

ORIGAMI Detailed Input for Rapid Spent Fuel 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



Gain experience creating from scratch an ORIGAMI input for fuel assembly depletion calculation



Become familiarized with the full set of ORIGAMI features, including axial and radial moderator density and power gradients

Create ORIGAMI input from scratch



ORIGAMI input from scratch – Create new input file

1	
csas5	- Criticality safety analysis using KENO V.a
csas5-shift	- Criticality safety analysis using Shift (KENO V.a Geometry)
csas6	- Criticality safety analysis using KENO-VI
csas6-shift	- Criticality safety analysis using Shift (KENO-VI Geometry)
devc	- 3D discrete ordinates eigenvalue analysis with Denovo
kenova	- Monte Carlo eigenvalue neutron transport module with simplified geometry
kenovi	- Monte Carlo eigenvalue neutron transport module with generalized geometry
kmart5	- KENO V.a multigroup post processor for fluxes and reaction rates
kmart6	- KENO-VI multigroup post processor for fluxes and reaction rates
mavric	- Shielding analysis with Monaco using automated variance reduction
mavric-shift	- Shielding analysis with Monaco using automated variance reduction using Shift
mcdancoff	- Monte Carlo Dancoff factor calculation using KENO-VI
monaco	- Monte Carlo fixed-source neutron and gamma transport
newt	- 2D discrete ordinates transport module
opus	- ORIGEN post processing utility
origami	- UO2 express form (configurable)
origami	- Depletion module for reactor assemblies
origen	- Depletion, decay, and activation analysis
polaris	- 2D light water reactor depletion with simplified
sampler	- Stochastic uncertainty analysis for any SCALE sequence
sams	- TSUNAMI sensitivity analysis module
shell	- Module to access operating system commands
t-depl	- 2D discrete ordinates depletion
t-depl-1d	- 1D discrete ordinance depletion
t-newt	- 2D discrete ordinates transport sequence
t-xsdn	- 1D discrete ordinates transport sequence
t-xsec	- Problem-dependent multigroup cross section processing with material aliasing
t5-depl	- Monte Carlo depletion using KENO V.a
t5-depl-shift	- Monte Carlo depletion using Shift (KENO V.a Geometry)
t6-depl	- Monte Carlo depletion using KENO-VI
t6-depl-shift	- Monte Carlo depletion using Shift (KENO-VI Geometry)
tsar	- Reactivity sensitivity analysis
tsunami-1d	- 1D sensitivity/uncertainty analysis
tsunami-2d	- 2D sensitivity/uncertainty analysis
tsunami-3d-k5	- 3D sensitivity/uncertainty analysis using KENO V.a
tsunami-3d-k5-shift	- 3D sensitivity/uncertainty analysis using Shift (KENO V.a Geometry)
tsunami-3d-k6	- 3D sensitivity/uncertainty analysis using KENO-VI
tsunami-3d-k6-shift	- 3D sensitivity/uncertainty analysis using Shift (KENO-VI Geometry)
tsunami-ip	- TSUNAMI indices and parameters for validation
tsurfer	- TSUNAMI data assimilation tool for validation
uslstats	- Legacy sequence for upper subcritical limit statistics
vader	- Validation and Data Evaluation Resource
xsproc	- Problem-dependent multigroup cross section processing

- CTRL + SPACE
- Select **origami** (not UO2 express form)
- **Has autocomplete capability**

ORIGAMI input from scratch – Select ORIGIN reactor library

```
1 =origami
2
3 title origami components here
4 prefix
5 asmid
6 options
7 print
8 srcopt
9 fuelcomp
10 libs
11 compmap
12 libmap
13 pxy
14 pz
15 meshz
16 modz
17 nuccomp
18 ggrp
19 ngrp
20 nonfuel
21 hist
```

- Put marker inside of **origami** sequence
- CTRL + SPACE
- Select **libs** (keep default for specs)

Note: default reactor library is **w17x17** for Westinghouse 17×17 PWR fuel assembly

ORIGAMI input from scratch – Input fuel composition

```
1 =origami
2 libs=[ w17x17 ]
3 fuelcomp{
4
5 uox
6 stdcomp
7 mix
8 end
9
```

origami components here

- Put marker inside of **origami** sequence
- CTRL + SPACE
- Select **fuelcomp**
- Put marker inside of **fuelcomp** sequence
- CTRL + SPACE
- Select **uox** (keep default for specs)

Note: default enrichment value for uox is 5% u235

ORIGAMI input from scratch – Input fuel mixture

- Put marker inside of **fuelcomp** sequence
- CTRL + SPACE
- Select **mix** (keep default specs)

```
1 =origami
2 libs=[ w17x17 ]
3 fuelcomp{
4 uox(fuel){ enrich=5.0 }
5
6 uox
7 stdcomp
8 mix
9 end
10
```

origami components here

Note: default for mix is 100% fuel

ORIGAMI input from scratch – Input irradiation history

- Put marker outside fuelcomp but inside **origami**
- CTRL + SPACE
- Select **hist**
- Put marker inside **hist**
- CTRL + SPACE
- Select **cycle**

Note: default for cycle is power 40MW/MTIHM, irradiation time 365 days, downtime 31 days

```
1 =origami
2 libs=[ w17x17 ]
3 fuelcomp{
4 uox(fuel){ enrich=5.0 }
5 mix(1){ comps[ fuel=100.0 ] }
6 }
7 hist[
8
9 cycle
10
11 ]
12
13 % insert origami components here
14
15 end
```

ORIGAMI input from scratch – First detailed input

Final input should look something like this.

Or cleaned up:

```
1 =origami
2 libs=[ w17x17 ]
3 fuelcomp{
4 uox(fuel){ enrich=5.0 }
5 mix(1){ comps[ fuel=100.0 ] }
6 }
7 hist[
8 cycle{ power=40 burn=365 down=31 }
9 ]
10 % insert cycles here
11
12 ]
13
14 % insert origami components here
15
16 end
17
```

```
1 =origami
2 libs=[ w17x17 ]
3 fuelcomp{
4   uox(fuel){ enrich=5.0 }
5   mix(1){ comps[ fuel=100.0 ] }
6 }
7 hist[ cycle{ power=40 burn=365 down=31 } ]
8 end
```

File is ready to run

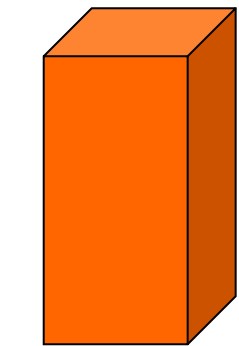
Results will be per MTU

```
30 Now depleting axial zone: 001, pin: 01-01
31
32 Scale job /Users/qv9/Desktop/origami/origami.inp is finished.
33 Output is stored in /Users/qv9/Desktop/origami/origami.out
34
35 Process finished with 0 return code; ran in 10 secs, finished at Wed Sep 24 11:53:31 2025
```

ORIGAMI spatial variation modeling

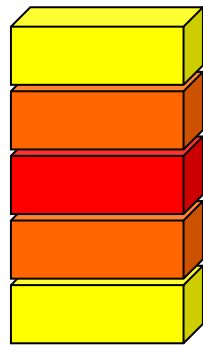
- Complexity level needed depends on the application and spent fuel metric of interest
 - 0D is generally adequate for assembly-average decay heat, activity, and nuclide inventory
 - 1D is needed if axial distributions are desired (e.g., for BWR), whereas 2D and 3D may be of interest to specific safeguards applications

power shape $\rightarrow$ axial & radial $\rightarrow p(z) \times p(x, y)$
moderator density $\rightarrow$ axial only $\rightarrow m(z)$
fuel composition $\rightarrow$ radial only $\rightarrow c(x, y)$



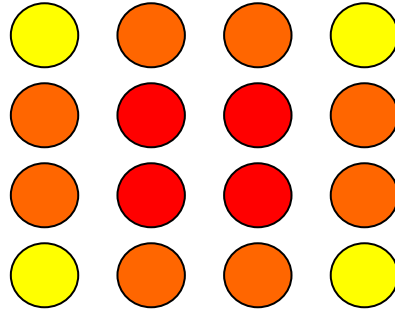
$\bar{p}$
 $\bar{m}$
 $\bar{c}$

**0D
assembly**



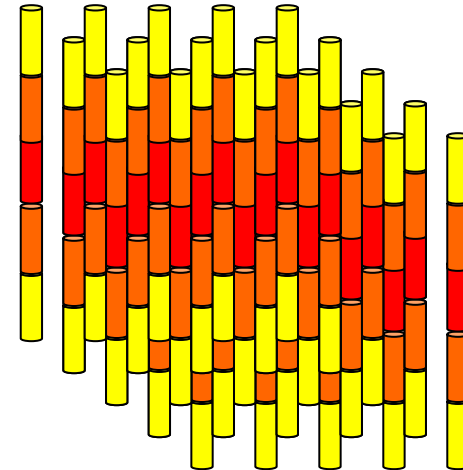
$p(z)$
 $m(z)$
 $\bar{c}$

**1D
assembly**



$p(x, y)$
 $\bar{m}$
 $c(x, y)$

**2D
assembly**



$p(x, y) \times p(z)$
 $m(z)$
 $c(x, y)$

**3D
assembly**

ORIGAMI input description: Main blocks

```
1 =origami
2   title="this-is-my-title"
3   options{ mtu=1.0 ft71=all }
4   libs=[ w17x17 ]
5   fuelcomp{
6     uox(fuel){ enrich=4.5 }
7     mix(1){ comps[ fuel=100.0 ] }
8   }
9   modz=[ 0.7332 ]
10  pz=[ 1.0 ]
11  hist[
12    cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
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{ down=1825 }
16  ]
17 end
```

- **title** descriptive title
- **options** various global options
- **libs** ORIGIN reactor library(ies) to use
- **fuelcomp** fuel mixtures and compositions (single mixture problems use 1)
- **modz** axial moderator density
- **pz** axial power shape
- **hist** operating history

ORIGAMI Input: Fuel composition

```
1 =origami
2   title="this-is-my-title"
3   options{ mtu=1.0 ft71=all }
4   libs=[ w17x17 ]
5   fuelcomp{
6     uox(fuel){ enrich=4.5 }
7     mix(1){ comps[ fuel=100.0 ] }
8   }
9   modz=[ 0.7332 ]
10  pz=[ 1.0 ]
11  hist[
12    cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
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{ down=1825 }
16  ]
17 end
```

- **uox(name){...}**
creates a SCALE StdCmp UO₂ mixture
- **mix(N)**
N mixture index
comps array of composition names and wt %

SCALE StdCmp has many common standard compositions (zirc4, gd2o3, etc.)

ORIGAMI Input: ORIGEN reactor data libraries

ORIGEN reactor library names in
arpdata.txt

libs=["w17x17"]

- declares library 1 to be "w17x17" (Westinghouse 17x17 PWR assembly)
- could be any reactor library
- a single entry is typical

```
1 =origami
2   title="this-is-my-title"
3   options{ mtu=1.0 ft71=all }
4   libs=[ w17x17 ]
5   fuelcomp{
6     uox(fuel){ enrich=4.5 }
7     mix(1){ comps[ fuel=100.0 ] }
8   }
9   modz=[ 0.7332 ]
10  pz=[ 1.0 ]
11  hist[
12    cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
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{ down=1825 }
16  ]
17 end
```

ORIGAMI Input: Power history

```
1 =origami
2   title="this-is-my-title"
3   options{ mtu=1.0 ft71=all }
4   libs=[ w17x17 ]
5   fuelcomp{
6     uox(fuel){ enrich=4.5 }
7     mix(1){ comps[ fuel=100.0 ] }
8   }
9   modz=[ 0.7332 ]
10  pz=[ 1.0 ]
11  hist[
12    cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
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{ down=1825 }
16  ]
17 end
```

Operating history

- **cycle**
 - power** assembly-average specific power in MW/MTIHM
 - burn** time in days at power
 - down** time in days of decay
 - nlib** number of library interpolation points per cycle

ORIGAMI Input: Moderator density (modz) & axial power profile (pz)

```
1 =origami
2   title="this-is-my-title"
3   options{ mtu=1.0 ft71=all }
4   libs=[ w17x17 ]
5   fuelcomp{
6     uox(fuel){ enrich=4.5 }
7     mix(1){ comps[ fuel=100.0 ] }
8   }
9   modz=[ 0.7332 ]
10  pz=[ 1.0 ]
11  hist[
12    cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
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{ down=1825 }
16  ]
17 end
```

Axial power and moderator distribution

- **modz**

density in g/cm³ of moderator
NOTE: not all reactor libraries have density variations

- **pz**

axial power shape (unitless)
NOTE: entered input is renormalized by default

Axial distribution: When is it useful ?

```
1=origami
2
3  title="W17x17 Assembly wtih Axial Distribution"
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
11 meshz=[0    50    100    200    300    350]
12 modz =[ 0.73  0.72  0.71  0.70  0.69  ]
13 pz   =[ 0.80  0.90  1.00  0.85  0.75  ]
14
15 hist[
16   cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
17   cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
18   cycle{ power=40 burn=333.33 nlib=4 down=0 }
19   cycle{ down=1825 }
20 ]
21
22 end
```

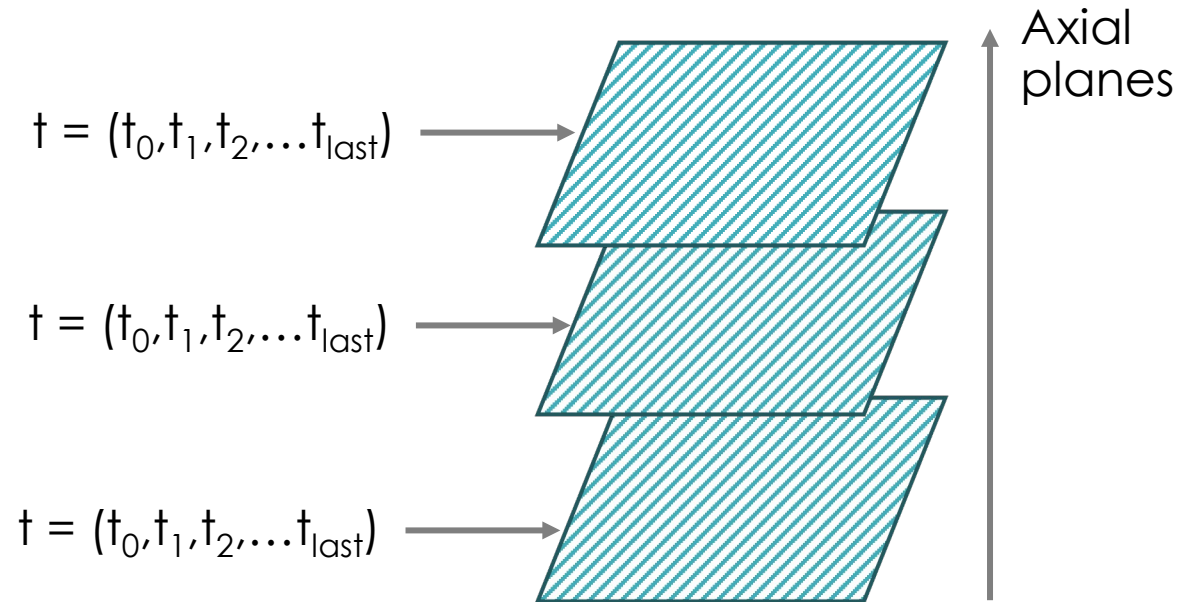
- to calculate axial decay heat in fuel assemblies
- to evaluate impact of moderator density variations on nuclide inventories, decay heat, or radiation emissions
- to model fork detector for NDA

meshz keyword useful for specifying non-uniform zone sizes (i.e., zone boundaries in cm)

ORIGAMI F71 files generated for models considering axial distributions of burnup and moderator density

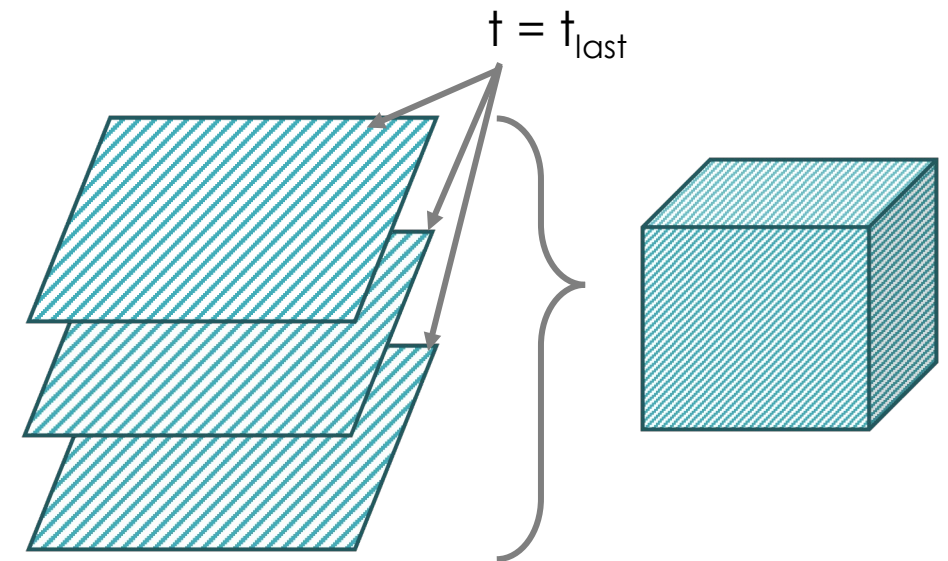
$\text{\$}\{\text{basename}\}\text{.f71}$

- Stores concentrations for each saved step for **every depleted zone**, for **each timestep saved**



$\text{\$}\{\text{basename}\}\text{.assm.f71}$

- Stores concentrations for **each axial zone** and the **total lumped per assembly**, at the **last time step**



ORIGAMI input: Other options

```
1 =origami
2   title="this-is-my-title"
3   options{ mtu=1.0 ft71=all }
4   libs=[ w17x17 ]
5   fuelcomp{
6     uox(fuel){ enrich=4.5 }
7     mix(1){ comps[ fuel=100.0 ] }
8   }
9   modz=[ 0.7332 ]
10  pz=[ 1.0 ]
11  hist[
12    cycle{ power=40 burn=333.33 nlib=4 down=16.67 }
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{ down=1825 }
16  ]
17 end
```

Miscellaneous input

- **mtu**
 - metric tons of initial heavy metal (not just U)
 - DEFAULT: 1 MTU
- **ft71**
 - flag to write f71 file (**last/cycle/all**)
- **other options**
 - **stdcomp** write SCALE StdComp blocks
 - **mcnp** write MCNP materials cards

ORIGAMI Input: Radiation sources

```
1 =origami
2 title="ORIGAMI depletion and decay 2x2 fuel array"
3 options{ mtu=1.0 ft7l=all stdcomp=yes }
4 libs=[ w17x17 w15x15 ]
5 libmap=[ 1 2
6          2 1 ]
7 fuelcomp{
8   uox(fuel){ enrich=5.0 }
9   mix(1){ comps[ fuel=100.0 ] }
10  uox(fuel2){ enrich=4.0 }
11  mix(2){ comps[ fuel2=100.0 ] }
12 }
13 compmap=[ 2 1
14           1 2 ]
15 pxy=[ 0.80 1.00
16       1.00 0.80 ]
17 pz= [ 0.20 0.50 0.80 1.00 1.10 1.15 1.20 0.90 0.60 0.30 ]
18 modz= [ 0.75 0.74 0.73 0.72 0.71 0.70 0.69 0.68 0.67 0.66 ]
19 meshz=[ 0 10 50 100 150 200 300 350 360 370 ]
20 nonfuel[ zirc4=200.0 co=0.02 ]
21 srcopt{
22   sublib=all
23   brem_medium=uo2
24   alphan_medium=case
25   print=no
26 }
27 hist[ cycle{ power=20 burn=2500.0 nlib=4 down=3650 } ]
28 ggrp=[ 3.0E+6 2.5E+6 1.5E+6 1.0E+6 0.7E+6 0.4E+6 ]
29 ngrp=[ 20.0E6 6.4E+6 3.0E+6 1.4E+6 0.9E+6 0.4E+6 0.1E+6 ]
30 end
```

- **scropt** – options for neutron and gamma sources
- **ngrp** – neutron group structure, given as energy boundary (eV) in descending order
- **ggrp** – gamma group structure, given as energy boundary (eV) in descending order

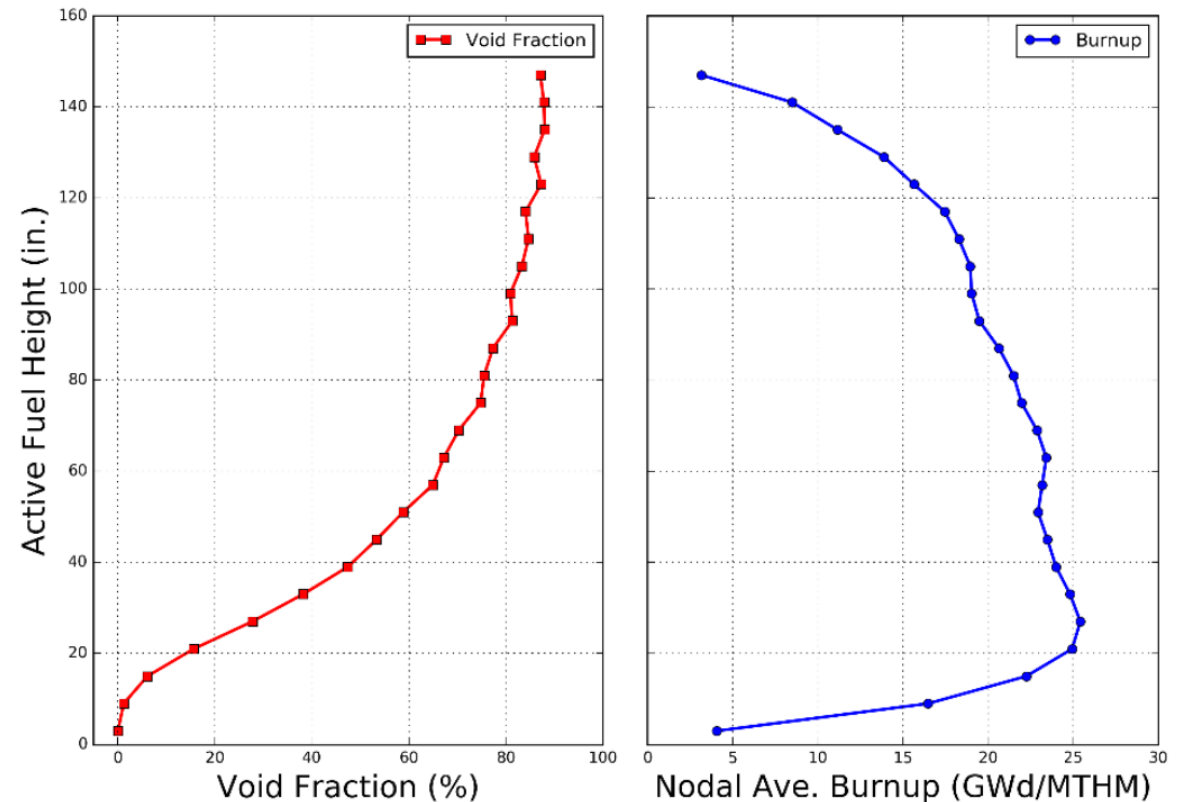
Tutorial:

Modeling axial variation
in burnup and coolant density
for a BWR assembly



Problem: Model a BWR assembly using 0D and 1D models

- Given a GE14 BWR assembly (ge10×10-8), what is the relative impact of assuming an **assembly-average** (0D) burnup and coolant density instead of **axial burnup and coolant density profiles** (1D) at discharge and 10-year cooling time on
 - Assembly Pu content
 - Assembly content for Pu isotopes
 - Assembly decay heat
 - Assembly total neutron source
- Evaluate** the impact using two models
 - Typical 0D ORIGAMI calculation (UOX Express)
 - 1D ORIGAMI detailed input calculation



Void fraction and discharge burnup profile for a representative BWR assembly (from NUREG/CR-7240)

<https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7240/index.html>

Part 1: 0D approximation

- **Irradiation history and assembly data**

- Discharge burnup: 45.2 GWd/MTU
- Assume uniform specific power (21.836 MW/MTU)
- 3 irradiation cycles, 690 days per cycle, 30 days downtime between cycles
- Library: ge10x10-8
- Enrichment: 4.5% ^{235}U
- Assembly-average coolant density: 0.3084 g/cm³
- Use nlib=2 (for speedup)

- **Evaluate at 10 years cooling time**

Start with the UOX express form!

Part 2: 1D approximation

- **Irradiation history and assembly data**

- Discharge burnup: 45.2 GWd/MTU
assume uniform specific power (21.836 MW/t)
- 3 irradiation cycles, 690 days per cycle, 30 days downtime between cycles
- Library: ge10x10-8
- Enrichment: 4.5% ^{235}U
- Axial power: use provided data
- Axial coolant density: use provided data
- Axial node height: use provided data
- Use nlib=2 (for speedup)

- **Evaluate at 10 years cooling time**

Axial node height (cm)	Axial node coolant density (g/cc)	Axial node power (relative)
30.48	0.750	0.549
45.72	0.710	1.189
91.44	0.557	1.340
106.68	0.370	1.283
121.92	0.330	1.256
137.16	0.297	1.226
182.88	0.270	1.238
213.36	0.210	1.162
228.60	0.200	1.104
259.08	0.170	1.030
304.80	0.147	0.975
381.00	0.124	0.559

Hint: We can reuse (expand upon) our prior input!

Tutorial solution:

Modeling axial variation
in burnup and coolant
density for a BWR assembly



ORIGAMI: 0D input

Decay heat:

- Discharge: 1.343 MW/MTU
- 10 years: 1.590 kW/MTU

Plutonium content (kg/MTU)

	Discharge	10 years
^{239}Pu	6.22	6.27
^{240}Pu	2.78	2.80
^{241}Pu	1.53	0.94
^{238}Pu	0.31	0.31
^{242}Pu	0.64	0.64
Total Pu	11.48	10.96

Neutron source (n/s/MTU)

	10 years
(alpha,n)	1.022E+07
SF	4.822E+08
Total	4.924E+08

^{244}Cm (g/MTU)

- Discharge: 62.8
- 10 years: 42.8

ORIGAMI: 1D input

Decay heat:

- Discharge: 1.340 MW/MTU
- 10 years: 1.643 kW/MTU

Plutonium content (kg/MTU)

	Discharge	10 years
^{239}Pu	6.24	6.30
^{240}Pu	2.70	2.73
^{241}Pu	1.44	0.89
^{238}Pu	0.33	0.32
^{242}Pu	0.69	0.69
Total Pu	11.40	10.92

Neutron source (n/s/MTU)

	10 years
(alpha,n)	1.186E+07
SF	6.969E+08
Total	7.087E+08

^{244}Cm (g/MTU)

- Discharge: 90.5
- 10 years: 61.7

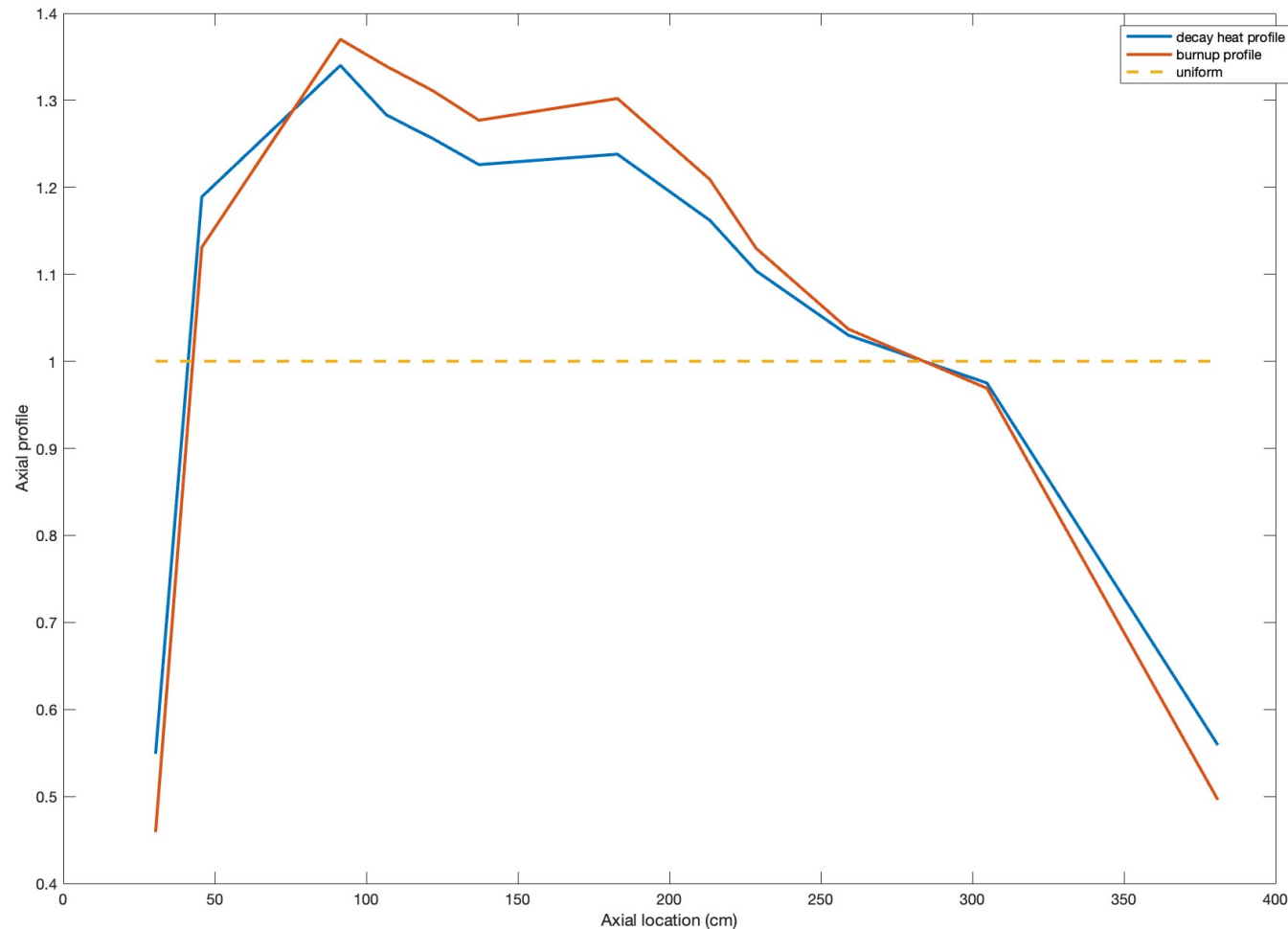
How large is the axial modeling effect for BWRs?

- **No change** in assembly total Pu @ 10 years
- **3.3% increase** in assembly decay heat @ 10 years
- **45% increase** in total neutron source @ 10 years
(**45% increase** in ^{244}Cm content @ 10 years)

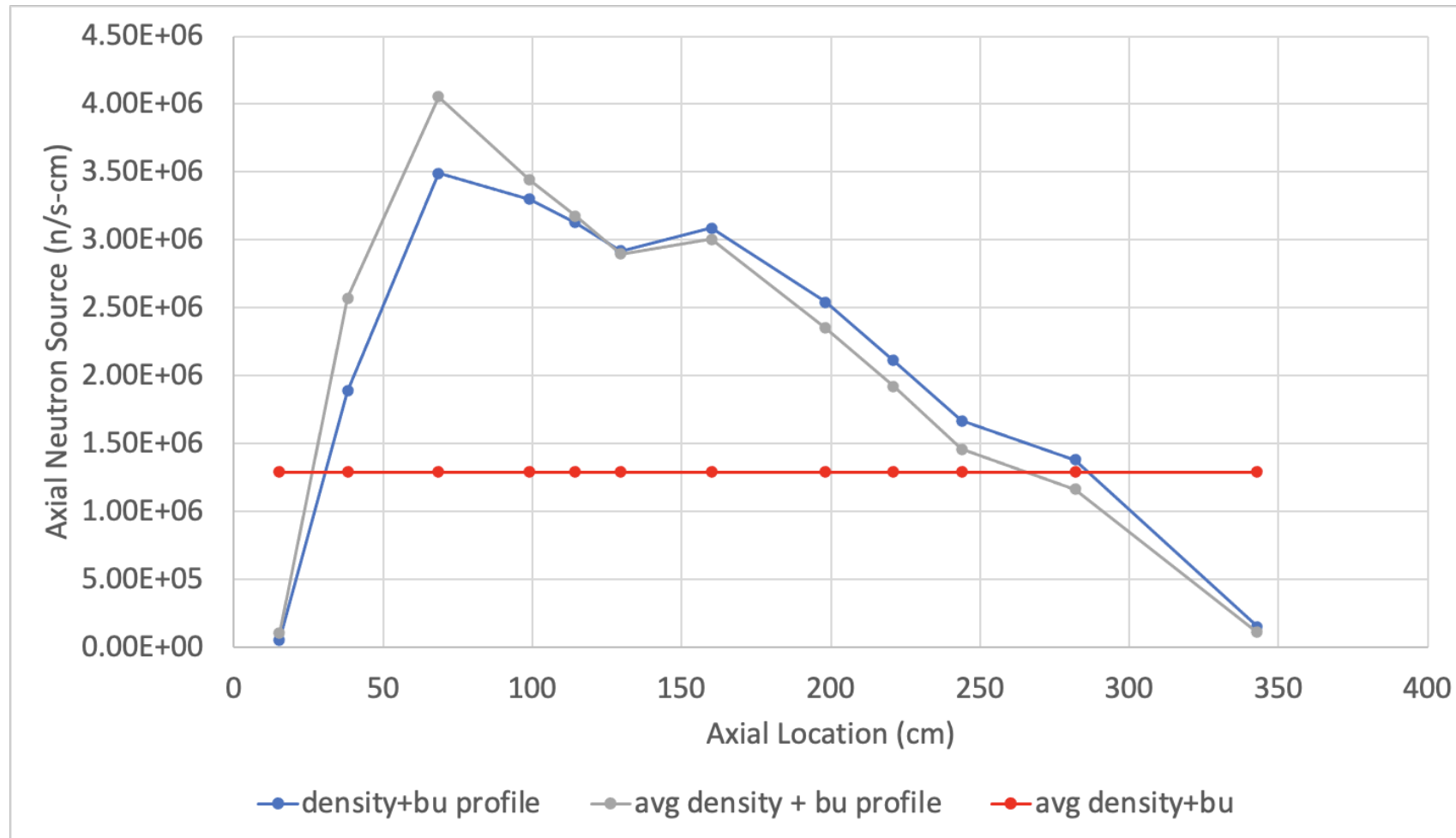
Conclusion?

Best-estimate models for BWR neutron source calculations benefit from including axial information

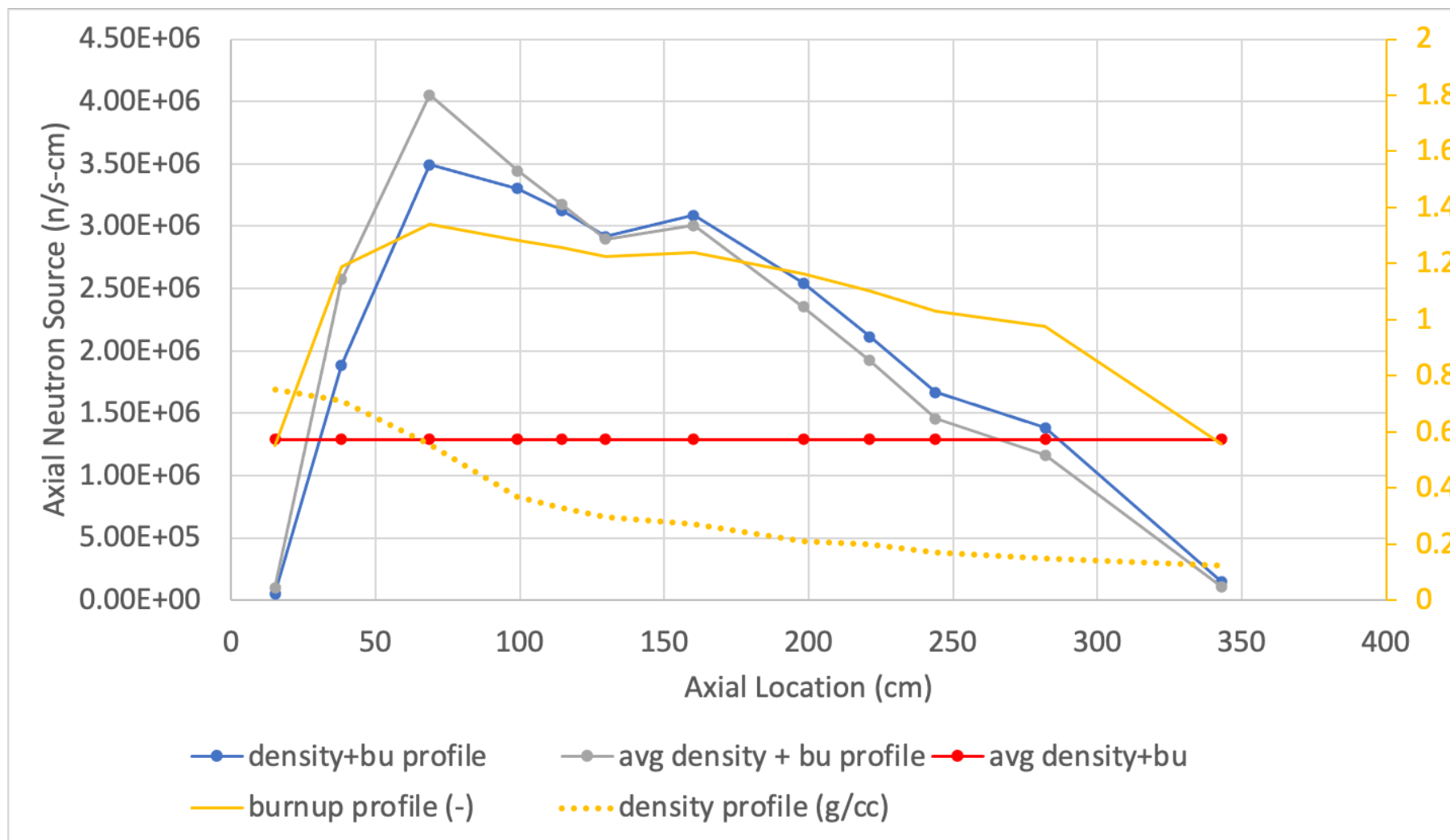
Impact on the decay heat: Axial decay heat profile vs. axial burnup profile



Impact of modeling assumptions on the neutron source



Impact of modeling assumptions on the neutron source with burnup and coolant density profiles



Frequently used input



Global-level options: The “**options**” block

Controls several calculation options, including

- Problem mass basis / normalization (**mtu**); default: 1.0
- Non-fuel mass fraction (**fracnf**)
- Axial power normalization (**relnorm**)
- Output file generation
 - **ft71** (binary), **output** (ascii)
- Auxiliary output file generation
 - **mcnp**, **stdcomp** (KENO), **decayheat**
- Calculation options
 - ORIGEN solver substeps: **nburn**, **ndecay**
 - ORIGEN solver: **cram**, **matrex**

Mass normalization: **mtu=**

- **Syntax:** `mtu=[float]`
- Default: `mtu=1.0`
- Will normalize mass across **all zones** to value in metric tons of heavy metal (not just uranium)
 - Assumes constant density across all axial / radial locations
 - Will apportion mass per axial zone **by volume** for non-uniform axial grid
- Used to convert to power in MW

Output control: **ft71=** and **output=**

- **Syntax:**

ft71=[all / cycle / last / none]

output=[all / cycle / last / none]

all: Print / save concentrations at the end of **each irradiation AND decay step** (if present) for **each cycle**

cycle: Print / save concentrations at the **last timestep** (irradiation or decay) for **each cycle**

last: Print / save concentrations at the **last timestep** (irradiation or decay) of the **entire history**

Non-fuel components: **nonfuel**

```
1 =origami
2 title="ORIGAMI depletion and decay 2x2 fuel array"
3 options{ mtu=1.0 ft71=all stdcomp=yes }
4 libs=[ w17x17 w15x15 ]
5 libmap=[ 1 2
6          2 1 ]
7 fuelcomp{
8   uox(fuel){ enrich=5.0 }
9   mix(1){ comps[ fuel=100.0 ] }
10  uox(fuel2){ enrich=4.0 }
11  mix(2){ comps[ fuel2=100.0 ] }
12 }
13 compmap=[ 2 1
14           1 2 ]
15 pxy=[ 0.80 1.00
16       1.00 0.80 ]
17 pz= [ 0.20 0.50 0.80 1.00 1.10 1.15 1.20 0.90 0.60 0.30 ]
18 modz= [ 0.75 0.74 0.73 0.72 0.71 0.70 0.69 0.68 0.67 0.66 ]
19 meshz=[ 0 10 50 100 150 200 300 350 360 370 ]
20 nonfuel[ zirc4=200.0 co=0.02 ]
```

- **Nonfuel** is given in **kg per MTU**
 - Give value pairs of standard compositions (i.e., zirc4), nuclides (i.e., zr-94) or elements (zr)
 - Can be used to account for activation of cladding, spacers, fuel channels, control rods, etc.
- **Optional:**
 - fracnf= in options block
 - Relative abundances of non-fuel components renormalized to **[fracnf] kg per MTU**

Print options

```
1 =origami
2 title="ORIGAMI depletion and decay 2x2 fuel array"
3 options{ mtu=1.0 ft71=all stdcomp=yes }
4 libs=[ w17x17 w15x15 ]
5 libmap=[ 1 2
6          2 1 ]
7 fuelcomp{
8   uox(fuel){ enrich=5.0 }
9   mix(1){ comps[ fuel=100.0 ] }
10  uox(fuel2){ enrich=4.0 }
11  mix(2){ comps[ fuel2=100.0 ] }
12 }
13 compmap=[ 2 1
14           1 2 ]
15 pxy=[ 0.80 1.00
16        1.00 0.80 ]
17 pz= [ 0.20 0.50 0.80 1.00 1.10 1.15 1.20 0.90 0.60 0.30 ]
18 modz= [ 0.75 0.74 0.73 0.72 0.71 0.70 0.69 0.68 0.67 0.66 ]
19 meshz=[ 0 10 50 100 150 200 300 350 360 370 ]
20 nonfuel[ zirc4=200.0 co=0.02 ]
21 print{
22   nuc{
23     sublibs=[lt ac fp]
24     total=yes
25     units=[ GRAMS CURIES WATTS ]
26   }
27 }
```

print - sets what is shown in the output for last time step

- select between nuclides (nuc) or elements (ele)
- select sublibs; can be light elements (lt), actinides (ac), fission products (fp) or all
- total=yes will print the total concentration for each output unit type

Special option for axial meshes (in options)

- Axial power normalization (relnorm) in options block
 - Yes (default): shape factors treated as **relative** modifiers of power
 - No: shape factors treated as **absolute** power fractions per axial zone

Workshop:

ORIGAMI detailed input for
rapid LWR spent fuel
calculations



Problem 1: Decay heat benchmark for BWR assembly

Determine the decay heat for BWR fuel assembly ID=13847 irradiated in Forsmark Unit 3 and compare to the reported measured decay heat; estimate the impact of cladding and structural materials activation on the calculated assembly decay heat

- Input data (source [NUREG/CR-6972](#))
 - SVEA 10×10 assembly design
 - Moderator density: 0.3816 g/cm³
 - Enrichment: 2.77 % ²³⁵U
 - Assume initial U contains U-235 and U-238 only
 - Assembly initial U loading 180.67 kg
 - Light element loading (accounts for clad and spacers activation): Table 1
 - Irradiation history data: Table 2
 - Cycle powers: Table 3
 - Reported measured decay heat: 169.6 W
 - Cooling time (from discharge to date of measurement): 4871 days

Problem 1: Decay heat benchmark for BWR assembly (cont.)

- Results required:
 - Calculated decay heat for assembly (W) using all provided input data
 - Calculated-to-measured (C/M) ratio
 - Difference between calculation and measurement C-M in W
 - Impact of cladding and structural materials activation on calculated decay heat (W) for the assembly
- Strategy:
 - Use SVEA100-0 reactor library
 - Optionally, can calculate decay heat for 1 MTU basis, then scale to assembly initial U loading
 - For estimating the impact of cladding and structural materials activation, perform calculation with or without the light element loading included

Problem 1: Decay heat benchmark for BWR assembly (cont.)

Table 1. Assembly light element content

Element	kg/MTU
C	1.882E-03
N	6.033E-03
Al	3.708E-02
Si	1.389E-02
S	3.321E-04
Ti	1.160E-01
Cr	6.958E-01
Mn	3.244E-02
Fe	3.154E-01
Co	3.017E-02
Ni	3.356E+00
Cu	2.325E-03
Nb	4.173E-02
O	1.384E+02
Zr	3.819E+02
Sn	2.712E+00

Table 2. Assembly 13847 irradiation history

Operational Data		Cycle 2	Cycle 3	Cycle 4	Cycle 5
Startup date		9/2/1986	8/9/1987	9/5/1988	7/13/1989
Shutdown date		7/10/1987	8/12/1988	6/9/1989	7/13/1990
Operating days		311	369	277	365
Downtime days		30	24	34	---
Assembly ID	Design	Assembly burnup (MWd/MTU)			
13847	SVEA-100	7857	17217	23779	31275

Problem 1: Decay heat benchmark for BWR assembly (cont.)

Table 3. Assembly 13847 power data

Cycle number	Sample power (MW/MTU)
2	25.26
3	25.37
4	23.69
5	20.54

Problem 2: Nuclide inventory benchmark for PWR spent fuel

Determine the nuclide inventory for Calvert Cliffs PWR spent fuel sample D047_MKP109_P and compare to experimental nuclide concentrations reported for this sample

- **Input data** (source [NUREG/CR-6968](#) report and [SFCOMPO](#) database):
 - CE 14×14 assembly design
 - Enrichment: 3.038% U-235
 - Initial U isotopic composition (wt%): 0.027 U-234, 0.014 U-236, 96.921 U-238
 - Sample irradiation history data: Table 3
 - Measured nuclide inventory data: Table 4
 - Cooling time (from discharge to measurement date): Table 4 footnote
 - Note on sample identifier: this sample was selected from fuel rod MKP109 of fuel assembly D047

Problem 2: Nuclide inventory benchmark for PWR spent fuel (cont.)

- Results required:
 - Calculated (C) nuclide concentrations (g/MTU) for the nuclides listed in Table 2
 - Calculated-to-measured (C/M) concentration ratios for the nuclides listed in Table 2
- Strategy:
 - Use ce14x14 reactor library
 - Multiple approaches to resolve the problem, based on ORIGAMI only, or adding to it ORIGEN and/or OPUS

Problem 2: Nuclide inventory benchmark for PWR spent fuel (cont.)

Table 3. Irradiation history data for Calvert Cliffs sample D047_MKP109_P

Reactor cycle #	Start date (month/day/year)	End date (month/day/year)	Duration (days)	Down (days)	Sample cumulative burnup (GWd/MTU)
2	3/22/77	1/22/78	306	71	9.52
3	4/3/78	4/20/79	382	81	21.93
4	7/10/79	10/18/80	466	85	34.14
5	1/11/81	4/17/82	461		44.34

Problem 2: Nuclide inventory benchmark for PWR spent fuel (cont.)

Table 4. Experimental data for Calvert Cliffs sample D047_MKP109_P

Nuclide	g/g U initial	Exp. uncert. (%)
u234	1.361E-04	1.6
u235	4.016E-03	1.6
u236	4.186E-03	1.6
u238	9.358E-01	1.6
pu238	3.049E-04	1.6
pu239	4.943E-03	1.6
pu240	2.885E-03	1.6
pu241	1.157E-03	1.6
pu242	9.530E-04	1.6
np237	5.338E-04	1.9
am241	4.331E-04	4.9
sr90	5.425E-04	5.7
tc99	8.944E-04	3.5
cs133	1.407E-03	0.0
cs135	4.874E-04	14.0
cs137	1.424E-03	3.5

Nuclide	g/g U initial	Exp. uncert. (%)
nd143	8.656E-04	1.0
nd144	1.864E-03	1.0
nd145	8.440E-04	1.0
nd146	9.416E-04	1.0
nd148	4.855E-04	1.0
nd150	2.360E-04	2.0
sm147	3.081E-04	3.8
sm148	2.246E-04	3.8
sm150	3.638E-04	3.7
sm151	1.072E-05	5.1
sm152	1.325E-04	4.2
sm154	5.640E-05	3.8
eu153	1.612E-04	3.4
gd155	9.959E-06	3.9

* measurement data reported at 1870 d after discharge for all nuclides except for Sm, Eu, Gd; for the latter, cooling time is 4656 d.

Problem 3: Decay heat axial variation for PWR assembly benchmark

Determine the impact of modeling axial variation of burnup on the total assembly decay heat for the decay heat benchmark for PWR assembly F32 irradiated in Ringhals 2 reactor

- **Input data** (sources [NUREG/CR-6972](#) and [NUREG/CR-6971](#)):
 - 15×15 assembly design
 - Enrichment: 3.197% ^{235}U
 - Initial U contains 0.02 wt% U-234 and 0.04 wt% U-236
 - Light element loading is neglected here
 - Discharge burnup 50.962 GWd/MTU
 - Assembly initial U loading 436.99 kg
 - Assembly irradiation history data: Table 5
 - Fuel rod active length 365.8 cm
 - Axial assembly power provided for 24 axial nodes of equal heights: Table 6
 - Reported assembly measured decay heat: 692.0 W
 - Cooling time (from discharge to measurement date): 5860 days

Problem 3: Decay heat axial variation for PWR assembly benchmark

- **Results required:**
 - Calculated decay heat for assembly (W) with and without considering assembly axial power
 - Calculated-to-measured (C/M) assembly decay heat ratio with and without considering assembly axial power
 - Difference between calculation and measurement C-E in W for assembly decay heat with and without considering assembly axial power
 - Calculated assembly axial decay heat (in 24 axial nodes)
 - Calculated assembly axial decay heat profile
 - Comparison of assembly axial decay heat profile and axial power (burnup) profile
- **Strategy:**
 - Use the w15x15 reactor library
 - Optional, can calculate decay heat for 1 MTU basis, then scale to assembly initial U loading
 - Perform two calculations using a 0D model and a 1D model
 - Calculate assembly decay heat profile by dividing the axial node decay heat by the assembly-average (over all nodes) decay heat value

Problem 3: Decay heat axial variation for PWR assembly benchmark

Table 5. Axial power profile for assembly F32

Axial node #	Axial node power (relative)
1	0.631
2	0.871
3	0.989
4	1.037
5	1.055
6	1.061
7	1.062
8	1.061
9	1.060
10	1.059
11	1.058
12	1.057
13	1.056
14	1.055
15	1.054
16	1.054
17	1.053
18	1.053
19	1.051
20	1.047
21	1.033
22	0.994
23	0.889
24	0.661

Cycle #	Duration (days)	Down (days)	Assembly power (MW/MTU)
1	290	80	36.390
2	317	84	33.467
3	273	778	30.736
4	320	63	24.253
5	297	54	22.320
6	329		21.334

Workshop solutions:

ORIGAMI detailed input
for rapid LWR spent fuel
calculations



Problem 1: Decay heat benchmark for BWR assembly (input)

```
=origami
title="Forsmark BWR Deacy Heat Benchmark"
options{ mtu=0.18067 ft71=all} % 180.67 kgU mass basis
libs=[ "svea100-0" ]
nonfuel=[ % nonfuel in kg/MTU mass basis
    C=1.882E-03   N=6.033E-03 Al=3.708E-02 Si=1.389E-02
    S=3.321E-04   Ti=1.160E-01 Cr=6.958E-01 Mn=3.244E-02
    Fe=3.154E-01  Co=3.017E-02 Ni=3.356E+00 Cu=2.325E-03
    Nb=4.173E-02  O=1.384E+02  Zr=3.819E+02 Sn=2.712E+00
]
fuelcomp{
    mix(1){ stdcomp(fuel){ base=uo2 iso[ 92235=2.77 92238=97.23 ] } } % 2.77% U-235 + U-238 only
}
modz=[ 0.3816 ] % 0.3816 g/cm³ average moderator density
pz=[ 1.0 ] % 1 axial zone
hist[
    cycle{ power=25.26 burn=311 nlib=4 down=30 }
    cycle{ power=25.37 burn=369 nlib=4 down=24 }
    cycle{ power=23.69 burn=277 nlib=4 down=34 }
    cycle{ power=20.54 burn=365 nlib=4 down=4871 }
]
end
```

Problem 1: Decay heat benchmark for BWR assembly (results)

- **Calculated** (C) assembly decay heat at 4871 days after discharge
 - 168.325 W (fuel only)
 - 169.21 W (fuel + structural material)
- **Measured** (M) decay heat: 169.6 W (per assembly)
- **Difference** calculation-measurement
 - $C-M = -1.275$ W (fuel); -0.39 W (fuel + structural material)
 - $C/M = 0.992$ (fuel); 0.998 (fuel + structural material)

Problem 2: Nuclide inventory benchmark for PWR spent fuel

Nuclide	Measured g/gU initial	Calculated g/gU initial	C/M
u234	1.361E-04	1.402E-04	1.030
u235	4.016E-03	3.627E-03	0.903
u236	4.186E-03	4.288E-03	1.024
u238	9.358E-01	9.348E-01	0.999
pu238	3.049E-04	3.099E-04	1.016
pu239	4.943E-03	4.933E-03	0.998
pu240	2.885E-03	2.862E-03	0.992
pu241	1.157E-03	1.131E-03	0.977
pu242	9.530E-04	9.578E-04	1.005
np237	5.338E-04	5.317E-04	0.996
am241	4.331E-04	3.804E-04	0.878
sr90	5.425E-04	1.038E-03	1.160
tc99	8.944E-04	1.402E-04	1.030
cs133	1.407E-03	1.444E-03	1.026
cs135	4.874E-04	4.887E-04	1.003
cs137	1.424E-03	1.410E-03	0.990

Nuclide	Measured g/gU initial	Calculated g/gU initial	C/M
nd143	8.656E-04	8.639E-04	0.998
nd144	1.864E-03	1.877E-03	1.007
nd145	8.440E-04	8.436E-04	0.999
nd146	9.416E-04	9.514E-04	1.010
nd148	4.855E-04	4.943E-04	1.018
nd150	2.360E-04	2.461E-04	1.043
sm147	3.081E-04	2.955E-04	0.959
sm148	2.246E-04	2.143E-04	0.954
sm150	3.638E-04	3.816E-04	1.049
sm151	1.072E-05	9.952E-06	0.928
sm152	1.325E-04	1.316E-04	0.993
sm154	5.640E-05	5.432E-05	0.963
eu153	1.612E-04	1.565E-04	0.971
gd155	9.959E-06	1.023E-05	1.027

Problem 3: Decay heat axial variation for PWR assembly benchmark

- Assembly calculated decay heat (W)
 - 0D model: 719.5
 - 1D model: 724.3
- Calculated - Measured (W)
 - 0D model: 27.5
 - 1D model: 32.3
- Calculated/Measured
 - 0D model: 1.040
 - 1D model: 1.047

Axial node #	Axial power profile	Axial decay heat profile
1	0.631	0.552
2	0.871	0.825
3	0.989	0.978
4	1.037	1.045
5	1.055	1.070
6	1.061	1.079
7	1.062	1.080
8	1.061	1.079
9	1.060	1.077
10	1.059	1.076
11	1.058	1.074
12	1.057	1.073
13	1.056	1.072
14	1.055	1.070
15	1.054	1.069
16	1.054	1.069
17	1.053	1.067
18	1.053	1.067
19	1.051	1.064
20	1.047	1.059
21	1.033	1.039
22	0.994	0.985
23	0.889	0.847
24	0.661	0.584

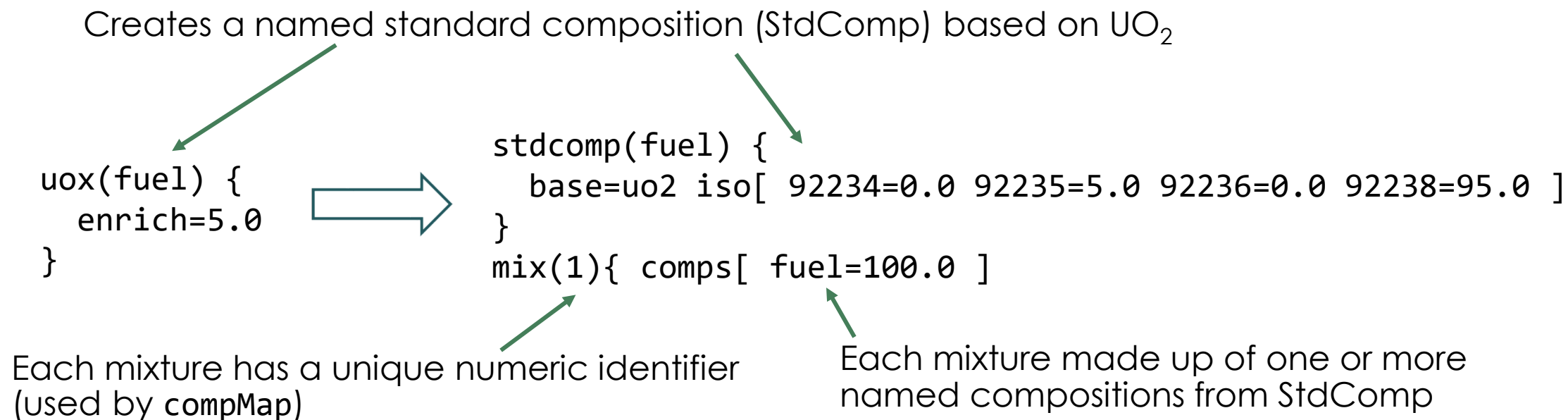
Beyond 1D: Radial modeling with ORIGAMI

- Powers, compositions, and complex mixtures



Beyond UOX: Using mixture compositions

- ORIGAMI is built on the SCALE standard composition engine
 - Mixtures, therefore, work like they do in other tools such as TRITON
- The uox= input is internally an *alias* for the more general syntax:



Creating complex mixtures for ORIGAMI using standard compositions

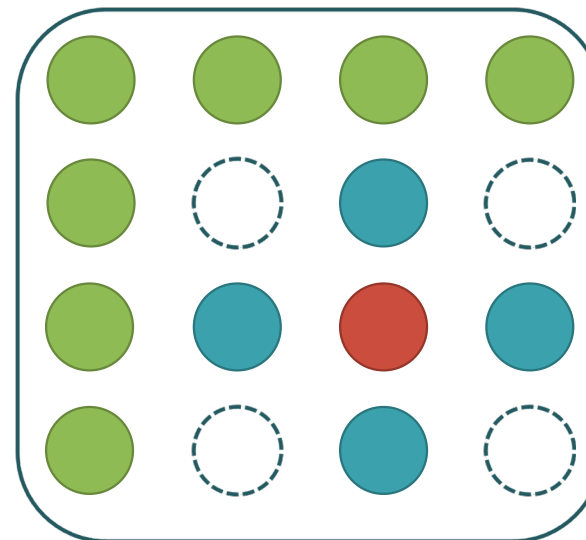
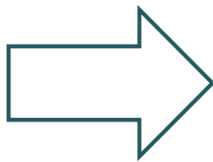
```
fuelcomp{
  stdcomp(fuel_5wt) {
    base=uo2 iso[ 92234=0.0 92235=5.0 92236=0.0 92238=95.0 ]
  }
  stdcomp(fuel_3wt) {
    base=uo2 iso[ 92234=0.0 92235=3.0 92236=0.0 92238=97.0 ]
  }
  mix(1){ comps[ fuel_5wt=100.0 ] % 100% UO2 @ 5.0% enrichment
  mix(2){ comps[ fuel_3wt=100.0 ] % 100% UO2 @ 3.0% enrichment
  mix(3){ comps[ fuel_3wt=97.0 Gd203=3.0 ] } % 97% UO2 @ 3.0% enrichment + 3% Gadolinia
  ...
}
```

Creating a heterogeneous array of pins: **compmap**

- Entries in **compmap** array correspond to fuel mixture numbers defined in the fuelcomp{ } block
- Entry format identical to TRITON / KENO arrays
 - Start at bottom row, left side => bottom-right
 - Proceed left-to-right along rows
 - “0” entry indicates a void (no material / fuel)

compmap=

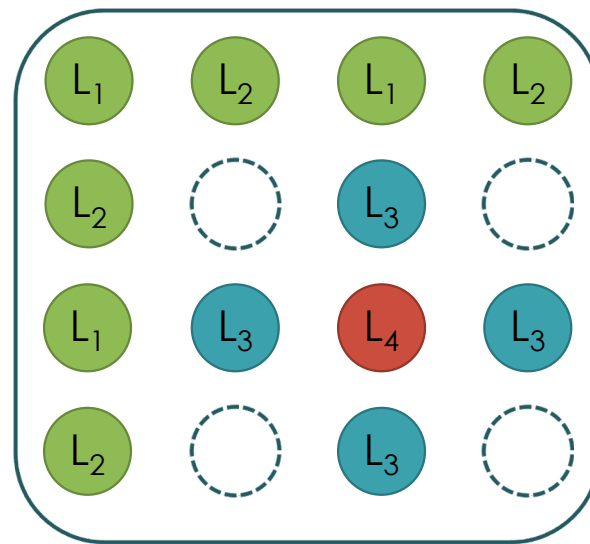
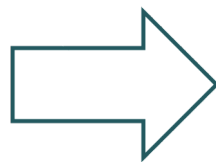
```
[ 1  0  2  0  
  1  2  3  2  
  1  0  2  0  
  1  1  1  1 ]
```



Defining different radial libraries: **libmap**

- Entries in **libmap** array correspond to index of entries defined in **libs[]** array (starting from 1)
- Same mapping syntax as **compmap[]**
 - **Default** is **libmap=[1]**, which will assign the same library to all pins
 - **Caveat:** if **libmap** is explicitly specified, must be consistent with **compmap**
 - “0” entry implies no pin at location

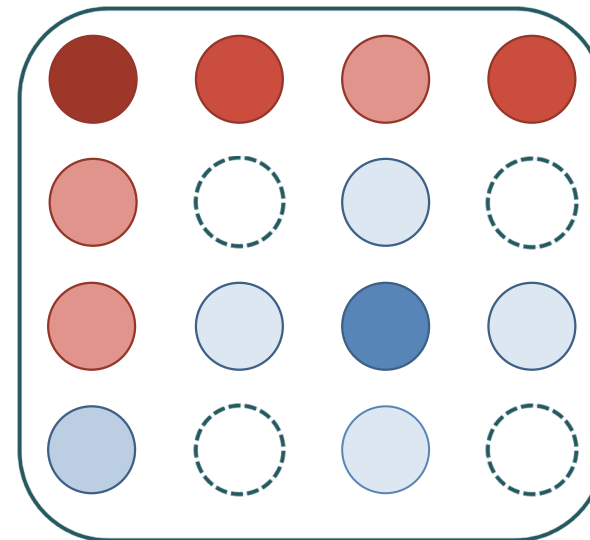
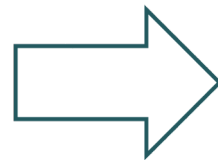
```
libmap=  
[ 2  0  3  0  
  1  3  4  3  
  2  0  3  0  
  1  2  1  2 ]
```



Introducing radial power gradients: **pxy**

- Radial power gradients are specified using the **pxy** array
- Imposes a **relative** power tilt
 - Mean of **pxy** automatically normalized to 1.0 for all **fuel** pins, preserving total assembly power

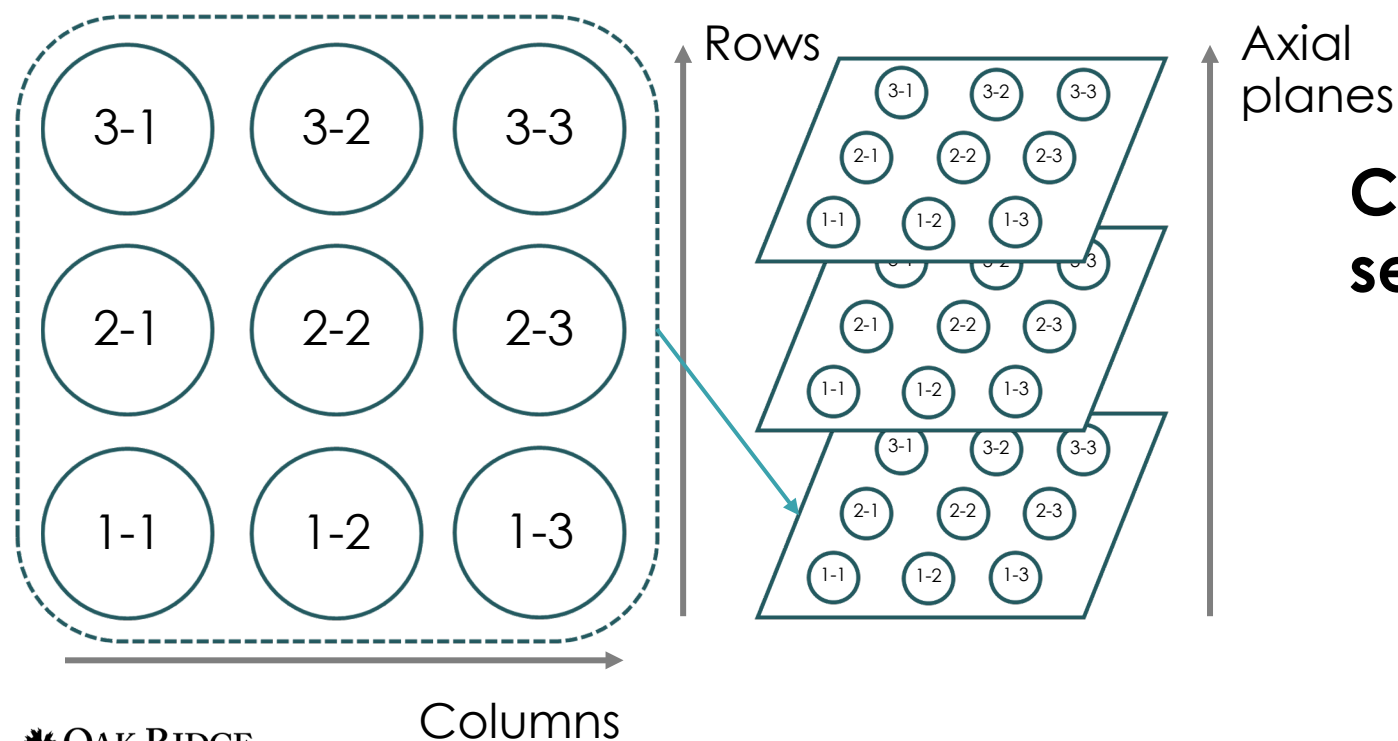
pxy= [0.75 0.0 0.80 0.0
1.1 0.8 0.65 0.8
1.1 0.0 0.8 0.0
1.2 1.15 1.1 1.15]



F71 binary concentrations file, revisited

Now that we've added **radial concentration** information **in addition** to **axial concentrations**, how do we navigate the F71 file?

$\{\text{basename}\}.f71$



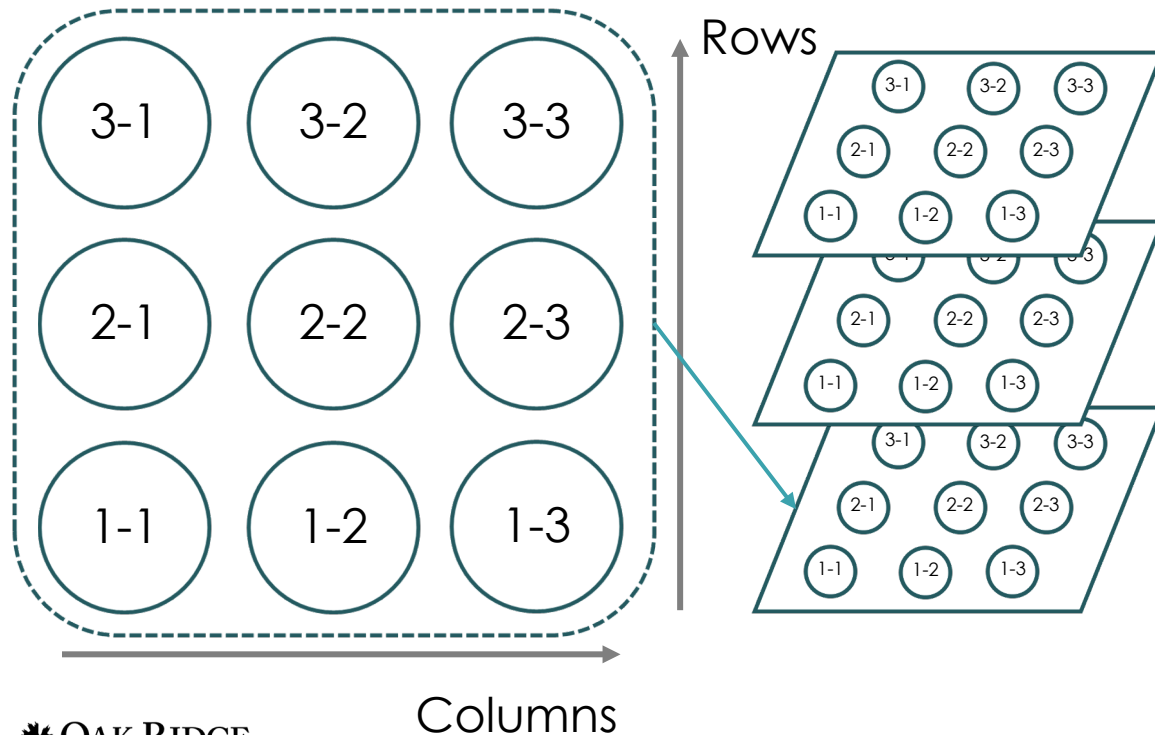
Concentrations now stored in sequential order by:

1. Time step ($t_0, t_1, t_2, \dots$)
2. Pin (1-1, 1-2, 1-3, 2-1...)
3. Axial zone (1, 2... N_{ax})

ORIGAMI radial and axial modeling: F71 files structures

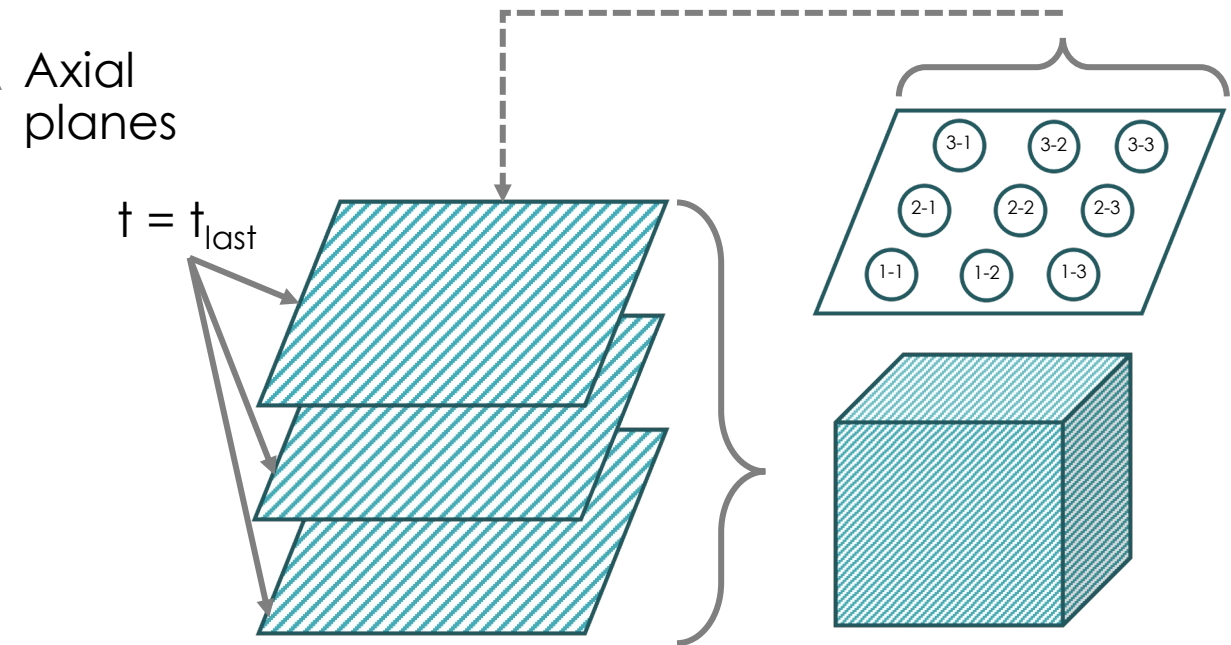
`${basename}.f71`

- Stores concentrations for **each saved step** for **every depleted zone**

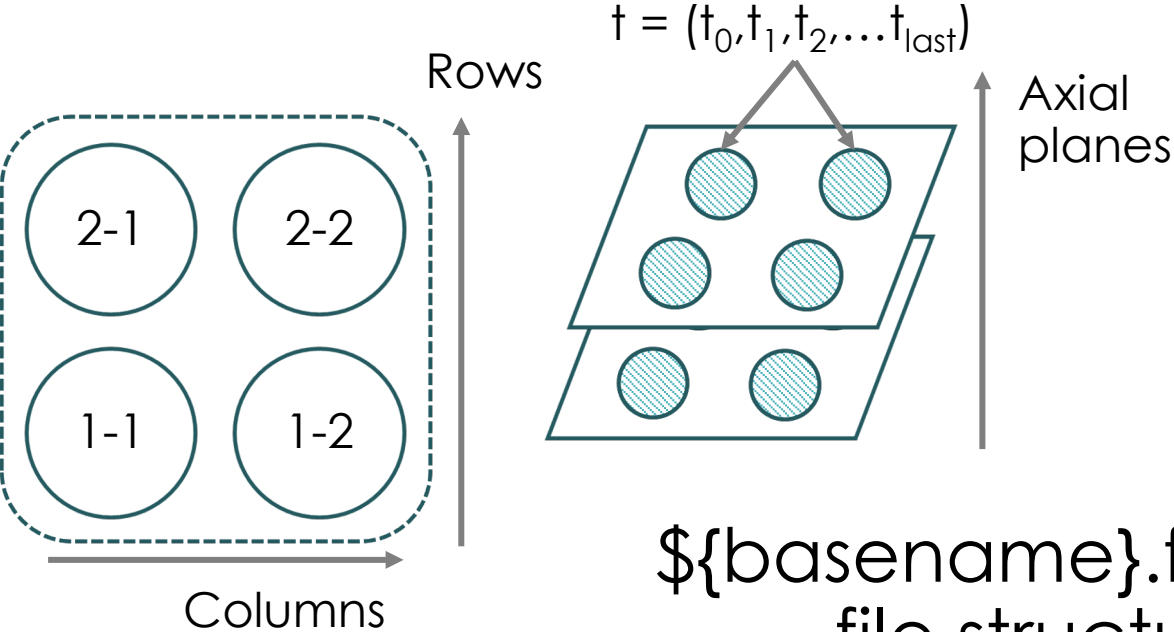


`${basename}.assm.f71`

- Stores concentrations for **each lumped axial zone** and the **total lumped assembly** at the **last saved timestep**



Parsing ORIGAMI concentrations files: pin-by-pin



`${basename}.f71`
file structure

Axial zone #1

Axial zone #2

Pin 1-1

Pin 1-2

Pin 2-1

Pin 2-2

Pin 1-1

Pin 1-2

Pin 2-1

Pin 2-2

t₀

t₁

t₂

t₀

t₁

t₂

t₀

t₁

t₂

t₀

t₁

t₂

t₀

t₁

t₂

t₀

t₁

t₂

t₀

t₁

t₂

t₀

t₁

t₂

Position:

1

2

3

4

5

6

7

8

9

10

11

12

13

14

15

16

17

18

19

20

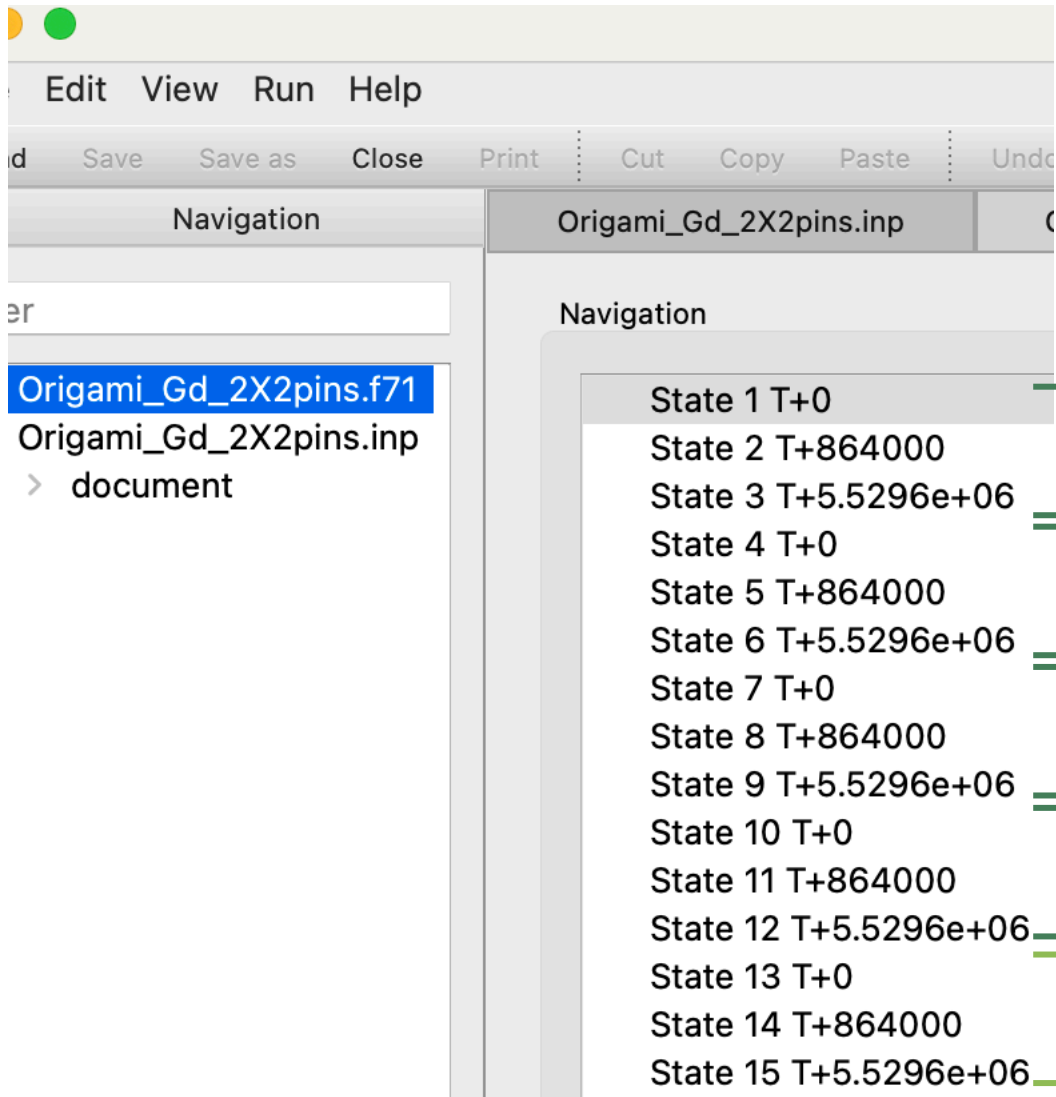
21

22

23

24

Making sense of the F71 file: Individual pins



Axial zone #1, Pin 1-1

Axial zone #1, Pin 1-2

Axial zone #1, Pin 2-1

Axial zone #1, Pin 2-2

Axial zone #2, Pin 1-1

- $t=0$ (initial)
- $t=10$ days (end of irradiation)
- $t=64$ days (+ 50 days decay)

Tutorial:

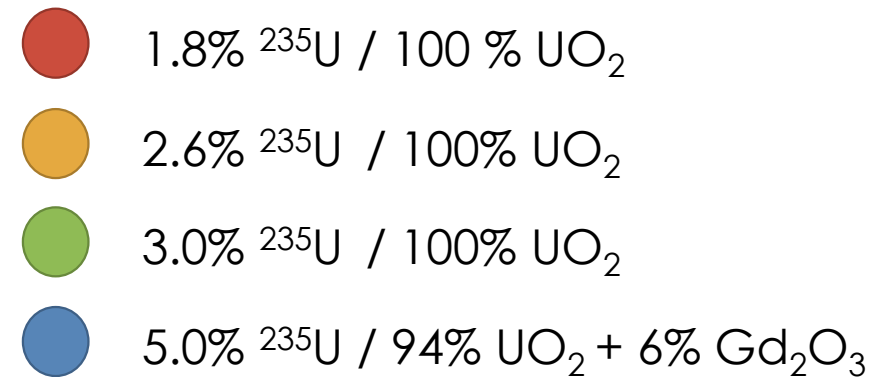
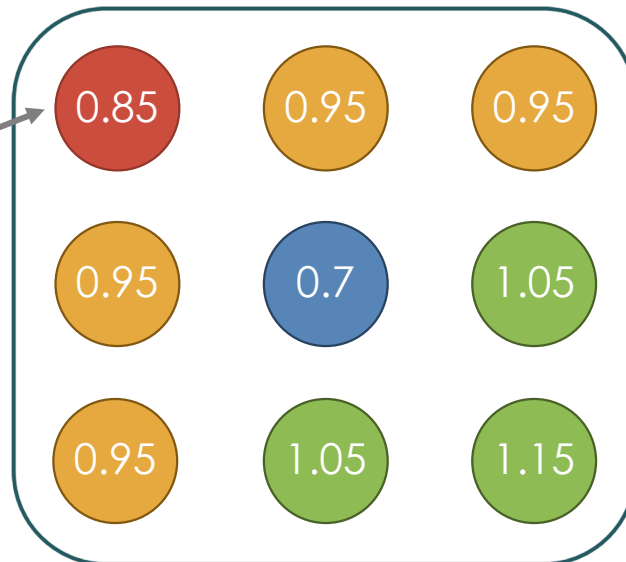
Gd-bearing BWR assembly:
Effect on decay heat and
Pu inventory



Tutorial: BWR assembly with Gd rods

- Calculate the decay heat and plutonium content for each fuel pin of a corner region for a Gd-bearing BWR assembly. Use two sets of ORIGEN libraries:
 - Same library for all pins (ge10x10-8)
 - Pinwise libraries (provided)
- How do your results change with the library used?

Numbers indicate relative power shaping factor



Tutorial: BWR assembly with Gd rods

- **Input data:**

- 1 cycle irradiation, 24-month
- Specific power: 25 MW/MTU
- Cooling time: 10 years
- Fuel compositions (wt %) are shown on previous page
- Pinwise libraries and associated arpdata.txt file provided
- Assume coolant density of 0.6 g/cc

- **Required results** (for each pin):

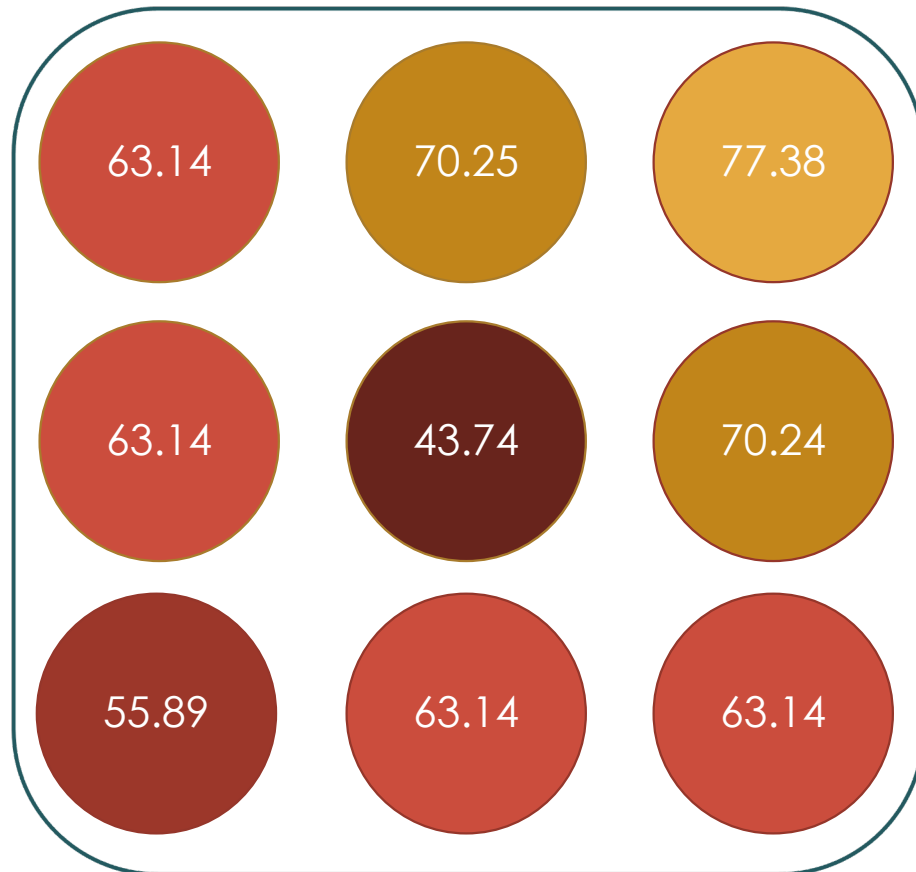
- Decay heat (W)
- Pu content (wt% Pu-239 / Pu, Pu mass)

- **Strategy:**

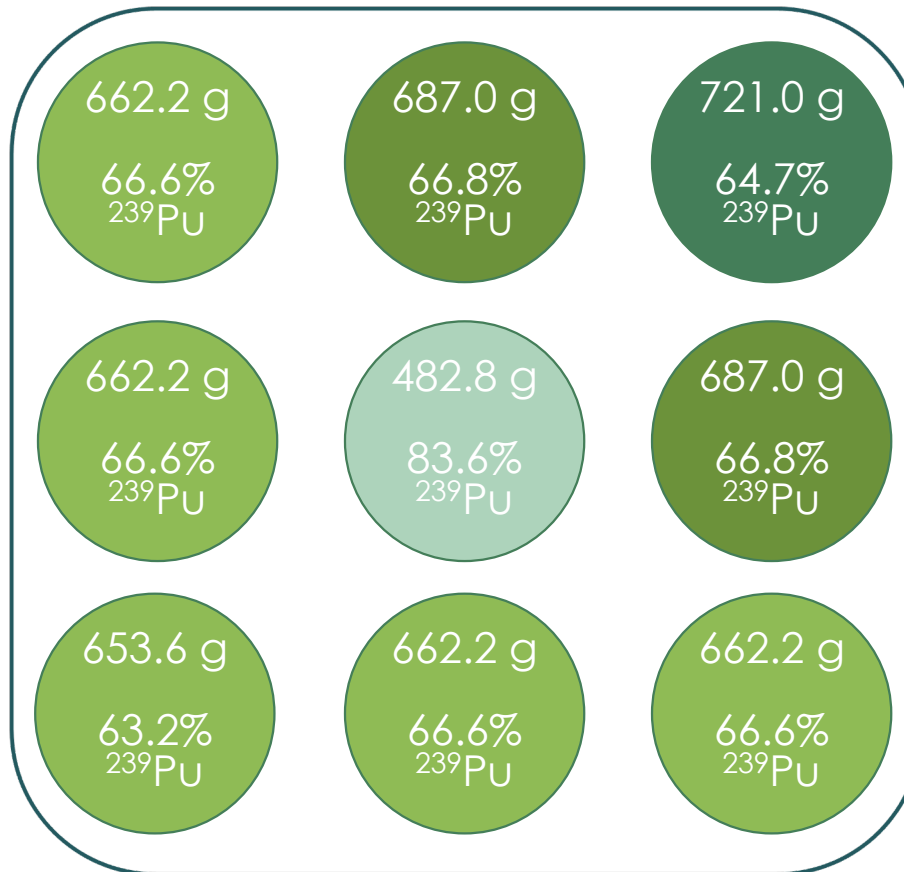
- Note that data Compositions are element wt. %; **6% of Gd pin mass** is Gd_2O_3
- Use a shell block at the beginning of the input to provide information about the custom arpdata.txt and pinwise libraries

Solution: BWR assembly with Gd rods (same library in all pins)

Decay heat (W) @ 10 years cooling

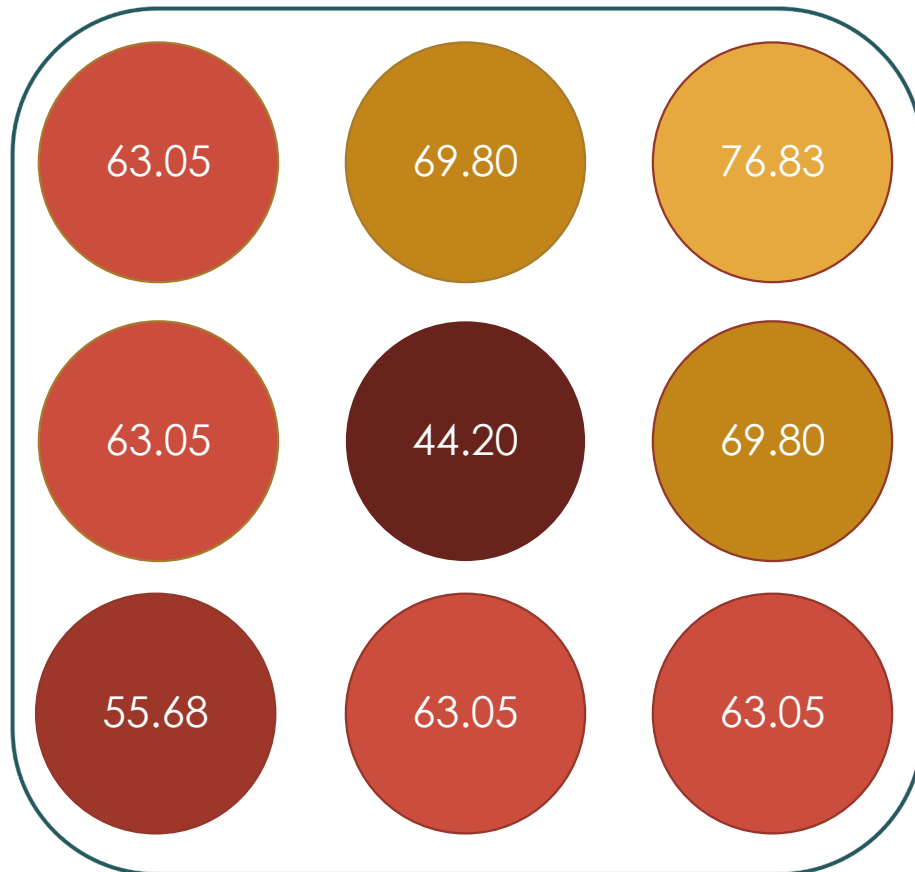


Plutonium content @ 10 years cooling



Solution: Relative change in decay heat using pin-specific libraries (compared to generic library)

**Decay heat (W) @ 10 years cooling
with pinwise libraries**

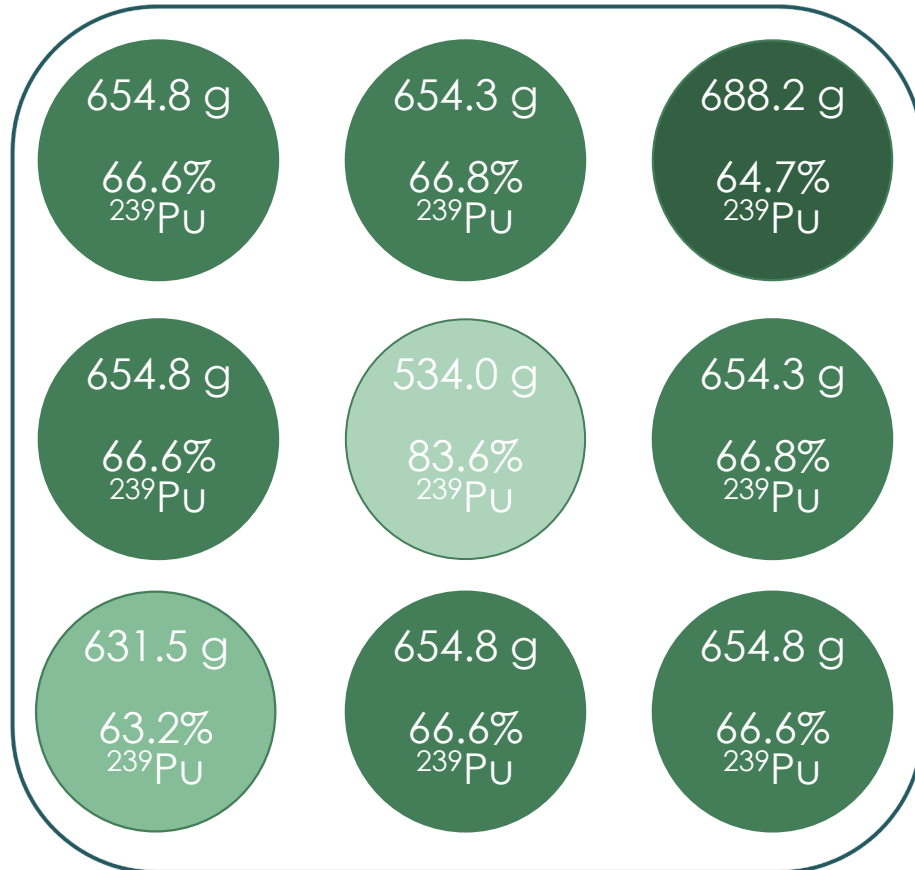


Decay heat (pinwise vs. generic library)

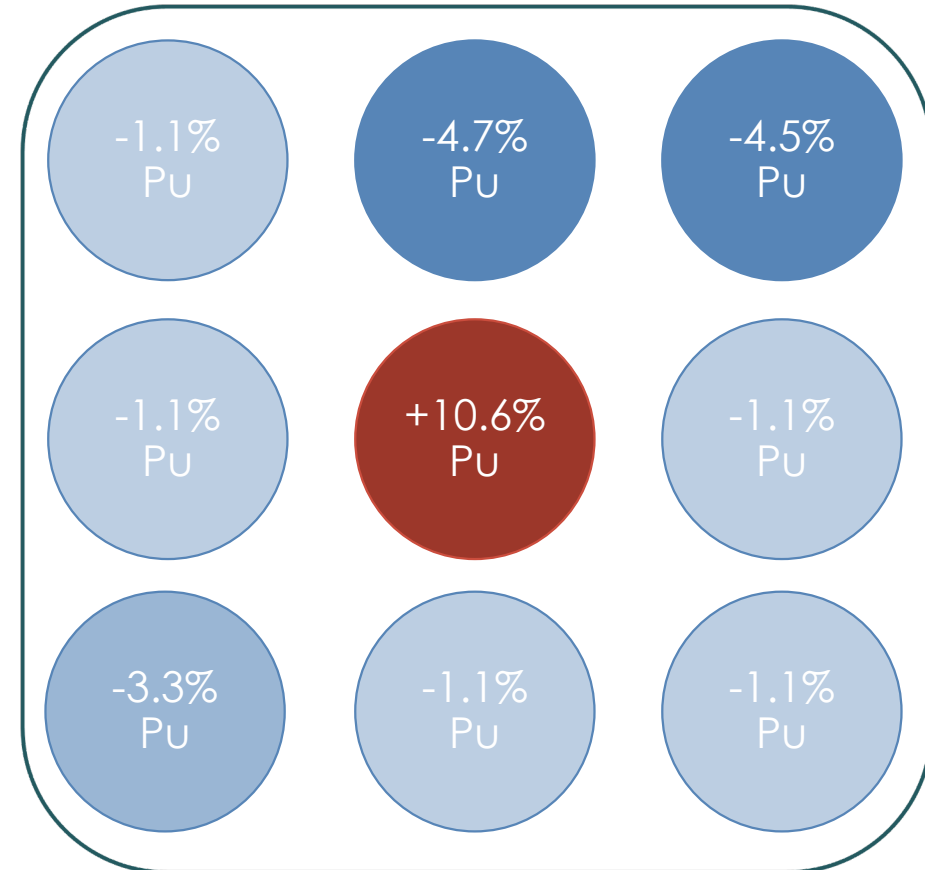


Solution: Relative change in plutonium using pin-specific libraries (compared to generic library)

**Plutonium content @ 10 years cooling
with pinwise libraries**



**Plutonium content
(pinwise vs. generic library)**



Workshop: Using ORIGAMI with a custom library



Advanced burner reactor with HALEU fuel

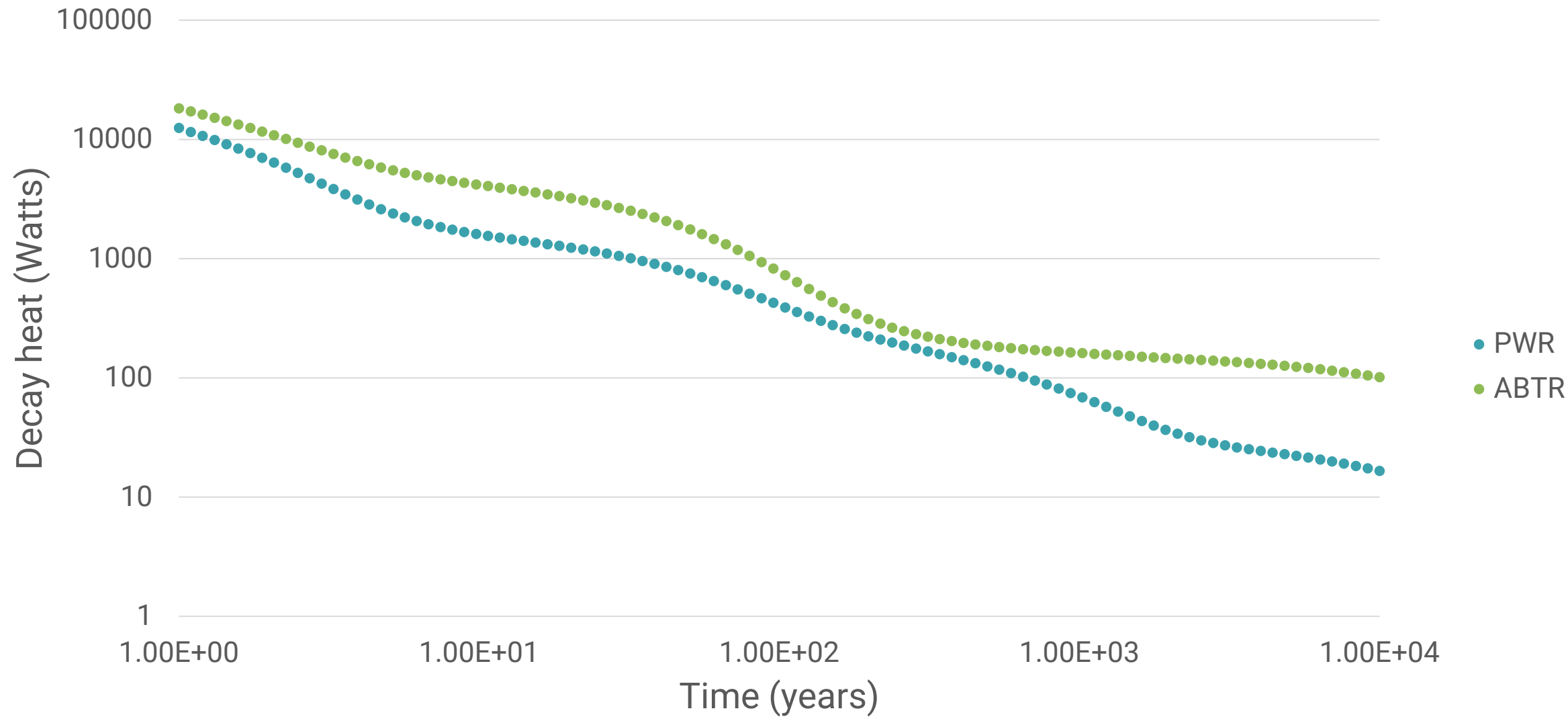
- Consider a fast reactor fueled by 16.5% enriched U-Zr fuel, burned to a discharge burnup of 150 GWd/MTU
 - Use the provided “ABTR_haleu” library & arpdata.txt
 - Copy into $\${TMPDIR}$ using shell before your ORIGAMI input
 - Assume a mixture of 90% U, 10% Zr with a density of 11.3726 g/cm³
 - Assume a specific power of 46.21523 MW/MTU
- Calculate the decay heat per MTU from 1 year to 10,000 years post-discharge
- **Compare** your time-dependent decay heat to an 5.0% enriched UO₂ LWR assembly irradiated to 45 GWd/MTU
 - Assume 40 MW/MTU specific power, w17x17 library

Tip: Use a log-log scale for decay heat comparisons over very long time intervals

Solution



Decay heat comparison: PWR vs. ABTR



Options for generating files for follow-on analyses



Options for generating data files for follow-on analysis

- Within **options** block
 - **stdcomp**: Generates SCALE standard composition files (*.compBlock) for axial zone compositions, for burnup credit nuclides (yes/no; **default**: no)
 - **fdens**: Sets fuel density (used for conversion to atom density for MCNP & SCALE standard composition outputs; **default**: 10.4 g/cm³)
 - **temper**: Sets fuel temperature in K (used for SCALE standard composition outputs; **default**: 293 K)
 - **decayheat**: Generates text output (*.AxialDecayHeat) of decay heat in Watts by axial zone (yes/no; **default**: no)
 - **mcnp**: Generates text output of material and source cards in MCNP input format (yes/no; **default**: no)

Nuclides included in **.compBlock** file

- **Light elements:** ^{16}O

- **Actinides**

^{234}U , ^{235}U , ^{236}U , ^{238}U , ^{237}Np , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , ^{241}Am , ^{243}Am

- **Fission products**

^{95}Mo , ^{99}Tc , ^{101}Ru , ^{103}Rh , ^{109}Ag , ^{133}Cs , ^{143}Nd , ^{145}Nd , ^{147}Sm , ^{149}Sm , ^{150}Sm , ^{151}Sm ,
 ^{152}Sm , ^{151}Eu , ^{153}Eu , ^{155}Gd

Gamma/neutron source options: **srcopt** block

- ORIGAMI allows for a limited subset of ORIGIN's source printing & calculation options (key/value pairs, defaults **highlighted**)

brem_medium: medium used for bremsstrahlung calculation (gamma)

- **uo2** (uranium oxide matrix)
- h2o (water)
- none (no bremsstrahlung calculation)

alphan_medium: medium used for (a,n) neutron calculation (neutron)

- uo2 (uranium oxide matrix)
- borosilicate (borosilicate glass)
- **case** (problem-specific mixture)

sublib: calculate / print gamma source terms by sublibrary

- lt (light elements)
- fp (fission products)
- ac (actinides)
- **all** (all nuclides)

print: print gamma/neutron source term to text output

- **yes:** generate text output files (*.AxialGammaSpec, *.AxialNeutronSpec) of gamma/neutron sources by axial zone
- **no:** sources only output to ft71 binary output

Examples of generated files for MCNP follow-on analyses

*.MCNP_mats.inp

```
C Axial zone: 01, Pin: 001
C Zone mass (grams): 9.073507E+04
m101      2004  -2.000317E-11
          8016  -1.182644E-01
          8017  -4.787782E-05
          82207 -3.165821E-18
          88223 -5.555578E-19
          88226 -4.426406E-15
          89227 -3.933551E-16
          90227 -9.115049E-19
          90230 -9.799348E-10
          90232 -5.310030E-12
          90234 -1.222845E-11
          91231 -3.824823E-11
          92234 -3.531152E-04
          92235 -3.967597E-02
          92236 -1.825094E-04
          92238 -8.414761E-01
C Axial zone: 02, Pin: 001
C Zone mass (grams): 9.073421E+04
m102      1001  -3.043445E-10
...
```

*.MCNP_gamma.inp

```
C Gamma source for axial zone 01, pin 001
C Total intensity (g/sec): 2.7917E+09
SI101      H      1.0000E-08  5.0000E-01  1.0000E+00  2.0000E+00
          1.0000E+01
SP101      D  0.0  9.8774E-01  8.1040E-03  4.1530E-03  1.4567E-06
C Gamma source for axial zone 02, pin 001
C Total intensity (g/sec): 1.0395E+15
SI102      H      1.0000E-08  5.0000E-01  1.0000E+00  2.0000E+00
          1.0000E+01
...
```

*.MCNP_neutron.inp

```
C Neutron source for axial zone 01, pin 001
C Total intensity (n/sec): 1.1489E+03
SI101      H      1.0000E-08  1.0000E-05  1.0000E-03  1.0000E-02
          1.0000E-01  1.0000E+00  2.0000E+01
SP101      D  0.0  1.3872E-08  1.4103E-05  4.3280E-04  1.3448E-02
          3.0902E-01  6.7708E-01
C Neutron source for axial zone 02, pin 001
C Total intensity (n/sec): 2.0845E+04
SI102      H      1.0000E-08  1.0000E-05  1.0000E-03  1.0000E-02
          1.0000E-01  1.0000E+00  2.0000E+01
...
```

Example of .compBlock file for SCALE follow-on analyses

```
o-16      1001  0  4.6291E-02  293.0 end
u-234     1001  0  6.6649E-06  293.0 end
u-235     1001  0  3.8451E-04  293.0 end
u-236     1001  0  1.4447E-04  293.0 end
u-238     1001  0  2.1426E-02  293.0 end
np-237    1001  0  1.3378E-05  293.0 end
pu-238    1001  0  4.4772E-06  293.0 end
pu-239    1001  0  1.4923E-04  293.0 end
pu-240    1001  0  5.1931E-05  293.0 end
pu-241    1001  0  2.0601E-05  293.0 end
pu-242    1001  0  1.0086E-05  293.0 end
am-241    1001  0  1.3683E-05  293.0 end
am-243    1001  0  1.8190E-06  293.0 end
mo-95     1001  0  5.5250E-05  293.0 end
tc-99     1001  0  5.5607E-05  293.0 end
ru-101    1001  0  5.1086E-05  293.0 end
rh-103    1001  0  3.0394E-05  293.0 end
ag-109    1001  0  3.8543E-06  293.0 end
cs-133    1001  0  5.7415E-05  293.0 end
nd-143    1001  0  4.1092E-05  293.0 end
nd-145    1001  0  3.2112E-05  293.0 end
sm-147    1001  0  1.1819E-05  293.0 end
sm-149    1001  0  1.7224E-07  293.0 end
sm-150    1001  0  1.2831E-05  293.0 end
sm-151    1001  0  5.3905E-07  293.0 end
sm-152    1001  0  4.3198E-06  293.0 end
eu-151    1001  0  4.3855E-08  293.0 end
eu-153    1001  0  4.6953E-06  293.0 end
gd-155    1001  0  2.4911E-07  293.0 end
```