

2024年2月16日

第2回NMB研究会･講演資料

1)「**Tutorial of IAEA's Fuel Cycle Code : NFCSS**」 IAEAサイトに掲載予定。2024年
2)「IAEAの核燃料サイクル計算コードVISTA」 吉岡、日本原子力学会誌 Vol.47, No.9(2005)
3)「IAEA の燃料サイクルコードNFCSS の改良（Th サイクル他）」 吉岡、林、三成、Rethinaraj、Sim、日本原子力学会2020年秋の大会、2H07

トリウム熔融塩国際フォーラム

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Tutorial of IAEA's Fuel Cycle Code : NFCSS

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International Thorium Molten-Salt Forum

Contents

Title	To the audience who.....
1) Introduction of fuel cycle (3 min.)	needs basic knowledge of fuel cycle
2) Introduction of NFCSS (4 min.)	wants to know its purpose and effectiveness
3) Details of NFCSS (6 min.)	plans to use NFCSS
4) CAIN Burnup Engine (13 min.)	plans to use NFCSS
5) Decay Heat Calculation (10 min.)	needs this area as backend study
6) Radiotoxicity Calculation (10 min.)	needs this area as backend study
7) Relation to Deep Geological Repository (5 min.)	needs this area as backend study
8) Actual Operation Video of NFCSS Case-1: Reproduce built-in case Case-2: Create user-specified case	needs practice for NFCSS

Please watch each section in the following slides.

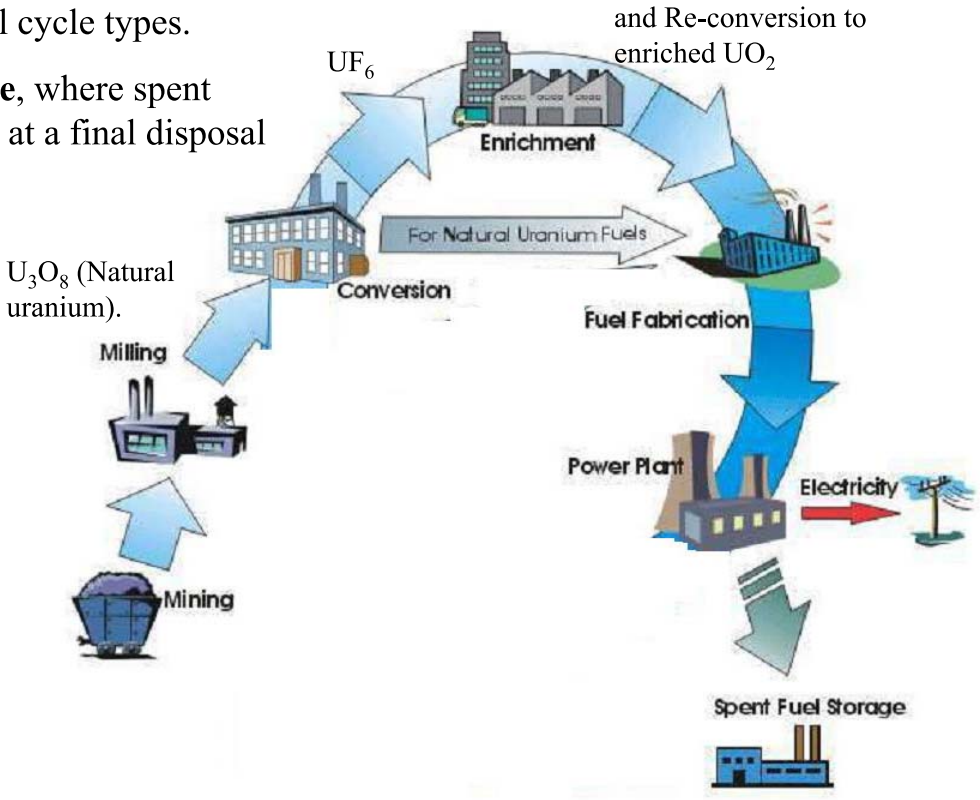
They are shown automatically, just after the narration of each slide.

1. Introduction of Fuel Cycle

Open Fuel Cycle

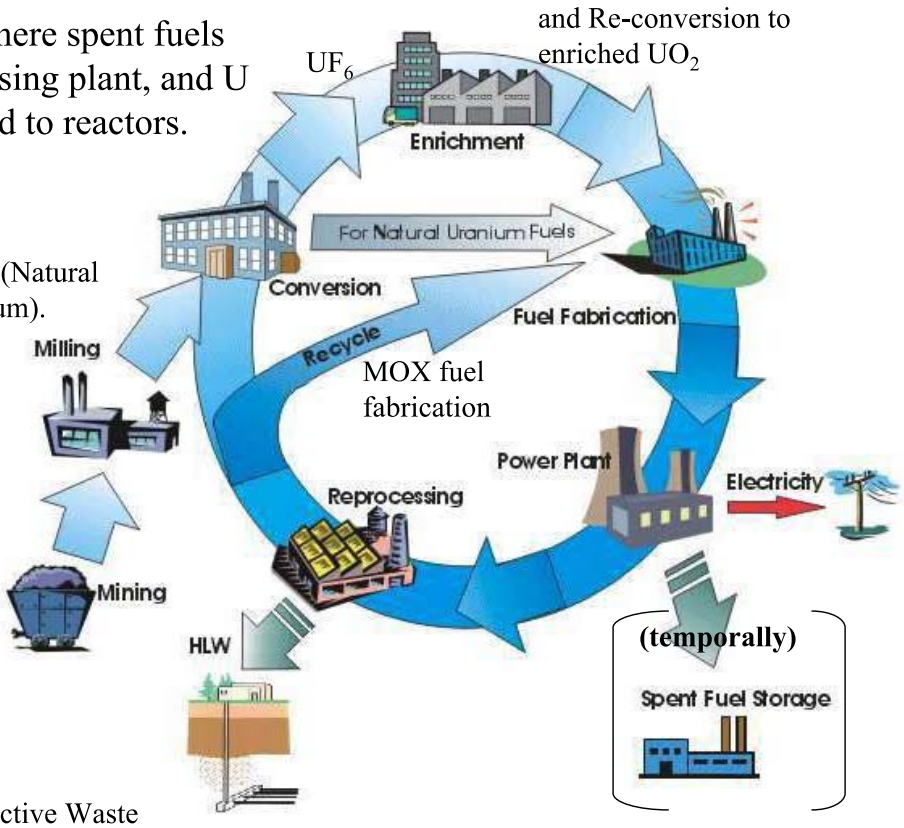
There are 2 fuel cycle types.

Open fuel cycle, where spent fuels are stored at a final disposal facility.



Closed Fuel Cycle

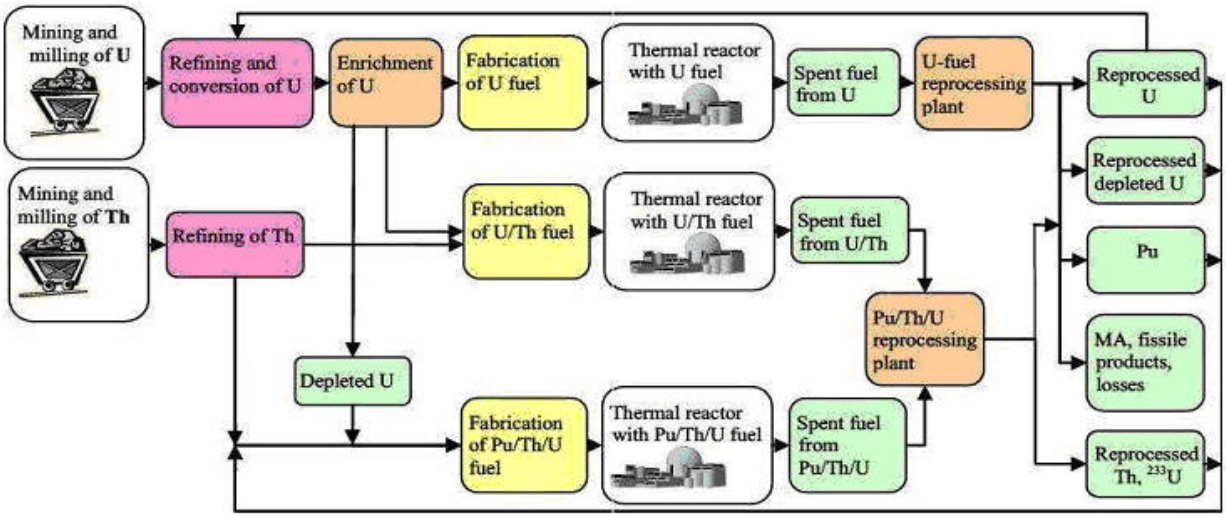
Closed fuel cycle, where spent fuels are sent to a reprocessing plant, and U and or Pu are recycled to reactors.



HLW:High Level radioactive Waste

Closed Fuel Cycle (Details)

This figure shows three fuel flows for closed fuel cycle.



Top line is a flow for uranium fuel, which is reprocessed at its end.
Middle line is a flow for thorium fuel.
Bottom line is a flow for recycled Pu/Th/U fuels.

2. Introduction of NFCSS

Features of Fuel Cycle Code NFCSS(1/3)

1) Fuel cycle code NFCSS (version-1) became open to public in 2007, and Version-2 became open to public in 2019. (It is “Free of charge”!)

2) NFCSS includes CAIN burnup engine with built-in nuclear cross section libraries, and does not need other design codes.

Purpose of NFCSS is;

To provide a fuel cycle evaluation tool for policy makers or institutions, where design codes are not available.

There are many fuel cycle codes in the world, but open codes (Free-of-charge) are rare, and most of them need design codes.

Features of Fuel Cycle Code NFCSS(2/3)

3) NFCSS can evaluate fuel cycle by calculating weights of 18 actinides in fuels, at loading, burn-up, and discharge stage, up to 200 years.

4) NFCSS can handle 8 reactor types, such as PWR, BWR, CANDU(PHWR), RBMK, VVER, GCR, AGR, including FBR.

5) Applicable fuel cycles are; UO_2 fuel cycle, MOX fuel cycle, and Thorium fuel cycle. (Currently, MOX and Th fuel can be applied to PWR&BWR.)

Th fuel can be [UO_2+ThO_2] or [PuO_2+ThO_2].

6) Minor Actinide fuel can be evaluated for PWR&BWR, such as [MA+UO_2] or [MA+MOX].

7) One reactor, one country or global fuel cycle with any combination of reactor types, can be evaluated.

8) Two fuel types, such as [UO_2fuel and MOX fuel] or [UO_2 fuel and ThO_2 fuel], can be loaded at one reactor. (currently for PWR&BWR.)

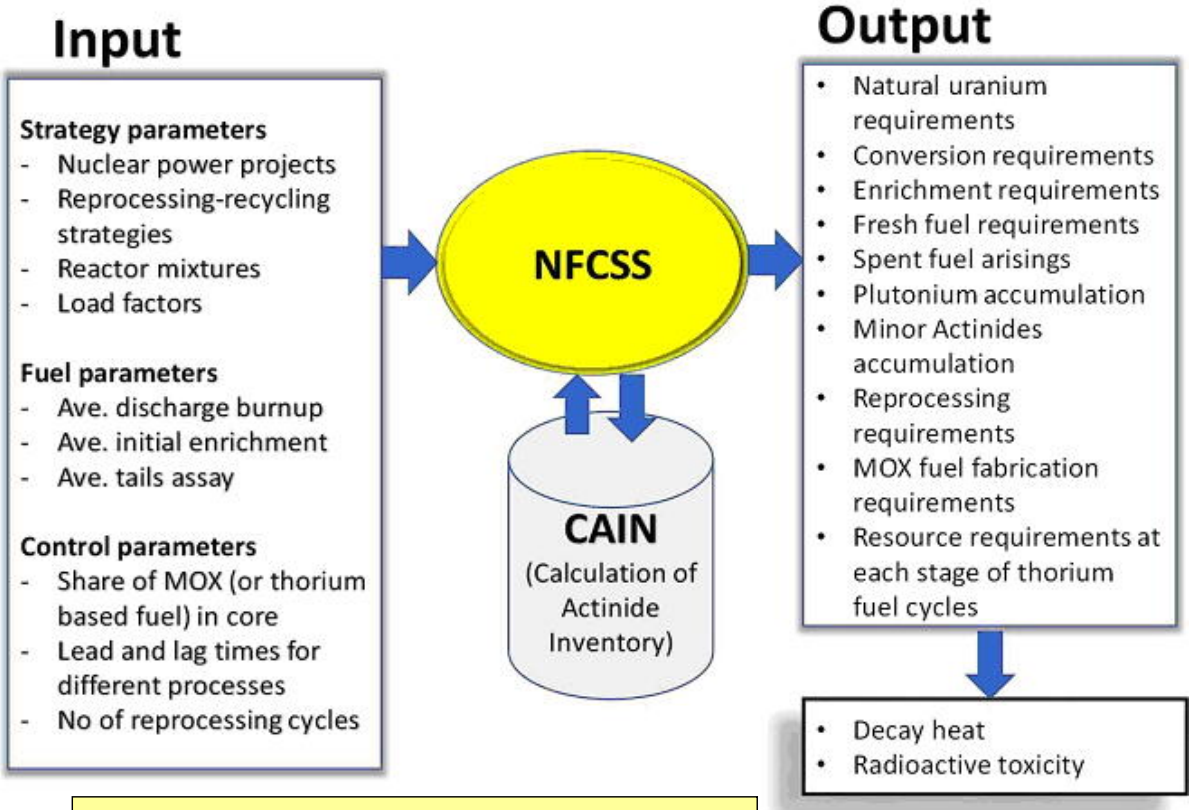
Features of Fuel Cycle Code NFCSS (3/3)

9) Decay heat and Radiotoxicity of discharged fuel can be calculated by a module in NFCSS, which utilizes NFCSS output (currently for PWR&BWR).

(Maybe only one fuel cycle code in the world, among open codes (Free-of-charge) without design codes?)

→Details in Section 5&6.

Configuration of NFCSS



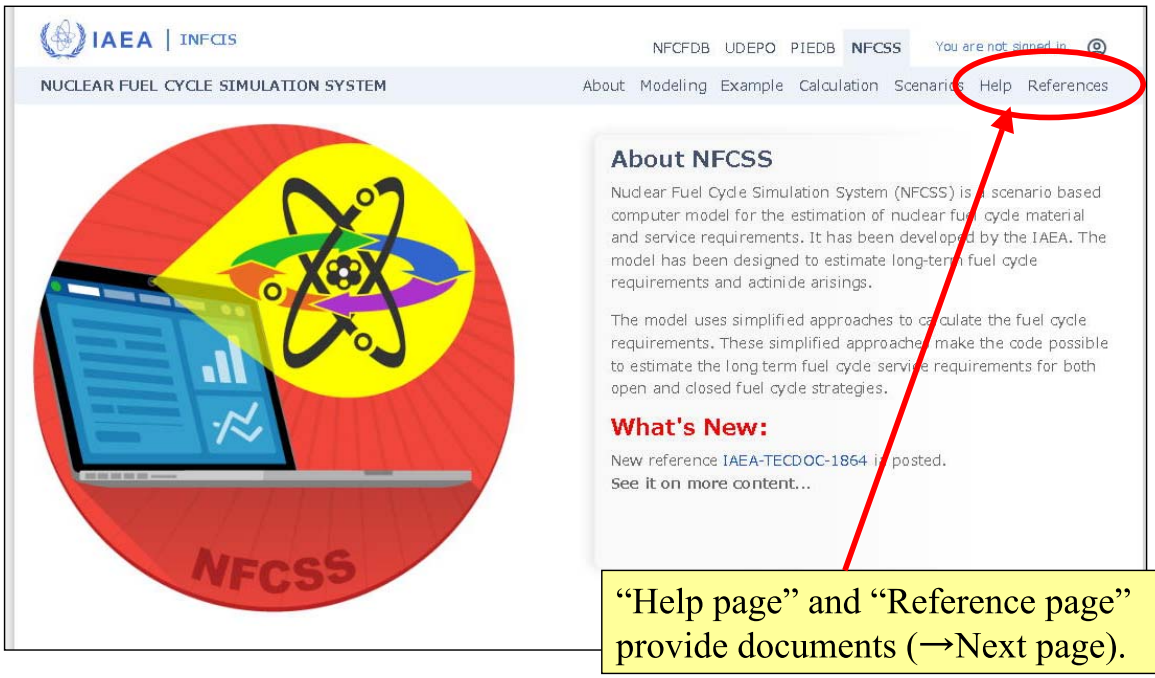
Each parameter is explained in next section.

3. Details of NFCSS

It is strongly recommended to watch previous Section-2 at first.

Gateway to NFCSS

NFCSS is available at IAEA site: (Free of charge!)
https://infcis.iaea.org/NFCSS/About



"Help page" and "Reference page" provide documents (→Next page).

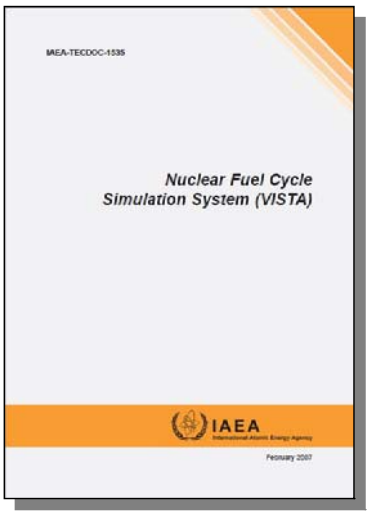
Documents for NFCSS

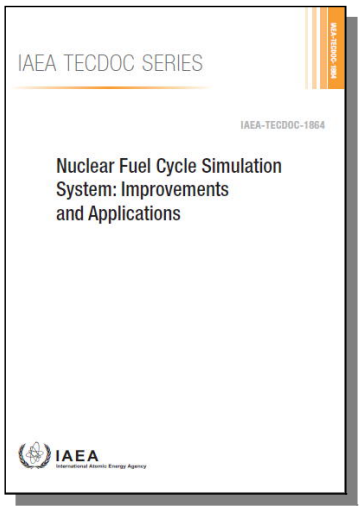
IAEA-TECDOC-1535, “Nuclear Fuel Cycle Simulation System (VISTA), 2007.

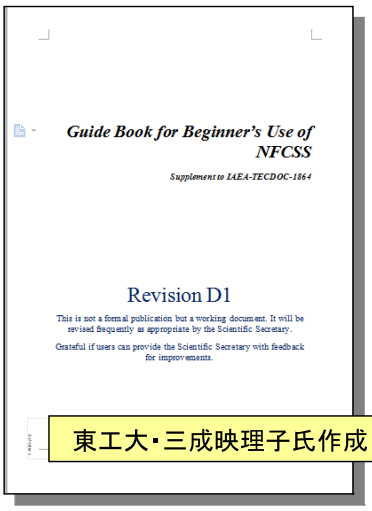
IAEA-TECDOC-1864, “Nuclear Fuel Cycle Simulations System: Improvements and Applications”, 2019. (including manual)

“Guide Book for Beginner’s Use of NFCSS”, 2019 (Revised in 2023)

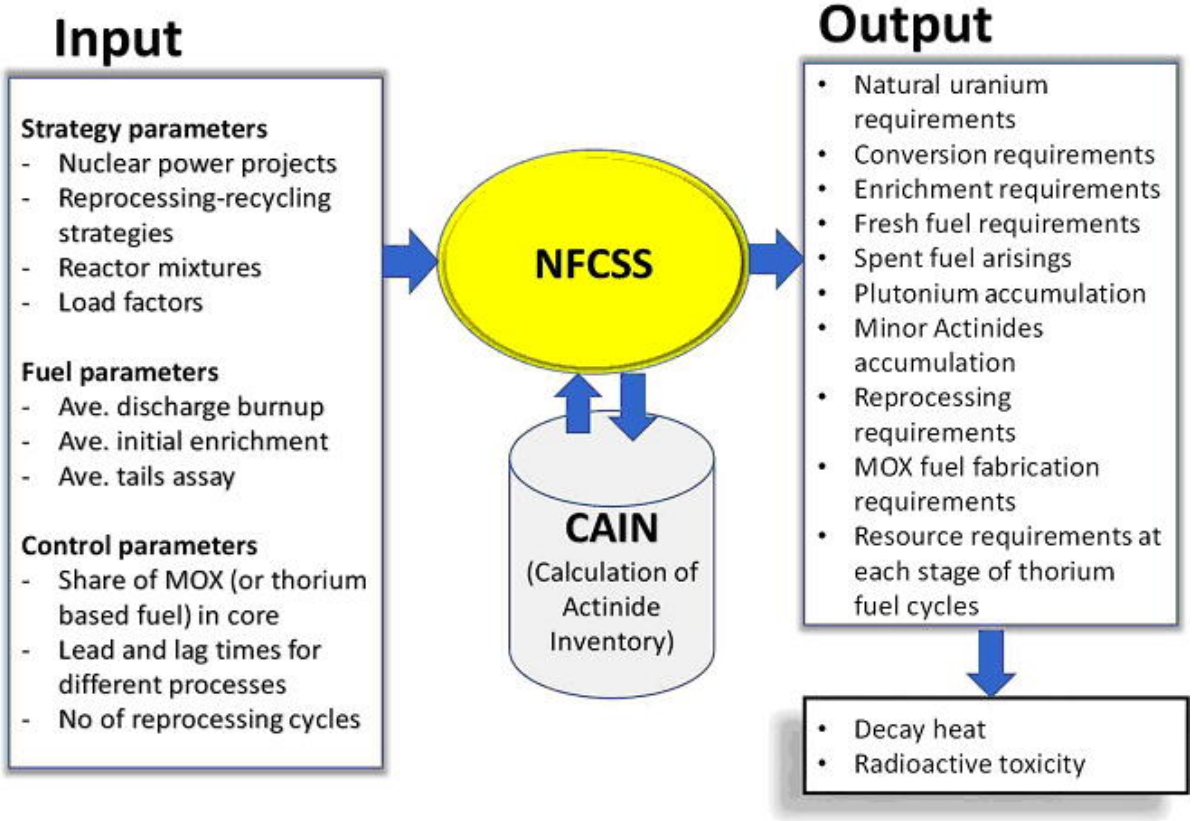
Other templates for Decay heat and Radiotoxicity.



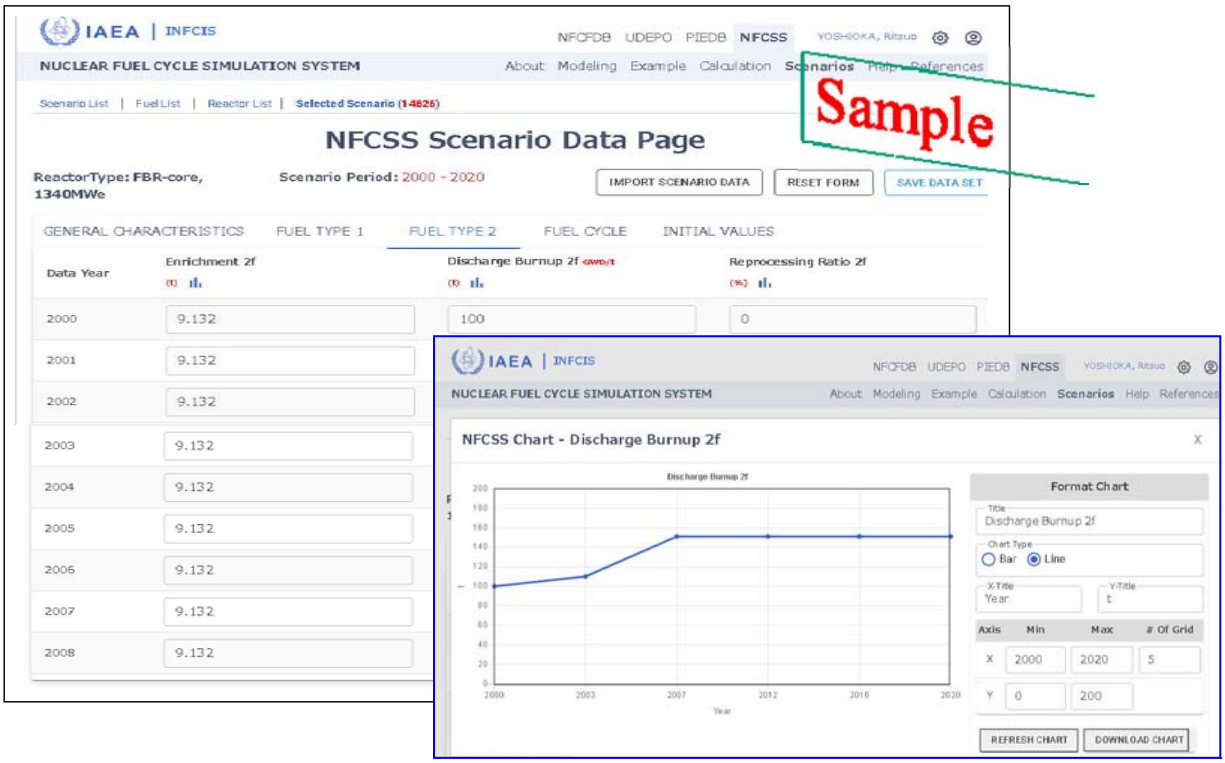




Input and Output of NFCSS

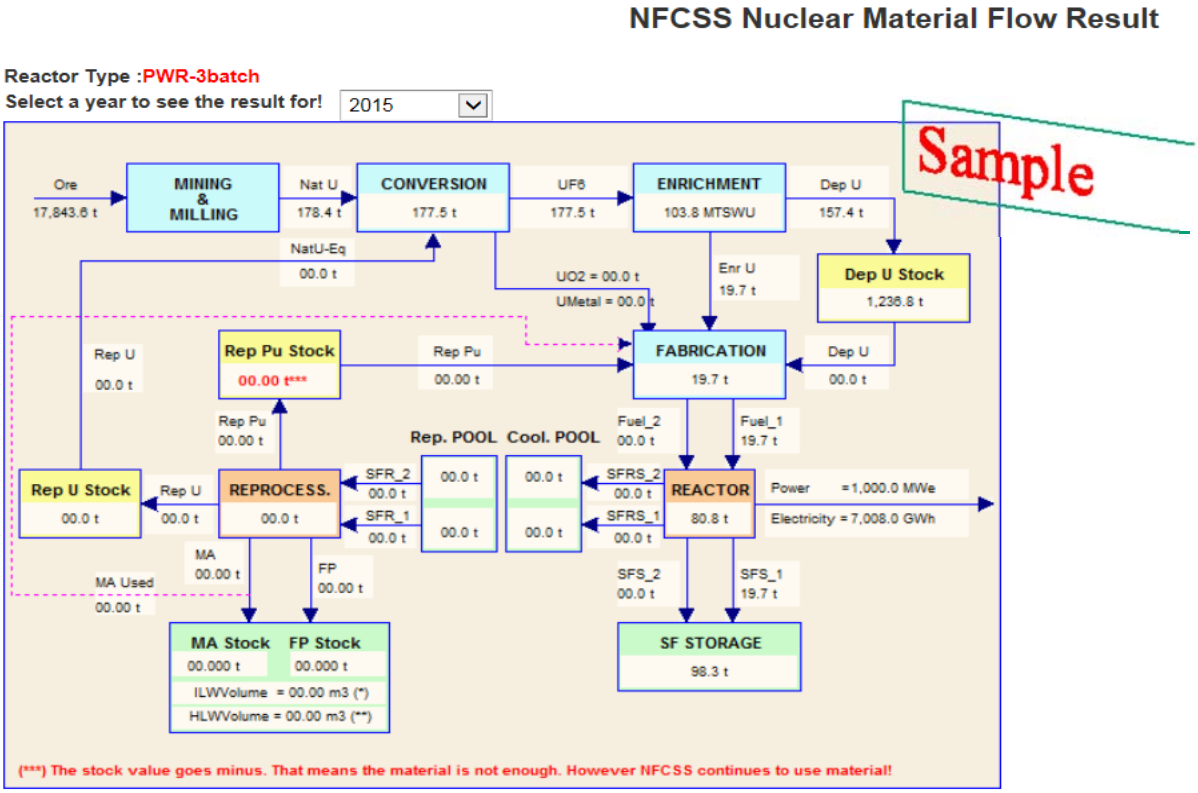


Sample Input of NFCSS



Inputs are shown in tables and in graphs.

Material Flow Diagram of NFCSS



Outputs are shown in figures, and.....

Material Balance in Excel Format

NFCSS Isotopic Composition Result

Sample

Reactor Type: PWR-3batch
Fuel: PWR UOX 1

Beginning of Cycle | End of Cycle | After Cooling

Export All Tables to MS Excel

Import Beginning of Cycle Table

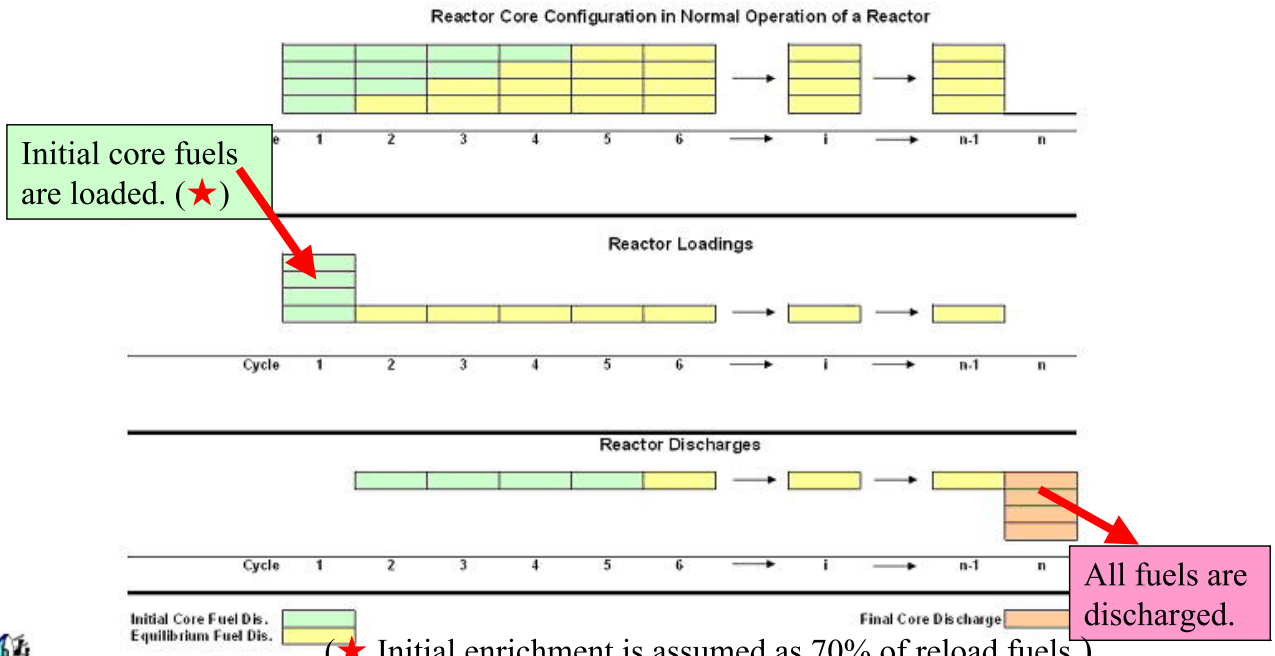
DataYear	Enrichment	Burnup	U235	U236	U238	Np237	Pu238	Pu239	Pu240	Pu241	Pu242	Am241	Am242m	Am243	Cm242	Cm244	TotPu	TotMA
2010	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2011	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2012	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2013	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2014	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2015	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2016	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2017	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2018	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2019	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2020	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2021	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2022	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2023	4.00000	45.00000	4.00000	0.00000	96.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

Outputs are shown in tables, and can be downloaded as Excel sheet.

Special Treatment at BOL & EOL

NFCSS assumes annual refueling, but initial core at Beginning-of-Life and decommissioning at End-of-Life are assumed as below.

See detail in p.171 of TECDOC-1864.



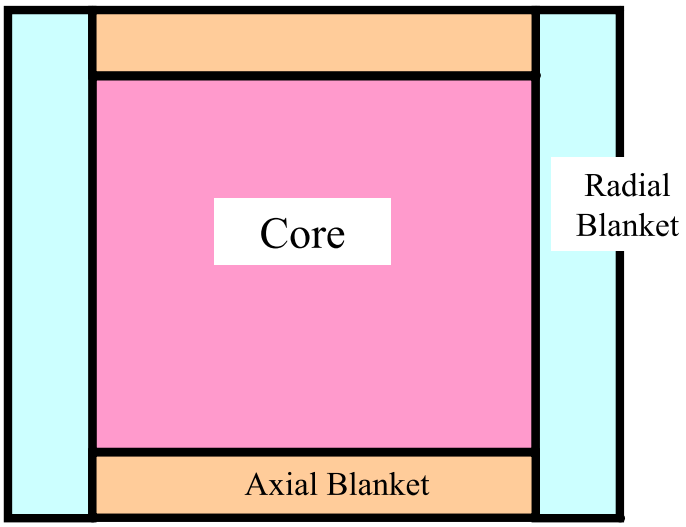
(★ Initial enrichment is assumed as 70% of reload fuels.)

Fast Reactor Simulation

NFCSS can handle a Fast Reactor (FR) by 3 separate reactors, such as Core, Axial blanket, and Radial blanket.

After executions of 3 reactor simulations, they are integrated as a whole FR simulation.

See detail in p.120 of TECDOC-1864



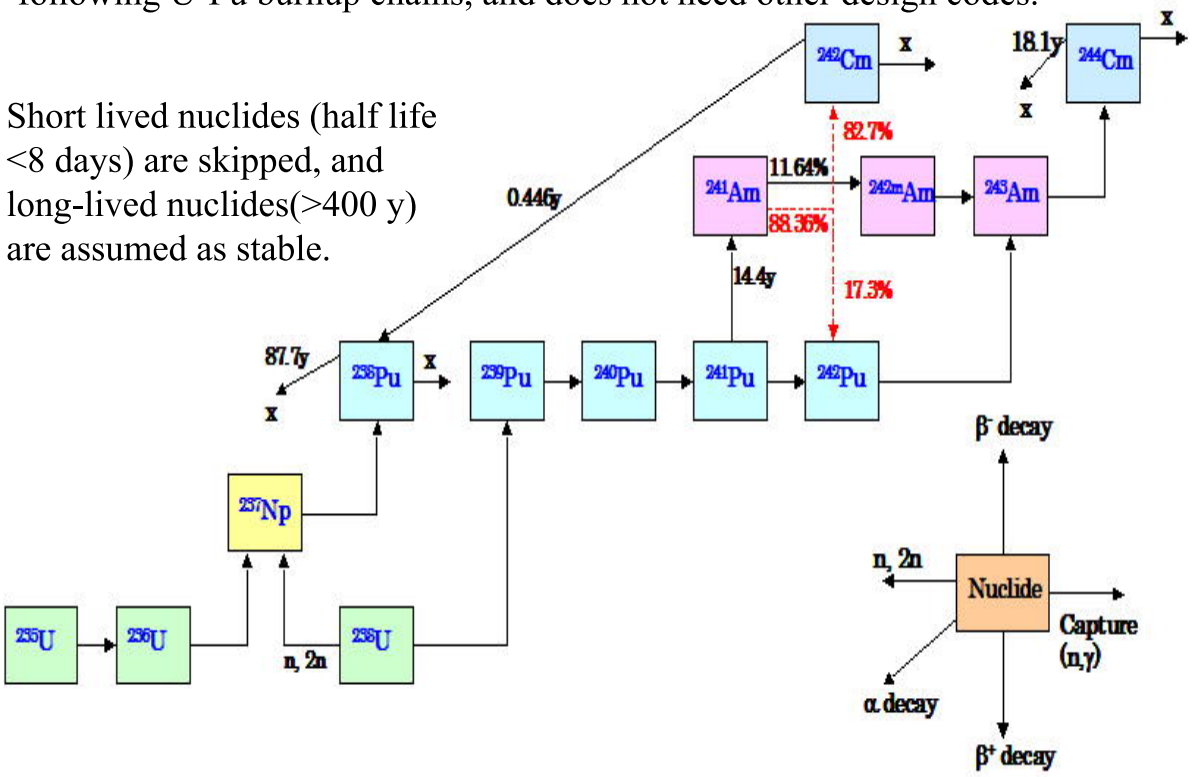
4. CAIN Burnup Engine

See detail in P.174 of TECDOC-1864

CAIN Burnup Chains for U-Pu cycle

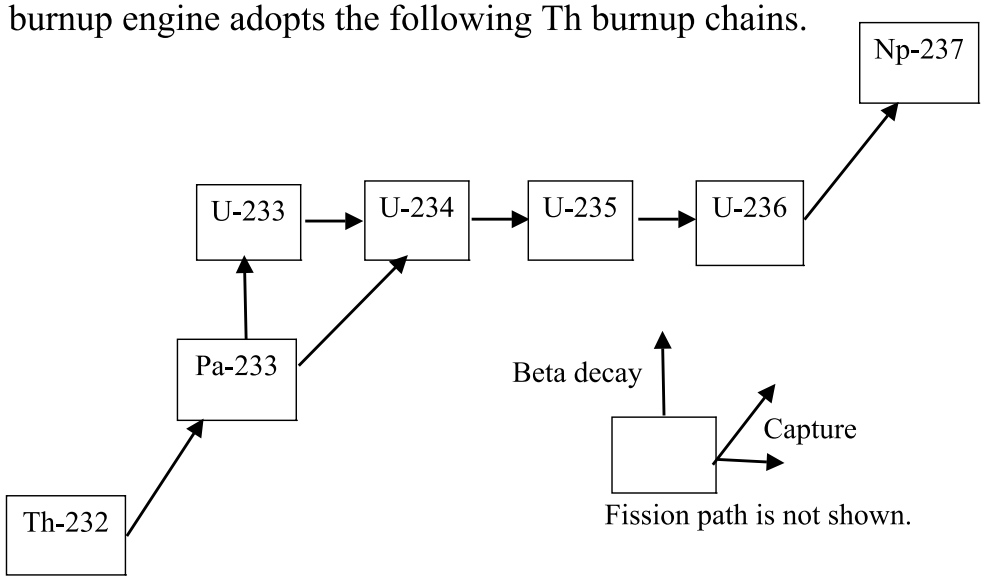
CAIN burnup engine with built-in nuclear cross section libraries adopts the following U-Pu burnup chains, and does not need other design codes.

Short lived nuclides (half life <8 days) are skipped, and long-lived nuclides(>400 y) are assumed as stable.



CAIN Burnup Chains for Th cycle

CAIN burnup engine adopts the following Th burnup chains.

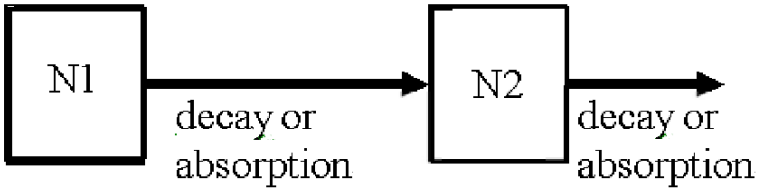


So, totally 18 nuclides are considered:

Th-232, Pa-233, U-233, U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Np-237, Am-241, Am-242m, Am-243, Cm-242 and Cm-244.

Bateman's Equation in CAIN (Decay only)

Most simple case, where only 2 nuclides exist, is shown below.



Bateman's equation of nuclide-1 is below:

$\mathbf{dN_1/dt = -\lambda_1 N_1}$

Solution of this equation is as follows.

$\mathbf{N_1 = N_1^0 \cdot exp(-\lambda_1 t)}$

N_1^0 : Atomic concentration of nuclide-1 at time zero,

λ_1 : Decay constant of nuclide-1 ($\lambda = 0.693/T_{1/2}$, where $T_{1/2}$ is half-life in sec),

t : Irradiation or cooling period (sec).

For nuclide-2, which is generated only by decay, equation is given as;

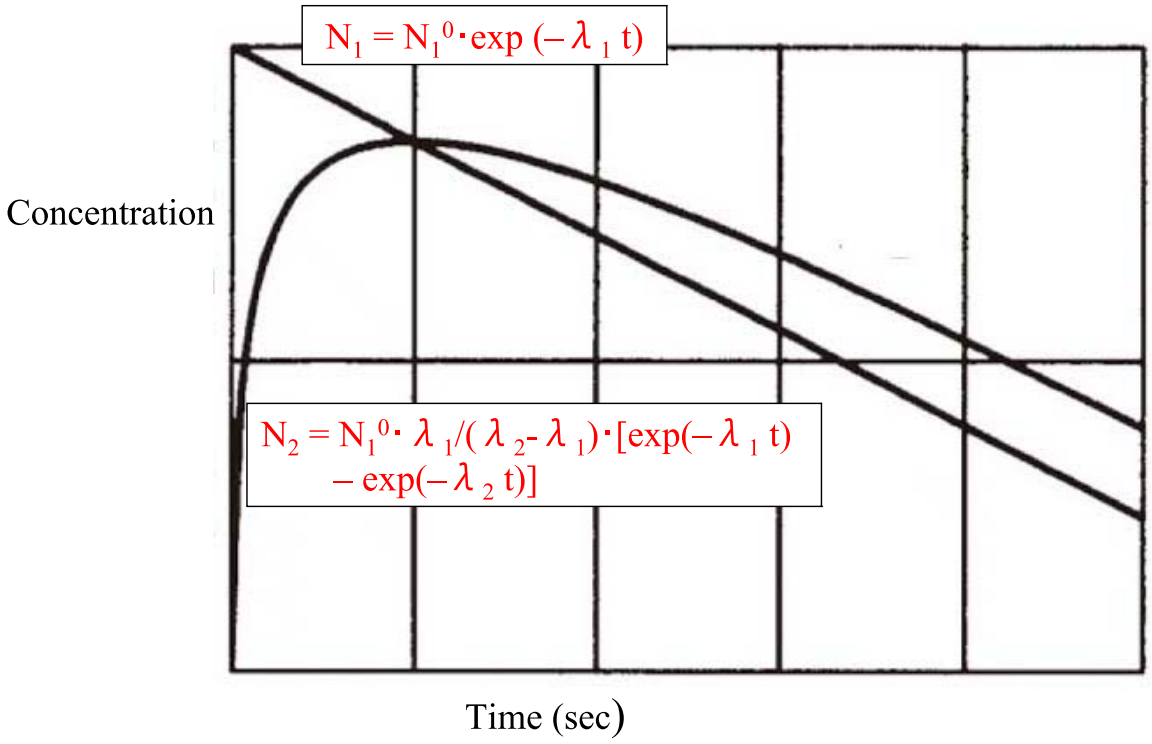
$\mathbf{dN_2/dt = -\lambda_2 N_2 + \lambda_1 N_1}$

Its solution is below.

$\mathbf{N_2 = N_2^0 \cdot exp(-\lambda_2 t) + N_1^0 \cdot \lambda_1/(\lambda_2 - \lambda_1) \cdot [exp(-\lambda_1 t) - exp(-\lambda_2 t)]}$

(Assuming initial $N_2 = 0$)

Solution of Bateman's Eq. (Decay only)



Bateman's Equation (Absorption only)

If transmutation is caused by neutron absorption, previous equation is modified, assuming constant cross section and constant neutron flux throughout burnup:

$\mathbf{dN_1/dt = -\sigma_1 \phi N_1}$

N_1 : Atomic concentration of nuclide-1,

σ_1 : Absorption cross section of nuclide-1,

10^{-24} cm^2(=1barn),

ϕ : Average neutron flux, n/cm^2/sec.

Solution of this equation is as follows.

$\mathbf{N_1 = N_1^0 \cdot exp\ (-\sigma_1 \phi\ t)}$

N_1^0 : Atomic concentration of nuclide-1 at time 0

t : Irradiation or cooling period (sec).

For nuclide-2, which is generated only by absorption, equation is given as;

$dN_2/dt = -\sigma_2 \phi N_2 + \sigma_1 \phi N_1$

σ_2: Absorption (fission+capture)

σ_1: Capture

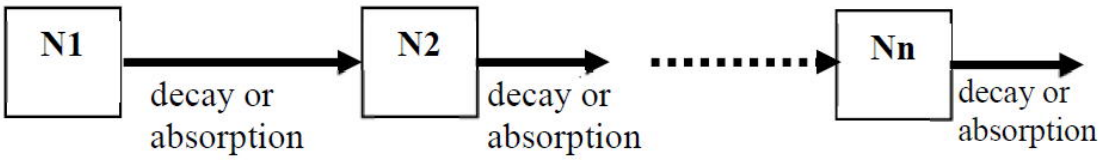
Its solution is below.

$N_2 = N_2^0 \cdot \exp(-\sigma_2 t) + N_1^0 \cdot \sigma_1/(\sigma_2 - \sigma_1) \cdot [\exp(-\sigma_1 \phi t) - \exp(-\sigma_2 \phi t)]$

(Assuming initial N_2 =0)

Generic Bateman's Equation

Generic burnup chain, Bateman's equation, and its solution are as follows.



$$\frac{dN_i}{dt} = -\sum_{j \neq i}\left[\lambda_{ji} + \sigma_{ji}^{tr}\varphi\right]N_i + \sum_{j \neq i}\left[\lambda_{ij} + \sigma_{ij}^{tr}\varphi\right]N_j$$

Decay

$$N_n = C_1 \exp(-\lambda_1 t) + C_2 \exp(-\lambda_2 t) \cdots + C_n \exp(-\lambda_n t)$$

$C_1 = N_1^0 \cdot [\lambda_1\lambda_2 \cdots \lambda_{n-1}] / [(\lambda_2-\lambda_1)\cdot(\lambda_3-\lambda_1)\cdots(\lambda_n-\lambda_1)]$

$C_2 = N_1^0 \cdot [\lambda_1\lambda_2 \cdots \lambda_{n-1}] / [(\lambda_1-\lambda_2)\cdot(\lambda_3-\lambda_2)\cdots(\lambda_n-\lambda_2)]$

$C_n =$

Actual solution is combination of these two.

Absorption

$$N_n = C_1 \exp(-\sigma_1 \phi t) + C_2 \exp(-\sigma_2 \phi t) \cdots + C_n \exp(-\sigma_n \phi t)$$

$C_1 = N_1^0 \cdot [\sigma_1\sigma_2 \cdots \sigma_{n-1}] / [(\sigma_2-\sigma_1)\cdot(\sigma_3-\sigma_1)\cdots(\sigma_n-\sigma_1)]$

$C_2 = N_1^0 \cdot [\sigma_1\sigma_2 \cdots \sigma_{n-1}] / [(\sigma_1-\sigma_2)\cdot(\sigma_3-\sigma_2)\cdots(\sigma_n-\sigma_2)]$

$C_n =$

Assumption to Solve Bateman's Equation

Bateman's equation can be solved analytically only cross sections and neutron flux are constant in burnup period.

Cross sections are provided as system library of NFCSS.

Reactor	UO_2 fuel	MOX fuel	ThO_2 fuel*1	MA-containing*2
PWR	○	○	○	○
BWR	○	○	○	○
PHWR	○			
RBMK	○			
VVER	○			
GCR	○			
AGR	○			
FBR-core		○		
FBR-blanket	○			

User can add any fuel library.

*1:[UO_2+ThO_2] or [PuO_2+ThO_2].

*2: [MA+UO_2] or [MA+MOX].

Neutron Flux

Constant neutron flux can be derived by the following formula.

$$\phi = SP / [1.602 \times 10^{-19} \times \Sigma (Ni \cdot \sigma i \cdot Ri)]$$

See p.183 of TECDOC-1864

SP : Specific Power (MW/ton),

φ: Neutron flux (n/cm2/sec),

Ni : Number of nuclide-i (in 1-ton fuel),

σi: Fission cross section of nuclide-i (10-24 cm2),

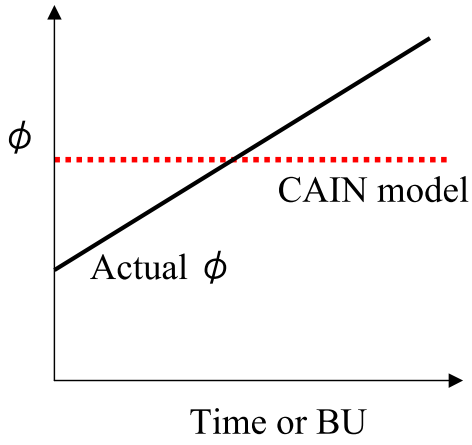
Ri: Fission energy of nuclide-i (typically 200 MeV/fission),

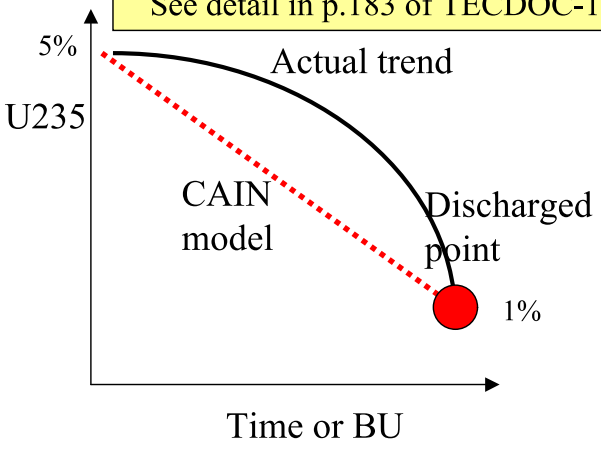
1.602×10^{-19}: Conversion factor from MeV/sec to MW.

Since this formula can be solved iteratively, users must input an initial guess of neutron flux.
And, NFCSS output is a converged value. (See p.131 of TECDOC-1864)

Why is Constant Neutron Flux OK?

See detail in p.183 of TECDOC-1864





Since neutron flux (ϕ) is smaller at BOC, U235 decreases slowly at first.

Then, ϕ becomes larger at EOC, and U235 decreases rapidly.

There is a principle: **1 gram fission ≒ 1MWD**. (→40GWd/t = 4%fission)

For example of 40 GWd/t fuel case, 5% U235 at BOC decreases to 1% at EOC, regardless of trajectory.

Only discharged point is used in fuel cycle, constant neutron flux is justified.

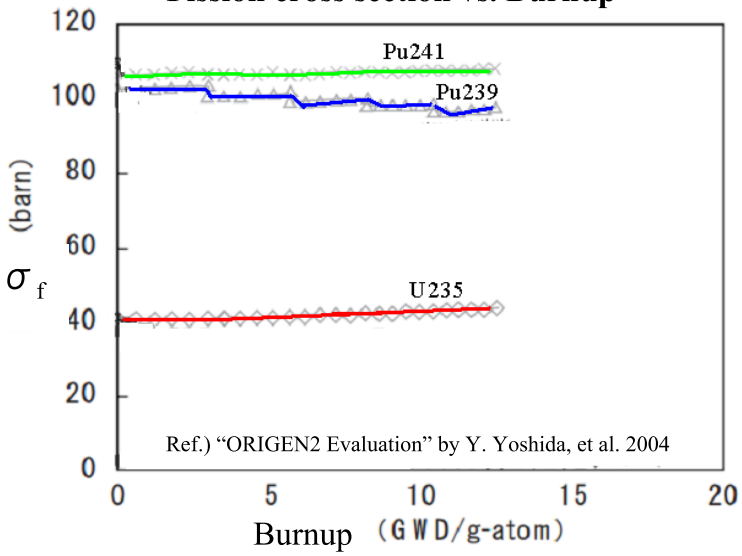
Why is Constant Cross Section OK?

ORIGEN-2 provides burnup dependency of fission cross sections (σ_f).

When fissile material decreases by burnup, neutron spectrum becomes softer, and it affects an average σ_f slightly.

But, its effect is small, and σ_f can be treated as constant, as is shown here.

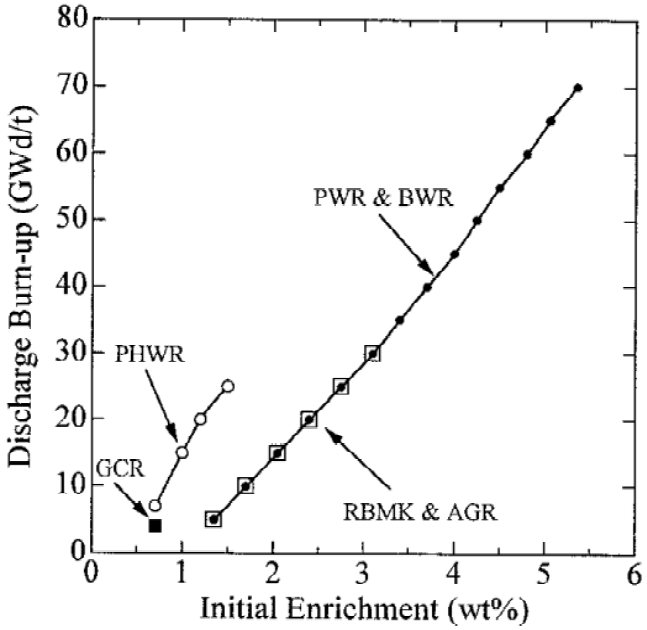
Fission cross section vs. Burnup



Initial Enrichment of UO_2 fuel

In order to input the enrichment of UO_2 fuel, a detail design code is required.

So, for users who do not own design codes, NFCSS provides the relation curve for specified discharge burnup.



If users do not specify enrichment, then this curve is used in NFCSS.

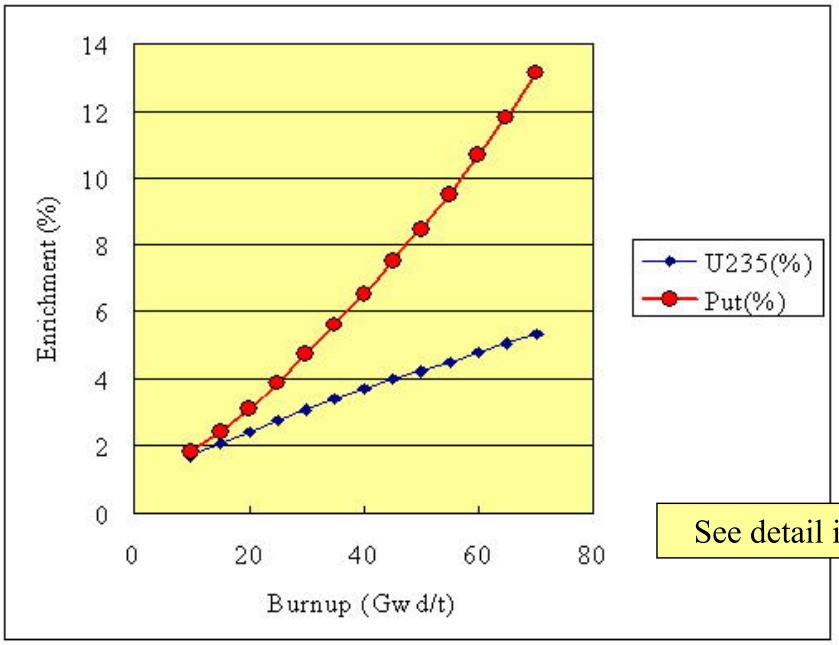
Of course, users can input enrichment if they have confident value by design codes.

See detail in p.182 of TECDOC-1864

Initial Fissile Content of MOX fuel

As for MOX fuel, the above UO_2 process is not provided.

So, for users who do not have design codes, the following curve is recommended.



See detail in p.182 of TECDOC-1864

Verification of CAIN & NFCSS

1) Burnup calculations by CAIN are verified by ORIGEN-2, and its accuracy is 1-2%.

Also, it is verified by other design codes.

2) Burnup calculations for UO_2 fuels are verified by the actual measurements for PWR and BWR fuels.

3) Material balance (Material flow diagram) is verified by other fuel cycle codes.

4) Details are shown in TECDOC-1535 and TECDOC-1864.

5. Decay Heat Calculation

See detail in Chapter-6 of TECDOC-1864

Introduction of Decay heat

Decay heat is one of important parameters for safety assessments in reactor accident analysis and spent fuel management.

Decay heat is generated due to the decay of actinides and fission products (FP), and thus it depends on fuel types/burnup and so on.

NFCSS is a unique fuel cycle code, where decay heat calculation of discharged fuels is integrated.

At this moment, decay heat can be calculated to PWR and BWR for UO_2 fuel, MOX fuel, and ThO_2-fuel.

Calculation Method for Decay Heat

ORIGEN-2 code was applied with original libraries from ORNL

[Ref.] A. G. Croff, A Users Manual for the ORIGEN-2 Computer Code, ORNL/TM-7175, 1980

U-PWR-core: 4wt%U235＋96wt%U238.

Th-PWR-core: 4wt%U233+96wt%Th232.

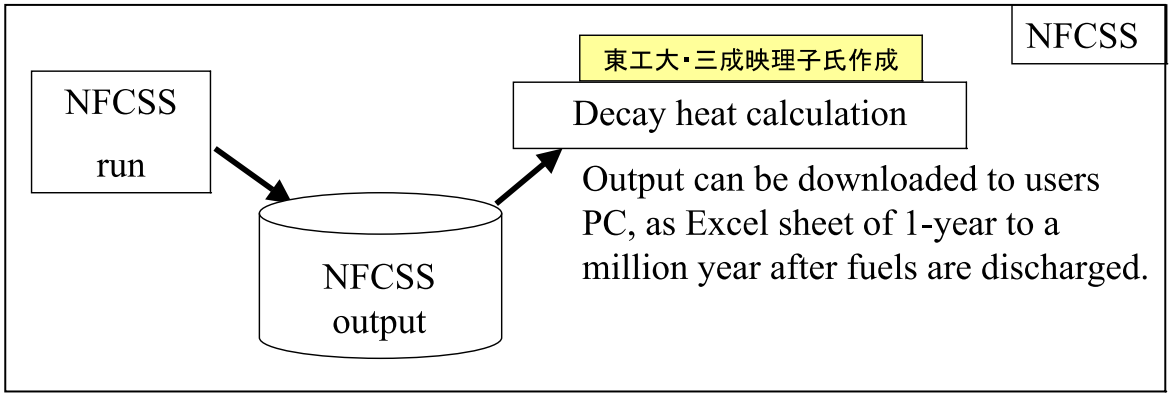
MOX-PWR-core: 7.50wt% Pu-total (019%Pu238, 4.10%Pu239, 1.96%Pu240, 0.71%Pu241, 0.54%Pu242, 92.50%U238)

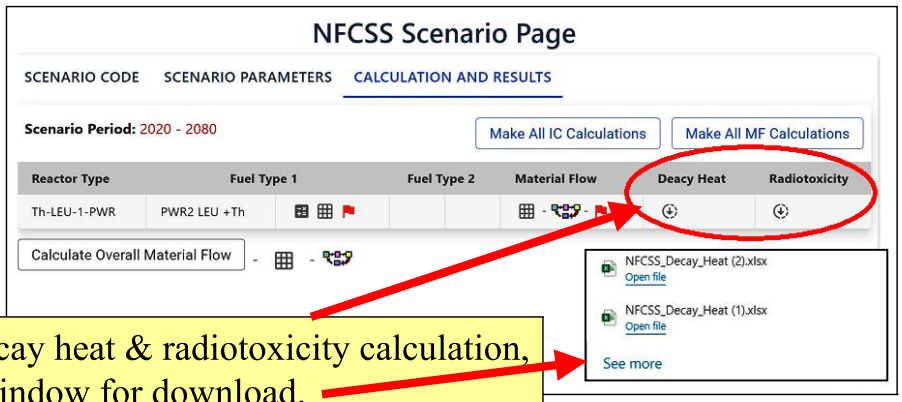
Rated power density ＝39.2MW/T

Output of decay heat by ORIGEN-2 is “Watt/tonHM” (HM:heavy metal weight).

Fuel is burnt up to 45GWD/t, and then decay is calculated from 1-sec to 1-million years.

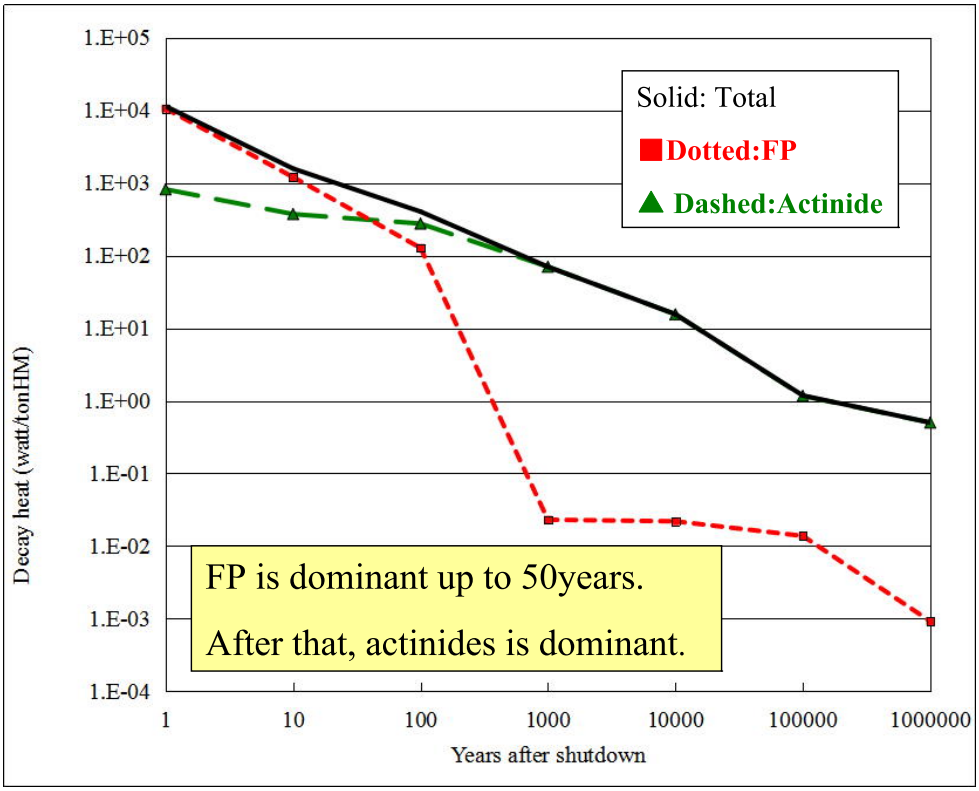
Decay Heat Module





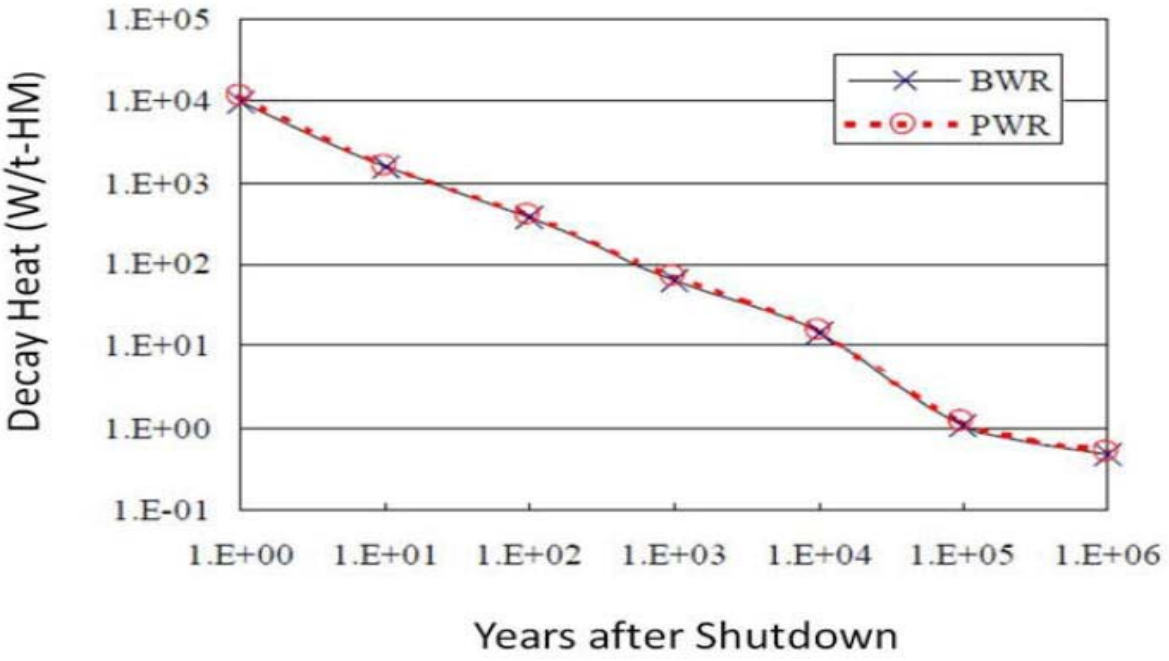
Buttons for decay heat & radiotoxicity calculation, with pop-up window for download.

Component of Decay Heat (U-core)



PWR vs BWR

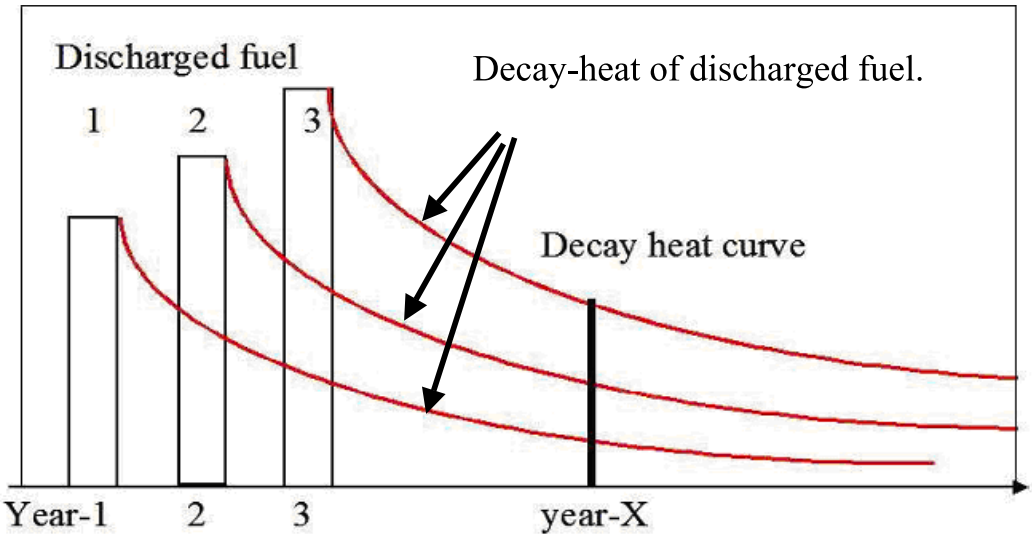
Since decay heat (or radiotoxicity) depends on the amount of fissions per weight of Heavy Metal (HM), PWR and BWR give almost identical results.



Accumulation for All Discharged Fuels

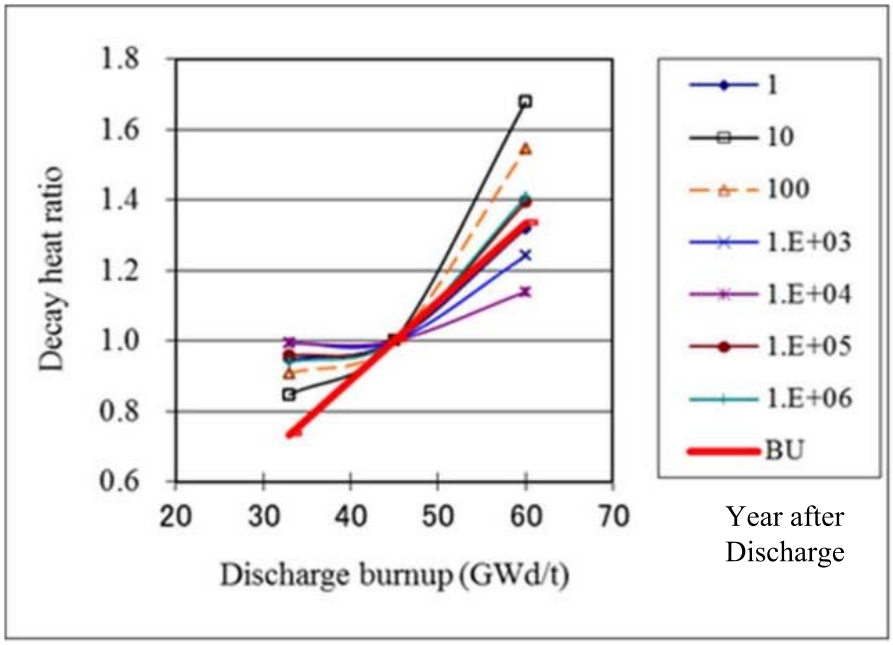
Decay-heat of discharged fuel at a certain year must be accumulated at a disposal facility.

Therefore, decay heat of all discharged fuels must be added at a certain year-X.



One Assumption in Calculation

Decay-heat of discharged fuel are stored only at a standard burnup of 45GWd/t.
Therefore, it is assumed that they are linearly proportional to discharged burnup.

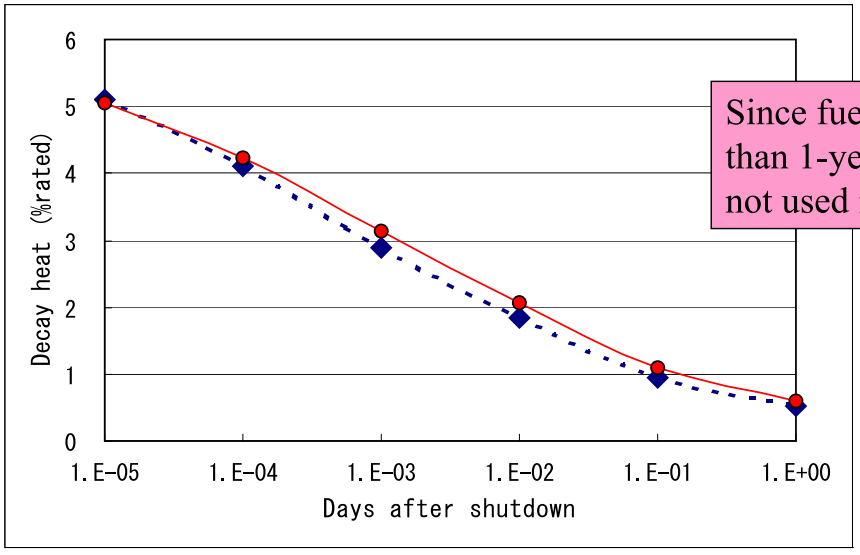


Decay Heat for Short Range

(% Rated vs. Days)

Dotted: U-core(U219-library)

Solid:Th-core(Th213-library)

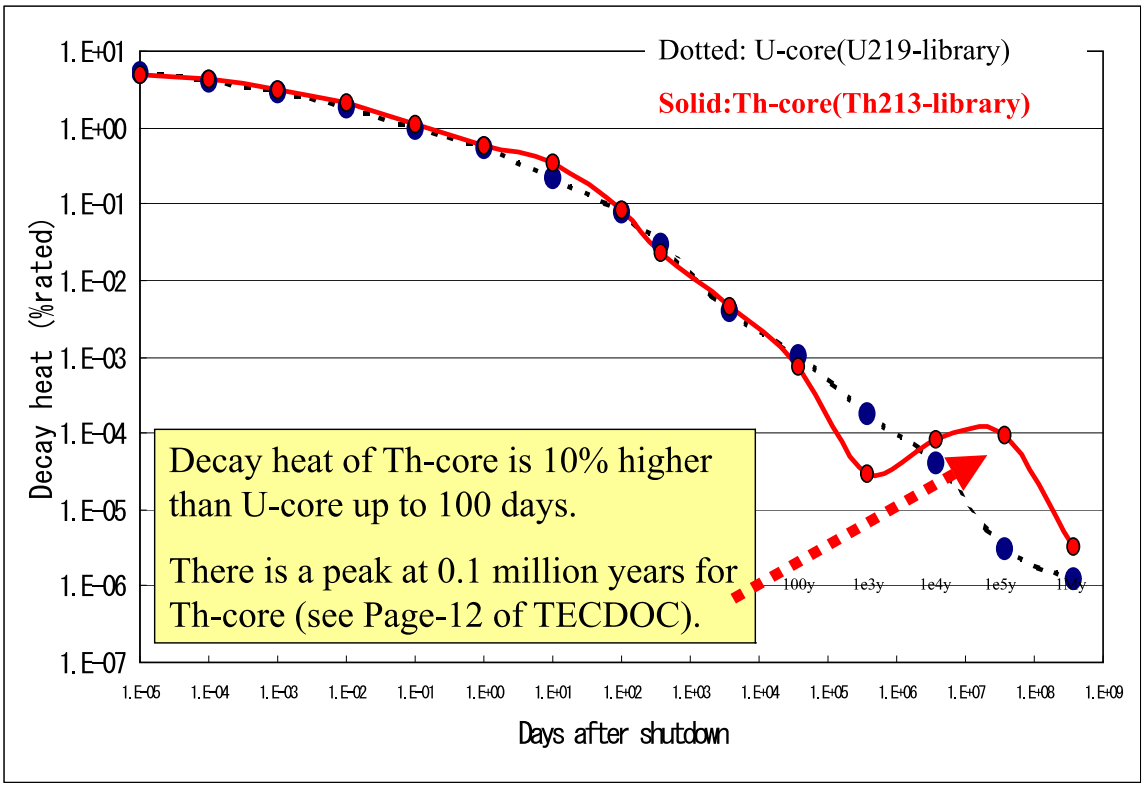


Since fuel cycle is longer than 1-year, these data are not used in NFCSS.

Th-core is 10% higher than U-core up to 1day, where is important for accident analysis.

Decay Heat for Long Range

(% Rated vs. Days)



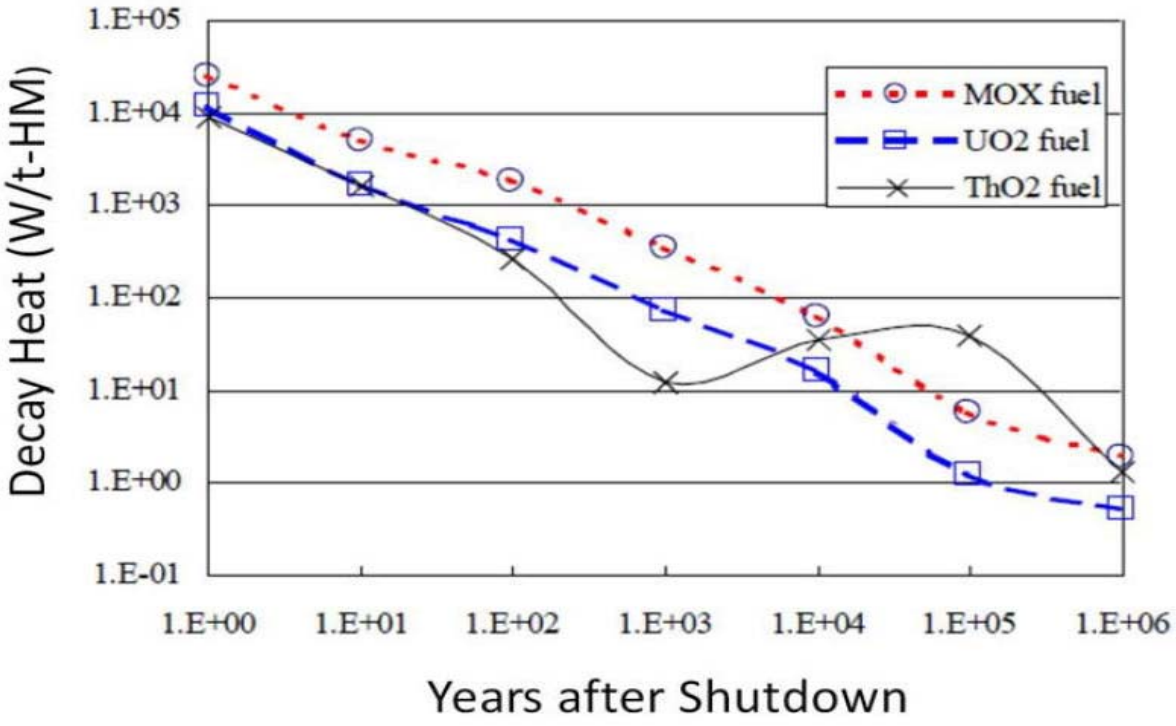
Verification of Decay Heat

1) Decay heat calculations by NFCSS are verified by experiments and other preceding papers.

2) Details are shown in TECDOC-1864.

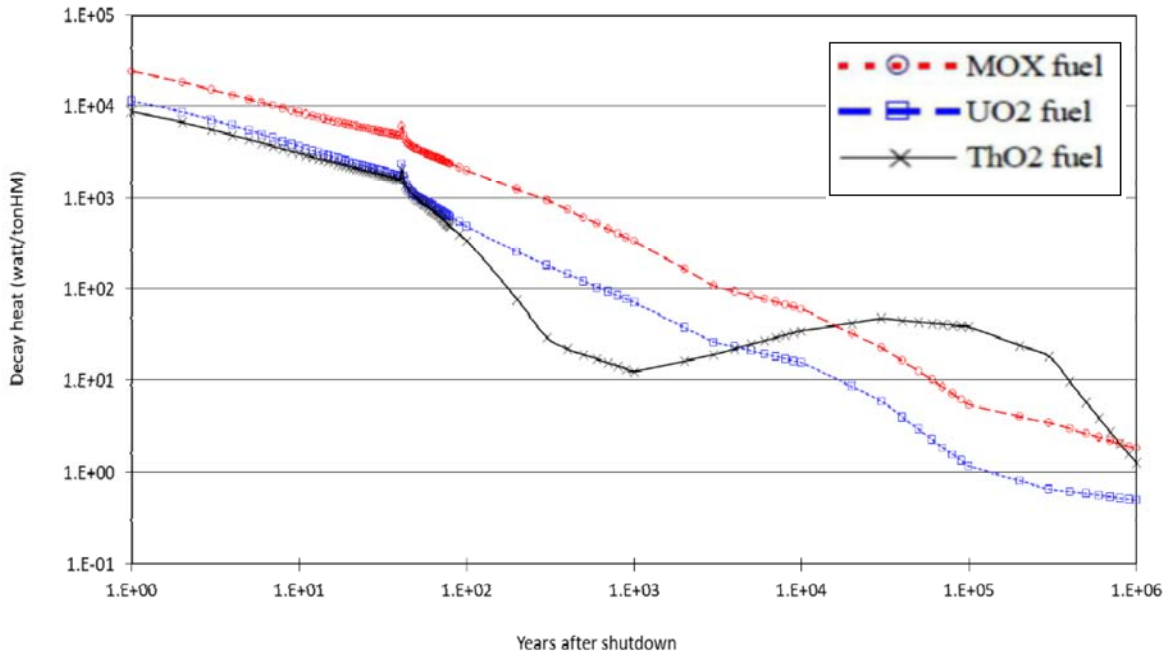
Decay Heat of Onetime Discharge

Decay heat of UO_2/ThO_2/MOX fuel (Watt/t-HM) after discharge, for 45GWd/t burnup fuel in 1GWe-PWR (or BWR).



Accumulated Decay Heat

Accumulated decay heat of UO_2/ThO_2/MOX fuel (Watt/t-HM) for 40-times discharge, after 40-years operation of 1GWe-PWR(or BWR).



Decay heat of MOX is larger than UO_2, because actinides contribution is dominant at these period, and there are more Pu and MA(Minor Actinide) in spent MOX fuel.

6. Radiotoxicity Calculation

See detail in Chapter-7 of TECDOC-1864

Introduction of Radiotoxicity

Radiotoxicity is the required information for managing spent fuel in temporary storage facilities or in geological repositories for permanent disposal.

Since radiotoxicity is caused by the decay of actinides and FPs, and thus it depends on fuel types/burnup and so on.

NFCSS is a unique fuel cycle code, where radiotoxicity calculation of discharged fuels is integrated.

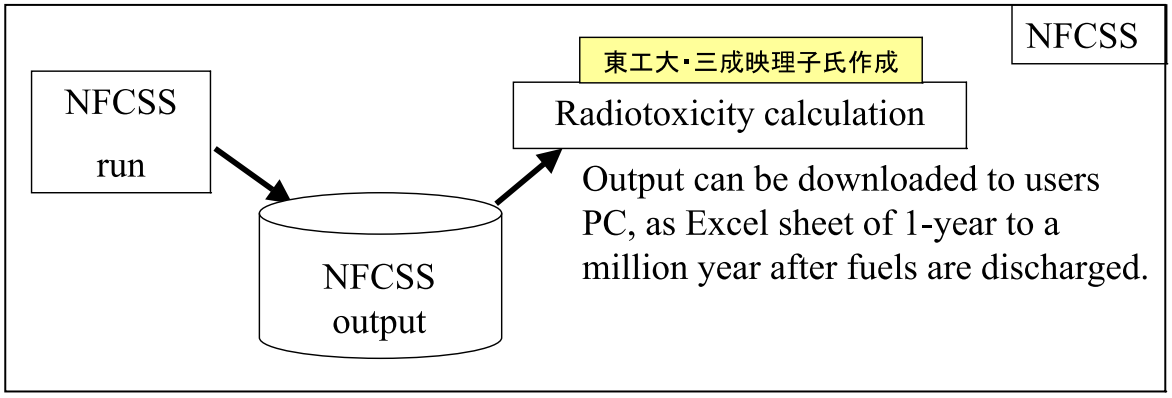
At this moment, radiotoxicity can be calculated on PWR and BWR for UO_2 fuel, MOX fuel, and ThO_2-fuel.

Radiotoxicity is an index of toxicity, when a nuclide is taken into a human body and causes **internal radiation exposure**. (That is, external exposure is not considered.)

As paths for internal expose, **ingestion** through mouth by drinking radioactive water etc. is considered. (That is, inhalation by breathing is not considered.)



Radiotoxicity Module



Same as decay heat calculation.

Calculation Method for Radiotoxicity

ORIGEN-2 code was applied with original libraries from ORNL

[Ref.] A. G. Croff, A Users Manual for the ORIGEN-2 Computer Code, ORNL/TM-7175, 1980

U-PWR-core: 4wt%U235＋96wt%U238.

Th-PWR-core: 4wt%U233+96wt%Th232.

MOX-PWR-core: 7.50wt% Pu-total (019%Pu238, 4.10%Pu239, 1.96%Pu240, 0.71%Pu241, 0.54%Pu242, 92.50%U238)

Fuel is burnt up to 45GWD/t, and then radioactivity is calculated from 1-sec to 1-million years.

Since ORIGEN-2 provides **Radioactivity**, it must be converted to **Radiotoxicity** (in Sievert) by the following formula.

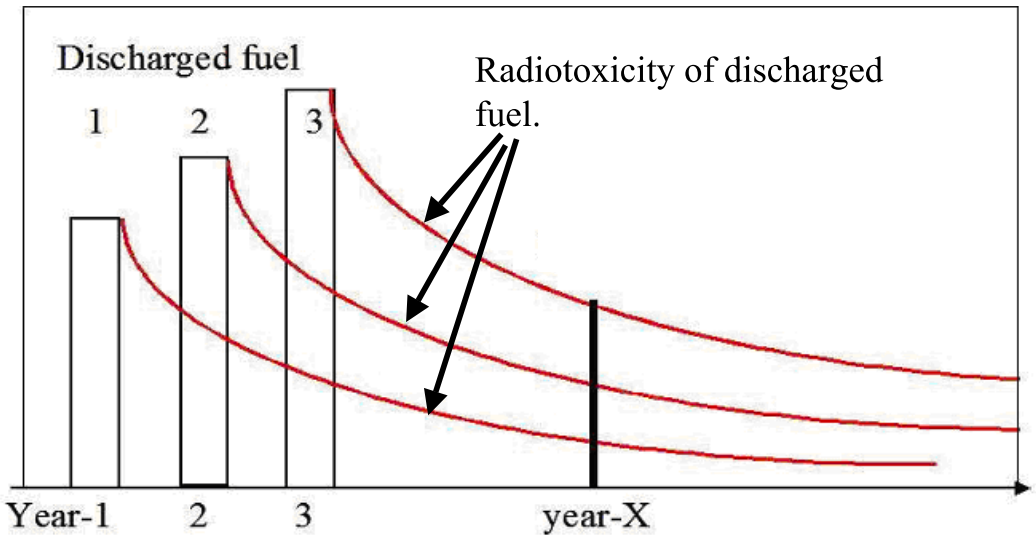
Radiotoxicity [Sv] = Radioactivity [Bq] * DCF [Sv/Bq]

Here, Radioactivity [Bq]= Radioactivity [Curies] x 3.7^{10} , and

DCF= Dose Conversion Factor (or Dose Coefficient), which are shown in the references.

Accumulation of Discharged Fuels

Radiotoxicity of discharged fuel at a certain year must be accumulated at a disposal facility.
Therefore, they must be added at a certain year-X.

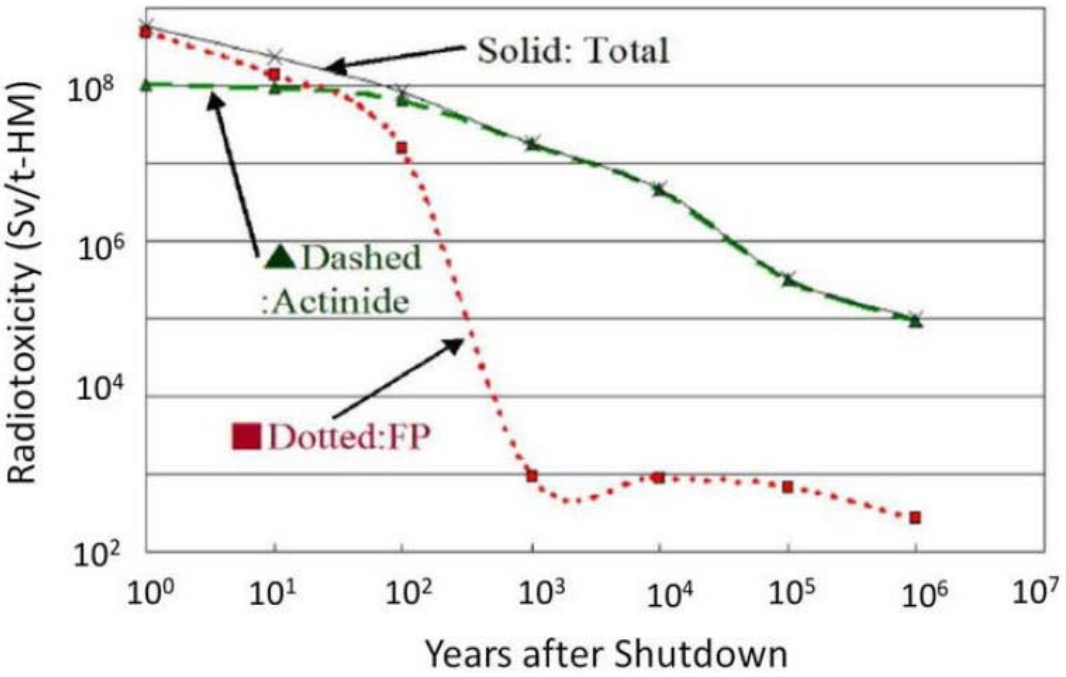


Same as decay heat calculation.

Component of Radiotoxicity

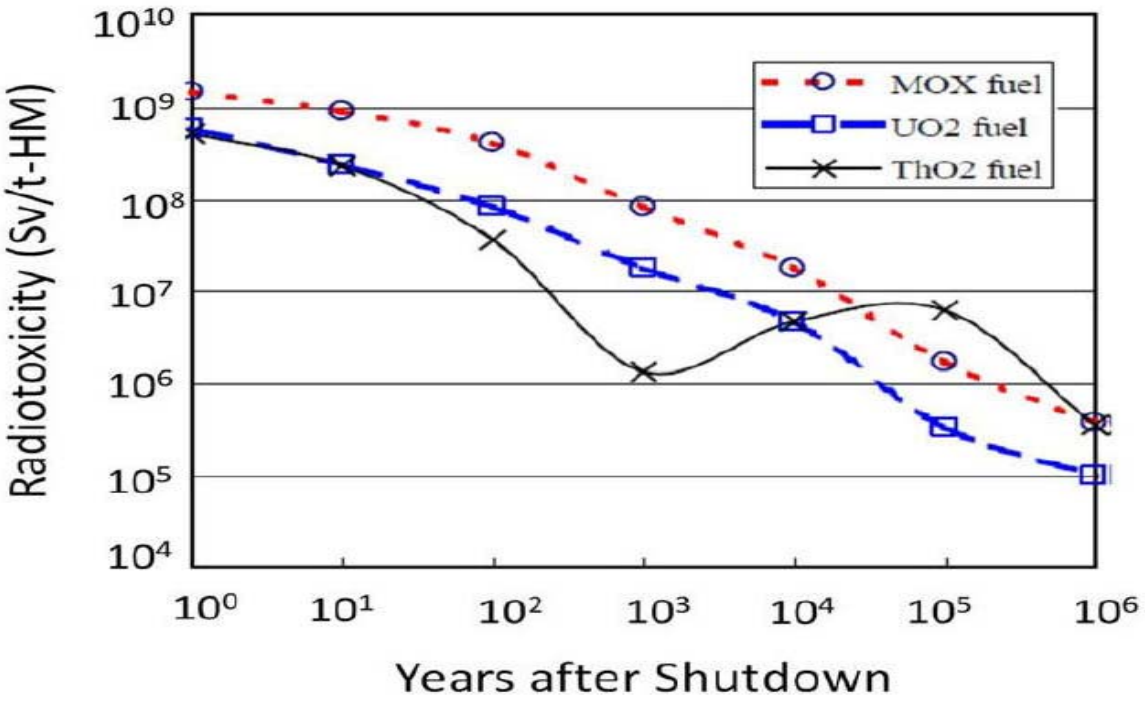
FP is dominant up to 50years.

After that, actinides is dominant.



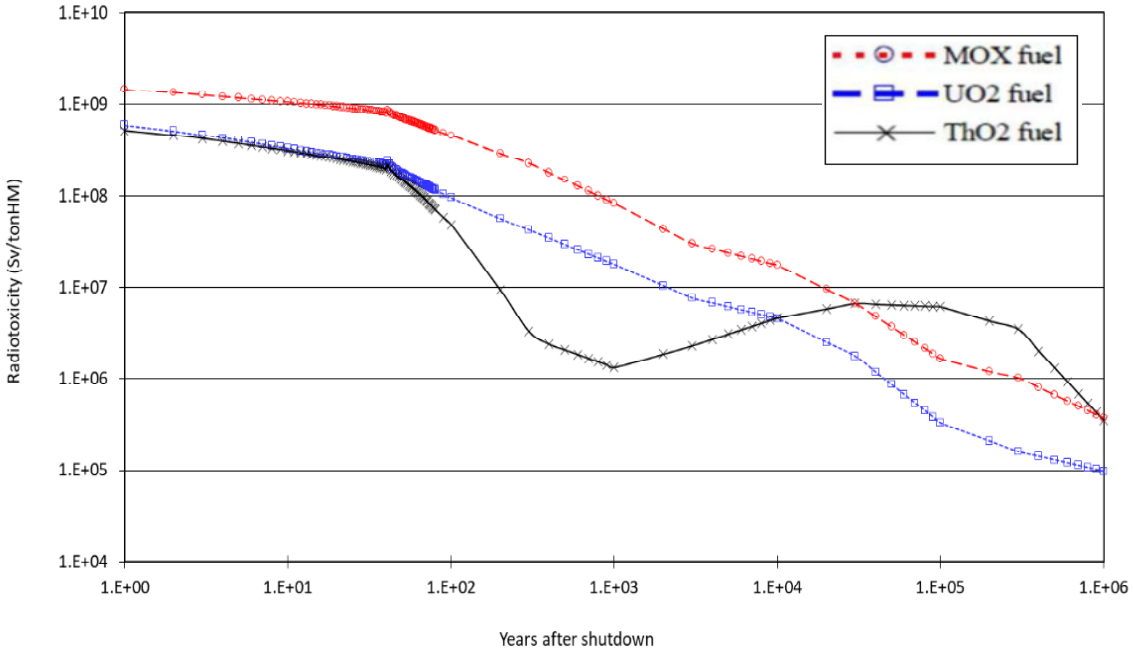
Radiotoxicity of Onetime Discharge

Radiotoxicity of UO_2/ThO_2/MOX fuel (Sv/t-HM), after discharge, for 45GWd/t burnup fuel in 1GWe-PWR (or BWR).



Accumulated Radiotoxicity

Accumulated radiotoxicity of UO_2/ThO_2/MOX fuel (Sv/t-HM) for 40-times discharge, after 40-years operation of 1GWe-PWR(or BWR).



Radiotoxicity of MOX is higher than UO_2, because actinides contribution is dominant at these period, and there are more Pu and MA(Minor Actinide) in spent MOX fuel.

Verification of Radiotoxicity

1) Radiotoxicity calculations by NFCSS are verified by preceding papers.

2) Details are shown in TECDOC-1864.

7. Relation to Deep Geological Repository

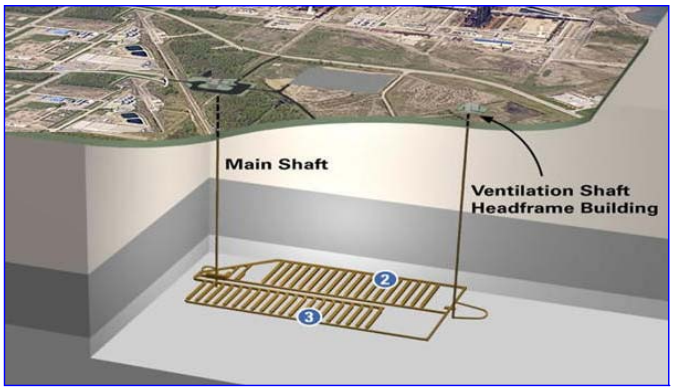
Deep Geological Repository (DGR)

Deep Geological Repository (DGR) is an excavated underground facility that is designed for safe and secure permanent disposal of radioactive waste. (by NRC)

There are 2 types DGR as follows;

(1) Direct disposal of spent nuclear fuels (open fuel cycle).

(2) Disposal of vitrified waste from spent fuel reprocessing (closed fuel cycle)



Details of DGR is shown in references, such as ["Planning and Design Considerations for Geological Repository Programmes of Radioactive Waste", IAEA-TECDOC-1755 , 2014].

Data from NFCSS(1/2)

This tutorial is not to provide knowledge on whole DGR, but only provide what kind of data can be provided by the fuel cycle code NFCSS for DGR study.

These data are as follows;

(1) Open fuel cycle case

Annual amount of discharged spent fuel (ton)

Actinide concentration of spent fuel (wt%)

Decay heat of spent fuel (1 year to 1 million year, after discharge)

Radiotoxicity of spent fuel (ditto)



Since NFCSS provides discharged weight, it is required to divide it by [weight/assembly] to get discharged assemblies.

Data from NFCSS(2/2)

(2) Closed fuel cycle case

Annual amount of discharged spent fuel (ton)

Actinide concentration of spent fuel (wt%)

Reprocessing amount

Recycled Pu amount

Stored MA amount

Stored HLW amount

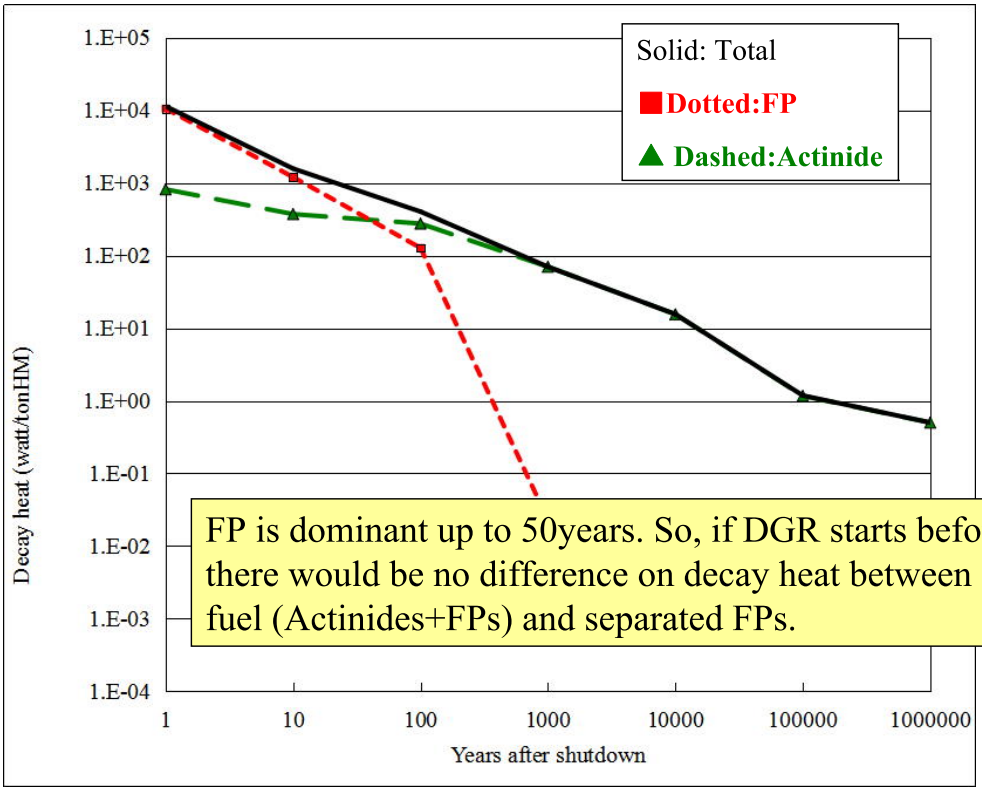
The following 2 data for MOX fuel.

Decay heat of spent fuel (1 year to 1 million year, after discharge)

Radiotoxicity of spent fuel (ditto)

Component of Decay Heat

Decay heat is an important parameter for DGR design.



FP is dominant up to 50years. So, if DGR starts before that, there would be no difference on decay heat between spent fuel (Actinides+FPs) and separated FPs.

Thank you for your attention!
Any questions/comments?



談話室

IAEAの核燃料サイクル計算コードVISTA

日本システム安全研究所　吉岡律夫

国際原子力機関（IAEA）の核燃料サイクル課では，核燃料サイクル計算コードVISTAを開発しているが，このほど試用版が完成したので，以下にご紹介する。同課では，核燃料サイクルに関連した下記４件のデータベースを提供している。このうち，本稿で紹介するVISTAはテスト用のアドレス（http://www-nfcistest.iaea.org/default_1.asp）であるが，その他は公式ホームページ(http://www-nfcis.iaea.org/）に公開されていて，いずれも，名前などを登録すれば，誰でも自由に利用できる。

①NFCIS	核燃料サイクル施設のデータベース
②UDEPO	ウラン鉱石資源のデータベース
③PIE	照射後試験施設のデータベース
④VISTA	核燃料サイクル計算コード
PRIS	発電炉設計情報と発電量実績データベース

（注：PRISは核燃料サイクル課の所掌ではないが，VISTAホームページにリンクがある。）

VISTA

もともと，VISTAは，大井昇氏が核燃料サイクル課長在任時の1995年頃に開発を指示され，筆者など数カ国の炉心設計技術者が開発に協力してきたシステムである。当初は，パソコンの能力が低く，オフラインで燃焼計算を実行するなど，オンラインでの利用は容易でなかった。その後，インターネットやBASICの利用が容易となり，ホームページから誰でも利用できるように改善がなされた。

VISTA開発の目的は，核燃料サイクルの分析ツール，すなわち，原子力発電所の物質収支を計算して核燃料サイクルに関係する各種データを予測するツールを提供することである。現在、PWR，BWR，PHWR(重水減速・重水冷却炉)，RBMK(黒鉛減速・沸騰軽水炉)，GCR(黒鉛減速・炭酸ガス冷却炉)，AGR(改良型GCR)，WWER(ロシア型PWR)の７種類の炉型が選択できる。なお，PWR，BWRについては，ワンススルー方式とPuサーマル方式の２方式が選択でき，再処理割合も指定できる。

今後，FBRや高温ガス炉などの新型炉も選択できるようにする予定であり，また，ウランサイクル以外に，近年，PuやMAの発生量が少ないということで注目されているトリウムサイクルも評価できるように検討中で，トリウム炉としては，高温ガス炉や溶融塩炉を選択できるようにする予定である。

VISTAは，過去の運転データを既に持っていて，その上で，将来の予測値，例えば，原子力発電容量や稼働率の予測，燃焼度や濃縮度の推移，ワンススルー方式かPuサーマル方式かの選択など約10種類のデータを入力すれば，CAINという内蔵の燃焼計算コードにより，使用済み燃料中のU，Pu，MAなどのアクチナイド量を計算し，天然ウラン必要量，濃縮必要量，U燃料成型加工量，使用済み燃料発生量，PuやMAの蓄積量，再処理必要量，MOX燃料成型加工量，などのデータを計算するシステムである。

なお，現在のホームページ上の試用版は，個々の炉型の平衡状態についてのみ，上記の計算が可能である。一方，世界全体の将来予測については，入力値やシナリオが固定された試算ケースが示されているだけである。しかし，最終公開版では，ユーザーがシナリオを選択又は入力して，世界全体の予測を得ることができるようにする。

なお，システム上の予測年数に制限はないが，もともとはIAEA加盟各国の政策立案などのための参考データを提供するという目的から，今後の数十年から100年程度の予測の要求が多いと見込んでいる。

CAIN

燃焼計算コードCAIN（Calculation of Actinide Inventory）は，１群の断面積を基に，前記７炉型の燃焼・崩壊計算を行なうコードで，筆者が主に開発を担当した。CAINは，基本的にORIGENと同様のコードであるが，VISTAに組み込むために，燃焼，崩壊に関するBateman 方程式の解析解を直接計算して，原子数を求めている。CAIN開発当初はパソコンの能力が低く，BASICで計算できる技法として，学生時代に三宅千枝先生に教わった「核化学と放射化学」という教科書を何十年ぶりに引っ張り出してきて，この方程式の解をプログラミングした。

1群の断面積は，PWR，BWRなどはORIGEN2のものをそのまま利用し，ORIGEN2にない炉型については，各国の技術者が詳細設計コードなどにより作成した。また，CAINの結果は，ORIGEN2や詳細設計コード，あるいはPIEデータなどと比較し，充分な精度があることを確認している。

また，取り出し時点のデータのみがVISTAに必要であるので，アクチナイドはU，Pu，Np，Cm，Amの元素のうち，半減期が8日以上の核種の燃焼・崩壊を計算しており，**第1図**に示す合計14核種のみが評価対象である。なお，半減期400年以上の核種は安定核種とみなしている。

VISTAの出力

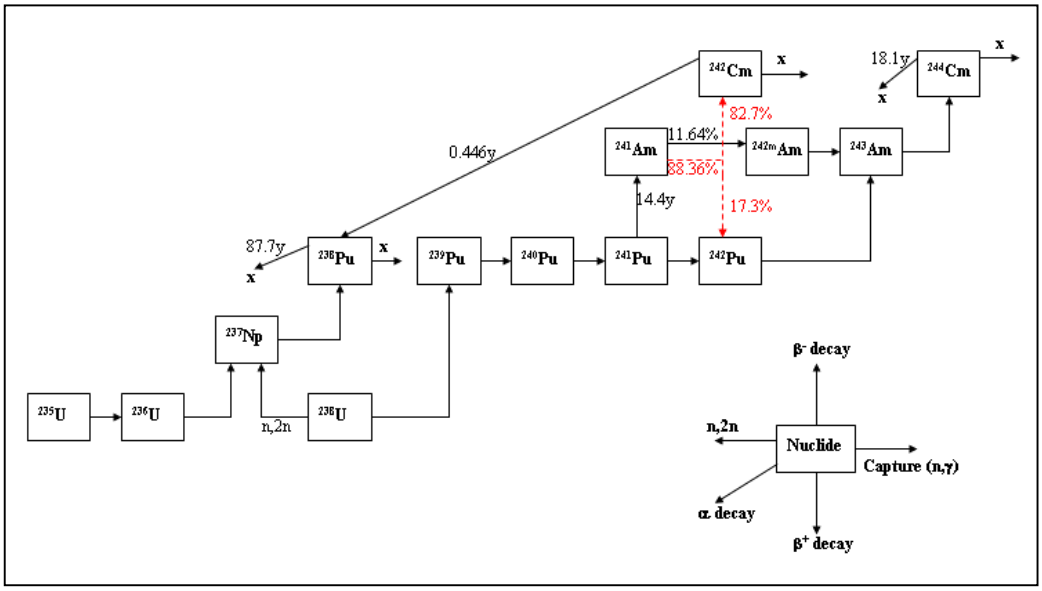
VISTAの出力としては，炉型毎に**第2図**のような物質収支が示され，上の14核種の重量も表形式で出てくる。

一方，試算ケースとして，**第3図**に示すような2050年までの天然U必要量累計値など約10種類のデータの計算結果が示される。前述のように，最終公開版では，ユーザーがシナリオを選択又は入力して，世界全体の予測を得ることができるようにする。

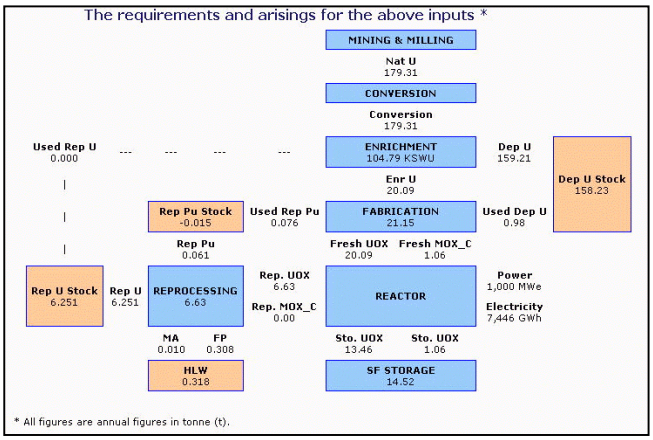
おわりに

上述のように，今後，本システムの応用範囲を拡張して，最終公開版とする予定である。本システムについて，皆様のコメントやアドバイスを学会事務局経由などで頂ければ幸いである。

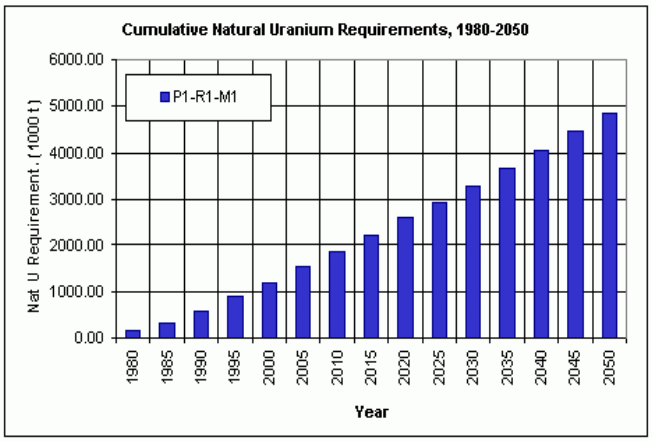
（2005年8月6日　記）



第1図　CAINコードでの燃焼・崩壊チェーン



第2図　VISTAコードの出力例1
（平衡状態 MOX-PWR の物質収支）



第3図　VISTAコードの出力例2
（全世界の天然U必要量累積予測値）

2H07

IAEA の燃料サイクルコード NFCSS の改良（Th サイクル他）

Improvement of IAEA's Fuel Cycle Code NFCSS (Th cycle etc.)

*吉岡 律夫[1], 林 秀行[2], 三成 映理子[3], T.S.Gopi Rethinaraj[4], Ki Seob Sim[5]

[1] トリウム熔融塩国際フォーラム, [2] 元 JAEA, [3] 東京工業大学, [4] インド理科大学院, [5]IAEA

抄録：

IAEA は 2019 年に燃料サイクル計算コード NFCSS の改良版を IAEA サイトに公開した。主な改良点は、PWR と BWR に対する Th 燃料計算、既存の 7 炉型の他に FBR 計算法の明確化、マニュアルの整備公開、NFCSS に連動して、PWR と BWR の取り出し燃料の崩壊熱と放射性毒性を計算するモジュールを付設したこと、等である。

キーワード：IAEA、燃料サイクル計算コード、NFCSS、トリウム燃料、崩壊熱、放射性毒性

１．緒言

IAEA は燃料サイクル計算コード NFCSS を 2005 年に公開したが、その後、加盟各国からの要望を受けて、幾つかの改良を行ない、2019 年に解説書[1]を発行すると共に、改良版コードを IAEA サイト[2]に公開した。今般の主な改良点について、報告する。

２．NFCSS の主な改良点

NFCSS は、元々、日本が提供した燃焼計算エンジンを内蔵し、断面積も内蔵しているので、炉心設計コードを必要とせず、単独で燃料サイクル評価ができるのが特長である。ユーザーは名前を登録するだけで、無償で利用できる。

今般の主な改良は、下記の通りである。

①PWR と BWR に対し、既存の U 燃料と MOX 燃料に加え、トリウム（Th）燃料計算を可能にしたこと。

②計算できるアクチナイド核種数を 14 から 18 に拡大したこと。

③既存の 7 炉型（PWR, BWR, PHWR, RBMK, GCR, AGR, WWER）に加え、FBR の計算方法を明確にしたこと。

④マニュアルを整備公開したこと。

⑤NFCSS に連動して、PWR と BWR の取り出し燃料の崩壊熱と放射性毒性を計算するモジュールを付設したこと。

その他に、従来は明確でなかった、⑥初期炉心・末期炉心の取り扱い手法の確認、⑦中性子束計算手法の確認、⑧他の燃料サイクルコードによる検証、も実施した。

３．改良の結果

図 1 に、Th 燃料に対して今回採用した燃焼チェーンを示す。NFCSS による燃料取り出し時のアクチナイド量を、他の計算コードや先行論文の結果を用いて妥当性を検証した。

図 2 に、PWR において、UO_2 燃料、MOX 燃料、ThO_2 燃料を取り出した後の崩壊熱の推移を示す。放射性毒性に関しても、同様の図が得られるが、ここでは省略する。

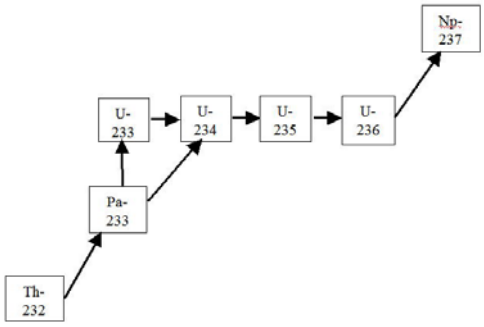


図 1　Th サイクルでの燃焼チェーン

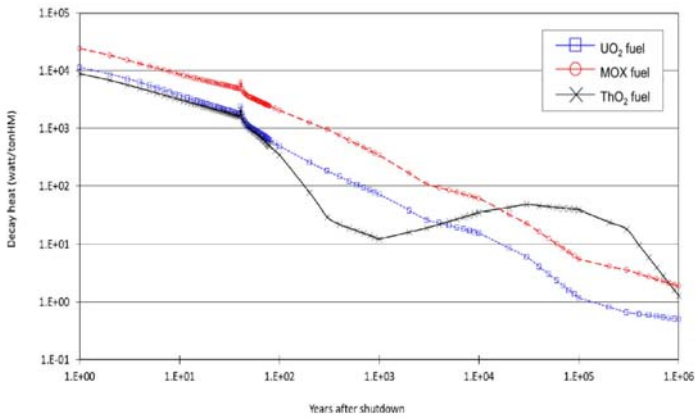


図 2　取り出し後の崩壊熱の推移

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