

Guidebook for Beginner's on Nuclear Fuel Cycle Simulation System (NFCSS)

Addendum to IAEA-TECDOC-1864

Revision 4.0 (Sept, 2024)

**This is not a formal publication but a working document. It will be
timely revised by the Scientific Secretary.**

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1. INTRODUCTIONS

1.1. BACKGROUND

The Nuclear Fuel Cycle Simulation System (NFCSS) is a scenario-based computer simulation tool that can model various nuclear fuel cycle options in various types of nuclear reactors. The NFCSS is mainly developed for UOX fuel, MOX fuel and ThOX fuel and 7 reactor types (BWR, PHWR, RBMK, AGR, GCR and WWER) for fuel cycle simulation. Recently, the NFCSS has been extended for fast reactor fuel cycle modelling and added new modules for decay heat and radiotoxicity calculations. DGR calculation module and cost calculation function are under development. Detailed information and theories involved in the NFCSS are described IAEA-TECDOC-1864^[1] and IAEA-TECDOC-1535^[2].

The NFCSS allows users to efficiently and accurately identify the nuclear mineral resources and technical infrastructure needed for the front end of the nuclear fuel cycle; the amounts of used fuel, actinide nuclides and high-level waste generated for a given reactor fleet size; and the impact of introducing recycling of used fuel on mineral resource savings and waste minimization.

For these assessments, the NFCSS is well optimized in terms of accuracy, simplicity and speed, and requires only a few input data and parameters for execution. This is possible because a complex reactor physics analysis is not necessary to run the NFCSS. This convenience allows even non-specialists, who do not have engineering/science background, to use the NFCSS for studies on future nuclear energy scenarios.

Recent statistics indicates that the NFCSS has been widely used by researchers in national labs and academia, by industry analysts, energy planners and policy analysts working on nuclear fuel cycle options. According to many feedbacks from users indicate that there is a need to develop an updated guidebook for beginner's use who are not familiar with the NFCSS.

The NFCSS website was developed in the past has undergone a major technological revamp in the current release 3.0.

The major changes are following.

- Implementation of scenario data input through an excel file;
- Implementation of fuel data through an excel file;
- Implementation of "Provided" data for BOC, EOC and After cooling through an excel file;
- Addition of fuel libraries for MA containing fuel;
- Addition of fuel libraries for MSR reactor (offline reprocessing);
- Implementation of Decay heat and Radio-toxicity calculations directly for PWR and BWR reactor scenarios;
- Implementation of Decay heat for Sodium Fast Reactor template excel sheet;
- Implementation of DGR size estimation excel template.

1.2. OBJECTIVES

This document is to provide user's guide for beginners who are not familiar with the NFCSS.

This document is an addendum to IAEA-TECDOC-1864 published in 2019 [1]. This document is a working document, and till it is officially published.

This document is located at the NFCSS webpage so that users can download it freely.

1.3. SCOPE

This Addendum is intended to demonstrate all steps required for the preparation of a complete set of input deck to run the NFCSS.

The following limitations should be taken account of:

- The NFCSS is recommended to use in Internet Explorer, and online access to the IAEA's web-site is required;
- Operation period of nuclear power reactors in the simulation is to 200 years maximum, as it is the limitation of the NFCSS;
- User's guides for decay heat and radiotoxicity calculations related to the NFCSS is included in this guide book.

User's guide and templates for calculating DGR sizing calculations are located in the Help page of the NFCSS web page (<https://infcis.iaea.org/NFCSS/NFCSSHelp.cshtml>) together with this User's Guide Book for Beginner's Use of NFCSS.

1.4. STRUCTURE

Section 2 provides an overall explanation on all steps and process required to execute the NFCSS via a walkthrough of a simulation case for UO₂ fuel cycle in a power reactor.

Section 3 provides an example of using the NFCSS for a simulation of a thorium-based fuel cycle in a power reactor.

Section 4 provides an example of using the NFCSS for a complex scenario with various reactor types.

Appendix I is intended to explain using MS Excel spreadsheet to import data for analysis. A working file of the MS Excel spreadsheet is attached to this User's Guidebook for Beginner's Use of NFCSS. Appendix II gives the mathematical model of NFCSS calculations. Appendix III give the source terms and alpha activity of the spent fuel at different burn up values. Appendix IV gives the details of benchmarking done for new modules for MA containing fuel, Fast reactors and Molten salt reactors.

2. GETTIGN STARTED: SIMULATION CASE WITH UO₂ FUEL CYCLE IN A POWER REACTOR

2.1. GENERAL

This section walks through the whole process of using NFCSS for UO₂ fuel cycle in a single reactor type. The logical arrangement of the information would enable the user to understand the sequence and flow of the system. Screen shots are shown under each heading to depict examples. The whole process of using the NFCSS can be explained with five steps as shown in Fig.1.

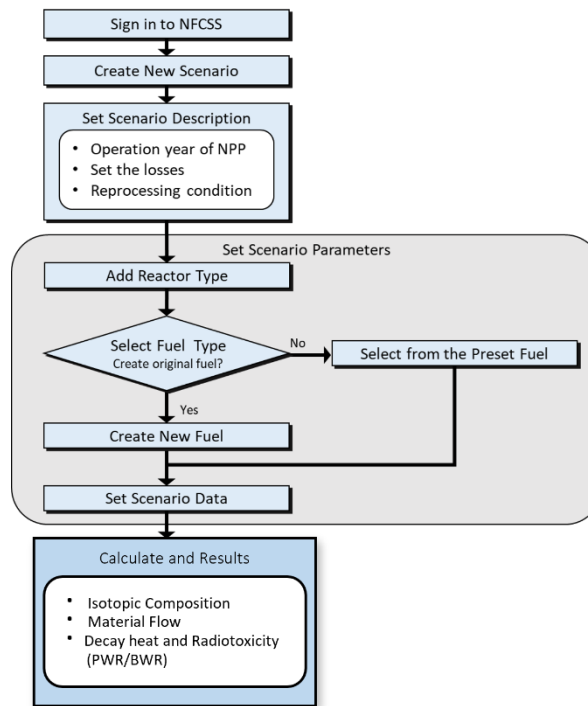


FIG.1 Overall steps to use the NFCSS.

2.2. STEP 1: SIGNING IN TO THE SYSTEM

The NFCSS is located on an IAEA's portal site named NUCLEUS. Before access to NFCSS, a registration to NUCLEUS is pre-requisite (Fig.2).

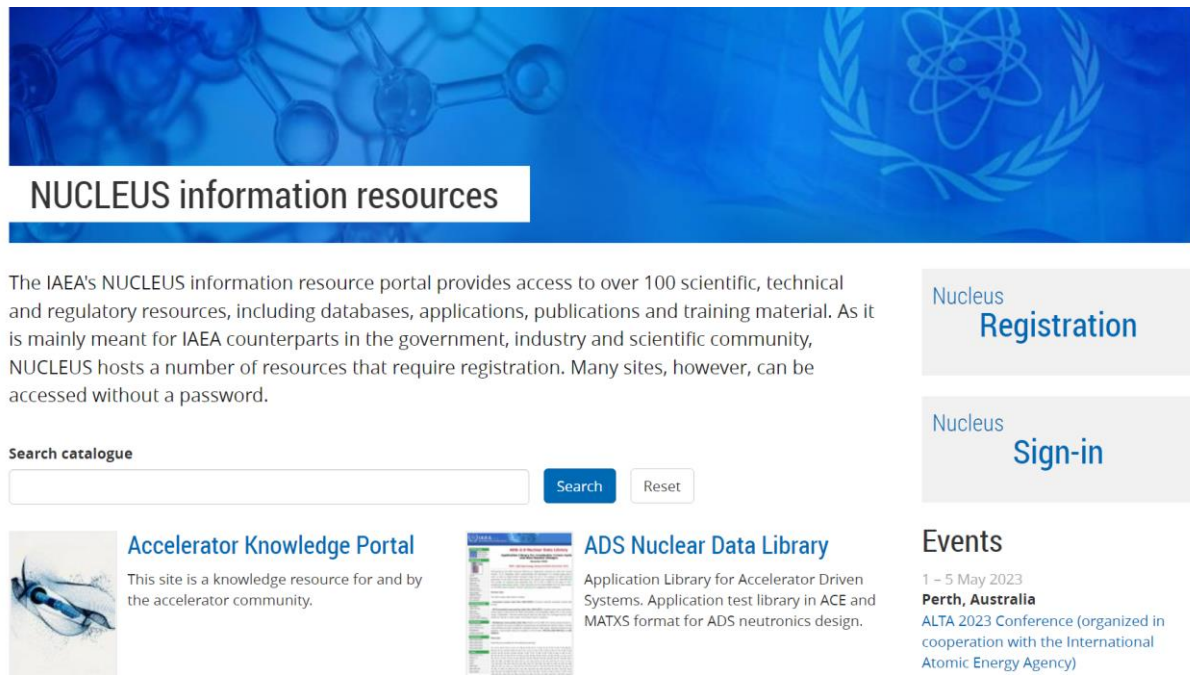


FIG.2 Register to NUCLEUS.

Then, on the computer go to [NFCSS home page](#), or <https://infcis.iaea.org/> and click [Sign In] button in the top right corner of the page (Fig.3).

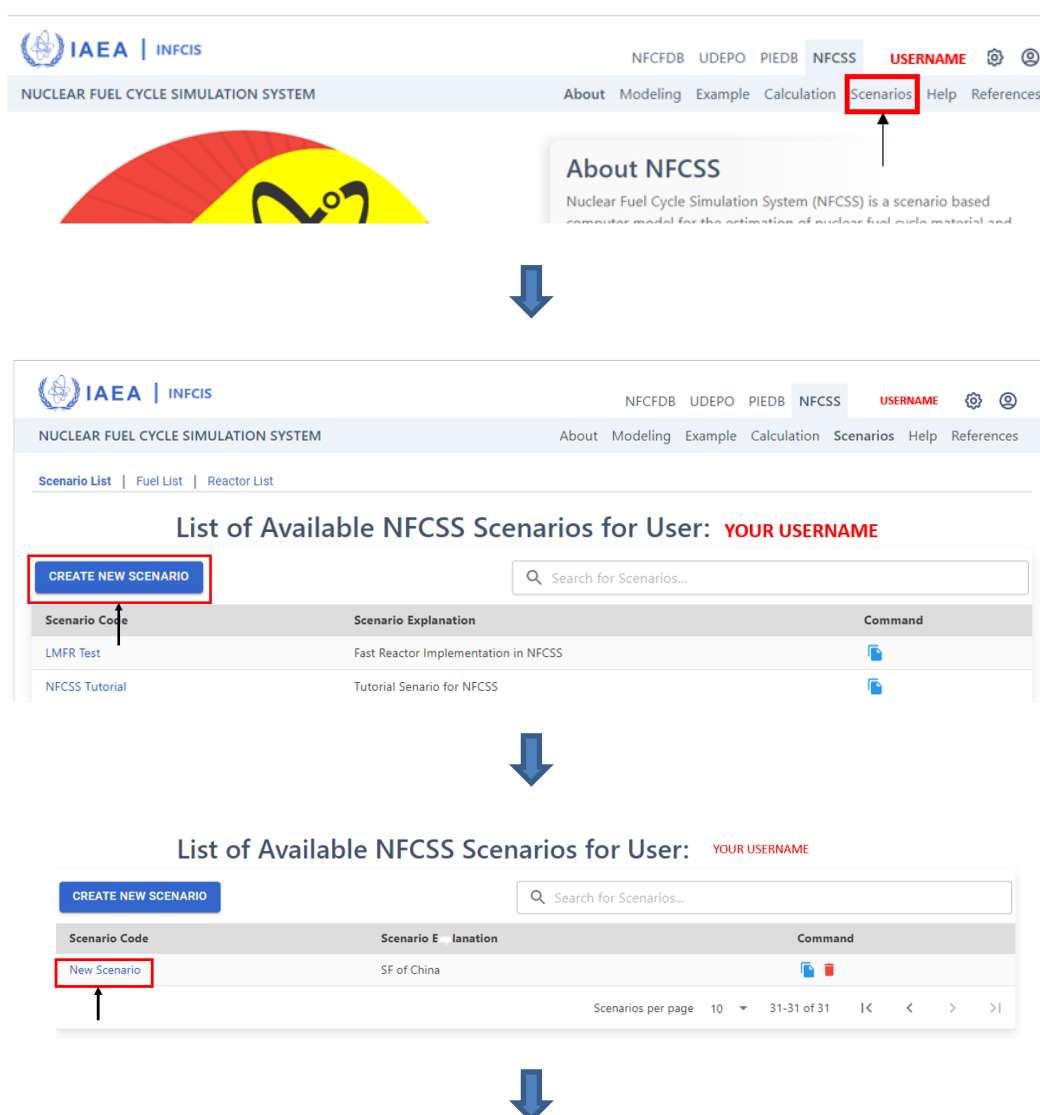


FIG.3 Step 1: Signing-in process.

2.3. STEP 2: CREATING A NEW SCENARIO

A new scenario is created (Fig.4):

- (a) Click [Scenarios] tab to start (Fig.4, 1st);
- (b) Click [Create New Scenario] button to create a new scenario (Fig.4, 2nd). After confirmation, the NFCSS will create a new scenario and displays the Scenario Page for the newly created scenario. The default code for the new scenario will be 'New Code' (Fig.4, 3rd).
- (c) If the new scenario is prepared by modifying an existing scenario (for example, the new scenario is generated by a modification of existing PWR tutorial scenario, click copy button (Fig.4, 4th). After a confirmation dialog, the NFCSS copies the selected tutorial scenario with all the data (Fig.5). Note that the isotopic composition and material flow results are not copied. Calculations must be done for the newly created scenario after changing the data if necessary.



Scenario List | Fuel List | Reactor List

NFCSS Scenario Page

Other Scenario Characteristics

↓

IAEA | INFCIS

NFCFDB UDEPO PIEDB **NFCSS** USERNAME

NUCLEAR FUEL CYCLE SIMULATION SYSTEM

About Modeling Example Calculation Scenarios Help References

Scenario List | Fuel List | Reactor List

List of Available NFCSS Scenarios for User: YOUR USERNAME

Scenario Code	Scenario Explanation	Command
LMFR Test	Fast Reactor Implementation in NFCSS	
NFCSS Tutorial	Tutorial Scenario for NFCSS	
PWR Tutorial	1000 MWe LWR	
REVISED LMFR TEST	Fast Reactor Implementation in NFCSS with revised fuel compositions data	
RESVISED NFCSS TUTORIAL	Tutorial Scenario for NFCSS with revised fuel compositions	
LMFR_1000MW	Fast Reactor Implementation in NFCSS	

FIG.4 Step 2: Scenario creation process

2.4 STEP 3: SETTING SCENARIO DESCRIPTION

This step is to select reactor types which are included in the NFCSS library or to define new reactor types by fuel type and fuel parameters for the reactor. The NFCSS currently focusses on 7 commercial reactor types: PWR, BWR, PHWR, RBMK, GCR, AGR and WWER. FBR (or FR) can be also modelled in the NFCSS in a limited way. The mathematical models of the simulation calculations are given in APPENDIX-II.

After creating a new scenario in Step 2,

- (a) First enter your details for feedback for user statistics for the Agency. Then, enter the general scenario description data. The explanation of each input parameter and the possible options are (corresponding numbers are shown in Fig.5, 1st):

- (1) Scenario Code: The identification code of the scenario. The text can be entered freely in this field up to 50 characters.
 - (2) Scenario Explanation: The explanatory information about the scenario. This field is a free text field of any length.
 - (3) Scenario Reference: The source(s) of information which is used to define the scenario, if any. For example, the user can use IAEA Nuclear Power Estimates (*) to analyse the worldwide situation in the future. User can also IAEA's PRIS database (www.pris.iaea.org/pris). This is a free text field with any length. (*: 2022 edition is shown in:
<https://www.iaea.org/publications/15211/nuclear-power-reactors-in-the-world>)
 - (4) Start Year: Beginning of the scenario period. The data should be entered in integer format.
 - (5) End Year: End of scenario period. The data must be entered in integer format. Note that the start and end year difference defined in the scenario must be not greater than 200 years.
 - (6) Loss in Conversion Facility (%), Loss in Enrichment Facility (%), Loss in Fabrication Facility (%), Loss in Reprocessing Facility (%): Loss coefficients at steps in the nuclear fuel cycle. The loss is given in percent. The material loss is calculated at the beginning of the step and remains unchanged for the rest of the calculation. If the scenario data does not match, a warning message will be prompted(Fig.6).
 - (7) Volume of HLW Generated in Reprocessing: The volume of the high-level waste generated during the reprocessing of spent fuel per tonne Heavy Metal of reprocessed spent fuel. These are assumed values from experience of reprocessing and can be changed by the user.
 - (8) Volume of LLW Generated in Reprocessing: The volume of the low-level waste generated during the reprocessing of spent fuel per tonne Heavy Metal of reprocessed spent fuel. These are assumed values from experience of reprocessing and can be changed by the user.
- (b) Second, click [Save Description] button at the upper right location of Scenario Page (Fig.5, 2nd).
 - (c) Third, go back to [Scenario List] tab and check whether it is saved properly (Fig.5, 3rd).

NFCSS Scenario Page

Scenario Code* 1 Scenario Type Standard	Scenario Explanation 2	Scenario Reference 3
Start Year* 4 2020	End Year* 5 2080	

Other Scenario Characteristics

Loss in Conversion Facility (%)* 6 0	Volume of HLW Generated in Reprocessing (m3/tHM)* 7 0.35
Loss in Enrichment Facility (%)* 6 0	Volume of LLW Generated in Reprocessing (m3/tHM)* 8 0.15
Loss in Fabrication Facility (%)* 6 0	
Loss in Reprocessing Facility (%)* 6 0	



NFCSS Scenario Page

Scenario Code* MSR CASE 2	Scenario Explanation Trial for MSR	Scenario Reference
Scenario Type Standard		
Start Year* 2020	End Year* 2080	

Other Scenario Characteristics

Loss in Conversion Facility (%)* 0	Volume of HLW Generated in Reprocessing (m3/tHM)* 0.15
Loss in Enrichment Facility (%)* 0	Volume of LLW Generated in Reprocessing (m3/tHM)* 0.35
Loss in Fabrication Facility (%)* 0	
Loss in Reprocessing Facility (%)* 0	



Scenario List |
 [Fuel List](#) |
 [Reactor List](#)

NFCSS Scenario Page

Scenario Code* MSR CASE 2	Scenario Explanation Trial for MSR	Scenario Reference
Scenario Type Standard		
Start Year* 2020	End Year* 2080	

Other Scenario Characteristics

Loss in Conversion Facility (%)* 0	Volume of HLW Generated in Reprocessing (m3/tHM)* 0.15
Loss in Enrichment Facility (%)* 0	Volume of LLW Generated in Reprocessing (m3/tHM)* 0.35
Loss in Fabrication Facility (%)* 0	
Loss in Reprocessing Facility (%)* 0	

FIG.5 Sub-Step 1-3: Scenario description.

NFCSS Scenario Page

Scenario Code* TEST	Scenario Explanation 	Scenario Reference
Scenario Type Standard		
Start Year* 102 Start Year is required!	End Year* 102 End Year is required!	

Other Scenario Characteristics

Loss in Conversion Facility (%)* 102 Value must be between 0 and 100	Volume of HLW Generated in Reprocessing (m3/tHM)* 0.15
Loss in Enrichment Facility (%)* 102 Value must be between 0 and 100	Volume of LLW Generated in Reprocessing (m3/tHM)* 0.35
Loss in Fabrication Facility (%)* 102 Value must be between 0 and 100	
Loss in Reprocessing Facility (%)* 102 Value must be between 0 and 100	

FIG.6 Warning for mismatch in scenario period.

The user can export whole scenario data to Microsoft Excel to locally store the scenario data and results or work on it in Microsoft Excel. To export scenario description including all the data to Microsoft Excel, [Export to Excel] button is used (Fig.7).

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Scenario Code* MSR CASE 2	Scenario Explanation Trial for MSR	Scenario Reference
Scenario Type Standard		
Start Year* 2020	End Year* 2080	

Other Scenario Characteristics

Loss in Conversion Facility (%)* 0.1	Volume of HLW Generated in Reprocessing (m3/tHM)* 0.15
Loss in Enrichment Facility (%)* 0.1	Volume of LLW Generated in Reprocessing (m3/tHM)* 0.35
Loss in Fabrication Facility (%)* 0.1	
Loss in Reprocessing Facility (%)* 0.1	

FIG.7 Export scenario data to Excel spreadsheet.

New Scenario will be shown in the bottom of the list (Fig.8).

Scenario List | Fuel List | Reactor List | **Selected Scenario (14422)**

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Export to Excel | Save Description | Reset Form

Scenario Code* MSR CASE 2	Scenario Explanation Trial for MSR	Scenario Reference
Scenario Type Standard		
Start Year* 2020	End Year* 2080	

Other Scenario Characteristics

Loss in Conversion Facility (%)* 0.1	Volume of HLW Generated in Reprocessing (m3/tHM)* 0.15
Loss in Enrichment Facility (%)* 0.1	Volume of LLW Generated in Reprocessing (m3/tHM)* 0.35
Loss in Fabrication Facility (%)* 0.1	
Loss in Reprocessing Facility (%)* 0.1	



List of Available NFCSS Scenarios for User: **YOUR USERNAME**

CREATE NEW SCENARIO

Scenario Code	Scenario Explanation	Command
PWR - LEU+TH	Overall mass flow comparison	
URAM PWR	For conference paper	
URAM MOX	For conference paper	
MSR	For Trial	
MSR Case 2	Trial	
FBRIN WITH BLANKETS	Fast reactors	
China Fleets	SF of China	
projected overall China Fleets	SF of China	
MSR CASE 2	Trial for MSR	

Scenarios per page 10 21-29 of 29 |< < > >|

FIG.8 Confirmation of new scenario.

2.4. STEP 4: SETTING SCENARIO PARAMETERS

This step is to select reactor types which are included in the NFCSS library or to define new reactor types by fuel type and fuel parameters for the reactor. The NFCSS currently focusses on 7 commercial reactor types: PWR, BWR, PHWR, RBMK, GCR, AGR and WWER. FBR (or FR) and MSR (off-line reprocessing) can be also modelled in the NFCSS in a limited way, details of which are given in TECDOC-1864 and ‘Amendment to TECDOC-1864’ (in progress).

2.4.1. General procedure for Step 4

Scenario parameters are provided as follows (Fig.9):

- (a) Click [Scenario Parameters] tab to specify reactor, fuel types and data for the scenario parameters for each of the reactor and fuel type;
- (b) Click [Add Reactor Type] button to add new reactor type to the scenario;
- (c) Select [Fuel Type 1] and [Fuel Type 2]. **Initial Core Fissile Content** is mandatory. One remark here is that selection of the fuel type is possible only from the built-in library of the NFCSS at this point, and if the user wants to use new fuel (not in the fuel list), see Section 2.4.2;
- (d) Click the drop-down box of each item in isotopic composition. Generally, the “calculated” option is selected; The user also can use excel template to customize isotope compositions in all 3 stages(s (**Initial content, depletion and Decay**)) by choosing ‘Provided’ option, as shown in Fig.10. Any combination of ‘**Provided**’ option can be used for all 3 stages. This feature allows user to have data different for all years and calculated from other codes. The precedence of scenario calculation will be given to the ‘provided’ data over ‘calculated’ option which takes data from fuel initial composition common for all years
- (e) Click [Save] button (rightmost button of the above screen).

Scenario List | Fuel List | Reactor List | **Selected Scenario (14432)**

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080 [ADD REACTOR TYPE](#)

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
There is no reactor type selected for this scenario!			

↓

Scenario List | Fuel List | Reactor List | **Selected Scenario (14432)**

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080 [ADD REACTOR TYPE](#)

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
There is no reactor type selected for this scenario!			

↓

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080 [ADD REACTOR TYPE](#)

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR	PWR UOX 1	REVISED PWR MOX-1	

Reactor Type*
PWR Select name the type of the Fuel Type 1 from fuel library.

Fuel Type 1
PWR UOX 1

RepU Use*
Yes Select name the type of the Fuel Type 2 from fuel library.

Fuel Type 2
PWR MOX 1

Nat Equivalence Ratio*
1

Initial Core Fissile Content (%)*
75

Isotopic Composition

Initial Content* Calculated	Initial Content* Calculated
Depletion* Calculated	Depletion* Calculated
Decay* Calculated	Decay* Calculated

DO NOT forget to save before moving to other steps

[RESET](#) [CANCEL](#) [SAVE CHANGES](#)

Calculated

Provided

The 'Calculated' option is generally selected; selecting the 'Provided' option allows the user to customize the input of isotopic compositions in all 3 stages.

FIG.9 Step 4: Scenario parameters.

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS **CALCULATION AND RESULTS**

Scenario Period: 2020 - 2080 [Make All IC Calculations](#) [Make All MF Calculations](#)

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Decay Heat	Radiotoxicity
PWR1	PWR UOX 1			!! Warning !!	!! Warning !!

[Calculate Overall Material Flow](#) [-](#) [+](#) [-](#) [+](#)

NFCSS Isotopic Composition Result

ReactorType: **PWR1**
Fuel: **PWR UOX 1**
Neutron Flux: 0 (n/(cm².s))

[SAVE DATA](#) **Import** [Export to Excel](#)

[BEGINNING OF CYCLE](#) [END OF CYCLE](#) [AFTER COOLING](#)

DataYear	Enrichment	Burnup	U235	U236	U238	Np237	Pu238	Pu239	Pu240	Pu241	Pu242	Am241	Am242m	Am243
There is no record														

NFCSS Isotopic Composition Result

ReactorType: **PWR1**
Fuel: **PWR UOX 1**
Neutron Flux: 0 (n/(cm².s))

[SAVE DATA](#) [Import](#) [Export to Excel](#)

[BEGINNING OF CYCLE](#) [END OF CYCLE](#) [AFTER COOLING](#)

[DataYear](#) [Enrichment](#) [Burnup](#)

Import and update existing Isotopic Composition from excel

Use the [template](#) to prepare the data to upload Isotopic Composition

[Click here to download the Excel template](#)

File Name:

[SELECT FILE](#)

[Click here to upload the filled Excel file](#)

[CANCEL](#) [SAVE CHANGES](#)

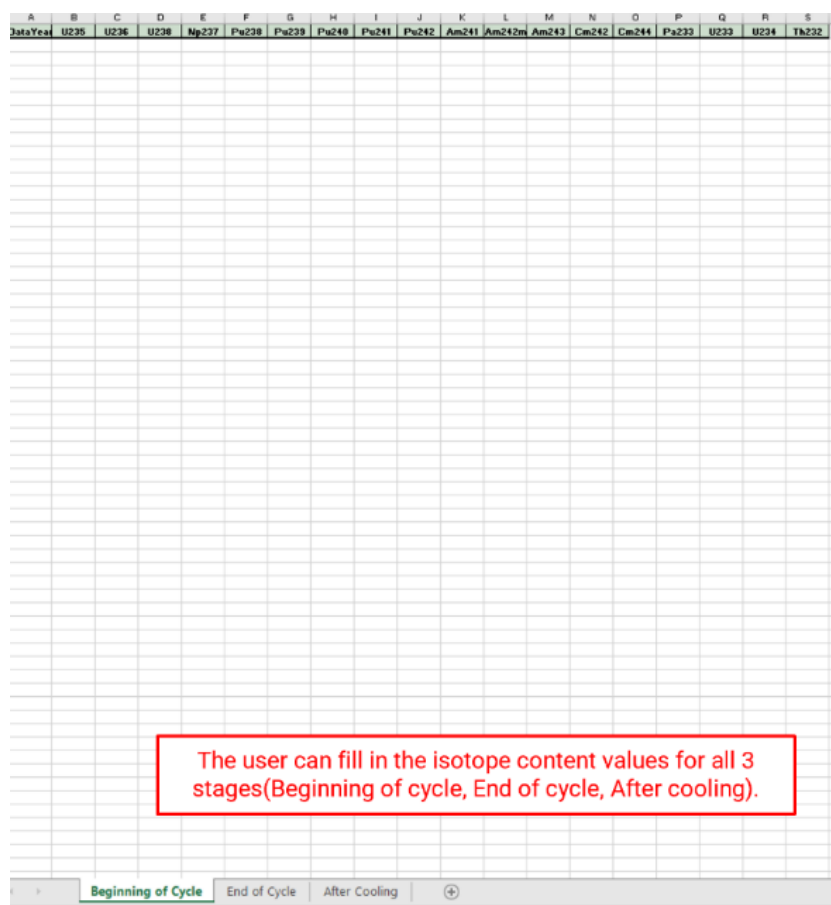


FIG.10 Optional: Steps to use 'Provide Function' for using excel template to customize isotope compositions.

2.4.2. Creating a new fuel type

In addition to the selection of the fuel type from the built-in library of the NFCSS, the user can add their own fuel types to the fuel list. Fuel types added by the user will only be available to the creator of the specific fuel type. Procedures are as follows:

- Click [Fuel List] tab to see the built-in library of fuel type (FIG.11A, 1st);
- If the user wishes to copy one of the existing fuel types from the list to create a new fuel type, copy and later modify the data for the newly created fuel type for the new scenario. Click [Copy] button in the command column of the fuel list table to copy (FIG.11A, 2nd);
- New copied fuel or newly crease fuel will appear at the bottom of the list (FIG.11A, 3rd);
- In the above (b), if the user wants to create a brand new fuel type, click [Create New Fuel] button to create a new fuel type and to add it to the library (FIG.11B). The newly created fuel type will be displayed at the end of the list. The user then goes to the fuel data page to define the details of the new fuel type by clicking on the name of the new fuel type;
- By clicking on the name of the fuel type, open the fuel data page for the selected fuel type (FIG.12A). Users can also enter fuel properties and cross-sections directly or import cross-sections using an Excel template (FIG.13).

IAEA | INFCIS

NFCFDB UDEPO PIEDB **NFCSS** **USERNAME**

NUCLEAR FUEL CYCLE SIMULATION SYSTEM
About Modeling Example Calculation Scenarios Help References

Scenario List **Fuel List** Reactor List

List of Available NFCSS Fuel for User: **YOUR USERNAME**

CREATE NEW FUEL

Fuel Id	Fuel Type	Owner	Command
1	PWR UOX 1	System	
2	PWR MOX 1	System	
3	PWR UOX 2	System	
4	PWR UOX 3	System	
5	BWR UOX 1	System	
6	BWR MOX 1	System	

List of Available NFCSS Fuel for User: **YOUR USERNAME**

CREATE NEW FUEL

Fuel Id	Fuel Type	Owner	Command
1	PWR UOX 1	System	
2	PWR MOX 1	System	
3	PWR UOX 2	System	
4	PWR UOX 3	System	
5	BWR UOX 1	System	
6	BWR MOX 1	System	

CREATE NEW FUEL

Fuel Id	Fuel Type	Owner	Command
5528	PHWR - CHINA	USERNAME	
5531	MSR Pu-Th	USERNAME	
5539	New Fuel	USERNAME	

Fuels per page 10 51-53 of 53

FIG.11A Creating of a new fuel (by copying an existing fuel case).

List of Available NFCSS Fuel for User: **YOUR USERNAME**

CREATE NEW FUEL			
Search for Fuels...			
Fuel Id	Fuel Type	Owner	Command
1	PWR UOX 1	System	
2	PWR MOX 1	System	
3	PWR UOX 2	System	
4	PWR UOX 3	System	
5	BWR UOX 1	System	
6	BWR MOX 1	System	

FIG.11B Creating a brand-new fuel case.

Scenario List | Fuel List | Reactor List

NFCSS Fuel Page

5241

RESET FORM DELETE FUEL SAVE FUEL DATA

DO NOT forget to save before moving to other stops

Name your own fuel type and set each parameters can be fully tailored to meet specific requirements PWR-MOX-new is an example.

Fuel Characteristics

Fuel Type Code*
UOX4

Fuel Type*
U Fuel

Reference Enrichment*
4 %

Power Density*
37.5 kW/kg

Conversion Type*
UF6

Cooling Time*
6 yr

Manufacturing Time*
1 yr

Reprocessing Time*
1 yr

Fuel Type Description
PWR UOX Fuel from ORIGEN Library

Fuel Initial Contents (*) (% Mass)

U235*
4

U236*
0

U238*
96

Np237*
0

Pu238*
0

Pu239*
0

Pu240*
0

Pu241*
0

Pu242*
0

Am241*
0

Am242m*
0

FIG.12A. Editing fuel data page.

NB: There are three fuel types in NFCSS library, U Fuel, U-Pu (MOX) Fuel, and Free Content fuel. For U- Fuel and Free Content, the sum of fuel initial content percentage

(mass) should be exactly 100. For U-Pu Fuel, the sum of Pu nuclide vectors percentage (mass) should be exactly 100 and only U235 % is to be given.

(1) In the case of “U Fuel,” users must input the U enrichment (%U235), then %U238 is internally calculated as [100- Enrichment].

(2) In the case of “U-Pu (MOX) fuel”, users must input [% Total Pu enrichment], the isotopic composition of Pu (Pu-vector), and [%U235] in [U235+U238]. Each nuclide fraction is internally calculated based on these inputs (FIG.12B).

Of course, users can use “Free content” input for MOX fuel, which is shown in (3).

FIG.12B. Input for “U-Pu (MOX) fuel”

(3) In the case of “Free content fuel”, the sum of fuel initial content percentage (mass) must be exactly 100, because NFCSS does not check it.

Thorium fuel is handled under ‘Free Content’ fuel, Conversion Type should be selected as ‘Th’ from the drop list.

Uranium fuel is usually set to Fuel Type 1. U-Pu fuel is usually set to fuel type 2. Free content is used for Thorium based fuels, MA containing fuels, and any fuels specified by users.

“Type of conversion” is necessary to manufacture the fuel. UF₆ is used for enriched fuel. UO₂ is typically used for PHWR fuels, and U Metal is used for GCR fuels.

NFCSS Fuel Page

IMPORT CROSS SECTIONS

RESET FORM

SAVE FUEL DATA

FUEL CHARACTERISTICS CROSS SECTIONS

Nuclide	Bu-Average Fission Cross_Section (barn)	Bu-Average Capture Cross_Section (barn)	Bu-Average n2n Cross_Section (barn)
U235	0	0	0
U236	0	0	0
U238	0	0	0
Np237	0	0	0



FUEL CHARACTERISTICS CROSS SECTIONS

Nuclide	Bu-Average Fission Cross_Section (barn)	Bu-Average Capture Cross_Section (barn)	Bu-Average n2n Cross_Section (barn)
U235	0	0	0
U236	0	0	0
U238	0	0	0
Np237	0	0	0
Pu238	0	0	0
Pu239	0	0	0
Pu240	0	0	0
Pu241	0	0	0
Pu242	0	0	0
Am241	0	0	0
Am242m	0	0	0

Import and update existing Fuel Cross Sections from excel

Click here to download the template

Use the [Template](#) to prepare the data to upload Fuel Cross Sections

Refer the first sheet in template for guidance

Click here to upload your completed file

File Name:



	A	B	C	D
	Nuclide	Bu-Average Fission Cross_Section (barn)	Bu-Average Capture Cross_Section (barn)	Bu-Average n2n Cross_Section (barn)
1				
2	U235	1.967000	0.569600	0.001323
3	U236	0.092750	0.507600	0.000825
4	U238	0.042390	0.298	0.521
5	Np237	0.322500	1.695	0.094
6	Pu238	1.086000	0.670	0.055
7	Pu239	1.842000	0.543	0.358
8	Pu240	0.361700	0.587400	0.000410
9	Pu241	2.588000	0.497300	0.001454
10	Pu242	0.259800	0.493100	0.000699
11	Am241	0.270000	1.607000	0.000131
12	Am242m	3.251000	0.450700	0.000629
13	Am243	0.196600	0.082350	0.000394
14	Cm242	0.690300	0.855100	0.000105
15	Cm244	0.463800	0.676600	0.000244
16	Th232	0.270600	0.542800	0.006998
17	Pa233	0.034810	0.809900	0.001154
18	U233	2.812000	0.277100	0.000202
19	U234	0.315900	0.625100	0.000292

Using this template, you can upload and create new fuel cross section in the system

FIG.13 Import of cross sections data using Excel template.

2.4.3. Setting scenario data

First, click  button to edit the scenario parameters data for each the reactor types and associated fuel types (FIG.14).

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2020 - 2120

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR2	UOX4	PWR MOX4	   

PRESS THESE BUTTONS: It enables you to set plant's related power, capacity, enrichment ratio, loading ratio of fuel type 2, reprocessing condition.

[ADD REACTOR TYPE](#)

FIG.14 Editing scenario parameter values.

Please enter the data manually in the first and second row and use the copy row function for filling other rows. The Net Total Capacity column need not be filled in as the system will automatically calculate and fill the column after saving the data. FIG.15A shows all the steps for entering scenario parameters. Users can also use the EXCEL template form to fill in the input scenario data, after uploading the file, the system will automatically fill in the form on the page, and the specific steps are shown in FIG.15B. FIG.16 shows the warning message for any mismatch of scenario data.

NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 1991 - 2050 [IMPORT SCENARIO DATA](#) [RESET FORM](#) [SAVE DATA SET](#)

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES			
Data Year	Added Capacity (MWe) 	Removed Capacity (MWe) 	Net Total Capacity (MWe) 	Efficiency (%) 	Load Factor (%) 	Tails Assay (%) 	Fuel Type 2 Share (%) 	Residence Time (year) 
  1991	326	0	0	33	80	0.3	30	4
  1992	0	0	0	33	80	0.3	30	4
  1993	0	0	0	33	80	0.3	30	4
  1994	0	0	0	33	80	0.3	30	4
  1995	0	0	0	33	80	0.3	30	4
  1996	0	0	0	33	80	0.3	30	4
  1997	0	0	0	33	80	0.3	30	4
  1998	0	0	0	33	80	0.3	30	4
  1999	0	0	0	33	80	0.3	30	4



NFCSS Scenario Data Page

User can also use the EXCEL template to prepare the data to upload Scenario Data

ReactorType: PWR

Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES			
Data Year	Added Capacity	Removed Capacity (MWe)	Net Total Capacity (MWe)	Efficiency (%)	Load Factor (%)	Tails Assay (%)	Fuel Type 2 Share (%)	Residence Time (year)
1991	326	0	0	33	80	0.3	30	4
1992	0	0	0	33	80	0.3	30	4
1993	0	0	0	33	80	0.3	30	4
1994	0	0	0	33	80	0.3	30	4
1995	0	0	0	33	80	0.3	30	4
1996	0	0	0	33	80	0.3	30	4
1997	0	0	0	33	80	0.3	30	4
1998	0	0	0	33	80	0.3	30	4
1999	0	0	0	33	80	0.3	30	4



NFCSS Scenario Data Page

ReactorType: PWR

Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES	
Data Year	Enrichment (%)	Discharge Burnup (GWD/t)	Reprocessing Ratio (%)			
1991	4	45	0			
1992	4	45	0			
1993	4	45	0			
1994	4	45	0			
1995	4	45	0			
1996	4	45	0			
1997	4	45	0			
1998	4	45	0			
1999	4	45	0			



NFCSS Scenario Data Page

ReactorType: PWR

Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Data Year	Enrichment 2f (t)		Discharge Burnup 2f GWD/t (t)		Reprocessing Ratio 2f (%)
1991	7		45		100
1992	7		45		100
1993	7		45		100
1994	7		45		100
1995	7		45		100
1996	7		45		100
1997	7		45		100
1998	7		45		100
1999	7		45		100



NFCSS Scenario Data Page

ReactorType: PWR

Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES		
Data Year	NatU Capacity (tHM)	Conversion Capacity (tHM)	Enrichment Capacity (MTSWU)	Fabrication Capacity (tHM)	Reprocessing Capacity (tHM)	Fabrication Capacity 2f (tHM)	Reprocessing Capacity 2f (tHM)
1991	0	0	0	0	1000	0	1000
1992	0	0	0	0	1000	0	1000
1993	0	0	0	0	1000	0	1000
1994	0	0	0	0	1000	0	1000
1995	0	0	0	0	1000	0	1000
1996	0	0	0	0	1000	0	1000
1997	0	0	0	0	1000	0	1000
1998	0	0	0	0	1000	0	1000
1999	0	0	0	0	1000	0	1000

NB: Nat U capacity, Conversion capacity, enrichment capacity, fabrication capacity and fabrication capacity 2F are not functional temporarily. As for reprocessing capacity, it is recommended to input a large enough value. If this value is smaller than discharged fuel amount, spent fuels are reprocessed within that capacity.



NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS	FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Initial Value Type	Initial Value	Insertion Year	Explanation	
Initial Reactor Power (MWe) :	0	N/A	The total power of reactors which are operational in equilibrium conditions at the beginning of the scenario period.	
Initial Rep. Pu Stock (t HM) :	0	N/A	The total amount of Reprocessed Pu which is available for recycling at the beginning of scenario period.	
Initial SF Stock - Fuel Type 1 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 1) which is ready for reprocessing at the beginning of the scenario period.	
Initial SF Stock - Fuel Type 2 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 2) which is ready for reprocessing at the beginning of the scenario period.	

NB: Fill in these values if applicable, otherwise 0 is default value.



NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS	FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Initial Value Type	Initial Value	Insertion Year	Explanation	
Initial Reactor Power (MWe) :	0	N/A	The total power of reactors which are operational in equilibrium conditions at the beginning of the scenario period.	
Initial Rep. Pu Stock (t HM) :	0	N/A	The total amount of Reprocessed Pu which is available for recycling at the beginning of scenario period.	
Initial SF Stock - Fuel Type 1 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 1) which is ready for reprocessing at the beginning of the scenario period.	
Initial SF Stock - Fuel Type 2 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 2) which is ready for reprocessing at the beginning of the scenario period.	

FIG.15A Scenario parameters steps

NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 1991 - 2050

IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES			
Data Year	Added Capacity (MWe)	Removed Capacity (MWe)	Net Total Capacity (MWe)	Efficiency (%)	Load Factor (%)	Tails Assay (%)	Fuel Type 2 Share (%)	Residence Time (year)
+ 1991	0	0	0	0	0	0	0	0
+ 1992	0	0	0	0	0	0	0	0
+ 1993	0	0	0	0	0	0	0	0
+ 1994	0	0	0	0	0	0	0	0
+ 1995	0	0	0	0	0	0	0	0
+ 1996	0	0	0	0	0	0	0	0
+ 1997	0	0	0	0	0	0	0	0
+ 1998	0	0	0	0	0	0	0	0
+ 1999	0	0	0	0	0	0	0	0



NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 1991 - 2050 IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1		FUEL TYPE 2		FUEL CYCLE		INITIAL VALUES	
Data Year	Added Capacity (MWe)	Removed Capacity (MWe)	Net Total Capacity (MWe)	Efficiency (%)	Load Factor (%)	Tails Assay (%)	Fuel Type 2 Share (%)	Residence Time (year)	
1991	0	0	0	0	0	0	0	0	
1992	0						0	0	
1993	0						0	0	
1994	0						0	0	
1995	0						0	0	
1996	0						0	0	
1997	0						0	0	
1998	0						0	0	
1999	0						0	0	

Import and update existing Scenario Data from excel

Use the [template](#) to prepare the data to upload Scenario Data

[Click here to download the Excel template](#) guidance

File Name:

SELECT FILE

[Click here to upload the filled Excel file](#)

CANCEL SAVE CHANGES



ScenarioDataTemplate (1).xlsx

File Home Insert Draw Page Layout Formulas Data Review View Automate Help

Clipboard Font Alignment Number Styles Cells

General Characteristics		Fuel Type 1		Fuel Type 2		Fuel Cycle		Initial Values	
Data Year	Added Capacity (MWe)	Removed Capacity (MWe)	Net Total Capacity (MWe)	Efficiency (%)	Load Factor (%)	Tails Assay (%)	Fuel Type 2 Share (%)	Residence Time (year)	
1991	0	0	0	0	0	0	0	0	
1992	0						0	0	
1993	0						0	0	
1994	0						0	0	
1995	0						0	0	
1996	0						0	0	
1997	0						0	0	
1998	0						0	0	
1999	0						0	0	

Guidelines with validation rules for filling the template

- 1 Using this template you can upload and create new reactor scenario data in the system
- 2 Please do not edit the worksheet names
- 3 Please do not edit the "DataYear" column in the <reactor name> worksheet
- 4 In case the values for reactor data are left blank the value will be considered as 0
- 5 The reactor data values do not allow negative number values.

FIG.15B Scenario parameters steps by using Excel template.

NFCSS Scenario Data Page

ReactorType: PWR
Scenario Period: 2020 - 2080

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Data Year	Enrichment 2f (%)		Discharge Burnup 2f GWd/t (%)		Reprocessing Ratio 2f (%)
2020	7		45		102 Value must be between 0 and 100
2021	7		45		102 Value must be between 0 and 100
2022	7		45		102 Value must be between 0 and 100
2023	7		45		102 Value must be between 0 and 100
2024	7		45		102 Value must be between 0 and 100
2025	7		45		102 Value must be between 0 and 100
2026	7		45		102 Value must be between 0 and 100

FIG.16 Warning for any mismatch of scenario data.

NB: After completing the scenario data setting, click [Selected Scenario] on the navigation bar to jump back to "NFCSS Scenario Page" for the next step of result calculation.

2.5. STEP 5: CALCULATION AND RESULTS

Calculations are divided into two parts: isotope composition calculations, material flow calculations. In order to perform correct isotopic composition calculations, initial composition of fresh fuel and other fuel parameters have to be provided.

As shown in FIG.17,

- (a) If all data is correctly entered for a fuel type, first click [Make All IC Calculations] button (FIG.17, 1st).
- (b) After completing isotopic composition calculations or providing the isotopic compositions, flag color will change from **RED** to **GREEN** (FIG.17, 2nd).

Then, click [Make All MF Calculations] button to make material flow calculations for each of the reactor types (FIG.17, 2nd).

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Make All IC Calculations
Make All MF Calculations

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Deacy Heat	Radiotoxicity
LMFR Core		REVISED LMFR MOX		!! Warning !!	!! Warning !!
LMFR Axial	REVISED LMFR AXIAL BLANKET			!! Warning !!	!! Warning !!
LMFR Radial	REVISED LMFR RADIAL BLANKET			!! Warning !!	!! Warning !!

Calculate Overall Material Flow



NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Make All IC Calculations
Make All MF Calculations

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Deacy Heat	Radiotoxicity
LMFR Core		REVISED LMFR MOX		!! Warning !!	!! Warning !!
LMFR Axial	REVISED LMFR AXIAL BLANKET			!! Warning !!	!! Warning !!
LMFR Radial	REVISED LMFR RADIAL BLANKET			!! Warning !!	!! Warning !!

Calculate Overall Material Flow

FIG.17 Step 5: calculation procedure.

If the calculation is finished without errors, screen can be found as shown in FIG.18.

The user can see calculation results of material flow in diagram by clicking the preview button. The user can get the chart for results of each of the fuel cycle steps by clicking on the box of steps.

The calculation order is important as mentioned. That is, the three-step calculation order should be:

- (a) Conduct isotopic composition calculations for each fuel type in each reactor type;
- (b) Conduct material flow calculations for each reactor type;
- (c) Conduct overall material flow calculations.

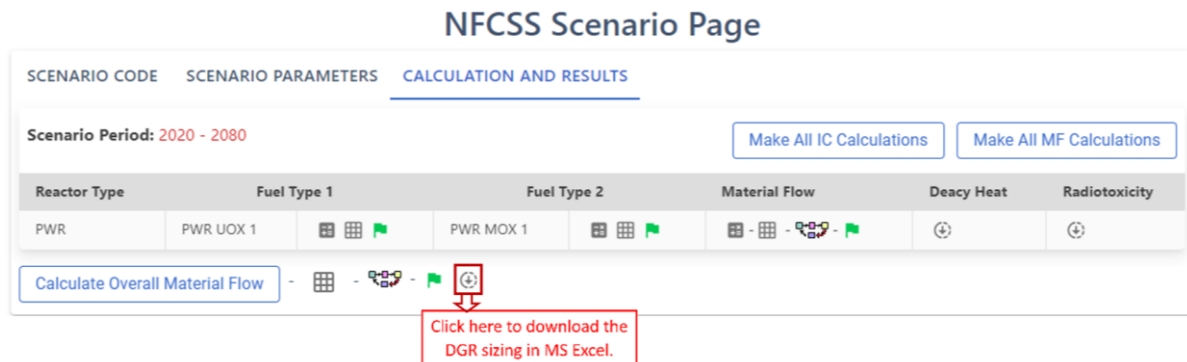


FIG.18 Screen after a calculation is completed without errors.

2.5.1. Isotopic composition calculation

To perform isotopic composition calculations for all reactor types,

- Click [**Make All IC Calculations**] button; or
- Click  button to perform isotopic composition calculation of each reactor type.

The calculation may take several seconds depending on the length or scenario period. The button is available for each of the fuel types for each of the reactor types.

The button  is used to see the result of isotopic composition calculation for each fuel type. If the isotopic composition is to be provided by the user,  button is again used. When the user clicks on  button, isotopic composition page is displayed (FIG.19). This page has three tab-pages for each of the cycle positions: 'Beginning of Cycle', 'End of Cycle' and 'After Cooling'.

The lower part of the isotopic composition page displays the data for each year and each isotope (and some isotope groups). Please note that all isotopic calculation results for BOC, EOC and After Cooling are given in weight percentage of the total fuel in the reactor core whereas the mass flow calculations and diagrams show absolute values in tons of radionuclides.

NFCSS Isotopic Composition Result

ReactorType: **PWR**
 Fuel: **PWR UOX 1**
 Neutron Flux: **1.41E14** (n/(cm2.s))

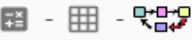
Export to Excel

BEGINNING OF CYCLE			END OF CYCLE		AFTER COOLING							
DataYear	Enrichment	Burnup	U235	U236	U238	Np237	Pu238	Pu239	Pu240	Pu241	Pu242	Am241
2020	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2021	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2022	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2023	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2024	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2025	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2026	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2027	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2028	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2029	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2030	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2031	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2032	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2033	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2034	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2035	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047
2036	4	45	0.679943	0.524254	93.046867	0.068695	0.024498	0.511368	0.263434	0.150562	0.06494	0.0047

FIG.19 Output of isotope composition calculations.

2.5.2. Material flow calculations for each reactor type

After completing isotopic composition calculations or providing the isotopic compositions,

- click [Make All MF Calculations] button; or
- The user can make material flow calculations for each reactor type. FIG.17 shows the command bar  for the material flow calculations, where the 1st button is used to perform material flow calculations for the selected reactor type, the 2nd button is used to see the material flow result in table form and the last button is used to see the material flow result in diagram form. If the isotopic compositions are not ready for any reactor type, the command bar will display a warning message.

When the user clicks on  button, NFCSS starts material flow calculations. The calculation is completed usually in less than a minute depending on the length of scenario period. During the calculations the NFCSS displays a wait message on the screen. After completing the calculations, the NFCSS returns to the Calculations and Results tab page.

Following the calculation, the user can see the results. The results can be displayed in two different formats: tabular and diagrammatic. When the user clicks  button, material flow result page is displayed for the selected reactor type (FIG.20). This page consists of four tab-pages: 'General', 'Front End', 'Back End' and 'Nuclide Group' tab pages provide the different results from the material flow calculation. The table displays the pre-selected result columns

from the NFCSS full result table. The selection of columns by the user will be implemented in future.

NFCSS Nuclear Material Flow Result

ReactorType: **PWR**

Export to Excel

GENERAL	FRONT END	BACK END	NUCLIDE GROUPS						
DataYear	TypeCapacity (MWe) 	Electricity (GWh) 	F2Share (%) 	Enri_1 (%) 	Bu_1 (GWd/t) 	Rep_1 (%) 	Enri_2 (%) 	Bu_2 (GWd/t) 	Rep_2 (%) 
2020	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2021	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2022	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2023	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2024	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2025	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2026	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2027	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2028	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2029	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2030	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2031	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2032	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2033	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2034	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000
2035	326	2285	30.000	4.000	45.00	100.000	7.000	45.00	100.000

FIG.20 Output of material flow calculations.

The user can see the graph for each of the result column by clicking on  button in each column heading of FIG.20. The chart form is shown in FIG.21.

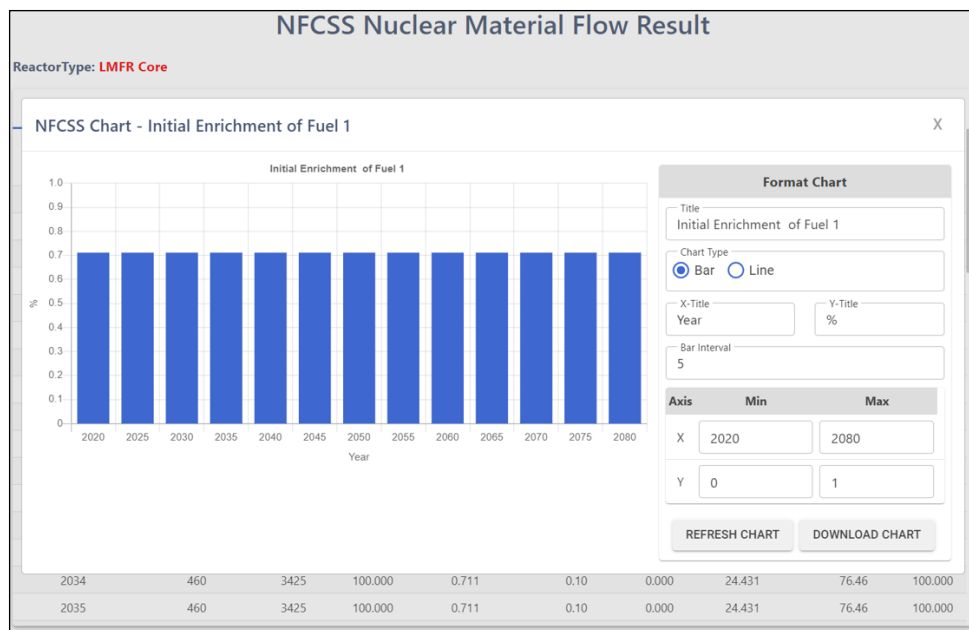


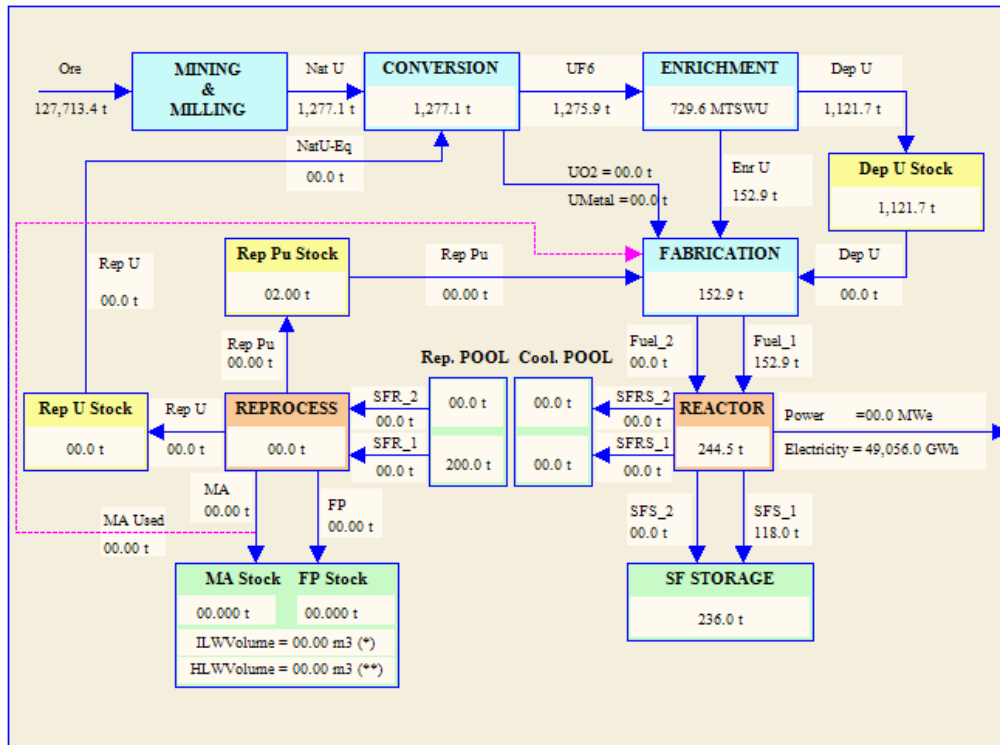
FIG.21 Graphical display of an output of material flow calculations.

The user can see calculation results of material flow in diagram by clicking on  button in FIG.18. The result is shown in FIG.22. The user can get the chart for results of each of the fuel cycle steps by clicking on the box of steps. The mouse cursor will automatically change when the mouse comes over a chartable parameter in the diagram.

Reactor Type: **PWR2**

Select a year to see the result for:

Year*
2020



(*) The factor to calculate LLW Volume is 0.35 m³/tonnes of HM.

(**) The factor to calculate HLW Volume is 0.15 m³/tonnes of HM.

FIG.22 Graphical display of material flow results.

The values shown are cumulative absolute values up to the end of the year selected from the drop-down year selection.

2.5.3. Overall material flow calculations

The material flow results of each reactor type can be combined using overall material flow calculations. [Calculate Overall Material Flow] button is used to make overall material flow calculations. Then, the result of the combination is displayed as table or diagram like the result for each reactor type (FIG.23).

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Make All IC Calculations Make All MF Calculations

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Decay Heat	Radiotoxicity
LMFR Core		REVISED LMFR MOX			
LMFR Axial	REVISED LMFR AXIAL BLANKET				
LMFR Radial	REVISED LMFR RADIAL BLANKET				

Calculate Overall Material Flow

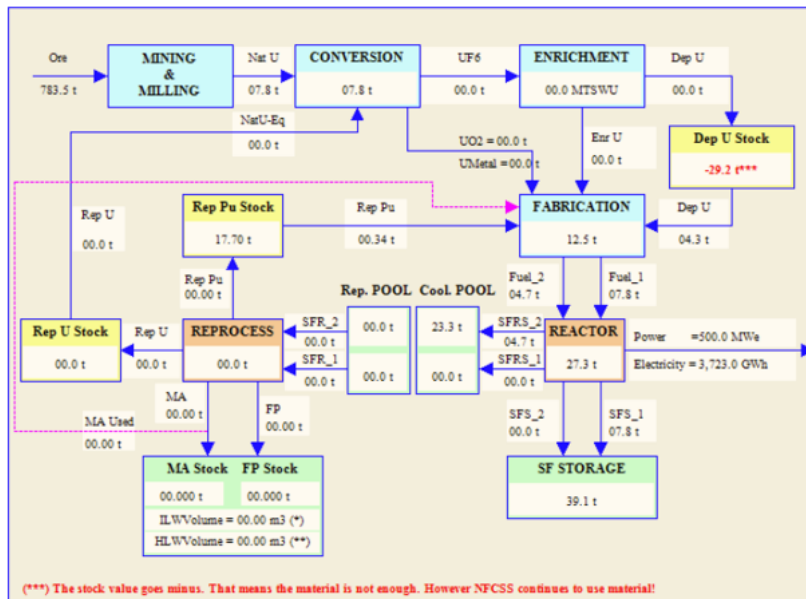


NFCSS Overall Nuclear Material Flow Result

Reactor Type: **All Types Combined**

Select a year to see the result for:

Year* 2025



(*) The factor to calculate LLW Volume is 0.35 m³/tonnes of HM.

(**) The factor to calculate HLW Volume is 0.15 m³/tonnes of HM.

(***) The stock value goes minus. That means the material is not enough. However NFCSS continues to use material!!

FIG.23 Overall material flow calculation and an example.

2.5.4. Decay heat and Radio-toxicity calculations

When the material flow calculation for each reactor type is completed, NFCSS can execute the calculation of decay heat and radiotoxicity. Click  button to start calculation and download the result table and graphs (FIG.24A) to users PC as Excel sheets. At this moment, decay heat and radiotoxicity calculations are only available for PWR and BWR.

For other types of reactors, Table 32 and Table 45 values, which are shown in TECDOC-1864, have to be entered using the templates given in the 'Help' tab, and the detailed steps for users to use Excel templates are shown in FIG.24B. Users can use the annual spent fuel discharge (SFD) and burnup calculated by NFCSS as "User input", and the values of table 32 and table 45 can be appropriately changed by the user.

Table 1 shows decay heat calculation for LMFR, which is being developed.

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS **CALCULATION AND RESULTS**

Scenario Period: 2000 - 2020
[Make All IC Calculations](#) [Make All MF Calculations](#)

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Decay Heat	Radiotoxicity
LMFR Core		REVISED LMFR MOX			
LMFR Axial	REVISED LMFR AXIAL BLANKET				
LMFR Radial	REVISED LMFR RADIAL BLANKET				

[Calculate Overall Material Flow](#) 



AutoSave  NFCSS_Decay_Heat (1... • Saved to this PC

File Home Insert Page Layout Formulas Data Review View Automate Help [Comments](#) [Share](#)

NFCSS Results for Reactor Type:							
		UO ₂ fuel	ThO ₂ fuel	MOX fuel	UO ₂ fuel	ThO ₂ fuel	MOX fuel
#	Time [Year]	SFS (tHM)	SFS (tHM)	SFS (tHM)	GWd/t	GWd/t	GWd/t
1	2000	0	0	0	0	0	136
2	2001	0	0	4.697952747	0	0	136
3	2002	0	0	4.697952747	0	0	136
4	2003	0	0	4.697952747	0	0	136
5	2004	0	0	4.697952747	0	0	136
6	2005	0	0	4.697952747	0	0	136
7	2006	0	0	4.697952747	0	0	136
8	2007	0	0	4.697952747	0	0	136
9	2008	0	0	4.697952747	0	0	136
10	2009	0	0	4.697952747	0	0	136
11	2010	0	0	4.697952747	0	0	136
12	2011	0	0	4.697952747	0	0	136
13	2012	0	0	4.697952747	0	0	136
14	2013	0	0	4.697952747	0	0	136
15	2014	0	0	4.697952747	0	0	136
16	2015	0	0	4.697952747	0	0	136
17	2016	0	0	4.697952747	0	0	136
18	2017	0	0	4.697952747	0	0	136
19	2018	0	0	4.697952747	0	0	136
20	2019	0	0	4.697952747	0	0	136
21	2020	0	0	4.697952747	0	0	136

Table 32 **User input** **DH_calculation results**



Name of the document	Title
User guide for use of Excel Templates	User guide for use of Excel Templates
Excel Template for calculating Decay heat	Excel Template for calculating Decay heat
Excel Template for calculating Decay heat Radiotoxicity	Excel Template for calculating Radiotoxicity
Guide book for beginner's use of NFCSS	Beginner's Guide Attachment to Beginner's Guide spreadsheet

Download the template which you need



A	B	C	D	E	F	G	H	I	J	K	L	M	N
Table 32													
year	1	3	10	30	100	3.00E+02	1.00E+03	3.00E+03	1.00E+04	3.00E+04	1.00E+05	3.00E+05	1.00E+06
UO ₂	1.141E+04	4.243E+03	1.580E+03	1.022E+03	4.038E+02	1.724E+02	7.017E+01	2.650E+01	1.560E+01	6.019E+00	1.184E+00	6.605E-01	5.018E-01
ThO ₂	8.881E+03	3.331E+03	1.595E+03	9.839E+02	2.574E+02	2.546E+01	1.225E+01	1.933E+01	3.549E+01	4.774E+01	3.920E+01	1.834E+01	1.289E+00
MOX	2.424E+04	8.616E+03	4.823E+03	3.413E+03	1.781E+03	9.190E+02	3.359E+02	1.079E+02	6.118E+01	2.297E+01	5.609E+00	3.608E+00	1.896E+00
Currently the above table is valid for PWR and BWR reactor types, for other reactor types these values can be appropriately changed by the user.													
The above table is reference values for PWR and BWR reactor types, for other reactor types these values can be appropriately changed by the user.													
Table 32	User input	cal_UO ₂	DH_UO ₂	cal_ThO ₂	DH_ThO ₂	cal_MOX	DH_MOX	DH_calculation results					



NFCSS Results for Reactor Type:								
#	Time [Year]	UO ₂ fuel	ThO ₂ fuel	MOX fuel	UO ₂ fuel	ThO ₂ fuel	MOX fuel	
		SFD (tHM)	SFD (tHM)	SFD (tHM)	GWd/t	GWd/t	GWd/t	
1	1970	19.66	19.66	19.66	45	45	45	45
2	1971	19.66	19.66	19.66	45	45	45	45
3	1972	19.66	19.66	19.66	45	45	45	45
4	1973	19.66	19.66	19.66	45	45	45	45
5	1974	19.66	19.66	19.66	45	45	45	45
6	1975	19.66	19.66	19.66	45	45	45	45
7	1976	19.66	19.66	19.66	45	45	45	45
8	1977	19.66	19.66	19.66	45	45	45	45
9	1978	19.66	19.66	19.66	45	45	45	45
10	1979	19.66	19.66	19.66	45	45	45	45
11	1980	19.66	19.66	19.66	45	45	45	45
12	1981	19.66	19.66	19.66	45	45	45	45
13	1982	19.66	19.66	19.66	45	45	45	45
14	1983	19.66	19.66	19.66	45	45	45	45
15	1984	19.66	19.66	19.66	45	45	45	45
16	1985	19.66	19.66	19.66	45	45	45	45
17	1986	19.66	19.66	19.66	45	45	45	45
18	1987	19.66	19.66	19.66	45	45	45	45
19	1988	19.66	19.66	19.66	45	45	45	45
20	1989	19.66	19.66	19.66	45	45	45	45
21	1990	19.66	19.66	19.66	45	45	45	45
22	1991	19.66	19.66	19.66	45	45	45	45
23	1992	19.66	19.66	19.66	45	45	45	45
24	1993	19.66	19.66	19.66	45	45	45	45
25	1994	19.66	19.66	19.66	45	45	45	45
26	1995	19.66	19.66	19.66	45	45	45	45
27	1996	19.66	19.66	19.66	45	45	45	45
28	1997	19.66	19.66	19.66	45	45	45	45
29	1998	19.66	19.66	19.66	45	45	45	45
30	1999	19.66	19.66	19.66	45	45	45	45
31	2000	19.66	19.66	19.66	45	45	45	45
32	2001	19.66	19.66	19.66	45	45	45	45
33	2002	19.66	19.66	19.66	45	45	45	45
34	2003	19.66	19.66	19.66	45	45	45	45
35	2004	19.66	19.66	19.66	45	45	45	45
36	2005	19.66	19.66	19.66	45	45	45	45
37	2006	19.66	19.66	19.66	45	45	45	45
38	2007	19.66	19.66	19.66	45	45	45	45
39	2008	19.66	19.66	19.66	45	45	45	45
40	2009	19.66	19.66	19.66	45	45	45	45
41	2010	80.81	80.81	80.81	45	45	45	45
42	2011	0	0	0	45	45	45	45
43	2012	0	0	0	45	45	45	45
44	2013	0	0	0	45	45	45	45
45	2014	0	0	0	45	45	45	45
46	2015	0	0	0	45	45	45	45
47	2016	0	0	0	45	45	45	45
48	2017	0	0	0	45	45	45	45

Users can use spent fuel discharge (SFD) and burnup of each year calculated by NFCSS as "User input".

Table 32

User input
cal_UO2
DH_UO2
cal_ThO2
DH_ThO2
cal_MOX
DH_MOX
DH_calculation results

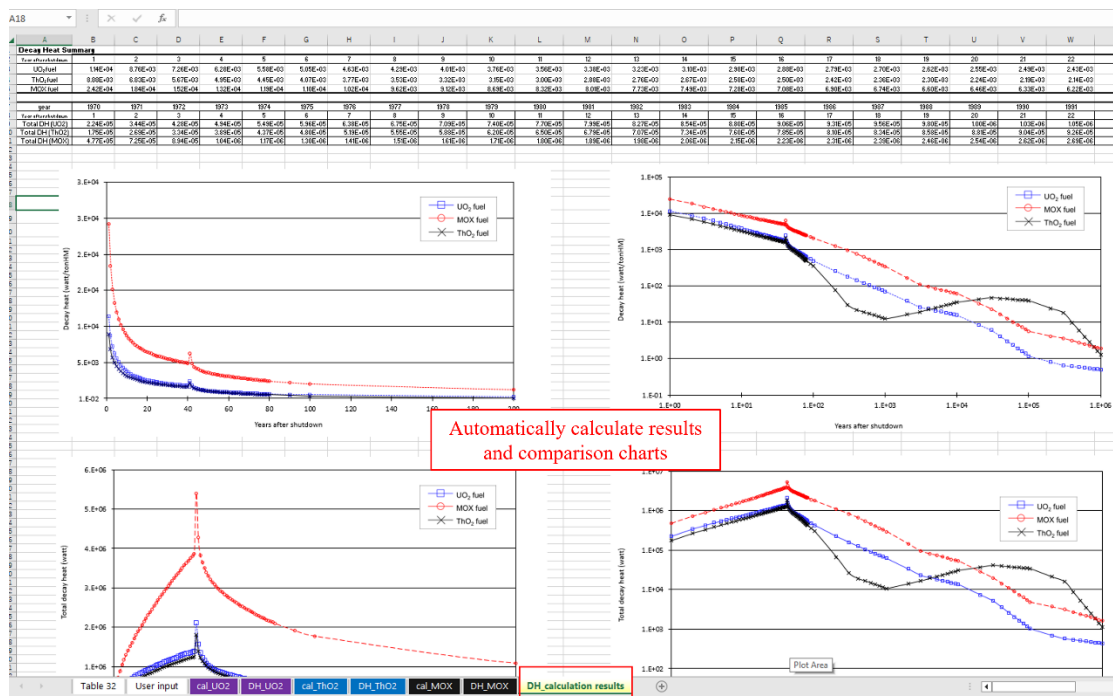


FIG.24B Decay heat and Radio-toxicity calculations by using EXCEL template

TABLE 1. DECAY HEAT FOR LMFR MOX FUEL (Watt/tonHM)

Year	1	10	100	1.00E+03	1.00E+04	1.00E+05	1.00E+06
Core 151448MWd/t	3.132E+04	9.378E+03	3.431E+03	1.027E+03	3.419E+02	1.650E+01	3.133E+00
Axial blanket 11828MWd/t	1.60E+03	3.463E+02	1.422E+02	9.812E+01	6.668E+01	4.561E+00	1.503E-01
Radial blanket 3807MWd/t	5.30E+02	1.347E+02	5.806E+01	4.448E+01	3.263E+01	2.372E+00	1.145E-01

The above values are decay heat at standard burnup shown in the Table. Decay heat at any other burnup is calculated internally, using linear relationship for different discharge burnup from standard burnup at core, axial blanket, and radial blanket. That is, if the discharge burnup is 2-times, then decay heat becomes 2-times.

2.5.5. DGR footprint calculations

NFCSS can execute the DGR footprint calculations as well when the material flow calculation for each reactor type is completed. Click a button to start calculation and download the result table to users PC as Excel sheets from calculation and results page. This template will require Spent fuel inventory or HLW inventory from reprocessing, number of fuel assembly per canister or canisters per tonne of HLW generated in vitrification, tunnel pitch which are governed by geo-technical analysis and package pitch which is determined from heat transfer calculations to limit the maximum temperature of the buffer material and skin temperature of canisters. In case of U, Pu, MA and fission products removal in reprocessing, the data will be given by NFCSS simulation exercise.

The table 2 shows the excel template for open cycle, and table 3 shows that for closed cycle. Green cells are inputs, and white cells are outputs. Among the inputs, the letters in yellow cells are the results obtained from NFCSS. Users can change inputs (values in green cells) and obtain the calculation results (values in white cells) automatically. The input values shown in table are Japan's case (DGR geological formation : Granite, 1,000m below the surface) as examples. The scenario considered here is a 1000 Mwe PWR operational for 60 years with other assumed parameters given in Table 4. In the open cycle, soft rock is excluded in Japan's case it cannot be used as a disposal site due to Decay heat constraints, thus the cells for soft rock in table 2 are not filled.

TABLE 2: DGR footprint calculation for open cycle by using EXCEL template

Open Cycle (Direct Disposal)			Disposal Type	
			Hard*	Soft
INPUT	NFCSS Output	1) SF Storage	[t]	1179.80
		SF Mass in Assembly	[kg/Assembly]	400.0
		No. of Assemblies per Cask	[Assemblies]	2
		Tunnel Pitch	[m]	20.00
		Package Pitch	[m]	6.18
OUTPUT		No. of Casks	[Casks]	1474.75
		No. of Casks per HM	[Casks/tHM]	1.25
		Occupied Area per Cask	[m ² /Cask]	123.60
		Disposal Area	[m ²]	182279.10

TABLE 3: DGR footprint calculations for closed cycle by using EXCEL template

Closed Cycle (Reprocessing)				Disposal Type	
				Hard*	Soft**
INPUT	NFCSS Output	1) Total SF Discharged	[t]	1179.80	1179.80
		2) MA Stock	[t]	1.402	1.402
		3) FP Stock	[t]	48.219	48.219
		4) Rep U Stock	[t]	979.30	979.30
		5) Rep Pu Stock	[t]	10.11	10.11
		Recovery Fraction of U and Pu	[%]	99.5	99.5
		Ratio of Vitrified Waste to HLW	[-]	1.13	1.13
		Waste Loading in Canister	[kg/Canister]	400.0	400.0
		HLW Content (Including 10.0wt% of Na ₂ O)	[wt%]	22.0	22.0
		Tunnel Pitch	[m]	10.00	9.99
		Package Pitch	[m]	4.44	3.13
	OUTPUT	HLW Mass in Canister	[kg/Canister]	42.3	42.3
		No. of Canisters	[Canisters]	1289.8	1289.8
		No. of Canisters per tHM	[Canisters/tHM]	1.09	1.09
		Occupied Area per Canister	[m ² /Canister]	44.40	31.27
		Disposal Area	[m ²]	57265.06	40328.92

NFCSS output in table 2, 3 are obtained by scenario based on NFCSS input in table 4 under the same conditions with Japan's standard for DGR [3][4]. The temperature of vitrified waste depend on conditions of fuel type, burnup, cooling time, etc. The occupied area per cask is determined according to the temperature of vitrified waste. Thus, input parameter in green cells in table 2,3 are only available in cases like this condition. Users need to change input values in green cells according to separate thermos-mechanical calculations for host rock.

TABLE 4: NFCSS input

	UOX
Enrichment (%)	4
Fuel Initial contents (% mass)	U-235 = 4, U-238 = 96
Share	100%
Burnup (GWd/tHM)	45
Reactor operation life	60 years
Neutron flux (n/ (cm ² .s)	2.99 E14
Power density (kW/kg)	37.5
Residence time	4
Load factor	80%
Efficiency	33
Tail Assay	0.30%
capacity	1000 MWe
Loss in conversion facility (%)	0.5
Loss in Enrichment facility (%)	0.3
Loss in Fabrication facility (%)	0.5
Loss in Reprocessing facility (%)	0.3
Volume of HLW generated in Reprocessing (m ³ /tHM)	0.15
Volume of LLW generated in reprocessing (m ³ /tHM)	0.35
RepU and RepPu recycling	No

3. SIMULATION CASE WITH THORIUM-BASED FUEL CYCLE IN A POWER REACTOR

3.1. GENERAL

This Section describes the actual input of the NFCSS for thorium (Th) fuel in a single PWR.

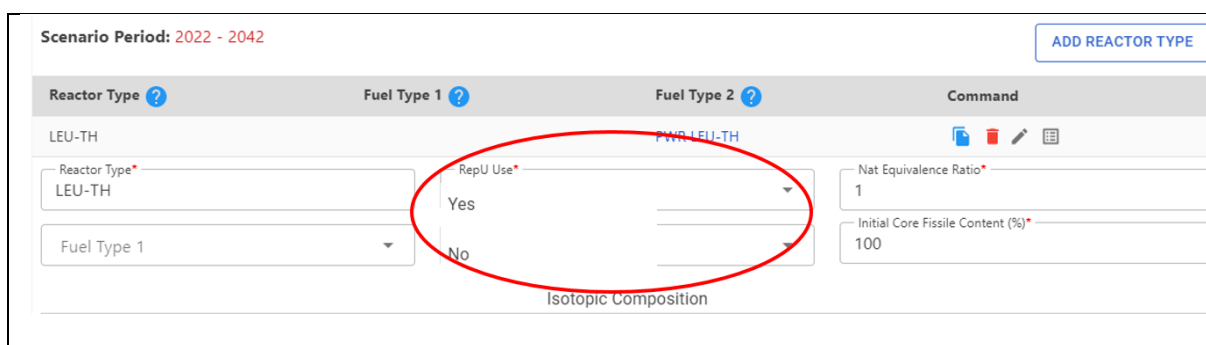
Since the starting procedure of the NFCSS is already described in this Guidebook, this section starts continued from the end of Section 2.2 (Fig.4-2nd; click [Create New Scenario] button). The first two steps, i.e., STEP-1 SIGNING INTO THE SYSTEM and STEP-2 CREATING A SCENARIO are identical as per 2.1 and 2.2. The following definitions should be kept in mind while setting scenario parameters for thorium fuel cycles.

FUEL TYPE DEFINITIONS (Other than described in TECDOC-1864, 2019)

NFCSS has currently three fuel types in its library, U Fuel, U-Pu Fuel and Free Content; the calculation of initial content is different for each fuel type. The way of input for the initial isotopic composition is different with the fuel types. In the case of “U Fuel,” users must input the U enrichment, in the case of “U-Pu Fuel,” must input total Pu enrichment and the isotopic composition of Pu (Pu-vector), and in the case of “Free Content,” must input isotopic composition of all nuclides. Thorium fuel is handled under ‘Free Content’ fuel, Conversion Type should be selected as ‘Th’ from the drop list. That is, the “fuel type” is related with the way of input for the initial isotopic composition.

There is another type of Type-1 and Type-2. These “Type” is referred to natural uranium requirement and SWU calculation. When users assign a fuel (any name of fuel which users defined) as Type-1, NFCSS calculates natural uranium requirement and SWU requirement from the initial isotopic composition. Generally, U fuel is set to fuel type 1, U-Pu fuel and Th fuel and MA containing fuel are set to fuel type 2.

There is one remark here for the above procedure. In a special scenario where reprocessed uranium is recycled to loaded fuels, users can assign it in NFCSS. As is shown in the following figure, if users select “YES” at “RepU use” column, then reprocessed uranium is used instead of natural uranium (FIG.25).



The screenshot shows the 'Scenario parameter for RU' form. At the top, it says 'Scenario Period: 2022 - 2042' and has an 'ADD REACTOR TYPE' button. Below this is a table with columns: 'Reactor Type', 'Fuel Type 1', 'Fuel Type 2', and 'Command'. The first row is for 'LEU-TH'. Under 'Reactor Type', there is a dropdown menu with 'LEU-TH' selected. Under 'Fuel Type 1', there is a dropdown menu with 'Fuel Type 1' selected. Under 'Fuel Type 2', there is a dropdown menu with 'PWR LEU-TH' selected. Under 'Command', there is a dropdown menu with 'RepU Use' selected, and a 'Yes' button is highlighted with a red circle. To the right of the 'RepU Use' dropdown, there are two input fields: 'Nat Equivalence Ratio' with a value of '1' and 'Initial Core Fissile Content (%)' with a value of '100'. At the bottom, there is a section for 'Isotopic Composition'.

FIG.25 Scenario parameter for RU

When users assign a fuel (any name of fuel which users defined) as Type-2, NFCSS regards the fuel does not need natural uranium for the fabrication. If the fuel includes uranium isotope, NFCSS regards it uses depleted uranium from enrichment facilities.

3.2 STEP 3: SETTING SCENARIO DESCRIPTION

Following [Step 2](#), rename the scenario code 'New Code' as 'PWR-LEU-Th' and set the start year and end year of the operation; in this case from 2020 to 2042 (FIG.26).

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2022 - 2042

Reset Form Save Description Export to Excel

Type the scenario name as you like

Scenario Code* PWR - LEU+TH

Scenario Type Standard

Scenario Explanation Overall mass flow comparison

Scenario Reference

Operation period for the calculation

Start Year* 2022

End Year* 2042

Other Scenario Characteristics

Loss in Conversion Facility (%)* 0

Volume of HLW Generated in Reprocessing (m3/tHM)* 0.15

Loss in Enrichment Facility (%)* 0.3

Volume of LLW Generated in Reprocessing (m3/tHM)* 0.35

Loss in Fabrication Facility (%)* 0.1

Loss in Reprocessing Facility (%)* 0.1

FIG.26 Scenario description.

3.3 STEP 4: SETTING SCENARIO PARAMETERS

3.3.1 Creating a new fuel type

Next action is to define Th (Thorium) fuel data, according to Section 2.4.2.

The input parameter 'fuel type' provides information on the type of fuel. Currently the NFCSS considers three types: 1) Uranium (U) fuel, 2) Uranium-plutonium (MOX) fuel, and 3) Free content fuel. For Thorium fuel, free content is recommended.

Thorium fuel cycles are only considered for PWRs and BWRs in the NFCSS for now. As is explained in TECDOC-1864 (Annex I.3), cross sections of PWR-Th fuel and BWR-Th fuel are identical. Also, cross sections of LEU-Th fuel and U233-Th fuel are identical.

There are several fuels (cross section libraries) are available in the NFCSS as listed in Table 4:

TABLE 4. AVAILABLE THORIUM FUELS IN NFCSS.

LWR-Th-U	ID 5009 (PWR-LEU-ThOX1)	ID 5013 (PWR-U233-ThOX1)
	ID 5015 (BWR-LEU-ThOX1)	ID 5018 (BWR-U233-ThOX1)
LWR-Th-Pu	ID 5014 (PWR-Pu-ThOX1)	ID 5019 (BWR-Pu-ThOX1)

Here, ID 5013 (PWR U233-ThOX1) is considered. Therefore, creating a new fuel case by copying ID 5013 case is shown in FIG.27.

IAEA | INFCS

NFCFDB UDEPO PIEDB NFCSS USERNAME

NUCLEAR FUEL CYCLE SIMULATION SYSTEM

About Modeling Example Calculation Scenarios Help References

Scenario List **Fuel List** Reactor List

List of Available NFCSS Fuel for User: YOUR USERNAME

CREATE NEW FUEL Search for Fuels...

Fuel Id	Fuel Type	Owner	Command
1	PWR UOX 1	System	
2	PWR MOX 1	System	
3	PWR UOX 2	System	
4	PWR UOX 3	System	
5	BWR UOX 1	System	
6	BWR MOX 1	System	

↓

Fuel Id	Fuel Type	Owner	Command
21	ADS 1	System	
22	LMFR Core MOX	System	
23	LMFR Axial Blanket	System	
54	LMFR Radial Blanket	System	
5009	PWR LEU-ThOX 1	System	
5013	PWR U233-ThOX 1	System	
5014	PWR Pu-ThOX	System	
5015	BWR LEU-ThOX 1	System	
5018	BWR U233-ThOX 1	System	
5019	BWR Pu-ThOX 1	System	

Thorium Fuel Data: Preset Thorium fuels by the system fuel 5009-t0-5019

Fuels per page 10 21-30 of 52

↓

Fuel Id	Fuel Type	Owner	Command
21	ADS 1	System	
22	LMFR Core MOX	System	
23	LMFR Axial Blanket	System	
54	LMFR Radial Blanket	System	
5009	PWR LEU-ThOX 1	System	
5013	PWR U233-ThOX 1	System	
5014	PWR Pu-ThOX	System	

PRESS HERE if you want to create your own fuel based on fuel #5013

FIG.27 New fuel type creation process

For editing the new fuel data,

- Change the fuel name to 'PWR-LEU-TH' (Note that this name is just an example) – see FIG.28 (1st);
- Select [Free Content] from fuel type (FIG.28, 1st);

- (c) Set the cooling period as 6 years. If reprocessing is NOT considered, this column data can be any number (FIG.28, 1st);
- (d) Fissile contents of the original fuel will be 4% 233U-Th fuel (4% U-233+96% Th-232) – see FIG.22. Note that reference enrichment column is invalid and will not be used for the calculation (FIG.28, 1st).
- (e) Click [Save Fuel Data] button after setting (FIG.28, 2nd).

NFCSS Fuel Page

5493

RESET FORM DELETE FUEL **SAVE FUEL DATA**

FUEL CHARACTERISTICS CROSS SECTIONS

Fuel Characteristics

Fuel Type Code*
PWR LEU-TH

Fuel Type*
Free Content

Reference Enrichment*
26 %

Power Density*
37.5 kW/kg

Conversion Type*
TH

Cooling Time*
6 yr

Manufacturing Time*
1 yr

Reprocessing Time*
1 yr

Fuel Type Description
PWR LEU-ThOX

Information Source

Initial guess of Neutron Flux*
2.55E+14 n/(cm2.s)

Fuel Initial Contents (*) (% Mass)

U235*
0

U236*
0

U238*
0

Np237*
0

Pu238*
0

FIG.28 Editing fuel data

3.3.2 Setting scenario data

- (a) Click [Scenario Parameters] tab, and click [Add Reactor Type] button (FIG.29, 1st);

- (b) Name the reactor type as 'PWR' and select aforementioned 'PWR-LEU-TH' for Fuel Type 1 or 2 (FIG.29, 2nd). Currently, it is recommended to select from Fuel Type 2 for thorium fuel cycle estimations if along with U fuel as Type Fuel-1.
- (f) Click the drop-down box of each item in isotopic composition, the user also can use excel template to customize isotope compositions in all 3 stages(s (Initial content, depletion and Decay)) by choosing "Provided" option, as shown in Fig.10.
- (c) In case of using [Free content], initial core fissile content will not be defined. Therefore, set as 100% (FIG.29, 2nd);
- (d) Click [Save] button (rightmost button of the following screen) (FIG.29, 3rd);
- (e) The screen is shown as the FIG.29 3rd figure. Then, click the scenario parameter button to endit the scenario data (FIG.29, 3rd).

NFCSS Scenario Page

SCENARIO CODE **SCENARIO PARAMETERS** CALCULATION AND RESULTS

Scenario Period: 2022 - 2042

Reactor Type ? Fuel Type 1 ? Fuel Type 2 ? Command

There is no reactor type selected for this scenario!

ADD REACTOR TYPE



NFCSS Scenario Page

SCENARIO CODE **SCENARIO PARAMETERS** CALCULATION AND RESULTS

Scenario Period: 2022 - 2042

ADD REACTOR TYPE

Reactor Type ? Fuel Type 1 ? Fuel Type 2 ? Command

PWR PWR_LEU_TH

Reactor Type* PWR

Fuel Type 1 Fuel Type 2 PWR_LEU_TH

RepU Use* Yes

Nat Equivalence Ratio* 1

Initial Core Fissile Content (%) 100

Isotopic Composition

Initial Content* Calculated

Depletion* Calculated

Decay* Calculated

Calculated

Provided

DO NOT forget to save before moving to other steps

RESET CANCEL SAVE CHANGES



NFCSS Scenario Page

SCENARIO CODE **SCENARIO PARAMETERS** CALCULATION AND RESULTS

Scenario Period: 2022 - 2042

ADD REACTOR TYPE

Reactor Type ? Fuel Type 1 ? Fuel Type 2 ? Command

PWR PWR_LEU_TH

PRESS THIS BUTTON: It enables you to set scenario parameters

FIG.29 Editing reactor type information.

Then, the screen is displayed as shown in FIG.30

NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 2022 - 2042

User can also use the EXCEL template to prepare the data to upload Scenario Data

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE		INITIAL VALUES		
Data Year	Added Capacity	Removed Capacity	Net Total Capacity	Efficiency	Load Factor	Tails Assay	Fuel Type 2 Share	Residence Time
	(MW _e)	(MW _e)	(MW _e)	(%)	(%)	(%)	(%)	(year)
2022	1000	0	1000	33	80	0.3	100	4
2023	0	0	1000	33	80	0.3	100	4
2024	0	0	1000	33	80	0.3	100	4
2025	0	0	1000	33	80	0.3	100	4
2026	0	0	1000	33	80	0.3	100	4
2027	0	0	1000	33	80	0.3	100	4
2028	0	0	1000	33	80	0.3	100	4
2029	0	0	1000	33	80	0.3	100	4
2030	0	0	1000	33	80	0.3	100	4

PREPARE THIS BUTTON: It enable you to copy the parameters
 User can also use the EXCEL template to prepare the data to upload Scenario Data

FIG.30 Editing scenario parameters.

Similarly fill all other data for tabs for 'Fuel Type 1, Fuel Type 2, Fuel Cycle, Initial Values as explained in 2.3.3. 'Import from excel' option is also available, this data will automatically get filled by the system, if pre-filled excel in the template as prompted is chosen.

Here, let's consider an operation of PWR with capacity of 1000 MWe from 2022 to 2042.

Parameters such as plant efficiency, burnup cannot be varied for the NFCSS simulation, therefore, provide input of the data for the year 2022 only. Also, set [Add Capacity] as 0.0 for years other than the first year. For other parameters, set as follows:

- Capacity:1000MWe
- Efficiency: 33%;
- Load Factor: 80%;
- Tail Assay: 0.3%;
- Fuel Type2 share: 100%;
- Fuel Residence Time: 4 yr;
- Tab Fuel Type 1: NA;
- Tab Fuel Type2: no need to set [enrichment] value;
- Discharge burnup: 45GWd/t;
- Reprocessing: 100% (In case of considering reprocessing) .

Tab Fuel Cycle [Reprocessing1] and [Reprocessing2] is used when setting the reprocessing amount, therefore, set a large number when considering reprocessing. For example, 1000t. No

need to input Reprocessing 1 when there is no 'Fuel Type 1'. FIG.31 shows all key scenario input parameters.

As mentioned in the previous chapter, these data can be copied by using  button located in the left side of each year in FIG.30. when clicking  button of one year, all the data blows will be copied from this year.

After filling out the form, don't forget to click on [SAVE DATA SET].

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2022 - 2042 ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
Reactor Type* PWR	RepU Use*		Nat Equivalence Ratio* 1
Fuel Type 1	Fuel Type 2 PWR_LEU_TH		Initial Core Fissile Content (%)* 100

Isotopic Composition

Initial Content* Calculated	Initial Content* Calculated
Depletion* Calculated	Depletion* Calculated
Decay* Calculated	Decay* Calculated

CANCEL
SAVE CHANGES



NFCSS Scenario Data Page

ReactorType: PWR
Scenario Period: 2022 - 2042

IMPORT SCENARIO DATA
RESET FORM
SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES			
Data Year	Added Capacity (MWe) 	Removed Capacity (MWe) 	Net Total Capacity (MWe) 	Efficiency (%) 	Load Factor (%) 	Tails Assay (%) 	Fuel Type 2 Share (%) 	Residence Time (year) 
  2022	1000	0	0	33	80	0.3	100	4
  2023	0	0	0	33	80	0.3	100	4
  2024	0	0	0	33	80	0.3	100	4
  2025	0	0	0	33	80	0.3	100	4
  2026	0	0	0	33	80	0.3	100	4
  2027	0	0	0	33	80	0.3	100	4
  2028	0	0	0	33	80	0.3	100	4
  2029	0	0	0	33	80	0.3	100	4
  2030	0	0	0	33	80	0.3	100	4



NFCSS Scenario Data Page

ReactorType: PWR

Scenario Period: 2022 - 2042

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS	FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Data Year	Enrichment 2f (%)	Discharge Burnup 2f GWD/t (t)	Reprocessing Ratio 2f (%)	
2022	4	45	100	
2023	4	45	100	
2024	4	45	100	
2025	4	45	100	
2026	4	45	100	
2027	4	45	100	
2028	4	45	100	
2029	4	45	100	
2030	4	45	100	



NFCSS Scenario Data Page

ReactorType: PWR

Scenario Period: 2022 - 2042

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1		FUEL TYPE 2		FUEL CYCLE	INITIAL VALUES	
Data Year	NatU Capacity	Conversion Capacity	Enrichment Capacity	Fabrication Capacity	Reprocessing Capacity	Fabrication Capacity 2f	Reprocessing Capacity 2f	
	(tHM) 	(tHM) 	(MTSWU) 	(tHM) 	(tHM) 	(tHM) 	(tHM) 	
2022	0	0	0	0	0	0	500	
2023	0	0	0	0	0	0	500	
2024	0	0	0	0	0	0	500	
2025	0	0	0	0	0	0	500	
2026	0	0	0	0	0	0	500	
2027	0	0	0	0	0	0	500	
2028	0	0	0	0	0	0	500	
2029	0	0	0	0	0	0	500	
2030	0	0	0	0	0	0	500	



NFCSS Scenario Data Page

ReactorType: PWR

Scenario Period: 2022 - 2042

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Initial Value Type	Initial Value	Insertion Year	Explanation		
Initial Reactor Power (MWe) :	0	N/A	The total power of reactors which are operational in equilibrium conditions at the beginning of the scenario period.		
Initial Rep. Pu Stock (t HM) :	0	N/A	The total amount of Reprocessed Pu which is available for recycling at the beginning of scenario period.		
Initial SF Stock - Fuel Type 1 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 1) which is ready for reprocessing at the beginning of the scenario period.		
Initial SF Stock - Fuel Type 2 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 2) which is ready for reprocessing at the beginning of the scenario period.		

FIG.31 Scenario input parameters.

3.4 STEP 5: CALCULATION AND RESULTS

First, select [Calculation and Result] tab to see the calculation results.

IC (Isotopic Composition) calculation is still not finished. To calculate isotopic composition of spent fuel, using button, or clicking [Make All IC Calculations] button.

Refer to FIG.17 for graphical explanation of calculation procedures. Users should be caution with calculation order, first isotope compositions and second material flow for all reactors individually and overall material flow.

The user can export to Microsoft Excel to locally store the scenario data and results. In order to export scenario description including all the data to Microsoft Excel, [Export to Excel] button is used (FIG.32). Please note that all isotopic calculation results for BOC, EOC and After Cooling are given in weight percentage of the total fuel in the reactor whereas the mass flow calculations and diagrams (FIG.33) show absolute values of nuclides.

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2022 - 2042

Reset Form Save Description **Export to Excel**

Scenario Code*
PWR - LEU+TH

Scenario Type
Standard

Start Year*
2022

End Year*
2042

Scenario Explanation
Overall mass flow comparison

Scenario Reference

Other Scenario Characteristics

Loss in Conversion Facility (%)*
0

Loss in Enrichment Facility (%)*
0.3

Loss in Fabrication Facility (%)*
0.1

Loss in Reprocessing Facility (%)*
0.1

Volume of HLW Generated in Reprocessing (m3/tHM)*
0.15

Volume of LLW Generated in Reprocessing (m3/tHM)*
0.35

FIG.32 Export to Excel spreadsheet.

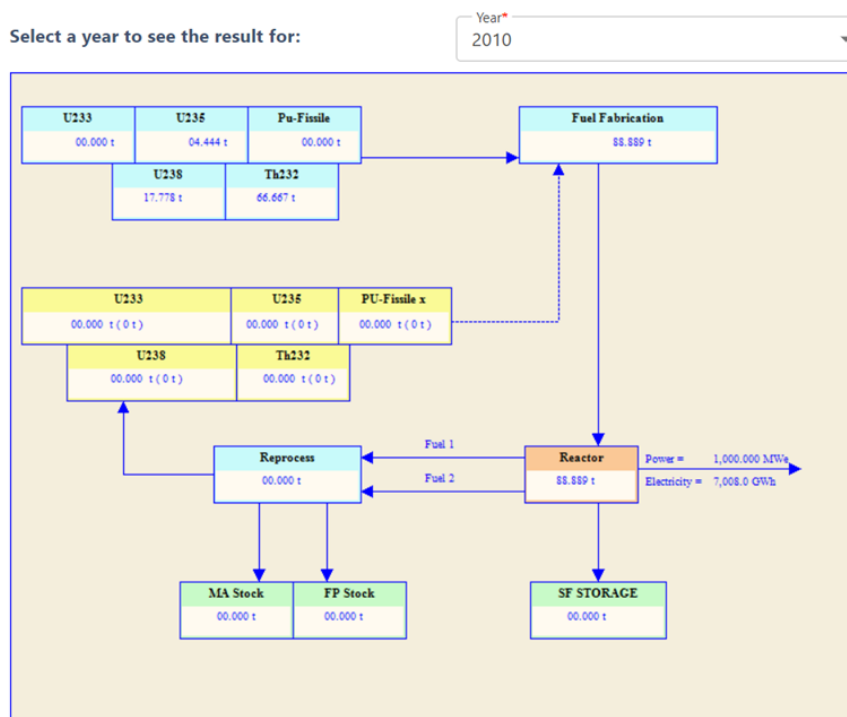


FIG.33 Material Flow Diagram for Thorium Fuel Cycle

NB: For mix of uranium and thorium fuel cycle the material flow calculations and overall material flow diagram are under progress.

4 SIMULATION CASE WITH A COMPLICATED SCENARIO

4.1 GENERAL

This Section describes how to run the NFCSS for a complicated scenario such as a fleet of dozens of the following four reactor types, that is a combination of PWR with MOX(&UO₂) fuel, PWR with UO₂ fuel, BWR with MOX(&UO₂) fuel, and BWR with UO₂ fuel. STEP 1 and STEP 2 are common as described in 2.1 and 2.2.

4.2 STEP 3: SETTING SCENARIO DESCRIPTION

Since the starting procedure of the NFCSS is already described in this Guidebook, this Section starts from the screenshot after creating a new scenario which is named as ‘MSR Case 2’ (FIG.34).

List of Available NFCSS Scenarios for User:

Scenario Code	Scenario Explanation	Command
PWR - LEU+TH	Overall mass flow comparison	 
URAM PWR	For conference paper	 
URAM MOX	For conference paper	 
MSR	For Trial	 
MSR Case 2	Trial	 
FBRIN WITH BLANKETS	Fast reactors	 
China Fleets	SF of China	 
projected overall China Fleets	SF of China	 
MSR CASE 2		 

Scenarios per page 10 21-29 of 29

FIG.34 A new scenario code creation.

After creating a new scenario, click [Scenario Parameters] tab to specify reactor, fuel types and data for the scenario parameters for each of the reactor and fuel type. And click [Add Reactor Type] button as shown in FIG.35.

Scenario List | Fuel List | Reactor List | Selected Scenario (17328)

NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR	PWR UOX 1	PWR MOX 1	   

FIG.35 Editing Scenario page.

4.3 STEP 4: SETTING SCENARIO PARAMETERS

4.3.1 Creating a new fuel type

As is described in TECDOC-1864, Table 5 shows a classification of fuel types (Type 1, Type 2) for UO₂ fuel cycle, MOX fuel cycle and thorium fuel cycle.

TABLE 5. CLASSIFICATION OF FUEL TYPES

Type	UO ₂ -cycle	MOX-cycle	Th-cycle
Type-1	LEU fuel	(LEU fuel)	(LEU fuel)
Type-2	-----	MOX fuel	Th-fuel

In this simulation, select ‘PWR UOX1’ for Type-1, and select ‘PWR-MOX1’ for Type-2. Since ‘PWR UOX1’ is in the NFCSS system library (Table 6), just select ‘PWR UOX1’ among the fuel list (FIG.36).

TABLE 6. NFCSS SYSTEM FUEL LIBRARIES

Fuel ID	Type	Fuel Composition
1	PWR UOX1	4% U235 + 96% U238- CS 1
2	PWR MOX1	0.3% U235 + 1.3652778% Pu238 + 60.104862% Pu239 + 21.629862% Pu240 + 11.7% Pu241 + 5.2% Pu242
3	PWR UOX2	4% U235 + 96% U238 CS 2
4	PWR UOX3	4% U235 + 96% U238 CS 3
5	BWR UOX1	4% U235 + 96% U238 CS 1
6	BWR MOX1	0.3% U235 + 1.3652778% Pu238 + 60.104862% Pu239 + 21.629862% Pu240 + 11.7% Pu241 + 5.2% Pu242
7	BWR UOX 2	4% U235 + 96% U238 CS 2
13	WWER UOX 1	4% U235 + 96% U238
14	WWER MOX 1	0.3% U235 + 0.90555555% Pu238 + 61.145832% Pu239 + 21.95764% Pu240 + 10.838195% Pu241 + 4.0590277% Pu242
15	WWER MOX 2	0.3% U235 + 0.02013889% Pu238 + 93.9625% Pu239 + 5.789583% Pu240 + 0.17847222% Pu241 + 0.029861111% Pu242
16	PBMR U 1	9.885% U235 + 90.115 % U238 CS 1
17	GTMHR U1	10% U235 + 90 % U238 CS 1
18	PBGCFR MOX 1	0.3% U235 + 1.4930556% Pu238 + 55.19097% Pu239 + 25.747223% Pu240 + 13.465972% Pu241 + 4.102778% Pu242
19	EFR Core MOX	0.2% U235 + 2.2722223% Pu238 + 59.052082% Pu239 + 25.893055% Pu240 + 6.80625% Pu241 + 5.977083% Pu242
20	EFR Blanket U	0.2% U235 + 99.8% U238 CS 1
22	LMFR Core MOX	0.25% U235 + 4% Pu238 + 48.9% Pu239 + 32.4% Pu240 + 3% Pu241 + 11.8% Pu242
5009	PWR LEU-ThOX1	5% U235 + 20% U238 +75% Th 232
5013	PWR U233-ThOX1	4% U233 +96% Th 232

5014	PWR Pu-ThOX	0.23%Pu238+5.11%Pu239+2.45%Pu240+ 0.89% Pu241 +0.67% Pu242+90.65%Th232
5015	BWR LEU-ThOX1	5% U235 + 20%U238 +75%Th 232
5018	BWR U233-ThOX1	4% U233 +96% Th 232
5019	BWR Pu-ThOX 1	0.23%Pu238+ 5.11% Pu239 +2.45% Pu240 + 0.89% Pu241 +0.67% Pu242 +90.65%Th232
5543	MSR-U233-TH	1.9696 %U233 +98.0304 %Th
5544	MSR Pu-Th	0.1584%Pu238+3.5229%Pu239+1.6852%Pu240+0.6124 %Pu241+0.4625%Pu242+93.5586%Th
5545	MSR-LEU-Th	0.1584%Pu238+3.5229%Pu239+1.6852%Pu240+0.6124 %Pu241+0.4625%Pu242+98.0304 %Th
5546	MSFR	11.154%U233+88.846%Th
5562	PWR-MA+UOX	5.153%U235+93.884%U238+0.588%Np237+0.233% Am241+0.142%Am243
5564	PWR-MA+MOX	0.174%U235+86.489%U238+0.588%Np237+0.367%Pu 238+6.194%Pu239+3.47%Pu240+1.453%Pu241+0.89% Pu242+0.233% Am241+0.142%Am243
5567	BWR-MA+UOX	5.153%U235+93.884%U238+0.588%Np237+0.233% Am241+0.142%Am243
5569	BWR-MA+MOX	0.174%U235+86.489%U238+0.588%Np237+0.367%Pu 238+6.194%Pu239+3.47%Pu240+1.453%Pu241+0.89% Pu242+0.233% Am241+0.142%Am243
5672	LMFR_CORE_1500 _JAEA	0.2463%U235+82.8627%U238+0.1858%Pu238+9.1374 %Pu239+5.4221%Pu240+0.7263%Pu241+0.6586%Pu24 2+0.3381%Am241+0.1690%Am243+0.1690%Am244
5673	LMFR_Axial blanket_1500_JAEA	0.3%U235+99.7%U238
5674	LMFR_Radial blanket_1500_JAEA	0.3%U235+99.7%U238

Note:Fuel IDs 5533, 5534, 5538, 5539, 5540, 5541, 5548 are not be used for simulation purpose as there is an error in the values.

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080
ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR	PWR UOX 1		
Reactor Type* PWR	RepU Use* Yes	Nat Equivalence Ratio* 1	
Fuel Type 1* PWR UOX 1	Fuel Type 2	Initial Core Fissile Content (%)* 70	
Isotopic Composition			
Initial Content* Calculated	Initial Content* Provided		
Depletion* Calculated	Depletion* Calculated		
Decay* Calculated	Decay* Calculated		
RESET CANCEL SAVE CHANGES			

The "Calculated" option is generally selected; selecting the "Provided" option allows the user to customize the input of isotopic compositions in all 3 stages.

FIG.36 Selection of Fuel Type 1 from in-line library.

However, 'PWR-MOX-new' is a new fuel and should be created by the user according to the following procedures:

- (a) Click [Fuel List] tab (FIG.37, 1st), and click [Command] button for copy at the rightmost of 'PWR MOX 1' row. Then, 'New Fuel' is shown at the bottom of the fuel list with identical fuel data for 'PWR MOX 1' fuel (FIG.37, 2nd). Then, it is possible to modify fuel data (enrichment, specific power etc.) from the system file data;
- (b) Change the name 'New Fuel' to 'PWR-MOX-new' (FIG.37, 3rd);
- (c) The following fuel data can be modified, and here, "Fuel Initial Contents" of each Pu nuclide weight % is modified by the user's data (FIG.37, 3rd). One remark here is that the total % must be 100.0 %; The 'Import fuel cross sections' option is also available through an excel template as prompted by the system.
- (d) Finally, click [Save Fuel Data] button (FIG.37, 3rd).

After clicking [Save Fuel Data] button, fuel name and fuel data are modified and stored as 'PWR-MOX- new' at the bottom of the fuel list, and it is defined as 'Owner is User'.

Now, the user can select 'PWR-MOX-new' for Type-2. That is,

- (a) Click the rightmost [Save] button among the command list in the following screen, and save the user specified reactor data, which are 'PWR UOX1' for Type-1 and 'PWR-MOX-new for Type-2 (FIG.38, 1st and 2nd);
- (b) Here, the input of the 'Initial Core Fissile Content (%)' is assumed as 70% for this case (FIG.38, 1st and 2nd);
- (c) Go back to reactor list by clicking [Scenario Parameter] tab, and confirm that new reactor is added to the list. That is, the first reactor type 'PWR-MOX' is introduced in the scenario with assigned fuel data (FIG.38, 3rd).

List of Available NFCSS Fuel for User:

CREATE NEW FUEL

Fuel Id	Fuel Type	Owner	Command
1	PWR UOX 1	System	
2	PWR MOX 1	System	
3	PWR UOX	System	
4	PWR UOX 3	System	
5	BWR UOX 1	System	
6	BWR MOX 1	System	



CREATE NEW FUEL

Fuel Id	Fuel Type	Owner	Command
5528	PHWR - CHINA	USERNAME	
5531	MSR Pu-Th	USERNAME	
5539	New Fuel	YOUR USER NAME	

Fuels per page 10 51-53 of 53 |< < > >|

New fuel appears to the bottom of the list



NFCSS Fuel Page

Fuel Id 5595 **IMPORT CROSS SECTIONS** **RESET FORM** **DELETE FUEL** **SAVE FUEL DATA**

FUEL CHARACTERISTICS **CROSS SECTIONS**

The user can use the EXCEL template to import cross sections

Fuel Characteristics Fuel Type Code* PWR-MOX-new Fuel Type* U Fuel Reference Enrichment* 4 % Power Density* 37.5 kW/kg Conversion Type* None Cooling Time* 6 yr Manufacturing Time* 1 yr Reprocessing Time* 1 yr Fuel Type Description PWR UOX Fuel from ORIGEN Library Information Source PWRUE from ORIGEN Library Initial guess of Neutron Flux* 2.99E+14 n/(cm2.s)	Fuel Initial Contents (*) (% Mass) U235 4 U236 0 U238 96 Np237 0 Pu238 0 Pu239 0 Pu240 0 Pu241 0 Pu242 0 Am241 0 Am242m 0 Am243 0 Cm242 0 Cm244 0 Th232 0 Pu231 0 U233 0 U234 0
---	--

FIG.37 new fuel creation.



NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2020 - 2080 ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR	PWR UOX 1	PWR-MOX-new	   
Reactor Type* PWR	Repl Use* Yes	Nat Equivalence Ratio* 1	
Fuel Type 1 PWR UOX 1	Fuel Type 2 PWR-MOX-new	Initial Core Fissile Content (%)* 70	
Isotopic Composition			
Initial Content* Calculated	Initial Content* Calculated		
Depletion* Calculated	Depletion* Calculated		
Decay* Calculated	Decay* Calculated		

RESET CANCEL **SAVE CHANGES**



NFCSS Scenario Page

SCENARIO CODE SCENARIO PARAMETERS CALCULATION AND RESULTS

Scenario Period: 2020 - 2080 ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR	PWR UOX 1	PWR-MOX-new	   
Reactor Type* PWR	Repl Use* Yes	Nat Equivalence Ratio* 1	
Fuel Type 1 PWR UOX 1	Fuel Type 2 PWR-MOX-new	Initial Core Fissile Content (%)* 70	
Isotopic Composition			
Initial Content* Calculated	Initial Content* Calculated		
Depletion* Calculated	Depletion* Calculated		
Decay* Calculated	Decay* Calculated		

RESET CANCEL **SAVE CHANGES**



NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080
ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR	PWR UOX 1	PWR-MOX-new	+ - ✎ ⌄

FIG.38 Selection of Fuel Type 1 and Fuel Type 2 to define Reactor Type.

Repeat above steps and create multiple reactors. As an example, the following 4 reactors are added, as is shown in the screen in FIG.39:

- (1) PWR-MOX, loading both UO₂ fuel and MOX fuel;
- (2) BWR-MOX, loading both UO₂ fuel and MOX fuel;
- (3) UO₂-PWR, loading only UO₂ fuel;
- (4) UO₂-BWR, loading only UO₂ fuel;

For the above 4 reactors, 2 fuels ‘PWR UOX 1’ and ‘BWR UOX 1’, whose parameters have been determined by the system, are used. Additionally, ‘PWR-MOX- new’ and ‘BWR-MOX-new’, which are created by the user, are used.

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080
ADD REACTOR TYPE

Reactor Type ?	Fuel Type 1 ?	Fuel Type 2 ?	Command
PWR 1	PWR UOX 1	PWR-MOX-new	+ - ✎ ⌄
PWR 2	PWR UOX 1		+ - ✎ ⌄
BWR 3	BWR UOX 1	BWR-MOX-new	+ - ✎ ⌄
BWR 4	BWR UOX 1		+ - ✎ ⌄

PRESS THESE
 BUTTONS: It enables you
 to set plant's rated power,
 capacity, enrichment
 ratio, loading ratio of fuel
 type 2, reprocessing
 condition

FIG.39 Complex scenario with four different reactor types.

4.3.2 Setting scenario data

As is explained in the above screen, if the button in the command list is clicked (see FIG.38), the user can modify the various data, such as introduction/removal of the electric power generation, thermal efficiency, load factor, U235 enrichment or Pu enrichment, discharged burnup, share of Type-2 fuel, residence year, reprocessing share, reprocessing capacity.

In this example, major parameters for the above 4 fuels are:

For Fuel Type 1,

- PWR UOX 1: Enrichment ratio 4%, Specific power 37.5 MW/t, Cooling period 6 years;
- BWR UOX 1: Enrichment ratio 4%, Specific power 25.9 MW/t, Cooling period 6 years.

For Fuel Type 2,

- PWR-MOX -new: Pu enrichment ratio 8%, Specific power 37.5 MW/t, Cooling period 7 years, with initial isotope compositions of Pu238: Pu239: Pu240: Pu241: Pu242 = 2.46: 54.69: 26.16: 9.51: 7.18;
- BWR-MOX -new: Pu enrichment ratio 8%, Specific power 25.9 MW/t, Cooling period 7 years, with initial isotope compositions of Pu238: Pu239: Pu240: Pu241: Pu242 = 2.46: 54.69: 26.16: 9.51: 7.18.

Each parameter to define reactor characterization is:

- **Common parameter:** 33% thermal efficiency, 80% load factor;
- **Residence time:** PWR 4 years, BWR 6 years.

Press [command] button for each reactor and set above.

There are five tabs under Scenario Data Page.

4.3.3 General Characteristics tab

To provide general data, click [General Characteristics] tab under Scenario data Page (FIG.40). The user needs to input the identical values for all years on the following fields:

- **Added Capacity (MWe):** Net electric capacity added to the system.
- **Removed Capacity (MWe):** Net electric capacity removed from the system.
- **Net Total Capacity (MWe):** Total net electric capacity. This column is automatically calculated by the system using the data given for initial power, added and removed capacity.
- **Thermal Efficiency (%):** The ratio of net electric capacity to the total thermal energy produced in the reactor.
- **Load Factor (%):** The ratio of the net electricity produced in the plant to the maximum producible net electricity in the plant for the whole year.
- **Enrichment Tails Assay (%):** The amount of U-235 in depleted uranium from the enrichment process.
- **Fuel Type 2 Share (%):** The average ratio of the amount of second fuel type to the total fuel for the reactor type. Fuel type 2 is MOX fuel for the most of the existing reactor types, if any.
- **Residence Time (year):** The time duration that each of the equilibrium loading fuels must stay in the reactor. In reality, multi-batch fuel management schemes are used in reactors and a fraction of core refuelled. If “n” is the number of fuel batches, the fraction of core refuelled is 1/n. If the operating cycle length is “Tc” then the residence time to be used in the NFCSS is equal to (nTc) (closest integer must be given to the NFCSS). The NFCSS calculates annual average loading or discharge instead of actual loading of every operating cycle.

NFCSS Scenario Data Page

ReactorType: PWR Scenario Period: 2020 - 2080 IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE		INITIAL VALUES		
Data Year	Added Capacity (MW _e)	Removed Capacity (MW _e)	Net Total Capacity (MW _e)	Efficiency (%)	Load Factor (%)	Tails Assay (%)	Fuel Type 2 Share (%)	Residence Time (year)
2020	1000	0	1000	33	80	0.3	30	4
2021	0	0	1000	33	80	0.3	30	4
2022	0	0	1000	33	80	0.3	30	4
2023	0	0	1000	33	80	0.3	30	4
2024	0	0	1000	33	80	0.3	30	4
2025	0	0	1000	33	80	0.3	30	4
2026	0	0	1000	33	80	0.3	30	4
2027	0	0	1000	33	80	0.3	30	4
2028	0	0	1000	33	80	0.3	30	4

FIG.40 Editing general characteristics.

4.3.4 Fuel Type 1 tab

The user needs to input the identical values for all years on the following parameters (FIG.41):

- **Enrichment (%)**: Initial fissile U-235 in Uranium fuel or Initial Total Pu content in MOX fuel for each of the fuel type;
- **Discharge Burnup (GW·d/t-HM)**: The average discharge burnup of the spent fuel at the discharge point for each of the fuel type;
- **Reprocessing Ratio (%)**: The ratio of amount of spent fuel to be reprocessed to the amount of total discharged spent fuel. Some portion of the spent fuel is delivered to cooling pool for reprocessing and the rest is delivered to interim storage facilities. The reprocessing ratio defines the amount of spent fuel which is sent to the cooling pool for reprocessing for given year and not the amount of reprocessed spent fuel for the given year.

NFCSS Scenario Data Page

ReactorType: BWR Scenario Period: 2020 - 2080 IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Data Year	Enrichment (%)		Discharge Burnup (GWD/t)		Reprocessing Ratio (%)
2020	4		45		0
2021	4		45		0
2022	4		45		0
2023	4		45		0
2024	4		45		0
2025	4		45		0
2026	4		45		0
2027	4		45		0
2028	4		45		0

FIG.41 Fuel Type 1 parameters data.

4.3.5 Fuel Type 2 tab

The user needs to input the identical values for all years on parameters including Enrichment (%), Discharge Burnup (GW·d/t-HM), and Reprocessing Ratio (%) (FIG.42).

NFCSS Scenario Data Page

ReactorType: BWR Scenario Period: 2020 - 2080 IMPORT SCENARIO DATA RESET FORM SAVE DATA SET

GENERAL CHARACTERISTICS		FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Data Year	Enrichment 2f (%)		Discharge Burnup 2f (GWD/t)		Reprocessing Ratio 2f (%)
2020	8		45		100
2021	8		45		100
2022	8		45		100
2023	8		45		100
2024	8		45		100
2025	8		45		100
2026	8		45		100
2027	8		45		100
2028	8		45		100

FIG.42 Fuel type 2 parameters data.

4.3.6 Fuel Cycle tab

The user needs to input the identical values for all years on the following field (FIG.43):

- **Reprocessing Capacity (t-HM):** The maximum amount of reprocessing which can be performed in a given year.

NFCSS Scenario Data Page

ReactorType: BWR Scenario Period: 2020 - 2080

GENERAL CHARACTERISTICS		FUEL TYPE 1		FUEL TYPE 2		FUEL CYCLE		INITIAL VALUES	
Data Year	NatU Capacity (tHM)	Conversion Capacity (tHM)	Enrichment Capacity (MTSWU)	Fabrication Capacity (tHM)	Reprocessing Capacity (tHM)	Fabrication Capacity 2f (tHM)	Reprocessing Capacity 2f (tHM)		
2020	0	0	0	0	0	0	100		
2021	0	0	0	0	0	0	100		
2022	0	0	0	0	0	0	100		
2023	0	0	0	0	0	0	100		
2024	0	0	0	0	0	0	100		
2025	0	0	0	0	0	0	100		
2026	0	0	0	0	0	0	100		
2027	0	0	0	0	0	0	100		
2028	0	0	0	0	0	0	100		

FIG.43 Fuel Cycle parameters data.

4.3.7 Initial Values tab

The user needs to input the identical values for all years on the following fields (FIG.44):

- **Capacities of fuel cycle facilities (t-HM or SWU):** The alternative running mode of the NFCSS is 'Capacity Driven Mode' which will be implemented in the future requires the capacities of each fuel cycle facility. Only reprocessing facility capacities have been implemented so far;
- **Initial Reactor Power (MWe):** Total net electric capacity of selected reactor type at the beginning of the scenario period. The reactor is assumed to be running in equilibrium mode at the beginning of the scenario period. This data should be supplied in the initial data section and it is not part of data import from Microsoft Excel;
- **Initial Reprocessed Pu Stock (t-HM):** The total amount of Reprocessed Pu which is available for recycling at the beginning of scenario period;
- **Initial SF Stock (t-HM):** The total amount of spent fuel (for each fuel type) which is ready for reprocessing at the beginning of the scenario period.

NFCSS Scenario Data Page

ReactorType: BWR

Scenario Period: 2020 - 2080

IMPORT SCENARIO DATA

RESET FORM

SAVE DATA SET

GENERAL CHARACTERISTICS	FUEL TYPE 1	FUEL TYPE 2	FUEL CYCLE	INITIAL VALUES
Initial Value Type	Initial Value	Insertion Year	Explanation	
Initial Reactor Power (MWe) :	0	N/A	The total power of reactors which are operational in equilibrium conditions at the beginning of the scenario period.	
Initial Rep. Pu Stock (t HM) :	0	N/A	The total amount of Reprocessed Pu which is available for recycling at the beginning of scenario period.	
Initial SF Stock - Fuel Type 1 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 1) which is ready for reprocessing at the beginning of the scenario period.	
Initial SF Stock - Fuel Type 2 (t HM) :	0	N/A	The total amount of spent fuel (fuel type 2) which is ready for reprocessing at the beginning of the scenario period.	

FIG.44 Initial Values parameters data.

4.3.8 Scenario with thorium fuel cycle and uranium fuel cycle

- The mass flow calculation in the results tab does not include thorium, protactinium, and U-234 and U-238, even though they are available in the ICR calculation; this will be fixed shortly.
- Mass flow diagram for this type is being developed and will be posted accordingly.

The easiest data entry is done through Microsoft Excel since Microsoft Excel is known to many computer users already and it provides easy editing features in tabular form. In this case, the data can be prepared in Microsoft Excel and then imported into the NFCSS using [Import from MS Excel] button. How to use this button is described in Appendix of this Guidebook. Also, how to calculate is described in this Guidebook.

4.4 STEP 5: CALCULATION AND RESULTS

In order to start the NFCSS calculations, the user first needs to make sure that isotopic compositions of initial fresh fuel, spent fuel at discharge time and spent fuel after a cooling period. To perform isotopic composition calculations, initial composition of fresh fuel and other fuel parameters have to be provided. If all data is correctly entered for a fuel type, click [Make All IC Calculations] button (FIG.45). For Th fuel cycle with 2 types of fuel, MFD may not be shown correctly.

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Make All IC Calculations

Make All MF Calculations

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Decay Heat	Radiotoxicity
PWR	PWR UOX 1	PWR-MOX-new		!! Warning !!	!! Warning !!
PWR	PWR UOX 1			!! Warning !!	!! Warning !!
BWR	BWR UOX 1	BWR-MOX-new		!! Warning !!	!! Warning !!
BWR	BWR UOX 1			!! Warning !!	!! Warning !!

Calculate Overall Material Flow

-
⌂
-
🚩
-
🚩

FIG.45 Calculation commands.

After completing isotopic composition calculations or providing the isotopic compositions, flag colour will change to GREEN from RED (FIG.46).

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Make All IC Calculations

Make All MF Calculations

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Decay Heat	Radiotoxicity
PWR	PWR UOX 1	PWR-MOX-new		!! Warning !!	!! Warning !!
PWR	PWR UOX 1			!! Warning !!	!! Warning !!
BWR	BWR UOX 1	BWR-MOX-new		!! Warning !!	!! Warning !!
BWR	BWR UOX 1			!! Warning !!	!! Warning !!

Calculate Overall Material Flow

-
⌂
-
🚩
-
🚩

FIG.46 Calculation flag.

Then, the user can make material flow calculations for each of the reactor types. Click [Make All MF Calculations] button (FIG.47).

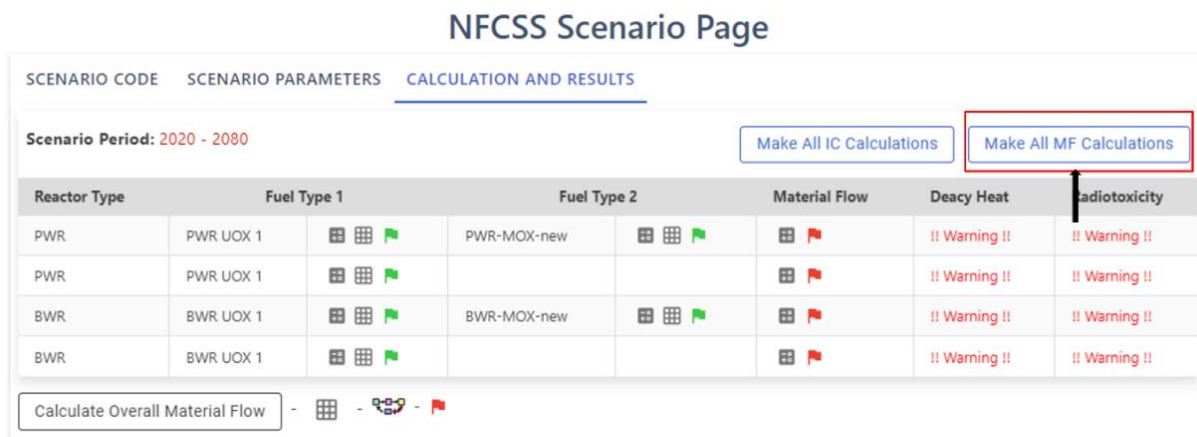


FIG.47 Material flow calculations.

If the calculation is finished without errors, following screen can be found (FIG.48).



FIG.48 Completed status of calculation for each reactor type.

The user can see calculation results of material flow in diagram by clicking [Calculate Overall Material Flow] button (FIG.49). The user can get the chart for results of each of the fuel cycle steps by clicking on the box of steps.

NFCSS Scenario Page

SCENARIO CODE
SCENARIO PARAMETERS
CALCULATION AND RESULTS

Scenario Period: 2020 - 2080

Make All IC Calculations

Make All MF Calculations

Reactor Type	Fuel Type 1	Fuel Type 2	Material Flow	Deacy Heat	Radiotoxicity
PWR	PWR UOX 1	PWR-MOX-new			
PWR	PWR UOX 1				
BWR	BWR UOX 1	BWR-MOX-new			
BWR	BWR UOX 1				

Calculate Overall Material Flow

Click these buttons can view results table and download results in excel file

FIG.49 Calculation with Overall Material Flow.

APPENDIX I.

NFCSS CALCULATION MODEL

The NFCSS calculates annual average fuel cycle material and service requirements and material arisings and, whenever relevant, cumulative amounts by summing annual amounts. The mathematical model of the fuel cycle calculations has been developed by tracking the nuclear material from the beginning of the cycle to the end of cycle based on material balance principle.

The formulas which are given below in details are limited to the model used for this publication. Calculations are performed separately for each reactor type. The MOX fuel is produced by using the plutonium separated by reprocessing spent fuel and the depleted uranium from enrichment facilities. The lead times between fuel discharges, reprocessing and MOX fuel fabrication are taken into account in the calculations. Use of reprocessed uranium in uranium fuel is also eliminated from the calculation.

II.1. Definition of parameters (*variable*):

- **Year:** Calculation year period.
- **Type:** Reactor type.
- **ft:** Fuel type.
- **Nuclear Capacity(year):** Total nuclear capacity by year.
- **Reactor Mix (type, year):** Ratio of nuclear power by reactor type and year.
- **Load Factor (type, year):** Average load factor by reactor type and year.
- **Efficiency (type, year):** Average thermal efficiency by reactor type and year.
- **Burnup (type, year, ft):** Average discharge burnup by reactor type, year, and fuel type.
- **Share2Type (type, year, ft):** Ratio of second fuel type (*ft=2 in all cases*).
- **Reprocessing (type, year, ft):** Reprocessing ratio by reactor type, year, and fuel type.
- **Type2Uranium (type, year):** Uranium source for second type fuel (Dep U, Rep U or Nat U) by reactor type and year (for this publication, uranium source for second fuel type is limited to Dep U from enrichment process).
- **Enrichment (type, year, ft):** Initial enrichment in fresh fuel by reactor type, year and fuel type (*it is the amount of total Pu in case of U-Pu MOX fuel*).
- **Multirecycling:** Multirecycling parameter; 0 for no-multirecycling, 1 for multirecycling.
- **Residence (type, year, ft):** Fuel residence time in core by reactor type, year and fuel type.
- **Tails Assay(year):** Average enrichment tails assay by year.
- **Enrichment Loss, Fabrication Loss, Reprocessing Loss, Conversion Loss:** Process loss coefficients for each of the respective processes.
- **Reprocessing Cooling (type, year, ft):** Cooling time before reprocessing by reactor type, year, and fuel type.
- **Fabrication Time (type, year, ft):** Fuel fabrication times by reactor type, year, and fuel type.
- **Reprocessing Time (type, year, ft):** Reprocessing time by reactor type, year, and fuel type.

II.2. Mathematical Model

The calculation steps are given below and follow the order of actual algorithm for each reactor type. Process losses are ignored in the formulas for easy reading.

Net nuclear capacity by reactor type (GWe)

$$TypeCapacity(type, year) = Reactor Mix(type, year) \times Total Capacity(year)$$

Equilibrium core (tHM)

$$FullCore(type, year) = \frac{TypeCapacity(type, year) \times 365 \times Residence(type, year) \times LoadFactor(type, year)}{(Efficiency(type, year) \times (Burnup(type, year)))}$$

Annual fresh fuel requirements (tHM)

$$FreshFuelTotal(type, year) = \frac{FullCore(type, year)}{Residence(type, year)}$$

Annual fuel loading (tHM)

Two categories of fuel have to be considered to calculate annual fuel loading in reactors. Index *ft* in below formulas corresponds to the fuel types in the material flow model. 2 and 1 refers to fuel type 2 and fuel type 1 respectively (ft=2, ft=1).

If there is enough reprocessed material in the stockpile

$$FreshFuel(type, year, 2) = FreshFuelTotal(type, year) \times Share2Type(type, year, 2)$$

else

$$FreshFuel(type, year, 2) = 0$$

$$FreshFuel(type, year, 1) = FreshFuelTotal(type, year) - FreshFuel(type, year, 2)$$

Annual fuel unloading (tHM)

Annual fuel unloading is calculated for each fuel type taking into account the residence time of fuels in reactor. It means any fuel is discharged after burning in reactor for its residence time.

$$SFD(type, year, ft) = FreshFuel(type, year) - Residence(type, year, ft)$$

$$SFD Total(type, year) = \sum_{ft} SFRS(type, year, ft)$$

Spent fuel route (tHM)

Unloaded spent fuel can be either reprocessed or disposed of directly, depending on the type of reactor and fuel, and on the strategy considered. Spent fuel of all reactor types except LWRs is never reprocessed according to the simulation. The shares of LWR fuels that are reprocessed vary depending on the strategy considered. Simulation considers that the initial content of the second fuel type is same and the fuel can be recycled unlimited times.

Spent fuel to be stored for recycling (tHM)

$$SFRS(type, year, ft) = SFD(type, year, ft) \times Reprocessing(type, year, ft)$$

$$SFRSTotal(type, year) = \sum_{ft} SFRS(type, year, ft)$$

Spent fuel to be stored for future use (or for disposal) (tHM)

$$SFS(type, year, ft) = SFD(type, year, ft) \times SFRS(type, year, ft)$$

$$SFRSTotal(type, year) = \sum_{ft} SFRS(type, year, ft)$$

Reprocessing (tHM)

$$SFR(type, year, ft) = SFRS(type, year1, ft)$$

Where

$$year2(ft) = ReprocessingCooling(type, year, ft) + ReprocessingTime(type, year, ft)$$

$$year1(ft) = year - year2(ft)$$

$$SFRTotal(type, year) = \sum_{ft} SFRS(type, year, ft)$$

Availability of reprocessed material for fuel type 2 (Pu in case of MOX fuel) (tHM)

The availability of reprocessed material is calculated by calculating the isotopic content of spent fuel. The fuel depletion code CAIN is used to calculate isotopic content of spent fuel.

There is also possibility to use actinide tables from other authoritative depletion codes such as ORIGEN. CAIN calculates isotopic composition of spent nuclear fuels for different initial fissile content, different discharge burnup and different cooling times.

$$NucStock(type, year, n) = \sum_{ft} SFR(type, year, ft) \times NucContent(type, year1(ft), ft, n, year2(ft))$$

where

$NucStock(type, year, n)$ is the available isotope **n** amount from reprocessing

$NucContent(type, year1, ft, n, year2(ft))$ is the content of isotope **n** in spent fuel type **ft** after cooling time **year2**

$$year2(ft) = ReprocessingCooling(type, year, ft) + ReprocessingTime(type, year, ft)$$

$$year1(ft) = year - year2(ft)$$

Used nuclear material (tHM)

$$UsedNuc(type, year, n) = \sum_{ft} NucContent(type, year1(ft), ft, n, year2(ft)) NucContent(type, year, ft, n, 0)$$

Natural uranium savings due to recycling of Pu (in MOX) (tHM)

$$K(type, year) = \frac{Enrichment(type, year) - TailsAssay(type, year)}{0.00711 - TailsAssay(type, year)}$$

$$NatUSavingPu(type, year) = K \times FreshFuel(type, year, 2)$$

Annual plutonium arising in discharged spent fuel

$$AnnualTotPu(type, year) = \sum_{ft} \sum_{n_{Pu}} SFD(type, year, ft) \times NucContent(type, year, ft, n, 0)$$

Annual enrichment service requirements (MTSWD)

$$SWURequirement(type, year) = \{Ve(type, year) + (K(type, year) - 1) \times Vt(type, year) - K(type, year) \times Vn(type, year)\} \times FreshFuel(type, year)$$

$$EnrichedU(type, year) = \begin{cases} fresh\ fuel\ (type, year, 1), & \text{if } enrichment(type, year, 1) > 0.711 \\ 0, & \text{if } enrichment(type, year, 1) < 0.711 \end{cases}$$

$$DepletedU(type, year) = EnrichedU(type, year) \times (K(type, year) - 1)$$

where

$$K(type, year) = \frac{Enrichment(type, year) - TailsAssay(type, year)}{0.00711 - TailsAssay(type, year)}$$

$$Ve(type, year) = (2 \times Enrichment(type, year)/100 - 1) \times LN\left\{\frac{Enrichment(type, year)/100}{Enrichment(type, year)/100}\right\}$$

$$Vt(type, year) = \{2 \times TailsAssay(type, year)/100 - 1\} \times LN\left\{\frac{Enrichment(type, year)/100}{1 - Enrichment(type, year)/100}\right\}$$

$$Vn(type, year) = \{2 \times 0.00711 - 1\} \times LN\left\{\frac{0.00711}{1 - 0.00711}\right\}$$

Annual conversion requirements (tHM)

$$ConvRequirement(type, year) = \begin{cases} EnrichedU(type, year) + DepletedU(type, year), & \text{if } Enrichment > 0.711 \\ FreshFuel(type, year, 1), & \text{if } Enrichment < 0.711 \end{cases}$$

Annual natural uranium requirements (tHM)

$$NatURequirement(type, year) = ConvRequirement(type, year)$$

APPENDIX II

SOURCE TERMS AND ALPHA ACTIVITY OF SPENT FUEL

TABLE I. SOURCE TERM (ACTINIDES AND LONG-LIVED FISSION PRODUCTS (FP) AND ITS COMPONENTS- VARIATION WITH TIME(Sv/TWhe) ^[5]

Time, Years	10 ²	10 ³	10 ⁴	10 ⁵	10 ⁶	10 ⁷
Total, Sv/TWhe	1.1×10 ⁹	3.1×10 ⁸	7.7×10 ⁷	4.2×10 ⁶	5.2×10 ⁵	1.4×10 ⁵
Components, %						
Rep U	-	-	0.1	6	29	79
Pu	85	90	97	88	50	17
Np	-	-	-	1.3	13	3
Am	10	9.2	2.5	2.7	6.8	1.4
Cm	0.4	0.3	0.4	-	-	-
FP	4.2	6×10 ⁻⁴	2.4×10 ⁻³	3.2×10 ⁻²	9.6×10 ⁻²	1.4×10 ⁻¹

TABLE II. ALPHA ACTIVITY OF SPENT FUEL OF THERMAL REACTORS (Ci/t HM) AS A FUNCTION OF BURNUP ^[6]

Burnup MW.d/kg HM	Pu total	²³⁸ Pu	²³⁷ Np	²⁴¹ Am	²⁴⁴ Cm	Total
33.00	3213	2360	0.15	1310	1490	6050
40.00	4430	3480	0.40	1500	2700	8750
50.00	6720	5740	0.51	1950	5700	14300
60.00	9270	8220	0.64	2430	11600	23500

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