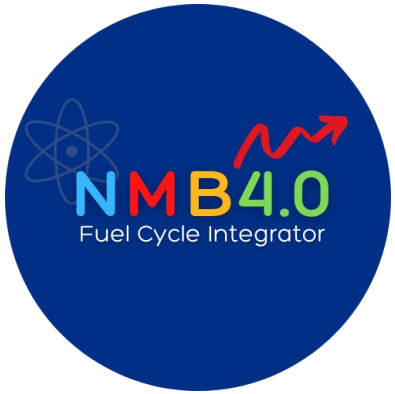
# Methodology of NMB 4.0



# Content

- Main items
  - Material tag
  - Plant throughput
  - Flow of analysis
  - Material handling of each plant
- Others
  - Spent Fuel/Inventory Search
  - Core-Fuel Combinations
  - Fast Breeder Reactor Blanket
  - Accelerator-driven system (ADS)

NMB Methodology

# Main

# Material tag

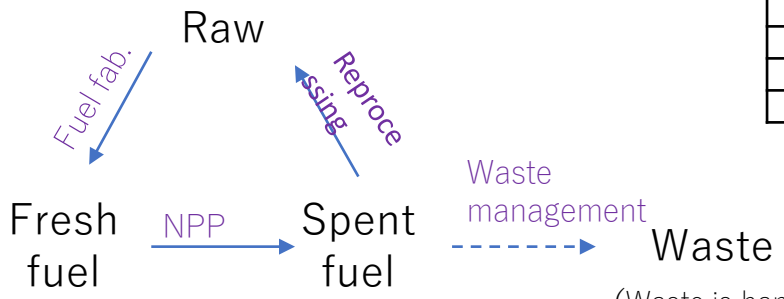
Fundamental function of NMB is to calculate material weigh $x_i(m,t)$ in each plant.

$x_i(m,t)$

$i$: isotope, $m$: material tag, $t$: year

Three kind of material tag
1. Raw material
2. Fresh fuel
3. Spent fuel

Material tags attached to substances change as they pass through the process (plant)




(Waste is handled by waste-tag)

Default material tag defined in NMB4.0

| Tag | Description | Tag | Description | Tag | Description |
|---|---|---|---|---|---|
| natTh | Natural Th | fUO2 | UO2 FF | sUO2 | UO2 SF |
| repTh | Recovered Th | fUO2_BWR | UO2 FF (BWR) | sUO2_BWR | UO2 SF (BWR) |
| natU | Natural U | fUO2_PWR | UO2 FF (PWR) | sUO2_PWR | UO2 SF (PWR) |
| enrU | Enriched U | fLMOX | MOX FF (LWR) | sLMOX | MOX SF (LWR) |
| depU | Depleted U | fLMOX_BWR | MOX FF (BWR) | sLMOX_BWR | MOX SF (BWR) |
| repU | Recovered U | fLMOX_PWR | MOX FF (PWR) | sLMOX_PWR | MOX SF (PWR) |
| Pu | Pu stock | fMOX | MOX FF | sMOX | MOX SF |
| dirtyPu | Dirty Pu Stock | fLMOX_RB | MOX FF (RBWR) | sLMOX_RB | MOX SF (RBWR) |
| gcrPu | Pu from gas-cooled reactor | fMOX_MA | MOX FF containing MA | sMOX_MA | MOX SF containing MA |
| gcrcbPu | Pu from CleanBurn | fROX | ROX FF | sROX | ROX SF |
| wepPu | Weapon-grade Pu | fUROX | ROX FF containing U | sUROX | ROX SF containing U |
| MA | MA stock | fMIX | MIX FF | sMIX | MIX SF |
| | | fBLK | FBR blanket FF | sBLK | FBR blanket SF |
| | | fBLK_AX | FBR axial blanket FF | sBLK_AX | FBR axial blanket SF |
| | | fBLK_RD | FBR radial blanket FF | sBLK_RD | FBR radial blanket SF |
| | | fMAN | MA nitride FF | sMAN | MA nitride SF |
| | | fPuN | Pu nitride FF | sPuN | Pu nitride SF |
| | | fCANDU | CANDU FF | sCANDU | CANDU SF |
| | | fGCR | FF for GCR | sGCR | SF for GCR |
| | | fGCRCB | FF for CleanBurn | sGCRCB | SF for CleanBurn |

FF=Fresh fuel, SF=Spent fuel, ROX=Rock-like oxide, LWR=Light water reactor, PWR=Pressurized water reactor, BWR=boiled water reactor, GCR=gas cooled reactor

# Nuclide to be evaluated

- 26 Actinides
  - Any fuel cycle can be calculated.
- 153 FPs
  - FPs are selected so that 99.9% of mass, decay heat, radio-activity and radio-toxicity is included.
  - Ignored nuclides are summed to descendant nuclide of each.

T. OKAMURA et al., JAEA-Data/Code 2020-023

Table 26 Actinides

| Th,Pa | U | Np | Pu | Am | Cm |
|---|---|---|---|---|---|
| Th232 | U232 | Np237 | Pu238 | Am241 | Cm242 |
| Pa231 | U233 | Np238 | Pu239 | Am242m | Cm243 |
| Pa233 | U234 | Np239 | Pu240 | Am243 | Cm244 |
| | U235 | | Pu241 | | Cm245 |
| | U236 | | Pu242 | | Cm246 |
| | U237 | | | | |
| | U238 | | | | |

Shortest $T_{1/2}$ = 2.117d (Np-238)

Table 153 FPs

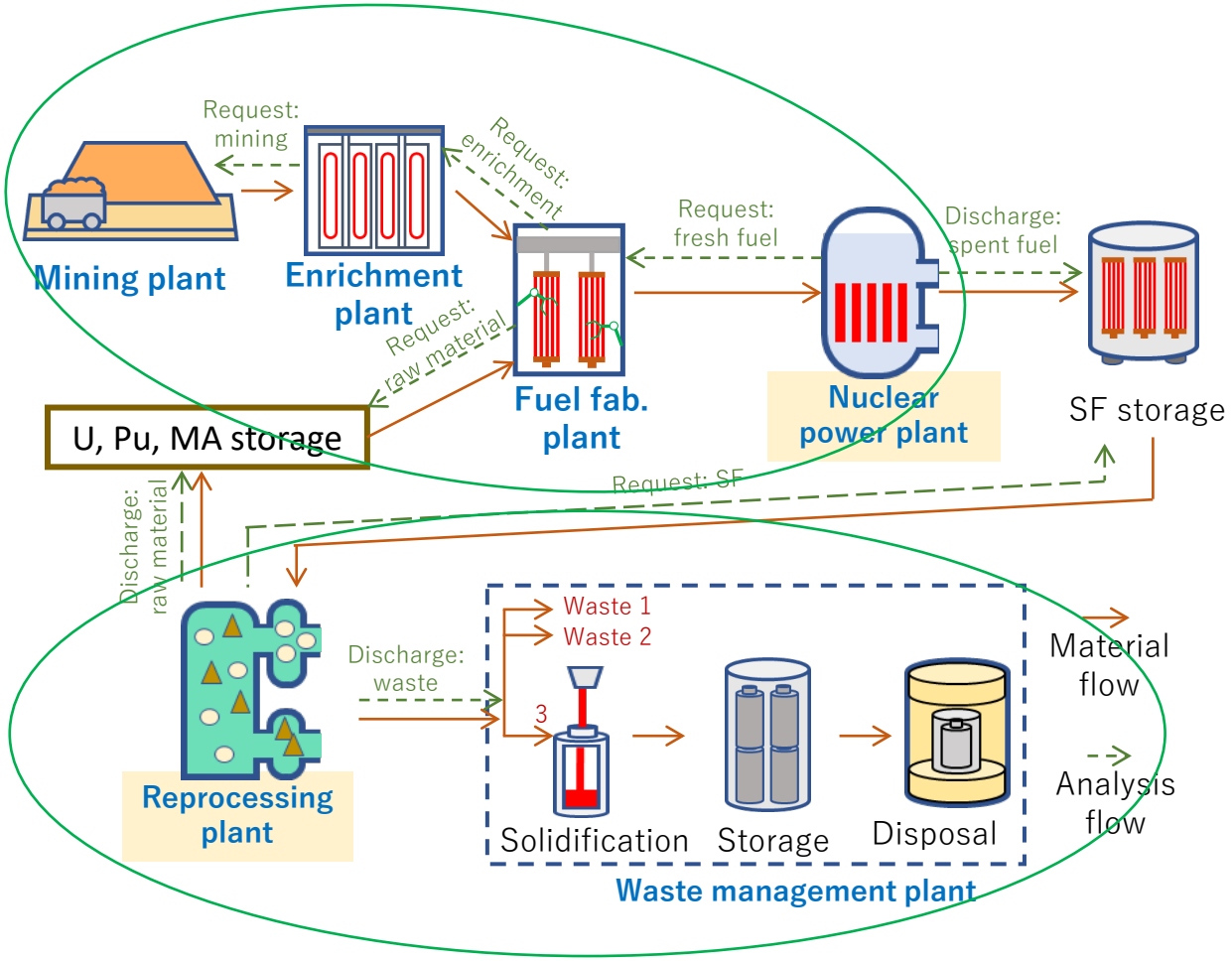
| | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| H3 | Rb85 | Zr91 | Mo95 | Ru103 | Pd108 | Cd114 | Sn122 | Te127m | Xe132 | Ba134 | Pr141 | Pm146 | Sm154 | Gd156 | Ho166m |
| Se78 | Rb87 | Zr92 | Mo96 | Ru104 | Pd110 | Cd116 | Sn124 | Te128 | Xe133 | Ba136 | Pr143 | Pm147 | Eu150 | Gd157 | Tm171 |
| Se79 | Sr86 | Zr93 | Mo97 | Ru106 | Ag109 | In115 | Sn126 | Te130 | Xe134 | Ba137 | Nd142 | Pm148 | Eu151 | Gd158 | Other |
| Se80 | Sr88 | Zr94 | Mo98 | Rh102 | Ag110m | Sn116 | Sb121 | Te132 | Xe135 | Ba138 | Nd143 | Pm148m | Eu152 | Gd160 | |
| Se82 | Sr89 | Zr95 | Mo99 | Rh103 | Cd109 | Sn117 | Sb123 | I127 | Xe136 | Ba140 | Nd144 | Sm147 | Eu153 | Tb158 | |
| Br81 | Sr90 | Zr96 | Mo100 | Rh105 | Cd110 | Sn118 | Sb125 | I129 | Cs133 | La139 | Nd145 | Sm148 | Eu154 | Tb159 | |
| Kr83 | Y89 | Nb93m | Tc99 | Pd104 | Cd111 | Sn119 | Sb126 | I131 | Cs134 | Ce140 | Nd146 | Sm149 | Eu155 | Tb160 | |
| Kr84 | Y90 | Nb94 | Ru100 | Pd105 | Cd112 | Sn119m | Te125 | Xe128 | Cs135 | Ce141 | Nd147 | Sm150 | Eu156 | Dy160 | |
| Kr85 | Y91 | Nb95 | Ru101 | Pd106 | Cd113 | Sn120 | Te125m | Xe130 | Cs137 | Ce142 | Nd148 | Sm151 | Gd154 | Dy161 | |
| Kr86 | Zr90 | Mo94 | Ru102 | Pd107 | Cd113m | Sn121m | Te126 | Xe131 | Ba133 | Ce144 | Nd150 | Sm152 | Gd155 | Dy162 | |

Shortest $T_{1/2}$ = 9.14h (Xe-135)

# Material flow versus Analysis flow

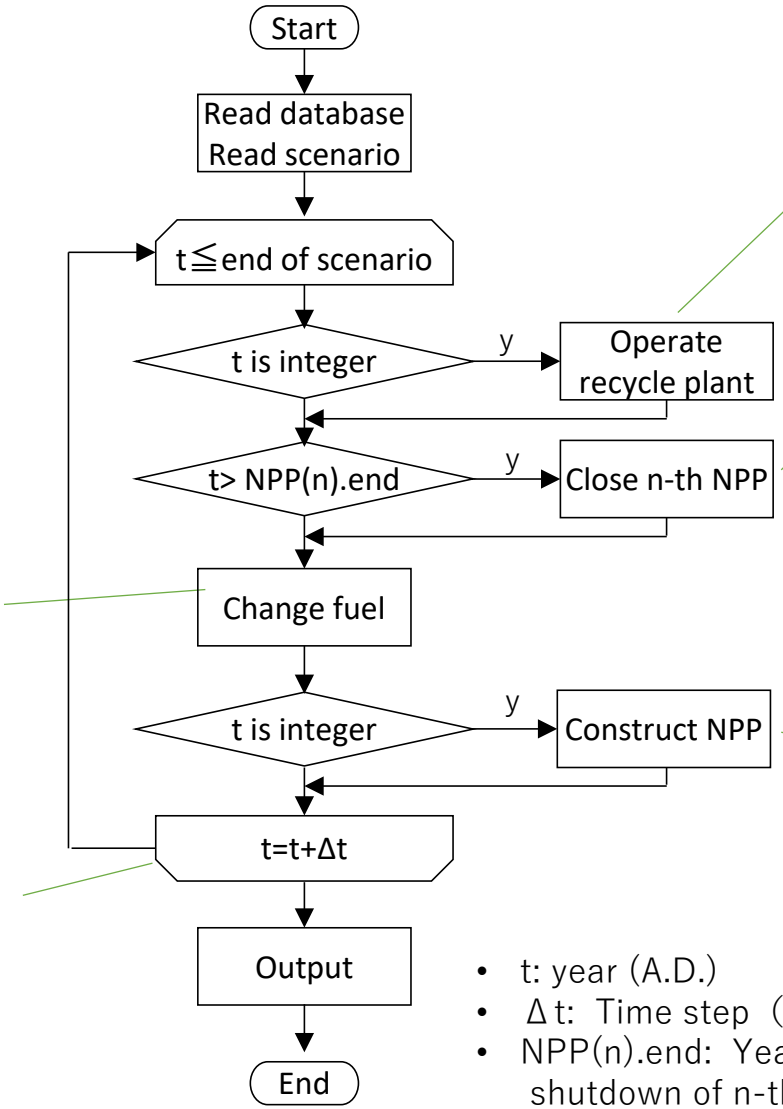
*Analysis flow*

- Fuel is produced in the fuel fabrication plant at the moment the reactor plant is operated at the user-specified capacity.
- If enriched uranium is required for fuel production, enrichment is performed in the enrichment plant at that moment.
- The natural uranium needed for enrichment is also mined at the same time in the mining plant.
- In reality, this is not possible due to the existence of an upper limit to the annual throughput in each plant.
- The NMB implicitly assumes that the demand from the reactors is anticipated beforehand and that an appropriate mining and fuel fabrication plan is developed to meet each plant's upper limit.
- The same method is applied to waste management.




The analysis starts from the reactor plant and the reprocessing plant, with two storages connecting them.

# Flow chart




Reprocessing conducted in integer years

Termination of NPP operation

Fuel exchange

Reactor construction in integer years

Increment by time step

- t: year (A.D.)
- Δt: Time step (～0.01 year)
- NPP(n).end: Year of plant shutdown of n-th NPP

# Definition of Plant throughput

There are six types of NMB plants. For each, one or several throughputs, $y(p,t)$, is calculated.

$$y(p,t), \quad p: \text{plant tag}, t: \text{year}$$

Six plants in NMB4.0

| Plant | Definition of throughput, y |
|---|---|
| Mining plant | Annual amount of mined resource （t/year） |
| Enrichment plant | Enriched amount of natural uranium (t/year), and separative work unit (SWU/year) |
| Fuel fabrication plant | Heavy metal weight of fabricated fuel (t/year) |
| Nuclear power plant | Generation capacity(GWe), and electricity generation (GWey, TWh) |
| Reprocessing plant | Weight of reprocessed spent fuel (t/year) |
| Waste management plant | Accumulated waste production (number), Stored waste (number), accumulated footprint (area) of disposal site ($m^2$) |

Red = user -defined maximum throughput for each year,
White = determined subordinately within the NMB

# Handling of material in each plant

Five kinds of handling is applied to each plant

(a) Generation

$x(m_0)$ is newly generated such as uranium mining.

(b) Division

$x(m_0)$ is divieded into $x(m_1)$~$x(m_n)$. $x_i(m_0) = \sum_k x_i(m_k)$

(c) Composition

$x(m_1)$~$x(m_n)$ is summed up to $x(m_0)$. $x_i(m_0) = \sum_k x_i(m_k)$

(d) Burnup

$x(m_0)$ is irradiated with neutron and changed to $x(m_1)$. $x(m_1)$ is a solution of $\frac{dx}{dt} = Ax$, where $A$ is burn-up matrix and $x(m_0)$ is initial condition.

(e) Cooling

$x(m_0)$is stored and changed to $x(m_1)$ after radioactive decay. $x(m_1)$ is a solution of $\frac{dx}{dt} = Dx$, where $D$ is decay matrix and $x(m_0)$ is initial condition.

# Mining plant

Material with U235-enrichment, $e_{\mathrm{enrU}}$, and weight, $w_{\mathrm{enrU}}$, is loaded to a NPP at time, $t$.

$$x_{\mathrm{U235}}(\mathrm{fUO2}, t) = e_{\mathrm{enrU}} w_{\mathrm{enrU}}, \quad x_{U238}(\mathrm{fUO2}, t) = (1 - e_{\mathrm{enrU}}) w_{\mathrm{enrU}}$$

fUO2=$UO_2$ fresh fuel

Natural uranium with material tag "natU" is automatically generated by NMB.

$$x_{\mathrm{U235}}(\mathrm{natU}, 0) = e_{\mathrm{natU}} w_{\mathrm{natU}}, \quad x_{\mathrm{U238}}(\mathrm{natU}, 0) = (1 - e_{\mathrm{natU}}) w_{\mathrm{natU}},$$

$$w_{\mathrm{natU}} = \frac{e_{\mathrm{enrU}} - e_{\mathrm{depU}}}{e_{\mathrm{natU}} - e_{\mathrm{depU}}} w_{\mathrm{enrU}}$$

natU=natural uranium

Throughput of mining plant, $y(\mathrm{mining}, t)$, is represented by weight of mined natural uranium. Here, time, $t$, is not the point of mining, but the point of demand generation (i.e., when the fuel is loaded into the reactor).

$$y(\mathrm{mining}, t) = w_{\mathrm{natU}}$$

Although the NMB code assumes that mining plants have unlimited throughput, in reality there is an upper limit for mining plants. Therefore, it implicitly assumes that an appropriate operating (import) plan is formulated to meet the upper limit by anticipating the demand from the reactors.

For this planning, it is convenient to think the number of years that demand will occur at the reactors, and the mining, enrichment, and fuel fabrication plants will be ready in time to meet the demand. Therefore, the amount of throughput is defined at the time the demand occurs.

# Enrichment plant

"Division" is operated to natural uranium $x(\mathrm{natU}, 0)$ in the case of 1t of enriched U.

enrU=enriched uranium

depU=depleted uranium

$$x_{U235}(\mathrm{enrU}, \mathrm{t}) = e_{\mathrm{enrU}}, \quad x_{U238}(\mathrm{enrU}, \mathrm{t}) = 1 - e_{\mathrm{enrU}}$$

$$x_{U235}(\mathrm{depU}, \mathrm{t}) = (w_{\mathrm{natU}} - 1)e_{\mathrm{depU}}, \quad x_{U238}(\mathrm{depU}, \mathrm{t}) = (w_{\mathrm{natU}} - 1)(1 - e_{\mathrm{depU}})$$

Throughput is weight of enriched natural uranium, $w_{natU}$, or, separative work unit. Time is the point of demand generation as same as mining plant.

$$y(\text{enrichment weight}, t) = w_{\mathrm{natU}},$$

$$y(\text{enrichment SWU}, t) = w_{\mathrm{enrU}} V(e_{enrU}) + w_{\mathrm{depU}} V\left(e_{\mathrm{depU}}\right) - w_{\mathrm{natU}} V(e_{\mathrm{natU}}),$$

$$V(e) = (1 - 2e) \ln \frac{1 - e}{e}$$

# Fuel fabrication plant 1/2

fLMOX=MOX fresh fuel for LWR

Pu=Pu stock
repU=Reprocessed uranium

$\alpha$ =Pu enrichment

Fuel is fabricated by "composition" handling.

$$x_i(\mathrm{fLMOX}, \mathrm{t}) = w_{fLMOX}\left(\alpha x_i(\mathrm{Pu}, \mathrm{t}) + (1-\alpha)x_i(\mathrm{repU}, \mathrm{t})\right)$$

$$w_{fLMOX} = \frac{(\text{Generation capacity GWe})}{(\text{Generation efficiency})} \quad (\text{Specific power MW t/t})$$

Throughput is defined as weight of fabricated fuel, where time is the point of demand generation, i.e., loading to NPP.

$$y(\text{fuel fabrication}, t) = w_{\mathrm{fLMOX}}$$

Combination of fuel type and raw material

| Type of fuel | Raw Th | Enriched U | U stock | Pu stock | MA sock | Adjusted content |
|---|---|---|---|---|---|---|
| UO2 | - | ○ | - | - | - | U235 enrichment |
| MOX for LWR | - | - | ○ | ○ | - | Pu ratio |
| MOX for FR | - | - | ○ | ○ | - | Pu ratio |
| MOX with MA | - | - | ○ | ○ | ○ | Pu ratio |
| Blanket of FR | - | - | ○ | - | - | NA |
| ADS | - | - | - | ○ | ○ | Pu ratio for initial core |
| Th-U233 | ○ | - | ○ | - | - | U ratio |
| Th-U235 | ○ | ○ | - | - | - | U235 enrichment |

# Fuel fabrication plant 2/2

Enrichment or Pu content ratio is adjusted for specified “infinite multiplication factor ($k_\infty$)”.

$$k_\infty = \frac{\sum_i \nu\Sigma_{f,i}\, x_i(\mathrm{fMOX}, t + aT)}{\sum_i \left(\Sigma_{c,i} + \Sigma_{f,i}\right) x_i\,(\mathrm{fMOX}, t + aT)}$$

$\nu\Sigma_{f,i}$ : microscopic fission neutron cross section of nuclide i.

$\Sigma_{c,i}$ : microscopic capture cross section of nuclide i.

$\Sigma_{f,i}$ : microscopic fission cross section of nuclide i.

$T$: Time spent in NPP (year)

$a$:one value from $\{0, 0.5, 1\}$
corresponding to initial-, mid- and end-stage of burnup

- The infinite multiplication factor is calculated from the fission neutron production cross section, fission cross section, and capture cross section, and does not include other reactions such as n,2n.
- In addition to the infinite multiplication factor adjustment, fuel fabrication can also be performed with a fixed Pu ratio.
- Another method for determining the Pu fraction is the Equivalent Fissile Method, which is not implemented in the NMB. The method determines Pu ratio so that $\sum_i \left(\nu\sigma_{f,i} - \sigma_{f,i} - \sigma_{c,i}\right) x_i = \mathrm{const.}$

# Nuclear power plant 1/2

Fresh fuel is changed to spent fuel by "burnup" handling.

fUO2=UO2 fresh fuel
sUO2=UO2 spent fuel

$$\frac{d\vec{N}(T)}{dt} = \boldsymbol{A}\vec{N}(T)$$

$$\vec{N}(T)\Big|_{T=0} = x_i(\mathrm{fUO2}, t_0)$$

$$x_i(\mathrm{sUO2}, t_1) = \vec{N}(T)\Big|_{T=t_1-t_0}$$

$$A = D + \phi\left(C_{fission} + C_{capture} + C_{n,2n} \cdots\right) \qquad \phi = \frac{P}{P_F \cdot F \cdot \vec{N}}$$

$\phi$: neutron flux、$P$: reactor power、$P_F$：vector representing thermal power per fission event、$D$：Decay matrix、$F$：fission cross section vector、$C$：matrix representing change by fission/capture/(n,2n) event

Throughput is generation capacity, GWe, or annual generated electricity (TWh/year)

$$y(\text{power plant}, t) = \mathrm{P}$$

$$y(\text{power generation}, t) = \mathrm{P} \cdot 365.25 \cdot 24 \cdot 10^{-6} \cdot \epsilon_{\mathrm{op}}$$

$\epsilon_{\mathrm{op}}$:Operation efficiency

# Nuclear power plant 2/2

Solution of burnup equation by conventional matrix exponential method (n-th degree)

$$\vec{N}(T+\Delta t)=\left(\boldsymbol{I}+\boldsymbol{A}\Delta t+\frac{1}{2!}(\boldsymbol{A}\Delta t)^2+\frac{1}{3!}(\boldsymbol{A}\Delta t)^3+\cdots\right)\vec{N}(T)$$

$$\frac{d\vec{N}(T)}{dt}=\boldsymbol{A}\vec{N}(T)$$

$$\vec{N}(T)=\overrightarrow{N_0}\exp(\boldsymbol{A}\mathrm{t})$$

$$\cong\overrightarrow{N_0}(1+\boldsymbol{A}t+\frac{1}{2}(\boldsymbol{A}T)^2+\cdots)$$

In practice, it cannot be solved this way because it is a matrix, but if $\boldsymbol{\Delta t}$ is sufficiently small, it will be a good accuracy.

Notation for each element, $i$

$$N_i^{(k+1)}=N_i^{(k)}+\frac{1}{k+1}\Delta t\sum A_{ij}N_j^{(\mathrm{k})}$$

$k$: degree {0,1,2,...}

$N_i^{(0)}$: Initial composition

$N_i^{(k)}$ : Compositon after $\Delta t$

Solution by new OEM method (Transpose, 1st degree). OEM is Okamura explicit method

$$\vec{N}(T+\Delta t)=\vec{N}(T)+\Delta\vec{t}\circ\left(\boldsymbol{A}\vec{N}(t)\right)$$

$$\widetilde{\Delta t_i}=\frac{1-\exp(-\lambda_i\Delta t)}{\lambda_i}$$

°: Element-wise product (Hadamard product)

$\lambda_i=-A_{ii}$ ,Decay constant + transmutation rate

Notation for each element, $i$

$$N_i^{(1)}=N_i+\widetilde{\Delta t_i}\sum_j A_{ij}N_i^{(0)}$$

Solution by OEM method (Transpose, 2nddegree).

Notation for each element, $i$

$$N_i^{(2)}=N_i^{(1)}+\frac{1}{2}\widetilde{\Delta t_i}\sum_{j\neq i}A_{ij}N_j^{(1)}$$

The method recommended by NMB, because it produces good accuracy at a small computational cost

# Supplement: Idea of OEM Method

■ Conventional matrix exponential method (MEM)

MEM-1st order: $X(t+\Delta t) = (I + A\Delta t)X(t)$

Ex.) If nuclide $i$ has a short half-life,

$A_{ii} = -\lambda_i$ (Large negative value)

$1 + A_{ii}\Delta t < 0$, Calculation fails.

Δt must be smaller enough than the shortest $T_{1/2}$ (9 hour).

■ Okamura explicit method (OEM)

$$\widetilde{\Delta t_i} \equiv \frac{e^{A_{ii}\Delta t}-1}{A_{ii}}, \quad \widetilde{\Delta t} = \begin{pmatrix} \widetilde{t_1} & \widetilde{t_i} & \widetilde{t_N} \\ \vdots & \vdots & \vdots \\ \widetilde{t_1} & \widetilde{t_i} & \widetilde{t_N} \\ \vdots & \vdots & \vdots \\ \widetilde{t_1} & \widetilde{t_i} & \widetilde{t_N} \end{pmatrix}$$

✓ OEM-normal

$$X(t+\Delta t) = \left(I + A \circ \widetilde{\Delta t}\right)X(0)$$

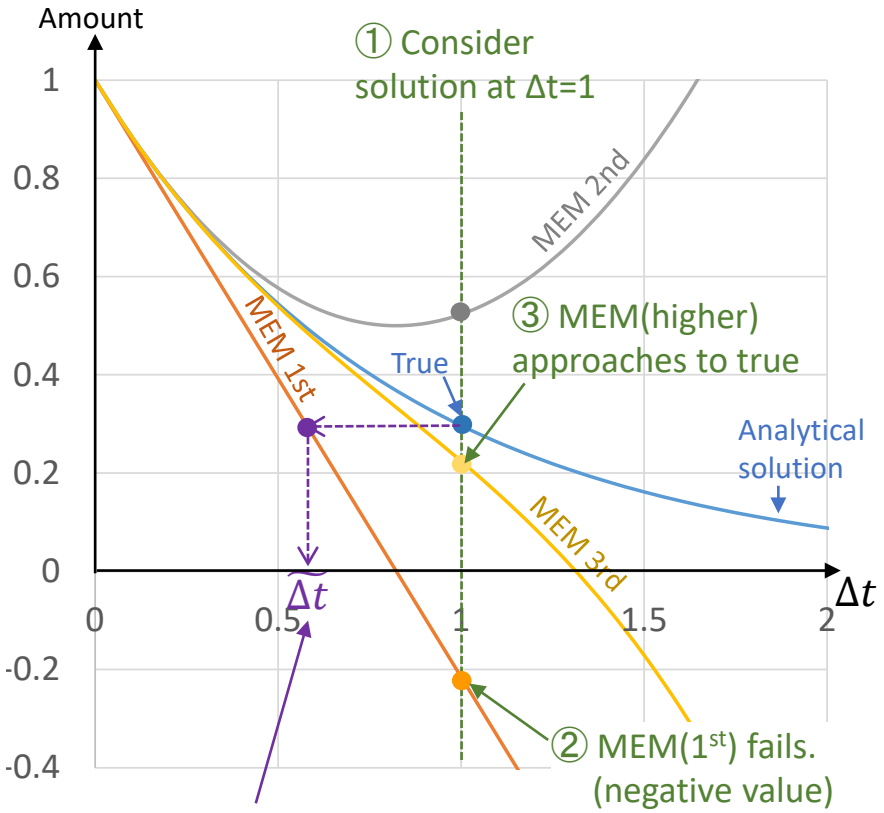
✓ OEM-transpose

$$X(t+\Delta t) = \left(I + A \circ \widetilde{\Delta t}^{\mathrm{T}}\right)X(0)$$

OEM-transpose performs better.

Δt can be very long.

Solution of single decay



# Reprocessing plant

Raw materials such as Pu are generated by "division" handling from spent fuels, and composition of whole waste is estimated at the same time.

$$x_i(\mathrm{Pu,t}) = r_i(\mathrm{Pu})\, x_i(\mathrm{sUO2,t})$$

$$x_i(\mathrm{repU,t}) = r_i(\mathrm{repU}) x_i(\mathrm{sUO2,t})$$

$$x_i(\mathrm{Waste_all,t}) = r_i(\mathrm{Waste_all})\, x_i(\mathrm{sUO2,t})$$

$$\sum_m r_i(m) = 1$$

sUO2=UO2 spent fuel
Pu＝Pu stock
repU=Reprocessed uranium
Waste_all=whole waste
m: Material tag to which fraction is distributed.

$r_i(\mathrm{Pu})$：Distribution ratio of nuclide $i$ from spent fuel to Pu stock

Throughput is defined as weight of reprocessed spent fuel (t/year)

$$y(\mathrm{reprocessing}, t) = w(\mathrm{sUO2}, t)$$

Order of reprocessing of spent fuel will be described afterword.

# Waste management plant 1/2

Whole waste distributed from reprocessing plant is "divided" to individual waste.

$$x_i(\text{Waste1}, \text{t}) = r_i(\text{Waste1})\, x_i(\text{Waste_all}, \text{t})$$

$r_i(\text{Waste1})$：Distribution ratio to waste 1

Example: LWR reprocessing produces three types of waste: noble gas waste, iodine waste, and the high-level radioactive waste (HLW).

Throughput of waste management plant expressed in terms of the accumulated number of waste generated* (number), the amount of waste stored (number), and the accumulated disposal area * (m2).

* For waste, an accumulation is often more convenient than a single year's generation or area.

# Waste management plant 2/2

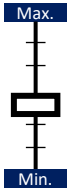
Waste management = Solidification + Storage + Disposal

Combination of solidification, storage and disposal is specified for each of waste.

Solidification

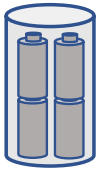
Content ratio (wt%)






Self-adjustable

Storage

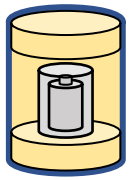


Storage period (year)




User specification

Disposal



Pitch between wastes (m)




Self-adjustable

## Possible Limitation on solidification of waste

| Item | Description | Designation |
|---|---|---|
| Weight | Oxide or element weight percentage in waste | 1 |
| Heat at solidification | Heat generation of a waste after solidification process (W/waste) | 10 |
| Maximum temperature of buffer material in underground | Maximum temperature of buffer material (℃) in repository when disposed of by specified disposal method after specified storage period | 100 |
| Heat at disposal | Heat generation of a waste after storage (W/waste) | 1,000 |
| MoO3 content | Weight percentage of molybdenum oxide | 10,000 |
| PGM content | Weight percentage of platinum group element oxides | 100,000 |

The six limits can be combined arbitrarily and the maximum waste content percentage that meets the specified limit is selected.

## Possible disposal concept

| Item | Description |
|---|---|
| PWR direct disposal | Direct disposal of PWR spent assembly. 0.125 to 4 assemblies/cask |
| BWR direct disposal | Direct disposal of BWR spent assembly. 0.125 to 12 assemblies/cask |
| Vertical emplacement in hard rock | Pitch of 1.11～28m between waste pit |
| Horizontal emplacement in hard | Pitch of 4.44～90mbetween waste tunnel |
| PEM emplacement in hard rock | Pre-fabricated EBS. EBS stands for engineered barrier system. Pitch of 9.6～90m between waste tunnel |
| Compact | Compact emplacement |

After the specified storage period, selected is the disposal method that meets the narrowest area and the limit on buffer temperature among several disposal methods. (Cannot be used concurrently with Solidification Limit destination of 100).

# Supplement: Analysis method of repository temperature

1. Heat transfer equation to be solved.

$$\sigma\rho\frac{d}{dt}u(r,t) = \kappa\nabla^2 u(r,t) + Q(r,t)$$

Temperature

Heat source

Temperature-INdependent parameters

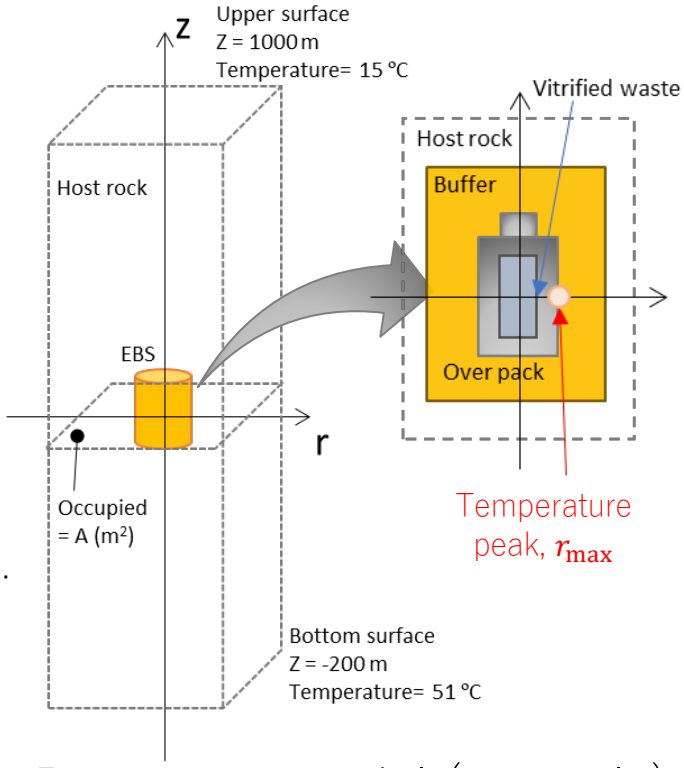
2. Source can be disintegrated by nuclide.

$$Q(r,t) = \sum_i A_i p(r) q_i(t)$$

Spatial distribution is unique.

3. Solve for each nuclide by FEM in advance.

$$\sigma\rho\frac{d}{dt}u_i(r,t) = \kappa\nabla^2 u_i(r,t) + p(r)q_i(t)$$




Repository model (example)

4. Sum up in NMB and get exact solution ☺

$$u(r,t) = \sum_i A_i u_i(r,t) \quad \text{or} \quad u(r_{max},t) = \sum_i A_i u_i(r_{max},t)$$

Inventory in waste (Watt)

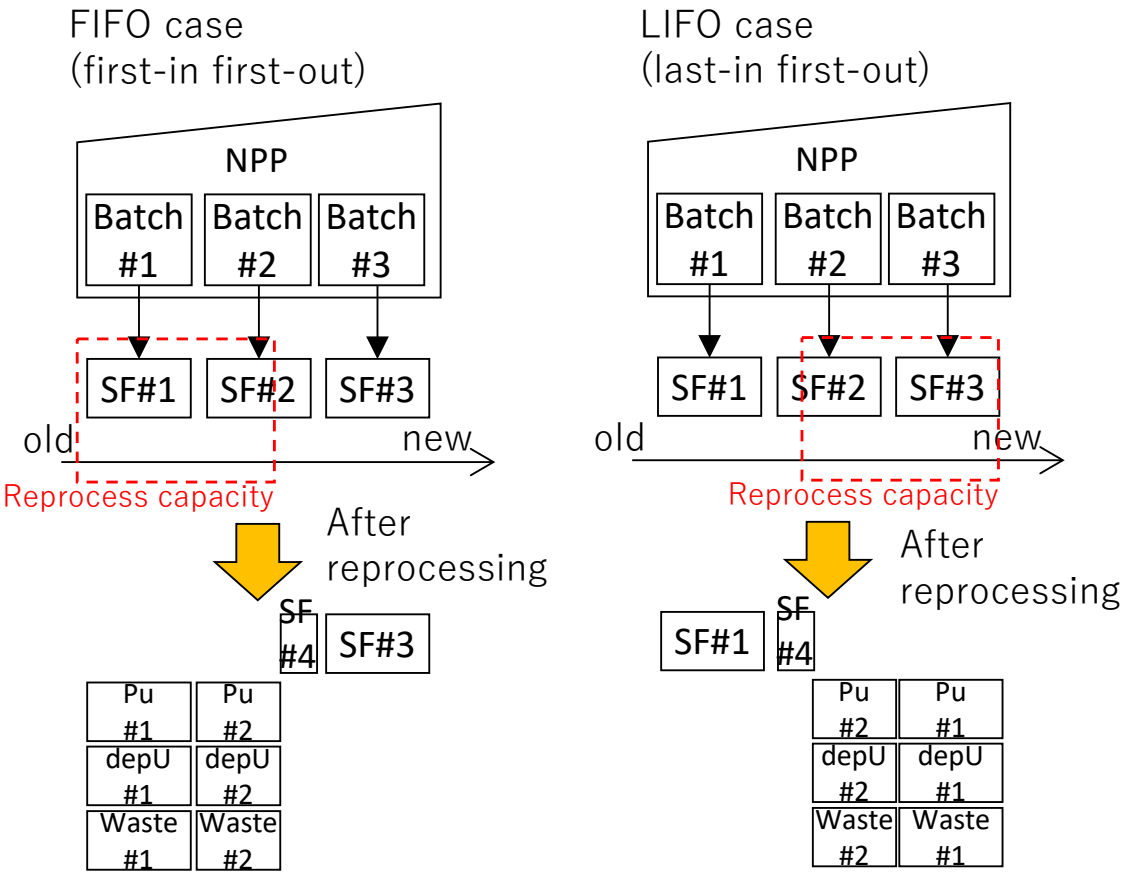
in database(K/Watt)

NMB Methodology

# Others

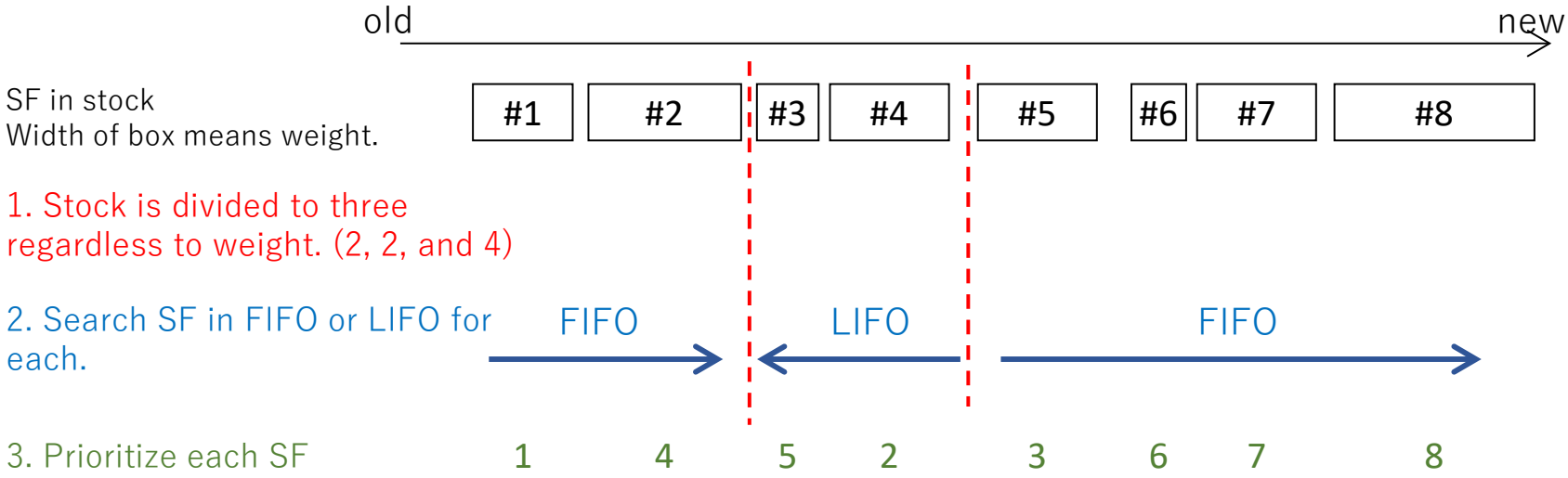
# Search of spent fuel and raw material 1/3

- The NMB “explicitly*” maintains information on each batch of spent fuel generated by an individual NPP.
- Individual spent fuels can be arranged in chronological order of discharge and reprocessed in order of oldest or newest.




* Another approach is to mix and store all the spent fuel present at the time in a “spent fuel buffer“ and take part of it. In such a case, the search method such as FIFO or LIFO cannot be used, and "average spent fuel” is always selected.

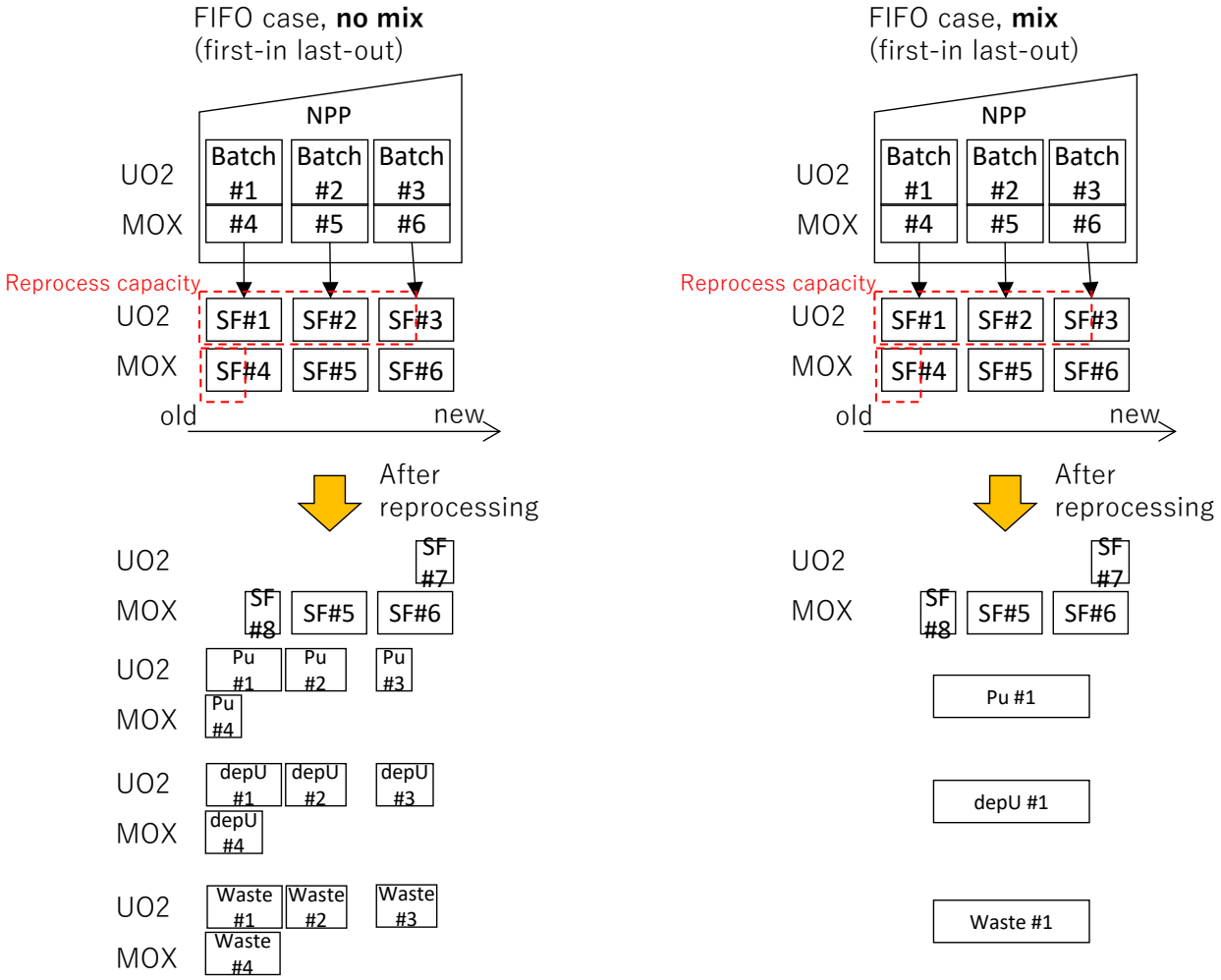
# Search of spent fuel and raw material 2/3




Search by "stock split" (in the case of a three-split)

- By stock splitting various search can be implemented: new and old fuels is alternately taken, or medium age fuels is taken first.
- When the number of divisions is sufficiently large, the average number of spent fuels of each age is selected.
- When the number of divisions is 1, it is the same as a normal LIFO/FIFO.

# Search of spent fuel and raw material 3/3



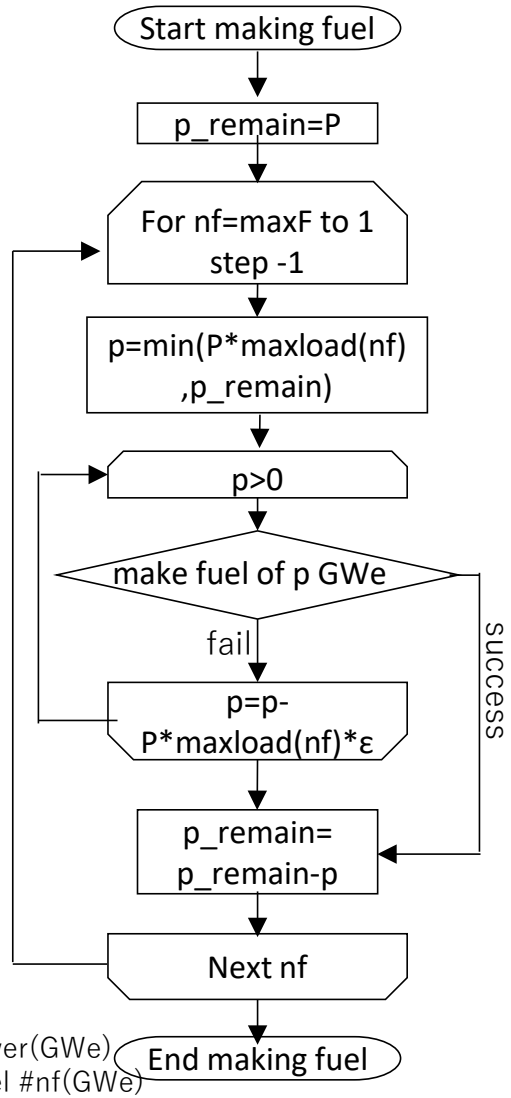

- In a real-world reprocessing process, the fuel assemblies may be mixed to some extent to create uniform Pu and waste.
- For example, LWR MOX fuel will be mixed with uranium spent fuel.
- To use this function, set [MixSpentFuel] to TRUE in the reprocessing plant designation.
- Spent fuels in only "1 year" period can be averaged.

One-year averaging at reprocessing plants

# Combination of Reactor and Fuel

- The NMB defines a nuclear power plant (NPP) as a combination of reactor and fuel, each of which is defined as a “type”. (e.g., Both of UO2 fuel and MOX fuel can be combined to PWR)
- The flowchart shows the case of loading maxF types of fuel in one reactor of 1GWe.
- As example, the fuel types are UO2 fuel and MOX fuel (maxF = 2 types) . Number of UO2 fuel is nf=1 and that of MOX fuel is nf=2.
- Maximum load of MOX fuel, maxload(2), is 0.33.
- On the other hand, the maximum loading ratio of UO2 fuel maxload(1) is not 0.67, but maxload(1)=1.0, because MOX fuel loading does not always take the maximum value depending on the plutonium supply, and the UO2 fuel must make up.
- At first, MOX fuel fabrication is attempt at P* maxload(2) GWe. If the attempt failed due to shortage of Pu, the next is attemped at P* maxload(2)*(1- ε) Gwe. If it failed again, the next is at P* maxload(2)*(1- 2ε) Gwe.
- If it succeeds or fails to the end, UO2 fuel (nf=1) is produced.

Flowchart: Multiple fuel loadingg to a reactor




- nf: fuel number
- maxF: number of fuel
- maxload(nf): maximum loading ratio of fuel #nf

- P: Reactor power(GWe)
- p: Power of fuel #nf(GWe)
- p_remain: Power of remaining fuel(GWe)
- ε: Percentage reduction in power when raw materials are in short supply (~0.05)

# Blanket of fast breeding reactor

fMOX=MOX fresh fuel
fBlk=Blanket fresh fuel
sUO2=MOX spent fuel
sBlk=Blanket spent fuel

Definition of breeding ratio

$$BR(t) = \frac{\sum_{\mathrm{fiss}}\{x_i(\mathrm{sMOX}, \mathrm{t}+T) + x_i(\mathrm{sBlk}, \mathrm{t}+T)\}}{\sum_{\mathrm{fiss}}\{x_i(\mathrm{fMOX}, \mathrm{t}) + x_i(\mathrm{fBlk}, \mathrm{t})\}}$$

"fiss" means fissile nuclides, i.e., U233, U235, Pu239, Np239, Pu241 in NMB

Method 1: The user specifies a breeding ratio, and the NMB internally calculates the blanket weight required to achieve that breeding ratio.

$$w_{\mathrm{fBlk}}(t) = -\frac{\mathrm{BR}(t)\sum_{fiss} x_i(\mathrm{fMOX}, \mathrm{t}) - \sum_{fiss} x_i(\mathrm{sMOX}, \mathrm{t})}{\mathrm{BR}(t)\sum_{fiss} a_i(\mathrm{fBlk}, \mathrm{t}) - \sum_{fiss} a_i(\mathrm{sBlk}, \mathrm{t})}$$

- $a_i$ means weight ratio. $w_{\mathrm{fBlk}}$ is weight of blanket fuel.
- $a_i(\mathrm{sBlk}, \mathrm{t})$ is calculated using predefined neutron flux at blanket region. Neutron flux at the blanket actually varies with blanket amount, but is approximated to be constant regardless of blanket fuel weight.

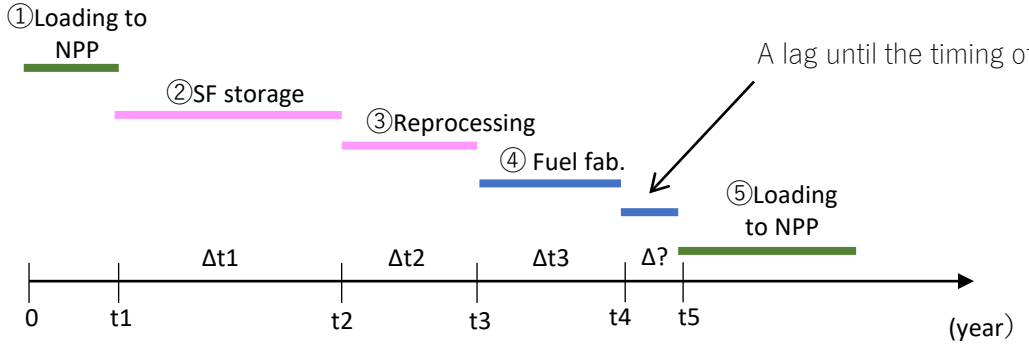
Method 2: The user specifies the blanket amount, the specific power of the blanket section and the power ratio of the blanket to the total.

$$w_{\mathrm{fBlk}}(\mathrm{t}) = \frac{(\mathrm{Generation\ capacity, GWe})(\mathrm{Power\ ration\ in\ blanket})}{(\mathrm{Thermal\ efficiency})\ (\mathrm{Specific\ power, MW\,t/t})}$$

- The breeding ratio depends on the composition of the MOX fresh fuel

# Detail timing of discharge-reprocess-fabrication-reload process

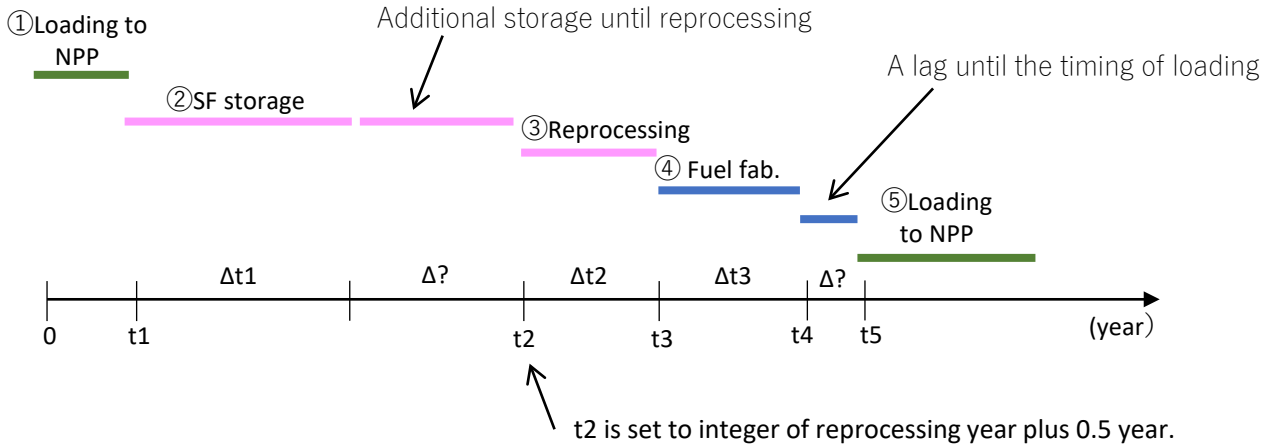
In the case that reprocessing capacity is sufficient and the SF storage period equals the minimum SF storage period required before reprocessing:.




Δt1~ Δt3 is specified by user:
Δt1 : minimum SF storage required before reprocessing
Δt2 : Time required for reprocessing
Δt3 : Time required for fuel fabrication

- Treated as fresh fuel in inventory estimation
- Treated as spent fuel
- Treated as raw material (such as Pu stock)

In the case that the SF storage period is longer than the minimum SF storage period:.



# Accelerator driven system (ADS)

- Since the fresh fuel composition is designed to vary with the number of irradiation in ADS for MA transmutation, the NMB provides a designation for ADS.
- At the first-loading, about 30~40% of Pu is added to adjust to an appropriate subcriticality (about 0.95~0.98), because pure MA is not critical enough.
- Pu is added only to the first-loading fuel; in subsequent fuel fabrication, only MA is added, as the criticality is compensated by fissile nuclide generated by the capture reaction of MA. It also contains a small amount of uranium generated during the transmutation process.
- At NMB, Pu enrichment is adjusted only for the first loading fuel using infinite multiplication factor. For subsequent fuels, the specific power is used to determine the required heavy metal weight, and the missing MA is supplied from the MA stock.

ADS fuel cycle

