A Study on Nuclear Fuel Cycle Model using AnyLogic Platform

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1. Introduction

The nuclear fuel cycle is the process of producing, using, and disposing of nuclear fuel. The mining and processing of uranium ore, enrichment of uranium, fabrication of fuel assemblies, use of the fuel in a nuclear reactor to generate electricity or for other purposes, storage and reprocessing of spent fuel, and disposal of radioactive waste are all part of it. The Korea Institute of Nuclear Nonproliferation And Control (KINAC) developed the nuclear fuel cycle model using the AnyLogic Platform [1]. This model, however, is insufficient to simulate the actual occurrence accurately because the series of analysis steps were written to run through a single continuous calculating process without an intermediate separation point. For this reason, it was changed to only allow events to occur at specific time points by including database information on when each event started and ended. In addition, a threshold value was set at which each event may occur, and the analysis was changed to only apply when the threshold value was met.

2. Nuclear Fuel Cycle Model using AnyLogic

In the AnyLogic Platform, each process of the nuclear fuel cycle mentioned above is implemented as an Agent object.

2.1 Natural Uranium Mining and Milling Model

The mining and milling process of natural uranium are shown in Fig. 1.

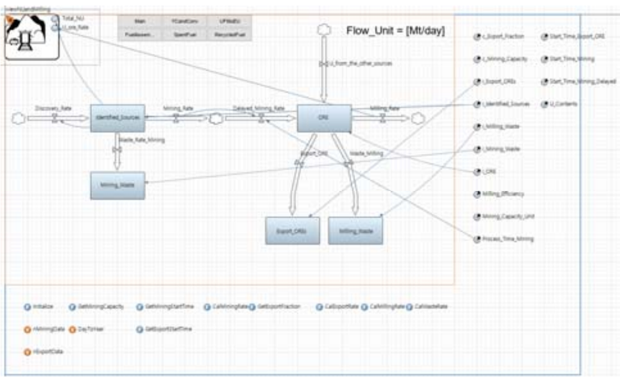

Fig. 1. Natural uranium mining and milling model.

The main Stock (AnyLogic Class) Object includes the variable Identifed_Sources, which represents the identified uranium reserves, and the variable ORE, which indicates the amount of uranium mined. The amount of newly identified uranium is provided by estimating the daily average using the Flow (AnyLogic Class) Object. The mining rate is determined by calculating the daily average mining rate based on the mining start and end dates specified in the database as well as the amount of natural uranium extracted during that time period. In the mining rate, the time delay is considered as much as the time required for the mining process. In addition, the final milling rate is calculated considering the proportion of unsold uranium ore, the uranium content in the ore, and the milling efficiency.

2.2 Uranium Conversion Model

U_3O_8, also known as yellow cake, is obtained during the milling process. It is converted to metallic uranium form to use natural uranium as fuel or to UF_6 form for enrichment of uranium. These processes are depicted in Fig. 2.

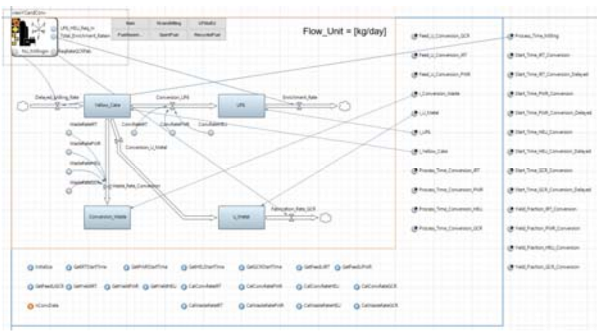

Fig. 2. Uranium conversion model.

The main Stock variables are the variable Yellow_Cake, which represents the amount of U stock contained in U_3O_8, the variable UF_6, which indicates the amount of U stock contained in UF_6, and the variable U_Metal, which represents the stock amount of metal uranium. The production of UF^6 and metallic uranium only takes place when Yellow_Cake stocks are available. The metal uranium generation rate is implemented as a Flow object, and its value is determined by the amount of U, the conversion yield which are presented in the database CONVERSION, and the time required for the conversion which can be adjusted by the user in the initial screen of the simulation. Similarly, the amount of UF_6 produced per day to produce PWR nuclear fuel, IRT nuclear fuel, and highly enriched uranium (HEU) for experimental use is determined by information provided by databases and user settings.

2.3 UF6 Enrichment Model

The enrichment process of UF6 was implemented as shown in Fig. 3. The uranium enrichment process proceeds only if there is a stock of UF6.

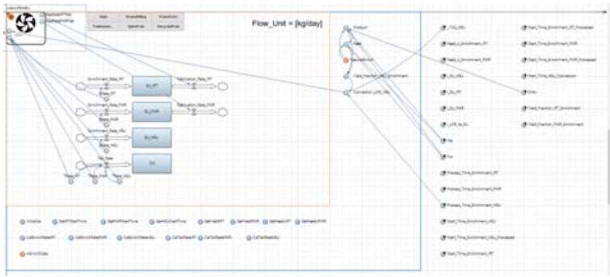

Fig. 3. UF6 enrichment model.

The most important factor considered in the enrichment process is the enrichment yield, which represents the amount of enriched uranium produced relative to the amount of natural uranium supplied. This has the same relationship as Eq. (3) and can be easily derived from the conservation equations (1) for the total amount of U and (2) for the total amount of U-235. Eq. (4) can also be used to express the ratio of enriched uranium to depleted uranium. In each formula, F means the amount of natural uranium provided for enrichment and P means the amount of enriched uranium produced. And T means the amount of depleted uranium released. x_f , x_p and x_t mean each enrichment values.

$$F = P + T. \tag{1}$$

$$x_f F = x_p P + x_t T. \tag{2}$$

$$\frac{P}{F} = \frac{x_f - x_t}{x_p - x_t}. \tag{3}$$

$$\frac{P}{T} = \frac{x_f - x_t}{x_p - x_f}. \tag{4}$$

If the separative work units is considered, it can be written as Eq. (5) [3].

$$W_{SWU} = P \cdot V\left(x_p\right) + T \cdot V\left(x_t\right) - F \cdot V\left(x_f\right). \tag{5}$$

where, $V\left(x\right)$ is the value function, defined as: [4]

$$V\left(x\right) = \left(2x - 1\right)\ln\left(\frac{x}{1-x}\right). \tag{6}$$

The natural uranium enrichment, x_f value is 0.00711. When a specific W_{SWU} value is given, the enriched uranium production P according to x_p and x_t can be calculated using Eq. (7). Other information required to calculate the enriched uranium production rate can be obtained from database ENRICHMENT and user settings in the initial screen of the simulation.

$$P = \frac{W_{SWU}}{V\left(x_p\right) + \frac{x_p - x_f}{x_f - x_t} V\left(x_t\right) + \frac{x_p - x_t}{x_f - x_t} V\left(x_f\right)}. \tag{7}$$

2.4 Fuel Assembly Manufacturing Model

When there is a stock of enriched uranium, the fuel manufacturing process for IRT and PWR is possible. Similarly, GCR fuel manufacturing process is possible when there is a stock of metallic uranium. These processes are depicted in Fig. 4.

Fig. 4. Fuel assembly manufacturing model.

The amount of uranium produced into fuel assemblies for each reactor is determined by the amount of uranium required to produce fuel assemblies, the yield in the fabrication process provided in the database FABRICATION, and the fabrication time defined by the user at the start of the simulation.

2.5 Reactor Operation Model

In this model, burnup calculations are performed for each reactor during its operating period. Euler's method and the 4th order Runge-Kutta method are provided in the AnyLogic Platform to solve initial-value problems for ordinary differential equations. The 4th order Runge-Kutta method was used to accomplish this in the previous model; however it was very slow in burnup calculation due to the need to divide the time interval by seconds in order to have a sufficiently converged solution. For this reason, a solver applying the series expansion method was applied to perform burnup calculation. Using this allowed burnup calculations for around 1000 days to be completed in less than a minute with satisfactory convergence.

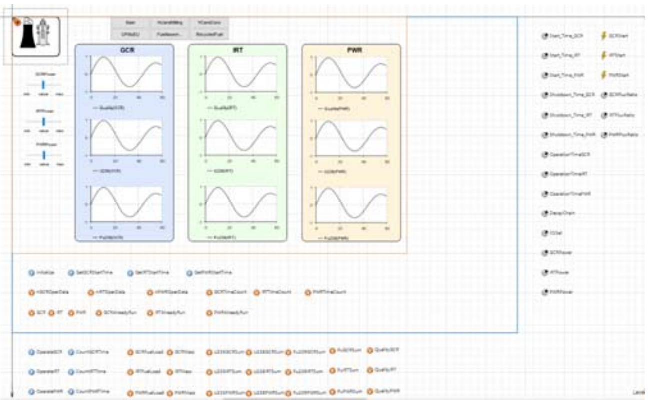

Fig. 5. Reactor operation model.

2.6 Spent Fuel Cooling Model

When the reactor is shutdown, the spent nuclear fuel is discharged from the reactor and cooled. During the cooling period, each nuclide's inventory decreases according to its respective half-life. On the simulation's initial screen, the user can specify the cooling period.

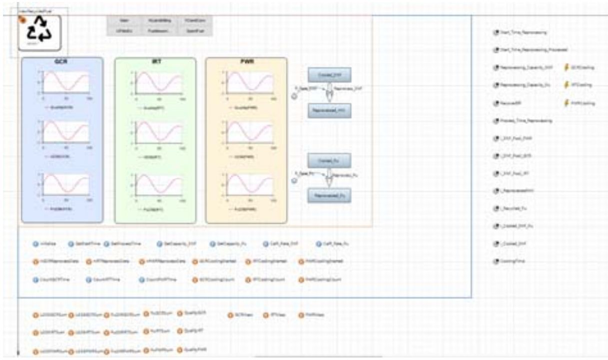

Fig. 6. Spent fuel cooling model.

3. Sensitivity Analysis

To accurately predict the amount of spent nuclear fuel discharged from each reactor, various reactor operation information must be provided. Yet, due to the restricted availability of information, it is frequently required to estimate the volume of spent nuclear fuel produced using only a limited piece of information. For this reason, it is necessary to evaluate how much the result differs by performing a sensitivity analysis on uncertain data and using the data that is believed to be accurate as a limiting condition. The neutron spectrum, which represents the ratio of thermal neutron flux to fast neutron flux, is the most influential factor in the production of Pu from spent nuclear fuel. In a prior work [1], burnup simulations were carried out under GCR reactor conditions, and the effect of changing the neutron spectrum on Pu-239 production was investigated.

Table I: Reference conditions

Thermal Power [MW_{th}]	Uranium Mass [MT]
25	50
Enrichment [w/o]	$\phi_2(0)/\phi_1(0)$
0.71	1.69

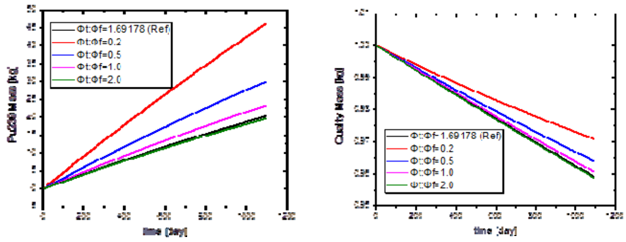

Fig. 7. Pu-239 production according to neutron spectrum.

3. Conclusions

The simulation process for each cycle stage was changed in the nuclear cycle analysis model developed on the AnyLogic Platform to more accurately reflect the more realistic process. In addition, the old 4th order Runge-Kutta method was replaced for evaluating the core burnup process with the series expansion method, which offers faster convergence in the linear system analysis, resulting in drastic reduction in analysis time. Finally, it was confirmed by a sensitivity test that Pu-239 production improves as the ratio of fast neutron flux to thermal neutron flux increases. This ratio has a significant impact on Pu-239 production.

ACKNOWLEDGMENTS

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KoFONS) using the financial resource granted by the Nuclear Safety and Security Commission (NSSC) of the Republic of Korea. (No. 1905008)

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