# Nuclear Fuel Cycle Simulator

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Team NMB

2024/12/3-4

# Content:

| Date | | |
|---|---|---|
| 12/3 13:10-14:20 | Lecture | 1. NFC simulator<br>2. NMB |
| 14:30-15:40 | Practice | Start up of NMB (2h) |
| 15:50-17:00 | Lecture | Toward Triple scenario in your country |
| 12/4 9:20-10:30 | Work | |
| 10:40-11:50 | Work | |
| 13:10-14:20 | Work | |
| 14:30-15:40 | Presentation | 20min x 7 ( 2.5h) |
| 15:50-17:00 | Presentation | |

NFC: Nuclear Fuel Cycle
NMB: Nuclear Material Balance

Goal: Make your own NFC scenario

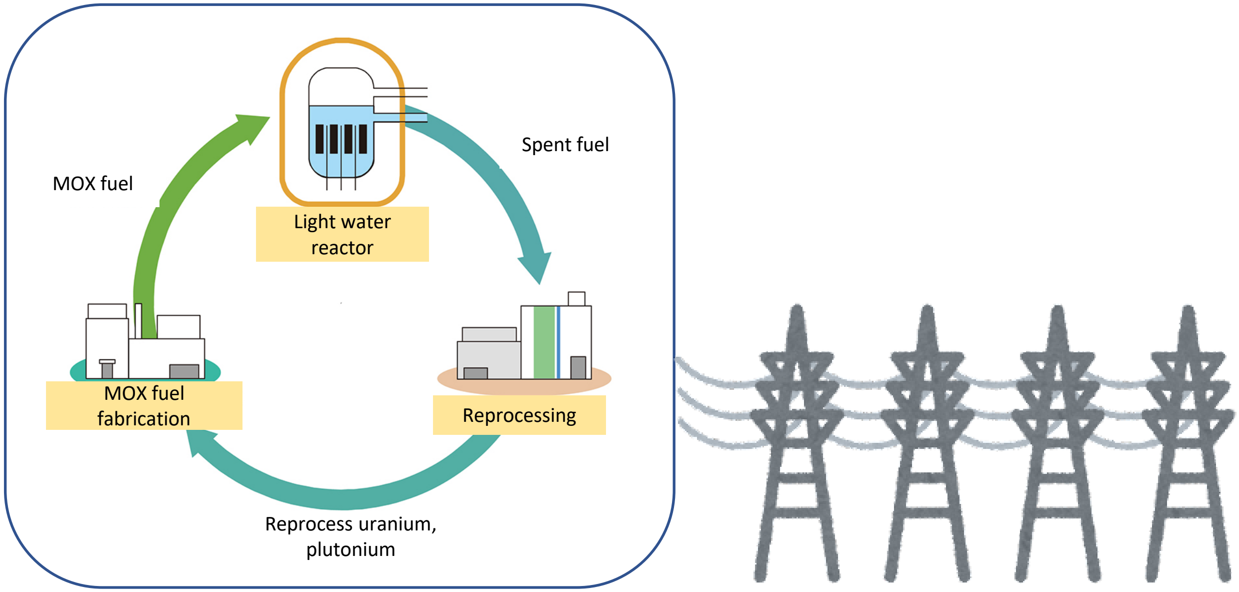
# Nuclear Fuel Cycle Simulator (NFC simulator)

# Purpose of NFC simulator

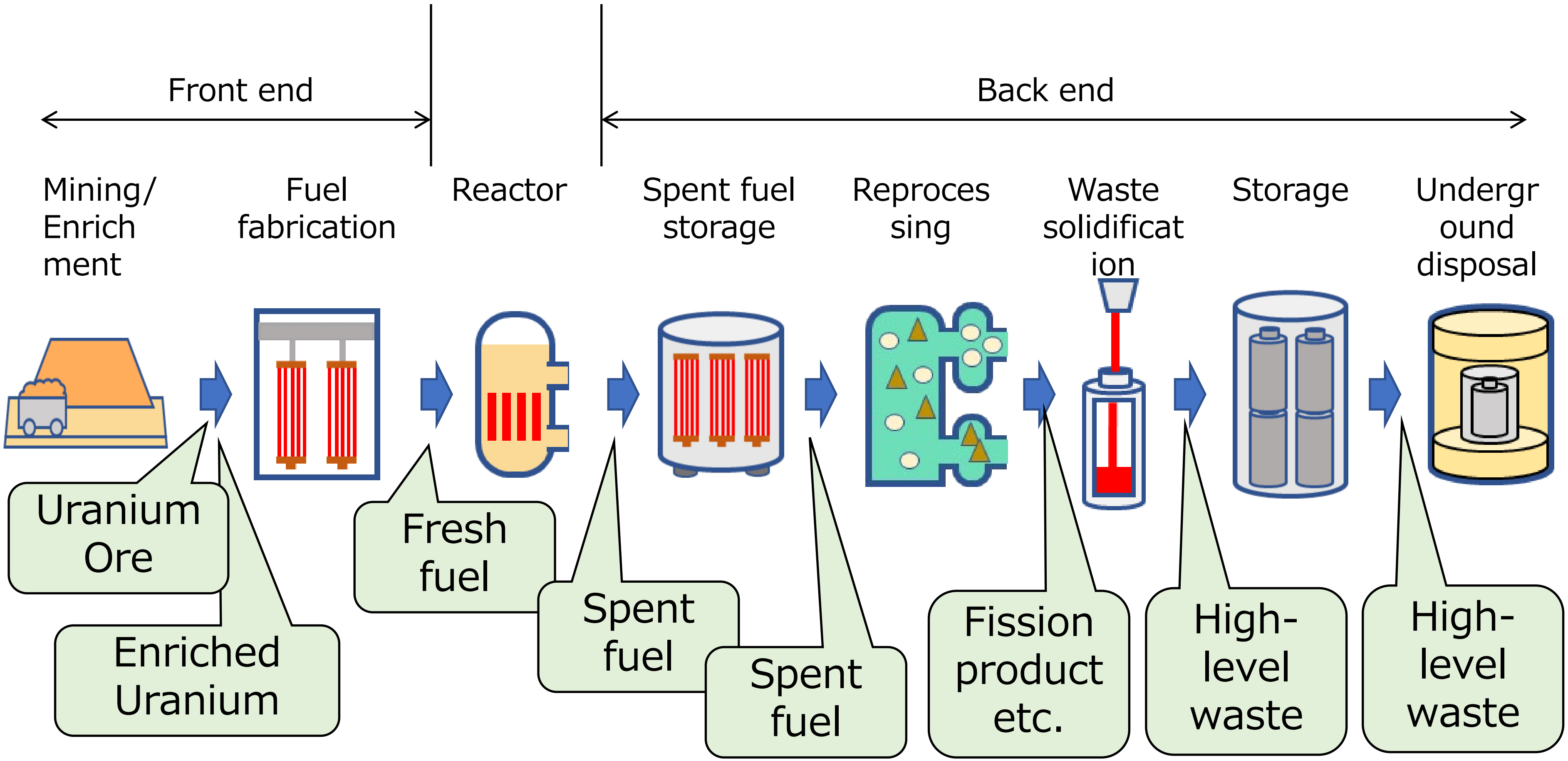
**To <u>quantify</u> <u>material flow</u> inside nuclear power industry**

- ✓ Future strategy of facility deployment by government or utilities
- ✓ R&D direction

Nuclear power industry

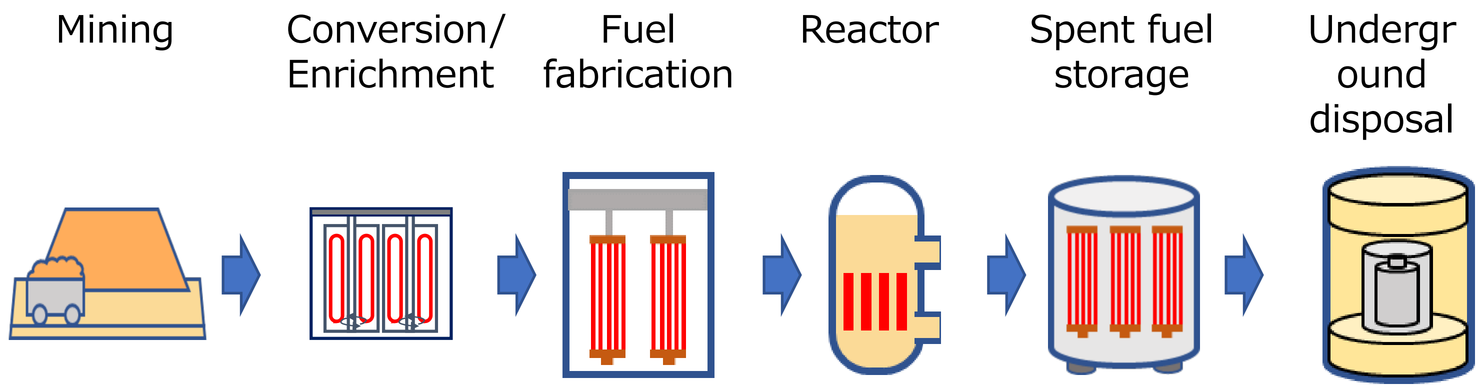


# Material flow




Material flow determines scale of each facility.

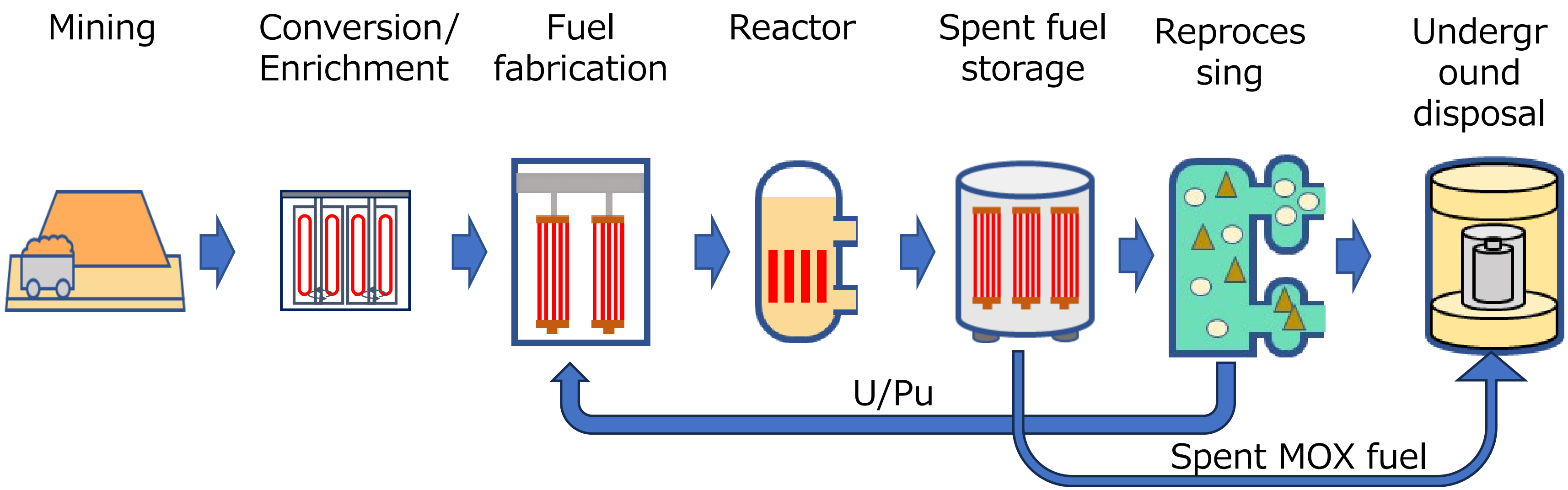
# Once through fuel cycle




If you want to operate a nuclear power plant (light water reactor), you have to know:

- How much uranium ore do I need?
- How many enrichment plants and fuel fabrication plants do I need?
- How much spent fuel will the reactor produce?
- How long will they be stored in the repository?
- What is the size of the repository?

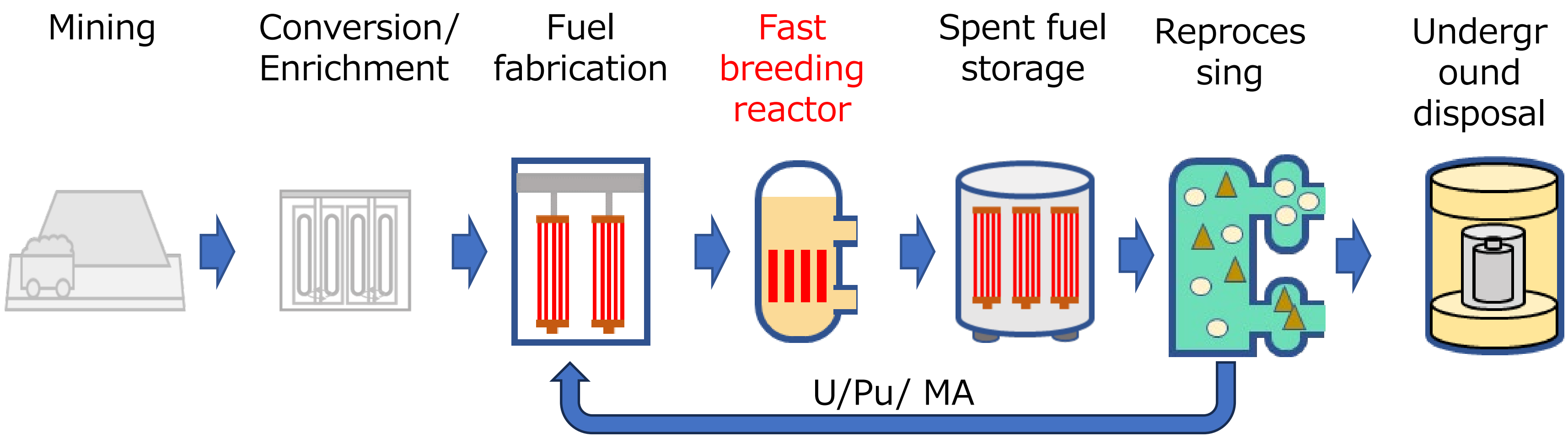
# Partially closed fuel fuel cycle




If you want to reduce necessary resource (uranium) and waste volume, you will proceed to the reprocessing technology.

- How much uranium can be saved?
- How much the high-level waste will be reduced? (
- Do not forget spent MOX fuel (plutonium fuel) is produced and can not be recycled any more in light water reactor technology.

# Closed fuel fuel cycle



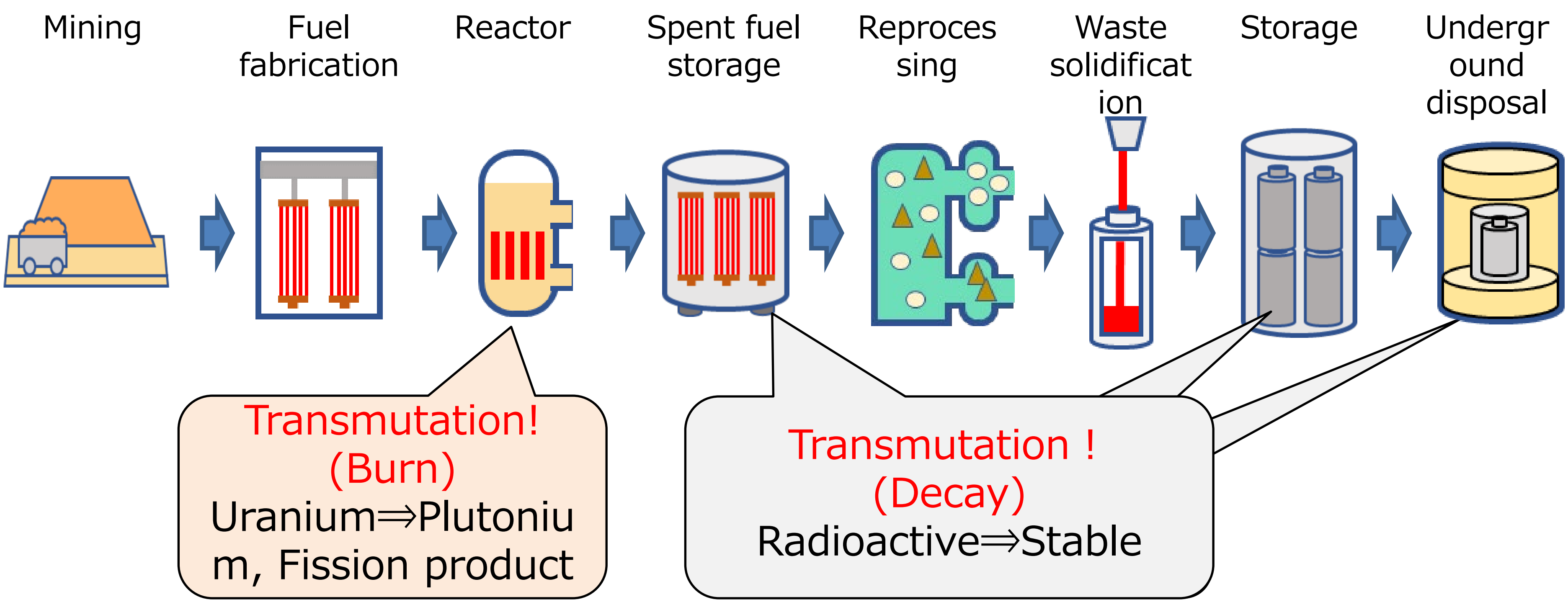

If you want to free from natural uranium resources, you will introduce fast breeding reactor which can accept reprocessed plutonium.

- In the beginning, FBR needs certain amount of plutonium from LWR.
- How long does it take to transit from LWR to FBR?

『 Nuclear power generation is an industry that extracts energy through various processes, involving the transmutation of nuclides. 』

Key 1

Key 2



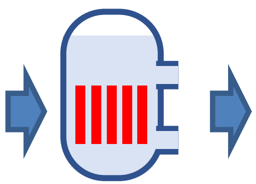
# Key point 1 :　Transmutation in nuclear reactors

- There is a wide variety of nuclide transmutation in the reactor, in particular, fuels using plutonium.
- The number of nuclides produced in a reactor exceeds 1000. Calculation time is very long when dealing with a large number of reactors and fuels.
- Inside the NFC simulator, improvement has been done to perform reasonable burnup calculations at high speeds.
  - Limit the number of nuclides ⇔ Limit the range of use of simulator results
  - Include approximations in calculation methods ⇔ Poor accuracy

Uranium fresh fuel

| | |
|---|---|
| U235 | 3% |
| U238 | 97% |

Only U235 enrichment varies with reactor type.



LWR

Uranium spent fuel

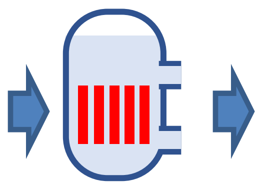
| | |
|---|---|
| U235 | 1% |
| U238 | 96% |
| Pu239 | 0.4% |
| … | … |

The composition is somewhat fixed. (can be calculated in advance)

Plu-thermal fresh fuel

| | |
|---|---|
| U238 | 92% |
| Pu239 | 4% |
| Pu240 | 2% |
| … | … |

The composition varies significantly depending on the isotopic composition of Pu.



LWR

Plu-thermal spent fuel

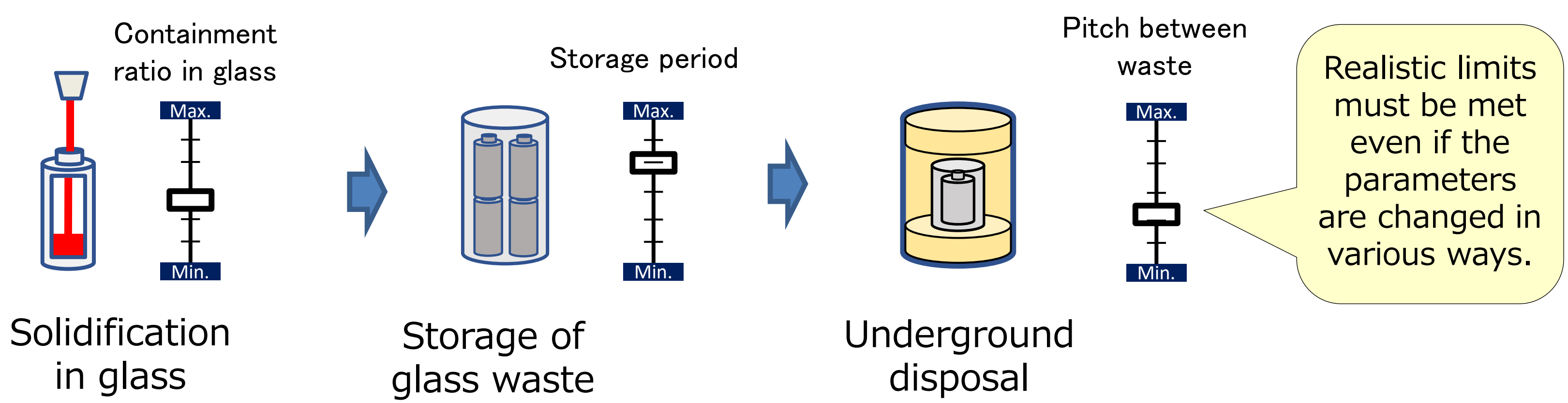
| | |
|---|---|
| U238 | 91% |
| Pu239 | 2% |
| Pu240 | 2% |
| ・・・ | ・・・ |

Composition cannot be predicted in advance. (calculations are required on a case-by-case basis).

Plu-thermal: mixed-oxide fuel for light water reactor

# Key point 2:　Modeling of various processes

- The nuclear fuel cycle other than nuclear reactors needs to be formulated.
- The front end (mining - conversion - enrichment - fuel fabrication) of uranium is in practical use and can be easily modeled.
- It is difficult to model the back end, where various options remain.
  - Which elements should be turned into which waste in reprocessing? How much can be put into vitrified waste?
  - How many years of cooling should the vitrified waste be allowed before disposal?
  - What layout will the waste be buried in when it is disposed of? (The repository has a temperature limit.)
  - If the solidification and disposal methods are addressed in a predetermined manner, as many simulators do, the method may not be feasible.



# Key point 3　Target user

| Target user | Elementary | Advanced |
|---|---|---|
| Scope/<br>Functions | Narrow | Wide |
| Computation<br>system | General purpose | Special |
| Making input | Easy | Difficult |
| Calculation cost | Low | High |
| Maintenance | Easy | Difficult |

Balance of functionality, ease of use, and maintainability is important.

# Key point 4　Open/Close

| | Close (private) | Open |
|---|---|---|
| Development efficiency/ structure | ○　Conducted intensively by an organization | ×　Voluntary conducted |
| Retention of Know-how | ○　The source can be kept secret | ×　Anyone in the world can see |
| Support staff | △　Limited to related parties. | △　Voluntary conducted |
| Number of Users | ×　Limited to related parties. | ○　Anyone |
| Social communication | ×　Information imbalance | ○　Can be discussed on a common platform |

Most of simulators in the world are closed.

Ideally, the simulator should be open source while non-profit universities and national research institutes retains core development capabilities.

# NFC Simulators in the world

Famous ones

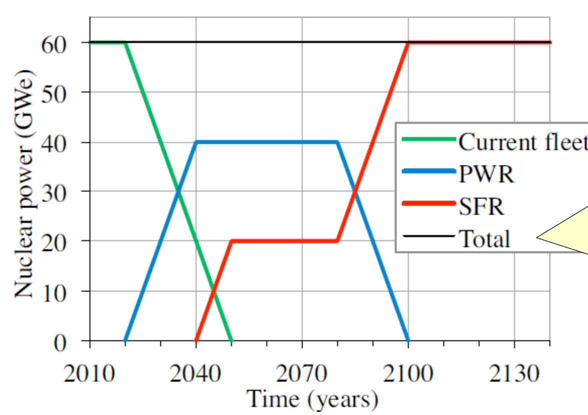
Oldest and Largest

# COSI 7

CLOSED

France
**CEA**

- 30 years of development
- Various burnup calculations by AI prediction, cycle optimization function, uncertainty assessment, economics, strategic substance (RE, In...) requirements, etc.
- Because of overly complex, cleanup from JAVA to C++ and creation of interface for policy makers in COSI7.
- CEA is the Mecca of NFC simulation. Many students conduct scenario studies and it has the largest number of users.




Typical assumption of COSI input

Specialized to MOX in LWR

# CLASS


Supported by IRSN (Regulator)

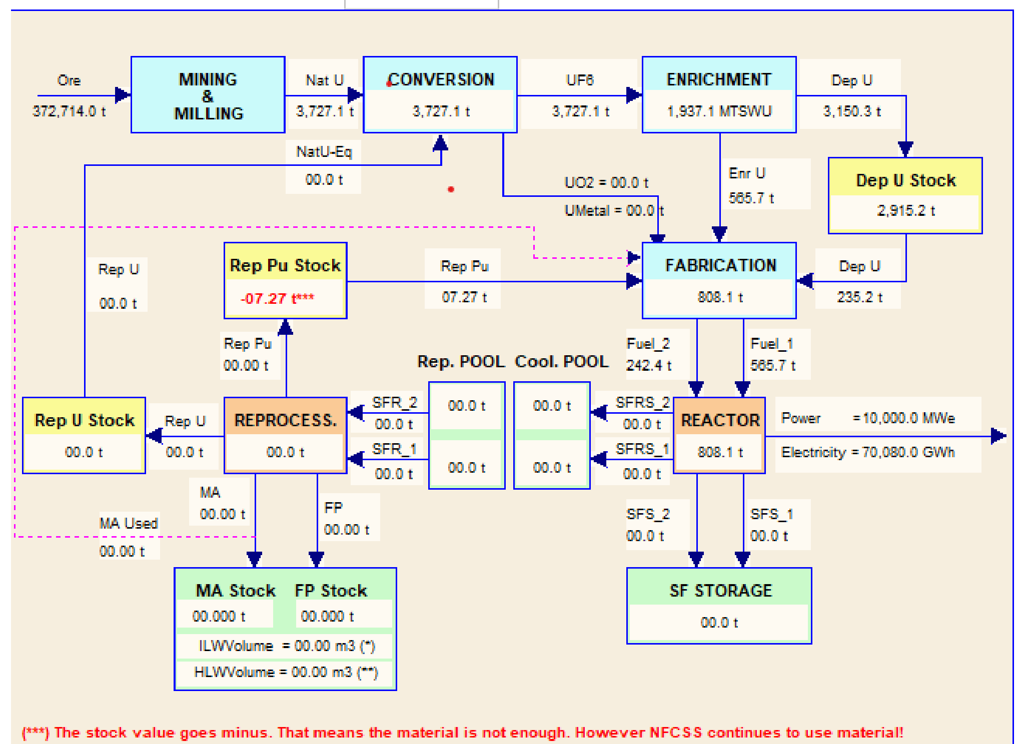
Funded by regulator

- 2012~Development by C++
- To improve the accuracy of composition determination and burnup calculation methods for new MOX fuels in LWR, a huge database of burnup calculations for various Pu loadings in PWR is maintained.

# For elementary users
# NFCSS


- IAEA is gathering experts from various countries for maintenance.
- Easy calculations on the web.
- The types of scenarios that can be analyzed are currently insufficient.

Reactor Type :PWR1
Select a year to see the result for! 2021




(*) The factor to calculate LLW Volume is 0.35 $m^3$/tonnes of HM.

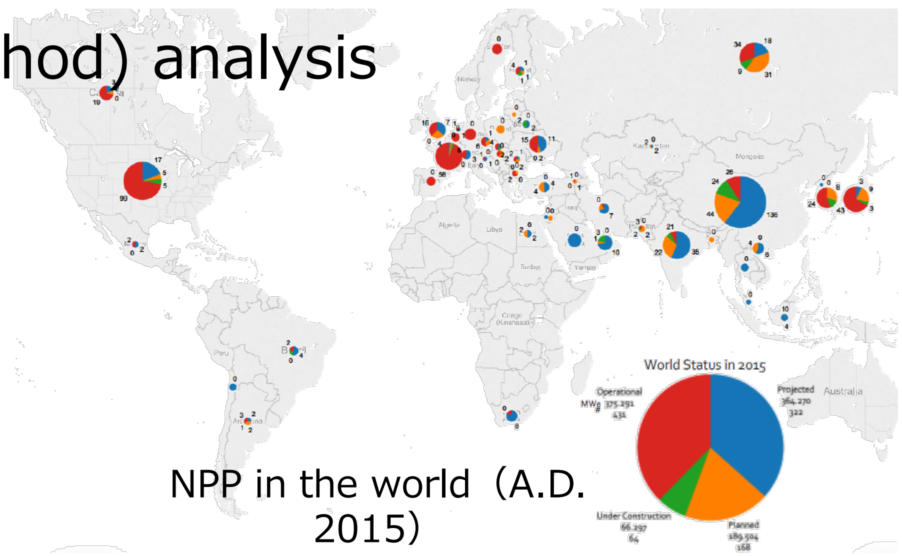
(**) The factor to calculate HLW Volume is 0.15 $m^3$/tonnes of HM.

Practical rather than accurate

# DANESS

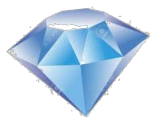

- 111 nuclides and a usual burnup calculation method, but...
- Abundant conversion of calculation results into evaluation indices (power generation cost, INPRO: an international index)
- Database covering existing nuclear power plants around the world
- Real options (investment method) analysis

Real Option Analysis: Analyzes how option values change depending on when an investment decision is made. ⇒Assists in decision making




NPP in the world（A.D. 2015）

On general purpose platforms
for system dynamics analysis

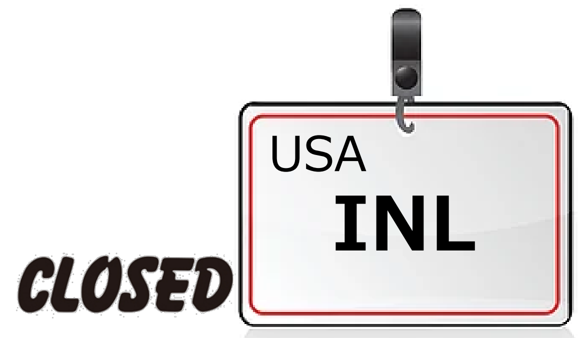


# DYMOND

Platform : AnyLogic


# VISION

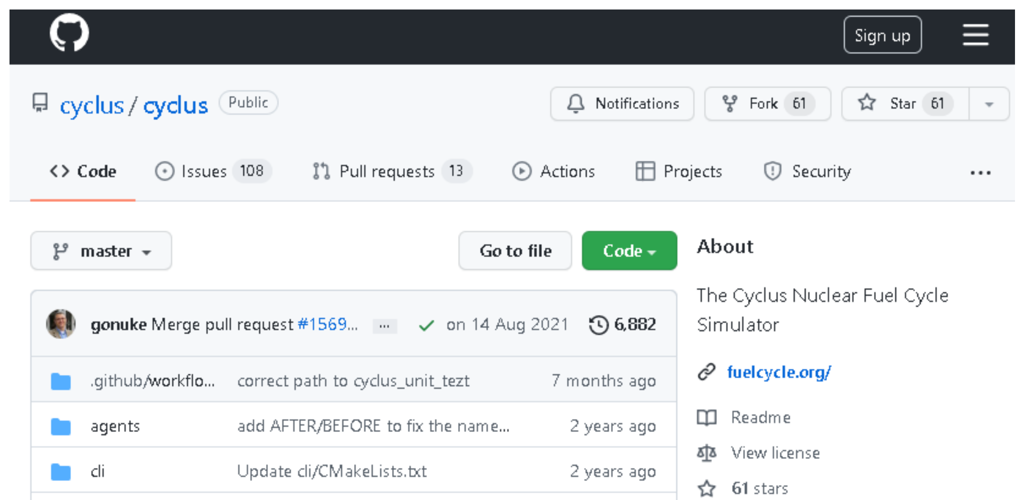
Platform : Powersim




System dynamics: A methodology that attempts to elaborate the dynamic characteristics of a target system by modeling the internal structure of a system that changes over time (dynamic system) and simulating its behavior. It is suitable for creating simulation models of social systems (business, policy, etc.).

# Collaborative Development with Git


- https://github.com/cyclus
- You can write input in XML, JSON, and Python. Developed in modules.
- Low functionality, e.g., no burnup calculation capability, fixed spent fuel composition, etc.

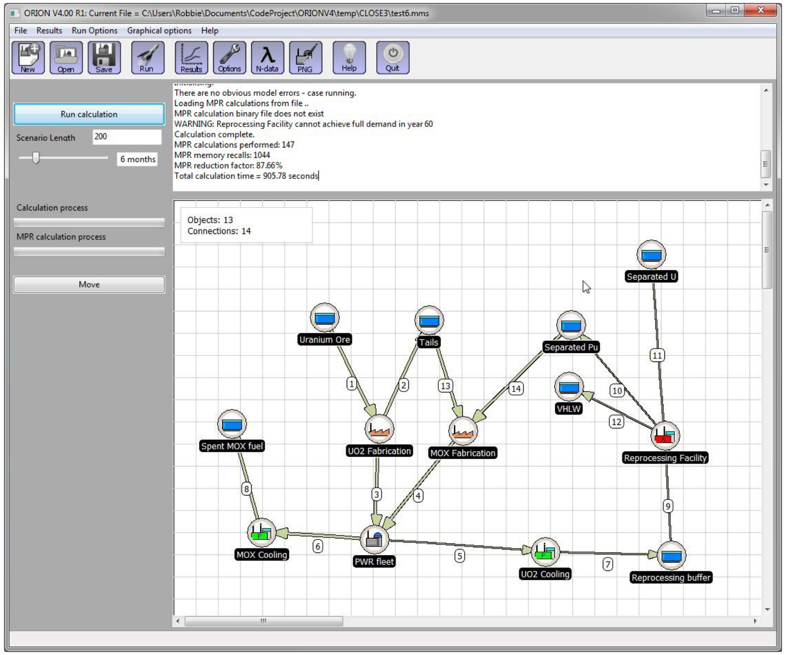



**Git repository**

Picking with a GUI is sure to be fun!

# ORION

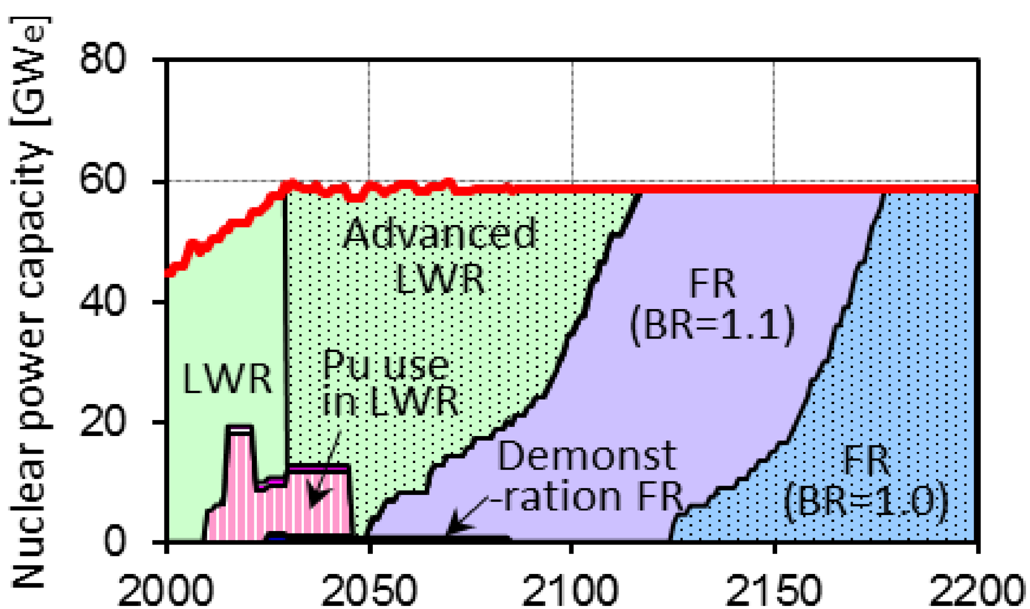

- Used for scenario analysis in UK and USA
- Can handle 2552 nuclides.
- Enhanced economic evaluation



Extensive experience in FBR

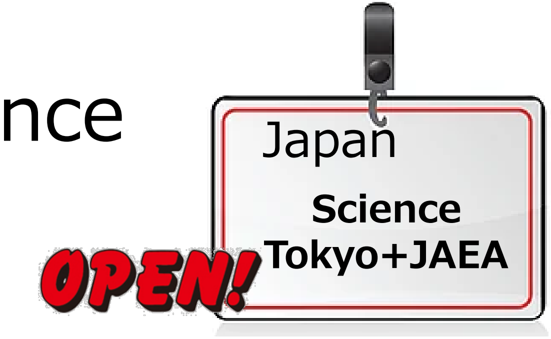
# FAMILY-21


- Long experience in analyzing fast reactor deployment scenarios in Japan.
- Active in policy discussions on fast reactor development strategies



# Easy to use but high performance

# NMB4.0




- Everyone loves EXCEL®: No preparation required. Open and use it right away.
- Fast burnup calculations for a quick and easy analysis: 1 minute to 1 hour max.
- The only repository heat transfer model in the world.
- One-sheet I/O: One scenario, one sheet.
- Most of the input data can be used with default values; just overwrite the I/O sheet with the data you want to change.

# Simulators in the world

| 名前 | COSI 7 | CLASS | DANESS | DY MOND | VISION | Cyclus | ORION | NFCSS | FAMILY-21 | NMB4.0 |
|---|---|---|---|---|---|---|---|---|---|---|
| Organization | CEA | CNRS | Nuclear 21 | ANL | INL | 大学連合 | NNL | IAEA | JAEA | Scie. Tokyo, JAEA |
| Country | France | France | | USA | USA | USA | UK | | Japan | Japan |
| Language | C++ | C++ | | Any Logic | Power sim | C++ | | JAVA | VBA | VBA |
| #nuclides | ～300 | | 111 | | 81 | | 2552 | 18 | 20 | 179 |
| Open? | | Yes | | | | Yes | | Yes | | Yes |

Others are:
TR-EVOL（Spain）、ANICCA（Belgium）、GENIUS（US）、NFCSim（US）、・・・

# Future of NFC simulator

- The World is moving toward nuclear again; many SMR is proposed, but will it be constructed?
- Dialogue with society is essential in today's mature democracy.
- Evaluation studies with NFC simulator to show advantages of new technology will become a essential part of R&D for all research area of the nuclear power industry.

- What NFC simulators must be developed in the future
  - Modeling using basic equations ⇒ Flexibility to adapt to new technologies
  - Ease of use and sophistication ⇒ Many users can use the tool and imagine the future
  - Conversion from physical quantity to indicators (economic efficiency, etc.) ⇒ Announce results to society

# NMB 4

(Nuclear Material Balance)

# Vision of NMB

1. To estimate materials in whole nuclear power industry
2. To derive indexes; economics, environment impact, safety, non-proliferation and security.
3. To contribute to formulate development and implementation strategies for nuclear fuel cycle

We are on stage 1.

# Feature of NMB 4

- ✓ Massive treatment of nuclide; 26 actinides and 153 FPs.
- ✓ Versatile for many kinds of nuclear fuel cycle including disposal.
- ✓ All in EXCEL®
- ✓ Open source (March, 2022)

# Methodology of NMB 4.1

# 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)
  - Verification by NEA Benchmark

# NMB Methodology

# Main

# Material tag

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

$$x_i(m,t)$$
$$i\text{: isotope}, m\text{: material tag}, t\text{: year}$$

Five kind of material tag

1. Raw material
2. Manufactured fuel
3. Fresh fuel
4. Spent fuel
5. Waste

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

| Change of material tag | Example of tag |
|---|---|
| Raw | natU (natural uranium) |
| Fabrication ↓ | |
| Manufac-tured fuel | mUO2 (manufactured UO2 fuel) |
| Loading ↓ | |
| Fresh fuel | fUO2 (fresh UO2 fuel in LWR) |
| Irradiation ↓ | |
| Spent fuel | sUO2 (spent UO2 fuel unloaded from LWR) |
| Waste management ↓ | |
| Waste | Glass (vitrified high level wate) |

Reprocessing

# 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

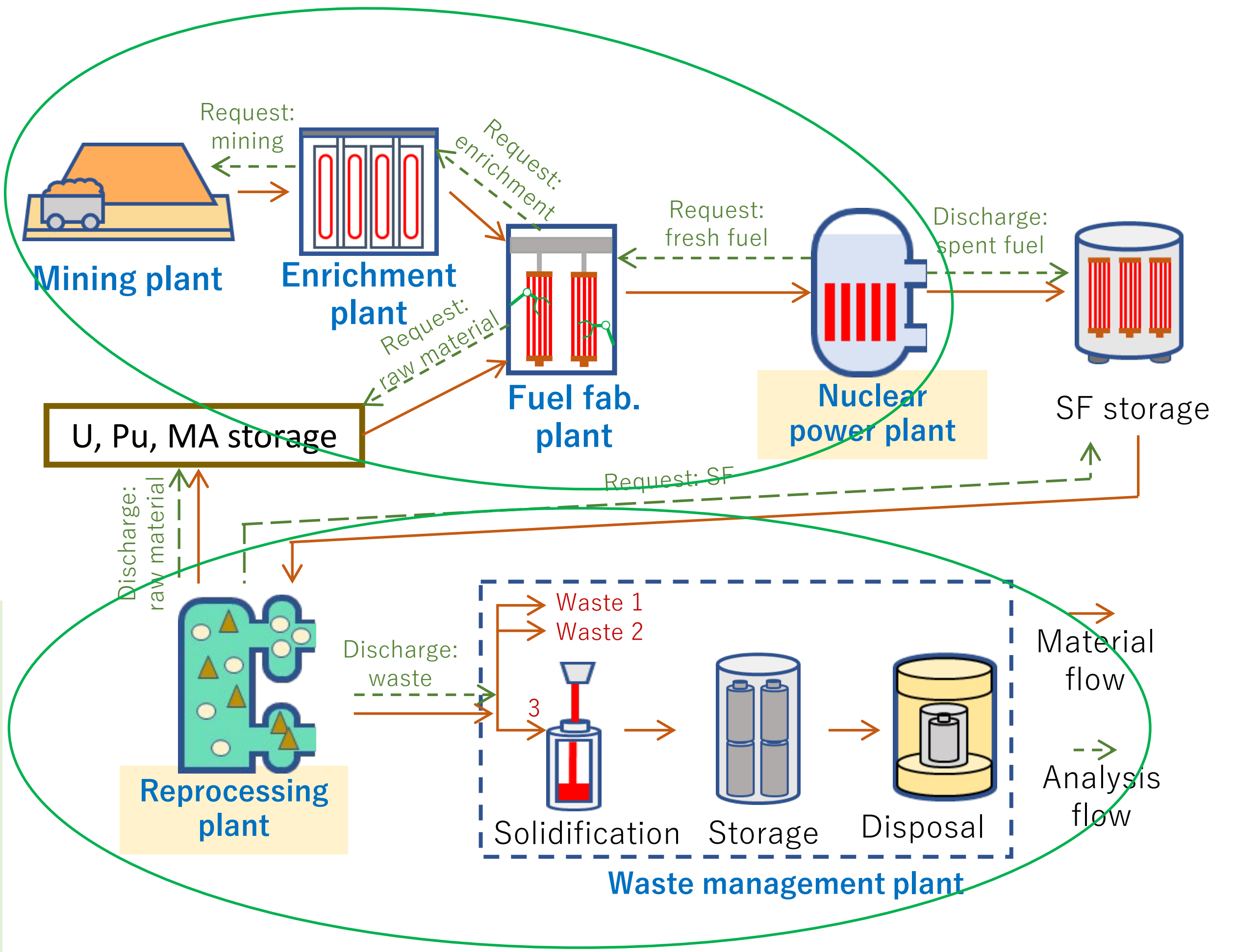
| | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| H3 | Rb85 | Zr91 | Mo95 | Ru103 | Pd108 | Cd114 | Sn122 | Te127m | Xe132 | Ba134 | Pr141 | Pm146 | Sm154 | Gd156 | Ho166m |
| Se78 | Rb87 | Zr92 | Mo96 | Ru104 | Pd110 | Cd116 | Sn124 | Te128 | Xe133 | 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.




The analysis starts from the reactor plant and the reprocessing plant, which are called as "master plant," while others are "slave plant." User is required to define nuclear power plant capacity at least.

# Definition of Plant throughput

There are eight types of NMB plants. For each, one or several throughputs is defined for each year.

Eight plants in NMB4.0

| Plant | Deta search string | Input amount |
|---|---|---|
| Mining pant | [Mining[t/y]] | The annual amount of mining（t/y） |
| Enrichment plant | [Enrichment[SWU/y]] | The annual amount of enrichment (SWUt/y) |
| Fuel fabrication plant | [FuelFabriation[t/y]] | The annual amount of fuel fabrication (tHM/y) |
| Reactor plant | [NewPlant1], etc. | Constructed power capacity (GWe), |
| Storage plant | [SF_Storage[t]] | Storage capacity (t) |
| Transport plant | [SF_Transport[t/y]] | The annual amount of transport of SF (tHM/y) |
| Reprocessing plant | [ReprocessingCapacity[t/y]] | The annual amount of reprocessed SF(t/y) |
| Waste management plant | NA | No limit |

Red = Master plant, White = Slave plant

Master plant does not work if user do not define. Slave plant works even if user do not define according to requirement of master plants.

# 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

Exercise: Calculate necessary natural uranium to obtain 1 ton of enriched uranium. Enrichment of natural, enriched and depleted uranium is 0.7%, 4% and 0.2%.

Answer:

$$w_{\mathrm{natU}} = \frac{e_{\mathrm{enrU}} - e_{\mathrm{depU}}}{e_{\mathrm{natU}} - e_{\mathrm{depU}}} w_{\mathrm{enrU}} = \frac{0.04 - 0.002}{0.007 - 0.002} = \frac{38}{5} = 7.6\ (ton)$$

# Enrichment plant

enrU=enriched uranium

depU=depleted uranium

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

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

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.

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

$$y(\mathrm{enrichment\ SWU}, t) = w_{\mathrm{enrU}}V(e_{enrU}) + w_{\mathrm{depU}}V\left(e_{\mathrm{depU}}\right) - w_{\mathrm{natU}}V(e_{\mathrm{natU}}),$$
$$V(e) = (1 - 2e)\ln\frac{1 - e}{e}$$

Tips: Energy necessary for enrichment is proportional to SWU.

# 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 of raw materials.

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

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

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 ($k_\infty$)".

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

$v\Sigma_{f,i}$ : 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 $\sum_i \left(v\sigma_{f,i} - \sigma_{f,i} - \sigma_{c,i}\right) x_i = \mathrm{const.}$

# Nuclear power plant 1/2

Fresh fuel is changed to spent fuel by “burnup” handling.

fUO2=UO2 fresh fuel
sUO2=UO2 spent fuel

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

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

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

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

$\phi$: 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

$$\frac{d\vec{N}(T)}{dt} = \boldsymbol{A}\vec{N}(T)$$
$$\vec{N}(T) = \overrightarrow{N_0}\exp(\boldsymbol{A}\mathrm{t})$$
$$\cong \overrightarrow{N_0}(1 + \boldsymbol{A}t + \frac{1}{2}(\boldsymbol{A}T)^2 + \cdots)$$

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

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

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

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

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

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

Notation for each element, $i$ $$N_i^{(1)} = N_i + \widetilde{\Delta t_i}\sum_j A_{ij} N_i^{(0)}$$

$\circ$: Element-wise product (Hadamard product)

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

Solution by OEM method (Transpose, 2nd degree).

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

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

# 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)

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

✓ OEM-normal

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

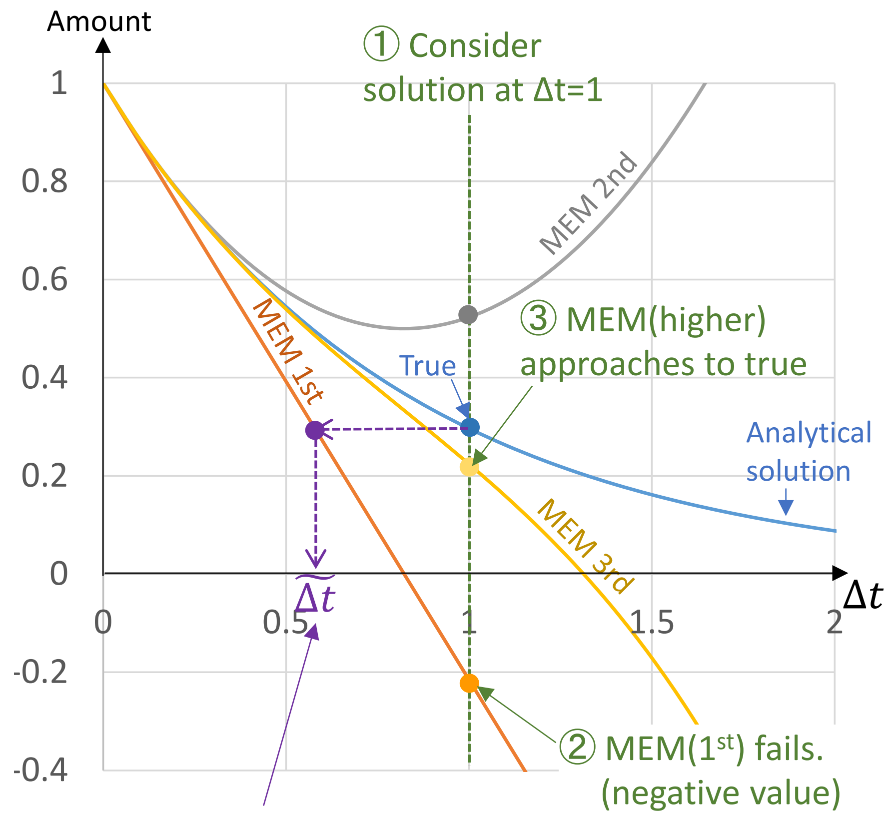
✓ OEM-transpose

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

OEM-transpose performs better.

Δt can be very long.

Solution of single decay




**④ How if $\widetilde{\Delta t}$ is used instead of $\Delta t$ in MEM-1st ?**
**⇒ New method !**

# 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.

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

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

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

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

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.

$$x_i(\mathrm{Waste1, t}) = r_i(\mathrm{Waste1})\ x_i(\mathrm{Waste\_all, t})$$

$r_i(\mathrm{Waste1})$ : 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.

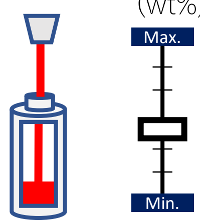
# Waste management plant 2/2

Waste management = Solidification + Storage + Disposal

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

Solidification

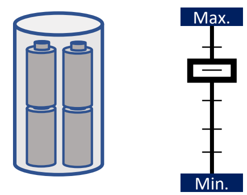
Content ratio (wt%)




Self-adjustable

Storage

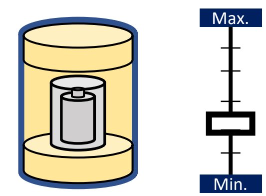
Storage period (year)




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

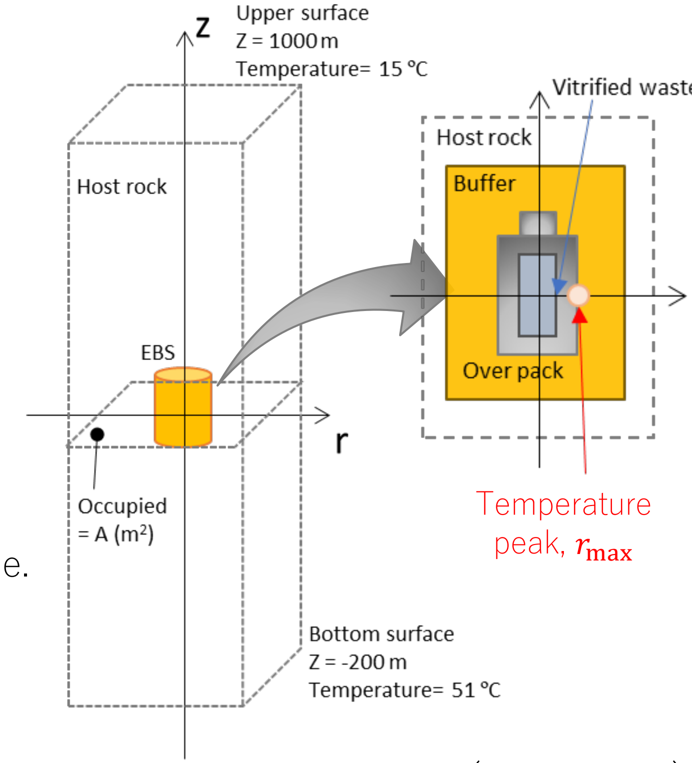
2. Source can be disintegrated by nuclide.

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

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)

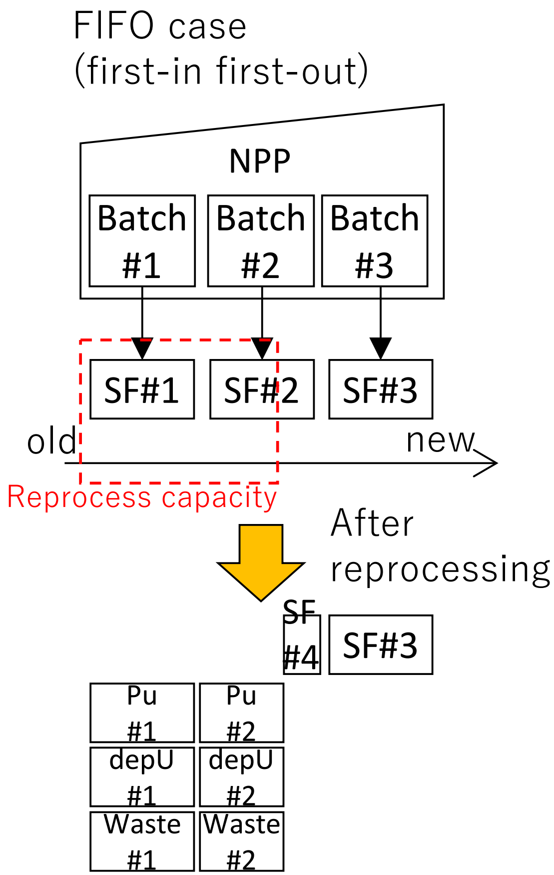
Supplement:

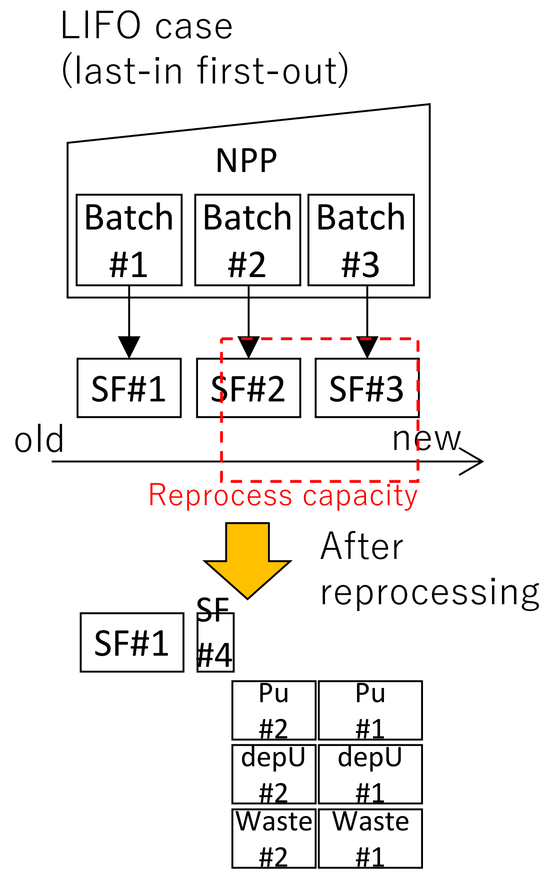
NMB Methodology

# Others

# Search of spent fuel and raw material　1/3

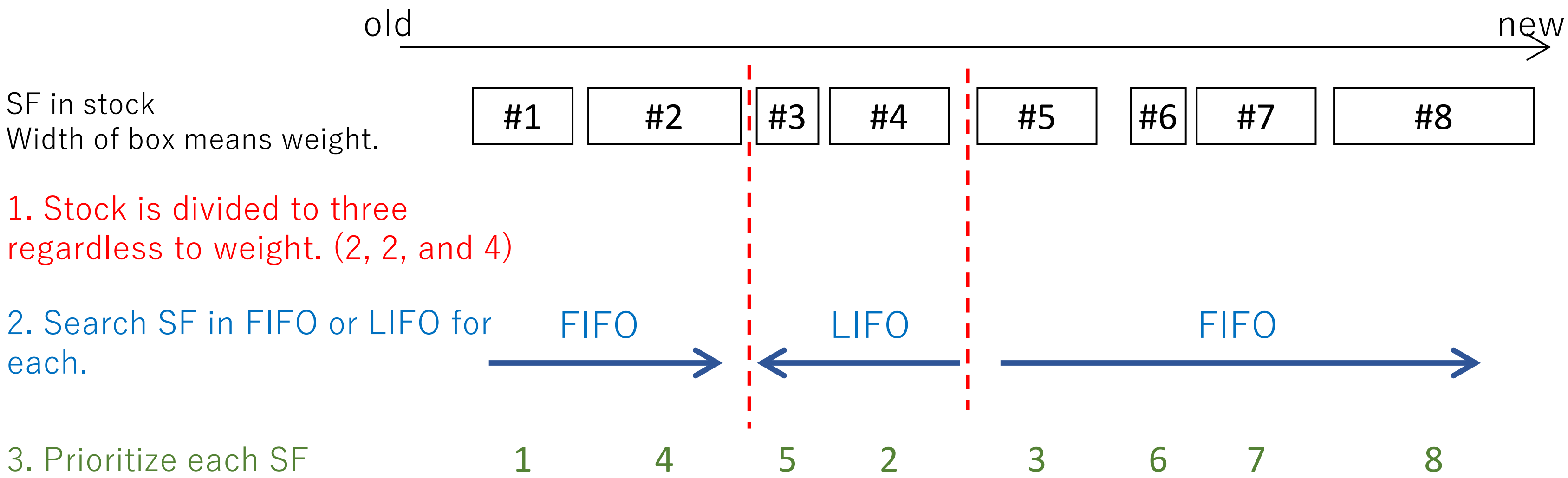
- The NMB “explicitly*” maintains information on each batch of spent fuel generated by an individual NPP.
- Individual spent fuels can be arranged in chronological order of discharge and reprocessed in order of oldest or newest.




* 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.

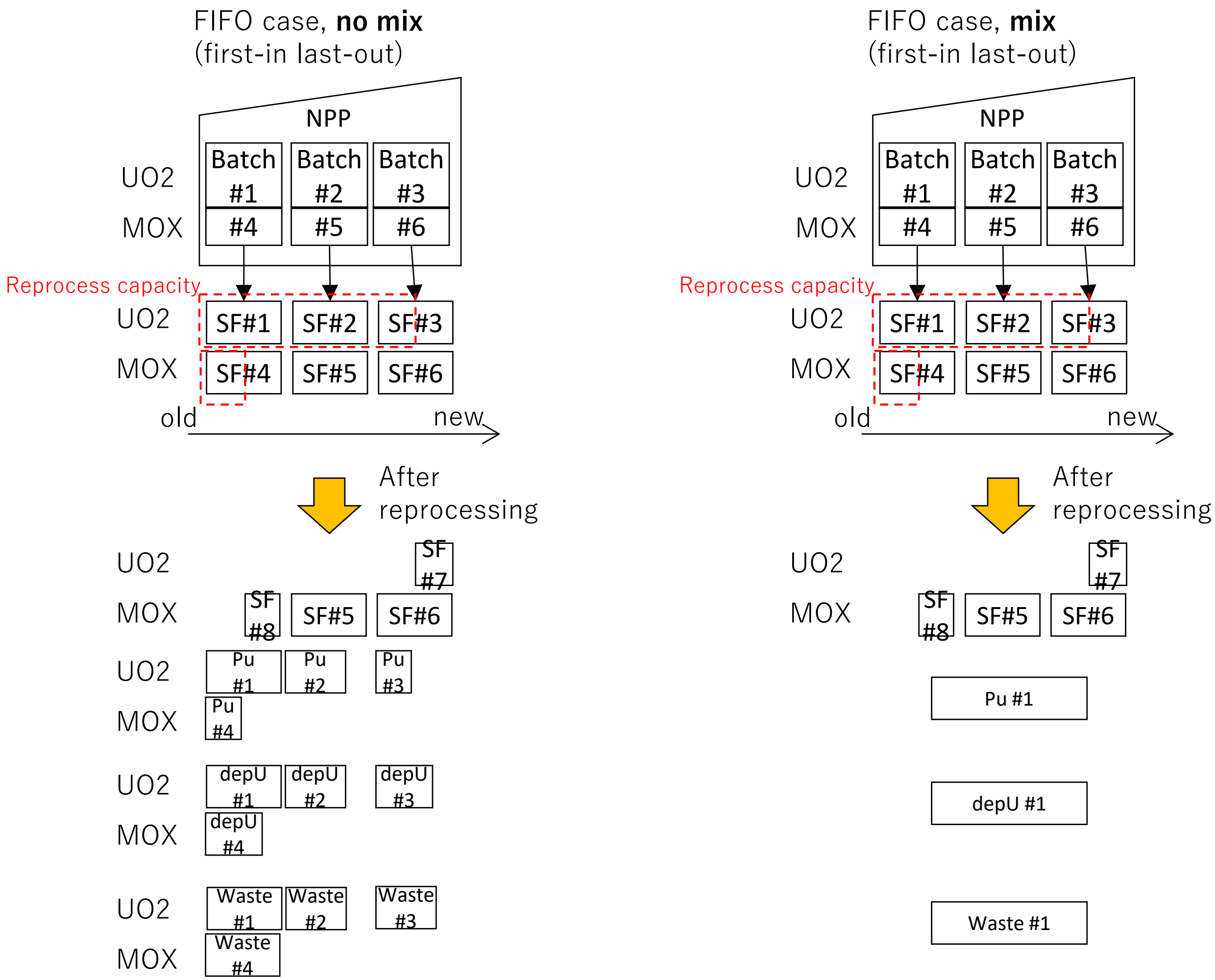
# Search of spent fuel and raw material　2/3




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



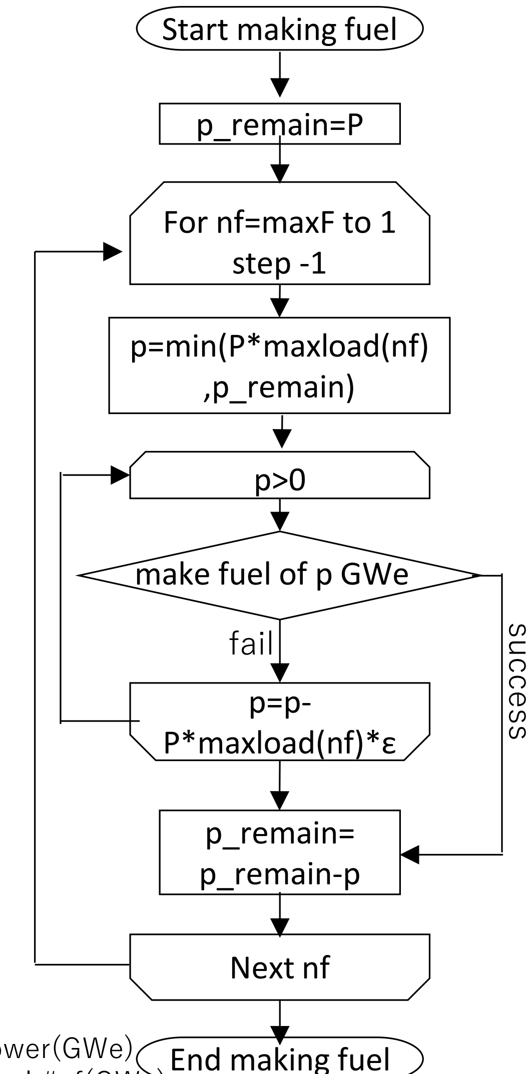

One-year averaging at reprocessing plants

- 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.

# 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**

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

"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.**

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

- $a_i$ means weight ratio. $w_{\text{fBlk}}$ is weight of blanket fuel.
- $a_i(\text{sBlk}, \text{t})$ 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.

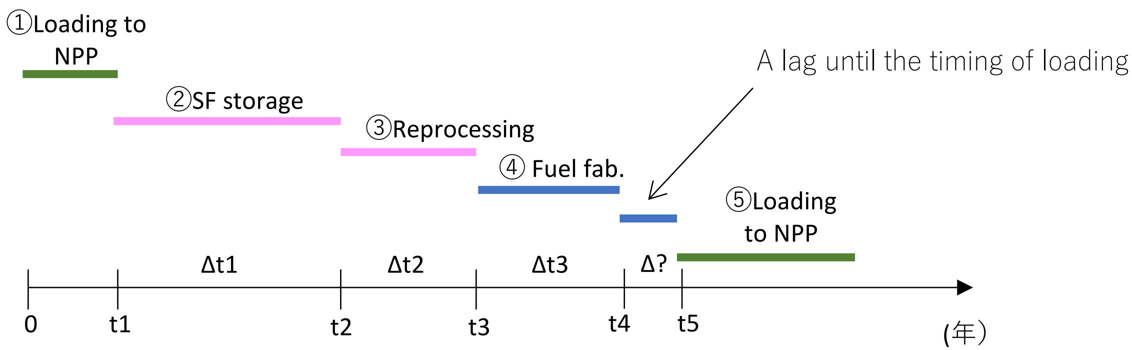
**Method 2: The user specifies the blanket amount, the specific power of the blanket section and the power ratio of the blanket to the total.**

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

- 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:.



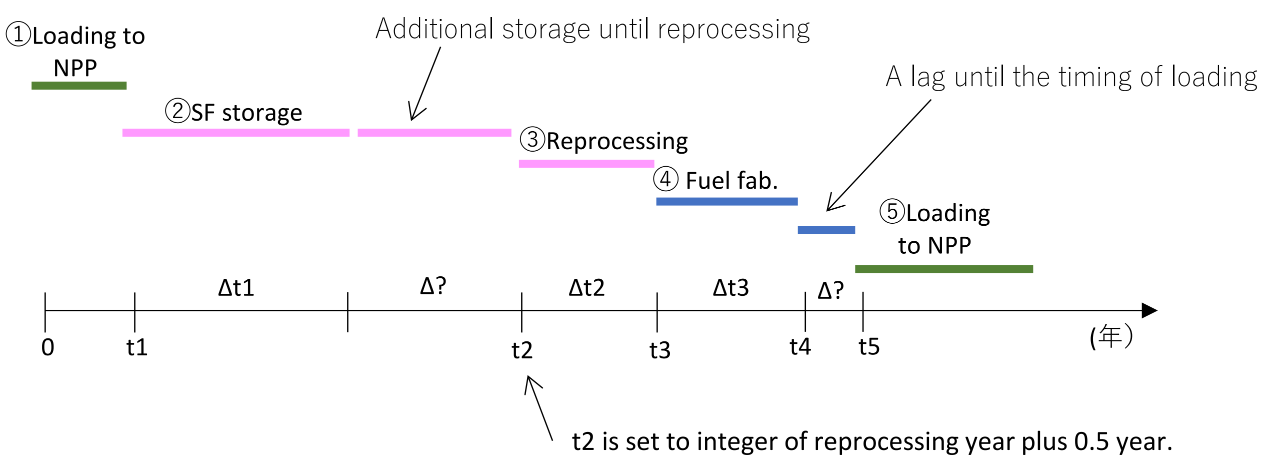

Δt1~ Δt3 is specified by user:
Δt1 : minimum SF storage required before reprocessing
Δt2 : Time required for reprocessing
Δt3 : Time required for fuel fabrication

Treated as fresh fuel in inventory estimation

Treated as spent fuel

Treated as raw material (such as Pu stock)

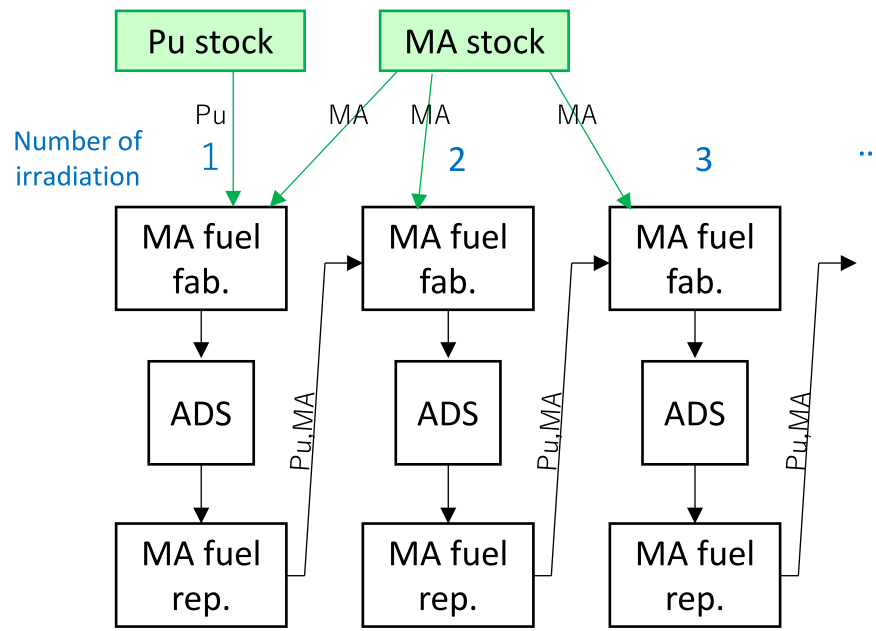
In the case that the SF storage period is longer than the minimum SF storage period:.



# 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



# Verification by NEA benchmark 1/3

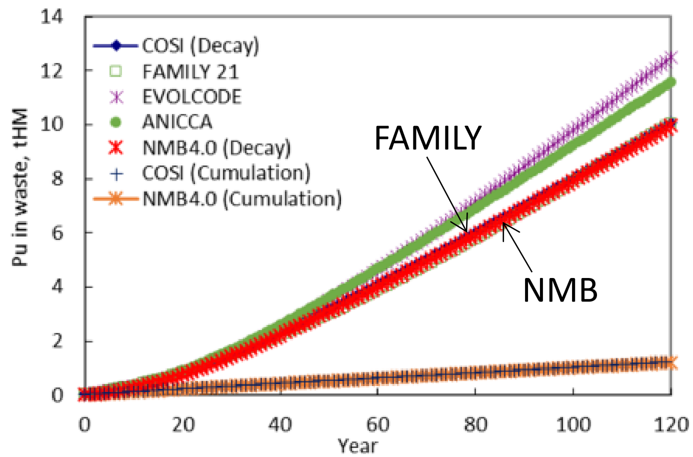
- Reference*1 verified NMB using NEA benchmark*2, to which 5 NFC simulator joins.
- Three simple benchmark problem was setup.

| | |
|---|---|
| Scenario 1 | Once-through, static |
| Scenario 2 | MOX utilization in LWR, static |
| Scenario 3 | FR transition, dynamic |

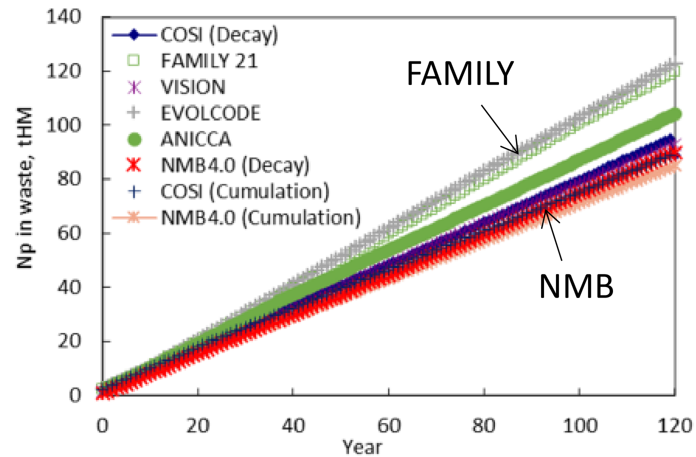
*1 T. Okamura, et. Al, NMB4.0: development of integrated nuclear fuel cycle simulator from the front to back-end, EPJ Nuclear Sci. Technol., 7, 19, 2021.
*2 Nuclear Energy Agency, in Benchmark study on nuclear fuel cycle transition scenarios analysis codes, NEA/NSC/WPFC/DOC (2012) vol. 16

# Verification by NEA benchmark 2/3

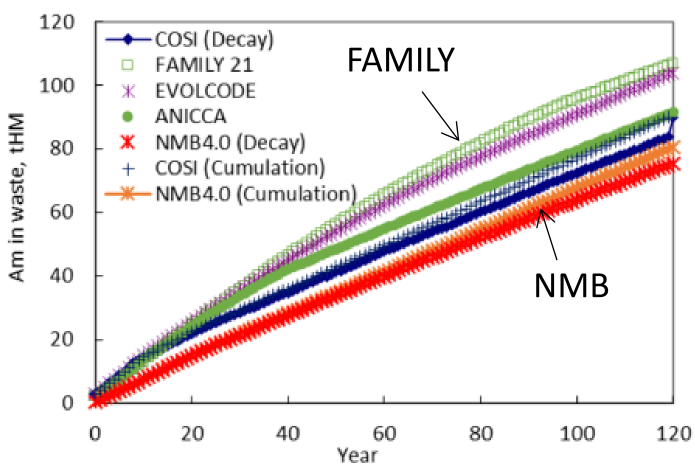
## Result of scenario 2 (MOX in LWR)



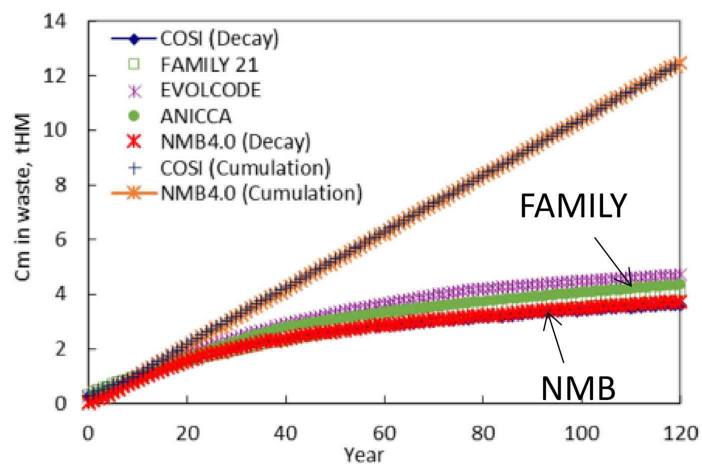

(a) Time course of Pu in waste




(b) Time course of Np in waste

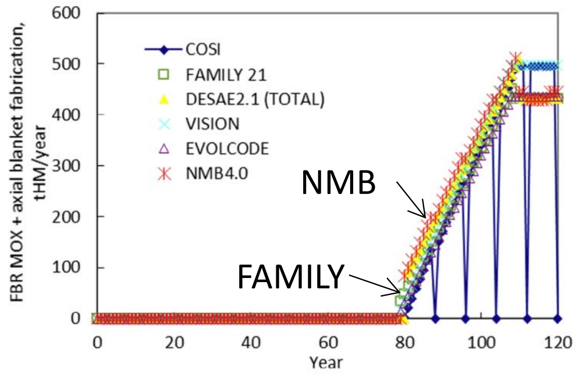



(c) Time course of Am in waste

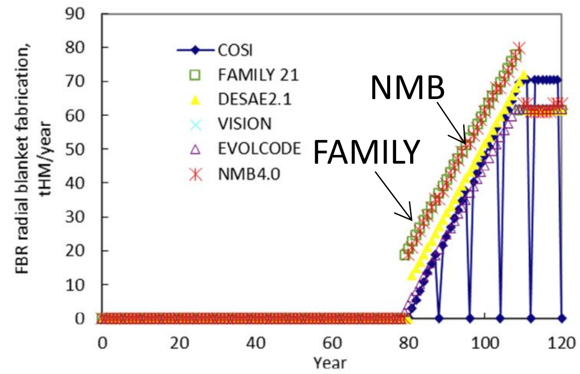



(d) Time course of Cm in waste

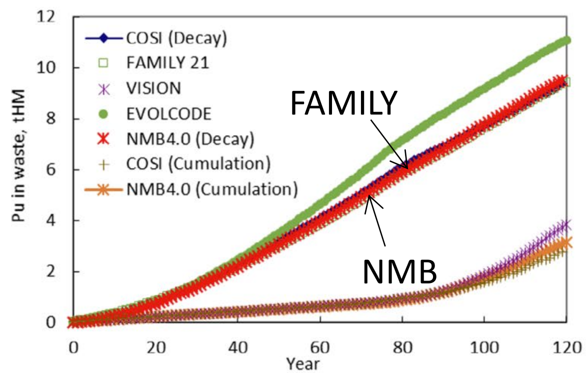
# Verification by NEA benchmark 3/3：Scenario 3 (FBR transition)



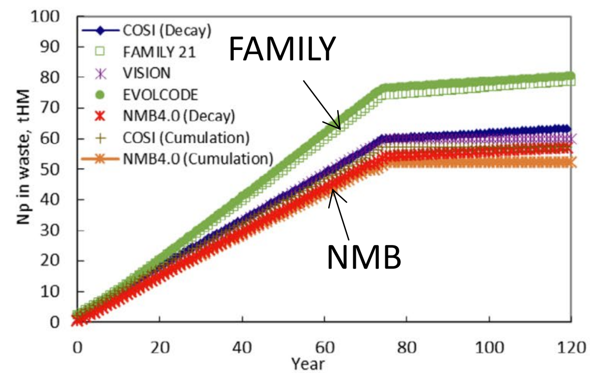

(a) Time course of FBR MOX and axial blanket fabrication



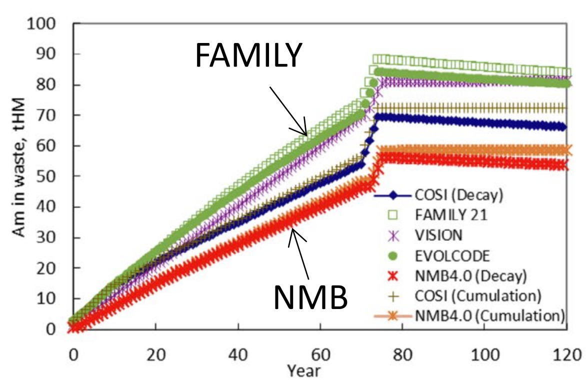

(b) Time course of FBR radical blanket fabrication



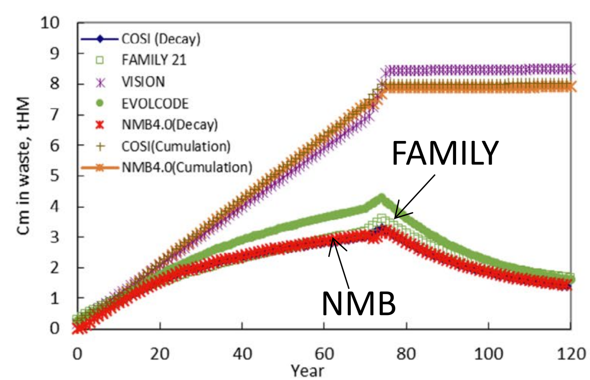

(c) Time course of Pu in waste




(d) Time course of Np in waste




(e) Time course of Am in waste




(f) Time course of Cm in waste

# Elementary Practice of NMB 4

# User application

1. Search “NMB code” on Web, or go to https://nmb-code.jp/english
2. Click “Application form” and fill it.
3. Wait for a reply from TEAM NMB. (A online interview will be required.)
4. Follow instruction from TEAM NMB to download NMB 4.0

# Tasks

| Number | Content |
|---|---|
| Task 0 | Minimum input |
| Task 1 | Operation of NPP |
| Task 2 | Direct disposal of spent fuel |
| Task 3 | Reprocessing of spent fuel |
| Task 4 | Pu utilization by MOX in LWR |
| Task 5 | Prolongation of cooling term (Practice for “Override function”) |
| Task 6 | Operation of fast breeding reactor |

# Excel settings 1 : Property



1. Right Click NMB4.?.?.xlsm file and see property.

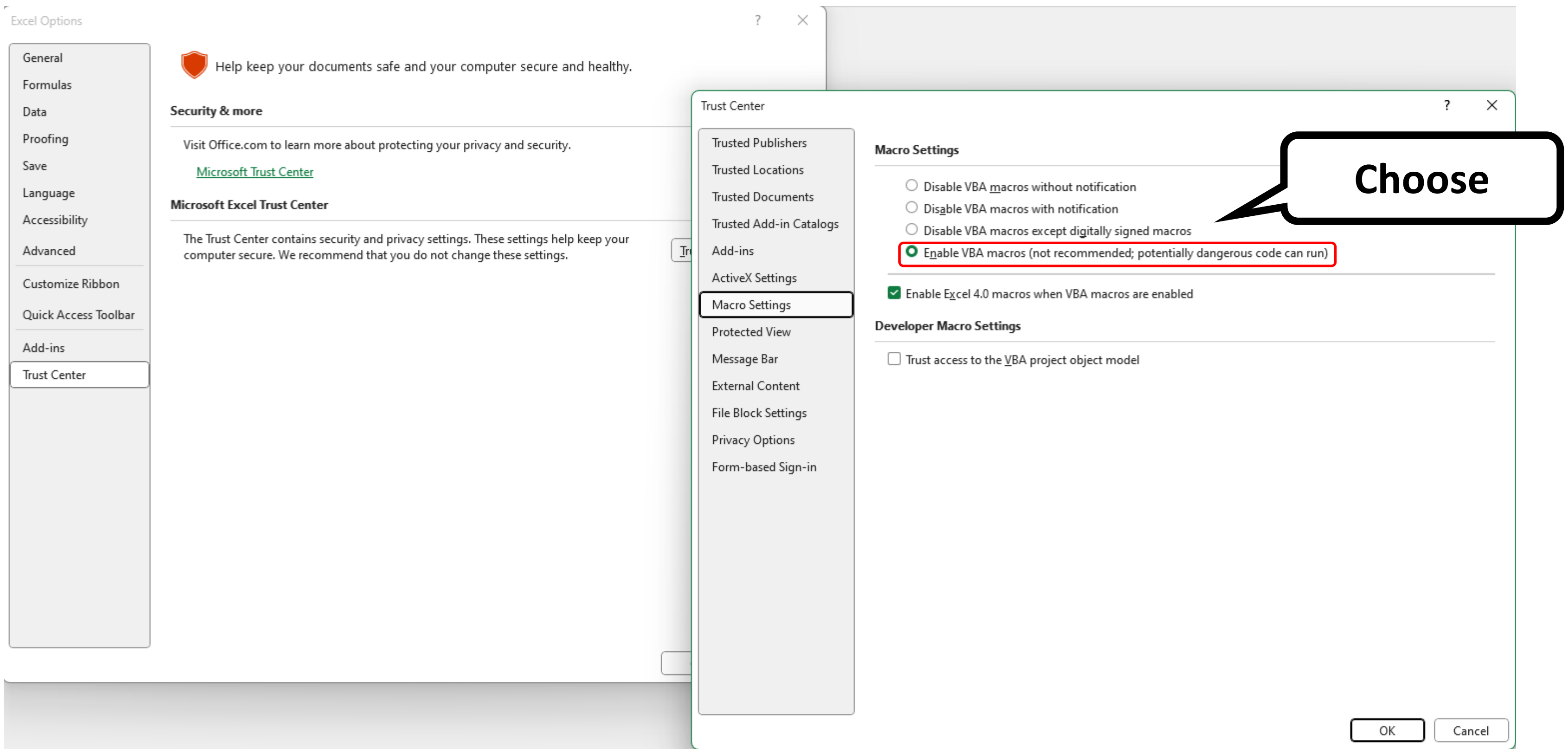
2. In General tab, place the check mark in the item of security.

If your files do not have the item, you do not have to do this setting.

# Excel setting 2 : Trust center

File→Option→Trust Center→Trust Center Settings

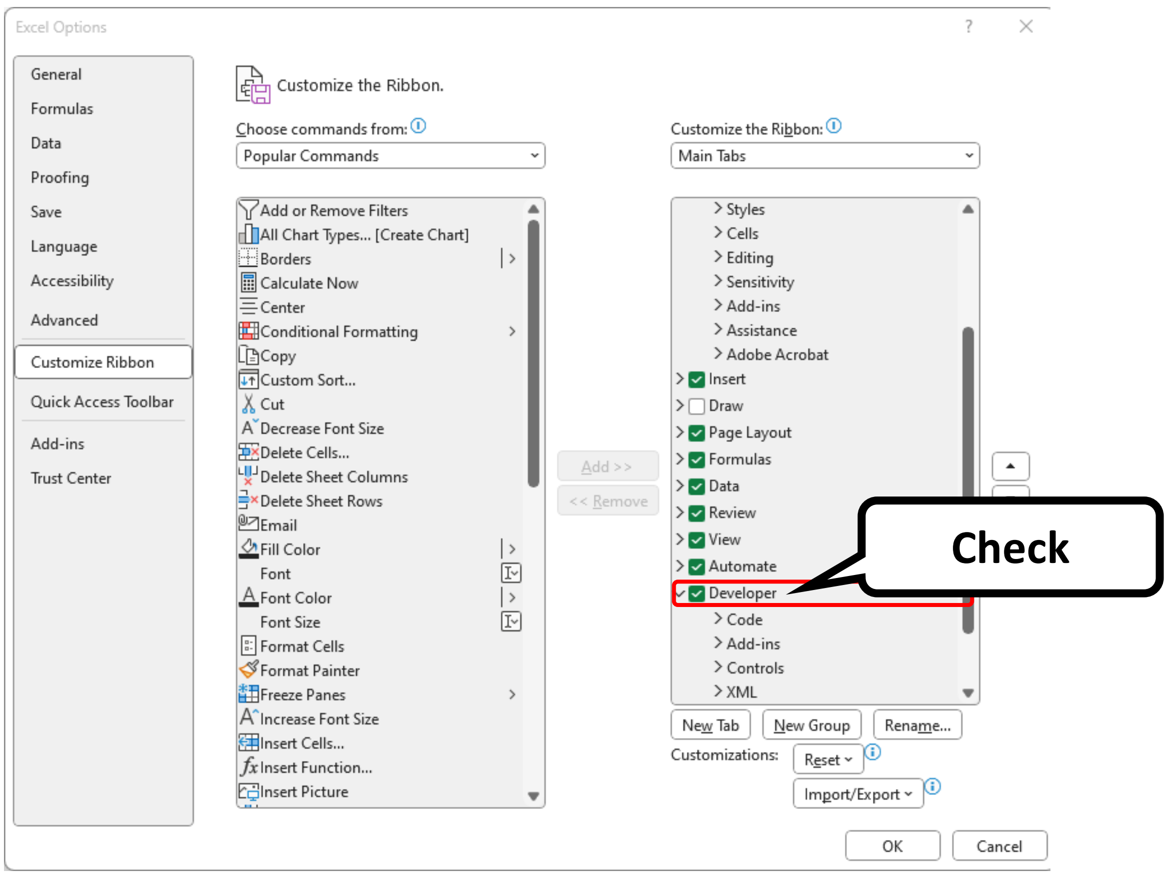
Choose “Enable VBA macros” in Macro setting



# Excel setting 3 : Developer tab

File→Option→Customize Ribbon

Place the check mark in “Developer”



# Task 0 Minimum input

1. Open NMB4.?.?.xlsm and your input book.
   NMB4.0.?.xlsm and MatrixCal64.dll must be at the same folder.
2. Display your input sheet.

- NMB finds a cell filled by "Search string" in your input sheet, then read/write to right or down from the cell. Search string is always enclosed in parentheses [].
- Any position of the cell is allowed as long as it does not interfere others.

*You may copy search strings from XLS manual.*

[Input for NMB4.0] or [Input for NMB4.1]

- User declares that this is an input sheet for NMB4.0.

[BasicData], [WasteData], [PlantData], [CycleData], [PredefineData], [CostData]

- User specifies XLS book and sheet name where NMB database is written.
- NMB reads two cells below each search string in order of "book name" and "sheet name".

*Take care for blank in search string*

[Messages]

- NMB displays analysis status in the lower right cell.

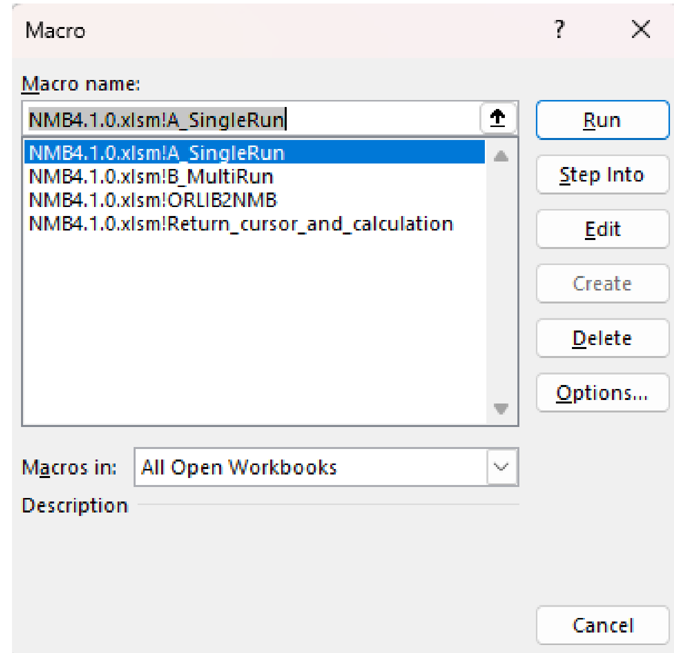
[A.D.]

*Start from 2000 and end at 2130.*

- User specifies year to be analyzed.
- NMB reads one column from two cells down to a blank cell.

# How to run NMB

Push Alt+F8, and execute "A_singleRun" macro!



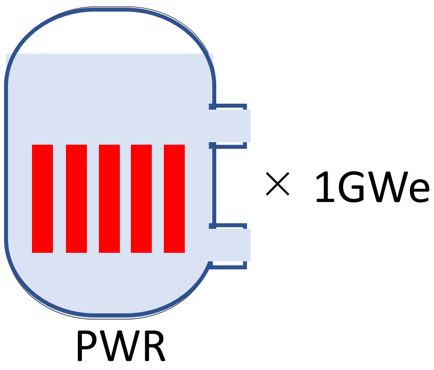

Typical error after running:

- If NMB stops at "Call CallInitialize", your excel in 32-bit. Please use NMB4.?.?.xlsm and MatrixCal32.dll in "32bit" folder in distributed package.
- If NMB stops at "Set shBasic = ….", your specification in [BasicData] is wrong.
- If NMB stops at "ChDrive ThisWorkbook.Path" or "ChDir ThisWorkbook.Path", you are using cloud environment. Please download all files to your local computer.

# Task 1　Introduce PWR

Condition
- A PWR (1GWe and 45GWd/tHM) is operated during 2000 to 2020.

*Write PWR-UO2-MB at [NewPlant1]*
*Write 1 at [PlantPower1]*
*Write 20 at  [PlantLife1]*




Output *
- Weight of consumed natural U (t/y) → *write [prtMiningDemand[t/y]]*
- U235 Enrichment → *write [prtU235EnrInNewFuel]*
- Annual electricity generation → *write [prtPowerGeneration[GWeY]]*
- Annual loading of FF → *write [prtNewFuel[t/y]]*
- Annual production of SF → *write [prtSpentfuel[t/y]]*
- SF storage → *write [prtInterimStorage[t]]*

* The maximum number of output column is specified in the right cell of search string.

*Write 4 as the maximum column*

***Run NMB and look into the result***

# Output of task 1

*[prtMiningDemand [t/y]], [prtNewFuel[t/y]]*

- They had highest values at initial loading (A.D. 2000).
- Fuel loading became 1/3 after 2001, because the PWR was operated by “3-batch fuel exchange”. Mining had a time lag because of the lead time of the fuel fabrication.

*[prtU235EnrInNewFuel]*

- Constant at 4.1wt%.

*[prtPowerGeneration[GWeY]]*

- Electricity generation (GWe x year) was not 1 GWeY but 0.85 GWeY because load factor was defined as 85% in the “PlantDataJ40” database sheet.

*[prtSpentfuel[t/y]]*

- Production of SF was 27 t/y at most of period. When the PWR was shutdown, it became 3 times larger.

*[prtInterimStorage[t]]*

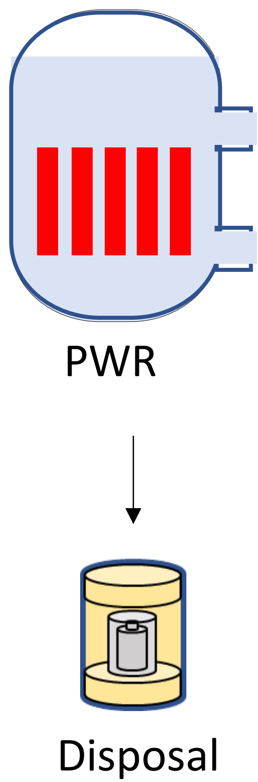
- SF storage increased up to 485 ton at the end. The value did not decrease because reprocessing or direct disposal was not introduced.

# Task 2　Direct disposal of spent fuel

*Copy the input sheet of "task 1", and rename to "task2".*

**Condition:**
- SF from PWR is directly disposed of without reprocessing.

*Write DD-PWR below [ReprocessingCapacity[t/y]]*
*Write maximum annual throughput of 100tHM/year till 2130.*




**Output:**
In addition to Task 1,
- Waste storage →*[prtWasteStorage]*
- Waste disposal →*[prtCummulativeCanisterDisposal]*
- Footprint of disposal site →*[prtCummulativeDisplsalArea]*

*Run the NMB and look into the result*

# Output of task 2

*[prtInterimStorage[t]]*

- SF Storage was kept at small amount comparing to Task 1.

*Note: SFs are stored for 50 years before direct disposal in practice. However, in NMB this storage is treated as a "waste storage" (not the "SF storage") Therefore, SFs are immediately removed from SF storage.*

*[prtWasteStorage]*

- The maximum amount of SF canister was 860. The value was determined taking account of the heat generation of the waste. It became 0 after the disposal.

*[NumDisposedCanister]*

- The final amount of SF canister was 860.

*[Waste_footprint[m2]]*

- Necessary area of disposal site was 106,000 $m^2$.

*EXCEL tips: In order to easily look excel result, activate "freeze panes".*

# Task 3　Reprocessing of spent fuel

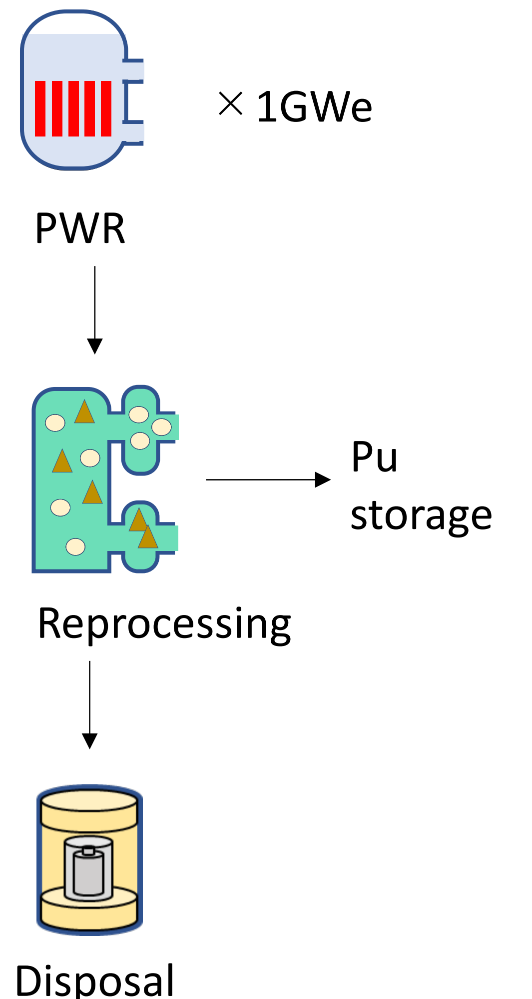
*Do not forget to copy the input sheet*

Condition:
- Run the reprocessing plant until all spent fuel is reprocessed, with a processing capacity same to annual production of spent fuel.

*Change DD-PWR in task 2 to UO2(Pu) with annual throughput of 27tHM/year.*

Output:
In addition to Task 2,
- Annual reprocessing amount *→[prtReprocessing[t/y]]*
- Pu storage *→[prtPu[t]] with 10 columns*

*Run NMB and compare results toTask2.*
- *[prtInterimStorage[t]]*
- *[WasteStorage]*
- *[prtCummulativeCanisterDisposal]*
- *[prtCummulativeDisplsalArea]*



# Output of task 3

*[prtReprocessing[t/y]]*

- Reprocessing was operated from 2004 till 2025.

*[prtInterimStorage[t]]*

- SF Storage was kept at small amount but larger than Task 1 because of 3 years of storage was needed before reprocessing.

*[prtWasteStorage] , [prtCummulativeCanisterDisposal]*

- 500 glass wastes was produced at the end.

*[prtCummulativeDisplsalArea]*

- Necessary area of disposal site was 22,000 $m^2$, which was 20% of direct disposal (task 2).

*[prtPu[t]]*

- All Pu in spent fuel was extracted and 4t of Pu remained.

# Task 4 Utilize Pu in PWR

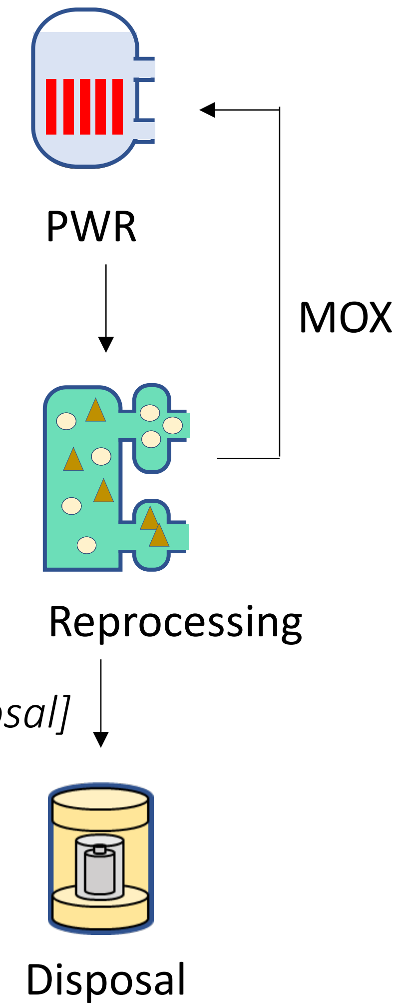
**Condition:**
- Recovered plutonium is used as MOX fuel in PWR

*Write [@PuThermalRatio] and fill a column by 15%.*

**Output:**
In addition to Task 3,
- Pu content in MOX fuel → *[prtPuRatioInNewFuel]*

*Run NMB and compare results toTask3.*

- Electricity generation by MOX fuel *[prtPowerGeneration[GWeY]]*
- SF storage *[prtInterimStorage[t]]*
- Waste production *[prtCummulativeCanisterDisposal]*
- Repository footprint *[prtCummulativeDisplsalArea]*
- Plutonium storage *[prtPu[t]]*




*Note about "@": For some parts of the Database, yearly input in the scenario sheet is allowed. To do so, fill in the time-dependent tags starting with @ in the Database sheet (e.g. @PuThermalRatio; note that strings starting with @ should be entered as '@ in EXCEL) Correspondingly, in the scenario sheet, the input for each year is read vertically in a single column (calculated years) from a time-dependent tag with [] key brackets added to the string. (e.g., [@PuThermalRatio])*

# Output of task 4

*[prtPowerGeneration[GWeY]]*

- Around 15% was generated by MOX.

*[prtPuRatioInNewFuel]*

- Pu content ratio was 5 to 8 %, which increased because fissile Pu content was rich at the initial loading fuel with low burn-up.

*[prtInterimStorage[t]]*

- Storage for UO2 spent fuel became zero, while that for MOX spent fuel remained.  MOX SF was not reprocessed.

*[prtCummulativeCanisterDisposal] , [prtCummulativeDisplsalArea]*

- Number of waste and footprint slightly decreased because production of UO2 spent fuel declined.

*[prtPu[t]]*

- Pu remained was 1.4 ton, that was much smaller than task 3.

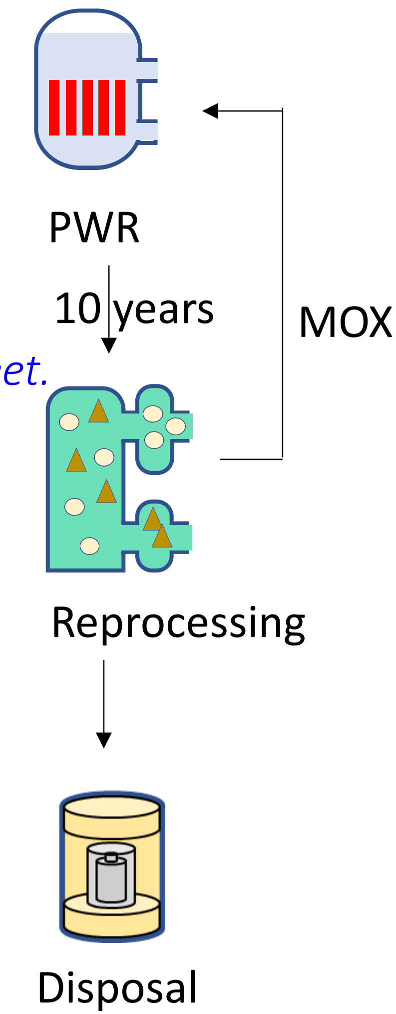
# Task 5 Change cooling period before reprocessing (Override function)

Condition:
- Cooling period of spent fuel before reprocessing is changed from 3 years to 10 years.

*To override one parameter in NMB original database,*

1. *Copy the cells from [RepIndex] to [MixSpentFuel] in PredefineData sheet.*
2. *Copy UO2(Pu) row below the copied cells.*
3. *Change 3 below the cells of cooling term to 10.*

***Direct change of the original database is NOT recommended!***




Output: Same to Task4

*Run NMB and compare results toTask4.*

- Electricity generation by MOX fuel *[prtPowerGeneration[GWeY]]*
- Waste production *[prtCummulativeCanisterDisposal]*
- Repository footprint *[prtCummulativeDisplsalArea]*
- Plutonium storage *[prtPu[t]]*

# Output of task 5

*[prtPowerGeneration[GWeY]]*

- MOX loading starts later than task 4

*[prtCummulativeCanisterDisposal], [prtCummulativeDisplsalArea]*

- The amount of waste from UO2 fuel should increase, but it is slightly less.
- Because of the extended cooling term, the amount of heat-generating nuclides in the spent fuel decreased due to decay. This allowed for a larger content of waste elements per waste.

*[prtPu[t]]*

- Pu inventory was larger than task4 because of reduced usage.

# Task 6 Introduction of FBR

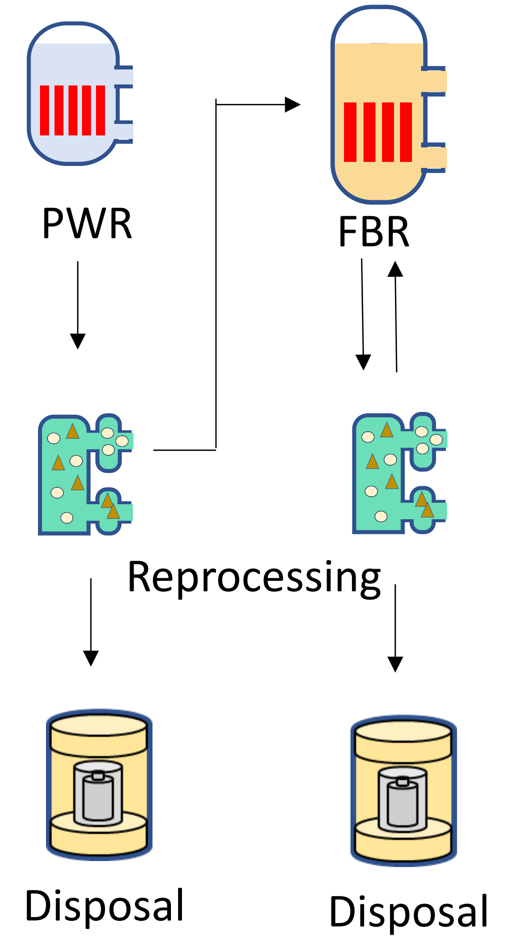
Condition
- FBR with high breeding ratio is operated from 2033 to 2072 with XXX GWe. Estimate XXX.
- Reprocessing of FBR MOX spent fuel is implemented from 2033 with 27 t/year.

1. *Write FBR_HB at [NewPlant1], 0.5 at [PlantPower1] and 40 at [PlantLife1].*
2. *Add MOX(Pu) in [ReprocessingCapacity[t/y]] filled with 27.*

Output:
In addition to task 5,
- Analysis information → [Information]

1. *Run NMB. FBR_HB construction will be failed as written in [information]*
2. *Change capacity to 0.25 GWe and run again.*
3. *Check the result*

- Uranium mining *[prtMining[t/y]]*
- Waste production *[prtCummulativeCanisterDisposal]*
- Repsitory footprint *[prtCummulativeDisplsalArea]*
- Pu storage *[prtPu[t]]*




*Note: NMB will skip construction of NPP if raw material is not enough, or construct [[BackupPlant1] if specified by user.*

# Output of task 6

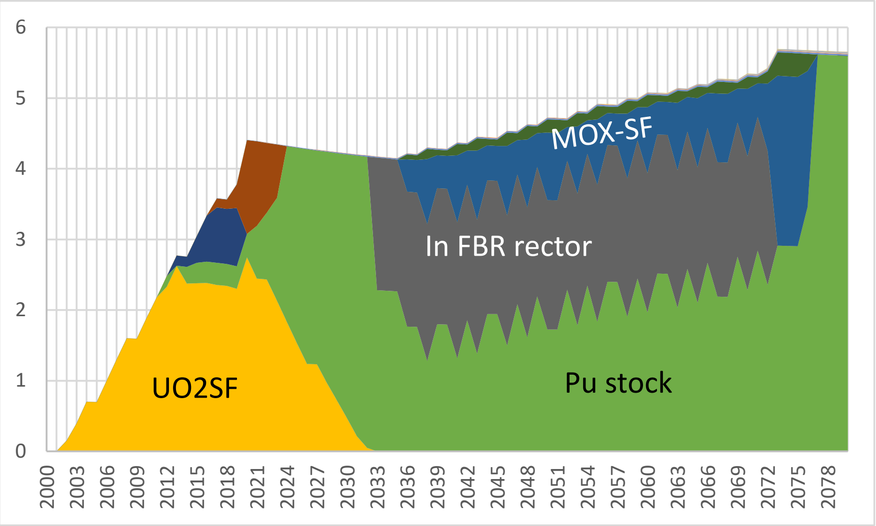
*[prtMining[t/y]]*

- No uranium was mined after 2033.

*[prtPu[t]]*

- Pu stock was around 4.1 t after PWR operation.
- Pu stock decreased to 1.3 t by introducing FBR.
- Pu stock gradually increased because breeding ratio of FBR_HB is 1.1. (see [BreadingRatio] in PlantdataJ40 sheet)

*EXCEL tips: stacked surface graph is convenient to look into Pu inventory in NFC.*



# Triple scenario

# Tripling nuclear energy by 2050


Up to now, 31 countries have joined to this declaration.

Armenia, Bulgaria, Canada, Croatia, Czech Republic, El Salvador, Finland, France, Ghana, Hungary, Jamaica, Japan, Kazakhstan, Kenya, South Korea, Kosovo, Moldova, Mongolia, Morocco, Netherlands, Nigeria, Poland, Romania, Slovakia, Slovenia, Sweden, Turkey, Ukraine, United Arab Emirates, United Kingdom, United States of America.

# Nuclear in 2020s

Table 3. **Global installed nuclear energy capacity by continent (2021)**

| | | | |
|---|---|---|---|
| **444** nuclear reactors in operation globally | **30** countries with nuclear reactors | **394** gigawatts global installed nuclear capacity | **66** gigatonnes of $CO_2$ cumulative emissions avoided since 1971 |

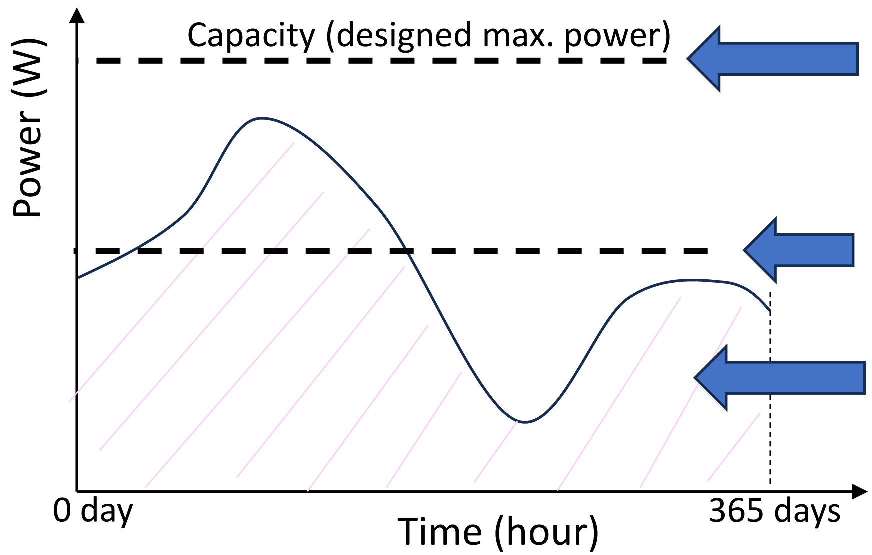
NEA No.7628 (2022)

**Table A.20: Nuclear generation** (TWh)

| | Historical | | | Stated Policies | | Announced Pledges | |
|---|---|---|---|---|---|---|---|
| | 2010 | 2021 | 2022 | 2030 | 2050 | 2030 | 2050 |
| World | 2 756 | 2 810 | 2 682 | 3 351 | 4 353 | 3 496 | 5 301 |
| North America | 935 | 916 | 898 | 925 | 934 | 926 | 1 195 |
| United States | 839 | 812 | 804 | 825 | 804 | 825 | 1 023 |
| Central and South America | 22 | 26 | 23 | 31 | 75 | 35 | 83 |
| Brazil | 15 | 15 | 15 | 24 | 45 | 24 | 45 |
| Europe | 1 032 | 889 | 750 | 808 | 781 | 894 | 970 |
| European Union | 854 | 732 | 607 | 623 | 576 | 703 | 698 |
| Africa | 12 | 12 | 11 | 24 | 43 | 29 | 72 |
| Middle East | 0 | 15 | 26 | 45 | 89 | 51 | 146 |
| Eurasia | 173 | 225 | 226 | 238 | 306 | 238 | 315 |
| Russia | 170 | 223 | 224 | 236 | 297 | 236 | 297 |
| Asia Pacific | 582 | 726 | 748 | 1 281 | 2 125 | 1 322 | 2 520 |
| China | 74 | 408 | 418 | 642 | 1 247 | 661 | 1 483 |
| India | 26 | 47 | 50 | 128 | 337 | 125 | 355 |
| Japan | 288 | 71 | 60 | 207 | 206 | 232 | 289 |
| Southeast Asia | 0 | 0 | 0 | 0 | 8 | 0 | 12 |

IEA: World Energy Outlook 2023

# What is “GWe” and “TWh”?




Maximum value is “W” or “We”.
Giga is 1e+9.
Typical power of an NPP is 1 GWe (1e+9 W).

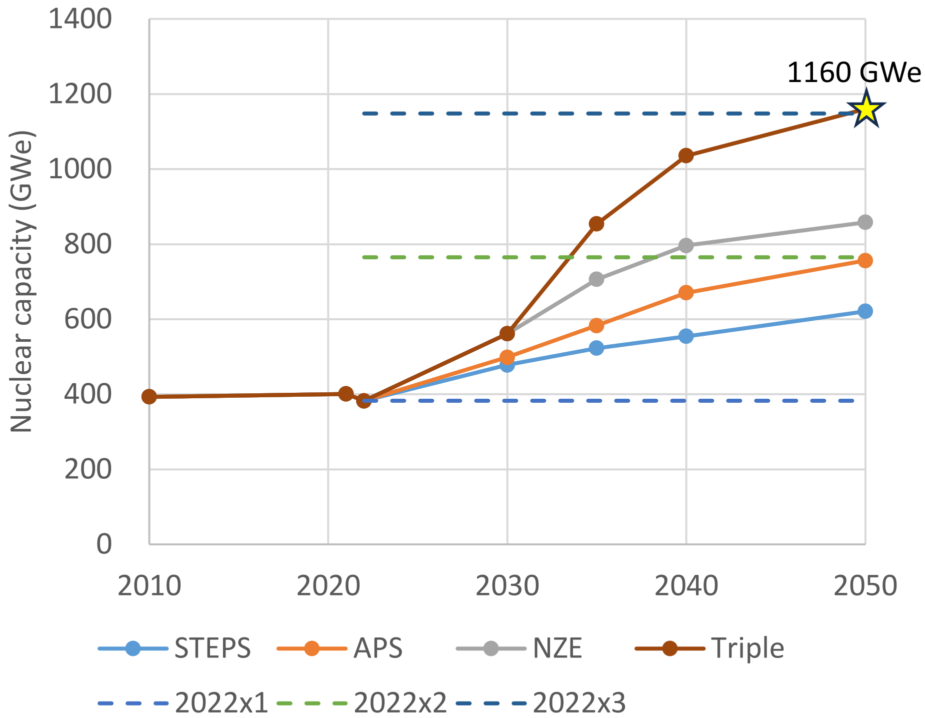
“Operation factor” is ratio of averaged power to the maximum power.

Area is “Wh”. Tera is 1e+12.

Exercise 1: How many electricity (TWh) is generated if you operate an NPP whose capacity 1GWe in operation factor of 80%.

Exercise 2: The capacity of global NPPs was 394 GWe and nuclear electricity generation was 2810 TWh in 2021. What is operation factor in this year?

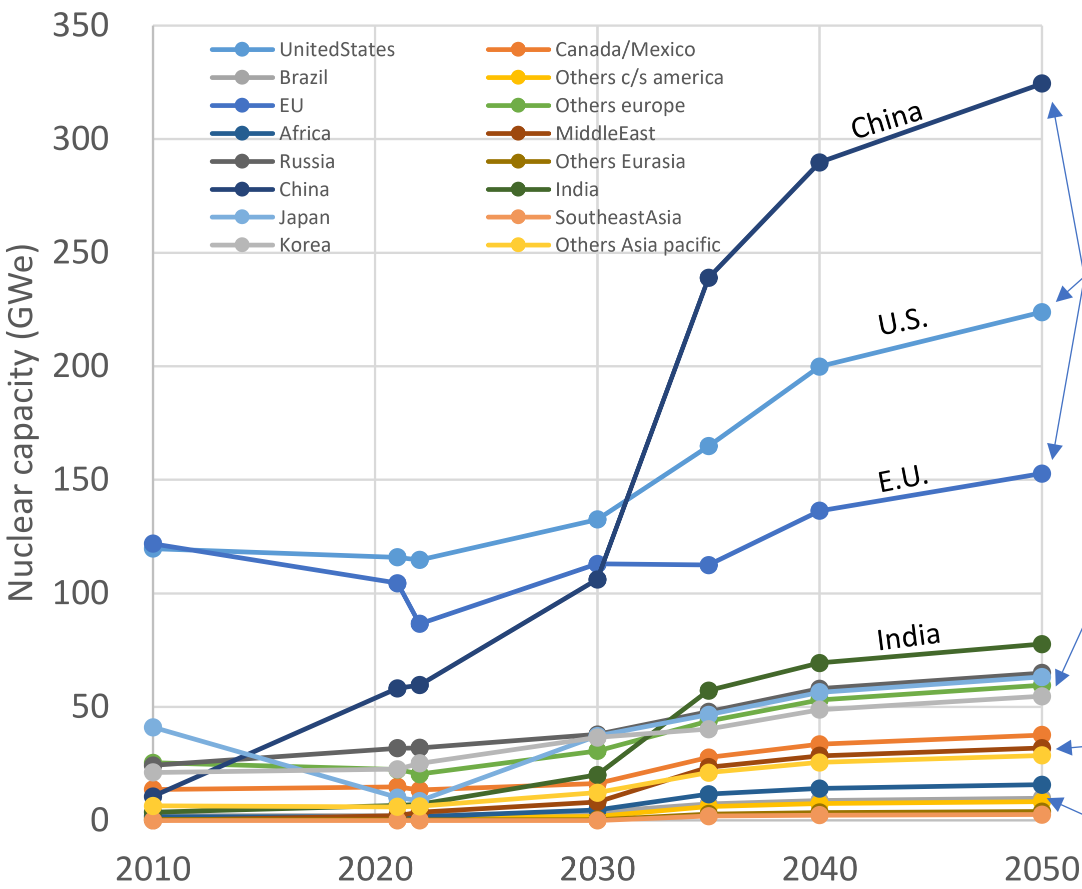
# Triple Nuclear in 2050




Nishihara converted TWh to GWe assuming 80% operation factor for STEPS, APS and NZE based on WEO 2023.
Triple scenario is generated by extrapolation of NZE scenario to target of 1160 GWe stated in IPCC 2018 document.

- The projections in the Stated Policies Scenario (STEPS) give a sense of the current direction of travel for the energy economy, based on the actual state of play in different sectors, countries and regions.
- The Announced Pledges Scenario (APS) shows how that future would be different if all countries were to hit their aspirational targets, including national and regional net zero emissions pledges, on time and in full.
- The updated Net Zero Emissions by 2050 (NZE) Scenario illustrates what more is required to limit global warming to 1.5 degree Celsius.
- Triple scenario can not be achieved by ambitious pledges (APS) or goal-oriented value (NZE).

# Breakdown to country (Triple)



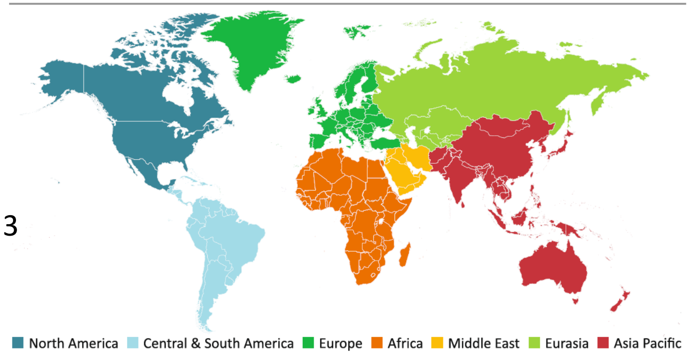

Big 3

2nd group: India Russia, Japan, Other Europe, Korea

3rd group: Canada and Mexico, M.E. Asia, Other Asia pacific

4th group: Africa, Brazil, Other C/S America, Other Eurasia, S.E. Asia

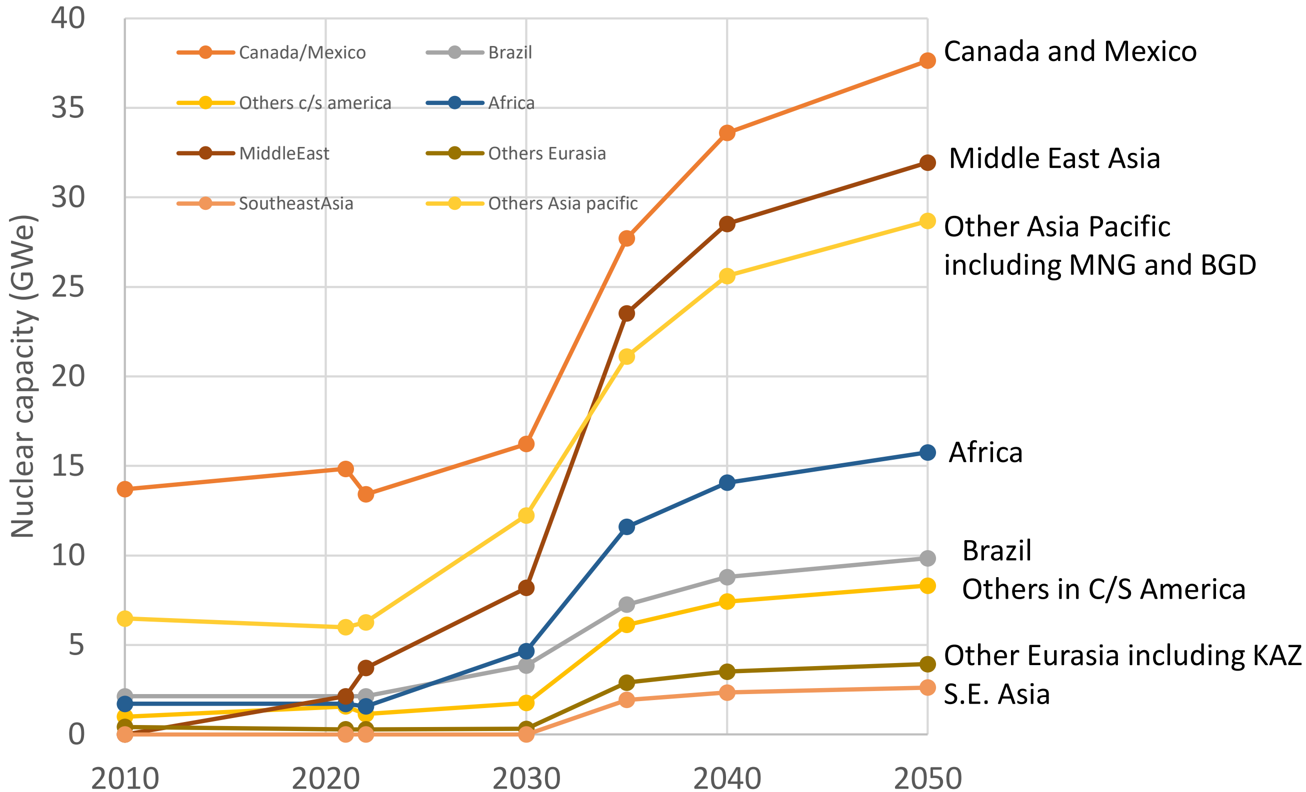
**Figure C.1** ▷ **Main country groupings**




Note: This map is without prejudice to the status of or sovereignty over any territory, to the delimitation of international frontiers and boundaries and to the name of any territory, city or area.

Nishihara made trend for each country assuming share of each country at 2035 and 2040 is similar to that at 2050 in NZE scenario of WEO2023.

# $3^{rd}$ and $4^{th}$ group




Contribution of South/East Asia is such small?

# Case of Vietnam


Address by Prime Minister Pham Minh Chinh on 12 November, 2024.

- Electricity demand increasing:
  - 220 TWh in 2020
  - 1000 TWh (1E+15 Wh) in 2045
- Government have decided to resume Ninh Thuận 1 and 2 nuclear plant projects (total: 4 GWe) which has been halted since 2016.

- *Exercise: How much electricity will be added by NT-1 and 2?*

# Advanced practice of NMB 4.0

# Task X: Scratch your own scenario towards "Triple-nuclear world"

- Imagine the future of the nuclear industry in <u>your country or your region</u> assuming global growth of nuclear industry.
- Illustrate your NFC on paper and ppt.
- Clarify the key stones (start time and scale of NPP, those of reprocessing, waste management etc.)
- Make NMB inputs.
    - NPP
    - Reprocessing
    - Waste management
- Check the result and feed back to inputs. Be aware that this procedure can continue very long!
- Finalize your calculations and summarize them for presentation.

# Hint: Presentation structure

Title example:

Future of nuclear power industry in Japan in triple nuclear scenario.

Content example:

1. Current national energy mix and status of nuclear energy
2. Government plan
3. Your proposal for future of nuclear power
4. Scenario analysis of NFC
5. Discussion/Conclusion

# Hint: Scenario analysis

1. Select concepts of future NFC. (Once-through?, LWR?, FBR?)
2. Set up inputs for each concept with same power generation capacity.
3. Run NMB, check result and modify input.
4. Edit result. (How much uranium resource, fabrication capacity, storage, disposal, …)
5. Compare results among several scenarios and propose your choice of future NFC.

# Presentation

- Use your PC and power point
- 10 min. for presentation
- 10 min. for discussion