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# Introduction to the ORIGEN + ARP depletion workflow

## SCALE 6.3.1



U.S. DEPARTMENT OF ENERGY

# Learning goals

**Introduce** users to how to perform interpolation of ORIGEN reactor library data to problem-specific conditions using ARP

**Demonstrate** how to use interpolated libraries with ORIGEN for nuclear fuel depletion calculations

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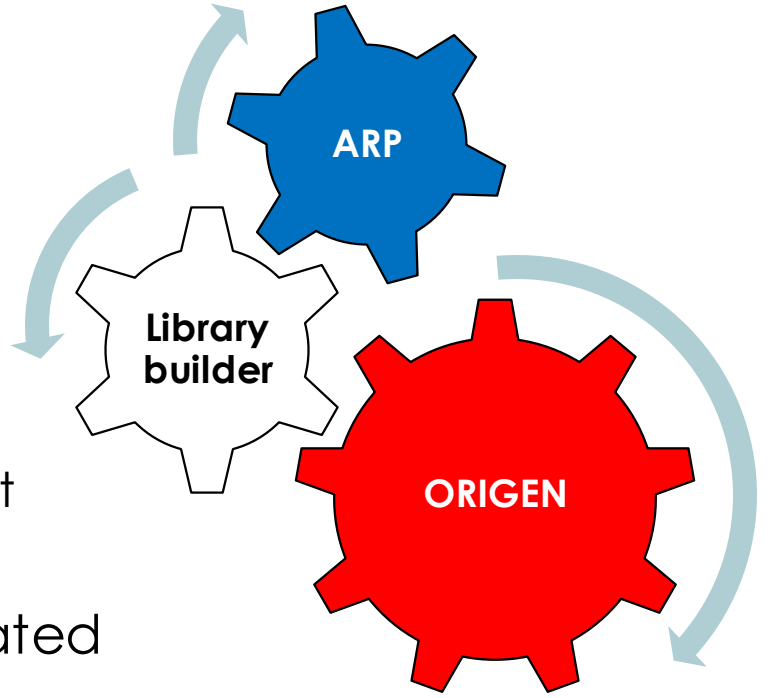
# ORIGEN Reactor Library interpolation using ARP

**A**utomated **R**apid **P**rocessing of pre-calculated ORIGEN reactor libraries for problem-specific applications

# ORIGEN-ARP methodology in SCALE
# How are ORIGEN Reactor Libraries used?

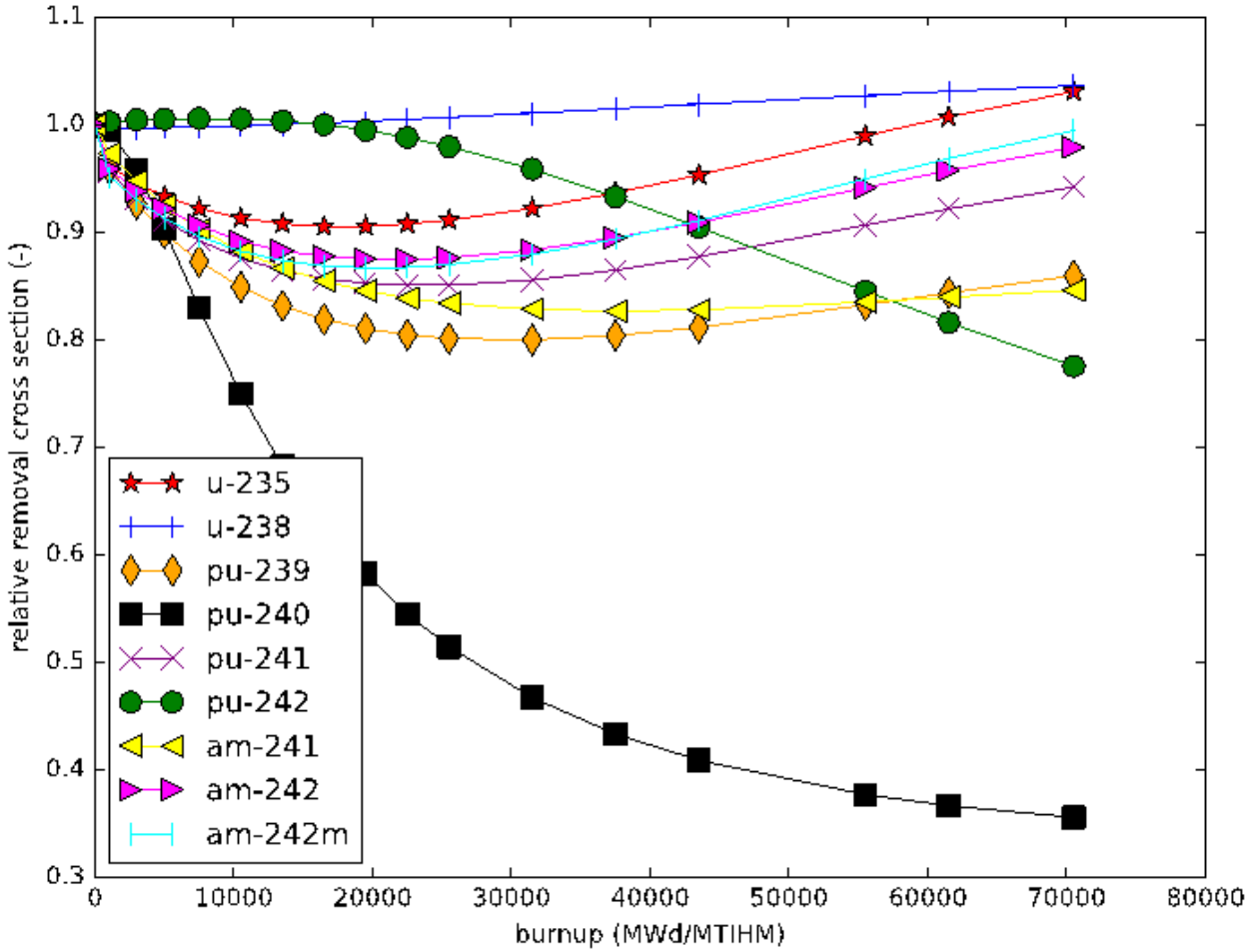
- **ORIGEN-ARP** = Innovative approach in SCALE since 1990's
  - Enables fast and accurate depletion simulations with ORIGEN for a given assembly design and user-defined burnup and assembly discrete parameters (e.g., enrichment) using pre-generated ORIGEN reactor library files
- **ARP** (Automated Rapid Processing) utility code
  - Interpolates pre-generated cross-sections to user-defined burnup and enrichment
  - Interpolation parameters for uranium fuels are burnup, enrichment and moderator density
- **ORIGEN** can perform depletion calculations with ARP-interpolated cross-section data



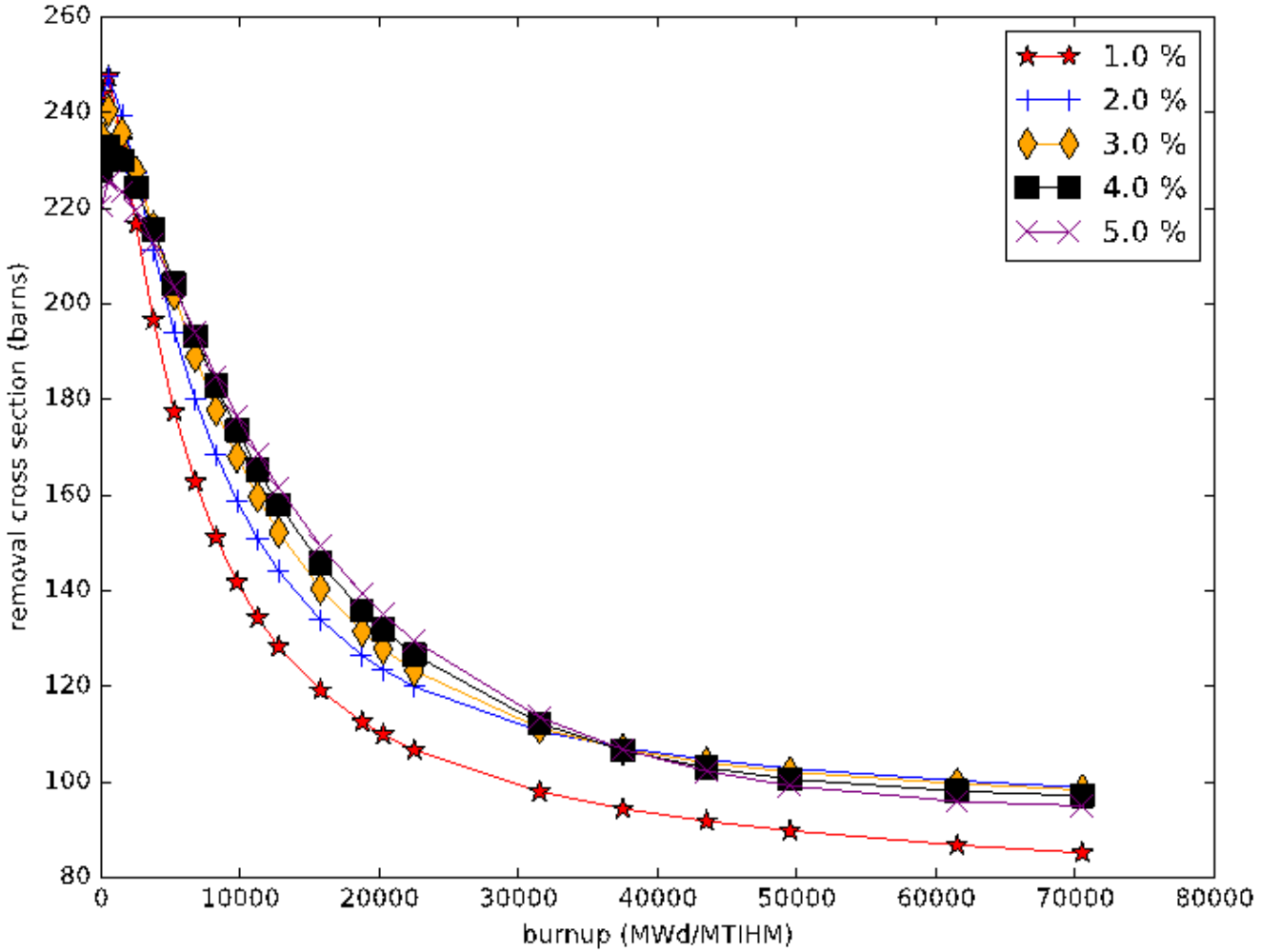

- **ORIGAMI automatically combines** ARP and ORIGEN to simplify the input and to perform calculations for axially- or radially-varying burnup and moderator density

Accuracy of assembly models that were used to generate the ORIGEN reactor libraries is maintained

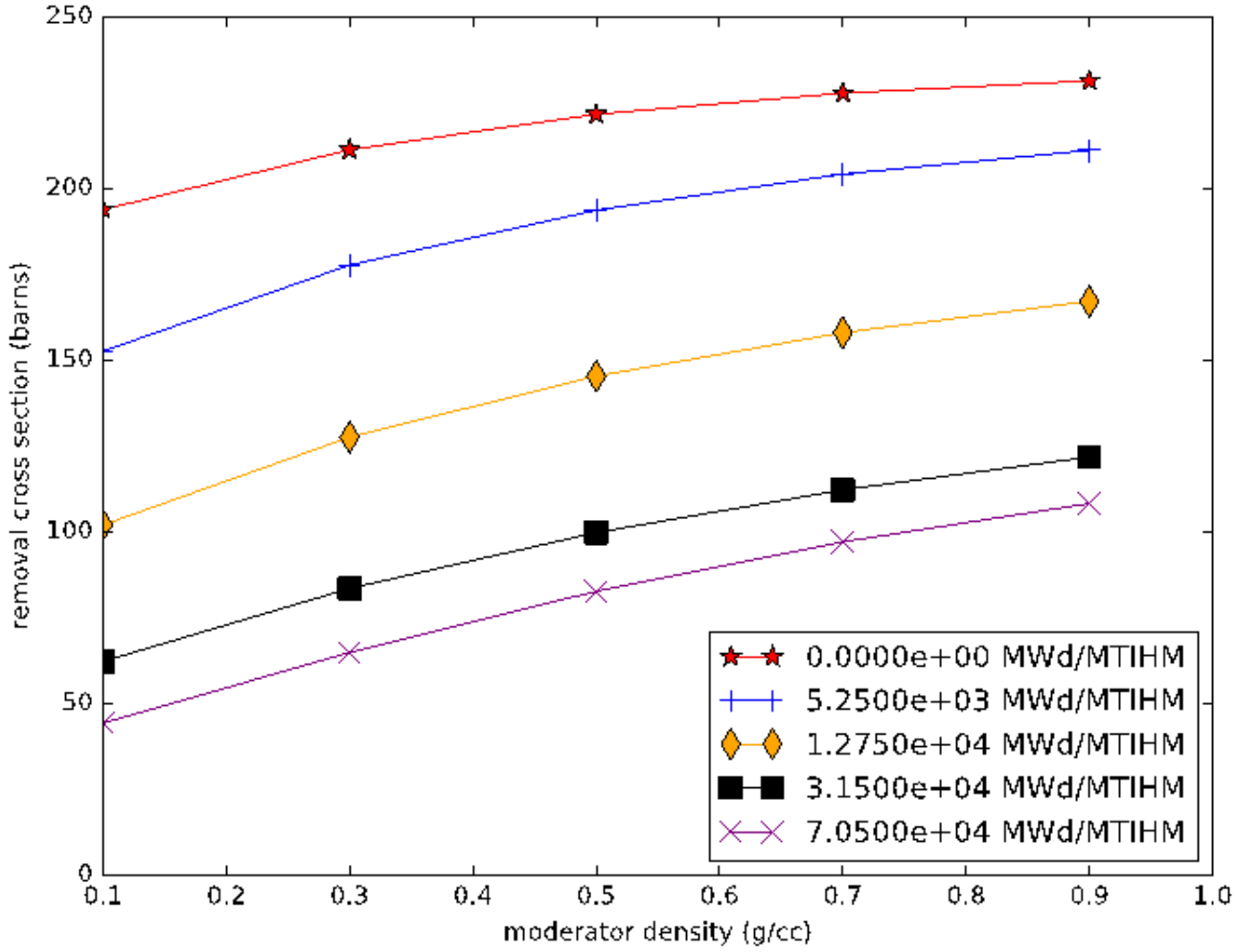
# Relative removal xs for various nuclides vs. burnup
W17x17



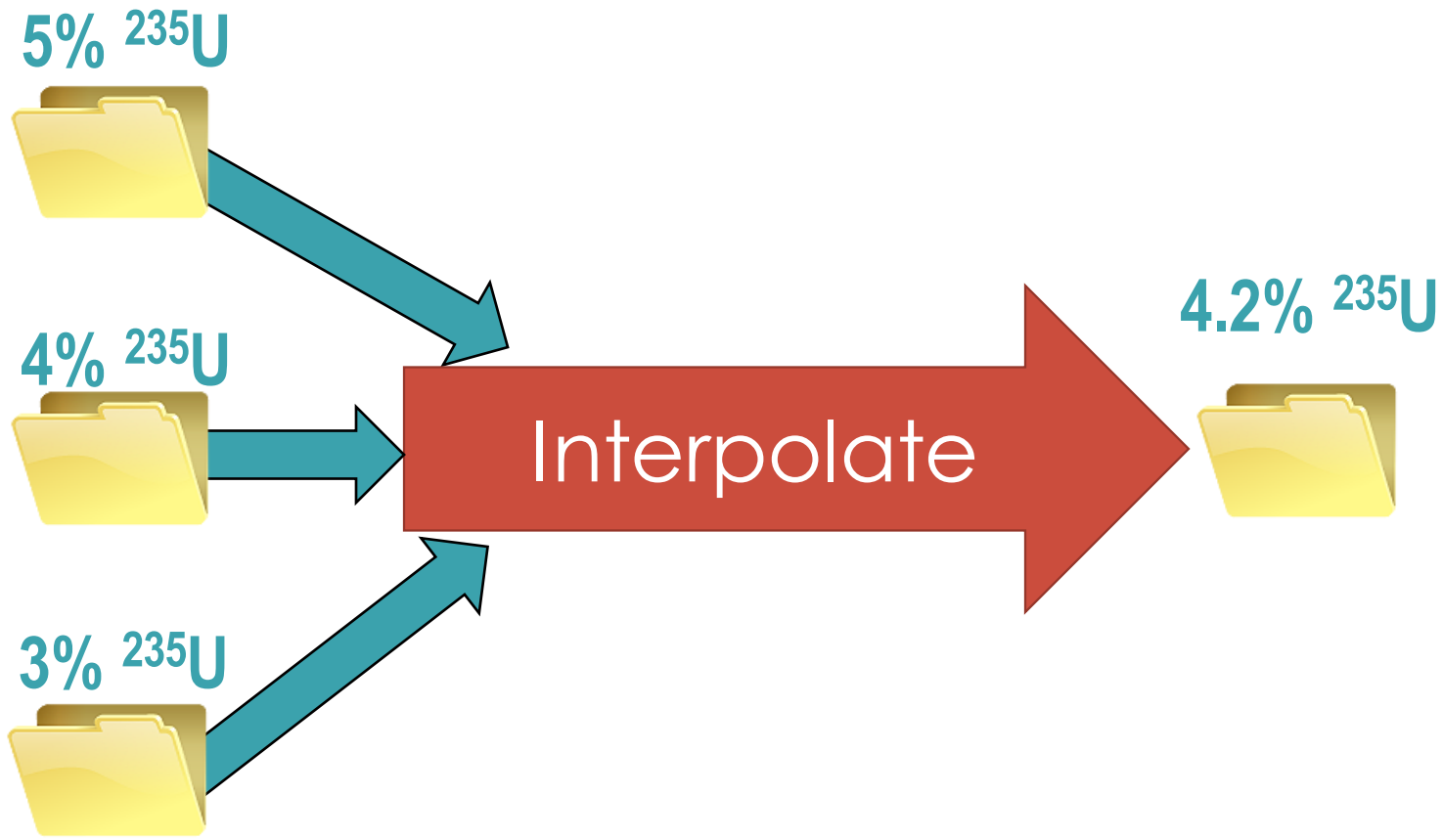
# Pu-240 removal xs vs. burnup and enrichment
## GE 10x10



# Pu-240 removal xs vs. burnup and moderator density
## GE 10x10



# ARP allows you to interpolate pre-generated ORIGEN reactor libraries (.f33) to problem-specific conditions



# ARP input format (LEU / UOX)

```
{ Configuration name }
{ enrichment [% U-235] }
{ # of cycles }
{ Fuel irradiation period(s) [days] }
{ Average cycle power [MW/MTU] }
{ # interpolations per cycle }
{ Moderator density [g/cm3] }
{ Interpolated library name }
```

e.g, w17x17

$^{235}$U initial wt. %

1 irradiation period per cycle

1 power (MW/MTU) per cycle

e.g.: w17_interpolated.f33

# ARP input example (UOX)

```
=arp
w17x17
3.65
3
400 400 350
40 35 30
1 1 1
0.732
w17_interpolated.f33
end
```

Westinghouse 17x17 lattice

3.65 % initial enrichment

3 irradiation cycles

400, 400, & 350 day cycle lengths

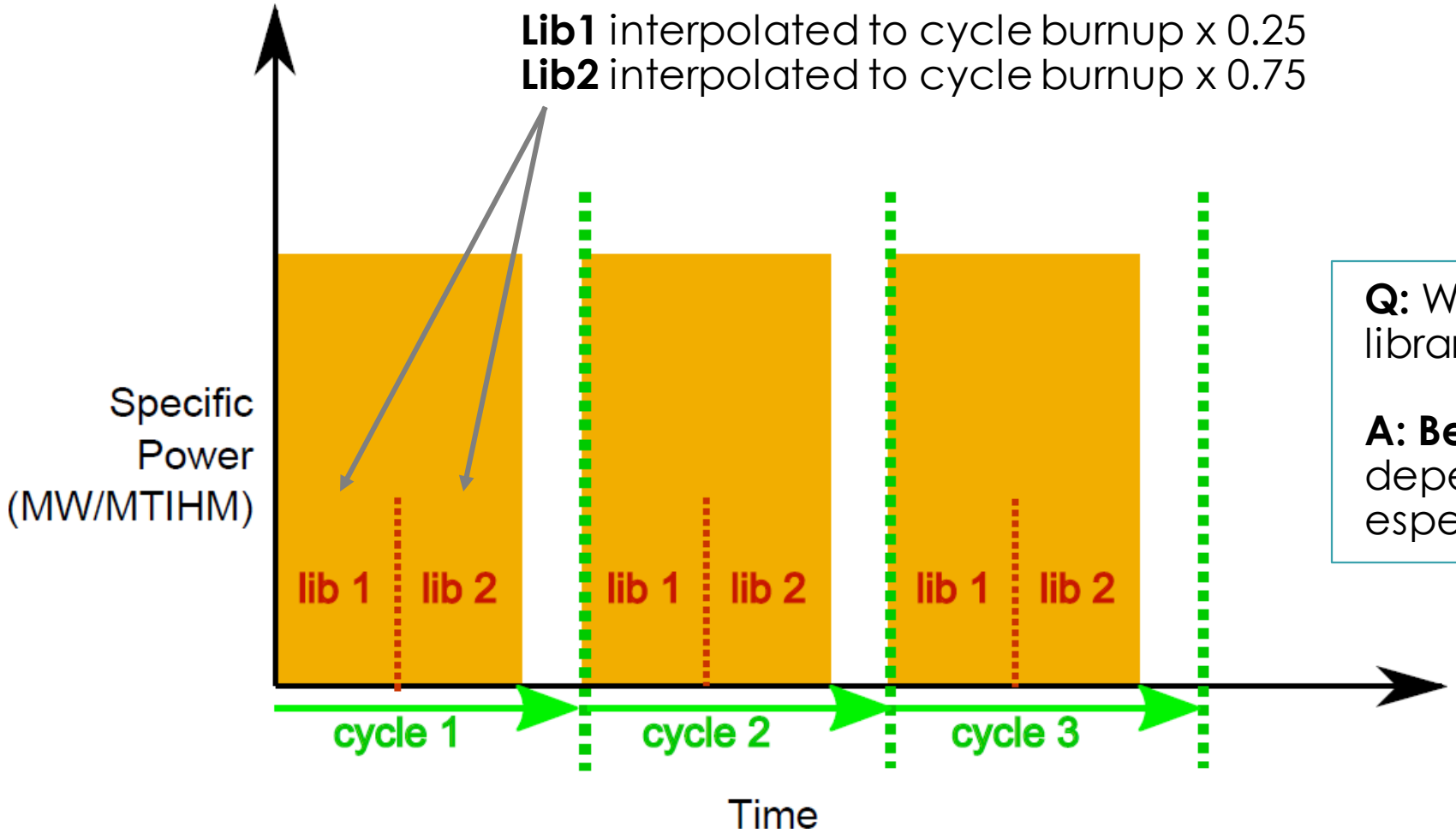
40, 35, & 30 MW/MTU

1 library per cycle

0.732 g/cm$^3$ mod. density

Interpolated library name

# Controlling cross section interpolation (libraries per cycle)




**Q:** Why would you use multiple libraries within a cycle?

**A: Better granularity** of burnup-dependent XS changes, especially at low burnups!

# Interpolation dimensions supported by ARP

**UOX fuel**

- Initial $^{235}U$ enrichment (wt.%)
- Average moderator density (g/cc)
- Burnup (MWd/MTHM)

**MOX fuel**

- Pu heavy metal fraction (%)
- $^{239}Pu$ fraction of Pu (%)
- Average moderator density (g/cc)
- Burnup (MWd/MTHM)

These dimensions have been found to be **generally** sufficient for capturing cross-section behavior in LWR assemblies!

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# ORIGEN depletion calculations with ARP-interpolated libraries

# Example: ARP-interpolated libraries to represent **problem-specific conditions** in ORIGEN calculations

```
=arp
w17x17
3.65
3
400 400 350
40 35 30
1 1 1
0.732
w17_interpolated.f33
end
```

```
=origen
case(cycle1){
  mat{ iso=[u235=36500 u238=963500] units=grams }
  lib{ file="w17_interpolated.f33" pos=1 }
  power=[10R40.0]
  time=[8I 40.0 400.0]
}
```

# Example: ARP-interpolated libraries to represent **problem-specific conditions** in ORIGEN calculations (cont.)

```
=arp
w17x17
3.65
3
400 400 350
40 35 30
1 1 1
0.732
w17_interpolated.f33
end
```

```
=origen
case(cycle1){
  mat{ iso=[u235=36500 u238=963500] units=grams }
  lib{ file="w17_interpolated.f33" pos=1 }
  power=[10R40.0]
  time=[8I 40.0 400.0]
}
```

Specify ARP-interpolated filename for first case in ORIGEN input

# Example: ARP-interpolated libraries to represent **problem-specific conditions** in ORIGEN calculations (cont.)

```
=arp
w17x17
3.65
3
400 400 350
40 35 30
1 1 1
0.732
w17_interpolated.f33
end
```

```
=origen
case(cycle1){
  mat{ iso=[u235=36500 u238=963500] units=grams }
  lib{ file="w17_interpolated.f33" pos=1 }
  power=[10R40.0]
  time=[8I 40.0 400.0]
}
```

Powers / burn time corresponds to first cycle entry in ARP input

ORIGEN works in units of **absolute** power (MW); ARP uses **specific power** (MW/MTU).
**Always check your ORIGEN mass basis!**

# Example: ARP-interpolated libraries to represent **problem-specific conditions** in ORIGEN calculations (cont.)

```
=arp
w17x17
3.65
3
400 400 350
40 35 30
1 1 1
0.732
w17_interpolated.f33
end
```

```
=origen
case(cycle1){
  mat{ iso=[u235=36500 u238=963500] units=grams }
  lib{ file="w17_interpolated.f33" pos=1 }
  power=[10R40.0]
  time=[8I 40.0 400.0]
}
case(cycle2){
  lib{ pos=2 }
  power=[10R35.0]
  time{ start=0 t=[8I 40.0 400.0] }
}
```

Repeat for each additional cycle, updating the library position for each irradiation interval

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# Workshop:

# Depletion calculations using ORIGEN + ARP

# Problem 1: PWR fuel at 10 years cooling time

- **Goal:** Determine the **concentrations** of significant nuclides (table at right) at 10 years post-discharge
- **Approach:**
  - Assume a W17x17 reactor library at 5.0% $^{235}U$ enrichment
  - Initial fuel composition in grams/basis (1 MTU basis)

    U-234 = 445

    U-235 = 50000

    U-236 = 230

    U-238 = 949325

    O = 1.34432e5

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

# Problem 1: PWR fuel at 10 years cooling time (cont.)

- Irradiation history data (use as input)
  - 3 reactor cycles
  - Cycle length: 540 days per cycle
  - Downtime between cycles: 30 days
  - Cooling time (after 3rd cycle): 10 years
  - Specific power (MW/MTU) for each cycle: 20, 19, 18
- Number of libraries per cycle for use with ARP: 2, 2, 1

# Problem 2: Forsmark BWR assembly decay heat

**Goal:** Determine the decay heat for Forsmark Unit 3 BWR fuel assembly ID=13847 and compare to measured decay heat at 4,871 days after discharge (NUREG/CR-6972)

**Hint:** calculate decay heat for 1 MTU basis, then scale to assembly initial U loading

**Assumptions:**

- SVEA100-0 cross section library
- Moderator density: 0.3816 g/cm$^3$
- Enrichment: 2.77 wt% $^{235}$U
- Assume that initial U contains only $^{235}$U & $^{238}$U
- Assembly initial U loading: 180.67 kg
- Light element loading: Table 1
- Irradiation history data: Table 2
- Power data: Table 3
- Measured decay heat: 169.6 W
- Assume 1 library per cycle

# Problem 2: Forsmark BWR assembly decay heat

**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 |

# Problem 2: Forsmark BWR assembly decay heat

**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 | 7,857 | 17,217 | 23,779 | 31,275 |

# Problem 2: Forsmark BWR assembly decay heat

**Table 3. Assembly 13847 power data**

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

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# Workshop :

# Depletion calculations with ORIGEN+ARP:

# **Solutions**

# Problem 1: PWR fuel at 10 years cooling time – ARP input

```
=arp
' Reactor library name
w17x17
' Enrichment (wt% U235)
5
' Number of cycles
3
' Cycle duration (days)
540 540 540
' Specific power per cycle (MW/MTU)
20 19 18
' Number of interpolated libraries per cycles
2 2 1
' Moderator density (g/cm3)
0.723
' Name for new generated library
w17x17.f33
end
```

# Problem 1: PWR fuel at 10 years cooling time

| Nuclide | g/MTU |
|---|---|
| U-235 | 21,959 |
| Pu-239 | 6,103 |
| Pu-240 | 1,715 |
| Cs-137 | 883.7 |
| Pu-241 | 665.1 |
| Am-241 | 467.2 |
| Pu-238 | 115.0 |
| Cm-244 | 5.308 |
| Gd-155 | 4.450 |
| Sm-149 | 3.490 |
| Cs-134 | 2.619 |

# Problem 2: Forsmark BWR assembly decay heat – ARP input

```
=arp
' Reactor library name
svea100-0
' Enrichment (wt% U235)
2.77
' Number of cycles
4
' Cycle duration (days)
311 369 277 365
' Specific power per cycle (MW/MTU)
25.26 25.37 23.69 20.54
' Number of interpolated libraries per cycle
1 1 1 1
' Moderator density (g/cm3)
0.3816
' Name for new generated library
forsmark.f33
end
```

# Problem 2: Forsmark BWR assembly decay heat

- Calculated decay heat at 4,871 days after discharge
  - 940.20 W/MTU
  - 169.87 W/assembly
- **Measured decay heat:** 169.6 W (per assembly)