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A simulation study on the radiation dose distribution in the hot cell environment in pyroprocessing

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1. Introduction

The storage and disposal of spent fuels from nuclear power plants have been the serious issue. Pyroprocessing is one of the prospective technologies to treat nuclear spent fuels. In the pyroprocessing, transuranic (TRU) materials are not separated, and they are recovered collectively. For this reason, the pyroprocessing is favorable in terms of nuclear proliferation resistance compared to conventional aqueous processes [1]. Development of the pyroprocess technology is expected to contribute to reduction of high-level radioactive waste and the effective use of nuclear resources.

Korea Atomic Energy Research Institute (KAERI) has researched the pyroprocessing since 1997. The engineering scale facility (PRIDE) for research and demonstration of the pyroprocessing was constructed and has been operated for several years [2]. Based on the demonstration operation in PRIDE, a larger-scale pyroprocessing facility to treat spent fuels is being planned and a study on the preliminary conceptual design and cost estimation is in progress [3].

In the pyroprocessing facility which has to ensure nuclear proliferation resistance, safeguards are an important part. In designing the pyroprocessing facility, a consideration on the high radiation dose environment in hot cells in pyroprocessing is essential to establish reliable safeguards. There are previous studies [4, 5] on the neutron flux/dose in the pyroprocess hot cell. However they have limitations such as no consideration of gamma-ray flux/dose, omission of unit processes for pretreatment, and calculation of the neutron flux/dose in the unit process individually. Therefore, a study on the radiation flux and dose distribution in the pyroprocess hot cell environment, which considering actual operation, is required.

In this study, the flux and dose distribution of neutron/gamma-ray in the hot cell environment of the pilot pyroprocessing facility are investigated. From the analysis of material flow of pyroprocessing, the material composition model at unit processes are established and the neutron/gamma-ray energy spectra from the nuclear material at unit processes are calculated. Based on the established material composition model and energy spectra, the neutron and gamma-ray flux/dose are calculated by MCNP6 simulation.

2. Establishment of a material composition model

The pyroprocessing can be classified into pretreatment of a spent fuel assembly (SFA) in the air cell and main chemical processes in the Ar cell. Figure 1 and 2 show the simplified flow of pyroprocessing in the air cell and Ar cell. The SFA that entered into the air cell is disassembled, and the Zr cladding of rods is removed. After oxidation and mixing processes, this rod materials are transformed into pellets for pyroprocessing. Gaseous fission products are removed at the stage of heat treatment.

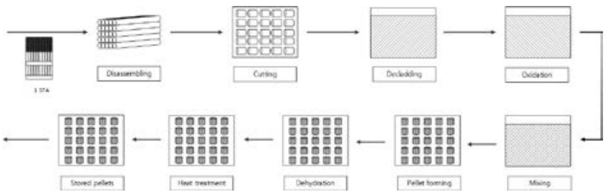

Fig. 1. Process flow in the air cell.

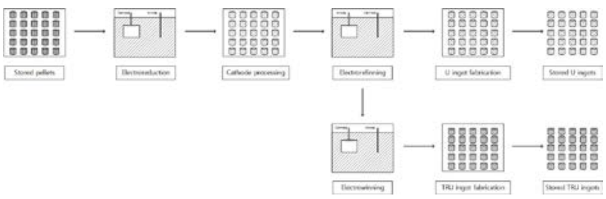

Fig. 2. Process flow in the Ar cell.

The formed pellets in the air cell are carried into the Ar cell. In the electroreduction process, this spent fuels of oxide form are converted into metallic form. The oxides soluble in molten salt (LiCl) are removed whereas the other metallic elements including uranium and TRU remains. A small portion of salt in the metal product is removed in the cathode processing. The electrorefinning process separates uranium from the other elements. Uranium collected on the cathode is transformed into uranium ingots through the salt distillation. In the electrowinning process, the remained TRU elements are collected and also converted to TRU ingots.

Based on the material flow of pyroprocessing and advices of KAERI, the material composition in unit processes in the air and Ar cell was calculated. The calculated material composition was used for material card specification in the following MCNP6 simulation. And the energy spectra of neutron/gamma-ray emitted from the radioisotopes in unit processes were calculated using ORIGEN. This energy spectra were also used for definition of source energy in the following MCNP6 simulation.

3. MCNP simulation

Unit processes in above figures were described in MCNP6 simulation. The described geometry is shown in the figure 3 and 4. Considering dimensions of planned facility, both hot cells were assumed as 50m length, 8m width, and 10m height. Walls of hot cells were defined as boron frits-baryte concrete of 15cm thickness.

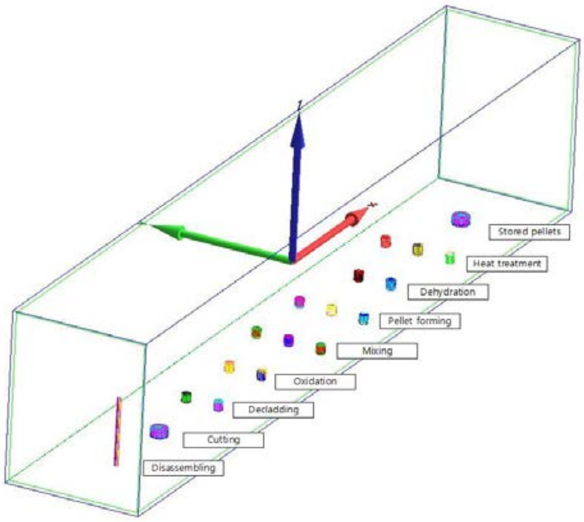

Fig. 3. Simulation geometry in the air cell simulation

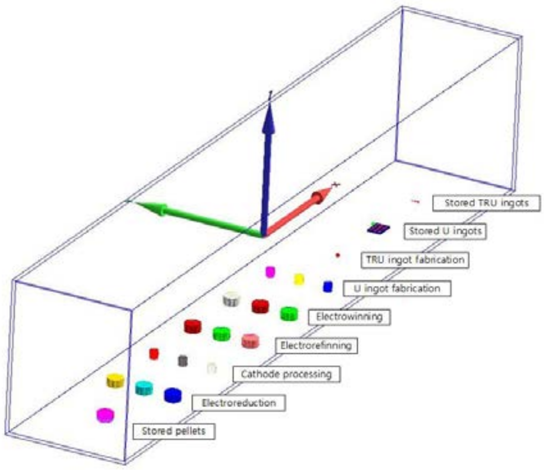

Fig. 4. Simulation geometry in the ar cell simulation

The processing equipment were described simply as the stainless cylinders which include materials, and empty space in the equipment were filled with air and Ar. Dimensions and thickness of these equipment were assumed based on throughput of the individual unit process.

To include all neutron/gamma-ray from many equipment with possible simulation NPS and reliable results, the simulation of neutron/gamma-ray emitted in each unit processes were performed individually, and the results of all processes in each cell were combined. The energy spectra of emitted neutron/gamma-ray were defined based on the previously ORIGEN calculation. In this way, 10^6 and 10^8 NPS were required in each neutron and gamma-ray simulation for the reliable results.

The cell and surfaces tallies were used for the calculation of neutron/gamma-ray flux and dose in each processed nuclear material and equipment. And to obtain spatial distribution of flux/dose, the mesh tally was applied as 1m apart in the whole space in the hot cell (Figure 5). For the calculation of neutron/gamma-ray dose, dose function (1971, normalization factor=1, LOGLIN energy interpolation) was used.

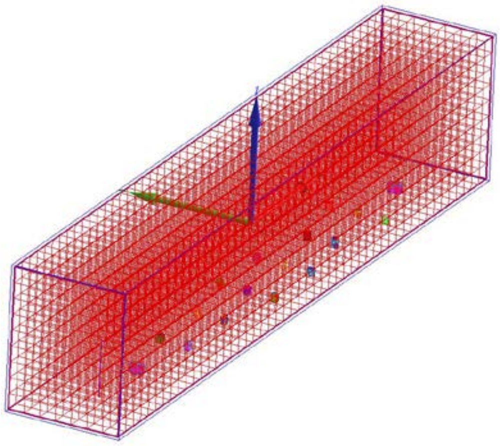

Fig. 5. Applied mesh tally in the simulation

4. Preliminary simulation result

The simulation results obtained from the F2 surface tally and F4 cell tally in MCNP6 simulation is the probability for 1 neutron or gamma-ray photon. Therefore, the emission rate of neutron/gamma-ray in each processed nuclear material is required for calculation of flux/dose at each tally. The emission rate of neutron/gamma-ray in individual unit process were calculated previously using ORIGEN and used for scaling.

The preliminary MCNP6 simulation results on the neutron/gamma-ray flux/dose in the air cell environment in pyroprocessing are summarized in the figure 6. The neutron/gamma-ray flux/dose at the hot cell walls (per cm^2) and each processing equipment are compared.

Fig. 6. Comparison of neutron/gamma-ray flux/dose at the hot cell walls and equipment

In the air cell, the neutron flux and dose vary on depending on only the mass of spent fuels so that total flux and dose emitted in unit processes are almost constant. On the contrary, the gamma-ray flux and dose is much smaller at the stage of stored pellets (the end of air cell processing) compared to the cutting process which has same mass of spent fuels. This is considered due to the reduction of gaseous gamma-ray emitters during heat treatment. This reduction of gamma-ray flux/dose is shown more clearly in the figure 7 and 8, which is the represent of spatial distribution of neutron/gamma-ray flux/dose in the entire air cell.

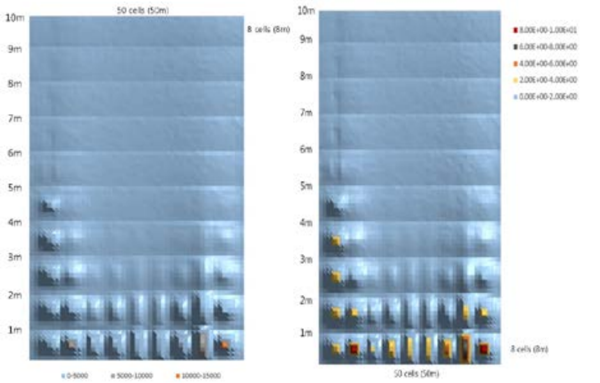

Fig. 7. Spatial distribution of neutron flux (left) and dose (right) in the air cell

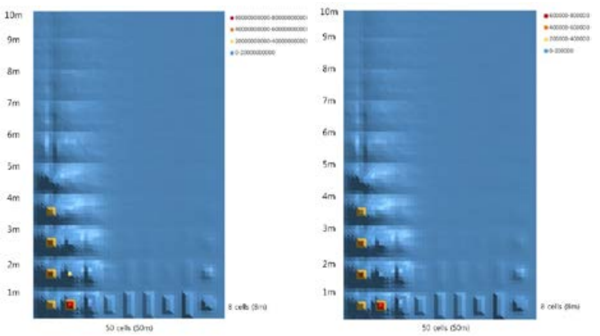

Fig. 8. Spatial distribution of gamma-ray flux (left) and dose (right) in the air cell

In the figure 7 and 8, the horizontal axis means the 50m length of the hot cell, the each vertical 8 cells mean the 8m width of the hot cell, and the 10 vertical groups mean 10m height.

Compared to the neutron flux and dose in the figure 7, gamma-ray flux and dose decrease is clearly shown in the figure 8. However, even they are decreased, the gamma-ray flux/dose are much higher than neutron flux/dose in the air cell processes. This results should be considered in development of the safeguards and shielding structures.

The remained results in the Ar cell are under analysis. The results in the Air and Ar cell will be presented at the spring meeting of Korean Nuclear Society.

4. Conclusion

In this study, the radiation flux and dose distribution in the hot cell environment of the pilot-scale pyroprocessing facility are investigated preliminarily by MCNP6 simulation. From the analysis of material flow of the major processes, the material composition model and neutron/gamma-ray energy spectra were established. With these information, the MCNP6 simulation is performed for calculation of neutron/gamma-ray flux/dose distribution in the air and Ar cell. The simulation results including the both cells will be presented at the spring meeting of Korean Nuclear Society.

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