# THE STATUS AND PROSPECT OF DUPIC FUEL TECHNOLOGY

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Since 1991, Korea, Canada and United States have performed the direct use of spent pressurized water reactor (PWR) fuel in the Canada deuterium uranium (CANDU) reactors (DUPIC) fuel development project. Unlike the Tandem fuel cycle, which requires a wet reprocessing, the DUPIC fuel technology can directly refabricate CANDU fuels from the PWR spent fuel and, therefore, is recognized as a highly proliferation-resistant fuel cycle technology, which can be adopted even in non-proliferation treaty countries. The Korea Atomic Energy Research Institute (KAERI) has fabricated DUPIC fuel elements in a laboratory-scale remote fuel fabrication facility. KAERI has demonstrated the fuel performance in the research reactor, and has confirmed the operational feasibility and safety of a CANDU reactor loaded with the DUPIC fuel using conventional design and analysis tools, which will be the foundation of the future practical and commercial uses of DUPIC fuel.

KEYWORDS : DUPIC, Proliferation-Resistance, OREOX, Compatibility, Economics, Safeguards

## 1. INTRODUCTION

Entering the 21st century as a developed country, Korea is facing a trilemma involving economics, energy and the environment caused by the intensifying public conflict between securing energy for national economic development versus alleviating environmental problems. In Korea, the energy consumption rate has been steadily increasing; however, most of the nation's energy resources are imported. Therefore, nuclear power, which is economical and free from green gas emission, is well accepted in Korea because of the nation's insufficient energy resources. Currently, nuclear power plants account for approximately 40% of the total electricity generation in Korea, and this percentage is expected to increase to 60%. Such an increase in nuclear power generation will inevitably result in an increasing need for management of spent nuclear fuel. Therefore, it is urgent that a fuel cycle technology be developed which can utilize spent nuclear fuel as a semi-domestic energy resource, thereby reducing the amount of imported fossil fuels and drastically curtailing the amount of high-level waste.

Technology for the direct use of spent pressurized water reactor (PWR) fuel in Canada deuterium uranium (CANDU) reactors (DUPIC) has been developed by South Korea, Canada, and the United States since 1991 in order to utilize the PWR spent fuel in CANDU reactors [1]. The optimal fuel fabrication process was termed the Oxidation and Reduction of Oxide Fuel (OREOX) process, which was determined based on the results of a feasibility study performed until 1993. The study considered fabrication complexity, operation/maintenance difficulty, and the productivity of the proposed process [2]. Because the OREOX process uses only a thermal/mechanical process for the refabrication of PWR spent fuel, the spent fuel standards are maintained throughout the process, and the process is recognized as a highly proliferation-resistant technology. In addition, because the amount of residual fissile isotopes ($^{235}U$ and $^{239}Pu$) in the PWR spent fuel is twice that of natural uranium, the fuel burnup of the DUPIC fuel can be twice that of the natural uranium fuel in a CANDU reactor. Therefore, as shown in Fig. 1, it is expected that (1) the direct disposal of PWR spent fuel will no longer be necessary, (2) natural uranium resources can be preserved, and (3) the amount of spent fuel from a CANDU reactor can be halved via the DUPIC fuel cycle.

This study will assess the technical feasibility and prospects of the practical use of the DUPIC fuel based on the research results obtained until now. In Chapter 2, the fuel fabrication technology, in-core performance, compatibility with a CANDU reactor, and the fuel cycle economics are assessed. Chapter 3 describes the hardware systems nece-

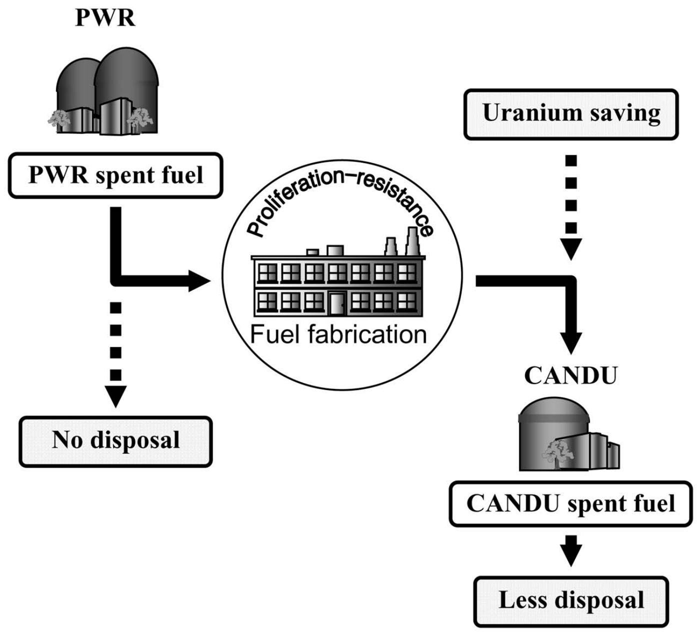



Fig. 1. DUPIC Fuel Cycle Concept

ssary for the practical use of the DUPIC fuel, such as the pilot-scale fuel fabrication facility, fuel transport equipment, fuel loading devices and the nuclear material safeguards system. Chapter 4 reviews the experimental verification required for a demonstration of DUPIC fuel performance in a power reactor and then Chapter 5 presents our conclusions.

## 2. TECHNICAL FEASIBILITY OF THE DUPIC FUEL CYCLE

In order to establish the DUPIC fuel cycle, it is necessary to confirm the fuel fabrication technology in terms of quality assurance and to demonstrate the fuel integrity through in-pile and out-pile tests. In addition, the existing design requirements of the CANDU reactor must be satisfied when the DUPIC fuel is loaded. For the economics of the fuel cycle, the DUPIC fuel cycle cost should compete with the existing fuel cycle cost.

### 2.1 Fuel Fabrication Technology

The Korea Atomic Energy Research Institute (KAERI) has been developing the basic DUPIC fuel fabrication technology since 1992, followed by the development of remote fuel-fabrication technology since 2002. By developing the DUPIC fuel fabrication technology, KAERI has acquired key fuel-cycle technologies for containment and shielding, remote operation of fuel fabrication and inspection devices in the hot-cell and the off-gas treatment, all of which are essential for the treatment of highly radioactive materials. In addition, the fuel cycle technology has been systematically established by developing the technologies for the spent fuel characterization, fuel composition analysis, and the quality control of the highly radioactive materials. The DUPIC fuel fabrication technologies that have been developed up to now are described in the following sections.

#### *2.1.1 Spent Fuel Recycling Facility Establishment*

In 1999, KAERI refurbished the M6 hot-cell of the irradiated material examination facility at KAERI and established the DUPIC fuel development facility (DFDF), as shown in Fig. 2, to process the PWR spent fuel and fabricate the DUPIC fuel on a laboratory scale. In this facility, about 25 pieces of fuel fabrication equipment are installed, as follows:

- Decladding machine, OREOX furnace, off-gas treatment system, attrition mill, and mixer to produce DUPIC fuel powder from the PWR spent fuel
- Compaction press, high temperature sintering furnace, centerless grinder, pellet cleaner and dryer, pellet stack length adjuster, and pellet loader to fabricate DUPIC fuel pellets
- Remote laser welder and welding chamber to fabricate

DUPIC fuel elements
- Quality inspection devices to characterize the DUPIC fuel powder, pellets and elements.

#### 2.1.2 DUPIC Fuel Pellet and Element Fabrication

In December 1998, the Facility Attachment was in force after the research activities of the spent fuel in the DFDF were approved by the International Atomic Energy Agency (IAEA). In April 1999, KAERI obtained a Joint Determination, for the first time in Korea, from the U.S. on the research activities that included the alteration of the forms and content of U.S.-origin PWR spent fuel. After resolving the international restrictions, KAERI produced the DUPIC fuel powder and pellets from the PWR spent fuel in March 2000. In addition, remote laser welding technology was developed for the fuel element end-cap welding, and small-sized DUPIC fuel elements were successfully fabricated in April 2000 for irradiation tests in the HANARO research reactor. Subsequently, the DUPIC fuel fabrication technology was established by successfully fabricating real-size DUPIC fuel elements in February 2001.

#### 2.1.3 DUPIC Fuel Quality Control and Assurance

As the remote fabrication technology of the DUPIC fuel was being established, the DUPIC fuel quality control and assurance system was being developed. In order to prepare for a fuel performance test under commercial reactor operating conditions, a qualification test was successfully conducted for fabricating DUPIC fuel elements by satisfying the specifications of the irradiation test fuel in the Canadian research reactor NRU. Following this, a certificate of quality assurance was issued by the Atomic Energy of Canada Limited (AECL) in July 2002; subsequently, the DUPIC fuel quality control and assurance system was established based on the Canadian standard CAN3-Z299.2-85. Currently, developments are underway to improve the accuracy and reliability of the remote fuel fabrication process and equipment technologies.

### 2.2 Fuel Performance Assessment

As the DUPIC fuel fabrication technology has been being developed, the performance analysis and irradiation tests of the DUPIC fuel have been carried out assiduously since 1998. The DUPIC fuel performance has been assessed in the following three areas:
- Thermal and mechanical material properties of the DUPIC fuel pellet
- Performance analysis code development for the DUPIC fuel
- Irradiation test and post-irradiation examination.

#### 2.2.1 DUPIC Fuel Pellet Material Properties

The thermal and mechanical properties of the DUPIC fuel were analyzed for material properties, which have a great effect on the in-core irradiation behavior of the fuel. For example, the thermal properties were analyzed for the thermal conductivity, thermal expansion, specific heat, densification, and the melting point of the DUPIC pellet, while the mechanical properties were examined for the Young's modulus, high-temperature hardness, and the fracture toughness, which are essential for the fuel integrity analysis. The thermal expansion of the DUPIC fuel pellet was measured via a neutron diffraction experiment and a dilatometer, and the thermal conductivity was measured by a laser flash method. In addition, the diffusion coefficient of the fission gas, which is one of the most important parameters of the fuel integrity, was measured for fission gas diffusion in the DUPIC fuel matrix through a post-irradiation annealing experiment. Each material property was measured several times to obtain a reliable value using the reference DUPIC fuel, of which the simulated fissile content was ~1.5 wt%.

A comparison of the DUPIC and natural uranium pellet material properties shows that the thermal expansion coefficient is higher for the DUPIC fuel pellet by ~5% in the high-temperature range above 1200°C, and the thermal conductivity is smaller for the DUPIC fuel by 8~23% in the range up to 1300°C. The Young's modulus is greater for the DUPIC fuel pellet by ~2% for the porosity condition of 4.5~7.5%. The high-temperature hardness is almost the same for both the DUPIC and natural uranium pellets in the low temperature range, but the value is higher for the DUPIC fuel pellet by 127~287% for the temperature range of 400~1000°C. The fracture toughness of the DUPIC fuel

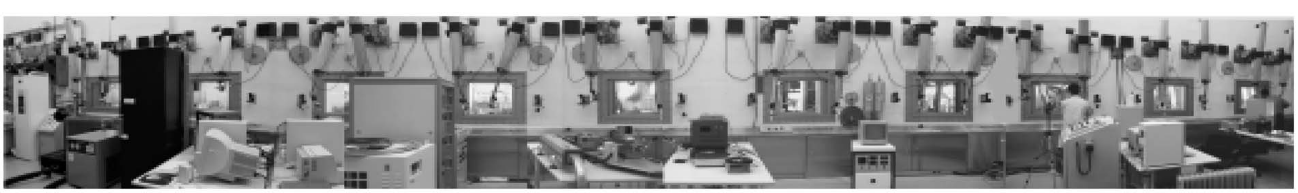


Fig. 2. DUPIC Fuel Development Facility

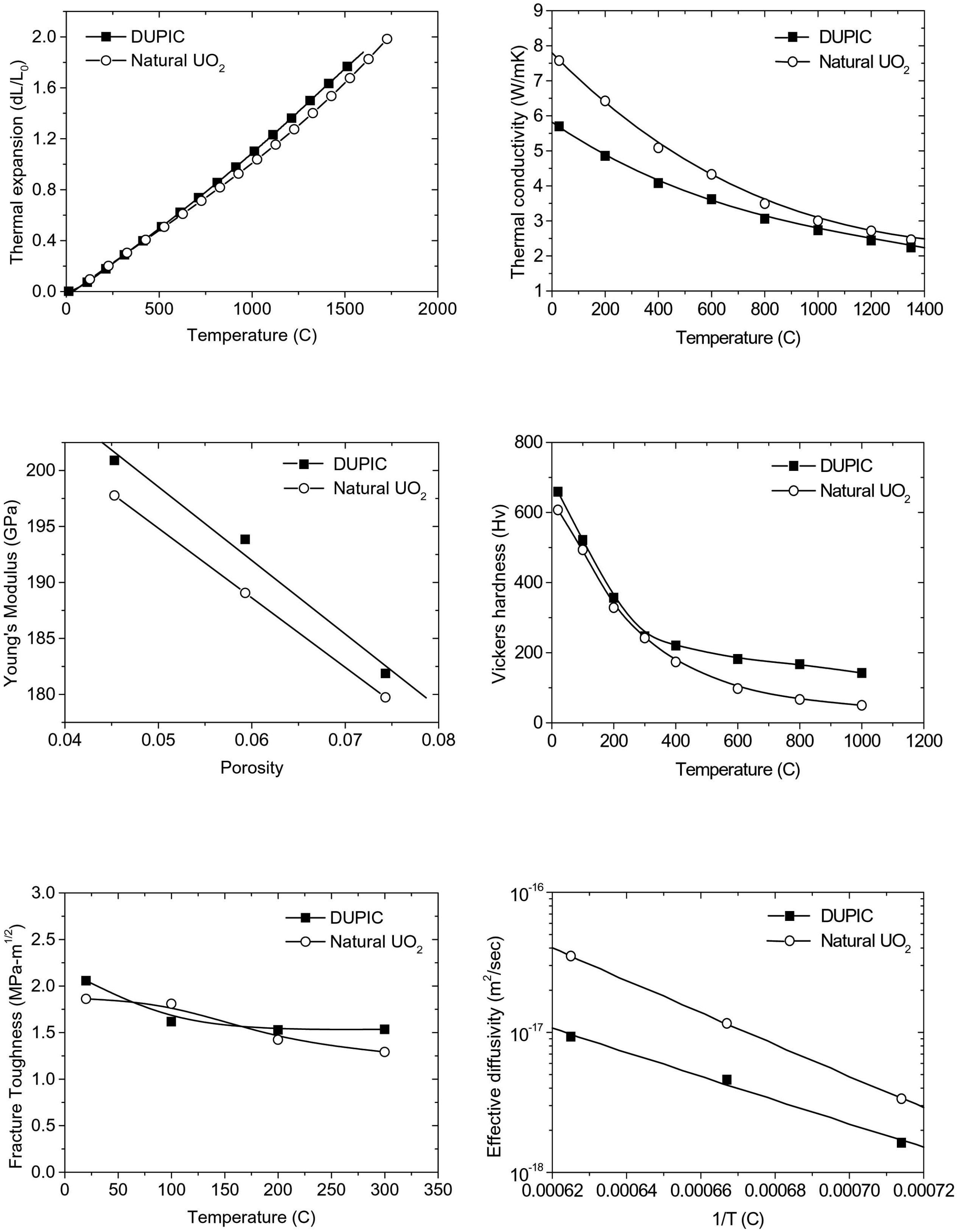
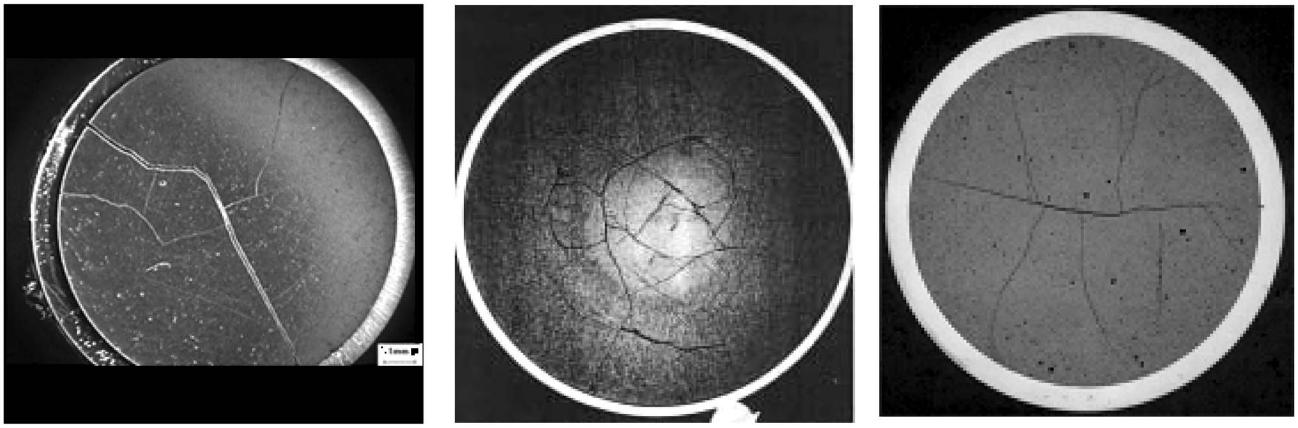



Fig. 3. Material Properties of the DUPIC Fuel Pellet



(a) DUPIC Fuel from HANARO Research Reactor

(b) $UO_2$ Fuel from Douglas Point (CANDU) Nuclear Power Plant [7]

(c) $UO_2$ Fuel from Kori (PWR) Nuclear Power Plant 2

Fig. 4. Comparison of Irradiated Fuel Pellets

pellet is not significantly different from that of the natural uranium pellet in the temperature range of 20~300°C. The diffusion coefficient of the fission gas for the DUPIC fuel matrix is estimated to be ~1/3 of that for the natural uranium, because the DUPIC fuel contains impurities (fission products) even at the fresh fuel condition. The experimental results, presented and compared with natural uranium properties in Fig. 3, have been integrated into the DUPIC fuel performance database system. Based on this database, the material characteristics models of the DUPIC fuel have been developed to construct the performance analysis code system. For the natural uranium in Fig. 3, the Young's modulus, hardness, fracture toughness and the diffusion coefficient were measured together with the DUPIC fuel, while others were obtained from the open literature. [3,4]

### 2.2.2 DUPIC Fuel Performance Analysis Code

The KAERI advanced oxide fuel-performance analysis code system (KAOS) has been developed to construct the DUPIC fuel performance analysis code system and to perform the safety analysis of the DUPIC fuel irradiation tests in the HANARO research reactor. The KAOS code system has two major functions: (1) to analyze macroscopic fuel behavior during irradiation in a reactor and (2) to assess fuel integrity in the reactor using a commercially-available finite element method. In addition, the ELESTRES-D code was developed to analyze the DUPIC fuel behavior in a CANDU reactor by combining the existing fuel analysis code, ELESTRES [5], and the DUPIC fuel pellet material properties obtained in this study, thereby establishing a fuel performance analysis system for the licensing and practical use of the DUPIC fuel.

### 2.2.3 Irradiation and Post-Irradiation Tests

Five DUPIC fuel irradiation tests were carried out in HANARO from 1999 to 2004. The first, second and fourth tests were non-instrumented tests, while the third one was an instrumented test to measure the thermal neutron flux of the irradiation hole, and the fifth one was an instrumented test to measure on-line the center temperature of the DUPIC fuel pellet. One fuel element irradiated in the third test was burned again in the fourth test. This element has a fuel burnup of 6700 MWd/tHM, which is the highest among all the fuel burnups obtained until now. The maximal and average linear element ratings of this element were estimated to be 34 kW/m and 25 kW/m, respectively.

A comparison of the pellet centerline temperature between the on-line measurement (1st instrumented irradiation test of the highly radioactive fuel in Korea) and the KAOS calculation showed that the calculation result was a little conservative for the 1st cycle of the irradiation but matched the measurement result within 8% for the temperature range of 800~1200°C [6]. Currently KAOS can predict the fuel behavior only for normal conditions. A post-irradiation examination was performed for the DUPIC fuel pellet, which was irradiated to the average fuel burnup of the standard CANDU fuel. As shown in the optical microscopy photo of Fig. 4, the irradiation behavior of the DUPIC fuel is not that different from that of the standard CANDU spent fuel or of PWR spent fuel of 40000 MWd/tHM.

Three DUPIC fuel elements (BB02, BB03 and BB04) were fabricated by AECL and irradiated in the fuel test loop

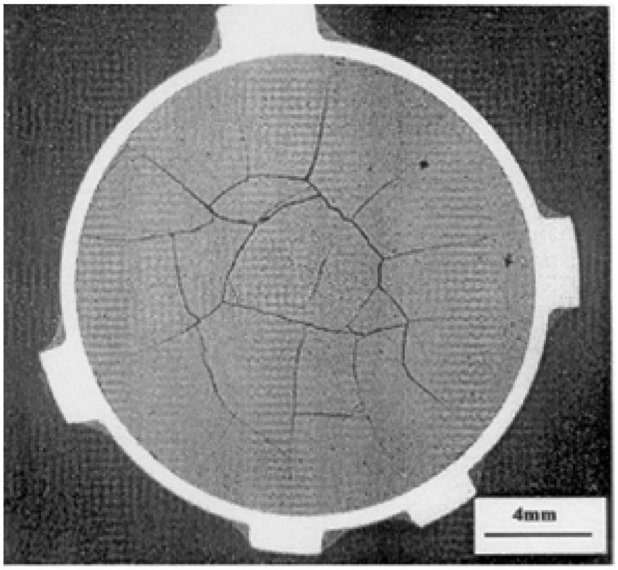



Fig. 5. DUPIC Fuel Irradiated in NRU

of the NRU research reactor under a Korea-Canada bilateral joint research program. The discharge burnups were 21000, 10000 and 16000 MWd/tHM for the BB02, BB03 and BB04 fuel elements, respectively. The maximum linear power of the BB03 and BB04 fuel elements was ~50 kW/m. Post-irradiation examinations were performed for the BB03 and BB04 fuel elements in 2001 and 2003, respectively, and the results for the BB03 fuel element are shown in Fig. 5. It can be seen that the BB03 fuel element shows normal irradiation behaviors under the CANDU reactor operation conditions. It was detected, however, that the amount of fission gas release was greater when compared to the standard CANDU fuel. For example, the fission gas release from BB03 and BB04 elements was 12.6 and 16.0 $cm^3$, respectively, while that of the standard CANDU fuel is ~4.0 $cm^3$ under nominal operating conditions. Nonetheless, the analysis showed that the DUPIC fuel element integrity was maintained.

The experimental results of the DUPIC fuel irradiation in the HANARO and NRU research reactors and of the post-irradiation are in good agreement with the performance analysis results by the KAOS code that was implemented with the mechanistic model (thermal and mechanical material properties) of the DUPIC fuel pellet. Therefore, it is expected that the DUPIC fuel integrity is also maintained in the CANDU reactor, which will be confirmed through a continuous generation of experimental results and a development of the empirical correlations.

## 2.3 Compatibility with a CANDU Reactor

The DUPIC fuel should be designed such that the CANDU reactor operates without compromising the safety and economics of the system. Therefore, the compatibility can be assessed by considering the general design requirements, as follows [8]:

- The fuel design, reactor operation, and reactor safety parameters should not exceed the design limits during normal and transient conditions.
- The total and local power levels should be controlled safely. It should be possible to trip the reactor and maintain a subcriticality, if necessary.
- Two independent shutdown systems are required, and each system should possess the capability of maintaining a subcriticality.
- The reactor control and safety system should not impair the economics and flexibility of the CANDU reactor.

Considering these design requirements, the compatibility of the DUPIC fuel with the CANDU reactor was assessed in terms of the fuel lattice physics characteristics, reactor power distribution, reactivity devices performance, and the operational margin.

### *2.3.1 DUPIC Fuel Physics Characteristics*

The reference DUPIC fuel composition was determined based on the PWR spent fuel data accumulated in South Korea until 1994 under the conditions that the fuel composition variation is minimized, the PWR spent fuel utilization is maximized, and the amount of fresh uranium used for the composition adjustment is minimized. At the same time, the DUPIC fuel composition was also adjusted such that the DUPIC fuel lattice property, core performance, and fuel cycle cost were optimized; and the reference composition was determined to be 1.0 wt% and 0.45 wt% for $^{235}U$ and $^{239}Pu$, respectively [9]. Because the delayed neutron fraction decreases due to the plutonium in the DUPIC fuel, the centre element of the fuel bundle was designed to be mixed with a burnable poison (natural dysprosium), which can compensate for the positive reactivity insertion in case of a loss of coolant accident (LOCA) [10]. The DUPIC fuel bundle adopts the 43-element CANDU Flexible (CANFLEX) model, of which the compatibility with the fuel channel and fueling machine was already demonstrated in the CANDU reactor.

At the equilibrium burnup state, the DUPIC fuel temperature coefficient is -1.4 $\mu$k/°K, which is more negative when compared to the natural uranium fuel. Therefore, a safety feature is enhanced for the DUPIC fuel in terms of the fuel temperature coefficient. The coolant and moderator temperature coefficients of the DUPIC fuel are smaller than those of the natural uranium fuel due to a thermal flux depression. The shutdown reactivity and purity coefficients are similar for both fuels. The void reactivity was estimated to be 12 and 14 mk for the DUPIC and natural uranium fuel lattice, respectively, at the equilibrium burnup. Because the DUPIC fuel bundle has a poisoned fuel element in the bundle centre, the void reactivity of the DUPIC fuel is smaller when compared to the natural uranium fuel. For a 713 MWe CANDU (CANDU-6) reactor, the half-core void reactivity was estimated to be 6.5 and 7.4 mk for the DUPIC and natural uranium fuel, respectively, by a detailed RFSP [11] simulation.

For the DUPIC fuel, the kinetic parameters represented by the delayed neutron fraction ($\beta_{eff}$) and the neutron generation time ( $\Lambda$ ) are reduced by 3% and 24%, respectively, when compared to the natural uranium fuel at the equilibrium burnup. Therefore, the slope of the power pulse (represented as the prompt inverse period) could be steeper for the DUPIC fuel in case of a LOCA. For the DUPIC fuel, however, the power pulse upon a LOCA is mitigated by the aid of a burnable poison, which reduces the void reactivity.

### 2.3.2 Compatibility with the Reactor Power Distribution

Because the fissile content ($^{235}U$ and $^{239}Pu$) of the DUPIC fuel is twice that of the natural uranium, a 2-bundle shift refueling scheme is adopted for the DUPIC fuel, by considering the reactivity insertion upon a refueling [12]. In order to simulate actual reactor operation, refueling calculations were conducted for 600 full power days (FPD), and the results are shown in Fig. 6. The peak maximum channel power (MCP) and maximum bundle power (MBP) of the DUPIC fuel core are 6998 kW and 827 kW, respectively, which are below the license limits of the natural uranium core (7300 kW and 935 kW). Compared to the natural uranium core, the average MCP and MBP of the DUPIC fuel core are also lower by 57 kW and 54 kW, respectively.

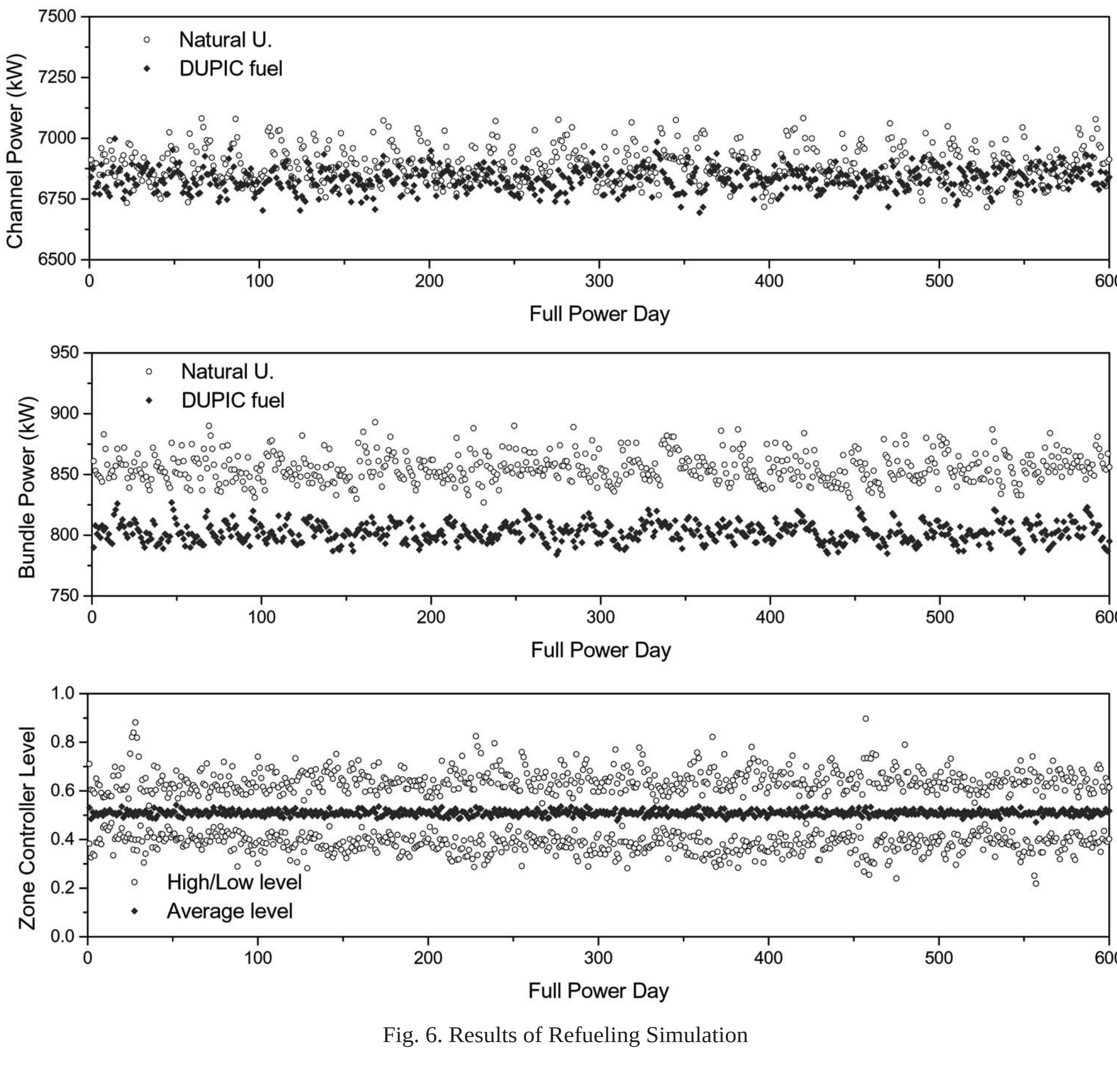



Fig. 6. Results of Refueling Simulation

The average channel power peaking factors (CPPF) of both cores are comparable. Regarding the refueling operation, the DUPIC fuel core requires four channels to be refueled per day. Therefore, the total number of fuel bundles loaded per day is approximately 8 for the DUPIC fuel core and 16 for the natural uranium core.

The composition heterogeneity of the DUPIC fuel causes variations of the lattice parameters, which in turn results in uncertainties for the core performance parameters. The uncertainties of the core performance parameters were estimated by both the deterministic and statistical methods [13,14]. In general, the results of the deterministic analysis were more conservative. As a result, the uncertainties of the MCP, MBP, and CPPF were estimated to be less than 1% for the simulated core.

### 2.3.3 Compatibility with the Reactivity Devices

*Zone Controller Unit (ZCU)*

The reactivity worth of the ZCU is 5.8 mk and 6.5 mk for the DUPIC and natural uranium fuel core, respectively. In order to assess the effectiveness of the ZCU system for suppressing a xenon-induced spatial oscillation, the refueling operation was simulated for channels J-14, S-3, and L-9. The channel J-14 is located in a relatively higher burnup region, and the S-3 and L-9 channels are located in a lower burnup region. After a refueling, the power tilts increase for the first 3.5 hrs and approach asymptotic values. For the natural uranium core, the damping behaviors of the zone power are similar to those of the DUPIC fuel core, except that the power tilts increase for 2.5 hrs and then reach the asymptotic values [15].

For the estimation of the ZCU draining effect, simulations were performed for a steady-state core with an equilibrium xenon concentration. Because the ZCU worth of the DUPIC fuel core is smaller than that of the natural uranium core, the power coefficient (a percentile change of the zone power per unit change of reactivity) is greater for the DUPIC fuel core. Therefore, the ZCU level of the DUPIC fuel core should change more than that of the natural uranium core to compensate for the same amount of reactivity perturbation. However, because the ZCU worth of the DUPIC fuel core itself is smaller, the effect of a ZCU draining is similar for both the DUPIC and natural uranium cores, which was confirmed by the refueling simulation [16].

*Adjuster Rod (ADJ)*

The ADJ's are designed to provide a positive reactivity when withdrawn from the core. The static reactivity worth of the ADJ is 10.2 and 16.6 mk for the time-average DUPIC and natural uranium fuel core, respectively. The xenon load is 6.8 mk at 30 min after a shutdown for the DUPIC fuel core, which is lower than the static reactivity worth of the ADJ by 3.4 mk and, therefore, the existing ADJ has the capability of overriding the xenon load at 30 min after a reactor shutdown. The ADJ bank reactivity insertion characteristics of the DUPIC fuel core are the same as those of the natural uranium core, except that the bank reactivity worth of the DUPIC fuel core is always smaller than that of the natural uranium core.

For the startup after a short shutdown, the xenon override time is 45 min for the DUPIC fuel core. The total required time to return to full power is 6.6 hrs, which is 3.2 hrs longer than that of the natural uranium core. For the startup after a long shutdown (poison-out), the required time to restart the reactor is 29 hrs for the DUPIC core, which is smaller than that of the natural uranium fuel core (36 hrs). The time required to return to a full power is 40 min for the DUPIC core, which is longer than that of the natural uranium core by 11 min.

In the event of a loss of refueling capability, the CANDU reactor can obtain excess reactivity by withdrawing the ADJ (shim operation). The simulation results showed that a successive withdrawal of the ADJ would permit an operation for more than 31 days without a fueling for the DUPIC fuel core, which is about 10 days shorter than that of the natural uranium core. For a step-back from 100% to 60% reactor power, it takes 4 more hours for the DUPIC fuel core to return to a full power when compared to the natural uranium core. The analysis results show that the ADJ system satisfies the design requirements for the DUPIC fuel core, even though the response time to the transient core condition is somewhat delayed.

*Mechanical Control Absorber (MCA)*

Four MCA's are used to provide rapid controlled reductions of the reactor power. The design requirement of the MCA is that the total worth of the four MCA's combined with a filling of the ZCU should be adequate to compensate for the maximum possible reactivity increase upon a power reduction from a nominal operation to a hot shutdown condition. The calculated total static reactivity worth of the four MCA's in an equilibrium core is 8.4 mk for the DUPIC core, which is 3 mk smaller than that of the natural uranium core. However, the reactivity increase following a hot shutdown of the equilibrium core was estimated to be 3 mk and, therefore, the MCA possesses enough reactivity to compensate for the reactivity increase following a reactor hot shutdown.

*Shut-off Rod (SOR)*

For the DUPIC fuel core, the static reactivity worth of 28 SORs (shutdown system 1) is 72.5 mk, and the worth drops significantly, to 40.5 mk, when the two most effective rods are missing. For the natural uranium fuel core, the static reactivity worth of the 28 and 26 SORs are 87 and 56 mk, respectively. The adequacy of the SOR design can be assessed by comparing the maximum thermal energy deposited in the fuel during the transient and the threshold value (840 J/g) of the fuel breakup [17]. The shutdown capability of the SOR was assessed against a 20% reactor inlet header (RIH) break LOCA by a coupled calculation between the reactor physics code, RFSP, and the full plant circuit code, CATHENA [18]. The simulation showed that the fuel breakup margin was 24% for the DUPIC fuel core, while it was 28% for the natural uranium core. For the assessment of the shutdown system 2, a 100% RIH break LOCA was chosen as the design basis accident. The simulations showed that the fuel breakup margins are 32% and 38% for the DUPIC and natural uranium fuel core, respectively. Therefore, it is believed that the existing CANDU reactor shutdown systems maintain their design requirements for the DUPIC fuel core.

#### 2.3.4 Compatibility with the Reactor Trip Set-point

The basic regional overpower protection (ROP) system design requirement is that the reactor is tripped for any flux shape and power ripple before any coolant channel reaches its critical channel power (CCP). For the ROP trip set-point calculation, the RFSP code is used to obtain the flux shapes and power distributions. The physics calculations are performed for design-base cases based on the time-average model, which includes all the normal operating configurations, all the single-device abnormal configurations, and certain types of double-device abnormal configurations. The on-power refueling is also considered by calibrating the rippled power to a 100% power level using the CPPF obtained from the 600-FPD refueling simulation. For the thermal-hydraulic calculation, the NUCIRC-MOD1.505 [19] code was used to calculate the CCP for each ROP case, using the $x_c$-$L_b$ correlation developed by AECL [20].

The ROP trip set-point of the DUPIC fuel core was estimated by the ROVER-F [21] code. The estimated ROP trip set-points were 123.4% and 122.9% for the DUPIC and natural uranium fuel core, respectively. Consequently, it is expected that DUPIC fuel loading in a CANDU-6 reactor does not deteriorate the current ROP trip set-point designed for the natural uranium fuel. In fact, a slight increase of the trip set-point appears to be attainable with the DUPIC fuel owing to the inlet-skewed channel power distribution and the 43-element CANFLEX fuel bundle geometry [22].

### 2.4 Economics of the DUPIC Fuel Cycle

The once-through or mixed oxide fuel cycle already has commercial-size fuel cycle facilities, which enhances their credibility to a certain extent in the evaluation of the fuel cycle cost. However, the DUPIC fuel cycle cost is estimated based on the fuel cycle component cost inferred through the conceptual design studies, which results in a relatively high uncertainty of the fuel cycle cost when compared to conventional fuel cycles.

#### 2.4.1 Component Cost of the DUPIC Fuel Cycle

A preliminary conceptual design for a commercial-scale DUPIC fuel fabrication facility with a capacity of 400 ton/yr was performed through a joint study with a U.S. company Scientech. Then, the conceptual design study was completed by implementing the remote fuel fabrication equipment development cost estimated by the Oak Ridge National Laboratory. The direct cost for the construction of the DUPIC fuel fabrication facility was estimated to be 580 million dollars, and the indirect cost including the design and licensing was estimated to be 450 million dollars. The levelized DUPIC fuel fabrication cost, including the contingency, operation and maintenance, and decommissioning costs, was 616 dollars/kgHM [23].

The cost estimation of the DUPIC fuel handling focused on the design change and additional equipment necessary for the fuel loading under the condition that the existing fuel handling devices are fully utilized. The direct cost of the DUPIC fuel handling was estimated to be 4.5 million dollars, and the levelized fuel handling cost was 5.1 dollars/kgHM [24]. The final disposal cost of the spent DUPIC fuel was estimated based on the Canadian repository model that uses the titanium container and borehole emplacement concept. The direct cost for the construction of the spent DUPIC fuel repository was estimated to be 1,700 million dollars, and the levelized disposal cost was 389 dollars/kgHM [25].

#### 2.4.2 DUPIC Fuel Cycle Cost

The fuel cycle cost was calculated using the levelized unit cost model presented by the Organization for Economic Cooperation and Development/Nuclear Energy Agency in 1993 [26]. The DUPIC fuel cycle can be most efficient

**Table 1**. Levelized Costs (mills/kWh) of the Once-through and DUPIC Fuel Cycle

| | Component | Once-through | | DUPIC |
|---|---|---|---|---|
| | | PWR | CANDU | |
| PWR | Uranium ($U_3O_8$) | 1.289 | - | 1.289 |
| | Conversion | 0.229 | - | 0.229 |
| | Enrichment | 2.084 | - | 2.084 |
| | Fabrication | 1.057 | - | 1.057 |
| | Transportation | - | - | 0.099 |
| | Transportation & Storage | 0.453 | - | - |
| | Disposal | 0.367 | - | - |
| CANDU | DUPIC Fuel Handling | | - | 0.019 |
| | Uranium ($U_3O_8$) | | 0.212 | - |
| | Conversion | | 0.038 | - |
| | Fabrication | | 0.301 | 1.197 |
| | Transportation | | - | 0.066 |
| | Transportation & Storage | | 0.191 | 0.156 |
| | Disposal | | 0.150 | 0.132 |
| | Total | 6.419 | | 6.346 |

when two 950 MWe PWRs and one 713 MWe CANDU reactor are linked. In this case, the DUPIC and once-through fuel cycle costs are 6.346 mills/kWh and 6.419 mills/kWh, respectively, under the condition that both fuel cycles generate the same amount of electricity. The breakdown of the fuel cycle cost is given in Table I. The cost difference between the DUPIC and once-through fuel cycles is marginally small and is believed to be within the uncertainty range of the fuel cycle cost by considering the uncertainties associated with the unit cost of the fuel cycle component. For the material flow of the DUPIC fuel cycle, the amount of natural uranium used and the spent fuel accumulation is reduced by 20% and 65%, respectively, when compared to the once-through cycle [27].

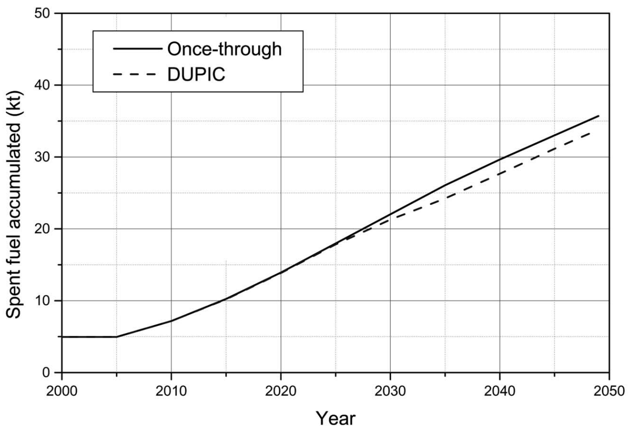



Fig. 7. Comparison of Spent Fuel Accumulation

### 2.4.3 Application to the Domestic Nuclear Fuel Cycle

As of July 2005, 16 PWRs and four CANDU reactors are operating in South Korea. The electricity generation by the nuclear plants is 15 GWe and it is expected to increase to 25 GWe in 2015. Assuming that the four CANDU reactors are loaded with the DUPIC fuel in sequence from 2015, the mass flow was analyzed up to 2050. The results showed that, for the considered period, natural uranium can be saved by 8000 ton , and the amount of spent fuel can be reduced by 1900 ton as shown in Fig. 7. When all four CANDU reactors are loaded with the DUPIC fuel, the annual saving of natural uranium is estimated to be valued at 20 million dollars. For an equilibrium fuel cycle, the spent fuels from

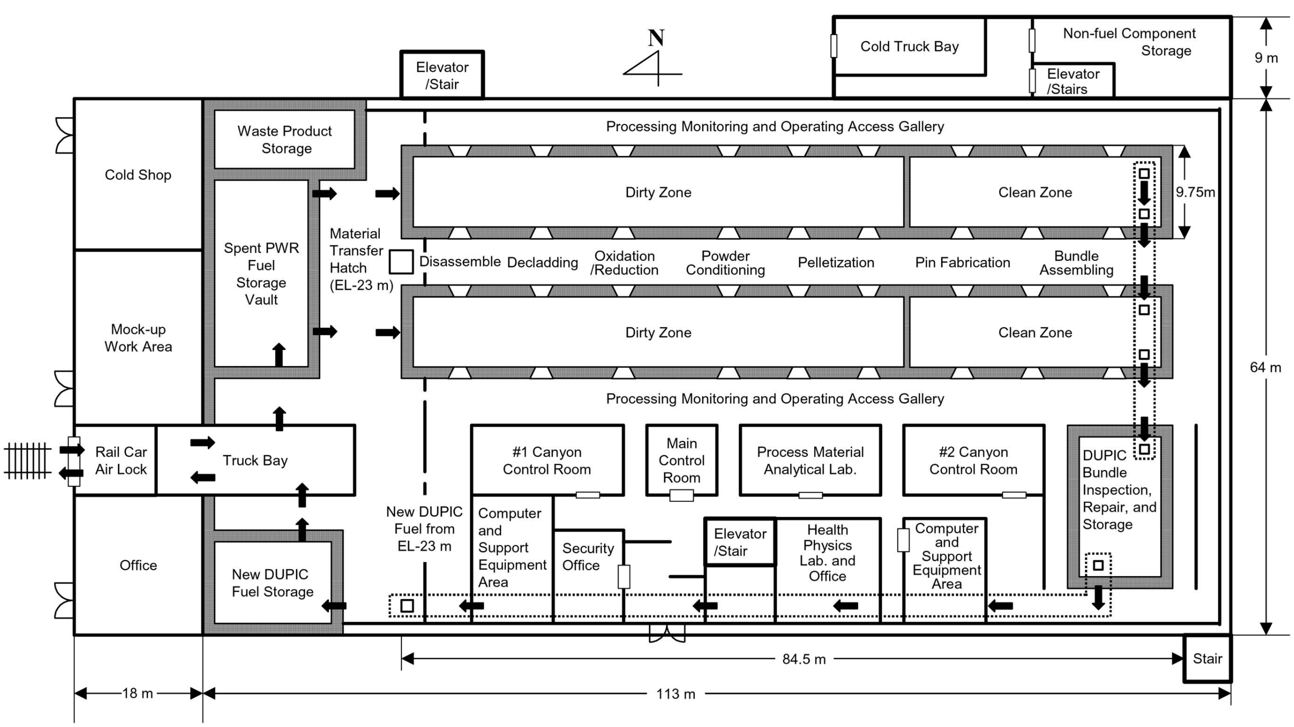



Fig. 8. DUPIC Fuel Fabrication Facility

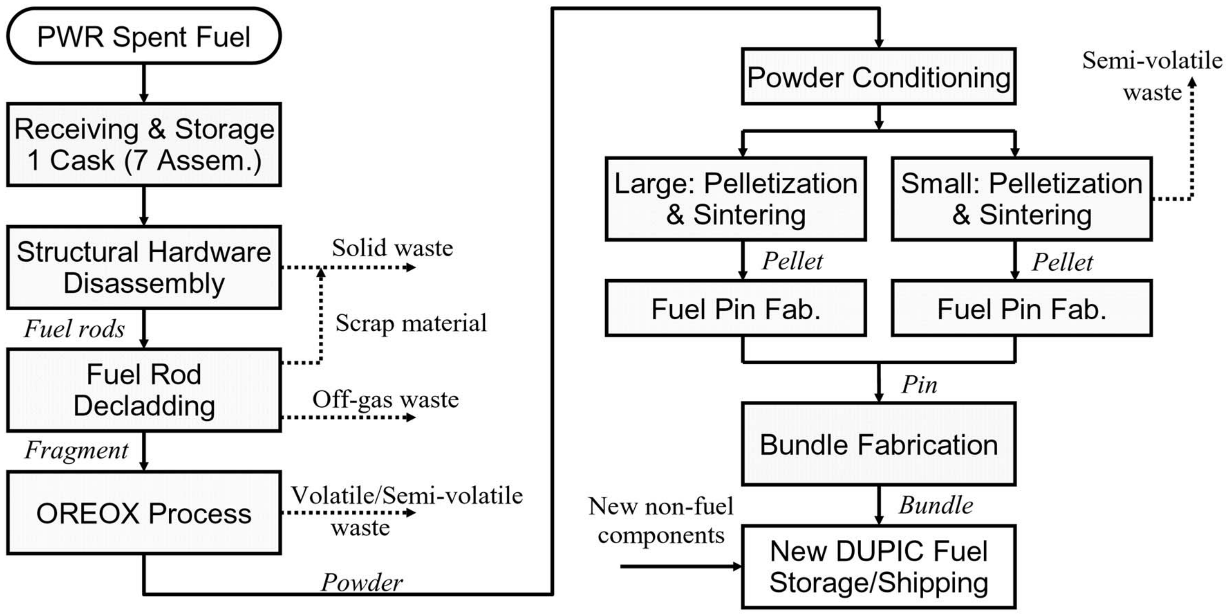



Fig. 9. DUPIC Fuel Fabrication Process

eight PWR units will be used for four CANDU reactors. Assuming that each refurbished CANDU reactor is operated for 20 years, the total electricity generation from the eight PWR units and four CANDU reactors is $1.2 \times 10^{12}$ kWh. Based on the cost saving of 0.073 mills/kWh for the equilibrium DUPIC fuel cycle, the total cost saving for 20 years is approximately 86 million dollars.

## 3. HARDWARE SYSTEMS FOR THE PRACTICAL USE OF THE DUPIC FUEL

The hardware systems considered for the practical use of the DUPIC fuel are a pilot-scale DUPIC fuel facility, transportation equipment, fuel loading facility and devices, and the nuclear material safeguards system for the DUPIC facility and reactor. The hardware systems were preliminarily evaluated and the results are given in the following sections.

### 3.1 DUPIC Fuel Fabrication Facility

The pilot-scale DUPIC facility is designed with a capacity of 50 ton/yr and a plant lifetime of 40 yrs. The design also considers the expansion of the facility to a commercial-scale plant. The DUPIC facility site is approximately 4000 $m^2$ in area, where the main process building is located in the centre, surrounded by auxiliary buildings. As shown in Fig. 8, two rectangular-shaped hot-cells are deployed in parallel in the main process building. The hot-cell consists of two floors: the key remote fabrication equipments are deployed on the 1st floor, while the gaseous waste treatment system and utility devices are deployed on the 2nd floor. In the basement, the scrap transfer, solid waste treatment, and the heating and ventilation air cleaning systems are located; the underground pathways for the nuclear material transfer are connected to the hatches on the 1st floor of the hot-cell.

The overall process of the pilot-scale DUPIC facility can be categorized into the DUPIC fuel fabrication, structural part recycling, and the radioactive waste treatment. Figure 9 shows a detailed flow path of the DUPIC fuel process, and the main processes are described below.

*PWR Spent Fuel Receiving and Storage*

The PWR spent fuels are transported to the DUPIC facility by using the transport cask. The fuel burnup and fissile contents are measured by a non-destructive method in the reception cell and the fuels are dry stored. The measured data is used for the selection of the PWR spent fuel assemblies that satisfy the DUPIC fuel fissile content requirements.

*Spent Fuel Disassembly and Decladding*

The fuel rods are mechanically removed from the PWR fuel assembly. The fuel assembly structural material is transferred to the solid waste treatment area and stored after going through the volume reduction and packaging process. The fuel rods are chopped into an appropriate size by a mechanical and/or laser cutting method, and the fuel material and cladding are mechanically separated. The fission gases released in this process are sent to the off-gas treatment system and are stored after going through the separation, treatment, and packaging processes. The cladding material is cleaned and decontaminated for more than a 99% recovery of the fuel material and is transferred to the solid waste treatment area.

*Fuel Powder Preparation*

The spent fuel materials are treated by repeated OREOX processes to form fuel powder that satisfies the powder characteristics requirements. The oxidation is performed at 450°C in air, and the reduction is performed at 700°C, which is repeated three times. Three OREOX furnaces are installed with automatic material transfer devices, which are connected to the off-gas treatment system.

*Fuel Pellet Fabrication*

The DUPIC fuel pellets are produced from the spent fuel powder through the pre-compaction, granulation, final compaction, sintering, and the grinding processes. The sintering is performed at 1750°C to obtain a pellet of more than a 95% theoretical density. Defective pellets and structural materials are transferred to the recycling station and reused.

*Fuel Element Fabrication*

The fuel pellets are loaded into the cladding tube manufactured outside the hot cell and the end cap is welded to form a fuel element. Non-destructive tests are performed for the end cap welding, such as a helium leak test following the quality assurance procedure. In this process, defective fuel elements are rejected and transferred to the structural material recycling station. The qualified fuel elements are transferred to the fuel bundle fabrication area after the surface decontamination and fissile content measurement.

*Fuel Bundle Fabrication*

Because the DUPIC fuel bundle adopts the CANFLEX geometry, two process lines are in use to process fuel elements with two different diameters. Once the fuel bundle is assembled, the non-destructive tests are conducted and the dimensions are measured. The fuel bundles that satisfy the quality assurance requirements are transferred to the storage area, while the defective fuel bundles are sent to the structural material recycling station to be reused.

## 3.2 Transportation of PWR Spent Fuel and Fresh DUPIC Fuel

Though it is, in general, efficient to use a high-capacity shipping cask to transport the PWR spent fuel, it is expected that ground-transportation may not be appropriate due to the heavy weight involved. Therefore, a comprehensive analysis of the transportation system is recommended, which includes transportation by truck, train, and ship. The cost to establish the transportation system is dependent on the shipping cask, transportation truck, ship, loading/unloading method of the shipping cask, harbor facilities, and the maintenance/repair facility, and is expected to be approximately 150~250 million dollars.

The DUPIC fuel bundles are placed in a basket, and several baskets are loaded into a shipping cask in the shielded facility. The shipping cask is ground-transported to the CANDU nuclear power plant (NPP), and the fuel bundles are unloaded in the storage room. It is also necessary to analyze comprehensively the transportation between the DUPIC facility and a CANDU NPP. It is expected that the DUPIC fuel transportation system will consist of the fuel basket, shipping cask, loading frame, and transportation truck. The estimated cost of the DUPIC fuel transportation is approximately 5 million dollars.

## 3.3 DUPIC Fuel Loading in the Wolsong Nuclear Power Plant

There are two ways of loading the DUPIC fuel into the Wolsong reactor, depending on the loading route: front-loading and rear-loading. Both options require modifications of the Wolsong reactor fuel loading system to accommodate the remote handling of the DUPIC fuel due to radiation emission.

### *3.3.1 Front-Loading Path*

This option requires a new hot-cell in the new fuel loading area inside the reactor building. The DUPIC fuel bundles are remotely and automatically pulled out from the cask in the new hot-cell. It is important to secure a space in the new fuel loading area to modify structures and install a new hot-cell, considering spatial restrictions for the hot-cell construction, remote handling equipment, hot-cell maintenance, and the shipping cask cleanup and transfer. The new fuel loading area and equipment should be shielded and modified such that the DUPIC fuel is remotely and automatically loaded. Extra fuel loading equipment is also required in case of decontamination and an exchange of the contaminated or failed fuel loading equipment. Finally, the front-loading facility design should minimize or eliminate the possibility of radiation exposure to the operators in the containment building.

### *3.3.2 Rear-Loading Path*

This option utilizes the existing spent fuel storage bay in the power plant. The DUPIC fuel bundles pulled out from the shipping cask are transferred to the reception bay and loaded into the fueling machine, following in reverse the existing discharge route of the spent fuel. In this option, the existing dry storage facility in the storage bay area should be modified to be used for opening the shipping cask and handling the DUPIC fuel bundle. Additional equipment is also required, such as a blow dryer to remove the light water from the DUPIC fuel, a ram device for inserting the DUPIC fuel into the spent fuel discharge port, and a gamma radiation detector for identifying the new and spent fuel. The computer program that controls the fuel movement should be modified and updated so that the new fuel can be loaded in the reverse direction. Finally, the reverse operation mechanisms of fuel transferring devices should be verified.

#### *3.3.3 Cost Evaluation*

The DUPIC fuel can be used in the Wolsong NPP either by the front-loading path or by the rear-loading path. For both options, it is essential to modify the existing fuel loading facility due to the high radiation field of the DUPIC fuel. Therefore, the optimal fuel loading option should be determined by considering safety, productivity, costs, operational efficiency, maintenance and regulatory requirements. As for the rear-loading option, the estimated direct cost is around 4.5 million dollars under the condition that the existing fueling facilities are fully utilized. This cost does not consider the replacement of the major components for life extension, the decontamination of the DUPIC fuel-loading route, or a spare fueling machine.

### 3.4 Nuclear Material Safeguards

The DUPIC Safeguards Neutron Counter (DSNC) and DUPIC Safeguards Neutron Monitor (DSNM) have been developed and used in the DFDF where the nuclear material is handled in a laboratory scale. The DSNC is a non-destructive neutron counting system for the verification of the nuclear material flow during the DUPIC process, which estimates the plutonium content of the nuclear material based on the quantitative measurement of the $^{244}Cm$ content, even in a high-gamma radiation environment. The DSNC together with the DSNM have been approved by the IAEA as the official nuclear material measurement and monitoring devices of the DFDF and have been used for IAEA and national inspections [28,29].

#### *3.4.1 DUPIC Fuel Fabrication Facility*

For the safety of the pilot-scale DUPIC fuel fabrication facility, the safeguards system should be developed for a fuel fabrication facility with a capacity of 50~200 ton/yr and a safeguardability analysis should be performed. The preliminary analysis done by Los Alamos National Laboratory in 1992 showed that the IAEA safeguards criteria could be satisfied if the uncertainty level of the nuclear material measurement and monitoring devices is within that of a conventional nuclear fuel cycle facility [30,31].

Safeguards at the pilot-scale DUPIC fuel fabrication facility include an item counting system for handling spent fuels before the DUPIC process, a bulk counting system for alteration of the nuclear material by the DUPIC process, and a bulk counting system for assembling fuel bundles. The safeguards equipment necessary for the pilot-scale DUPIC fuel fabrication facility are as follows:

- Feed material measurement: PWR spent fuel rod scanning system
- Process material measurement: assay and verification system for the fuel powder, hold-up, and the process material in the canister
- Process monitoring: Unattended continuous hot-cell monitoring system
- Final product measurement: CANDU fuel bundle assay system.

A Near Real Time Accounting (NRTA) system is also necessary to maintain the integrity and continuity of the accounting data between the facilities. The NRTA system is integrated with individual nuclear material measurement systems.

#### *3.4.2 CANDU Nuclear Power Plant*

The safeguards equipment currently used in a CANDU NPP are an integrated fuel monitor, a core discharge monitor, a fuel bundle counter, and inspection devices for spent fuel management. In addition, the following safeguards approaches are being considered:

- Annual physical inventory verification and quarterly interim inspections
- Unattended monitoring of the spent fuel material flow
- Extensive use of the containment surveillance system
- Extensive surveillance of workers during fuel transfer to the dry storage.

When the DUPIC fuel is loaded into a CANDU reactor, a new safeguards approach should be deployed, because all the monitoring systems are remotely operated and the material flow and classification system are different from those of the current CANDU reactor system. South Korea has been negotiating with the IAEA on the integrated safeguards system since the Additional Protocol entered into force in 2004. Therefore, it is expected that the current safeguards system can be utilized for the CANDU reactor with the DUPIC fuel if the current system is modified and supplemented through consultations with the main inspection organizations, such as the National Nuclear Management and Control Agency (NNCA) and the IAEA.

## 4. EXPERIMENTAL DESIGN VERIFICATION PLAN FOR A PRACTICAL USE OF THE DUPIC FUEL

In order to demonstrate the DUPIC fuel performance under power reactor operating conditions, a lead test assembly (LTA) irradiation test should be performed. For the LTA irradiation, a series of in-pile and out-pile tests is required to prepare the fuel design documents and evaluation reports to be submitted to the regulatory body.

### 4.1 Physics Design Verification

The physics design provides the excess reactivity and power distribution of the DUPIC fuel. The design calculation is mostly performed by the WIMS-AECL (WIMS) [32] and RFSP for the lattice and core simulations, respectively. The benchmark calculation of the WIMS/RFSP was carried out for the Deuterium Critical Assembly (DCA), which is a heavy-water moderated, light-water cooled, and pressure-tube type research facility [33,34]. The DCA has $UO_2$ (1.2 wt%) or $PuO_2$-$UO_2$ (1.2 or 1.49 wt%) fuel in a cluster form. The calculation has shown that the root-mean-square errors of the $k_{eff}$ and the void reactivity are within 0.55% $\delta$k and 0.25% $\delta$(1/k), respectively. The calculation of the assembly power distribution showed that the results are consistent with the measured value within 8.5% [35].

However, the experimental data that can be used for the validation of the DUPIC fuel physics design is limited because of the complexity of the fuel composition. Therefore, it is recommended to conduct a few physics experiments using either the actual DUPIC fuel or simulated DUPIC fuel. The simulated DUPIC fuel shall have the reference DUPIC fuel composition of $^{235}$U and $^{239}$Pu as well as some of the major fission products. The experiments shall include a measurement of the critical buckling, detailed reaction rates, and neutron density distributions across a fuel bundle with and without coolant in the channel.

### 4.2 Thermal-hydraulic Design Verification

The purpose of the fuel channel thermal-hydraulic design is to determine the heat removal capability in all the fuel channels and to meet the performance and safety criteria. For example, the fuel and fuel management should be designed such that the steam quality at the channel exit is less than 4% during a normal reactor operation to meet the minimum channel flow rate design requirement [36]. For the thermal-hydraulic design of the CANDU fuel, the NUCIRC, a single channel analysis code is typically used to estimate the channel flow rate, pressure drop, critical channel power and the channel exit quality.

Because the radial power distribution of a fuel bundle and the axial power distribution in a fuel channel for the DUPIC fuel are different from those for the standard 37-element fuel and 43-element CANFLEX natural uranium fuel, it is recommended to develop a new $x_c$-$L_b$ correlation of the NUCIRC code for various radial power and non-uniform axial power distributions [37]. Based on the water critical heat flux test results, the NUCIRC code shall be validated for the thermal-hydraulic design of the DUPIC fuel.

### 4.3 Mechanical Design Verification

The DUPIC fuel shall be designed to be mechanically compatible with the primary heat transport system, fuel channel, fuel handling system, and the fuel management system. Because the DUPIC fuel bundle adopts the CANFLEX geometry, it is believed that mechanical compatibility can be verified by either an experimental or an analytical method. For compatibility with the primary heat transport system, the pressure tube fretting and spacer grid fretting experiments shall be required, while analytical methods can be used for the end plate fatigue and pressure tube corrosion. However, it is expected that the pressure drop tests between two fuel bundles and over each fuel bundle string are not required.

For compatibility with the fuel channel, the clearance between the fuel bundle string and the shield plug can be analytically evaluated, while the experiments for the fuel bundle/channel interaction and path through the pressure tube rolled joint are not necessary. For compatibility with the fuel handling system, it is expected that the cross-flow fretting experiment is required, while experiments for compatibility with the fueling machine sensors and separators, fuel bundle strength against the hydraulic drag during refueling, and the fuel bundle strength against the refueling impact are not necessary.

In order to assess the in-core integrity and geometrical stability of the DUPIC fuel bundle, the high power and power ramp irradiation tests are required, and these are used to produce the stress corrosion cracking (SCC) threshold curve of the DUPIC fuel, typically used to assess the integrity of the CANDU fuel during the normal and operational transients. It is also required to measure the melting temperature of the DUPIC fuel to estimate the safety margin of the DUPIC fuel, because the thermal conductivity of the DUPIC fuel is lower than that of the natural uranium fuel.

## 5. CONCLUSIONS

The DUPIC fuel cycle is a unique spent nuclear fuel management technology that can be implemented in South Korea. In the past, the Tandem fuel cycle development [38], which recycles mixed oxide fuel in a CANDU reactor through a reprocessing, was not successful. The Korea Hydro Nuclear Power also tried reprocessing outside of Korea, but this work was unsuccessful due to the increasing concern about proliferation and adherence to the non-proliferation treaty as it concerns the Korean peninsula. Nonetheless, the accumulation of spent fuel is an urgent issue that should be resolved. Therefore, a technology should be developed that can be implemented in Korea under the non-proliferation policy. The DUPIC fuel cycle is known to be the most representative example that has technically overcome the international and domestic restrictions involved with the Tandem fuel cycle.

Though it is yet too early to launch the commercialization of DUPIC fuel based on the basic DUPIC fuel technology currently developed, it is also true that the key technologies have been developed for the DUPIC fuel cycle. Therefore, it is expected that there should be no technical problems to develop commercial DUPIC fuel technology once the DUPIC fuel technology and its performance are demonstrated through a practical use of the DUPIC fuel, which will be an important turning point in the history of nuclear power development. By utilizing spent fuel via an internationally-proven proliferation-resistant technology, it is expected that the burden of spent fuel accumulation will be relieved not only in the domestic nuclear grid but also in the worldwide nuclear power industry.

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