

## Simulation of the Material Flow and Calculation of the MUF Uncertainty in the Pyroprocess

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### 1. Introduction

A simulation program for the analysis of nuclear material flow in a pyroprocess facility and for the calculation of the MUF uncertainty has been developed. This simulation program calculates the mass of each nuclide (in grams), the decay rate (in curies), the thermal power (in watts), the radiation dose (in mSv), and the degree of criticality. This program calculates the isotopic composition of the spent fuel that will be used in a facility and its criticality by connecting ORIGEN Code[1] and MCNPX Code[2]. The formula for calculating the material flow change by process is applied to the mass of each nuclide. The decay rate, thermal power, and radiation dose are calculated using DB. Also, the hold-up in the process equipment and the measurement error are estimated, and these are applied to the material flow calculation. The estimated measurement error and the calculated MUF uncertainty would be utilized in the verification of the design of a pyroprocess facility.

### 2. Development of modules

In this section, the modules that were used to develop the simulation program are described. These modules include those on the linking of the ORIGEN code, the calculation of the material flow, the analysis of the material properties, the calculation of the MUF uncertainty, and the visualization.

#### 2.1 Development of a module on linking the ORIGEN code

A module has been developed to produce output files by running a code, after producing input files that can execute the ORIGEN Code using the information on the spent fuel assemblies (ID, fuel type, initial concentration, burn-up cycle, operation period, cooling period, average burn-up, moderator concentration, and total mass of the assemblies) stored in the database. The isotopic composition of the spent fuel is determined from the mass of each nuclide extracted from the output file.

#### 2.2 Development of a module for calculating the material flow

The mass of each nuclide of the spent fuel assemblies is calculated using the isotopic composition obtained from the result of the execution of the ORIGEN code.

The isotopic composition is identically applied until one assembly is treated completely, and the material flow of each process is calculated with the calculation formula classified by nuclide using the parameter DB, in which the characteristics of the process are stored. The calculation formula for the nuclide is classified by group of nuclides, and the group of nuclides is presented in Table 1.

Table I: Group of Nuclides

Group	Nuclide
U	U
TRU	Ac, Th, Pa, Np, Pu, Am, Cm, Bk, Cf, Es, S
NM	Tl, Pb, Bi, B, Co, Ni, Cu, Zn, Ga, Ge, Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Hf
REFP	Y, La, Ce, Pr, Nd, Pm, Sm, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu
AMFP	Fr, Ra, Li, Cs
AEFP	Ba, Be, Sr
FP	Po, At, Rn, H, He, C, N, Ne, As, Se, Br, Kr, Xe, Te, I, Sn, Sb, Eu

The process is carried on with each batch as a unit, and the criteria move to the next process can be set by the user. The interface is established to modify the parameters of each process used in the calculation of the material flow. The calculated value of the material flow is stored automatically in the DB, and can be stored in an EXCEL file if needed.

#### 2.3 Development of a module for analyzing the material properties

A module for calculating the decay rate of each nuclide, the thermal power, and the radiation dose has been developed using an isotope DB in which the decay rate per gram, thermal power, and radiation dose are stored. In the electro-reduction process and the electro-refining process, the criticality of the process is extracted from the output file produced by the MCNPX Code, after the input file of the MCNPX Code is produced using the mass value of each nuclide. The state of the nuclear material by process could be analyzed using this information, and this information could also be utilized to verify the safety of the process.

## 2.4 Development of a module for calculating the MUF uncertainty

The estimated measurement error is applied to the result of the nuclear material flow calculation. The random uncertainty, systematic uncertainty, and total MUF uncertainty are calculated by applying the calculation formula for the MUF uncertainty [3] to the result of the calculation of the material flow of each nuclide using the MUF parameters DB (i.e., the KMP determination information, designated place of each KMP, measurement method, batch size of each KMP, container size, number of analyses per sample, and random/systematic error by method).

The calculated MUF uncertainty is always indicated on the upper part of the screen, and the value of the MUF uncertainty is updated by re-calculating it each time the campaign is completed. The MUF uncertainty would be utilized to analyze the safeguardability of a pyroprocess facility from the perspective of accounting.

## 2.5 Development of a visualization module

The mass value by group, radiation dose, and degree of criticality are indicated in the flow diagram of the pyroprocess shown in Fig. 1. If a mass value by group, as shown on the flow diagram, is selected, the mass value of each nuclide, the decay rate, and the thermal power can be checked in detail.

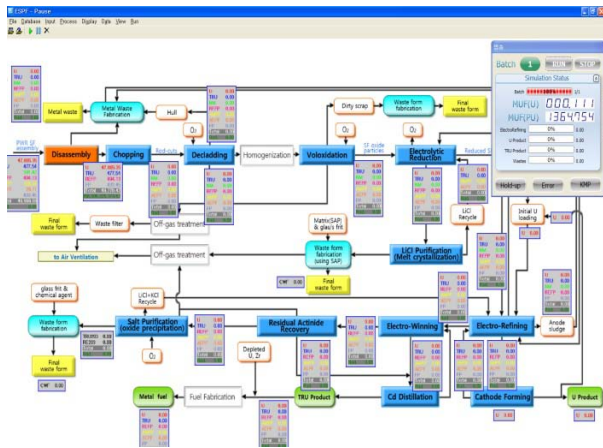


Fig. 1. Flow diagram in the simulation program

If a process shown on the flow diagram is selected, the information on the input and output materials of the process is presented, as shown in Fig. 2. Besides, the function of the production of a graph on the change in the product by batch, the change in the mass by process, and the change in the process by batch is added to analyze the data, as shown in Fig. 3.

All the parameters used to calculate the material mass flow and the MUF uncertainty, and to produce the ORIGEN input file, can be easily modified on the user interface (UI). The results of the simulation, which are stored in the DB, can be checked with the tables.

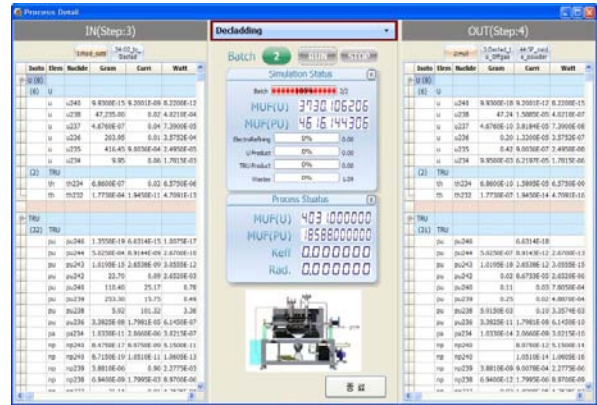


Fig. 2. Process status information



Fig. 3. Graph on the change in the mass by process (Electro-winning Process)

## 3. Conclusions

In this research, a simulation program for the analysis of the nuclear material flow in a pyroprocess facility and for the calculation of the MUF uncertainty and mass of each nuclide (in grams), the decay rate (in curies), the thermal power (in watts), the radiation dose (in mSv), and the degree of criticality has been developed.

By improving the module on the calculation of the material flow by process, the mass of recycled nuclear materials could be calculated and users would be able to control the addition and deletion of processes that operate independent of each other.

## REFERENCES

- [1] I.C. Gaulland and O.W. Hermann, ORIGEN-S, 2005.
- [2] D.B. Pelowitz et al., MCNPX Users' Manual Version 2.5.0, LANL Report, 2005.
- [3] IAEA Safeguards Technical Manual, IAEA-TECDOC-261, 1982.