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PREFACE

This document(draft) contains DUPIC fuel cycle development activities to be carried out for 5 years(April 2002 ~ March 2007) in addition to the work scope described in the report KAERI/AR-510/98-rev.1, which was attached to Joint Determination for Post-Irradiation Examination of Irradiated Nuclear Fuels, signed by MOST and US Embassy in Seoul, Korea on April 8, 1999.

This document is prepared for the purpose to review that overall DUPIC fuel development activities are within the scope and contents in compliance to Article 8(C) of ROK-U.S. cooperation agreement.

This document(draft) contains;

-Chapter Ⅰ : Manufacturing Program of DUPIC Fuel in DFDF

-Chapter Ⅱ : Post Irradiation Examination Program of DUPIC Fuel

-Chapter Ⅲ : Safeguards of Nuclear Materials

CONTENTS

Chapter Ⅲ. Safeguards of Nuclear Materials

List of Tables

List of Figures

Chapter I

Manufacturing Program of DUPIC Fuel in DFDF

1. Objectives and Necessity

Since Korea has PWRs and CANDU reactors for the electricity generation, the development of the new fuel, DUPIC(Direct use of spent PWR fuel in CANDU reactor), would have many benefits in terms of uranium utilization, reduction of accumulation of spent fuel and non-proliferation. It is believed that the spent PWR fuel, which contains about 1.5% fissile, can be reused without the conventional reprocessing in CANDU reactor.

Following the DUPIC feasibility study, the DUPIC research is now in the stage of experimental verification of DUPIC fuel performance, which has been performed since 1993 in cooperation of AECL in Canada, USA and IAEA.

The main objective of the DUPIC research is to manufacture several DUPIC fuel bundles, including pellets and fuel rods, from spent PWR fuel by only utilizing thermal and mechanical processes, irradiate them at research reactors and perform PIE study to experimentally verify the performance of DUPIC.

Manufacturing program of DUPIC fuels consists of disassembling of spent PWR assemblies at PIEF(Post Irradiation Examination Facility) and fabrication of several DUPIC fuel bundles, including pellets, mini elements and fuel elements, at DFDF(DUPIC Fuel Development Facility), which is developed at IMEF(Irradiated Materials Examination Facility) at KAERI. Irradiation tests of DUPIC fuel bundles including pellets and fuel elements at research reactor, post-irradiation examination at PIEF/IMEF after irradiation and treatment of radioactive waste at RWTF(Radioactive Waste Treatment Facility) at KAERI are to be performed.

The objectives of the DUPIC research are :

- To manufacture several DUPIC fuel bundles, including pellets, mini elements and fuel elements, from spent PWR fuel for the performance evaluation by irradiation test and PIE

- To verify the feasibility of remote fabrication of DUPIC fuel
- To accumulate technical experience of equipment development, operation and maintenance of hot cell equipment
- To perform PIE study to experimentally verify the feasibility and the performance of DUPIC fuel

2. Specifications of DUPIC Fuel

2.1 DUPIC Fuel Bundle

The DUPIC fuel will be manufactured in accordance with CANFLEX(CANdu FLEXible) and CANDU-6 fuel geometry. The specifications of the DUPIC fuel bundle are shown in Table I-1 and Fig. I-1.

Table I-1 Specifications of DUPIC Fuel Bundle

	Specifications(CANFLEX)	Specifications(CANDU-6)
No of elements Bundle length Bundle diameter Outer diameter of element	43 (8 are larger and 35 are smaller dia. elements) 495.30 ± 0.75 mm 102.50 mm max. - larger element : 13.50 mm - smaller element : 11.50 mm	37 495.30 ± 0.75 mm 102.50 mm max. 13.08 mm
Cladding - Material - Thickness	 Zircaloy-4 0.33 mm for smaller, 0.36 mm for larger element	 Zircaloy-4 0.42 mm
Weight - Fuel oxide+FP - Zircaloy	 ~21 kg ~2 kg	the same as left
Composition - U-235 - U-236 - U-238 - Pu-239 - Pu-240 - Pu-241 - Pu-242 - Am-241 - Am-243 - Total Pu	 0.164 kg 0.056 kg 17.500 kg 0.086 kg 0.034 kg 0.012 kg 0.007 kg 0.008 kg 0.001 kg 0.160 kg	the same as left

* In case adding fresh depleted and/or low enriched UO_2 powder and/or burnable poison material is decided to meet the linear element rating and fuel performance requirements, this composition could vary.

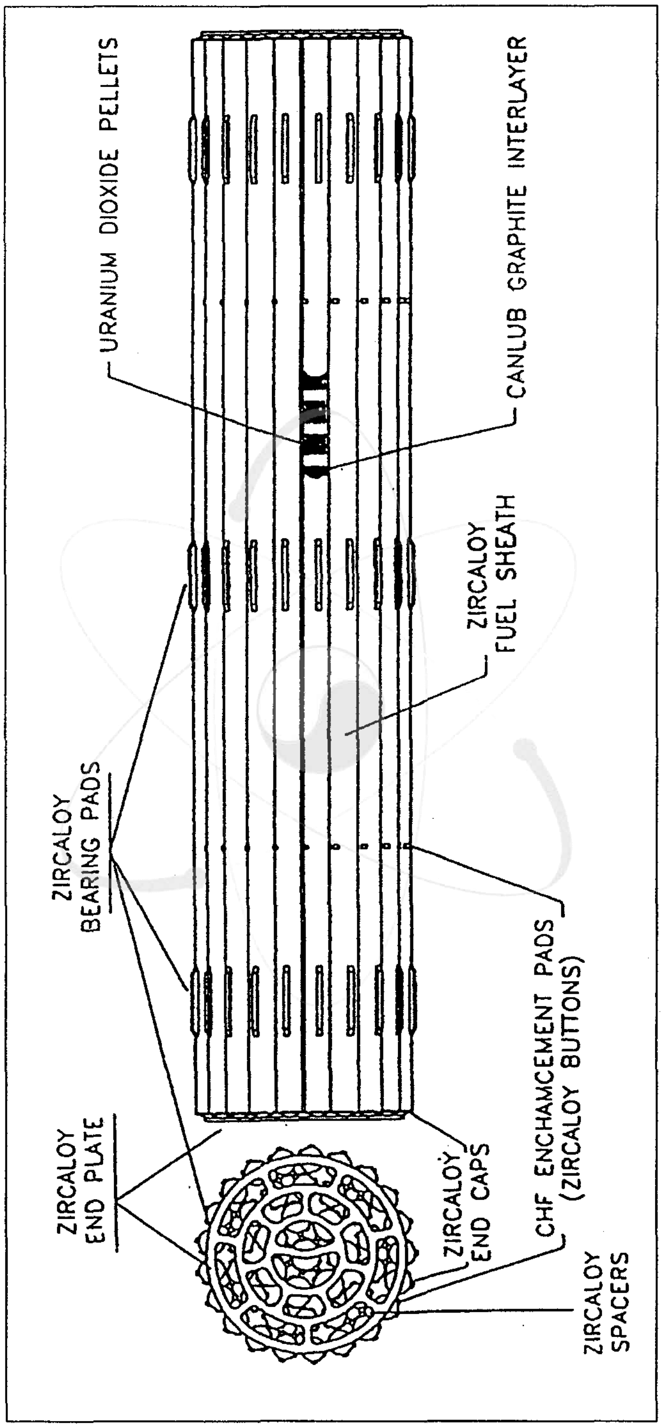


Fig. I-1. Schematic Diagram of the DUPIC Bundle(CANFLEX)

The specifications are based on the assumption that the DUPIC fuel bundles are made from only spent PWR fuel. In case small amount of slightly low enriched or depleted uranium dioxide might be added to satisfy the linear element rating requirement in the research reactor and the burnable poison material, Dy-164, might be also added to some of the elements to meet the performance requirement in CANDU reactor, its composition could vary a little.

Besides, in case the inner elements of DUPIC fuel bundle might be made from the natural uranium dioxide, the composition could also vary a little. Several DUPIC fuel may be fabricated in accordance with standard CANDU-6 fuel geometry, which has 37 fuel elements, in order to satisfy the requirements for the irradiation tests.

2.2 Spent PWR Fuel

Several DUPIC fuel bundles, including mini elements and fuel elements, are planned to be manufactured in accordance with CANFLEX or CANDU-6 fuel geometry. Considering the low yield rates due to its pioneering work of remote fabrication of DUPIC fuel, 200 kg U-base of spent PWR fuel would be required for manufacturing of several DUPIC fuel bundles. The total amount of spent PWR fuel to be used for the DUPIC fuel development could be reduced by the development of recycling technology of the clean scrap from the DUPIC fuel fabrication processes. Since KAERI has several spent PWR fuel assemblies which were transported from Kori nuclear power plant and are currently stored at PIEF, it would be convenient to use them. Among them, the G23 fuel assembly was selected on the basis of the calculation results of estimating the average burnup of spent PWR fuels discharged from the Korean Kori nuclear power plant.

The brief specifications and irradiated history of the G23 spent PWR fuel assembly are as follows;

• specifications

- 14×14 type fuel assembly irradiated in Kori unit 1 power plant
- enrichment : 3.21%
- the average burnup of the assembly : 35,502 MWD/MTU(cycle 4~cycle7)
- discharged date : 1986. 10. 24
- transported date to PIEF : 1990. 6. 1

• irradiated history

- cycle 4 : 261.17 EFPD
- overhaul : 51 days
- cycle 5 : 320.99 EFPD
- overhaul : 84 days
- cycle 5 : 288.41 EFPD
- overhaul : 75 days
- cycle 5 : 286.70 EFPD
- total EFPD : 1157.27 days

Other 14×14 type or 16×16 type spent PWR fuel assemblies might be used for the future manufacturing of DUPIC fuel.

The specifications of typical spent PWR fuel assembly are shown in Table I -2. Table I -2 is based on the spent fuel of 14×14 type in discharge burnup of 35,000 MWD/MTU as a reference.

Table I-2 Specifications of 14×14 Type Spent PWR Fuel Assembly

	Items	Specification
Fuel Assembly	- Rod array - Rod pitch, mm - Dimensions, mm - No. of instrument/guide thimbles	14×14 14.122 197.12×197.12 17
Fuel Rod	- Outside diameter, mm - Inside diameter, mm - Clad meterial - Length, mm	10.72 9.48 Zircaloy-4 3,857.6
Pellet(fresh)	- Material - Enrichment - Fuel stack length, mm	UO_2, sintered 3.21 wt% of U-235 3,657.6
Spent PWR Fuel Compositions (kg/MTU)	- U-234 - U-235 - U-236 - U-238 - Pu-239 - Pu-240 - Pu-241 - Pu-242 - Fissile Pu - Total Pu - Total fuel(kg)	0.136 7.625 4.211 941.39 5.305 2.229 1.320 0.556 6.624 9.410 374.9

3. Transportation

3.1 Specifications of Transportation Cask

Since the spent PWR fuel assemblies in PIEF are to be utilized for the manufacturing of DUPIC fuel, no additional transportation of spent PWR fuel assemblies from nuclear power plants is required.

However, the cask for the transportation of rod-cuts of spent PWR fuel elements from PIEF to IMEF cells for the fabrication is required.

Table I-3 shows the specifications of transportation casks for transporting rod-cuts, sample, mini-element, irradiation rig and solid waste.

Table I-3 Specifications of Transportation Casks

Cask	Specifications	Remarks
○ Padirac Cask(RD-10)		
- Weight(ton)	1.4	DUPIC sample transportation
- Inner diameter(mm)	340	
- Inner length(mm)	400	
- Lead thickness(mm)	100	
- Steel thickness(mm)	5	
○ Padirac Cask(RD-15)		
- Weight(ton)	2.6	
- Inner diameter(mm)	340	Rod-cuts and DUPIC sample transportation
- Inner length(mm)	400	
- Lead thickness(mm)	150	
- Steel thickness(mm)	5	
○ Hanaro Cask		
- Package type	B(U)	
- Burn-up(MWD/MTU)	120,000	Mini-element, element, irradiation rig transportation
- Design press(kg/cm^2)	26	
- Decay heat(W)	1,300	
- Inner length(mm)	1,067	
- Inner diameter(mm)	124	
○ Solid Waste Cask		
- Weight(ton)	6	
- Inner diameter(mm)	400	In-cell solid waste transportation
- Inner length(mm)	430	
- Lead thickness(mm)	150	

3.2 Transportation Route

The rod-cuts of spent PWR fuel elements which are cut at PIEF hot cells will be loaded into the padirac cask(RD-15) through back door of PIEF hot cell and transported to IMEF following the outside road. It will be unloaded through back door of DFDF hot cell.

The chemical samples of DUPIC powder and pellet will be loaded into the padirac cask(RD-15 or RD-10) through the back door of DFDF and transported to PIEF following the outside road. It will be unloaded through back door of PIEF 9406 hot cell.

The solid waste generated in DFDF will be loaded into the solid waste cask through the roof door of DFDF and transported to monolith of RWTF following the outside road after consultation with IAEA.

The DUPIC mini-elements and elements which are packed in an irradiation rig will be loaded into the HANARO cask through the pool of IMEF and transported to the pool of HANARO following the outside road.

The transportation method of DUPIC mini-elements, elements and bundles to research reactors in foreign countries will be determined later.

Also, the transportation method of Q.A. chemical analysis samples to foreign countries(having research reactors) will be determined later.

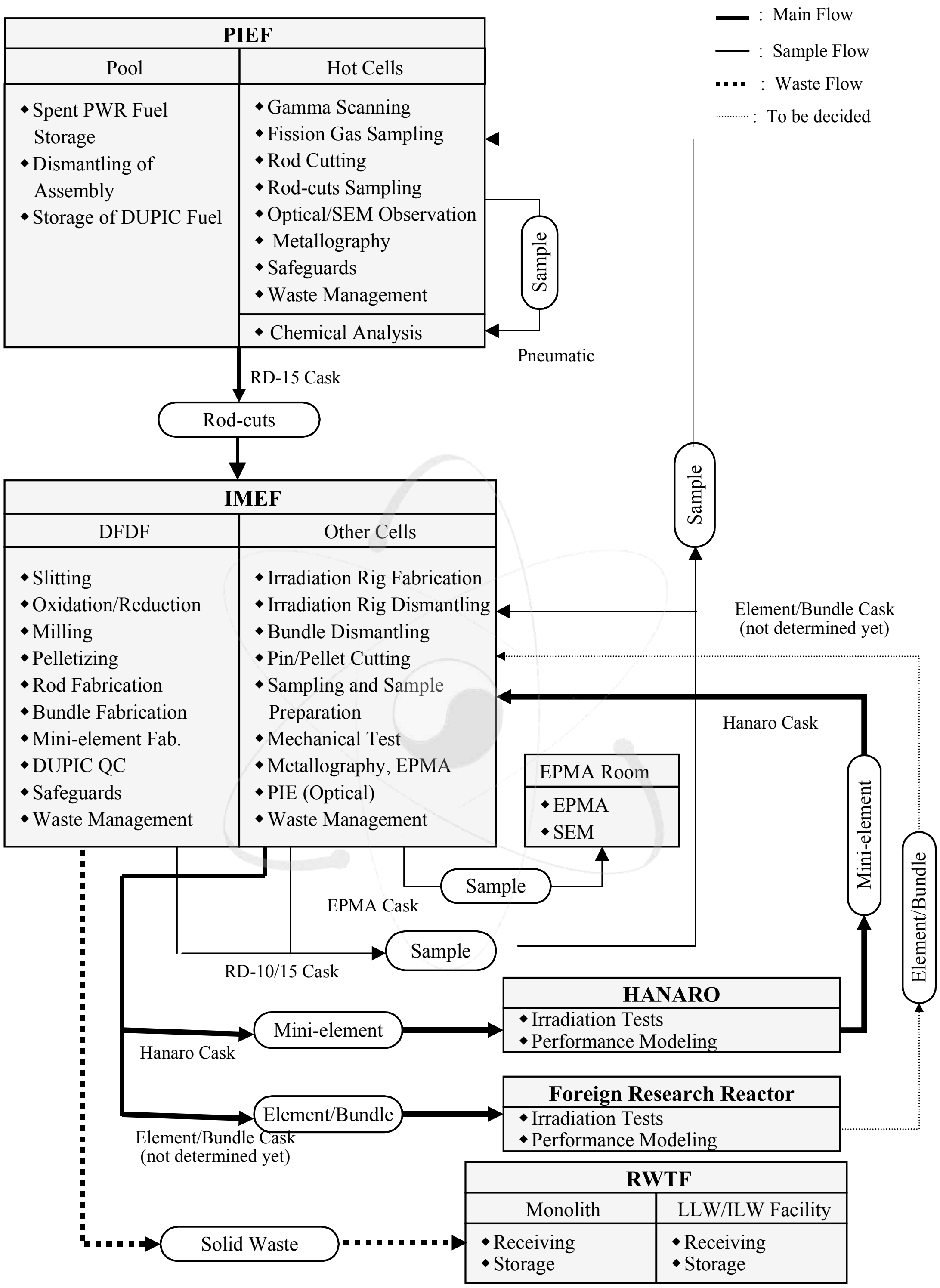


Fig. I-2. The Flow of Nuclear Materials in the DUPIC Fuel Development

4. Work Scope of DUPIC Fuel Development

4.1 Disassembling of Spent PWR Fuel at PIEF

- Disassembling the spent PWR fuel in pool at PIEF
- Inspection of extracted spent PWR fuel rods
- Cutting of the spent PWR fuel rods to rod-cuts of about 25 cm long

4.2 Manufacturing of DUPIC Fuel Bundles Including Pellets, Mini-elements and Elements at DFDF

- Decladding of rod-cuts by slitting and/or oxidation
- Powder preparation by oxidation/reduction and milling
- Pelletizing process by compaction, sintering and grinding
- Element manufacturing by welding
- Bundle manufacturing by welding

4.3 Quality Control at DFDF, IMEF and PIEF Cells

- Powder characterization at DFDF, IMEF and/or PIEF
- Dimension measurement at DFDF, IMEF and/or PIEF
- Nondestructive examination at DFDF, IMEF and/or PIEF
- Chemical analysis in the chemical analysis laboratory at PIEF

4.4 Equipment for DUPIC Fuel Manufacturing at DFDF

The overall equipment layout at DFDF is shown in Fig. I-3. The equipment for the element fabrication are installed at DFDF. But the equipment for the DUPIC fuel bundle such as blender, end-plate welder and bundle QC equipment, etc. will be installed later in accordance with the project schedule. If necessary, some equipment like oxidation/reduction furnace, precompacting press may be additionally installed to increase the present production capacity. The major equipments are as follows;

- Slitter
- O/M ratio measurement furnace
- OREOX furnace
- Attrition mill
- Mixer
- Press
- Sintering furnace
- Grinder
- End-cap welder
- Assembler
- End-plate welder
- QC equipments : micrometers, sieves and shaker, chemical balance, Hall flow meter(or Scott volumeter), mess cylinder, magnifier, Vernier calipers, surface roughness tester, He leak tester, QC furnace, Gamma-spectrometer, etc.
- Off-gas treatment system

5. Manufacturing Procedures

Overall manufacturing process flow is shown in Fig. I-4. The details of the processes would be improved as the research progresses.

5.1 Disassembling of Spent PWR Fuel

The end fittings of spent PWR fuel assemblies are cut and removed at PIEF pool. The spent PWR fuel rods are extracted and inspected at PIEF pool and transported to PIEF hot cell. Further tests such as fission gas analysis and burnup measurement are performed at PIEF hot cells if necessary. The spent PWR fuel rods are cut to rod-cuts of about 25 cm long for the transportation to DFDF.

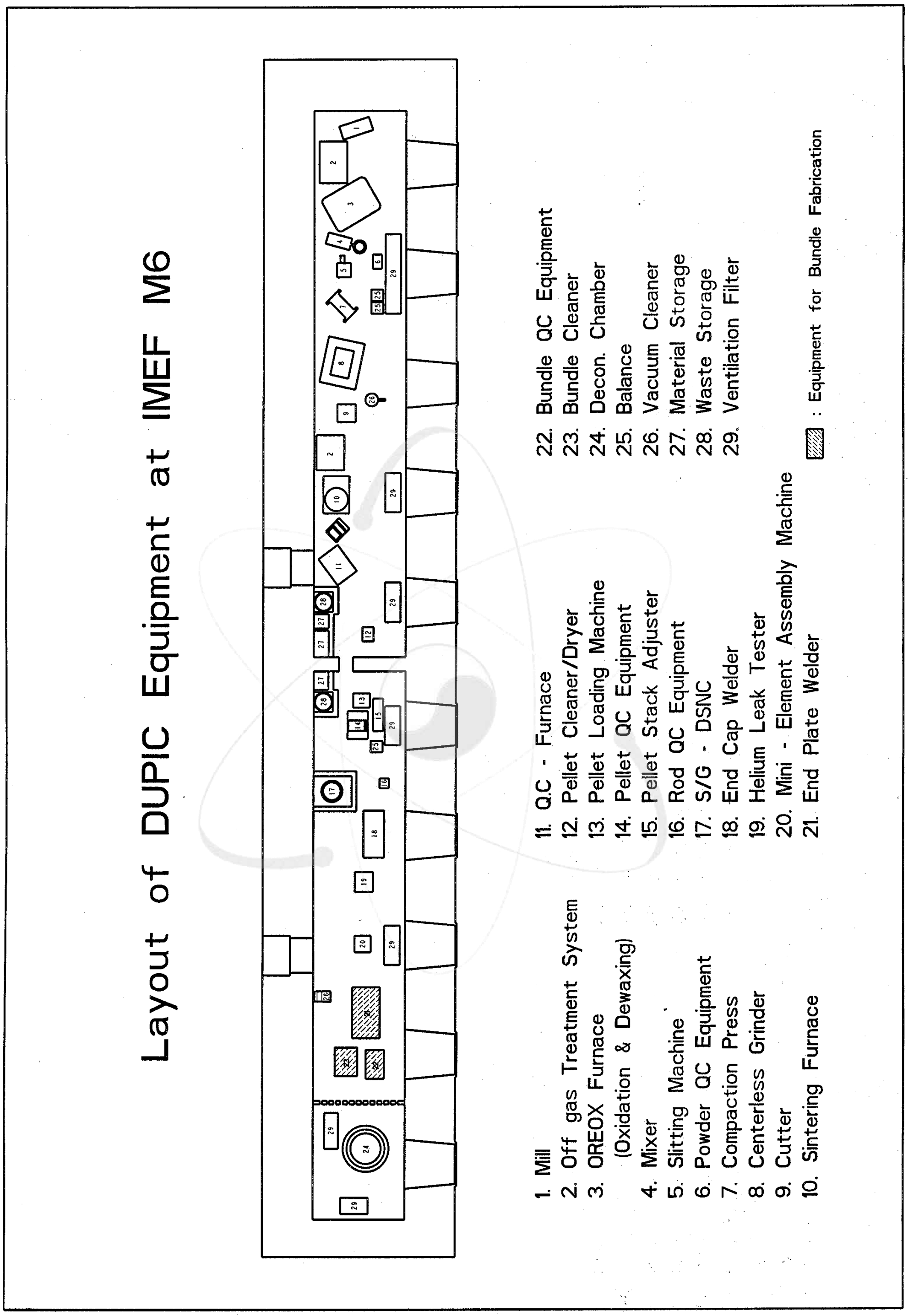


Fig. I-3. The Overall Equipment Layout for DUPIC Fuel Manufacturing at DFDF

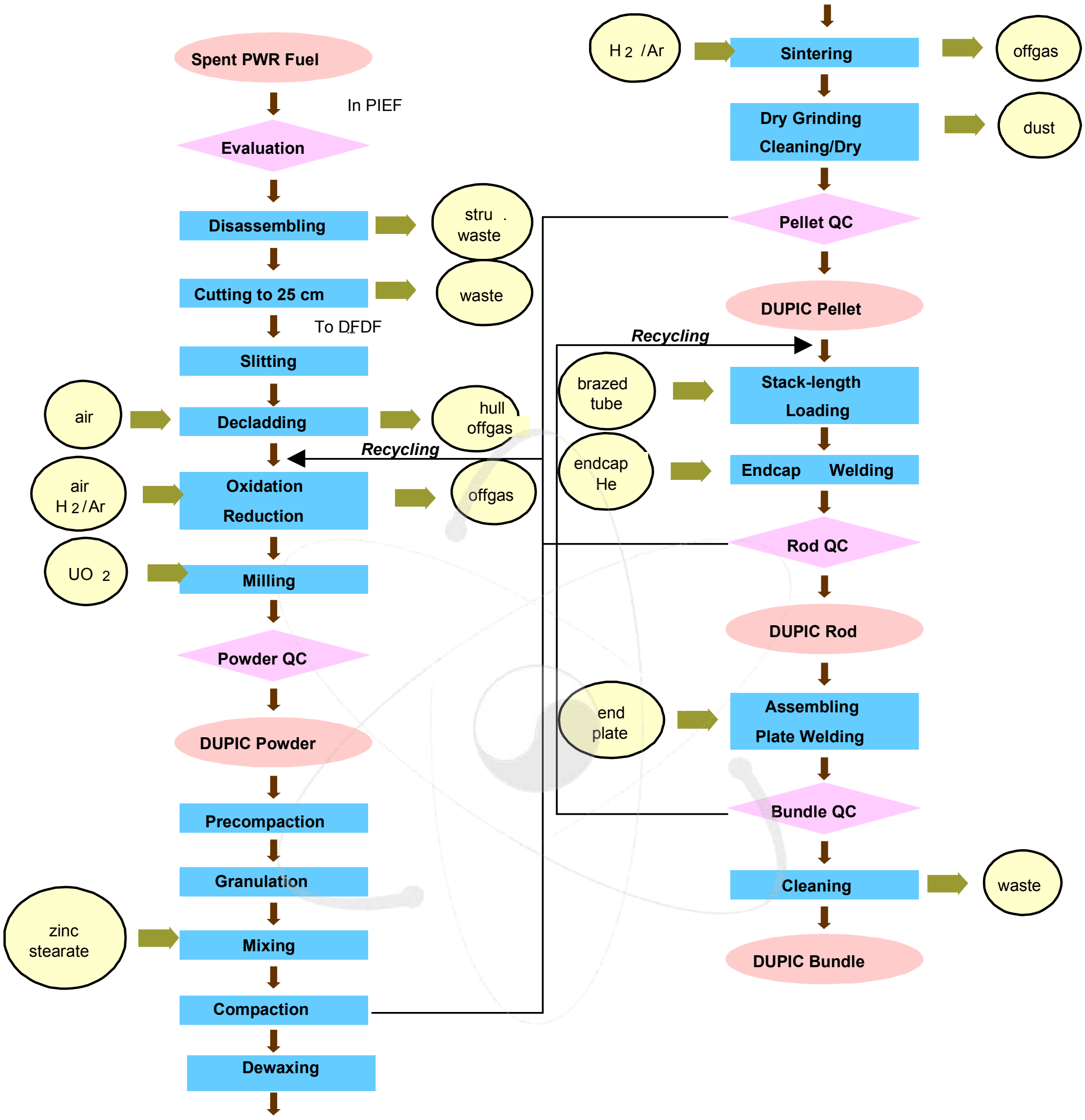


Fig. I-4. DUPIC Fuel Bundle Manufacturing Process

5.2 Powder Preparation

After slitting cladding tubes, the oxidative decladding is, if necessary, performed by low temperature oxidation at 400℃ in air. The decladded powder is further processed to convert it to resinterable powder. It is reduced at 700℃ in Ar/4%H2 and reoxidized at 450℃ in air. 3 cycles of the oxidation and reduction(OREOX) treatments are performed to obtain resinterable powder. The optimum conditions for the oxidation and reduction will be established from the experimental results. After this thermal treatment, the powder is subject to mechanical milling to improve the sinterability of the powder. During milling process, some amount of fresh enriched or depleted uranium dioxide powder can be mixed with the OREOX-treated powder to meet LER(linear element rating) requirement for the irradiation test, if it is required.

5.3 Pelletizing

The powder is precompacted, crushed and granulated to improve its flowability. The granulated powder is mixed with zinc-stearate which acts as lubricant, and is compacted to green pellets. The green pellets are sintered at about 1700℃ and above in reducing atmosphere(Ar/4%H2) to produce the sintered pellets. The sintered pellets are ground by dry method to adjust the diameter.

5.4 Element Manufacturing

The ground pellets are loaded into the cladding tubes, which have been prepared outside hot cell and have been one end welded. The loaded cladding tubes are filled with He gas and welded. Mini-elements for the pellet irradiation will also be fabricated in similar way.

5.5 Bundle Manufacturing

The elements are assembled into bundle type using the specially designed fixture. The bundle is manufactured by welding between the end caps and end plates. The irradiation rig for pellet irradiation tests will be fabricated by mechanical assembling. The details of 7-element bundle assembling for the irradiation test at HANARO will be determined later.

5.6 Quality Control Program

The following materials which are produced in the manufacturing process have to be tested for quality control purpose according to relevant testing procedures, and the results will be recorded and reported.

- o Spent PWR fuel
- o OREOX powder
- o DUPIC powder
- o Green pellets
- o Sintered pellets
- o Fuel elements, including mini-elements
- o Fuel bundles, including irradiation rig

The major tests for each material are as follows:

5.6.1 Spent PWR fuel

- Chemical analysis

5.6.2 OREOX powder

- Chemical analysis

5.6.3. DUPIC powder

- Chemical analysis
- Sinterability
- Flowability

- O/M Ratio
- Particle size
- Powder morphology
- Apparent density of powder

5.6.4. Green pellets

- Dimension
- Density
- Visual inspection

5.6.5. Sintered pellets

- Chemical analysis
- O/M Ratio
- Microstructure
- Homogeneity
- Dimension
- Immersion density
- Stack length
- Visual inspection
- Surface roughness

5.6.6. Fuel element including mini-elements

- Defects of end cap weld
- He leak test of fuel element
- Dimension of fuel element
- Visual inspection
- Contamination of element surface

5.6.7. Fuel bundles including irradiation rigs

- Dimension
- Torque test on the welded point
- Surface defects
- Contamination of fuel bundle

Most of tests are performed at DFDF by using QC equipments of DFDF. However, chemical analysis of spent fuel, OREOX powder and sintered pellet is carried out at chemical analysis laboratory of PIEF or may be conducted at the laboratory in foreign country to meet the requirements of irradiation test at foreign research reactors. Powder morphology is observed by SEM at IMEF and/or PIEF. Microstructure of sintered pellet is investigated by microscope at IMEF and/or PIEF. Homogeneity of sintered pellet is measured by autoradiography at PIEF 9409 cell with the same specimen used in the observation of microstructure.

Besides above-mentioned quality control items of each material, the supplementary inspection and tests during DUPIC fuel bundle manufacturing may be performed, if necessary. Table I-4 shows the QC program and the nuclear material quantity to be used for QC throughout the DUPIC fuel fabrication in DFDF.

Table I-4 Amount of Materials to Be Used for QC

Year	Test Material	Test Item	Test Place	Amount of Sample
1999.4~ 2000.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	-
	OREOX powder	Chemical Analysis	Chemical Lab.	-
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at IMEF	-
	Sintered Pellets	Chemical Analysis	Chemical Lab.	1 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	-
			Subtotal	1 g
2000.4~ 2001.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	1 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at PIEF	10 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	2 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	30 g
			Subtotal	46 g
2001.4~ 2002.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	2 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at PIEF	15 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	10 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	90 g
			Subtotal	120 g
Total(1999.4 ~ 2002.3)				167 g

(continued)

Year	Test Material	Test Item	Test Place	Amount of Sample
2002.4~ 2003.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	2 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at IMEF	15 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	10 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	90 g
			Subtotal	120 g
2003.4~ 2004.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	2 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at IMEF	15 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	10 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	90 g
			Subtotal	120 g
2004.4~ 2005.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	2 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at IMEF	15 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	5 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	75 g
			Subtotal	100 g
2005.4~ 2006.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	2 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at IMEF	10 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	5 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	60 g
			Subtotal	80 g
2006.4~ 2007.3	Spent PWR fuel	Chemical Analysis	Chemical Lab.	2 g
	OREOX powder	Chemical Analysis	Chemical Lab.	3 g
	DUPIC powder	Chemical Analysis	Chemical Lab.	-
		Powder Morphology	EPMA room at IMEF	10 g
	Sintered Pellets	Chemical Analysis	Chemical Lab.	5 g
		Microstructure	DFDF→IMEF M3→ IMEF M7→(PIEF 9409)	60 g
			Subtotal	80 g
Total (2002.4 ~ 2007.3)				500 g
Grand Total (1999.4 ~ 2007.3)				667 g

6. Radioactive Waste Management

6.1 Radwaste Arisings

Radwaste arisings are estimated based on the following assumptions.

- Spent PWR fuel elements of 35,000 MWD/MTU burnup having cooling period of 10 years are assumed to be used in calculating gaseous waste arising in Table I-5.
- All the gaseous phases except hydrogen are released during slitting, oxidation, reduction, sintering processes, etc.
- About 50% of hydrogen arising is released during slitting, oxidation, reduction, sintering processes, etc.
- Decontamination wastes of DFDF cell and equipment which can be generated after manufacturing DUPIC fuel bundles are not counted in this report.
- Radwaste arising is calculated based on using 200kgU of spent PWR fuel.

6.1.1 Gaseous waste arising

Table I-5 Gaseous Waste Arising

	arising, g	arising, ml	activity, Ci
H_2	5.88×10^{-3}	66	97.76
He	0.203	1.137	
Br_2	4.62	647	
Kr	77.2	20,636	1,046.2
I_2	48.28	4,258	6.54×10^{-3}
Xe	1,120.4	191,142	
RuO_2	601.6	101,322	108.74
Cs_2O	538.2	42,781	18,568
CO_2	70.79	36,039	0.114
Particulate (base: U_3O_8)	590		154.4
Total	3,051	398,028	19,975

+ Amount of particulate waste is assumed as 0.25% of treated UO_2 weight.

6.1.2 Solid waste arising

Table I-6 Solid Waste Arising

Wastes	Quantities in storage	Remarks
Cladding hulls	~ 46 kg	About 2 kg-U may be included.
Dirty scrap		-To be generated during grinding and QC processes, etc. -About 4 kg-U may be included.
Used process filters, adsorbents and in-cell filters		About 1.0 kg-U may be included.

6.2 Waste Management

6.2.1 Gaseous waste

- The unique systems for trapping off-gas from each fuel manufacturing process step will be applied.

6.2.2 Cladding hulls

- Cladding hulls are to be compressed and/or packed in a container, and then stored in monolith at RWTF prior to be disposed of after consultation with IAEA.

6.2.3 Dirty scrap

- Dirty scrap is to be solidified and/or packed in a container, and then stored in monolith at RWTF until final disposal after consultation with IAEA.

6.2.4 Used process filters, adsorbents and in-cell filters

- Used process filters, adsorbents and in-cell filters are to be compressed and/or packed and then stored in the low and medium level radwaste storage building or monolith of RWTF until final disposal after consultation with IAEA.

6.2.5 Miscellaneous waste

- Miscellaneous wastes generated during decontamination of hot cell and equipment are to be packed in a container and stored in the low and medium level radwaste storage building or monolith of RWTF.

6.2.6 Liquid wastes

- Low or medium level waste such as cleaning solution of the test pellets, decontamination solution of equipment and solution from chemical analysis is to be stored in storage tank for medium level liquid waste storage at IMEF and transferred to RWTF for further treatment.

7. DUPIC Fuel Fabrication & Irradiation Test Plan

7.1 DUPIC Fuel Fabrication Plan

Table I-7(a) Plan for DUPIC Fuel Fabrication(1998.9～2002.3)

Year	R & D Activities	Amount of Spent PWR Fuel To Be Used. (kg)
1998.9 ～ 2000.9	○ Powder/Pellet characterization study at PIEF	1
1999.4 ～ 2000.3	○ Mini-element fabrication for irradiation test - 5 pellets(10g×5)/mini-element × 3 mini-elements/rig ○ DUPIC pellet process qualification test - 0.5 kg/batch × 4 batch test - tests of decladding, OREOX, powder treatment, pressing, sintering and grinding process	0.15 2
	Sub-total	2.15
2000.4 ～ 2001.3	○ Mini-element fabrication for irradiation test - 5 pellets(10g×5)/mini-element × 3 mini-elements/rig × 10 rigs (back-up) ○ DUPIC element process qualification test - 0.5 kg/batch × 5 batch test - tests of pelletizing, endcap welding and QC process ○ DUPIC fuel element fabrication - 2 elements (0.7kg×2 elements) × 2 sets(back-up)	1.5 2.5 2.8
	Sub-total	6.8
2001.4 ～ 2002.3	○ Mini-element fabrication for irradiation test - 5 pellets(10g×5)/mini-element × 3 mini-elements/rig × 10 rigs (back-up) ○ DUPIC element process qualification test - 0.5 kg/batch × 5 batch test ○ DUPIC fuel element fabrication for demonstration - 3 elements (0.7kg×3 elements) × 3 sets (back-up) ○ DUPIC fuel element fabrication for demountable bundle - 18 elements (0.7kg×18 elements) × 2 sets (back-up)	1.5 2.5 6.3 25.2
	Sub-total	35.5
Total		45.45

Table I-7(b) Plan for DUPIC fuel fabrication (2002.4～2007.3)

Year	R & D Activities	Amount of Spent PWR Fuel To Be Used(kg)
2002.4 ~ 2003.3	○ Experiment for manufacturing DUPIC element & bundle - 0.5 kg/batch × 6 batch ○ Mini-element fabrication for irradiation test - 5 pellets(10g×5)/mini-element × 3 mini-elements/rig × 4 rigs (back-up) ○ DUPIC bundle process qualification test - 0.5 kg/batch × 10 batch test ○ DUPIC fuel element & bundle for irradiation test - 1 bundle (23 kg/bundle)	3 0.6 5 23
	Sub-total	31.6
2003.4 ~ 2004.3	○ Experiment for improvement of DUPIC pellet quality (homogeneity, grain size, density etc.) - 0.7 kg/element × 10 elements ○ Mini-element fabrication for irradiation test - 5 pellets(10g×5)/mini-element × 3 mini-elements/rig × 4 rigs (back-up) ○ DUPIC fuel element & bundle for irradiation test - 1 bundle (23 kg/bundle)	7 0.6 23
	Sub-total	30.6
2004.4 ~ 2005.3	○ Experiment for improvement of DUPIC fuel quality (weldability, dimension etc.) - 0.7 kg/element × 10 elements ○ Mini-element fabrication for irradiation test - 5 pellets(10g×5)/mini-element × 3 mini-elements/rig × 4 rigs (back-up) ○ DUPIC fuel element & bundle for demonstration irradiation test - 1 bundle (23 kg/bundle)	7 0.6 23
	Sub-total	30.6
2005.4 ~ 2006.3	○ 7-element rig fabrication for irradiation test - 50 large pellets(22.5g×50)/element × 7 elements ○ DUPIC fuel element & bundle for demonstration irradiation test - 1 bundle (23 kg/bundle)	7.9 23
	Sub-total	30.9
2006.4 ~ 2007.3	○ 7-element rig fabrication for irradiation test - 50 large pellets(22.5g×50)/element × 7 elements ○ DUPIC fuel element & bundle for demonstration irradiation test - 1 bundle (23 kg/bundle)	7.9 23
	Sub-total	30.9
Total		154.6

7.2 DUPIC Fuel Irradiation Test Plan

DUPIC fuel irradiation test plan from April 2002 to March 2007, which is currently planed, is as follows.

- April, 2002 ~ March, 2007 : irradiation of DUPIC pellets and elements at HANARO in KAERI
  - 3 times irradiation tests for one year each with 3 mini-elements in irradiation Rig, and each mini-element contains 5 DUPIC pellets of about 10 grams each, which can be fabricated with either of DUPIC, SIMFUEL, UO_2 or NU-Dy powder.
  - 2 times irradiation tests for one year each with 7-elements in irradiation Rig, and each element contains 50 DUPIC pellets of about 20 grams each.
- April, 2002 ~ March, 2006 : irradiation of demountable DUPIC bundle at NRU reactor in AECL
  - 1 time irradiation up to 20,000 MWd/tU
  - Maximum 18 DUPIC fuel elements and natural uranium elements in a demountable bundle.
  - Each element contains about 30 DUPIC pellets or natural uranium of about 20 grams each.
- April, 2002 ~ March, 2006 : irradiation of DUPIC bundle at NRU in AECL
  - 1 time irradiation up to 20,000 MWd/tU of a bundle with 37 or 43 elements
  - Each element contains about 30 DUPIC pellets of about 20 grams each.
- April, 2003 ~ March, 2006 : irradiation of DUPIC element or bundle at foreign research reactor
  - 2 times irradiation for transient and high power tests at foreign research reactor
  - The available research reactor will be determined later.
- April, 2002 ~ March, 2007 : PIE of irradiated DUPIC fuels

The overall DUPIC fuel irradiation test and measuring plan of the fuel property with instrument during irradiation at KAERI are summarized as shown in Table I-8.

Table I-8. Overall DUPIC Fuel Irradiation Test Plan at KAERI

Fiscal year	Irradiation Quantity	Irradiation Test	Remarks
2002 (April 2002 ~March 2003)	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	1 time irradiation for a year with 3 mini-elements in a Rig at HANARO	- SPND(Self-Power Neutron Detector) - Pellet centerline temperature measurement - DUPIC and/or NU-Dy poison pellets - Target burnup:12,000 MWd/tU - Some of irradiation pellets might be made either of SIMFUEL, UO2 or NU-Dy
2003 (April 2003 ~March 2004)	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	1 time irradiation for a year with 3 mini-elements in a Rig at HANARO	- SPND - Pellet centerline temperature measurement - Cladding inner pressure - DUPIC and/or NU-Dy poison pellets - Target burnup: 12,000 MWd/tU - Some of irradiation pellets might be made either of SIMFUEL, UO2 or NU-Dy
2004 (April 2004 ~March 2005)	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	1 time irradiation for a year with 3 mini-elements in a Rig at HANARO	- SPND - Pellet centerline temperature measurement - DUPIC and/or NU-Dy poison pellets - Target burnup: 20,000 MWd/tU - Some of irradiation pellets might be made either of SIMFUEL, UO2 or NU-Dy
2005 (April 2005 ~March 2006)	7 elements × 50 large pellets/ element× 22.5 g/pellet = 7.9 kg	1 time irradiation up to 10,000 MWd/tU at HANARO	- 10,000 MWd/tU burnup - Using Fuel Test Loop Facility to be developed at HANARO
2006 (April 2006 ~March 2007)	7 elements × 50 large pellets/ element× 22.5 g/pellet = 7.9 kg	1 time irradiation up to 20,000 MWd/tU at HANARO	- 20,000 MWd/tU burnup - Using Fuel Test Loop Facility to be developed at HANARO

Chapter II

Post Irradiation Examination Program of DUPIC Fuel

Chapter II. Post Irradiation Examination Program

1. Backgrounds

DUPIC fuel specifications and quantities described below are maximum estimated amounts of the post-irradiation examination for DUPIC fuel development during terms from April 2002 to March 2007. And they are estimated by considering the irradiated fuels and irradiation assembly rigs irradiated in HANARO reactor. The proposed post-irradiation examination plan of irradiated DUPIC fuel is described below. This chapter described the works to be performed for the irradiated DUPIC fuel study at KAERI.

2. Transportation

The DUPIC fuel irradiated in rig or FTL at the HANARO will be transported from the HANARO pool to the IMEF by using the same cask as that of HANARO drive fuel. DUPIC fuel means bundles, demountable bundles, elements and mini-elements. Samples for chemical analysis, and/or metallography, optical/SEM observation, out-of-pile annealing experiments will be transported from IMEF to PIEF by using the padirac.

3. Fuel specifications

Table II-1 Fuel Specifications

	Rig	Bundle	Demountable bundle	Mini-element
Weight	3.05 kg	23 kg	25 kg	50 g
Length	980 mm	500 mm	500 mm	200 mm
Stack length	960 mm	480 mm	480 mm	50 mm
Diameter	60 mm	13.5 mm	13.5 mm	12 mm
Composition	Aluminium	spent fuel	spent fuel	spent fuel
Remarks	For 3 mini-element	CANFLEX bundle (43 elements), or CANDU-6 bundle (37 elements)	CANFLEX bundle (43 elements), or CANDU-6 bundle (37 elements)	element assembly

4. Post-irradiation examination

4.1 Scope of Work

- Nondestructive examination in hot cells
  - Dismantling of DUPIC fuel
  - Visual examination and photography
  - Measurement of dimensional change
  - Eddy current test
  - X-ray radiography
  - Gamma-scanning

- Destructive examination in hot cells
  - Fission gas analysis
  - Out-of-pile annealing experiments
  - Cutting/Grinding/Polishing/Etching
  - Microstructure analysis (optical/SEM/EPMA)

- Density measurement
- Hardness
- Chemical analysis
- Physical properties measurement

4.2 Methods

- Cask receiving and unloading
  - Checking the dose rate close to the cask transport trailer.
  - Transfer the cask to the decontamination room (IMEF), where the surface washing is performed.
  - Transfer the cask to the unloading pool and remove of the cask lid by over-head crane.
  - Unloading of the fuel from the cask by handling tool

- Fuel transfer from pool to hot cell
  - The fuels are transferred to the M1 Cell of the IMEF through the channel connected with the pool by the use of bucket elevator.

- Fuel dismantling
  - Transfer to a dismantling machine by which the end plate of the fuel assembly is removed

- Nondestructive test
  - Place the fuel on the rod examination multi-bench which is vertically positioned in the M1 Cell
  - Perform visual inspection and photography, profilometry, eddy current test, X-ray radiography, and gamma-scanning

- Fission gas collection and analysis
  - Fuel element puncturing is performed by a puncturing device installed in 9404 Cell of PIEF.
  - Fission gas collection is made by a gas collection system installed outside the cell, which is connected with a puncturing device installed in 9404 Cell.
  - Fission gas collected is sent to the Chemical Lab. of the PIEF for composition analysis.
  - For the study of local gas distribution inside of the irradiated pellets, out-of-pile annealing experiments of irradiated pellets would be performed at PIEF hot cell. A slice of the irradiated pellet would be transported to PIEF for this experiments.

- Fuel element cutting
  - Cutting of the fuel element is carried out in the M2 Cell.
  - For the PIE of the bundles, elements are selected from a bundle and usually samples of 5cm in length are taken from each element for destructive examination. The fuel segments which are not examined are put in a container and then are stored in the hot cell and/or pool for temporary storage.
  - For the PIE of the elements and mini-elements, each element is cut to prepare destructive examination.

- Metallographic sample preparation
  - Sectioning, mounting, grinding, polishing and chemical etching are performed in the M3 Cell, and/or at PIEF.

- Metallography
  - Microstructural observation (optical/SEM) is performed in the M7 Lead Cell, and/or at PIEF.

- EPMA sample preparation
  - Sample preparation for EPMA (Electron Probe Micro Analyzer) is conducted in the M3 Cell.
  - EPMA examination is carried out at the EPMA room of IMEF.

- TEM sample preparation
  - TEM (Transmission Electron Microscope) sample preparation is performed in the M4 Cell
  - Replica for TEM observation is prepared in the M4 Cell.

- Density measurement
  - The density of the fuel samples is measured by using a precision balance installed in the M7 Cell.

- Hardness measurement
  - The hardness of the fuel samples is measured by using a microhardness tester installed in the M7 Cell.

- Burnup measurement
  - The burnup is determined by the measurement of ^{148}Nd separated chemically from the fuel sample by means of mass spectrometer.
  - Burnup measurement is performed in the Chemical Lab. of the PIEF.
  - The samples to be analyzed will be transported by a small padirac cask from the M4 Cell of the IMEF to the Chemical Lab. of the PIEF.

- Physical property measurement
  - Thermal expansion, thermal conductivity and creep tests on the fuel sample are performed in the M5a Cell

5. Storage of Unexamined Fuel

The unexamined fuel elements and element-cuts are put into a stainless steel container and then sent to the hot cell and/or pool for temporary storage.

6. Radioactive Waste Treatment

Radioactive wastes are categorized into fuel specimen, solid waste and liquid waste, for which their treatment methods are described in the followings

- Fuel specimen
  - The fuel specimens examined are put in a container and then are stored in the hot cell and/or pool for temporary storage.

- Solid waste treatment
  - High level active wastes including the dismantled rig are put in 50 ℓ stainless container which is water-tight, while low level active wastes are put in vinyl bags and in 200 ℓ drum.
  - These are transferred to the RWTF for temporary storage.

- Liquid waste treatment
  - The liquid wastes come mainly from the pool water treatment unit, and are transferred to the low level active tanks located in the RIPF(Radio-Isotope

Production Facility) and then to the RWTF.

- A small quantity of liquid wastes produced during fuel cutting, grinding and polishing etc. in the hot cells is plastered in a can inside the cell and then treated like a high active solid waste.
- The low active liquid wastes which come from intervention area and other contaminated areas are sent to the low level active waste tanks located in the RIPF and then to the RWTF.

7. Maximum Quantity Estimated for PIE at KAERI

Table II-2 Maximum Quantity Estimated for PIE During April 2002~March 2007

Fiscal year	Source of PIE	Quantity	Remarks
2002 (April 2002 ~March 2003)	3 mini-elements irradiated at HANARO in May 2001	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	-Burn-up: 8,000MWd/tU
2003 (April 2003 ~March 2004)	3 mini-elements irradiated at HANARO in April 2002	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	-Burn-up: 12,000MWd/tU -Some of irradiation pellets might be made either of SIMFUEL, UO2 or NU-Dy
2004 (April 2004 ~March 2005)	3 mini-elements irradiated at HANARO in April 2003	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	-Burn-up: 12,000MWd/tU -Some of irradiation pellets might be made either of SIMFUEL, UO2 or NU-Dy
2005 (April 2005 ~March 2006)	3 mini-elements irradiated at HANARO in April 2004	3 mini-elements × 5 pellets/element × 10 g/pellet = 150 g	-Burn-up: 20,000MWd/tU -Some of irradiation pellets might be made either of SIMFUEL, UO2 or NU-Dy
2006 (April 2006 ~March 2007)	7 elements irradiated at HANARO in April 2005	7 elements × 50 large pellets/ element× 22.5 g/pellet = 7.9 kg	-Burn-up: 10,000MWd/tU
Total		8.5 kg	

* The PIE of the last irradiated bundle (7.9 Kg) in March 2006 will be postponed to the next scope of Joint Determination.

Chapter Ⅲ

Safeguards of Nuclear Materials

1. Introduction

The DUPIC safeguards activities under the '98 JD Report (KAERI/AR-510/98-rev.1) for last 2 years has been successfully carried out in terms of IAEA safeguards implementation as described in Facility Attachment No. 23(KO-Y) and in compliance with ROK/US nuclear corporation agreement. As DUPIC R&D project proceeds with its official plan together with safeguards knowledges and experiences gained so far, safeguards application to DUPIC R&D project needs to be extended to future DUPIC R&D plan.

In consideration of its characteristics of bulk handling of US origin spent fuel material, DUPIC process safeguards is differentiated from that of PIE activities. The major concerns of DUPIC safeguards in handling of approximately 200 kgU of US origin material are focused on how to manage accurate accountability in process inventory, how to maintain the knowledge of continuity in US origin material flow from one facility to another facility, and how well the DUPIC activity will be under IAEA containment and surveillance system, so as to give assurance of non-diversion of US origin material.

As complementary procedures, in case those three concerns not being properly controlled, IAEA inspectors is required to be in presence for on-the-spot verification or to have additional monitoring system installed for the particular DUPIC process activity.

As for the independent verification of neutron measurement system as well as its measurement data integrity, IAEA's certified neutron calibration source is available for the periodical system check.

The overall DUPIC process material balance is materialized by Cm-244 measurement using neutron coincidence counting method. This concept has been theoretically proved jointly by KAERI, LANL, and IAEA. This recommended concept is to use the spontaneous fission neutron emissions from Cm-244 to indirectly quantify the plutonium contents of fuel materials at the DUPIC facility. Because there is no chemical reprocessing involved in the DUPIC fuel cycle for any particular batch of spent PWR fuels, the ratio of Cm-244 to plutonium should be constant at the input, output, and process steps of the DUPIC fuel cycle. Therefore, it would be possible to establish the associated plutonium inventory by knowing the Cm-244 to plutonium ratio and a measured value of Cm-244.

With these concepts recommended by LANL, Coincidence Neutron Counter, so-called DSNC(DUPIC Safeguards Neutron Counter), for material accounting of DUPIC process has been developed by KAERI and LANL since 1995. The DSNC was installed at DFDF at the beginning of 1999 and performance test for use of nuclear material accounting was finished successfully by IAEA in cooperation with KAERI and LANL during November to December in 1999. A report on the DSNC's performance test has been issued as LA-UR-99-6217, titled "The Calibration of the DSNC for the measurement of Cm-244 and Plutonium". Now, the DSNC is functioning as designed, and the DSNC system has been used successfully for material accounting of IAEA safeguards in DFDF.

In addition to the above material accounting system, automated nuclear material accounting system and unattended continuous monitoring system are being developed by KAERI jointly with LANL and publicized at the international institutions such as INMM, ANS, global technical conferences and IAEA safeguards symposium. Progress reports have been prepared periodically for KAERI, AECL, U.S. and IAEA Quadri-Partite Project Review Meetings.

2. Safeguards Implementation Plan

2.1 Material Flow Relevant to DUPIC

○ The utilization plan of KAERI facilities for the DUPIC project is described as in Fig. Ⅲ-1. The types of nuclear materials used are varied from the spent PWR assembly to the irradiated DUPIC bundles.

○ One half of a spent PWR fuel assembly in PIEF is cut into about 25㎝ long and transferred to IMEF in a shielded cask. The solid waste (saw dust) produced from cutting in PIEF rod is temporarily stored in PIEF and transferred to RWTF for final disposal when it is ready.

○ Several DUPIC bundles and rods, including mini-elements and element assemblies are manufactured in IMEF M6 cell(DFDF). M6(DFDF) cell is designated as an independent material balance area separating from main IMEF activity and operated with 8 inventory Key Measurement Points. DUPIC fuel fabrication process has several types of materials, which are OREOX powder and pellets, mini-elements, rods, element assemblies, bundles, associated wastes, and fresh UO2 powder for the adjustment of U-235 enrichment. In order to establish the quantities of nuclear material within this area, several safeguards R&D programs are in process(Refer to Section 4 of R&D Status).

○ Fabricated DUPIC bundles and rods, including mini-elements and element assemblies are to be transferred to HANARO and/or foreign research reactor and performed fuel irradiation test. The fissile contents of U, U-235 and Pu are estimated by NDA measurement. The irradiated DUPIC fuel bundles and rods including mini-elements and element assemblies become subjected to and are reported to Canada under bilateral obligation.

○ The part of the irradiated DUPIC fuels are transferred to IMEF and/or PIEF for PIE associated with reactor safety, fuel design and fabrication technology improvement. The unexamined DUPIC fuels return to HANARO pool for storage. Whenever the nuclear material is transferred to the different MBAs facility, operators should notify the date and fissile material quantity and provide the nuclear material control record between MBA to safeguards office. The requirements for IAEA safeguards implementation are described in detail in Appendix Ⅰ.

KAERI has prepared the required procedures for nuclear material accounting for and physical protection in compliance with the Ministry of Science and Technology Notice of Korea.

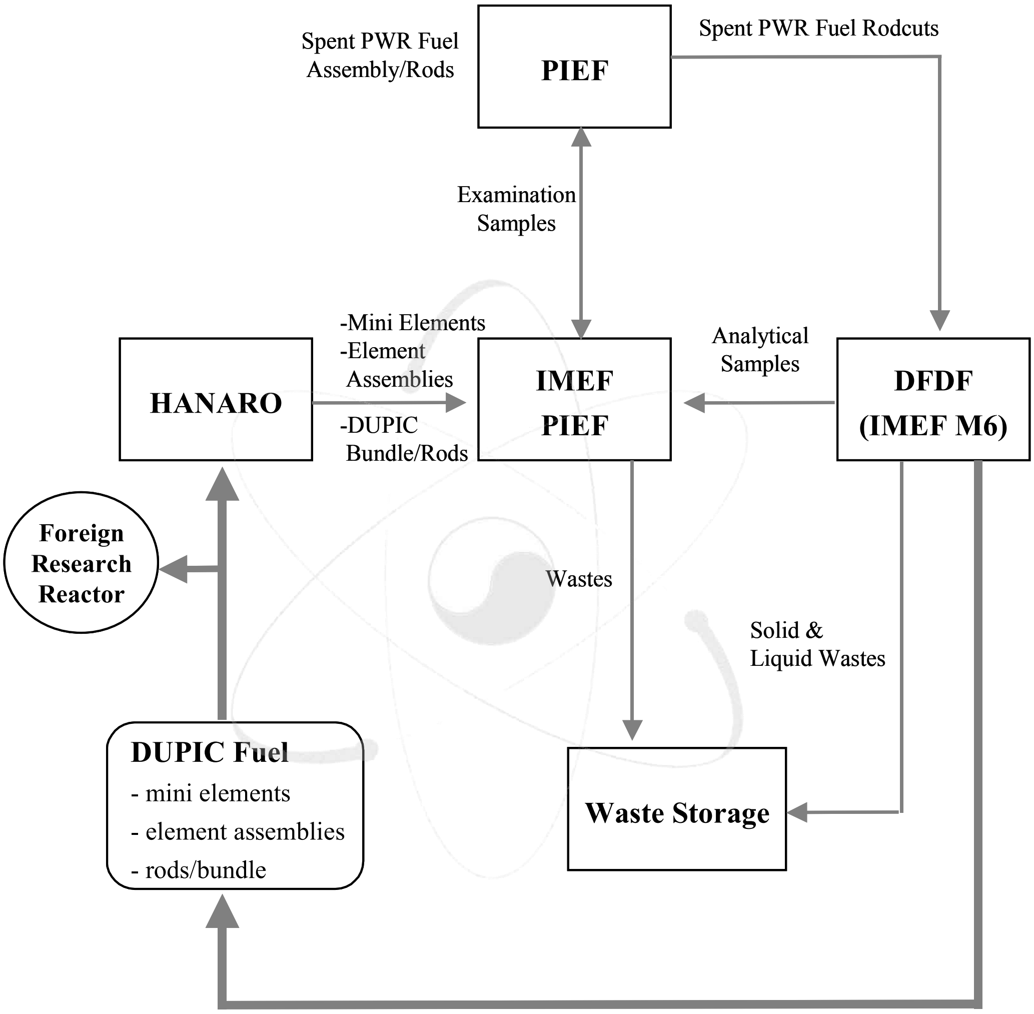
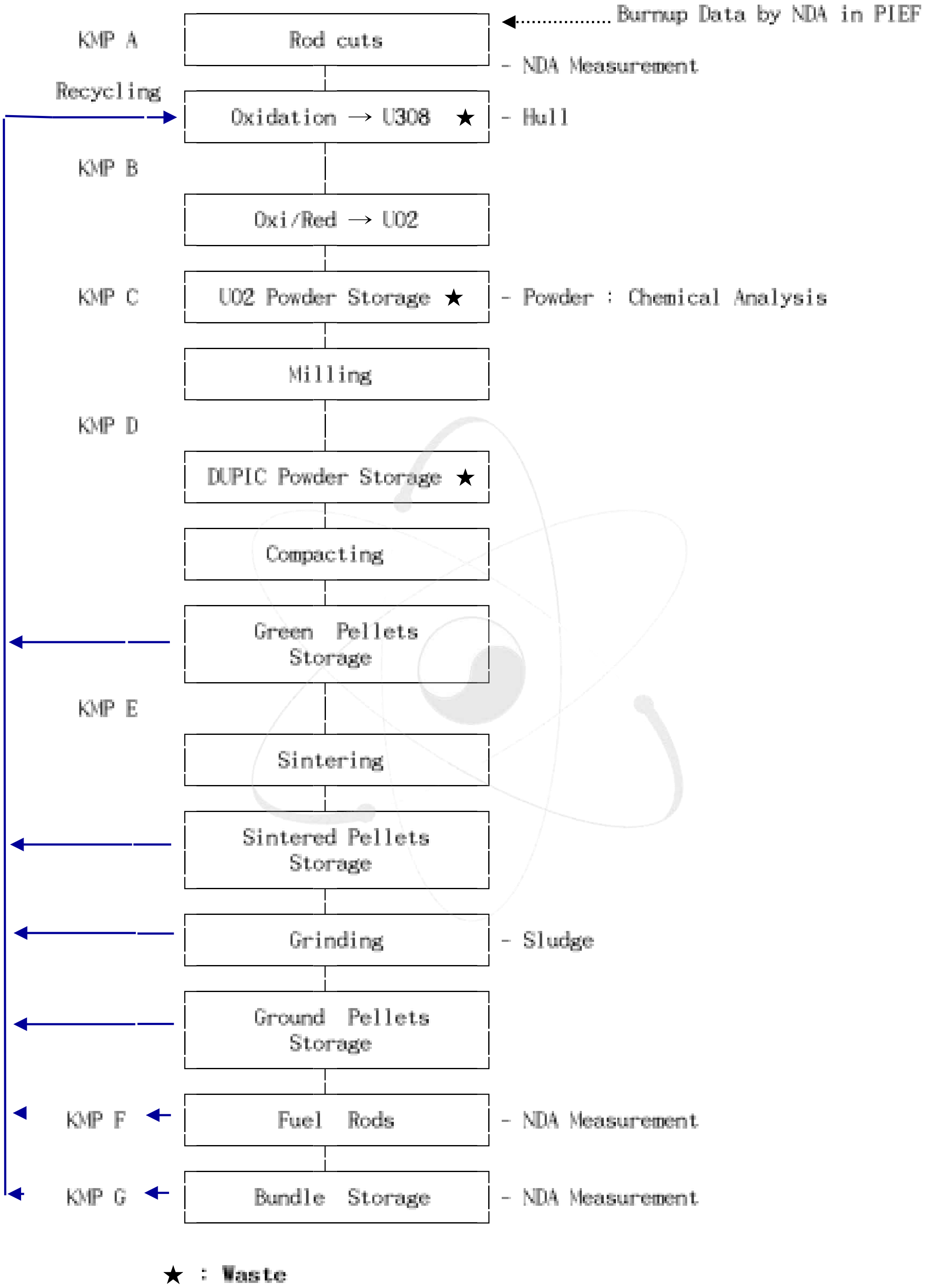


Fig. III-1. Utilization of KAERI Facilities for DUPIC

2.2 Accounting Plan for DUPIC Manufacturing Process



Waste (Hull, Dirty scrap) : NDA or Chemical Analysis

\+ Hold-up and scrap material can be produced from each process line.
++ Weighing data is obtained from each process line.

The KMPs associated with both manufacturing flows and inventories are described in Fig. Ⅲ-2. Detailed descriptions of the KMPs are as follows.

○ KMP-1 : **Receipt of nuclear material, de-exemption and accidental gain**

Feed input to MBA is spent PWR fuel rod-cuts of about 25㎝ long. The maximum capacity of the padirac cask is 56 rod-cuts, and the receiving frequencies will be determined by manufacturing plan. NDA measurement (DSNC: DUPIC Safeguards Neutron Counter) data is available as declared values of SNM content. And fresh UO2 powder for blending is also received. The fissile contents are obtained by the shipping document.

○ KMP-2 : **Shipment of nuclear material, exemption and accidental loss**

Product output is from the facility. Several bundles and rods are fabricated throughout the project until March 2007 [mini-element, element assembly, elements and bundles] and shipped to HANARO for the irradiation test. DSNC data is available as declared values of SNM content.

○ KMP-3 : **Measured discard and transfer to retained waste**

Scrap waste is to be held as retained waste until safeguards termination. The wastes used for chemical samples are also to be retained.

○ KMP-4 : **Retransfer from retained waste**

Retained waste are to be retransferred to safeguards for the purpose of safeguards termination, after consultation of IAEA.

○ KMP-A : **Spent PWR rodcuts storage**

Spent PWR fuel rod-cuts of about 25㎝ long are to be controlled as inventory of this area. DSNC data for the receipt is available for the safeguards purpose.

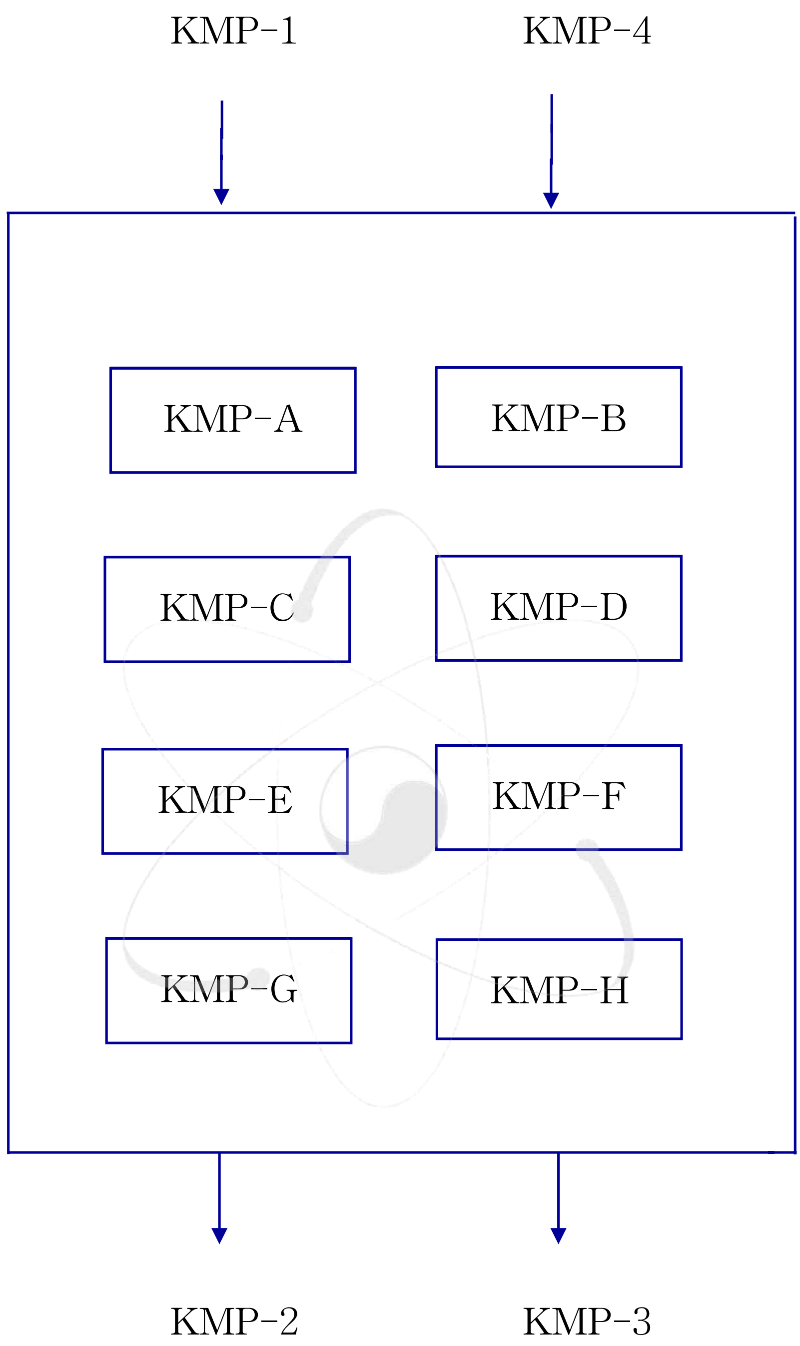


Fig.Ⅲ-2. KMPs Structure for DFDF

○ KMP-B : U_3O_8 powder storage

This powder is weighed for QC.

○ KMP-C : UO_2 powder storage

This powder is to be analyzed for QC and safeguards. The facility performs destructive analysis to estimate the SNM contents. U, U-235 and Pu factors are to be determined by chemical analysis.

○ KMP-D : DUPIC powder storage

UO_2 powder can be mixed with fresh UO2 powder, if it is necessary, to adjust the enrichment.

○ KMP-E : UO_2 pellet storage

The sound(green and sintered) pellets are weighed to get the fissile content and analyzed for QC data (microstructure, density, dimension, etc). Or, the SNM data for powder can be applied. some remaining nuclear material of used batch will be also stored in this KMP.

○ KMP-F : Fuel rod storage

The fabricated rods including mini-elements are stored and powder/pellet data can be applied to get the SNM contents.

○ KMP-G : Fuel bundle storage

The fuel bundles including element assemblies are stored in this area. The SNM data for rods can be applied.

○ KMP-H : Scrap/Waste storage

The stored clean scrap is crushed and recycled through the process. Miscellaneous waste is stored and can be measured by NDA system to declare the quantity.

2.3 Safeguards System Installed in DFDF

Fig. III-3 shows the safeguards system installed in DFDF showing the connection of accounting system including DSNC as well as C/S system of IAEAs and KAERIs instruments related to the DSNC.

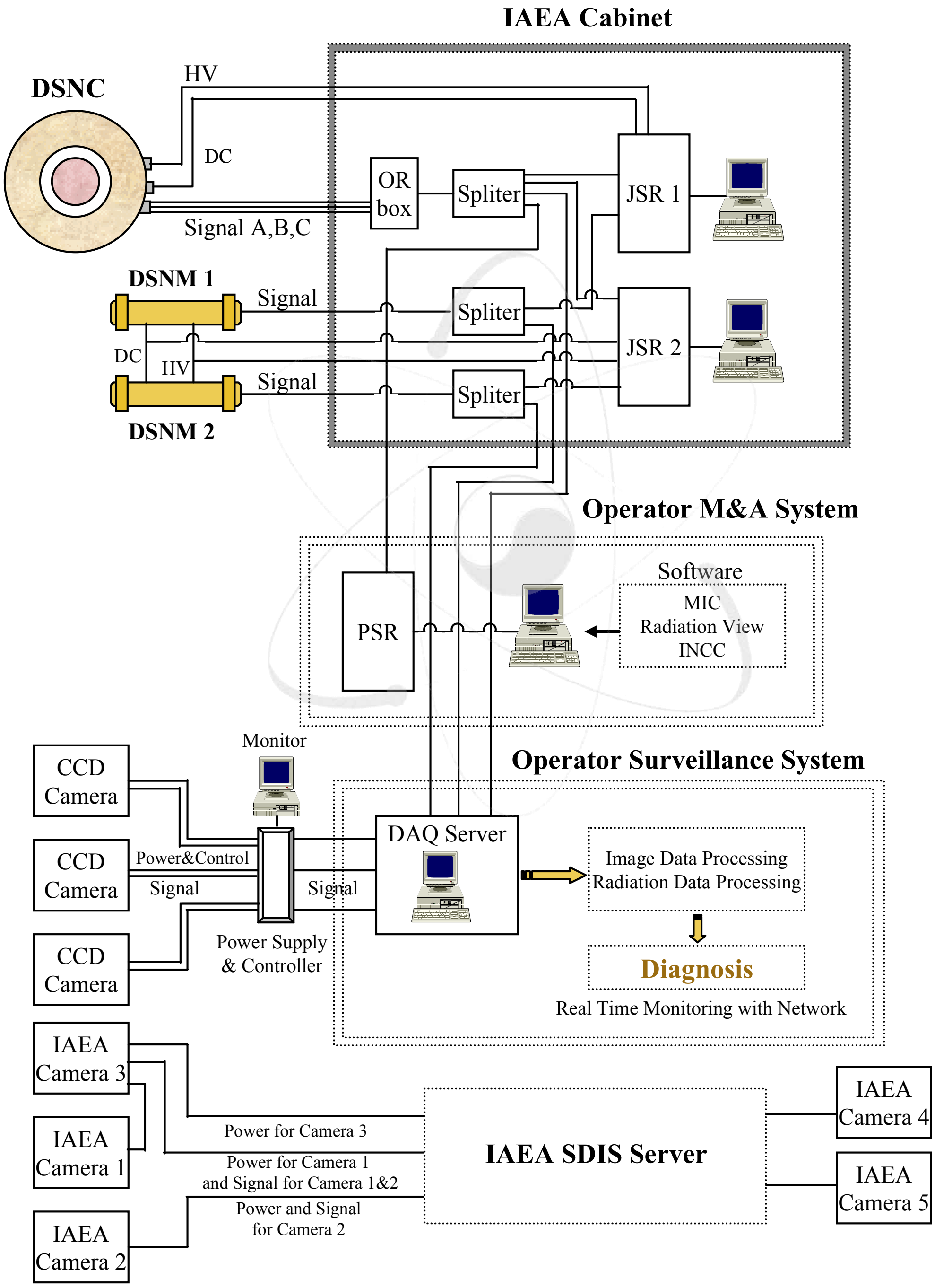


Fig. III-3. Safeguards System of DFDF

3. The Current Status of Safeguards Implementation

3.1 The Status of DIQ

The first DIQ for DFDF was submitted to IAEA through MOST in Sept. 1995 and then major items of the first DIQ for DFDF was discussed in Safeguards Review Meeting between Korea and IAEA in Oct. 1996.

The second DIQ for DFDF were submitted to IAEA through MOST in Sept. Jan. 1997 and DFDF Facility Attachment was received from IAEA in Dec. 1998. The facility attachments of DFDF is attached in the appendices as a separated document.

3.2 The Status of Safeguards Implementation

DFDF has been received safeguarding by IAEA and related domestic body since hot operation started at the beginning of the year 2000. Until now, several times interim inspections were carried out by IAEA. All measurement of nuclear material have been performed by NDA system(DSNC).

Fig. III-4 shows the nuclear material flow in DFDF as of October 20, 2000. The first shipment of DUPIC process material as one batch was introduced from PIEF on Jan, 27, 2000 after the selected PWR rod was cut in 25 cm long pieces and measured by DSNC and calculated by Origen Code to obtain the quantity of nuclear material such as U element, U-235 isotope, and Pu element total. Immediately after transfer to DFDF the whole batch was measured with the DSNC to get the Curium total, then, Curium Ratio is established for each type of nuclear material. These Curium Ratios are maintained until the specific batch operation is finished.

DUPIC mini-element for irradiation test in HANARO research reactor and samples for IMEF test were taken out from DFDF on April 8, 2000 as shown in Fig. III-4. Another three batches with different burnup were introduced in Oct. 7, 2000.

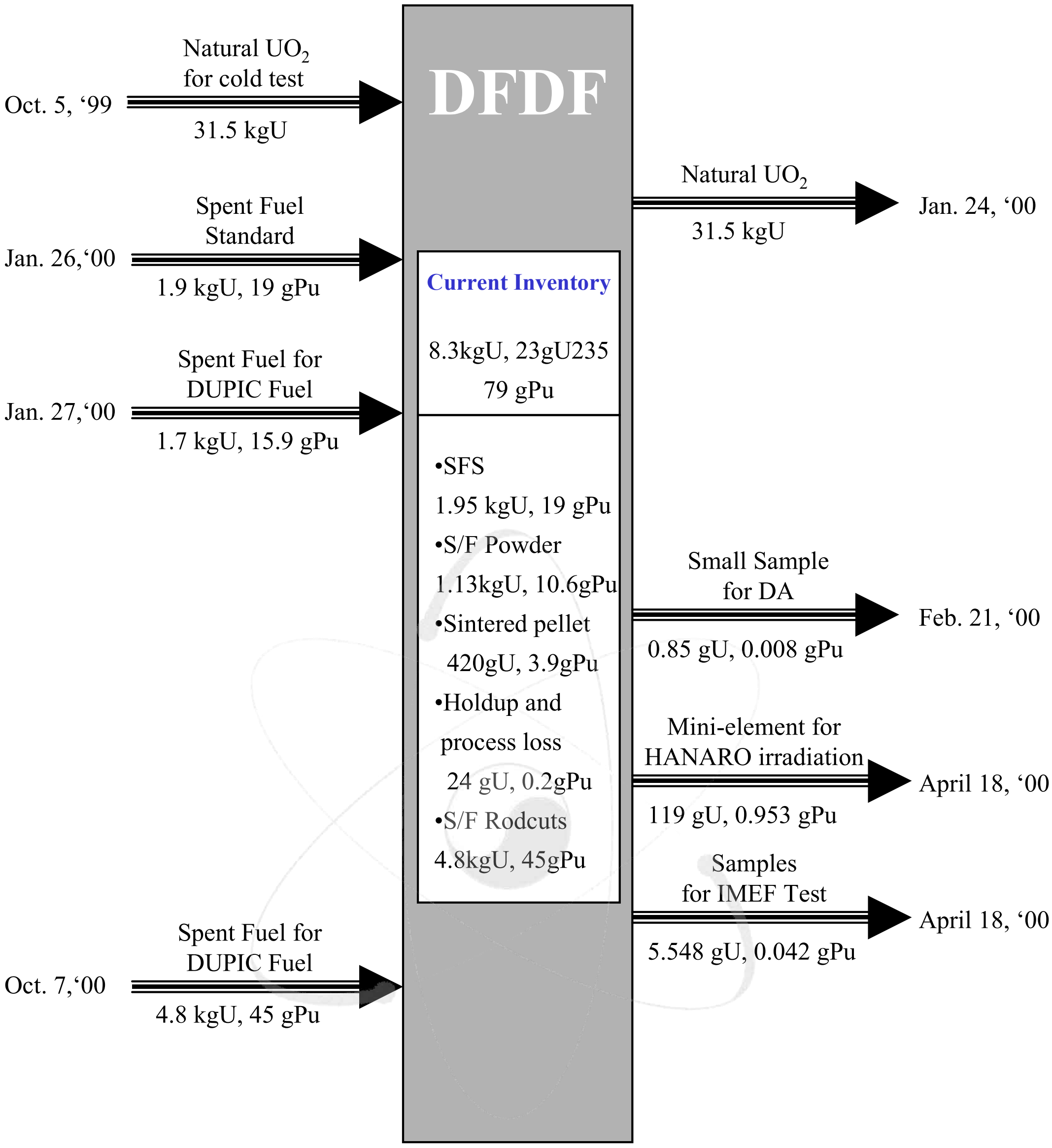


Fig. III-4. Material Flow in DFDF

4. Current Status and Future R&D

DUPIC safeguards R&D works have been carried out as one of Intermediate and Long Term Nuclear R&D Programs sponsored by Government. The major safeguards technology involved here was to design and fabricate a neutron coincidence counting system, so-called DSNC, for process accountability, and also an unattended continuous monitoring system in association with independent verification by the IAEA. This combined technology was to produce information of nuclear material content and to maintain knowledge of the continuity of nuclear material flow. In addition to hardware development, diagnosis software is being developed to assist data acquisition, data review, and data evaluation based on a neural network system.

4.1 Development of DUPIC Safeguards Neutron Counter (DSNC)

■ Hardware Description and its Installation

The DSNC, which is a well-type neutron coincidence counter, is for inferring the amount of Curium from measuring spontaneous fission neutrons at various process stages in the DUPIC fuel cycle. The DSNC was developed by jointly KAERI and LANL. The DSNC design focused on all types of DUPIC process material that are remotely measurable(CANDU type bundle, powder, rod-cut, hulls, and wastes) in a hot cell during lab scale operation. A total of 18 3He tubes with nitrogen quenching were symmetrically located in a high density polyethylene moderator and each of the 3He tubes was connected to an individual preamplifier to reduced the gamma-ray pileup problem. A preamplifier status lamp was attached to each tube in order to visually monitor its normal operation in the hot cell. Another unique feature of the DSNC, compared to other conventional coincidence counters, is that substantial shielding is added to protect the 3He tube/electronics from the intense gamma-rays of the DUPIC process material with a maximum surface dose of ~104 R/h.

The DSNC manufactured by KAERI was installed in the M6b hot-cell of DFDF for DUPIC material safeguards. Lead bricks for gamma shielding and boron lined high-density polyethylene bricks for neutron shielding were installed outside of the DSNC to reduce background signals from the processing hot materials remaining in the M6 hot-cell. Stainless steel plates were also covered shielding materials. In order to commonly use the data from the DSNC, an IAEA safeguards instrument cabinet for monitoring the DSNC data was installed near KAERIs measurement system located outside the hot cell. All related cabling work was done in the operation area of the DFDF hot-cell during from Oct. 4th to Oct. 8th, 1999.

Fig. III-5 shows the connection of cabling DSNC as well as IAEAs and KAERIs instruments related to the DSNC. In this figure, the neutron signal from DSNC was split into 3 neutron signals through a splitter that was inside the IAEA cabinet. Two signals from the splitter connected to two JSR-12s of the IAEA system, respectively. The one of them is for backup, and the remaining one was connected to the PSR-B of the KAERI system.

In addition, the IAEA neutron source, K-868 (Cf-252 activity 2.022E+04 n/s as of Dec. 19th, 1999), which was received in March 1999, was located under the VACOSS seal near the DSNC in the M6 hot-cell to determine the detector characteristics in advance of DUPIC material measurement.

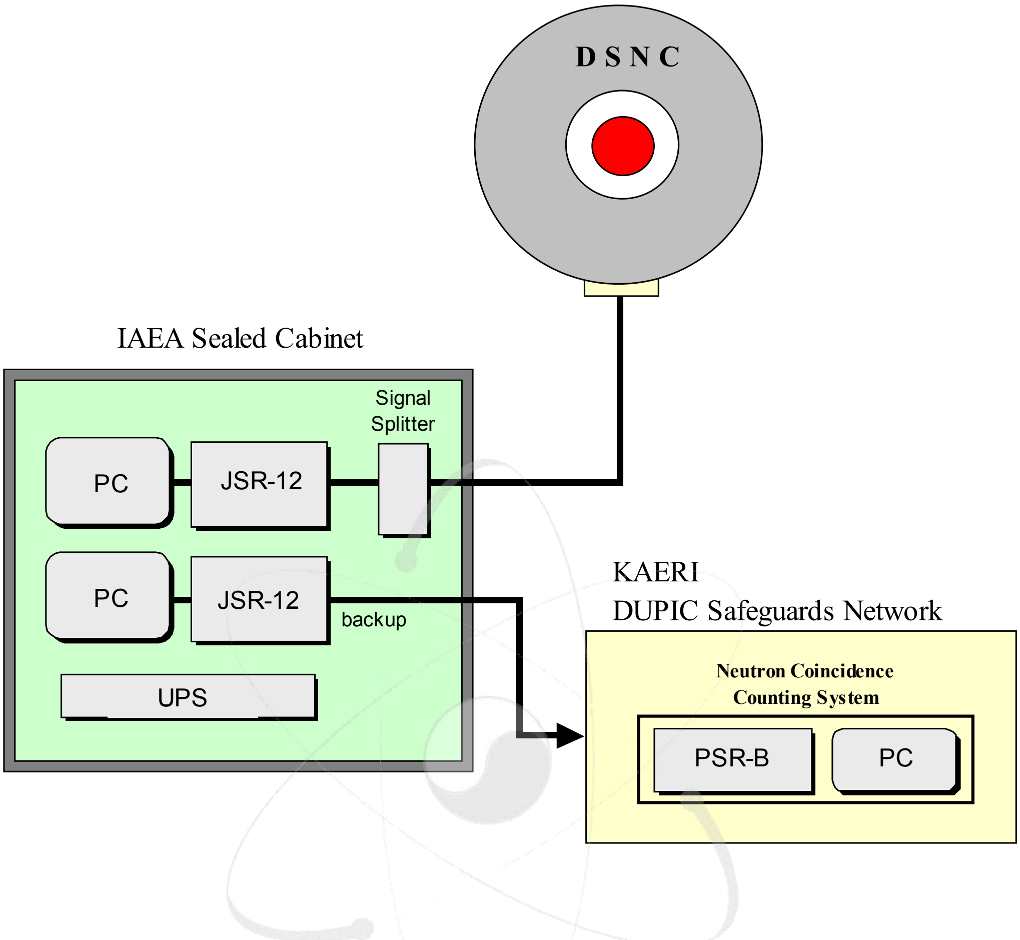


Fig. III-5. Connection of the DSNC under IAEA Safeguards

■ Software Setup

Collect and analysis programs, MIC (Multi-Instrument Collect, Version 1.652), RAD (Radiation Review, version 2.05) and INCC (IAEA Neutron Coincidence Counter, version 4.04) for a neutron multiplicity counter developed by LANL for IAEA safeguards were transferred to KAERI for DSNC operation and installed not only on a PC in the IAEA cabinet, but also on a KAERI PC during from Oct. 4th to Oct. 8th, 1999. The revised INCC program includes the calculation function of Pu, U and U-235 isotope contents from measured Cm-244 content and the Curium ratio.

■ Performance and IAEA Authentication Tests

To officially use the DSNC for DUPIC safeguards, the authentication from the IAEA is needed. For this, the DSNC was calibrated and authenticated two times at the DFDF by the IAEA with assistance from the KAERI and LANL safeguards teams on Oct. 8, 1999 and Dec. 9, 1999.

From the hot experiment of the DSNC with IAEA standard source and Spent Fuel Standards (SFS), major setup parameters including operating high voltage under a high-level gamma-ray background, detection efficiency and a calibration constant were determined. The DSNC detection efficiency was 13.48% using the IAEA neutron source(K868) and verified its constancy within ±0.02% during one month. The Cm-244 calibration constant of DSNC, which means the slope of a straight line between the curium mass and measured double, was also derived from the singles rate of K868 combined with the singles/doubles ratio from the SFSs by LANL as shown in Figure III-6. The Slope, 1.19×10^5 cps/g Cm-244, will be used to convert the measured Doubles rate to Cm-244 content.

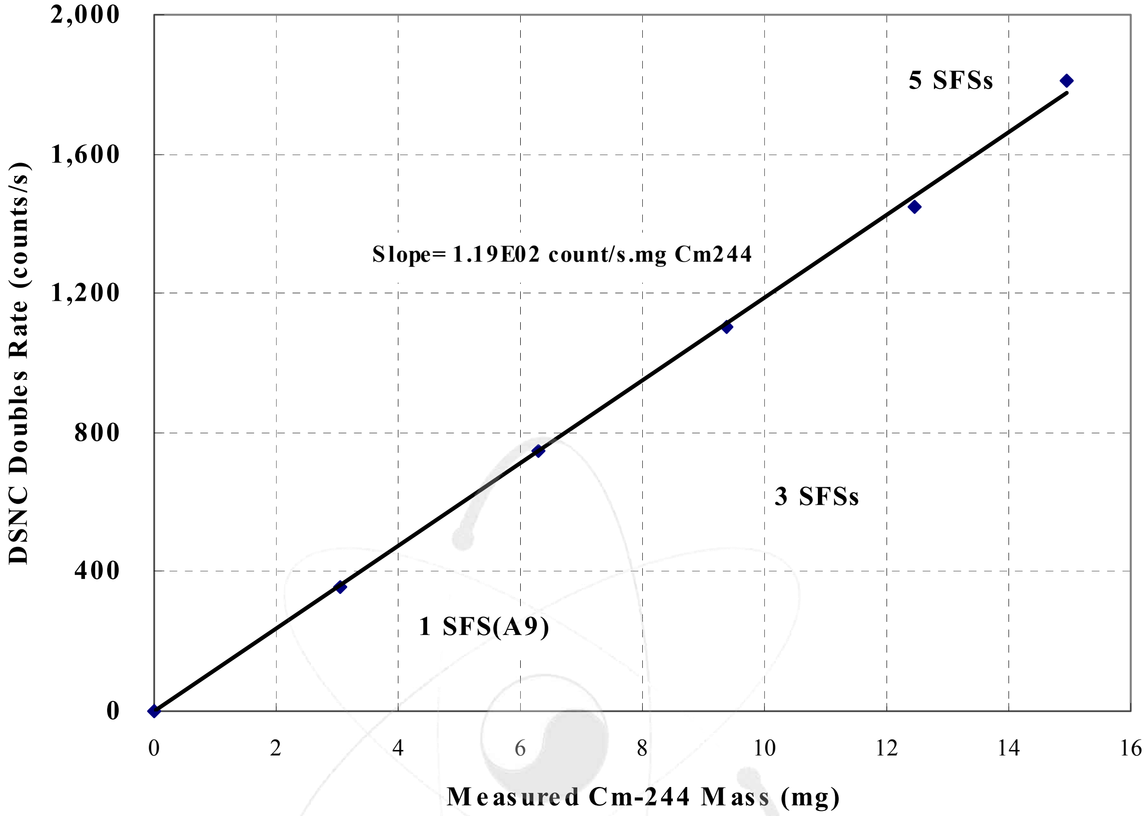


Fig. III-6. Calibration Slope for DSNC Cm-244 Measurements

4.2 Containment and Surveillance System for DUPIC

R&D efforts in the C/S area are directed to an unattended, continuous, integrated surveillance system to meet and improve the basic function of other unattended continuous surveillance systems. In system development, particular effort is made for digital analysis of events by incorporating an advanced diagnosis mechanism to selectively draw a conclusion on only the significant events throughout the monitoring period. This was done by integrating the video and radiation sensors in a common time dimension through image processing, and designing a computer interface for the neutron counting sensor. This system is able to alert spent fuel material movement to and from a typical hot cell system.

■ Hardware Description

Figure III-7 shows the Containment and Surveillance(C/S) system for the safeguards of the DUPIC Fuel Development Facility(DFDF) under development.

There are two unsealed doors and one sealed door on the DFDF. Three CCD cameras are positioned at each door in order to monitor some activities related to the SNM movement to and from the doors. Two DSNMs(DUPIC Safeguards Neutron Monitor), as shown in Figure III-8, are located near the unsealed doors to detect any transportation of nuclear material through the doors. The cameras and DSNMs installed on the outside surface of the DFDF are cabled to the surveillance server (a personal computer) located in the working area of the DFDF. The personal computer takes the image signal and the radiation signal periodically, analyzes them, and diagnoses the transportation status to report the result to the remote client as shown in Figure III-7. The signals of DSNC and DSNMs are shared with the cabinet of the IAEA.

Each neutron monitor(DSNM) installed at the DFDF includes two pairs of He-3 gas proportional counter tubes and a preamp to increase efficiency and reliability. They have no gamma shielding because they are located outside of the hot cell, and their structures are constructed with high density polyethylene, which also has the function of neutron moderation. The signals from the DSNMs are acquired with a DAQ, card which has four input channels and simply counts the number of neutrons detected.

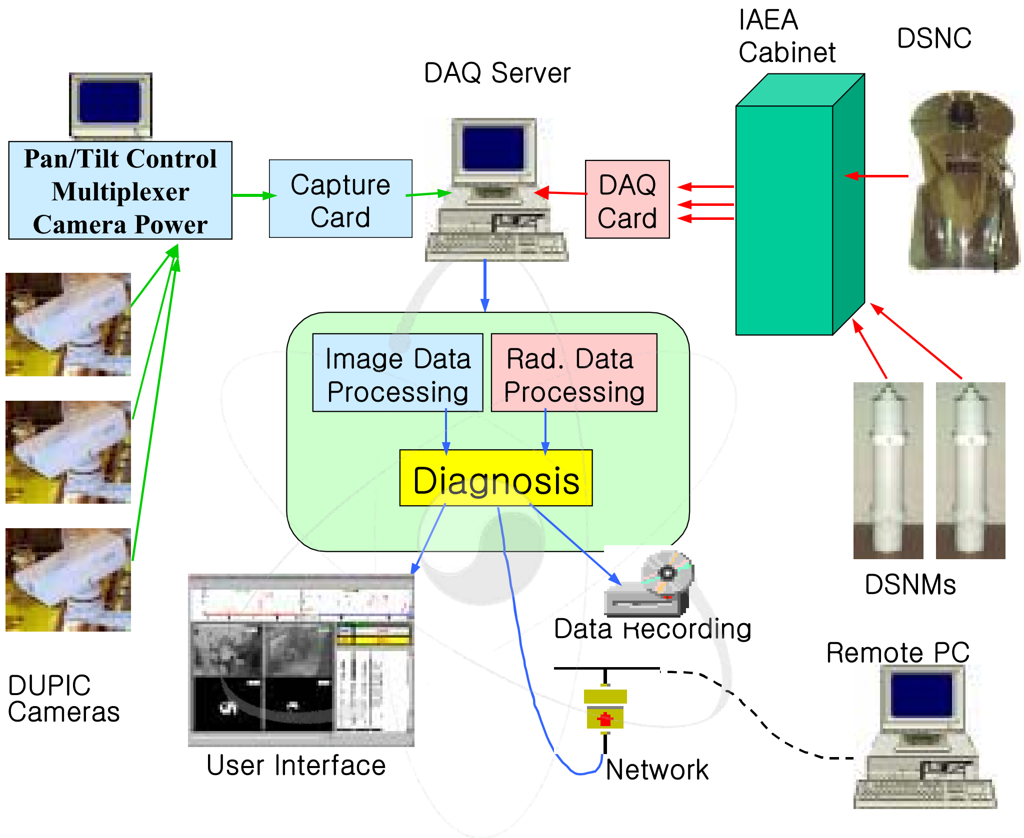


Fig. III-7. Configuration of the DUPIC C/S System

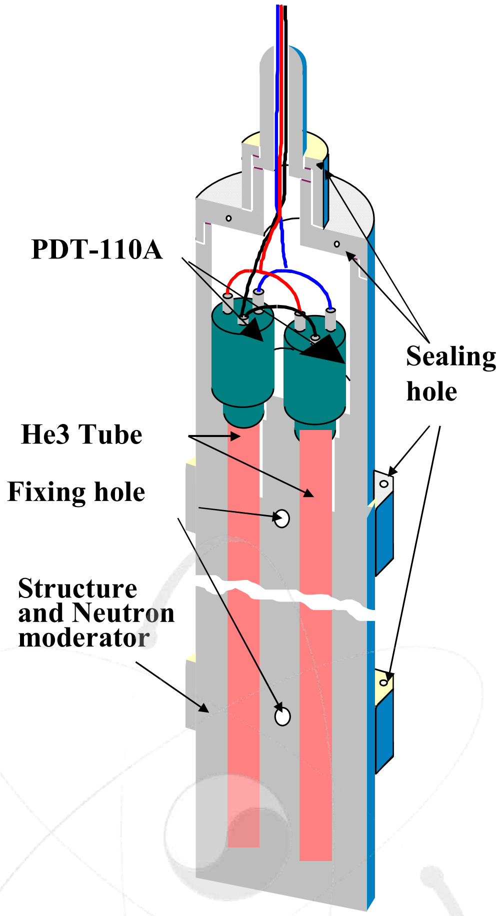


Fig. III-8. DUPIC Safeguards Neutron Monitor

■ Software Description

The data periodically acquired is processed to be fed to the transportation diagnosis routine, where the level and the variance of the radiation data and the position and the size of the objects in the image data are evaluated by the data processing software. Image processing, based on the differentiation of images and the comparison of objects size, is still very preliminary.

The transportation diagnosis program has two stages of the individual mode and the overall mode. In the individual mode, the radiation transportation status based on the radiation data and the object transportation status based on the image data are determined separately, and in the overall mode the two results are unified to make an overall diagnosis about transportation. In the individual mode, the diagnosis routines of both radiation and image determine the transportation status to one of the four cases of 'No Detection', 'Fade In', 'Rest', and 'Fade Out'. In the overall mode, a more sophisticated diagnosis based on the results of the individual mode and the raw data will be performed.

■ Preliminary Hot Test of the Surveillance System

Since the overall mode diagnosis is under development, only an example of the individual mode diagnosis is presented here as shown in Figure III-9.

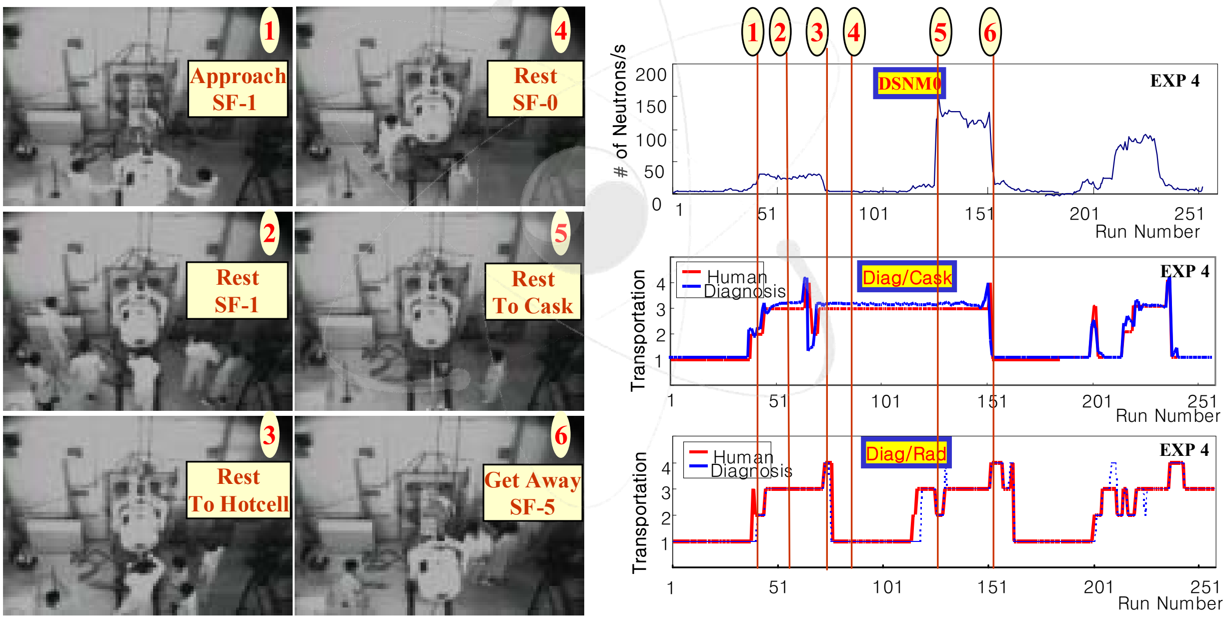


Fig. III-9. Hot Test of the Surveillance System Using a Cask with One Spent Fuel Rod-cut

In this experiment, 10cm spent fuel rod-cuts were transported into the hot cell. (1) A cask containing one spent fuel rod-cut is approaching to the hot cell door(Cask Fade In, Rad Fade In), (2) the cask with one rod-cut in it is at rest(Cask Rest, Rad Rest), (3) the one rod-cut in the cask is being brought into the hot cell(Cask Rest, Rad Fade Out), (4) the empty cask is at rest(Cask Rest, Rad Rest), (5) five rod-cuts are being brought into the cask from the hot cell(Cask Rest, Rad Fade In), (6) the cask with five rod-cuts in it is going away(Cask Fade Out, Rad Fade Out). As shown in the above graphs, the diagnosis results well describe the transportation status. The top graph shows the radiation level, the middle one is object diagnosis, and the bottom one means radiation diagnosis. The blue lines in the middle and bottom plots are the diagnosis results by software and the red plots are by a human. The axis values in the middle and bottom plots mean that 1 is no detection, 2 is fade in, 3 is rest, and 4 is fade out.

4.3 Development of Nuclear Material Accountability System

The Nuclear Material Accountability System will be developed for the accountability management of nuclear materials treated by DUPIC facility. This system is able to track the controlled nuclear materials while maintaining the material inventory in near-real time and to generate the required material accountability reports. It consists of basic nuclear material database and a series of programs for inventory management of these materials. Also it has easy and useful user interface. The facility, which treats nuclear material, manages nuclear material safely under the related provision and reports periodically to IAEA, MOST and so on. Thus the nuclear material management using database system is a prerequisite, and the performance of this system has a great effect on the operation of nuclear facility.

We will design the system in the basis of E/ZMAS (Easy Materials Accountability System) developed by Los Alamos National Laboratory. The system is a web-based, internet MC&A(Nuclear Materials Control and Accounting) system, which offers a solution for tracking nuclear material holdings and keeping records that manage and control the inventory of those holdings. Preliminary design considerations for this system are easiness to use, simplicity, and economy. These design themes will be incorporated into every aspect of these system, from initial installation of the system to running the program and manipulating the information. The system runs in a workstation/server mode. Data is stored in Relational Database Management System which resides on a server computer running the MS windows NT operating system. Access control and all of the security features required in a materials accountability system are managed on the server. The user interface, which provides functions for entering, retrieving, and modifying data, runs on a workstation and accesses the server via a Web browser. The system software will be written in MS visual Basic script and HTML, which provide a rapid Web screen development environment. The workstation are networked to the server by an Ethernet connection or telephone connection.

Core database consists of various tables concerned with nuclear material information, container information, decay information, project management, process management, facility information, measuring device, measuring data, user and group management information and authentication, etc. Function modules treat the material movements, such as stored location change in one MBA(Material Balance Area) and movement between MBAs.

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Project Manager and Dept.	Ho-dong Kim (Spent Fuel Technology Development Team)
Researcher and Dept.	Myung-Seung Yang, Won-Il Ko, In-Ha Jung, Jang-Jin Park, Kee-Chan Song, ,Jung-Won Lee, Sang-Yoon Lee, Jang-Ho Ha
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Abstract (About 300 Words)
This document contains DUPIC fuel cycle R&D activities to be carried out for 5 years beyond the scope described in the report KAERI/AR-510/98, which was attached to Joint Determination for Post-Irradiation Examination of irradiated nuclear fuel, by MOST and US Embassy in Korea, signed on April 8, 1999. This document is purposely prepared as early as possible to have ample time to review that the over-all DUPIC activities are within the scope and contents in compliance to Article 8(C) of ROK-U.S. cooperation agreement, and also maintain the current normal DUPIC project without interruption. Manufacturing Program of DUPIC Fuel in DFDF and Post Irradiation Examination of DUPIC Fuel are described in Chapter I and Chapter II, respectively. In Chapter Ⅲ, safeguarding procedures in DFDF and on-going R&D on DUPIC safeguards such as development of nuclear material accounting system and development of containment/surveillance system are described in details.

Subject Keywords (About 10 Words)
DUPIC fuel fabrication, DUPIC powder/pellet characterization, Safeguards approach, Facility Attachments, Joint Determination, ROK-US bilateral agreement

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<tr><td colspan="2">연구책임자 및 부서명</td><td colspan="6">김 호 동 (사용후핵연료기술개발팀)</td></tr>
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정 인 하, 송 대 용, 이 상 윤, 하 장 호, 이 호 희
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<tr><td colspan="8">초록(300단어 내외)</td></tr>
<tr><td colspan="8">본 보고서는 기 작성된 기술현황보고서 “KAERI/AR-510/98”에 이어서 작성되는 것으로서, 향후 5년간의 DUPIC 연구개발 계획을 담고 있다. 이 보고서는 DUPIC 연구를 계속적으로 수행하기 위하여 2002년 3월에 종료되는 기존의 JD(Joint Determination)를 연장하기 위하여 한-미 원자력협력협정 제8조 C항에 근거하여 사용후 핵연료 사용에 대한 미국측의 사전동의를 받기 위해 작성되는 것이다. 조사재시험시설 M6 핫셀(DFDF)에서 수행되는 DUPIC 연료 제조 과정 및 계획, 제조된 핵연료의 연구로에서의 조사계획, DUPIC 시설 안전조치 연구개발 방향 등 향후 5년간의 DUPIC 과제의 수행계획을 총 망라하여 기술하였다.
제 1 장에서는 DFDF 시설에서의 DUPIC 핵연료 특성, 제조 절차, 폐기물 관리절차 등이, 그리고 제 2장에서는 제조된 DUPIC 핵연료의 국내 및 국외 연구로에서의 조사계획 및 이를 위한 수송계획 등이 기술되어 있다. 또한 제 3장에서는 DFDF 시설을 위한 안전조치 이행 절차 및 DUPIC 시설 안전조치 관련 연구개발 현황 등이 구체적으로 기술되어 있다.</td></tr>
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