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# PRELIMINARY SAFETY STUDY OF ENGINEERING-SCALE PYROPROCESS FACILITY

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Pyroprocess technology has been considered as a fuel cycle option to solve the spent fuel accumulation problems in Korea. The Korea Atomic Energy Research Institute has been studying pyroprocess technology, and the conceptual design of an engineering-scale pyroprocess facility, called the Advanced Fuel Cycle (AFC) facility, has been performed on the basis of a 10tHM throughput per year. In this paper, the concept of the AFC facility was introduced, and its safety evaluations were performed. For the safety evaluations, anticipated accident events were selected, and environmental safety analyses were conducted for the safety of the public and workers. In addition, basic radiation shielding safety analyses and criticality safety analyses were conducted. These preliminary safety studies will be used to specify the concept of safety systems for pyroprocess facilities, and to establish safety design policies and advance more definite safety designs.

KEYWORDS : Pyroprocess, Pyroprocess Facility, Safety Evaluation, Hazard Analysis

## 1. INTRODUCTION

Spent fuel (SF) is inevitable byproduct of nuclear power generation. Spent fuel is highly radioactive waste, which contains uranium (U), transuranic elements (TRU), and fission products. The direct disposal and interim storage of spent fuel require wide and isolated areas, and thus it is not easy to find proper sites in Korea. Therefore, the development of an effective management or recycling technology of spent fuel is essential to enhance nonproliferation and environmental friendliness.

In Korea, pyroprocess technology has been considered as a fuel cycle option to solve spent fuel accumulation problems. Pyroprocessing is one of the key technologies used to recover actinide elements and long-lived fission products from the spent fuel in LiCl or LiCl-KCl molten salt by an electro-chemical reaction, and it is known that the technology is more advantageous than existing PUREX in terms of nonproliferation. KAERI (Korea Atomic Energy Research Institute) has been developing a pyroprocess technology for the recycling of spent fuels. PRIDE (PyRoprocess Integrated inactive DEmonstration facility) had been developed from 2007 to 2012 as a cold test facility to support integrated pyroprocessing and an equipment demonstration, which is essential to verify the pyroprocess technology [1, 2]. In PRIDE, depleted uranium is used for the process, and the maximum throughput is 10tHM per year. As the next stage of PRIDE, the design requirements of an engineering-scale demonstration facility are being developed, and a conceptual design of the facility is being performed. INL (Idaho National Laboratory) conducted a conceptual design of an AFCF (Advanced Fuel Cycle Facility) and accident analyses for AFCF to support the development of advanced technologies related to safeguards and security, instrumentation, process control and integration, and to provide data on the reliability and scale-up for full-scale separations and fuel fabrication facilities [3-6]. Also, JNC (Japan Nuclear Cycle Development Institute) have proposed the concept of safety systems in pyrochemical reprocessing systems and performed safety evaluations [7].

In this paper, the concept of the AFC (Advanced Fuel Cycle) facility was introduced, and its preliminary safety evaluations were performed. For the safety evaluations, anticipated events and accident events were selected, and environmental safety analyses were conducted for the safety of the public and workers. In addition, basic radiation shielding safety analyses and criticality safety analyses were conducted.

## 2. CONCEPTUAL DESIGN OF AFC FACILITY

The AFC facility for the pyroprocess demonstration consists of (a) processing equipment, (b) a hot cell facility, and a building structure to shield and isolate the process equipment, (c) h ot cell remote operation equipment for safety operation and maintenance, (d) an argon system to control the inert atmosphere of a process cell, (e) a utility supply facility, (f) material receipt and storage areas for spent fuel, (g) and a waste treatment area and a shipping facility.

The main process is composed of the disassembly and rod cutting of a spent fuel assembly, chopping and decladding, voloxidation, electrolytic-reduction, electro-refining, electro-winning, salt purification and recovery, waste form fabrication, off-gas treatment, and so on. Fig. 1 shows a flow diagram of the reference pyroprocess developed by KAERI.

### 2.1 Design Requirements

The AFC facility allows a maximum of 10tHM/yr of pressurized water reactor (PWR) fuel. The other top-tier requirements such as the operation rate, product and waste storage facility, reference spent fuel, facility design life, and so on were given in Table 1.

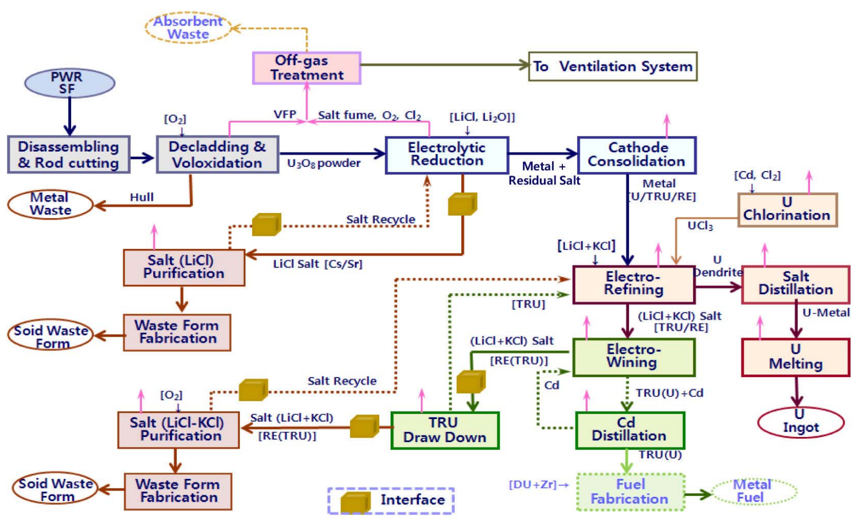



Fig. 1. Process Flow Diagram of the AFC Facility

**Table 1.** Top-tier Requirements of AFC Facility

| Item | Requirements |
|---|---|
| Throughput | ·10tHM/year |
| Reference Spent Fuel | ·16×16 PWR Type, 4.5wt.% U-235<br>·55,000 MWD/MTU, 10 Years Cooling |
| Availability | ·70% (in Consideration of O&M Outage)<br>·200 Equivalent Full Operating Calendar Days |
| Design Life | ·40 Years (Building and Cell Structure)<br>·20 Years (Equipment) |
| Input | ·PWR Spent Fuels |
| Output | ·U Metal Ingot as LLW, U-TRU-RE Metal Ingot for SFR Fuel<br>·Wastes (Ceramic, Metal, Virtrified Form) |
| Main function | ·Temporary Material Storages (PWR Spent Fuel, Metal Ingot, Waste) |
| | ·PWR Spent Fuel Disassembling, Rod Chopping |
| | ·Decladding, Voloxidation, Electrolytic Reduction, Electro-Refining, Electro-winning, CD distillation, Cathode Processing<br>·U and TRU Metal Ingot Fabrication |
| | · Salt Waste Recycling, Waste Treatment, Off-gas Treatment |

The Safety Class, Seismic Class, and Quality Class of the structures, systems and components (SSCs) of a nuclear facility are classified according to their functions. In the case of the AFC facility, there are no SSCs considered as Safety Classes 1 and 2. A hot cell structure and other SSCs requiring an equivalent structural integrity with the hot cell are classified in Safety Class 3, which can be assigned to the SSCs of which a loss of function can cause the radiological dose limit at the site boundary to be exceeded. In the AFC facility, all of the Safety Class SSCs are considered as Seismic Category I, and the SSCs having the possibility to affect the loss of safety functions of Seismic Category I SSCs under an earthquake were considered as Seismic Category II SSCs. The main building structure and overhead crane in the AFC facility can be categorized as Seismic Category II. Table 2 shows the main SSCs' classification for the AFC facility. The hot cell structure and hot cell inlet/outlet filter performs a safety function isolating the radioactive material, and its loss of safety function can cause the radiological dose limit at the site boundary to be exceeded, and thus they were considered as the SSCs of Safety Class 3, Seismic Category I, and Quality Class Q. The hot cell linear, radiation shielding window, transfer lock, rear door, and feed through were also classified as the same class as the hot cell structure.

In the AFC facility, the argon system was designed to control the impurities and maintain negative pressure in the argon atmosphere cell. The argon system consists of an argon supply unit, an argon gas cooling and circulation unit, and an argon gas purification unit. The argon gas purification unit has a function to maintain less than 15ppm water vapor and 40ppm oxygen in the cell atmosphere. The argon gas pressure release unit controls the pressure in the cell from excess overpressure of 75mmAq and an underpressure of -300mmAq.

## 2.2 Facility Layout

The AFC facility is divided into a main process building and support buildings, as shown in Fig. 2. The hot cells are contained within 3 stories of a large, single 7-story main process building including 1 basement level. The building has a length of 100m, a width of 40m, and a height of 48m, including a 9m high basement. The auxiliary buildings used to support the main process building are composed of an administration building, a fire house, a workshop building, a gas storage building, a utility building,

**Table 2.** Main Equipment Classification of AFC Facility

| Main Systems and Components | Safety Class | Seismic Class | Quality Class |
|---|---|---|---|
| 1. Hot Cell Sturcture | 3 | I | Q |
| 2. Hot Cell Liner | 3 | I | Q |
| 3. Raidation Shielding Window | 3 | I | Q |
| 4. Transfer Lock | 3 | I | Q |
| 5. Feed-through | 3 | I | Q |
| 6. Read Door | 3 | I | Q |
| 7. Manipulator | NNS | III | A |
| 8. In-cell Crane/EMM or TM | NNS | III | A |
| 9. Inter-cell Door | NNS | III | A |
| 10. Fire Protection System in Hot Cell | NNS | III | S |
| 11. Process Equipment in Hot Cell | NNS | III | S |
| 12. Electrical System in Hot Cell | NNS | III | S |
| 13. Ar System | NNS | III | A |
| 14. Hot Cell Inlet/Outlet Filter | 3 | I | Q |
| 15. AHU/ACU | NNS | II | A |
| 16. Compressed Air System | NNS | III | S |
| 17. Cooling Water System | NNS | III | S |
| 18. Main Building | NNS | II | A |
| 19. Overhead Crane(Main Building) | NNS | II | A |

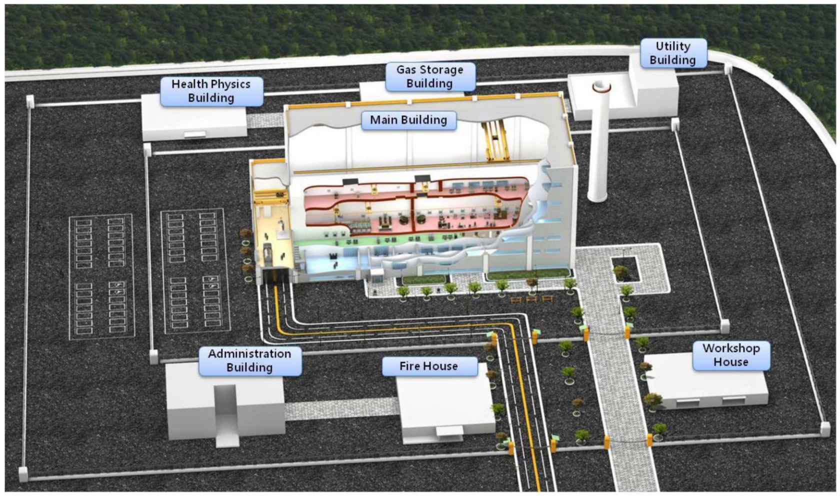



Fig. 2. Conceptual Design Layout of AFC Facility

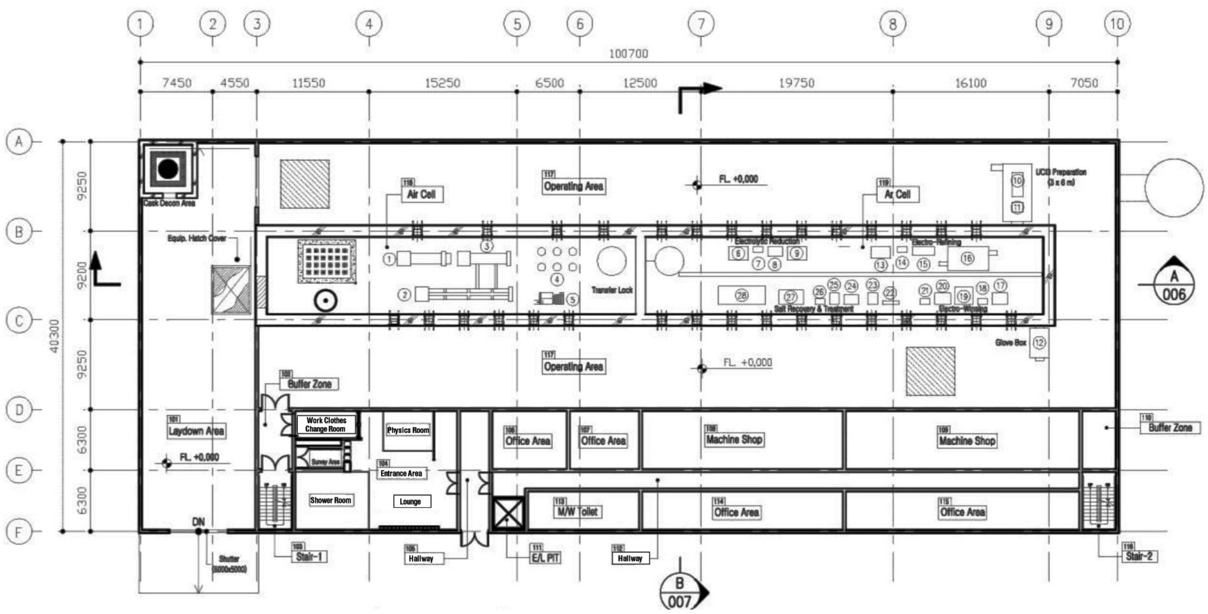


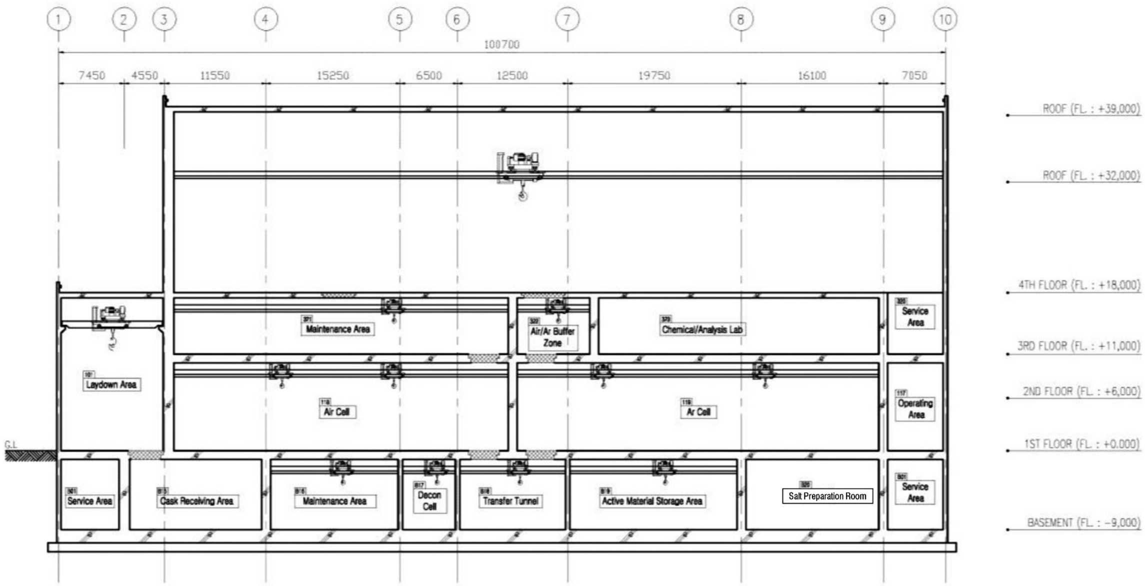

Fig. 3. Conceptual Design Layout of 1st Floor

and a health physics building, which are located around the main process building.

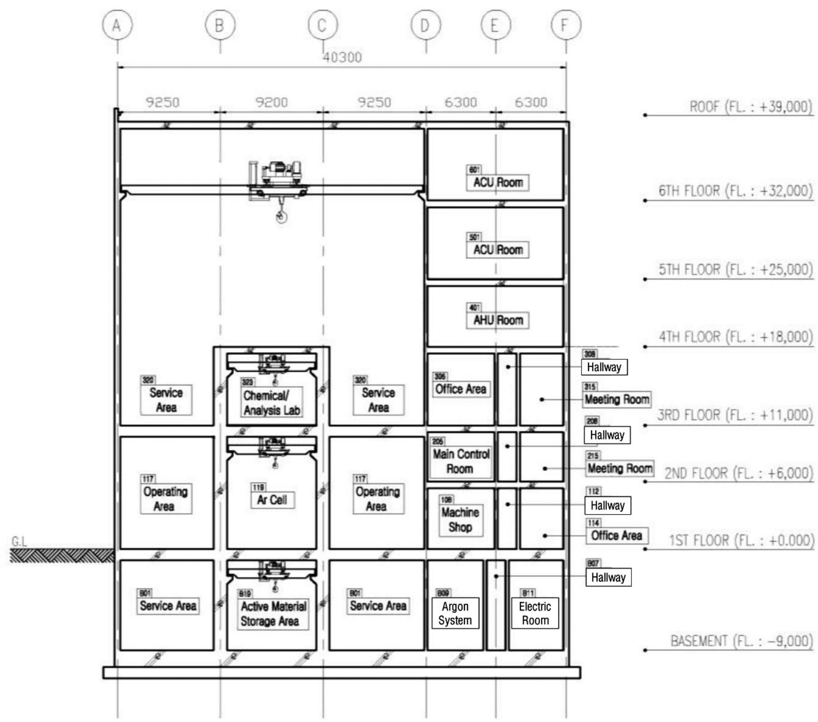
The 1st floor provides space for the process cells, operating area, service area, main entrance area, truck bay, office area, and so on, as shown in Fig. 3. The decontamination cell, a storage room for the waste and process products, an electric room, an argon system, a service area, a utility supply system area, and so on were arranged to be located in the 1st basement level. A maintenance cell, a chemical analysis laboratory, an office area, and a showing area are provided on the 3rd floor, and a HVAC (Heating, Ventilation, Air Conditioning) room was arranged on the 4th – 6th floors. Sectional views of the main process building are shown in Fig. 4, where the overall layout can be seen.

## 3. PRELIMINARY SAFETY ANALYSIS

The AFC facility was designed to treat spent fuel and chemically toxic materials, and thus the safety for the public and workers should be protected from the radiological hazards and chemical hazards of facility operation. For successful safety evaluations, the three key elements should be required such as operation and functional requirements




(a) Front Section View




(b) Side Section View

Fig. 4. Sectional View of Conceptual Design Layout

for safety SSCs, hazard analysis technique, and safety analysis technique. Fig. 5 shows the safety evaluation procedure for safety SSCs. As a result of a hazard analysis, the design basis accident scenarios are determined, and the initial design concept for safety SSCs should then be changed according to the safety analysis results. This design process can be very iterative, and thus this procedure should be applied in the initial design stage.

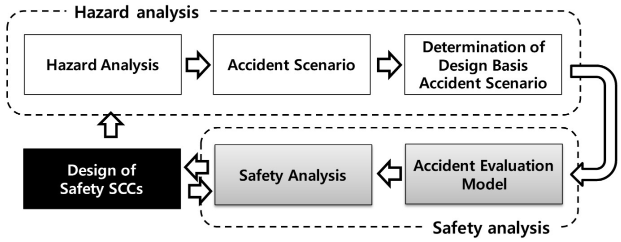



Fig. 5. Safety Evaluation Procedure

**Table 3.** Risk Ranking Matrix

| Consequence Category | Frequency Category | | | | |
|---|---|---|---|---|---|
| | I | II | III | IV | V |
| A | 1 | 1 | 2 | 2 | 3 |
| B | 1 | 1 | 2 | 3 | 4 |
| C | 2 | 2 | 3 | 3 | 4 |
| D | 3 | 3 | 4 | 4 | 4 |

**Table 4.** Representative Hazard Evaluation Results

| Hazard Type | Accident Scenario | Frequency Categroy | Consequence Categroy | Risk Ranking |
|---|---|---|---|---|
| Radiological | Release of radioactive materials due to hot cell fire | III | A | 2 |
| Toxic | Release of chlorine gas due to pipe rupture | III | A | 2 |
| Toxic | Release of argon gas due to argon supply pipe rupture | III | A | 2 |

### 3.1 Determination of Accident Scenarios

The hazard analysis is performed to identify and evaluate potential accidents, and to identify bounding accident scenarios (design basis accident scenarios) that require further quantitative development. In addition, the technical safety requirements (TSRs) for defense in depth and the significant safety functions performed by SSCs are established by hazard evaluation results.

Hazard identification was conducted to identify and characterize hazardous materials and energy sources associated with the operations and inventory of the AFC facility. The fundamental hazards affecting the AFC facility can be categorized into process-related hazards, natural hazards, and manmade external hazards, and spent fuel, radioactive materials, toxic materials, and combustibles are included in process-related hazard materials. The hazard identification activities were conducted, and some process-related hazards were identified. However, no external events were identified as a unique hazard. In this study, a preliminary hazard analysis (PHA) was used to evaluate hazards. The results of the PHA serve as the basis for hazard ranking so that bounding accident scenarios can be selected. Hazard ranking is determined by qualitatively assigning frequency and consequence estimates to each hazard or accident scenario developed by the PHA. The hazard frequency is categorized into 5 grades: I, II, III, IV, and V, and the hazard consequence severity is classified into 4 grades: A, B, C, and D [8]. Table 3 shows the risk ranking matrix used to compare all hazards and accident scenarios identified in the PHA. Table 4 shows the representative accident scenarios finally selected by applying PHA and a hazard ranking matrix.

### 3.2 Environmental Safety

Most significant processes and operations in the AFC facility take place within the confined hot cell, where both the air and argon in the hot cell would be released through the 2nd stage HEPA filters. Therefore, it is expected that various types of accident conditions may have little effect on the public, workers, and environment. A representative accident was analyzed to verify that the operation of the AFC facility is safe and no threat to man or nature.

The accident analysis was performed for the case of a hot cell fire, which is considered as the greatest accident event influencing exposure dose at the site boundary. The hot cell fire scenario is the damage accident of off-gas treatment equipment by fire in the hot cell, and thus the collected radioactive materials are released into the environment. The key assumptions used for the calculation of radioactive material emission rate are as follows:

(1) 100% of the collected radioactive materials, which is accumulated for 1 year, in the off-gas treatment equipment is released, but only 50% of Xe and Kr, which is accumulated for 6 months, is released because the radioactive materials are retained in the equipment for 6 months.
(2) In-cell filter and ACU (Air Conditioning Unit) do not function due to fire.
(3) All of the radioactive materials are released into the environment within 2 hours.
(4) The final release fraction values are as listed below. These values are for a Hazard Category 2 facility, produced by the U.S. DOE (Department of Energy) [9].

- Gases ($^3$H, Kr, Xe, Ar, Rn, Cl): 1.0
- Highly volatile/combustible (P, S, K, I, Na, Br): 0.5
- Semi-volatile (Se, Hg, Cs, Po, Te, Ru, C): 0.01

The atmospheric dispersion model, PAVAN (Potential Accident Consequence Assessments at Nuclear Power Plants), was used to provide the short-term atmospheric dispersion factors ($\chi$/Q) for an assessment of the consequences of the accident. The following assumptions were used for the calculations.

(1) Effective release height: 0m, ground level height
(2) Meteorology based on 2 year accumulation, which is referred to in a preliminary safety analysis report of Shin-Kori 1 and 2 nuclear power plants
(3) Site boundary location: 560m
(4) Wet and dry depositions of radioactive material are zero for individual receptors
(5) Inhalation and external exposure from a plume
(6) Breathing rate: $3.47 \times 10^{-4} m^3/s$ [10]

The $\chi/Q$ values are calculated for 16 sectors at a distance from the AFC facility. Table 5 represents the short-term atmospheric dispersion factors for the accident release from the AFC facility. The maximum $\chi/Q$ at the site boundary is $4.055 \times 10^{-4} s/m^3$ in the NNW sector and this value is used to calculate the dose. The maximally exposed individual (MEI) dose is calculated using conservative assumptions, including the MEI at the site boundary in the NNW direction with along the plume centerline [2].

The effective dose from external exposure and equivalent dose due to the thyroid received by inhalation was calculated and summarized in Table 6. The dose rate limit for unlikely accidents is 250mSv to the whole body, and 3,000mSv of equivalent dose rate to the thyroid, which is embodied in 10CFR100.11 [11]. The ratio of the effective dose to the limit is 2.1% and 21.2% for effective dose due to external exposure and equivalent dose due to the thyroid, respectively, and thus the results show that the design requirement is satisfied.

### 3.3 Radiation Shielding Safety

The AFC facility is dedicated to the mission of the pyroprocessing of spent fuel. The radiation shielding analysis is conducted to determine the thickness of high-density concrete walls, which ensure that radiation doses to the workers from radiation exposures are maintained below the regulatory limits. The dose limits was presented in Table 7, which were determined by considering the maximum dose constraint for workers, 20mSv in a year, as recommended in ICRP-60 and working hours at each area of hot cells.

The following assumptions were used to determine the source term for the shielding analysis.

**Table 5.** Short-term Atmospheric Dispersion Factors (0 ~ 2 hour) for Accident Releases from AFC Facility

| Direction | Atmospheric Dispersion Factors (at Site Boundary) |
|---|---|
| S | $3.189 \times 10^{-4}$ |
| SSW | $3.002 \times 10^{-4}$ |
| SW | $2.641 \times 10^{-4}$ |
| WSW | $2.303 \times 10^{-4}$ |
| W | $2.773 \times 10^{-4}$ |
| WNW | $4.003 \times 10^{-4}$ |
| NW | $4.083 \times 10^{-4}$ |
| NNW | $4.055 \times 10^{-4}$ |
| N | 0.000 |
| NNE | 0.000 |
| NE | $1.880 \times 10^{-4}$ |
| ENE | $2.074 \times 10^{-4}$ |
| E | $2.103 \times 10^{-4}$ |
| ESE | $1.897 \times 10^{-4}$ |
| SE | $2.264 \times 10^{-4}$ |
| SSE | $3.098 \times 10^{-4}$ |

**Table 6.** Calculated Effective Doses

| | Effective Does | Equivalent Does for Thyroid |
|---|---|---|
| Dose (mSv) | 5.2 | 635 |
| Dose Limit (mSv) | 250 | 3,000 |
| Ratio (%) | 2.1 | 21.2 |

**Table 7.** Radiation Shielding Result; Wall Thickness Satisfied with Dose Limit

| Objective | | Dose Limit (mSv/hr) | Wall Thickness (m) |
|---|---|---|---|
| Temporary Spent Fuel Storage Vault | Upper Part | 10 (High Radiation Area) | 0.4 |
| | Side and Lower Part | 0.05 (Service Area) | 0.7 |
| Head-end and Main Process Cell | Side Part | 0.01 (Operating Area) | 1.0 |
| | Upper Part | 10 (High Radition Area) | 0.5 |
| Active Material Storage Cell | Side Part | 0.05 (Service Area) | Not Determined |

(1) 10tHM of spent fuels (24 spent fuel assembly), which is the throughput per year, was stored in the temporary spent fuel storage vault.
(2) 5tHM of spent fuels was contained in the head-end or main process cell by considering the process characteristics.

The MCNP-X code was used to evaluate the potential dose from the source term, and the γ and neutron emission rate for the reference fuel were calculated by ORIGEN-ARP of SCALE code Ver. 5.1. Fig. 6 shows the model of temporary spent fuel storage vault for shielding analyses, and 24 (4×6 array) spent fuel assemblies are stored in the 10mm-think steel cask. Fig. 7 shows the analysis model of the head-end or main process cell. In the case of an active material storage area, the shielding walls for the product and waste casks should be additionally installed to meet the dose limits, and thus the design and shielding analysis for the storage area should be performed during the final design stage of the AFC facility. It was assumed that the density of concrete, steel, and air is 3.457, 7.870, and 0.001293 g/cm$^3$, respectively. The shielding walls were modeled in a 1.0m thickness, and a series of shielding analyses were conducted to determine the radiation shielding performance with various detection locations at intervals of 0.1m from the inner surface of the shielding wall.

The results of the calculations are summarized in Table 7, and some of the results were plotted in Fig. 8. The thickness of the side and lower walls, and the upper wall of the spent fuel storage vault are 0.7m and 0.4m, respectively, to meet the dose limits. Also, in the case of the main process cell, a 1.0m side wall and 0.5m upper wall are needed to meet the standard limits.

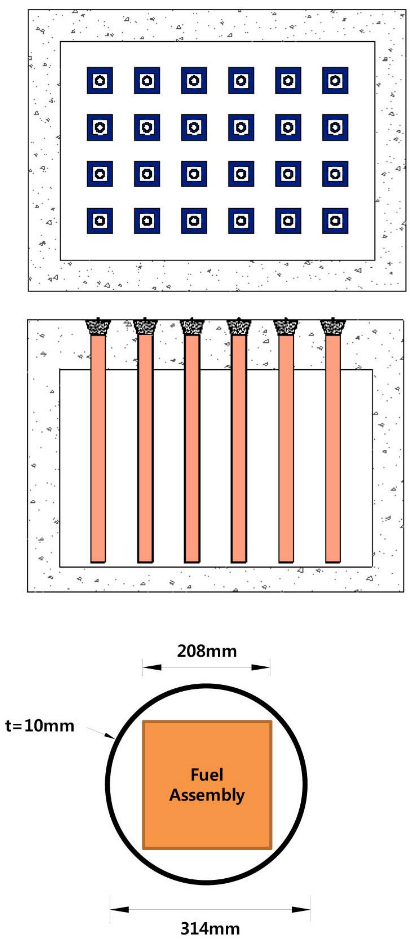



Fig. 6. Shielding Analysis Model of Temporary Spent Fuel Storage Vault

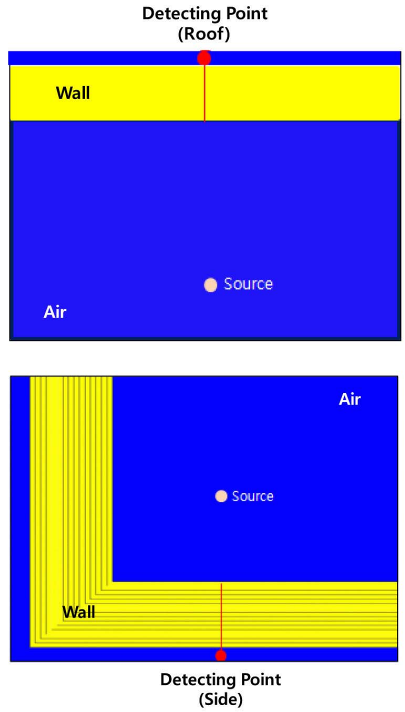



Fig. 7. Shielding Analysis Model of Temporary Spent Fuel Storage Vault

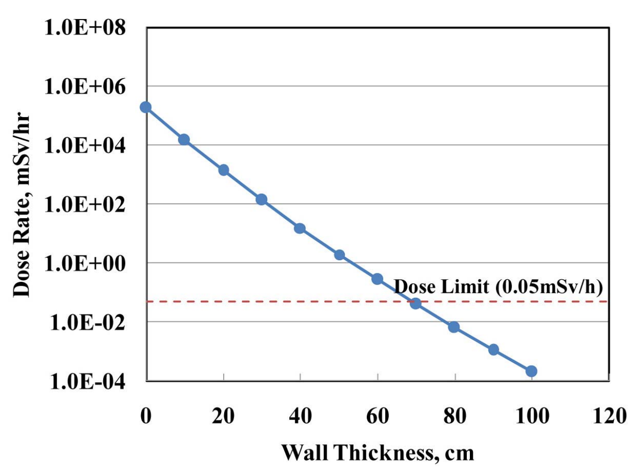



Fig. 8. Example of Shielding Analysis Results; Dose Rate at Side Wall of Spent Fuel Storage Vault

### 3.4 Criticality Safety

The nature of AFC operations makes a criticality event highly unlikely, however, the AFC facility will process and store fissile materials in sufficient quantity, and thus it is necessary to confidently guarantee that criticality cannot occur in the AFC facility under all normal, abnormal, and accident conditions. A maximum value for the effective multiplication factor ($K_{eff}$) including uncertainty and bias is used to evaluate the criticality safety. The $K_{eff}$ should be less than 0.98 under the condition of the highest anticipated reactivity, assuming optimum moderation as recommended by NUREG-0800, and less than 0.95 under the submerged and water filled condition as recommended in ANSI 57.3-1983. The $K_{eff}$ must include allowance for all relevant uncertainties and tolerances.

Criticality calculations were estimated with the MCNP-X code in a 3D geometry of the TRU metal ingot equipment and storage vaults for the process products shown in Fig. 9 and 10. It was assumed that the minimum critical mass of fissile materials (TRU) is 5.6kg, which is the critical mass of $^{239}$Pu, and the operational mass limit of fissile materials was determined as 4.5kg of $^{239}$Pu on the assumption that the fissile material is composed of only pure $^{239}$Pu and the safety factor is 0.8.

The following assumptions were used for the criticality calculations:

(1) Submerged and water filled conditions

(2) Cylinder shaped container for TRU ingots

The calculation results for the TRU metal ingot casting equipment were presented in Tables 8 and 9. It was calculated that the effective multiplication factor of the device would be sub-critical if the plutonium cylinder diameter is below 5.0cm, and the array of the container has little effect on the effective multiplication factor. Table 10 shows the calculated effective multiplication factor by various arrays of a TRU storage container with a diameter of 5cm and a distance of 20cm. In the case of a 3 layer array, the result shows that the TRU storage container slightly exceeded the limit. It is thought that the detailed criticality calculations considering the composition of plutonium and the array of containers should be conducted in the detailed design stage.

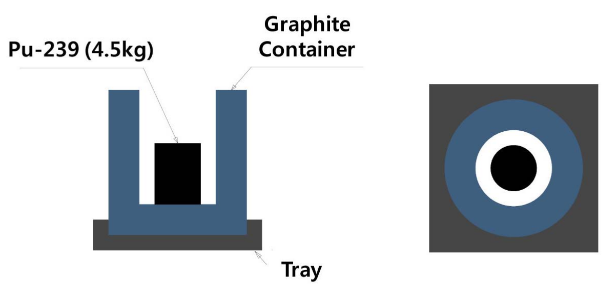



Fig. 9. TRU Metal Ingot Equipment

**Table 8.** Calculated Multiplication Factor by Various Diameter of Cylinder

| Diameter (cm) | Multiplication Factor ($k_{eff}$) |
|---|---|
| 6.0 | 0.9568±0.0028 |
| 5.0 | 0.9222±0.0028 |
| 4.4 | 0.8929±0.0030 |
| 4.0 | 0.7298±0.0024 |

**Table 9.** Calculated Multiplication Factor by Various Array of Cylinder

| Array | Multiplication Factor ($k_{eff}$) |
|---|---|
| 1×1 | 0.8929±0.0030 |
| 1×2 | 0.8905±0.0083 |
| 1×3 | 0.8938±0.0027 |

**Table 10.** Calculated Multiplication Factor by Various Array of Storage Container

| Array | Multiplication Factor ($k_{eff}$) |
|---|---|
| 1 Layer 2×4 | 0.9319±0.0031 |
| 2 Layer 2×4 | 0.9342±0.0030 |
| 3 Layer 2×4 | 0.9353±0.0162 |

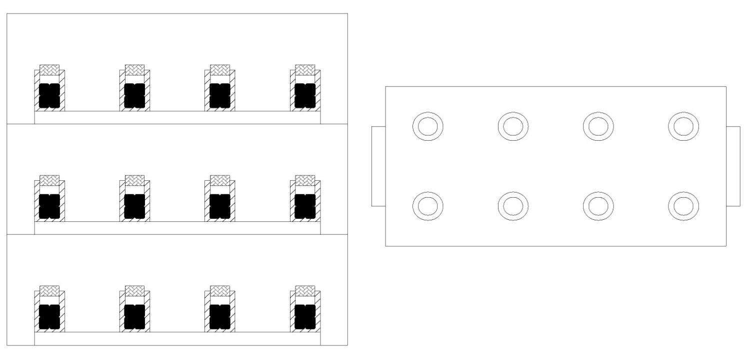


Fig. 10. TRU Metal Ingot Storage Container

### 3.5 Structural Safety

The structural safety during earthquakes was not considered because the structural framing plan is not determined in this conceptual design stage.

## 4. SUMMARY

The development of pyroprocess facilities for an effective management of spent fuel is essential to the long-term success of nuclear energy policy in Korea. In this paper, the conceptual design concept of an engineering-scale pyroprocess facility, Advanced Fuel Cycle (AFC) facility, developed by the Korea Atomic Energy Research Institute, was reviewed, and its preliminary safety evaluations were conducted. The key results are as follows:

(1) As a result of hazard analysis, a hot cell fire scenario was selected as the greatest accident event influencing on the exposure dose at the site boundary, and it was verified that dose rates don't exceed the standard limits.

(2) Radiation shielding analyses were conducted to determine the thickness of a hot cell wall assuring of radiation shielding safety.

(3) Criticality calculations were carried out to design the manufacturing equipment and storage container for TRU metal ingots to prevent criticality.

These preliminary safety studies will be used to specify the concept of safety systems for pyroprocess facilities, and to establish safety design policies and advance more definite safety designs.

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