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Simulations of the Chinese Nuclear Fuel Cycle Scenario Using a New Code

YOUPENG ZHANG

Master of Science Diploma Work

Department of Reactor Physics

Royal Institute of Technology

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Abstract

One of the most important affairs in the nuclear industry is the fuel cycle situation prediction. It affects the energy company's profit, environment and even the safety of reactor operation. For these reasons, a series of computer codes have been generated to simulate the fuel cycle scenario including NFCSim, ORION and so on. At the Department of Reactor Physics, a new fuel cycle simulation code is under development and this code will be used in the present thesis.

In order to simulate the nuclides transmutation chains, MCNP was first used to calculate the neutron spectrum and cross section data for the reactor cores, using JEF 3.0 and EAF 99 data libraries.

The main task of this project is to simulate the present and future status of all the facilities in Chinese reactor park. Three consecutive scenarios (present, near-term and long-term) are defined for this comparison, simulation time scale is set to be 208 years (1992~2200) and four groups of nuclides (major actinides, minor actinides, major fission products and safety related nuclides) are defined and presented.

Power balance scenario, plutonium self-sustained scenario and CIAE proposals are discussed individually as choices of reactor parks' future development. The result is that at least 70 years (cooling storage time is not included) are needed to transmute the minor actinides inventory after the large-scale FBR (Fast Breeder Reactor) technology is mature enough for large scale commissioning in plutonium-sustained scenario.

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List of Abbreviations

ADS	Accelerator Driven System
AECL	Atomic Energy of Canada Limited
AEE&ZAES	Atomenergoexport, Russia&Russia Production Association
BE	Binding Energy
BOC	Begin Of Cycle
CANDU	CANada Deuterium Uranium
CEFR	China Experimental Fast Reactor
CIAE	China Institute of Atomic Energy
CNNC	China National Nuclear Corporation
COGEMA	Compagnie générale des matières nucléaires
CPR	China Pressurized-water Reactor
dpa	Displacements per Atom
EAF	European Activation File
EOC	End of Cycle
FBR	Fast Breeder Reactor
HLW	High Level Waste
HPPA	High Power Proton Accelerator
HTGR	High Temperature Gas-cooled Reactor
JEFF	Joint Evaluated Fission and Fusion (JEFF) project
kgHM	kilogram Heavy Metal
LBE	Lead-Bismuth Eutectic
LINAC	Linear Particle Accelerator
LFBR	Liquid metal Fast Breeder Reactor
MA	Minor Actinide
MCR	Monte Carlo Reactor code

MCNP	Monte Carlo N-Particle Transport Code
MCNPX	Monte Carlo N-Particle eXtended
MFBR	Module Fast Breeder Reactor
MOST	China Ministry of Science and Technology
MOX	Mixed OXide
NEA	Nuclear Energy Agency
NFCSim	Nuclear Fuel Cycle Simulation code
ORIGEN	Oak Ridge Isotope Generation code
ORNL	Oak Ridge National Laboratory
PUREX	Plutonium and Uranium Recovery by EXtraction
RBMK	Reaktor Bolshoy Moshchnosti Kanalniy
RDT	Reactor Doubling Time
TBP	TriButyl phosphate
VVER	Voda-Vodyanoi Energetichesky Reactor
XSDIR	Cross Sections DIRectory

Contents

1. Introduction

When mentioning the sentence "China is the biggest country in the world", generally, we are talking about the population. Although the population of China is 1.3128 billion (until 23rd, Sep, 2006) [1], the average energy usage is only 0.9 ton crude oil / capita (11.5% of the value for USA) [2]. The irrational energy structure and the imbalance in energy source distribution leads to heavy burdens both for the economic development and the environment, shown in Figure 1 and 2:

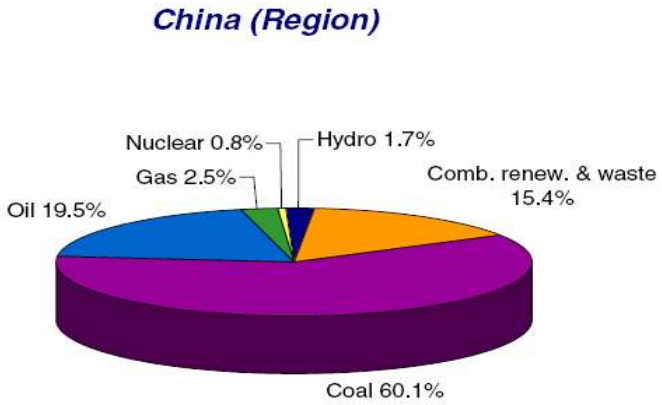


Figure 1: Share of Total Primary Energy Supply (2003) [3]

According to Figure 1, we can find out that the main energy source supporting China is coal. Coal generates 96% CO_2, SO_2 and NOx of the total emissions in the electricity generation industry while it contributes only to 60% of the electricity. Moreover, it can be observed from Figure 2 that most of the coalmines are located in the northern part of China (black circle region), and they are at least 2000 kilometers away from the economic center (red circle region). This kind of situation brings heavy burden to transportation system and pollution control task.

Figure 2: Locations of Coalmine and Economic Center

For these reasons, the Chinese government started the nuclear development plan from 1980s and installed the first commercial reactor - Qin Shan (Phase 1) by Dec.1st, 1991. Accompanied by the economic boom during 1990s, the energy demand increased sharply as shown in Figure 3. It is not difficult to construct more fossil fuel power plants. Actually, about 1000 thermal power plants were built up from 1950 to 2000. However, the coalmines production is limited by the source amount and series of accidents, the coal price in the east and south part of China has risen by a factor of three in the recent 10 years.

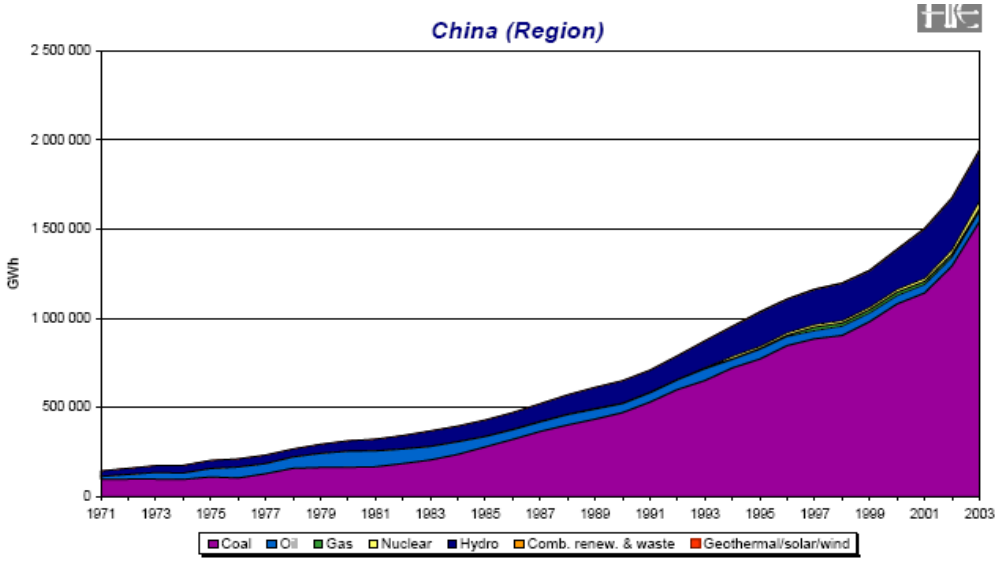


Figure 3: Electricity Generation from Fuel (1971~2003) [3]

This situation forces the energy companies that are far away from the coalmines to pay more attention on new energy sources, including nuclear power. Since the first nuclear reactor's connection to the grid in 1991, 11 reactors (listed in Table 1) have been commissioned in the mainland of China, which have raised the nuclear energy's share in the whole energy market from less than 0.1 % to 0.8%. In the next 14 years (until 2020), about 32 reactors will be arranged according to the short-term energy plan of the Chinese government, which will further enhance the proportion of nuclear energy to 6%.

Name	Type	Location	El Output	Start Date	Reactor Supplier
Qinshan Phase 1	PWR	Haiyan, Zhejiang Province	310MW	Dec.1st, 1991	CNNC
Qinshan Phase 2A	PWR, CNP 650	Haiyan, Zhejiang Province	670MW	Apr 15th, 2002	CNNC
Qinshan Phase 2B	PWR, CNP 650	Haiyan, Zhejiang Province	670MW	Apr 1st, 2003	CNNC
Qinshan Phase 3A	CANDU 6,700	Haiyan, Zhejiang Province	728MW	Feb 12th, 2003	AECL
Qinshan Phase 3B	CANDU 6,700	Haiyan, Zhejiang Province	728MW	Nov 12th, 2003	AECL
Daya Bay Unit1	PWR	Lingao, Guangdong Province	700MW	Feb 1st, 1994	AREVA
Daya Bay Unit2	PWR	Lingao, Guangdong Province	700MW	May 7th, 1994	AREVA
Lingao Phase 1A	PWR, M310	Lingao, Guangdong Province	990MW	May 28th, 2002	AREVA
Lingao Phase 1B	PWR, M310	Lingao, Guangdong Province	990MW	Mar 15th, 2003	AREVA
Tianwan Phase 1A	VVER, AES 91	Lianyun Gang, Jiangsu Province	1060MW	May 1st, 2006	AEE&ZAES
Tianwan Phase 1B	VVER, AES 91	Lianyun Gang, Jiangsu Province	1060MW	Dec 30th, 2006	AEE&ZAES

Table 1: Present Chinese Reactor Park

Following this trend, several challenges will arise: 1. Increased demands for fuel elements manufacture capability. Currently, there are two fuel factories (listed in Table 2) to supply refueling elements for current 11 reactors. It is urgent to

expand the current capacity; 2. Search for the solution of the nuclear waste; 3. Upgrade the reactor safety system; 4.Checking the necessity of using advanced technology such as ADS, FR, HTGR and so on. Simulation of the fuel cycle scenario is a key role in these four factors realization.

Uranium Mine	Start Date	Capability (tons/year)
Native and Kazakhstan	1980.01	750
Australia	2005.01	2000
Niger	2006.10	1000
Fuel Factories	Location	Fuel Type
Yibin (812) Factory	Yibin, Sichuan	AFA-2G, AFA-3G
Baotou (202) Factory	Baotou, Neimenggu	CNP650 and CANDU 6
Fast Reactor	Start Construction	Location
CEFR (China Experimental Fast Reactor)	2000.05	Beijing
Waste Depository	Commission Date	Location
Beishan HLW (High-level Waste) Depository	1985	Beishan, Gansu Province

Table 2: Nuclear Establishments Relevant to Chinese Reactor Park

The nuclear industry has been evolving for more than fifty years in global range so a series of codes have been composed to accomplish the fuel cycle simulation task, such as, ORION, NFCSim and so on [23]. Compared with the former codes, the new one is upgraded to make the simulation simpler and faster. Although this advantage is compensated by a lower accuracy of the result, you will find out that the results generated from this new code is good enough in some areas, like policy shaping, budget setting, trend tracing etc.

In the following chapters of this thesis, the new fuel cycle code will be compared with classic ones and descriptions of the components included in it will be mentioned as well. Then, Chinese nuclear park scenario is created step by step with both assumptions and deterministic calculations. Eventually, analysis of the results generated by new code will be given.

2. Fundamentals of Nuclear Reaction

2.1 Binding Energy

The nuclear reactions interesting for energy production can be divided into two chief areas: the fusion reaction and fission reaction. Both of them are based on the same formula -- $\Delta E=\Delta Mc^2$ proposed by A. Einstein. From this equation, it can be observed that mass change of one substance corresponds to either exothermic or endothermic reactions. Fission reaction of actinides is severe exothermic reaction [4]. Before talking about the nuclear reaction, two terms should be clarified: one is "Binding Energy" and the other is "Cross Section".

When two or more entities merge into a new entity, some energy will be released because of the mass change after such kind of reaction. This energy is named Binding Energy (or BE in short). The relation between mass defect (ΔM) and the BE can be shown in the formula $BE=\Delta Mc^2$.For a given kind of nuclide ${}^{A}_{Z}X$ (A means Atomic number and Z means Proton number), the Binding Energy can be described as:

$$BE/c^2 = \Delta M = Z*m_p + (A-Z)*m_n - m({}^{A}_{Z}X) \quad (1)$$

m_p means proton mass, m_n means neutron mass and $m({}^{A}_{Z}X)$ means the mass of the nucleus. Since atomic masses can easily be obtained from mass tables, it is more convenient to rewrite Equation (1) using atomic masses as Equation (2).

$$BE/c^2 = Z*\left[M({}^{1}_{1}H) - m_e + BE_{1e}/c^2\right] + (A-Z)*m_n - \left[M({}^{A}_{Z}X) - Z*m_e + BE_{Ze}/c^2\right] \quad (2)$$

Hence, in Equation (2), M represents atomic mass and m_e corresponds to electron mass. The first term uses the hydrogen nucleus (contains no neutron) to get the proton mass. The third term is the transformation one from Equation (1), which treats the nuclide as the combination of electrons (e) and nucleus (${}^{A}_{Z}X$). After rearrangement, the Equation (2) can be rewritten into Equation (3).

$$BE/c^2 = Z*M({}^{1}_{1}H) + (A-Z)*m_n - M({}^{A}_{Z}X) + (Z*BE_{1e} - BE_{Ze})/c^2 \quad (3)$$

The binding energy in the third term of the right side of the Equation (3) can be omitted because the binding energy between electrons and protons is millions of times smaller than the one in nucleus. Therefore, the third term can

be neglected and the Equation (4) will be deduced, which can be considered as the final expression of the Binding Energy.

$$BE / c^2 = Z * M({}_1^1H) + (A - Z) * m_n - M({}_Z^AX) \quad (4)$$

From the Figure 4 below, it can be noticed that the uranium nuclide has lower binding energy per nucleon than medium mass nuclides in the range around from 40 to 160, which means that if the uranium nucleus can be split into two or more lighter nuclides, parts of the binding energy will be released. This is the source of nuclear energy. To realize such kind of reaction, two conditions must be fulfilled: fissionable nuclides should participate and the initial binding energy per nucleon must be lower than boundary binding energy level.

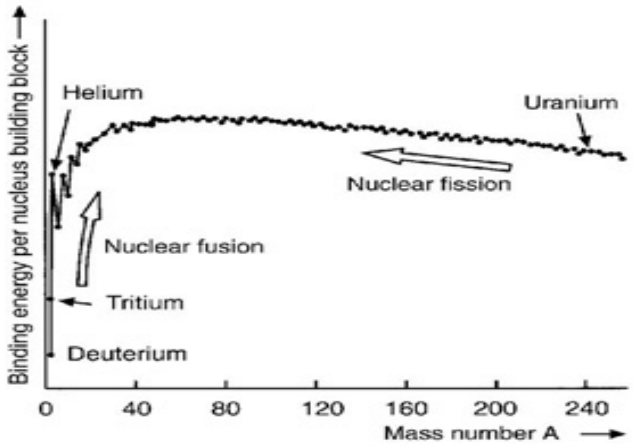


Figure 4: Binding Energy Diagram [5]

Fissile nuclides can be considered as the subset of "fissionable" nuclides, which means the nuclides that can undergo nuclear fission reaction. Besides this property, fissile nuclide just needs to absorb a neutron that has very small kinetic energy [7] (such as in thermal region) in order to fission. For the fissile nuclides, if one nucleus absorbs a neutron, its BE/A value will decrease, which will move the nucleus to the meta-stable state. Then, this nucleus will split into two lighter nuclei with higher BE/A value (more stable state), fast neutrons (needed in the chain reaction) and other energetic radiation (such as beta and gamma radiation). In this process, the binding energy decrease mainly converts to the kinetic energy of the fission products.

Fissile nuclides play a significant role in thermal reactors, in which prompt neutrons generated from reactions are moderated to thermal state and then

absorbed, which causes additional fission reactions. Some common fissile nuclides are U-233, U-235 and Pu-239 [4].

2.2 Cross Section

"Cross Section" is a term that can be considered as a description of the probability for nuclide at given energy level to react with another given nuclides. In this definition, a kind of effective disc-shaped area around the nucleus is specified to face the incident projectile. For example, when a neutron in a reactor comes into this area, it can interact with the given nucleus. This area is named as the "Cross Section" for this nucleus. Its common unit is "barn" (10^{-28} m^2), which is not a SI unit, but easier to write and remember.

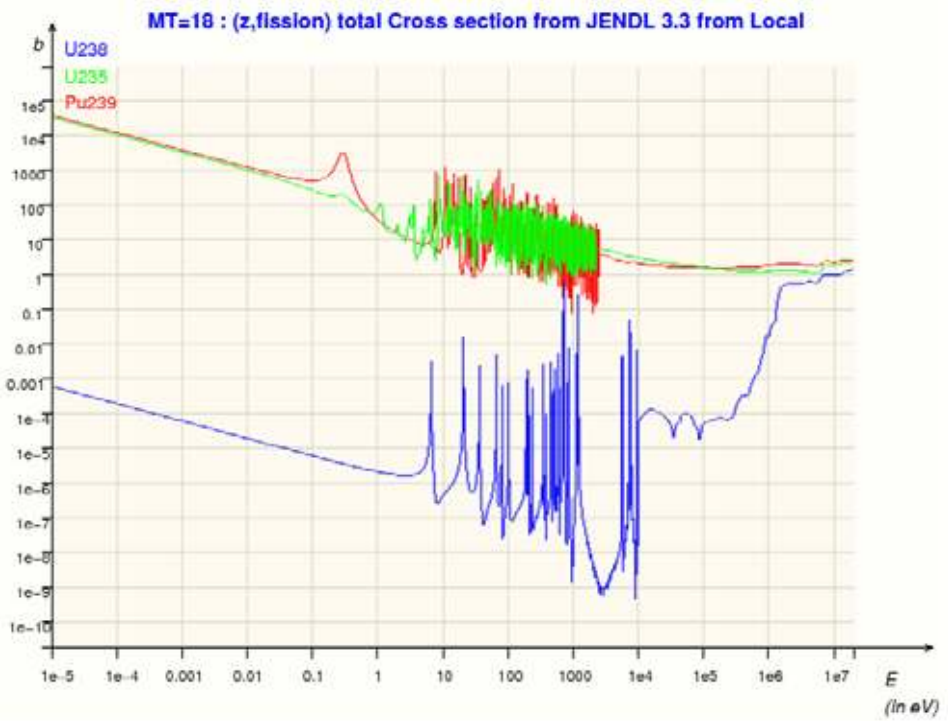


Figure 5: Energy Dependence of Fission Cross section for U-235, U-238 and Pu-239 [8]

From the definition described above, firstly, the neutron cross section depends on the nucleus type. These deviations are related to not only different nuclei properties (proton number and neutron number) but also nuclei states (excited and ground).

The incident neutron energy is another important variables in reactor physics. U-235 could a proper example because of its chief role in thermal reactor engineering. In Figure 5, the neutron energy scale can be divided into three parts from left to right: slow region (with energy less than 1 eV), resonance region and fast region. The slow region gives high fission cross-section to the fissile nuclides like U-235, Pu-239 etc. The cross-section in the first part is proportional to 1/E. That is the reason why the neutrons generated by the fission reaction should be slow down. The resonance (from 1eV to 0.01 MeV) part plays a crucial role in safety consideration because its temperature dependency property affects the possibility of the incident neutrons to be absorbed during slowing down process. Furthermore, it influences the whole neutron economy as well. The fast region is given most attention when we talk about transmutation and breeder reactors, which will be discussed from Chapter 5.

Energy Source	Energy (MeV)	Heat Produced
Fission Fragments	168	168
Fast Neutrons	5	5
Prompt Gammas	7	7
Decay Gammas	7	7
Capture Gammas	--	5
Decay Betas	20	8
Total	207	200

Table 3: Energy Distributions of U-235's Fission Products [6]

During the fission reaction, about 200MeV energy (for the U-235 case) will be released and transferred to the fission products (main heat source of nuclear reaction), fast neutrons (will be useful for the following chain reaction), decay betas (related with the waste handling) and so on. The energy share of these individuals is listed in the Table 3.

3. Components of the New Fuel Cycle Code

3.1 Bateman Equation

Since the new fuel cycle code will be used in simulation of large-scale nuclear system (like Chinese reactor park that consists of nearly 100 different facilities), high processing speed is necessary to be fulfilled. To realize such kind of property, the new code uses a simpler way to estimate the nuclide transmutation rate by solving the following first order differential Equation (5) [9].

$$\frac{dN_i}{dt} = -\sum_{j\neq i}[\lambda_{ji}^{d} + \int \varphi(E,t)\sigma_{ji}^{tr}(E)dE]N_i + \sum_{j\neq i}[\lambda_{ij}^{d} + \int \varphi(E,t)\sigma_{ij}^{tr}(E)dE]N_j \qquad (5)$$

This formula represents the transmutation rate of nuclide i. On the right side the equation, λ^{d} is the decay constant. The ij and ji indices symbolize the direction of the transmutation course (ji means decay from nuclide i to nuclide j and ij is in the opposite). φ (E, t) is the particle flux (mainly neutron) as a function of transmutation energy at time t and cross section σ^{tr} (E). N_j represents the amount of nuclide j.

It can be found out that the first sum-up group shows the consumption of nuclide i through both decay and reaction. While the second group shows the accumulation of nuclide i, so that it shows the change of nuclide i when suming these two parts together.

In the Equation (5), both the particle flux φ and induced nuclide cross-section σ^{tr} will be integrated over energy. This integration needs a full energy-scale neutron transportation simulation that will be time consuming task.

Since the purpose of present fuel cycle code is to simulate a big system, it is assumed that the transmutation cross section for a given kind of reactor type is constant, which is pre-calculated by a transport code (like MCNP). Then, Equation (5) will be rewritten as Equation (6) and effective transmutation cross section can be given by Equation (7). Moreover, Equation (6) is named with

"Bateman Equation" because of its developer — H. Bateman in 1910 [21].

$$\frac{dN_i}{dt} = -\sum_{i \neq j}(\lambda_{ji}^{d} + \overline{\sigma}_{ji}^{tr} * \overline{\varphi}) * N_i + \sum_{j \neq i}(\lambda_{ij}^{d} + \overline{\sigma}_{ij}^{tr} * \overline{\varphi}) * N_j \tag{6}$$

$$\overline{\sigma}_{ji}^{tr} * \overline{\varphi} = \int \varphi(E)\sigma_{ji}^{tr}(E)dE \tag{7}$$

Equation (6) can be transferred into vector form, as shown in Equation (8). After integration process, Equation (9) can be deduced.

$$\frac{d\vec{N}}{dt} = \vec{\Lambda} * \vec{N} \tag{8}$$

$$\vec{N}(t) = \vec{N}(0) * e^{\vec{\Lambda} * t} \tag{9}$$

Besides exponential matrix method, the Bateman equation can be solved by the transmutation trajectory method. The concentration of i th nuclide from 1st nuclide's decay is expressed in Equation (10), which is given by Bateman too.

$$n_i(t) = \lambda_1 \lambda_2 \ldots \lambda_{i-1} n_1(0) \sum_{j=1}^{i} \frac{e^{-\lambda_j t}}{\prod_{\substack{k=1 \\ k \neq j}}^{i} (\lambda_k - \lambda_j)} \tag{10}$$

ORIGEN applies exponential matrix method [24] and trajectory method is applied in some other kinds of numerical code such as BISON, CINDER etc [21]. Comparing to the former one, trajectory method is stated with a group of linear chain equations just as Equation (10), so that the contribution to i nuclide's concentration from any original nuclide can be traced.

In the new fuel cycle code case, there is no necessity for tracing contributor in the decay chain and matrix solving is faster than solving concentration equation for all nuclides one by one. Hence, the new code's composer selects matrix calculus as reference methodology.

3.2 Neutron Flux Spectrum

As described above, integration of flux-weighted transmutation cross sections with energy is necessary in order to solve the Bateman Equation. Such kind of integration process should be based on a spectrum that can be generated from neutron transport code such as MCNP (**M**onte-**C**arlo **N**eutron-**p**article

Transport Code).

Since the new fuel cycle simulation code is used in PWR majored scenario, in order to simplify the present study, only the neutron spectrum of PWR will be used in the transmutation calculation process. This method is reasonable for BWR mainly with three reasons: 1. Both of these two kinds of reactors work in thermal neutron region as normal LWR; 2. The fissile material of these two reactors is all U-235; 3. The chief moderator elements of them are hydrogen and oxygen nuclides, the similar scattering performance of which makes their moderating behavior (such as speed and level) tend to be accordant. However, neutron spectrum of CANDU is more thermal (realized by deuterium's lower absorption cross section compared to hydrogen) than in the LWRs, hence, allowing lower fuel enrichment. As mentioned in Chapter 1, two CANDU 6 units are also under operation in China but the details of its fuel pin design is not available till now so they are simply treated as common PWR. This will not influence the final result very much because the CANDU's limited number compared to the whole system scale and no plan to expand CANDU 6's usage in near term. In conclusion, the fuel pellet for all reactors in China can be simplified to own the same neutron spectrum in the simulation process.

In order to obtain representative one-group transmutation cross-sections, we use the simplified pin-cell model specified in the OECD/NEA benchmark. The OECD/NEA Benchmark geometry is a fairly short but concise one. It divides the fuel assembles into individual parts: one fuel pin and a hexahedral water region surrounding each fuel pin [25]. The parameters (listed in Table 4) of all cells are considered as the same including fuel composition (U-235 proportion chiefly) and moderator composition (Boron concentration chiefly). The boron concentration value is appropriated by averaging of the whole boron concentration curve with 1500 ppm at BOC and 0 ppm at EOC. The boron concentration value used in present work is 500 ppm, which is from OECD/NEA benchmark as well [25]. Such kind of assumption is logical when talking about a PWR core area at normal operation state because of the even neutron flux distribution in core area and no control rod's insertion in this case. Actually, the difference between the neutron spectrum from this simple model and the one from more complex models (considering radial uneven power

distribution, axial uneven neutron distribution, fuel burn-up with time, adjusting neutron flux in reactor by changing boron concentration etc.) is in the reasonable region when checking them with more detailed calculation.

Cell	**Fuel Pellet**			**Cladding**	**Coolant (Moderator)**				**Shell**
Density (g/cm^3)	10.02			6.52	0.71395				7.48
Composition	U-235	U-238	O	Zr-40	H	O	B10	B11	Fe56
Isotope Fraction	0.0905	2.0645	4.3100	1	4.772	2.386	3.93×10^{-4}	1.59×10^{-3}	1
Temperature (K)	900	900	900	600	600	600	600	600	600

Table 4: Simplified Fuel Pin Compositions [25]

While, the upper and down ends of the geometry are reflection surface, which is not that proper because of neglected influence from the large water amount outside these two ends. So that I do some modification to the benchmark geometry including extending the water hexahedral till reactor vessel shell, adding stainless steel at both new ends to present the steel vessel shell and then set surface reflection again. The old and new geometry parameters are compared in Table 5 and sketched out in Figure 7 correspondingly.

OECD/NEA Benchmark Geometry		Evolution Geometry	
Pellet Radius (cm)	0.4095	Pellet Radius (cm)	0.4095
Pin Radius (cm)	0.4750	Pin Radius (cm)	0.4750
Pin Height (cm)	365.8	Pin Height (cm)	365.8
Water Height (cm)	365.8	Water Height (cm)	765.8
Water Length (cm)	1.3133	Water Length (cm)	1.3133
Shell Thickness (cm)	----	Shell Thickness (cm)	20

Table 5: Geometry Parameters for Old and New Design

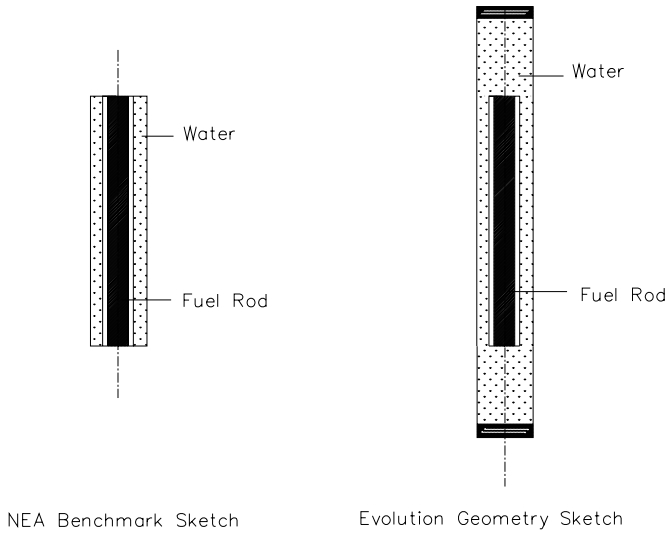


Figure 7: Comparisons between Old and New Fuel Pin Geometry

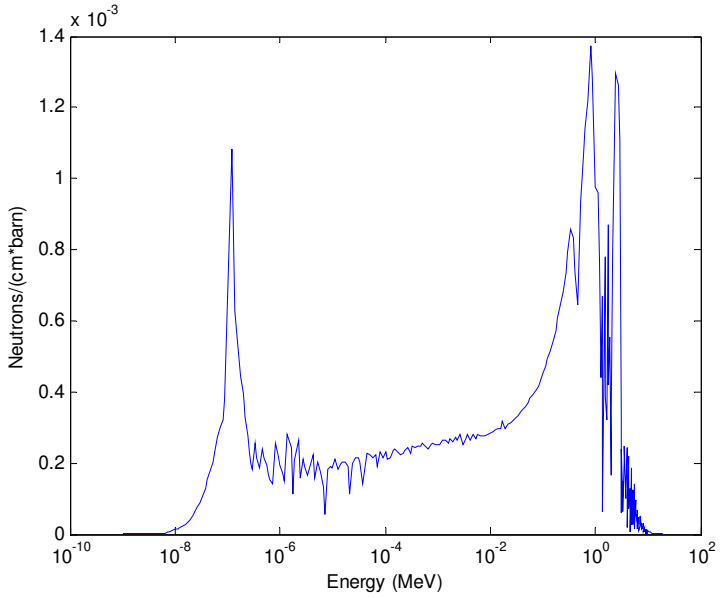


Figure 8: Neutron Spectrums from MCNP for Benchmark PWR Fuel Pins

From Figure 8, we can get that there is a peak value of the neutron flux between the region 1 □ 10^{-8} MeV and 1 □ 10^{-6} MeV. This area matches the slow neutron region, which means that about one fifth of the neutrons in the fuel pin are well moderated and the fissile nuclides can enjoy higher fission cross section. The spectrum-averaged cross sections are listed in Table 6.

Nuclide	σ_f (barn)	Nuclide	σ_f (barn)
92235	4.11×10^{1}	94242	4.42×10^{-1}
92238	9.98×10^{-2}	95241	1.31×10^{0}
93237	5.07×10^{-1}	95242	1.49×10^{2}
94238	2.57×10^{0}	95342	6.49×10^{2}
94239	1.13×10^{2}	95243	4.24×10^{-1}
94240	6.10×10^{-1}	96244	9.89×10^{-1}
94241	1.12×10^{2}	96245	1.10×10^{2}

Table 6: Fission Cross Sections for Pin-cell Model

Oscillating part in the slow region (from 1×10^{-6} MeV to 1×10^{-2} MeV) is caused by capture cross-section resonance of U-238 in the same induced neutron energy region. Furthermore, oxygen's oscillating scattering cross sections bring several dips to neutron density in part of the fast neutron region (from 5×10^{-1}

MeV to 5 MeV), which plays an important role in the neutron's slowing down process.

3.3 Cross Section Tables in ORIGEN Format

The structure of the cross section libraries in the new fuel cycle code has been adapted from that of ORIGEN libraries, which contain data for more nuclides (1343 nuclides) than we may calculate using JEF 3.0 and EAF 99.

The ORIGEN libraries contain data for transmutation and consist of two parts: cross section data part in the first line and fission yield data in the second line. In order to fetch these data from the database, a kind of dir files are generated to record the linkage between calculation demand and the corresponding location of data in database. The dir files for cross section are named of xsdir files. The cross section data part in the xsdir starts with nuclide mass number, then following by (n, γ), (n, 2n), (n, 3n)[1] / (n, alfa), (n, fission)[2] / (n, p), (n, gamma*) and (n, 2n*) cross section data. All of these parameters are arranged in the same order as this. While, if there is no data for some parameters or data for this parameter is still not clear, a zero value is put into that location. Only fission products have their yield data so that only the third part of the ORIGEN library (fission product nuclide part) has the second line. The second line contains yield proportion from given reactions that correspond with the reaction type in the first line. According to the description above, the data table for nuclides in ORIGEN will be like the example below:

	Mass No.	(n, γ)	(n, 2n)	(n, α)	(n, p)	(n, γ*)	(n, 2n*)	
206	360820	7.249E 00	1.195E-04	2.737E-06	2.330E-05	3.804E 00	0.0	1.0
206	9.68E-09	1.83E-06	1.02E-07	4.59E-09	5.01E-06	3.14E-08	3.13E-08	3.13E-08

However, accompany with the new code's testing, some problems come out with the libraries from ORIGEN: 1. Most of the fission product nuclides included in the ORIGEN libraries have only the yield data (the data to present how a given kind of nuclide is generated) but no reaction cross section data; 2.

[1] Only for Actinide nuclides
[2] Only for Actinide nuclides
* Means fission product at excited state

The cross section data from ORIGEN is relatively old compared with JEF 3.0 and EAF 99, which means that data from JEF 3.0 and EAF 99 libraries are more reliable; 3. Most of the nuclides that are included in ORIGEN not in new data libraries (consists of JEF 3.0 and EAF 99) have a relative short half-life time, which limits these nuclides to get reaction with neutrons. So that it is not necessary to check their cross section data then. For example, the (n, 2n) and (n, 3n) cross section data of nuclide Am-242 (ground state) are zero in new data library because the half-life of this kind of nuclide is very short (only 16.01 h), so it is hard to get its parameters. Thus, the data included in ENDF B/IV are kind of assumption data and they are set the same as the ones for excited state.

Besides the nuclide type content difference of ENDF B/IV and new data libraries, in the mutual nuclides part, the cross section data of them are different and some of them even enjoy tremendous ones. As an example for this, I sketch out the Table 7 to show the difference for some actinide nuclides.

	$\sigma_{(n,\gamma)}$ (%)	$\sigma_{(n,2n)}$ (%)	$\sigma_{(n,3n)}$ (%)	$\sigma_{total,f}$ (%)
92235	10.50	27.80	60.90	13.70
92238	1.39	21.80	61.90	0.57
93237	9.31	63.60	23.90	3.51
94238	15.20	85.50	71.90	4.05
94239	11.40	24.20	1.66	6.35
94240	131.00	59.90	72.30	4.22
94241	6.18	1010.00	229.00	5.92
94242	1.77	4.91	104.00	6.09
95241	15.00	42.80	79.60	14.30
95242	19.50	Infinite	Infinite	24.60
95342	52.10	49.90	188.00	28.10
95243	2030.00	86.90	100.00	6.54
96244	24.80	27.20	100.00	11.60
96245	41.00	13.80	100.00	55.60

Table 7 Relative Differences in Fission Cross-Sections between New Data Library and ENDF B/IV Based Values

The data included in Table 7 are calculated out with Equation (10), which can be treated as evaluation of differences between new data library data and ENDF B/IV cross section data.

$$\frac{(\sigma_{XSDIR} - \sigma_{ORIGEN}) \times 100}{\sigma_{XSDIR}} \tag{10}$$

We can observe that differences are about 10 times bigger in minor actinides part than that in major actinides part. The reason is that minor actinides' properties especially their transmutation cross sections have been paid more attention in the latest decades, which gives large demand of these values' precision in new cross section databases.

The other interesting points are two infinite values in Table 7, which means the cross section values of Am-242's (n, 2n) and (n, 3n) reactions are 0 in new data library while non-zero in ENDF B/IV. Actually, these data included in ENDF B/IV are suspicious because they are same with the ones for Am-242m. However, precise cross sections for these reactions are still under so the evaluators of JEF 3.0 put 0 here just to clarify this situation not present their real values.

Gathering all the factors mentioned above, 756 nuclides (307 nuclides from JEF 3.0 and 448 nuclides from EAF 99) are brought into cross section generation step and inserted into new fuel cycle code eventually. The full list of the chosen nuclides is presented in Appendix B.

4. Simulation Code Overview

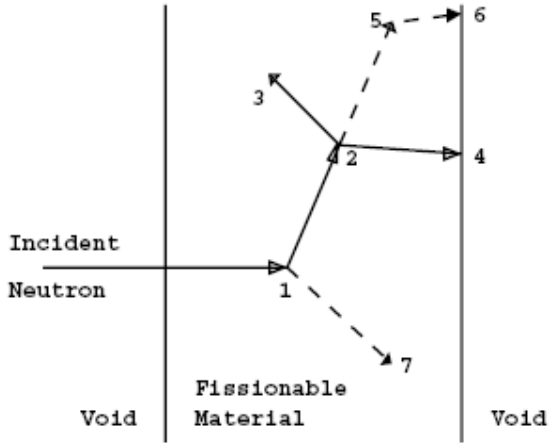
Generally, most of the mathematical problems can be solved in either deterministic way or non-deterministic way. Deterministic result comes from logical deductions of theorems so its precision is high enough to be treated as truth in most cases. While, the neutron transport affair is more difficult to be simulated precisely because all the factors such as emitting energy, emitting direction and nuclide reaction consequence etc. are random in this course so that it is hard to distill some principles to describe this process precisely and incisively. Therefore, Monte Carlo methodology is used into this problem's solution. Monte Carlo methodology is a typical non-deterministic technique because the fundamentals of it are not equations and mathematical deduction but more likely logical assumptions from large amount of statistic investigation. One classic example of this methodology is the dice game, which face of the dice will be upward is uncontrollable just like neutron transportation status in reactor core. While by throwing and checking the result of the dice for enough times, we can conclude that probability of each face is in almost the same value (one sixth), which can be considered as a statistic result of this gambling problem.

In the following parts of this chapter, an overview of codes used for simulation of neutron transport and transmutation will be given including MCNP (neutron transport), MCB (classic burn-up simulation code) and ORIGEN (transmutation simulation tool).

4.1 MCNP

MCNP (**M**onte-**C**arlo **N**-**P**article Transport Code) was composed by Los Alamos National Laboratory (New Mexico, US). The main function of it is to simulate the transport of multi-particles including neutrons, photons and electrons (both individually and together) with continuous energy distribution and obtain cross section data of nuclides included or generated in reactor core [10].

In the MCNP case, neutrons are emitted from defined neutron source with given energy level and direction one by one. Then, their transport progress is concluded by solving the Boltzmann Equation. The possible transport process consists of six interrelated reaction between incident neutrons, induced particles and target nuclides. All of these reactions are sketched out in the Figure 6. By increasing the amount of such kind of random neutron emitting and tracing their motion in the core, just like throwing the dice for enough times, a collection of neutron tracks are obtained, which approaches the physical neutron flux in the limit of large number of neutron histories.



1. Neutron Scattering 2. Neutron Fission 3. Neutron Capture 4. Neutron Leakage 5. Photon Scattering 6. Photon Leakage 7. Photon Capture

Figure 6: Transportation Scenarios of Incident Neutrons

In order to simulate a given transportation scenario, some components must be pre-defined into the input file of MCNP, commonly including five parts: title, geometry card, material card, tally card and source card. Between each part, some interval symbols (like c ---) should be added for the sake of clear structure and reader friendly configuration.

Title line accompany with comments (sentences start with c) after command lines can be considered as the explanation to input file such as card or tally content description.

The object that will be analyzed by the MCNP should be specialized in the

geometry cards firstly. There are two kinds of cards: surface cards and cell cards. A surface card is defined by specifying the surface type (such as P (Plane), S (Sphere) etc.), size and location parameter. Then, all of these surfaces can cut the universe into series of separated spaces called cells and useful cells for specific input file case could be selected out by adding positive and negative signs before serial number of surfaces.

The Material Card is used to define the relative density of nuclides in cells containing the given material. It also specifies which cross section library to be used in the neutron transport simulation. Nuclide type, temperature and class of data will be specified in this card. In the Tally Card, the user can specify what kind of derived data they want to get from the MCNP's simulation such as flux, power, effective cross sections etc. Finally, the Source Card is defined to generate incident neutrons for the transportation process. All of the source factors should be described in this card including: neutron energy spectrum, source shape, and source location.

MCNP contains a series of codes such as MCNP4B, MCNP4C2, and MCNP4C3 etc., which are updated versions of the former MCNP code with time. Moreover, based on the same fundamentals, there are other codes named MCNPX and MCB. MCNPX can do the same simulation work for another 31 more particles' interaction besides neutrons, photons and electrons in larger energy scale comparing with MCNP code (only limited to 0~20MeV for neutrons). So that MCNPX can be used to simulate the transport process of high-energy particles correctly, which is definitely necessary for ADS (Accelerator Driven System) calculations. MCB is also based on Monte Carlo methodology and it can be used to simulate niclides' transmutation and burnup status, i.e. the evolution of nuclide densities with time.

4.2 MCB

MCB (**M**onte **C**arlo **B**urn-up code) is an integrated code based on MCNP4C and can be used in burnup simulation for nuclides' transmutation process.

Adding three cards to the original MCNP input file and solving the Bateman

Equations with these complementary parameters realize the burnup simulation function:

1. BURN Card in the form of BURN n1 n2 … nN. This card lists the materials that will be included into the burn-up simulation work.
2. Power status specification can be realized by application of one card in the following two candidates: PERIOD Card with the form of PERIOD duration (1) unit (1)…duration (n) unit (n). It defines the period of time during which the reactor core undertakes a constant power (given in POWER Card) or neutron source (given in SRCST Card); POWER Card in the form of power (1) power (2)…power (n). It describes the magnitude of constant power in the predefined time period referred in PERIOD card.
3. SRCST Card in the form of strength (1) strength (2)…strength (n). It specifies the constant neutron source value in the given period characterized in PERIOD Card.

The composition change generated from the BURN card will be brought into common MCNP module to process transportation simulation described in Chapter 3.1 and the time-dependent criticality can be obtained then.

For the reason that the MCNP part of the MCB input file works similarly with pure MCNP input file, the tally material card used in simplified PWR fuel pin simulation can be transplanted into MCB simulation input file (including the ones of BWR, ADS, CAPRA and Corail from Daniel Westlén).

4.3 ORIGEN

ORIGEN code is a kind of multi-function computer code that can be used to simulate complex transmutation and decay of large amount of isotopes. Such kind of function can be realized by matrix exponential method.

Although there are several advantages of ORIGEN, the truth is that its libraries are relatively old so some data in them will be less precise or even not clarified (for some short-lived isotopes) compared with the new libraries such as JEF3.0, EAF99 and so on. However, the data format of the libraries can be preserved for its comprehensive and general structure.

5. Advanced Nuclear Facilities

5.1 Reprocessing Plant

Reprocessing is an advanced technique to separate different elements (such as plutonium, uranium and MA) out from the nuclear waste generated from traditional LWR. Until now, only two kinds of methods are adopted into deeper research: hydrochemical reprocessing (also known as wet reprocessing) and pyrochemical (also known as dry reprocessing).

Hydrochemical Reprocessing

The basic fundamental of the hydrochemical reprocessing method is the extraction phenomenon between organic and inorganic solutions. When referring to the uranium and plutonium extraction, the most popular hydrochemical technique is PUREX (**P**lutonium and **U**ranium **R**ecovery by **Ex**traction) that was developed by ORNL (**O**ak **R**idge **N**ational **L**aboratory) in 1949.

PUREX uses the blending of TBP (**T**ributyl **P**hos**p**hate, 30%) and kerosene (70%) as the organic solution. The nuclear waste is dissolved into aqueous nitric acid as the inorganic solution. Then both of these two solutions are mixed together sufficiently by intense agitation to increase reaction efficiency. During this process, the uranium and plutonium ions with valence four in the nitride solution will react with TBP to form a complex compound. For the reason that the TBP can dissolve into both organic and inorganic solutions, it can be treated as a solvent agent between these two kinds of solutions. The uranium and plutonium ions will be transported from nitride solution to kerosene continuously. Then the organic kerosene solution will be extracted out from mixed compound. For the sake of plutonium's including, the kerosene solution should undergo another reaction to change the valence of plutonium from to and it is not complex to separate them away then. The whole process can be shown in following sketch:

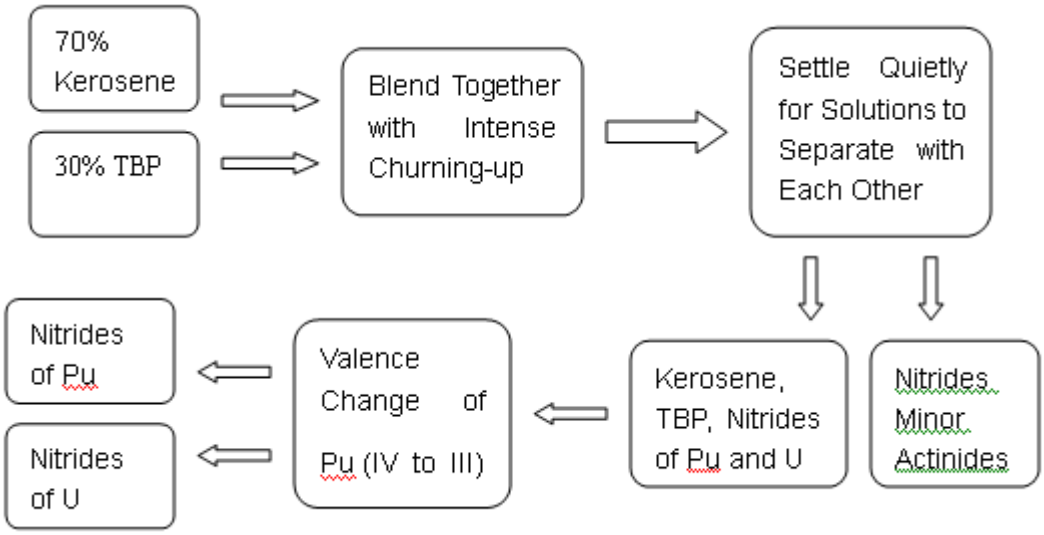


Figure 9 Flow Chart of Reprocessing Technique

PUREX is the most economical and popular reprocessing method until now because of its low materials loss (lower than 0.1%) [14], sufficient operation experience and relatively low operation cost. Hence, there are about 10 units of reprocessing plants running throughout the world, most of which are based on this method. The biggest one named COGEMA project with the capacity of 1700 tons HM/year in 1996 [22] (2000 tons HM/year in 2003 [26]) was commissioned in La Hague of France [22], which takes care of half of the solution load for LWR waste reprocessing work in the whole world.

Till 2006, only one pilot reprocessing facility is under operation in the mainland of China, which is located in Gebi desert of Xinjiang province and applying the PUREX technique. Its capacity is only 100 tons HM/year because the original purpose of its establishment is serving for military industry. It supports high purity fissile material for some experimental FBR units (such as CEFR project in Beijing) and other research application as well.

Pyrochemical Reprocessing

The dry reprocessing (pyrochemical reprocessing) method does not work in the dry environment actually. The main fundamental of this method is electrolysis phenomenon. The nuclear waste solid is put into an anode basket and sunk into molten chlorides or fluorides salt electrolyte (for example

LiCl+KCl or LiF+CaF_2). Then add specific current between the two poles in the system and the metallic uranium will be separated out from the waste to the solid metal cathode. While at the liquid cathode place (made of molten cadmium or bismuth), the other actinides (including the lanthanide series nuclides) will be separated out as well.

Although the course and equipments in pyrochemical processing are briefer than hydrochemical processing, the low efficiency and the solution problem of used salt limit its large-scale usage greatly.

5.2 Fast Breeder Reactor

Before the discussion about Fast Breeder Reactor, we should mention the term "Fast Neutron Reactor" firstly. As stated above in Chapter 2, the reaction situation depends mainly on two factors: binding energy and cross section. From the cross section spectrum of chief fissile nuclides U-235 and Pu-239, we can notice that the fission cross-section value increases with decreasing neutron energy in the thermal region. For this reason, in the thermal reactor (PWR, BWR, RBMK etc.), neutrons from source or fission reaction should be slow down by moderator (such as H_2O, D_2O, Graphite etc.). Although using moderator will deteriorate neutron economy and minimize the energy density of the core, it can be compensated by higher fission probability.

The other way of thinking this phenomenon is that replacing moderation function with higher enrichment fissile material (from 2~4% to 10~20%) will bring chain fission reaction in the core too. This is the main fundamental of fast neutron reactor.

From the Equation (11), we can observe that the fission neutron number from Pu-239 is higher than that from U-235 and will increase with incident neutrons' energy. So that in the fast reactor using PuO_2 as the fissile material, prompt neutrons generated from each fission reaction (normally 2.94 per fission) [15] is higher than in thermal reactor. In these 2.94 prompt neutrons, 1 neutron will be used in the following generation fission reaction, 0.7 will leak out from the core and the other 1.24 neutrons will generate new Pu-239 nuclides by neutron

capturing.

$$\nu[239] = 2.844 + 0.138E$$
$$\nu[235] = 2.349 + 0.150E \quad (11)$$

From above description, we can know that one Pu-239 nucleus' fission could create 1.24 new Pu-239 nuclides ideally, which means that the fissile nuclides inventory will increase in the fast reactor core with the burning process. This kind of phenomena is called breeding and the core that can undergo this process is named with FBR (**F**ast **B**reeder **R**eactor).

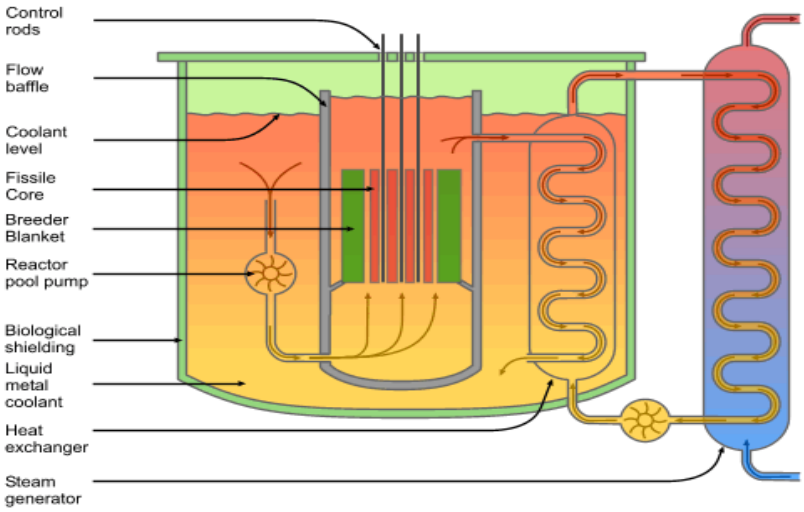


Figure 10: Sketch of Fast Breeder Reactor

FBR consists of five main components: fissile core, breeder blanket, control rod, coolant circulation and heat exchanger. The fissile core is the source of breeding neutrons. For the internal FBR type, the core is mainly made of PuO_2 (10~30%, typically ≥20%) and UO_2 (90~70%). The responsibility of this part is to sustain the chain reaction just like traditional thermal reactor core (except using fast neutrons for Pu-239's fission) and generate excess neutrons to breed Pu-239 from U-238 not only in the core but also in the blanket around. However, for the current sodium cooled FBR design, MA fraction (especially americium isotopes) is limited to 2~3% by safety considerations, which means prototype FBR will be able to burn MA from LWR waste. The americium fraction can be increased to 5% in the Gen-IV design so that FBRs may be used for burning MA generated from LWRs as well.

Technically, FBR can be considered as an ideal facility with high burn-up (up to about 150GWd/T for LMFBR comparing to 50GWd/T for common PWR) and no MA generation problem. For these reasons, in the following part of the report, FRB function will be discussed in two directions: as a power generation facility to meet society's energy requirement and as a waste transmutation facility to ease repository load.

Till now, almost all the large-scale fast breeder reactors are based on LMFBR technology, in which the liquid metal is used as coolant in the core. There are several candidate materials that own required properties (low capturing and scattering cross-section, high thermal capacity) at the beginning such as molten lead, sodium, sodium-potassium eutectic and mercury, while some of them have serious disadvantages like: mercury's tremendous density (heavy burden for circulation pump), high toxicity and high vapor pressure; lead's high density and corrosion influence on steel vessel; sodium-potassium's high price. So majority accepts sodium-cooled scheme eventually in spite of potential safety problems from its severe chemical reaction with water.

5.3 ADS (Accelerator Driven System)

When talking about the nuclear industry, one of the most serious issues is the long-lived nuclear waste. Although there are several FP with relatively long half-life (such as Tc-99 with 2.14×10^4 years and I-129 with 1.60×10^7 years), they are still not a problem in the waste management for some the reasons: 1. The high level waste repository field in China is Beishan (Gansu Province), which enjoys almost the same granite geological structure as Sweden does and the technetium will not have mobile form in this condition; 2.Iodine is a sublimation into atmosphere during reprocessing process and then it will be diluted thoroughly [14]; From the Figure 11, it can be observed that the fission yield proportions of technetium and iodine are only 3% and 1% correspondingly, which are relatively low compared with those of transuranic nuclides.

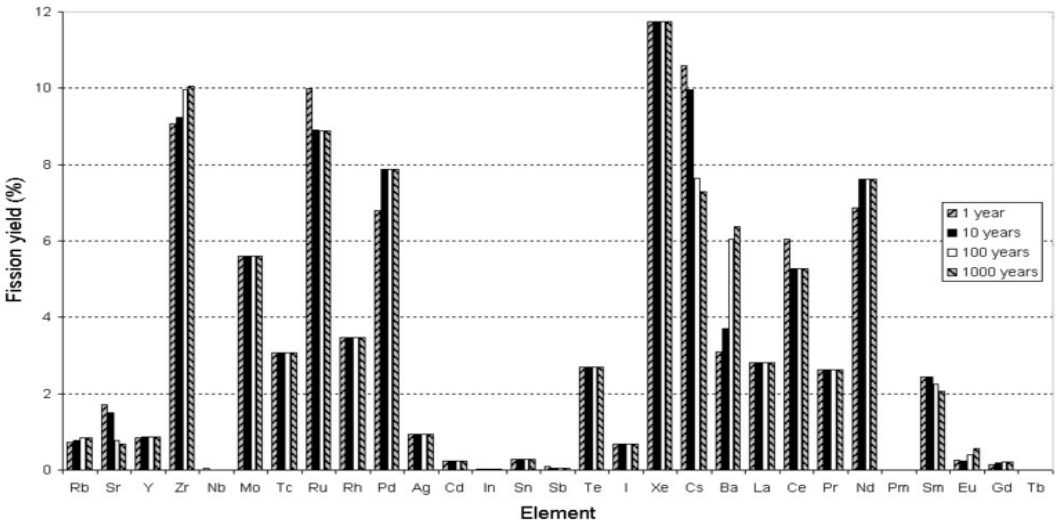


Figure 11: Fission Yield Data

Moreover, radiotoxic index value (Ingestion Does Coefficient in our case) of major transuranic nuclides are about 10 times bigger than those of major fission products as shown in Table 8.

Transuranics	e (Sv*Bq-1)	Fission Products	e (Sv*Bq-1)
Uranium 235	4.70×10^{-8}	Strontium 90	2.80×10^{-8}
Plutonium 238	2.30×10^{-7}	Technetium 99	6.40×10^{-10}
Plutonium 239	2.50×10^{-7}	Iodine 129	1.10×10^{-7}
Plutonium 240	2.50×10^{-7}	Cesium 135	2.00×10^{-9}
Plutonium 241	4.80×10^{-9}	Cesium 137	1.30×10^{-8}
Plutonium 242	2.40×10^{-7}	Samarium 149	——
Americium 241	2.00×10^{-7}		
Americium 243	2.00×10^{-7}		
Curium 244	1.20×10^{-7}		
Curium 245	2.10×10^{-7}		
Californium 251	9.10×10^{-6}		
Californium 252	5.00×10^{-6}		

Table 8: Ingestion Does Coefficient of TRU and FP [18]

For these reasons, what should be cared about in the waste management course are mostly the transuranic nuclides (with atomic number bigger than 92). FBR can help to erase uranium and plutonium inventories but not easily MA for the reasons referred above. ADS is a more efficient facility to transmute these

nuclides because the up-limit of the americium fraction is increased to 50% that means it can burn much more MA than it generates.

There are three main parts in ADS as shown in Figure 12: accelerator, target and core. While the core referred in ADS has some differences with the ones in PWR and FBR. Firstly, ADS is used to transmute the transuranic materials so that it is necessary to replace the traditional fissile fuels (U-235 and U-233) with transuranic isotopes; Secondly, the fission probability of transuranic nuclides with even number of neutrons are bigger in the fast neutron region so that the prompt neutrons should not be moderated a lot, which means that the coolant should be substituted by LBE (Pb-Bi eutectic), lead, sodium or gas. Thirdly, in order to transmute Am-241 in the fast neutron core efficiently, higher fraction of MA should be introduced into the fuel, which will deteriorates the Doppler effect and effective delayed neutron magnitude very much in the core (shown in Table 9). Therefore, the ADS core is set to be sub-criticality to broaden the safety margin into reasonable level again.

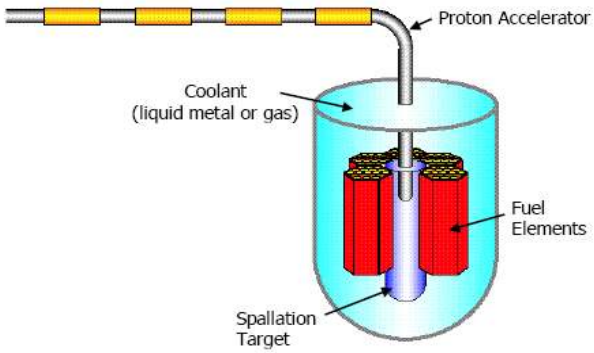


Figure 12: Sketch Map of ADS

U (%)	Pu (%)	Am (%)	Zr (%)	K_D = Tdk/dT (pcm)	β_{eff} (pcm)
80	20	--	--	-810 ± 20	342 ± 10
50	20	30	--	-20 ± 20	204 ± 11
--	20	--	80	-420 ± 20	206 ± 13
--	20	30	50	-20 ± 20	143 ± 11

Table 9: Relation of Am Proportion with
Doppler Feedback and Effective Delayed Neutron Fraction [12]

In order to maintain the power production, an external and hard neutron source should be added. A proton accelerator and heavy metal target take this job. The accelerator is a tunnel after the proton source to accelerate the proton by electromagnetic fields. To generate high-energy neutrons from bombarding the target, the proton energy must be high enough (1000 MeV will gives highest efficiency) [13]. Moreover, 50~70% of proton beam energy will deposit in target as heat depend on the proton energy. For 600 MeV and 2.33 mA beam in XT-ADS design, 1.4 MW heat will be generated from the target. In this tough circumstance, the properties of both the beam window and target will deteriorate intensely. Although, windowless design and target cooling system have been created, more work still should be fulfilled in this area.

The development situation of Chinese ADS system is much slower than that of EU. It consists of several parts [19]:

1. Conceptual study from 1994 to 1999 sponsored by MOST (China **M**inistry **o**f **S**cience and **T**echnology). During this process, more attention is paid on theoretical design of HPPA (**H**igh **P**ower **P**roton **A**ccelerator).
2. Set up verification facility with 150MeV/3mA LINAC installed in and 3.5 MW thermal power modified swimming pool light water reactor design.
3. MEND code is applied in theoretical model's analysis. Simulation work for Pb target (60 cm length and 20 cm diameter) with 800, 1000, 1500,

2000 MeV incident protons are accomplished.

4. Mechanical properties' test has been done to the candidate beam window material (9Cr2WVTi steel).
5. Damage on 316L stainless steel in given temperature region (from 25 ºC to 802 ºC) and given irradiation does (from 21 dpa to 100 dpa) has been investigated.

6. Scenario Composition, Results and Analysis

6.1 Scenario Creation

Without Advanced Facility

The nuclear technology history of China is really dramatic because we have a relatively long nuclear energy usage history (mainly for military area, which can be dated back to the first nuclear weapon's blast in 1964) but quite short civil application time compared with Sweden (Oskarshamn Unit 1 was set up in 1966). Actually, real research on the nuclear power plant technique was not started until 1984. Then, the lack of fund, low coal price and limited international academic communication restricted the development for about 10 years. Until Dec.1st of 1991, there was no commercial reactor commissioned in the mainland of China. That is the reason why 1990 is selected to be the start point of the scenario. The first nuclear power plant set up in China is named Qinshan (Phase 1) located in Zhejiang Province with 310MW power capacity. While this unit is more like a proof of Chinese research ability not a real epoch of Chinese civil nuclear industry because no follow-up projects (Daya Bay project mainly supplies electricity to Hong Kong for some political reason) were put forward in the next ten years.

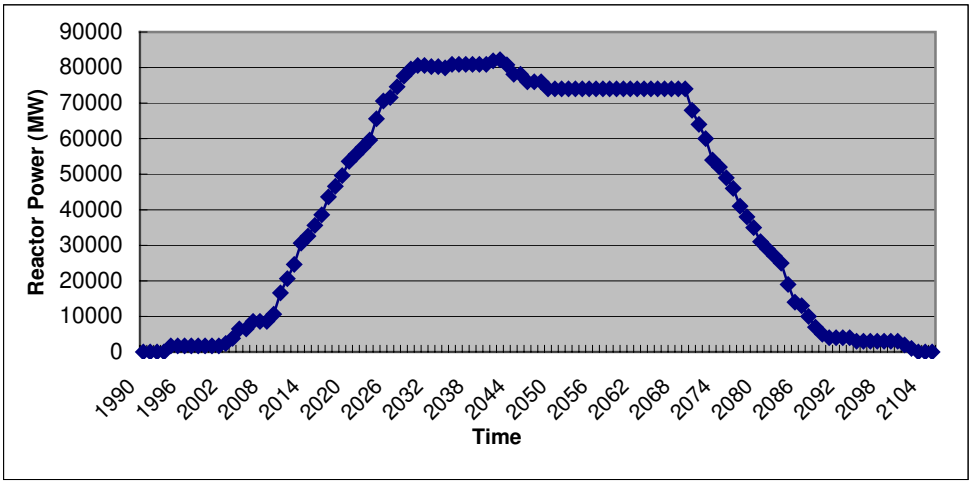


Figure 13: Chinese Nuclear Park Development Tendency

From Figure 13, it can be noticed that the spring of Chinese civil nuclear industry comes in 2000 because of fossil fuel price's leaping and large-scale energy shortage in the south part of China (mentioned in Chapter 1). From 2002 to 2006, 8 reactors are constructed and thrown into operation, in which three main reactor types (PWR, VVER and CANDU 6) are included. This period is considered as investigation stage of different reactor type for further expansion. Therefore, in the first scenario, no more reactors are built after 2006 and this time period can be regarded as the present situation.

The near-term plan is actually set by Chinese government and it ends in 2020. Moreover, information of the power plants (including scheduled start time, location, total unit number, reactor type, power capacity, etc.) is also stated in the government layout documents. While, in the documents, there are only start dates of the first phase for each project so proper assumptions are necessary in this case. Using the information from the first stage, 3 main assumptions are generated: 1.The time gap between phases in each project is 6 years, which is the average value of the time gaps from Qinshan Phase 1 to Phase 2 (10 years) and Phase 2 to Phase 3 (2 years); 2.The time gap between units in each phase is 1 year; 3.The life-time of reactors is 40 years (CPR 650 criteria) before 2010 and 60 years (CPR 1000 criteria) after 2010; 4. Reactors in the same phase are of the same type.

Besides time-scale confirmation, reactor core refueling affairs should also be brought into consideration. In the new fuel cycle code, three parameters are taken into account: fuel batch number, fuel recycling time and total fuel mass of the core.

Fuel batch number shows the times of refueling to replacing whole fuel in the core. In other words, inverse of this number is the fuel proportion that will be replaced with fresh ones in each refueling process. The value used in Chinese scenario is 4, which is the average value of Swedish PWRs.

The time gap between two refueling process is called fuel recycling time. This parameter is defined by several factors including: fuel assembles design,

uranium enrichment, power level and also reactor type. The last three factors could not be modified because they affect total reactor economy deeply. So that the main fuel element suppliers around the world (such as Framatome ANP, Westinghouse etc.) spend lots of their efforts on the evolution of fuel pin arrangement, material selection and frame construction to increase the burnup ratio of elements. For example, AFA-2G from Framatome has burn-up value of 30GW*day/tonU corresponding to the fuel recycling period of 360 days, while, AFA-3G has 50GW*day/tonU burn-up value corresponding to 540-day recycling time. From 1998, Framatome started the project to transfer AFA-3G manufacturing technology to Chinese Yibin Fuel Assembly Factory and the first reload elements to Daya Bay was produced by Yibin in 2001. Then, with the manufacturing ability's expanding, Yibin can support all the reload demand of Qinshan, Daya Bay and Lingao nuclear power plants till 2006. Moreover, Chinese new version CPR1000 with advanced fuel assembly design leading to high burnup ratio (up to 70 GW*day/tonU) will join into large-scale commercial usage from 2010 hopefully.

The total mass of the cores can be obtained by using a simple estimation with Equation 12.

$$FuelMass = \frac{T \times n \times P}{P0 \times \eta} \tag{12}$$

In this equation, T is Fuel Recycling Time, n is Fuel Batch Number, P is the Electricity Power of the Core, P_0 is Benchmark Burn-up Rate (obtained from the technical brochure of AREVA ANP) and η is the Electricity Efficiency of the Reactor Unit.

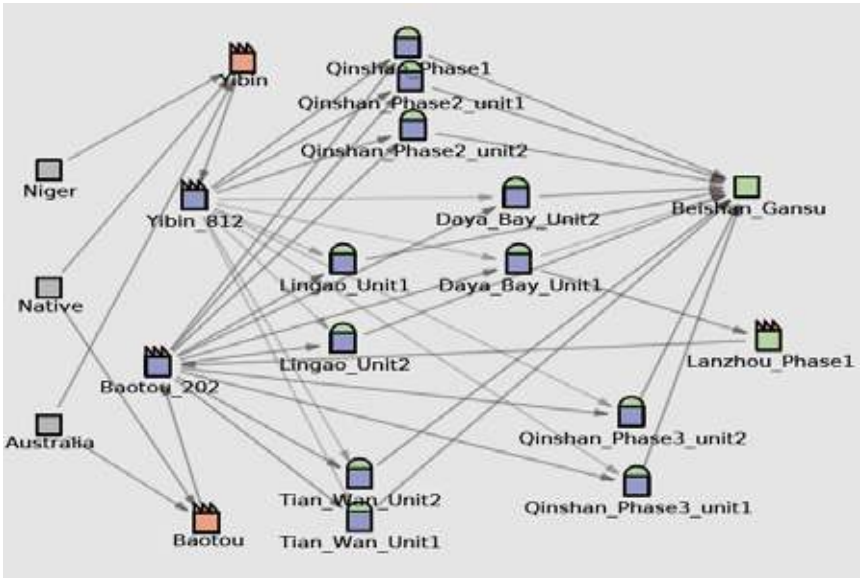


Figure 14: Present Scenario of Chinese Reactor Park

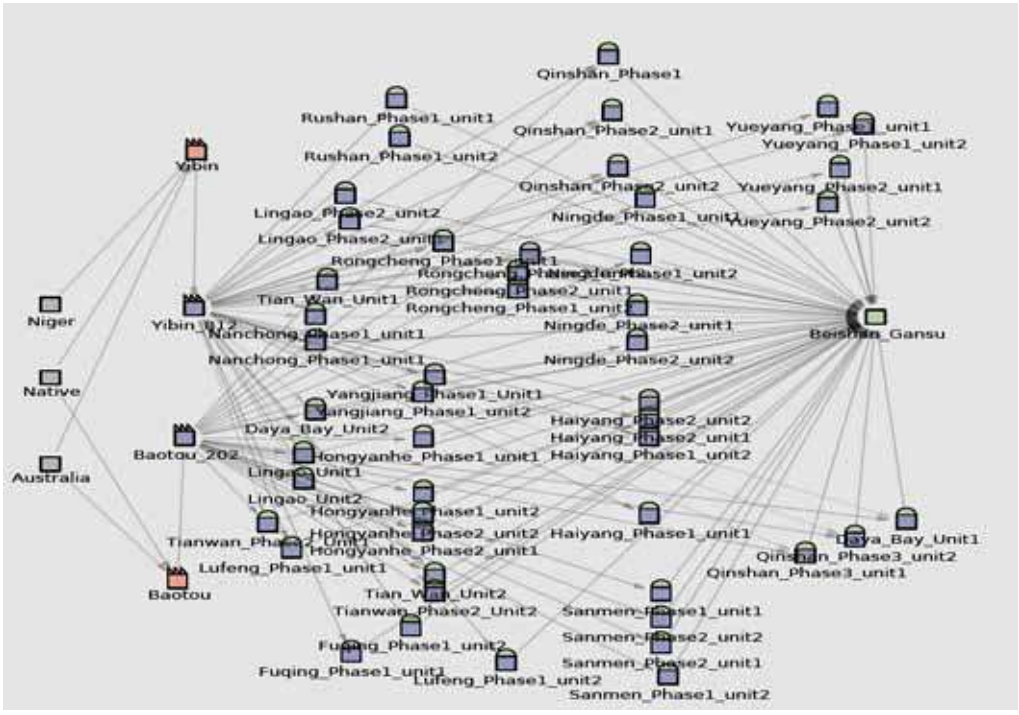


Figure 15 Near-term Scenario of Chinese Reactor Park

With these criteria and information, the short-term scenario input file for the new code can be created. The graphical representation of this scenario is shown in Figure 14.

According to the time plan of Chinese FBR project, the first prototype FBR will be commissioned by 2030 hopefully and we also do not want to use the

valuable plutonium into MOX fuel assembly's fabrication so that the reprocessing facility is still not included into the short-term scenario (till 2020) although there is one pilot reprocessing plant already existing in Lanzhou, Gansu Province since 1997 with capability 100 tons HM/year. Eventually, we can sketch out the near-term scenario without neither FBRs nor reprocessing plants as shown in Figure 15.

Including Fast Breeder Reactor

As mentioned in Chapter 5.3, FBR enjoys profits in two different directions (both energy generation and waste solution) so that when thinking about FBR development plan, we should divide it into three routes: energy demand balancing based on repository Pu-fueled FBR, energy demand balancing based on breeding Pu-fueled FBR and repository Pu-fueled fast reactors with breeding ratio equal to 1.

For the first route, we should focus on electricity magnitude obtained from FBR. The long-term FBR plan from Chinese government is [16]:

1. The first experimental unit CEFR (**C**hinese **E**xperimental **F**ast **R**eactor) gets criticality at the end of 2007 with thermal power of 65MW and electricity power of 20MW [19] ;
2. Setting up mid-scale prototype unit by 2030 with electricity power of 300MW;
3. Accomplish the construction of commercial unit by 2040 with electricity power of 1000~1500MW;
4. Realize the fast reactor's popularization in China by 2050 with 1500MW and 40 years lifetime electricity power to replace existing LWR park step by step.

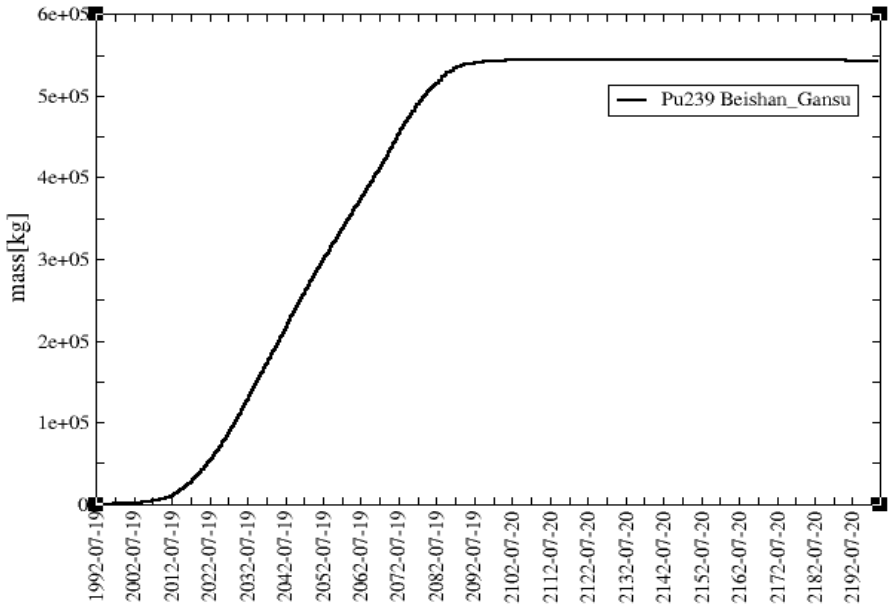


Figure 16: Pu-239 Inventory in Waste Repository

Besides the FBR's progress itself, some other factors are also need to be fulfilled and the most important two among them are reprocessing technology status and plutonium repository amount. In 1997, one pilot reprocessing plant (Lanzhou Phase 1, Gansu Province) with the capacity of 100 tons HM/year was accomplished by Chinese government. The main mission of this plant is to support pure isotopes for experimental reactors and limited MOX fuel fabrication. Its production capacity (about 1 ton plutonium per year) is only enough to support a mid-scale FBR so that a new ambitious plan (Lanzhou Phase 2, Gansu Province) is put out with capacity of 550 tons HM per year and it will be thrown into operation by 2020 hopefully to support large-scale spread of FBR technology in the future. Then, about 5.5 tons plutonium will be separated out for the FBR or MOX usage each year. For the plutonium source inventory, it is not possible for Chinese government to buy from abroad or obtain from nuclear weapon decommission so the plutonium source we are talking about here is mainly fetched from PWR waste repository. The Pu-239 inventory in nuclear repository can be visualized in Figure 16.

P_{el} (MW)	280	Fuel Pellet Density (%TD)	85
P_{th} (MW)	714	Fuel Pellet Diameter (mm)	5.4
Core Height (m)	0.93	No.of Pins per Assembly	169
Peak Linear Power (kW/m)	36	Average Fuel Burnup (MWd/kg)	80
Average Linear Power (kW/m)	20	No. of Assemblies	198

Table 11: Design Parameters of MONJU Prototype FBR [20]

The plutonium proportion in the PuO_2-UO_2 fuel of FBR is from 10~30% and typically 20%. Moreover, from Table 11, we can calculate out the core mass of FBR with 280MW P_{el} is 8.20 tons for MONJU prototype design and only 1.97 tons fissile plutonium is enough to start up a 300MW FBR in 2030, which can be realized from both reprocessing capability and raw source amount points of view. This FBR can compensate the power loss from Qinshan Unit 1's phasing out. From 2040 on, we can accomplish the construction of commercial FBR with 1500MW electricity power and 50 tons core mass, which means that more than 10 tons of fissile plutonium is needed for each unit's startup. From Figure 15, we can observe that we can obtain 200 tons (till 2040), 400 tons (till 2060) and 550 tons (till 2080) plutonium from PWR repository by reprocessing process. Therefore, we can obtain enough fissile plutonium inventory from LWR waste repository to start up about 55 FBR units (with 1500 MW electricity power capacity) that equals to at least 82500 MW electricity generation ability, which is definitely high enough to compensate the LWR units' phasing out.

Reprocessing capability in Chinese reactor park will be 650 tons per year (100 tons from Lanzhou Phase1 and the other 550 tons from Lanzhou Phase 2), which equals to about 6.5 tons of plutonium generation. Comparing with the demand (10 tons per year), a small gap (3.5 tons per year) exists between generation and requirement. So that another reprocessing facility with 350 tons

HM per year capacity should be commissioned before 2074 from when large-scale of LWRs' phasing out will start.

In conclusion, a power-sustained scenario (the first route) can be realized based on former assumptions. The transition from a LWR park to a FBR park is shown in Figure 17.

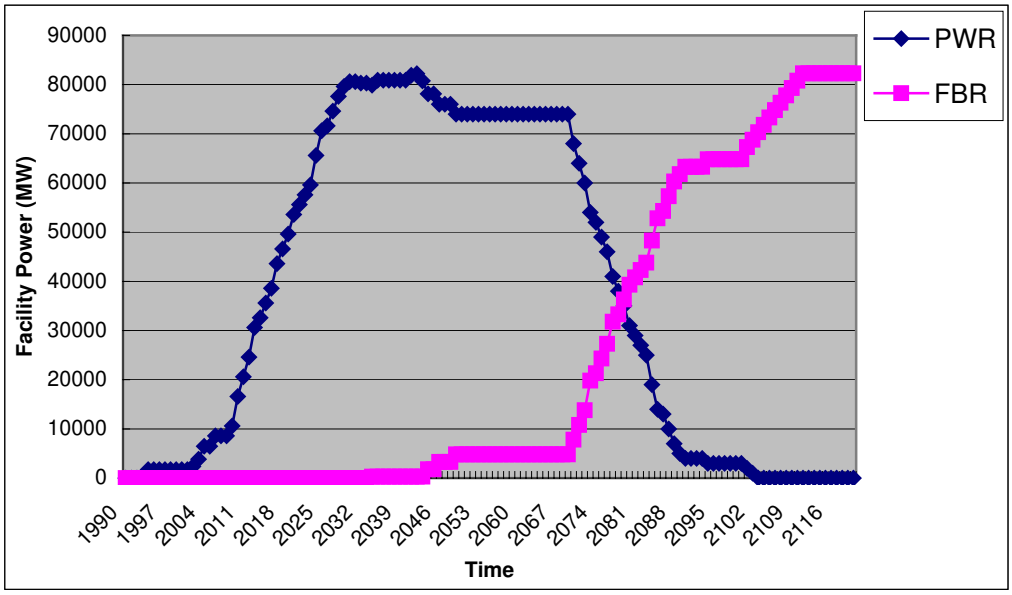


Figure 17: Nuclear Park Transition from PWR to FBR

Although enough fissile plutonium inventory can be fetched from waste repository, increasing the fraction of fast spectrum facilities is still preferable because the fission probability of high radiation emitter (such as Cm-244, Cm-245 and Cf-252 etc.) is much higher in fast neutron spectrum than in thermal neutron spectrum that means high radiation nuclide inventory will be smaller during fuel fabrication. Then the Pu self-sustained scenario is valuable to be estimated in this case.

Before going into deeper discussion, RDT (**R**eactor **D**oubling **T**ime) should be explained first. In short, RDT is a time scale that the magnitude of fissile material in FBR increasing to twice of its origin magnitude. This time can be obtained from Equation (13) [20]. In this equation M_0 (set to be 6.4 tons, 64% of 10 tons total plutonium inventory) is the initial fissile inventory mass; G (set to be 0.30) is breeding gain, which is the ratio of fissile material gained per

cycle and fissile material destroyed per cycle; P (set to be 3825 MW, efficiency obtained from MONJU, Japan) is the thermal power of core; f (set to be 0.80) is the fraction of time at rated power; α (set to be 0.2, for Pu-239 at fast neutron spectrum) is the capture-to-fission ratio. Then, we can estimate the RDT for Chinese scenario is approximate 15 years. Following with 5 years storage cooling time, the whole cycle will take 20 years.

$$RDT \cong \frac{2.7 \times M_0(kg)}{G \times P(MW) \times f \times (1+\alpha)} \tag{13}$$

After bring these two factors into scenario, we can get the conclusion that: 1. Total plutonium amount needed for the startup of FBRs is 452 tons till 2102; 2. Plutonium generated from breeding process is 386 tons till 2102; 3. The distribution of the breeding plutonium amount is uneven, which means that the plutonium magnitude is not enough till 2102 but about 200 tons of extra plutonium will be generated from 2102 to 2112. If plutonium is a normal commercial stuff, purchasing plutonium or waste nuclear fuel from abroad at the deficient time and selling out the extra plutonium at the sufficient time can solve this problem easier. The budget will be balanced eventually at some extents. Since plutonium is related with politics and military affairs, it is unlikely possible to make the free trade real.

For the second route, we should pay more attention on plutonium inventory that can be acquired at different time period because what we want to obtain from this route is a Pu self-sufficient scenario. Firstly, we need to define when we can put first FBR into operation. In order to breed more plutonium for the future FBR park usage, it is certain that we should introduce FBR as soon as possible at the given breeding ratio. As mentioned above, a prototype FBR will be set up by 2030 with electricity power of 300MW and breeding ratio of 1.25, which means about two tons fissile plutonium should be added to reach the criticality and also means two tons more fissile plutonium will be generated from this core after 20 years. Till 2040, we can fetch 198 tons fissile plutonium (2 tons of repository fissile plutonium has been used in the first FBR) from PWR repository and 1 tons breeding plutonium from the first FBR. It is certainly enough for a 1200MW FBR core. Therefore, 1500MW electricity power will be generated by FBR to compensate old PWR phasing out till 2040.

1500MW is a perfect capacity, because there will be three reactors (Qinshan Unit1, Daya Bay Unit1 and 2) whose total power is just 1500 MW. This means that if we can realize FBR project according to the existing time plan, the constant electricity level of the whole reactor park can be obtained quite well.

According to this principle, we can get the situation of the Chinese reactor park's evolution that is listed in the Table 12.

Year	Note	El Power (MW)	Pu Requirement (ton)
2020~2030	Prototype FBR	300	2.0
2030~2040	1 Commercial Pilot Unit	1200	8.0
2040~2050	1 Commercial Unit	1500	10.0
2050~2060	1 Commercial Unit	1500	10.0
2060~2070	3 Commercial Units	4500-300*	30.0
2070~2080	3 Commercial Units	4500-1200	30.0
2080~2090	4 Commercial Units	6000-1500	40.0
2090~2100	5 Commercial Units	7500-1500	50.0
2100~2110	7 Commercial Units	10500-4500	70.0
2110~2120	9 Commercial Units	13500-4500	90.0
2120~2130	13 Commercial Units	19500-6000	130.0
2130~2140	17 Commercial Units	25500-7500	170.0
2140~2150	23 Commercial Units	34500-10500	230.0
Total	62 Commercial Units	93000	870.0

* Minus power comes from former unit's phasing out

Table 12: Plutonium Self-sustained Situation from 2030 to 2150

From the data listed in Table 12, we can conclude that it is impossible to compensate the electricity loss of the PWRs' phasing out with FBRs that are based on the FBR breeding process till 2100. Actually, the inventory gap is so huge that the FBRs can only supply less than one third of the total consumption (22500 MW out of 82226 MW) even the breeding ratio we have assumed (1.30) is much higher than present level (1.0~1.1). While we can notice an obvious tendency that the breeding plutonium inventory is increasing

intensely with the increasing of FBR units number, which means that the replacing task can be accomplished if extending time scale till 2150.

The non-breeding route is the simplest one and also the most feasible plan that is published by CIAE (**C**hina **I**nstitute of **A**tomic **E**nergy). The non-breeding we mentioned means the FBR with breeding ratio of 1.0. In this case, the plutonium reprocessed from LWR waste fuel can be added into FBR core and it will breeds itself in the core without external plutonium replenishment anymore ideally. The amount of Pu-239 that should be introduced in for the startup is lower than the ones in high breeding ratio FBR (10 tons). We can assume that the Pu-239 requirement is 8 tons for BR equals to 1.0 and 1500 MW electricity power fast reactor. Till 2030, there will be 120 tons Pu-239 in waste repository as plotted in Figure 15 and it is definitely enough for a prototype FBR with 300MW power listed in time plan. The pilot commercial (1200 MW) FBR's fissile plutonium demand can be fulfilled as well till 2040 (200 tons Pu-239 will be in storage). The third one can be set up by 2050 with 1500 MW electricity capacity. Till then, approximate 300 tons of fissile plutonium can be fetched from LWR park's repository. From 2050 on, both technology and fissile plutonium inventory are ready for large scale expansion of FBR. Necessary assumptions should be made to this process: 1. Storage cooling time for waste fuel is 15 years so only 150 tons of fissile plutonium (repository amount till 2035) can be reprocessed by Lanzhou Phase 1 and 2; 2. Construction time of FBR unit is 5 years and time gap between each unit is 1 year; 3. All the FBR units are with the same electricity power level of 1500 MW. 4. Fissile plutonium requirement of one unit is 8 tons; 5. FBR unit's lifetime is 40 years; 6. Fuel inventory of decommissioned units can be used in new unit's starting. Then the full list of the FBRs' commissioning status is stated in the Table 13.

Year	Note	El Power (MW)	Pu Requirement (ton)
2030	Prototype FBR	300	1.5
2040	1 Commercial Pilot Unit	1200	6.0
2050	1 Commercial Unit	1500	8.0
2060	5 Commercial Unit	7500	40.0
2070	10 Commercial Unit	15000-300 *	80.0-1.5
2080	10 Commercial Unit	15000-1200 *	80.0-6.0
2090	10 Commercial Unit	15000-1500 *	80.0-8.0
2100	10 Commercial Unit	15000-7500 *	80.0-40.0
Total	40 Commercial Units	60000	320.0

* Minus power comes from former unit's phasing out

Table 13: Pu Self-sustained Situation (Breeding Ratio 1.0)
from 2030 to 2100

It can be observed that the FBR can only supply 60000MW electricity generation till 2100 less than the energy market's requirement on nuclear park.

There is another project proposed by CIAE to realize FBR's transmutation role more quickly and economically. In this special plan, the 300MW prototype FBR will be accomplished by 2025 and 4 (or 6 depending on the funding situation and the public satisfaction level of prototype FBR's operation) MFBR (**M**odular **F**ast **B**urner **R**eactor) with electricity power of 300 MW directly based on the prototype one will be setting up in the following 5 to 10 years [17]. The main role of the MFBR is not generating electricity for the energy market but transmuting MA nuclides stored in waste repository especially the long-lived ones. So the electricity power level is not set to be very high for the reason that high capacity brings safety problem and will postpone the commissioning date of FBR transmutation task. Although the preparation time of changing prototype FBR into several pilot MFBR is limited to only 10 years, it is still feasible by some extend because only few modification should be introduced for this transformation. Moreover, all of these so-called MFBRs are in the same module, which means that the construction efficiency will be higher for relatively large-scale development (such as continuously building up

4 or 6 units' case). Following above proposal, we can assume some details of the project: 1.Construction time for a single MFBR unit is 5 years; 2.The time gap between the starting dates of two MFBR units (except for the first one) is 1 year; 3.The starting time of the first unit is 2015; 4. Lifetime of MFBR is 40 years. So that the commissioning dates of MFBR units are 2025, 2031, 2032, 2033, 2034, and 2035. Eventually, there will be 6 units thrown into operation till the end of 2035.

6.2 Results and Analysis

Because the present and short-term scenarios are the subclasses of the long-term scenario, it is not necessary to analyze the results from the first two cases individually. With the advanced facilities' insertion, not only the scenario structure itself but also the transportation process of both transuranic nuclides and fission products will be influenced intensely.

In order to make the analysis more clearly, major nuclides are separated into three groups (shown in Table 15) according to their magnitude and properties.

Primary Actinide	U-234, U-235, U-236 Pu-239, Pu-240 Pu-241, Pu-242	Minor Acitinide	Np-237, Np-239, Am-241, Am-243 Cm-242, Cm-243, Cm-244 Cm245
Long-lived FP	Sr-90, Tc-99, I-129 Cs-135, Cs-137		

Table 15: Nuclides Groups for Plotting

Primary Actinides

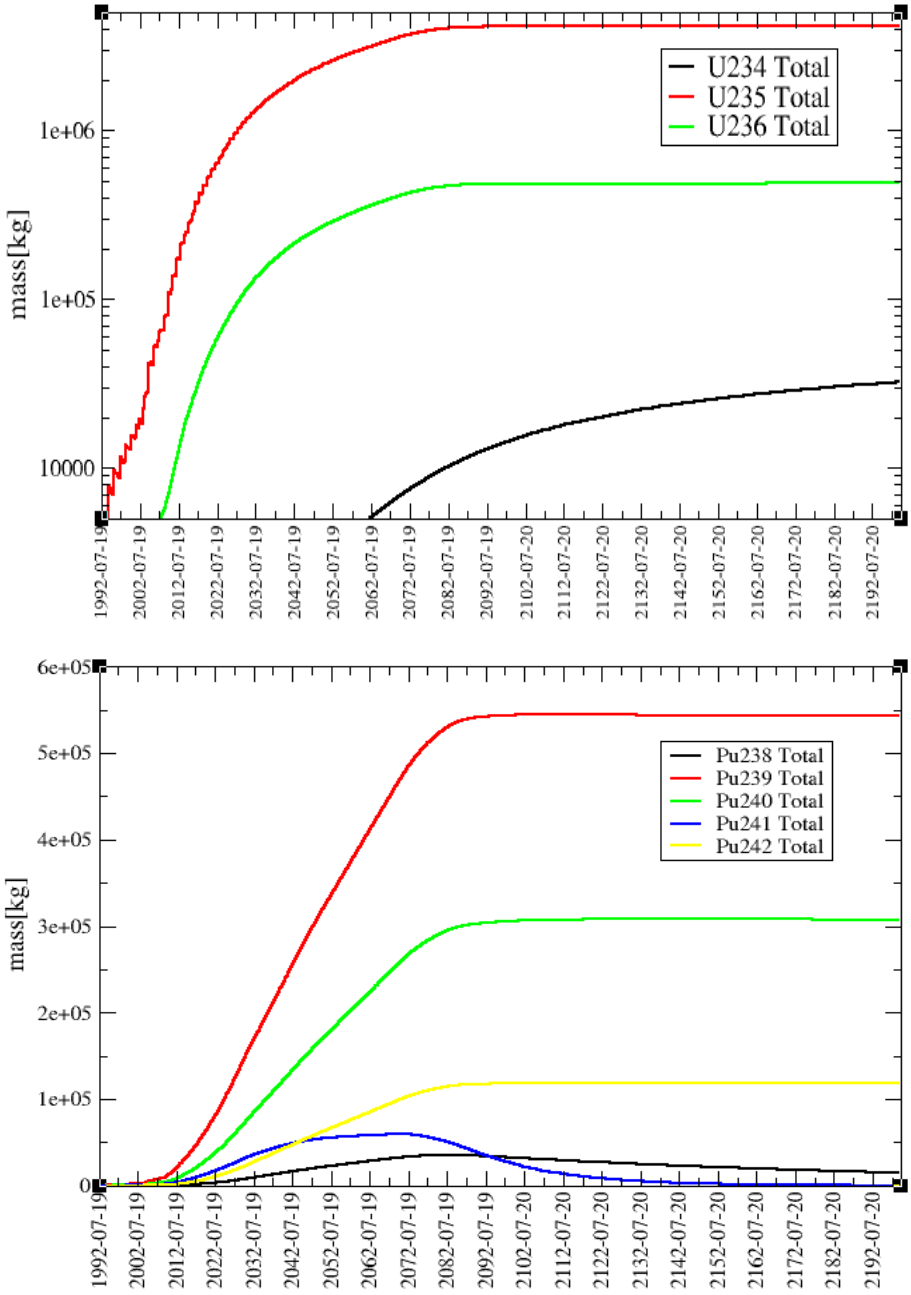


Figure 19: Primary Actinides in Waste Repository

We can observe that the main part of the actinides in the waste repository (except U-238) is U-235 because the enrichment of U-235 in the PWR fuel is commonly 3~5% (default value is 3% in the new fuel cycle code), while there is still about 0.83% U-235 left in the spend fuel. The reason is that the macro

fission cross-section of U-235 is linear with nuclide concentration at given neutron flux. In the reactor core, the neutrons are not only consumed in the fission process but also been absorbed by non-fissile nuclides (like U-238 especially at the resonance absorption region, hydrogen and so on) or depleted through leakage etc. So the fission neutrons will not be enough to balance the demand of depletion and the following chain reactions. In these case, the role of given fuel assembles will become neutron consumer and cannot give out abundant energy to keep the core's power level. So that it becomes necessary to replace these assembles with fresh ones.

As mentioned above, U-238 will be transmuted into transuranic isotopes by continuous absorption of neutrons (especially in the resonance region). Such series of reactions can be presented in Equation 14.

$$
\begin{aligned}
&{}^{238}_{92}U + {}^{1}_{0}n \rightarrow {}^{239}_{92}U + \gamma \xrightarrow{2..35d} {}^{239}_{94}Pu + \beta^{-} \\
&{}^{239}_{94}Pu + {}^{1}_{0}n \rightarrow {}^{240}_{94}Pu + \gamma \\
&{}^{240}_{94}Pu + {}^{1}_{0}n \rightarrow {}^{241}_{94}Pu + \gamma \xrightarrow{14.35\,y} {}^{241}_{95}Am + \beta^{-} \\
&{}^{242}_{96}Cm \xrightarrow{162.8d} {}^{238}_{94}Pu + {}^{4}_{2}\alpha \\
&{}^{237}_{93}Np + {}^{1}_{0}n \longrightarrow {}^{238}_{93}Np + \gamma \xrightarrow{2.117d} {}^{238}_{94}Pu + \beta^{-} \\
&{}^{238}_{94}Pu \xrightarrow{87.7\,y} {}^{234}_{92}U + {}^{4}_{2}\alpha \\
&{}^{241}_{94}Pu + {}^{1}_{0}n \longrightarrow {}^{242}_{94}Pu + \gamma
\end{aligned}
\qquad (14)
$$

The daughter nuclides described in the decay chain should be cared about not only for their big inventory and long half-life time (except the Pu-241 with 14.35 years) but also for other problematic properties.

The U-236 curve can be divided into two parts: one part corresponds to U-235's neutron capturing; the other part comes from the α-decay of Pu-240. Part of U-236 inventory undergoes the fission process and it is a weak α-emitter with half-life of 2.342×10^{7} years as well.

Pu-239 and Pu-241 are chief nuclides in the reprocessing technology because they are fissile materials that are the main fissile nuclides in the MOX fuel and FBR core. Pu-239 is a weak α-emitter with the half-life time of 2.411×10^{4} years, which will be generated by neutron capturing of U-238. Because the concentration of U-238 is about 20 times higher than U-235, together with the

absorption cross-section of 0.961 and 9.41 correspondingly, the amount of induced Pu-239 should be 2 times higher than U-236, which consistent with Figure 19. On the other hand, the U-236 will also come from α-decay of Pu-240 with half-life time of 6563 years. This reaction corresponds to the opposite-directioned steps between the curves of Pu-240 and U-236.

Same thing happens between the second part of the curves for Pu-238 and U-234 shown in Figure 20 for the reason of alfa decay with the half-life time of 87.7y.

Another important property of Pu-241 is its strong beta emitting ability with short half-life time of 14.35y that makes the decay curve obvious. The product of Pu-241's beta decay reaction is Am-241 that is an alfa emitter and source of (α, n) reactions, generating difficulties for reprocessing and danger for fuel fabrication.

Pu-240 and Pu-242 are considered to be neutron absorbers so that their inventories can affect the general neutron economy.

Minor Actinides

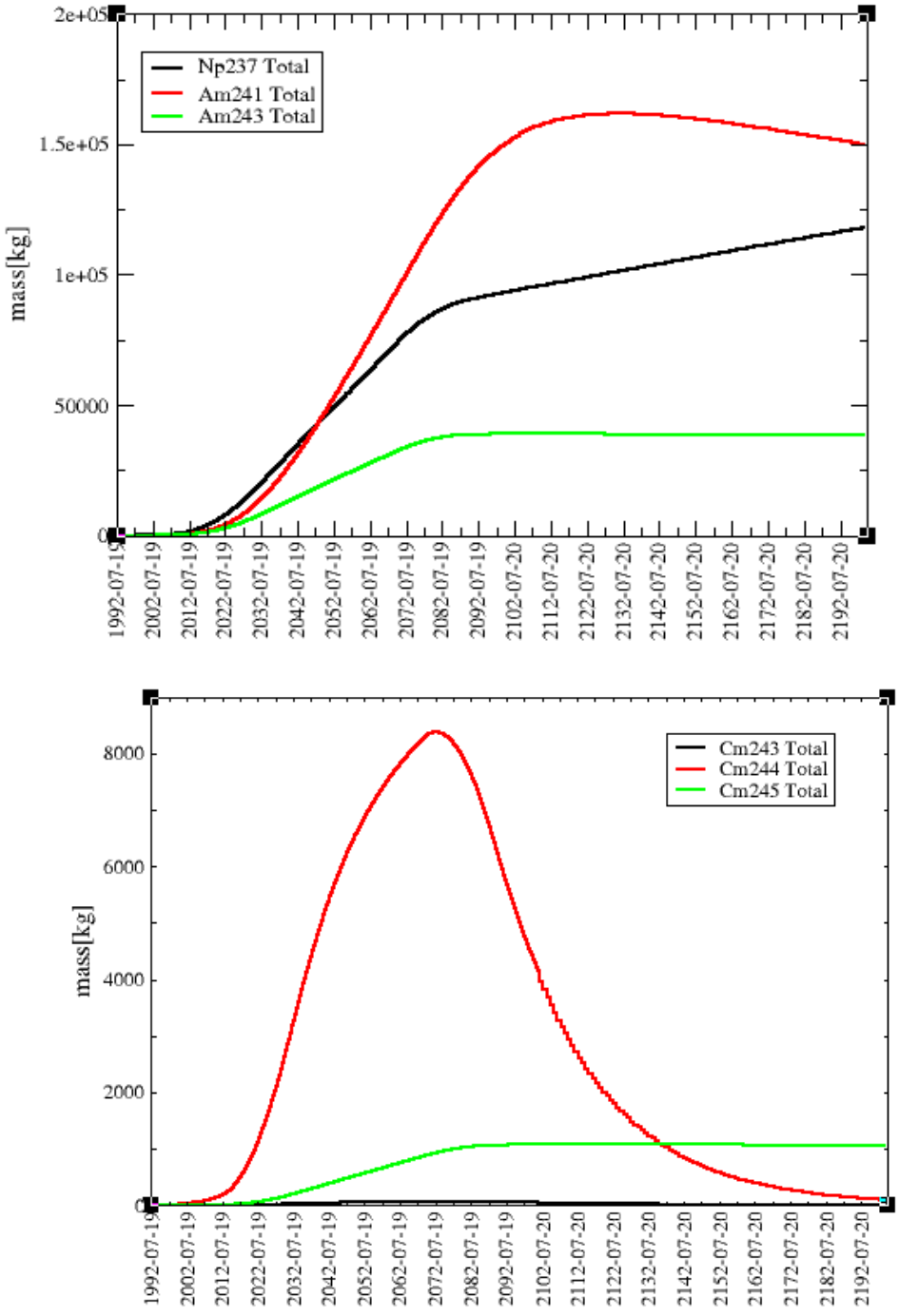


Figure 20: Inventories of Minor Actinides

Minor actinide is a term complementary to the major actinide. Basically, it consists of three elements: americium (mainly from plutonium's decay), curium (mainly from beta decay of americium isotopes) and neptunium (from the α-decay of americium isotopes and beta decay of uranium isotopes).

From the Figure 20, we can observe that “Mirror Function" happens again between decay curve of Np-237 and Am-241. After all the reactors' phasing out, new Pu-241 can not be generated any more so that the deposited Pu-241 decay with time and causing the increasing of Am-241, which is a typical "Mirror Function" mentioned above. Besides, Am-241 will decay to Np-237 with half-life time of 432.2y to cause a new "Mirror Function" after most Pu-241 is decayed.

Another two interesting curves are Cm-243 and Cm-244 that are short-lived nuclides with half-life time of 29.1y and 18.1y correspondingly. Although their magnitude is quite small (peak value is only approximate 100 kg and 8000 kg) comparing to inventories of primary actinides, they are still strong α-emitters and spontaneous fission nuclides so that they gives out large amounts of heat and neutrons after taking out from the core, which brings necessity of cooling storage before reprocessing.

The total amount of MA that can be transmuted is considered as the sum of americium and curium isotopes’ inventories, which equals to about 210 tons till 2200. The expected MA transmutation rate of MFBR is 50kg/(Gwe*y), which means that 15 kg MA can be transmuted by one MFBR unit per year (or 75 kg for one commercial unit with 1500 MW electricity power) [17].

$$T(year) = \frac{m_{MA}(kg)}{n(unit) \times TR(kg/GWe/year) \times P(GWe/unit)} \qquad (15)$$

In Equation (15), T represents the transmutation time, m_{MA} is mass of repository MA, n is number of FBR unit, TR is the transmutation rate and P is rated electricity power of FBR unit. Therefore, the transmutation time is 2333 years for MFBR route (6 MFBR units will be commissioned) and 70 years for commercial LFBR route (40 LFBR units will be commissioned).

Long-lived Fission Products

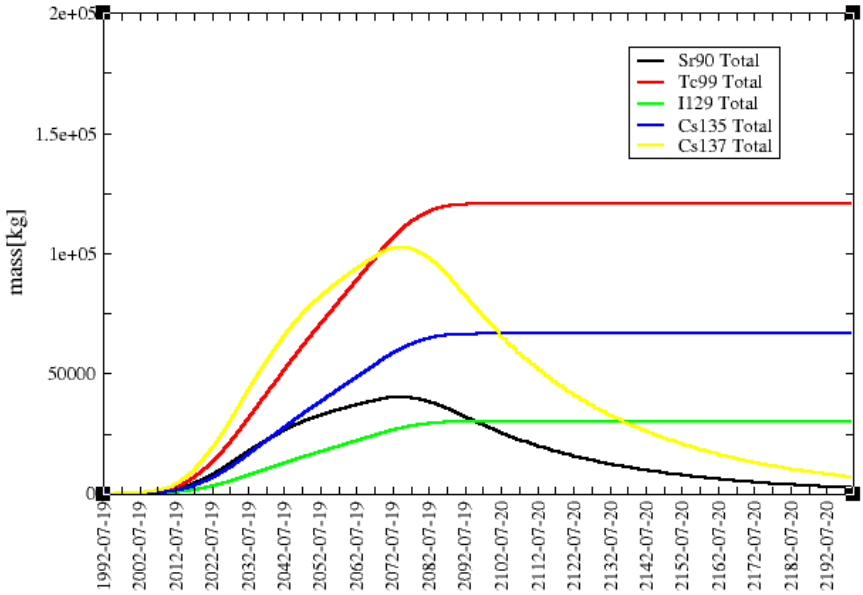


Figure 21: Long-lived Fission Products

There are more than 700 different fission yield nuclides in the typical LWR, from which only five nuclides (Tc-99, Sr-90, Cs-135, Cs-137 and I-129) are selected for the new code's simulation. The criteria for nuclide selecting are long half-life, large magnitude (obtained from the Figure 21), high radioactivity (obtained from Table 16) and high mobility.

Nuclide Type	e ($Sv*Bq^{-1}$)	Nuclide Type	e ($Sv*Bq^{-1}$)
Uranium 235	4.7×10^{-8}	Curium 242	1.2×10^{-8}
Uranium 236	4.7×10^{-8}	Curium 243	1.5×10^{-7}
Plutonium 239	2.5×10^{-7}	Curium 244	1.2×10^{-7}
Plutonium 240	2.5×10^{-7}	Curium 245	2.1×10^{-7}
Uranium 234	4.9×10^{-8}	Strontium 90	2.8×10^{-8}
Plutonium 238	2.3×10^{-7}	Technetium 99	6.4×10^{-10}
Plutonium 241	4.8×10^{-9}	Iodine 129	1.1×10^{-7}
Plutonium 242	2.4×10^{-7}	Cesium 135	2.0×10^{-9}
Neptunium 237	1.1×10^{-7}	Cesium 137	1.3×10^{-8}
Neptunium 239	8.0×10^{-10}	Samarium 149	——
Americium 241	2.0×10^{-7}		
Americium 243	2.0×10^{-7}		

Table 16: Ingestion Dose Coefficient of Nuclides [18]

Sr-90 and Cs-137 have a small half-life (only 18.78y and 30.07y correspondingly), which can be shown in Figure 21. This means that their radiotoxicity will decrease below background after 300 years storage.

On the contrast, long-lived I-129 will sublime to atmosphere (or dissolve into water) and deposit in some of the human glands to become a long-term inside radiation source; Tc-99 will come into water in the form of acid ion; Cs-135 will dissolve into water and deposit in organic substances.

According to the above description, large amount of fission products should be taken care in the reactor waste deposit process. Moreover, they are even more difficult to solve because they have various existence state, complex compound possibilities and they cannot be transmuted through fast neutron transmutation process as the transuranic nuclides.

Conclusion

According to the all depiction above, following results can be proved or concluded:

1. Reasonable results for different nuclides and different facilities can be obtained from the new fuel cycle code. However, because of the constant neutron spectrum, old version cross-section data and many assumptions, results generated from new fuel cycle code are not quite accurate till now.
2. Huge amount of nuclear waste inventory will be generated from Chinese reactor park if following Chinese government's time plan. As simulated from the new fuel cycle code, more than 700 tons of plutonium (including isotopes Pu-239, Pu-240, Pu-241 and Pu-242) and 230 tons of MA (including different isotopes of neptunium, americium and curium) will accumulate in the waste repository place till 2100. Even worse thing is that most of these transuranic nuclides are long-lived or high radiotoxic ones that give a heavy burden for the future environment. All of these factors give obvious necessity of the fast neutron transmutation facility's setting up.
3. For power-balance scenario, plutonium inventory is enough and 350 tons HM per year's more reprocessing ability should be added; for plutonium sustained scenario with 1.30 breeding ratio scenario, only one third of nuclear power generation (82226 MW) can be fulfilled by 2100; for CIAE LFBR proposal based on waste repository inventory, 60000 MW (out of 82226 MW) electricity generation can be fulfilled.
4. There are two choices to start the fast reactor transmutation project: one is accomplishing six MFBRs' commission with electricity power of 300 MW by 2035; the other one is starting LFBR construction with electricity power of 1500 MW according to CIAE's time plan from 2050. For the first choice, more than 2333 years will be needed. While, if using the second choice and same MA load, only 70 years are enough.
5. Transmutation work with FBR park is not efficient, so ADS facilities should be reconsidered for the MA transmutation.

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Appendix A

Summary of JEF 3.0 and EAF 99

JEF 3.0						
1001	34080	46104	53131	61151	74183	96247
1002	34082	46105	53135	62144	74184	96248
1003	35079	46106	54124	62147	74186	96249
2003	35081	46107	54126	62148	75185	96250
2004	36078	46108	54128	62149	75187	97249
3006	36082	46110	54129	62150	82204	97250
3007	36083	47111	54130	62151	82206	98249
4009	36084	48106	54131	62152	82208	98250
5010	36085	48108	54132	62153	83209	98251
5011	36086	48110	54133	62154	90227	98252
7014	37085	48111	54134	63151	90228	98254
7015	37086	48112	54135	63152	90229	99254
8016	37087	48113	54136	63153	90230	99255
8017	38087	48114	55133	63154	90232	
9019	38088	48315	55134	63155	90233	
11023	38089	48116	55135	63156	90234	
13027	38090	49115	55136	63157	91231	
14028	39089	50112	55137	64152	91233	
14029	39090	50114	56130	64154	92232	
14030	39091	50115	56132	64155	92233	
15031	40090	50116	56134	64156	92234	
21045	40091	50117	56135	64157	92235	
24050	40092	50118	56136	64158	92236	
24052	40093	50119	56137	64160	92237	
24053	40094	50120	56138	65159	92238	
24054	40095	50122	56140	65160	93236	
25055	40096	50123	57138	66160	93237	
26054	41093	50124	57139	66161	93238	

26056	42094	50125	57140	66162	93239	
26057	42095	50126	58140	66163	94236	
26058	42096	51121	58141	66164	94237	
27059	42097	51123	58142	67165	94238	
28058	42098	51124	58143	68162	94239	
28061	42099	51125	58144	68164	94240	
28062	42100	51126	59141	68166	94241	
28064	43099	52120	59142	68167	94242	
29063	44096	52122	59143	68168	94243	
29065	44098	52123	60142	68170	94244	
32070	44099	52124	60143	71175	95241	
32072	44100	52125	60144	71176	95242	
32073	44101	52126	60145	72174	95342	
32074	44102	52327	60146	72176	95243	
32076	44103	52128	60147	72177	95344	
33075	44104	52329	60148	72178	96241	
34074	44105	52130	60150	72179	96242	
34076	44106	52132	61147	72180	96243	
34077	45103	53127	61148	73181	96244	
34078	45105	53129	61348	73182	96245	
34079	46102	53130	61149	74182	96246	

E AF 99									
4007	21046	32068	45301	55131	65156	71377	77190	82202	99252
4010	21047	32069	45302	55132	65356	72170	77191	82203	99253
6012	21048	32071	45099	56128	65157	72171	77192	82205	99354
6013	21344	32077	45399	56129	65158	72172	77193	82207	100252
6014	22044	33071	46100	56131	65161	72173	77194	82209	100253
8018	22045	33072	46101	56133	66154	72175	78200	82210	100255
10020	22046	33073	46103	56333	66155	72181	77393	83203	100257
10021	22047	33074	46109	56335	66156	72182	77394	83205	
10022	22048	33076	46112	56139	66157	72380	77396	83206	
11022	22049	33077	47105	57135	66158	72378	77392	83207	

11024	22050	34072	47306	57137	66159	72379	78188	83208	
12024	23048	34073	47308	57141	66165	73177	78189	83210	
12025	23049	34075	47310	58134	66166	73179	78190	83310	
12026	23050	35076	48109	58135	67163	73180	78191	84206	
12028	23051	35077	48115	58136	67164	73183	78192	84207	
13026	24048	35082	48313	58138	67166	73380	78193	84208	
14031	24051	36076	49114	58139	67364	74178	78194	84209	
14032	25052	36079	50113	58337	67366	74180	78195	84210	
15032	25053	36081	50121	60140	68161	74181	78196	86211	
15033	25054	37084	50317	60141	68165	74185	78197	86222	
16032	26055	38082	50319	60149	68169	74187	78198	88223	
16033	26059	38083	50321	61143	68171	74188	78395	88224	
16034	26060	38085	51119	61144	68172	75181	79300	88225	
16035	27055	39086	51122	61145	69165	75182	79194	88226	
16036	27056	39087	51127	61146	69166	75183	79196	88228	
17035	27057	39088	51320	61150	69167	75184	79198	89225	
17036	27058	40088	52118	62145	69168	75186	80200	89226	
17037	27060	40089	52119	62146	69169	75188	80201	89227	
18036	27358	40097	52121	63145	69170	75189	80202	90231	
18037	28056	41091	52127	63146	69171	75384	80203	91228	
18038	28057	41092	52129	63147	69172	75386	80204	91229	
18039	28059	41094	52319	63148	70166	76182	79398	91230	
18040	28060	41095	52321	63149	70168	76183	80193	91232	
18041	28063	41096	52323	63150	70169	76184	80194	92230	
18042	28066	41391	52325	63350	70170	76185	80195	92231	
19039	29064	41392	52131	63352	70171	76186	80196	93234	
19040	29067	41393	53123	64146	70172	76187	80197	93235	
19041	30064	41395	53124	64147	70173	76188	80198	94246	
19042	30065	42093	53125	64148	70174	76189	80199	95240	
19043	30066	43095	53126	64149	70175	76190	81200	96240	
20040	30067	43096	53128	64150	70176	76191	81201	97245	
20041	30068	43097	53133	64151	71169	76192	81202	97246	
20042	30070	43098	54122	64153	71170	76193	81203	97247	
20043	30072	43395	54125	64159	71171	76194	81204	97248	

20044	30369	43397	54127	65151	71172	76391	81205	97348	
20045	31067	44097	54329	65152	71173	77185	80393	98246	
20046	31069	45100	54331	65153	71174	77186	80395	98248	
20047	31071	45101	54333	65154	71374	77188	80397	98253	
20048	31072	45102	55129	65155	71177	77189	82200	99251	

Appendix B

Simulation Process Tips

1. In order to display the figures plotted under LINUX system in Windows system, some X server program should be applied into windows system. Two X server programs (winaXe and cygwin X) are tested and the result is: Only the test version of winaXe is free to download; Cygwin X is purely free for all users; Cygwin X is 35~50% faster than winaXe for the same scenario simulation.
2. When downloading cygwin X folders through ftp, all the ssh, ssl and X11 folder should be included for the sake of later usage;
3. Setting different time steps before reactor' phasing out and after phasing out accelerates simulation process obviously, especially when the non-reactor time is defined to be long to check short half-life time nuclides such as Cs137's full-scale decay situation.

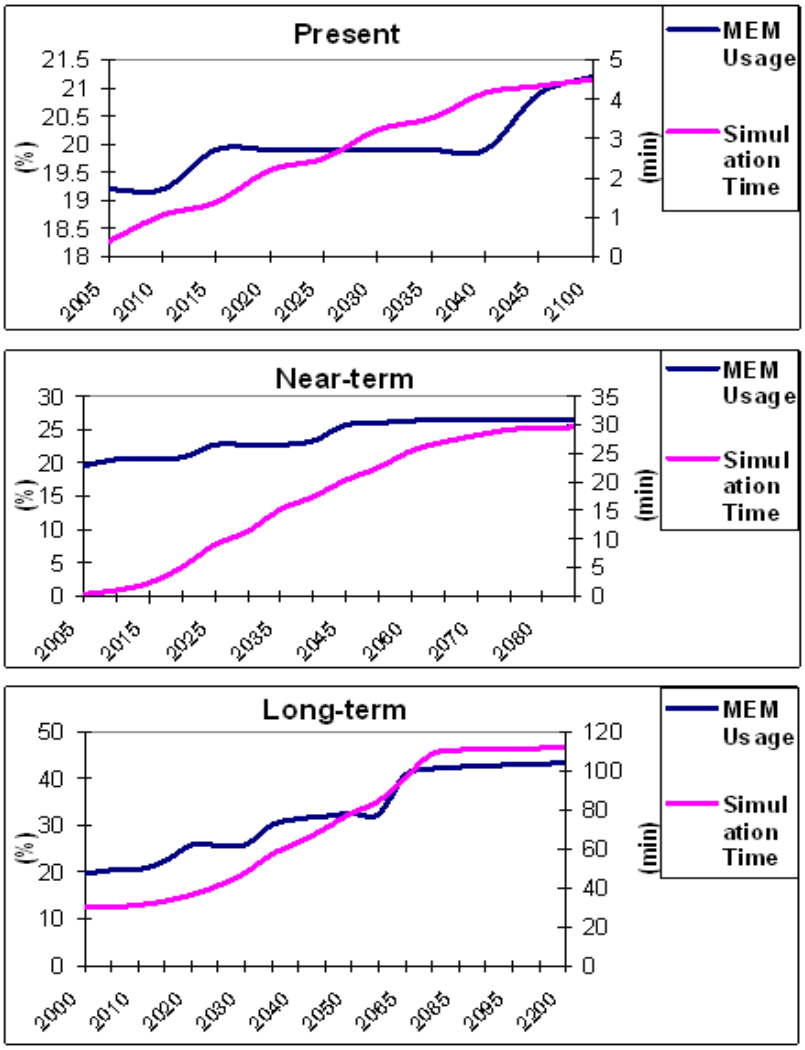

Figure 18 Computer Statuses of Test Run

4. Not only time scale and time step but also facility number and type will influence the simulation process (simulation time and MEM usage status). The investigations of these effects are listed in Figure 18