Strategy Consideration and R&D Activities of Closed Nuclear Fuel Cycle in China

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1. The Strategy Study of China Nuclear Energy Development

1.1 Nuclear Energy Requirement and PWR-FBR Development Strategy

Based on the overseas and domestic experiences on nuclear application, it has been started in China mainland to develop nuclear power plants as a subsidiary role in the regions which are lack of electricity and energy resources.

Up to now, only 6.7 GWe NPPs with nine units in Operation, 2 GWe NPPs with two units under Pre-Operation testing.

Other 7.2 GWe NPPs with eight units, some of them are under Pre-construction based on our own technology and some of them are intended to import from other countries.

Among these nineteen units only two HWRs, others are all PWRs.

According to the 2006～2020 National Nuclear Development Program Suggestion (to be approved by the Government), the nuclear energy utilization will be further developed in the future: The total nuclear power capacity will reach 40 GWe in 2020.

The nuclear capacity by 2005 envisaged will reach to 250 GWe in 2050, almost same as preceding estimated.

For such huge capacity 250 GWe NPPs it is impossible to use only PWRs due to the Uranium resources technically and economically exploited are limited

in China or in the world. So the nuclear fuel cycle police banes on PWR-FBR system is the only way for sustainable development of nuclear energy in China.

In order to estimate the maximum contribution by matched PWR-FBR development, assuming:

(1) the total capacity of PWRs in 2020 and 2030 will reach 32GWe and 50GWe respectively;

(2) In each period the capacity increases linearly;

(3) Large Fast Breeder Reactors will be deployed from 2030, and with metal fuel closed cycle.

So, by matched PWR-FBR development and their related closed fuel cycle, the fission nuclear energy will be sustainable one to support the national economic development and public living standard improvement, and the nuclear could be really clean energy.

1.2 Status of China Experimental Fast Reactor

In the framework of the National '863' High-Tech Program the China Experimental Fast Reactor has been executed since 1990.

CEFR timetable

Conceptual Design	1990-1992.7
Consultation with Russian FBR Association and Optimization	1993
Technical Co-Design with R-FBR-A	1994-1995
Preliminary Design	1996-1997
Detail Design	1998-2003
Preliminary Safety Analysis Report Review	1998.5-2000.5
Architecture Construction (first pot of concrete) started	2000.5
Reactor Building construction completed	2002.8

Recent Status of CEFR

More than 90% components and systems have been ordered.

More than 400 components have been installed in the building.

70% non-sodium systems have been installed,but for Na systems only 20%.

Planning

2005.8 Starting installation of Reactor Block

2005～2007 Pre-operation testing

2008.6～2008.7 Physics start-up and first criticality

2008.12 40% full power incorporated to the grid

1.3 China FBR Development Strategy Study

Some famous experts of China were organized by CNNC recent years studying FBR development pace to meet future technology requirement. A road-map has been worked out as expert's proposal for further study seeing follow table.

Table 1 China FBR Development Strategy

Reactor	Power(MWe)	Design Beginning	Commissioning
CEFR	25	1990	2008
CPFR	600	2005	2020
CMFR	n×300	(If needed)	
CDFR	1000～1500	2010	2025
CCFR	1000～1500	2018	2030～2035

2. Nuclear Fuel Cycle Consideration

2.1 Overall Target and Fuel Cycle Program

Based on the following consideration, uranium and plutonium recycle and

MA/LLFP burning has been put forward, seeing following figu.1:

• Uranium resources should be sufficiently utilized including by-products Pu and MA

• The volume of high radioactive wastes to be geologically buried should be as less as possible

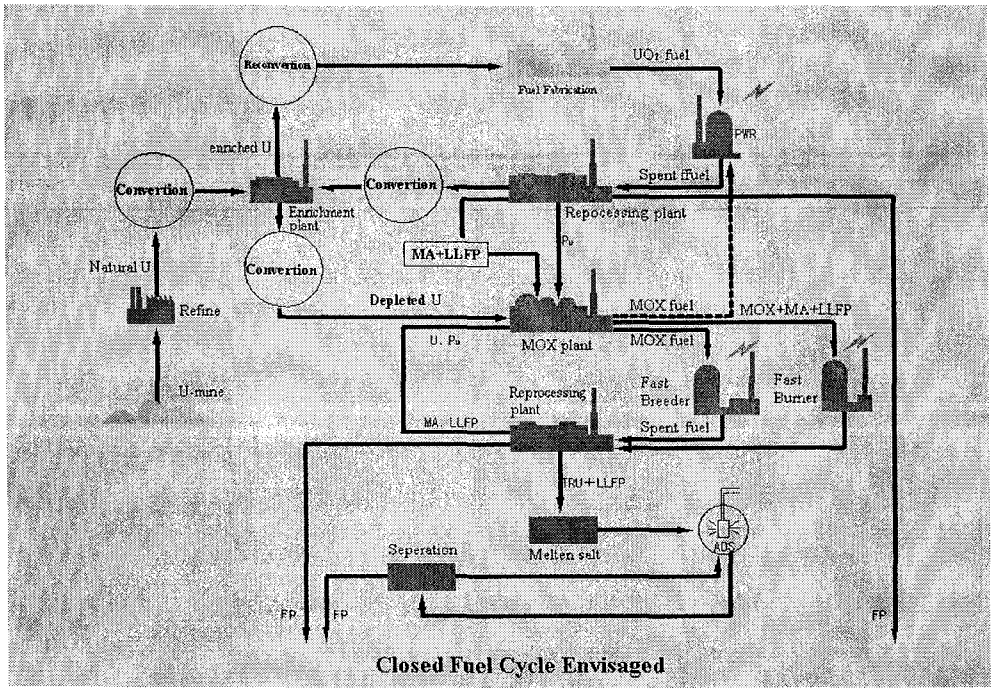
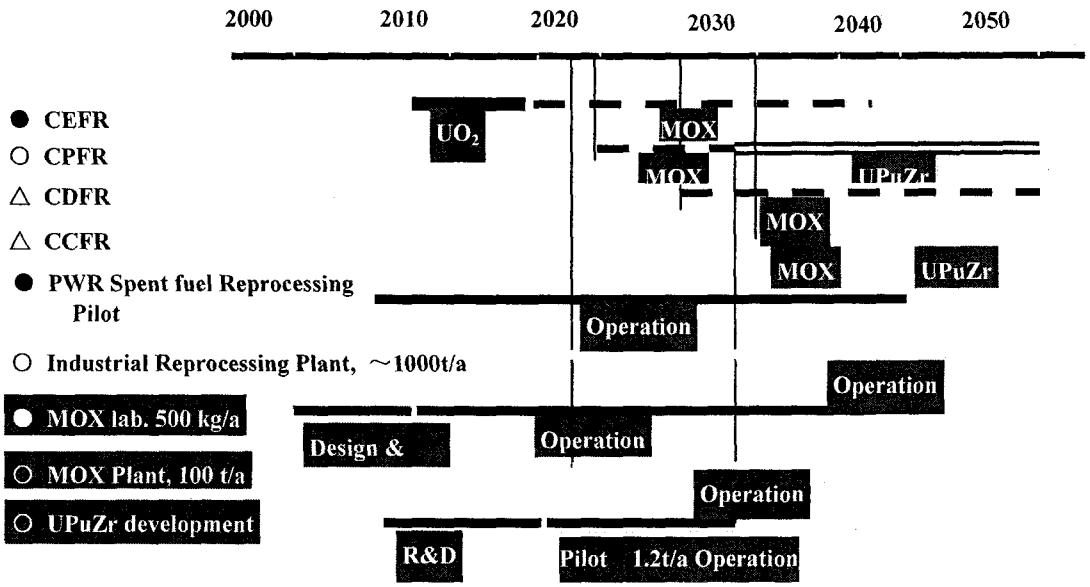


Fig.1 closed fuel cycle envisaged

The target fuel cycle for FBR is in-site, metal fuel closed cycle. MOX fuel only as transit and standby closed cycle for Fast Breeder or Burner reactor.

For this target a program has been roughly framed:



Figu.2 fuel cycle road-map

3. PWR spent fuel reprocessing

3.1 Reprocessing pilot plant

In China, the R&D programs on the civilian spent fuel reprocessing have been implemented. A central wet storage facility with the capacity of 550 tHM for intermediate storage of the spent fuel is completion. The pilot plant is under construction and its hot commissioning will be initiated by the end of 2007. RPP's function is as the follows:

- demonstration of the processes, equipment and instrumentation under hot conditions,
- experience accumulation of the design, construction and operating;
- training of the operation personnel,
- recovering highly enriched uranium (HEU) from spent fuel of research and test reactors,
- R&D of reprocessing technology for MOX spent fuel in future.

RPP's main process for spent LEU fuel reprocessing is based on a mechanical head-end process followed by the Purex-type process comprising two TBP

extraction cycles and tail end process. The designed and studied steps include as follows: a shear-leach process, a clarification process including pulse filtration, a porous stainless steel sintered tubes and centrifuge, a removing radioiodine process for the dissolution off-gas, a spent solvent treatment system using hydrazine carbonate, a new Pu valence adjustment with N_2O4_4 liquid, a continues precipitation of Pu(Ⅲ)oxalate, denitration of uranyl nitrate solution to UO_3 or U_3O_8 by microwave heater etc. The simulated tests of some key equipment and instrumentation, and remote operation have been carried out recently.Recovered uranium product will be re-enriched in other plant while separated plutonium will be fabricated into MOX fuel to recycle to FBR in a small scale demonstration facility.

At present, a lot of auxiliary research works is undertaken in institutes such as CIAE and BINE, to ensure RPP's stable and safe operation after 2003 with a modificational 2 extraction cycle Purex process to get qualified U and Pu products. Some R&D works are also undertaken to develop advanced Purex process for future.

3.2 Commercial reprocessing plant program

Considering the needs of further development of nuclear power, the construction of a large-size commercial reprocessing plant has been under the consideration of government at present. Pre-investigation work has already be started in CNNC. A reprocessing plant with the capacity of 800t/a spent fuel is planed to completed around 2020. Its main digests are:

- The initial uranium enrichment is 4.45%
- The burn-up for PWR spent fuel reach 50000MWd/t
- Matured process PUREX has been selected, UO_3 or U_3O_8 and PuO_2 as Products
- High Activity wastes are temporarily stored and not vitrified to wait the

suitable extraction process for Actinides which is under development in CIAE and Tsinghua University.

3.3 R&D on investigation on the flowsheet of U-Pu co-recovery process:

MOX fuel has been used in light reactor in the scal of industry, in which the reprocessing plutonium can be utilized to generate electricity efficiently. At present uranium product and plutonium product can be obtained through reprocessing, then U and re-processing Pu are mixed with certaint ratio to fabricate the MOX fuel.This process is complicated and unfavourable to safeguards. A improved Purex flowsheet has been studied in CIAE showed in Fig.3.In 1A extractor U and Pu are co-extracted into organic phase, in 1B extractor Pu and partial U are stripped into aquoes phase. There is no Pu product in this process, and then the U+Pu product is changed into MOX fuel.The ratio of Pu to U+Pu vary from 5% to 30% suiting for various purpose.

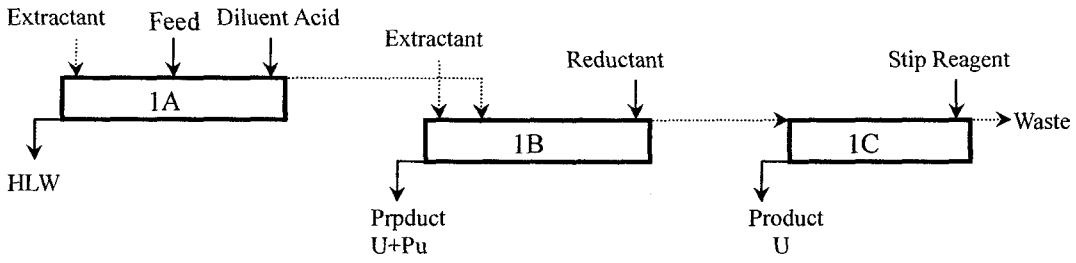


Fig.3 First cycle flowsheet of improved Purex Process for the back-extraction of Plutonium together with partly Uranium

Considering the needs of further development of nuclear power, a large-size commercial reprocessing plant will be put production possibly around 2020, after sufficient experience of the pilot plant is obtained, and certain volume of the spent fuel accumulated.

3.4 Study on elimination of Zr influence on separation of actinides for high burn-up spend fuel

Zirconium is a troublesome fission product element in PUREX process especially for high burn-up spent fuel. Zirconium easily form compounds with degradation products of TBP, then polymer, and finally interface crud, which interfere seriously extraction process. Similarly Zr interfere the extraction process for the separation of HLLW if most Zr go into HLW from PUREX process. So it is necessary to remove zirconium form the solution of spent fuel before PUREX process. Silica gel usually is used to remove zirconium.

The adsorption behavior of zirconium by silica gel was studied, including the effect of acidity, determination of capacity (both static and dynamic).The main result is:

(1) In a simple simulated system, the adsorption capacity of zirconium adsorbed by silica gel is about 0.02-0.03g zirconium per gram silica gel.

(2) The adsorption capacity of zirconium by silica gel drops with the increasing of the acidity of solution.

(3) Dynamic experiment shows that 0.5g/l zirconium solution break through the column at a volume of 30ml, high concentration of HNO_3 can remove zirconium from the silica gel column.

4. Partitioning of HLLW in China

The high level liquid waste (HLLW) has high radioactive and high toxicity. It concentrates more than 95% radioactivity in spent nuclear fuel. The safe disposal of HLLW is one of the key problems that effect on the development of nuclear energy.

A Total Partition(TP) process, which consists of a TRPO(mixed trialkyl phosphine oxide) process for separating TRU and ^{99}Tc, a CESE(Crow Ether Strontium Extraction) process for removing ^{90}Sr and a KtiFC ion exchanger process for segregating ^{137}Cs from HLLW,was developed by INET, Tsinghua

university.A hot test with real high saline HLLW shows that these processes are well compatible and the HLLW could be transferred to ILLW.Table 5 shows the DFs of some nuclides in the hot test.

Table 2 The DFs of obtained in the hot test of TP

Nuclides	Total α	^{241}Am	^{239}Pu	^{237}Np	U	^{99}Tc	^{90}Sr	^{137}Cs
Material balance	93	93	94	124	131	104	119	--
DF	588	666	695	15.2	1000	125	>250	>2000

Another promising process for separation α-nuclides from HLLW to make it non-α-waste is the Podand Amide Process developed by CIAE, which comprising of 3 batteries(Fig4).With 0.2mol/L TBOPDA/ 40% octanol-kerosene actinides and lanthanides were coextracted into organic phase with the recovery of 99.98 % for U(VI),>99.98 for Pu(IV),and >99.99% for Am(III) and Eu(III) respectively in battery A, in battery R1, 99.99% U, 86.2% Pu and a part of Am or Eu were stripped into aqueous phase by 0.01mol/L HNO_3+0.2mol/L aceto-hydroxyamic acid (AHA). In battery R2, Am , Eu and remained Pu were stripped into aqueous phase by 0.2mol/L AHA. This separation process possesses has some advantages: it is a salt-free process, the extractant can be completely incinerated and the extraction can be done in relative high concentration of nitric acid solution without dilution of HLLW. As for strotium, it can be extracted by N,N,N'N'-tetraisobutyl-3-oxa-pentanediamide (TiBOPDA) directly from 2~3mol/l nitric acidic solutions.

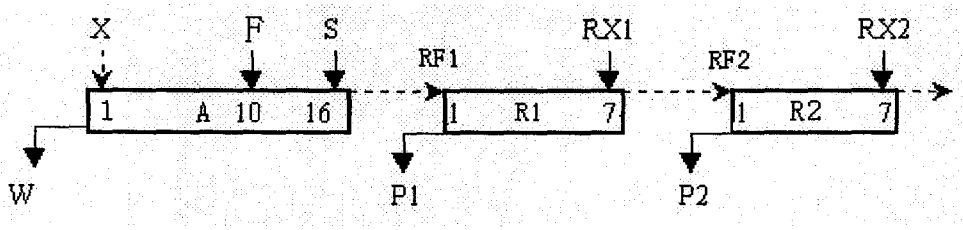


Fig.4 The flowsheet diagram of the separation experiment with a set of miniature mixsettler using 0.2mol/LTBOPDA as extractant.

F---HLLW, F:X:S=1:1.5:0.5
X---0.2mol/LTBOPDA/40%octanol-kerosene, RF1:RX1=1:1
S---2.0mol/LHNO_3, RF2:RX2=1:1
RX1---0.01mol/LHNO_3+0.2mol/LTBOPDA F=1.0ml/min
RX2---0.2mol/L AHA X=RF1=RF2=1.5ml/min
S=0.5ml/min

Crystal Silicotitanate (CST,its chemical fomular is $H_{0.85}Na_{1.15}Ti_{1.84}O_{6.68}$ $1.78H_2O$) -- a novel inorganic ion exchanger, was sythesized by INET, which has high capacity and selectivity for cesium. A new cesium ion-sieve (Cs-IS) was developed by Tianjin University . Cs-IS carries forward the large exchange capacity for cesium and good chemical stability in the medium of 3mol/l HNO_3. In the static exchange with the strong acid simulated HLLW Cs- IS exhibits high exchange rate for cesium (above 96%) and large separation factor $\beta_{Cs/M}$ (more than 958).

5. Management of high radioactive waste

5.1 HLLW

Vitrification was widely used for solidifying HLLW in many countries. Liquid-Fed Ceramic Melter(LFCM) technology has been development for long time in China. A full-scale mock-up vitrification facility was build on Sichuan Nuclear Fuel Plant. It has been put into operation since Feburary this year. The main features of the melter are shown in Table 6.It is estimated that a hot vitrification facility could be built around 2010 in Sichuan Nuclear Plant.

Table 3 Main features of the ceramic melter

Designed feed throughput	65L/h
Waste glass throughput	38.4kg/h
Pool surface	1.4m^2
Heating electrodes	3 plates with air cooling,material Inconel 690
Melter refractory	ER2161,ER1681,ER1711

Pouring system	Bottom and overflow
Overall dimension	3372×2162×2682mm
Weight	28t

5.2 strategy of high level waste disposal

COSTIND(the commission of science, technology and industry for national defense) held a conference on high level radioactive waste disposal in Beijing in August,2005. The strategy program has been outlined through this conference and the guide will be published afterwards. The main digests are as follows:

The Deep Geological Disposal Repository for HLW is planed to be completed around 2060. The program is divided into 4 phases:

Phase 1:2006-2020 laboratory preliminary study and site selection

Phase 2:2021-2040 underground experiment,

Phase 3:2041-2050 protocol repository design

Phase 4:2051-2060 construction of repository

During Phase 1 main task is to constitute concrete strategy, program, code, and standard, to conduct the study of depository engineering, to conduct the study of depositor geology, to conduct the study of depository chemistry,to conduct the study of security and environment appraise. In phase 2 an underground research laboratory will be built near the selected site and all full-scale *insitu* test and disposal demonstration will be conducted in the laboratory. The final repository design and construction will be finished in 2050 and around 2060 respectively.

The geology disposal of HLW is a long-term comprehensive task. Disposal chemistry and site pre-selection have been conducted since 1989. Site pre-select focused on northwest China, with granite considered as the candidate host rock. The following aspects of the area have been studied: earthquakes, structural framework, active faults, crustal stability, lithilogy, hydrogeology, and engineering geology. Some imported parameter such as diffusion, adsorption and dispersion etc has been obtained for some key elements Pu, Am, Np, and Tc through the

experiment with underground water, underground granite and simulating geological conditions.

In addition to the above, other studies are also being conducted on:

- Site preselection for an underground research laboratory;
- Natural analogues;
- Buffer/backfill materials and their geotechnics;
- Speciation of transuranic elements in solutions;
- Heater test;and
- Models for safely and environmental assessments.

The safe disposal of high level radioactive waste is a worldwide challenging task. Although China has made much progress in this field, there is still a long way to go.

6. Treatment of L/ILLW

6.1 solid waste

Compaction, supercompaction and incineration were selected to reduce the volume of solid waste. Some incompactable and unburnable waste will be encapsulated by cementation.

A compaction facility with supercompactor of 2000KN for LLW has been constructed in CIAE. It is planned to put into commmissioning in 2007.Both 180L drum and 200L drum could be compacted in this facility.

A multipurpose incineration facility with capacity of 25—30kg/h has been built in CIRP. It can be used for incinerating solid waste with 40% plastic or rubber, or with 20% spent resin.It can also incinerate waste oil. The cold tests show that the waste can be fully pyrolyzed and combusted. The residual carbon in ash was below 5%(wt.), the DF of particulate was $>10^7$, and the absorption efficiency for HCl,SO_x and NO_x was higher than 95%,98% and 83% respectively.

6.2 Disposal of L/ILLW

In light of the policy on LLW and ILLW regional disposal, the national regional repositories have been constructed in the area where nuclear facilities are comparatively concentrated to dispose of local LLW and ILLW.

Tne Northwest Repository is one of the LLW and ILLW regional repositories in China, mainly to dispose of the LLW and ILLW solid wastes which were in stock arising from the nuclear industry in past years, has been already produced and will be produced during decommissioning. It is located in Gansu Province, and the first phase of the project (20,000 m^3 waste in capacity) went into trial operation in 1999. The first 140m^3(635drums) LLW disposed is contaminated sandy soil from Lanzhou Nuclear Fuel Complex.

Guangdong Beilong Repository is close to Daya Bay and Ling-ao NPPs, its designed capacity is 80,000 m^3.The first phase of the project with the capacity of 8,000 m^3 can be completed by the end of 2000. It will dispose of the LLW and ILLW solid wastes mainly produced by NPPs in Guangdong Province and in adjacent area and by the applications of isotopes.

Zhejiang Repository is now in the process of planning and siting, which will dispose of wastes arising from the province and NPPs in adjacent area.