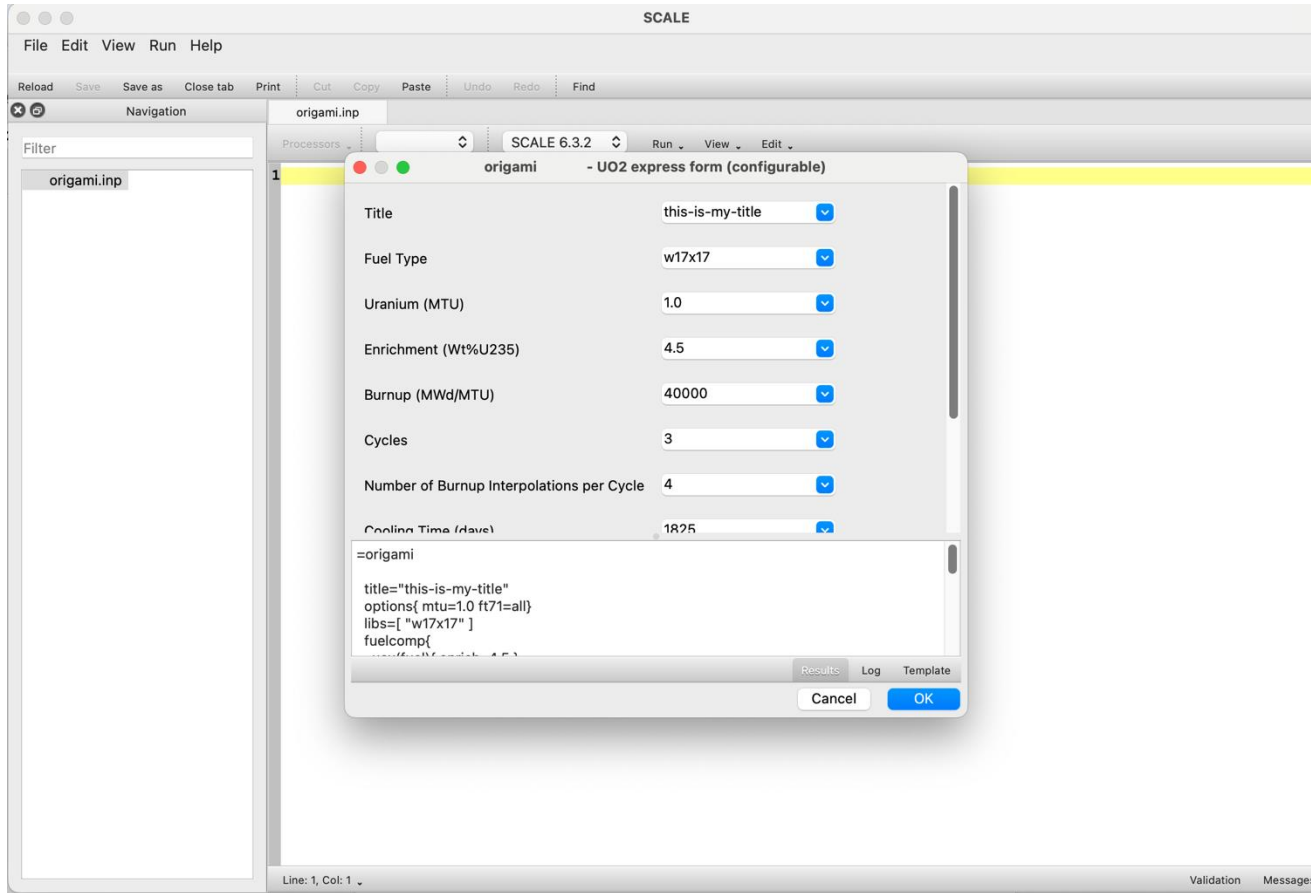
```
=origami
title='fully lumped assembly model'
options{ mtu=0.38 ft71=all }
libs=[ cel4x14 ]
fuelcomp{
  stdcomp(fuel){
    base=uo2 iso[92234=0.02848 92235=3.2
                 92236=0.01472 92238=96.7568] }
    mix(1){ comps[fuel=100] }
  }
nonfuel=[ cr=3.366 mn=0.1525 fe=6.309 co=0.0302
          ni=2.366 zr=516.3 sn=8.412 gd=2.860 ]
hist[
  cycle{ power=40 burn=284 nlib=4 down=54 }
  cycle{ power=38.6 burn=300 nlib=4 down=28 }
  cycle{ power=25.2 burn=250 nlib=3 down=30 }
]
print{
  nuc {
    sublibs=[ac fp]
    units=[grams moles]
    total=no }
}
end
```

ORIGAMI “Express Form” – How to start?



- Create empty file (origami.inp)
- CTRL+SPACE inside empty file
- Choose "origami – UO2 express form (configurable)"

ORIGAMI "Express Form" – How to use?



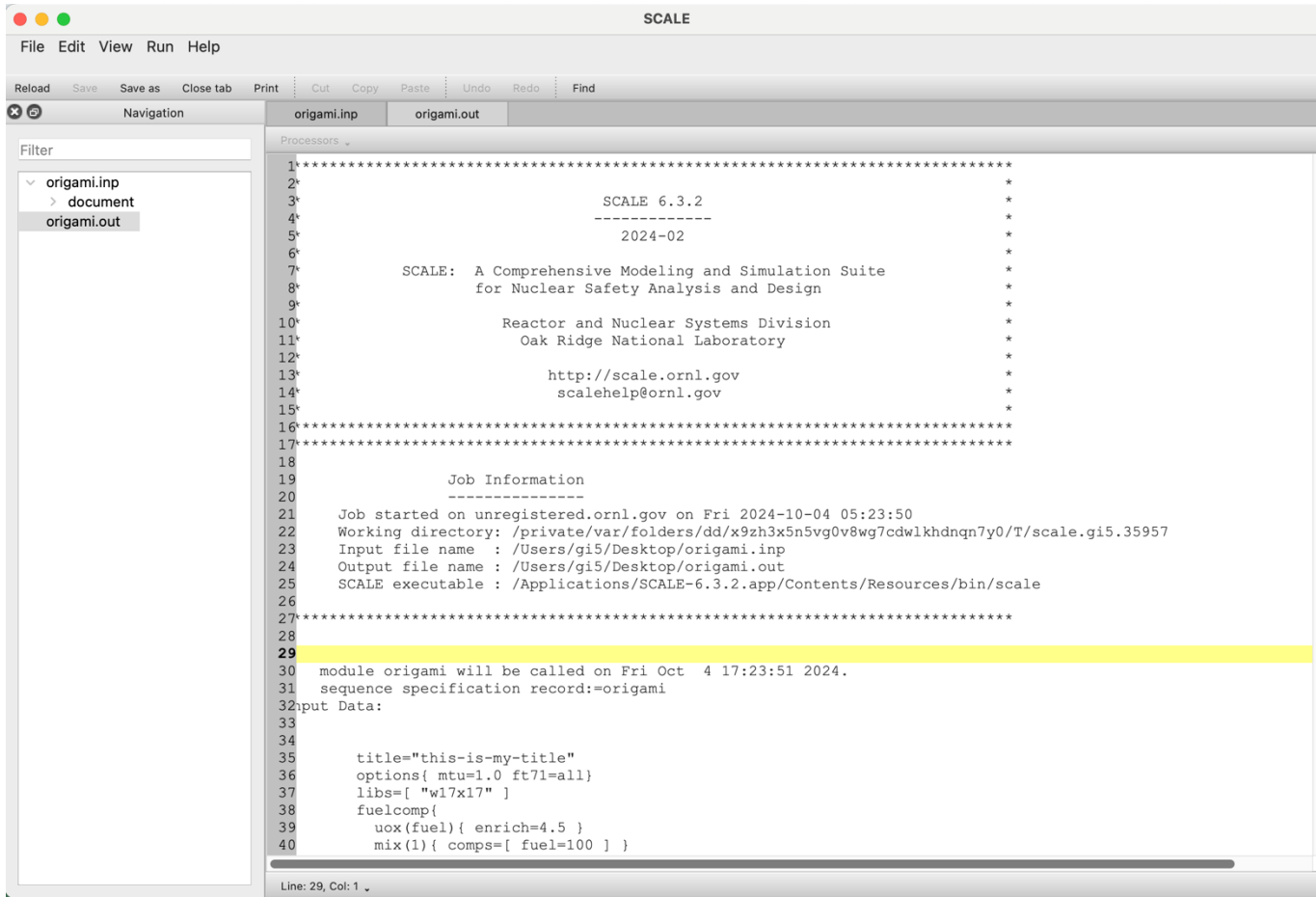
- Configurable window appears
- Click "OK"

ORIGAMI "Express Form" – Input behind the form

```
1 =origami
2
3 title="this-is-my-title"
4 options{ mtu=1.0 ft71=all}
5 libs[ "w17x17" ]
6 fuelcomp{
7   uox(fuel){ enrich=4.5 }
8   mix(1){ comps=[ fuel=100 ] }
9 }
10 modz=[ 0.7332 ]
11 pz=[ 1.0 ]
12 hist[
13   cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
14   cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
15   cycle{ power=40 burn=333.33 nlib=4 down=0 }
16   cycle{ down=1825 }
17 ]
18
19 end
20
21
22
```

- This is a valid ORIGAMI input!
- Click Run

ORIGAMI "Express Form" – Execution



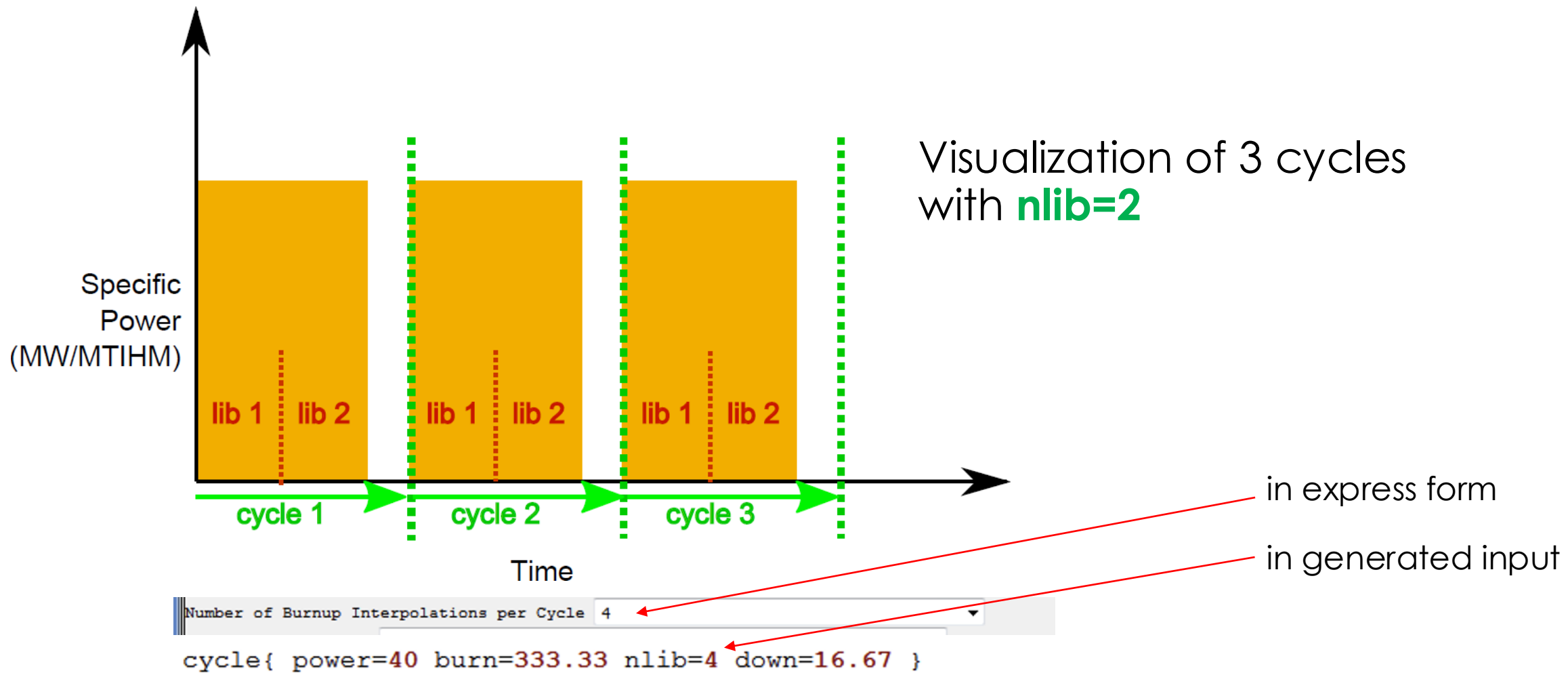
```
1 *****
2
3          SCALE 6.3.2
4          -----
5          2024-02
6
7          SCALE: A Comprehensive Modeling and Simulation Suite
8          for Nuclear Safety Analysis and Design
9
10         Reactor and Nuclear Systems Division
11         Oak Ridge National Laboratory
12
13         http://scale.ornl.gov
14         scalehelp@ornl.gov
15 *****
16 *****
17 *****
18
19         Job Information
20         -----
21         Job started on unregistered.ornl.gov on Fri 2024-10-04 05:23:50
22         Working directory: /private/var/folders/dd/x9zh3x5n5vg0v8wg7cdwlkhdnqn7y0/T/scale.gi5.35957
23         Input file name : /Users/gi5/Desktop/origami.inp
24         Output file name : /Users/gi5/Desktop/origami.out
25         SCALE executable : /Applications/SCALE-6.3.2.app/Contents/Resources/bin/scale
26 *****
27 *****
28
29 module origami will be called on Fri Oct 4 17:23:51 2024.
30 sequence specification record=:origami
31put Data:
32
33
34         title="this-is-my-title"
35         options{ mtu=1.0 ft7l=all}
36         libs= [ "w17x17" ]
37         fuelcomp{
38             uox(fuel){ enrich=4.5 }
39             mix(1){ comps=[ fuel=100 ] }
40
```

ORIGAMI takes a little while...

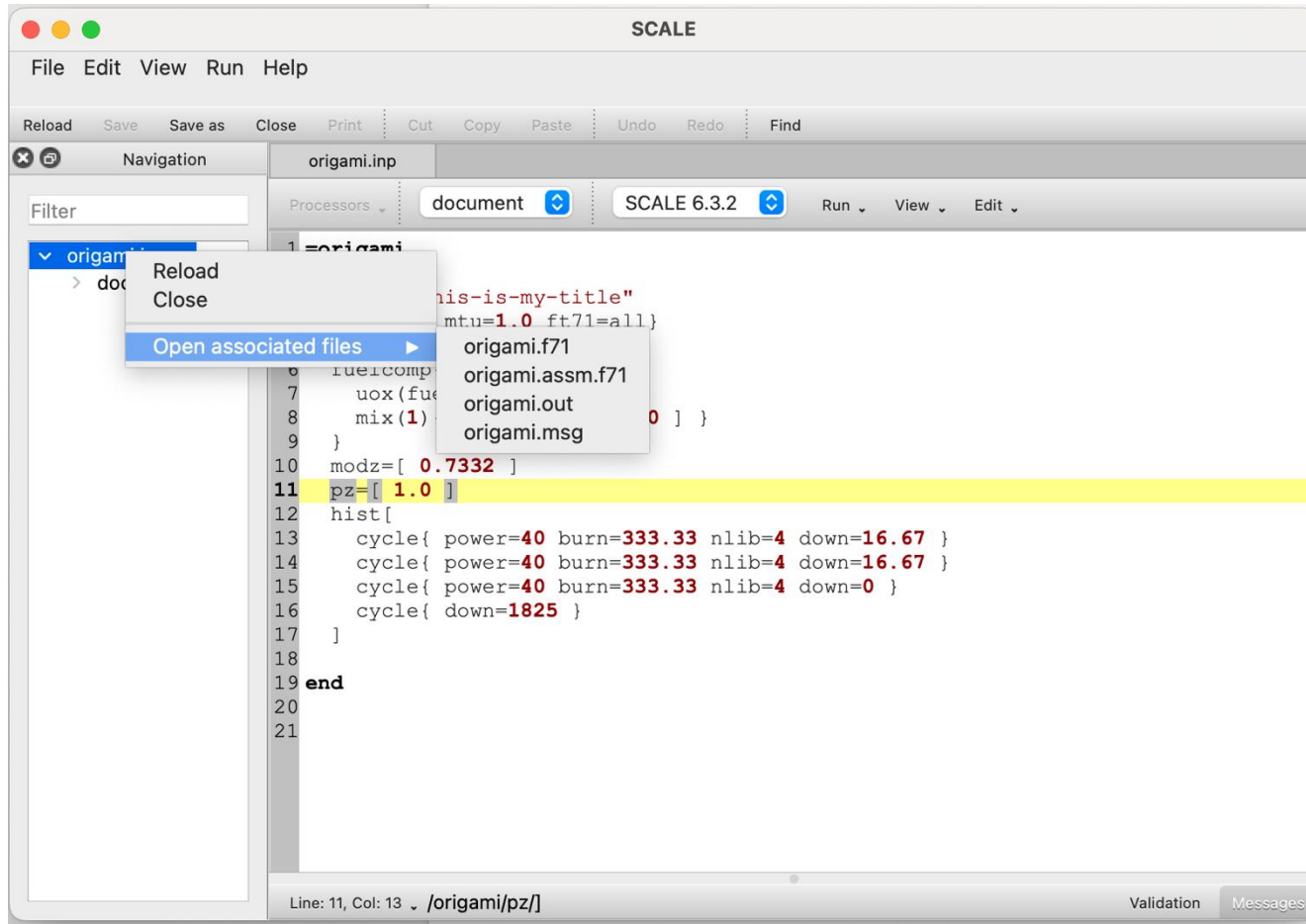
- Performs 10 substeps by default per depletion / decay step
- $nlib=4$ would require 4 cases in ORIGIN
- The input on previous page involves $(4+4+4+1)*10 = 130$ depletion solves

Controlling cross section interpolation

nlib (libraries per cycle)

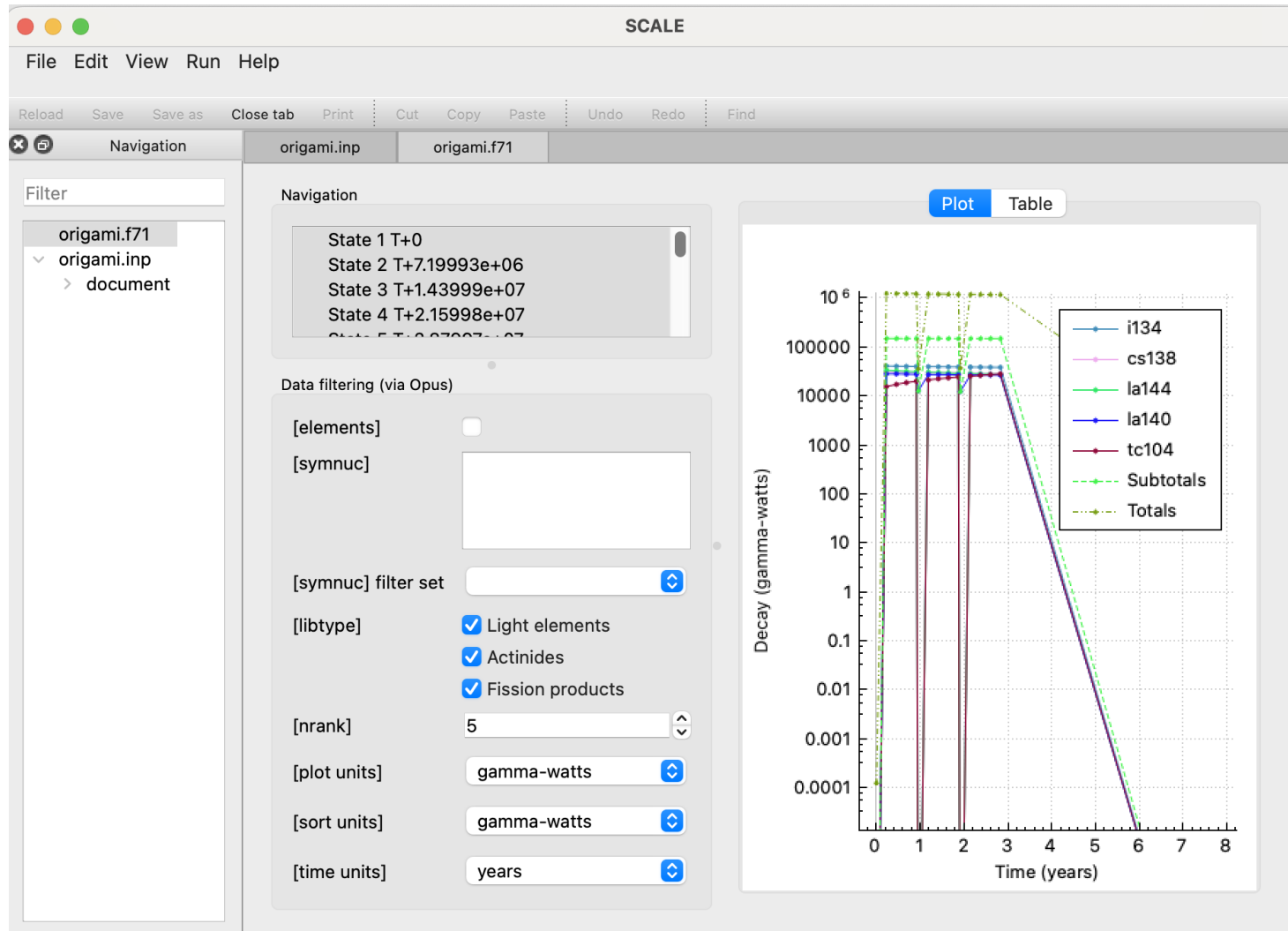


ORIGAMI "Express Form" – F71 files produced



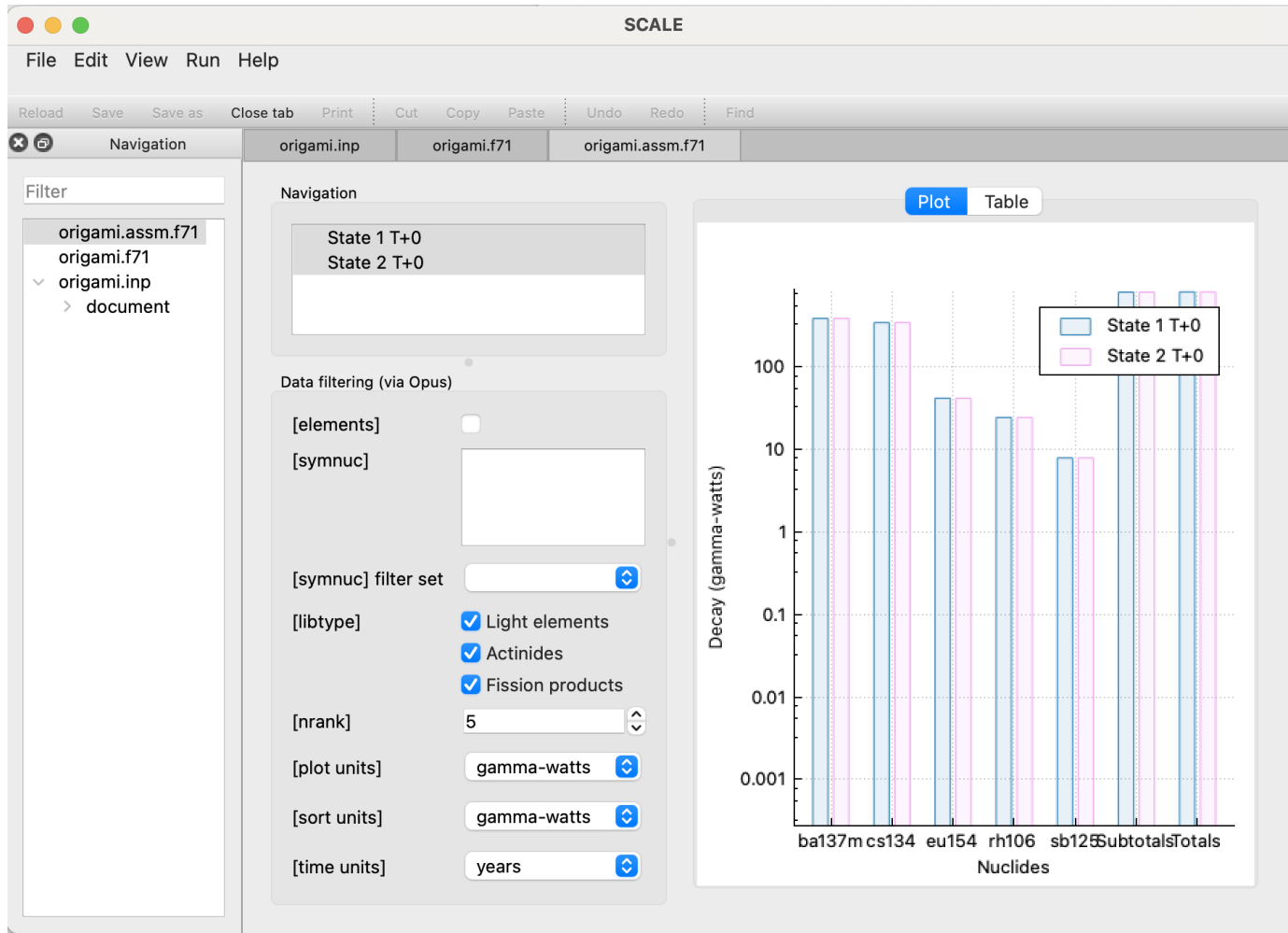
- ORIGAMI produces by default two F71 files:
 - $\text{\textcolor{green}{\${basename}.f71}}$
 - $\text{\textcolor{green}{\${basename}.asm.f71}}$
- How to open .f71
 - Right-click on origami.inp in navigation pane
 - Open associated files
 - Select origami.f71

ORIGAMI "Express Form" – F71 files coverage



- `${basename}.f71`
- Has data for all times in the problem

ORIGAMI "Express Form" – F71 files coverage



- `${basename}.asm.f71` has data only for final time in the problem
- The final time in this case corresponds to the end of the cooling time
- Because this is a 0D problem, both States 1 and 2 contain the same data
 - (For 1D, 2D, and 3D problems, the available states contain different data. This will be covered in a later ORIGAMI module.)

ORIGAMI-UOX Express: Ringhals-2 Sample F32

- Open the ORIGAMI UOX Express form
- Populate the **average** cycle history parameters and run
- Use the w15x15 library

Average cycle history

- 3.197 wt. % enrichment
- 436.99 kgU
- 50,962 MWd/MTU burnup
- 5,860 days cooling time
- **Average power:** 27.91 MW/MTU

For reference: detailed cycle history

Cycle	Burnup (MWd/MTU)	Cycle length (d)	Down time (d)	Avg. power (MW/MTU)
1	10,553	290	80	36.39
2	21,162	317	84	33.47
3	29,553	273	91	30.74
4	37,314	320	161	24.25
5	43,943	297	71	22.32
6	50,962	329	—	21.33

How accurate is ORIGAMI for calculating decay heat?

Method	Decay power (W)	C/E – 1 (%)	
Measured Power	692	-	
ORIGAMI (average history)	720.8	4.2	Generic reactor data library (w15x15)
ORIGAMI (detailed history)	715.3	3.4	
ORIGAMI (avg. with custom library)	706.7	2.1	Design-specific (F32) reactor data library
ORIGAMI (detailed with custom library)	701.5	1.4	

Notes on the axial effect

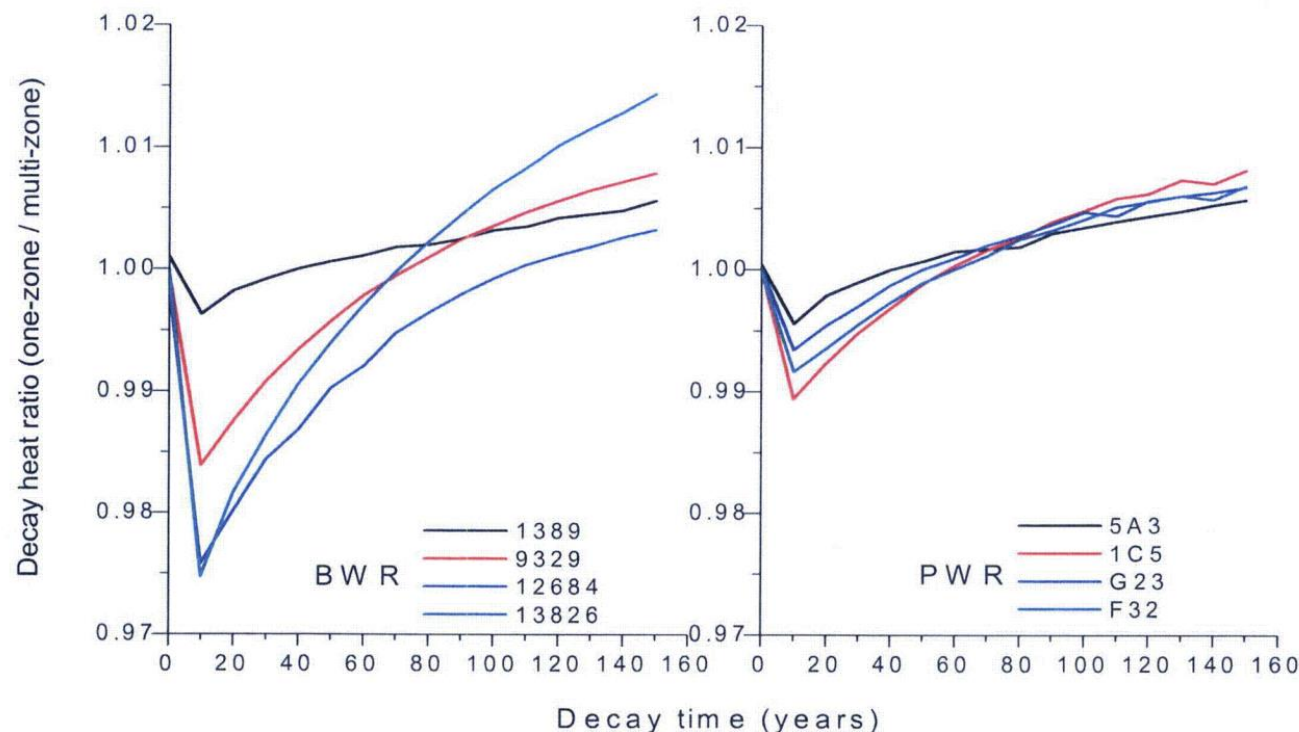
- Depletion is non-linear with respect to cross sections and decay constants, i.e. expected/average values

$$\langle e^{-\lambda t} \rangle \neq e^{-\langle \lambda \rangle t}$$

$$\langle \lambda t + c \rangle = \langle \lambda \rangle t + \langle c \rangle$$

- Therefore, assembly-average decay heat calculated from a single “average” ORIGAMI simulation is **NOT** the same as the average of either
 - Axial-dependent simulation
 - Pin-by-pin simulation
- Luckily the effect appears to be 1-2%

Although detailed axial calculations may not be practical for routine decay heat analysis, and this effect is relatively small (the magnitude in many cases being smaller than the experimental uncertainties), the user needs to be aware of the bias introduced in assembly models that use average data instead of detailed axial data. Importantly, it can be seen from Figure 6.17 that there is no single average value for the assembly burnup (or coolant density) that will yield results that are equivalent to a detailed axial calculation.



Calculated decay heat of selected Swedish BWR and PWR assemblies: Decay heat ratio of one-zone model vs. multi-zone representation of the assembly*

Steps to fast inventory calcs with ORIGAMI

Developer Steps

1. Sensitivity studies are performed with TRITON (or Polaris if LWR) to understand different conditions important for representing the neutron energy spectrum as a function of burnup/time in a particular fuel system.
Examples for LWRs:
 - **UO₂ PWR**: assembly design, enrichment, burnup
 - **UO₂ BWR**: assembly design, enrichment, void fraction, burnup
 - **MOX PWR/BWR**: same as UO₂ but Pu fraction + Pu fissile fraction instead of enrichment
2. ORIGEN Reactor Libraries are generated on a grid, e.g. for a specific assembly design, 10 enrichments, 3 moderator densities, 30 burnups → 10x3x30 grid of 1-group cross section data
3. ORIGEN Reactor Libraries are “registered” in a special text file called **arpdata.txt**

Runtime Steps

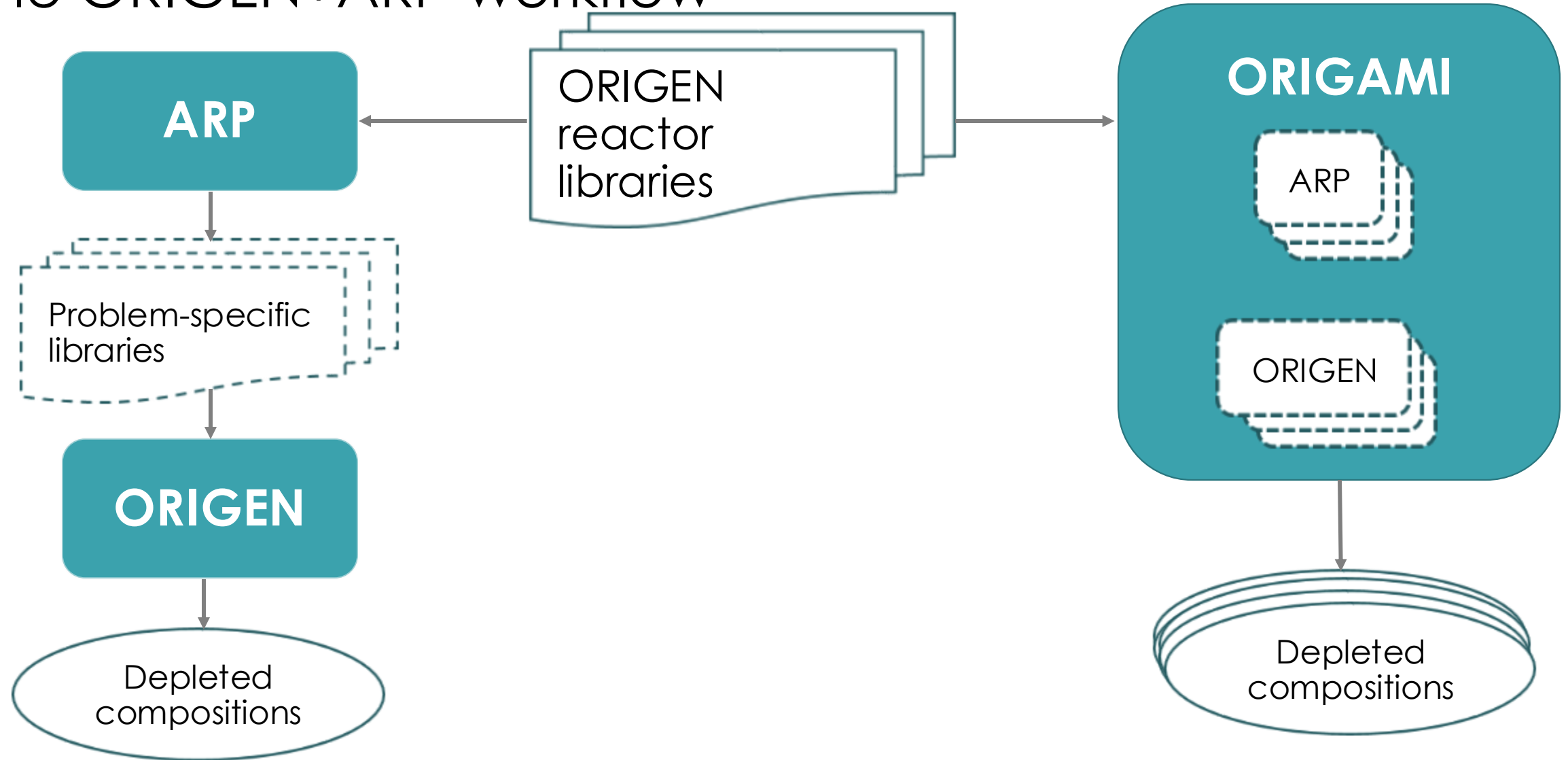
4. Based on user input, ORIGAMI finds the specified ORIGEN reactor library and interpolates non-burnup dimensions, e.g. 4.3% enrichment, 0.735 g/cc moderator density. The result is a burnup-dependent set of 1-group cross section data, **A(bu)**
5. ORIGAMI then interpolates this **A(bu)** data on burnup according to the user-provided power, **P(t)** and repeatedly requests the ORIGEN kernel solver to compute the end-of-step nuclide inventory **n(bu_{i+1})** based on **n(bu_i)**, **A(bu_{i+1/2})**
6. Finally, ORIGAMI writes the f71 with requested nuclide inventory results

ORIGAMI vs ORIGEN+ARP

Selecting the right tool for your workflow



ORIGAMI is a more specialized and compact version of the ORIGIN+ARP workflow



Which tool is right for me?

ARP + ORIGEN

- Depletion of one or a few regions
- Complex irradiation histories and/or advanced processing
- Any fuel type (incl. MOX, CANDU, etc.)

ORIGAMI

- Depletion of many regions / compositions (several depletion problems)
- Relatively simple irradiation histories
- Enables follow-on analysis using tools such as KENO and MCNP
- UOX only (presently)

Tutorial:

Nuclide inventory calculation with ORIGAMI UO₂ Express



Problem: Nuclide inventory in typical PWR fuel at 10 years cooling

- Required results:

- Concentrations in g/MTU of important actinides and fission products (see table with nuclide list)
- Total amount of Pu in g/MTU
- Weight percentage of Pu-238, Pu-239, Pu-240 and Pu-241 relative to total Pu

- Input assumptions:

- W17x17 reactor library
- Initial fuel enrichment: 5.0% ^{235}U
- Cooling time: 10 years
- Number of reactor cycles: 3
- Keep all other input default values in the form

Nuclide	g/MTU
U-235	
Pu-238	
Pu-239	
Pu-240	
Pu-241	
Am-241	
Cm-244	
Cs-134	
Cs-137	
Sm-149	
Gd-155	

Solution: Nuclide inventory in typical PWR fuel at 10 years cooling

Nuclide	g/MTU
U-235	16413.5
Pu-238	193.6
Pu-239	6504.6
Pu-240	2266.5
Pu-241	908.1
Am-241	603.2
Cm-244	18.3
Cs-134	5.1
Cs-137	1170.6
Sm-149	4.7
Gd-155	7.0

Nuclide	wt% Pu
Pu-238	1.9
Pu-239	63.0
Pu-240	22.0
Pu-241	8.8
Pu-242	4.3

```
=origami

  title="PWR typical inventory at 10-yr cooling"
  options{ mtu=1.0 ft71=all}
  libs=[ "w17x17" ]
  fuelcomp{
    uox(fuel){ enrich=3.5 }
    mix(1){ comps=[ fuel=100 ] }
  }
  modz=[ 0.7332 ]
  pz=[ 1.0 ]
  hist[
    cycle{ power=32 burn=468.75 nlib=4 down=46.88 }
    cycle{ power=32 burn=468.75 nlib=4 down=46.88 }
    cycle{ power=32 burn=468.75 nlib=4 down=0 }
    cycle{ down=3652.5 }
  ]
end
```

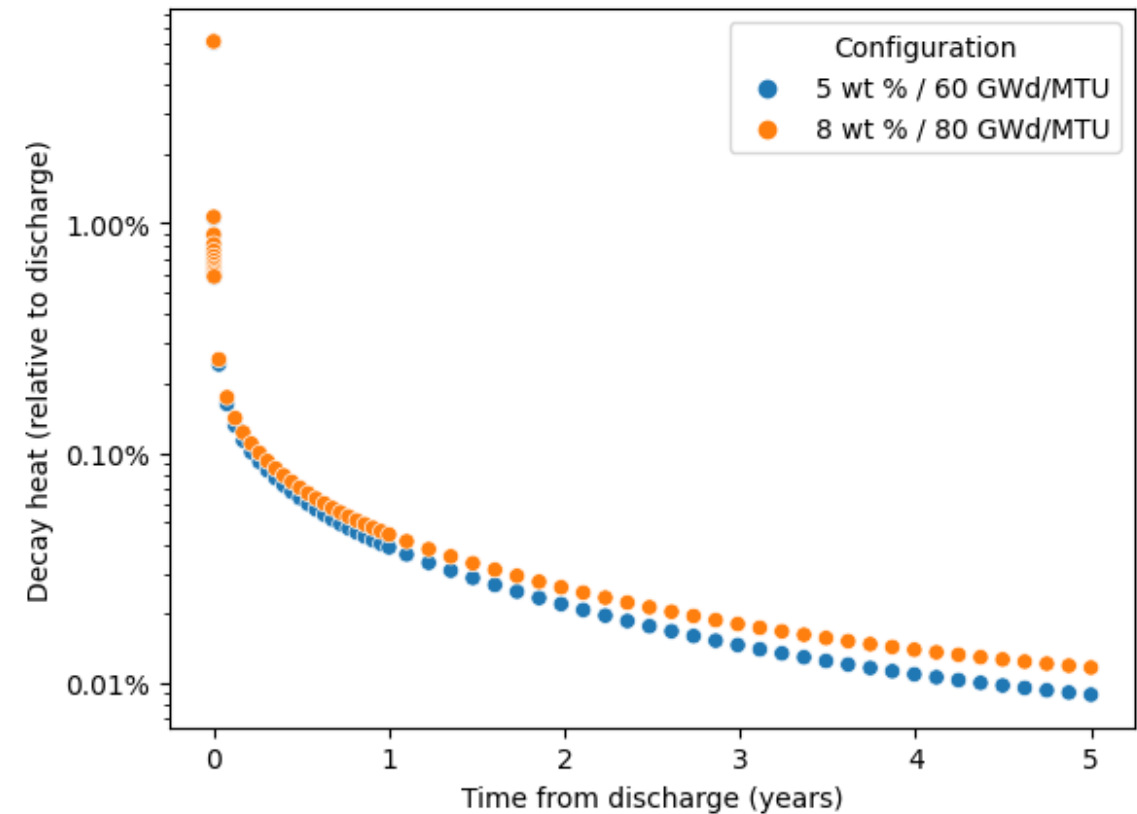
Tutorial:

Compare decay heat of LEU to LEU+/HBU



Goals

- With ORIGAMI, generate spent fuel inventory for W17x17 for two variants producing two f71 files
 - 40 MW/MTU, 5 wt% @60 GWd/MTU, w5-60.discharge.f71
 - 40 MW/MTU, 8 wt% @80 GWd/MTU, w8-80.discharge.f71
- Plot decay heat up to five years as % of full power
 - Import each f71 at discharge into a simple ORIGEN decay input producing two more f71 files
 - w5-60.decay.f71
 - w8-80.decay.f71
 - Load these f71 files in Fulcrum, change units to decay heat, copy/paste data from Table view into spreadsheet, reformat as needed



Workshop:

Fast spent fuel calculations
with ORIGAMI UO₂ Express

**Compare calculated key spent
fuel metrics (activity, decay heat,
nuclide concentrations) of typical
PWR and BWR typical fuel**



Description of problems



Problem 1a: Decay heat and activity of PWR spent fuel (W17x17) at 10-year cooling time

- **Input data:**

- 3.5% U-235 enrichment
- 3 irradiation cycles
- 45,000 MWd/MTU burnup
- 10-year cooling after discharge
- 90% uptime
- 32 MW/MTU average power

- **Results required:**

- Total decay heat (W/MTU)
- Gamma component of decay heat (W/MTU)
- Activity (Bq/MTU)

- **Strategy:**

- Use ORIGAMI Express with “w17x17” reactor library; use defaults for non-specified input values
- Can extract results using Fulcrum with F71 file generated by ORIGAMI, or by editing the generated ORIGAMI input to add OPUS blocks at the end of the input and rerun the case

Problem 1b: Decay heat and activity of BWR spent fuel (GE 10x10) at 10-year cooling time

- **Input data:**

- 3.5% U-235 enrichment
- 3 irradiation cycles
- 45,000 MWd/MTU burnup
- 10-year cooling after discharge
- 90% uptime
- 25 MW/MTU average power
- 0.45 g/cm³ moderator density

- **Results required:**

- Total decay heat (W/MTU)
- Gamma component of decay heat (W/MTU)
- Activity (Bq/MTU)

- **Strategy:**

- Use ORIGAMI Express with “ge10x10-8” reactor library; use defaults for non-specified input values
- Can extract results using Fulcrum with F71 file generated by ORIGAMI, or by editing the generated ORIGAMI input to add OPUS blocks at the end of the input and rerun the case

Problem 1c: Compare decay heat and activity between typical PWR and BWR spent fuel at 45 GWd/MTU burnup and 10-year cooling

- **Input data:**
 - PWR assembly initial U load: 460 kg
 - BWR assembly initial U load: 180 kg
- **Results required:**
 - Total decay heat: difference in Watts and % (per MTU, and per assembly)
 - Gamma decay heat: difference in Watts and % (per MTU, and per assembly)
 - Activity: difference in Bq and % (per MTU, and per assembly)
- **Strategy:**
 - Use inputs and results from Problem 1a and Problem 1b

Problem 2a:

Nuclide inventories in once-, twice-, thrice-burned PWR fuel

Calculate mass data of selected U, Pu, Nd isotopes for a typical PWR W17x17 fuel irradiated in the reactor for 1, 2, and 3 cycles

- **Input data:**

- Enrichment 4.5 wt% ^{235}U
- Power 30 MW/MTU
- Reactor library w17x17
- Burnup 15, 30, and 45 GWd/MTU for 1-, 2-, and 3-cycle irradiation (3 cases)
- Keep default values for everything else

- **Strategy:**

- Start with ORIGAMI express form
- Can extract the data using Fulcrum for the F71 file generated by ORIGAMI, or by editing the generated input to add OPUS blocks at the end of the input and rerun the case

- **Results required:**

- U-235 mass (kg/MTU) at discharge
- Percentage of initial U-235 that was burned during the irradiation
- Pu mass (kg/MTU) at 0- and 10-year cooling time
- Percentage of fissile Pu at 0- and 10-year cooling times ($\text{Pu-239} + \text{Pu-241}$)/Pu
- Nd-148 mass (g/MTU) at discharge [note this is a burnup indicator fission product]

Problem 2b:

Nuclide inventories in once-, twice-, thrice-burned BWR fuel

Calculate mass data of selected U, Pu, Nd isotopes for a typical BWR 8x8 fuel irradiated in the reactor for 1, 2, and 3 cycles

- **Input data:**

- Enrichment 4.5 wt% ^{235}U
- Power 30 MW/MTU
- Moderator density 0.45 g/cm³
- Burnup 15, 30, and 45 GWd/MTU for 1-, 2-, and 3-cycles irradiation (3 cases)
- Assembly type ge8x8-1
- Keep default values for everything else

- **Strategy:**

- Start with ORIGAMI express form
- Can extract data using Fulcrum with F71 file generated by ORIGAMI, or by editing the generated input to add OPUS blocks at the end of the input and rerun the case

- **Results required:**

- U-235 mass (kg/MTU) at discharge
- Percentage of initial U-235 that was burned during the irradiation
- Pu mass (kg/MTU) at 0- and 10-year cooling time
- Percentage of fissile Pu at 0- and 10-year cooling time (Pu-239+Pu-241)/Pu
- Nd-148 mass (g/MTU) at discharge [note this is a burnup indicator fission product]

Problem 2c:

Nuclide inventories in once-, twice-, thrice-burned LWR fuel

Compare mass data of selected U, Pu, Nd isotopes for a typical PWR 17x17 fuel and a BWR 8x8 fuel irradiated in the reactor for 1, 2, and 3 cycles

- **Input data:**
 - For PWR fuel, as for Problem 2a
 - For BWR fuel, as for Problem 2b
- **Strategy:**
 - Use inputs or results from Problem 2a and Problem 2b
- **Results required:**
 - Compare U-235 mass (kg/MTU) at discharge
 - Compare percentage of initial U-235 that was burned during the irradiation
 - Compare Pu mass (kg/MTU) at 10-year cooling time
 - Compare percentage of fissile Pu at 10-year cooling time $(\text{Pu-239} + \text{Pu-241}) / \text{Pu}$
 - Compare Nd-148 mass (g/MTU) at discharge

Solutions



Problem 1a: Decay heat and activity of PWR spent fuel (W17x17) at 10-year cooling time

- Total decay heat:
1.651e+03 W/MTU at 10-year cooling
- Gamma component of decay heat:
4.934e+02 W/MTU (29.9% of total) at 10-year cooling
- Activity:
1.941E+16 Bq at 10-year cooling

```
=origami
  title="PWR decay heat"
  options{ mtu=1.0 ft71=LAST}
  libs=[ "w17x17" ]
  fuelcomp{
    uox(fuel){ enrich=3.5 }
    mix(1){ comps=[ fuel=100 ] }
  }
  modz=[ 0.7332 ]
  pz=[ 1.0 ]
  hist[
    cycle{ power=32 burn=468.75 nlib=4 down=46.88 }
    cycle{ power=32 burn=468.75 nlib=4 down=46.88 }
    cycle{ power=32 burn=468.75 nlib=4 down=0 }
    cycle{ down=3652.5 }
  ]
end
=opus
  data="${BASENAME}.assm.ft71"
  units=watts
end
=opus
  data="problem1a_pwr_dh.assm.ft71"
  units=gamwatts
end
=opus
  data="problem1a_pwr_dh.assm.ft71"
  units=becquerels
end
```

Problem 1b: Decay heat and activity of BWR spent fuel (GE 10x10) at 10-year cooling time

- Total decay heat:
1.586e+03 W/MTU at 10-year cooling
- Gamma component of decay heat:
4.736e+02 W/MTU (29.9% of total) at 10-year cooling
- Activity:
1.843E+16 Bq at 10-year cooling

```
=origami
  title="BWR decay heat"
  options{ mtu=1.0 ft71=last}
  libs=[ "ge10x10-8" ]
  fuelcomp{
    uox(fuel){ enrich=3.5 }
    mix(1){ comps=[ fuel=100 ] }
  }
  modz=[ 0.45 ]
  pz=[ 1.0 ]
  hist[
    cycle{ power=25 burn=600.00 nlib=4 down=60.00 }
    cycle{ power=25 burn=600.00 nlib=4 down=60.00 }
    cycle{ power=25 burn=600.00 nlib=4 down=0 }
    cycle{ down=3652.5 }
  ]
end
=opus
  data="${BASENAME}.assm.f71"
  units=watts
end
=opus
  data="${BASENAME}.assm.f71"
  units=gamwatts
end
=opus
  data="${BASENAME}.assm.f71"
  units=becquerels
end
```

Problem 1c: Compare decay heat and activity between typical PWR and BWR spent fuel at 35 GWd/MTU burnup and 10-year cooling

Comparison for 1 MTU

	PWR	BWR	abs. difference	rel. difference
Mass (kg initial U)	1000	1000		
Decay heat (W)	1650.6	1586.0	-64.6	-3.9%
Gamma component of decay heat (W)	493.4	473.6	-19.8	-4.0%
Gamma component of decay heat (%)	29.9%	29.9%		
Activity (Bq)	1.941E+16	1.843E+16	-9.84E+14	-5.1%

Comparison for 1 assembly

	PWR	BWR	abs. difference	rel. difference
Mass (kg initial U)	460.0	180.0		
Decay heat (W)	759.3	285.5	-473.8	-62.4%
Gamma component of decay heat (W)	227.0	85.2	-141.7	-62.4%
Gamma component of decay heat (%)	29.9%	29.9%		
Activity (Bq)	8.93E+15	3.32E+15	-5.61E+15	-62.9%

Problem 2a: U-235 and Nd-148 mass at discharge (PWR)

Discharge burnup	Cooling time = 10-year	
(GWd/MTU)	U-235	Nd-148
15	29.63 kg/MTU (34.2% depleted)	169.8 g/MTU
30	18.63 kg/MTU (58.6% depleted)	336.7 g/MTU
45	10.89 kg/MTU (75.9% depleted)	501.4 g/MTU

Problem 2a: Pu mass and percentage fissile (PWR)

Discharge burnup	Cooling time (years)	
(GWd/MTU)	0	10
15	5.60 kg/MTU (85.3% fissile)	5.51 kg/MTU (85.1% fissile)
30	9.14 kg/MTU (76.6% fissile)	8.78 kg/MTU (75.6% fissile)
45	11.49 kg/MTU (69.0% fissile)	10.94 kg/MTU (67.3% fissile)

Problem 2b: U-235 and Nd-148 mass at discharge (BWR)

Discharge burnup	Cooling time = 10-year	
(GWd/MTU)	U-235	Nd-148
15	29.45 kg/MTU (34.6% depleted)	170.2 g/MTU
30	18.03 kg/MTU (59.9% depleted)	337.7 g/MTU
45	9.99 kg/MTU (77.8% depleted)	503.0 g/MTU

Problem 2b: Pu mass and percentage fissile (BWR)

Discharge burnup	Cooling time (years)	
(GWd/MTU)	0	10
15	5.05 kg/MTU (84.5% fissile)	4.97 kg/MTU (84.3% fissile)
30	8.13 kg/MTU (74.4% fissile)	7.84 kg/MTU (73.5% fissile)
45	10.15 kg/MTU (65.5 fissile)	9.72 kg/MTU (63.8% fissile)

Problem 2c: U-235 and Nd-148 mass at discharge (PWR vs BWR)

Discharge burnup	Ratio (PWR vs BWR) at 10-year cooling	
(GWd/MTU)	U-235	Nd-148
15	1.006	0.998
30	1.033	0.997
45	1.090	0.997

Problem 2c: Pu mass and percentage fissile (PWR vs BWR)

Discharge burnup	Ratio (PWR vs BWR) at 10-year cooling	
(GWd/MTU)	Pu	Pu % fissile content
15	1.109	1.010
30	1.120	1.029
45	1.125	1.055