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Nuclear Fuel Cycle Analysis and Simulation Tool (FAST)

2005. 6

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제 출 문

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CONTENTS

NOMENCLATURE

AAA : Advanced Accelerator Applications

ADS : Accelerated Driven System

AFCI : Advanced Fuel Cycle Initiatives

AGCR : Advanced Gas Cooled Reactor

BWR : Boiling Water Reactor

CANDU : Canadian Deuterium Uranium

CEA : French Atomic Energy Commission

CoreRatioMT : Core Ratio of MOX Fuel in PWR in MOX Thermal Option

CoreRatioMTO : Core Ratio of MOX Fuel in PWR in MOX Thermal Only Option

COSI : Simulation Software for a Pool of Reactors and Fuel Cycle Plant

DUPIC : Direct Use of Spent PWR Fuel in CANDU

DYMOND : Dynamic Model of Nuclear Development

EndYearMT : Ending Year of MOX Fuel in PWR in MOX Thermal Option

FAST : Nuclear Fuel Cycle Analysis and Simulation Tool

FCCG : Fuel Cycle Crosscut Group of GEN-IV Program

FirstStepA : Annual Grid Ratio Increase up to First Target Year of ADS in NPP Grid

FirstStepH : Annual Grid Ratio Increase up to First Target Year of HTGR in NPP Grid

FirstStepS : Annual Grid Ratio Increase up to First Target Year of SFR in NPP Grid

FirstTargA : ADS Grid Ratio in First Target Year of ADS in NPP Grid

FirstTargH : Heat Generation in First Target Year of HTGR in NPP Grid

FirstTargS : SFR Grid Ratio in First Target Year of SFR in NPP Grid

FR : Fast Reactor

GCR : Gas Cooled Reactor

GEN IV : Generation IV

HLW : High Level Waste

HTGR : High Temperature Gas Cooled Reactor

IAEA : International Atomic Energy Agency

ICECAT : Integrated Cost and Needs of the Fuel Cycle Analysis Tool

INPRO : International Project on Innovative Nuclear Reactors and Fuel Cycles

IntYearA : Introduction Year of ADS in NPP Grid

IntYearH : Introduction Year of HTGR in NPP Grid

IntYearMT : Introduction Year of MOX Fuel in PWR in MOX Thermal Option

IntYearMTO : Introduction Year of MOX Fuel in PWR in MOX Thermal Only Option

IntYearS : Introduction Year of SFR in NPP Grid

KAERI : Korea Atomic Energy Research Institute

KALIMER : Korea Advanced Liquid Metal Reactor

LANL : Los Alamos National Laboratory

LWR : Light Water Reactor

MOCIE : Ministry of Commerce, Industry and Energy of Korea

MOX : Mixed Oxide

NFC : Nuclear Fuel Cycle

OECD/NEA : Nuclear Energy Agency of Organization for Economic Co-operation and Development

PGFHTGR : Power Generation Fraction of HTGR

PHWR : Pressurized Heavy Water Reactor

PowerGenScenario : Power Generation Scenario

PWR : Pressurized Water Reactor

ReactorStra : Reactor Strategies

SecondStepA : Annual Grid Ratio Increase up to Second Target Year of ADS in NPP Grid

SecondStepH : Annual Gird Ratio Increase up to Second Target Year of HTGR in NPP Grid

SecondStepS : Annual Grid Ratio Increase up to Second Target Year of SFR in NPP Grid

SecondTargA : ADS Grid Ratio in Second Target Year of ADS in NPP Grid

SecondTargH : Heat Generation in Second Target Year of HTGR in NPP Grid

SecondTargS : SFR Grid Ratio in Second Target Year of SFR in NPP Grid

SFR : Sodium Fast Reactor

ShareADS : Power Ratio of ADS

ShareCANDU : Power Ratio of CANDU

ShareHTGR : Power Ratio of HTGR

SharePWR : Power Ratio of PWR

ShareSFR : Power Ratio of SFR

TargYearA : First Target Year of ADS in NPP Grid

TargYearH : First Target Year of HTGR in NPP Grid

TargYearS : First Target Year of SFR in NPP Grid

VISTA : Nuclear Fuel Cycle Simulation System

Chapter 1 Introduction

Recently, international programs to develop innovative nuclear energy systems, including both reactors and fuel cycle technologies, have been initiated under the names of the Generation IV International Forum (GIF) led by the United States [1] and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) organized by the International Atomic Energy Agency (IAEA) [2]. The Republic of Korea has been participating in both programs as well as performing its own comprehensive R&D programs in which several different reactor types, such as the sodium fast reactor (SFR), the high temperature gas cooled reactor (HTGR) and the accelerator driven system (ADS), are taken into consideration.

As a result of all these vigorous research activities, future nuclear energy systems in most countries will consist of various mixes of nuclear reactors and fuel cycle technologies. Although such a situation requires a dynamic systems analysis which includes all the aspects of economics, safety, proliferation and radioactive waste, there have been little efforts in that regard. With this realization, the Korea Atomic Energy Research Institute (KAERI) has recently developed a computer program for simulating various nuclear fuel cycles that have potential application in Korea, and thus making it easy to understand policy implications when such fuel cycles are deployed in this country.

The report intends to describe this computer program, the Nuclear Fuel Cycle Analysis and Simulation Tool (FAST) which now allows a mass flow analysis and is being extended to include cost and environmental analyses. Results of several tryout-simulations from the FAST are also discussed in the report in order to demonstrate how this program works. The users are advised that, since it has not yet reached the testing and benchmarking stage, the results discussed here cannot be certified. With completion of this program, however, it is hoped to contribute to the decision making process of shaping the future nuclear fuel cycle development paths in Korea.

Chapter 2 Previously Developed Nuclear Fuel Cycle System Analysis Tools

In order to assess the viability of nuclear power plants and their fuel cycles, the estimation of their material flows are firstly required. Currently, several computer tools are available for such estimation: the Nuclear Fuel Cycle Simulation System (VISTA) which has been developed by the IAEA [3], the Integrated Cost and Needs of the Fuel Cycle Analysis Tool (ICECAT) by the OECD/NEA [4], the Dynamic Model of Nuclear Development (DYMOND) by the GIF [5], the Simulation Software for a Pool of Reactors and Fuel Cycle Plant (COSI) by the CEA [6], the system analysis modeling for the AFCI program [7], and the Dynamic Analysis of Nuclear Energy System Strategies (DANESS) by the Argonne National Laboratory (ANL) [8].

This chapter introduces briefly these programs.

2.1 VISTA

The Nuclear Fuel Cycle Simulation System (VISTA) [3] is a scenario-based computer program for the estimation of fuel cycle service requirements. This simulation tool was originally used to assess the different scenarios outlined by the Working Groups of the International Symposium on *Nuclear Fuel Cycle and Reactors Strategies: Adjusting to New Realities* held from 3 to 6 June 1997 in Vienna, Austria.

The VISTA takes three groups of parameters as input: strategy parameters, fuel parameters and control parameters. The strategy parameters include nuclear capacity variants, reprocessing-recycling strategies, reactor type mixtures and load factors for each type of reactor. The reactor types taken into consideration in this program are the pressurized water reactor (PWR), the boiling water reactor (BWR), the pressurized heavy water reactor (PHWR), the advanced gas cooled reactor (AGCR), the gas cooled reactor (GCR), WWER, all of which are currently being commercially operated. The fuel parameters include average discharge burnup, average initial enrichment and average tail assay of enrichment plants on an annual basis. The control parameters are share of MOX (Mixed Oxide) fuel in reactor fuel, lead and lag times for different processes and the number of spent fuel reprocessing cycles.

Taking these three groups of parameters as input, the VISTA gives various output data including natural uranium, conversion and enrichment service requirements, spent fuel arisings, actinide contents of spent fuel and reprocessing service requirements.

The current version of the simulation tool is in MS Excel spreadsheet format. In the future the program will be converted into a web application, and made available for the use of interested Member States and specialists with their own scenarios.

2.2 ICECAT

The Integrated Cost and Needs of the Fuel Cycle Analysis Tool (ICECAT) is a program designed for simulating the future development of installed nuclear capacity in order to forecast demand for fuel as well as the associated costs [4]. It has been created by the OECD/NEA for its own use and potentially that of experts from interested Member countries.

The ICECAT, a program driven by energy demand, operates under the iThink software environment. Data on the existing reactors and scenarios for energy demand are stored in a MS Access formatting database, while output files are formatted in the MS Excel to allow users easily update this database with the most recent data available.

Currently, the ICECAT offers a choice of 7 fuel cycle scenarios:

- LWR (Light Water Reactor) Once-through-Cycle (OTC)
- LWR mono Pu recycling
- LWR multi Pu recycling
- LWR OTC + FR (Fast Reactor) multi Pu recycling
- LWR mono Pu recycling + FR multi Pu recycling
- LWR mono Pu recycling + FR multi Pu recycling + Pu feedback
- Direct use of Spent PWR Fuel in CANDU (DUPIC)

The current version of the ICECAT is not final and thus, continues to be modified. Other fuel cycles, such as the thorium fuel cycle and fast breeder fuel cycle, as well as the management of minor actinide are being added, while the ADS option is considered to be added in the near future.

2.3 DYMOND

The Dynamic Model of Nuclear Development (DYMOND) was originally created by Mr. Anton Moisseytsev, a Texas A&M graduate student, as a summer project when he was employed by the Reactor Analysis and Engineering Division at the Argonne National Laboratory (ANL) in 2001 [5]. It has been modified and extended by Dr. Latif Yacout at ANL by adding the thorium cycle option in this model. Later, it had used in the GEN IV Fuel Cycle Crosscut Group (FCCG) [5].

The DYMOND employs the iThink dynamic modeling platform to model 100-year dynamic evolution scenarios for postulated global nuclear energy parks. The scenarios use the worldwide deployments of fuel cycle facilities and power plants in the year of 2000 as initial conditions.

The VISTA gives various output data including the front and back-end mass flows, inventories of spent fuel, plutonium and minor actinides in interim storage. The code also quantify the scales of deployment of mining/milling, enrichment services, fuel fabrication plants, reprocessing capacities, required capacities of interim storage facilities and final disposal repositories.

2.4 COSI

The Simulation Software for a Pool of Reactors and Fuel Cycle Plant (COSI) was originally developed by the French Atomic Energy Commission (CEA) in 1991 and used for the material flow calculation in the OECD/NEA report, *Trends in the Nuclear Fuel Cycle*, published in 2001 [6].

2.5 AFCI System Analysis Modeling

The Los Alamos National Laboratory (LANL) has been using a series of computer models in order to assess/evaluate the future, advanced nuclear fuel cycle scenarios in terms of three main issues of economics, environmental impacts and proliferation risk. The figure below illustrates the relationships among these models of DELTA, FCOPT and NFCSim [7].

The DELTA, a steady-state (equilibrium) model, evaluates scenarios based on key performance indicators such as cost, waste mitigation, proliferation risk and resource utilization. Based on these equilibrium analyses, optimizations and simulations are performed by use of FCOPT and NFCSim. The FCOPT, created by the Los Alamos National Laboratory, is to optimize a comprehensive set of nuclear fuel cycle options. Using Linear Programming (LP) methods and taking the time factor into consideration, the FCOPT tracks a number of nuclear materials and related processes. Based on the demand for nuclear-electric energy, the FCOPT considers available technologies and then determine the optimal nuclear fuel cycle scenario.

Together with the above two models, the NFC Simulation (NFCSim) model completes the troika of systems models used for the AFCI program. The NFCSim has been benchmarked by the COSI under an NFC modeling benchmark collaboration between the CEA and the LANL.

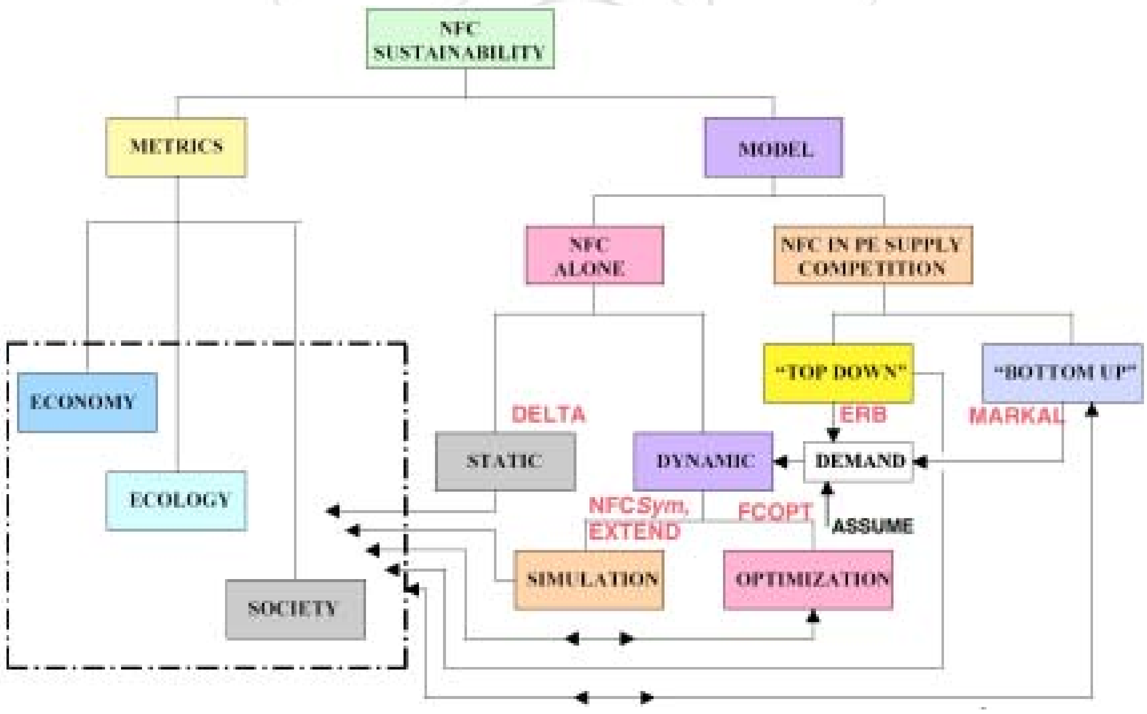

Fig. 1 Spectrum of System Analysis Modeling of AFCI Program

2.6 DANESS

The Dynamic Analysis of Nuclear Energy System Strategies (DANESS) is an integrated nuclear process model intended for the dynamic analysis of multiple development paths for nuclear energy systems which has recently gained interest worldwide [8]. Using of iThink software and being accompanied with a MS Access database and the MS Excel, the DANESS allows users to simulate all aspects of a varying mix of reactor and fuel types. According to the ANL, the model allows simulating up to 20 different reactor types and up to 20 different fuel types in one simulation. In its current version, the code only allows mass flow analysis and economics, but is being extended to include life-cycle analysis data, non-proliferation metrics and non-nuclear energy sources.

Chapter 3 Nuclear Fuel Cycle Analysis and Simulation Tool

3.1 Overview

As described in Chapter 2, several international bodies have developed computer tools that calculate material flows of various nuclear fuel cycles. Unfortunately, being developed for their own uses, all of those are not commercially available. Besides, none of those are useful when considered future nuclear fuel cycles that have potential application in Korea. With this realization, the Korea Atomic Energy Research Institute (KAERI) has recently developed its own computer program, the Nuclear Fuel Cycle Analysis and Simulation Tool (FAST).

The Nuclear Fuel Cycle Analysis and Simulation Tool (FAST) is a computer program that estimates quantity requirements for each nuclear fuel cycle step, such as ore mining, conversion, enrichment and fuel fabrication. Such estimation could be made on a yearly basis, or over a particular time frame, up to the year of 2100.

The Fig. 2 gives an overview of the Nuclear Fuel Cycle Analysis and Simulation Tool (FAST) linked with the FORECAST, which projects a future nuclear energy need in Korea and whose output is transferred automatically to the FAST. The boxes at the top and the bottom sides of the figure show input and output parameters, respectively.

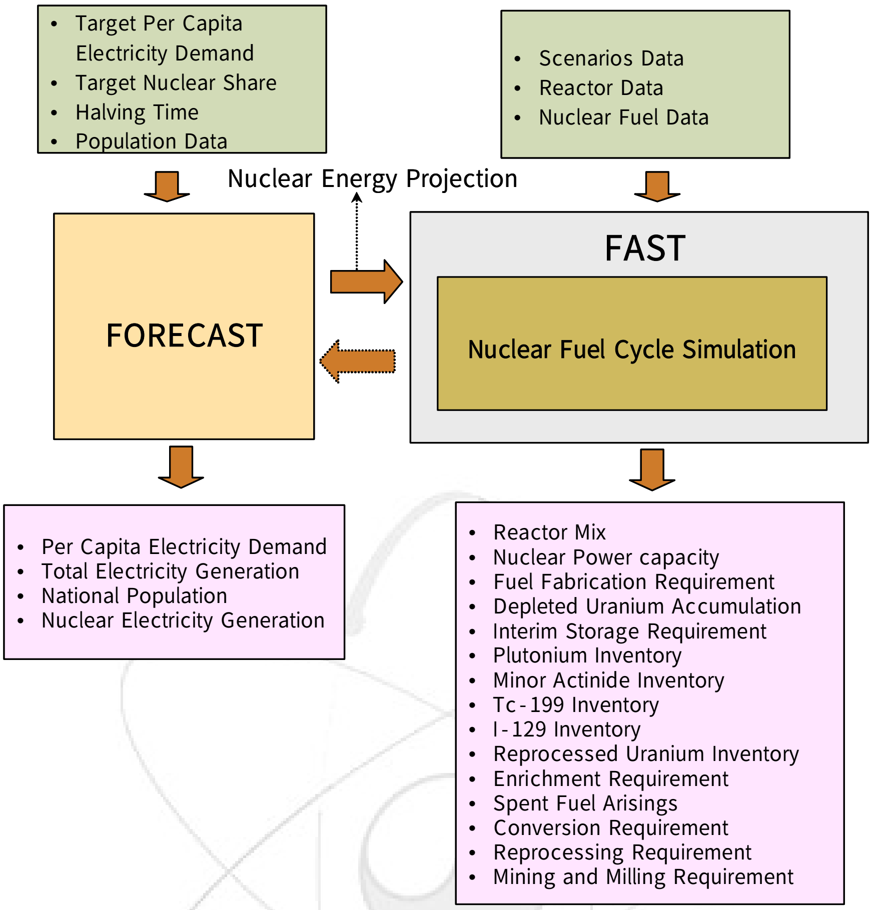

Fig. 2 Schematic Diagram of FAST Linked with FORECAST

3.2 FORECAST: Projection of a Future Nuclear Energy Need

A study on the nuclear fuel cycle must begin with an estimation of the growth in nuclear power during a time frame considered. The growth in nuclear power will depend on both nuclear power market share and total electricity generation. The total electricity generation will also depend on per capita electricity demand in a specific country [9, 10]. This logic, as shown in Fig. 3, is used in the FORECAST.

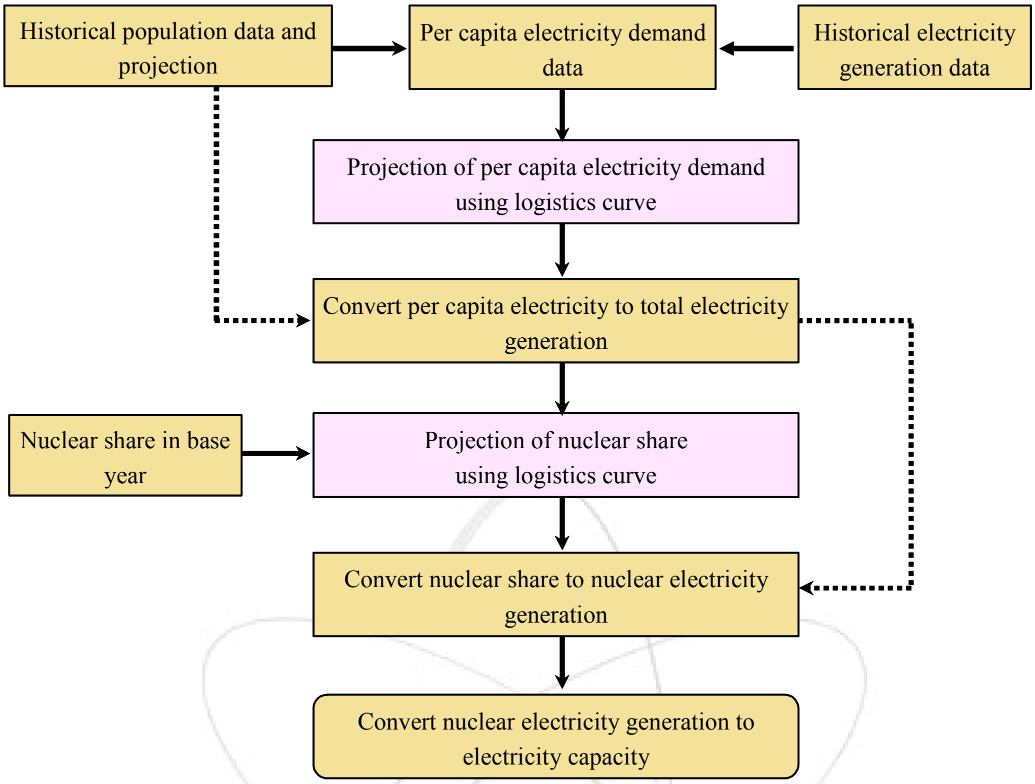

Fig. 3 Calculation Procedure of FORECAST Model

The formula for estimating the electricity demand per capita is given as follows:

$$E_t = \frac{E_\infty}{1 + e^{-(a_E + b_E T)}} \qquad (1)$$

Where;

E_t: electricity demand per capita in year t

E_∞ : asymptotic limit for the demand E_t

T: time in years since the base year

a_E and b_E,: parameters estimated by the regression

$$a_E + b_E T = \log \frac{E_t}{E_\infty - E_t} \qquad (2)$$

Given that the asymptotic electricity consumption per capita and the historically-based electricity consumption per capita, the parameters, a_E and b_E, can be estimated by the regression analysis. As a result, electricity demand per capita can be projected from the equation (1).

To project nuclear electricity market share, the logistics curve model [9] is used. The logistics curve for this purpose is defined by an initial point and two parameters, nuclear electricity market share and halving time (eq. (3)). The initial point is the nuclear electricity market share of the base year. The first parameter is the nuclear electricity market share at a given date in the long term or the asymptotic limit. The second parameter, halving time, is defined as the time taken, counted from the base year, for nuclear electricity market share of total electricity generation to reach a value half-way between the value of the base year and the asymptotic limit.

$$S_t^N = \frac{S_\infty^N}{1 + e^{-(a_N + b_N T)}} \qquad (3)$$

Where

S_t^N : nuclear electricity market share of total electricity generation in year t

T: time in years since the base year

S_∞^N : asymptotic limit for nuclear electricity market share S_t^N

a_E and b_E : parameters estimated by the regression

$$a_N = \log_e \frac{S_0^N}{S_\infty^N - S_0^N} \tag{4}$$

$$b_N = \frac{1}{H_N} \log_e \frac{S_0^N + S_\infty^N}{S_0^N} \tag{5}$$

S_0^N : nuclear electricity market share of total electricity generation in the base year

H_N : halving time, i.e., the time taken, counted from the base year, for nuclear electricity market share of total electricity generation to reach a value half-way between the value at the base year and the asymptotic limit.

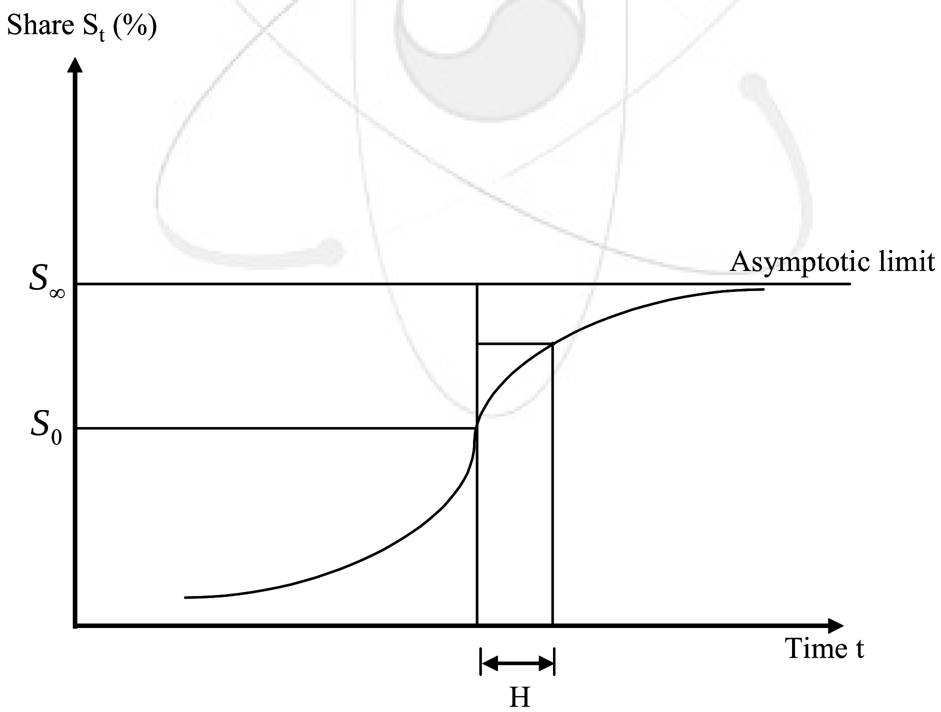

Fig. 4 Logistics Curve for Forecasting Nuclear Power Capacity

3.3 FAST: Estimation of Fuel Cycle Service Requirements & Calculating of Material Flows

3.3.1 Scenario Description

In the FAST, nuclear fuel cycles are categorized into 11 scenario groups, combinations of PWR, CANDU, SFR, ADS and HTGR. These reactor types taken into consideration are those in operation now (PWR and CANDU) or those that have potential application in Korea. All the scenarios groups are based on the current power plant grid and the 2nd Basic Plan of Electricity Demand and Supply [11] which was announced by the Korean Ministry of Commerce, Industry and Energy (MOCIE) in 2004. All the related data is pre-provided in its database.

3.3.1.1 Scenario Group 1 (PWR + CANDU (Constant ratio))

Scenario group 1 considers a situation in which only the existing reactor types of PWRs and CANDU reactors would be operated in Korea. As all the other scenarios, the group 1 is based on the 2nd Long-term Nuclear Power Generation Plan which blueprints the nuclear power situation in Korea up to the year of 2017. After 2017, this group assumes that the ratio of CANDU reactors to total nuclear electricity capacity continues to be constant until 2100. The ratio, according to the 2^{nd} Plan, will be 0.102 in 2017 with the completion of the plan.

The spent PWR and CANDU fuels are permanently disposed of in centralized storage facilities.

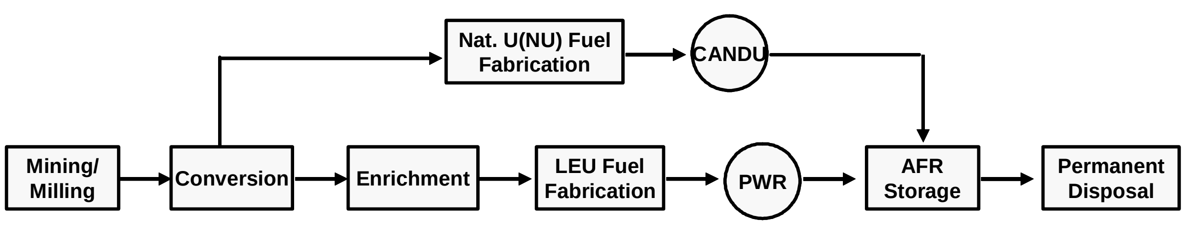

Fig. 5 Schematic Diagram of Scenario Group 1

3.3.1.2 Scenario Group 2 (PWR + CANDU (Phase out))

Scenario group 2 is identical with the Group 1 except that the ratio of CANDU reactors to total nuclear electricity capacity is phased out. This means a new CANDU reactor will be no more constructed up to the year of 2100.

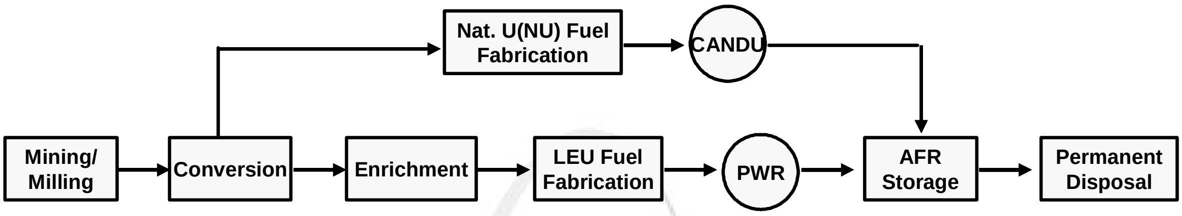

Fig. 6 Schematic Diagram of Scenario Group 2

3.3.1.3 Scenario Group 3 (PWR + CANDU + HTGR)

In scenario group 3, the HTGR, which is used both in hydrogen production and electricity generation, is taken consideration in addition to the PWR and CANDU reactors. As in Group 2, the CANDU reactors are phased out in this group after the end of their life time.

3.3.1.4 Scenario Group 4 (PWR (Thermal recycle) + CANDU)

Scenario group 4 deals with the thermal recycle in which the PWR spent fuel is wet-reprocessed and reused as MOX fuel in the existing PWRs while the CANDU spent fuel is permanently disposed of without reprocessing. As in Groups 2 and 3, the CANDU reactors are phased out after the end of their life time. The remaining PWR and MOX spent fuels are permanently disposed of at centralized storage facilities.

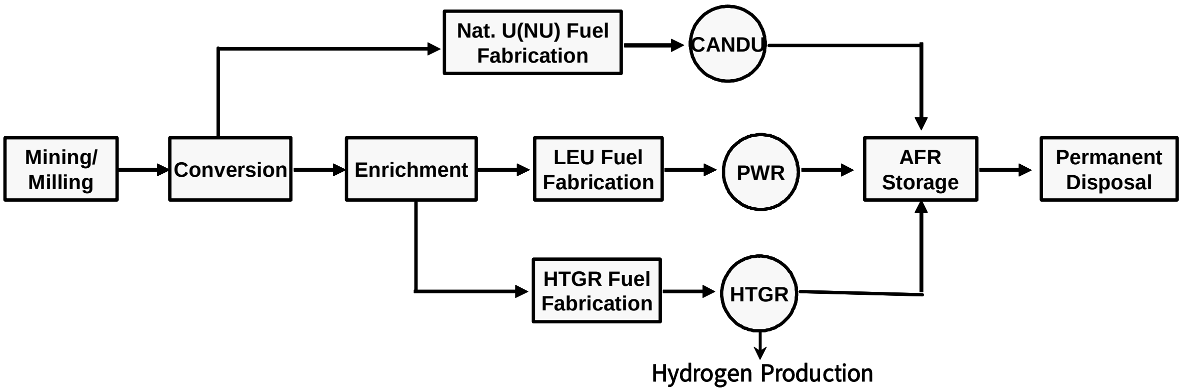

Fig. 7 Schematic Diagram of Scenario Group 3

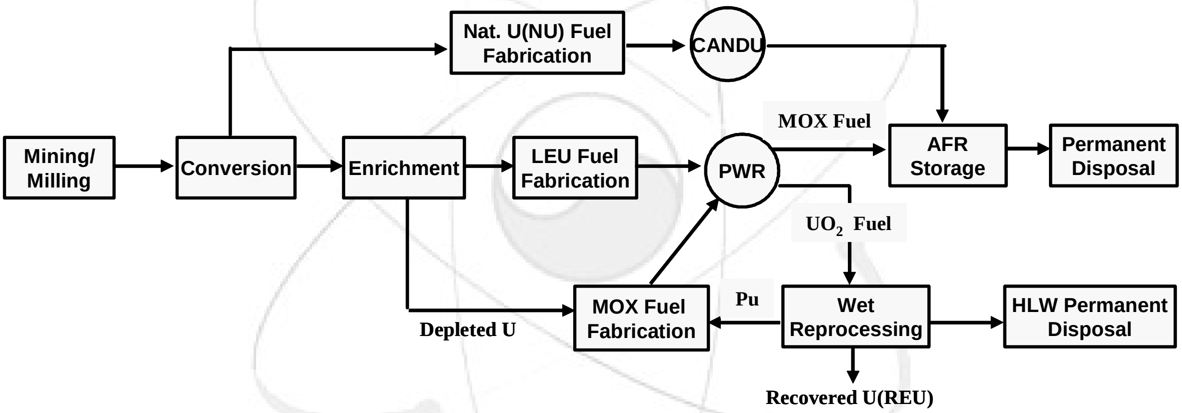

Fig. 8 Schematic Diagram of Scenario Group 4

3.3.1.5 Scenario Group 5 (PWR + CANDU + SFR)

Scenario group 5 introduces to Korea the SFR in which reprocessed PWR spent fuel is used. The spent SFR fuel can also be reprocessed and continuously recycled in the SFR. The use of blanket fuel in the SFR is optional. On the other hand, CANDU spent fuel is not reprocessed and permanently disposed of at centralized storage facilities. The CANDU reactors, as in the Groups of 2, 3 and 4, are phased out after the end of their life time.

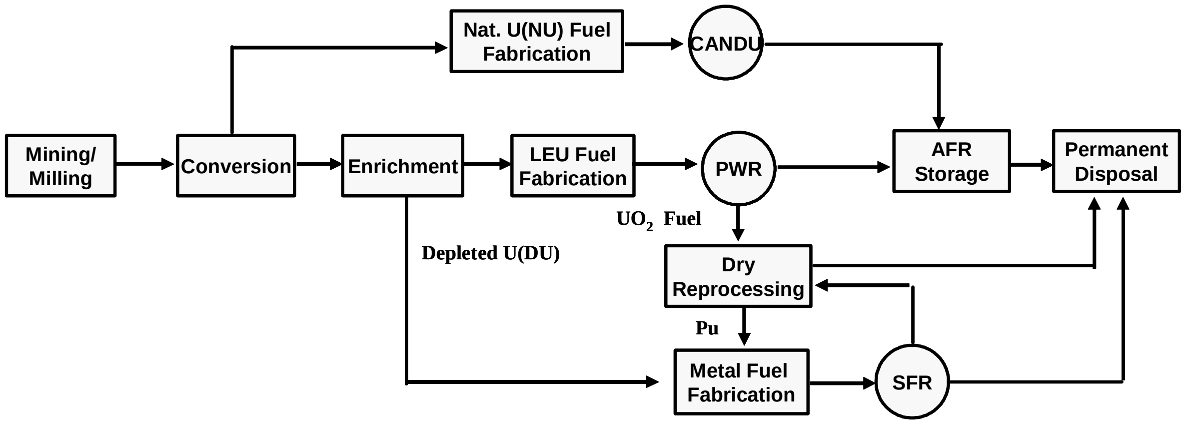

Fig. 9 Schematic Diagram of Scenario Group 5

3.3.1.6 Scenario Group 6 (PWR + CANDU + HTGR + SFR)

In this scenario, the HTGR, both for hydrogen production and electricity generation, is introduced. Except that the HTGR is taken consideration, the deployment of Scenario Group 6 is the same as the Group 5 in which both PWR and SFR spent fuels are reprocessed and reused in the SFR.

3.3.1.7 Scenario Group 7 (PWR + CANDU + ADS + SFR)

In scenario group 7, the ADS is included for transmuting actinide and long-lived fission products. As in the previous scenario, both PWR and SFR spent fuels are reprocessed and reused in the SFR. The PWR spent fuel can also be reused in the ADS. Tc and I from PWR and SFR spent fuels are fabricated as a target fuel and transmuted in the ADS.

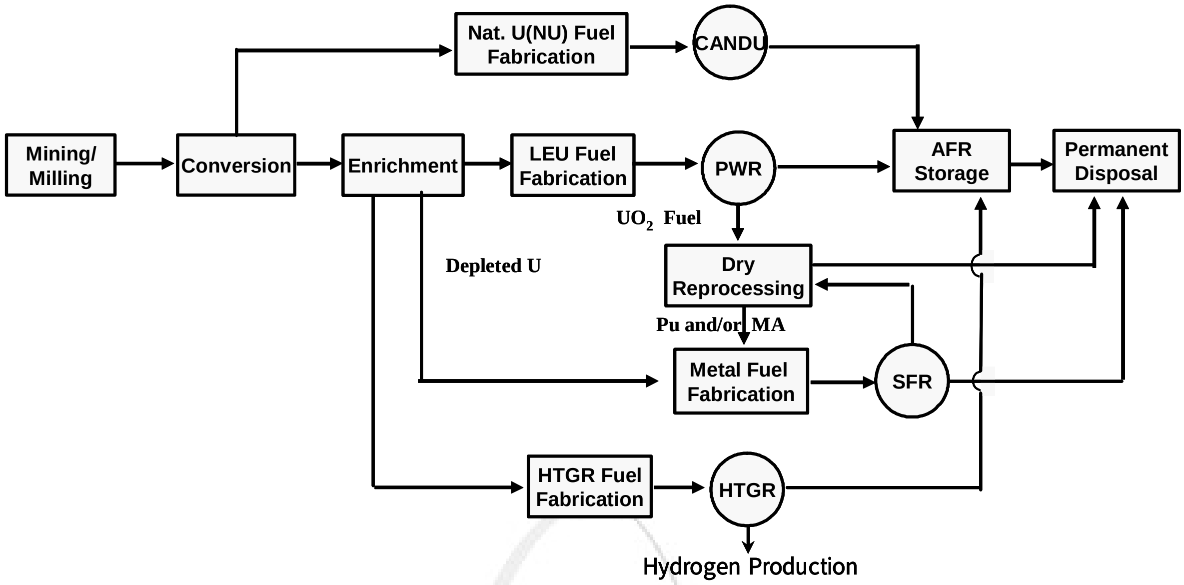

Fig. 10 Schematic Diagram of Scenario Group 6

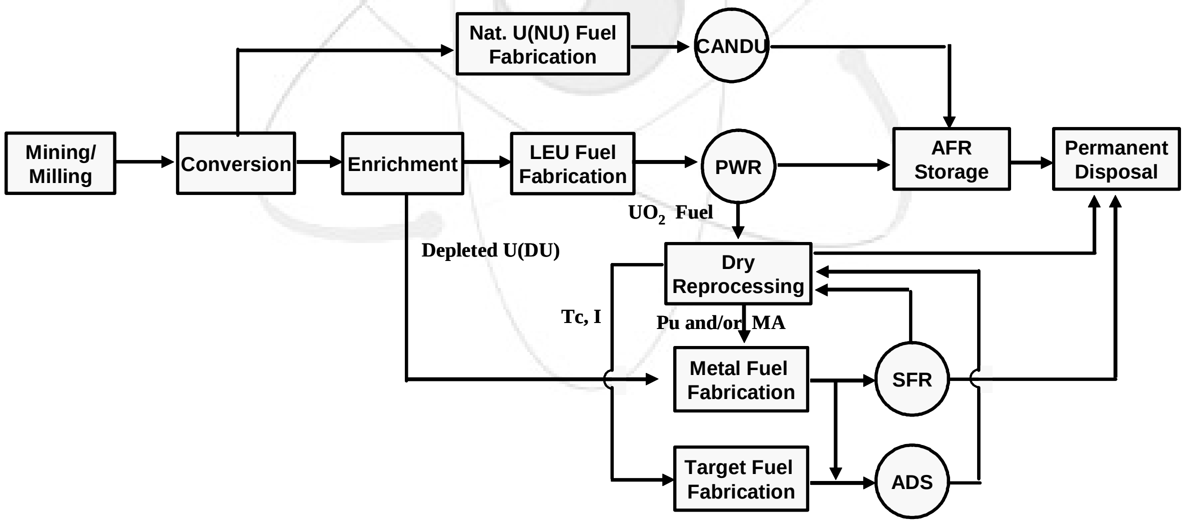

Fig. 11 Schematic Diagram of Scenario Group 7

3.3.1.8 Scenario Group 8 (PWR + CANDU + ADS)

In scenario group 8, the PWR, CANDU reactors and the ADS are deployed. PWR and ADS spent fuels are reprocessed and reused in the ADS, instead of the SFR and Tc and I in the PWR spent fuel are fabricated as a target fuel and transmuted in the ADS.

3.3.1.9 Scenario Group 9 (PWR + CANDU + HTGR + ADS)

In scenario group 9, the PWR, CANDU reactors, the HTGR and the ADS are deployed. As in the previous scenario group 8, PWR and ADS spent fuel is reprocessed and reused in the ADS. Tc and I in the PWR spent fuel are fabricated as a target fuel and transmuted in the ADS. The HTGR, both for hydrogen production and electricity generation, is also deployed.

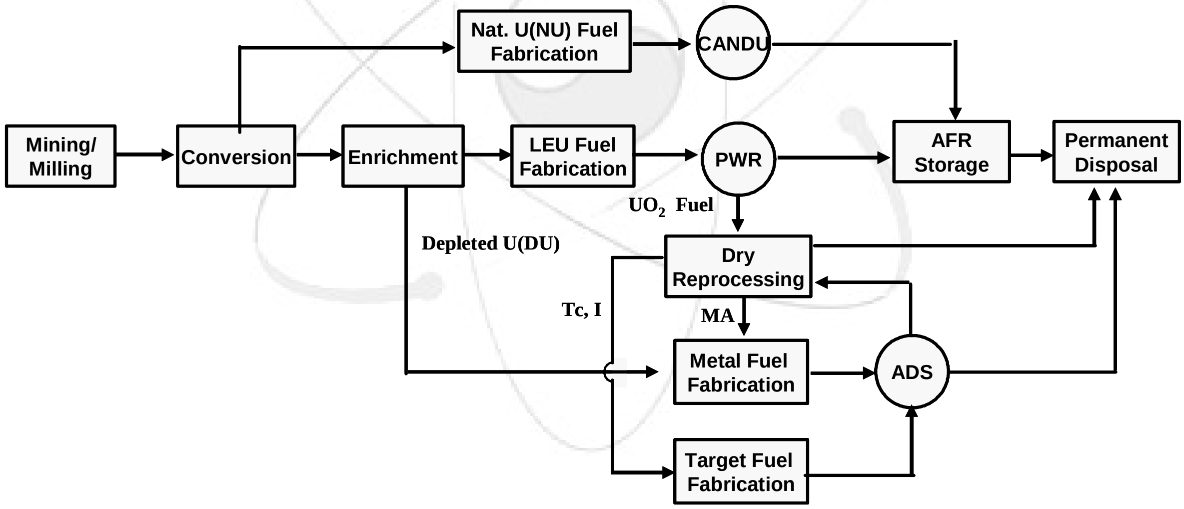

Fig. 12 Schematic Diagram of Scenario Group 8

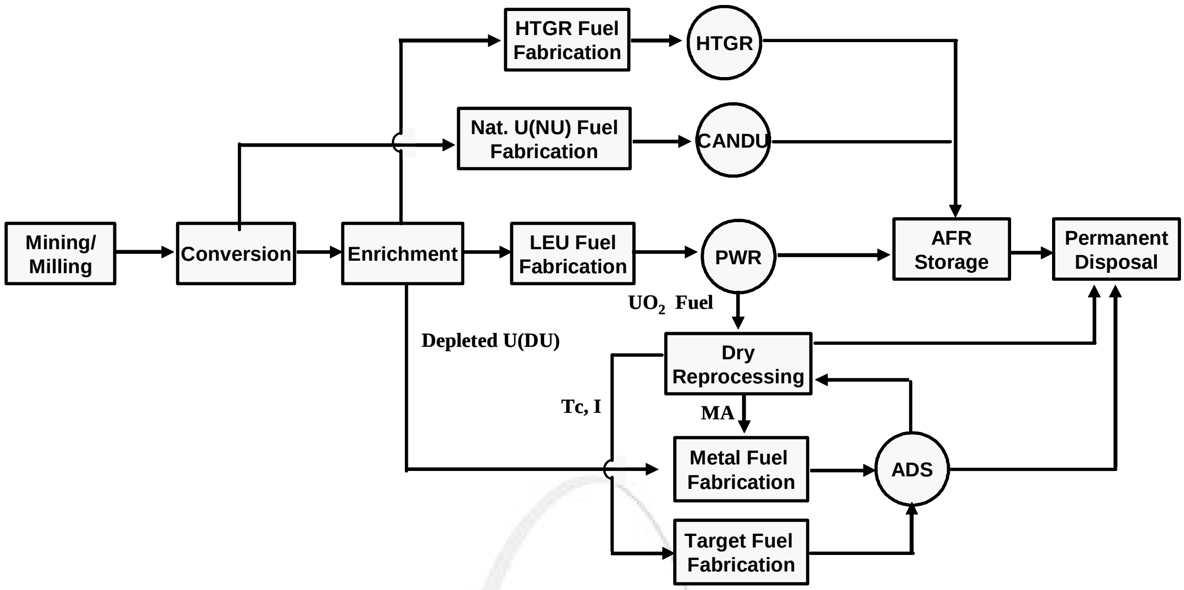

Fig. 13 Schematic Diagram of Scenario Group 9

3.3.1.10 Scenario Group 10 (PWR + CANDU + HTGR + SFR + ADS)

In scenario group 10, all the reactor types that have potential application in Korea are deployed. PWR spent fuel is reprocessed and reused in the SFR and/or the ADS, while spent fuels from the SFR and the ADS are continuously recycled in the ADS. Tc and I in the PWR spent fuel are fabricated as a target fuel and transmuted in the ADS. The HTGR, both for hydrogen production and electricity generation, is also deployed.

3.3.1.11 Scenario Group 11 (PWR (Thermal Recycle) + CANDU + HTGR + ADS)

This is the same scenario as the previous group except that the thermal recycle of MOX fuel in the PWR is added. It is similar to the “double strata” concept of Japan and Europe: At first, the thermal recycle using MOX fuel in the PWR is performed. Then, actinide and long-lived fission products are burned in the ADS and/or the SFR.

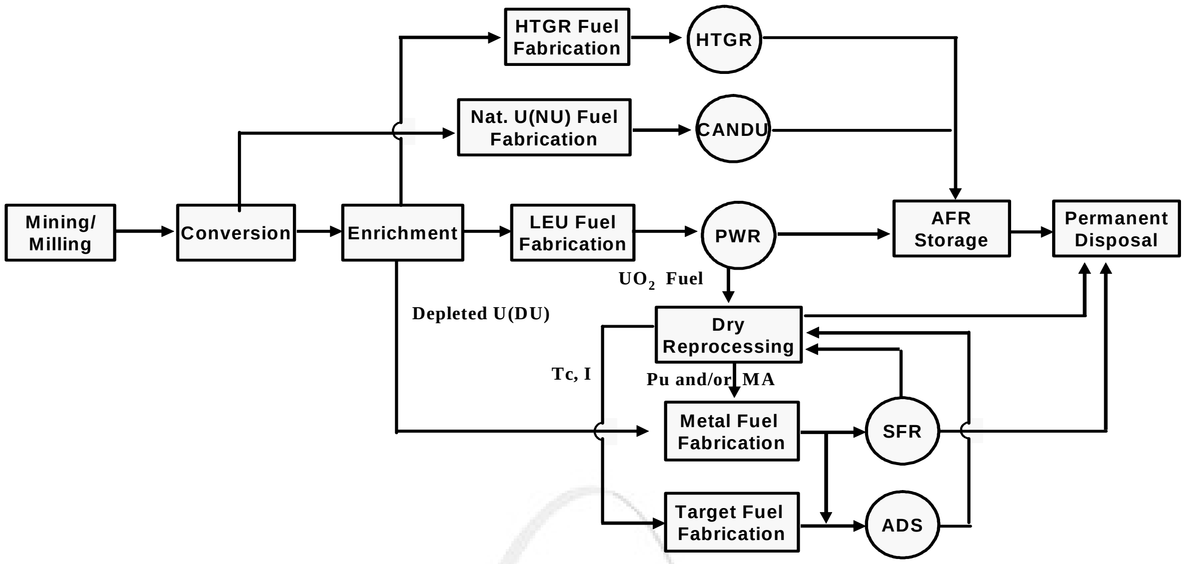

Fig. 14 Schematic Diagram of Scenario Group 10

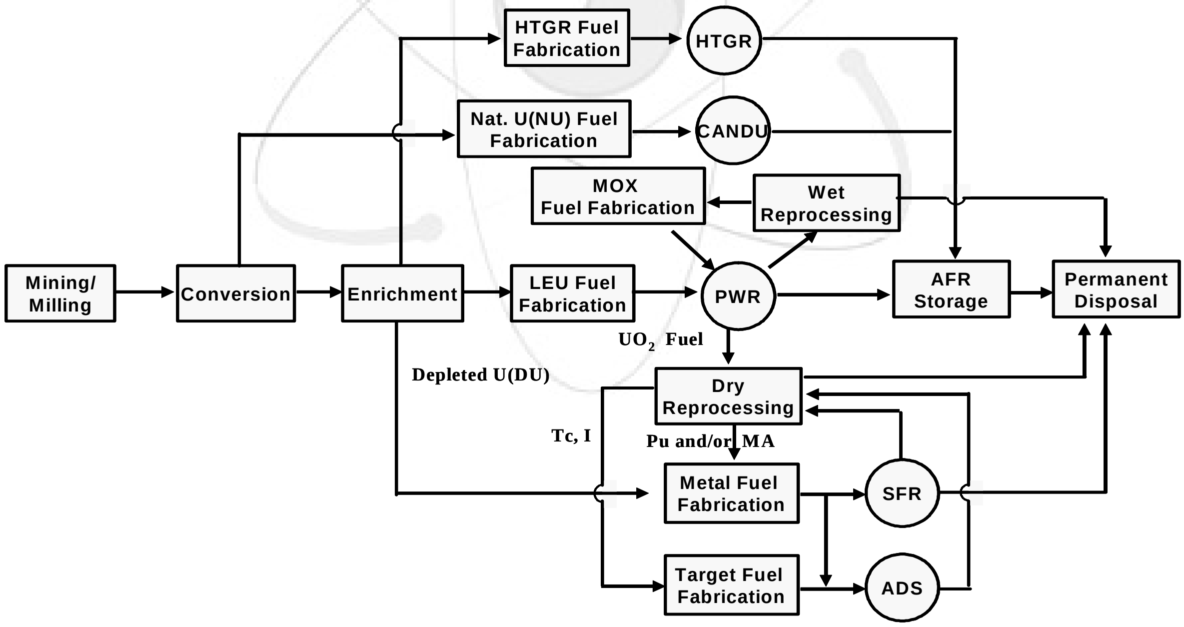

Fig. 15 Schematic Diagram of Scenario Group 11

3.3.2 Reactor Grid Calculation

Given that the projection of the nuclear power generation is determined, the power generations are allocated to each reactor grid. The reactor grids except for PWR and CANDU are inputted by use of two step target grid. With two step target grid ratio, each reactor grid deployed in Korean nuclear reactor system can be automatically calculated year by year with following rationale.

The grid ratio means the electricity capacity portion of a specific nuclear type of all nuclear reactor types deployed. Therefore, the sum of nuclear grid ratios will be as followings:

ShareHTGR + SharePWR + ShareCANDU + ShareADS + ShareSFR = 1

Where,

-ShareHTGR : the ratio of HTGR power capacity,

-SharePWR : the ratio of PWR power capacity,

-ShareCANDU : the ratio of CANDU power capacity,

-ShareADS : the ratio of ADS power capacity, and

-ShareSFR : the ratio of SFR power capacity.

HTGR Grid Ratio

The HTGR grid is calculated from the introduction year to first step target year with constant increase ratio. Annual increase of power (GWe/year) up to first target year of HTGR is as follows:

FirstStepH = FirstTargH / (TargYearH – IntYearH + 1) x Efficiency

Where,

-FirstStepH : Annual increase (GWe/year) up to first target year of HTGR,

-FirstTargH : Heat generation (GWth) in first target year of HTGR,

-IntYearH : Introduction year of HTGR in NPP Grid, and

-TargYearH : First target year of HTGR in NPP Grid.

Annual increase of power up to second target year of HTGR is also calculated with similar concept as follows:

SecondStepH = (SecondTargH - FirstTargH) / (SecondTargYearH - TargYearH) x Efficiency

Where,

-SecondStepH : Annual increase (GWe/year) up to second target year of HTGR,

-SecondTargH : Heat generation (GWth) in second target year of HTGR, and

-SecondTargYearH : Second target year of HTGR.

The main purpose of HTGR introduction is to make hydrogen and for the case heat generation (GWth) is important. Actually annual increase (GWe/year) for the electricity generation, therefore, power generation fraction of HTGR to hydrogen production should be considered.

The grid ratio for the HTGR is described as follows:

$$ShareHTGR = 0 \qquad (Year < IntYearH, \text{ or } Year > SecondTargYearH)$$

$$ShareHTGR = \sum_{year} FirstStepH \times PGFHTGR \qquad (IntYearH \le Year \le TargYearH)$$

$$ShareHTGR = \sum_{year} SecondStepH \times PGFHTGR \qquad (TargYearH < Year < SecondTargYearH)$$

Where, the PGFHTGR is power generation fraction of HTGR.

SFR Grid Ratio

The SFR grid is also calculated from the introduction year to first step target year with constant increase ratio. Annual grid ratio increase up to first target year of SFR in NPP Grid is as follows:

$$FirstStepS = FirstTargS / (TargYearS - IntYearS + 1) \times Efficiency$$

Where,

-FirstStepS : Annual grid ratio increase (%) up to first target year of SFR in NPP grid,

-FirstTargS : SFR grid ratio (%) in first target year of SFR,

-IntYearS : Introduction year of SFR in NPP grid, and

-TargYearS : First target year of SFR in NPP Grid.

Annual grid ratio increase up to second target year of SFR in NPP grid is also calculated as follows:

SecondStepS = (SecondTargS - FirstTargS) / (SecondTargYearS - TargYearS) x Efficiency

Where,

-SecondStepS : Annual grid ratio increase (%) up to Second target year of SFR in NPP grid,

-SecondTargS : SFR grid ratio (%) in second target year of SFR, and

-SecondTargYearS : Second target year of SFR.

The grid ratio for the SFR can be described as follows:

$$\text{ShareSFR} = 0 \qquad (\text{Year} < \text{IntYearS, or Year} > \text{SecondTargYearS})$$

$$\text{ShareSFR} = \sum_{year} \text{FirstStepS} \qquad (\text{IntYearS} \leq \text{Year} \leq \text{TargYearS})$$

$$\text{ShareSFR} = \sum_{year} \text{SecondStepS} \qquad (\text{TargYearS} < \text{Year} < \text{SecondTargYearS})$$

ADS Grid Ratio

The SFR grid is calculated by use of the same concept as the SFR case. Annual grid ratio increase up to first target year of ADS in NPP Grid is as follows:

FirstStepA = FirstTargA / (TargYearA – IntYearA + 1) x Efficiency

Where,

-FirstStepA : Annual grid ratio increase (%) up to first target year of ADS in NPP grid,

-FirstTargA : SFR grid ratio (%) in first target year of SFR,

-IntYearA : Introduction year of SFR in NPP grid, and

-TargYearA : First target year of SFR in NPP Grid.

Annual grid ratio increase up to second target year of ADS in NPP grid is also calculated as follows:

SecondStepA = (SecondTargA - FirstTargA) / (SecondTargYearA - TargYearA) x Efficiency

Where,

-SecondStepA : Annual grid ratio increase (%) up to second target year of ADS in NPP grid,

-SecondTargA : ADS grid ratio (%) in second target year of ADS, and

-SecondTargYearA : Second target year of ADS.

The grid ratio for the ADS can be described as follows:

ShareADS = 0 (Year <IntYearA, or Year > SecondTargYearA)

$$\text{ShareADS} = \sum_{year} \text{FirstStepA} \qquad (\text{IntYearA} \le \text{Year} \le \text{TargYearA})$$

$$\text{ShareADS} = \sum_{year} \text{SecondStepA} \qquad (\text{TargYearA} < \text{Year} < \text{SecondTargYearA})$$

PWR and CANDU Reactor Grid Ratio

For the CANDU reactor grid, two cases are assumed considering the 2nd Basic Plan of Electricity Demand and Supply which includes plant construction plans up to 2017. One is that the CANDU reactor grid ratio remains constant after 2017. The other is that after 2017 CANDU reactor is phased out.

Finally, the PWR grid ratio will be calculated by the following equation.

SharePWR = 1 – (ShareHTGR + ShareCANDU + ShareADS + ShareSFR)

3.3.3 Material Flow Calculation

The model is designed for estimating fuel cycle service requirements form uranium mining to final disposal. Such estimation is calculated on a yearly basis up to the year of 2100. The calculating method and the algorithm are described below in detail.

The input variables and their notations used in the model are as follows:

- Total nuclear capacity by year (NuclearCapacity (year)) (MWe)

- Ratio of nuclear power by reactor type by year and type (ShareType (type, year))(%) (Here, the "type" means reactor type)

- Average load factor by reactor type and year (LoadFactor (type, year)) (%)

- Average load factor by all reactor type (AvgLoadFactor) (%)

- Average thermal efficiency by reactor type and year (Efficiency (type, year)) (%)

- Average discharge burnup by reactor type and year (Burnup (type, year, ft)) (GWd/MtU) (Here, the "ft" means fuel type, and it is assumed that PWR fuel has different-typed burnup and initial enrichment)

- Consumption ratio of heavy metal during burning in reactor (ConsumpRate (type, year, ft) (%)

- Reprocessing ratio by reactor type, year and fuel type (Reprocessing (type, year, ft)) (%)

- Initial fissile content in fresh fuel by reactor type, year and fuel type (Enrichment (type, year, ft)) (wt %)

- SFR type (TypeSFR): 1 for blanket type SFR (150MW type KALIMER), 1 for no blanket type SFR (600MW KALIER)

- Average enrichment tails assay by year (TailsAssay (year)) (wt%)

- Process loss coefficients by year (EnrichLoss (type, ft), FabLoss (type, ft), ReproLoss (type, ft), ConvLoss (type, ft)) (%)

Using the above input parameters, the followings can be calculated year by year and reactor type by reactor type.

- Total Power Capacity (year) = Total Power Generation (year) / AvgLoadFactor, where, the Total Power Generation comes from the FORECAST module.

- Power Capacity (type, year) = ShareType (year, type) X Total Power Capacity (year)

- Annual fresh fuel requirements (type, year, ft) (ton)

  = Power Capacity (type, year) X 365 X (LoadFactor (type, year) / ((Efficiency (type, year)

- X Burnup (type, year, ft) X 1000)

- Annual spent fuel arisings (type, year, ft)

  = Annual fresh fuel requirements (ton) X (1-ComsumpRatio (type, year, ft))

- Annual fuel fabrication requirement (type, year, ft) (ton)

  = Annual fresh fuel requirements (type, year, ft) / (1- FabLoss (type, ft))

- Annual Enrichment Requirement (type, year, ft) (SWU)

  = EnProduct (type, year, ft) X V_p + EnTail (type, year, ft) X V_t – EnFeed (type, year, ft) X V_f

Where,

EnProduct = mass of uranium to be charged in the fuel fabrication facility,

EnFeed = mass of uranium feed in enrichment plant (and output of conversion plant), and

EnTail = mass of uranium discharged from the enrichment plant (i.e., depleted uranium).

Where, V_p is expressed as follows;

$$V_X = (2e_X - 1)\ln\frac{e_X}{(1-e_X)}$$

where x is the subscript for f, p or t,

e_p = fraction of ^{235}U in the uranium feed,

e_t = fraction of ^{235}U in the tails, and

e_f = fraction of ^{235}U of uranium to be charged in enrichment plant.

Then, EnFeed = EnProduct X $\frac{(e_p - e_t)}{(e_f - e_t)}$, and EnTail = EnFeed / (1-ReproLoss(type, ft)) – EnProduct.

If EnProduct and three fractions of the ^{235}U in enrichment plant are known, then the SWU (Separated Work Unit) as well as EnFeed and EnTail can be calculated from the above equations.

- Annual Conversion Requirement (type, year, ft) (ton)

  = EnFeed (type, year, ft) / (1- ConvLoss (type, ft))

- Annual Natural Uranium Requirement (type, year, ft) (Mt-U_3O_8)

  = Annual Conversion Requirement (type, year, ft) X $\frac{W_{U_3O_8}}{W_{U_3}}$ where

  $W_{U_3O_8}/W_{U_3}$ is the weight fraction of uranium in the uranium resources (U_3O_8).

- Interim Storage Requirement of Spent Fuel (type, year)

  = Spent Fuel Accumulation (type, year) – $\sum_{year}$ Spent Fuel Storage Pool Capacity (type, year)

- Spent Fuel Accumulation (year, type)

  $= \sum_{year}$ Spent Fuel Arisings (type, year) - $\sum_{year}$ Reprocessing Amount (type, year)

- Spent Fuel Storage Pool Capacity (type, year)

  = Existing Spent Fuel Pool Capacity (type) (Year < 2004)

  = Annual Spent Fuel Arisings (type, year) X Pool Reserve Years (type, year) (2004 ≤ Year)

- Reprocessing Requirement

  PWR Reprocessing Requirement = Repro. amount for MOX(year) + Repro. amount for SFR (year) + Repro. amount for ADS(year)

Unloaded spent fuel can be either reprocessed or disposed of directly, depending on types of reactor and fuel, as well as the strategy considered. For example, it is assumed that the CANDU spent fuel is not reprocessed, while the ADS and the SFR fuels are multiple-recycled. The PWR spent fuel can be also reprocessed and reused.

Concerning the lead and lag time, all actions are performed next step one year before. For example, the fuel fabrication is done one year before loading in a reactor and reprocessing is also carried out one year before fuel fabrication.

Chapter 4 Description of Interface

As shown in Fig. 2, the FAST, based on the nuclear power generation data from the FORECAST, estimates quantity requirements for each nuclear fuel cycle step and calculates material flows for various nuclear fuel cycles. As a computational simulation tool, the FAST has several advantages. First, it has a logistics function which links the code to the FORECAST, a program that projects a future nuclear energy need in Korea and whose output is transferred automatically to the FAST. Second, the FAST employs a MS Excel spread sheet with the Visual Basic Application. Such an application allows users manipulate the program very easily. The speed of the calculation is also quick enough to make comparisons among different options in a considerably short time. The Fast can also be used by non-nuclear fuel specialists putting only a few basic data and develop different energy scenarios.

The FAST code has one input sheet for FORECAST model and three input sheets for FAST model : Scenario sheet, Reactor sheet and Fuel sheet. Each sheet is explained below.

4.1 FORECAST Sheet

The Fig. 16 shows the main sheet of the FORECAST for projecting the nuclear electricity generation. In the upper part of the sheet, users can input an asymptotic electricity demand per capita and a base year. There are three options that the users could choose for the asymptotic electricity demand per capita: low, reference and high. In the down part of the sheet, nuclear electricity market share of the base year, halving time and a targeted market share of nuclear power are chosen. An average load factor is used to calculate the capacity of the nuclear. Given above data and historical data such as per capita electricity demand, total electricity generation, national population and nuclear electricity generation, coefficients described in Equation 1, 4 and 5 are calculated automatically by the regression analysis, and then the projections for per capita electricity demand, total electricity generation, national population and nuclear electricity generation are calculated.

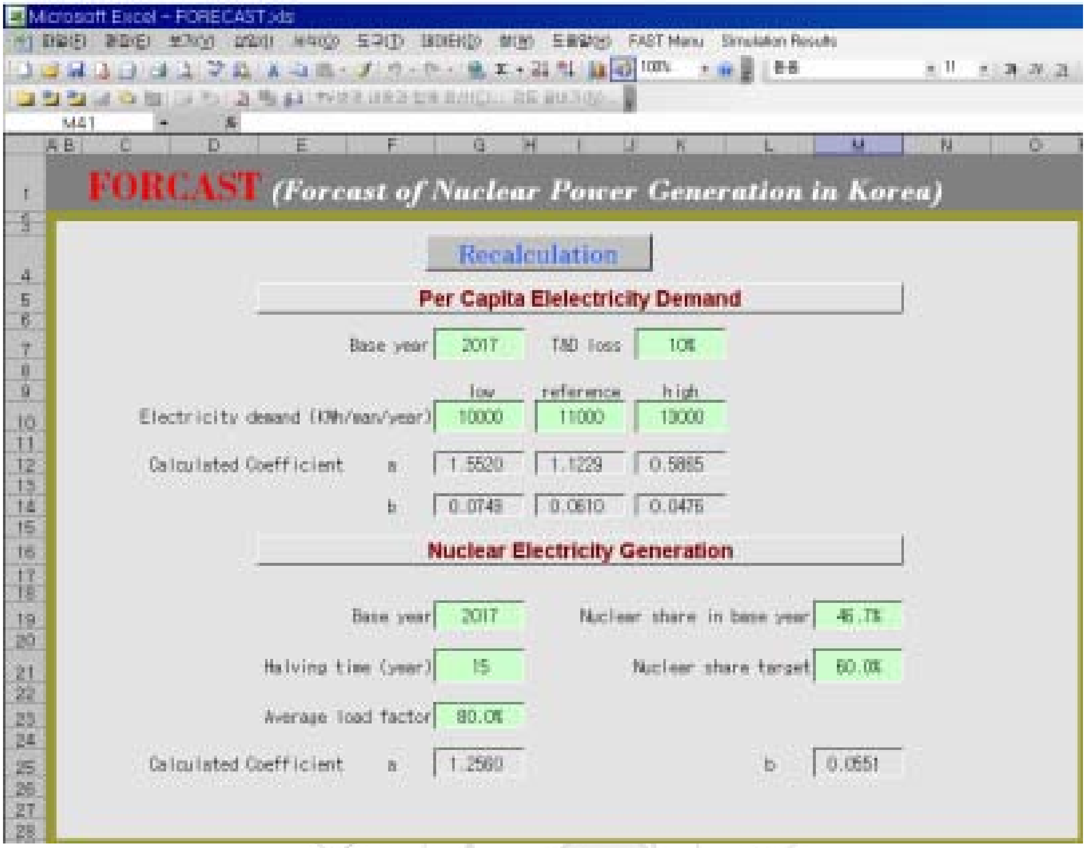

Fig. 16 Illustration of FORECAST Main Sheet

4.2 Scenario Sheet

The scenario sheet, shown in Fig. 17, considers two sets of scenarios: the level of nuclear power generation and the reactor type mixtures.

The Power Gen. Scenario, which is in the combo box of the left side, uses the projection data of the nuclear power generation calculated from FORECAST. Users can choose one of the three options for the nuclear power generation scenario in Korea: low, reference and high. In the Reactor Strategies box, a reactor mix can be chosen. The total of 11 scenarios, which are generated from the combination of PWR, CANDU, HTGR, SFR and ADS, are considered in the FAST. As shown in the Fig. 17, these 11 scenarios are categorized into four clusters: the once-through cycle, the MOX thermal recycle, SFR and ADS related cycles. The once-through cycle covers scenario groups 1 through 3 while the MOX thermal recycle covers Group 4. As for the SFR, the FAST considers both 150MW KALIMER type, which uses several different blanket fuels and 600MW KALIMER type, which uses only driving fuel without any blanket fuels.

Other input cells in the Scenario Sheet are for the calculation of each reactor grid by the method described in section 3.3.2. After taking the two main input parameters above, the "Enter" button should be pressed for inputting further data. When the button is pressed, the color of the input cells in the three input sheets is changed into green. If all the data were given appropriately in the green cell boxes, outputs are automatically calculated, which can be obtained from the drop down menu of the "Simulation Results" in the upper line.

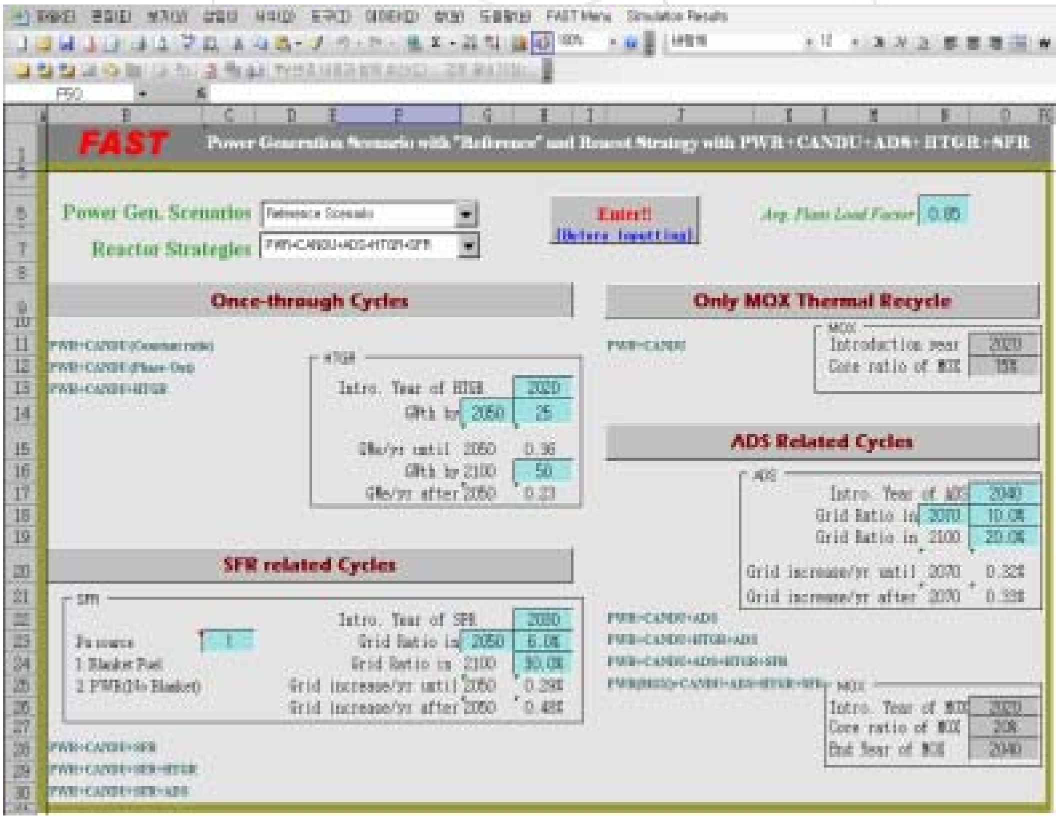

Fig. 17 Illustration of Scenario Sheet

The average plant load factor on the right upper side is to induce nuclear power capacity from the nuclear power generation which calculates in the FORECAST.

4.3 Reactor Sheet

As shown in Fig. 18, the Reactor Sheet includes input parameters related to reactor information such as load factors, burnup, efficiency and pool capacity. There are also specific SFR and ADS information sections in the middle of the sheet since the SFR and the ADS are required more reactor information.

As for the SFR, if the 150MW KALIMER type is chosen in the Scenario Sheet, the number of fuel assembly of each fuel type and their power fraction are needed for the calculation of the average weighted burnup. As for the ADS, there are TRU (transuranium) fuel and target fuel made by technetium and iodine. Users have to put other information such as TRU fuel and target fuel weight per core and TRU contribution ratio of thermal power.

For a CANDU reactor, the plant life time is required because CANDU reactors are phased out except for scenario group 1.

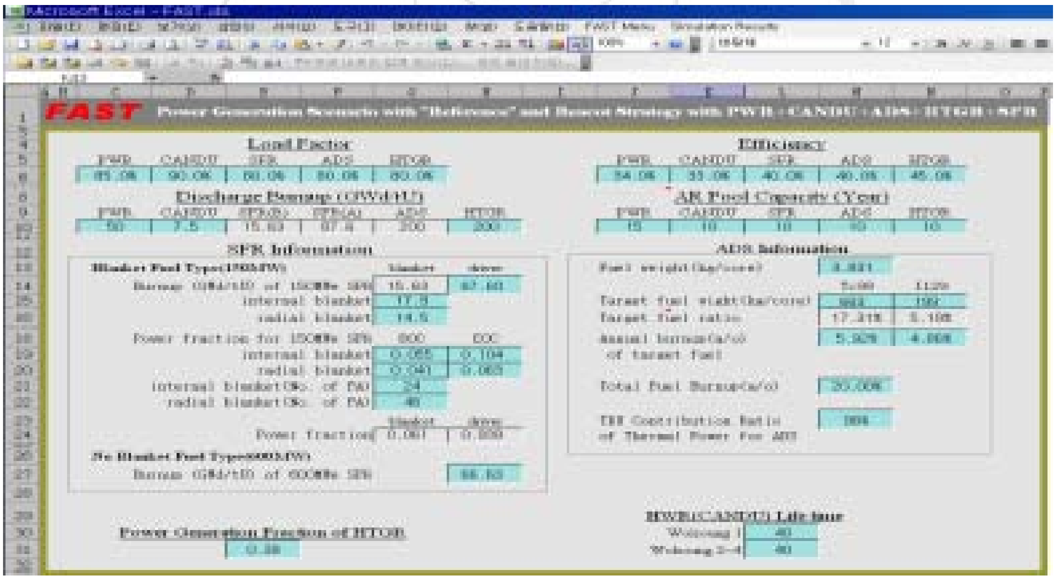

Fig. 18 Illustration of Plant Sheet

4.4 Fuel Sheet

The Fuel Sheet, as shown in Fig. 19, deals with information on fresh fuels and spent fuels of each reactor type. In addition, process losses in various steps, which are in the down part of this sheet, are put in this sheet. The fuel contents consist of Pu, U^{235}, U, MA, Tc, I which are put with unit of weight percent of heavy metal. On the right side, there is a heavy metal consumption rate which means burning amount of each fuel in reactors.

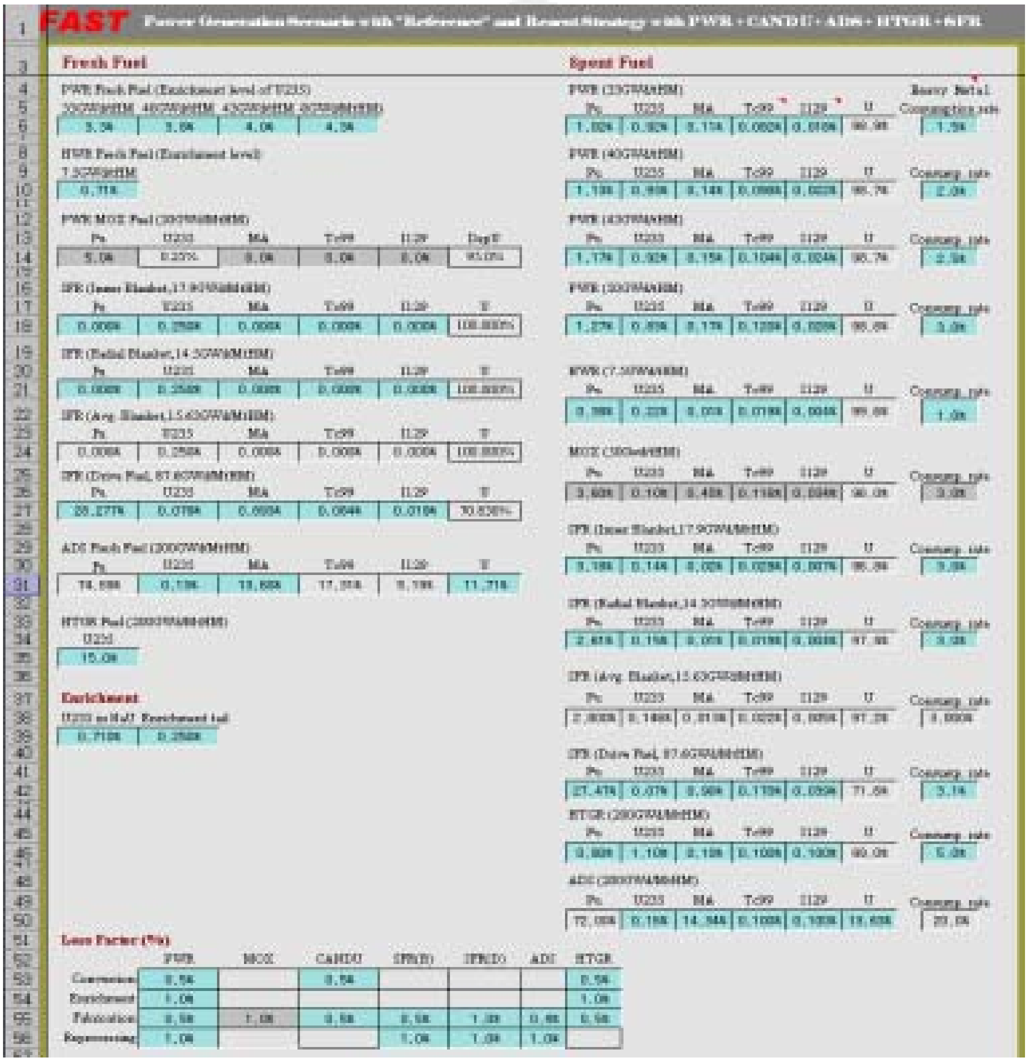

Fig. 19 Illustration of Fuel Sheet

Chapter 5 Example of Applications

The following simulations have been performed for several different reactors and fuel cycle options by using the FAST. It should be noted that these results are only given as examples of how to use the model. Since the software has not yet reached the testing and benchmarking stage. However, there is an indirect testing method for this code. That is to compare the simulation data on spent fuel arisings with historical data. Fig. 20 shows the result of the FAST code simulation on spent fuel arisings for Scenario Group 1. The dot mark in this figure means the real data on spent fuels generated in Korea from 1978 to 2000. As seen in this figure, the simulation results are nearly consistent with the real data.

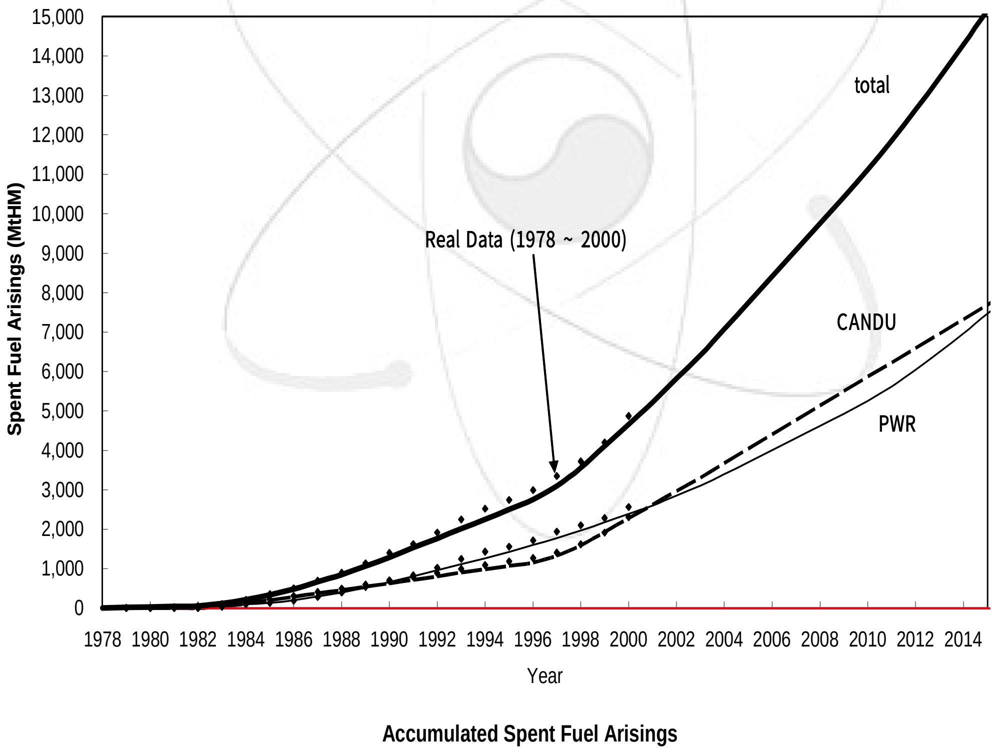

Fig. 20 Comparison of Simulation Results with Real Data on Spent Fuel Arisings in Korea

5.1 Nuclear Electricity Generation from FORECAST

In this section, nuclear electricity generation is projected by the use of the FORECAST. First, the per capita electricity demand (kWh/man/year) is estimated with three options (low, reference, high) and then the share of electricity is calculated using the logistics curve. The inputs used in this case study are as follows.

5.1.1 Input Data

- Assumed asymptote to which per capita demand trend

  -Low : 10,000kWh/man/year

  -Reference : 11,000kWh/man/year

  -High : 13,000kWh/man/year

- Base year : 2017

- Historical and future population : used the database of Korea National Statistical Office [12]

- The nuclear share of total electricity generation in the base year : 46.7% in 2017 [11]

- An asymptote to which the nuclear share of total electricity generation trends : 60%

For the nuclear electricity data, up to 2003, the historical data was used, while form 2004 to 2017 the 2nd Basic Plan of Electricity Demand and Supply which includes plant construction plans up to 2017 was used.

5.1.2 Output

The outputs of the FORECAST are shown in From Fig. 21 to Fig 24.

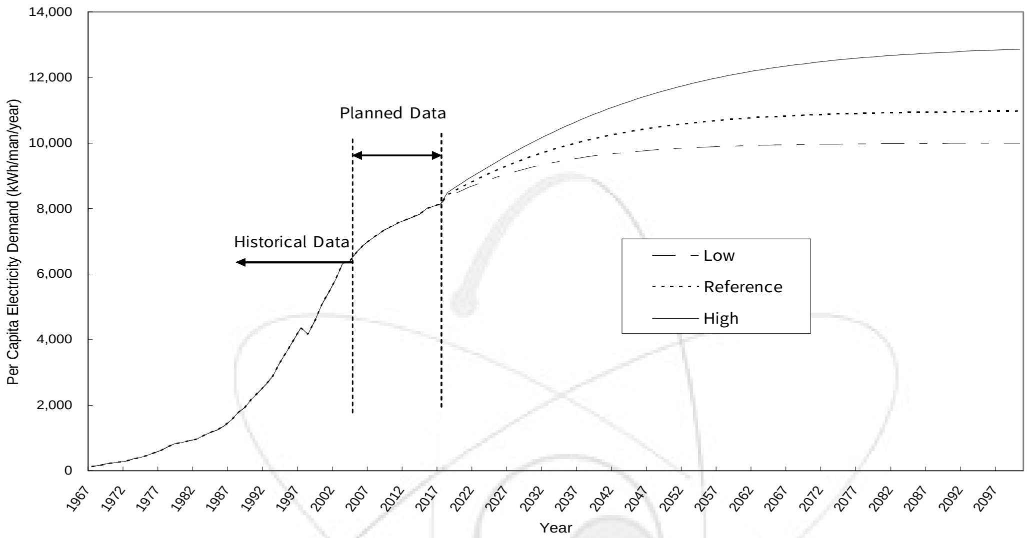

Fig. 21 Trend of Per Capita Electricity Demand (kWh/man/year)

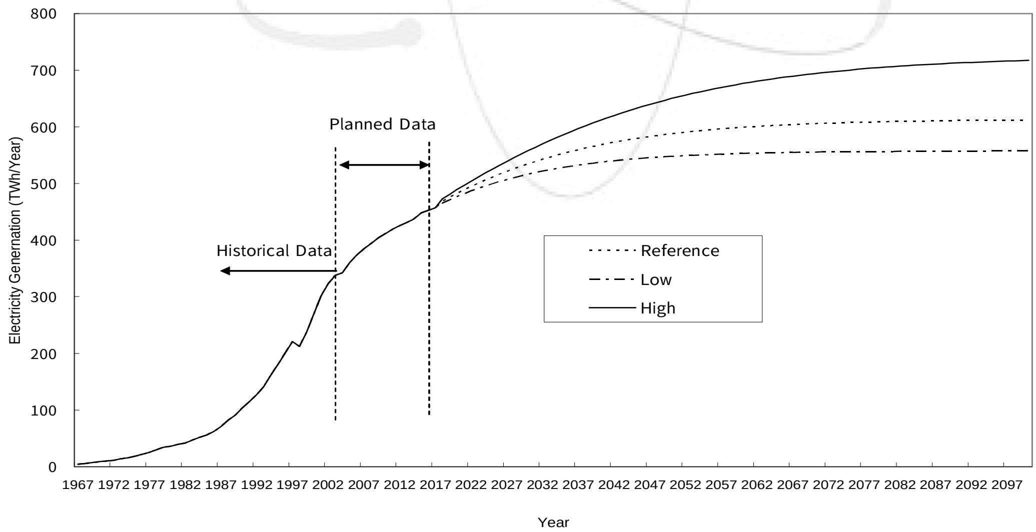

Fig. 22 Trend of Total Electricity Generation

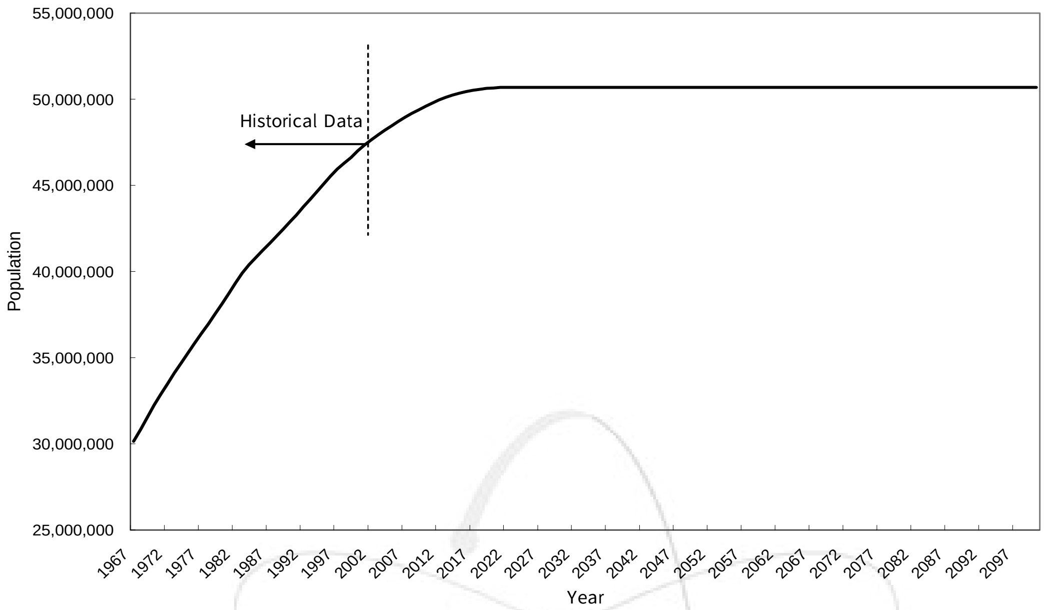

Fig. 23 Trend of Future Koran Population

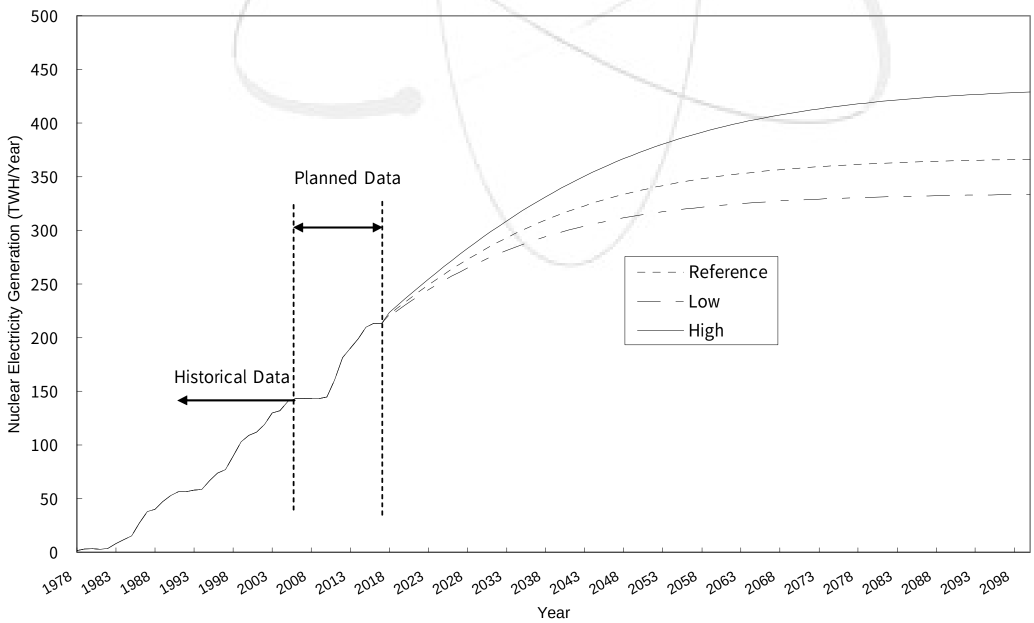

Fig. 24 Trend of Nuclear Electricity Generation

5.2 Scenario Group 1

5.2.1 Scenario Parameters (PWR + CANDU)

Scenario Group 1 is the once-through cycle with the PWRs and the CANDU reactors, and after 2017, the CANDU reactor grid ratio remains constant. Reference Scenario for nuclear electricity generation is used in example cases.

5.2.2 Reactor Parameters

- Load factor: Up to 2003, the historical data were used. After 2004, 85% for the PWR and 90% for the CANDU were assumed.
- Efficiency : PWR 34%, CANDU 33%
- PWR discharge burnup :

  -1978 ~ 1984 : 33 GWd/MTU

  -1985 ~ 1995 : 40 GWd/MTU

  -1995 ~ 2015 : 43 GWd/MTU

  -2016 ~ : 50 GWd/MTU
- CANDU discharge burnup : 75 GWd/MTU
- Pool capacity : Up to 2003, the historical data were used. After 2004, storage pools of PWR can employ spent fuels generated for 15 years and storage pools of CANDU can employ spent fuels generated for 10 years.

5.2.3 Fuel Parameters

- Initial enrichment of the PWR fuels

33GWd/tHM	40GWd/tHM	43GWd/tHM	50GWd/MtHM)
3.3%	3.8%	4.0%	4.3%

- Initial enrichment of the CANDU fuel : 0.71%

- Spent fuel contents

PWR (33GWd/tHM)						Heavy Metal Consumption rate
Pu	U235	MA	Tc99	I129	U	
1.02%	0.92%	0.11%	0.082%	0.018%	92.1%	1.5%
PWR (40GWd/tHM)						
Pu	U235	MA	Tc99	I129	U	
1.13%	0.93%	0.14%	0.098%	0.022%	89.1%	2.0%
PWR (43GWd/tHM)						
Pu	U235	MA	Tc99	I129	U	
1.17%	0.92%	0.15%	0.104%	0.024%	88.0%	2.5%
PWR (50GWd/tHM)						
Pu	U235	MA	Tc99	I129	U	
1.27%	0.83%	0.17%	0.120%	0.028%	85.1%	3.0%
HWR (7.5GWd/tHM)						
Pu	U235	MA	Tc99	I129	U	
0.38%	0.22%	0.01%	0.019%	0.004%	98.5%	1.0%

- Loss factors

	PWR	CANDU
Conversion	0.5%	0.5%
Enrichment	1.0%	
Fabrication	0.5%	0.5%

- Enrichment : natural U of 0.71% and enrichment tail of 0.25% were assumed.

5.2.4 Outputs

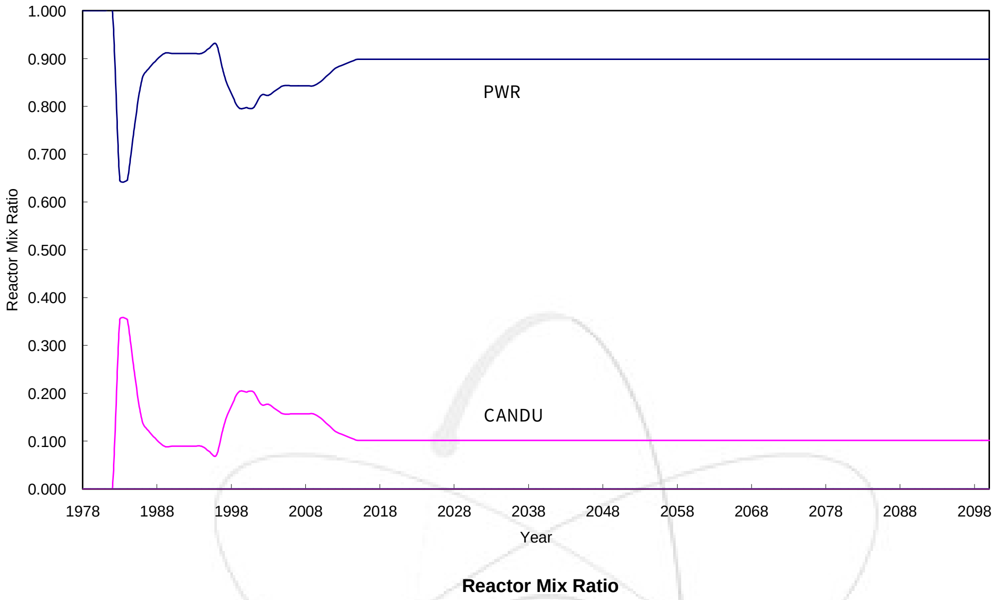

Fig. 25 Reactor Mix Ratio of Scenario Group 1

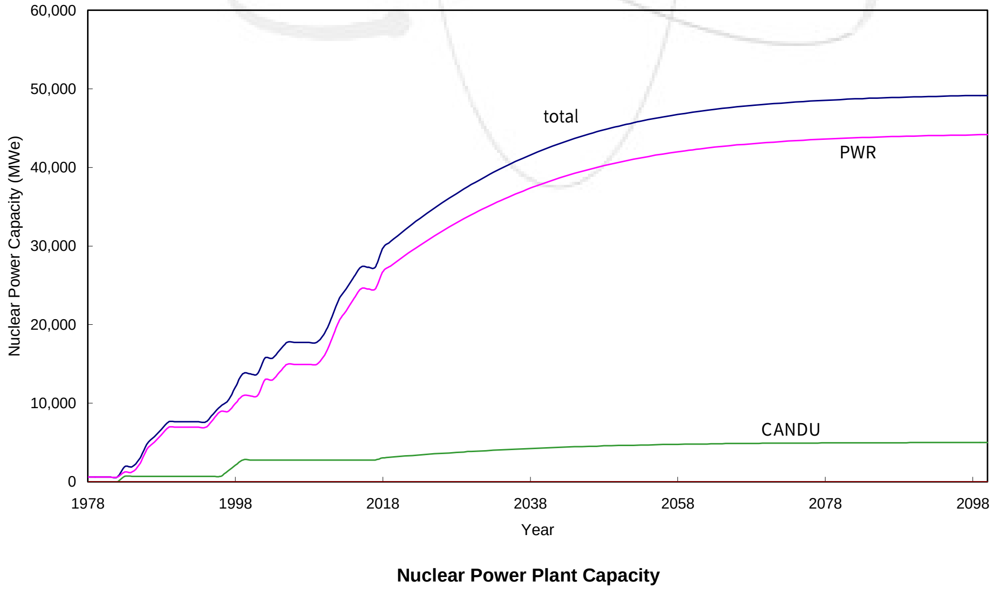

Fig. 26 Nuclear Power Plant Capacity of Scenario Group 1

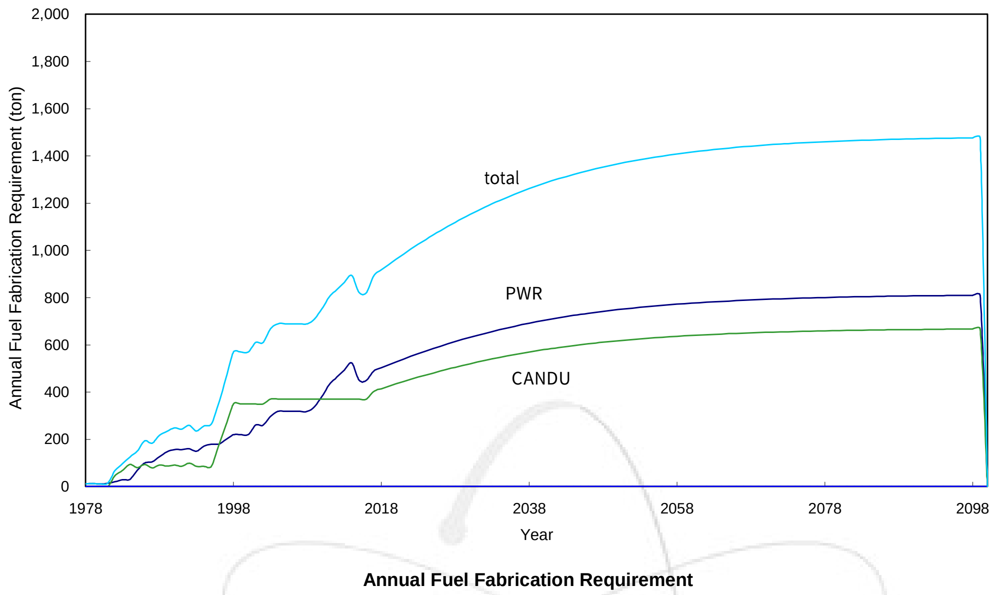

Fig. 27 Annual Fuel Fabrication Requirement of Scenario Group 1

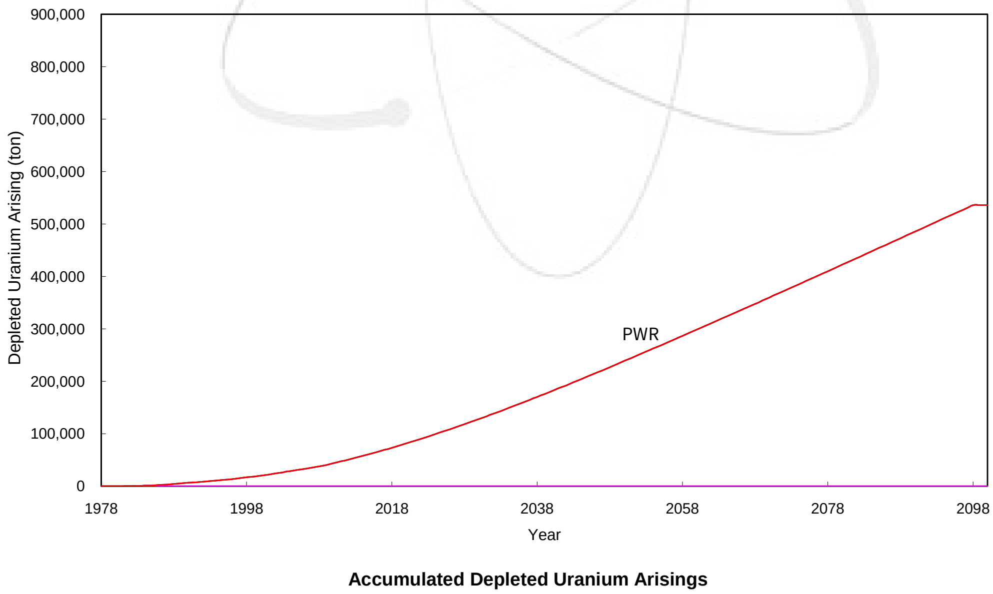

Fig. 28 Accumulated Depleted Uranium Arisings of Scenario Group 1

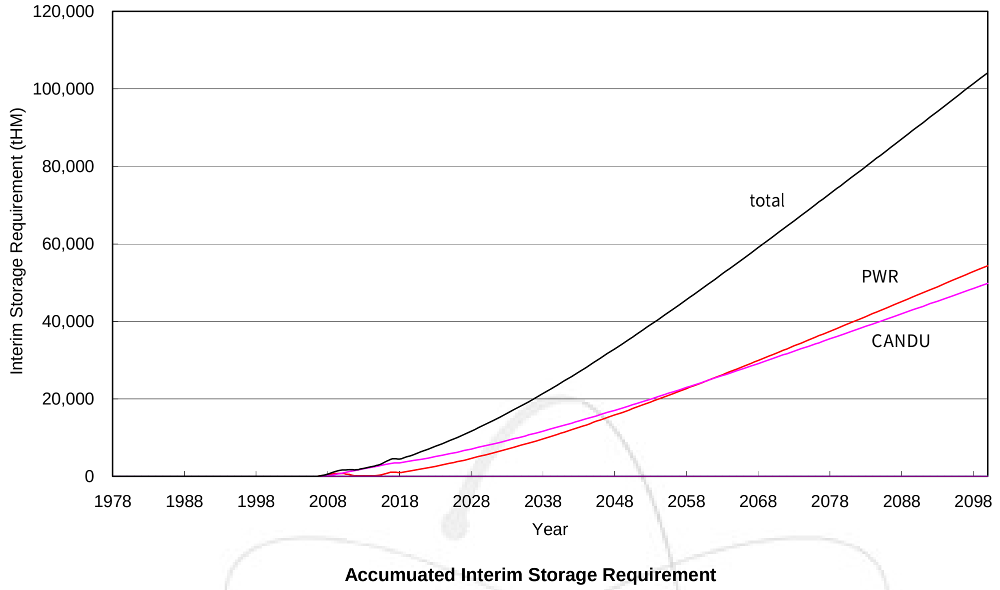

Fig. 29 Accumulated Interim Storage Requirement of Scenario Group 1

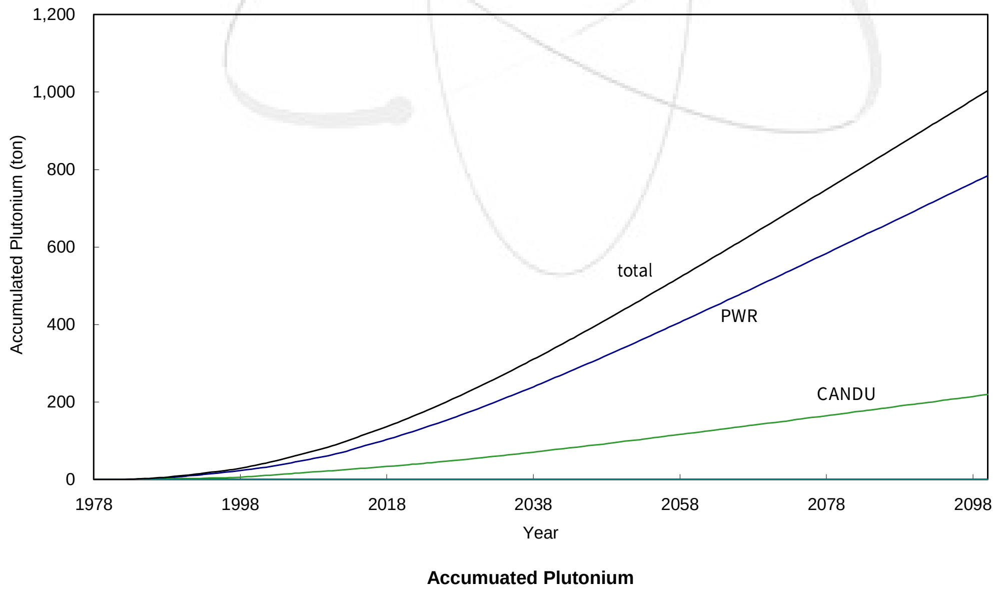

Fig. 30 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 1

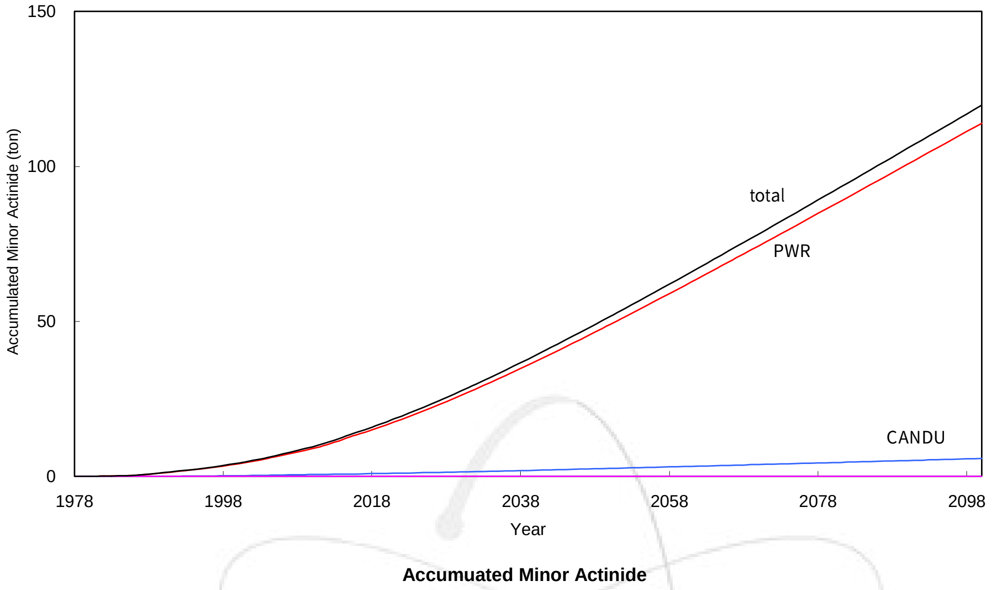

Fig. 31 Accumulated MA Embedded in Spent Fuels of Scenario Group 1

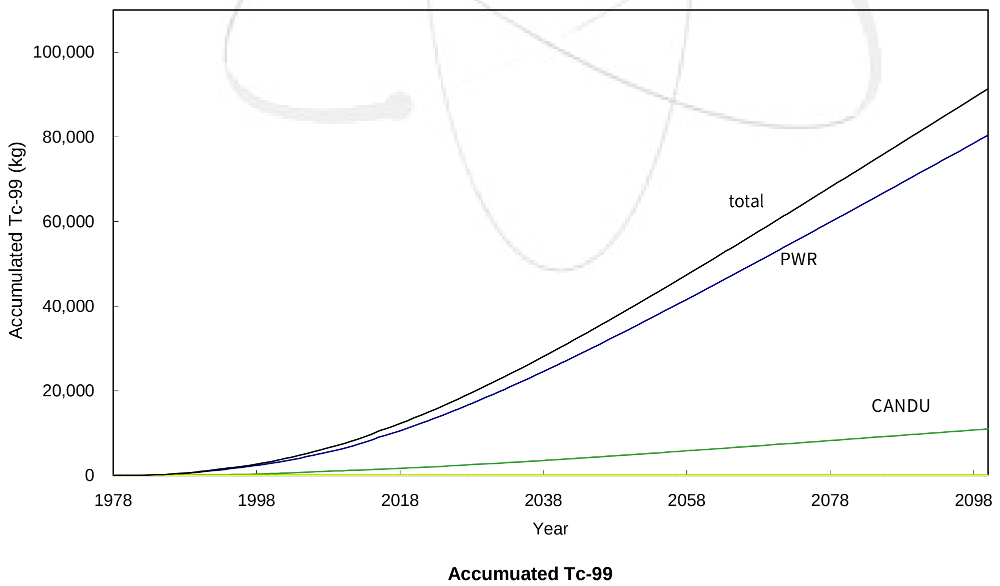

Fig. 32 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 1

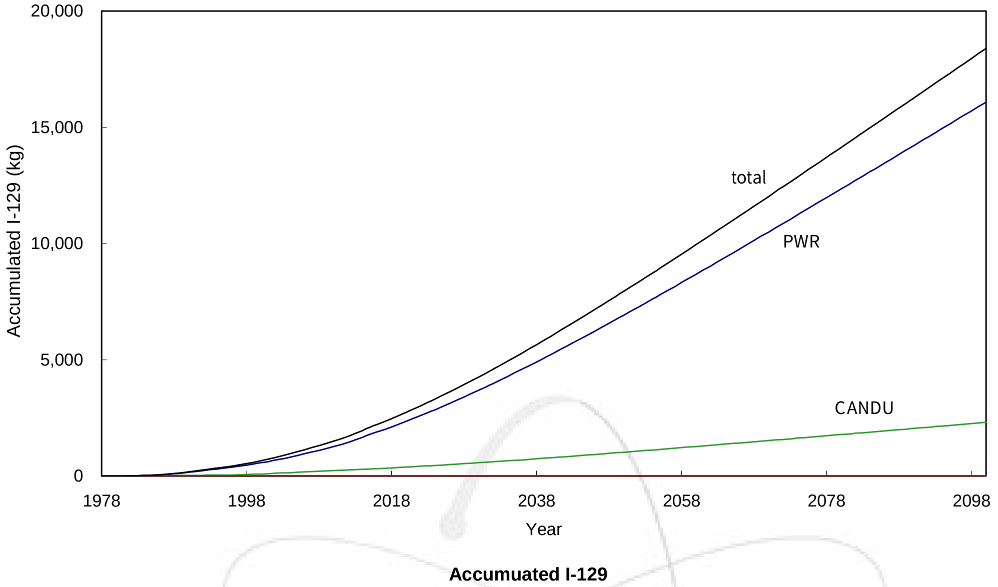

Fig. 33 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 1

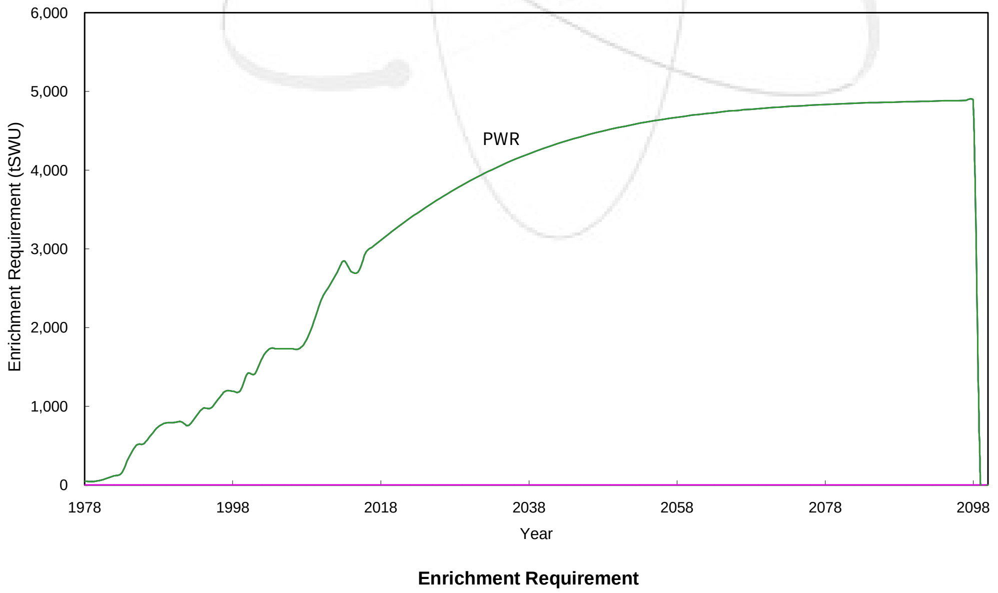

Fig. 34 Enrichment Requirement of Scenario Group 1

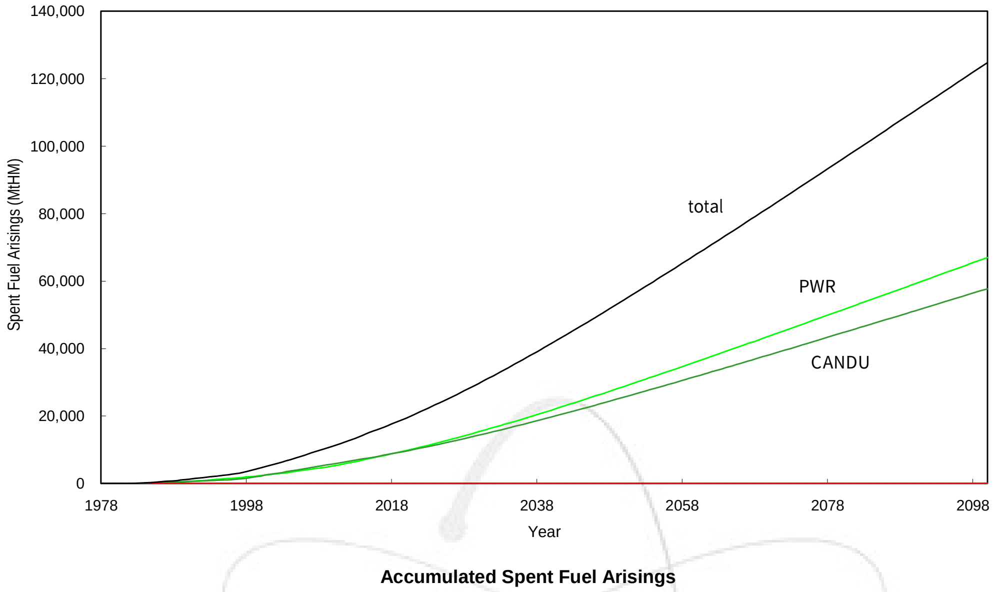

Fig. 35 Accumulated Spent Fuel Arisings of Scenario Group 1

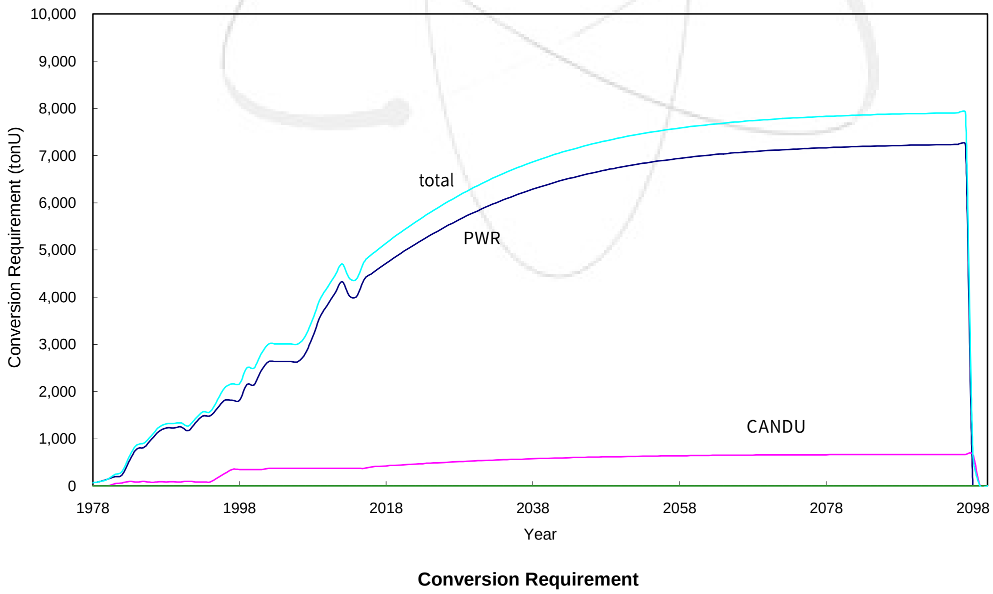

Fig. 36 Conversion Requirement of Scenario Group 1

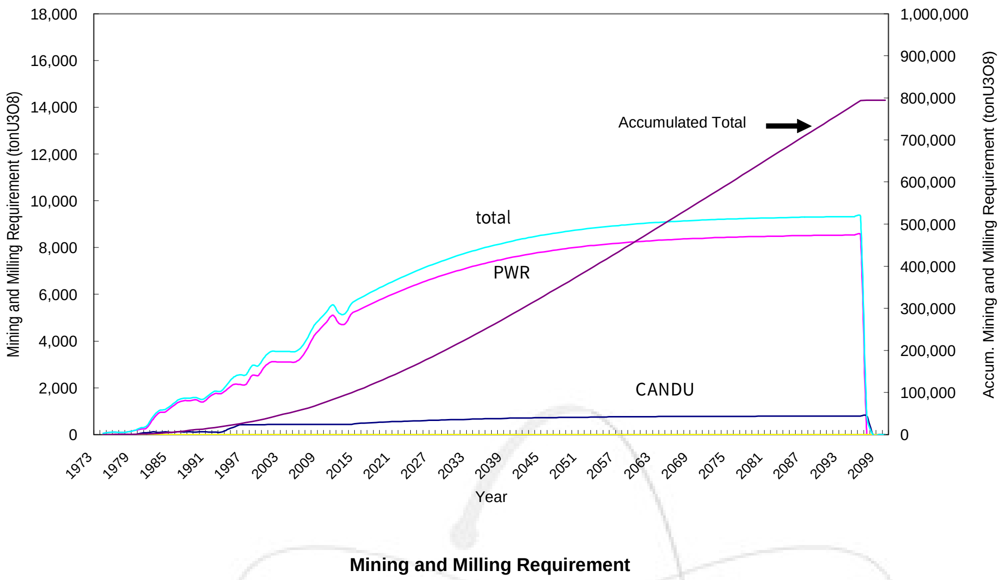

Fig. 37 Mining and Milling Requirement of Scenario Group 1

5.3 Scenario Group 2

5.3.1 Scenario Parameters (PWR + CANDU)

Scenario Group 2 is the same as Group 1 except that after 2017 CANDU reactor is phased out.

5.3.2 Reactor Parameters

- Load factor and efficiency of the reactor, fuel discharge burnup and pool capacity are the same as those of the Scenario Group 1.

- For this group, the lifetime of the CANDU reactors is needed because the CANDU reactors are phased out. The 40 years of lifetime for all CANDU reactors are assumed in this case study.

5.3.3 Fuel Parameters

- Initial enrichment of fuel, fuel content and enrichment parameters are the same as those of the Scenario Group 1.

5.3.4 Outputs

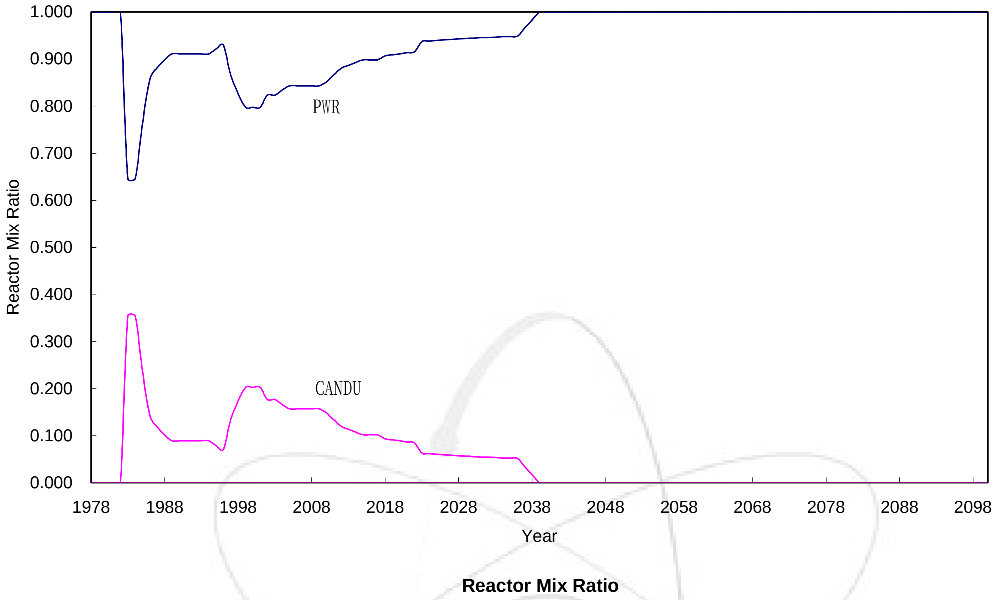

Fig. 38 Reactor Mix Ratio of Scenario Group 2

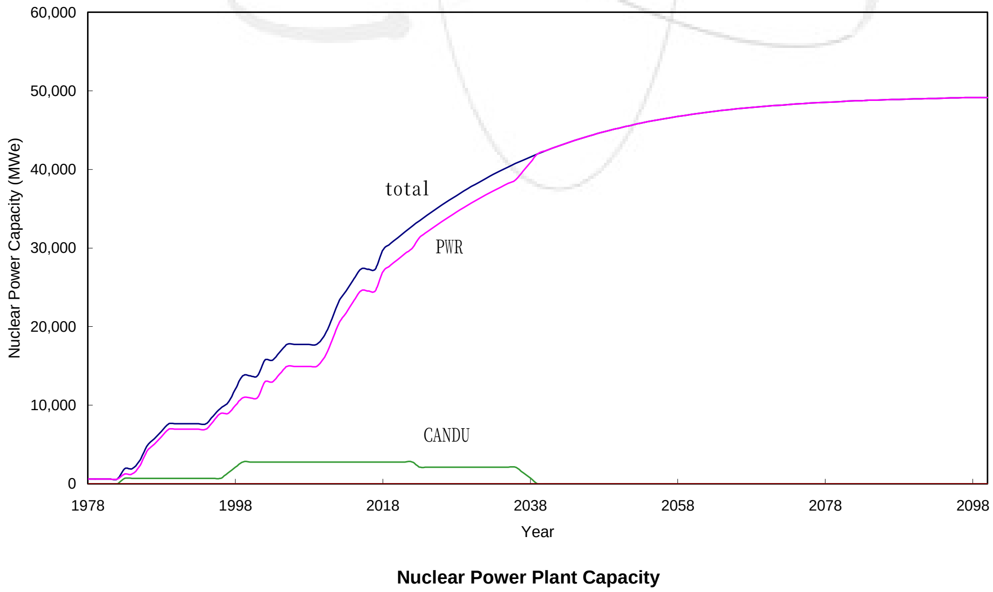

Fig. 39 Nuclear Power Plant Capacity of Scenario Group 2

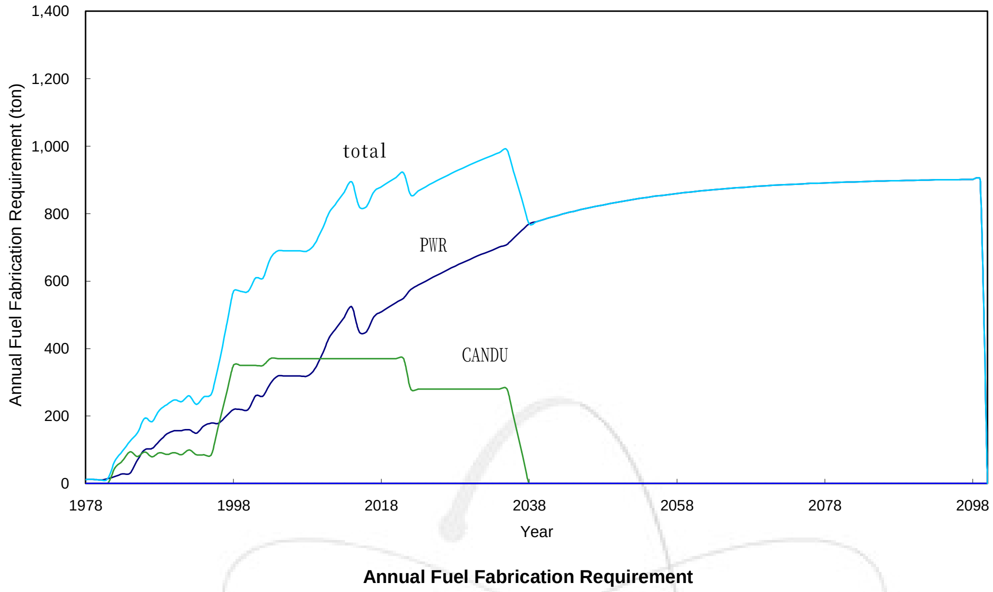

Fig. 40 Annual Fuel Fabrication Requirement of Scenario Group 2

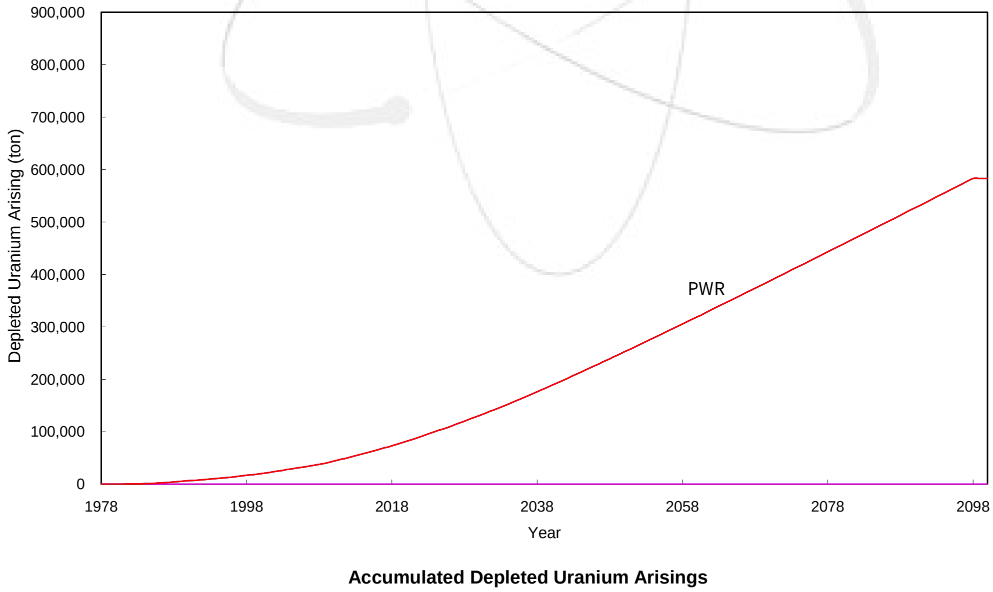

Fig. 41 Accumulated Depleted Uranium Arisings of Scenario Group 2

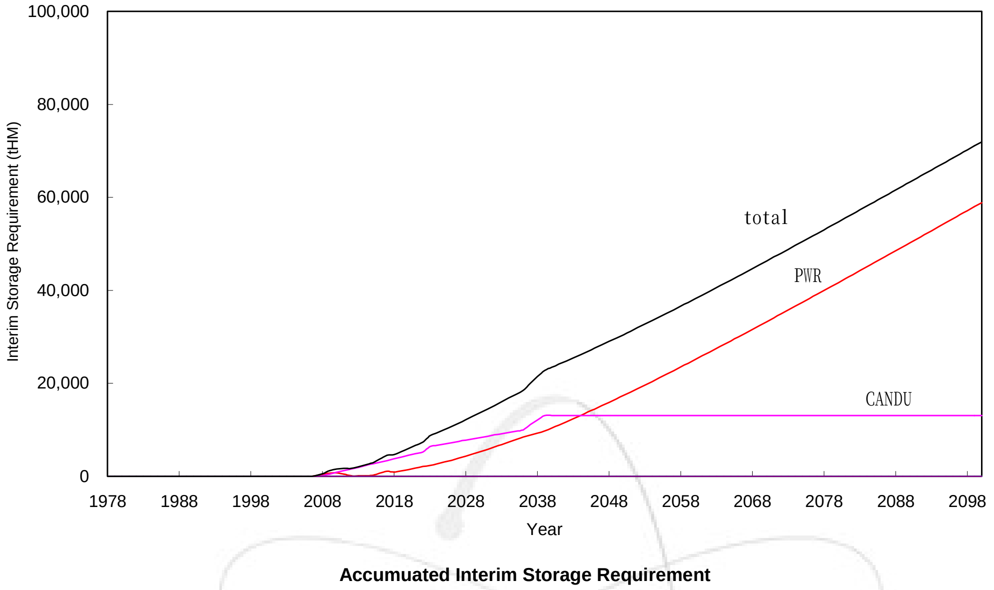

Fig. 42 Accumulated Interim Storage Requirement of Scenario Group 2

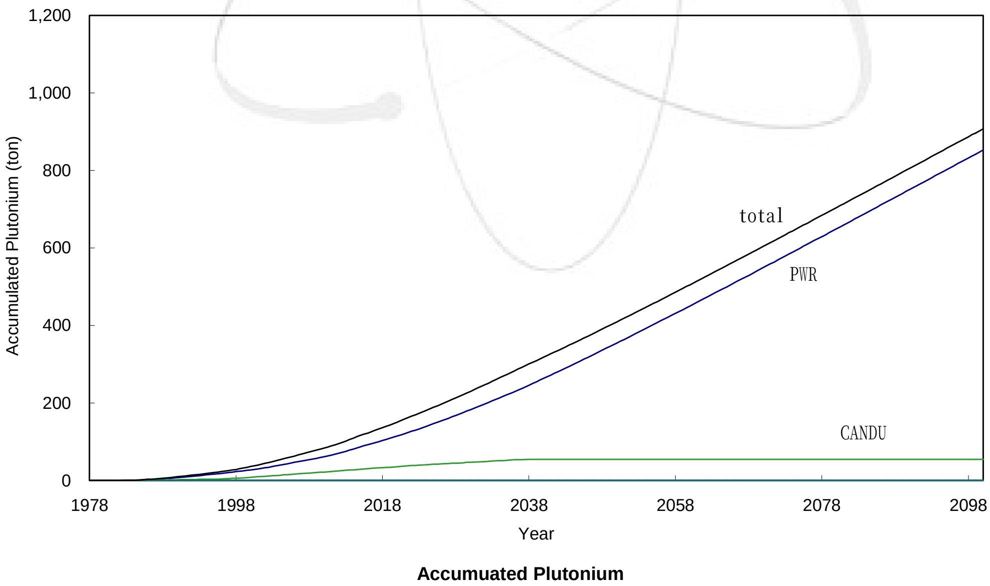

Fig. 43 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 2

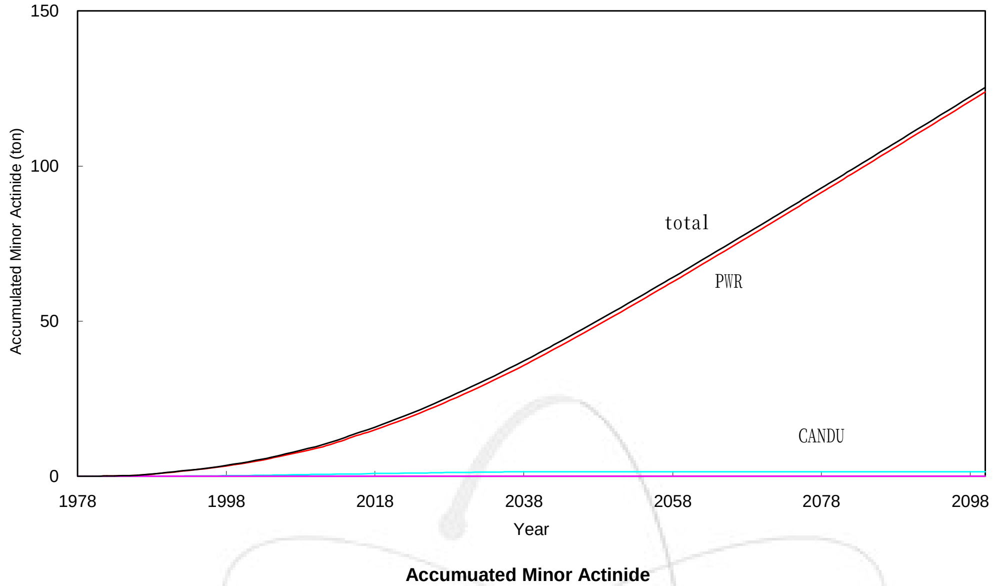

Fig. 44 Accumulated MA Embedded in Spent Fuels of Scenario Group 2

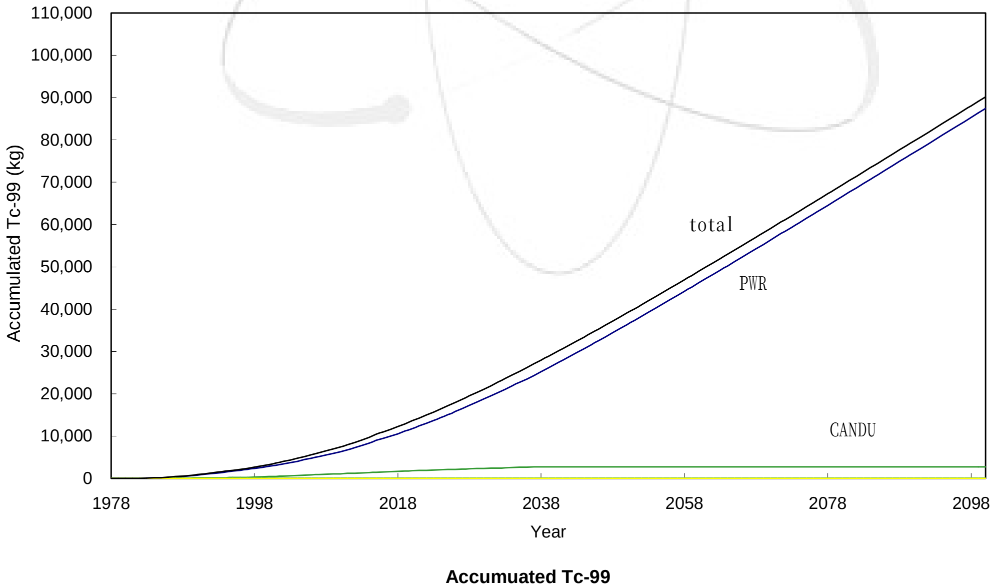

Fig. 45 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 2

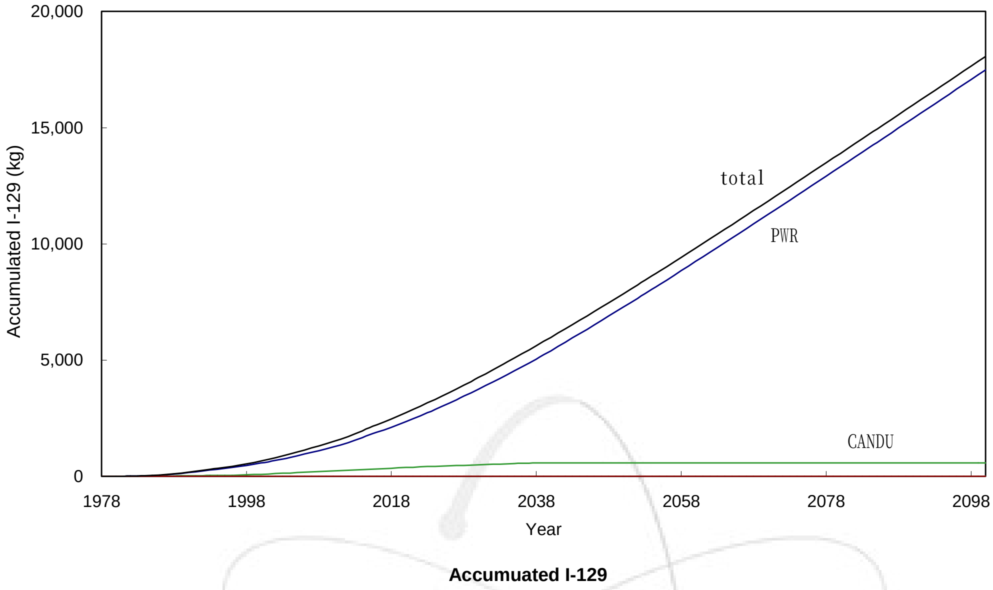

Fig. 46 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 2

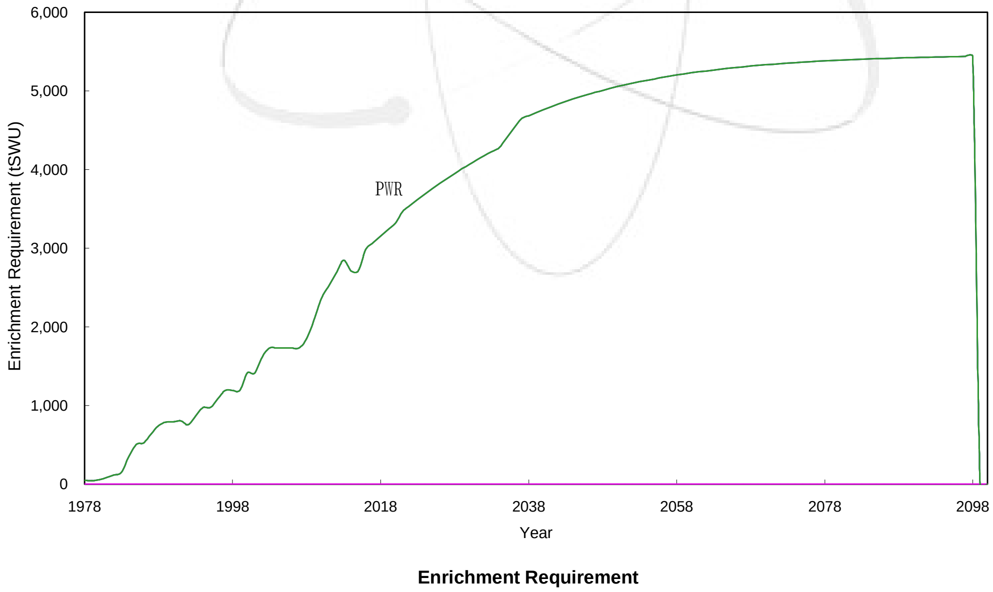

Fig. 47 Enrichment Requirement of Scenario Group 2

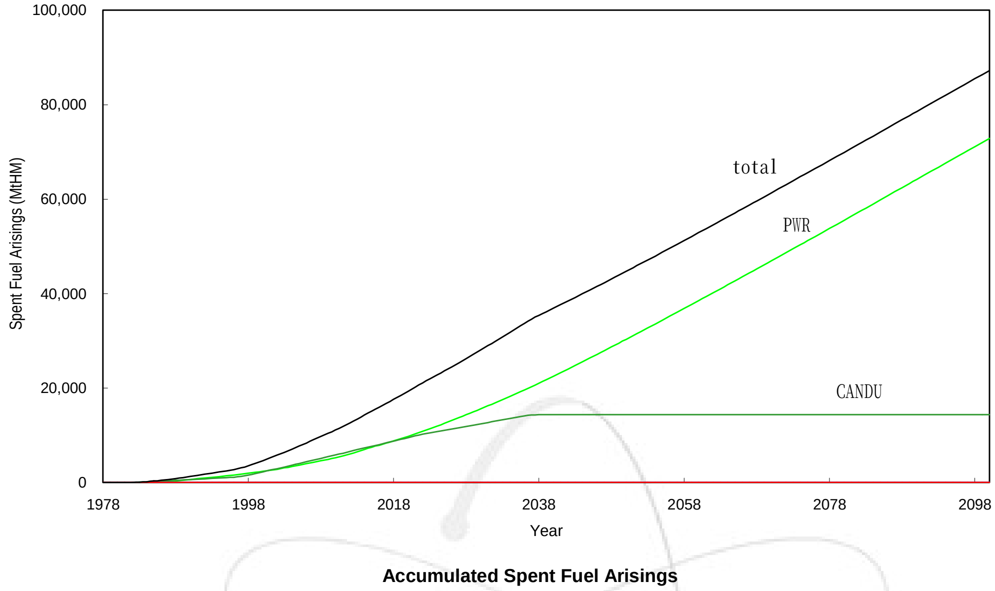

Fig. 48 Accumulated Spent Fuel Arisings of Scenario Group 2

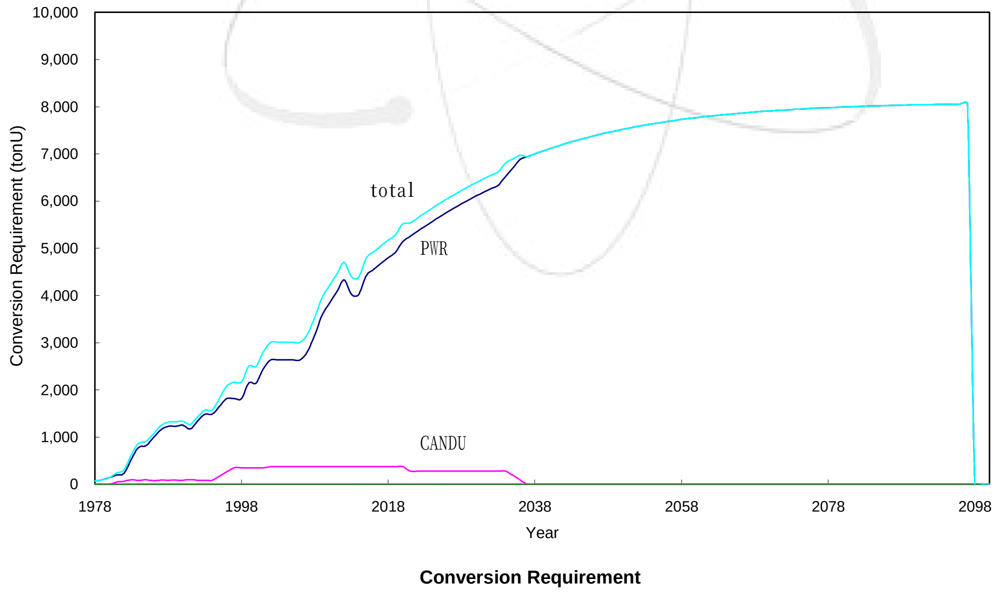

Fig. 49 Conversion Requirement of Scenario Group 2

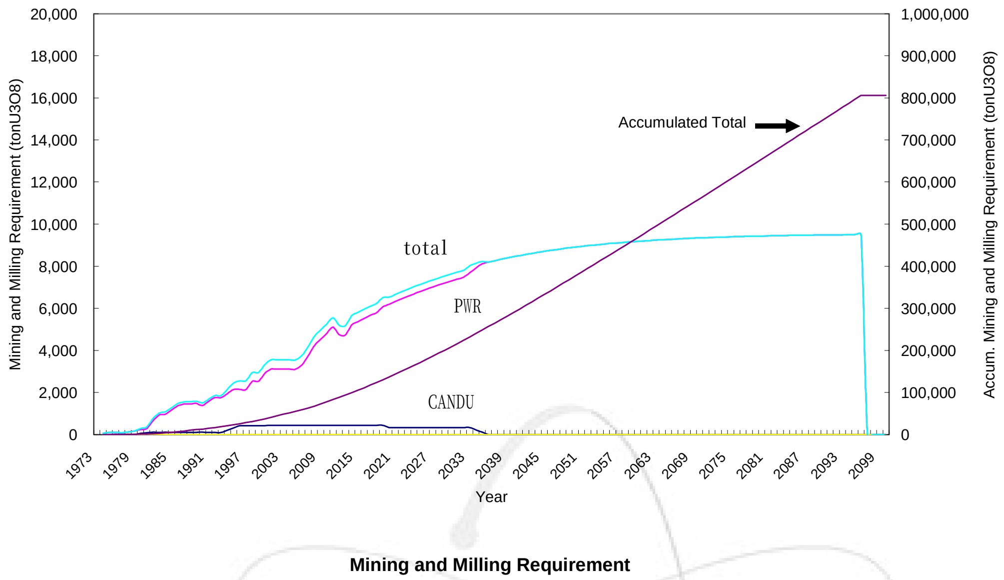

Fig. 50 Mining and Milling Requirement of Scenario Group 2

5.4 Scenario Group 3

5.4.1 Scenario Parameters (PWR + CANDU + HTGR)

Scenario Group 3 is the once-through cycle with the PWR and CANDU and the HTGR. The HTGR uses both of hydrogen production and electricity generation. It is assumed that after 2015, CANDU reactors are phased out and no more constructed up to 2100. It is also assumed that the HTGR is introduced from 2020 and their capacity is expanded up to 25GWth by 2050 and 50 GWth by 2100.

5.4.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Load factor of HTGR : 80%
- Thermal efficiency of HTGR : 45%
- Power generation fraction of HTGR : 39%. (The remaining (41%) of thermal out will be used for hydrogen production.)
- HTGR discharge burnup : 200 GWd/MTU
- Pool capacity : The storage pool of HTGR can employ spent fuels generated for 10 years.

5.4.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Initial enrichment of HTGR : 15%
- Spent fuel contents

HTGR (200GWd/tHM)						Heavy Metal Consumption rate
Pu	U235	MA	Tc99	I129	U	
0.90%	1.10%	0.10%	0.100%	0.100%	99.0%	5.0%

- Loss factors

	HTGR
Conversion	0.5%
Enrichment	1.0%
Fabrication	0.5%

- Enrichment : natural U of 0.71% and enrichment tail of 0.25% were assumed.

5.4.4 Outputs

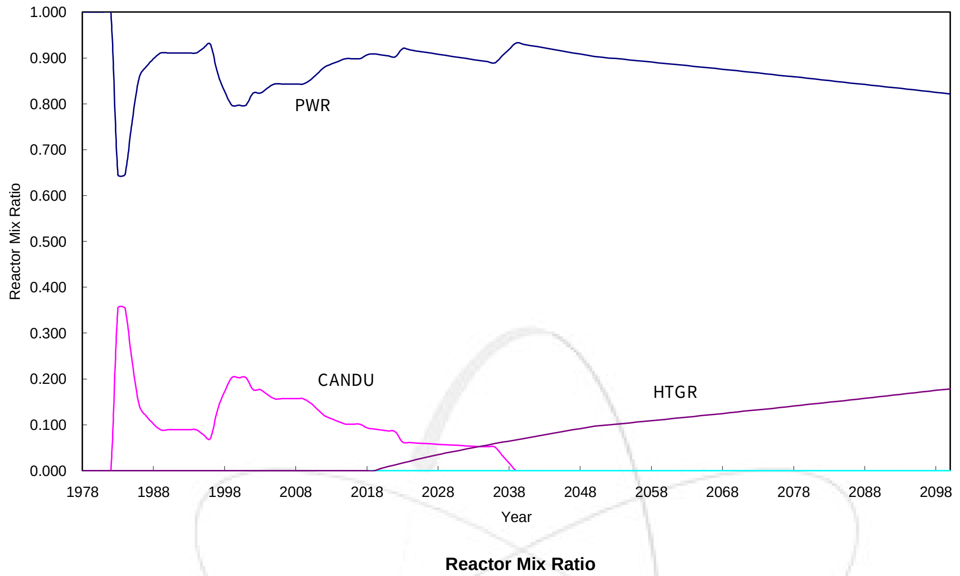

Fig. 51 Reactor Mix Ratio of Scenario Group 3

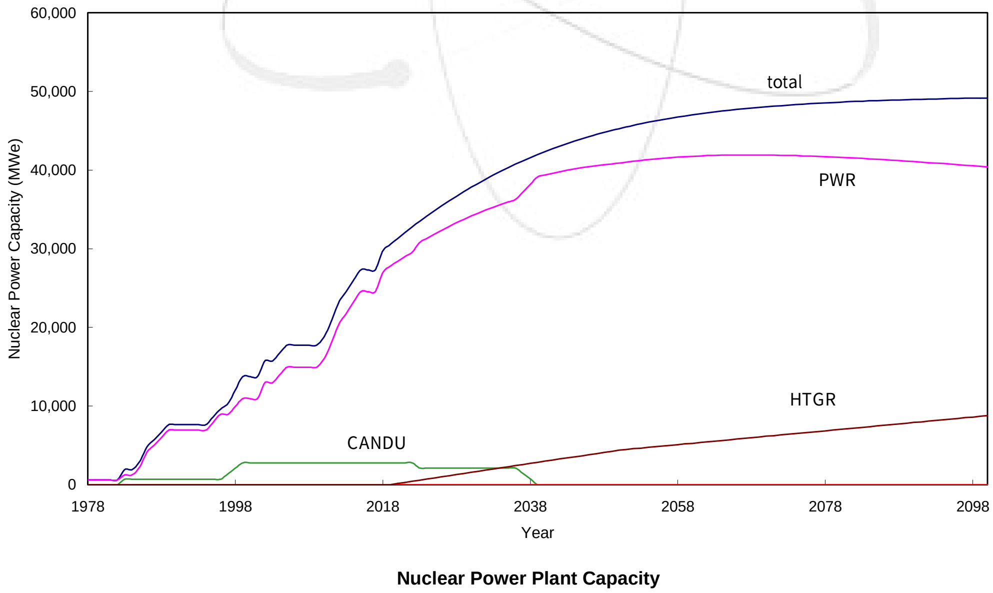

Fig. 52 Nuclear Power Plant Capacity of Scenario Group 3

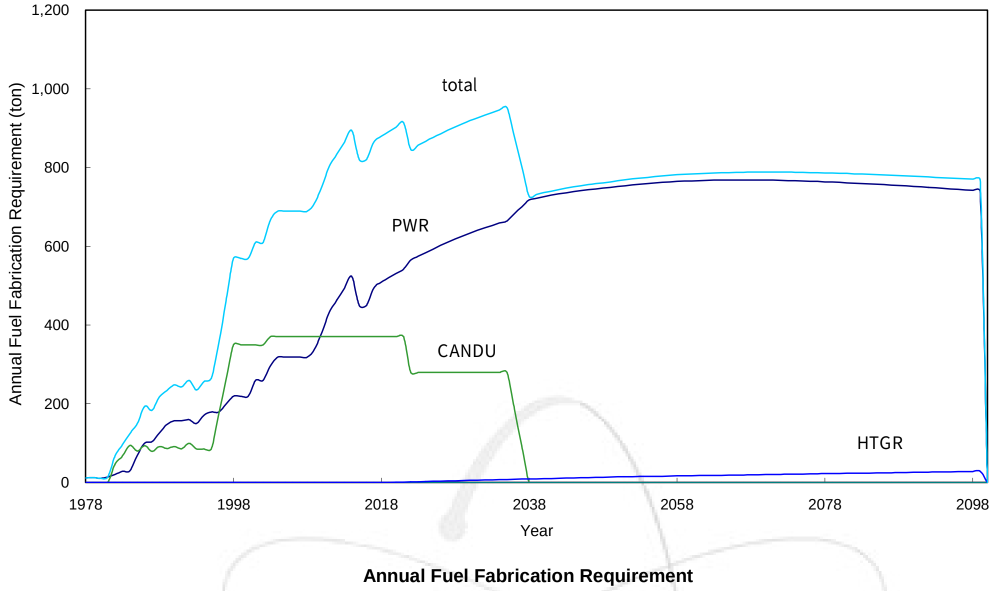

Fig. 53 Annual Fuel Fabrication Requirement of Scenario Group 3

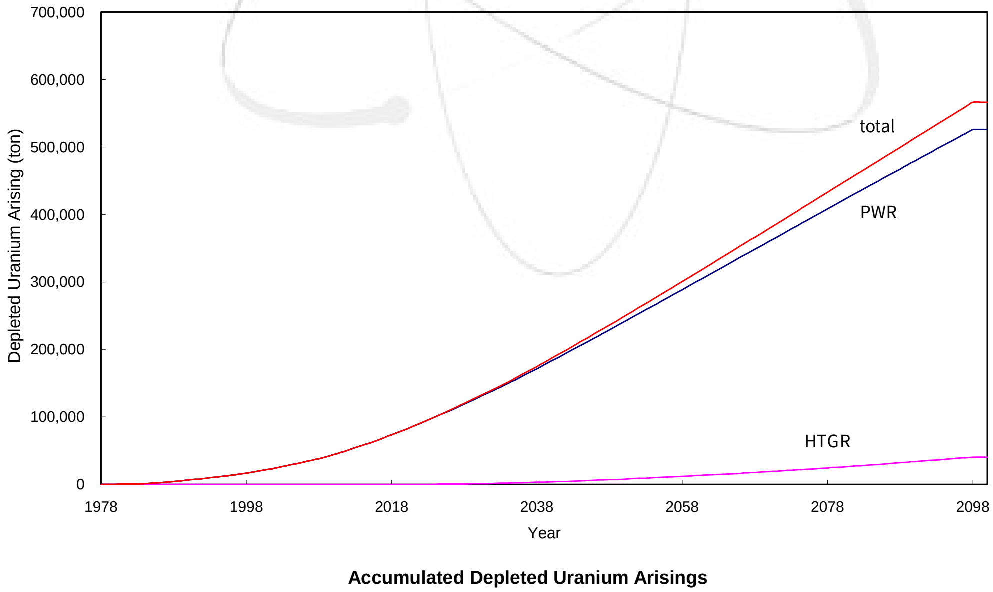

Fig. 54 Accumulated Depleted Uranium Arisings of Scenario Group 3

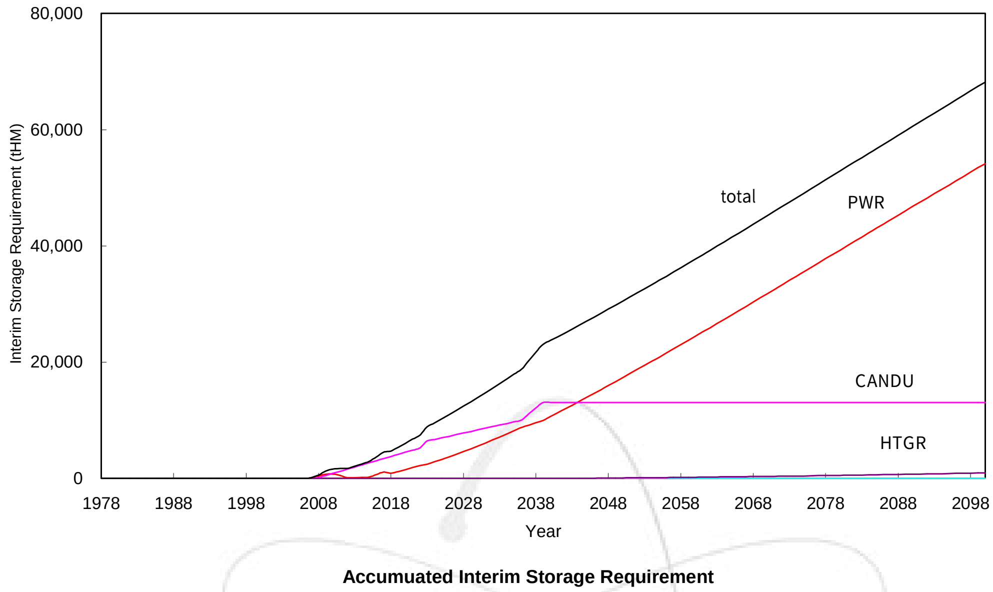

Fig. 55 Accumulated Interim Storage Requirement of Scenario Group 3

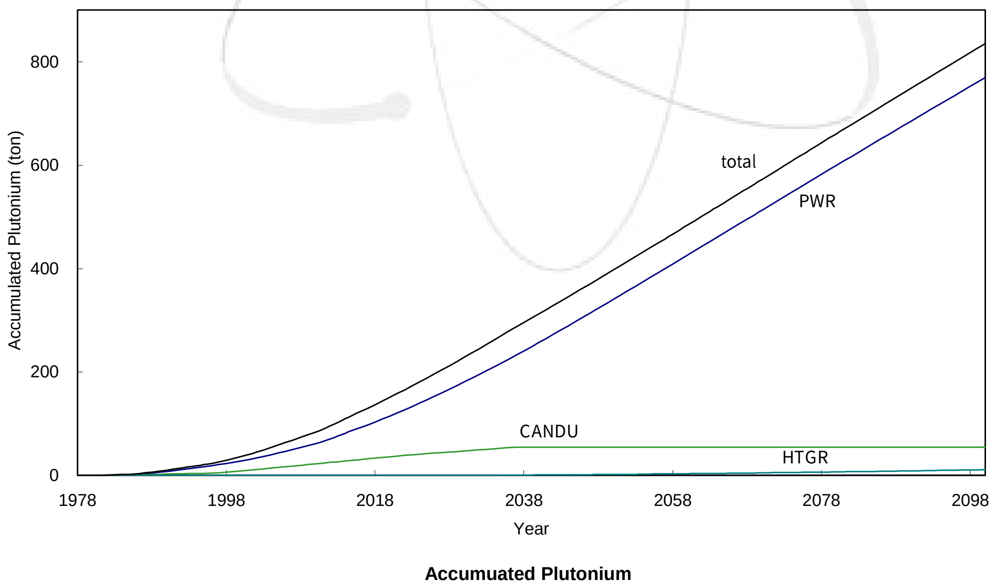

Fig. 56 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 3

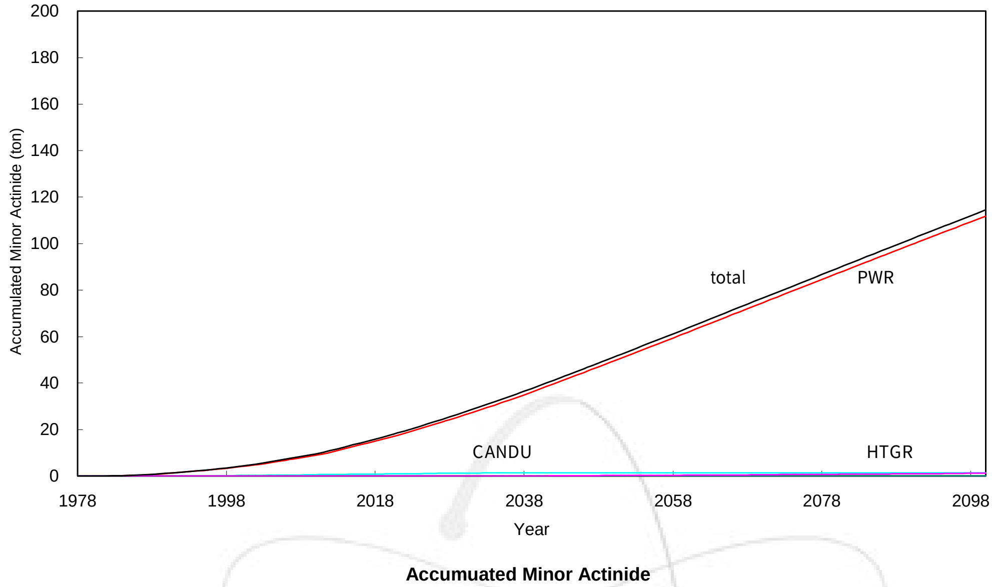

Fig. 57 Accumulated MA Embedded in Spent Fuels of Scenario Group 3

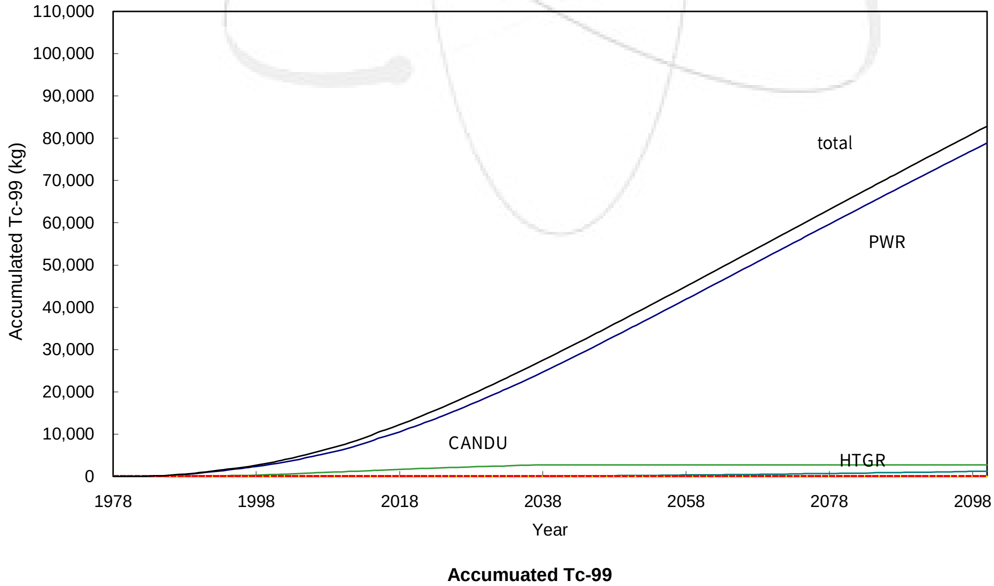

Fig. 58 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 3

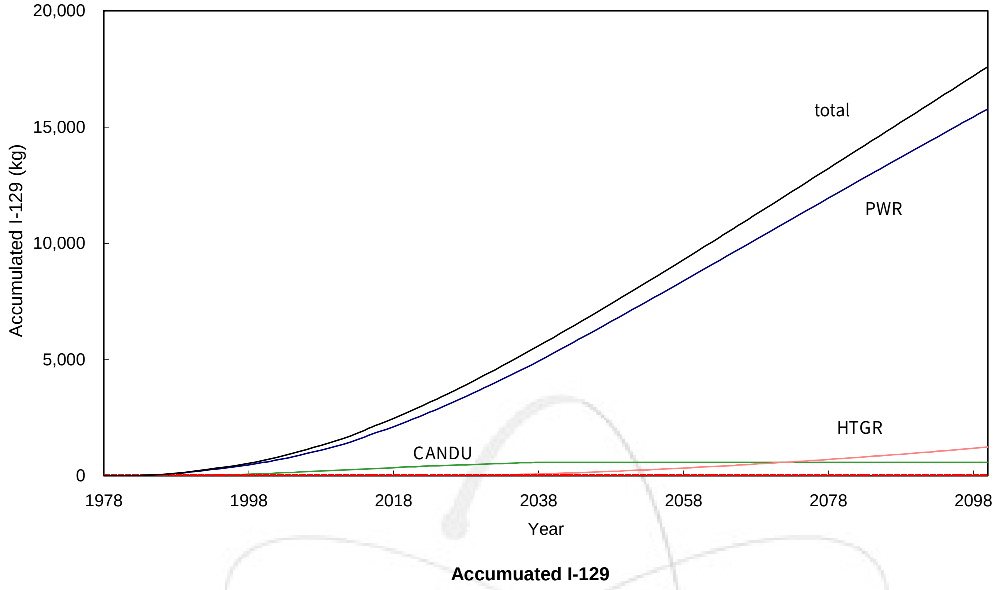

Fig. 59 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 3

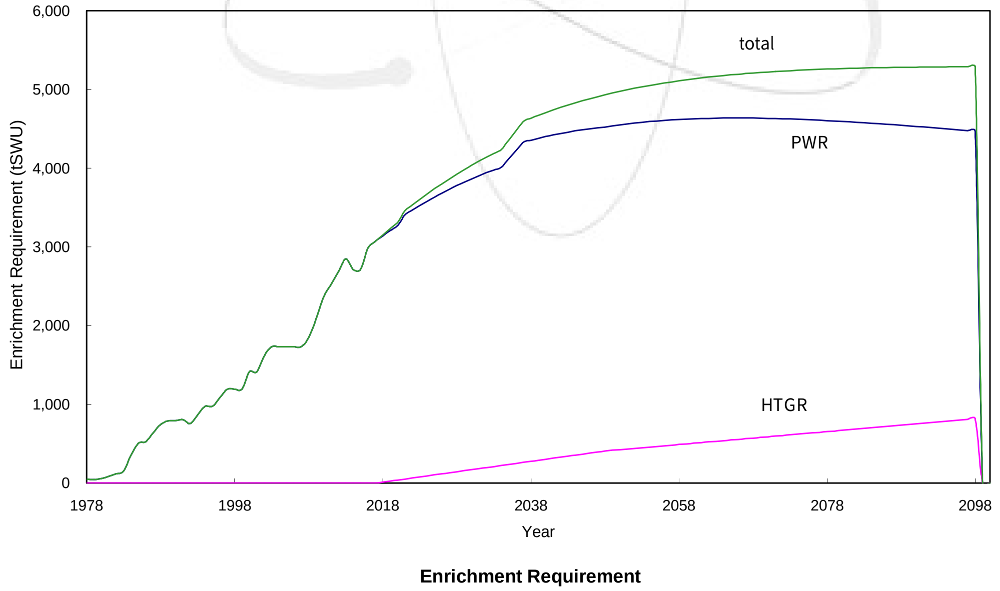

Fig. 60 Enrichment Requirement of Scenario Group 3

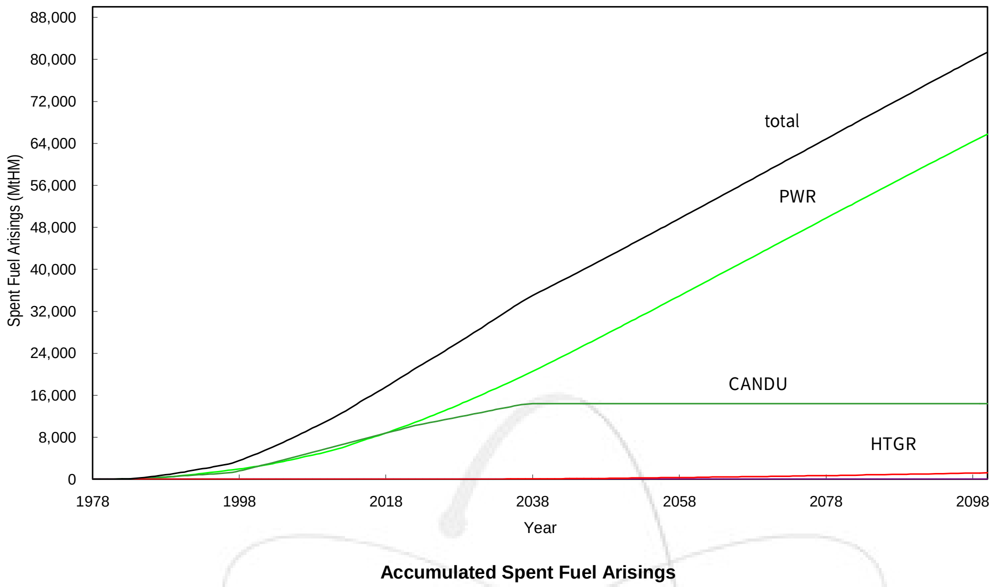

Fig. 61 Accumulated Spent Fuel Arisings of Scenario Group 3

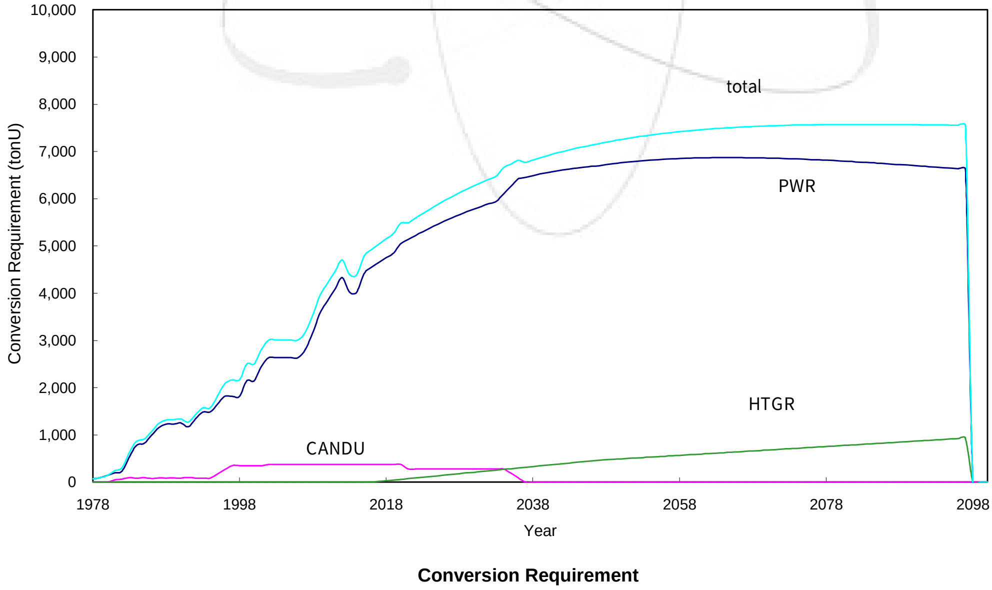

Fig. 62 Conversion Requirement of Scenario Group 3

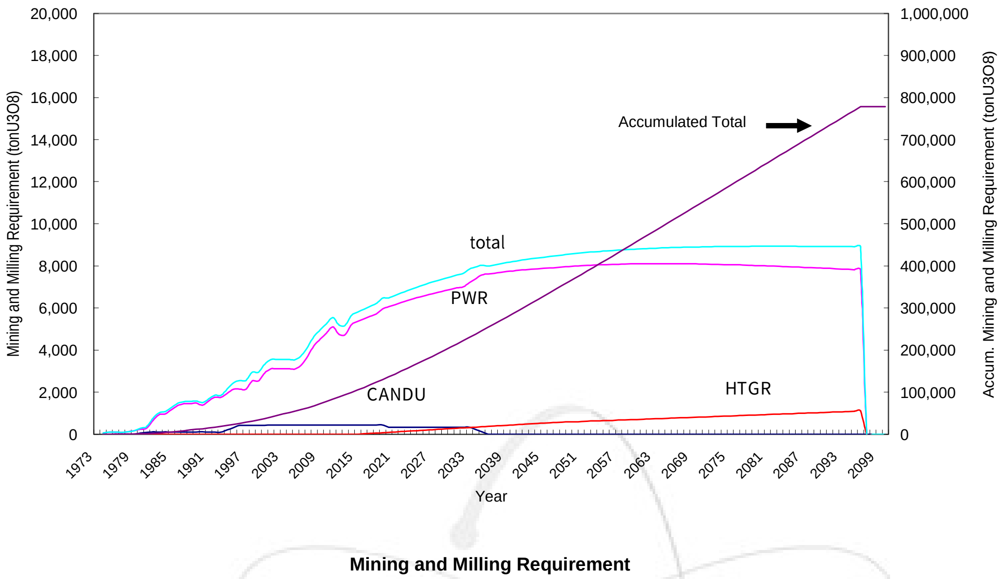

Fig. 63 Mining and Milling Requirement of Scenario Group 3

5.5 Scenario Group 4

5.5.1 Scenario Parameters (PWR (MOX) + CANDU)

Scenario Group 4 is the thermal recycle system in which PWR spent fuel is wet-reprocessed and reused as MOX fuel in existing PWRs. The MOX fuel is made by use of depleted U and reprocessed Pu. After 2017, CANDU reactors are gradually phased out. It is assumed that the MOX fuel is introduced from 2020 and the core ratio of MOX fuel is 15%. The reprocessing of PWR spent fuels and MOX fuel fabrication will start from 2018 and 2019, respectively.

5.5.2 Reactor Parameters

- Reactor parameters for the PWR and the CANDU are the same as those of the Scenario Group 2.

5.5.3 Fuel Parameters

- Fuel parameters for LEU of PWR and CANDU are the same as those of the Scenario Group 2.

- Fresh MOX Fuel Content

Pu	U235	MA	Tc99	I129	DepU
5.0%	0.25%	0.0%	0.0%	0.0%	95.0%

- Spent MOX Fuel Contents

Pu	U235	MA	Tc99	I129	U	Consumption rate
3.60%	0.10%	0.45%	0.116%	0.034%	56.5%	3.0%

5.5.4 Outputs

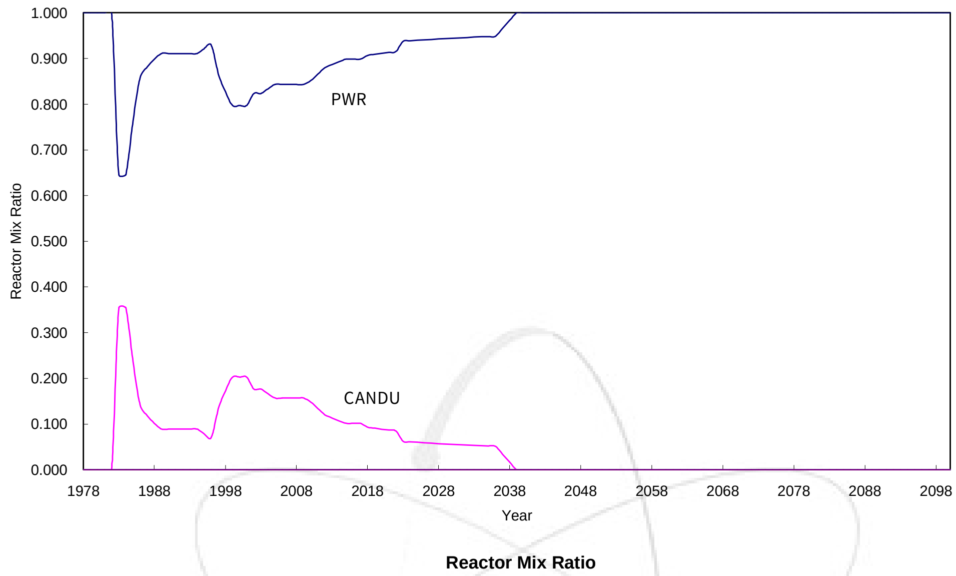

Fig. 64 Reactor Mix Ratio of Scenario Group 4

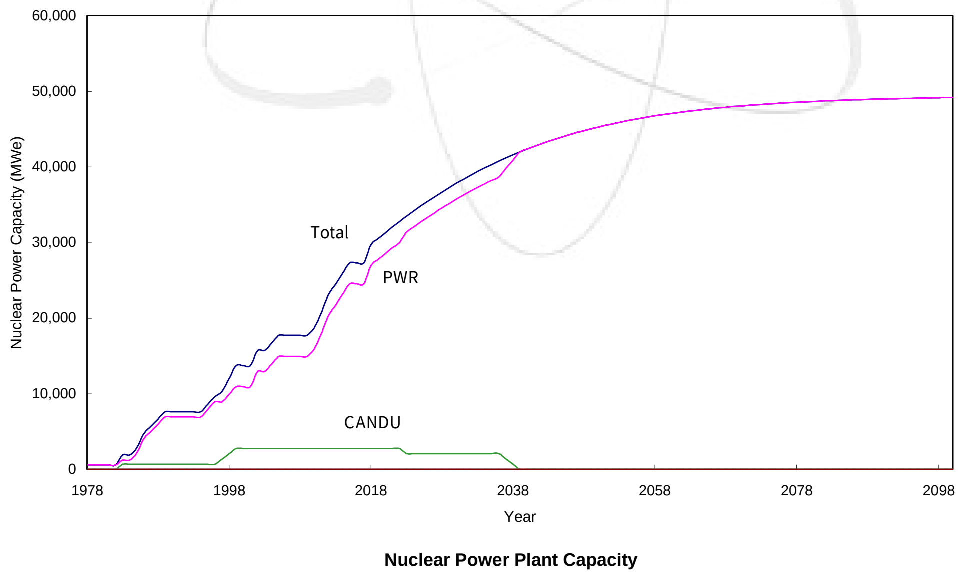

Fig. 65 Nuclear Power Plant Capacity of Scenario Group 4

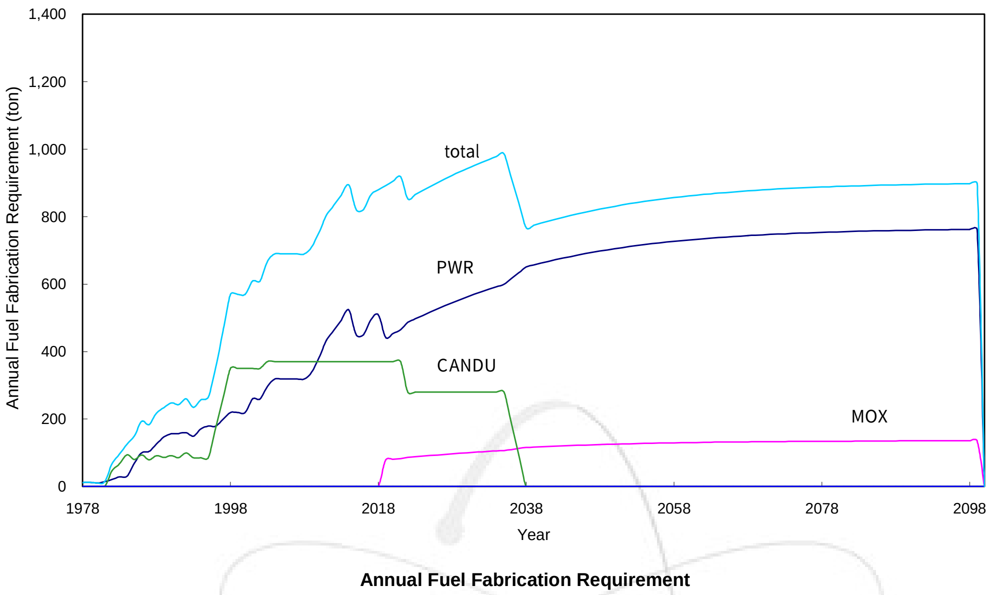

Fig. 66 Annual Fuel Fabrication Requirement of Scenario Group 4

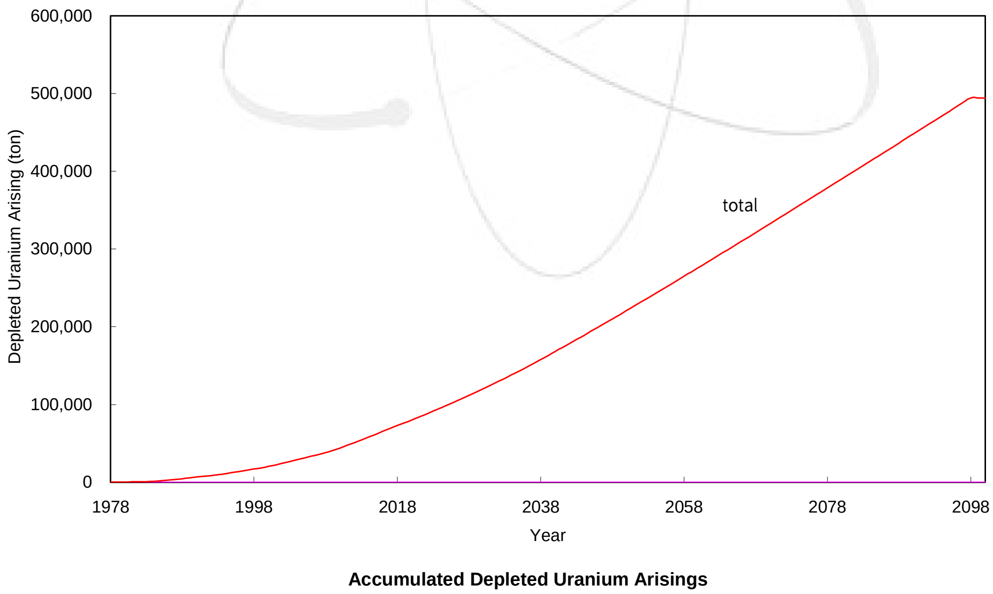

Fig. 67 Accumulated Depleted Uranium Arisings of Scenario Group 4

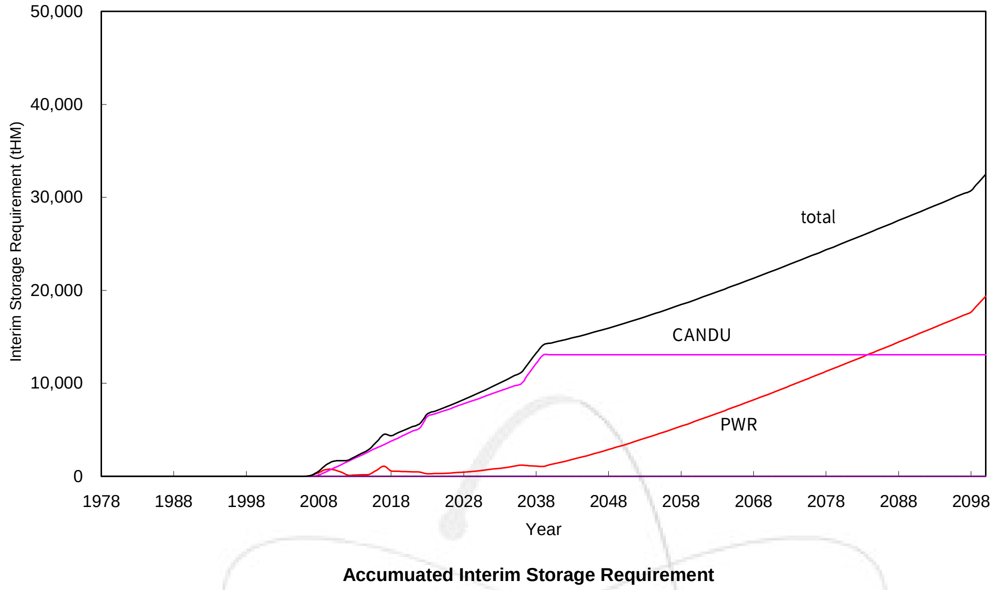

Fig. 68 Accumulated Interim Storage Requirement of Scenario Group 4

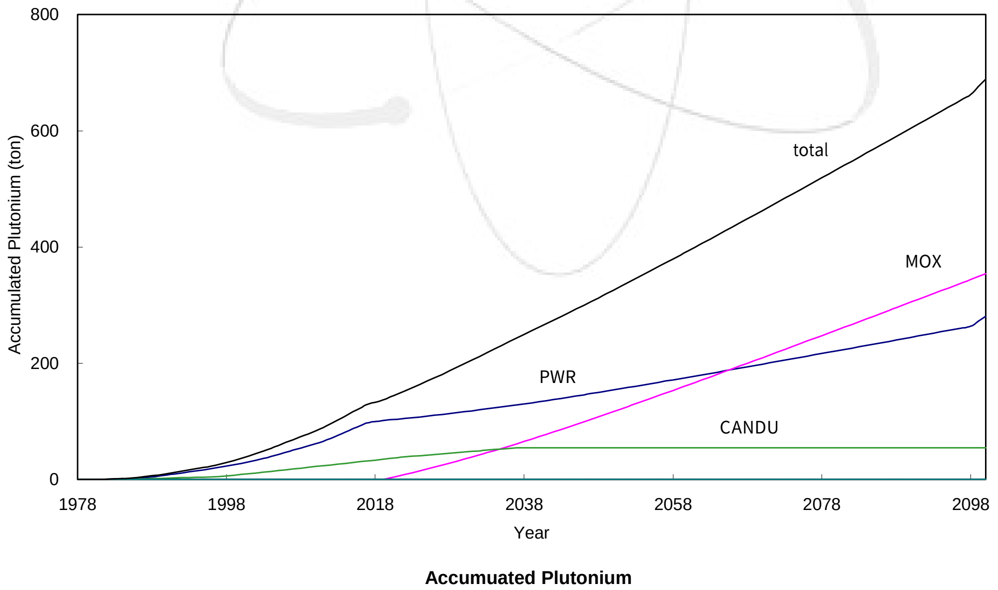

Fig. 69 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 4

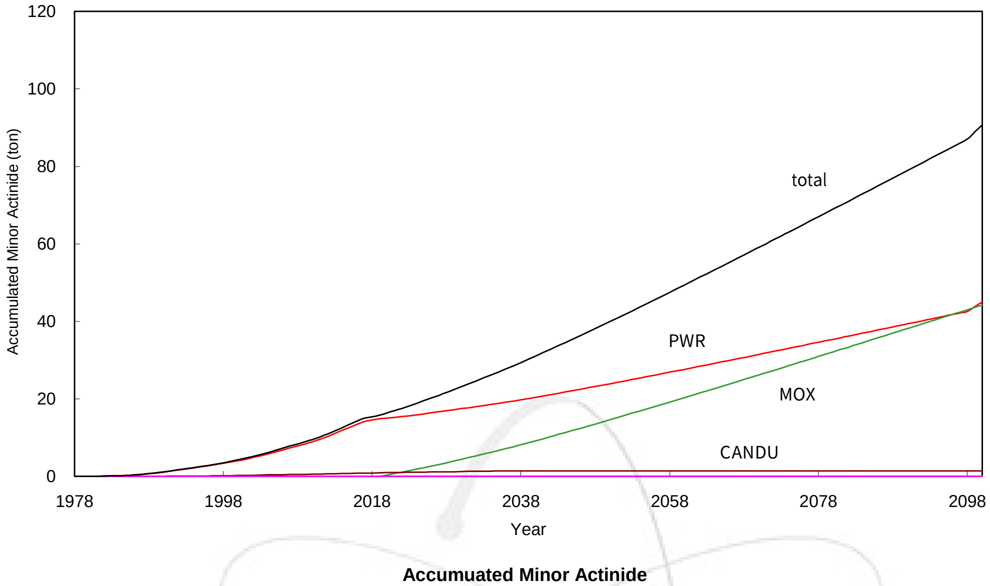

Fig. 70 Accumulated MA Embedded in Spent Fuels of Scenario Group 4

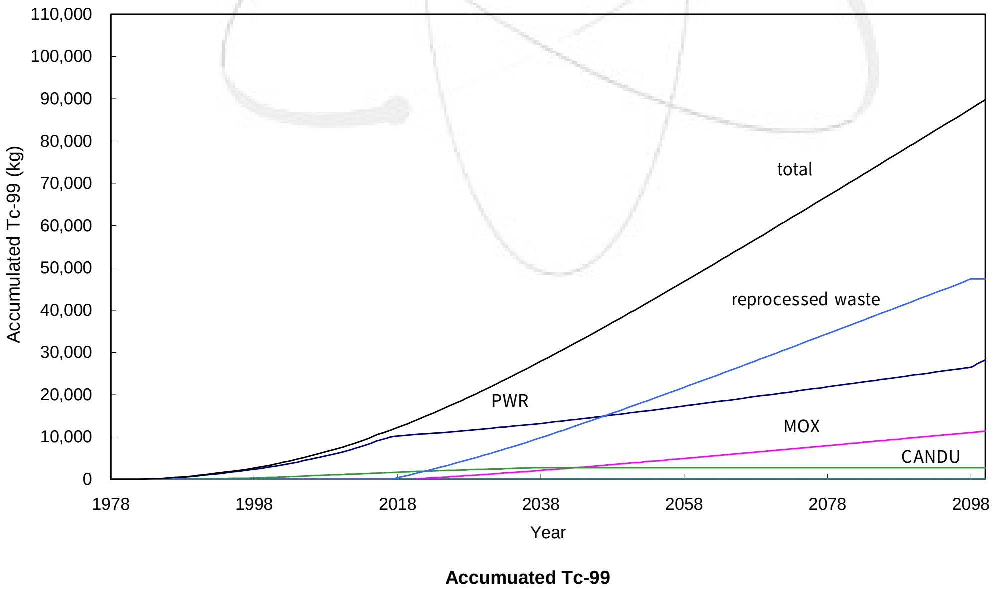

Fig. 71 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 4

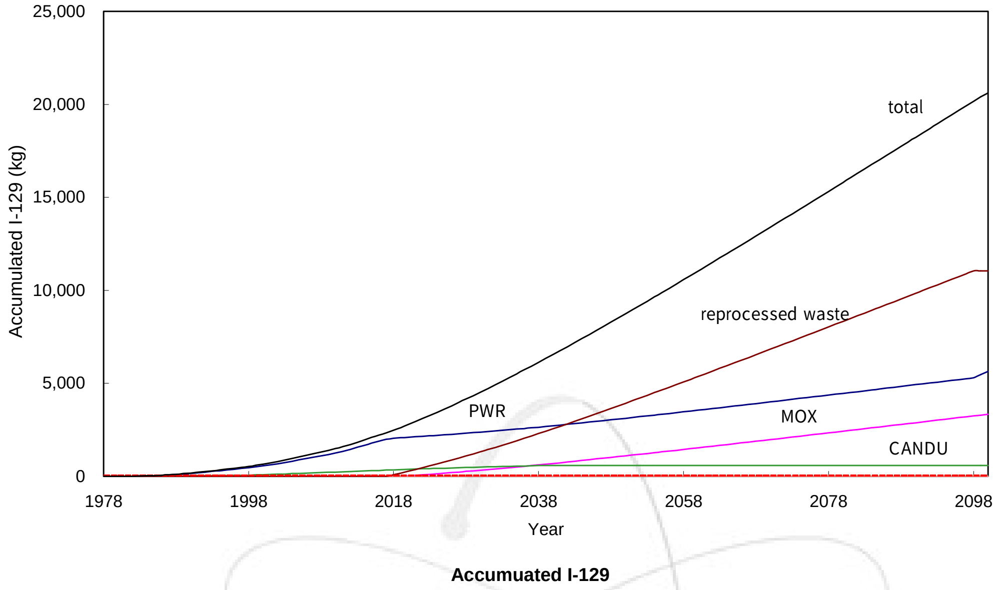

Fig. 72 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 4

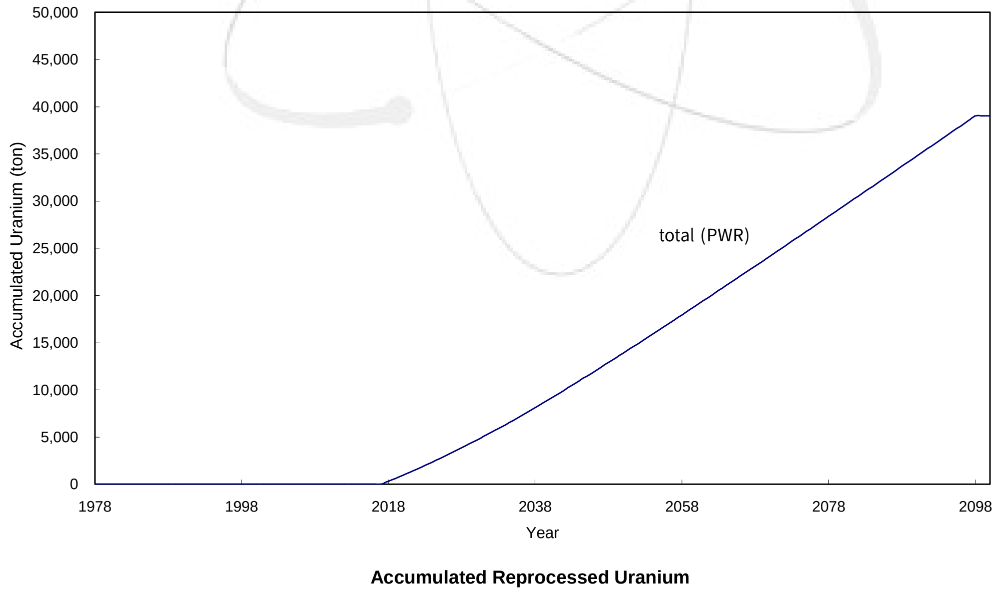

Fig. 73 Accumulated Reprocessed Uranium of Scenario Group 4

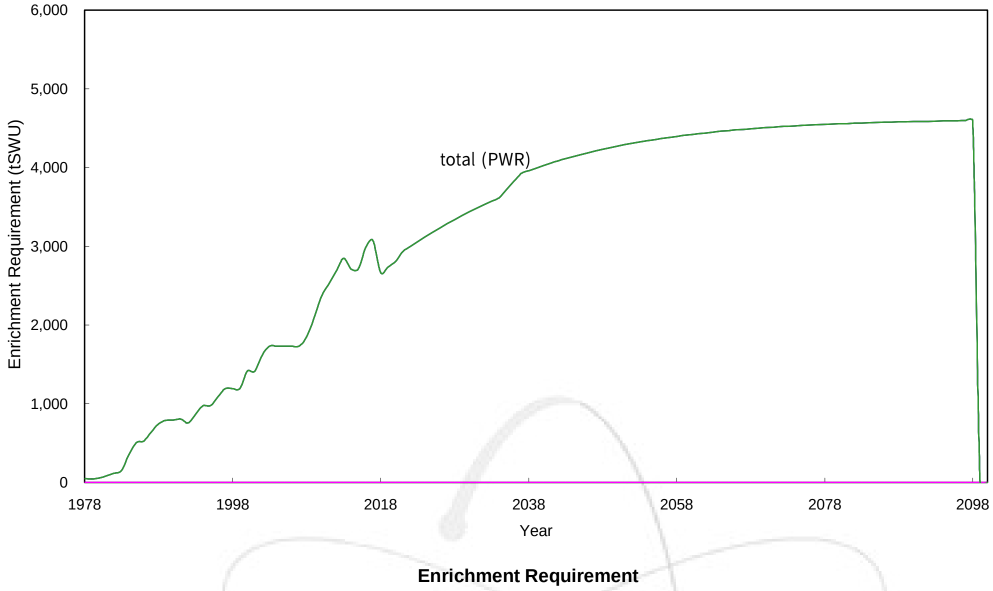

Fig. 74 Enrichment Requirement of Scenario Group 4

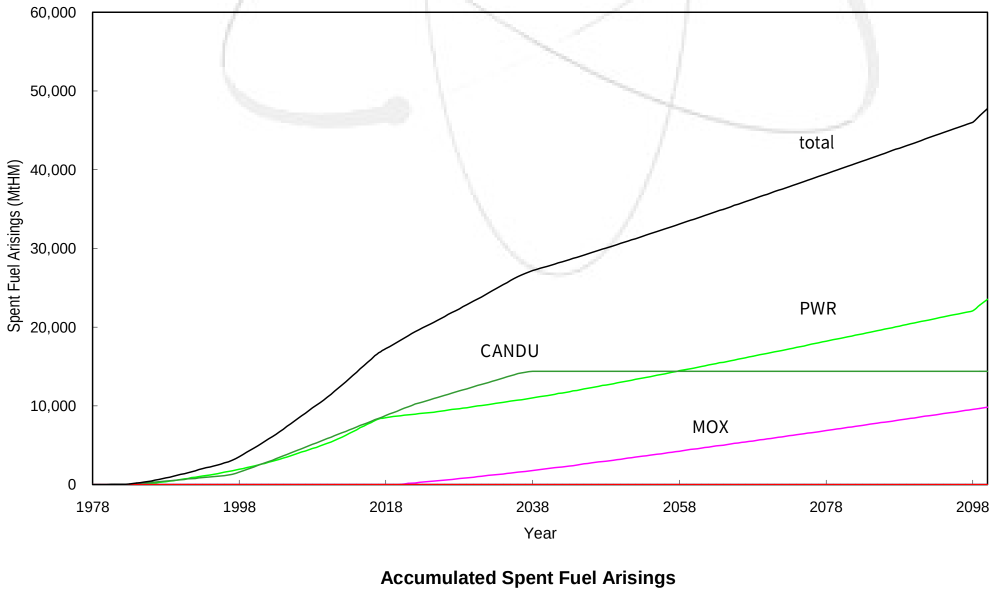

Fig. 75 Accumulated Spent Fuel Arisings of Scenario Group 4

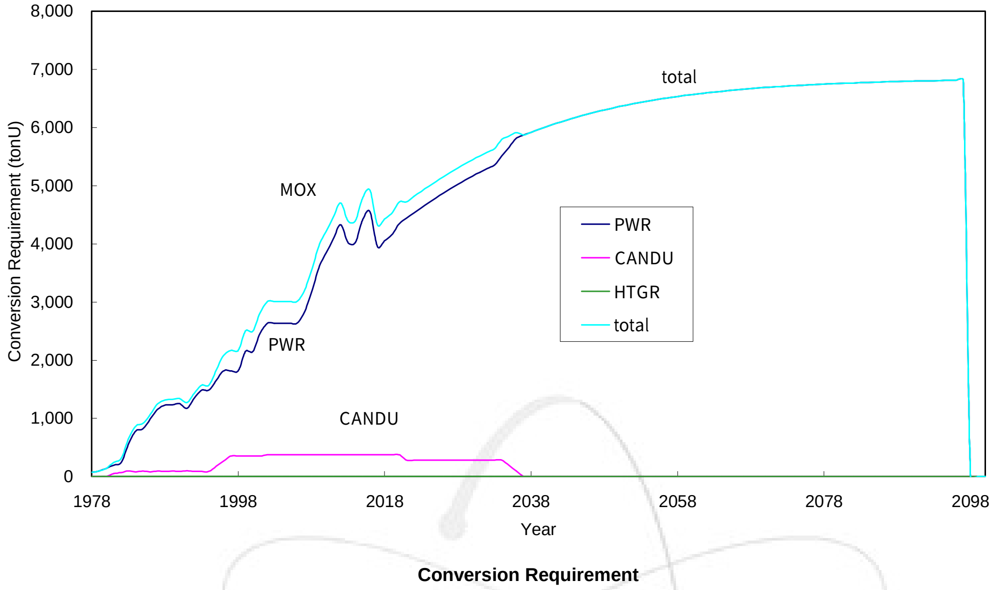

Fig. 76 Conversion Requirement of Scenario Group 4

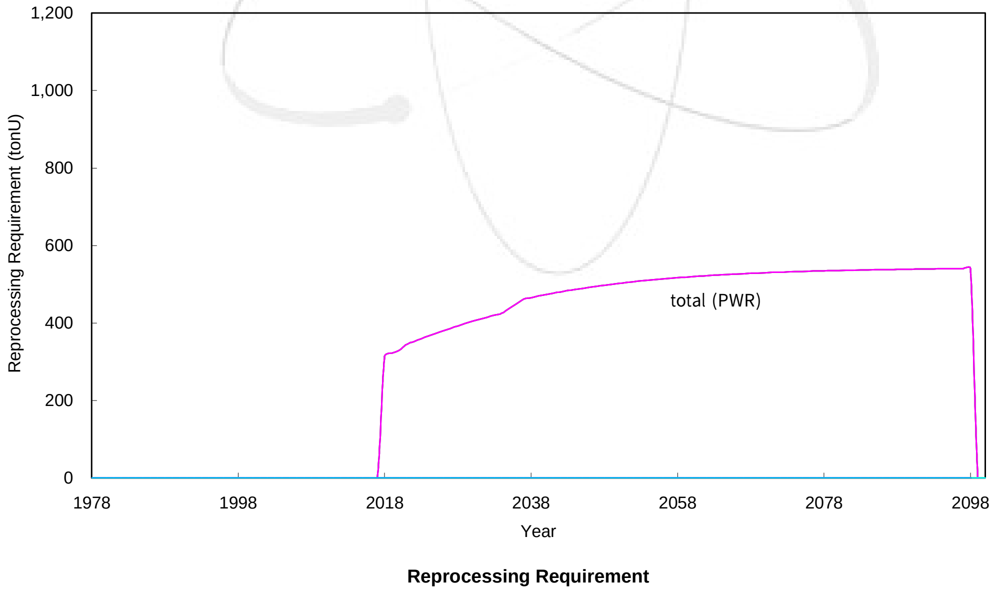

Fig. 77 Reprocessing Requirement of Scenario Group 4

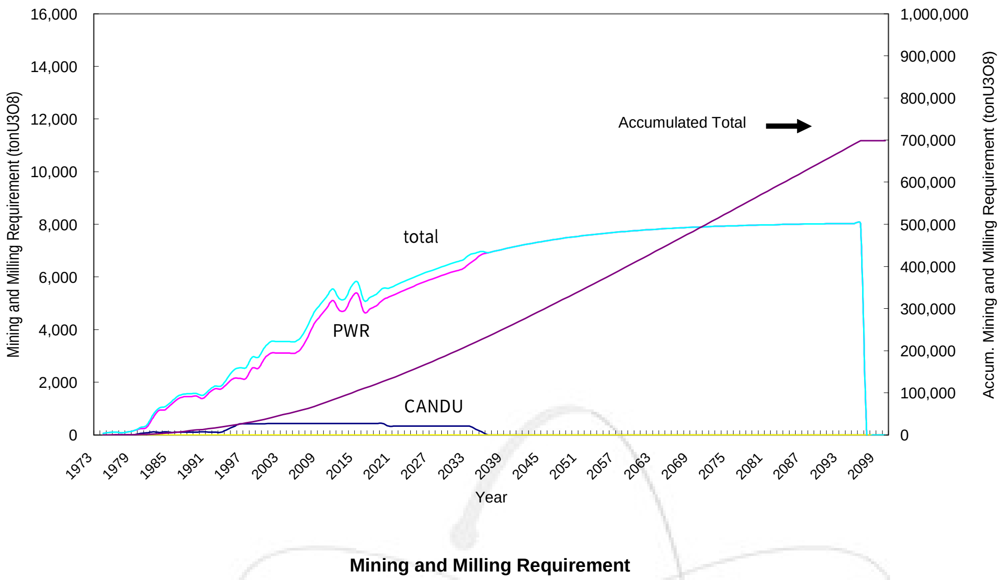

Fig. 78 Mining and Milling Requirement of Scenario Group 4

5.6 Scenario Group 5

5.6.1 Scenario Parameters (PWR + CANDU + SFR)

Scenario Group 5 is including the SFR system in which the spent SFR fuel is also reprocessed and continuously recycled in SFRs. PWR spent fuel is also reprocessed and reused in SFRs according to the balance of the plutonium production in SFR. The blanket fuel in the SFR in which the blanket fuel is made by depleted uranium is optional. In this case study, it is assumed that only driver fuel is used which means the power capacity of 600 MWe.

It is also assumed that the SFR is introduced from 2030 and its grid ratio is expanded up to 6% by 2050 and 30 % by 2100.

5.6.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the group 2.
- Load factor of SFR : 80%
- Thermal efficiency of SFR : 40%
- Pool capacity : The storage pool of SFR can employ spent fuels generated for 10 years.
- Blanket Fuel Type (150MW)

		blanket	driver
Burnup	Average burnup (GWd/tU) of 150MWe SFR	15.63	87.60
	internal blanket	17.9	
	radial blanket	14.5	
Power fraction	Average power fraction	0.061	0.939
	Power fraction for 150MWe SFR	BOC	EOC
	internal blanket	0.055	0.104
	radial blanket	0.041	0.063
	internal blanket(No. of FA)	24	
	radial blanket(No. of FA)	48	
		blanket	driver
	Average power fraction	0.061	0.939

- The average burnup and power fraction are calculated considering the ratio of the number of each fuel assembly.

- Burnup of 600MWe SFR : 66.63 GWd/tU

5.6.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as those of the Scenario Group.

- Fresh SFR fuel content : The average content is calculated considering the ratio of the number of each fuel assembly.

SFR (Inner Blanket,17.9GWd/MtHM)					
Pu	U235	MA	Tc99	I129	U
0.000%	0.250%	0.000%	0.000%	0.000%	100.000%
SFR (Radial Blanket,14.5GWd/MtHM)					
Pu	U235	MA	Tc99	I129	U
0.000%	0.250%	0.000%	0.000%	0.000%	100.000%
SFR (Avg. Blanket,15.63GWd/MtHM)					
Pu	U235	MA	Tc99	I129	U
0.000%	0.250%	0.000%	0.000%	0.000%	100.000%
SFR (Drive Fuel, 87.6GWd/MtHM)					
Pu	U235	MA	Tc99	I129	U
28.277%	0.078%	0.893%	0.084%	0.019%	70.830%

- Spent SFR fuel contents

SFR (Inner Blanket,17.9GWd/MtHM)						
Pu	U235	MA	Tc99	I129	U	Consump. rate
3.18%	0.14%	0.02%	0.029%	0.007%	96.8%	3.0%
SFR (Radial Blanket,14.5GWd/MtHM)						
Pu	U235	MA	Tc99	I129	U	Consump. rate
2.61%	0.15%	0.01%	0.019%	0.004%	97.4%	3.0%
SFR (Avg. Blanket,15.63GWd/MtHM)						
Pu	U235	MA	Tc99	I129	U	Consump. rate
2.800%	0.146%	0.013%	0.022%	0.005%	97.2%	3.000%

SFR (Drive Fuel, 87.6GWd/MtHM)						
Pu	U235	MA	Tc99	I129	U	Consump. rate
27.47%	0.07%	0.90%	0.170%	0.039%	71.6%	3.1%

- Loss factor

	PWR	CANDU	SFR(B)	SFR(D)
Conversion	0.5%	0.5%		
Enrichment	1.0%			
Fabrication	0.5%	0.5%	0.5%	1.0%
Reprocessing	1.0%		1.0%	1.0%

5.6.4 Outputs

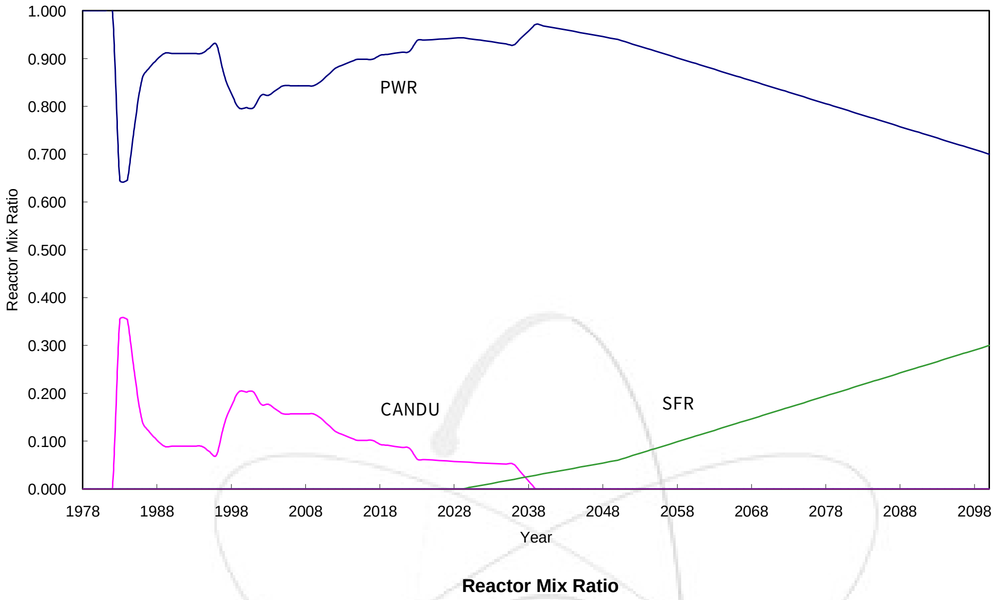

Fig. 79 Reactor Mix Ratio of Scenario Group 5

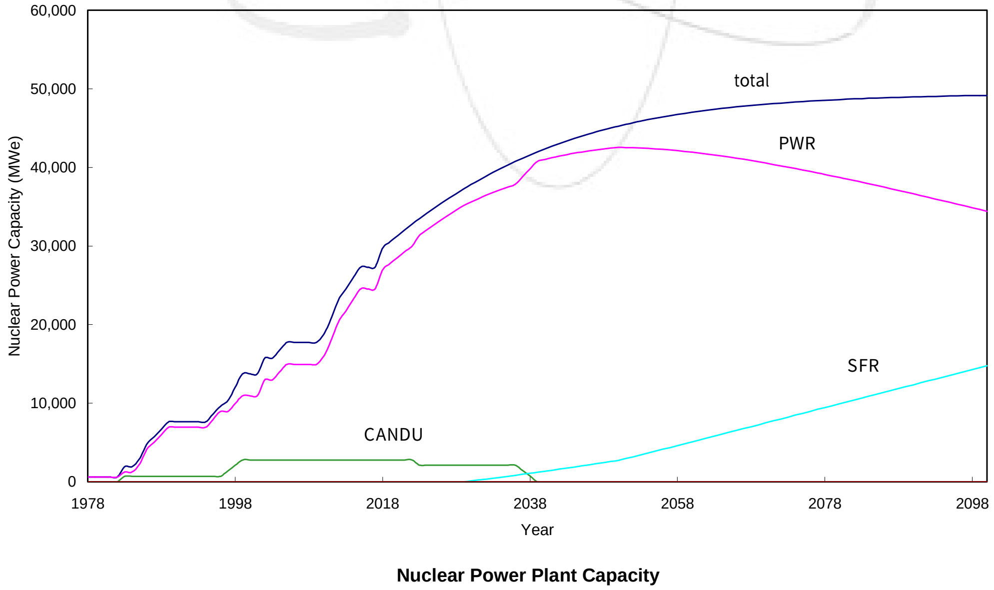

Fig. 80 Nuclear Power Plant Capacity of Scenario Group 5

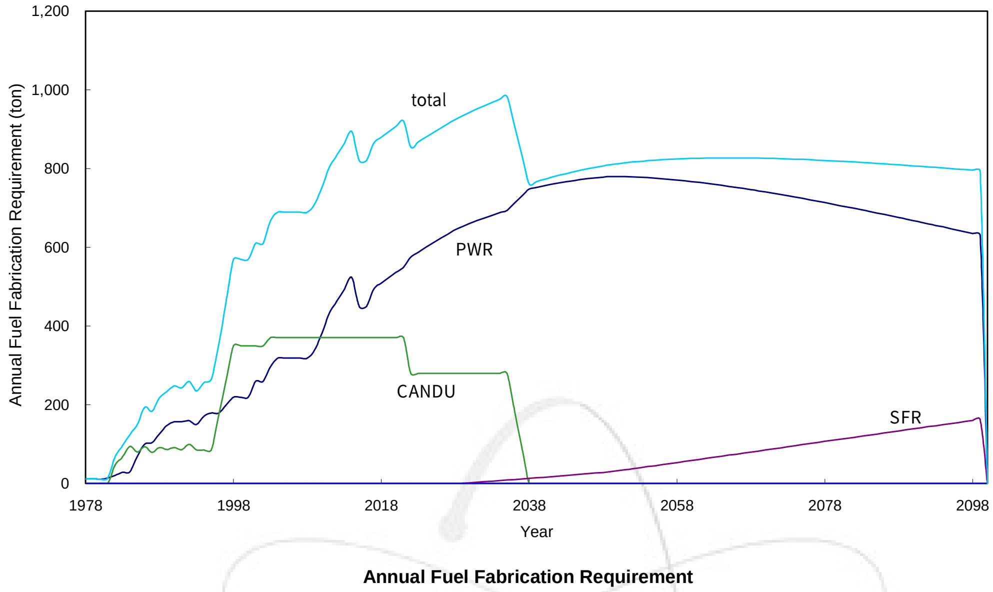

Fig. 81 Annual Fuel Fabrication Requirement of Scenario Group 5

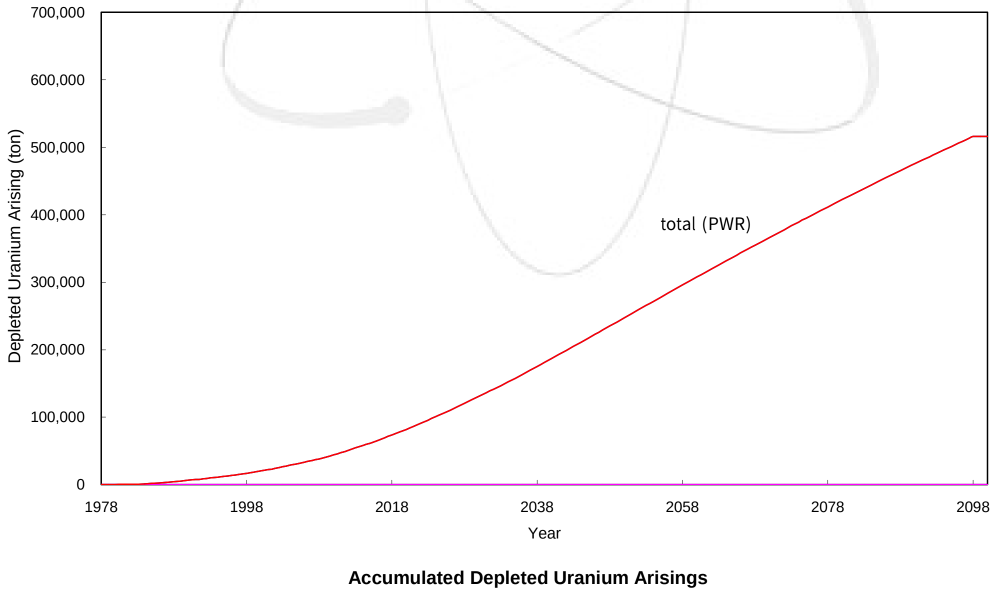

Fig. 82 Accumulated Depleted Uranium Arisings of Scenario Group 5

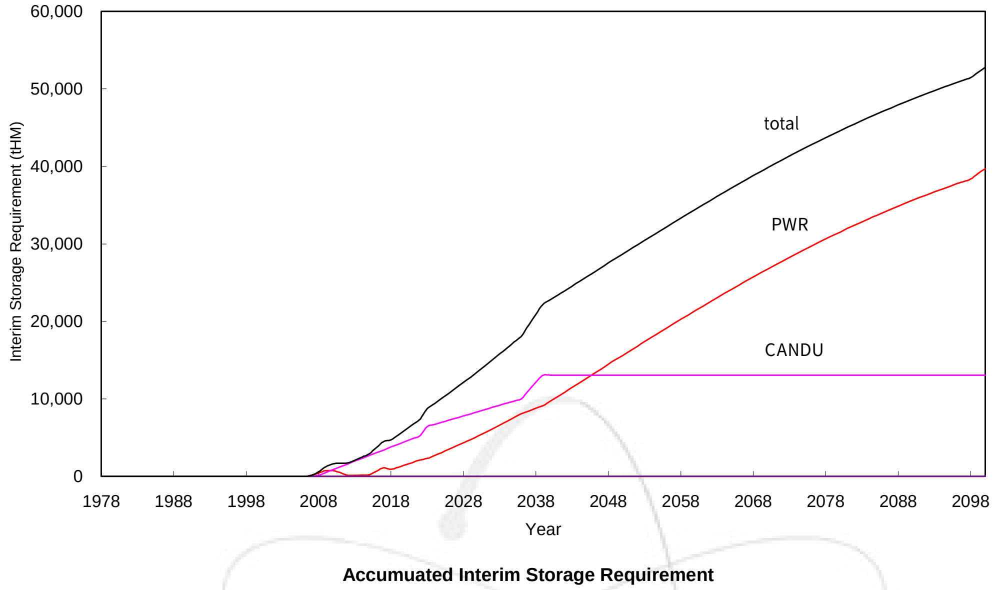

Fig. 83 Accumulated Interim Storage Requirement of Scenario Group 5

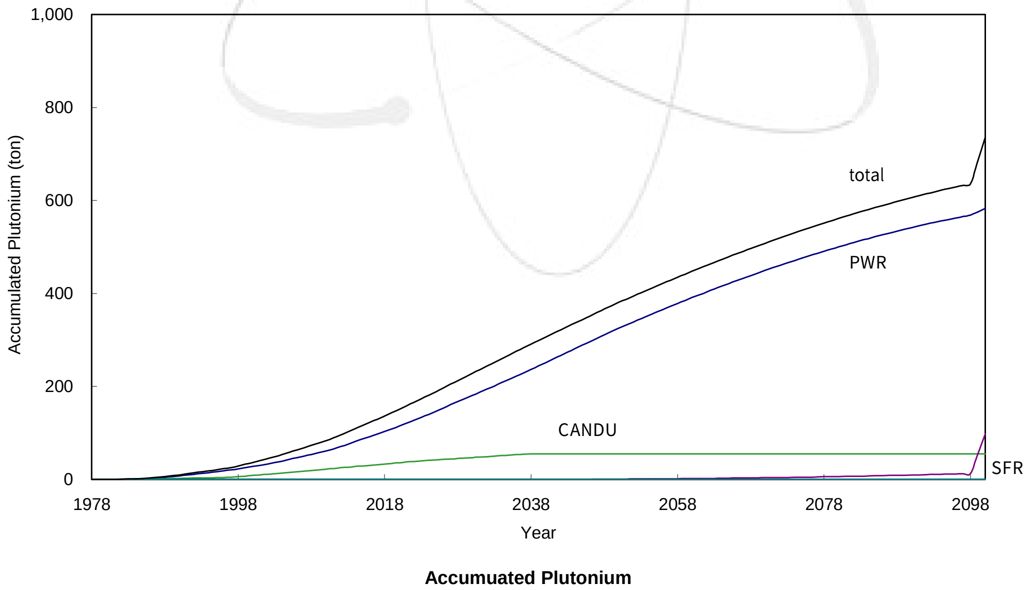

Fig. 84 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 5

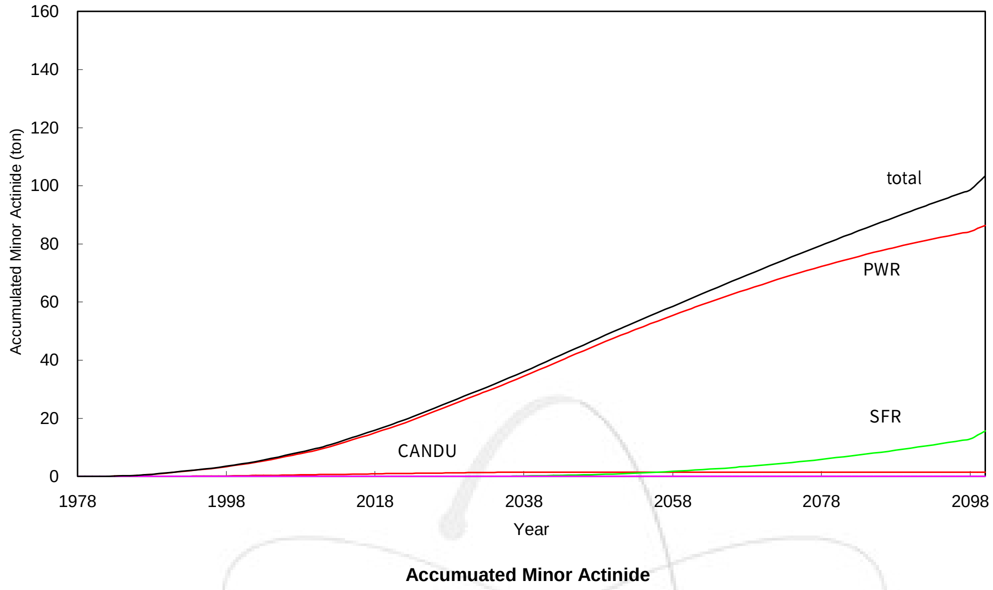

Fig. 85 Accumulated MA Embedded in Spent Fuels of Scenario Group 5

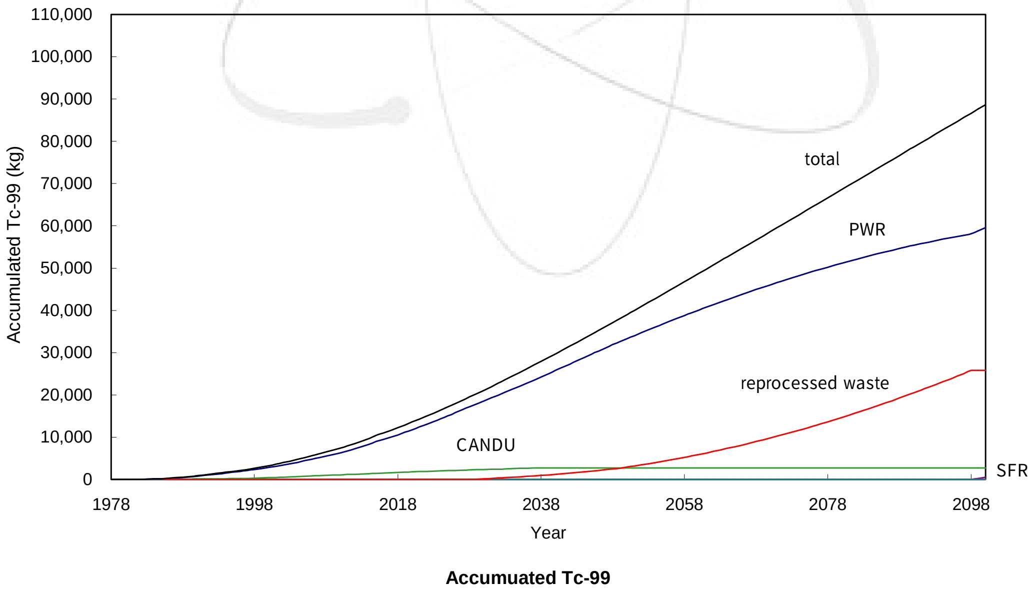

Fig. 86 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 5

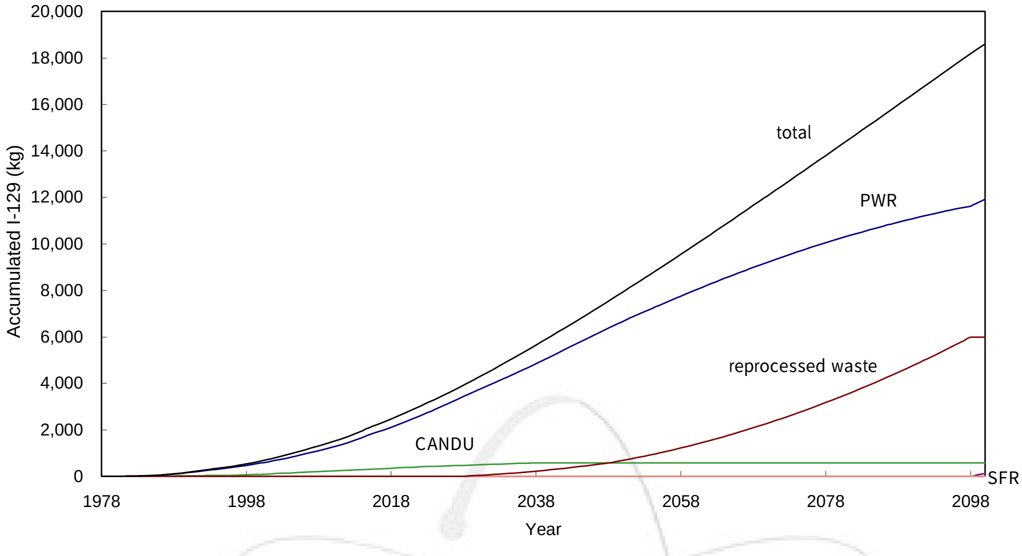

Fig. 87 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 5

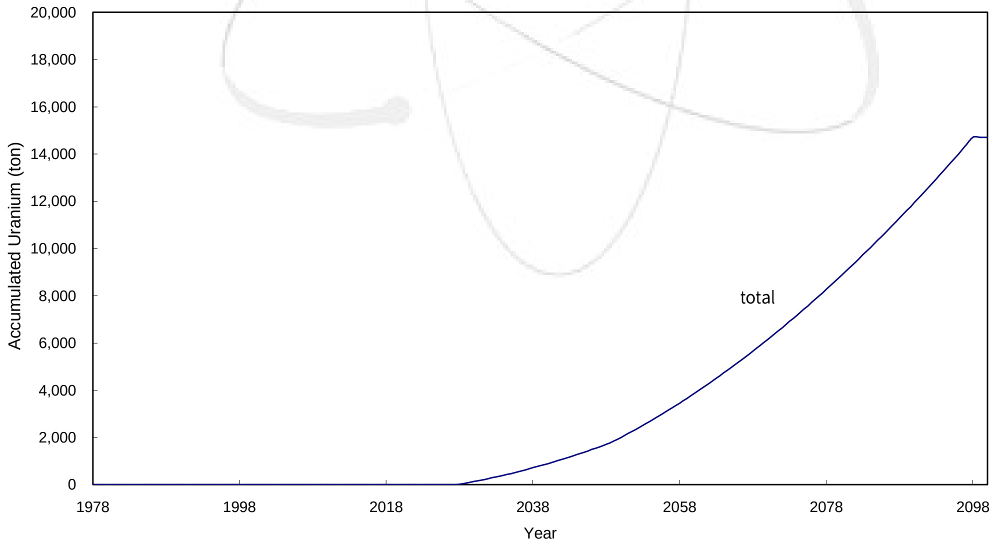

Fig. 88 Accumulated Reprocessed Uranium of Scenario Group 5

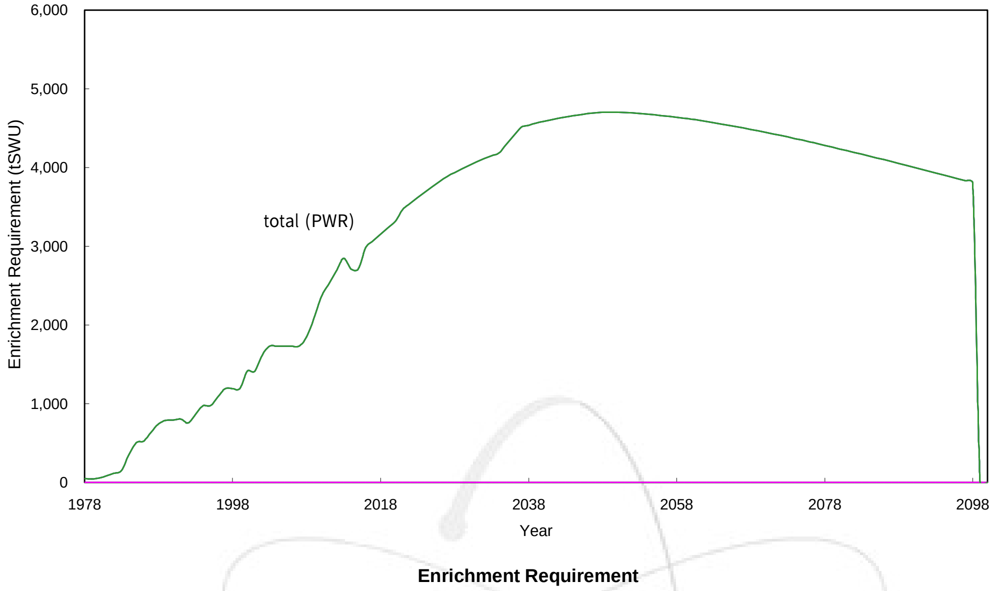

Fig. 89 Enrichment Requirement of Scenario Group 5

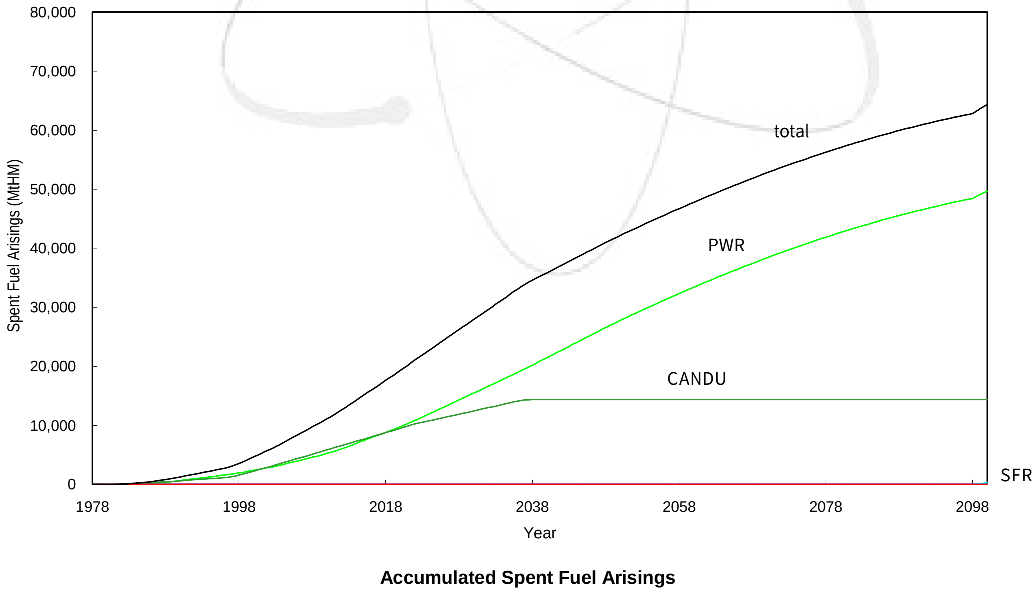

Fig. 90 Accumulated Spent Fuel Arisings of Scenario Group 5

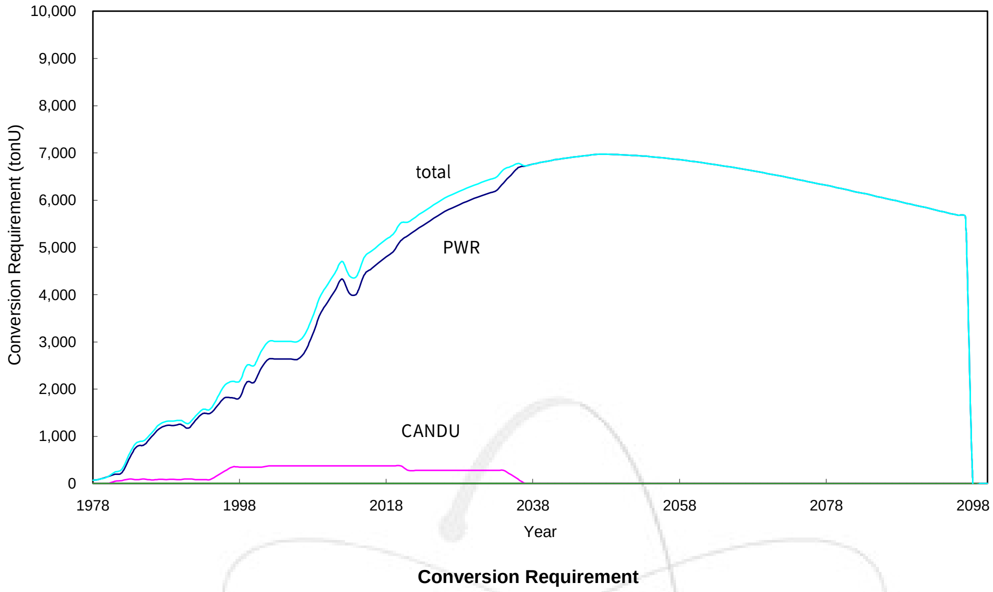

Fig. 91 Conversion Requirement of Scenario Group 5

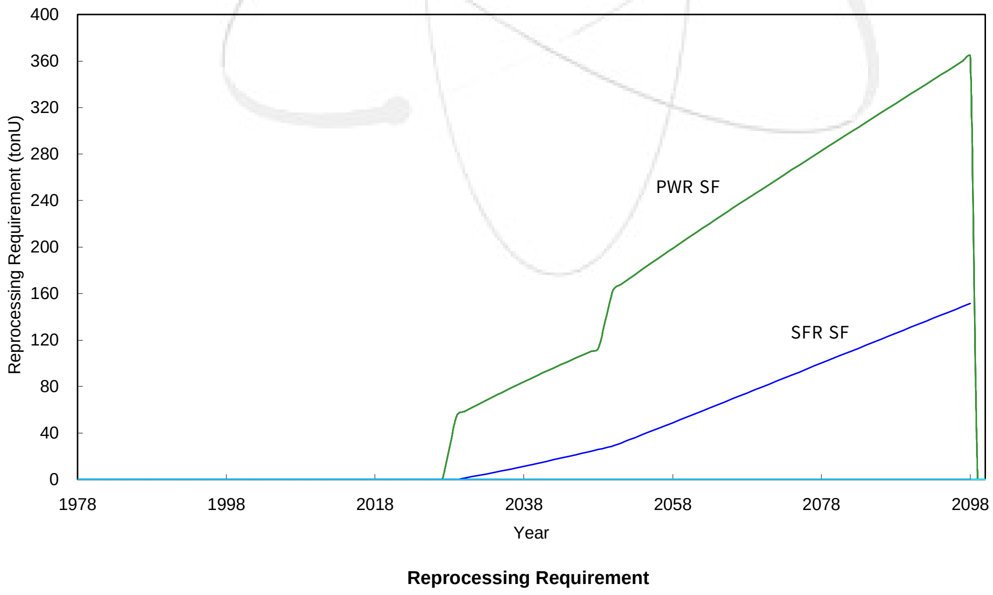

Fig. 92 Reprocessing Requirement of Scenario Group 5

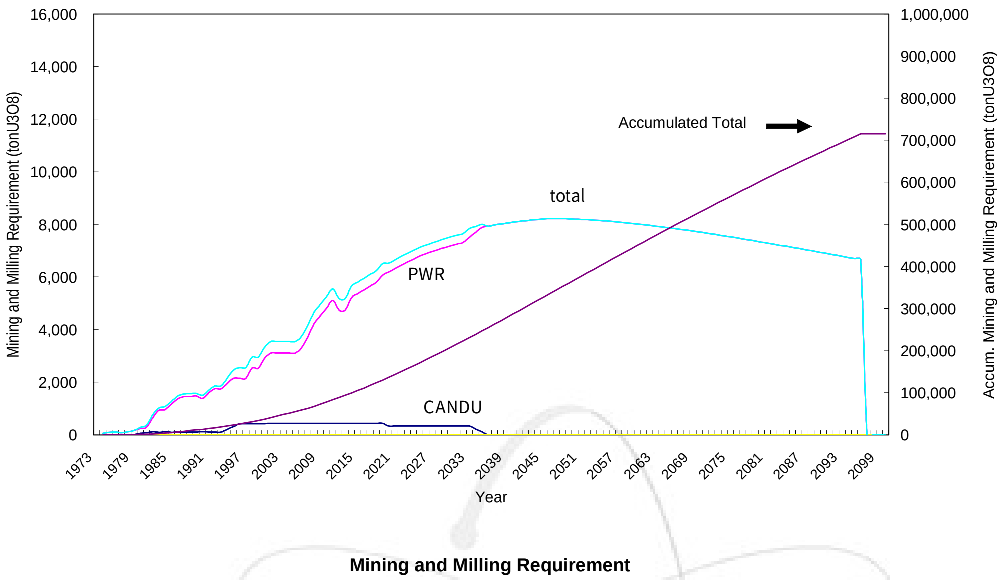

Fig. 93 Mining and Milling Requirement of Scenario Group 5

5.7 Scenario Group 6

5.7.1 Scenario Parameters (PWR + CANDU + HTGR + SFR)

Scenario Group 6 is similar to Scenario Group 5 except that the Scenario Group 6 has the HTGR in its grid. The PWR spent fuel is reprocessed and reused in SFRs. The spent SFR fuel is also reprocessed and continuously recycled in SFRs. It is assumed that the SFR is introduced from 2030 and its grid ratio is expanded up to 6% by 2050 and 30 % by 2100.

The HTGR uses both of hydrogen production and electricity generation. It is assumed that the HTGR is introduced from 2020 and their capacity is expanded up to 25GWth by 2050 and 50 GWth by 2100.

5.7.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Reactor parameters for SFR are the same as those of the Scenario Group 5.
- Reactor parameters for HTGR are the same as those of the Scenario Group 3.

5.7.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as the Scenario Group 2.
- Fuel parameters for SFR are the same as the Scenario Group 5.
- Fuel parameters for HTGR are the same as the Scenario Group 3.

5.7.4 Outputs

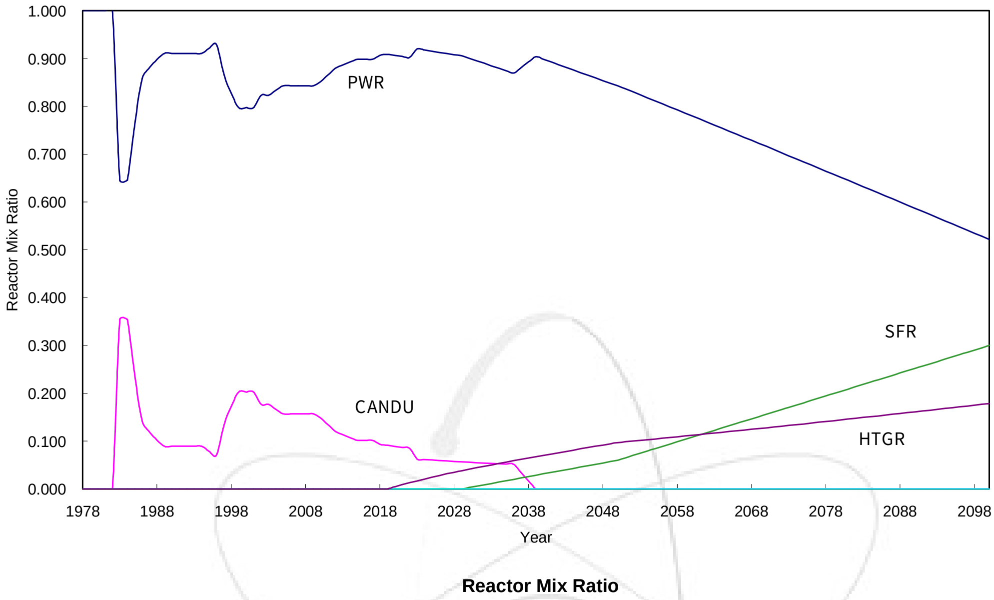

Fig. 94 Reactor Mix Ratio of Scenario Group 6

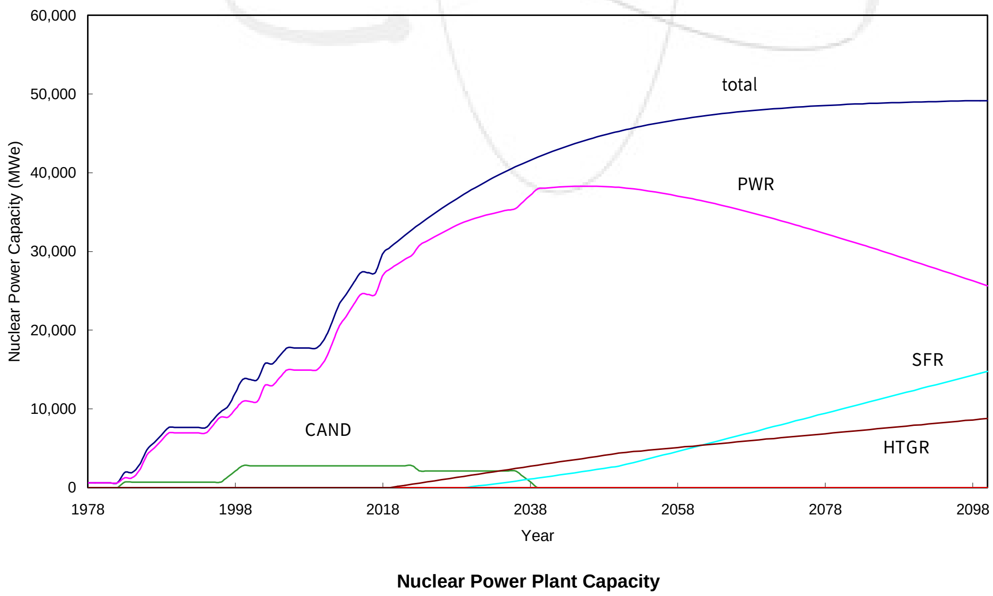

Fig. 95 Nuclear Power Plant Capacity of Scenario Group 6

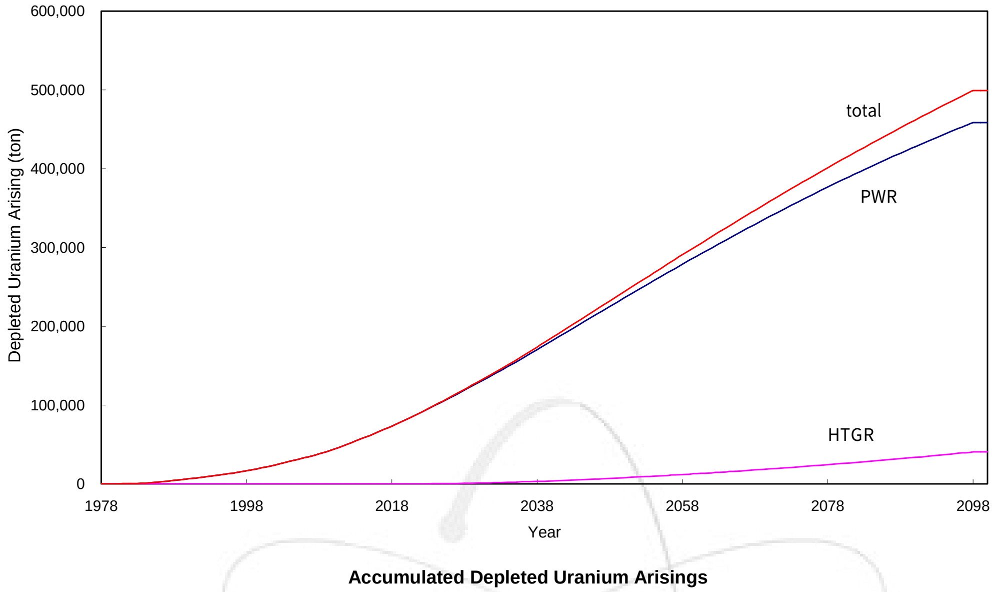

Fig. 96 Accumulated Depleted Uranium Arisings of Scenario Group 6

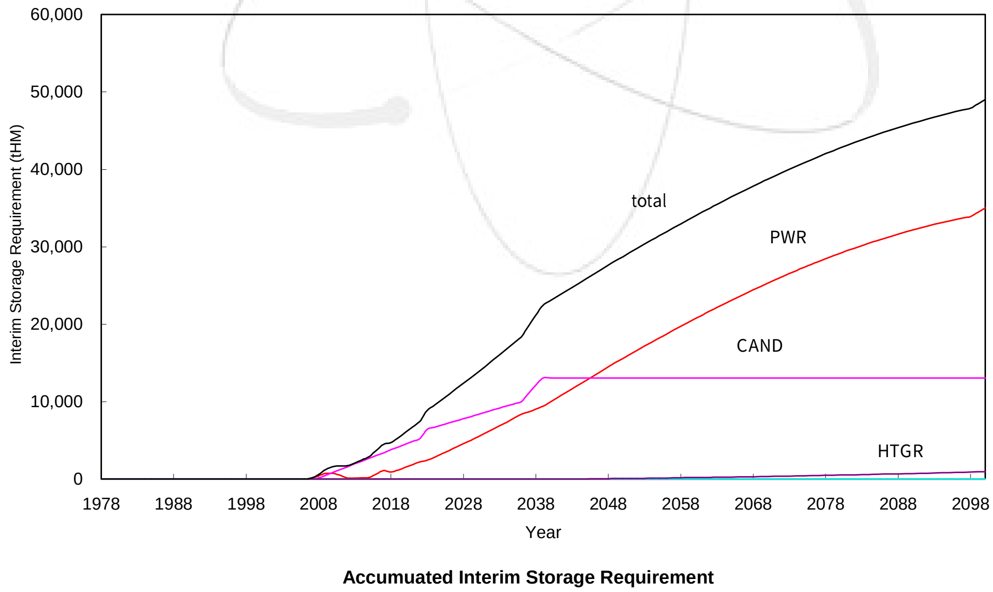

Fig. 97 Accumulated Interim Storage Requirement of Scenario Group 6

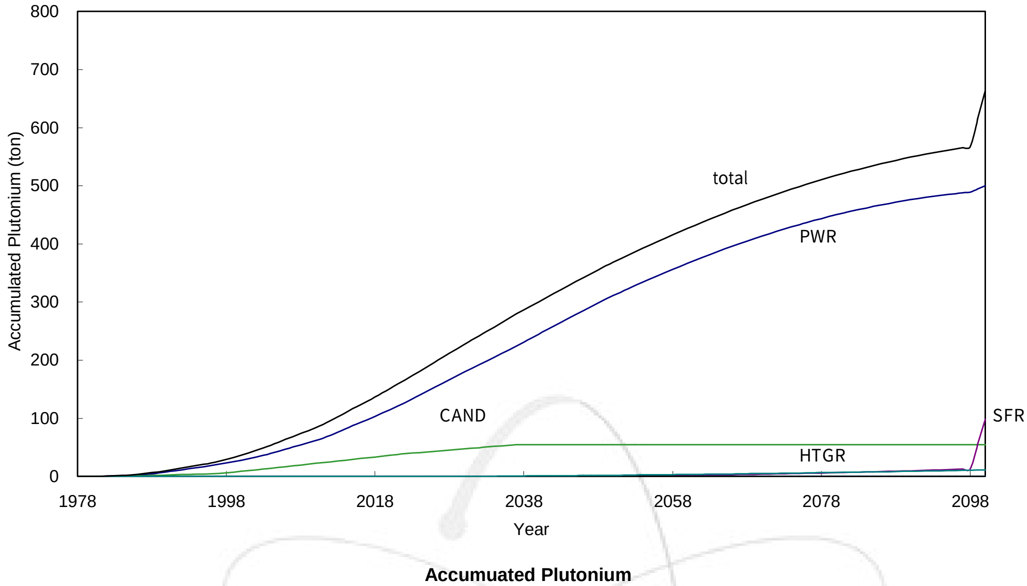

Fig. 98 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 6

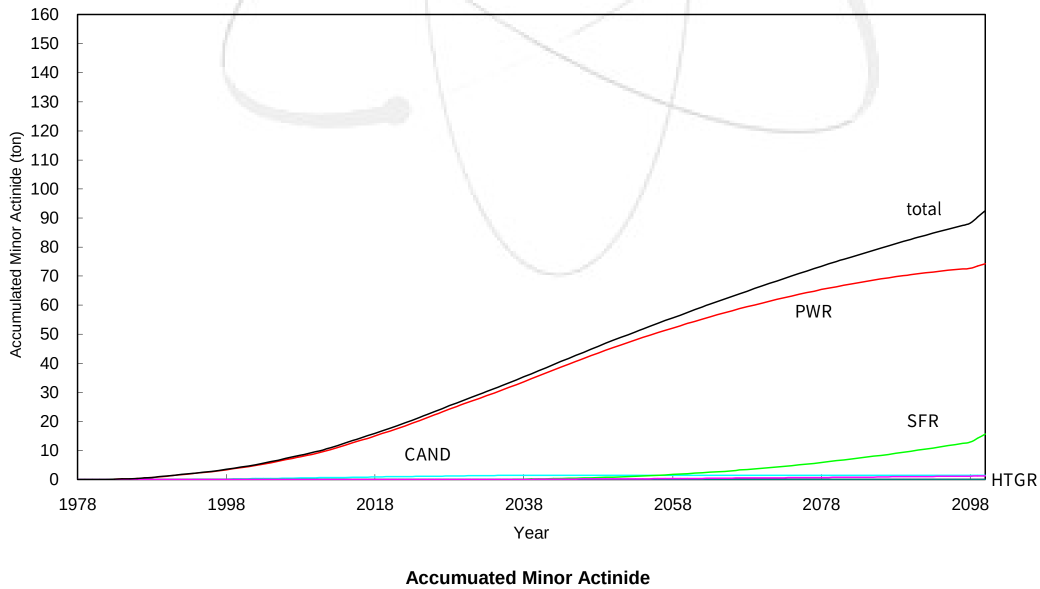

Fig. 99 Accumulated MA Embedded in Spent Fuels of Scenario Group 6

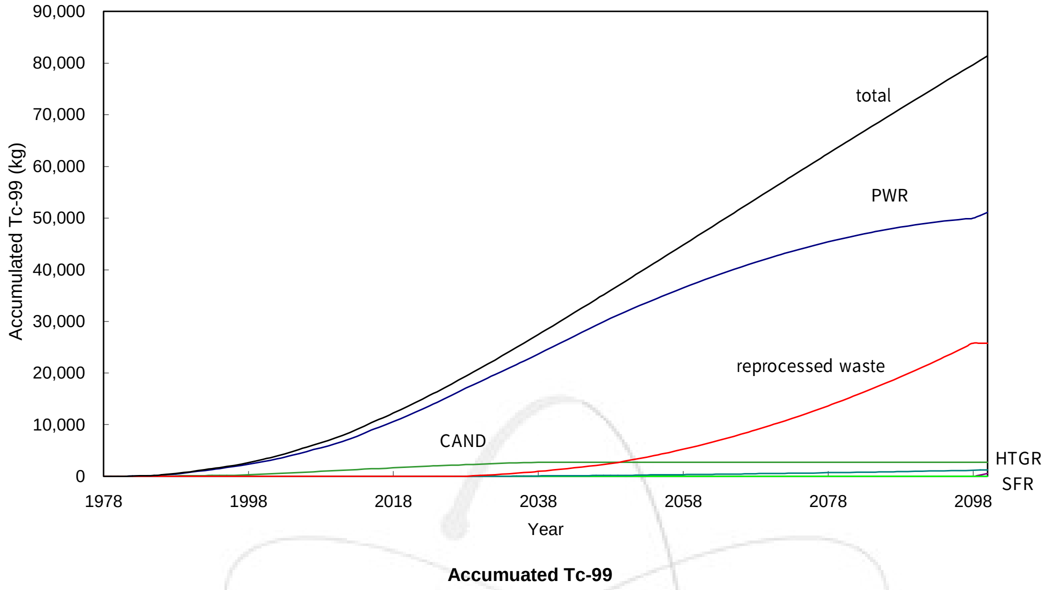

Fig. 100 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 6

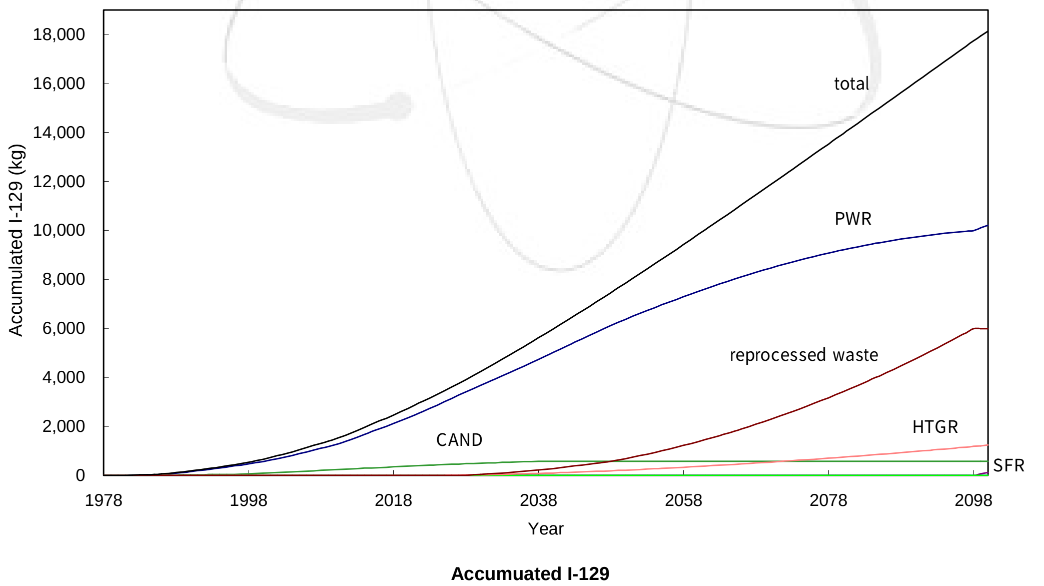

Fig. 101 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 6

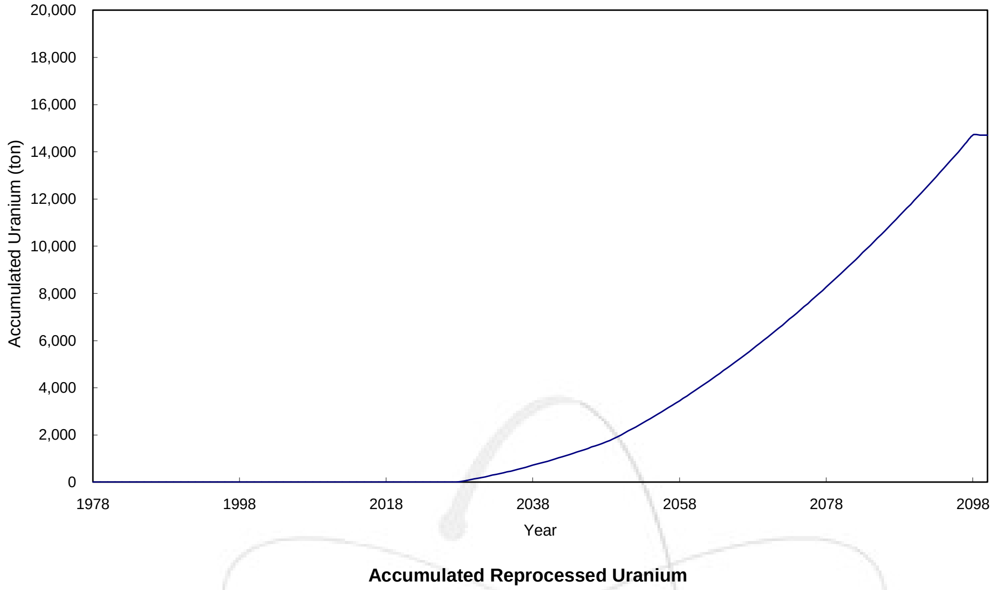

Fig. 102 Accumulated Reprocessed Uranium of Scenario Group 6

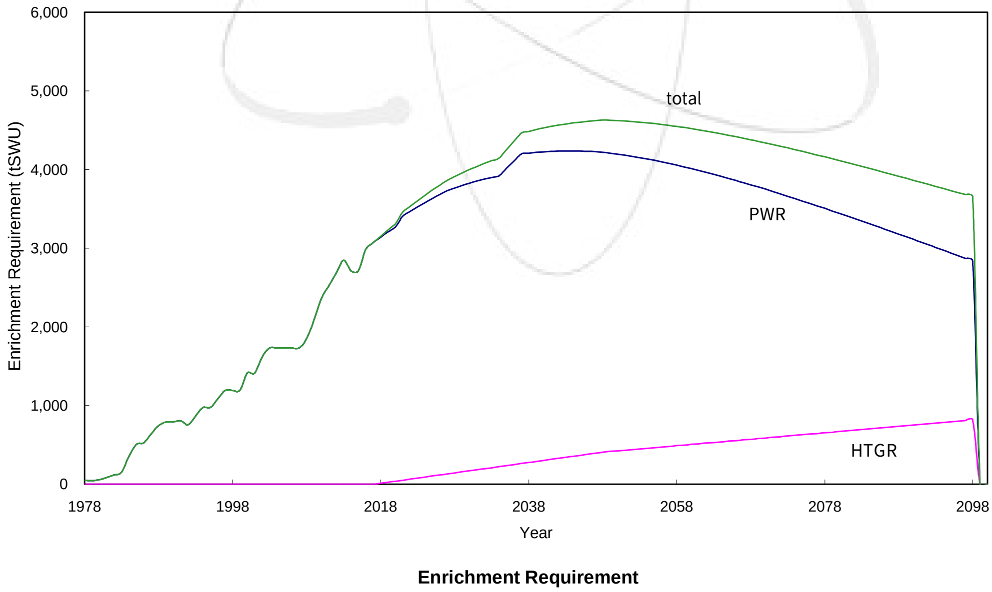

Fig. 103 Enrichment Requirement of Scenario Group 6

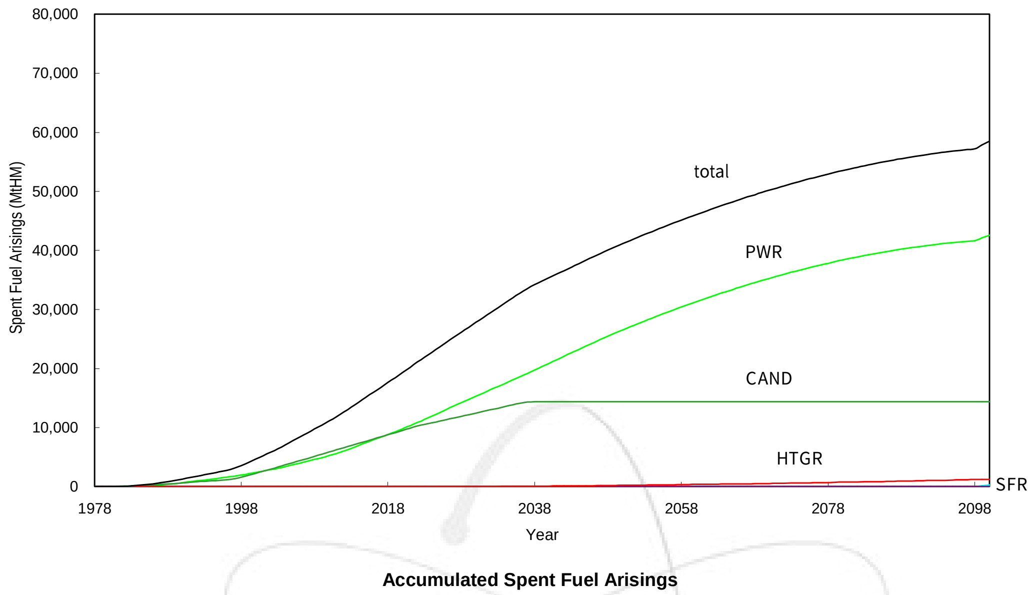

Fig. 104 Accumulated Spent Fuel Arisings of Scenario Group 6

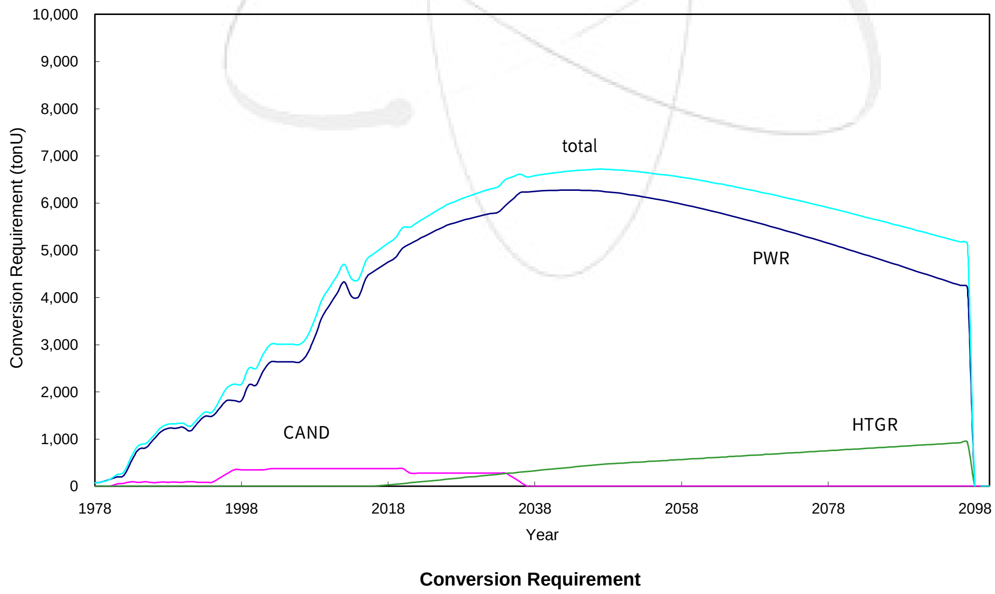

Fig. 105 Conversion Requirement of Scenario Group 6

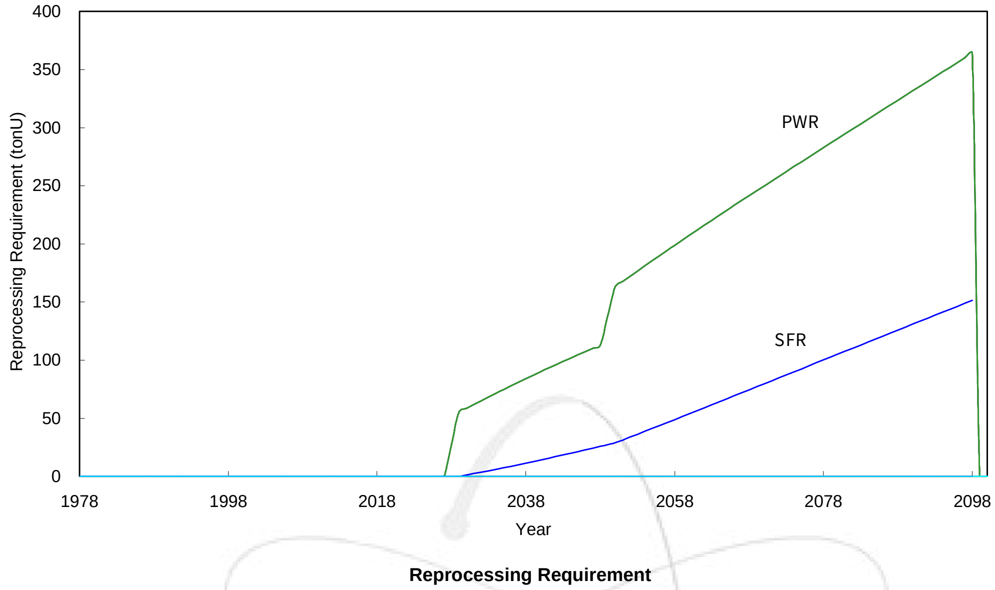

Fig. 106 Reprocessing Requirement of Scenario Group 6

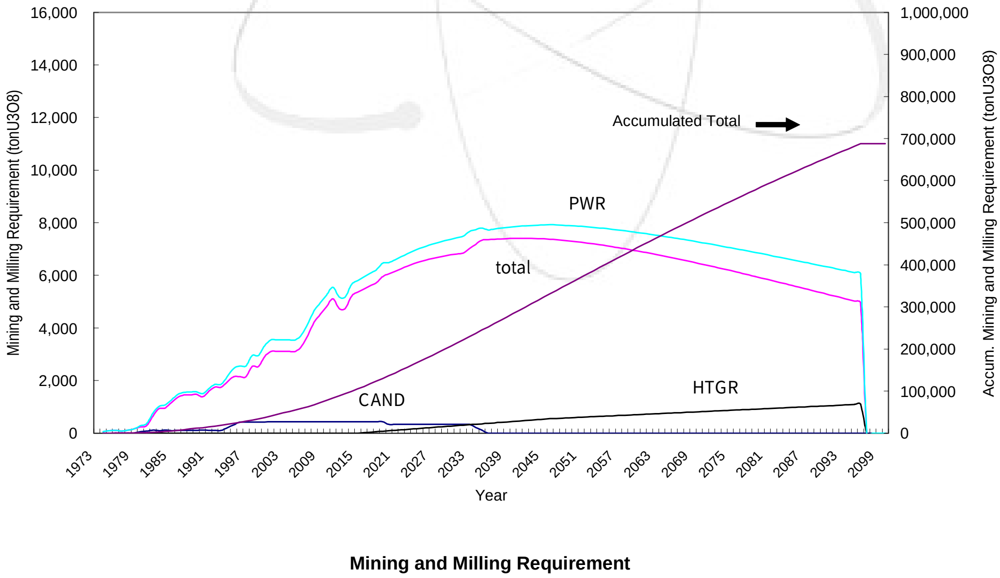

Fig. 107 Mining and Milling Requirement of Scenario Group 6

5.8 Scenario Group 7

5.8.1 Scenario Parameters (PWR + CANDU + ADS + SFR)

Scenario Group 7 is the recycle system including transmutation of actinide and long-lived fission product in which the PWR spent fuel is reprocessed and reused in SFRs and/or ADSs. The spent SFR fuel is reprocessed and continuously recycled in SFRs. The long-lived fission products, Tc and I, from the PWR and the SFR spent fuel are fabricated as a target fuel and transmuted in the ADS.

The Scenario parameters for the SFR are the same as the Scenario Group 5 and 6. It is assumed that the ADS is introduced from 2040 and its grid ratio is expanded up to 10% by 2070 and 20 % by 2100.

5.8.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the Scenario Group 2.

- Reactor parameters for ADS are the same as those of the Scenario Group 5 and 6.

- Load factor of ADS : 80%

- Thermal efficiency of ADS : 40%

- Pool capacity : The storage pool of ADS can employ spent fuels generated for 10 years.

- ADS Reactor Information

Fuel weight(kgHM/core)	3,831	
Target fuel weight(kg/core)	Tc99	I129
	663	199
Target fuel ratio	17.31%	5.19%
Annual burnup(a/o) of target fuel	5.92%	4.86%
Total Fuel Burnup(a/o)	20.00%	
TRU contribution ratio of thermal power for ADS	98%	

5.8.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Fuel parameters for SFR are the same as those of the Scenario Group 5 and 6.
- Fresh ADS fuel content : The average content is calculated considering the ratio of the number of each fuel assembly.

ADS Fresh Fuel (200GWd/MtHM)					
Pu	U235	MA	Tc99	I129	U
74.69%	0.13%	13.60%	17.31%	5.19%	11.71%

- Spent SFR fuel contents: The average content is calculated considering the ratio of the number of each fuel assembly.

Pu	U235	MA	Tc99	I129	U	Consumption rate
72.00%	0.16%	14.34%	0.100%	0.100%	13.63%	20.0%

- Loss factors

	PWR	CANDU	SFR(B)	SFR(D)	ADS	HTGR
Conversion	0.5%	0.5%				0.5%
Enrichment	1.0%					1.0%
Fabrication	0.5%	0.5%	0.5%	1.0%	0.4%	0.5%
Reprocessing	1.0%		1.0%	1.0%	1.0%	

5.8.4 Outputs

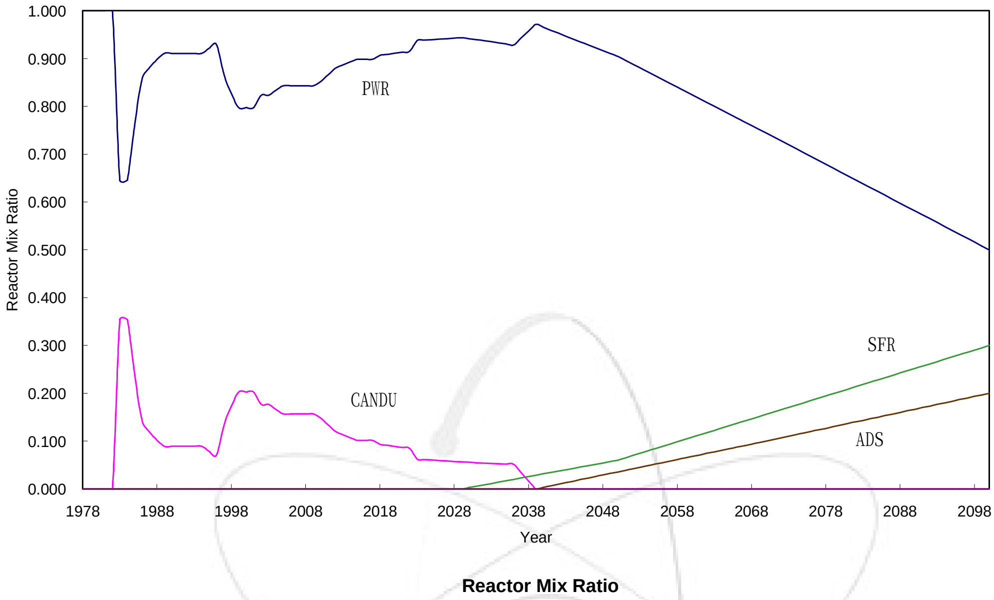

Fig. 108 Reactor Mix Ratio of Scenario Group 7

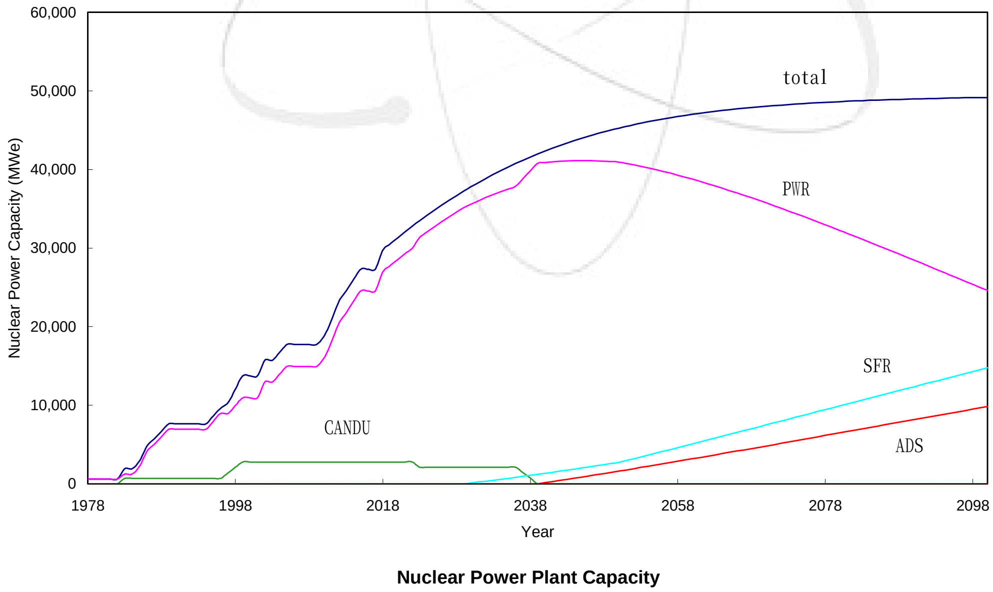

Fig. 109 Nuclear Power Plant Capacity of Scenario Group 7

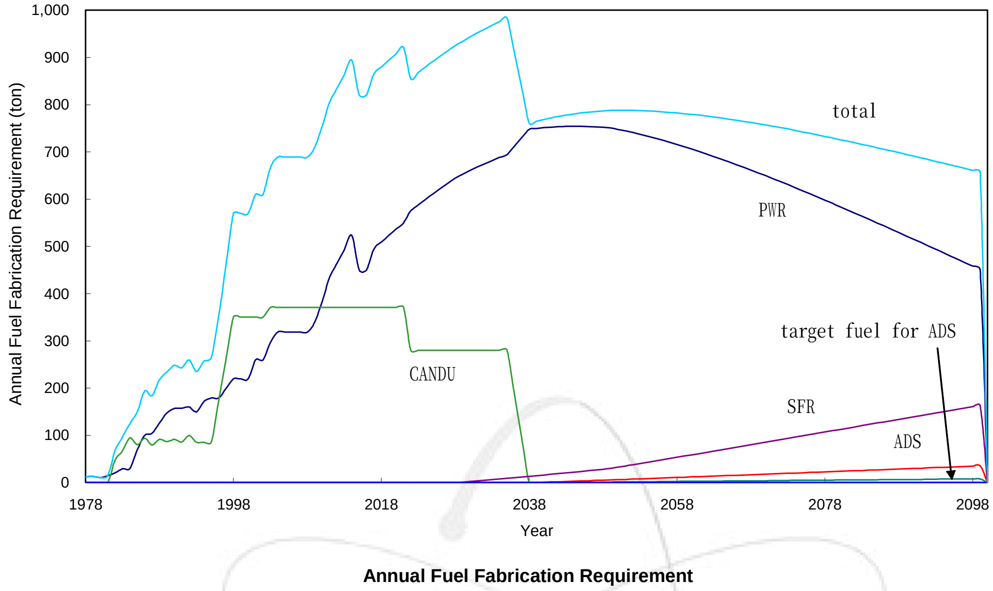

Fig. 110 Annual Fuel Fabrication Requirement of Scenario Group 7

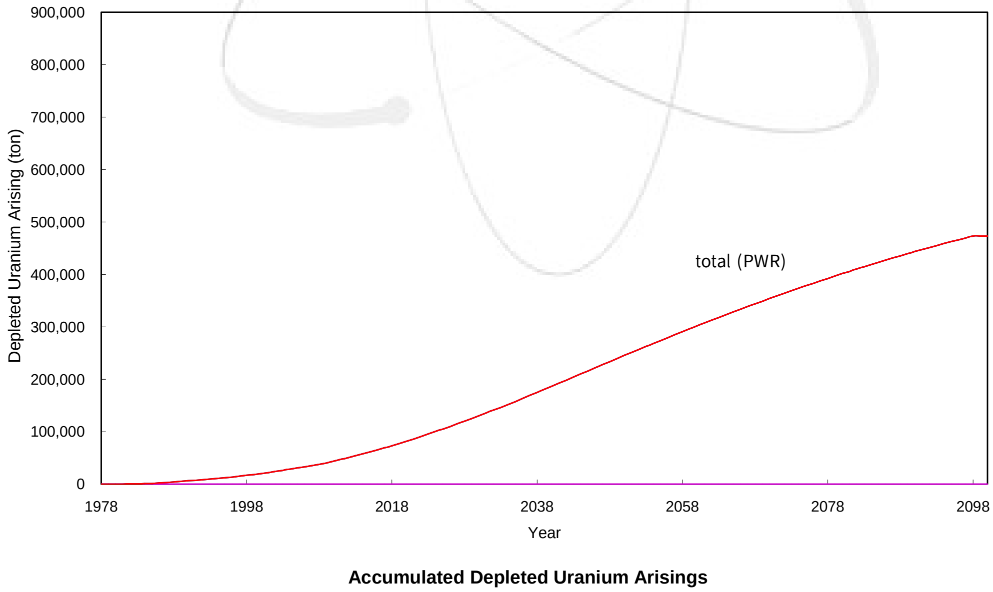

Fig. 111 Accumulated Depleted Uranium Arisings of Scenario Group 7

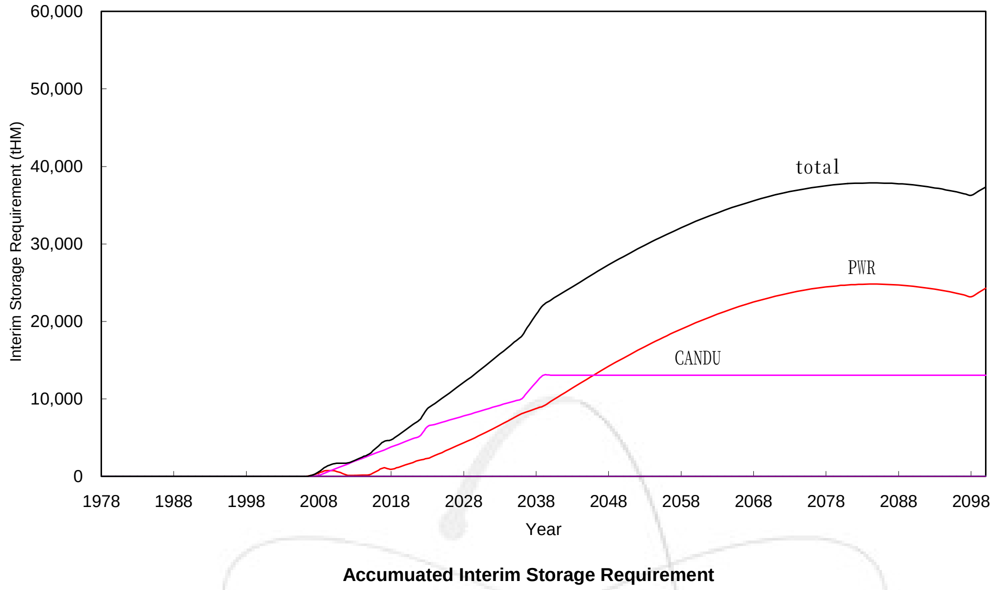

Fig. 112 Accumulated Interim Storage Requirement of Scenario Group 7

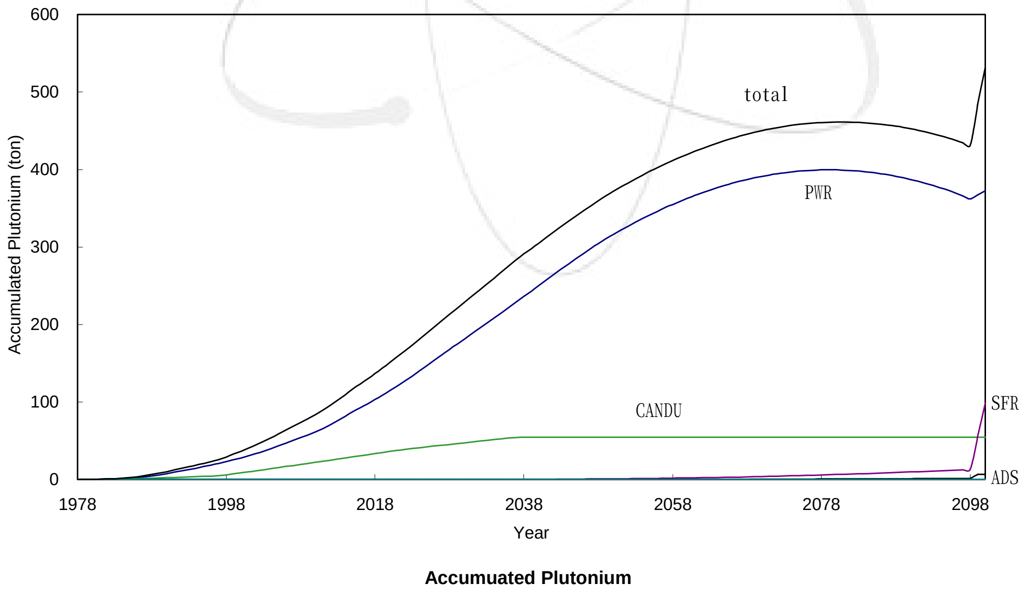

Fig. 113 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 7

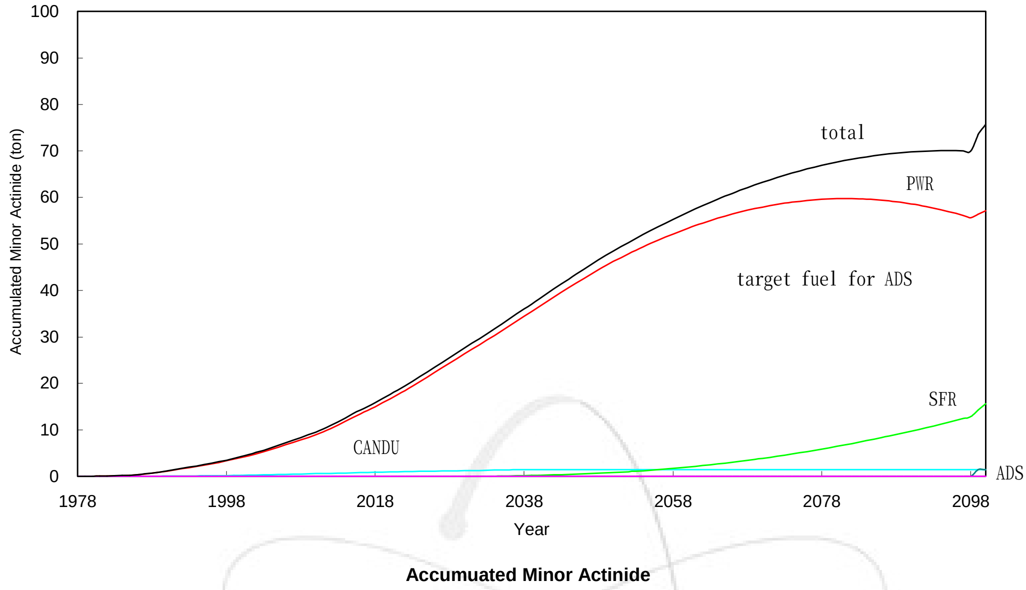

Fig. 114 Accumulated MA Embedded in Spent Fuels of Scenario Group 7

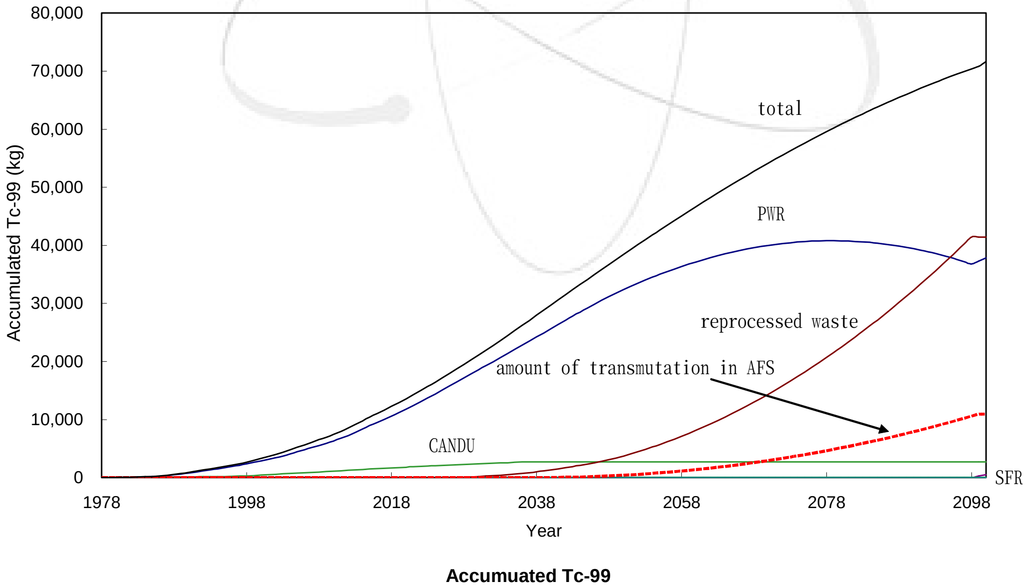

Fig. 115 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 7

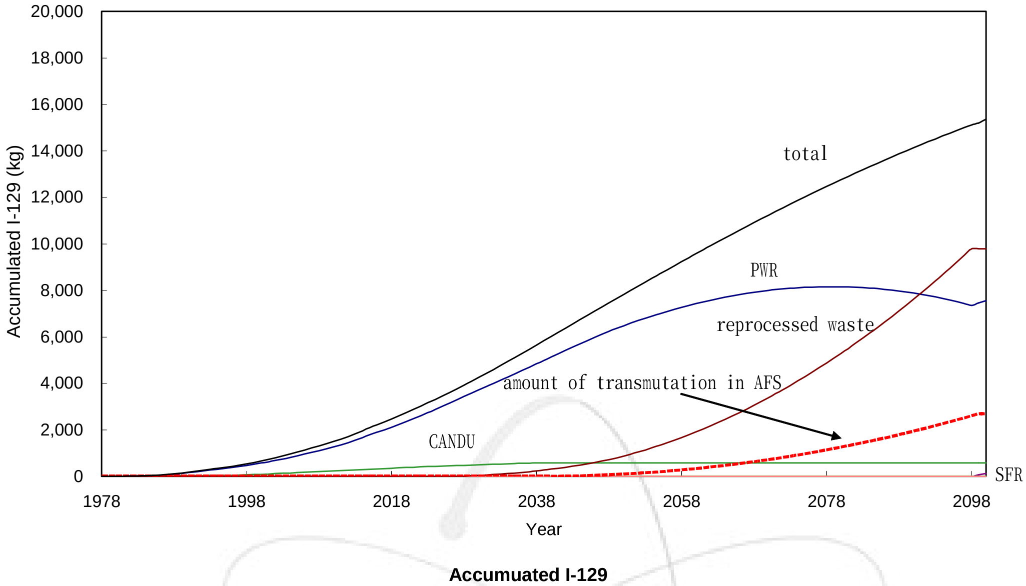

Fig. 116 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 7

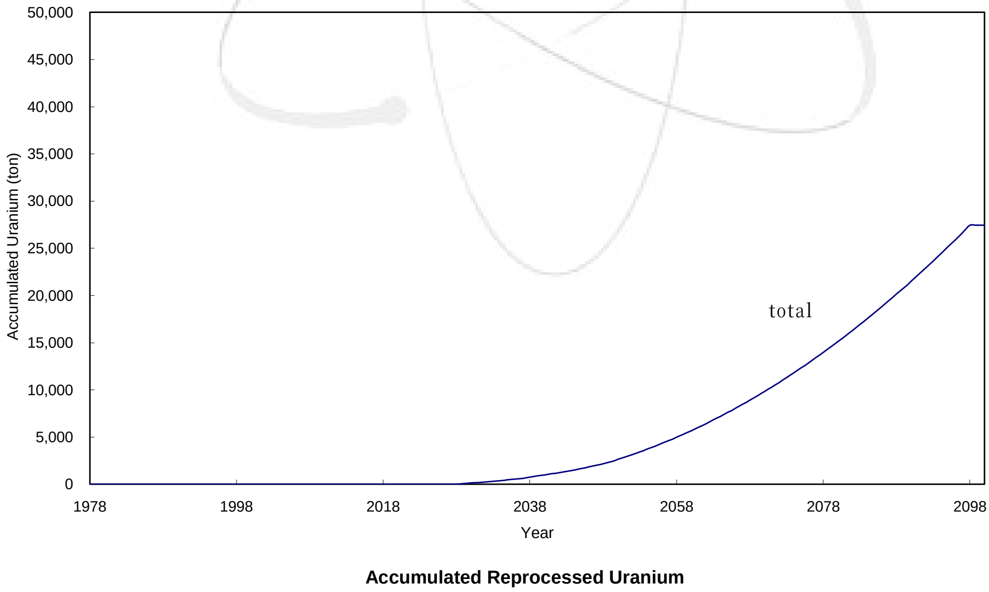

Fig. 117 Accumulated Reprocessed Uranium of Scenario Group 7

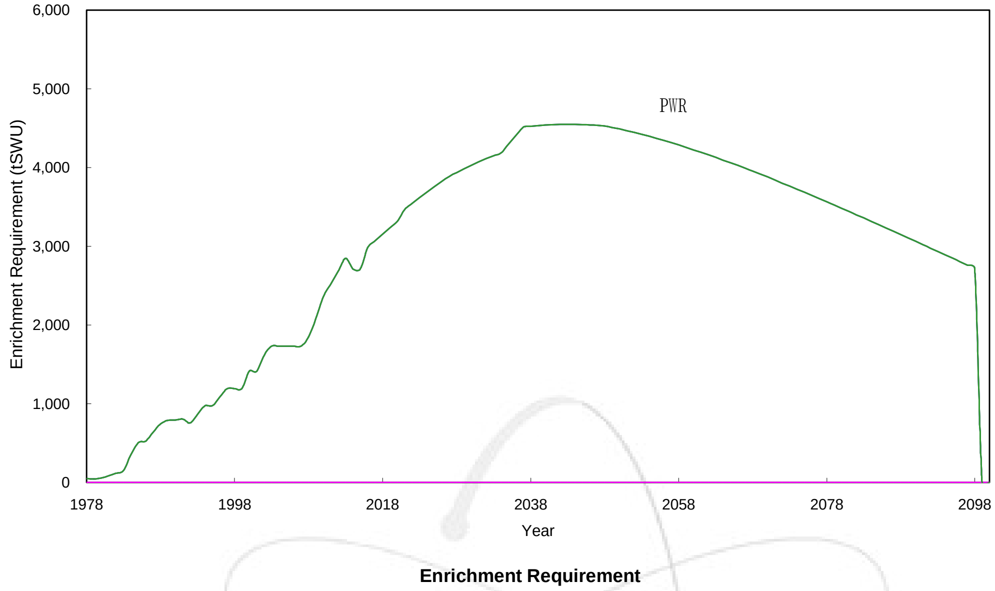

Fig. 118 Enrichment Requirement of Scenario Group 7

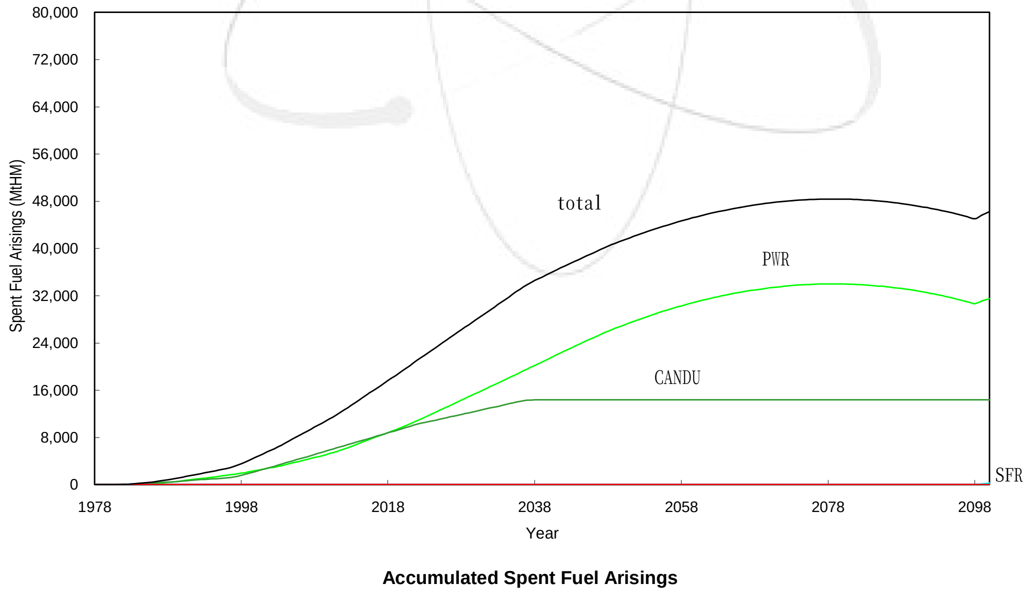

Fig. 119 Accumulated Spent Fuel Arisings of Scenario Group 7

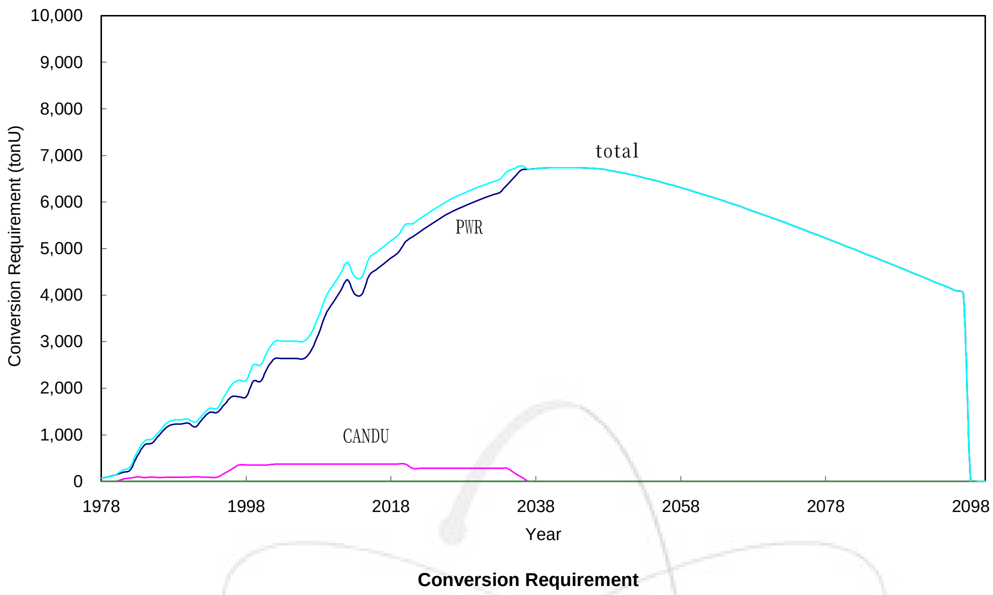

Fig. 120 Conversion Requirement of Scenario Group 7

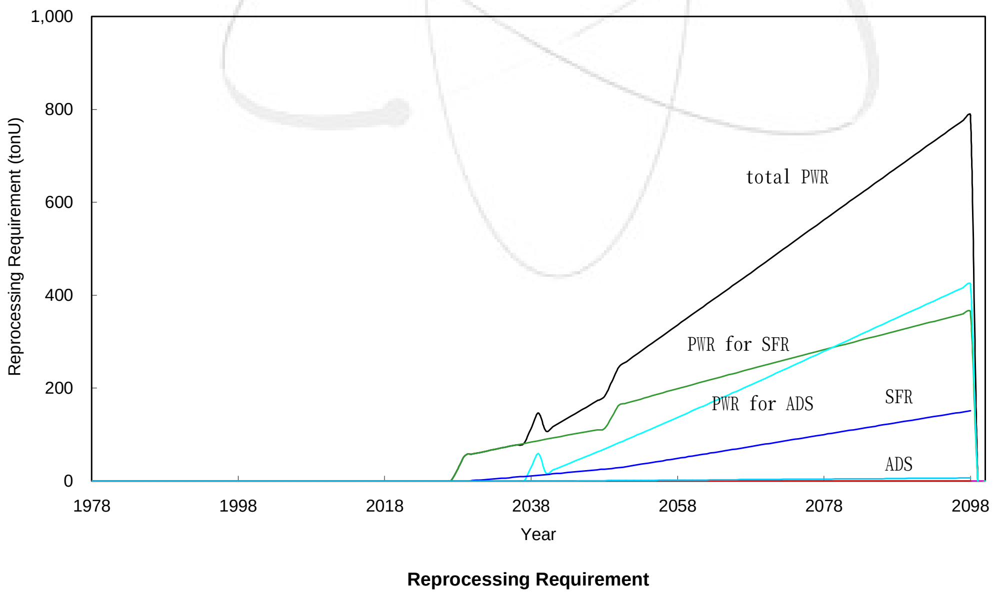

Fig. 121 Reprocessing Requirement of Scenario Group 7

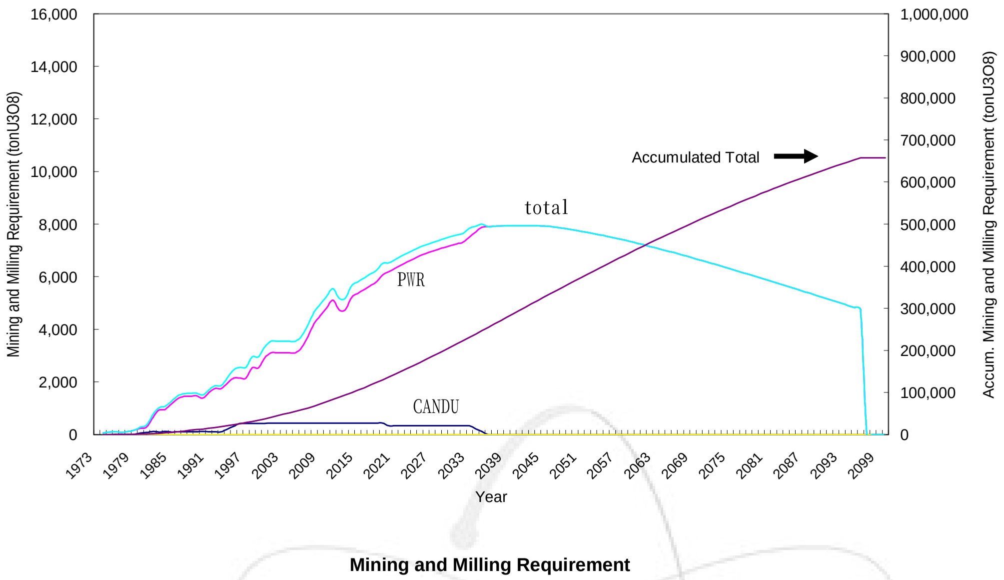

Fig. 122 Mining and Milling Requirement of Scenario Group 7

5.9 Scenario Group 8

5.9.1 Scenario Parameters (PWR + CANDU + ADS)

Scenario Group 8 is the same as the Scenario Group 7 except that there is no SFR in this system. PWR spent fuel is reprocessed and reused in ADSs. Spent fuel from the ADS is continuously recycled in ADSs. The Tc and I generated from the PWR are fabricated as a target fuel and transmuted in ADS

It is assumed that the ADS is introduced from 2040 and its grid ratio is expanded up to 10% by 2070 and 20 % by 2100.

5.9.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are those of the same as the Scenario Group 2.

- Reactor parameters for ADS are the same as those of the Scenario Group 7.

5.9.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are those of the same as the Scenario Group 2.

- Fuel parameters for ADS are the same as those of the Scenario Group 7.

5.9.4 Outputs

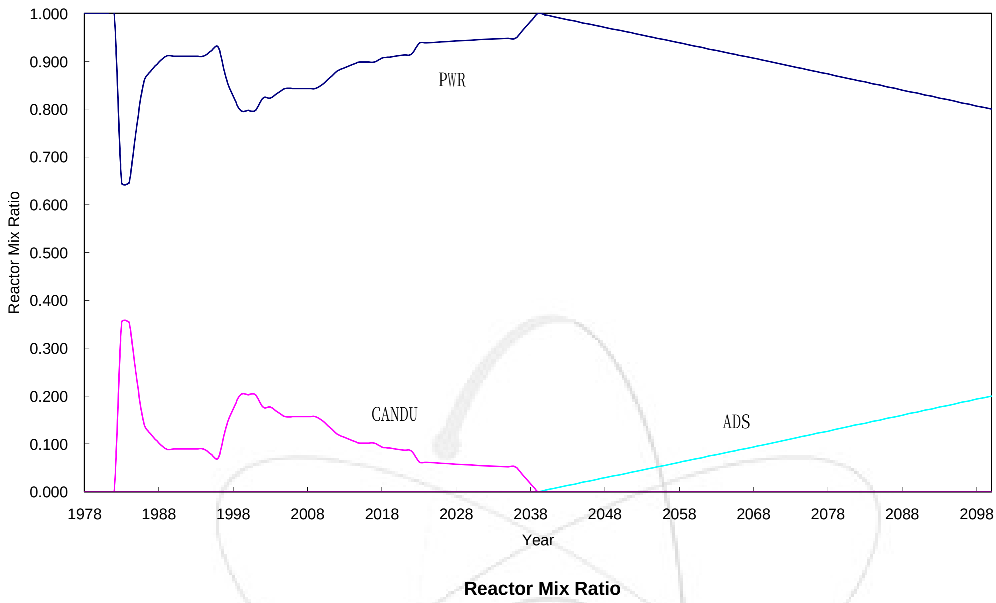

Fig. 123 Reactor Mix Ratio of Scenario Group 8

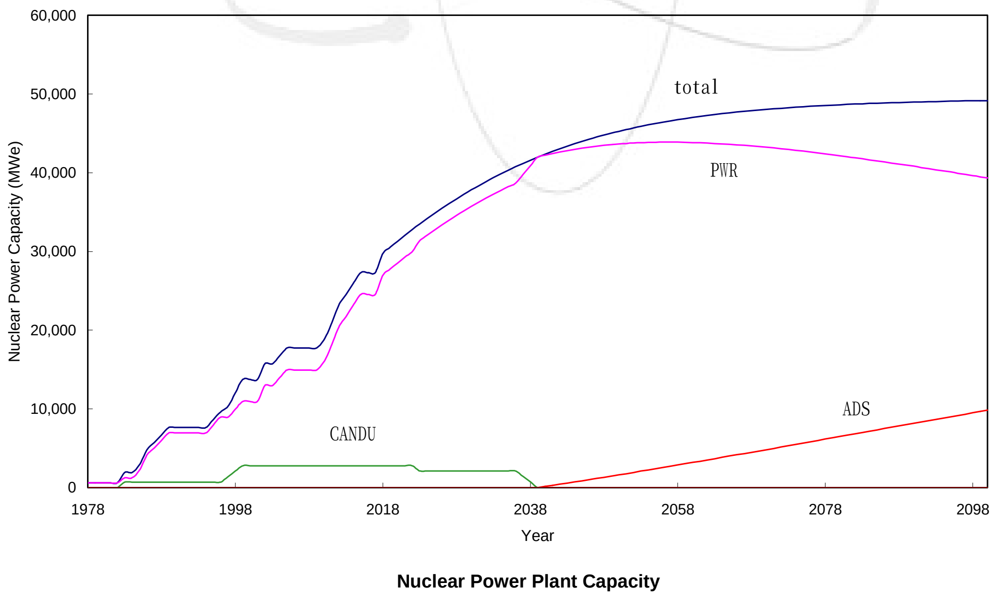

Fig. 124 Nuclear Power Plant Capacity of Scenario Group 8

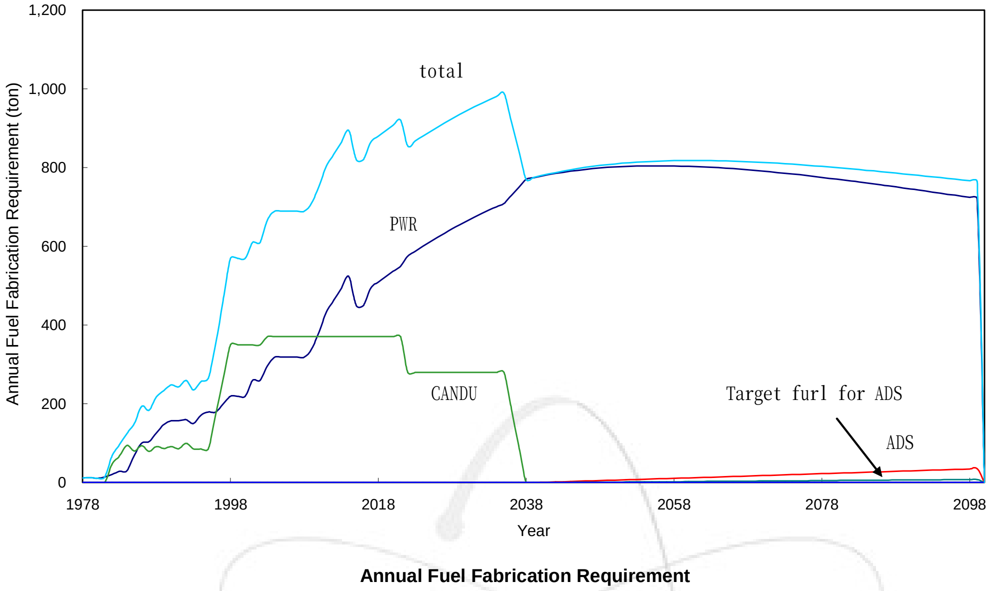

Fig. 125 Annual Fuel Fabrication Requirement of Scenario Group 8

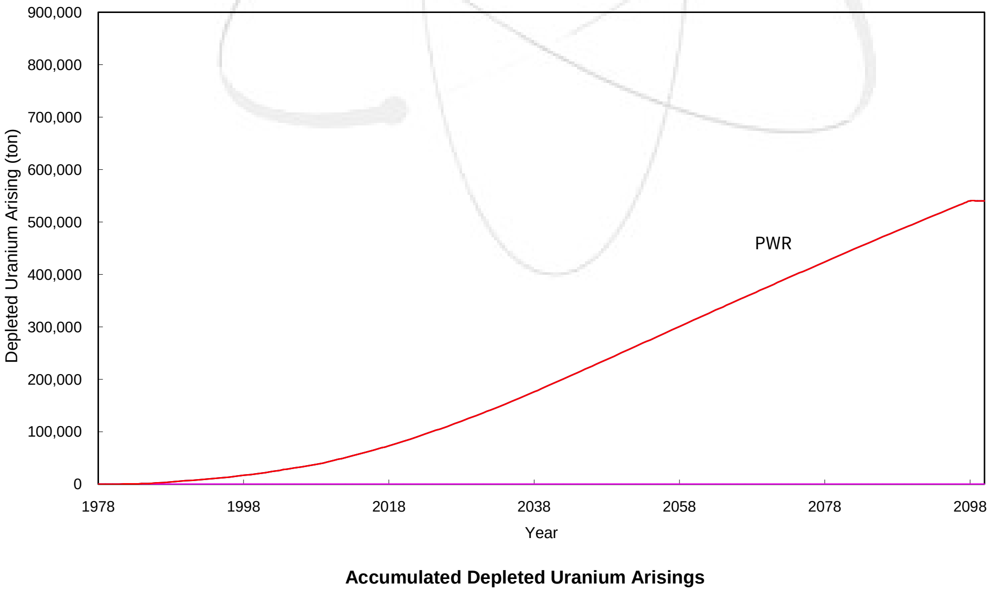

Fig. 126 Accumulated Depleted Uranium Arisings of Scenario Group 8

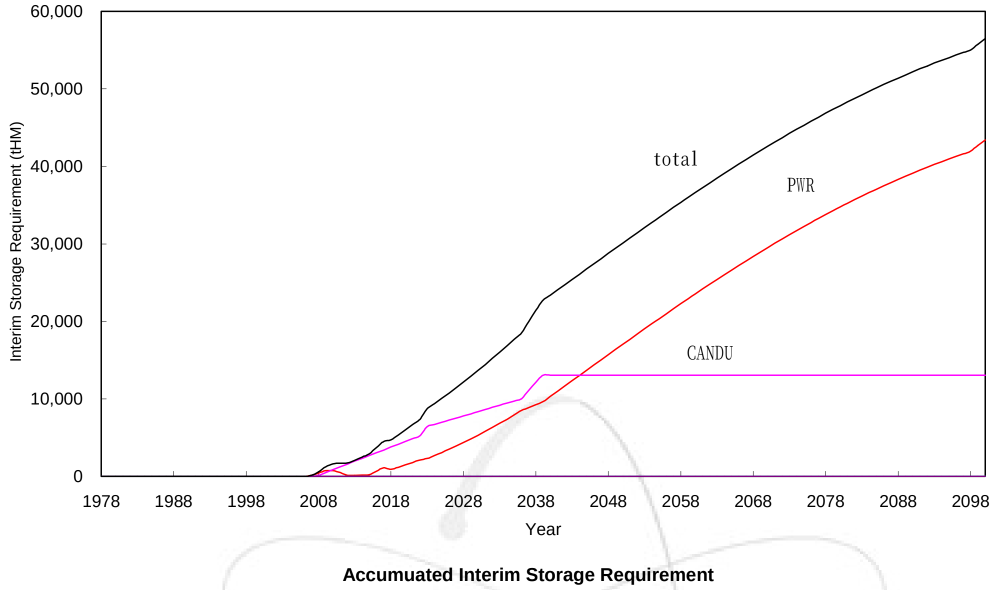

Fig. 127 Accumulated Interim Storage Requirement of Scenario Group 8

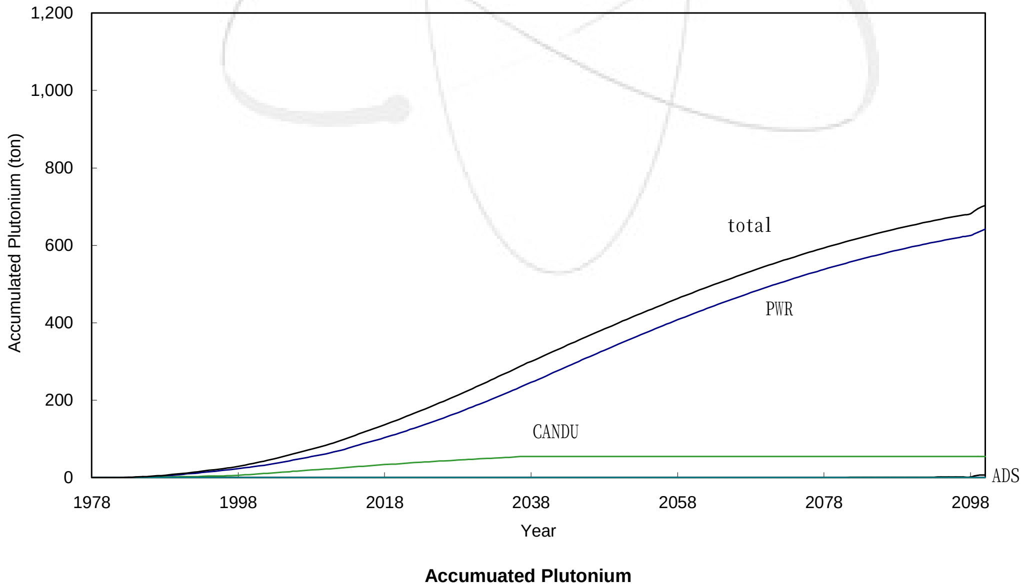

Fig. 128 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 8

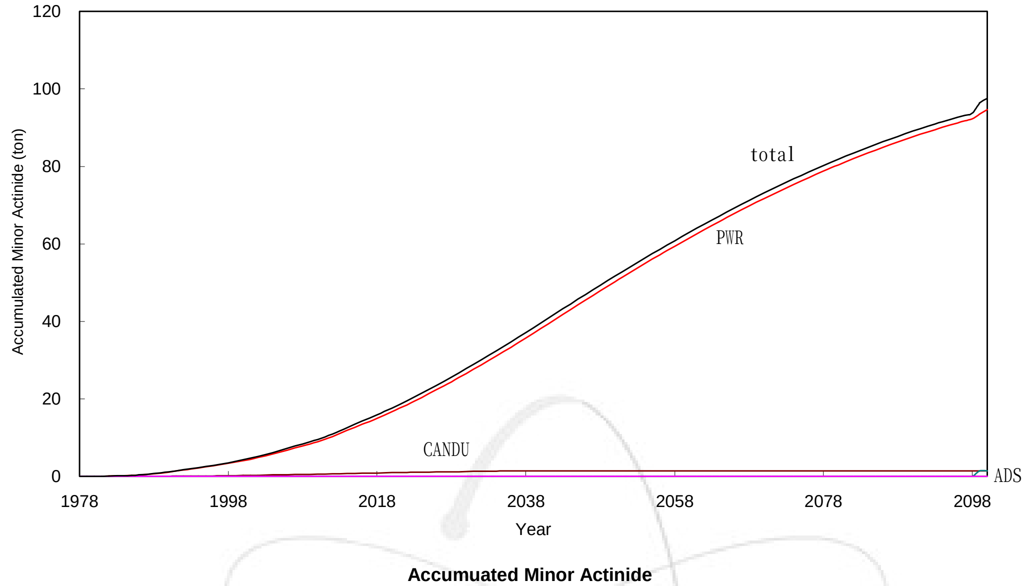

Fig. 129 Accumulated MA Embedded in Spent Fuels of Scenario Group 8

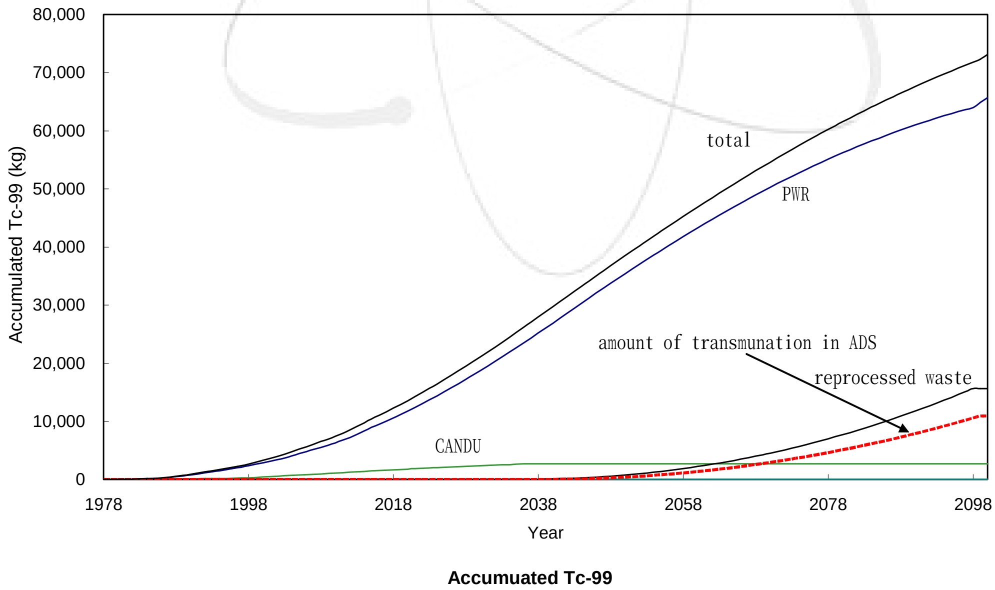

Fig. 130 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 8

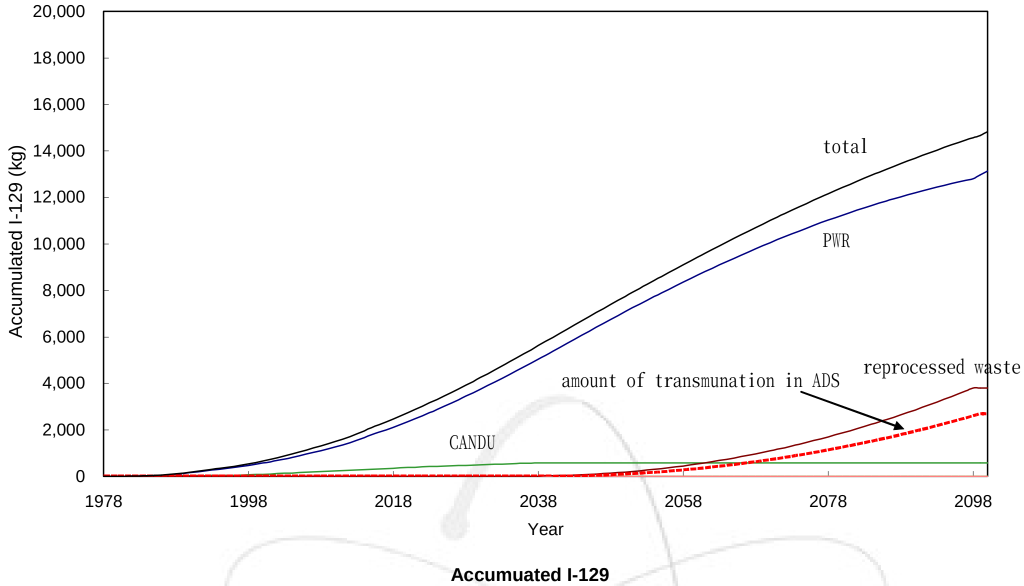

Fig. 131 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 8

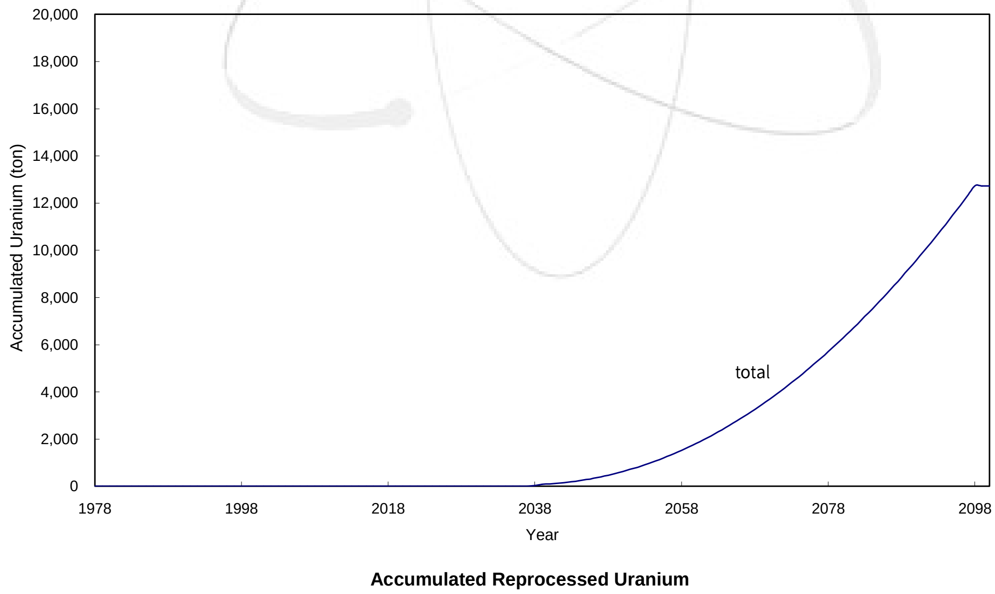

Fig. 132 Accumulated Reprocessed Uranium of Scenario Group 8

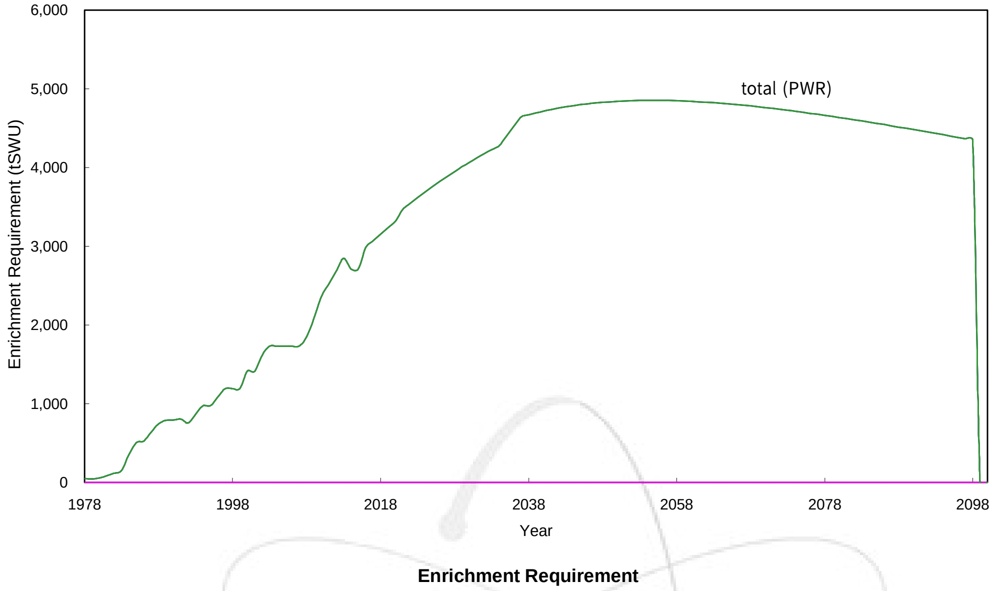

Fig. 133 Enrichment Requirement of Scenario Group 8

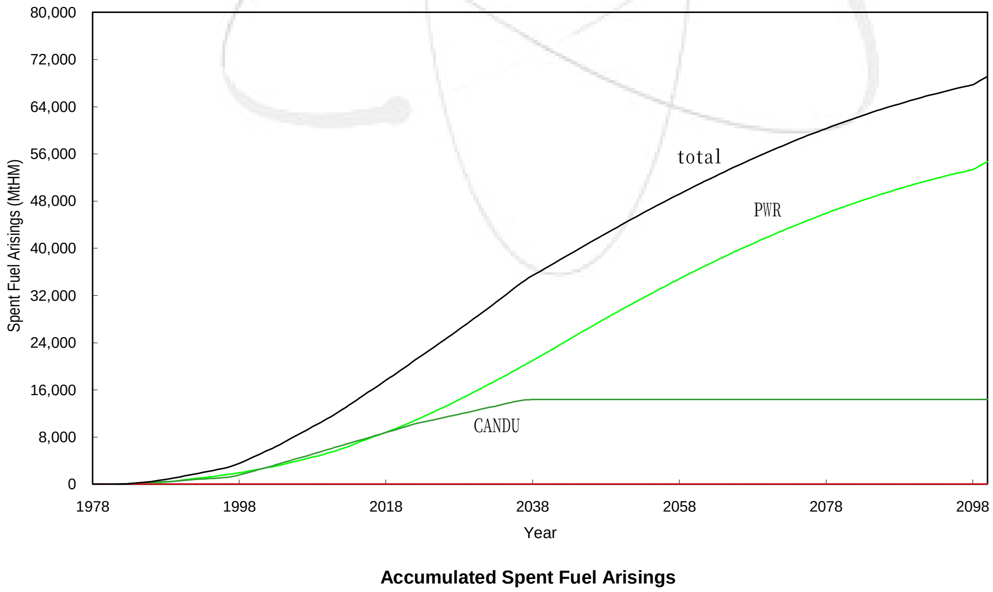

Fig. 134 Accumulated Spent Fuel Arisings of Scenario Group 8

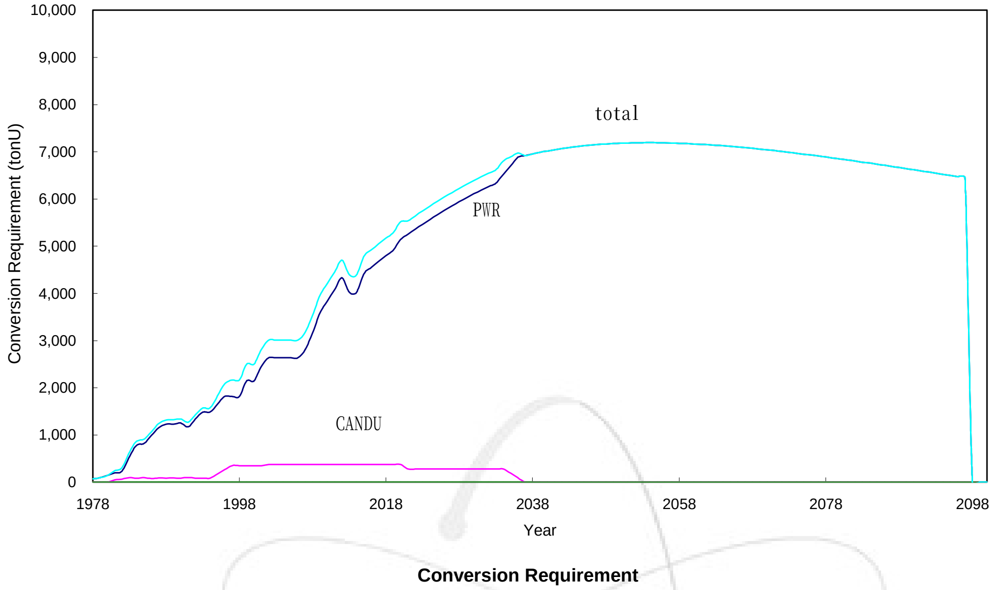

Fig. 135 Conversion Requirement of Scenario Group 8

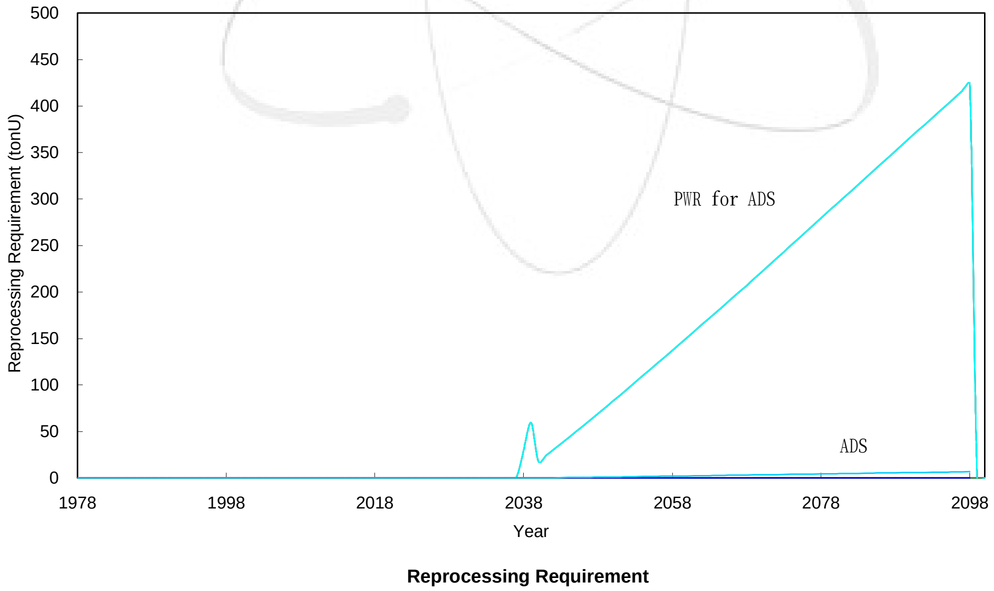

Fig. 136 Reprocessing Requirement of Scenario Group 8

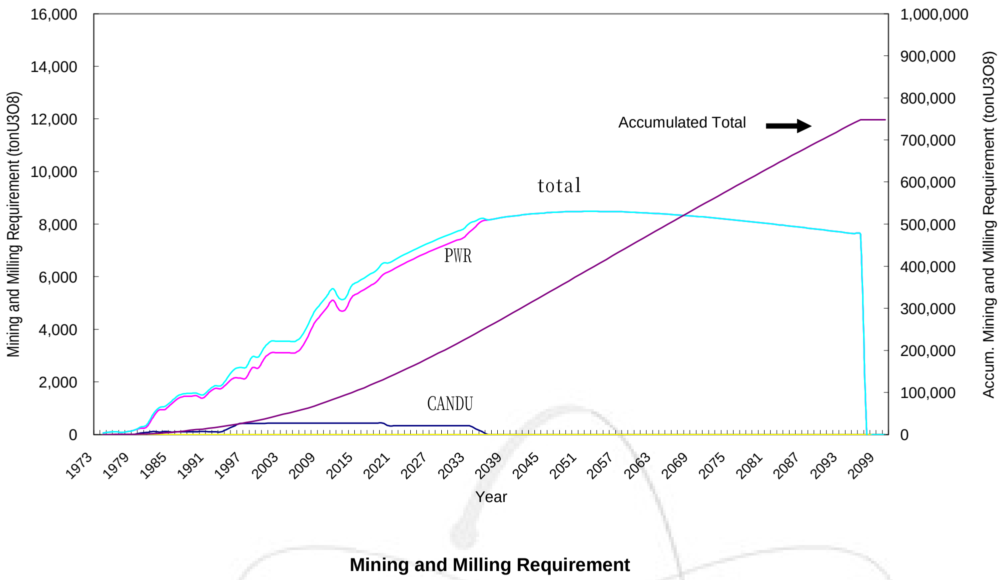

Fig. 137 Mining and Milling Requirement of Scenario Group 8

5.10 Scenario Group 9

5.10.1 Scenario Parameters (PWR + CANDU + HTGR + ADS)

Scenario Group 9 is the same as the Scenario Group 8 except that the HTGR is added. PWR spent fuel is reprocessed and reused in ADSs. The spent fuel from the ADS is continuously recycled in ADSs. Tc and I from the PWR are fabricated as a target fuel and transmuted in ADSs. The Scenario parameters for the HTGR are the same as the Scenario Group 3.

5.10.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the Scenario Group 2.

- Reactor parameters for HTGR are the same as those of the Scenario Group 3.

- Reactor parameters for ADS are the same as those of the Scenario Group 7 and 8.

5.10.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as the Scenario Group 2.

- Fuel parameters for HTGR are the same as the Scenario Group 3.

- Fuel parameters for ADS are the same as the Scenario Group 7 and 8.

5.10.4 Outputs

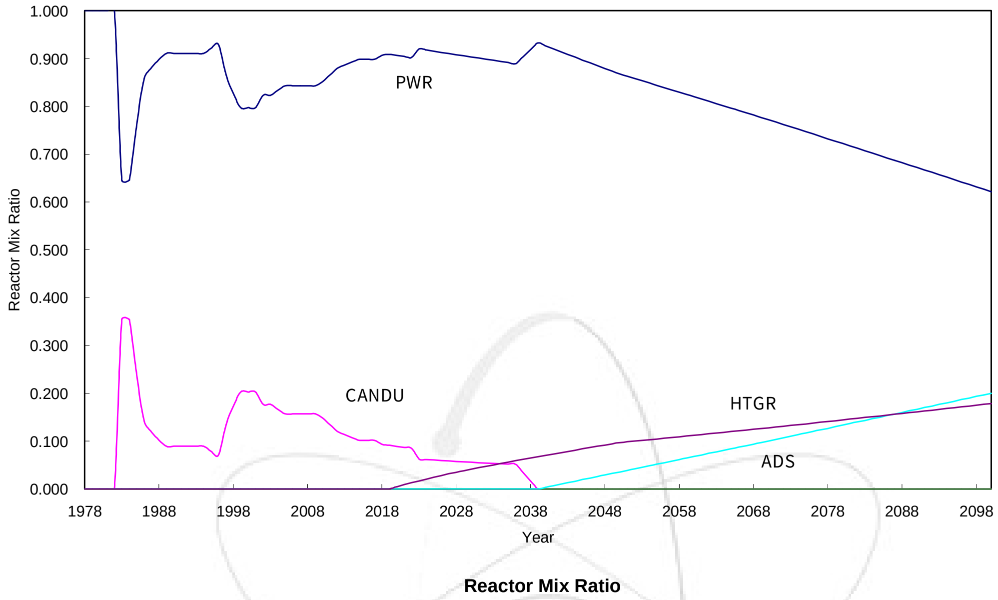

Fig. 138 Reactor Mix Ratio of Scenario Group 9

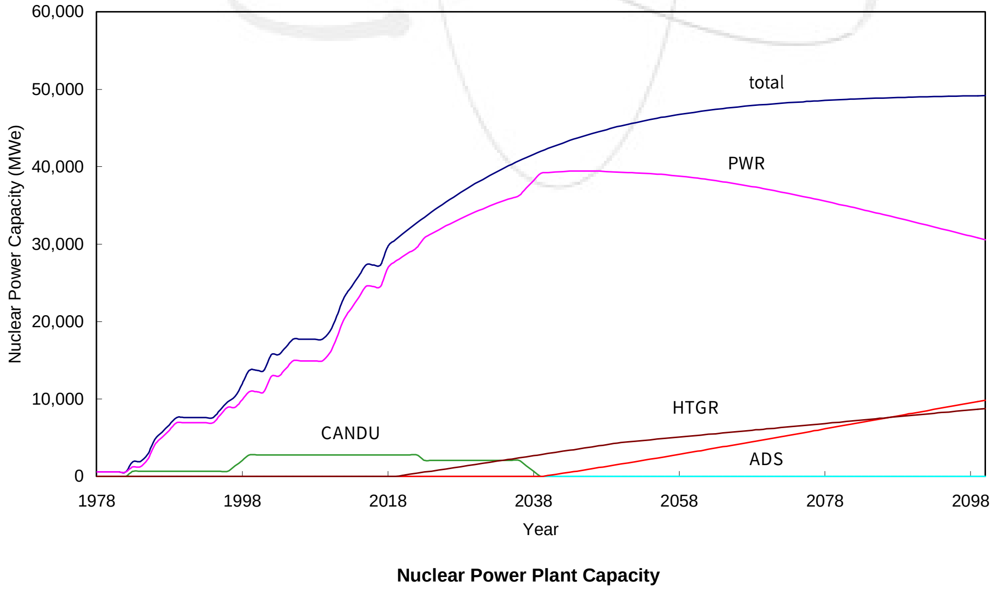

Fig. 139 Nuclear Power Plant Capacity of Scenario Group 9

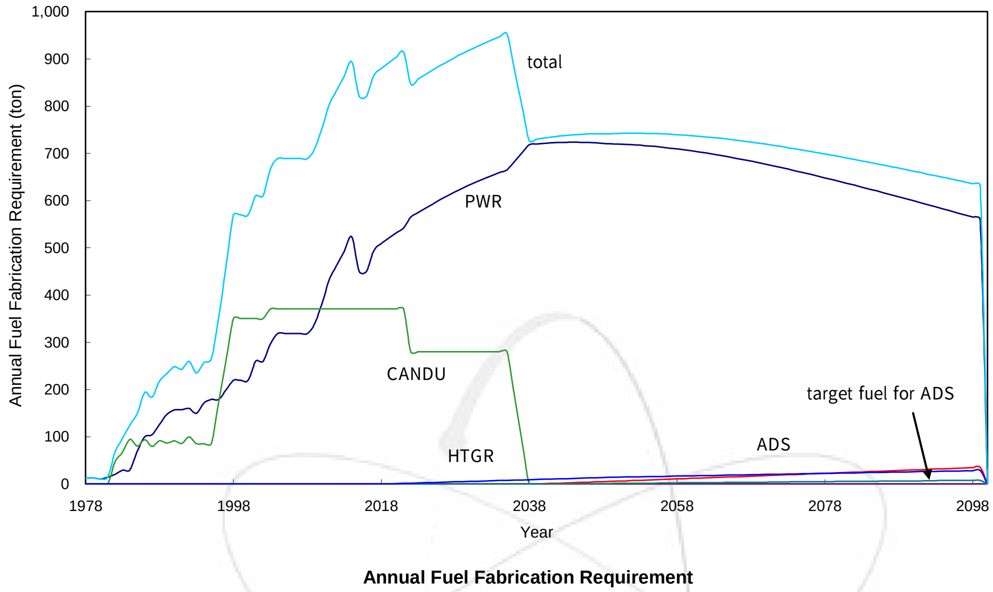

Fig. 140 Annual Fuel Fabrication Requirement of Scenario Group 9

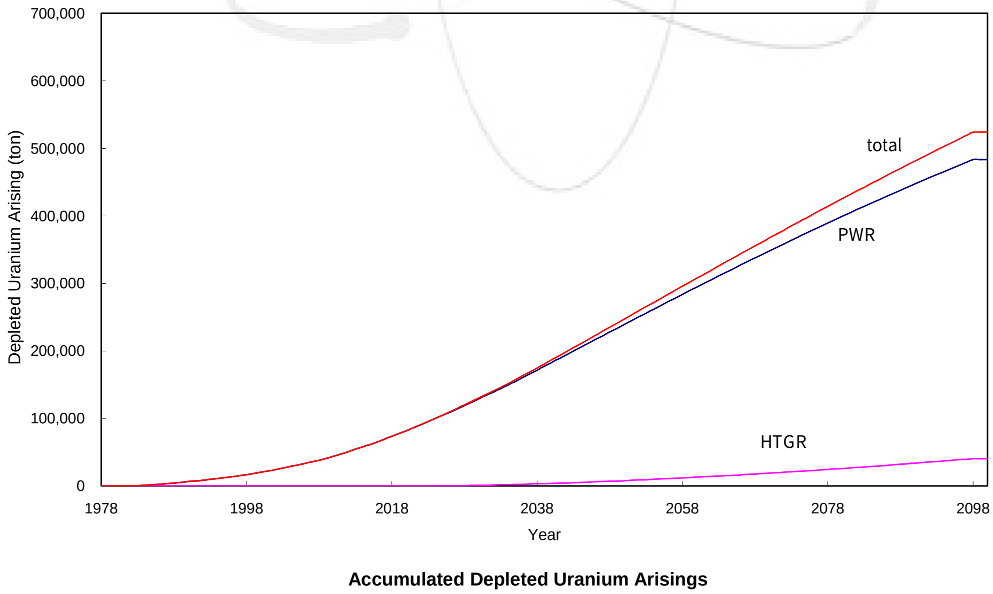

Fig. 141 Accumulated Depleted Uranium Arisings of Scenario Group 9

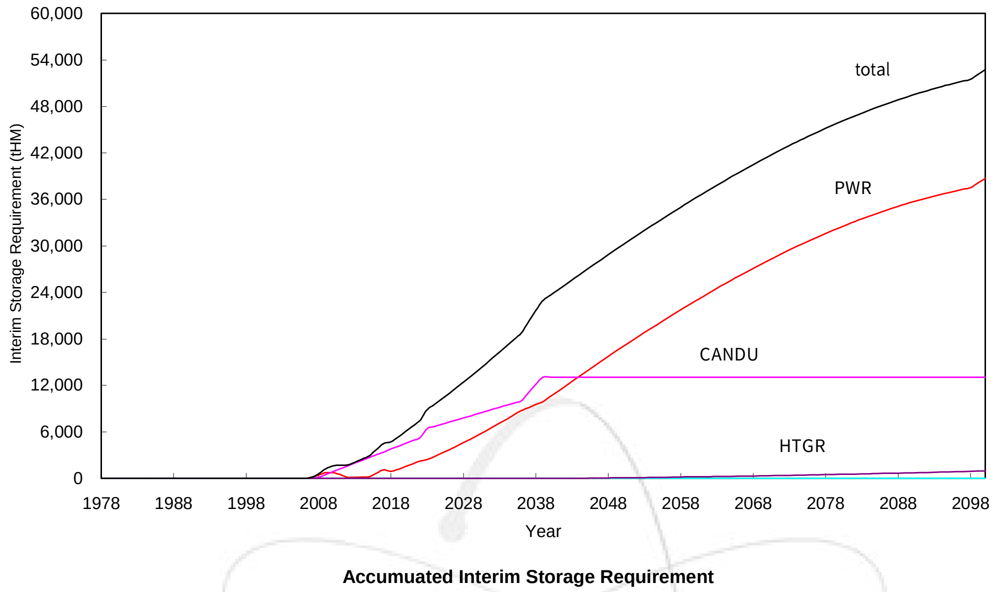

Fig. 142 Accumulated Interim Storage Requirement of Scenario Group 9

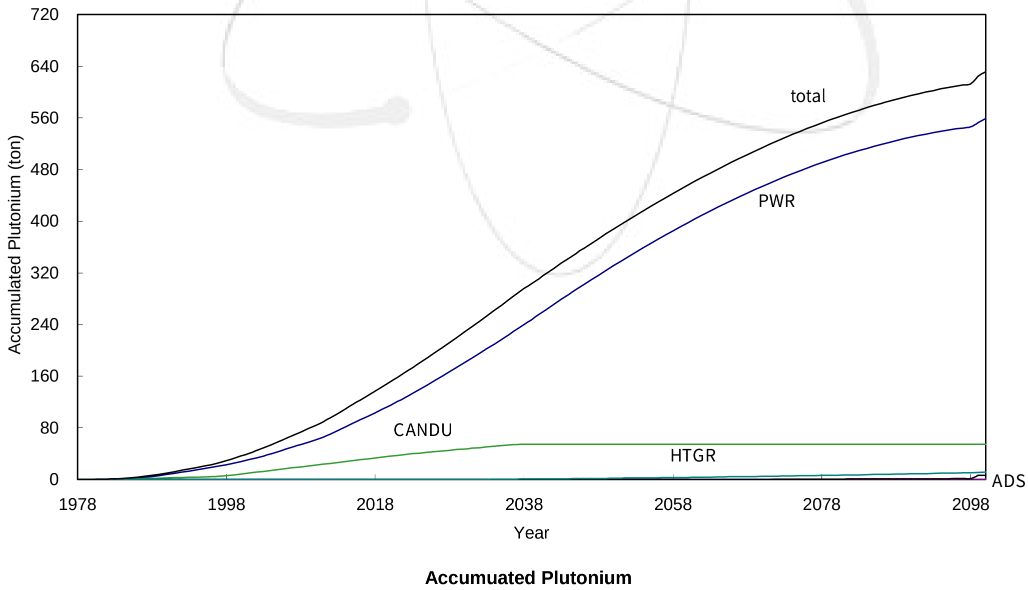

Fig. 143 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 9

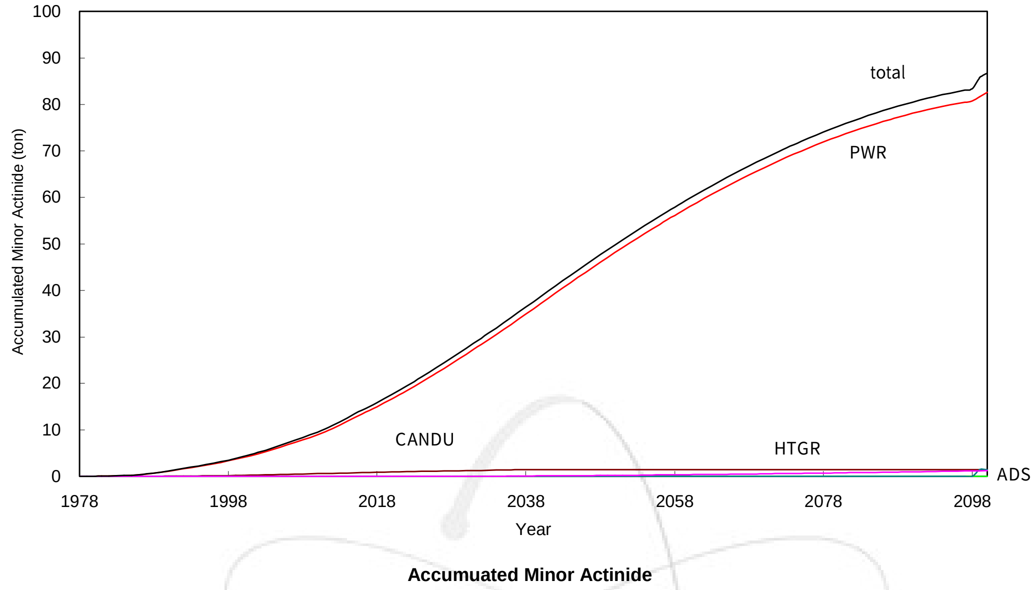

Fig. 144 Accumulated MA Embedded in Spent Fuels of Scenario Group 9

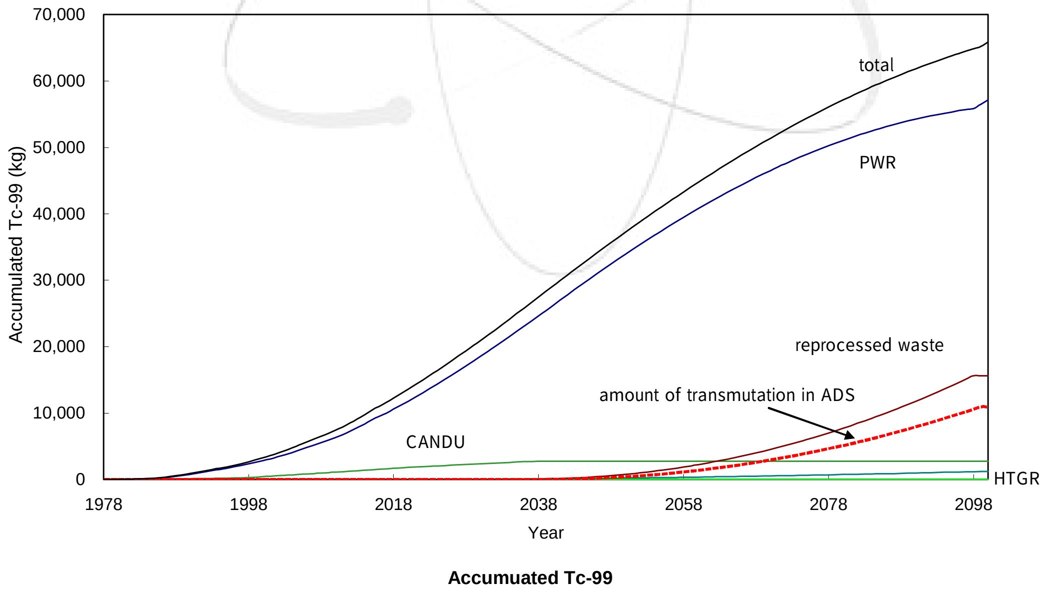

Fig. 145 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 9

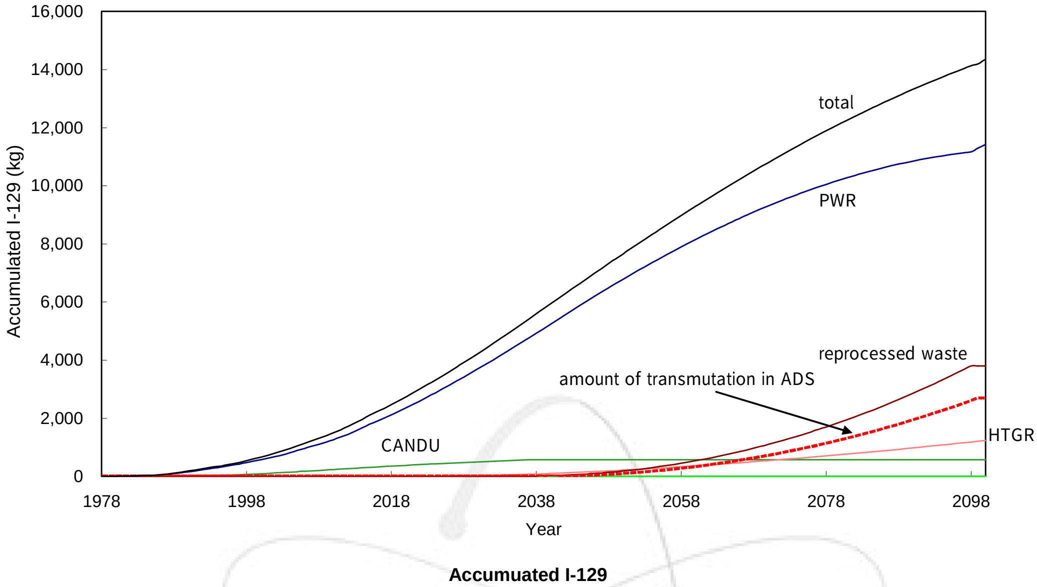

Fig. 146 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 9

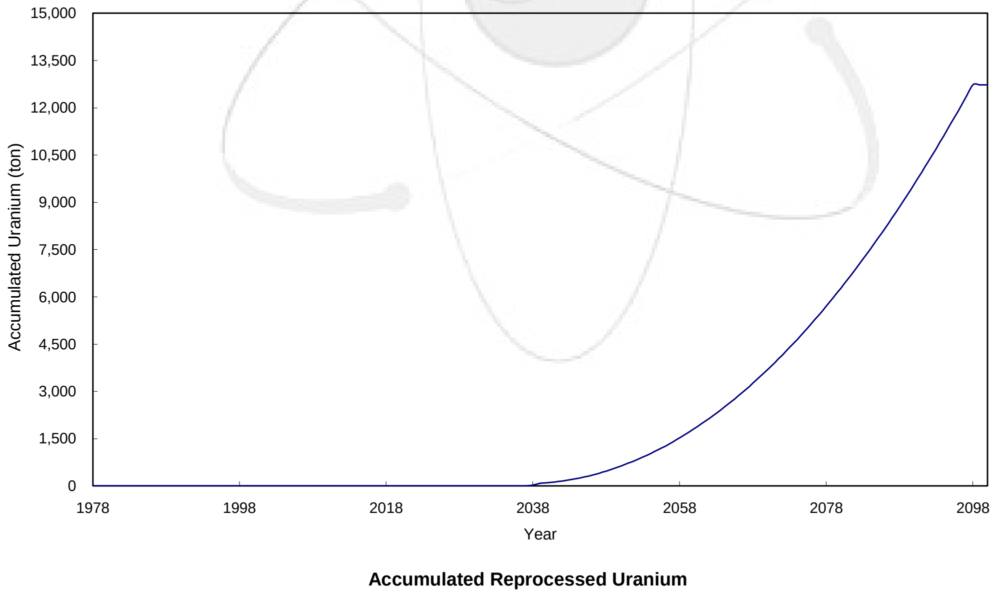

Fig. 147 Accumulated Reprocessed Uranium of Scenario Group 9

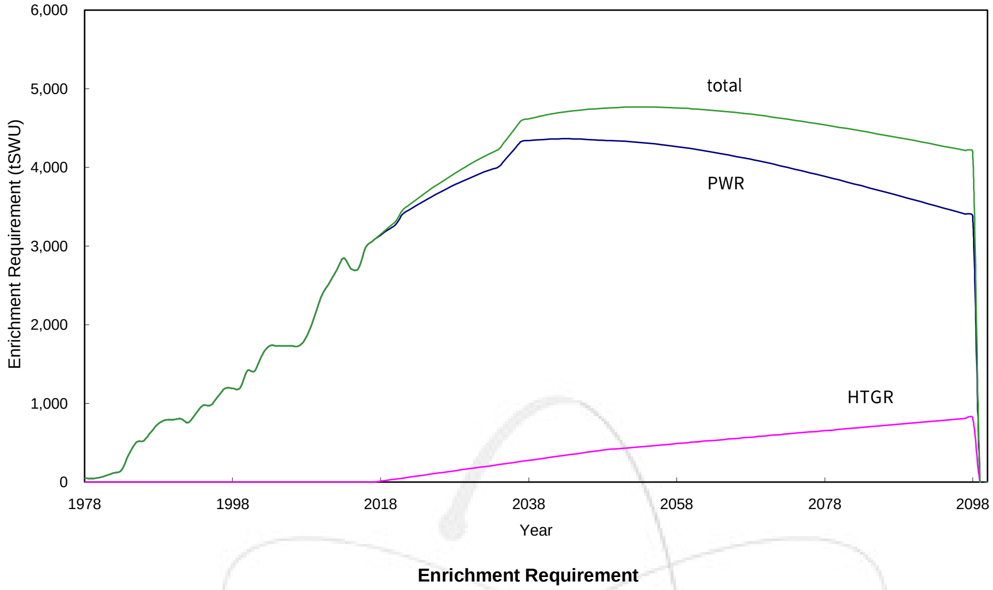

Fig. 148 Enrichment Requirement of Scenario Group 9

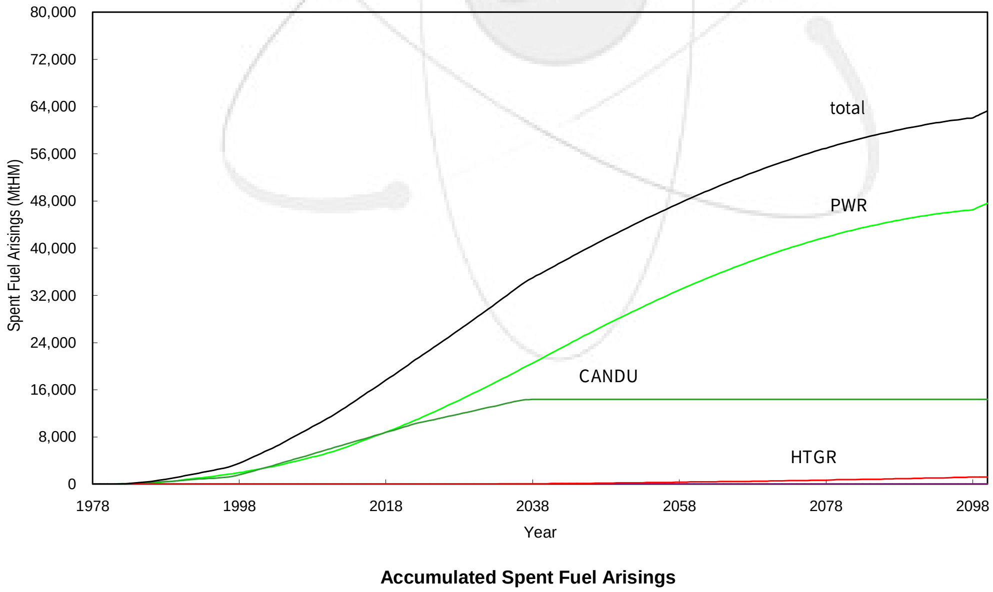

Fig. 149 Accumulated Spent Fuel Arisings of Scenario Group 9

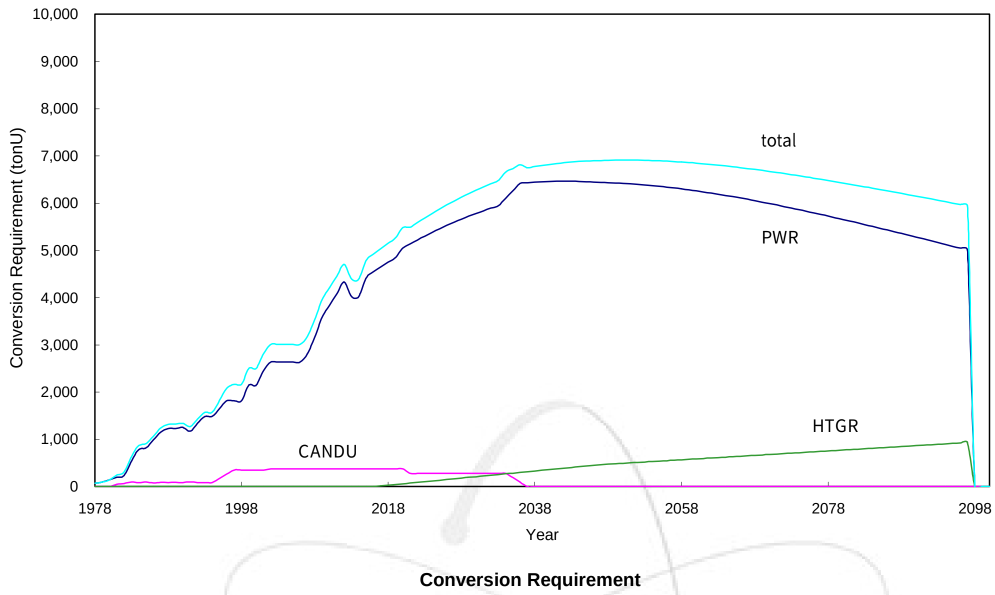

Fig. 150 Conversion Requirement of Scenario Group 9

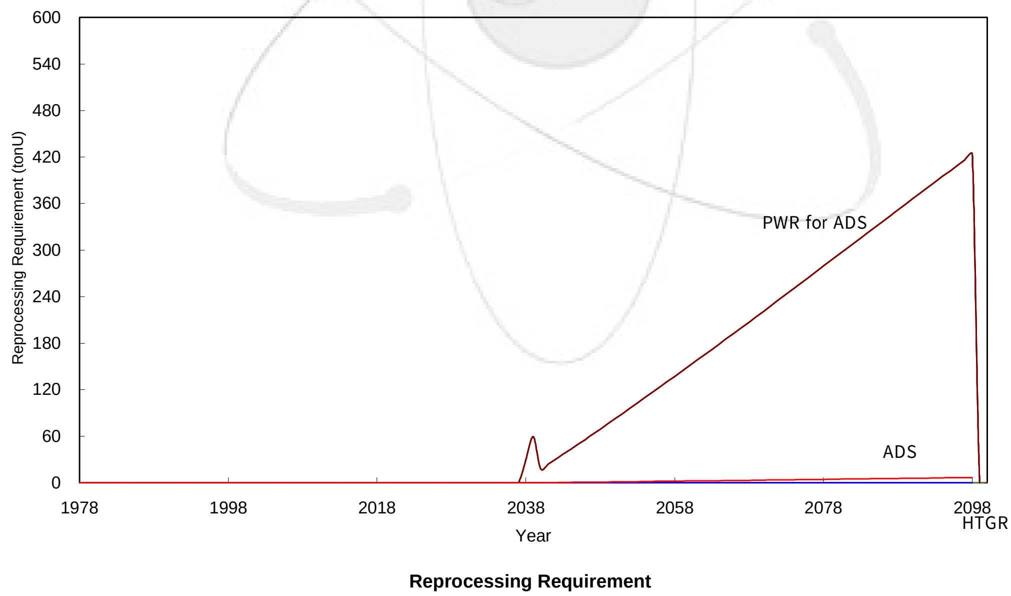

Fig. 151 Reprocessing Requirement of Scenario Group 9

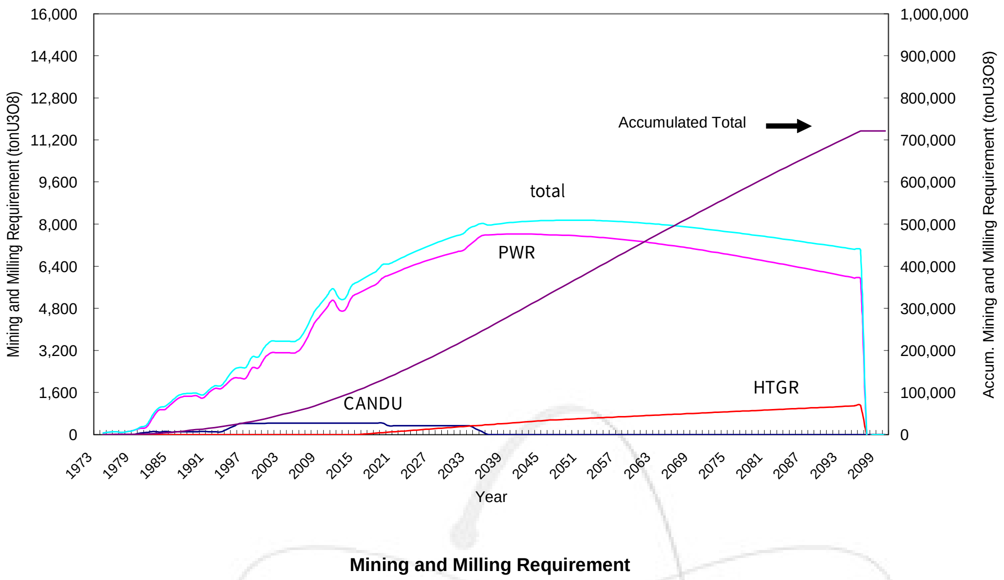

Fig. 152 Mining and Milling Requirement of Scenario Group 9

5.11 Scenario Group 10

5.11.1 Scenario Parameters (PWR + CANDU + HTGR + SFR + ADS)

Scenario Group 10 is the mixed system including the fast recycle system by use of SFRs and the once through system in PWRs and HTGRs. This group is the same as the Scenario Group 7 except that this group is including the ADS system. PWR spent fuel is reprocessed and reused in SFRs and/or ADSs. Spent fuels from the SFR and the ADS are continuously recycled in ADSs. The Tc and I generated from PWR are fabricated as a target fuel and transmuted in ADSs.

5.11.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Reactor parameters for HTGR are the same as those of the Scenario Group 3.
- Reactor parameters for SFR are the same as those of the Scenario Group 5.
- Reactor parameters for ADS are the same as those of the Scenario Group 7.

5.11.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Fuel parameters for HTGR are the same as those of the Scenario Group 3.
- Fuel parameters for SFR are the same as those of the Scenario Group 5.
- Fuel parameters for ADS are the same as those of the Scenario Group 7.

5.11.4 Outputs

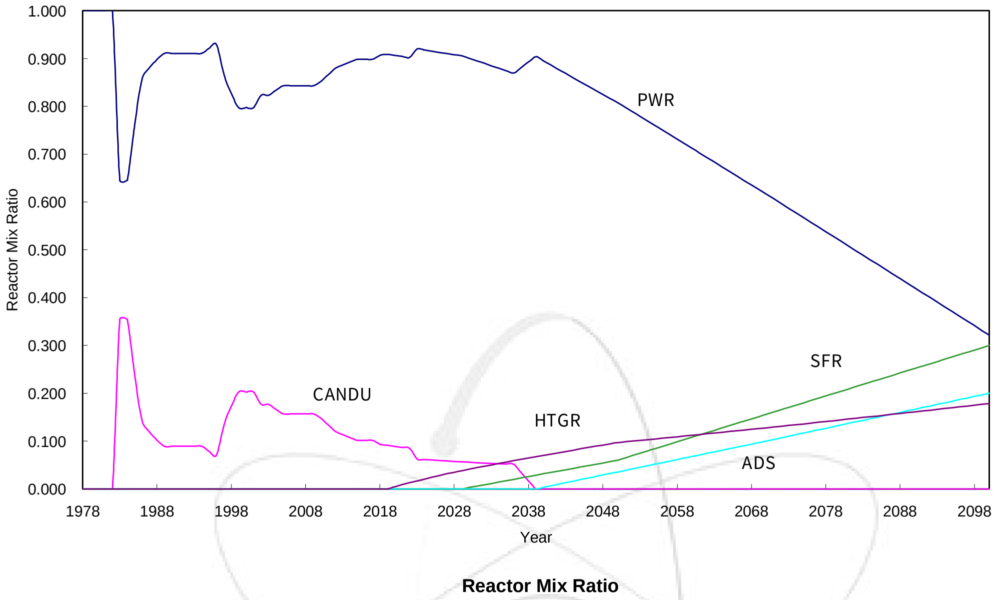

Fig. 153 Reactor Mix Ratio of Scenario Group 10

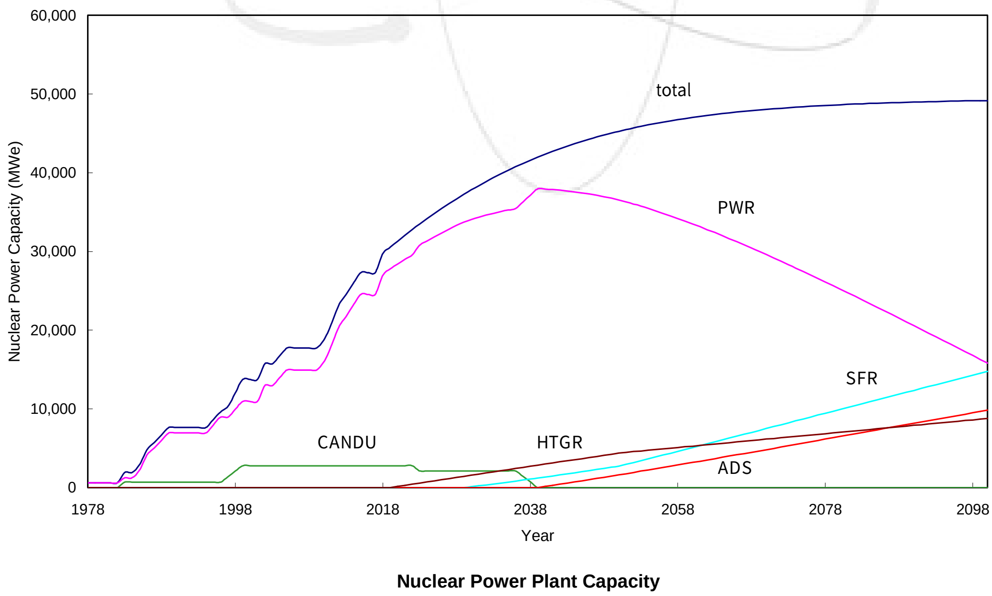

Fig. 154 Nuclear Power Plant Capacity of Scenario Group 10

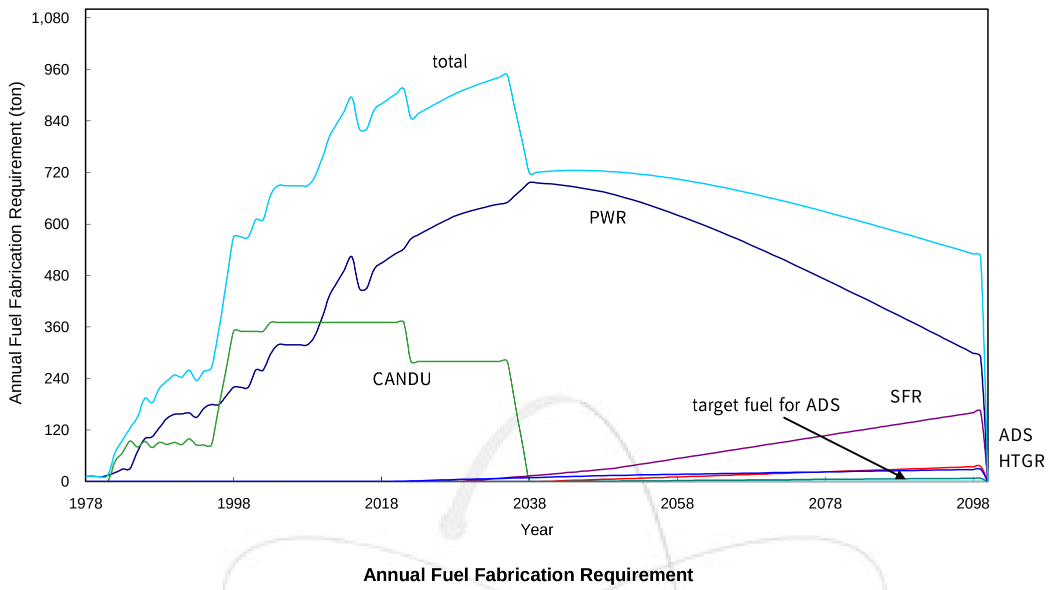

Fig. 155 Annual Fuel Fabrication Requirement of Scenario Group 10

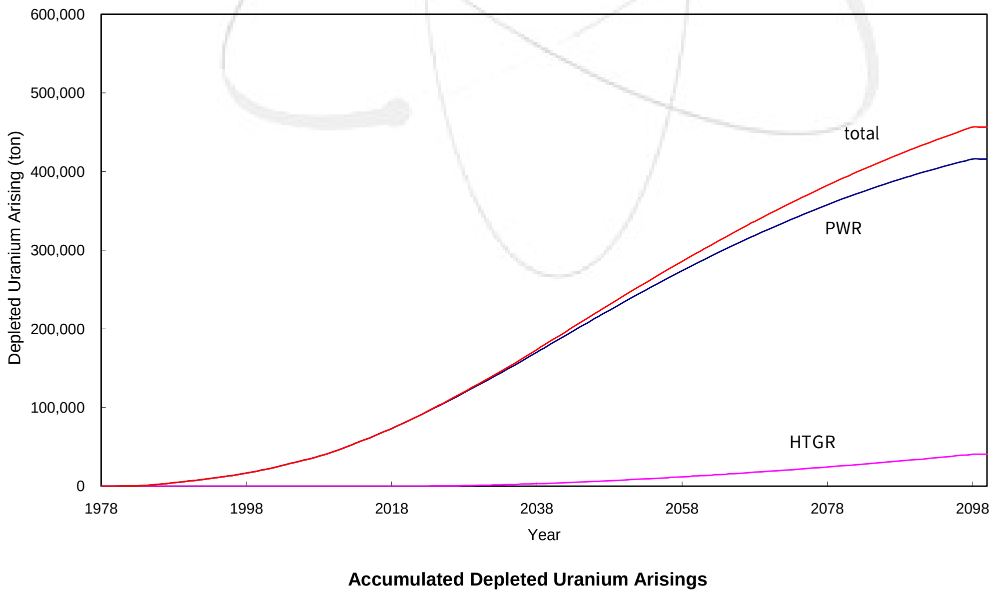

Fig. 156 Accumulated Depleted Uranium Arisings of Scenario Group 10

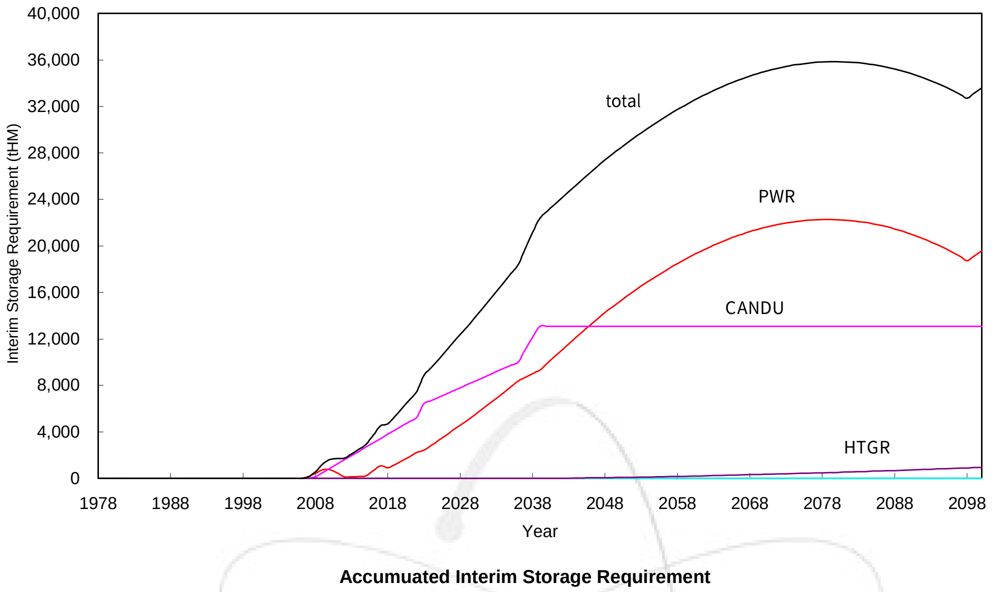

Fig. 157 Accumulated Interim Storage Requirement of Scenario Group 10

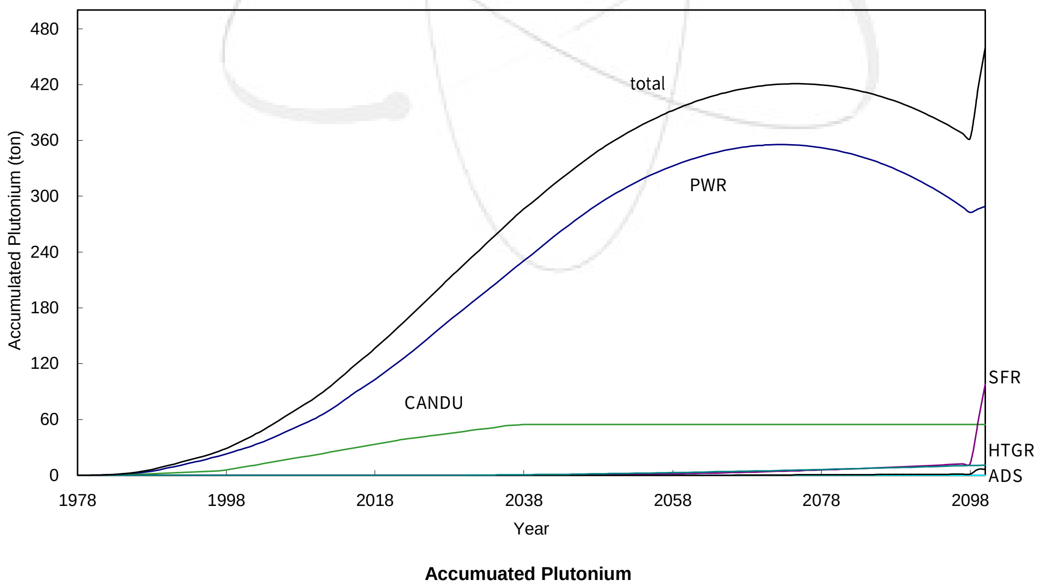

Fig. 158 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 10

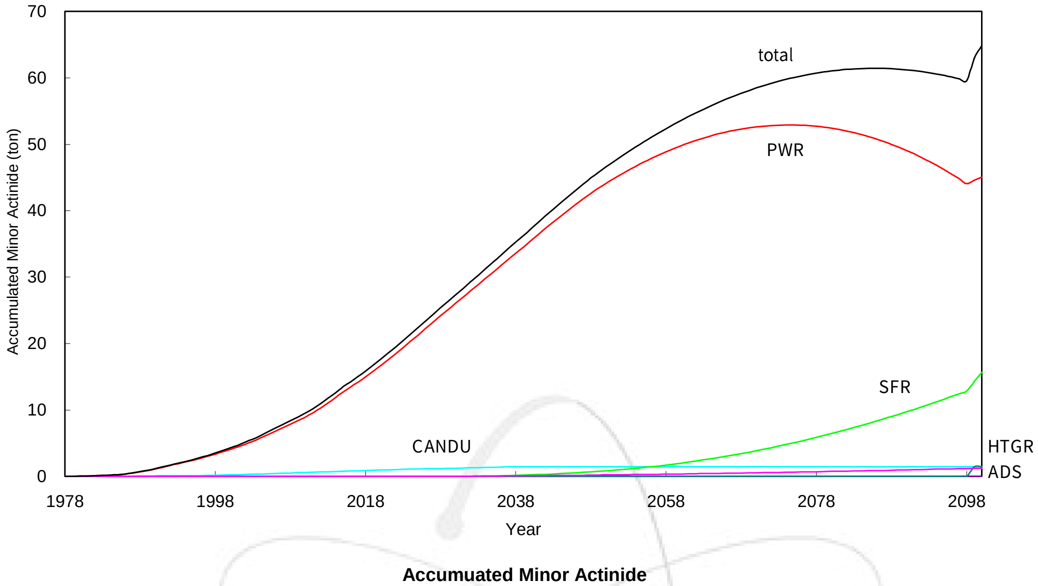

Fig. 159 Accumulated MA Embedded in Spent Fuels of Scenario Group 10

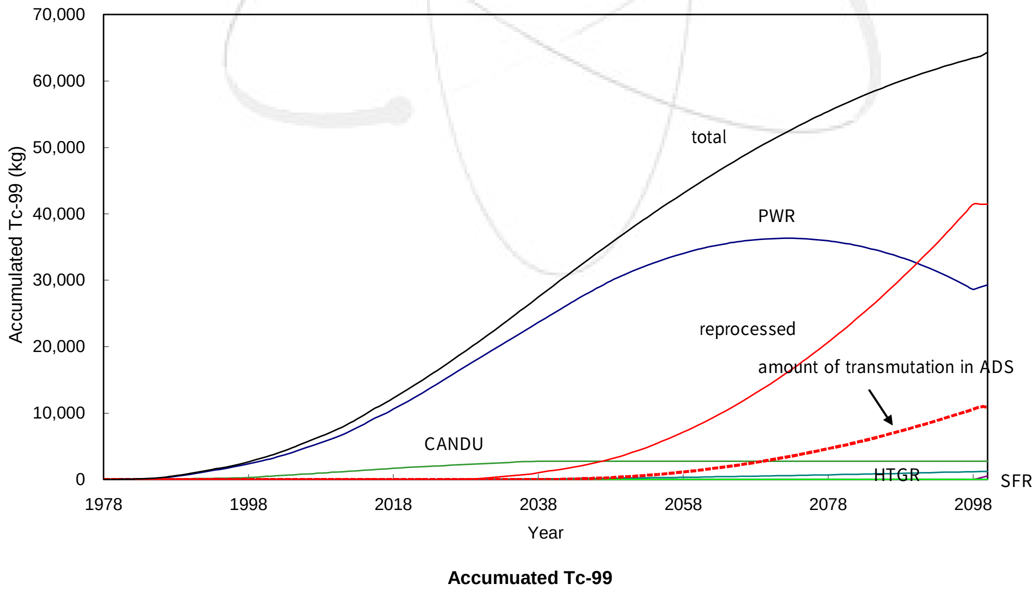

Fig. 160 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 10

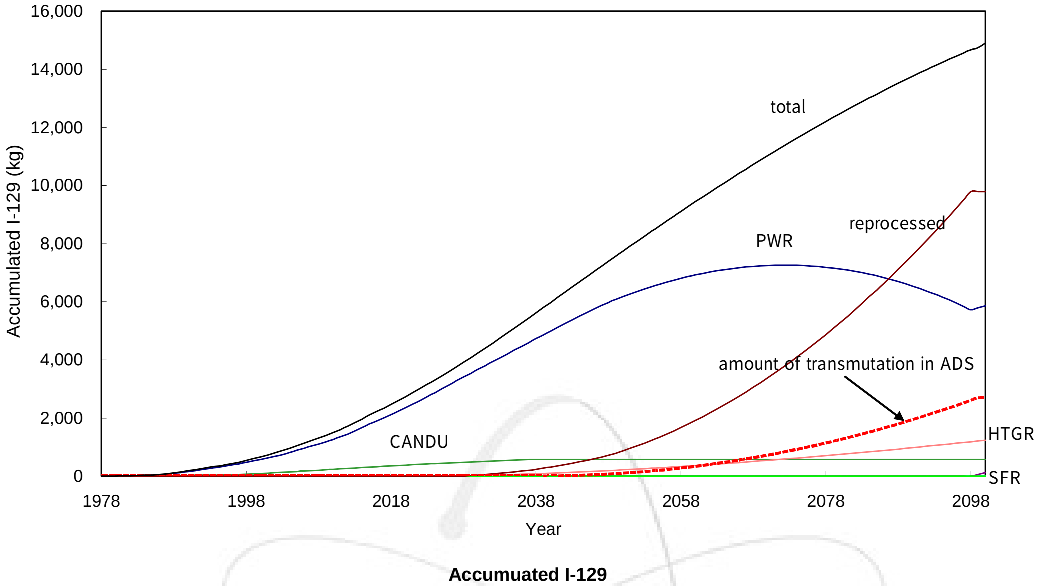

Fig. 161 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 10

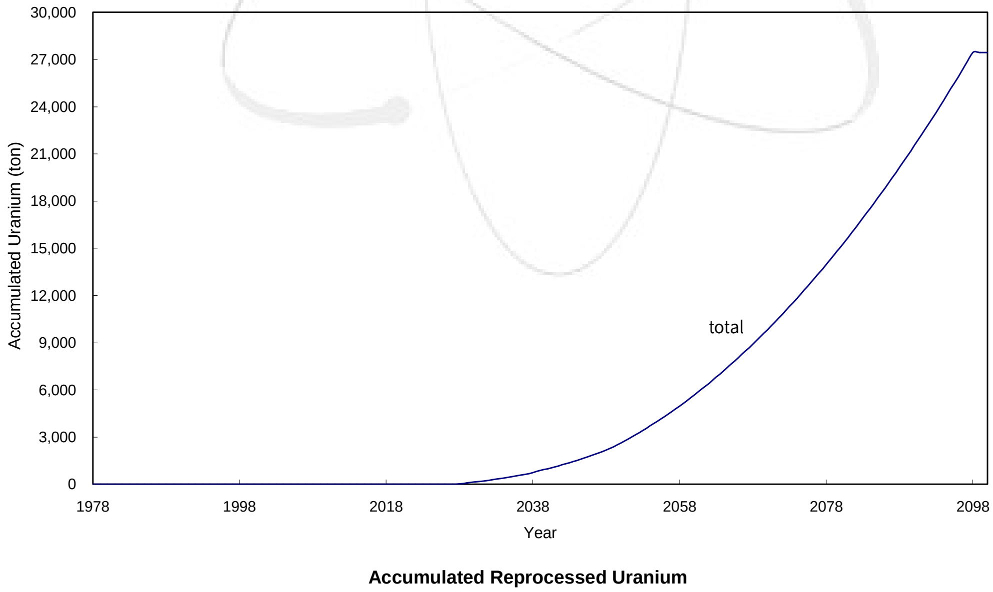

Fig. 162 Accumulated Reprocessed Uranium of Scenario Group 10

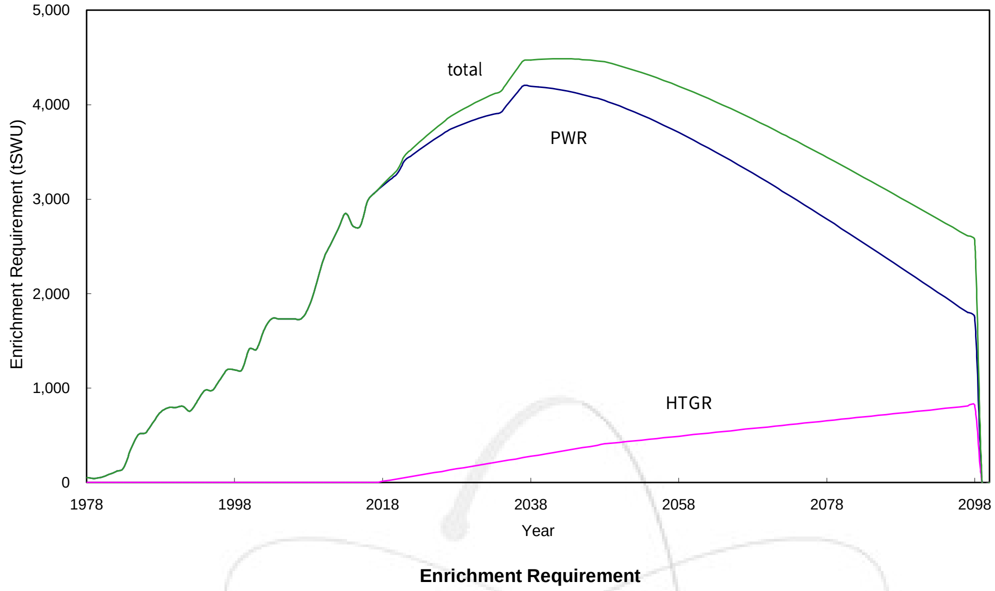

Fig. 163 Enrichment Requirement of Scenario Group 10

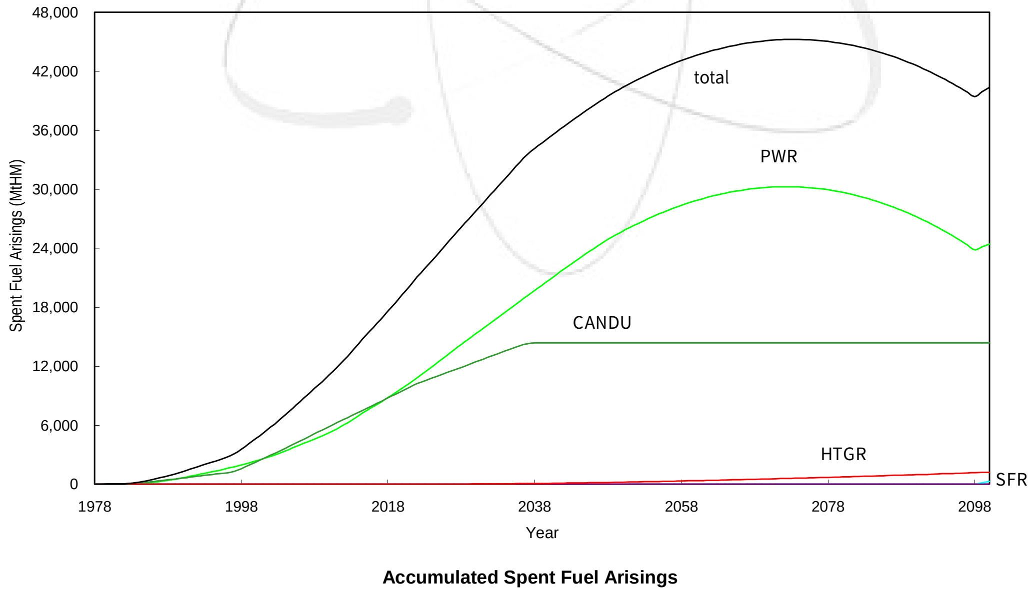

Fig. 164 Accumulated Spent Fuel Arisings of Scenario Group 10

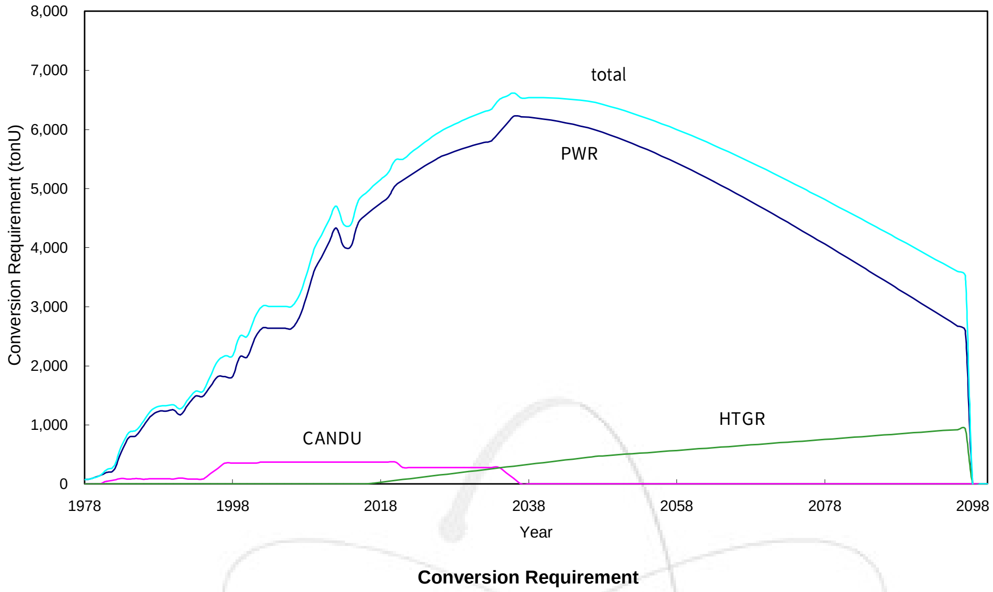

Fig. 165 Conversion Requirement of Scenario Group 10

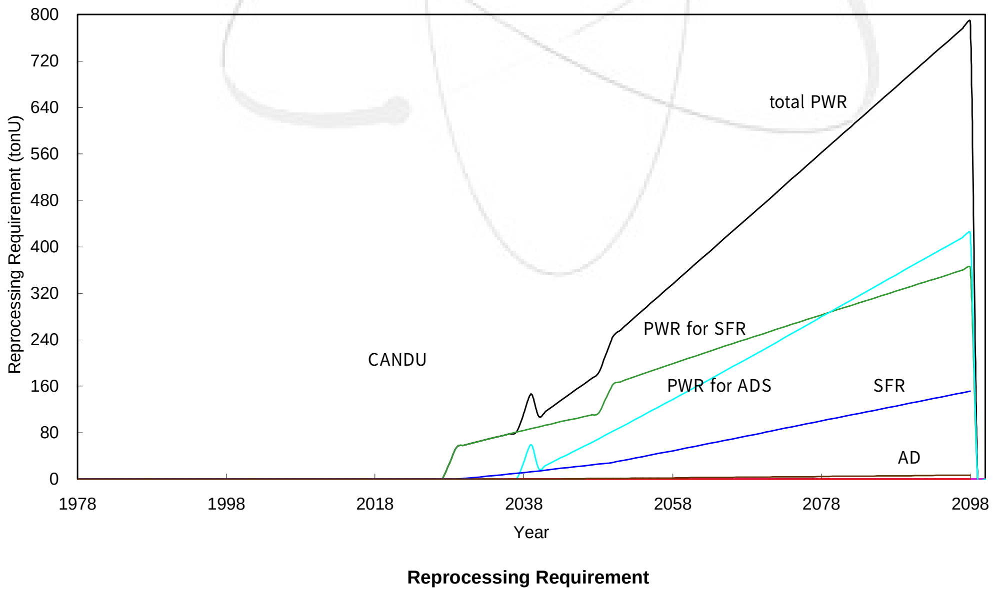

Fig. 166 Reprocessing Requirement of Scenario Group 10

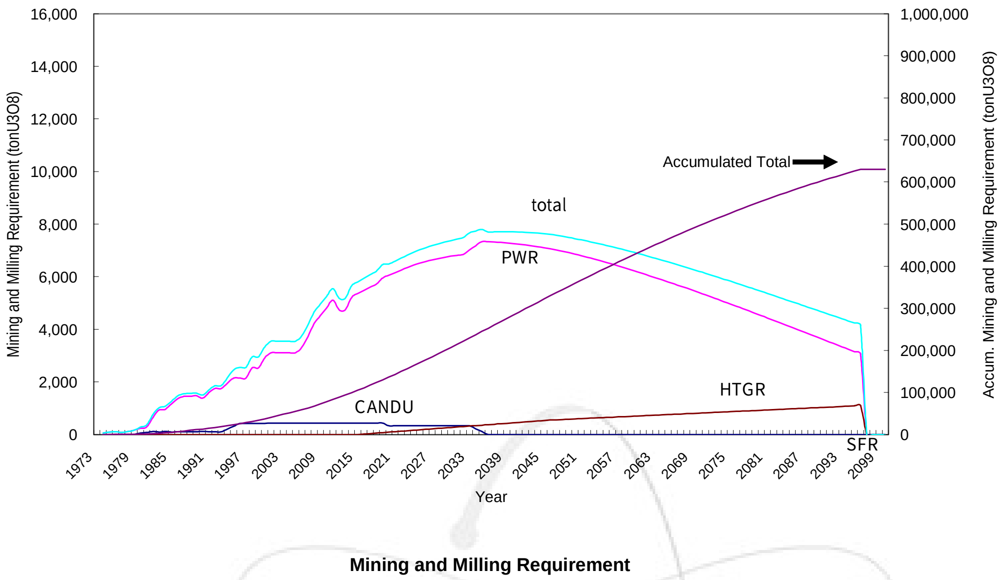

Fig. 167 Mining and Milling Requirement of Scenario Group 10

5.12 Scenario Group 11

5.12.1 Scenario Parameters (PWR + CANDU + HTGR + SFR + ADS)

Scenario Group 11 is the case that all possible reactors and nuclear fuel cycles are included. This group is the same as the Scenario Group 10 except that thermal recycle of MOX fuel in PWR is added. This scenario is similar to double strata concept in Japan and Europe and double tier concept in the US. The MOX fuel scenario in this group is a little different from that of Scenario Group 4. The ending year of the MOX fuel is added. It is assumed that the MOX fuel is used from 2020 and end in 2040. The core ratio of MOX fuel is assumed to be 20%.

5.12.2 Reactor Parameters

- Reactor parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Reactor parameters for HTGR are the same as those of the Scenario Group 3.
- Reactor parameters for SFR are the same as those of the Scenario Group 5.
- Reactor parameters for ADS are the same as those of the Scenario Group 7.

5.12.3 Fuel Parameters

- Fuel parameters for PWR and CANDU are the same as those of the Scenario Group 2.
- Fuel parameters for MOX are the same as those of the Scenario Group 4.
- Fuel parameters for HTGR are the same as those of the Scenario Group 3.
- Fuel parameters for SFR are the same as those of the Scenario Group 5.
- Fuel parameters for ADS are the same as those of the Scenario Group 7.

5.12.4 Outputs

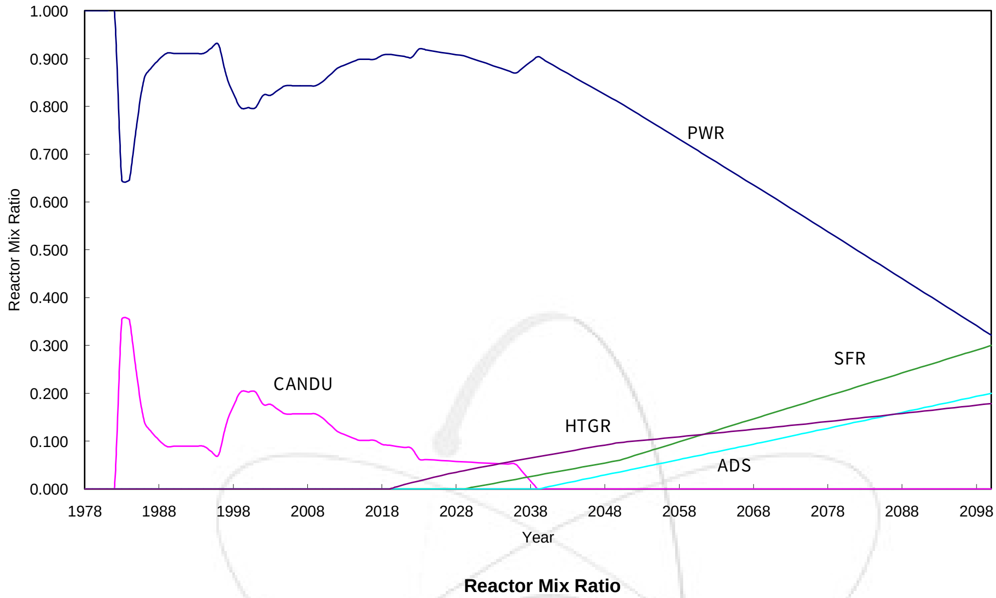

Fig. 168 Reactor Mix Ratio of Scenario Group 11

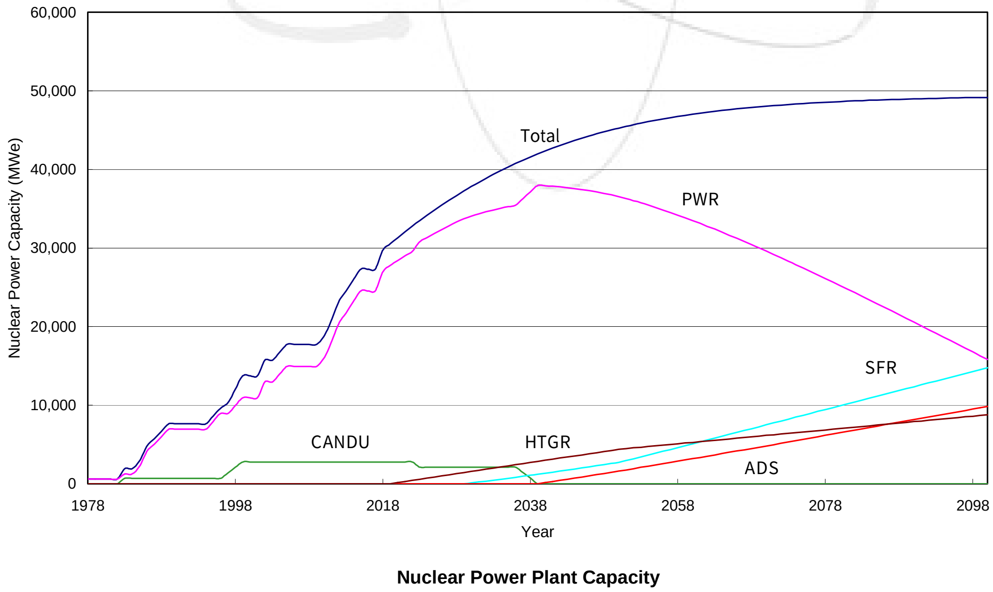

Fig. 169 Nuclear Power Plant Capacity of Scenario Group 11

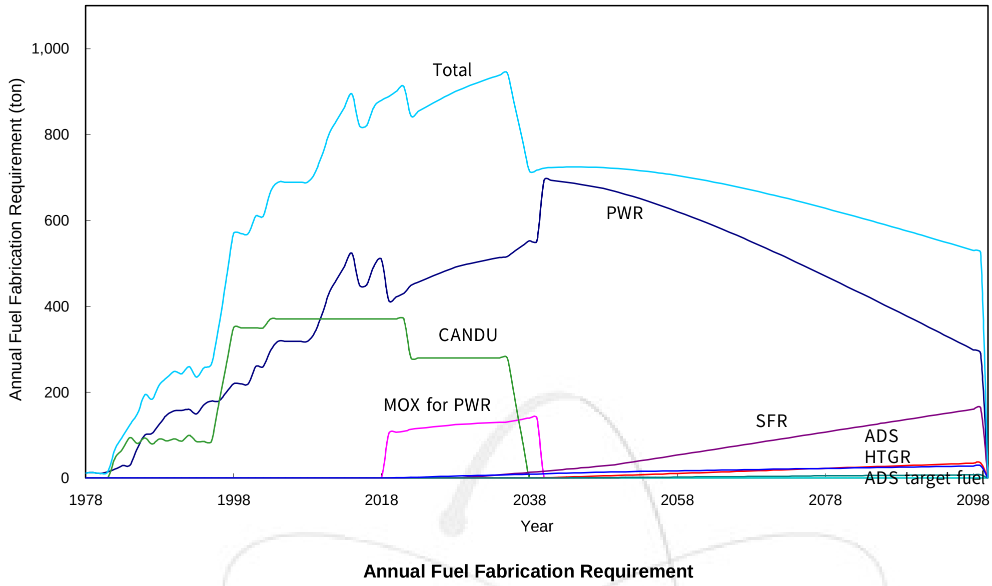

Fig. 170 Annual Fuel Fabrication Requirement of Scenario Group 11

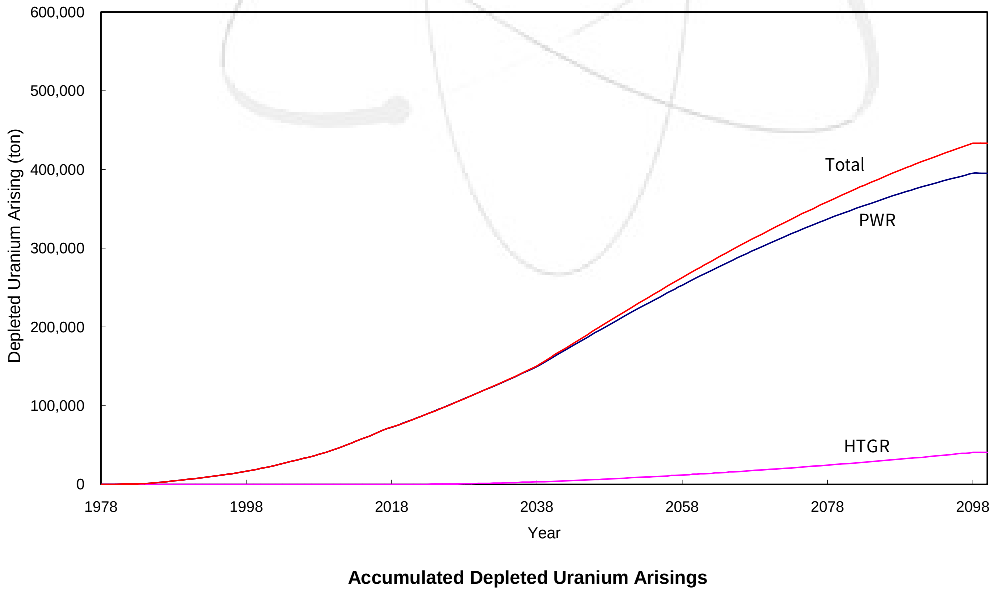

Fig. 171 Accumulated Depleted Uranium Arisings of Scenario Group 11

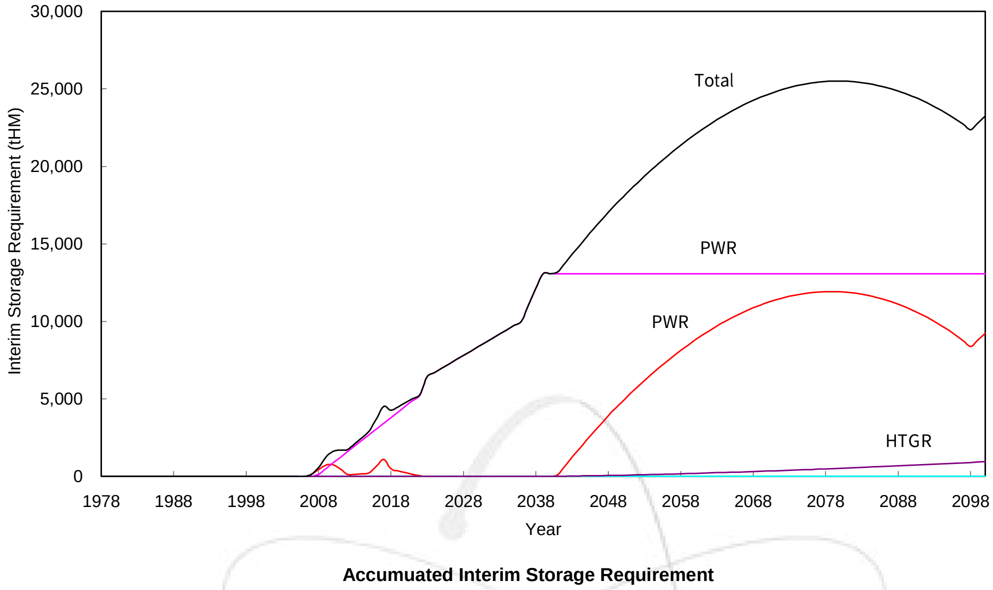

Fig. 172 Accumulated Interim Storage Requirement of Scenario Group 11

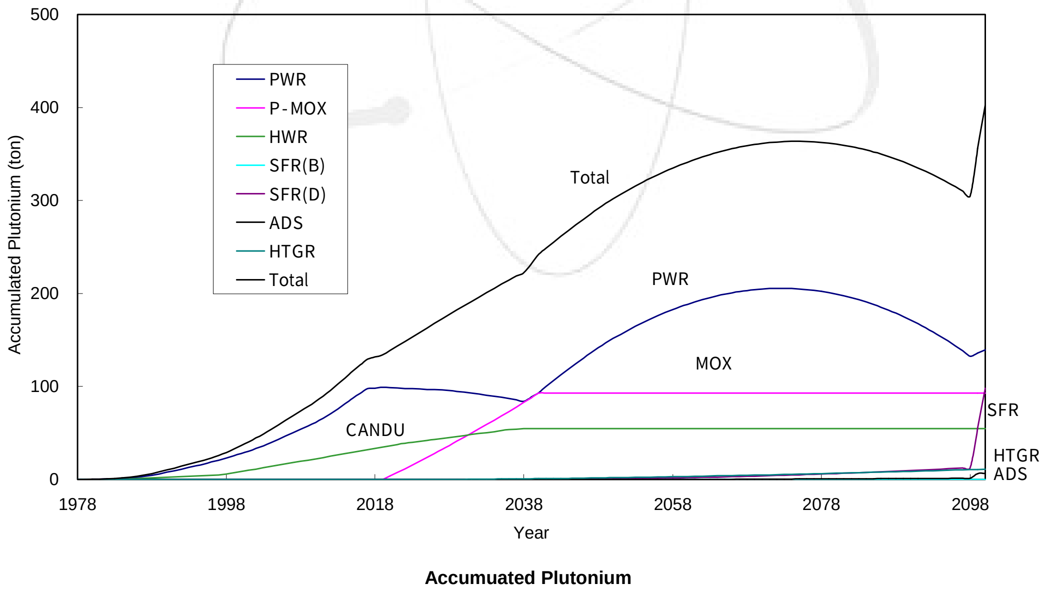

Fig. 173 Accumulated Plutonium Embedded in Spent Fuels of Scenario Group 11

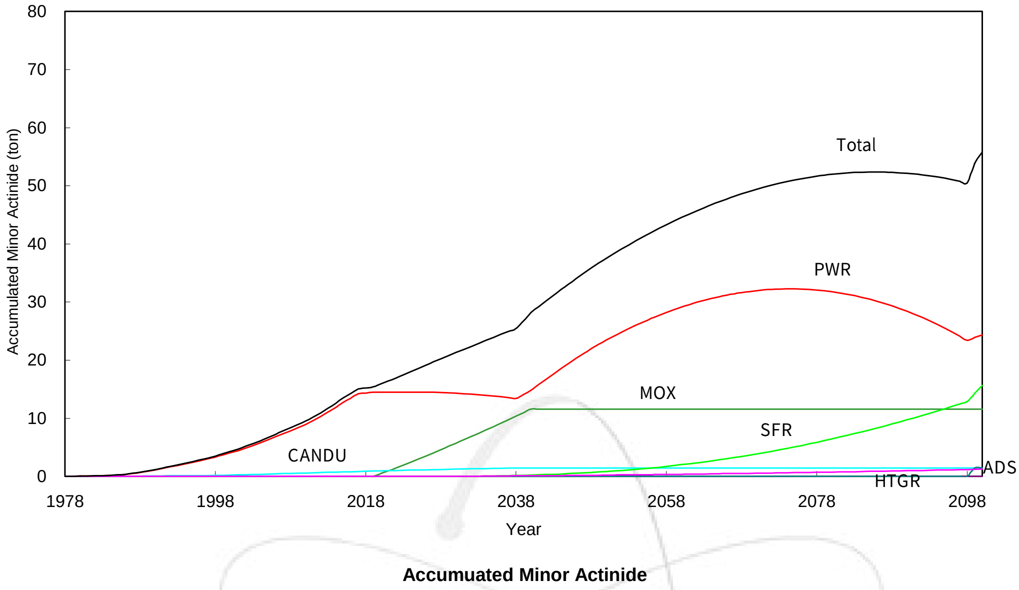

Fig. 174 Accumulated MA Embedded in Spent Fuels of Scenario Group 11

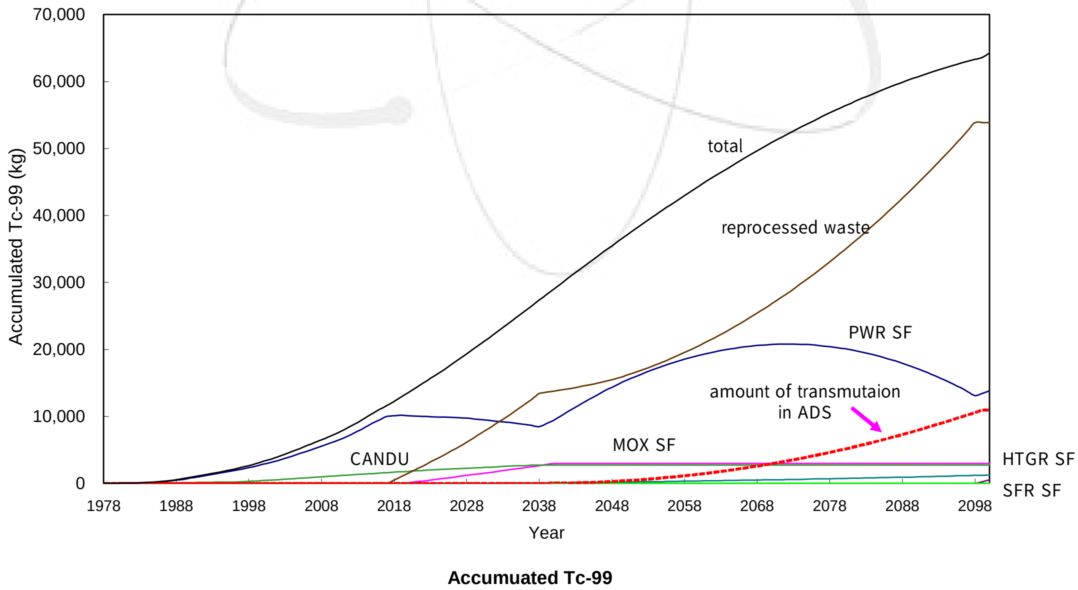

Fig. 175 Accumulated Tc-99 Embedded in Spent Fuels of Scenario Group 11

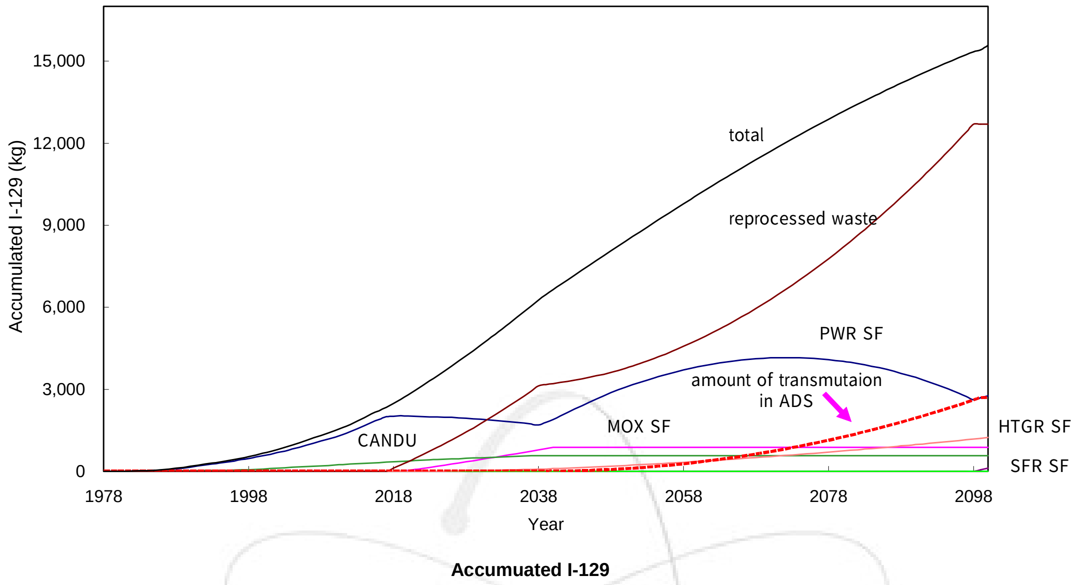

Fig. 176 Accumulated I-129 Embedded in Spent Fuels of Scenario Group 11

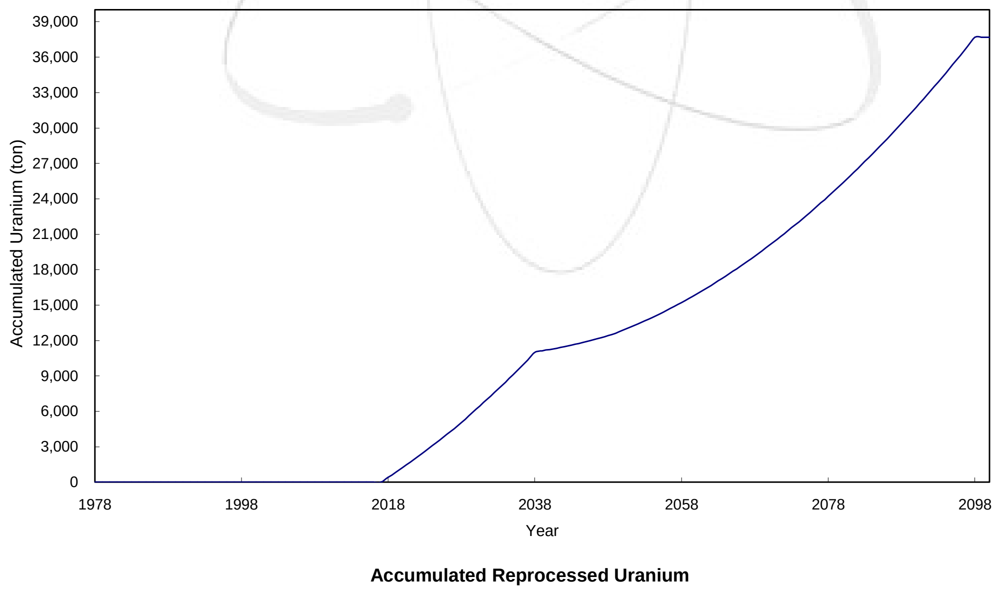

Fig. 177 Accumulated Reprocessed Uranium of Scenario Group 11

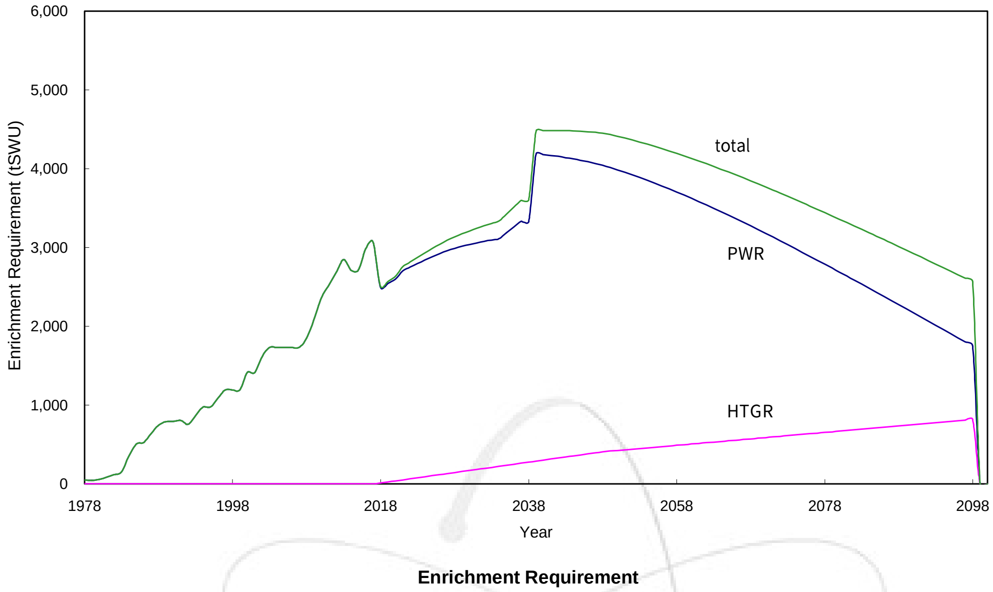

Fig. 178 Enrichment Requirement of Scenario Group 11

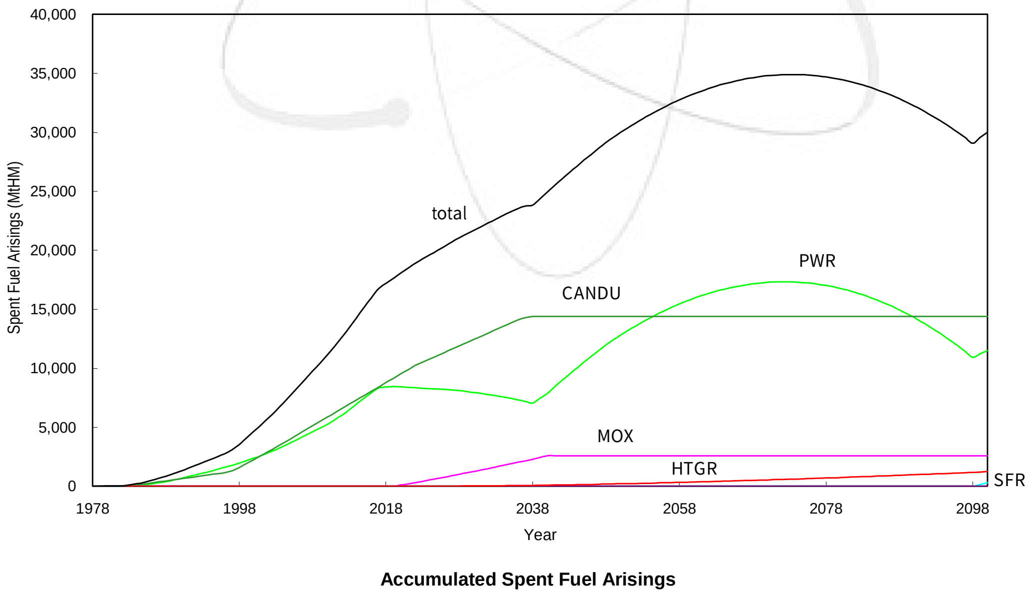

Fig. 179 Accumulated Spent Fuel Arisings of Scenario Group 11

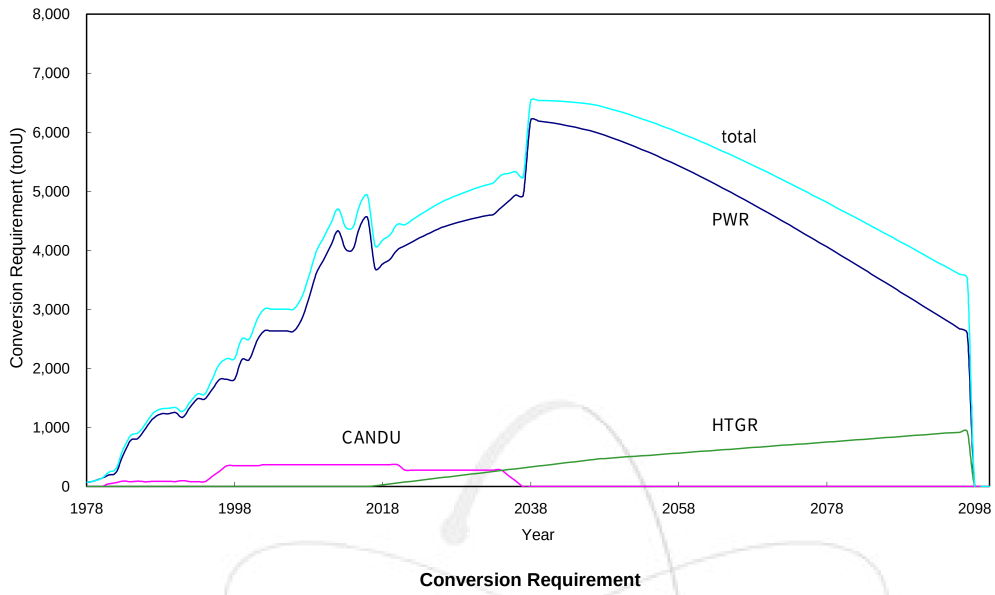

Fig. 180 Conversion Requirement of Scenario Group 11

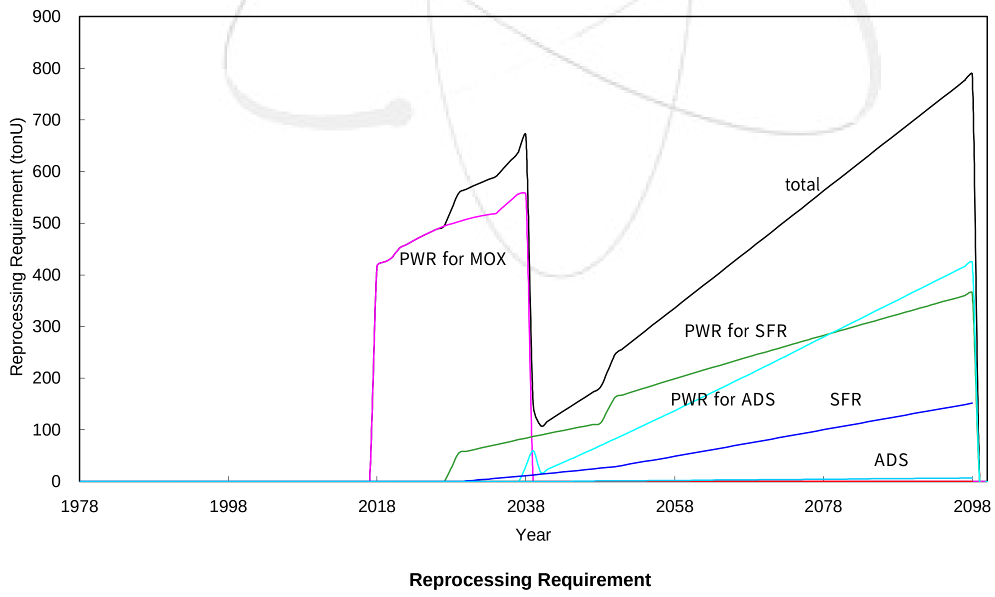

Fig. 181 Reprocessing Requirement of Scenario Group 11

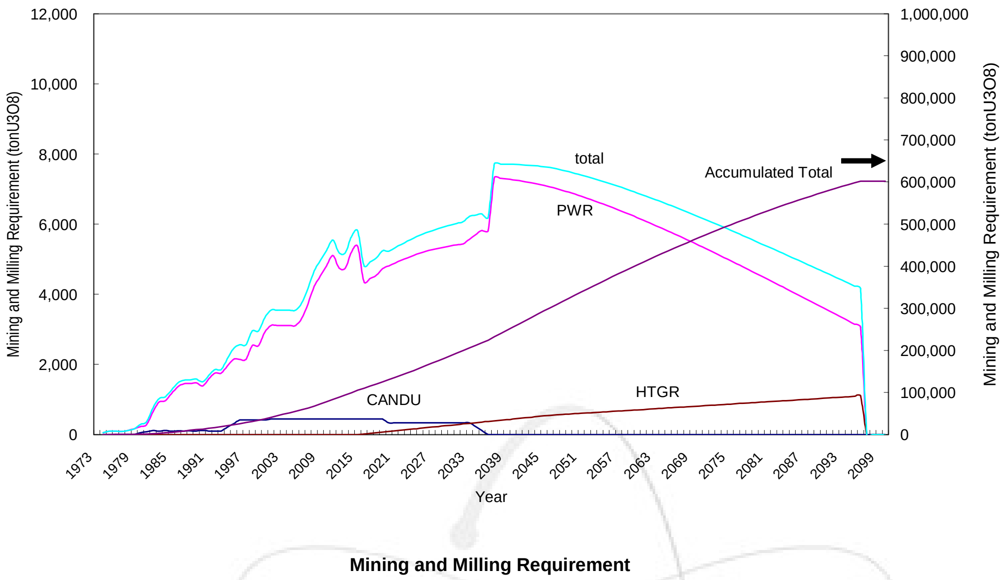

Fig. 182 Mining and Milling Requirement of Scenario Group 11

6. Conclusions

This paper describes the Nuclear Fuel Cycle Analysis and Simulation Tool (FAST) which has been developed by the Korea Atomic Energy Research Institute (KAERI). As described in Chapter III, the FAST categorizes various mixes of nuclear reactors and fuel cycles into 11 scenario groups. The reactors that the FAST is taking into consideration are the pressurized water reactor (PWR), the Canadian deuterium uranium reactor (CANDU), the sodium fast reactor (SFR), the accelerator driven system (ADS) and the high temperature gas cooled reactor (HTGR). The FAST then calculates all the required quantities for each nuclear fuel cycle component, such as mining, conversion, enrichment, fuel fabrication, interim storage and final disposal for each scenario. Such calculation could be performed year by year or for a particular time period, up to the year of 2100.

Compared with other simulation codes, the FAST has several advantages. First, it has a logistics function which links the code to the FORECAST, a program that projects a future nuclear energy need in Korea and whose output is transferred automatically to the FAST. Such a function makes the output from the FAST much more reliable, for the analysis is made based on the most updated nuclear energy demand.

Second, the FAST employs a MS Excel spread sheet with the Visual Basic Application. Such an application allows users to manipulate the program with ease. The speed of the calculation is also quick enough to make comparisons among different options in a considerably short time.

This user-friendly simulation code now allows mass flow analysis and is being extended to include cost and environmental aspects on it. With completion of the program, more comprehensive analysis will be able to be made, and hopefully will contribute to the decision making process of shaping the future nuclear fuel cycle development paths in Korea.

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Abstract (15-20 Lines)

This paper describes the Nuclear Fuel Cycle Analysis and Simulation Tool (FAST) which has been developed by the Korea Atomic Energy Research Institute (KAERI). Categorizing various mixes of nuclear reactors and fuel cycles into 11 scenario groups, the FAST calculates all the required quantities for each nuclear fuel cycle component, such as mining, conversion, enrichment, fuel fabrication, interim storage, and final disposal for each scenario. A major advantage of the FAST is that the code employs a MS Excel spread sheet with the Visual Basic Application, allowing users to manipulate it with ease. The speed of the calculation is also quick enough to make comparisons among different options in a considerably short time. This user-friendly simulation code is expected to be beneficial to further studies on the nuclear fuel cycle to find best options for the future, all proliferation risk, environmental impacts and economic costs considered.

Subject Keywords (About 10 words)	Nuclear Fuel Cycle, Material Flow, Nuclear Energy, FAST Code

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연구위탁기관		계약 번호	

초록 (15-20줄내외)

본 보고서는 최근 한국원자력연구소에서 개발된 핵연료주기 분석 코드인 FAST에 대해서 코드 설명과 사용 방법, 그리고 수학적 모델을 기술하고 있다. 또한 각 시나리오에 대한 Case Study 결과를 포함하고 있어 사용자가 친숙하게 사용할 수 있도록 하고 있다. FAST 코드는 고속로, 경수로, 중수로, 고온가스로 등이 혼합된 11개의 핵연료주기 시나리오를 다룰 수 있으며, 우라늄 채광에서부터 고준위폐기물 처분까지의 핵연료주기 구성 부문별 물질흐름을 계산할 수 있다. 특히 FAST는 로지스틱 함수를 이용하여 우리나라의 원자력 발전 전망을 예측할 수 있는 FORECAST 모델을 포함하고 있다. FAST 코드는 MS Excel spread sheet와 Visual Basic Application를 이용하여 만들었기 때문에 계산속도도 충분히 빠르고, 사용자가 이해하기 쉬우며, 사용자 요건에 맞게 수정하여 사용할 수 있는 장점이 있다. 본 코드는 향후 우리나라의 핵연료주기 분석하는데 유용하게 사용될 수 있을 것으로 기대된다.

주제명키워드 (10단어내외)	핵연료주기, 물질흐름, 원자력 에너지, FAST, 시뮬레이션