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# Depletion Calculations with TRITON

## SCALE 6.3.3



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

U.S. DEPARTMENT OF ENERGY

# Learning goals

- **Understand** TRITON's depletion approach
  - Predictor–corrector scheme
  - Power normalization
  - Available neutron transport codes
- **Understand** TRITON's various output files
- **Learn** how to set up a TRITON model
  - DEPLETION and BURNDATA blocks
  - ALIAS and KEEP blocks
  - TIMETABLE block
  - COLLAPSE, HOMOGENIZE, ADF, and BRANCH blocks
  - ADDNUX parameter

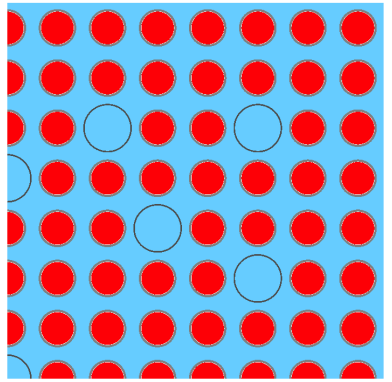
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# What is TRITON?

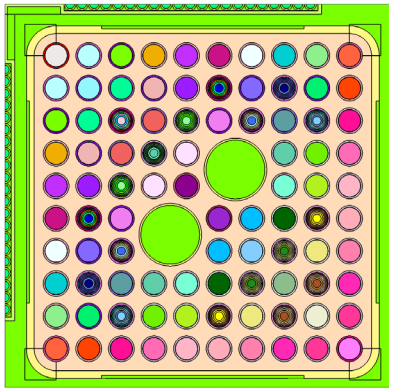


# What is TRITON?

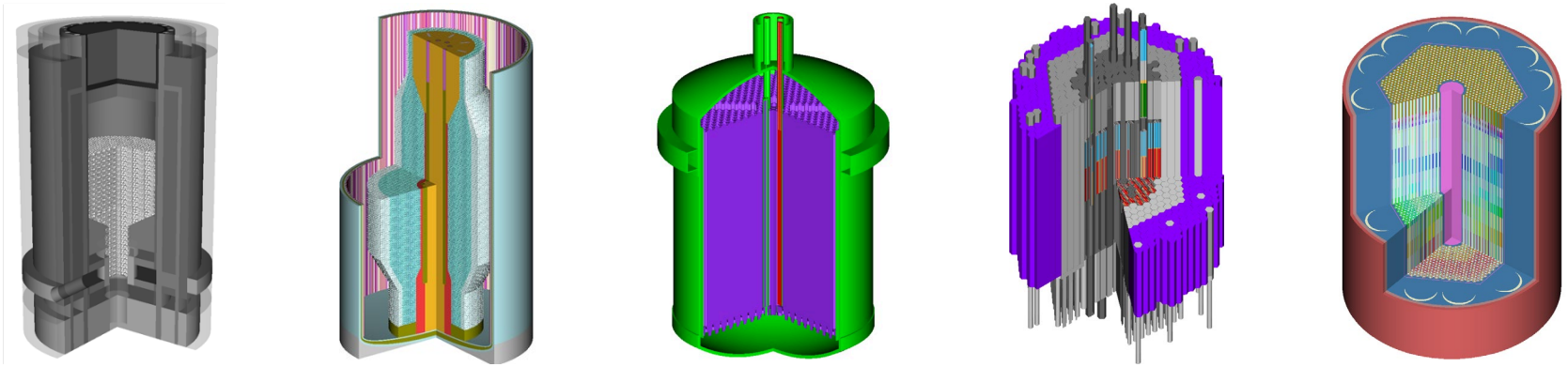
- TRITON is SCALE's general reactor physics sequence
- Main purpose:
  - Generation of accurate **burnup-dependent fuel inventory**
  - Generation of nodal data (few-group cross sections)



PWR 1/4th assembly



BWR assembly



HTGR FHR MSR SFR HPR

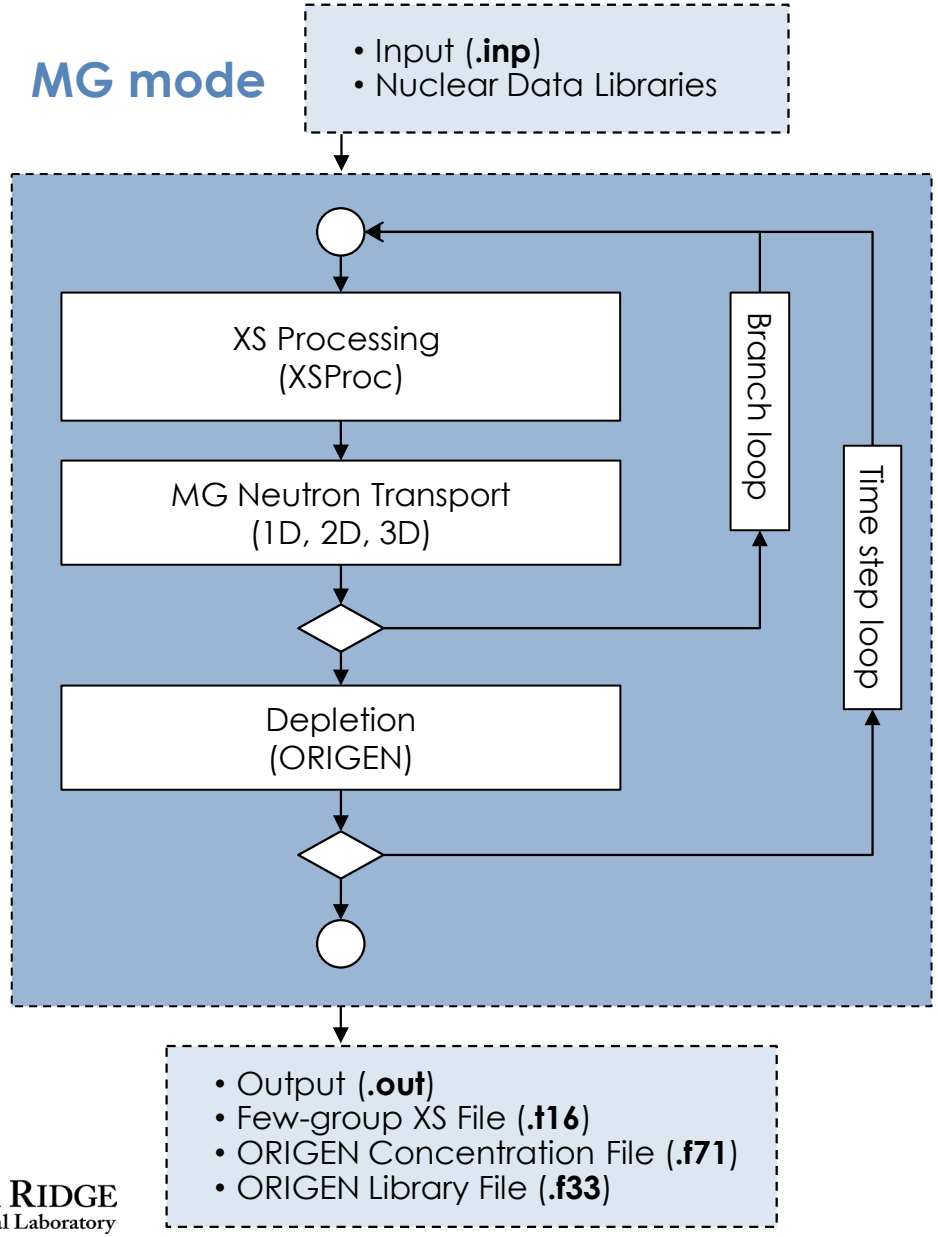
**Advanced reactor (non-LWR) models**

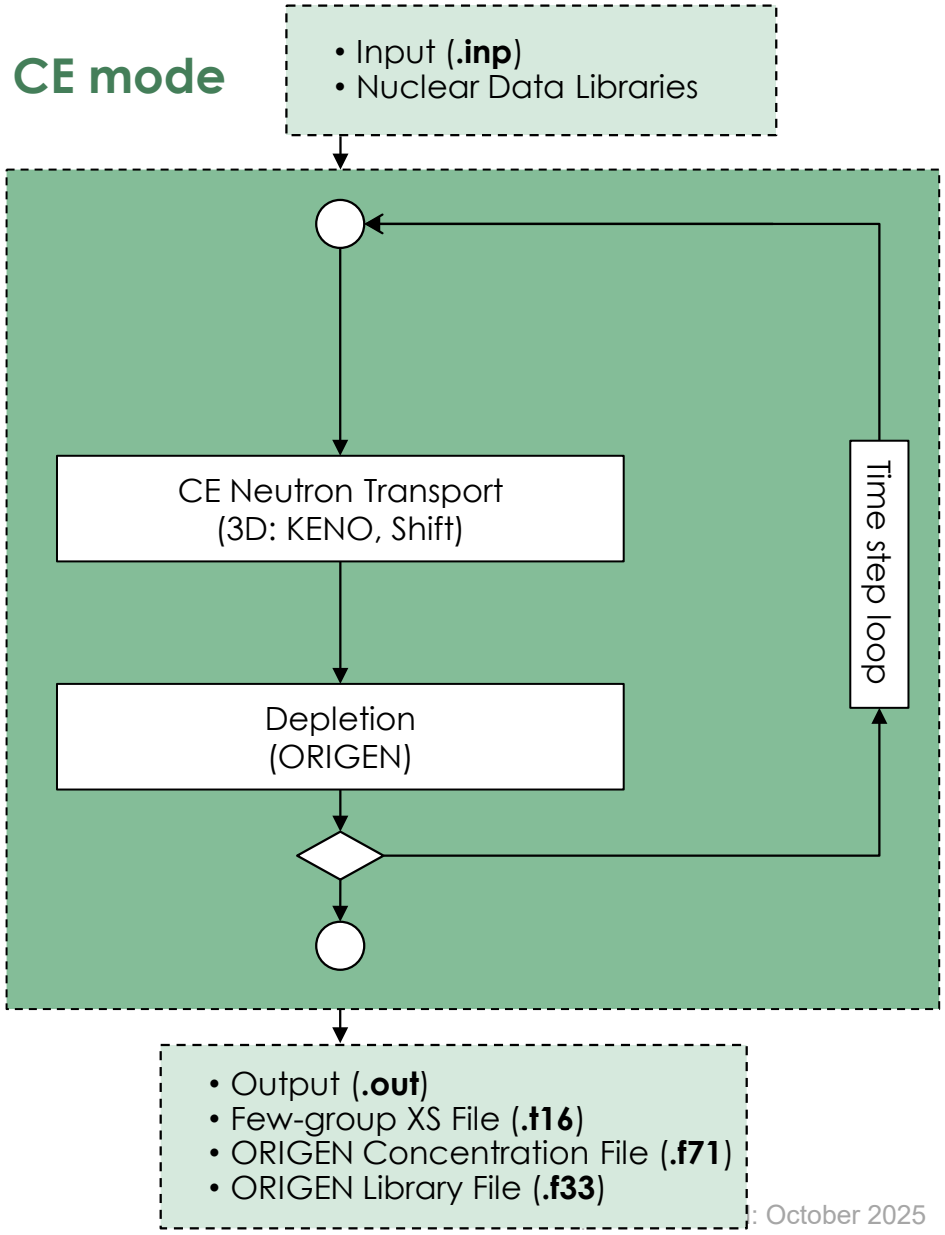
**LWR models**

# What is TRITON?

- TRITON uses other SCALE modules to perform coupled neutron transport–depletion calculations:
  - XSProc for **cross-section processing** (in MG mode)
  - XSDRN (1D), NEWT (2D), KENO V.a (3D), KENO-VI (3D), or Shift (3D) for **neutron transport**
  - ORIGEN for **depletion**
- TRITON manages communication between the modules and prepares the data:
  - Normalizes power/flux levels
  - Sets up the required tallies for the neutron transport kernel
  - Prepares transition matrices for ORIGEN
  - Manages time-stepping (predictor–corrector)
  - Performs user-requested model updates (mixture temperature, nuclide concentration in the coolant [e.g., boron concentration], and mixture swaps [e.g., control rod insertion])
- TRITON offers different geometries and physics with different accuracy and runtime:
  - 1D, 2D, or 3D geometry
  - Multi-group (MG) or continuous-energy (CE) physics

# TRITON flowchart



# Main input blocks

- **Sequence identifier and parameters**
  - t-xsdrn, t-newt, t-depl-1d, t-depl, t5-depl, t6-depl, t5-depl-shift, t6-depl-shift
  - addnux option: selection of nuclides considered during neutron transport
- **Alias**
  - Definition of aliases for compositions to simplify input
- **Composition**
  - Material specifications (nuclides, densities, temperatures)
- **Celldata** (MG only)
  - Cells to be used for cross section self-shielding calculation with XSProc
- **Depletion**
  - List of mixtures to be depleted
  - Power normalization
- **Burndata**
  - Specific power [MW/MTIHM]
  - Depletion and decay intervals [days]
  - Number of depletion intervals
- **Timetable** (optional)
  - Time-dependent nuclide concentration and temperature changes
- **Collapse, Homogenize, ADF, Branch** (optional)
  - Lattice physics parameter generation
- **Opus** (optional)
  - Generate text files for time-dependent fuel concentrations
- **Keep** (optional)
  - Enable additional module output
- **Transport model**:
  - Parameters, geometry, volume, etc.

We need to learn these general input components first before moving on to depletion.

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# Use TRITON-NEWT or TRITON-KENO/Shift?



# NEWT vs. KENO/Shift

## NEWT deterministic code:

- Is a 2D discrete-ordinates neutron transport code based on the Extended Step Characteristic (ESC) approach
- Uses spatial discretization of the 2D geometry
- Uses an arbitrary mesh structure
- Supports Sn level symmetric quadrature and product quadrature sets
- Provides a fast solution

## KENO/Shift Monte Carlo codes:

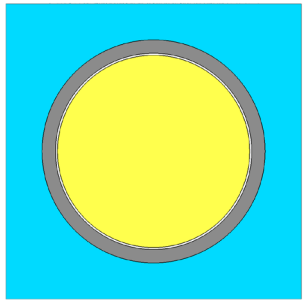
- Uses an exact 3D geometry
- Performs stochastic neutron transport
- Allows calculations in both continuous-energy (CE) and multigroup (MG) mode
- Results are reported with a statistical uncertainty
  - Relative uncertainty can be high in low-flux or low-probability regions / reaction channels
- Requires longer computation times

# KENO/Shift vs. NEWT (cont.)

- KENO/Shift can model 3D problems
  - LWRs: model axial distribution of fuel/poison
  - CANDU: model horizontal fuel, vertical control rods
  - Deplete control rod tips
  - Model complex reactor designs (pebble-bed reactor models)
- KENO/Shift can be used to verify NEWT solutions
  - Is more accurate in space and angle
  - Allows both CE and MG calculations
- NEWT can be used for lattice physics modeling
  - Efficiently generates few-group constants for core simulators
  - KENO/Shift depletion sequences do not support BRANCH calculations (see later slides)

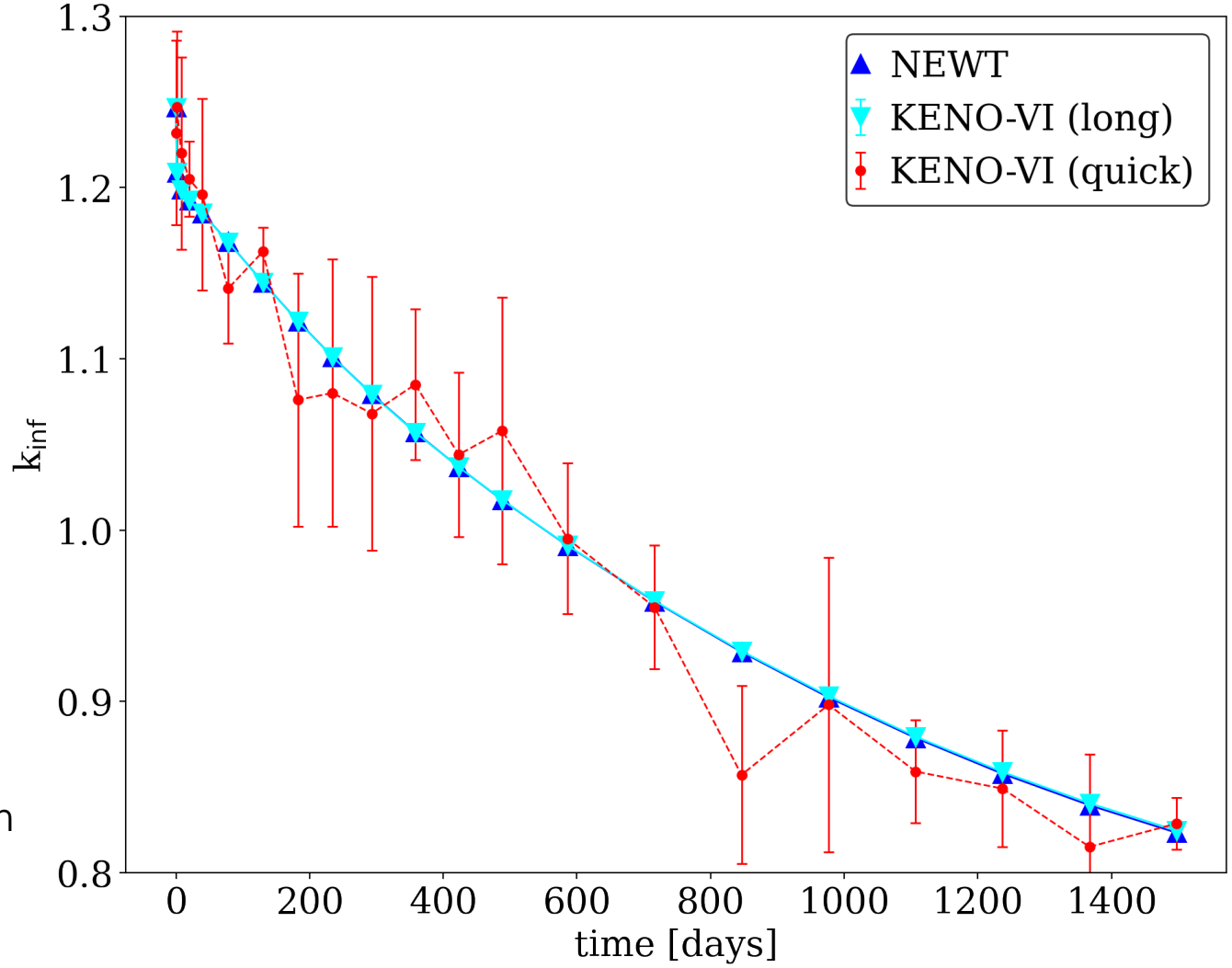
# Impact of Monte Carlo statistical uncertainty

**Example application:**



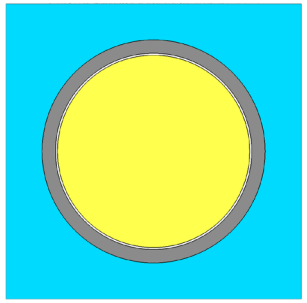
LWR fuel pin

- Model: Depletion:
  - Final burnup 60 GWd/tU
  - 20 depletion steps
  - 21 neutron transport calculations
- Neutron transport:
  - NEWT: converge to 1 pcm
  - KENO-VI long: converge to 10 pcm
  - KENO-VI quick: converge to ~3,000 pcm



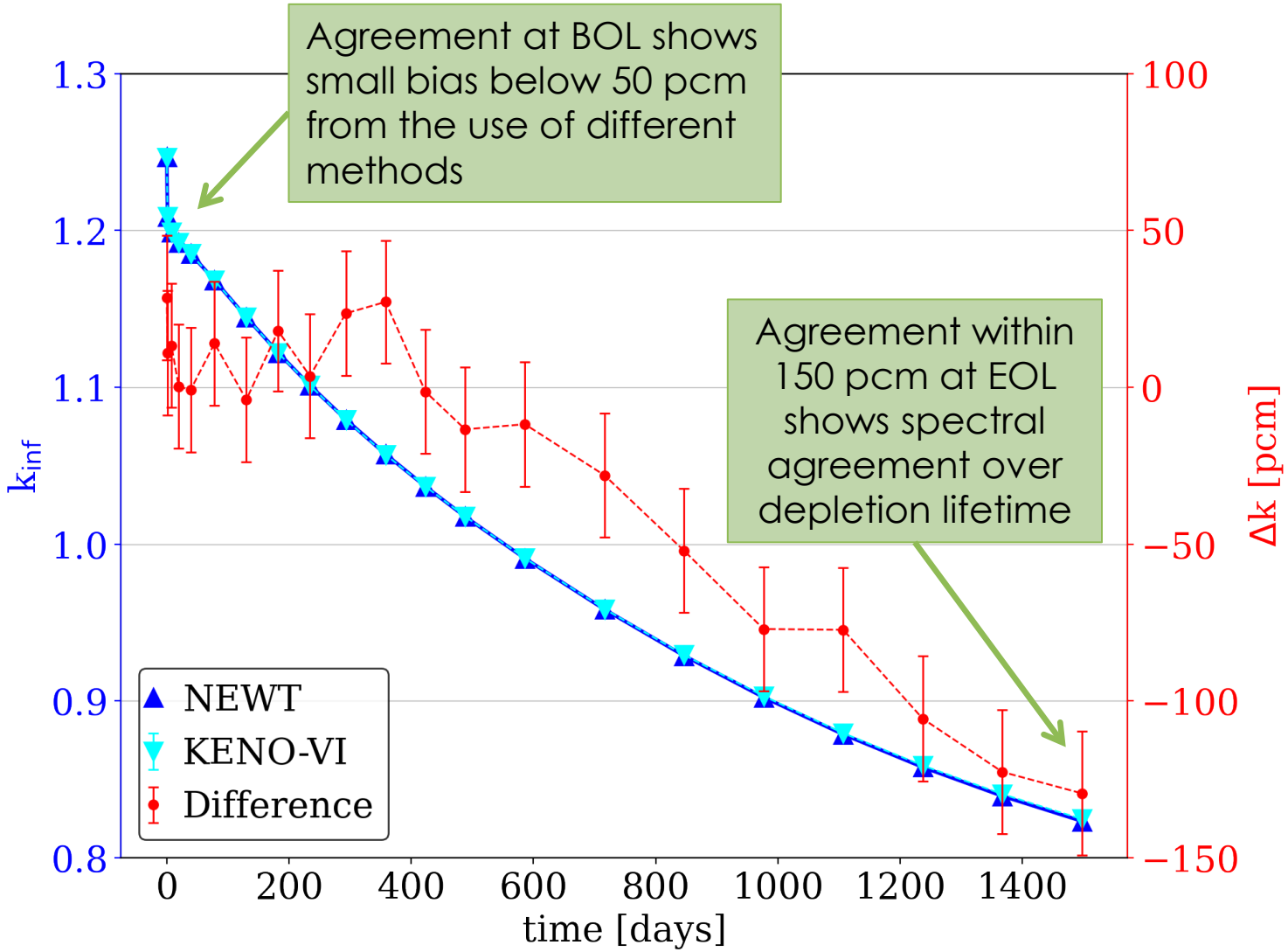
# Multiplication factor comparison

**Example application:**



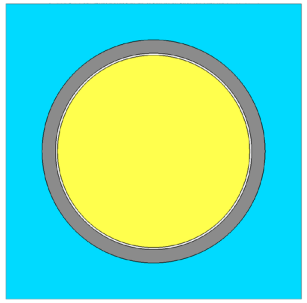
LWR fuel pin

- Model: Depletion:
  - Final burnup 60 GWd/tU
  - 20 depletion steps
  - 21 neutron transport calculations
- Neutron transport:
  - NEWT: converge to 1 pcm
  - KENO-VI: converge to 10 pcm



# Runtime comparison

**Example application:**



LWR fuel pin

- Model: Depletion:
  - Final burnup 60 GWd/tU
  - 20 depletion steps
  - 21 neutron transport calculations
- Neutron transport:
  - NEWT: converge to 1 pcm
  - KENO Va./KENO-VI/Shift: converge to 10 pcm

| Sequence | CPU hours |
|---|---|
| **TRITON-NEWT** | 0.2 |
| **TRITON-KENO-V.a** | 43.9 |
| **TRITON-KENO-VI** | 50.5 |
| **TRITON-Shift** | 111.4 |

**Major runtime contributors:**

1. Nuclear data loading
2. Cross section processing for MG calculation
3. Neutron transport calculation

***Note: Runtime difference between the Monte Carlo codes is problem-dependent!***

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# TRITON depletion method



# Terminology

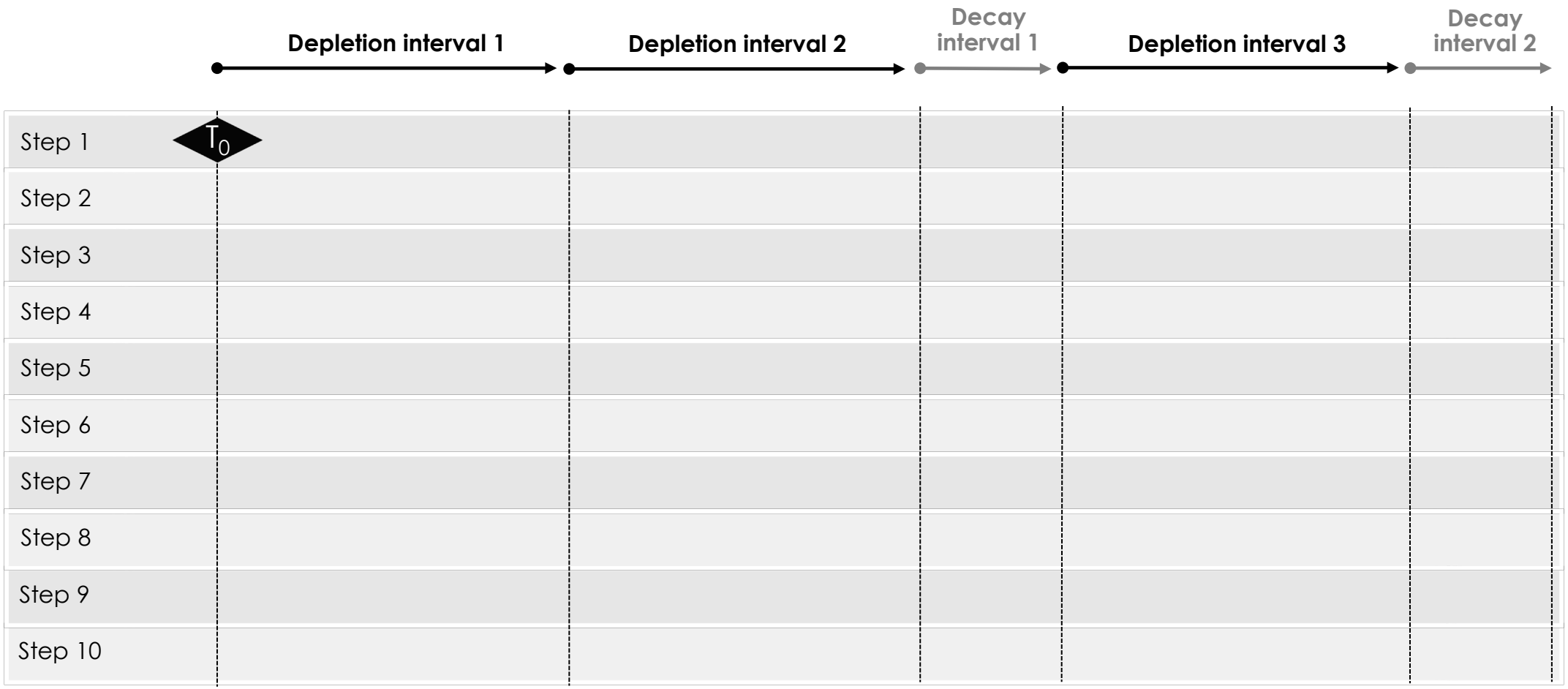
- **Depletion interval** – time interval in which power or flux level is constant
- **Decay interval** – zero-power time interval that follows depletion interval (optional)
- **ORIGEN intervals** – ORIGEN calculates nuclide concentration changes using its own set of smaller time intervals to cover a given TRITON depletion or decay interval; TRITON automatically determines ORIGEN time intervals

The same total depletion time realized through different depletion intervals will yield different results.

Fine intervals are recommended for the beginning of depletion, whereas coarser depletion steps may be sufficient later.

- Observe differences in 1-group cross sections over time
- Run sensitivity calculations with double/triple/… the time steps
- Note that the required interval size depends on the quantities of interest

# Predictor–corrector approach




$T_x$ Neutron transport

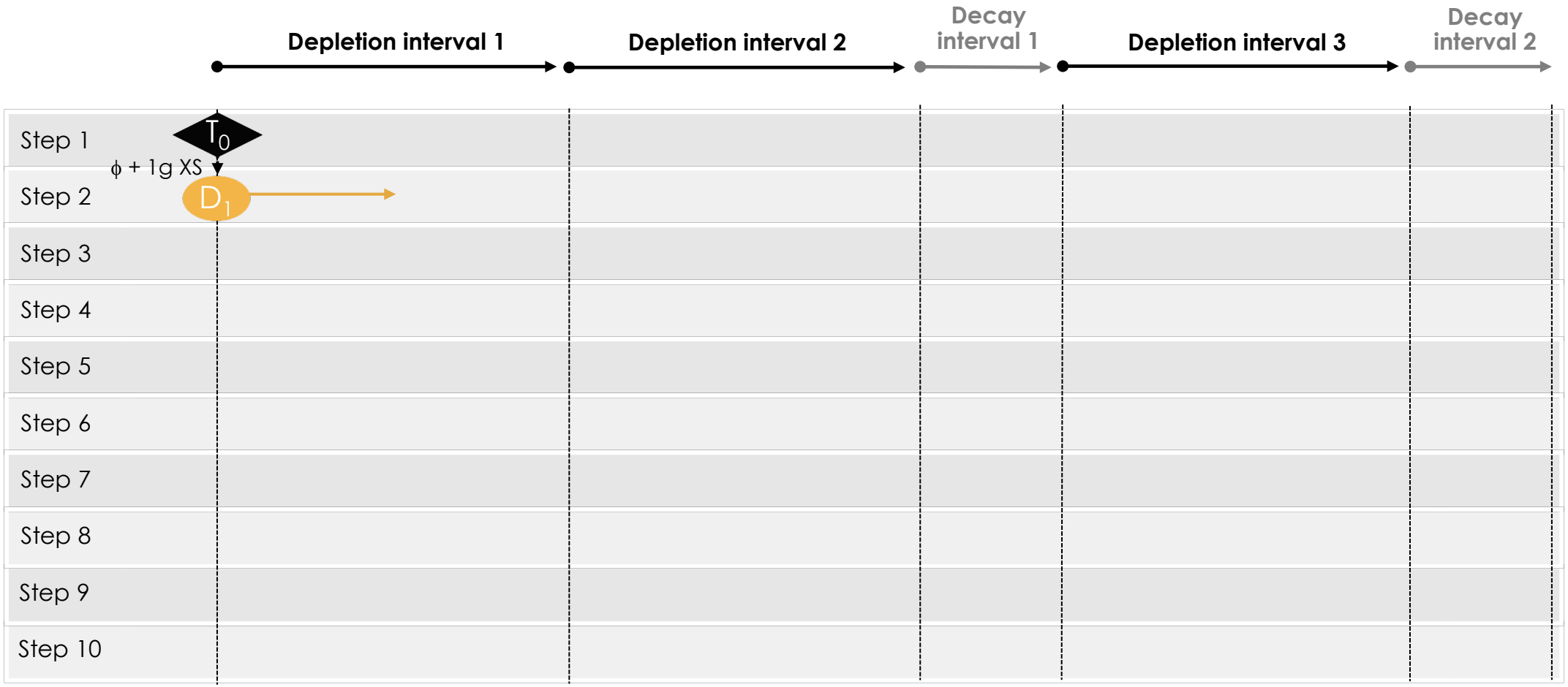
$D_x$ Depletion: predictor

$D_x$ Depletion: corrector

ND: nuclide densities

**Step 1:**
$T_0$: Cross section processing and transport calculation using initial nuclide densities.

# Predictor–corrector approach




$T_x$ Neutron transport

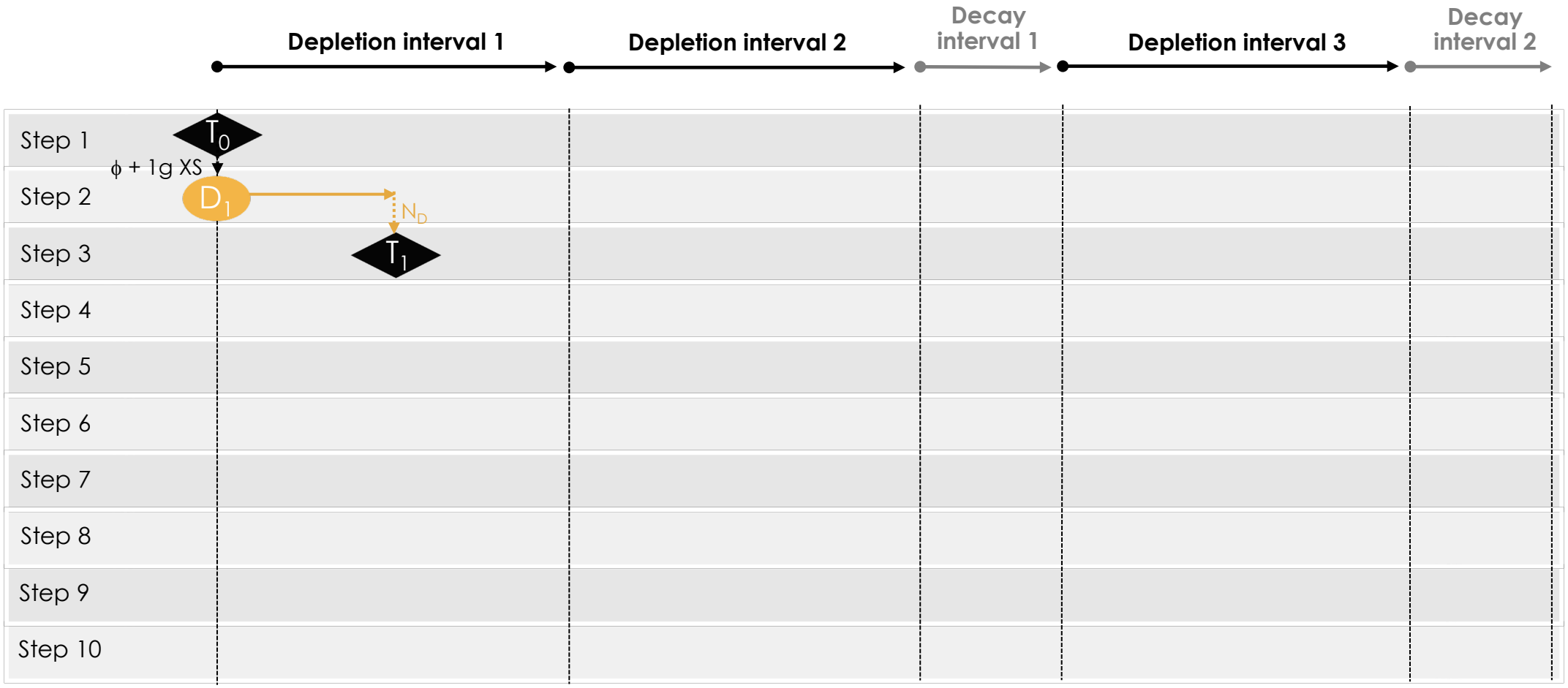
$D_x$ Depletion: predictor

$D_x$ Depletion: corrector

ND: nuclide densities

**Step 2:**
$D_1$: Depletion calculation from t=0 to the midpoint of the first depletion interval. This "predictor" step uses t=0 1-group XS and flux distributions.

# Predictor–corrector approach




$T_x$ Neutron transport

$D_x$ Depletion: predictor

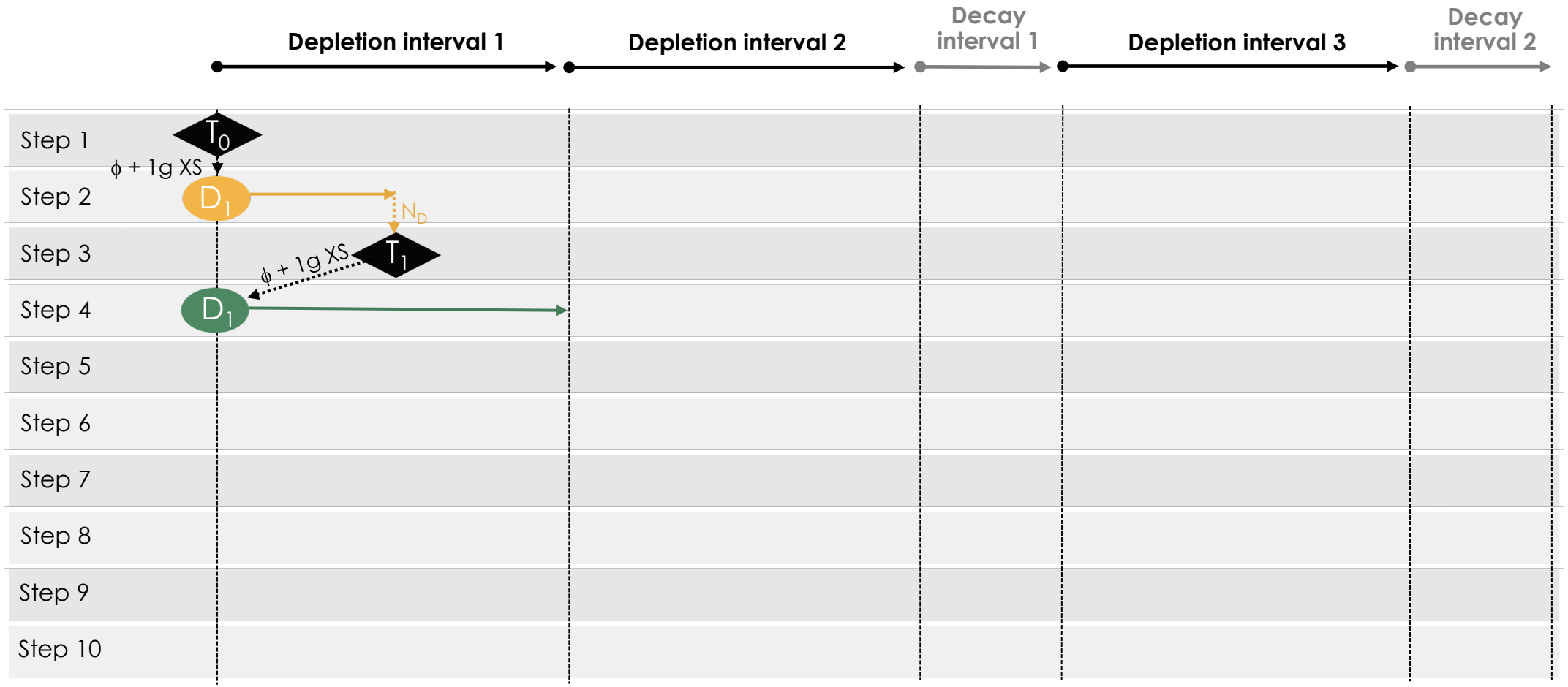
$D_x$ Depletion: corrector

ND: nuclide densities

**Step 3:**

$T_1$: Cross section processing and transport calculation at the midpoint of the first depletion interval.

# Predictor–corrector approach




Tx Neutron transport

Dx Depletion: predictor

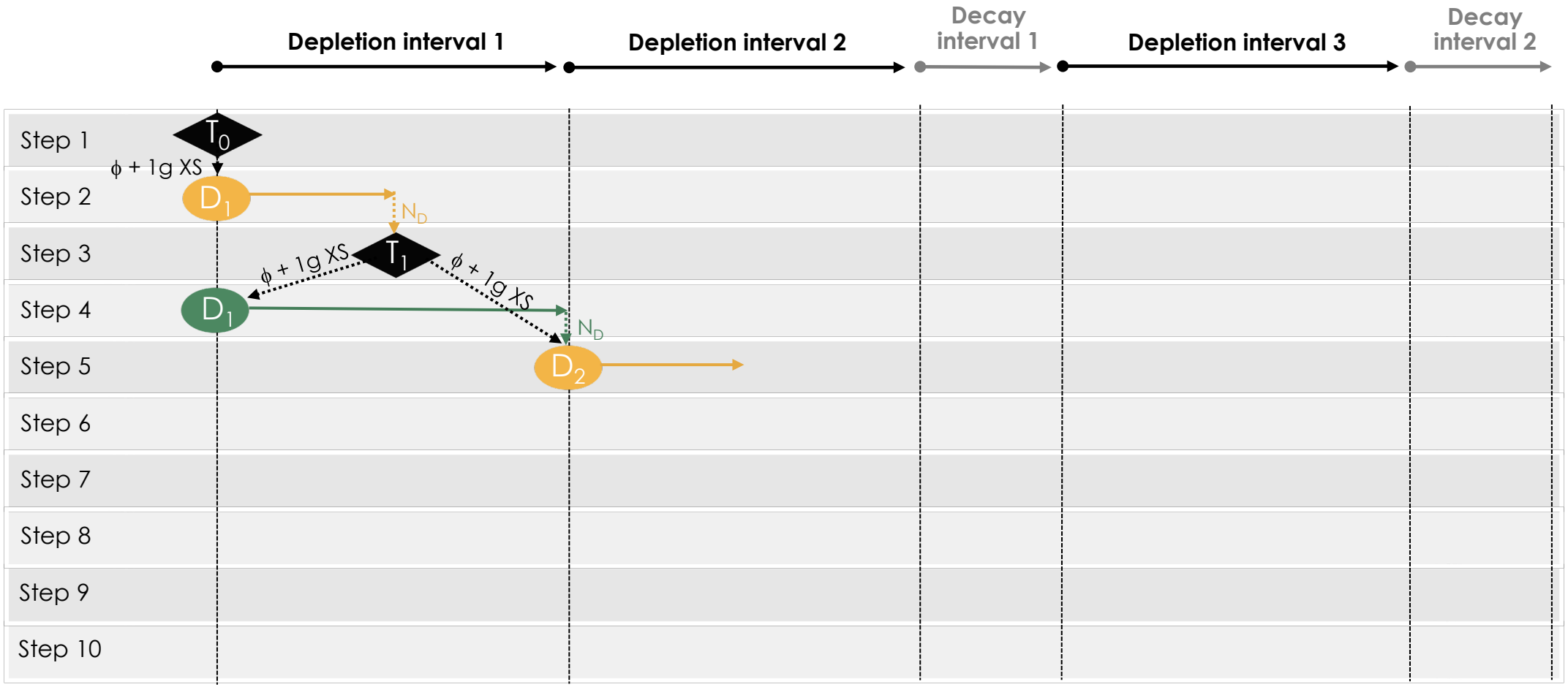
Dx Depletion: corrector

ND: nuclide densities

**Step 4:**

$D_1$: Depletion calculation for the first depletion interval. This "corrector" step uses 1-group XS and flux distributions from the interval midpoint.

# Predictor–corrector approach




Tx Neutron transport

Dx Depletion: predictor

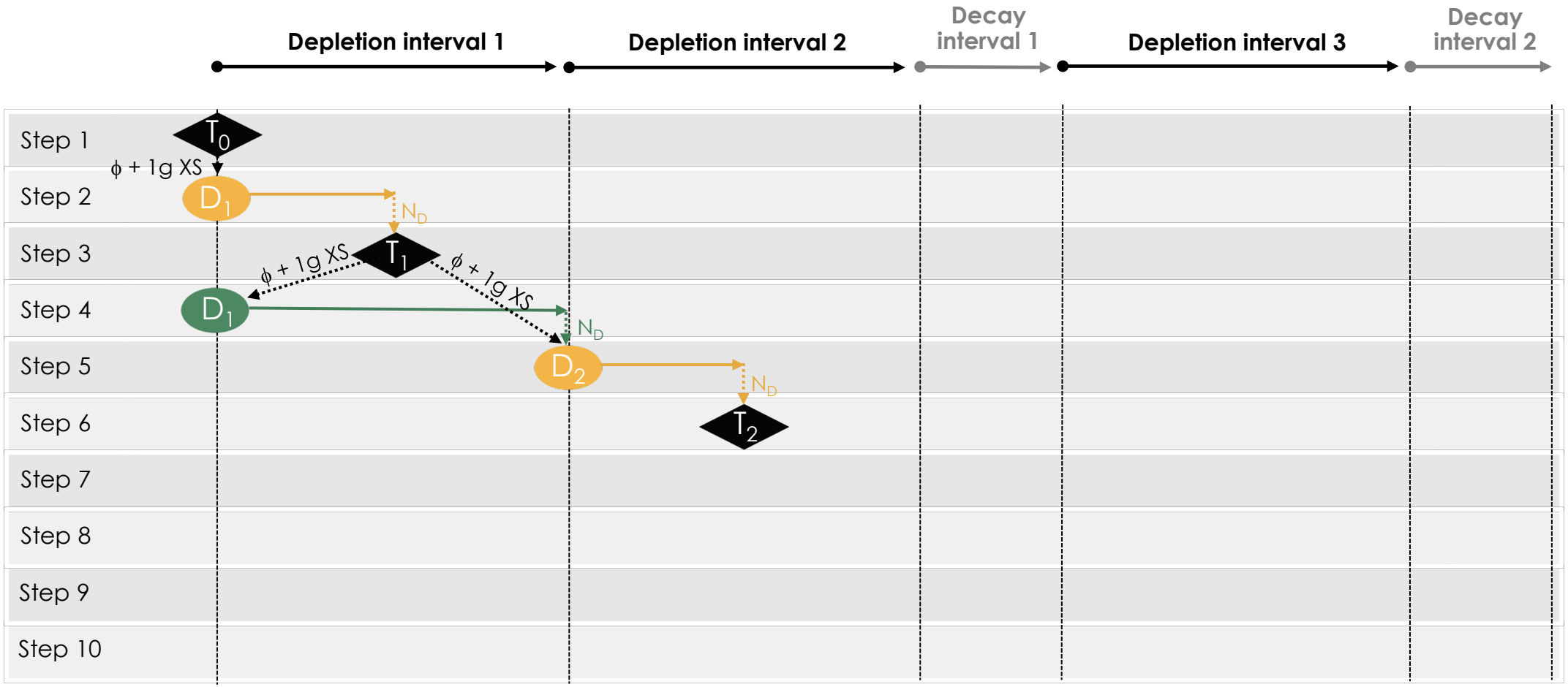
Dx Depletion: corrector

ND: nuclide densities

**Step 5:**

$D_2$: Depletion calculation to the midpoint of the second depletion interval. This "predictor" step uses 1-group XS and flux distributions from the **previous** interval midpoint.

# Predictor–corrector approach




Tx Neutron transport

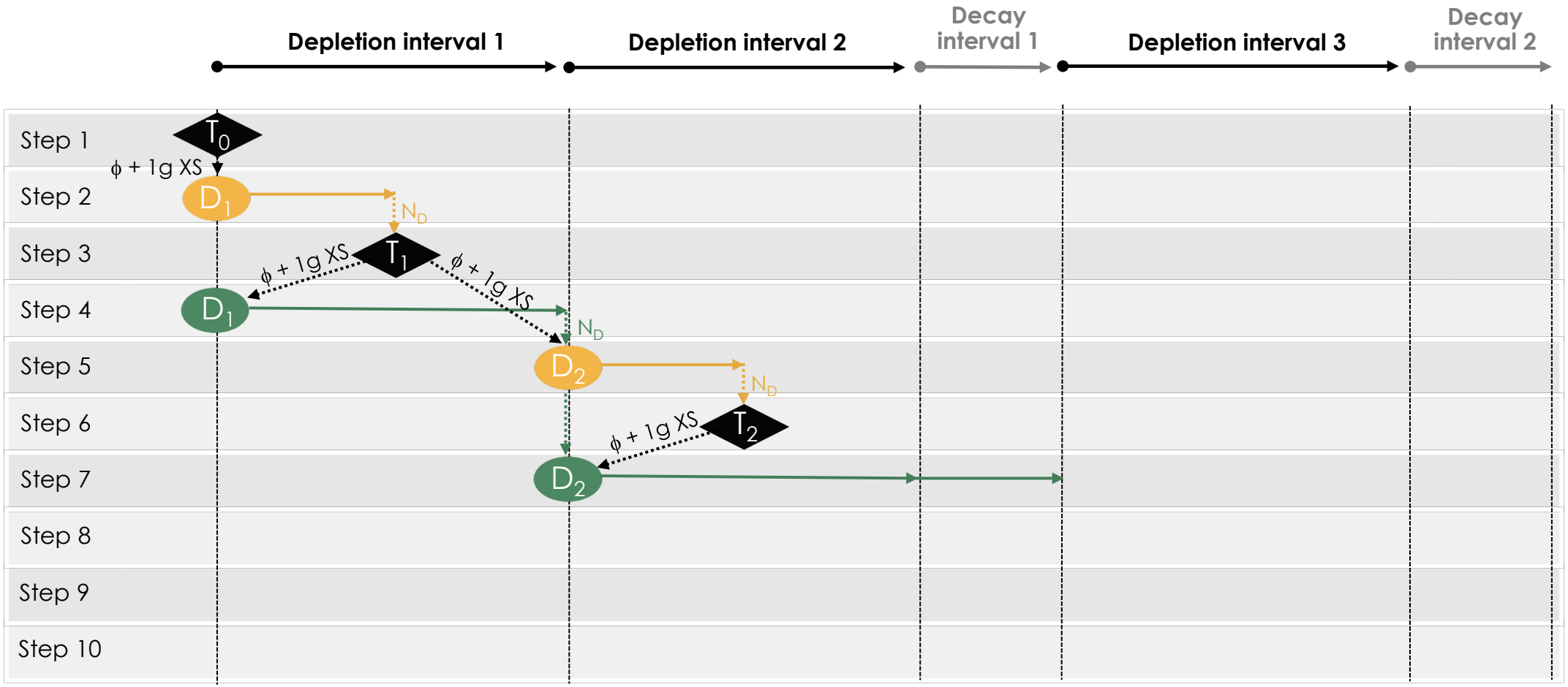
Dx Depletion: predictor

Dx Depletion: corrector

ND: nuclide densities

**Step 6:**
$T_2$: Cross section processing and transport calculation at the midpoint of the second depletion interval.

# Predictor–corrector approach




**Step 7:**

$D_2$: Depletion calculation for the second depletion interval, followed by the decay calculation at the end of the first depletion interval. This "corrector" step uses 1-group XS and flux distributions from the interval midpoint.

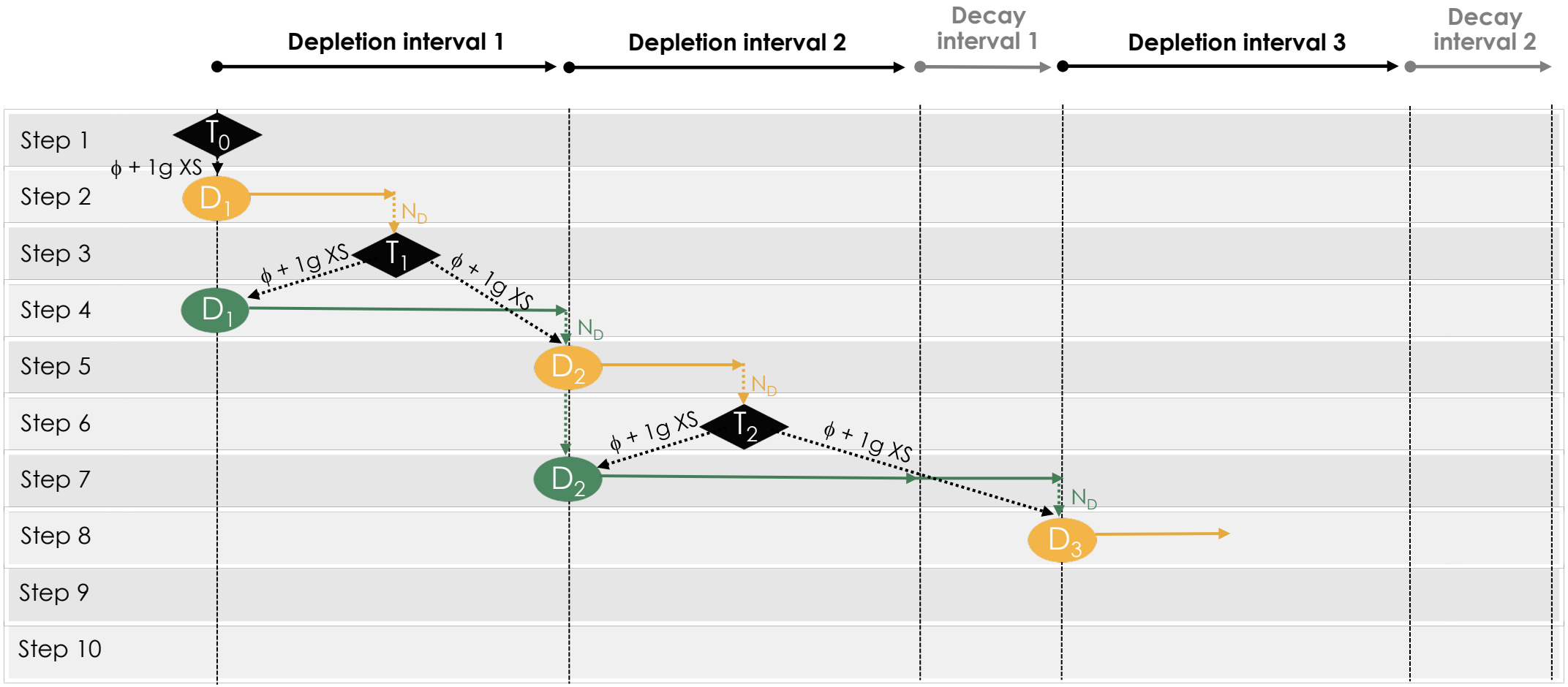
$T_x$ Neutron transport

$D_x$ Depletion: predictor

$D_x$ Depletion: corrector

ND: nuclide densities

# Predictor–corrector approach




Tx Neutron transport

Dx Depletion: predictor

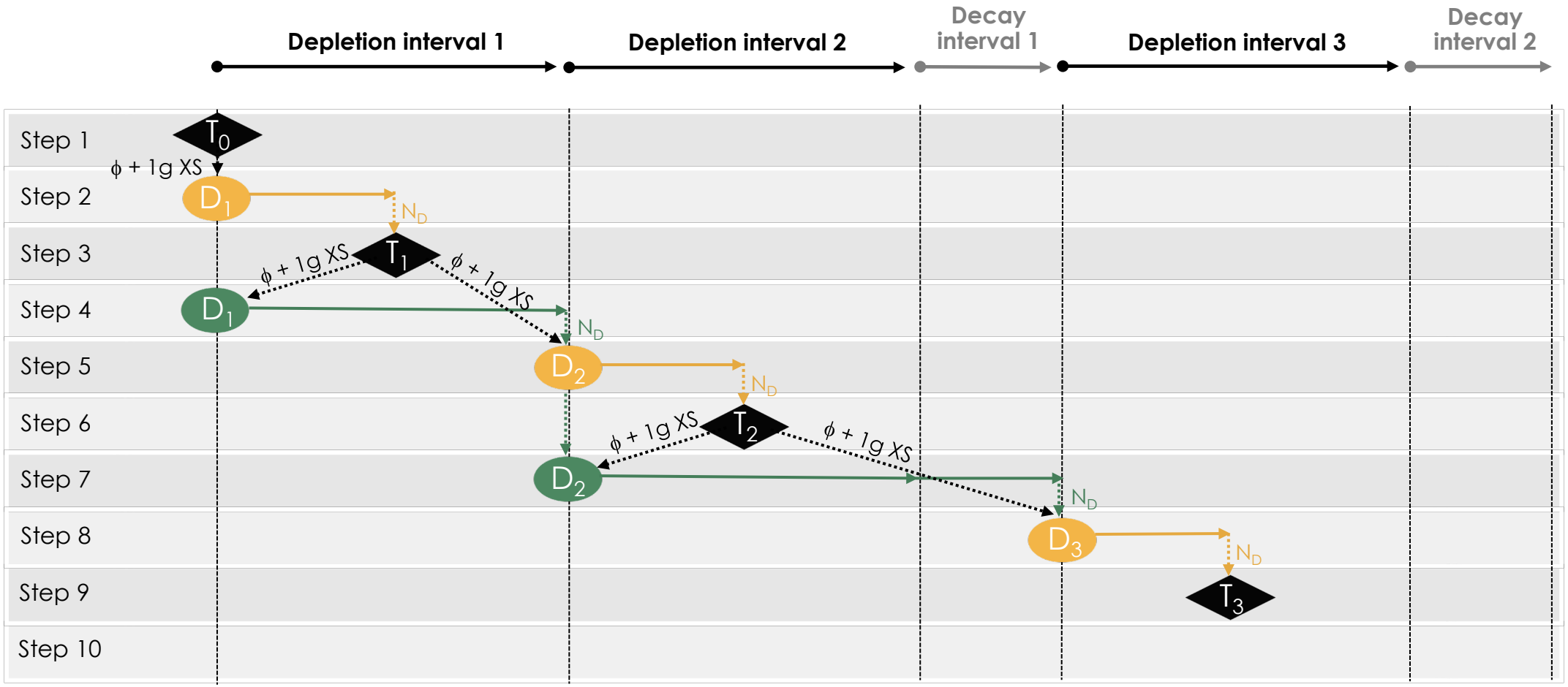
Dx Depletion: corrector

ND: nuclide densities

**Step 8:**

$D_3$: Depletion calculation for the third depletion interval. This "predictor" step uses 1-group XS and flux distributions from the **previous** depletion interval midpoint.

# Predictor–corrector approach




Tx Neutron transport

Dx Depletion: predictor

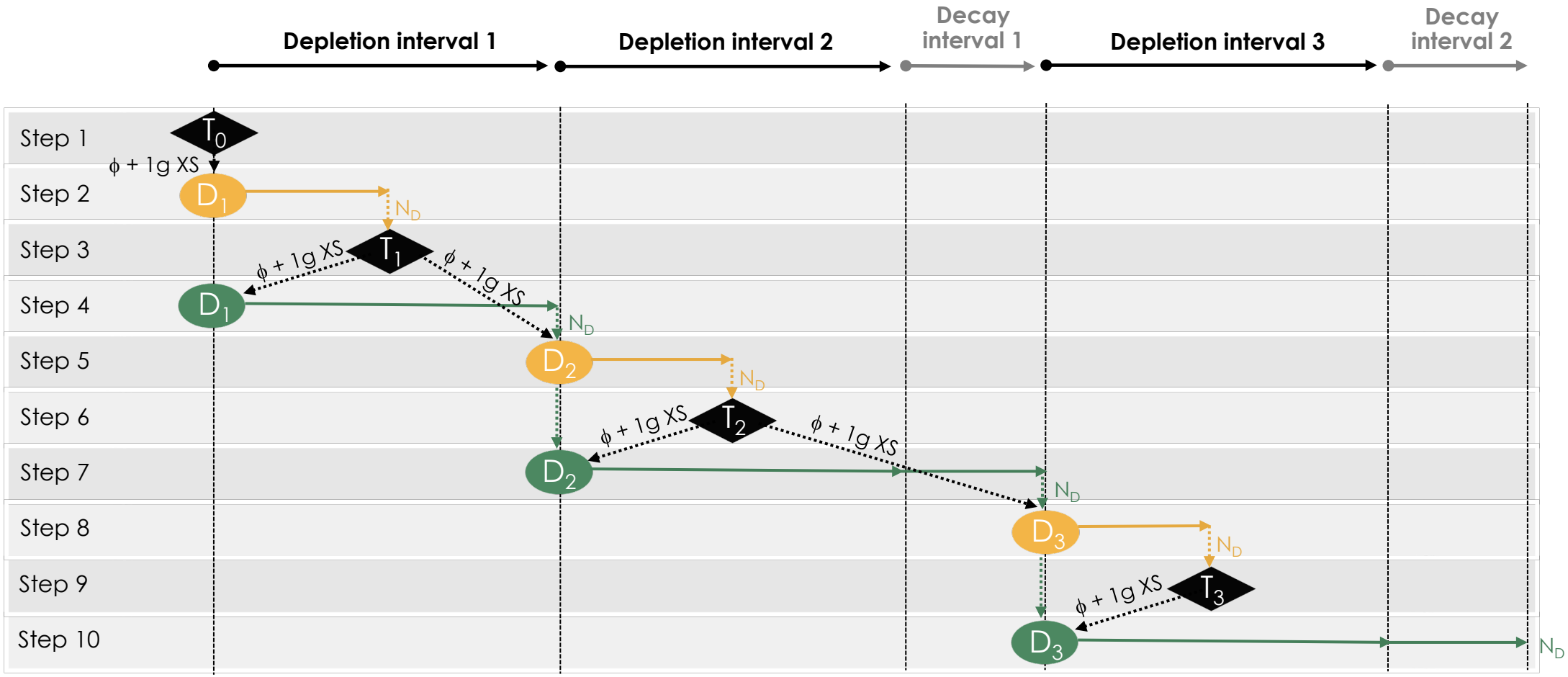
Dx Depletion: corrector

ND: nuclide densities

**Step 9:**

$T_3$: Cross section processing and transport calculation at the midpoint of the third depletion interval.

# Predictor–corrector approach




Tx Neutron transport

Dx Depletion: predictor

Dx Depletion: corrector

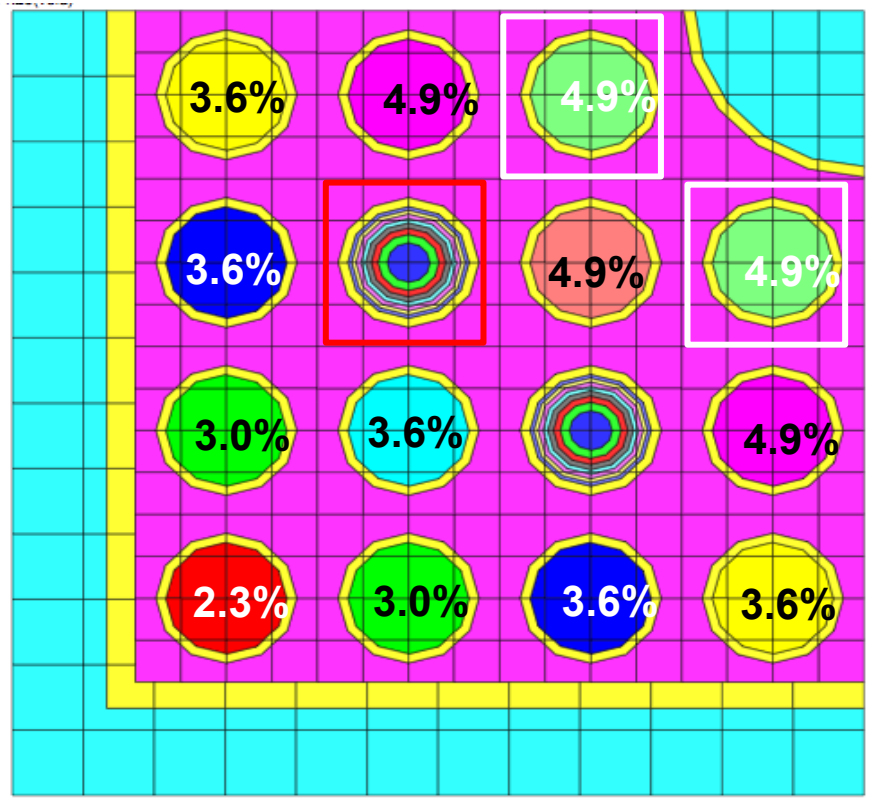
ND: nuclide densities

**Step 10:**

$D_3$: Depletion calculation for the third depletion interval, followed by a decay calculation. This "corrector" step uses 1-group XS and flux distributions from the interval midpoint.

# Multi-mixture depletion

- TRITON can deplete each fuel location in a system using local fluxes
- A unique depletion mixture must be specified for each region to be individually depleted
- ORIGEN depletion is performed based on the average flux within each mixture, irrespective of location(s)
- One fuel element can be modeled as multiple depletion mixtures (rings or regions within a contiguous volume)
- Conversely, large discontinuous regions (i.e., sets of pins) can be modeled as a single depletion mixture



# Mass normalization

- TRITON normalizes the system's mixture masses to yield a total system mass of 1 metric ton of initial heavy metal (mass of all nuclide with Z ≥ 90)
- A normalization factor is determined **once**, at the beginning of a TRITON calculation, and is then applied to all mixtures
- The mixture masses are internally scaled before ORIGEN materials are initialized, so ORIGEN concentration files (f71) always include mixtures with normalized masses
- You can scale back the data (e.g., the grams) from the f71 files using the normalization factor
- The correct scaling of mixture masses requires correct volumes

# Selection of nuclides for depletion vs. transport

- ORIGEN tracks ~1,700 nuclides during the depletion calculation and, therefore, needs 1-group cross sections for all these nuclides
  - 1-group cross sections are generated through the neutron transport calculation *for nuclides that are contained in the depleted mixtures*
  - 1-group cross sections for nuclides not considered in the neutron transport calculation are obtained based on a JEFF cross section data resource
- The neutron transport calculation does not need to consider all nuclides to generate accurate results for the neutron flux, reaction rates, etc.
  - The nuclides considered in the neutron transport calculations are defined by the initial mixture compositions plus a set of nuclides deemed relevant for neutron transport (controlled by sequence parameter ADDNUX, see later slides)
  - Trace quantities of the nuclides deemed relevant are added to the composition of depleted mixtures

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# Relevant TRITON output



# Relevant output – Depletion/decay intervals

```
**************************************************************************************
Based on the supplied burnup history, triton will use the following time history to perform
depletion calculations.  This breakdown has been calculated so as to permit TRITON depletion
steps of no more than 400 days.
**************************************************************************************

  8 Time-dependent libraries will be created
```

| Sub-Interval No. | Depletion Interval | Sub-interval in interval | Specific Power(MW/MTIHM) | Burn Length (d) | Decay Length (d) | Library Burnup (MWd/MTIHM) |
|---|---|---|---|---|---|---|
| 0 | ****Initial Bootstrap Calculation**** | | | | | 0.00000E+00 |
| 1 | 1 | 1 | 20.000 | 55.000 | 0.000 | 5.50000e+02 |
| 2 | 1 | 2 | 20.000 | 55.000 | 0.000 | 1.65000e+03 |
| 3 | 1 | 3 | 20.000 | 55.000 | 0.000 | 2.75000e+03 |
| 4 | 1 | 4 | 20.000 | 55.000 | 0.000 | 3.85000e+03 |
| 5 | 1 | 5 | 20.000 | 55.000 | 0.000 | 4.95000e+03 |
| 6 | 1 | 6 | 20.000 | 55.000 | 30.000 | 6.05000e+03 |

**NOTE: Library Burnup is the cumulative burnup computed at the midpoint of the depletion sub-interval.**
**Specific Power and Library Burnup depend on basis material(s) selected in DEPLETION block.**

**Here you can check that this is the depletion history you requested.**

# Relevant output – Power and burnup

## Power/burnup information **at time of transport**:

```
--- Material powers for depletion pass no.    1 (MW/MITHM) ---
      Time =     1.30 days (   0.004 y), Burnup =     0.050     GWd/MTIHM, Transport k=  1.2210
```

| Mixture Number | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) | Mixture Burnup (GWd/MTIHM) | Mixture Thermal Flux n/(cm^2*sec) | Mixture Total Flux n/(cm^2*sec) |
|---|---|---|---|---|---|---|
| 201 | 38.279 | 0.99684 | 38.279 | 0.050 | 4.4004e+13 | 3.1752e+14 |
| 2 | 0.000 | 0.00000 | N/A | N/A | 4.6313e+13 | 3.1728e+14 |
| 3 | 0.031 | 0.00080 | N/A | N/A | 4.7722e+13 | 3.1728e+14 |
| 4 | 0.091 | 0.00237 | N/A | N/A | 5.0346e+13 | 3.1808e+14 |
| Total | 38.400 | 1.00000 | | | | |

**Here you can check total power and power normalization.**

```
NOTE:    Total Power is the Mixture Power per 1 metric ton of HM of the initial  system mass.
         Mixture Power is the Mixture Power per 1 metric ton of HM of the initial mixture mass.
         Mixture Burnup is the Mixture Burnup per 1 metric ton of HM of the initial mixture mass.
         Mixture Thermal Flux determined using 0.625 eV cutoff: Groups 41 through 56.
```

## Power/burnup information **after depletion/decay interval**:

```
      end-of-step summary at time =    13.021  days (    0.036  y), system-average burnup* =    0.498  GWd/MTIHM
```

| mixture (-) | power (MW) | flux (n/cm2-s) | fluence (n/cm2) | burnup* (MWd/MTIHM) | initialhm (MTIHM) |
|---|---|---|---|---|---|
| 201 | 3.82783e+01 | 3.19074e+14 | 3.58614e+20 | 4.98416e+02 | 1.00000e+00 |

**This is the result of the ORIGEN calculation, considering the mass normalization.**

```
* Burnup is only calculated for mixtures with initial HM mass fraction greater than 1e-6.
```

# Relevant output – ORIGEN concentration file (F71)

- TRITON stores compositions for each mixture listed in the DEPLETION block and for some special mixtures in the F71 file at each depletion step
- Fulcrum allows viewing compositions in various units and converting to quantities like decay heat and activity
- The content of the F71 files can be viewed with the command-line utility OBIWAN (see later training)

```
File ft71f001 contains origen/opus-formatted binary data for
      - each of the 1 depletion materials,
      - a set for the sum of all depletion materials,
      - a set for the sum of selected materials (from optional opus block),
      - a set for the sum of all fuel materials
    for always 21 time steps.
    Isotopic data locations are listed according to the following table.

    Position  Time Step  Cycle Time (d)  Cumulative Time (y)
        1          0       0.0000e+00    0.0000e+00   Depletion material no. 201 (ft71 case=201)
        2          1       2.6042e+00    7.1298e-03
        3          2       1.0417e+01    3.5649e-02
        4          3       1.3021e+01    7.1298e-02
        5          4       2.6042e+01    1.4260e-01
        6          5       5.2083e+01    2.8519e-01
        7          6       5.2083e+01    4.2779e-01
        ...
       22          0       0.0000e+00    0.0000e+00   Weighted sum of concentrations of all depleted materials (ft71 case=0)
       ...

       43          0       0.0000e+00    0.0000e+00   Weighted sum of concentrations for selected materials (ft71 case=-1)
       ...
       64          0       0.0000e+00    0.0000e+00   Weighted sum of concentrations for fuel materials (ft71 case=-2)
```

**Here you can find out what is stored where on the ORIGEN concentration (F71) file.**

*Example: Concentrations for nuclides in mixture 201 for 21 time steps (depletion and/or decay steps) on positions 1–21.*

# Relevant output – ORIGEN cross section library (F33)

*Note common terminology:*
*ORIGEN cross section library*
*= ORIGEN library*
*= ORIGEN 1-group library*
*= F33 file*

- TRITON produces 1 F33 file for each mixture listed in the DEPLETION block and for a system-average mixture:
  - **`{BASENAME}.mix<number>.f33`**
  - **`{BASENAME}.system.f33`**
- These libraries store the data that were used by ORIGEN to perform the depletion of the individual mixtures
  - Problem-independent data: Decay constant, molar masses, etc.
  - **Problem-dependent data for each burnup step**: Transition coefficient matrix, 1-group removal/loss cross section, 1-group fission cross section, neutron yields
- The libraries can be used in standalone ORIGEN calculations
- As of SCALE 6.3.3., the burnup-dependent 1-group fission and loss cross sections can be displayed with Fulcrum
- The content of the F33 files can be viewed with the command-line utility OBIWAN (see later training)

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# Workshop

- Light-Water Reactors
- TRITON-2D

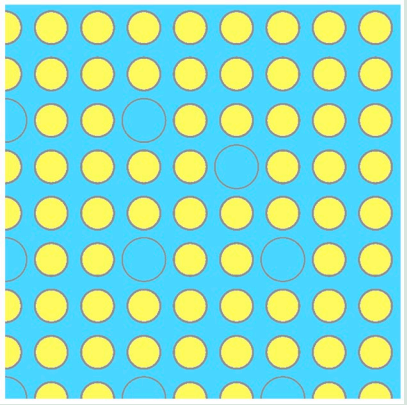


# **Problem A-1**: Use Fulcrum to inspect TRITON input and output

**Goal:** Find relevant sections in the output, understand impact of predictor–corrector procedure, and understand mass normalization

1. Inspect example TRITON input “w17_assm.inp” from the “demo” folder:
   a) Check input validation
   b) Visualize geometry
2. Inspect example TRITON output:
   a) Find for which points in time keff is provided, and for which points in time the nuclide densities are provided
3. Inspect example ORIGEN concentration file (F71 file):
   a) How do you know which displayed "State" belongs to which mixture and time step?
   b) Visualize the total heavy metal mass, the $^{235}U$ and $^{239}Pu$ mass (in kg) as a function of time and discuss the values considering TRITON’s mass normalization
4. Inspect example ORIGEN cross section file (F33 file) for mixture 1:
   a) Display 1-group fission and loss cross section as a function of burnup

# **Solution A-1**: Use Fulcrum to inspect TRITON input and output



- Depletion steps:

```
read burndata
power=40.0 burn=2.0   down=0 end
power=40.0 burn=48.0  down=0 end
power=40.0 burn=100   down=0 nlib=2  end
power=40.0 burn=1950  down=0 nlib=26 end
end burndata
```

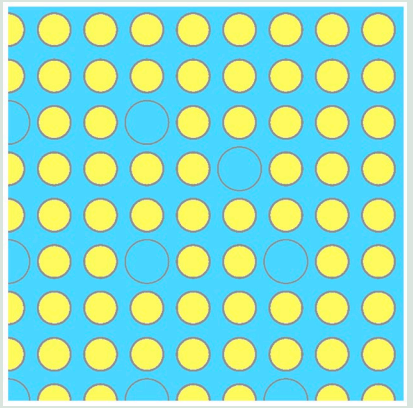
Depletion steps (days):
2, 50, 100, 150, 225, 300, 375, 450, 525, 600, 675, 750, 825, 900, 975, 1050, 1125, 1200, 1275, 1350, 1425, 1500, 1575, 1650, 1725, 1800, 1875, 1950, 2025, 2100

- Neutron transport calculations are provided in the **middle** of a depletion step (and at t=0) → $k_{eff}$ is a result of the transport calculation

**TRITON's mixture power tables, at time of transport**

```
Time =     0.00 days (   0.000 y), Burnup =    0.000     GWd/MTIHM, Transport k=  1.3616
Time =     1.00 days (   0.003 y), Burnup =    0.040     GWd/MTIHM, Transport k=  1.3267
Time =    26.00 days (   0.071 y), Burnup =    1.040     GWd/MTIHM, Transport k=  1.3039
Time =    75.00 days (   0.205 y), Burnup =    3.000     GWd/MTIHM, Transport k=  1.2848
...
Time =  1837.50 days (   5.031 y), Burnup =   73.500     GWd/MTIHM, Transport k=  0.8168
Time =  1912.50 days (   5.236 y), Burnup =   76.500     GWd/MTIHM, Transport k=  0.8053
Time =  1987.50 days (   5.441 y), Burnup =   79.500     GWd/MTIHM, Transport k=  0.7946
Time =  2062.50 days (   5.647 y), Burnup =   82.500     GWd/MTIHM, Transport k=  0.7848
```

# **Solution A-1**: Use Fulcrum to inspect TRITON input and output



- ORIGEN provides fuel compositions at the **end** of a depletion step

### ORIGEN summary at the end of a depletion step

```
end-of-step summary at time =     2.000  days (    0.005  y), system-average burnup* =    0.080  GWd/MTIHM
end-of-step summary at time =    50.000  days (    0.137  y), system-average burnup* =    1.995  GWd/MTIHM
end-of-step summary at time =   100.000  days (    0.274  y), system-average burnup* =    3.991  GWd/MTIHM
end-of-step summary at time =   150.000  days (    0.411  y), system-average burnup* =    5.986  GWd/MTIHM
...
end-of-step summary at time =  1875.000  days (    5.133  y), system-average burnup* =   74.779  GWd/MTIHM
end-of-step summary at time =  1950.000  days (    5.339  y), system-average burnup* =   77.767  GWd/MTIHM
end-of-step summary at time =  2025.000  days (    5.544  y), system-average burnup* =   80.755  GWd/MTIHM
end-of-step summary at time =  2100.000  days (    5.749  y), system-average burnup* =   83.743  GWd/MTIHM
```

### Overview of nuclide concentrations at the end of the TRITON output

```
+++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++
+                                          Depletion mixture no. 1, opus case 1                                                          +
+++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++++
            | nuclide concentrations
            | time: days
     at/bcm |    0.00e+00d       2.00d      50.00d     100.00d   ...   1875.00d    1950.00d    2025.00d    2100.00d
------------+-------------------------------------------------------------------------------------------------------------------------
       u234 |   1.0445e-05  1.0435e-05  1.0211e-05  9.9796e-06   ...   3.6509e-06  3.4798e-06  3.3175e-06  3.1640e-06
       u235 |   1.1757e-03  1.1734e-03  1.1190e-03  1.0650e-03   ...   9.7482e-05  8.4936e-05  7.3824e-05  6.4022e-05
```

# **Solution A-1**: Use Fulcrum to inspect TRITON input and output

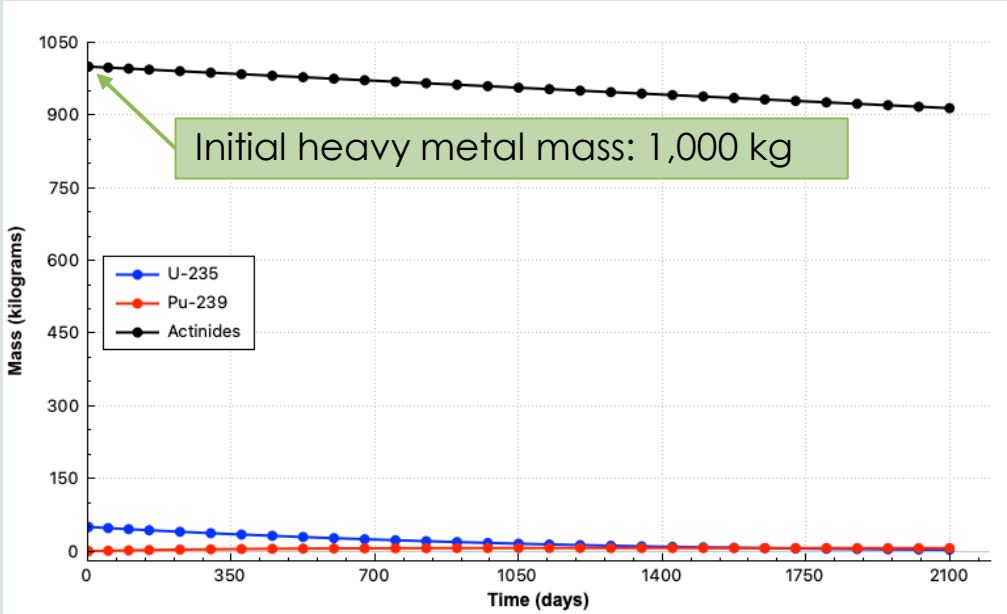
- Data on ORIGEN F71 concentration file:
  - Table at end of TRITON output describes content of F71 file
  - Mark States 1 through 62 to plot data for mixture 1
- Heavy metal content on F71 file:
  - Select to plot only Actinides (libtype), switch plot units to kilograms, and display table instead of plot
  - TRITON normalizes the initial heavy metal mass of the model to 1 metric ton
  - Initial mass of total actinides in table is 1,000 kg

```
File ft71f001 is the ORIGEN binary concentration file, containing
concentrations for
  - each of the 1 depletion mixtures,
  - a set for the sum of all depletion mixtures,
  - a set for the sum of selected mixtures (from optional opus block),
  - a set for the sum of all fuel mixtures
for always 31 time steps. Isotopic data locations are listed according
to the following table.

Use OPUS, Fulcrum, or command line utility OBIWAN to interrogate this
file.

NOTE: TRITON normalizes the masses to a system total mass of 1 metric
ton of initial HM. The data provided on the ORIGEN
      binary concentration file follows this normalization.
...
```

| | u235 | pu239 | Totals |
|---|---|---|---|
| 0.000000e+00 | 50 | 4.32529e-16 | 1000 |
| 2.000000e+00 | 49.9012 | 0.00914811 | 999.916 |
| 5.000000e+01 | 47.5877 | 0.828455 | 997.891 |
| 1.000000e+02 | 45.2907 | 1.59834 | 995.785 |



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# TRITON Input

# Part I: DEPLETION and BURNDATA block



# Main input blocks

- **Sequence identifier and parameters**
  - t-xsdrn, t-newt, t-depl-1d, t-depl, t5-depl, t6-depl, t5-depl-shift, t6-depl-shift
  - addnux option: selection of nuclides considered during neutron transport
- **Alias**
  - Definition of aliases for compositions to simplify input
- **Composition**
  - Material specifications (nuclides, densities, temperatures)
- **Celldata** (MG only)
  - Cells to be used for cross section self-shielding calculation with XSProc
- **Depletion**
  - List of mixtures to be depleted
  - Power normalization
- **Burndata**
  - Specific power [MW/MTIHM]
  - Depletion and decay intervals [days]
  - Number of depletion sub-intervals
- **Timetable** (optional)
  - Time-dependent nuclide concentration and temperature changes
- **Collapse, Homogenize, ADF, Branch** (optional)
  - Lattice physics parameter generation
- **Opus** (optional)
  - Generate text files for time-dependent fuel concentrations
- **Keep** (optional)
  - Enable additional module output
- **Transport model**:
  - Parameters, geometry, volume, etc.

# Burndata block

- Specification of the specific power, time steps, and decay time for the depletion

```
read burndata
   power=P1 burn=B1 nlib=N1 down=D1 end
   power=P2 burn=B2 nlib=N2 down=D2 end
   power=P3 burn=B3 nlib=N3 down=D3 end
   ...
end burndata
```

- **P** = average specific power in the depleted mixture(s)*, in megawatts per metric ton of initial heavy metal [MW/MTIHM]
- **B** = length of depletion interval at power P [days]
- **N** = number of depletion sub-intervals for depletion interval *(optional, default=1)*
- **D** = length of decay interval (zero power) following the depletion interval [days] *(optional, default=0.0)*

* "Depleted mixture" refers to mixtures that are listed in the DEPLETION block.

# Burndata block – Example

```
read burndata
   power=30 burn=120 nlib=5  down=30  end
   power=20 burn=50  nlib=10          end
   power=55 burn=99           down=100 end
end burndata
```

1. Deplete at **30** MW/MTIHM for **5** sub-intervals that are each **24** days long, then decay for **30** days
2. Deplete at **20** MW/MTIHM for **10** sub-intervals that are each **5** days long
3. Deplete at **55** MW/MTIHM for **99** days, then decay for **100** days

# Burndata block – Special case

```
read burndata
   power=30 burn=0 end
   power=30        end
end burndata
```

← These lines are equivalent

- As of SCALE 6.3, TRITON supports a neutron transport–only calculation: a single neutron transport calculation at t=0 without a subsequent depletion/decay calculation
- This allows a neutron transport calculation using all TRITON default settings (e.g., with trace nuclides in fuel compositions)
- This allows a calculation of mixture-specific power to construct a power profile (while we are working on enabling an actual power mesh tally)

# Depletion block

- Specification of mixtures to be considered **depletion materials**

```
read depletion
   MX1 MX2 MX3 … MXn
end depletion
```

- Mixtures **$MX_1$ $MX_2$ $MX_3$ … $MX_n$** must be specified in the composition block and must be used in the geometry block
- Can be any mixtures – they do not have to be a fissile material
- TRITON calculates a power (flux) for each mixture in the problem, regardless of whether the mixture is specified in the depletion block
- But: TRITON only updates mixture compositions that are specified in the depletion block

# Depletion block – Example

```
read depletion
   3   6   7
end depletion
```

- Mixtures 3, 6, and 7 are specified in the composition block and used in the geometry block
- Compositions of mixtures 3, 6, and 7 get updated after each depletion step, with nuclide concentrations calculated by ORIGEN
- The composition of all other mixtures is not changed

# Advanced depletion options

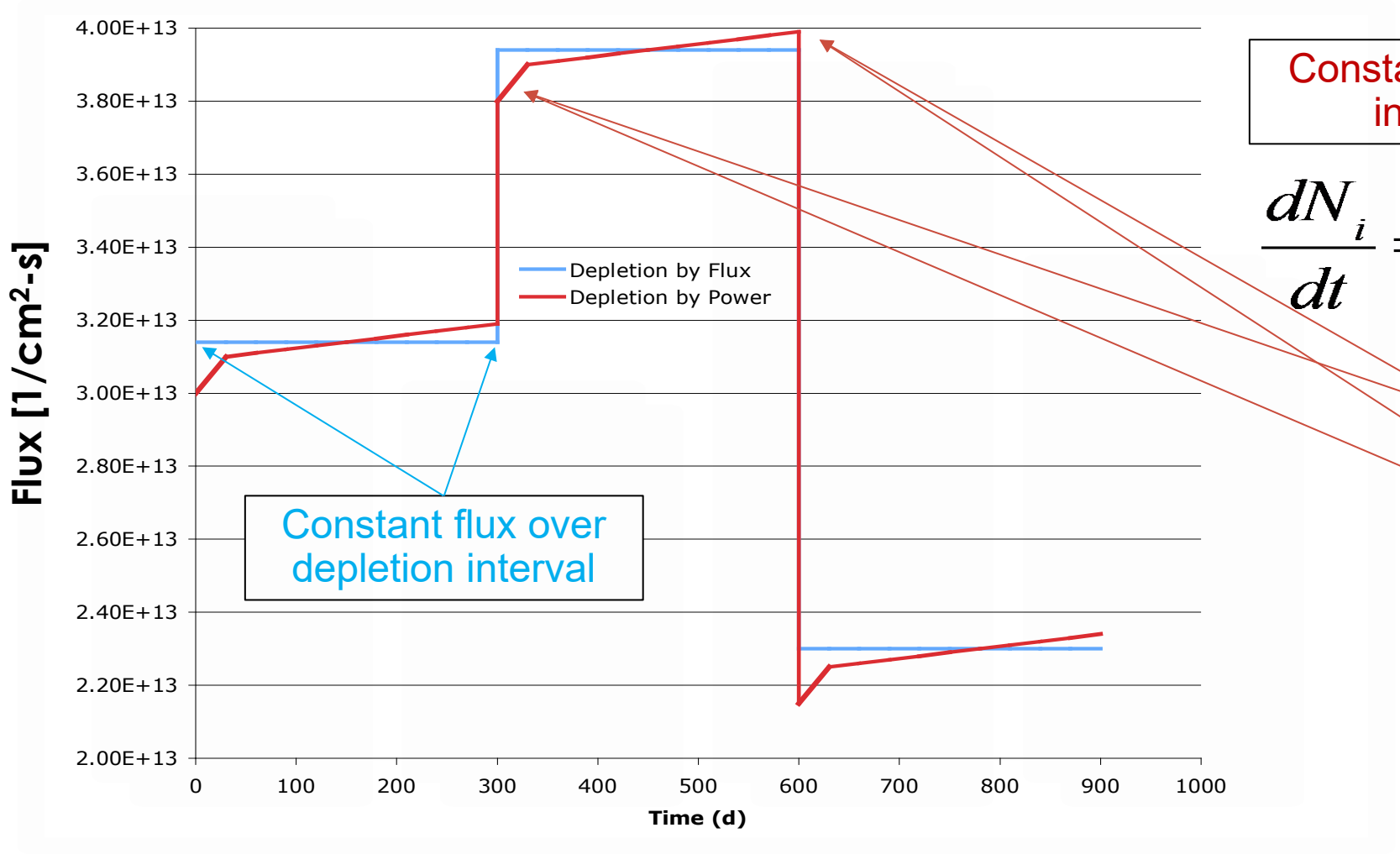
- The DEPLETION block has a few advanced options
- **Deplete by constant power or by constant flux**
  - Instructs ORIGEN on what assumption to use when depleting a material and is useful when depleting or activating non-fuel mixtures
- **Power normalization**
  - Allows the user to identify those mixtures for which the specific power is known, with additional mixture powers being adjusted accordingly
- **Mixture assignments (assign function, t-depl only)**
  - Allows reduction in cross section processing for problems that have many of the same or similar depletion materials (only with t-depl, and only recommended for experienced users)

# Deplete by constant power or by constant flux

- ORIGEN solves the Bateman equations by assuming
  - Constant flux, or
  - Constant power (default)
- For constant power, the flux is varied at each ORIGEN depletion step to keep the mixture power constant over the irradiation interval
- Practical guide: use constant power except for structural materials and strong absorbers
- LWRs:
  - Regular $UO_2$ fuel pins: use constant power
  - Gd-bearing fuel pins: use constant flux
  - WABA pins: use constant flux
  - IFBA pins: use constant flux

# Deplete by constant power or by constant flux

**Example: three TRITON depletion intervals, each using two ORIGEN time intervals**




Constant power over depletion interval → varied flux

$$\frac{dN_i}{dt} = \sum_{j=1}^{m} l_{ij}\lambda_j N_j + \bar{\Phi}\sum_{k=1}^{m} f_{ik}\sigma_k N_k - (\lambda_i + \bar{\Phi}\sigma_i + r_i)N_i$$

# Deplete by constant power or by constant flux

- Example:
  - Instructs TRITON to deplete mixtures 3, 6, and 7 by constant power, mixture 22 by constant flux

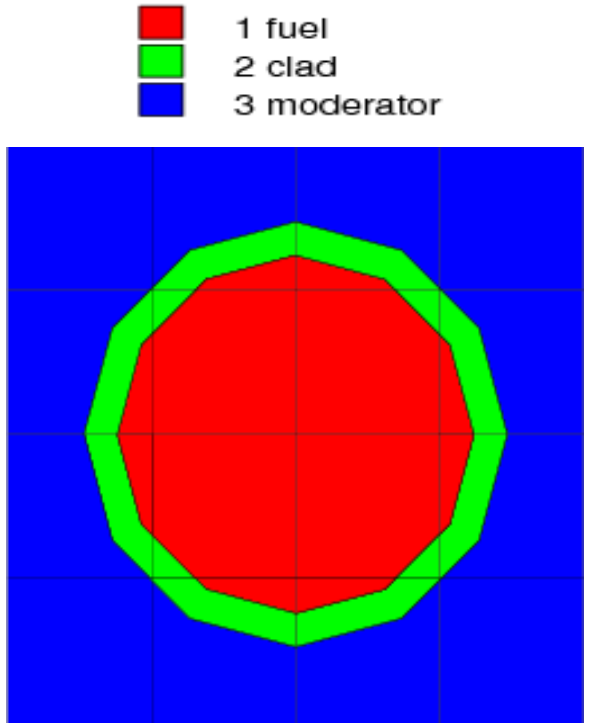
```
read depletion
  3 6 7 flux 22
end depletion
```

  - Instructs TRITON to deplete mixtures 3 and 6 by constant power, mixtures 7 and 22 by constant flux

```
read depletion
  3 6 flux 7 22
end depletion
```

# Power normalization

- TRITON accounts for all power sources in a model
  - Fluxes obtained during the transport calculation are used to calculate fission and n-gamma reaction rates, as well as to obtain power generation rates in all mixtures in the model
- For example:
  - Power in moderator (3) from n-gamma
  - Power in clad (2) from n-gamma
  - Power in fuel (1) from n-gamma and fission
- TRITON has several options for power normalization



# Power normalization – System

By default, power is normalized to the system power.
In the examples below, the system power is
Mixture 1 power + Mixture 2 power + Mixture 3 power.

- Example 1
  - Deplete only mixture 1
  - The system power is set in the Burndata block
- Example 2
  - Deplete mixtures 1 and 2
  - The system power is set in the Burndata block
- Example 3
  - Deplete all mixtures
  - The system power is set in the Burndata block

```
read comp
  uo2          1  end
  zirc2        2  end
  h2o          3  end
end comp
```

```
read depletion
  1
end depletion
```

```
read depletion
  1 2
end depletion
```

```
read depletion
  1 2 3
end depletion
```

**In each example, the power in mixtures 1, 2, and 3 remains the same because the power is the system total power in MW/MTIHM.**

# Power normalization – System

```
read burndata
  power=20 burn=100 nlib=2 end
end burndata
```

Example 1

```
read depletion
  1
end depletion
```

| Mixture Number | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) |
|---|---|---|---|
| 1 | 19.949 | 0.99743 | 19.949 |
| 2 | 0.018 | 0.00090 | N/A |
| 3 | 0.034 | 0.00168 | N/A |
| **Total** | **20.000** | 1.00000 | |

Example 2

```
read depletion
  1 2
end depletion
```

| Mixture Number | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) |
|---|---|---|---|
| 1 | 19.949 | 0.99743 | 19.949 |
| 2 | 0.018 | 0.00090 | N/A |
| 3 | 0.034 | 0.00168 | N/A |
| **Total** | **20.000** | 1.00000 | |

Example 3

```
read depletion
  1 2 3
end depletion
```

| Mixture Number | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) |
|---|---|---|---|
| 1 | 19.949 | 0.99743 | 19.949 |
| 2 | 0.018 | 0.00090 | N/A |
| 3 | 0.034 | 0.00168 | N/A |
| **Total** | **20.000** | 1.00000 | |

1. The powers are identical because **no power normalization was applied**. The provided power of 20 MW/MTIHM is applied to the whole system (i.e., all mixtures) in all cases.
2. The listing of mixtures in the depletion block only affects which mixtures are considered irradiated and therefore obtain an updated composition after each depletion step.

# Power normalization – Mixture

Normalize by a subset of depletion mixtures by selecting normalization for each desired mixture.

- Example 1
  - Deplete mixtures 1 and 2
  - Normalize to mixture 1 power
  - The mixture 1 power is set in the Burndata block
- Example 2
  - Deplete all three mixtures
  - Normalize to the sum of mixture power 1 and mixture power 2
  - The sum of mixture power 1 and mixture power 2 is set in the Burndata block
- Example 3
  - Deplete only mixture 1
  - Normalize to mixture 1 power
  - The mixture 1 power is set in the Burndata block

Minus sign denotes mixture power normalization

```
read comp
  uo2          1  end
  zirc2        2  end
  h2o          3  end
end comp
```

```
read depletion
  -1 2
end depletion
```

```
read depletion
  -1 -2 3
end depletion
```

```
read depletion
  -1
end depletion
```

# Power normalization – Mixture

```
read burndata
  power=20 burn=100 nlib=2 end
end burndata
```

Example 1

```
read depletion
  -1 2
end depletion
```

| Mixture Number | | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) |
|---|---|---|---|---|
| 1 | * | 20.000 | 0.99743 | 20.000 |
| 2 | | 0.018 | 0.00090 | N/A |
| 3 | | 0.034 | 0.00168 | N/A |
| Total | | 20.052 | 1.00000 | |

Example 2

```
read depletion
  -1 -2 3
end depletion
```

| Mixture Number | | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) |
|---|---|---|---|---|
| 1 | * | 19.982 | 0.99743 | 19.982 |
| 2 | * | 0.018 | 0.00090 | N/A |
| 3 | | 0.034 | 0.00168 | N/A |
| Total | | 20.034 | 1.00000 | |

Example 3

```
read depletion
  -1
end depletion
```

| Mixture Number | | Total Power (MW/MTIHM) | Fractional Power (---) | Mixture Power (MW/MTIHM) |
|---|---|---|---|---|
| 1 | * | 20.000 | 0.99743 | 20.000 |
| 2 | | 0.018 | 0.00090 | N/A |
| 3 | | 0.034 | 0.00168 | N/A |
| Total | | 20.052 | 1.00000 | |

# General guidance

- Depletion approximations
  - Reactor operation is more closely approximated by the depletion-by-power model
  - Results of both by-power and by-flux approximations are in good agreement unless chosen depletion steps are too large for the model
  - Large depletion steps introduce errors in other approximations (use of constant cross sections over a long time interval)
  - For some systems, depletion-by-power will fail (e.g., depletion of a strong absorber or low power region)
- Depletion step size:
  - Multi-group and therefore 1-group cross sections change with time as nuclide inventories vary
  - Changes are generally second order, so cross section updates are not needed on a fine temporal basis
  - Cross section updates are required more often when properties change rapidly (e.g., Gd depletion)

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# Conversion of neutron transport input to depletion input



# Conversion from **T-NEWT** input to **TRITON** input

## Option 1: In-place modification

- Change sequence to "t-depl"
- Add a DEPLETION and a BURNDATA block

```
=t-depl parm=( )
title-goes-here
xslib-goes-here

read comp
' TODO: define comp
end comp

read celldata
' TODO: define celldata
end celldata

read depletion
' TODO: define depletion
end depletion

read burndata
' TODO: define burndata
end burndata

read model

title-goes-here
read parameters
' TODO: define parameters
end parameters

read materials
mix=1 end
' TODO: define materials
end materials

read geometry
' TODO: define geometry
end geometry

end model
end
```

## Option 2: Block-wise copying

- Create a new input with Fulcrum
- Use auto-completion to create the skeleton of a "t-depl" input
- Copy/paste title, cross section library, and composition, self-shielding, and neutron transport blocks
- Add information to DEPLETION and BURNDATA blocks

```
=t-depl parm=( )
title-goes-here
xslib-goes-here

read comp
' TODO: define comp
end comp

read celldata
' TODO: define celldata
end celldata

read depletion
' TODO: define depletion
end depletion

read burndata
' TODO: define burndata
end burndata

read model

title-goes-here
read parameters
' TODO: define parameters
end parameters

read materials
mix=1 end
' TODO: define materials
end materials

read geometry
' TODO: define geometry
end geometry

end model
end
```

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# Workshop

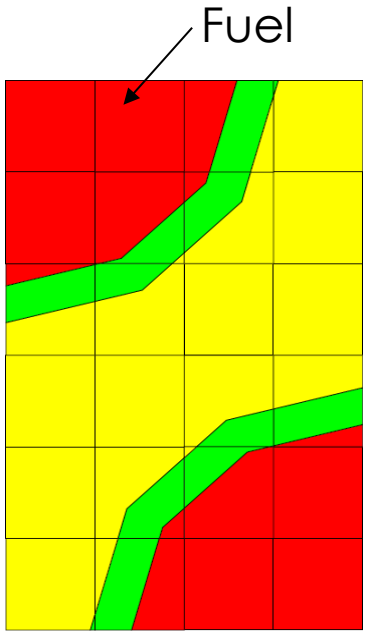
- Light-Water Reactors
- TRITON-2D



# **Problem A-2**: Depletion of a VVER pin cell

1. Copy "vver_pin_tnewt.inp" to a new file "vver_pin_tdepl.inp"
2. Update the model:
   - Change the sequence to a "t-depl" sequence and add parm=(bonami)
   - Deplete mixture 1 with the following depletion history:

| Power (MW/MTU) | Depletion interval (days) | Decay interval (days) | No. of depletion sub-intervals |
|---|---|---|---|
| 20 | 100 | 100 | 2 |
| 30 | 300 | 30 | 1 |
| 20 | 300 | 0 | 1 |

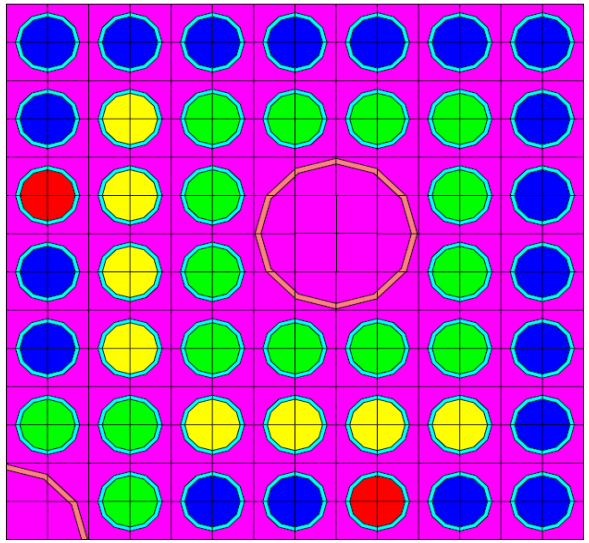



**VVER pin cell with reflective boundary conditions**

3. Run the model and investigate the output:
   - Check $k_{eff}$, mixture-specific power, and burnup
   - How does the total power compare to the power for the fuel mixture?

# **Problem A-3:** Depletion of a PWR assembly

1. Copy "ce14_assm_tnewt.inp" to a new file "ce14_assm_tdepl.inp"
2. Update the model:
   - Change the sequence to a "t-depl" sequence and add parm=(bonami)
   - Suppose post-irradiation exam reveals that combined power for the red fuel pins is 20 MW/MTU
   - Deplete assembly such that mixture 1 power is 20 MW/MTU for 330 days, followed by a 30-day decay interval; use 6 depletion sub-intervals
3. Run the model and investigate the output:
   - Check $k_{eff}$, mixture-specific power, and burnup
   - How does the total power compare to the power for the fuel mixture?

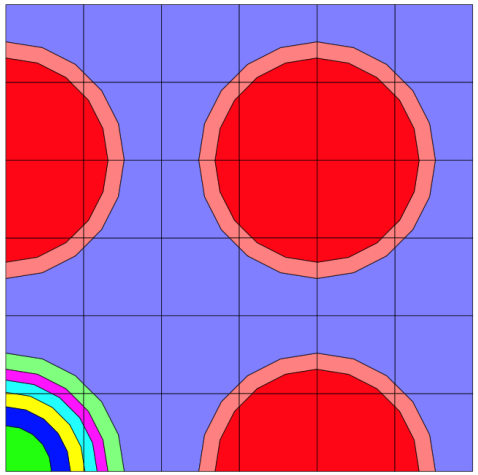


**CE 14x14 fuel assembly**

Mixture 1: red fuel color
Mixture 4: green fuel color
Mixture 7: blue fuel color
Mixture 10: yellow fuel color

# **Problem A-4:** Gadolinium depletion

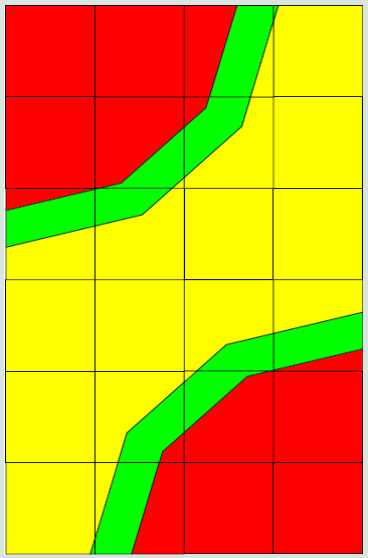
1. Copy "bwr_mini_assm_tnewt.inp" to a new file "bwr_mini_assm_tdepl.inp"
2. Update the model:
   - Change the sequence to a "t-depl" sequence and add parm=(centrm)
   - Deplete assembly at 25 MW/MTU for 200 days using 4 depletion sub-intervals
   - Mixture 1 is $UO_2$ fuel; deplete by constant power
   - Mixtures 21–25 are 5 fuels rings of $UO_2$-$Gd_2O_3$ fuel; deplete by constant flux
3. Run the model and investigate the output:
   - Check $k_{eff}$, mixture-specific power, and burnup
   - How does the total power compare to the power for the fuel mixture?



**BWR mini assembly**

Note: ~15 minute runtime → Time to take a break

# **Solution A-2:** Depletion of a VVER pin cell



**VVER pin cell**

- $k_{eff}$
  - Grep $k_{eff}$ result directly from neutron transport output

```
k-eff =       1.24513893     Time=      0.00d Nominal conditions
k-eff =       1.20615301     Time=     25.00d Nominal conditions
k-eff =       1.19456773     Time=     75.00d Nominal conditions
k-eff =       1.13408854     Time=    350.00d Nominal conditions
k-eff =       1.07102125     Time=    680.00d Nominal conditions
```

  - Or grep $k_{eff}$ from TRITON mixture power overview table

```
Time =     0.00 days (   0.000 y), Burnup =     0.000     GWd/MTIHM, Transport k=  1.2451
Time =    25.00 days (   0.068 y), Burnup =     0.500     GWd/MTIHM, Transport k=  1.2062
Time =    75.00 days (   0.205 y), Burnup =     1.500     GWd/MTIHM, Transport k=  1.1946
Time =   350.00 days (   0.958 y), Burnup =     6.500     GWd/MTIHM, Transport k=  1.1341
Time =   680.00 days (   1.862 y), Burnup =    14.000     GWd/MTIHM, Transport k=  1.0710
```

# **Solution A-2:** Depletion of a VVER pin cell

Without normalization, the input specific power is the total specific power:

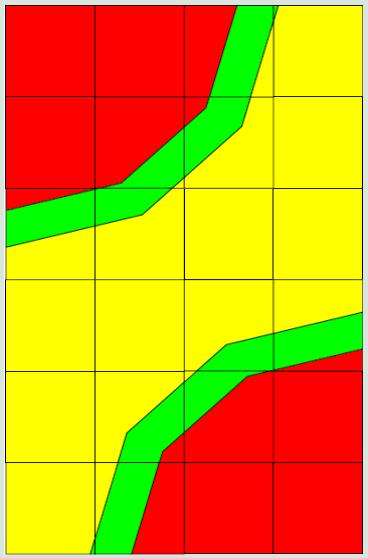
```
--- Mixture powers for depletion pass no.    0 (MW/MITHM) ---
       Time =      0.00 days (    0.000 y), Burnup =     0.000     GWd/MTIHM, Transport k=  1.2451

                  Total          Fractional  Mixture       Mixture       Mixture           Mixture
       Mixture    Power (a)      Power       Power (b)     Burnup (c)    Thermal Flux (d)  Total Flux (e)
       Number     (MW/MTIHM)     (---)       (MW/MTIHM)    (GWd/MTIHM)   n/(cm^2*sec)      n/(cm^2*sec)
            1         19.945       0.99723        19.945         0.000     1.6897e+13        1.8044e+14
           20          0.022       0.00110           N/A           N/A     1.8092e+13        1.8109e+14
           30          0.034       0.00168           N/A           N/A     1.8940e+13        1.8149e+14
           10          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
           11          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
           21          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
           31          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
        Total         20.000       1.00000
```

```
 --- Mixture powers for depletion pass no.    6 (MW/MITHM) ---
       Time =    680.00 days (    1.862 y), Burnup =    14.000     GWd/MTIHM, Transport k=  1.0710

                  Total          Fractional  Mixture       Mixture       Mixture           Mixture
       Mixture    Power (a)      Power       Power (b)     Burnup (c)    Thermal Flux (d)  Total Flux (e)
       Number     (MW/MTIHM)     (---)       (MW/MTIHM)    (GWd/MTIHM)   n/(cm^2*sec)      n/(cm^2*sec)
            1         19.943       0.99716        19.943        13.961     1.4900e+13        2.1127e+14
           20          0.025       0.00125           N/A           N/A     1.6287e+13        2.1213e+14
           30          0.032       0.00160           N/A           N/A     1.7236e+13        2.1261e+14
           10          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
           11          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
           21          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
           31          0.000       0.00000           N/A           N/A     0.0000e+00        0.0000e+00
        Total         20.000       1.00000

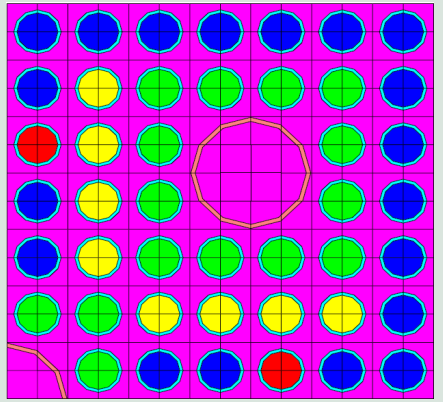
   NOTE: (a) Total Power is the Mixture Power per 1 metric ton of HM of the initial system mass.
         (b) Mixture Power is the Mixture Power per 1 metric ton of HM of the initial mixture mass.
         (c) Mixture Burnup is the Mixture Burnup per 1 metric ton of HM of the initial mixture mass.
         (d) Mixture Thermal Flux determined using 0.625 eV cutoff: Groups 41 through 56.
         (e) Normalized flux from the neutron transport calculation was scaled by a factor of 3.1991e+12 .
```



**VVER pin cell**

This burnup is the library burnup, calculated by multiplying time with power.

# **Solution A-3:** Depletion of a PWR assembly



**CE 14x14 fuel assembly**

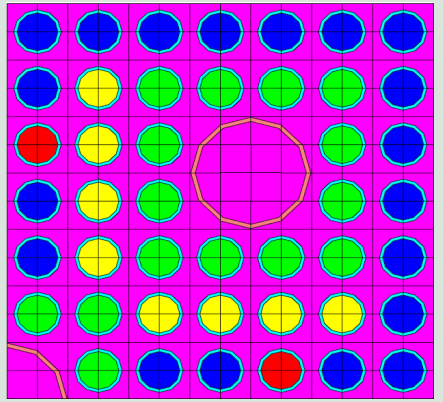
- $k_{eff}$
  - Grep $k_{eff}$ result directly from neutron transport output

```
 k-eff =       1.28880003     Time=      0.00d Nominal conditions
 k-eff =       1.23919473     Time=     27.50d Nominal conditions
 k-eff =       1.22576190     Time=     82.50d Nominal conditions
 k-eff =       1.21161634     Time=    137.50d Nominal conditions
 k-eff =       1.19685937     Time=    192.50d Nominal conditions
 k-eff =       1.18212091     Time=    247.50d Nominal conditions
 k-eff =       1.16770424     Time=    302.50d Nominal conditions
```

  - Or grep $k_{eff}$ from TRITON mixture power overview table

```
Time =     0.00 days (   0.000 y), Burnup =     0.000     GWd/MTIHM, Transport k=  1.2888
Time =    27.50 days (   0.075 y), Burnup =     0.550     GWd/MTIHM, Transport k=  1.2392
Time =    82.50 days (   0.226 y), Burnup =     1.650     GWd/MTIHM, Transport k=  1.2258
Time =   137.50 days (   0.376 y), Burnup =     2.750     GWd/MTIHM, Transport k=  1.2116
Time =   192.50 days (   0.527 y), Burnup =     3.850     GWd/MTIHM, Transport k=  1.1969
Time =   247.50 days (   0.678 y), Burnup =     4.950     GWd/MTIHM, Transport k=  1.1821
Time =   302.50 days (   0.828 y), Burnup =     6.050     GWd/MTIHM, Transport k=  1.1677
```

# **Solution A-3:** Depletion of a PWR assembly



**CE 14x14 fuel assembly**

--- Mixture powers for depletion pass no. 0 (MW/MITHM) ---
Time = 0.00 days ( 0.000 y), Burnup = 0.000 GWd/MTIHM, Transport k= 1.2888

| Mixture Number | Total Power (a) (MW/MTIHM) | Fractional Power (---) | Mixture Power (b) (MW/MTIHM) | Mixture Burnup (c) (GWd/MTIHM) | Mixture Thermal Flux (d) n/(cm^2*sec) | Mixture Total Flux (e) n/(cm^2*sec) |
|---|---|---|---|---|---|---|
| 1 | * 0.909 | 0.04220 | 20.000 | 0.000 | 2.3725e+13 | 1.6118e+14 |
| 2 | 0.018 | 0.00085 | N/A | N/A | 2.7535e+13 | 1.6276e+14 |
| 3 | 0.041 | 0.00191 | N/A | N/A | 3.1340e+13 | 1.6483e+14 |
| 4 | 7.929 | 0.36808 | 23.257 | 0.000 | 2.7949e+13 | 1.6404e+14 |
| 5 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 6 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 7 | 8.843 | 0.41052 | 20.478 | 0.000 | 2.4362e+13 | 1.6161e+14 |
| 8 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 9 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 10 | 3.799 | 0.17635 | 20.893 | 0.000 | 2.4362e+13 | 1.6161e+14 |
| 11 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 12 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 13 | 0.002 | 0.00009 | N/A | N/A | 4.1599e+13 | 1.8605e+14 |
| Total | 21.540 | 1.00000 | | | | |

* - Power normalized to this mixture.

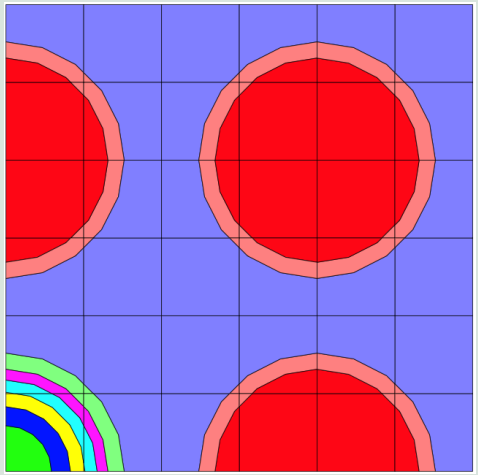
--- Mixture powers for depletion pass no. 6 (MW/MITHM) ---
Time = 302.50 days ( 0.828 y), Burnup = 6.050 GWd/MTIHM, Transport k= 1.1677

| Mixture Number | Total Power (a) (MW/MTIHM) | Fractional Power (---) | Mixture Power (b) (MW/MTIHM) | Mixture Burnup (c) (GWd/MTIHM) | Mixture Thermal Flux (d) n/(cm^2*sec) | Mixture Total Flux (e) n/(cm^2*sec) |
|---|---|---|---|---|---|---|
| 1 | * 0.909 | 0.04239 | 20.000 | 6.050 | 2.2386e+13 | 1.7687e+14 |
| 2 | 0.020 | 0.00092 | N/A | N/A | 2.6661e+13 | 1.7866e+14 |
| 3 | 0.041 | 0.00191 | N/A | N/A | 3.0892e+13 | 1.8100e+14 |
| 4 | 7.852 | 0.36616 | 23.032 | 7.017 | 2.7163e+13 | 1.8002e+14 |
| 5 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 6 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 7 | 8.831 | 0.41180 | 20.450 | 6.191 | 2.3102e+13 | 1.7733e+14 |
| 8 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 9 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 10 | 3.789 | 0.17671 | 20.842 | 6.315 | 2.3733e+13 | 1.7760e+14 |
| 11 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 12 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| 13 | 0.002 | 0.00010 | N/A | N/A | 4.1599e+13 | 1.8605e+14 |
| Total | 21.444 | 1.00000 | | | | |

* - Power normalized to this mixture.

Because the power was normalized to mixture 1, the power of this mixture is equal to the input power in the BURNDATA block. All other powers are scaled accordingly.

# **Solution A-4:** Gadolinium depletion



**BWR mini assembly**

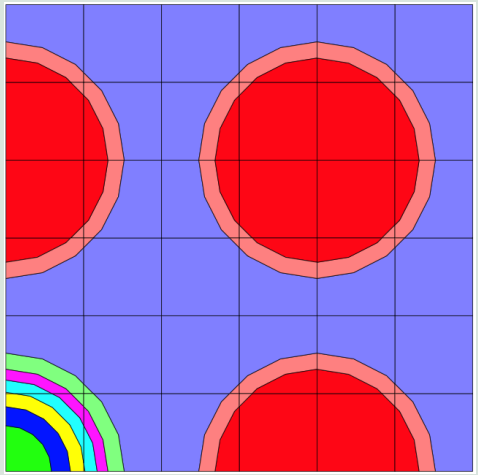
- $k_{eff}$
  - Grep $k_{eff}$ result directly from neutron transport output

```
k-eff =       1.01625682     Time=      0.00d Nominal conditions
k-eff =       1.00202903     Time=     25.00d Nominal conditions
k-eff =       1.03132563     Time=     75.00d Nominal conditions
k-eff =       1.06674489     Time=    125.00d Nominal conditions
k-eff =       1.10660719     Time=    175.00d Nominal conditions
```

  - Or grep $k_{eff}$ from TRITON mixture power overview table

```
Time =     0.00 days (   0.000 y), Burnup =     0.000     GWd/MTIHM, Transport k=  1.0163
Time =    25.00 days (   0.068 y), Burnup =     0.625     GWd/MTIHM, Transport k=  1.0020
Time =    75.00 days (   0.205 y), Burnup =     1.875     GWd/MTIHM, Transport k=  1.0313
Time =   125.00 days (   0.342 y), Burnup =     3.125     GWd/MTIHM, Transport k=  1.0667
Time =   175.00 days (   0.479 y), Burnup =     4.375     GWd/MTIHM, Transport k=  1.1066
```

# **Solution A-4:** Gadolinium depletion



**BWR mini assembly**

```
--- Mixture powers for depletion pass no.   0 (MW/MITHM) ---
 Time =     0.00 days (   0.000 y), Burnup =    0.000     GWd/MTIHM, Transport k=  1.0163
```

| Mixture Number | Total Power (a) (MW/MTIHM) | Fractional Power (---) | Mixture Power (b) (MW/MTIHM) | Mixture Burnup (c) (GWd/MTIHM) | Mixture Thermal Flux (d) n/(cm^2*sec) | Mixture Total Flux (e) n/(cm^2*sec) |
|---|---|---|---|---|---|---|
| 1 | 23.442 | 0.93769 | 26.285 | 0.000 | 2.7374e+13 | 2.2966e+14 |
| 3 | 0.021 | 0.00084 | N/A | N/A | 2.9282e+13 | 2.2878e+14 |
| 5 | 0.023 | 0.00093 | N/A | N/A | 3.0404e+13 | 2.2874e+14 |
| 21 | 0.213 | 0.00852 | 9.847 | 0.000 | 7.6117e+12 | 2.0398e+14 |
| 22 | 0.227 | 0.00910 | 10.523 | 0.000 | 8.2368e+12 | 2.0435e+14 |
| 23 | 0.252 | 0.01007 | 11.637 | 0.000 | 9.0675e+12 | 2.0490e+14 |
| 24 | 0.300 | 0.01202 | 13.898 | 0.000 | 1.0294e+13 | 2.0610e+14 |
| 25 | 0.518 | 0.02074 | 23.961 | 0.000 | 1.3391e+13 | 2.0921e+14 |
| 4 | 0.002 | 0.00009 | N/A | N/A | 2.0360e+13 | 2.1619e+14 |
| 6 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| Total | 25.000 | 1.00000 | | | | |

For $UO_2$-$Gd_2O_3$ fuel, as Gd absorbers burn out, mixture-specific power increases from BOC to EOC.

(Note Gd-155 and Gd-157 nuclide densities over time.)

```
--- Mixture powers for depletion pass no.   4 (MW/MITHM) ---
 Time =   175.00 days (   0.479 y), Burnup =    4.375     GWd/MTIHM, Transport k=  1.1066
```

| Mixture Number | Total Power (a) (MW/MTIHM) | Fractional Power (---) | Mixture Power (b) (MW/MTIHM) | Mixture Burnup (c) (GWd/MTIHM) | Mixture Thermal Flux (d) n/(cm^2*sec) | Mixture Total Flux (e) n/(cm^2*sec) |
|---|---|---|---|---|---|---|
| 1 | 22.776 | 0.91105 | 25.538 | 4.536 | 2.5483e+13 | 2.1646e+14 |
| 3 | 0.020 | 0.00079 | N/A | N/A | 2.7526e+13 | 2.1605e+14 |
| 5 | 0.022 | 0.00090 | N/A | N/A | 2.9116e+13 | 2.1678e+14 |
| 21 | 0.363 | 0.01453 | 16.785 | 2.220 | 1.4426e+13 | 2.0320e+14 |
| 22 | 0.407 | 0.01629 | 18.845 | 2.523 | 1.7237e+13 | 2.0575e+14 |
| 23 | 0.433 | 0.01730 | 19.994 | 2.887 | 1.9637e+13 | 2.0783e+14 |
| 24 | 0.454 | 0.01815 | 20.996 | 3.292 | 2.1337e+13 | 2.0936e+14 |
| 25 | 0.522 | 0.02089 | 24.136 | 3.929 | 2.2675e+13 | 2.1039e+14 |
| 4 | 0.002 | 0.00009 | N/A | N/A | 2.4214e+13 | 2.1129e+14 |
| 6 | 0.000 | 0.00000 | N/A | N/A | 0.0000e+00 | 0.0000e+00 |
| Total | 25.000 | 1.00000 | | | | |

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# TRITON Input

## Part II: Sequence-level parameters, ALIAS block, KEEP block



# Main input blocks

- **Sequence identifier and parameters**
  - t-xsdrn, t-newt, t-depl-1d, t-depl, t5-depl, t6-depl, t5-depl-shift, t6-depl-shift
  - addnux option: selection of nuclides considered during neutron transport
- **Alias**
  - Definition of aliases for compositions to simplify input
- **Composition**
  - Material specifications (nuclides, densities, temperatures)
- **Celldata** (MG only)
  - Cells to be used for cross section self-shielding calculation with XSProc
- **Depletion**
  - List of mixtures to be depleted
  - Power normalization
- **Burndata**
  - Specific power [MW/MTIHM]
  - Depletion and decay intervals [days]
  - Number of depletion intervals
- **Timetable** (optional)
  - Time-dependent nuclide concentration and temperature changes
- **Collapse, Homogenize, ADF, Branch** (optional)
  - Lattice physics parameter generation
- **Opus** (optional)
  - Generate text files for time-dependent fuel concentrations
- **Keep** (optional)
  - Enable additional module output
- **Transport model**:
  - Parameters, geometry, volume, etc.

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# Sequence-level parameters for TRITON



# Sequence-level TRITON parameters: CHECK

- *parm=check*
  - Used to check your input file
  - Faster than trying to run the input file in TRITON and then receiving an error
  - Can be used with TRITON-NEWT to generate plots of the geometry (*.ps files)
    - Best way to perform a visual check of the geometry and grid spacing before running the full calculation

# Sequence-level TRITON parameters: WEIGHT

- parm=(weight=N) or parm=weight
  - Used to create a new, weighted broad-group (56g) library from a fine-group (252g) master library
  - First neutron transport calculation is performed using the 252g library, and the solution spectrum is used to collapse the initial library to a 56g master library
    - The 56g master library is named *newxnlib* and is available in the temporary directory
    - The new master can be copied from the temporary directory and renamed to be used in other transport calculations
  - Depletion calculation then restarts with a reduced group master library (this calculation will run faster)
  - If N is specified, weighting is performed using the calculated flux in mixture N
  - If N is not specified, weighting is performed using the system-averaged flux
  - Note:
    - If using an additional COLLAPSE block, collapsing must be specified based on 56 groups (see later slides on COLLAPSE block)
    - **Only used for LWR analysis; not recommended for non-LWRs**
    - **Only for experienced users**

# Sequence-level TRITON parameters: ADDNUX

- parm=(addnux=N) – selects set of nuclides in depleted mixtures
  - A limited number of nuclides are considered by default in the neutron transport calculation to reduce the transport runtime
  - ORIGEN tracks concentrations of 2,237 nuclides in 3 sublibraries (~1,682 unique nuclides)
  - Only the concentrations of nuclides that are initially part of a depleted mixture are updated in the neutron transport model after each depletion step
  - To consider the most important nuclides during the transport calculation, a set of 94 nuclides are by default added (in trace quantities) to each fuel mixture so that these mixtures have a large nuclide vector that gets updated
    - These selected nuclides are known to have a significant impact on flux as a function of burnup
  - User control of added nuclides: ADDNUX
  - Also note that one-group cross sections prepared for ORIGEN are updated after each transport calculation:
    - For nuclides that are part of the transport model: based on results of the neutron transport calculation, considering cross section self-shielding
    - For all other nuclides: based on the JEFF data resource

# Sequence-level TRITON parameters: ADDNUX

- parm=(addnux=0)
  - Adds no nuclides – only those specified in the input by the user are updated
  - ADDNUX=0 only used for single state point calculation (burn=0) or for testing purposes
- parm=(addnux=1)
  - Adds 15 actinides to user-supplied inventories
  - Usually, a poor approximation after one cycle, when fission products become significant
  - ADDNUX=1 is only used for testing purposes
- parm=(addnux=2) (Default)
  - Adds an additional 79 fission products and 15 actinides to user-supplied inventories
  - ADDNUX=2 is generally adequate for depletion calculations

# Sequence-level TRITON parameters: ADDNUX

- parm=(addnux=3)
  - Adds a total of 230 nuclides to user-specified inventory
  - These nuclides may be needed for accurate $k_{eff}$ calculations at high burnup
  - These nuclides have little effect on the neutron flux spectrum
  - Consider that additional nuclides require additional cross section processing time
  - ADDNUX=3 is primarily used for spent fuel characterization or high-burnup analysis
- parm=(addnux=4)
  - Adds a total of 388 nuclides to user-specified inventory
  - ADDNUX=4 primarily used for spent fuel characterization, high-burnup analysis, and advanced reactor analysis

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# KEEP block

Control printing of individual module output



# KEEP block

- TRITON calculations can result in thousands of pages of output (and very large file sizes)
- Much of the output is not needed unless debugging or checking certain parameters
  - Important:
    - NEWT (k-eff, check few-group cross sections)
    - KENO (k-eff, check Monte Carlo convergence)
  - Not important:
    - XSPROC, ORIGEN
- TRITON provides the ability to selectively delete certain types of outputs

# KEEP block

```
read keep
   newt xsproc
end keep
```

- Valid names are:
  - xsproc, origen
  - xsdrn, newt, kenova, kenovi, shift*
- By default, only the output of the neutron transport code (e.g., NEWT) is printed to the TRITON output
- When adding a KEEP block, only the output for the listed modules is printed
  - An empty KEEP block disables all module output and limits the output to only TRITON itself (e.g., the power/burnup tables)
  - Adding a single module name enables the output of only that single module

* The shift keyword in the KEEP block requires a time array: **`shift 0 1 LAST end`**. It also specifies the time step at which the large Shift HDF5 result file is saved. Time arrays will be permitted for all keywords starting with SCALE 7.

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# ALIAS block

Input option to shorten and simplify TRITON input



# ALIAS block

- The ALIAS block was developed to simplify the setup of composition and cell data when numerous depletion materials are used
  - Example: PWR assembly with 24 unique fuel locations (reduced by 1/8 symmetry) — all 24 positions have identical initial loadings but burn at different rates
  - To be tracked individually, identical fresh fuel mixtures must be specified

# Example of a long COMPOSITION and CELLDATA block

```
read comp
'Fuel
uo2 1 den=10.045 1 790 92234 0.027 92235 3.038 92236 0.014 92238 96.921 end
uo2 2 den=10.045 1 790 92234 0.027 92235 3.038 92236 0.014 92238 96.921 end
uo2 3 den=10.045 1 790 92234 0.027 92235 3.038 92236 0.014 92238 96.921 end
uo2 4 den=10.045 1 790 92234 0.027 92235 3.038 92236 0.014 92238 96.921 end
wtptzirc  5 6.44 4 40000 97.91 26000 0.5 50116 0.86 50120 0.73 1 620 end
wtptzirc  7 6.44 4 40000 97.91 26000 0.5 50116 0.86 50120 0.73 1 620 end
wtptzirc  8 6.44 4 40000 97.91 26000 0.5 50116 0.86 50120 0.73 1 620 end
wtptzirc  9 6.44 4 40000 97.91 26000 0.5 50116 0.86 50120 0.73 1 620 end
h2o 6 den=0.7575 1 557 end
h2o 10 den=0.7575 1 557 end
h2o 11 den=0.7575 1 557 end
h2o 12 den=0.7575 1 557 end
wtpt-bormod 6  0.7332 1 5000 100 330.8e-6 557 end
wtpt-bormod 10 0.7332 1 5000 100 330.8e-6 557 end
wtpt-bormod 11 0.7332 1 5000 100 330.8e-6 557 end
wtpt-bormod 12 0.7332 1 5000 100 330.8e-6 557 end
end comp

read celldata
latticecell squarepitch pitch=1.4732 6 fuel=0.47815 1 clad=0.5588 5 end
latticecell squarepitch pitch=1.4732 10 fuel=0.47815 2 clad=0.5588 7 end
latticecell squarepitch pitch=1.4732 11 fuel=0.47815 3 clad=0.5588 8 end
latticecell squarepitch pitch=1.4732 12 fuel=0.47815 4 clad=0.5588 9 end
end celldata
```

# Example of a simplified input using ALIAS

```
read alias
$fuel 1-4 end
$clad 5, 7-9 end
$mod 6 10 11 12 end
end alias

read comp
'Fuel
uo2          $fuel den=10.045 1 790 92234 0.027 92235 3.038 92236 0.014 92238 96.921 end
wtptzirc     $clad 6.44 4 40000 97.91 26000 0.5 50116 0.86 50120 0.73 1 620 end
h2o          $mod den=0.7575 1 557 end
wtpt-bormod $mod 0.7332 1 5000 100 330.8e-6 557 end
end comp

read celldata
latticecell squarepitch pitch=1.4732 $mod fuel=0.47815 $fuel clad=0.5588 $clad end
end celldata
```

# ALIAS block

- In the alias block, aliases are defined to represent a set of mixture IDs
- Format:

```
read alias
   $alias1name [list of mixtures] end
   $alias2name [list of mixtures] end
   $alias3name [list of mixtures] end
   ...
end alias
```

- *aliasname* is any string of non-blank alphanumeric values up to 12 characters, preceded by a dollar sign ($)

# ALIAS block

- Mixture lists can be specified in multiple forms
  - Separated by spaces and/or commas
  - A series can be specified as **first-last**
  - Any combination of the above
- Example:

```
read alias
   $fuel 1-3 10 end
   $clad 4,5,6,11 end
   $mod 7 8-9 12 end
end alias
```

- Mixture numbers can be repeated if needed
  - Note that this is likely not desired. Since it is valid, it will not be caught by the input validator or parser!
- Aliases are then used in place of mixture numbers
- A pre-processor parses input, expands all aliases, and rewrites a modified input

# ALIAS block

- Aliases can be used in the following data blocks:
  - comp
  - celldata
  - depletion
  - timetable
  - branch
  - material (NEWT only)
- The form of the alias expansion is block-specific

# Aliases within the COMPOSITION block

```
read alias
   $fuel 1 2 10 end
end alias
read composition
uo2 $fuel den=10.045  1  841 92235 2.5 92238 97.5   end
end composition
```

is equivalent to

```
read composition
uo2   1 den=10.045  1  841 92235 2.5 92238 97.5   end
uo2   2 den=10.045  1  841 92235 2.5 92238 97.5   end
uo2  10 den=10.045  1  841 92235 2.5 92238 97.5   end
end composition
```

# Aliases within the CELLDATA block

```
read alias
    $fuel 1-3 10 end
    $clad 4,5,6,11 end
    $mod 7 8-9 12 end
end alias
read celldata
latticecell squarep pitch=1.26 $mod fuelr=0.41 $fuel cladr=0.50 $clad end
end celldata
```

is equivalent to

```
read celldata
latticecell squarep pitch=1.26 7  fuelr=0.41  1 cladr=0.50  4 end
latticecell squarep pitch=1.26 8  fuelr=0.41  2 cladr=0.50  5 end
latticecell squarep pitch=1.26 9  fuelr=0.41  3 cladr=0.50  6 end
latticecell squarep pitch=1.26 12 fuelr=0.41 10 cladr=0.50 11 end
end celldata
```

**Note: Aliases are not supported in multiregion and doublehet cells!**

# Aliases within the DEPLETION block

- In the depletion block, the alias is simply replaced by the list of mixtures

The alias definition

```
read alias
  $fuel 1 - 6 end
end alias
```

and a depletion block set up as:

```
read depletion 7 8 9 $fuel end depletion
```

becomes:

```
read depletion 7 8 9 1 2 3 4 5 6 end depletion
```

# Aliases within the TIMETABLE block

- In the timetable block, a duplicate timetable is created for each mixture number associated with the alias

  For the alias definition

```
read timetable
density $allmod 2 5010 5011
  0.0    1.0
  1203   1.0
  1274   1.419
  1655.7 1.419
  1737   1.523
  2203   1.523 end
end timetable
```

  One new timetable will be created for each mixture ID associated with alias `$allmod`

# ALIAS block – Notes

- Alias expansion:
  - Fulcrum can expand aliases: Click the arrow next to "Run" and select "Expand aliases"
  - You can check the output file in which the expanded input is also printed
  - CSAS does not support aliases: when converting TRITON inputs to CSAS inputs, you need to expand aliases and then copy/paste the input blocks that used aliases
- Do not terminate an alias definition with a single "end" keyword on an otherwise empty line

correct:

```
read alias
$fuel 1-10 end
end alias
```

wrong:

```
read alias
$fuel 1-10
    end
end alias
```

- CELLDATA block:
  - Multiregion and Double-het are not supported
  - CENTRM DATA and MORE DATA statements are not expanded

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# Workshop

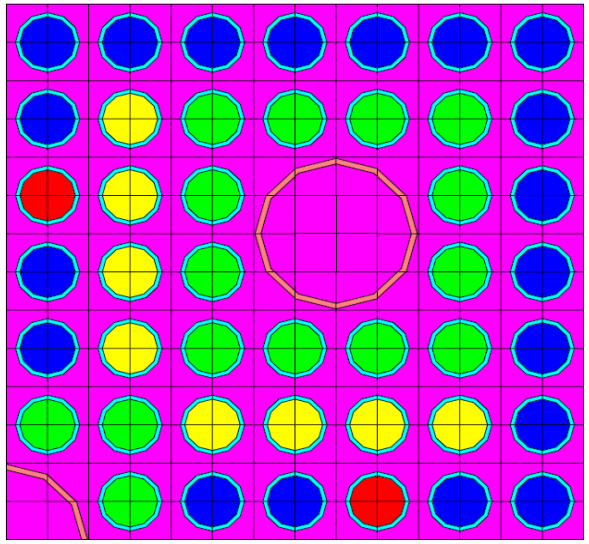
- Light-Water Reactors
- TRITON-2D



# **Problem A-5:** Use of ALIAS in PWR assembly depletion

1. Copy your input "ce14_assm_tdepl.inp" from the earlier exercise to a new file "ce14_assm_tdepl_alias.inp"
2. Simplify the COMPOSITION block using an ALIAS block
   - Make sure to check all parameters: densities, compositions, temperatures
3. Run the model and confirm that it produces identical results to those of Problem 3
4. Bonus: Add a KEEP block and perform runs with different KEEP block contents:
   - empty
   - xsproc
   - xsproc, newt, origen

   For all runs, check the presence of the NEWT, XSPROC and ORIGEN output

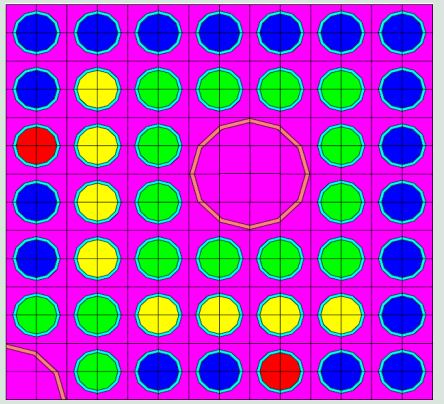


**CE 14x14 fuel assembly**

Mixture 1: red fuel color
Mixture 4: green fuel color
Mixture 7: blue fuel color
Mixture 10: yellow fuel color

# **Solution A-5:** Use of ALIAS in PWR assembly depletion

- Fuel, cladding, and moderator compositions can be simplified
- Zirc4 mixture 13 has a different temperature than the other Zirc4 mixtures and therefore cannot be combined



**CE 14x14 fuel assembly**

```
read alias
$fuel 1 4 7 10 end
$clad 2 5 8 11 end
$mod  3 6 9 12 end
end alias
read composition
uo2   $fuel 0.95 873
            92235 3
            92238 97 end
zirc4 $clad 1 620 end
h2o   $mod den=0.7332 1 570 end
boron $mod den=0.7332 0.0003 570 end
zirc4 13 1 570 end
end composition
```

Bonus: KEEP block

```
read keep
end keep
```

➡ No module output is printed

```
read keep
 newt
end keep
```

➡ Only NEWT output is printed; same as having no KEEP block

```
read keep
 newt origen xsproc
end keep
```

➡ NEWT, XSProc, and ORIGEN output is printed

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# TRITON Input

## Part III: TIMETABLE block



# Main input blocks

- **Sequence identifier and parameters**
  - t-xsdrn, t-newt, t-depl-1d, t-depl, t5-depl, t6-depl, t5-depl-shift, t6-depl-shift
  - addnux option: selection of nuclides considered during neutron transport
- **Alias**
  - Definition of aliases for compositions to simplify input
- **Composition**
  - Material specifications (nuclides, densities, temperatures)
- **Celldata** (MG only)
  - Cells to be used for cross section self-shielding calculation with XSProc
- **Depletion**
  - List of mixtures to be depleted
  - Power normalization
- **Burndata**
  - Specific power [MW/MTIHM]
  - Depletion and decay intervals [days]
  - Number of depletion intervals
- **Timetable** (optional)
  - Time-dependent nuclide concentration and temperature changes
- **Collapse, Homogenize, ADF, Branch** (optional)
  - Lattice physics parameter generation
- **Opus** (optional)
  - Generate text files for time-dependent fuel concentrations
- **Keep** (optional)
  - Enable additional module output
- **Transport model**:
  - Parameters, geometry, volume, etc.

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# TIMETABLE block – Part 1

Time-dependent changes in concentrations and temperatures



# TIMETABLE block – Temperature

- **Temperature** specifications

**temperature I $t_1$ $K_1$ $t_2$ $K_2$ …$t_z$ $K_z$ end**

where

**I** = Mixture ID number

**$t_i$** = Time (d) in calculation where temperature $K_i$ is set

**$K_i$** = Temperature (in K) of specified nuclides at time $t_i$

# TIMETABLE block – Density

- **Density** specifications

**density I C $N_1$ $N_2$ ...$N_C$ $t_1$ $D_1$ $t_2$ $D_2$ …$t_z$ $D_z$ end**

where

**I** = Mixture ID number
**C** = Number of nuclides to which this change is applied
$N_i$ = Nuclide ID (SCALE ZA no.) to which change is applied
$t_i$ = Time (d) in calculation where density $D_i$ is set
$D_i$ = Density multiplier (fractional change) of specified nuclides at time $t_i$

If **C** = 0, then no nuclides are specified, and the density multiplier is applied to all nuclides in the mixture.

# TIMETABLE block – Example

| Cycle | Boron concentration (ppm) | | Temperature (K) |
|---|---|---|---|
| | BOC | EOC | |
| 1 | **1000** | **100** | **615** |
| 2 | **1250** | **130** | **685** |
| 3 | **980** | **100** | **610** |

```
read timetable
temperature 3
    0.0   615    121.0   615
  121.01  685    322.5   685
  352.5   610    738.75  610 end
density 3 2 5010 5011
    0.0   1.0    121.0   0.1
  121.01  1.25   322.5   0.13
  352.5   0.98   738.75  0.1 end
end timetable
```

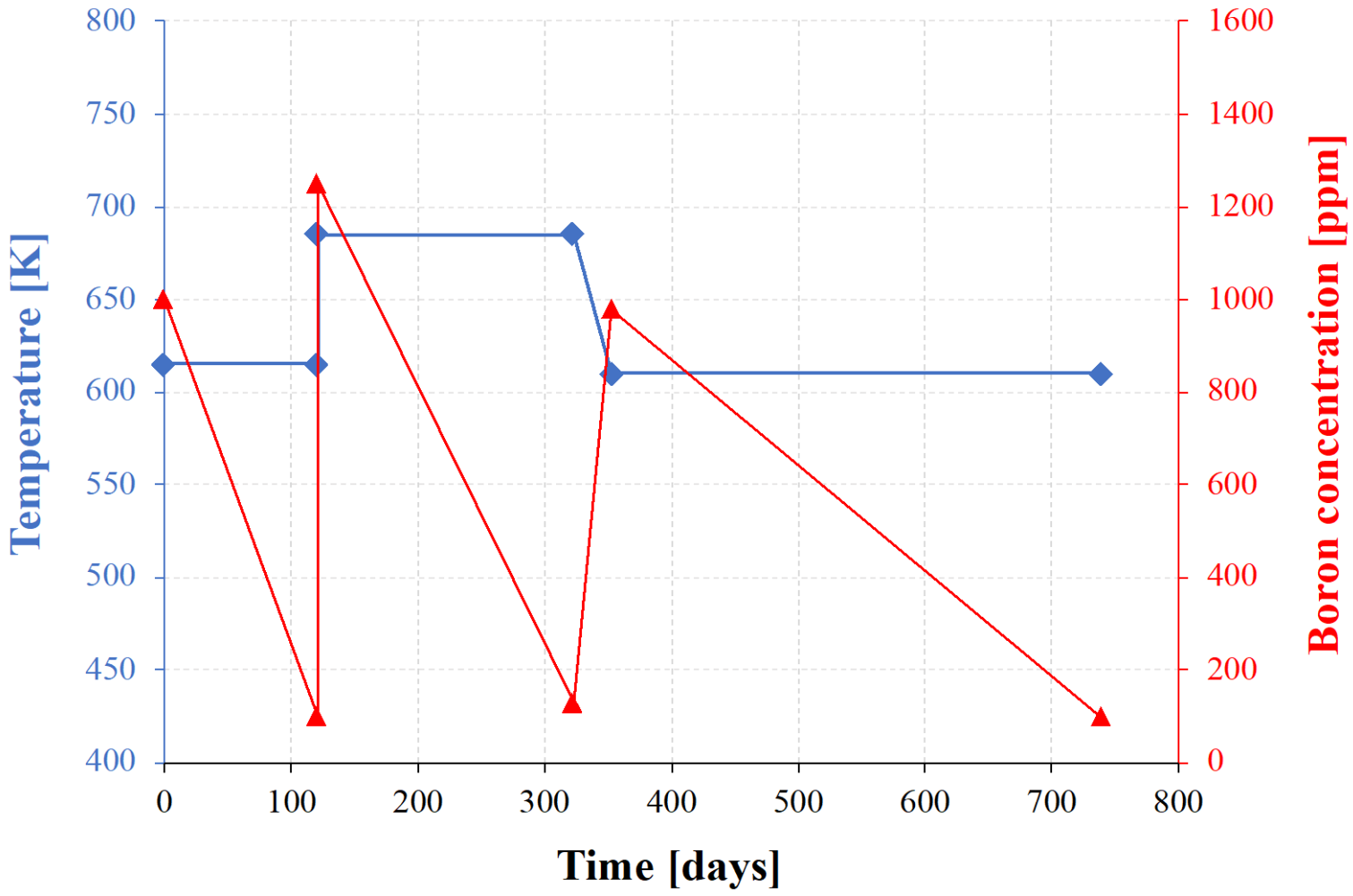
Initial boron concentration (in the COMPOSITION block): 1000 ppm

Cycle 1: burn from 0 to 121 d, no down time.

Cycle 2: burn from 121 to 322.5 d, down from 322.5 to 352.5 d.

Cycle 3: burn from 352.5 d to 738.75 d, down time unimportant.

# TIMETABLE block – Example




```
read timetable
temperature 3
     0.0   615    121.0   615
   121.01  685    322.5   685
   352.5   610    738.75  610 end
density 3 2 5010 5011
     0.0   1.0    121.0   0.1
   121.01  1.25   322.5   0.13
   352.5   0.98   738.75  0.1 end
end timetable
```

# TIMETABLE block – Density/temperature notes

```
read timetable
temperature 3
  121.0  615
  121.01 685
  322.5  685
  352.5  610
  738.75 610 end
density 3 2 5010 5011
    0.0  1.0
  121.0  0.1
  121.01 1.25
  322.5  0.13
  352.5  0.98
  738.75 0.1 end
end timetable
```

- Timetable curves are used to determine temperatures and densities at the time of each transport calculation
- **Linear interpolation is used between points**
- Standard comp specifications are used unless altered by a timetable entry
- Select nuclide densities can be varied for boron letdown, mixture exchange

# TIMETABLE block – Swap

- **Swap** specifications

**swap $I_1$ $I_2$ $t_1$ $S_1$ $t_2$ $S_2$ $t_3$ $S_3$ ... $t_C$ $S_C$ end**

where

**$I_1$, $I_2$** = first and second mixture ID
**$t_i$** = time (days) in calculation where swap ID is set, from **i=1** to **C**
**$S_i$** = swap value 0/1 at time **$t_i$**, **i=1** to **C**
**C** = number of time steps

**$S_j$ = 0**: model the swap materials as defined in the nominal model
**$S_j$ = 1**: swap every **$I_1$** for every **$I_2$** and swap every **$I_2$** for every **$I_1$**

# TIMETABLE block – Swap example

```
read timetable
  swap 40 50
        0.000      1
      500.000      0
     1000.000      1  end
end timetable
```

- Mixture 40 is the mixture used in nominal model: water
- Mixture 50 is the mixture to be swapped: control rod

- Control rod **in** for the first 500 days
- Control rod **out** for days 500–1000 days
- Control rod **in** for anything after 1000 days

# TIMETABLE block – Swap notes

- User should construct the model so that explicit mixtures are used for *control rod* ***out*** and *control rod* ***in*** regions
- User should construct the model such that control rods are **out** in the nominal case
- Use the timetable to swap the *control rod* ***in*** materials for the *control rod* ***out*** materials
- Materials swapped **out** of the model are not depleted or decayed; their concentrations remain static
- Use a simple test case to make sure swap is operating correctly before implementing it in a long-running case

# TIMETABLE block – Comments on MG calculations

- TIMETABLE entries alter self-shielding calculation
- TIMETABLE entries should modify mixtures in all cells affected, even if not directly used in the transport model – they are used in resonance processing

```
read celldata
  latticecell squarepitch pitch=1.3 3 fueld=0.9 1 cladd=1. 2 end
  latticecell squarepitch pitch=1.3 6 fueld=0.9 4 cladd=1. 5 end
  latticecell squarepitch pitch=1.3 9 fueld=0.9 7 cladd=1. 8 end
end celldata
```

- Example:
  Transport model may only use mixtures 1, 2, 3, 4, and 7; however, timetable data should specify boron letdown or moderator temperature for all moderators

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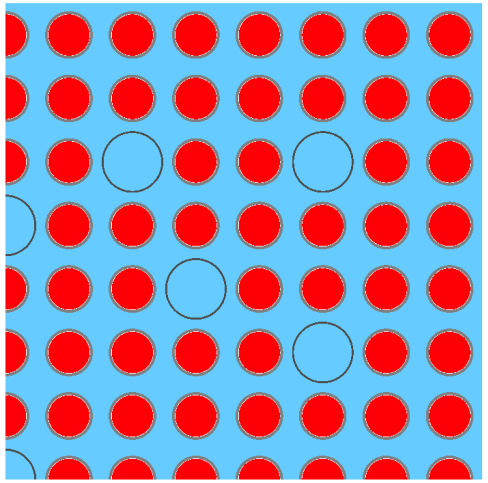
# Workshop

- TRITON-2D



# **Problem A-6:** Time-dependent boron concentration

Create a TRITON depletion model for a Westinghouse 15×15 assembly



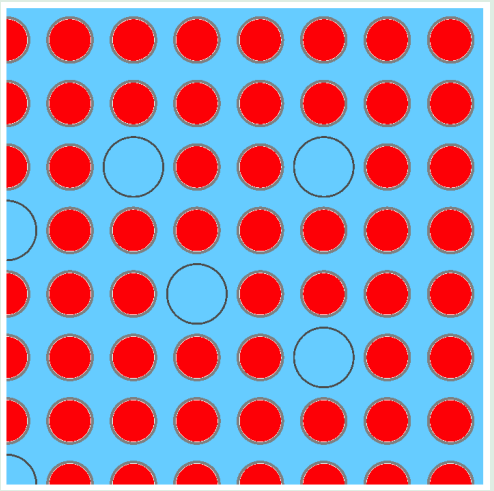
1. Open "w15_assm_tnewt.inp" and save as "w15_assm_depl.inp"
2. Update the model:
   - Add the depletion and timetable data based on the table below
   - Deplete only fuel mixture (ID 1)
   - Use one depletion sub-interval per depletion interval
3. Investigate the output:
   - Find the boron concentration of the moderator
   - Why is the moderator mixture not found on the F71 file?

| Depletion Interval | Power (MW/MTU) | Depletion Interval (d) | Decay Interval (d) | BOC Boron (ppm) | EOC Boron (ppm) |
|---|---|---|---|---|---|
| 1 | 25 | 330 | 35.25 | 1000 | 100 |
| 2 | 20 | 330 | 35.25 | 800 | 100 |
| 3 | 15 | 330 | 35.25 | 900 | 200 |
| 4 | 15 | 330 | 35.25 | 500 | 250 |

**Note** that the moderator is initially defined to contain **1000 ppm boron**

# **Solution A-6:** Time-dependent boron concentration

- Since the moderator mixture (mix 3) is not depleted, it is not included in the F71 file (ORIGEN doesn't know about this mixture)
- You find the composition of all mixtures in NEWT's mixing table at the time of each transport calculation if you set "prtmxtab=yes":



| Time of transport [days] | $^{10}$B concentration [atoms/b-cm] |
|---|---|
| 0.00 | 7.90917E-06 |
| 165.00 | 4.35004E-06 |
| 530.25 | 3.55913E-06 |
| 895.50 | 4.35004E-06 |
| 1260.75 | 2.96594E-06 |

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# TRITON Input

## Part IV: TRITON-2D blocks COLLAPSE, HOMOGENIZE, ADF, and BRANCH



# Main input blocks

- **Sequence identifier and parameters**
  - t-xsdrn, t-newt, t-depl-1d, t-depl, t5-depl, t6-depl, t5-depl-shift, t6-depl-shift
  - addnux option: selection of nuclides considered during neutron transport
- **Alias**
  - Definition of aliases for compositions to simplify input
- **Composition**
  - Material specifications (nuclides, densities, temperatures)
- **Celldata** (MG only)
  - Cells to be used for cross section self-shielding calculation with XSProc
- **Depletion**
  - List of mixtures to be depleted
  - Power normalization
- **Burndata**
  - Specific power [MW/MTIHM]
  - Depletion and decay intervals [days]
  - Number of depletion intervals
- **Timetable** (optional)
  - Time-dependent nuclide concentration and temperature changes
- **Collapse, Homogenize, ADF, Branch** (optional)
  - Lattice physics parameter generation
- **Opus** (optional)
  - Generate text files for time-dependent fuel concentrations
- **Keep** (optional)
  - Enable additional module output
- **Transport model**:
  - Parameters, geometry, volume, etc.

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# COLLAPSE block

Collapsing cross sections to a reduced group structure

Only available with TRITON-NEWT:
T-NEWT and T-DEPL



# COLLAPSE block

- Contains instructions to collapse the cross sections to a reduced group structure
- For each fine-group energy group, a broad-group assignment is made
- Broad-group numbers must be contiguous (constant or increasing)
- Cross sections are collapsed using the converged flux in each mixture to collapse each nuclide in each mixture
- FIDO-type repeat factors are permitted

# Example COLLAPSE block

- Collapse 56 group cross sections to 3 groups (56 entries required):

```
read collapse
  1 1 1 1 1 1 1 1 1 1 1 1 1 2 2 2 2 2 2 2 2 2 2 2 2 2
  2 2 2 2 2 2 2 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
  3 3 3 3 3 3 3 3 3 3 3 3
end collapse
```

- Condensed version of the same operation (14+20+32=56):

```
read collapse 14r1 20r2 32r3 end collapse
```

or

```
read collapse 14*1 20*2 32*3 end collapse
```

# LWR COLLAPSE blocks

- To generate cross sections for core simulators, we recommend a 2-group structure using a 0.625 eV boundary

```
read collapse
'252G->2G
  213r1 39r2
end collapse
```

```
read collapse
'56G->2G
  40r1 16r2
end collapse
```

# COLLAPSE output

- If setting *prtbroad=yes or prtbroad=1d*, collapsed cross sections are printed in the regular output
  - Search for "Collapsing cross-sections" or "Broad group parameters by nuclide"
  - The *1d* option disables the printing of the 2D scattering tables (depending on the group structure, those can be lengthy)

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# HOMOGENIZATION block

Collapsing mixtures into a single, reaction-rate-equivalent mixture

Only available with TRITON-NEWT:
T-NEWT and T-DEPL



# HOMOG(enization) block

- Homogenization is the process of collapsing cross sections over space (often in tandem with an energy collapse)
- Multiple mixtures can be homogenized into a single effective mixture with cross sections that preserve reaction rates in the homogenized region
- Transport methods usually provide the ability to homogenize over a cell or zone

# HOMOG(enization) block

- Due to the arbitrary nature of the NEWT geometry, there are no predefined cells or zones
- NEWT performs homogenization over a user-specified set of mixtures
  - The placement of these mixtures defines the cell or zone

# HOMOG(enization) block

- The homogenization block has the following form:

```
read homog
<Hom_Mix_ID1> <MixName1> <list of mixtures> end
<Hom_Mix_ID2> <MixName2> <list of mixtures> end
...
end homog
```

- A homogenized cross section library is created with the homogenized ***Hom_Mix_ID****(s)* as ID(s) and ***MixName****(s)* as the names

- The ***MixName*** must be contiguous text of up to 12 characters with no embedded white space

# Example HOMOG block

- Mixture ID numbers are arbitrary
- A mixture may appear in more than one homogenization set
- LWR user guidance:
  - Fuel Assembly Models: One set that includes all mixtures that are listed in the MATERIAL block
  - Reflector Assembly Models: Two sets – (1) reflector materials, (2) fuel materials
- Example:

```
 read homog
   500 fuel  201 202 203 204 205 end
   501 fuel_cell 201 202 210 220 end
end homog
```

- Word of caution – TRITON does not remind you to homogenize all materials in the model; it is easy to forget a material

# HOMOG(enization) block results

- The homogenized mixture macroscopic cross sections are printed in the regular output
  - Search for "Flux weighted cross-sections for homogenized regions"
- The presence of the HOMOG block enables the generation of a few-group cross section file, a "t16" file
  - A binary file "xfile016" is also created and contains the same information; this file must be manually copied from TMPDIR
  - t16 files can be used to prepare cross sections for use in a core simulator (e.g., GenPMAXS processes t16 files into the format for core simulator PARCS)

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# ADF block

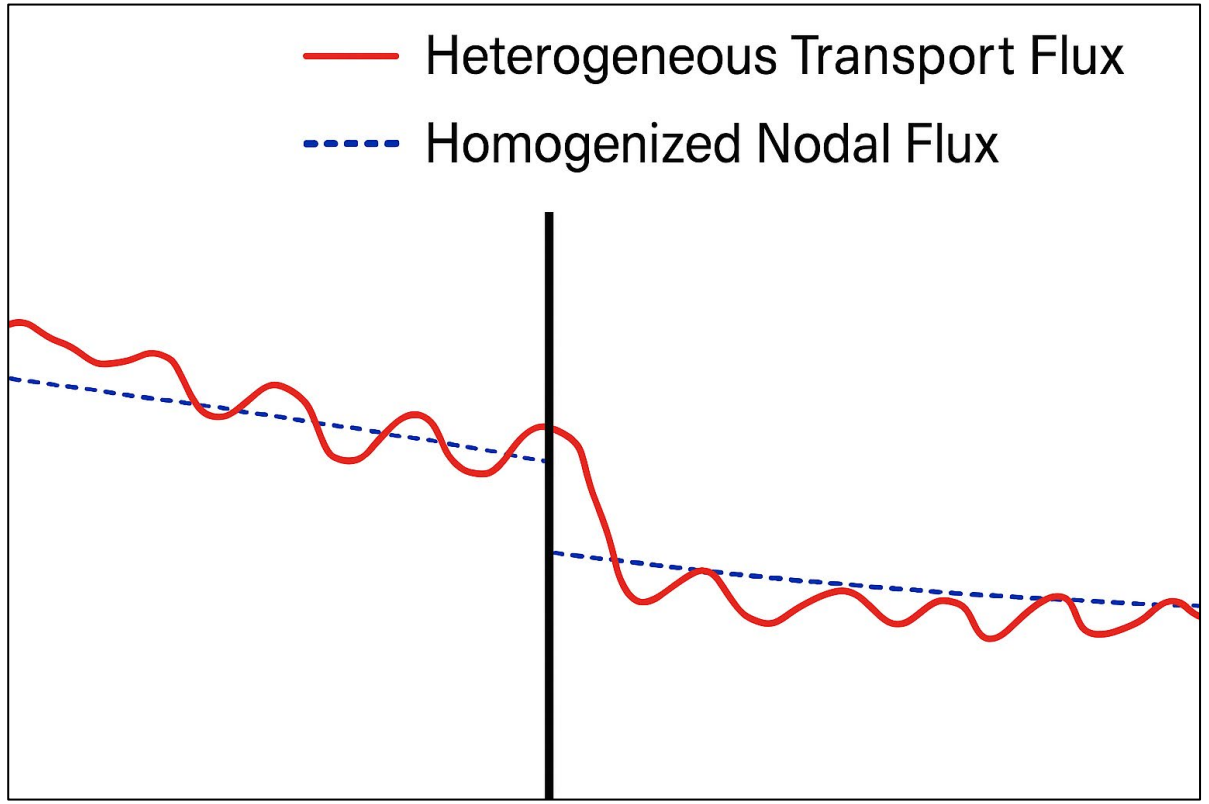
Calculation of assembly discontinuity factors for lattices

Only available with TRITON-NEWT:
T-NEWT and T-DEPL



# Assembly discontinuity factors

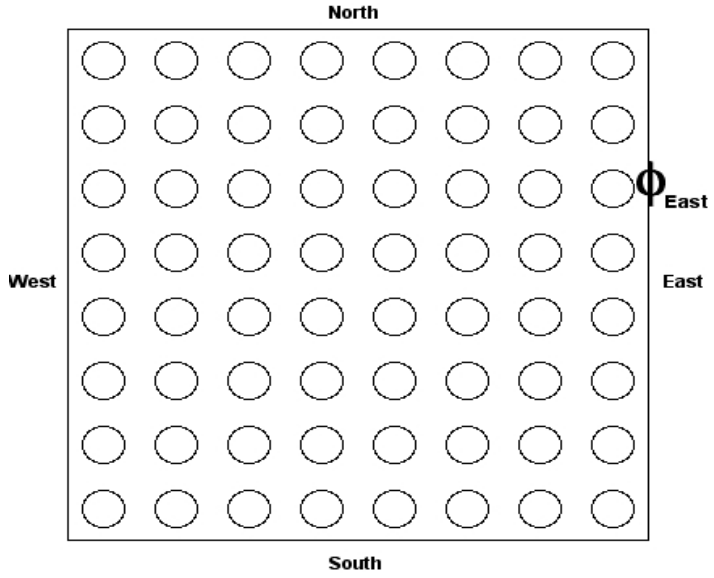
- ADFs are used by nodal diffusion codes to force continuous currents at nodal boundaries



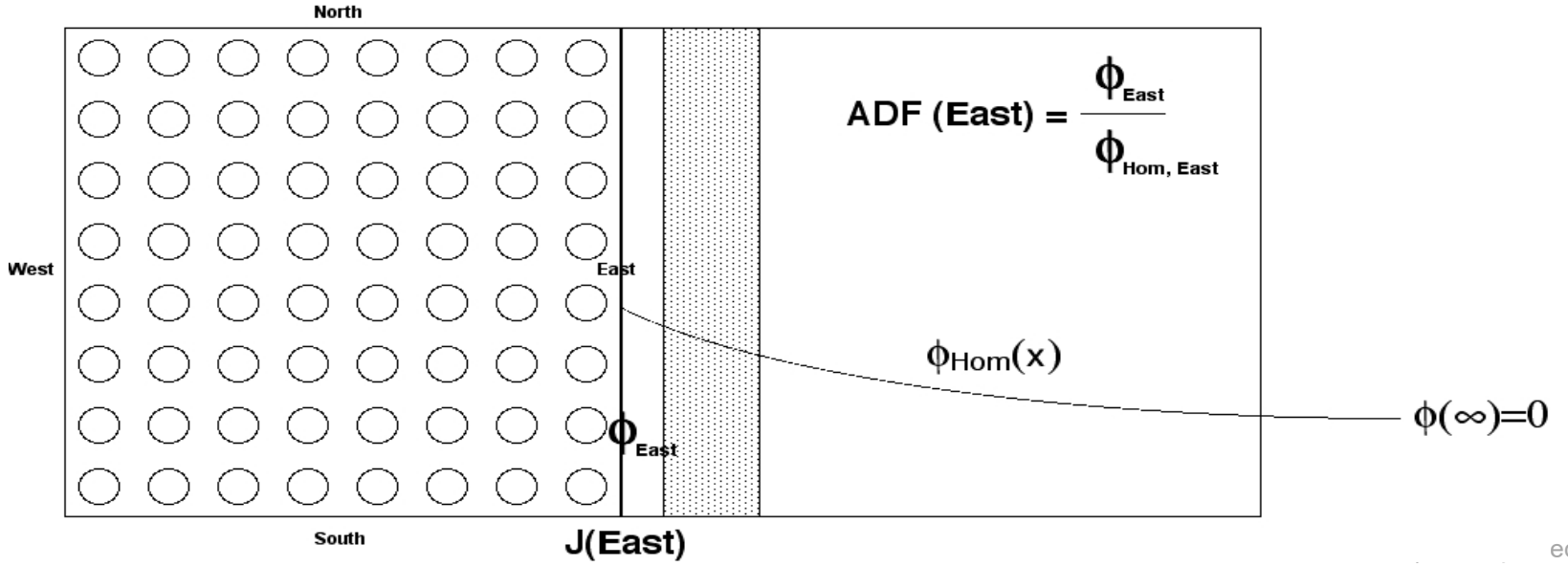
# Assembly discontinuity factors

- Defined as the ratio of the heterogeneous flux to the homogeneous flux on assembly boundaries
- The heterogeneous flux is computed by NEWT
- For a single assembly, the homogeneous flux is the spatially averaged flux
- For a reflected assembly, NEWT uses a one-dimensional diffusion solution in the reflector to obtain the homogeneous flux at the assembly–reflector boundary

# Assembly discontinuity factors




$$ADF\ (East) = \frac{\phi_{East}}{\overline{\phi}_{Assembly}}$$




$$ADF\ (East) = \frac{\phi_{East}}{\phi_{Hom,\ East}}$$

# ADF block

- ADF specifications require prior specification of:
  - Homogenized assembly (unreflected assembly)
  - Homogenized assembly and reflector (reflected assembly)
- ADF input specifies:
  - Global x coordinate of east, west faces
  - Global y coordinate of north, south faces

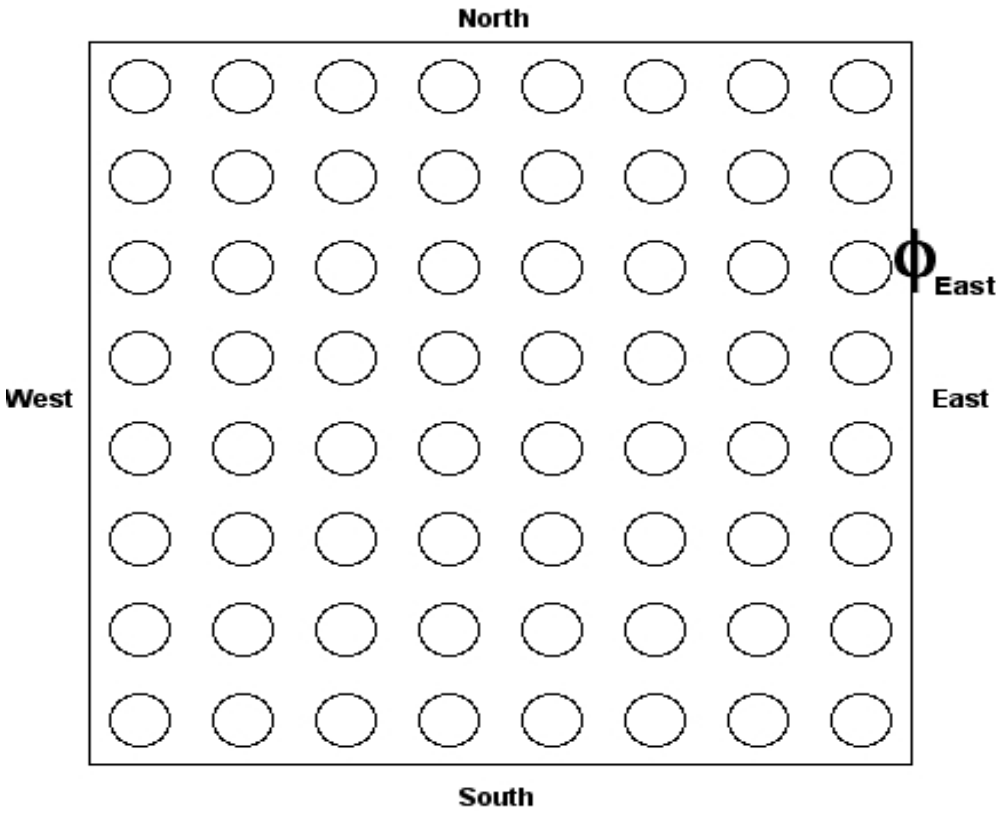
# ADF block examples

- Unreflected (interior) assembly

**Index 1 indicates this is an unreflected fuel assembly**

```
read homog
500 fuel 1 25 26 end
end homog

read adf
1 500 n=10.3124 s=0.0 e=10.3124 w=0.0
end adf
```



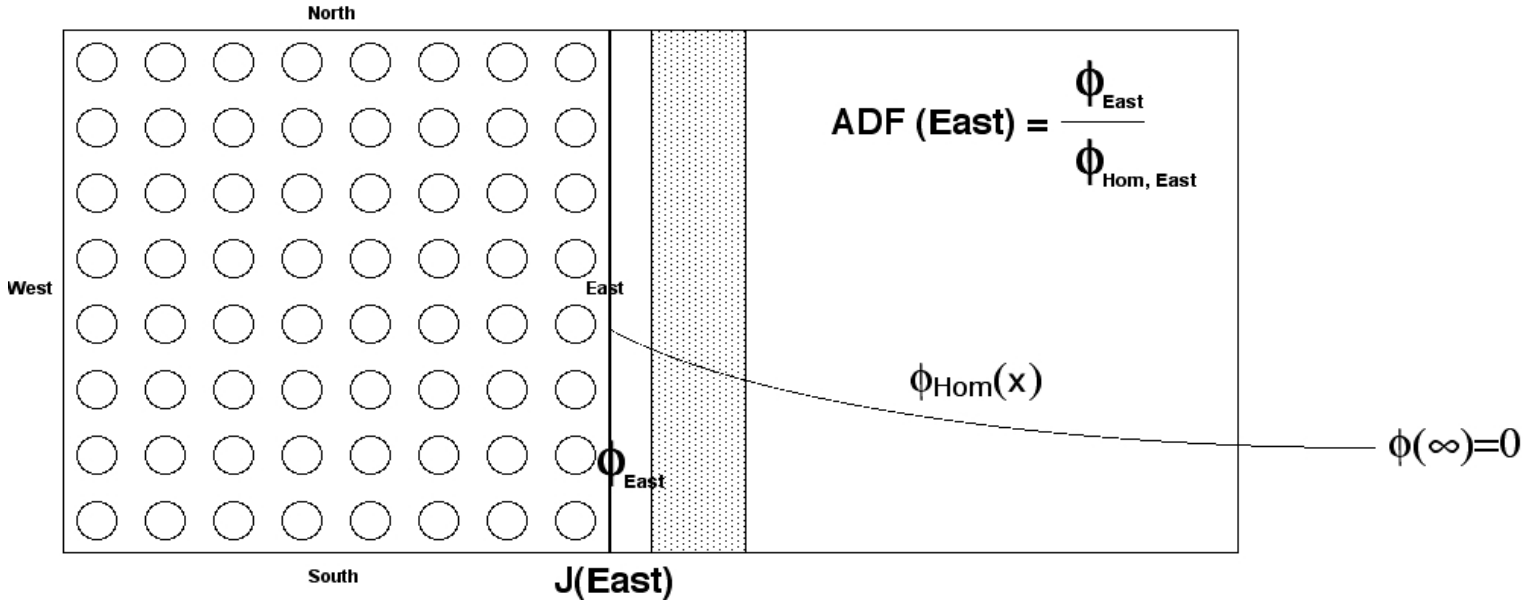
# ADF block examples

- Reflected (outer) assembly

**Index 2 indicates a reflected fuel assembly. The first homogenized mix will be fuel, the second the reflector. Only one ADF location (E/W or N/S) can be specified, which represents the interface location.**

```
read homog
500 fuel 1 25 26 end
501 refl 30 end
end homog

read adf
2 500 501 e=10.3124
end adf
```



# Notes on ADFs

- ADFs are calculated for each energy group in the few-group set
- Fuel ADFs must always be calculated
- Reflector ADFs must be calculated if the assembly has a neighboring reflector

# ADF block results

- ADFs are included in the few-group cross section file, the “t16” file
  - Order of ADFs: west, south, east, north
- ADFs are also printed in the regular output along the homogenized mixture macroscopic cross sections
  - Search for “Assembly Discontinuity Factors”
  - This printout includes the information on the west, south, east, and north face

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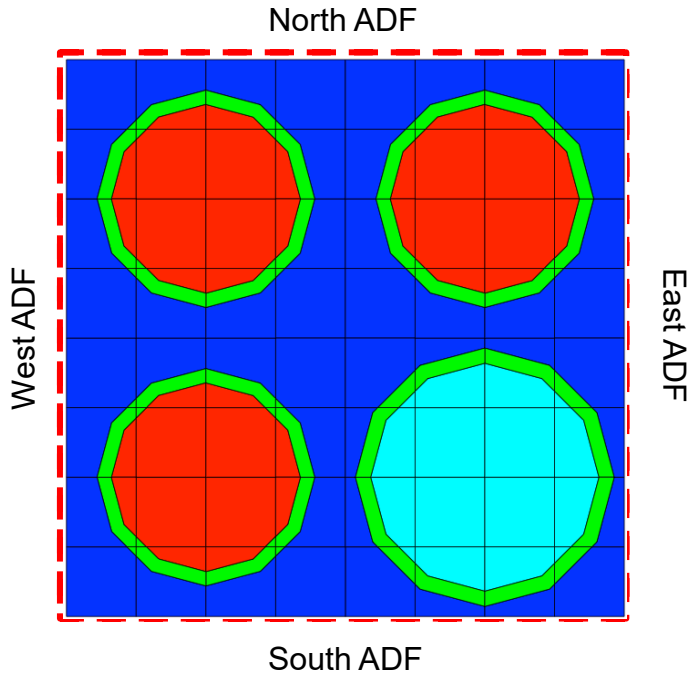
# Workshop

- TRITON-2D



# **Problem A-7:** Few-group XS generation for a PWR mini assembly

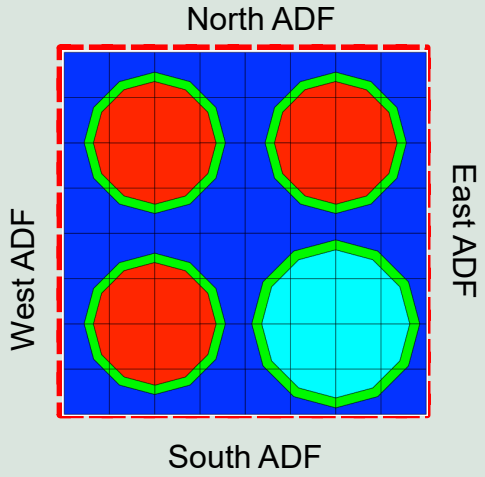
1. Copy "pwr_mini_assm_tnewt.inp" to a new file "pwr_mini_assm_tdepl_fgxs.inp"
2. Update the model:
   - Change the sequence to a "t-depl" sequence
   - Deplete assembly: 25 MW/MTU, 1,000 days, 3 sub-intervals
   - Mixture 1 is $UO_2$ fuel; deplete by constant power
   - Homogenize all used materials with a HOMOG block
   - Collapse the cross sections from 56 energy groups to 2 energy groups (fast and thermal) with a COLLAPSE block
   - Add assembly discontinuity factors for the north, south, east, and west assembly boundaries with an ADF block
3. Run the model and investigate the output:
   - Open the 2-group flux plot file and explain what you see
   - Find the printout in the regular output file
   - Open the t16 file and record fission cross section and ADFs




**PWR mini assembly**

# **Solution A-7:** Few-group XS generation for a PWR mini assembly

*Flux plots generated for both fine-group and broad-group via prtflux=yes*




**Fast flux (>0.625 eV)** **Thermal flux (<0.625 eV)**

**Nodal data at t=0**

| | fission | elastic scattering | flux | diffusion coefficient |
|---|---|---|---|---|
| fast | 0.003 | 0.556 | 2.531 | 1.512 |
| thermal | 0.056 | 1.417 | 0.540 | 0.340 |

**ADFs at t=0**

| | west | south | east | north |
|---|---|---|---|---|
| fast | 1.015 | 0.976 | 0.976 | 1.015 |
| thermal | 0.901 | 1.224 | 1.224 | 0.901 |

```
read depletion
  1
end depletion
read burndata
  power=25 burn=1000 nlib=3 end
end burndata
…
read collapse
 40r1 16r2
end collapse
read homog
  500 mini  1 20 30 40 50 end
end homog
read adf
  1 500 n=1.875 s=-1.875 e=1.875 w=-1.875
end adf
```

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# BRANCH block

Auxiliary calculations to generate few-group cross sections of core simulation

Available with TRITON-NEWT: T-DEPL

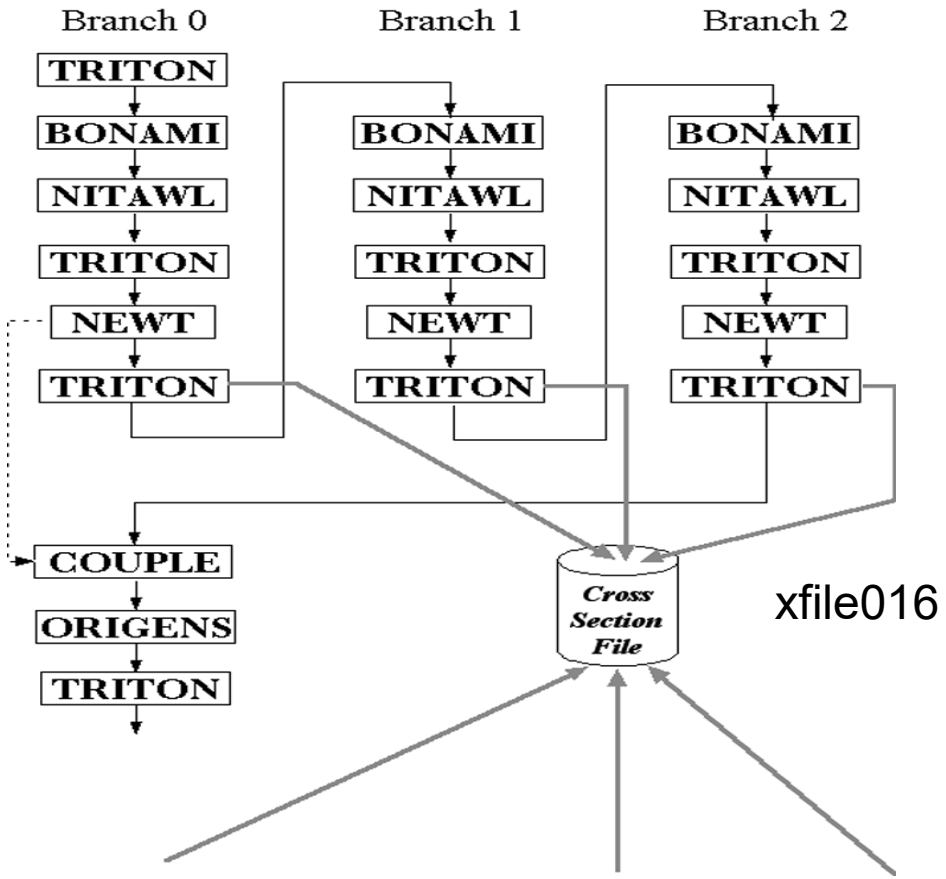


# BRANCH block

- Branch calculations are supported in TRITON-NEWT for lattice physics analysis
  - Branch calculations are not currently supported in TRITON-3D sequences
- Branch calculations are commonly used to generate few-group cross sections for one or more perturbed states for each depletion step
- Branch calculations allow for a nominal condition transport calculation followed by perturbation of one or more of the following parameters:
  - Fuel temperature
  - Moderator temperature
  - Moderator density
  - Soluble boron concentration
  - Control rod insertion

# BRANCH block

- Cross sections are computed for the nominal state and each perturbed state
- At each state:
  - Cross section processing calculations are performed
  - NEWT transport calculation is performed
  - Few-group cross sections are generated (HOMOG and COLLAPSE block)



# BRANCH block

- The BRANCH data block consists of two parts:
  - The first part defines mixture associations to be used in branch calculations:
    - Mixture numbers are associated with keywords "fuel" or "mod"
    - Mixtures present in the transport model during control rod insertion and withdrawal are defined
  - The second part defines the actual branch states
- For N branches, N+1 transport calculations are performed
- Nominal or base conditions are considered "branch 0"
- Depletion calculations are all based on branch 0 conditions
- The first branch in the branch block must be the branch 0 (nominal) conditions

# BRANCH block

```
read branch
define <deftype> <I1 I2 … In> end
...
tf=<fueltemp> tm=<modtemp> dm=<moddens> sb=<boronconc> cr=<inout> end
...
end branch
```

| | | |
|---|---|---|
| **deftype** | = | 'fuel,' 'mod,' 'crout', or 'crin' |
| **$I_i$** | = | list of mixtures associated with type definition **deftype** |
| **fueltemp** | = | branch fuel temperature (K) |
| **modtemp** | = | branch moderator temperature (K) |
| **moddens** | = | branch moderator density (g/cm$^3$) |
| **boronconc** | = | soluble boron concentrations (ppm) |
| **inout** | = | control rod state (out=0, in=1) |

# Example BRANCH block

6 branch specifications → 6 calculations: 1 base, 5 branches

```
read branch
define fuel  11 12 end                      fuel branches are applied to mixtures 11 & 12
define mod   13 14 end                      mod branches are applied to mixtures 13 & 14
define crout 20 21 end                      mixtures 20 & 21 are present for C/R out
define crin  30 31 end                      mixtures 30 & 31 are present for C/R in
dm=0.75 cr=0 sb=655 tf=900 tm=650 end       0) initial (base) conditions
dm=0.80 cr=0 sb=655 end                     1) mod den changed to 0.8
dm=0.80 cr=1 sb=655 end                     2) C/R in, mod den changed to 0.8
sb=20   dm=0.8 end                          3) soluble B reduced to 20ppm, mod den 0.8
sb=1300 dm=0.8 end                          4) soluble B increased to 1300ppm, mod den 0.8
tf=559 end                                  5) fuel temp changed to 559, mod den to base value
end branch
```

Tip: If a state (moderator density, etc.) does not differ from the nominal state, it can be omitted.

# Advanced branches: Dancoff factors

- Users can define a Dancoff factor set to be used in a certain branch
- TRITON will replace all **N**'s in **dan2pitch(N)** statements, where N is the fuel mixture number and DF is the MCDANCOFF-calculated Dancoff factor

```
read branch
  define fuel    $1fl240 $1fl280 $1fl320 $1fl360 $1fl395 $1fl440 $1fl490
                 $fuel440 $fuel240 $fuel280 $fuel320 $fuel360 $fuel395
                 $fuel490 $1f490gd700 $1f440gd700 $1f440gd800
                 $f490gd700 $f440gd700 $f440gd800 $1fl440E $fuel440E
                 $1fl490E $fuel490E                    end
  define mod     $mod                                  end
  define crin    $cbpelin $cbcladin $cbwingin          end
  define crout   $cbout                                end
  define d2pset  100    140 0.360   280 0.360   320 0.560 360 0.560 395 0.560   449 0.560   499 0.560 end
  define d2pset   80    140 0.300   280 0.300   320 0.460 360 0.460 395 0.460   449 0.460   499 0.460 end
  define d2pset   40    140 0.230   280 0.230   320 0.340 360 0.340 395 0.340   449 0.340   499 0.340 end
  define d2pset    1    140 0.175   280 0.175   320 0.260 360 0.260 395 0.260   449 0.260   499 0.260 end
  dm=0.7397      tf=863.15    tm=559.15   sb=1    cr=0     d2p=  1      end
  dm=0.458432    tf=863.15    tm=559.15   sb=1    cr=0     d2p= 40      end
  dm=0.177164    tf=863.15    tm=559.15   sb=1    cr=0     d2p= 80      end
  dm=0.03653     tf=863.15    tm=559.15   sb=1    cr=0     d2p=100      end
end branch
```

# LWR lattice physics user guidance: BRANCH and TIMETABLE

- Common mistakes in the BRANCH block:
  - SCALE requires that each cell definition in the CELLDATA block use unique mixture IDs for the moderator
  - However, only one moderator ID is typically used in NEWT models

→ BRANCH and TIMETABLE moderator definitions MUST include ALL moderator mixtures for correct cross section processing calculations

```
read celldata
  latticecell squarepitch pitch=1.3 3 fueld=0.9 1 cladd=1.0 2 end
  latticecell squarepitch pitch=1.3 6 fueld=0.9 4 cladd=1.0 5 end
  latticecell squarepitch pitch=1.3 9 fueld=0.9 7 cladd=1.0 8 end
end cell data
```

# LWR lattice physics user guidance – injected input

- TRITON redirect options allows pieces of an input file to be read from an external text file, an "injection file"
- This feature can be useful in the BRANCH block (and anywhere else in the input)
  - A user might often have multiple models for which the same branch structure is used
  - The branches can be read from an injection file
  - The injection file must be copied to TMPDIR so that SCALE can find it
  - Fulcrum may not correctly validate or visualize a model that uses an injection file

```
=shell
 cp ${INPDIR}/tree1.inj  tree1.inj
end
=t-depl
...
read branch
  define fuel  1 201 end
  define mod   5 205 15 end
  define crout 15 end
  define crin  4  end
<tree1.inj
end branch
```

Where the external file named "tree1.inj" contains a set of branches:

```
tf=900 tm=500 dm=0.8458 sb=900  cr=0 end
tf=900 tm=500 dm=0.9    sb=900  cr=0 end
tf=900 tm=500 dm=0.7    sb=900  cr=0 end
tf=900 tm=500 dm=0.7    sb=2000 cr=0 end
```

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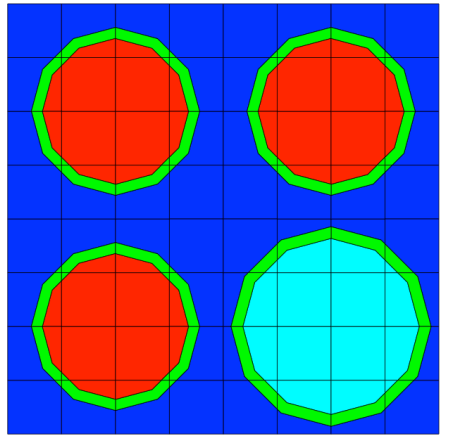
# Workshop

- TRITON-2D



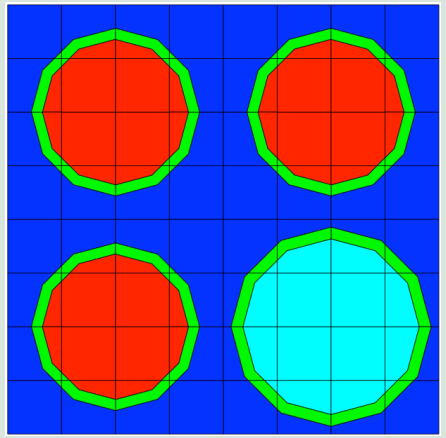
# **Problem A-8:** PWR mini assembly with BRANCH

1. Copy "pwr_mini_assm_tdepl_fgxs.inp" from the previous exercise to a new file "pwr_mini_assm_tdepl_branch.inp"
2. Update the model:
   - Note the control rod (b4c) in southeast corner
   - Add a branch block:
     - Add branch *define* statements: fuel, mod, crout, and crin
     - Add 2 branch conditions
       - Tf=900 dm=0.75 sb=900 cr=0 (Branch 0)
       - Tf=900 dm=0.75 sb=400 cr=0 (Branch 1)
       - Tf=900 dm=0.75 sb=900 cr=1 (Branch 2)
3. Run the model and investigate the output:
   - Find the printout that indicates the model updates for the branch
   - Inspect the t16 files and identify where the data for the branches are stored
   - Record $k_{eff}$ and confirm that the trends make sense



**PWR mini assembly**

# **Solution A-8:** PWR mini assembly with BRANCH



**PWR mini assembly**

T16 file contains nodal data for all branches. Note headers for individual data sets "' Depletion pass no.  N, branch no. N"

Property change printed in TRITON output below
k-eff result: "Property changes for this branch"

```
read branch
  define fuel 1  end
  define Mod 30 50 end
  define Crout 50 end
  define Crin  40 end
  tf=900 tm=540 dm=0.75 sb=900 cr=0 end
  tf=900 dm=0.75 sb=400 cr=0 end
  tf=900 dm=0.75 sb=900 cr=1 end
end branch
```

**Branch 0**

**Branch 1**

**Branch 2**

```
k-eff =    1.38776610   Time=     0.00d  Branch=1
k-eff =    0.48795302   Time=     0.00d  Branch=2
k-eff =    1.30478533   Time=     0.00d Nominal conditions
k-eff =    1.30102290   Time=   166.67d  Branch=1
k-eff =    0.46658807   Time=   166.67d  Branch=2
k-eff =    1.22638094   Time=   166.67d Nominal conditions
k-eff =    1.22000256   Time=   500.00d  Branch=1
k-eff =    0.43623686   Time=   500.00d  Branch=2
k-eff =    1.14936779   Time=   500.00d Nominal conditions
k-eff =    1.14566177   Time=   833.33d  Branch=1
k-eff =    0.40582781   Time=   833.33d  Branch=2
k-eff =    1.07741840   Time=   833.33d Nominal conditions
```

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# Questions?

**TRITON training goals**

- Understand TRITON's depletion approach
- Understand TRITON's various output files
- Learn how to set up a TRITON model

