

PRESENT STATUS OF THE OMEGA PROGRAM IN JAPAN

PREPARED FOR  
SECOND GENERAL MEETING FOR INFORMATION EXCHANGE MEETING  
ON ACTINIDE AND FISSION PRODUCT SEPARATION AND TRANSMUTATION  
11-13 NOVEMBER 1992

Masato NAKAMURA  
Science and Technology Agency  
Chiyoda-ku , Tokyo , Japan

Hiroyuki YOSHIDA  
Japan Atomic Energy Research Institute  
Tokai-mura , Ibaraki-ken , Japan

Satoshi TANI  
Power Reactor and Nuclear Fuel Development Corporation  
Minato-ku , Tokyo , Japan

Tadashi INOUE  
Central Research Institute of Electric Power Industry  
Komae-shi , Tokyo , Japan

## PRESENT STATUS OF THE OMEGA PROGRAM IN JAPAN

### 1. INTRODUCTION

The management of high level radioactive waste (HLW) generated from the reprocessing of spent fuel is very important as well as the safety assurance, to further develop nuclear electricity generation.

National policy of managing this HLW in Japan is to solidify in a stable form and to dispose of ultimately into deep geological formations after 30 to 50 years of storage for cooling. In addition, the Japan's Atomic Energy Commission submitted in October 1988 a report entitled "Long-Term Program for Research and Development on Nuclide Partitioning and Transmutation (P-T)", from the viewpoints of conversion of HLW into useful resources and its disposal efficiency. The program plots a course for technological development up to the year 2000 and is called "OMEGA" which is the acronym derived from Options for Making Extra Gains from Actinides and fission products.

The program was jointly stimulated by the collaborative efforts of the Japan Atomic Energy Research Institute (JAERI) and the Power Reactor and Nuclear Fuel Development Corporation (PNC). In the private sector, the Central Research Institute of Electric Power Industry (CRIEPI) has been also carrying out R&D on this subject.

The Japanese government (represented by Science and Technology Agency) proposed an international cooperation for information exchange relevant to partitioning and transmutation technology under the framework of the OECD Nuclear Energy Agency, in January 1989. The first information exchange meeting on this subject was held in Japan in November 1990. Eleven OECD member countries and two international organizations, namely IAEA and Joint Research Center of CEC participated in the meeting. A workshop on partitioning was held in Japan in November 1991, and a specialist meeting on accelerator-driven transmutation was held in Switzerland in March 1992.

### 2. OMEGA PROGRAM IN JAPAN

The program is to proceed in two phases: the phase-I and -II. The phase-I covers a period up to about 1996 and the phase-II covers a period from about 1997 to about 2000. In general, the basic studies and testing are to be conducted in the phase-I to evaluate various concepts and to develop required technologies. In the phase-II, engineering tests of technologies and/or demonstration of concepts are planned. After the year 2000, pilot facilities will be built to demonstrate the P-T Technology.

## 2.1. PARTITIONING

The principal technologies to be developed consist of the following three subjects and their R&D are to be conducted according to the R&D schedule shown in Fig.1.

Nuclide partitioning technology for HLW,  
Technology for recovering useful metals in insoluble residue from reprocessing, and  
Technology for utilizing the separated elements.

The partitioning is to separate HLW into four groups:

- i) TRU elements
- ii) Sr-Cs elements
- iii) Tc-Platinum elements
- iv) Others

Harmonized and optimized system is to be studied to integrate the partitioning process with the PUREX process and to perform the best utilization and management of nuclear resources contained in spent fuel. Feasibility and applicability of dry process with molten salt, and sublimation/volatilization processes are also to be studied.

The recovery technology is to be developed to obtain useful elements, such as Ru, Rh and Pd, from insoluble residues of dissolved solution from reprocessing. Purification and isolation techniques are also to be developed, for example, for removing long-lived Pd-107 from the recovered Pd element.

The fabrication technology of the separated TRU and fission products is to be developed for their further use or disposal. Immobilization of the separated Sr and Cs is needed for either utilization or disposal"

## 2.2. TRANSMUTATION

The principal technologies to be developed consist of the following two subjects and their R&D are to be conducted along with the R&D schedule shown in Fig.2 and Fig.3.

Application of nuclear reactor on transmutation

Application of accelerator on transmutation.

Nuclear reactor provides an extremely rational means for the possible earlier realization of transmutation technology for TRU nuclides. Sodium-cooled fast breeder reactor with MOX or metallic fuel can be applied for transmutation of many kinds of TRU nuclides due to its relatively large high-energy fissioning. TRU burner fast reactor is another candidate to be developed due to its more efficient transmutation capability than conventional fast reactor. Thermal neutron reactor is also an alternative, when plutonium is used as nuclear fuel.

Owing to the recent remarkable development of accelerator technology, application of accelerators is becoming more attractive as a means of transmutation. Proton accelerator is to be developed to transmute TRU and long-lived FPs by spallation reaction and associated large number of emitted neutrons. Electron accelerator is also to be developed to transmute the nuclides having small neutron cross-section, such as Sr-90 and Cs-137, by photo-nuclear reaction. Hybrid system combining an accelerator with a subcritical assembly will be studied to

improve the overall energy balance of the transmutation system.

### 2.3. RELATED BASIC RESEARCH

Reliable data base of TRU and long-lived FPs is indispensable to proceed the OMEGA program. Underlying studies on physical and chemical properties of TRU and FPs will improve understanding of the science and technology for separation and recovery of these nuclides from HLW, for fabrication of TRU fuel for recycling to reactors or accelerator-driven systems for transmutation, and for utilization of these nuclides. Nuclear data and thermodynamic data of these nuclides will be measured, compiled and evaluated for reactor Physics and material development.

The areas covered by the OMEGA program and R&D activities are illustrated in Fig.4.

### 3. RESEARCH AND DEVELOPMENT ACTIVITIES

Under the framework of the OMEGA program, the following R&D activities are under way at JAERI, PNC AND CRIEPI.

#### 3.1. PARTITIONING

Wet processes of partitioning are development. at JAERI AND PNC. Dry processes are developed at CRIEPI AND PNC.

##### A. STUDIES AT JAERI

A partitioning process has been developed to separate elements in the HLW into four groups. This process consists of three major steps as shown in Fig.5. In the first step, the HLW denitrated with formic acid is fed to the solvent extraction system where the DIDPA (diisodechl phosphoric acid) is used as a solvent. The DIDPA is an excellent solvent to extract not only tetra and hexa valent ions but also penta valent ion with addition of hydrogen peroxide. The TRU elements, such as residual Pu In HLW and minor actinides (Np, Am and Cm), are coextracted with rare earths (RE) into "the DIDPA, and then separation of the TRU from" the RE is carried out by using the DTPA (diethylene triamine pentaacetic acid) as a completing reagent. In the second step, raffinate is fed through activated carbon column for separation final step, the high-decay-heat generating elements, Sr and Cs, are captured by inorganic ion exchangers of titanitic acid and zeorite respectively, leaving the final solution virtually free of both ultra-long-lived radioactive nuclides and high-decay-heat generating nuclides.

In the mixer-settler experiments with the actual HLW, it was demonstrated that more than 99.99% of Am and Cm were extracted with the DIDPA. It was also Verified in a series of the tests with the synthesized HLW that more than

99.95% of Np was extracted when hydrogen peroxide was added as indicated in Fig.6. Regarding separation of the TRU from the RE, a preferential back extraction of the TRU from the DIDPA has been studied by using the DIDPA as the completing reagent to hold the RE in the organic phase. Distribution ratios of Am and RE were measured in the batch shaker tests and compared in Fig.7 by using a stripping solution composed of 0.05M DTPA, 1M lactic acid and NH<sub>4</sub>OH at PH value of 3.6. The separation factors were observed to be larger than 10 between Am and RE. Mixer-settler experiments are in progress to verify the effect of PH variations due to extraction of NH<sub>4</sub><sup>+</sup>.

The separation of Tc has been demonstrated by adsorption with the activated carbon column. More than 99% of Tc was adsorbed from the synthesized HLW adjusted to 0.5M nitric acid concentration and the absorbed Tc was eluted from the column very efficiently by using a eluant , 2M KSCN-4M NaOH ( alkaline thiocyanate solution). The adsorption and elution of platinum-group metals were also examined. And more than 99.9% of Sr and Cs were captured by the inorganic ion exchangers described above.

The proposed partitioning process is planned to be tested in the sense' of chemical engineering and in parallel , to be tested with the actual HLW at the Nuclear Fuel Cycle Safety Engineering Facility (NUCEF) of which cold operation will start in 1994.

## B. STUDIES AT PNC [11][21][31]

PNC's basic approach is to develop partitioning process as a part of advanced reprocessing system where the improved PUREX process is closely combined with TRUEX-like TRU separation process as indicated in Figure.8 with emphasis on TRU recycle in FBR system.

As an effort to improve PUREX process , studies are in progress to control Np valence by electrochemical and photochemical methods so that it is completely recovered in Pu product stream for recycling in FBR. Preliminary experiments are also in progress to electrochemically recover platinum-group metals and Tc from dissolver solution.

A series of extraction tests have been conducted to recover TRU from the HLW by using CMPO(octyl phenyl N,N diisobutyl carbamoyl methyl phosphine oxide) mixed with TBP as an extractant. Fundamental batch tests were carried out to determine the distribution ratios of TRU and FP elements as well as to clarify the fundamental behaviors of the CMPO-TBP solvent such as the third phase formation and the precipitation. Following these fundamental studies counter-current experiments with miniature mixer-settlers were conducted based on the simplified TRUEX flowsheet using the actual HLW generated from FBR fuel reprocessing tests. Concentration profiles for Am, Pu and typical FP uclides are shown in Fig.9. It has been observed that Am and Pu were completely extracted by the CMPO-TBP solvent whereas such FP elements as Cs and Sb were well retained in the raffinate. However , the RE elements such as Ce were

co-extracted as expected from the batch shaker measurements. Also extracted into the solvent was a significant fraction of Ru. Future studies are planned to improve the process for Pu back-extraction, to improve extraction efficiency of Np, and to develop a process to separate TRU from RE element.

As an effort to recover noble metals from the nuclear waste, the Pb extraction tests have been conducted with the actual insoluble residue produced in the dissolution of irradiated mixed oxide fuel. The tests verified that noble metals such as Ru, Rh and Pd were efficiently recovered by this process.

In addition to the development of aqueous separation processes, experimental studies are in progress to explore the feasibility of dry separation process. One of the processed methods under investigation is the FP separation by super high temperature treatment of the HLW as illustrated in Fig.10. In the preliminary experiments, the simulated HLW calcined at 700 C was heated in inert atmosphere at 900 C at first and then further heated up to about 1800 C. It was observed that more than 90% of Cs was eliminated by sublimation in 2 hours at 900 C, and most of noble and transition metals were separated as molten alloy from the oxides of remaining FP elements (RE, Zr, alkali earths and TRU) at 1800 C.

#### C. STUDIES AT CRIEPI [41]

CRIEPI has proposed a concept to separate TRUS from HLW by dry process with pyrometallurgical methods and then to transform to short-lived nuclides in a metallic fuel FBR under developed [4]. This dry process can be applied on the removal of TRUS from undissolved residue and cladding after the leaching by nitric acid without much efforts for the modification. According to this concept, TRUS produced in LWR fuel cycle are confined in the FBR cycle. Fig.11 shows the flow diagram of the dry process for partitioning of TRUS. This process consists of (1) denigration to obtain oxides, (2) chlorination to oxides to chlorides, (3) reductive extraction to reduce TRUS from molten chlorides in liquid cadmium by using of lithium as reductant, and (4) electrorefining to increase the purity of TRUS recovered. Both the denigration and chlorination steps are pre-treatment process prior to the application of pyrometallurgical process. The first target on the development of dry process is to recover more than 99% of TRUS containing in the HLW by allowing the introduction of the same amount of rare earths as TRUS.

Microwave heating device was selected as the preferred method for denigration of HLW because of the expectation of the production of small amount secondary wastes and operating safety in a hot cell. The technical feasibility of this method was examined experimentally, and a design study was carried out for the assumed scale of the denigration step based on the experiments. The specification of the facility with suitable condition is presented in Table.1.

Chlorination using chlorine gas together with carbon as reductant was selected as the preferred method because of

recycling of chlorine gas by processing of salt wastes. The feasibility examined experimentally, by using the apparatus shown in Fig.12, showed that nearly 100% of chlorination can be attained under suitable condition with evaporation of some amount of specified elements with chloride form, which can be recovered for the following processing of" reductive extraction. The assumed scale of chlorination step was designed based on the experimental study. The specification of the facility is also presented in Table.1.

The TRUS, accompanied with some amount of rare earths, contained in molten salts are reduced into liquid cadmium added lithium as reductant. The principle of the reductive extraction with the subsequent step of electrorefining is schematically shown in Fig.13. In order to evaluate the purity of TRUS recovered in liquid cadmium, it is necessary to obtain the distribution coefficients between salt and cadmium phases, and the related thermodynamic data, especially activity coefficients and standard free energy of formation of chlorides for TRUS and rare earths. The distribution coefficients measured for actinides of U and Np and for rare earths of La, Ce, Pr, Nd and Y predict the separation factor between actinides and rare earths 151[61].

The electrorefining technology can be applied as preferable method in order to get the TRUS with rare earth as low Concentration as possible. The electrorefining with liquid cadmium anode or inert anode and solid cathode covered with molten salt electrolyte, i.e. KCl-LiCl eutectic salt, has been applied. The separation study using rare earths has been carried out in CRIEPI, and, moreover, the anode dissolution and cathode deposition study using actinides has started within the framework of the TRUMP-S Program organized by CRIEPI, Kawasaki Heavy Industry (KHI), Rockwell International corp. (RI) and Univ.of Missouri. The development of process technology including pre-treatment process and the reductive extraction and electrorefining steps will start from the beginning of 1993 in CRIEPI.

The activity coefficients and free energy of formation of chlorides in the eutectic salt and activity coefficients in liquid cadmium were obtained by EMF measurements for rare earths, of La, Ce, Pr, Nd, Sm, Eu, Gd and Y, and for actinides of U, NP, Pu and Am [7]. The data gives the prediction of separation efficiency of actinides from rare earths, as shown in Fig.14. The figure suggests that recovery more than 99% of TRUS can be attained with the accompaniment of rare earth less than 10% to the initial amount in HLW.

Cadmium used in the pyrometallurgical process is recycled to the reductive extraction step after recovery of TRUS and another elements, such as noble metals, by the distillation. The waste chloride salts containing fission products mainly of alkali and/or disposal. It was clarified that the alkali and alkaline earth elements were reduced to metals by using the graphite anode and liquid cadmium or zinc cathode. The chlorine gas produced in the process can be" recycled to the chlorination process.

### 3.3. TRANSMUTATION WITH FISSION REACTORS

Optimization of minor actinide recycling into MOX-FBR is studied at PNC. Studies of minor actinide recycling into metallic fuel FBR are conducted at CRIEPI. Actinide burner reactor design study and integral experiments to improve cross section data of minor actinides are carried out at JAERI.

#### 3.3.1. MOX-FBR AND METALLIC FUEL FBR

##### A. STUDIES ON TRU TRANSMUTATION IN MOX-FBR POWER PLANT AT PNC [8][9]

Feasibility studies of TRU transmutation in LMFBRs have been performed to establish TRU (exactly minor actinides) transmutation technology. Systematic parameter survey calculations were performed for a conventional 1000MWe LMFBR core to investigate basic characteristics of TRU transmutation in an LMFBR core and also to establish TRU loading method which has no serious influence on core design.

A heterogeneous TRU-loading method, where a few number of subassemblies with concentrated TRU fuel (target S/As) are loaded in the core, can have an advantage in manufacturing and managing the TRU-Loaded fuel since the number of the TRU-loaded fuel subassemblies is smaller than that of the homogeneous method. An effort has been made so as to make the heterogeneous method feasible. The fuel pin design and the loading pattern of the target S/A were studied. It was found that the heterogeneous TRU-loading method with target S/As is feasible by reducing the fuel pin diameter loaded with TRU and optimizing the loading pattern of target S/As as shown in Fig.15.

Study on an innovative core concept for TRU transmutation was also carried out taking into account the results of systematic parameter survey calculations. Feasibility from neutronic point of view was studied on 1000MWe and 300MWe Super Long-Life Cores with no need of fuel exchange during plant life. It was found that remarkable reduction of both reactivity change and power variation during burnup is possible by optimizing the amount and zoning of TRU loaded. The reactivity change and power swing during 30 years of "the optimized core are less than half of those of the Super Long-Life Cores with no TRU loaded. The Super Long-Life Cores loaded with TRU fuel were found feasible from the neutronic point of view.

R&D program are being pursued to establish the transmutation technology of TRU nuclides in LMFBRs. Irradiation of MOX fuel pin including small amount of  $^{241}\text{Am}$  was started in JOYO from August 1992.

##### B. STUDIES ON TRANSMUTATION IN METALLIC FUEL FBR AT CRIEPI

TRUS recovered by pyrometallurgical process will be trans-



formed. to short-lived nuclides by fissioning in metallic fuel with the matrix of U-Pu-Zr in FBR. The research is directed to (1) the analysis of transmutation rare by simulation code and design study of fuel element and core, and the characterization study of U-Pu-Zr alloy contained minor actinides (Np, Am and Cm) and irradiation study. The latter activity has been proceeding by the cooperation with the European Institute for Transuranium Elements (ITU) of the Commission of the European Communities in Karlsruhe, Germany.

The CITATION-TRU code, developed by modifying the burn-up calculation code of the CITATION, pointed out that a metallic fuel FBR is superior in the transmutation TRUS because of the hard neutron spectrum. The transmutation rate of 13 to 14 wt% per year of the loaded TRUS is expected in the metallic. fuel FBR. If the metallic fuel allows 5 wt% of minor actinides in the matrix, a metallic fuel FBR with the capacity of 1000MWe can accept the minor actinides produced from 5 LWR plants with the same capacity per year at the equilibrium recycle condition [10]. Moreover, the CITE-BURN code, which can make the burn-up analysis at the fuel rod unit, was developed for the analysis of irradiation study. The analysis of sodium void effect gave no large effect' on reactor operation so far as the content of minor actinides is up to 5% [11]. In addition, using the analytical data for oxide samples with the bench mark calculations for transmutation was carried out in the cooperation with KfK, ITU and CRIEPI. This study showed that the prediction by calculation of transmutation of minor actinides is accordance with the experimental results.

As the first step, the alloys with binary alloys of actinides, and U-Zr with rare earths, and U-Pu-Zr with minor actinides and rare earths were fabricated for the miscibility and -characterization studies in the Institute for Transuranium Elements. Through the study, the volubility of minor actinides and rare earths in the U-Pu-Zr alloy as well as the miscibility of actinides were clarified. It was found that the rare earth inclusion with americium distributed uniformly along the grain boundaries in the matrix, when the content of minor actinides was up to 5 wt% [12]. In addition, the important parameters of U-Pu-Zr alloy with minor actinides and rare earths, such as metallography and phase composition, melting point, Young's modulus thermal conductivity, dilatometry, and redistribution of alloy components in a temperature gradient, were measured. The chemical interaction between the alloy with minor actinides and stainless steel, and compatibility of the alloy with sodium were also examined prior to the irradiation study. Base on the results, feasibility study was carried out for the irradiation in CEA. Fig.16 shows the composition of the irradiation fuel and the figure of the irradiation capsule in the PHENIX fast reactor.

### 3.3.2. ACTINIDE BURNER REACTOR STUDIES AT JAERI

Conceptual design study has been carried out on Actinide Burner Reactor (ABR) which is a sort of fast reactor with very

hard neutron spectrum, specially designed for efficiently burning of minor actinides (MA). Two types of reactor design with about 1000MW thermal power have been evolved, which are Na-cooled MA alloy fuelled ABR (M-ABR) and He-cooled MA nitride particle fuelled ABR (P-ABR).

Low thermal conductivity and melting point of MA alloy fuel are the limiting factors for the high MA burnup in the M-ABR. Therefore, the particle bed reactor concept was selected as the reference, which has the high efficiency in heat transfer since small particle size produces a large heat transfer surface per volume. The bed of coated particle fuel contained in double concentric porous frits is directly cooled by helium. The fuel is a microsphere of MA nitride which is coated with a refractory material such as TiN. The fuel concept of P-ABR is compared with that of M-ABR in Fig.17.

In a cold fuel concept as adopted to the P-ABR, fuel temperature is to be kept lower than one third of its melting point to reduce mass transport. Since reduced mass transport results in smaller swelling and gas release, thickness of coating layer can be minimised to give large heavy-metal density in order to obtain hard neutron spectrum in a core.

The reactor core design parameters of M-ABR and P-ABR at their equilibrium state are shown in Table.2. In the P-ABR, the neutron flux is very high and the MA burnup per year is consequently very high (17%), owing to the efficient heat removal characteristics of particle fuel. A fuel cycle based on the P-ABR is being studied.

Reliable database of MA is indispensable to increase the accuracy of the ABR design. The integral experiments have been conducted in the fast critical facility (FCA) of JAERI to evaluate and refine neutron cross-sections for MA, and accuracy of nuclear data of MA compiled in JENDL-3 nuclear data library has also been examined. The phase diagram and thermodynamic studies on MA compounds are being carried out under the cooperation with the Oak Ridge National Laboratory (ORNL).

#### 3.4. TWINSMUTATION WITH ACCELERATORS

A transmutation system driven by a proton accelerator is studied and development of an intense proton accelerator is under way at JAERI. At PNC, accelerator assisted or fusion driven 'systems for Sr-90 and Cs-137" transmutation are evaluated and an intense electron accelerator is under development.

##### A. STUDIES OF A TRANSMUTATION SYSTEM DRIVEN BY AN INTENSE PROTON ACCELERATOR AT JAERI

A conceptual design study on proton accelerator-driven transmutation plant has been made at JAERI, which consist of a high intensity proton accelerator, a sodium-cooled subcritical core, with MA alloy fuels and a tungsten target. As shown in Fig.18, the central tungsten target is surrounded by the annular MA fuel region, which in turn is surrounded by radial and

axial reflectors of stainless steel. A proton beam window is located right above the target region and is designed to be replaceable. High energy protons are injected vertically downward through the window into the target. The tungsten target acts as a spallation neutron source. The principal design parameters of the plant is shown in Table.3. The table shows that the plant transmutes about 250kg of MA by fission and produces 820MW thermal power by using a 1.5GeV proton beam with a current of 39mA. Electric power of 246MW is generated with a conventional steam turbine, and supplies sufficient electricity to power the accelerator.

Study is also in progress on the system with a chloride molten salt target as shown in Fig.19. The system has a potential to establish a continuous MA transmutation concept with on-line separation facilities for the spallation and fission products, and to avoid the complicated processes being followed by batch reprocessing for solid spent fuels and/or targets. The preliminary system design parameters and the schematic image of the separation system are given in Table.4 and Fig.20, respectively. In addition, study on transmutation of Tc and I by using accelerator is also under way, in which Bi-Pb fluid target surrounded by a heavy water blanket is preliminarily used.

In order to verify the reliability of the high-energy nuclear data and the computer codes developed at JAERI, the integral experiment with a lead bulky target and 500MeV protons has been carried out since 1990. The experiment to use tungsten and TRUS will also be conducted in near future.

In order to conduct various engineering tests for the accelerator-driven transmutation plant, a plan to construct the Engineering Test Accelerator (ETA) with a proton energy of 1.5GeV and a current of 10mA shown in Fig.21 has been proposed. As the first step of developing the ETA, the Basic Technology Accelerator (BTA) with the proton energy of 10MeV and the current of 10mA is ready to be constructed to study a lower energy portion of the ETA.

## B. STUDIES OF TRANSMUTATION BY USE OF ACCELERATOR AT PNC [13] [14] [15]

The transmutation methods of Cs-137 using accelerators have been studied. The transmutation rate and the transmutation energy have been evaluated by Monte Carlo calculation codes to examine the feasibilities. The new transmutation methods have been also proposed and studied to achieve both high transmutation rate and low transmutation energy.

A high current electron linear accelerator with an electron energy of 10MeV and a maximum current of 100mA (average 20mA) is under development. A schematic drawing of the accelerator technology development facility is shown in Fig.22.

Fabrication of prototype components is in progress. Prototypes of main components such as an accelerating tube and a klystron were assembled and their RF power test is being conducted. Reflecting the test results, the design will be modified.

fied.

The construction of the whole linac system is scheduled to start in 1993, and the first operation is scheduled in 1996.

#### 4. CONCLUSIONS

Various R&D activities under the framework of the OMEGA program in Japan were presented here. These activities will be reviewed by Atomic Energy Commission, and the partitioning and transmutation technology will become more and more worthy to reduce the long-term burdens of nuclear waste disposal and to enhance the effective utilization of resources. However, it should be recognized that the OMEGA program is not to seek a short-term alternatives for established or planned fuel cycle back-end policies, but is conceived as a research effort to pursue benefits for future generations through long-term basic R&D .

## REFERENCES

- [1] T.kawata et al.;" Preliminary Study on the Partitioning of Transuranium Elements in High Level Liquid Waste" , First OECD/NEA Information Exchange Meeting on Separation and Transmutation of Actinides and Fission Products, Mito(1990).
- [2] Y.Wada et al.; "Recovery and Utilization of Valuable Metals from Spent Nuclear Fuel", Proc. 1989 Joint International Waste Management Conf., Kyoto(1989).
- [3] M.Horie "Super-High-Temperature of High-Level Liquid Waste", Trans. Am. Nucl. Soc. ,62, 111 (1990)
- [4] T.Inoue et al.;"Development of partitioning and transmutation technology for long-lived nuclides", Nucl. Technol. 93 206 (1991)
- [5] M.Sakata et al.;"Equilibrium distribution of rare earth elements between molten KCl-LiCl eutectic salt and liquid cadmium", J.Nucl. Mater. 185 56 (1991)
- [6] M.Sakata et al.;"Development of pyrometallurgical partitioning of transuranium elements prior to transmutation in metallic fuel FBR", Int. Conf. on Fast Reactor and Related Fuel Cycles, Ott.28 to Nov.1, 1991, Kyoto, Japan (1991)
- [7] J.J.Roy et al.;"Standard potential of lanthanide and actinide trichlorides in molten eutectic LiCl-KCl electrolyte", Mater. Sci. Forum 83 547 (1991)
- [8] M.Yamaoka et al.;"Feasibility Study of TRU Transmutation LMFBRs", Int. Conference on Fast Reactors and Related Fuel Cycles, Vol.III, Kyoto, Oct. 28 - Nov.1 (1991)
- [9] M.Yamaoka et al.;"Design Study of a Super Long Life Core Loaded with TRU Fuel", Int, Conference on Design and Safety of Advanced Nuclear Power Plants, P3-3, Tokyo, Oct. 25-29 (1992)
- [10]A.Sasahara et al.;" Transmutation of transuranium nuclides with FBR", Proc. Int. Conf. Nuclear Data for Science and Technology, Mito, Japan, May 30 - June 3, 1988, p .955 (1988)
- [11]A.Sasahara et al.;"The status of minor actinide transmutation study in CRIEPI", Int. Working Group ON Fast Reactors Specialist Meeting on Use of Fast Breeder Reactors for Actinide Transmutation, Obninsk, IPPE, Sep.22-24, 1992
- [12]T.Inoue et al.;"Characterization of alloy fuels with minor actinides", Trans. Am. Nucl. Sot. , 64 552 (1991)
- [13]T.Kase et al.; "Transmutation of Cs-137 Using Muon Catalyzed Fusion Reaction", Muon Catalysed Fusion, 6 (1991) 521
- [14]K.Konashi et al.;"New Transmutation Method by Using Moving Target", Fusion Technology, 20 (1991) 664
- [15]K.Haga et al.;"Progress of Super-Power CW Electron Linac Development in PNC", to be submitted to the Second OECD/NEA Information Exchange Meeting on Separation and Transmutation of Actinides and Fission Products, Chicago (1992)

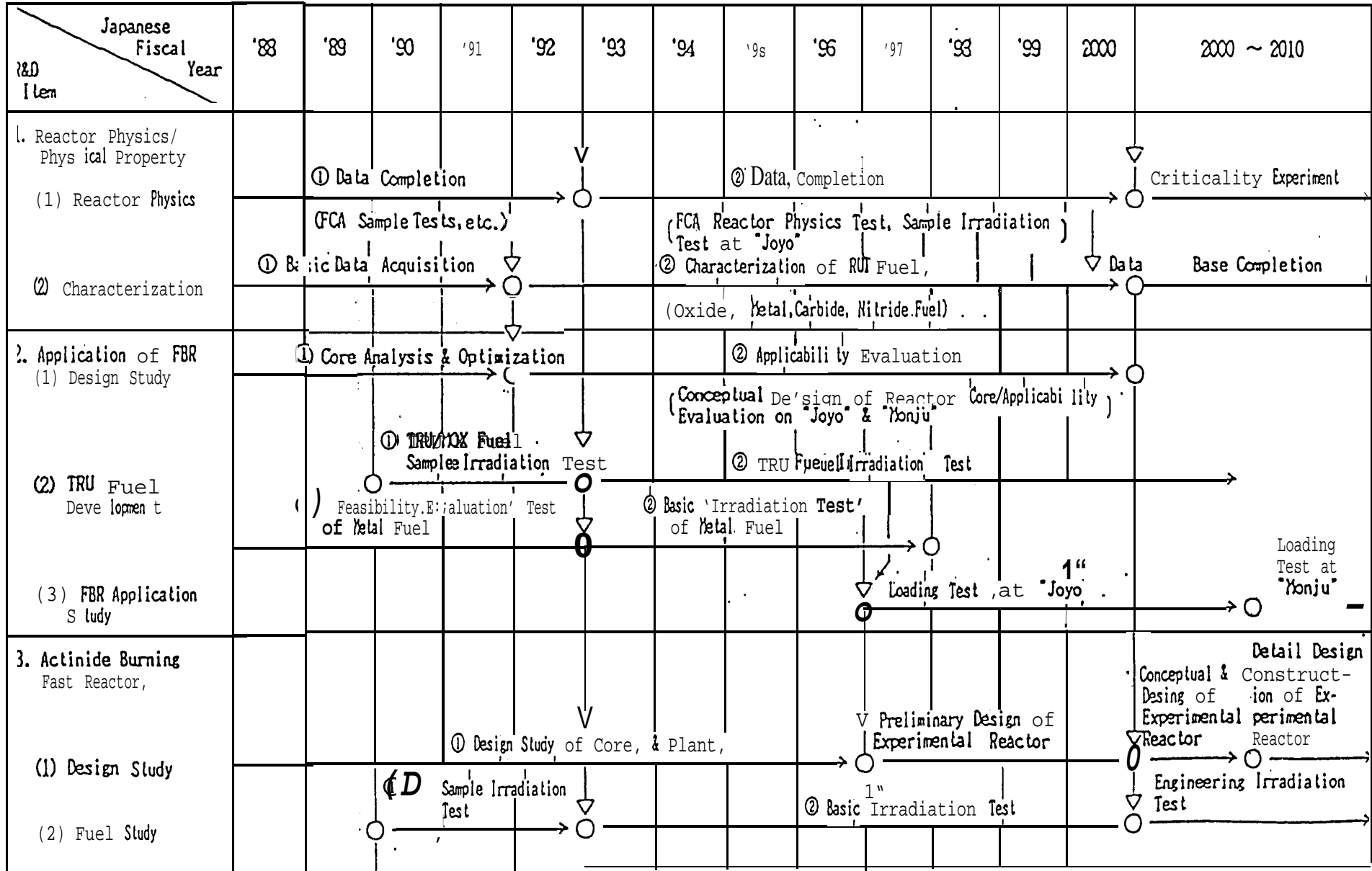
Fig. 1. R&D SCHEDULE OF NUCLIDES PARTITIONING

① Phase I ② Phase II  
V: Check & Review

R&D Item	Japanese Fiscal Year	'88	'89	'90	'91	'92	'93	'94	'95	'96	'97	'98	'99	2000	2000 ~ 2010		
1. R&D on Nuclides Partitioning of Reprocessing HLLW	(Aqueous process)			① General Basic Test							▽				▽	Pilot-scale Test/ Practicability Test	
	(Non-Aqueous process)			(Process Establishment/ Preliminary Engineering Test)	① Technology Verification Test						○	② Process Engineering Test					
2. R&D on Recovering Technology for Useful Metals Contained in Insoluble Residue					① General Basic Test						▽					▽	
				(Basic Recess Test, Basic Engineering Test)							○	② Demonstration Test				○	Practicability Test
3. R&D on Utilization Technology of Separated Elements				① Basic Test,							▽		② General Engineering Test			▽	Pilot-scale Test/ Practicability Test
				(Separation and Purification, Utilization, Fabrication)							○					○	

Fig. 2 R&D SCHEDULE OF TRANSMUTATION BY REACTOR

① Phase I ② Phase II  
V: Check & Review



88

Fig. 3 R&D SCHEDULE OF TRANSMUTATION BY ACCELERATOR

① Phase I ② Phase II  
 ▽: Check & Review

R&D Item	Japanese Fiscal Year	'83	'89	'90	'91	'92	'93	'94	'95	'96	'97	'98	'99	2000	2000 ~ 2010	
1. Proton Accelerator	(1) Design Study		① Computer Code Development			▽				② Conceptual Design					▽ Design Optimization	Practicability Study
	(2) Target System Study		① Medium-scale Reactor Physics Experiment			▽				② Large-scale Reactor Physics Experiment					▽ Pilot-scale Test	↻
	(3) Construction of Accelerator			① Elemental Technology Development		▽				② Construction of Accelerator (Proton Lineac)					▽ Development of Elemental Technology for Practical-use Accelerator	↻
										(High Current/Higher Energy)						(10mA-1.5GeV Class)
																(300mA-1.5GeV Class)
2. Electron Accelerator	(1) Design Study		① Computer Code Development			▽				② Conceptual Design/Detail Design					▽ Design Optimization	
	(2) Target System Study		① Basic Experiment 'for' Reactor Physics Data			▽				② Conceptual Design/Detail Design					▽	
	(3) Construction of Accelerator			① Elemental Technology Development		▽				② construction of Accelerator					▽ Pilot-scale Test/ Hybrid System	
										(Beam Stabilization)						(10A-100MeV Class)
																(1A-100MeV Class)



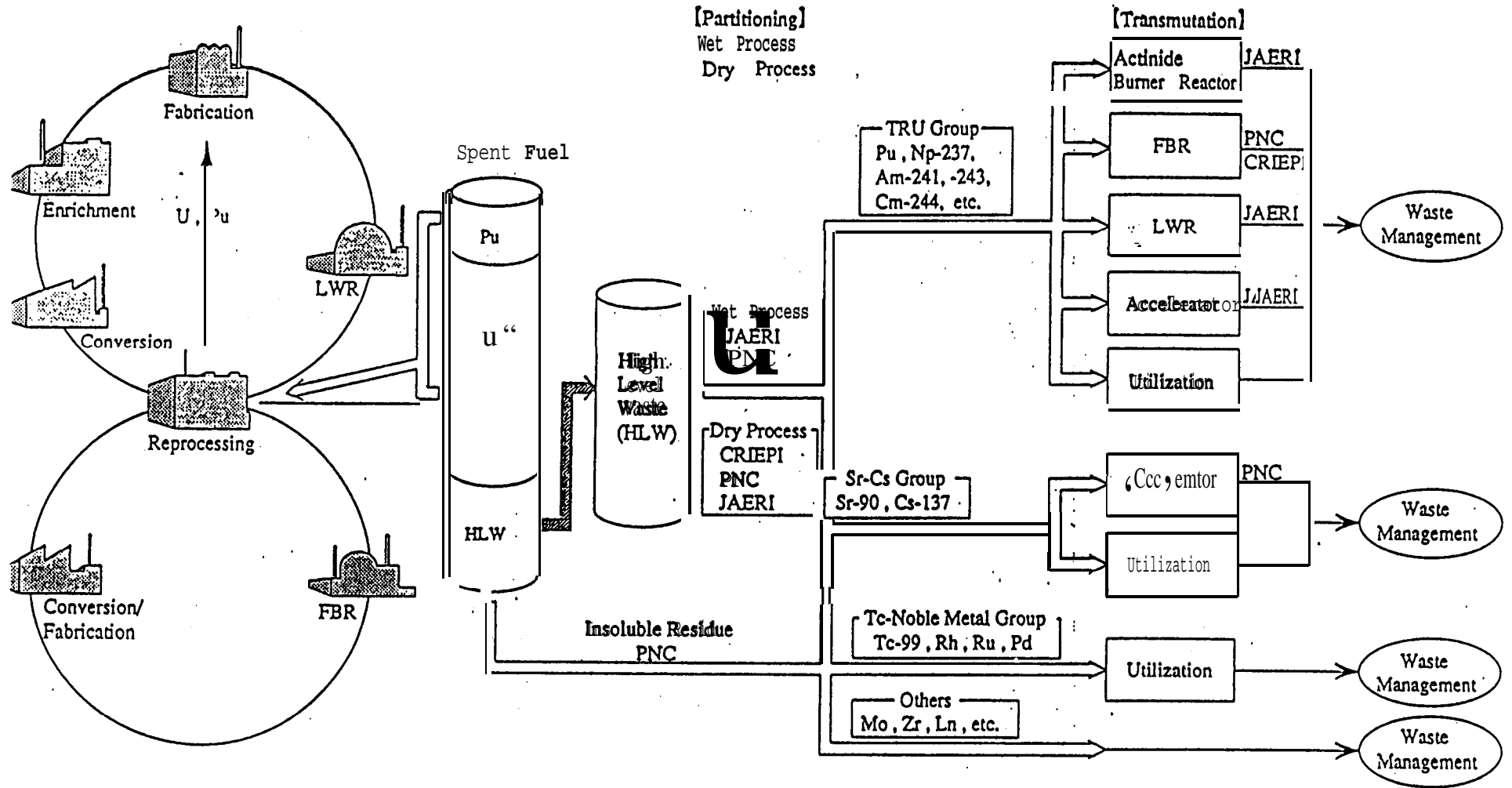


Fig. 4 P-T R/D ACTIVITIES UNDER OMEGA PROGRAM

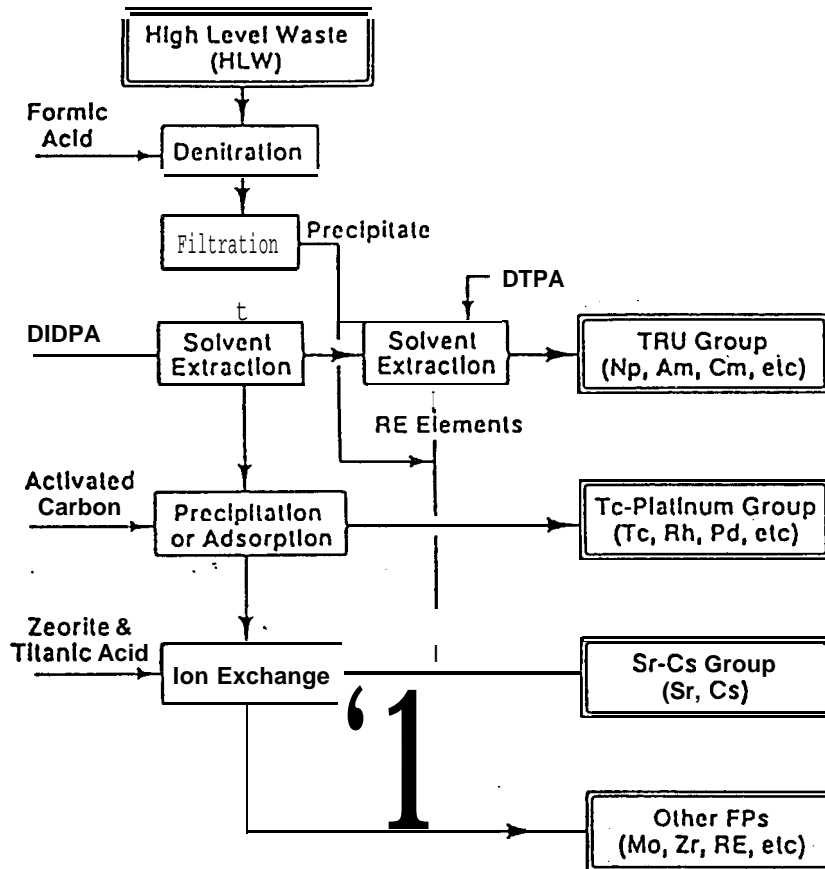


Fig.5 FLOW CHART OF ADVANCED PARTITIONING PROCESS DEVELOPED BY JAERI

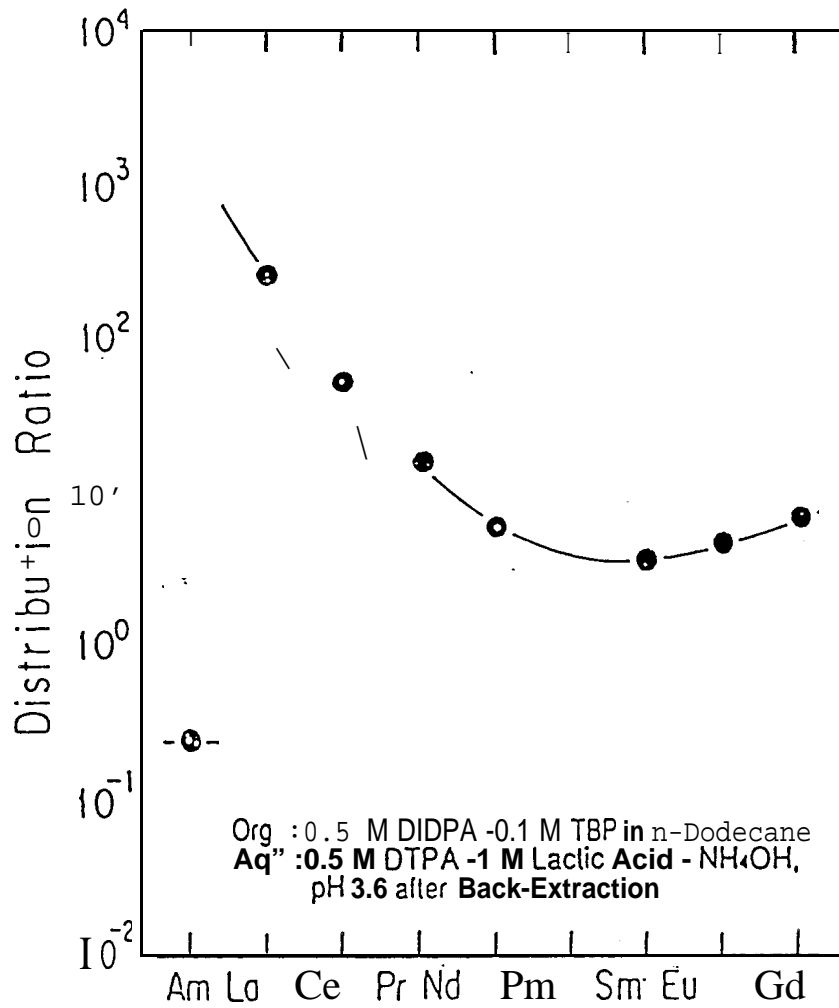
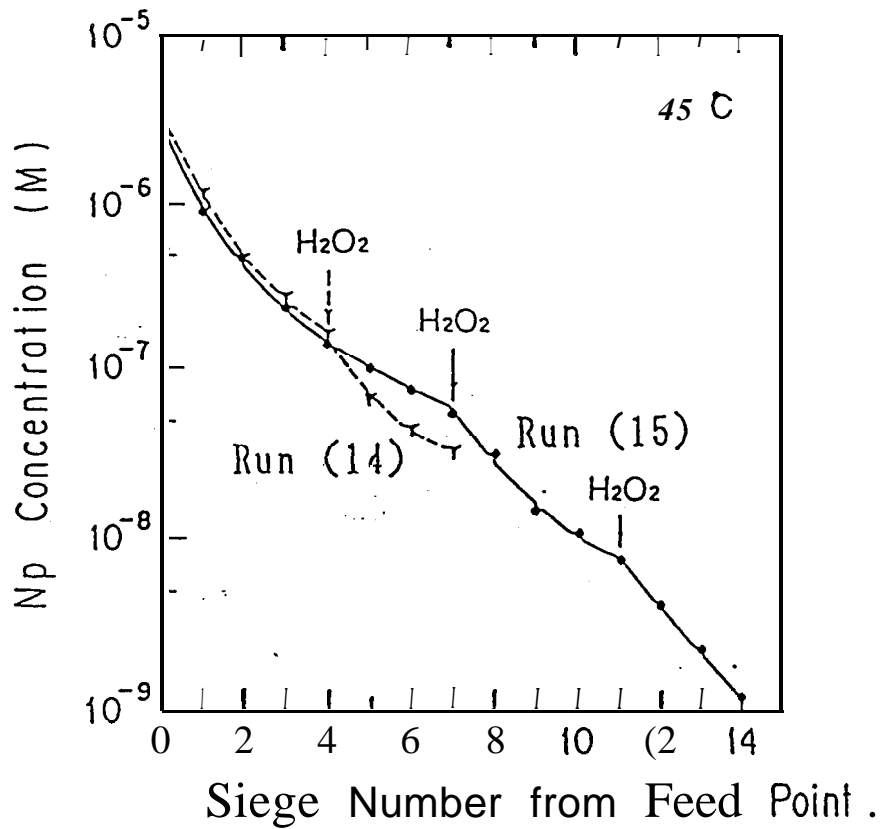
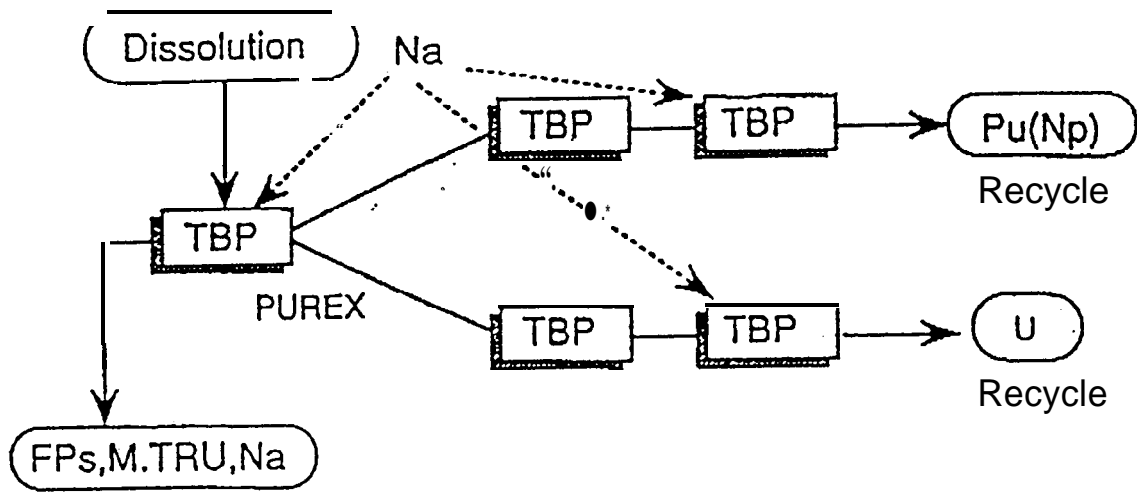


Fig.6 COMPARISON OF DISTRIBUTION RATIOS BETWEEN Am AND RARE EARTHS

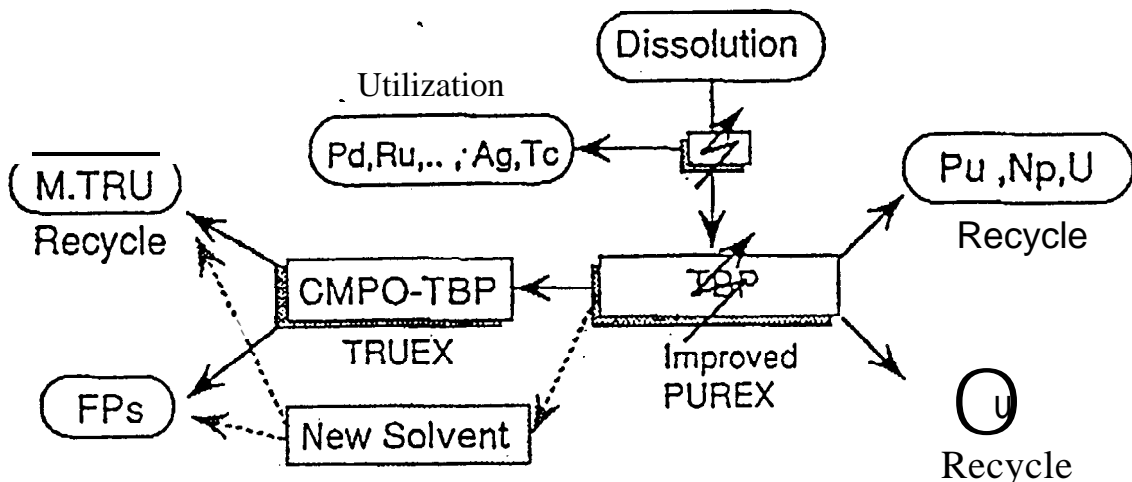


Np Recovery Fraction = 99.0 % for Run (14) (7 stages)  
 > 99.96% for Run (15) (14 stages)

Fig. 7 CONCENTRATION PROFILE OF NP IN AQUEOUS PHASE AT THE EXTRACTION SECTION



Present Reprocessing Concept



⚡ : Salt - Free method  
( Electrochemical!, Photochemical!)

Advanced Reprocessing Concept

Fig. 8 **ADVANCED REPROCESSING SYSTEM**

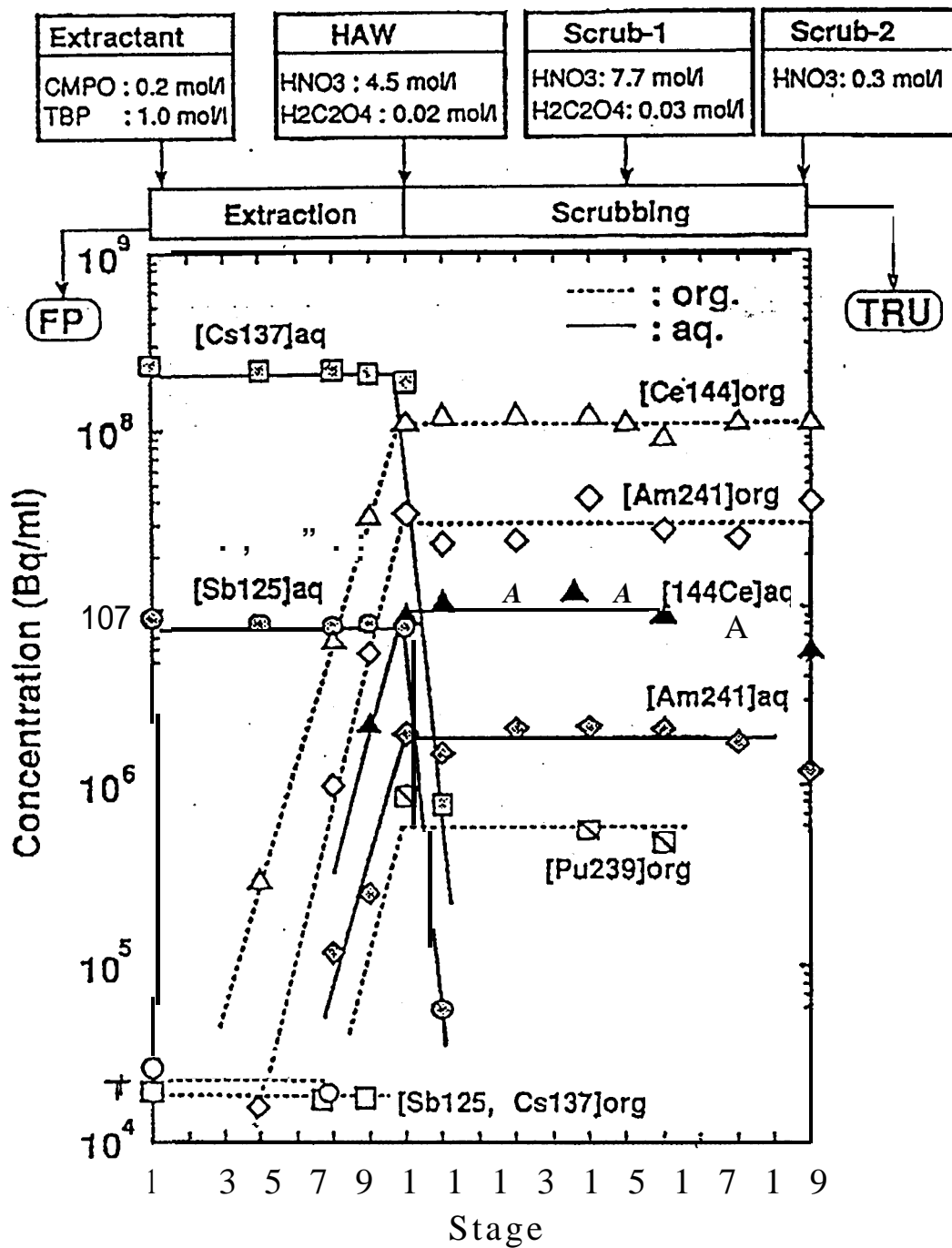


Fig.9 CONCENTRATION PROFILES OF Cs, Sb, Ce, Am AND Pu IN MIXER-SETTLER EXPERIMENT WITH ACTUAL HLW

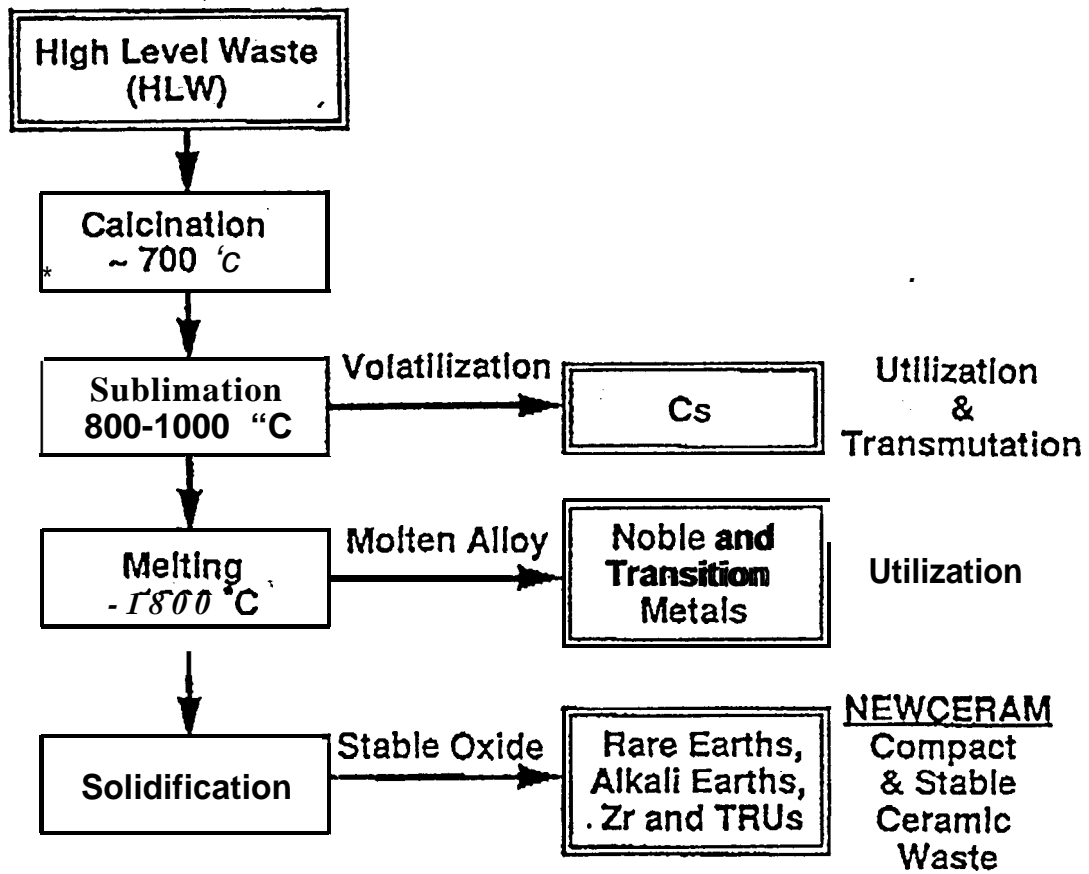


Fig.10

**SUPER HIGH TEMPERATURE TREATMENT OF HLW**

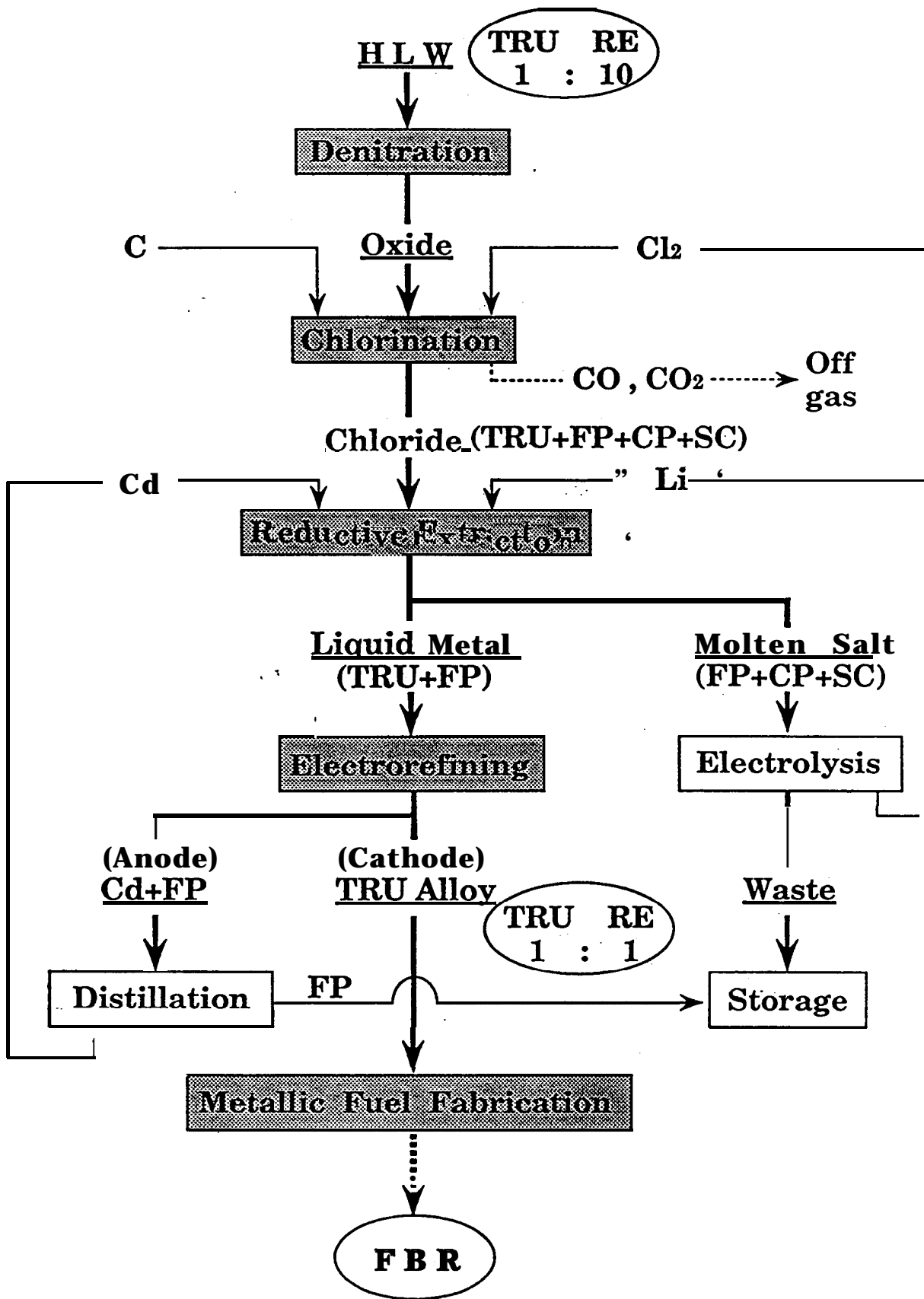


Fig. 11 Flow diagram of pyrometallurgical process for partitioning of TRUS from HLW in CRIEPI



Table 1. The Specification of Facilities and the Operating Conditions.

(1) for Denigration

Conditions for Design

HLW	<b>2,0001 / Day</b>
Condition	Temperature <b>700 °C ~ 800 °C</b> Absorption Efficiency of Microwave 0.7 Microwave Power per Unit Volume of Oven <b>0.03 KW / l (Tentative Value)</b>

Specifications

Major Apparatus	Oven 2 m W X 2 m L X 2 m H Microwave Oscillator 100 KW X 2
Number of Batch	4 Batches/Day
Hot Cell	5 m W X 8 m L X 4 m H

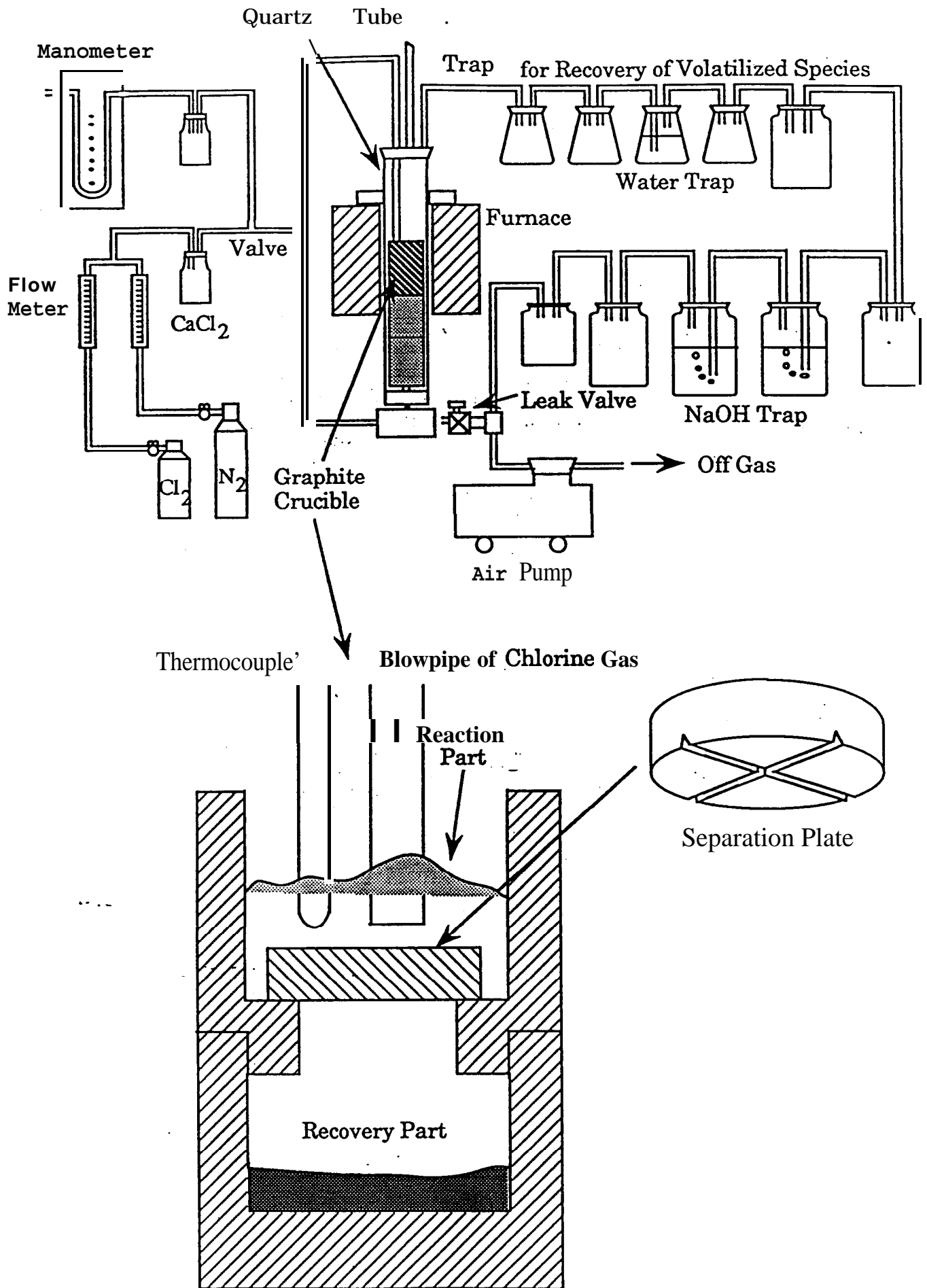
(2) for Chlorination

Conditions for Design

Denigrated Material	<b>300 kg/ Day</b>
Conditions	Temperature <b>780 °C</b> Carbon 36 kg Chlorine Gas 1100 kg

Specifications

Major Apparatus	Chlorination Furnace 1.2 mD X 2.3 mH Condenser 0.9 mD X 1.9 mH Powder Blender 1.8 mW X 0.5 mL X 0.45 mH
Number of Batch	3 Batches/Day
Hot Cell	5.5 m W X 6 m L X 8.5 m H



**Fig. 12** Schematic Illustration of Chlorination Test Apparatus

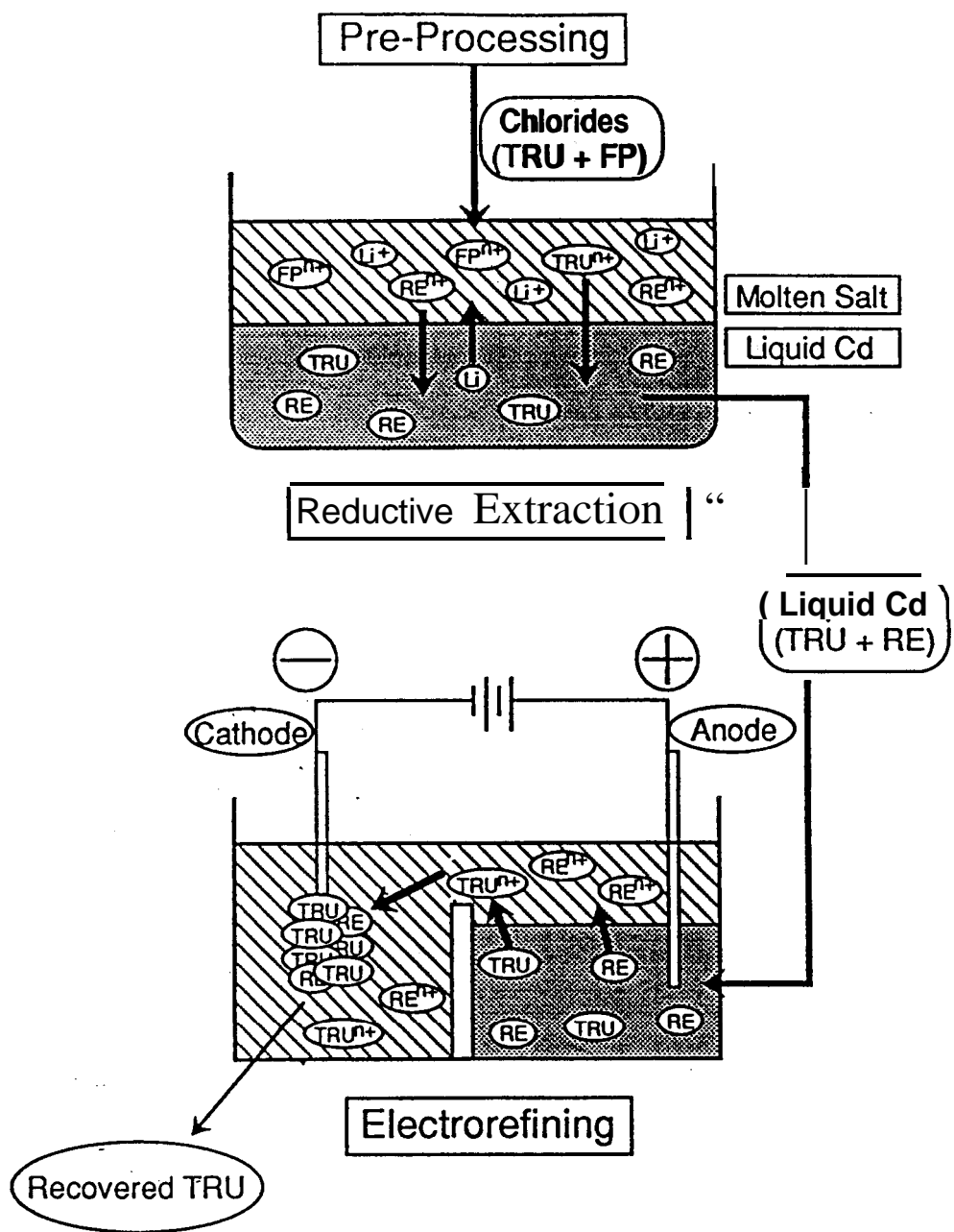
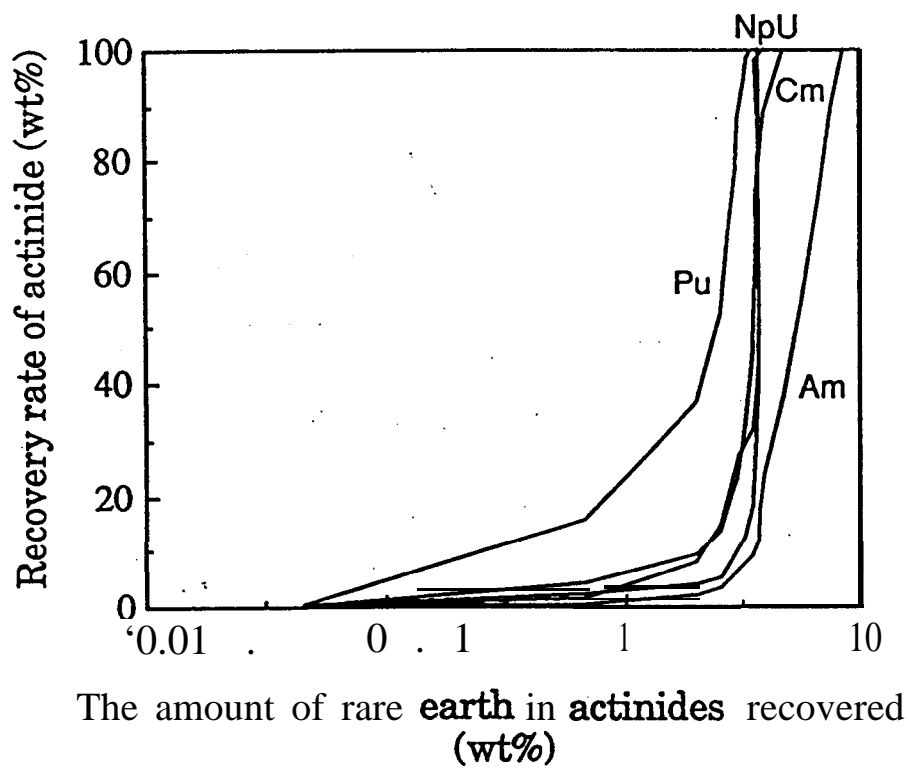
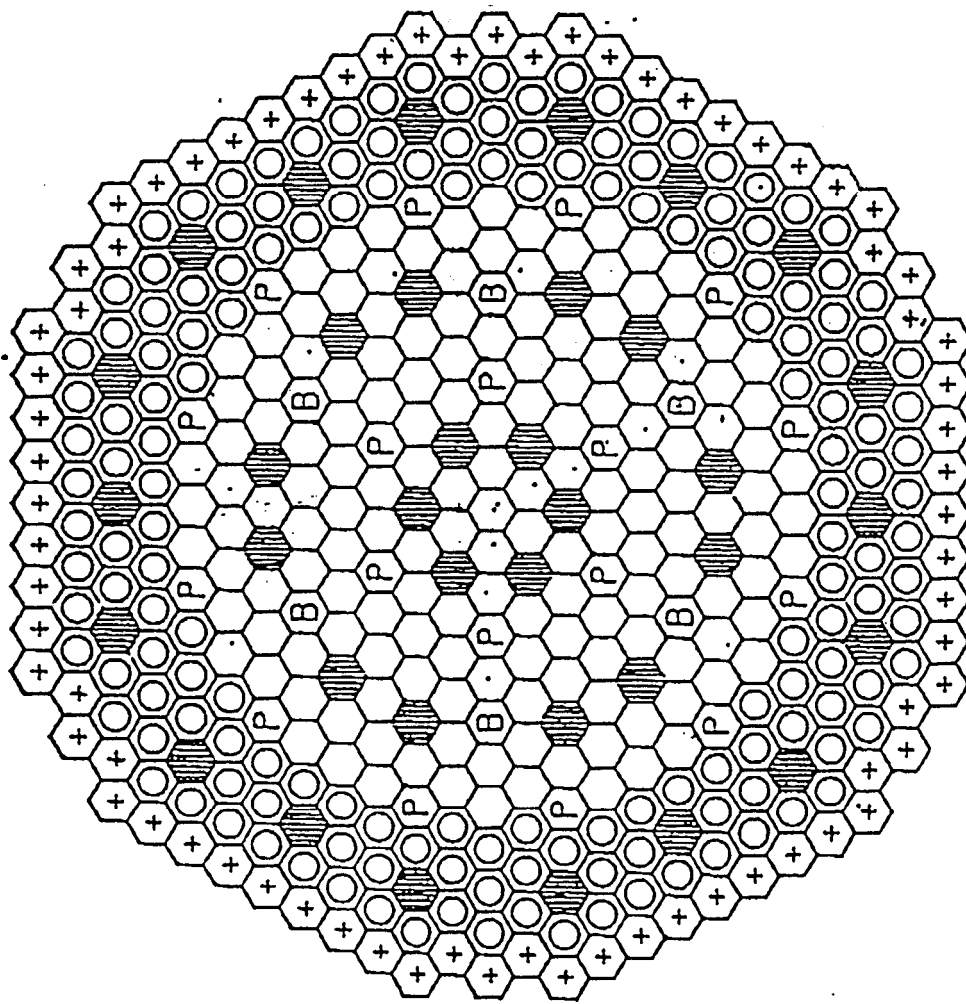


Fig. 13 Schematic Illustration of Pyrometallurgical Partitioning Process



**Fig.14** Prediction of recovery rate of **actinides** as function of the amount of rare earth contained in actinides recovered by **electrorefining** using Cd anode



- Inner Core
- ⊕ Outer Core
- Ⓟ Primary Control Rod
- Ⓟ Backup Control Rod
- ⊕ Radial Blanket
- ▨ Target Fuel (39)

● Fraction of Target Fuels = 11% (39 / 355)

Fig. 15 Typical Loading Pattern of Target Fuels

Actinide Fuel ;

U 19Pu 10Zr 0.6Am 1.2Np 0.2Cm 2RE

U 19Pu 10Zr 1.6Am 3.0Np 0.4Cm 5RE

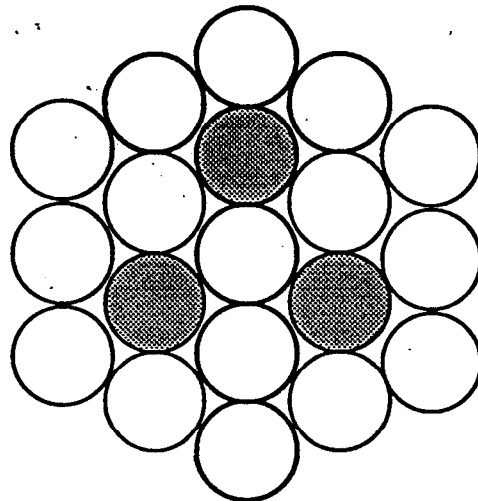
Reference Fuel ;

U 19Pu 10Zr

Operating conditions

$T_{\text{metal}} \leq 750 \text{ } ^\circ\text{C} \ll 1150 \text{ } ^\circ\text{C}$  (fuel melting)

$T_{\text{clad}} \leq 520 \text{ } ^\circ\text{C} \ll 725 \text{ } ^\circ\text{C}$  (eutectic  
U-Pu-clad)

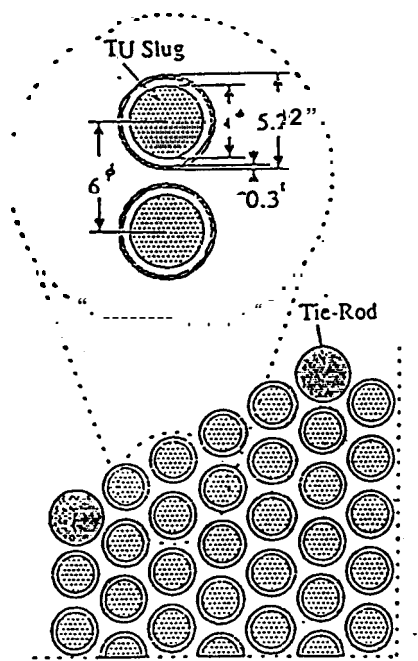


**O** :3 metal pins

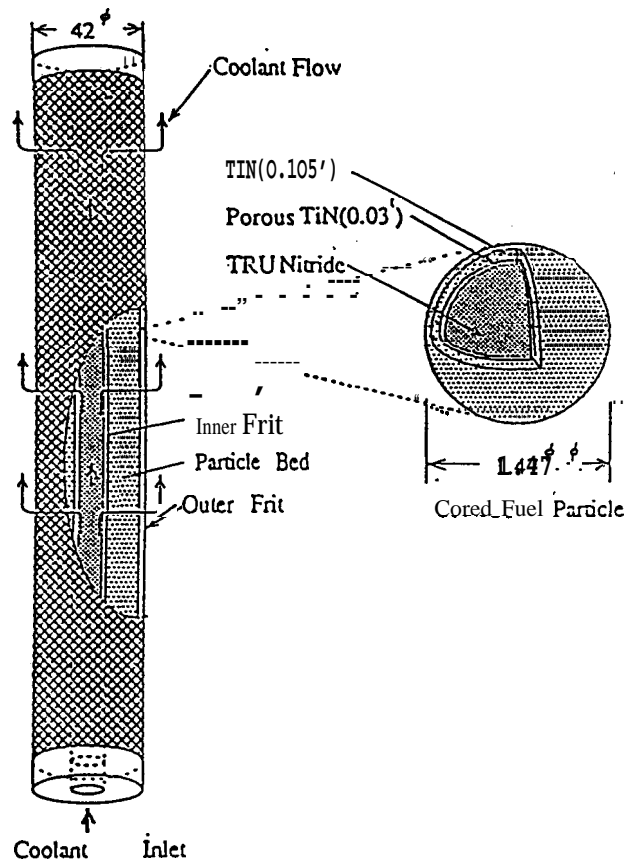
**●** :16 oxide pins

Irradiation capsule

Fig. 16 Figure of Irradiation Study



Ductless Fuel Assembly of M-ABR



Particle Fuel of P-ABR

Fig. 17 Fuel Concept of ABR (unit mm)

Tab 1 e. 2 Reactor design parameters of Actinide Burner Reactors

	M-ABR <sup>1)</sup>	P-ABR <sup>2)</sup>
Fuel concept material	pin-bundle IC <sup>3)</sup> :Np-22Pu-20Zr OC:AmCm-35Pu-5Y	coated particle (66NpAmCm-34Pu) <sub>1,2</sub> N <sub>1,2</sub>
MA initial loading, <sup>4)</sup> kg Np/Am,Cm/Pu	666 “ 255/199/212	2065 765/598/702
Reactor power, MWth	1 7 0	1 2 0 0
Coolant material	S o d i u m	Helium
velocity, m/s	8	total flow, kg/s inlet pressure, MPa pressure drop, kPa
inlet temperature, °C	300	1088 10 13
outlet temperature(core max), °C	IC:484 OC:440	127 340
Fuel temperature, °C max <sup>5)</sup>	IC:834 OC:809	722
Clad temperature, °C max <sup>6)</sup>	1C517 OC:484	Frit temperature, max 5 6 0
Neutron flux, 10 <sup>15</sup> n/cm <sup>2</sup> ·sec <sup>-1</sup>	IC:4.1 OC:3.4	8.4
Neutron fluence (E>0.1MeV), 10 <sup>23</sup> n/cm <sup>2</sup>	IC:2.2 OC: 1.7	2.2
Core averaged mean neutron energy, keV	IC:766 OC:785	743
<b>Reactivity (% Δk/k)</b>		
Na-void reactivity/core	2.52	
Doppler reactivity/core (Δt=300°C)	-0.01	-0.01
<b>Kinetic parameters</b>		
β <sub>eff</sub>	135X10 <sup>3</sup>	1.72X10 <sup>7</sup>
L <sub>p</sub> , sec	6.84 X 10 <sup>4</sup>	10.8X10 <sup>5</sup>
Cycle length, full-power days <sup>7)</sup>	730	300
MA transmutation, %/cycle	26.0	25.3
MA burnup, %/cycle	17.8	17.3

1) M-ABR: MA metallic fuel burner reactor

2) P-ABR : MA particle fuel burner reactor

- 3) IC: Inner Core, OC: Outer Core

4) After 1st cycle, only Np, Am, Cm are added.

5) Predicted melting point of fuel 900°C for M-ABR

Max. allowable temp. of fuel 727°C (1/3 of M.P. 3000K) for P-A BR

6) Max. allowable temp. of cladding/frit (HT-9) 650°C

7) Fuel irradiation time



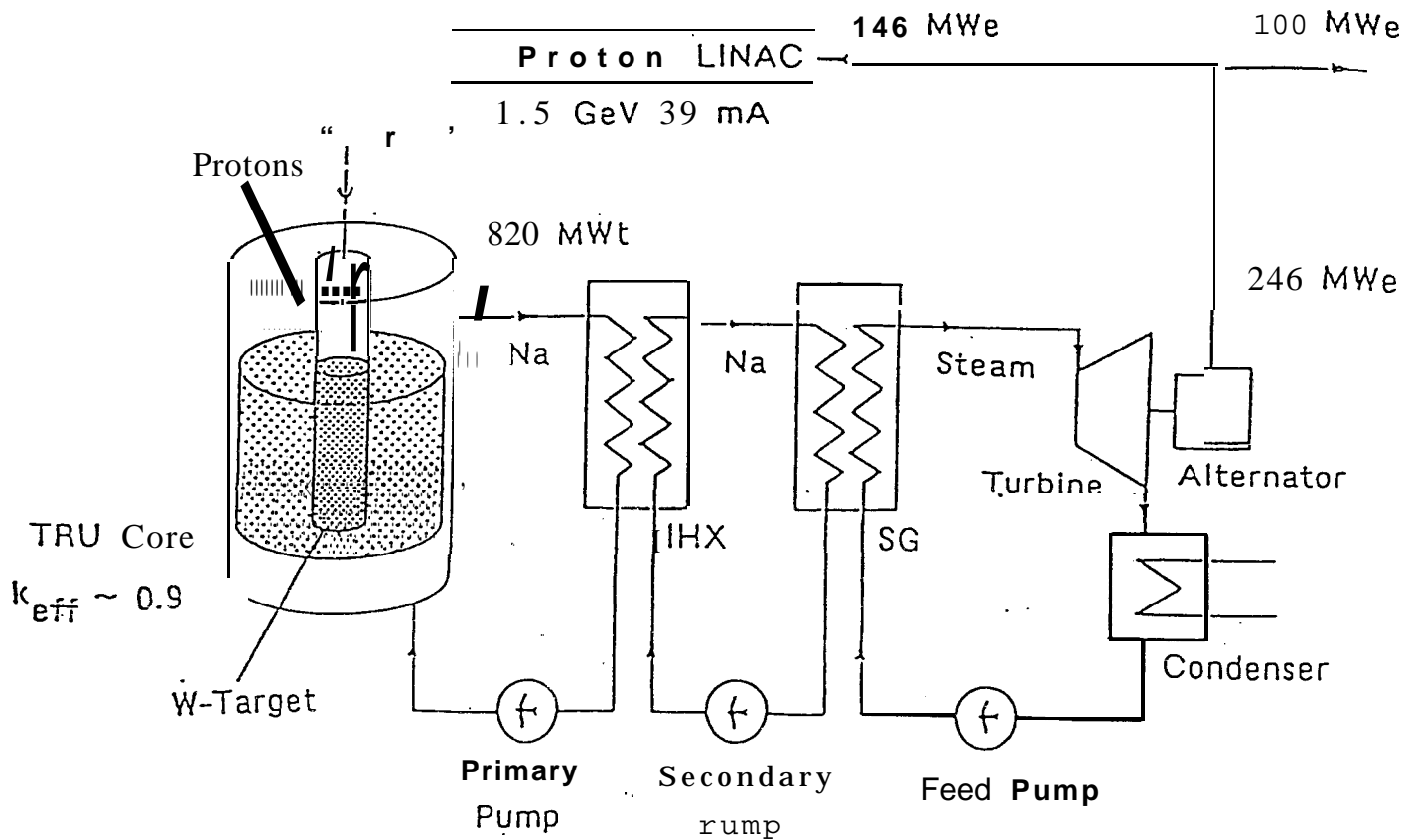


Fig.18, TRANSMUTATION PLANT CONCEPT DRIVEN BY AN INTENSE PROTON ACCELERATOR

Table. 3 Design parameter of minor actinide transmutation system assisted by an intense proton LINAC

Proton beam current		39 mA
Actinide inventory		3160 kg
Effective neutron multiplication factor		0.89
No. of neutrons per incident photon		40 n/p
No. of fissions"	(>15MeV)	0.45 f/p
	(<15MeV)	100 f/p .
Neutron flux		$4 \times 10^{15}$ n/cm <sup>2</sup> . s
Mean neutron Energy		690 keV
Burnup		250 kg/y
<hr/>		
MA thermal output	fuel	800 MW
	clad	20 Mw
	total	820 MW
<hr/>		
Power density .	maximum	930 MW/m <sup>3</sup>
	average	400 MW/m <sup>3</sup>
Linear power ratio	maximum	61 kW/m
	average	27 kW/m .
Coolant temperature	maximum	473 °C
	average	430 °C /
Maximum, temperature	fuel	
	center	890 °C
	surface	548 °C
	clad	
	inside	528 °C
	outside	484 °C
<hr/>		

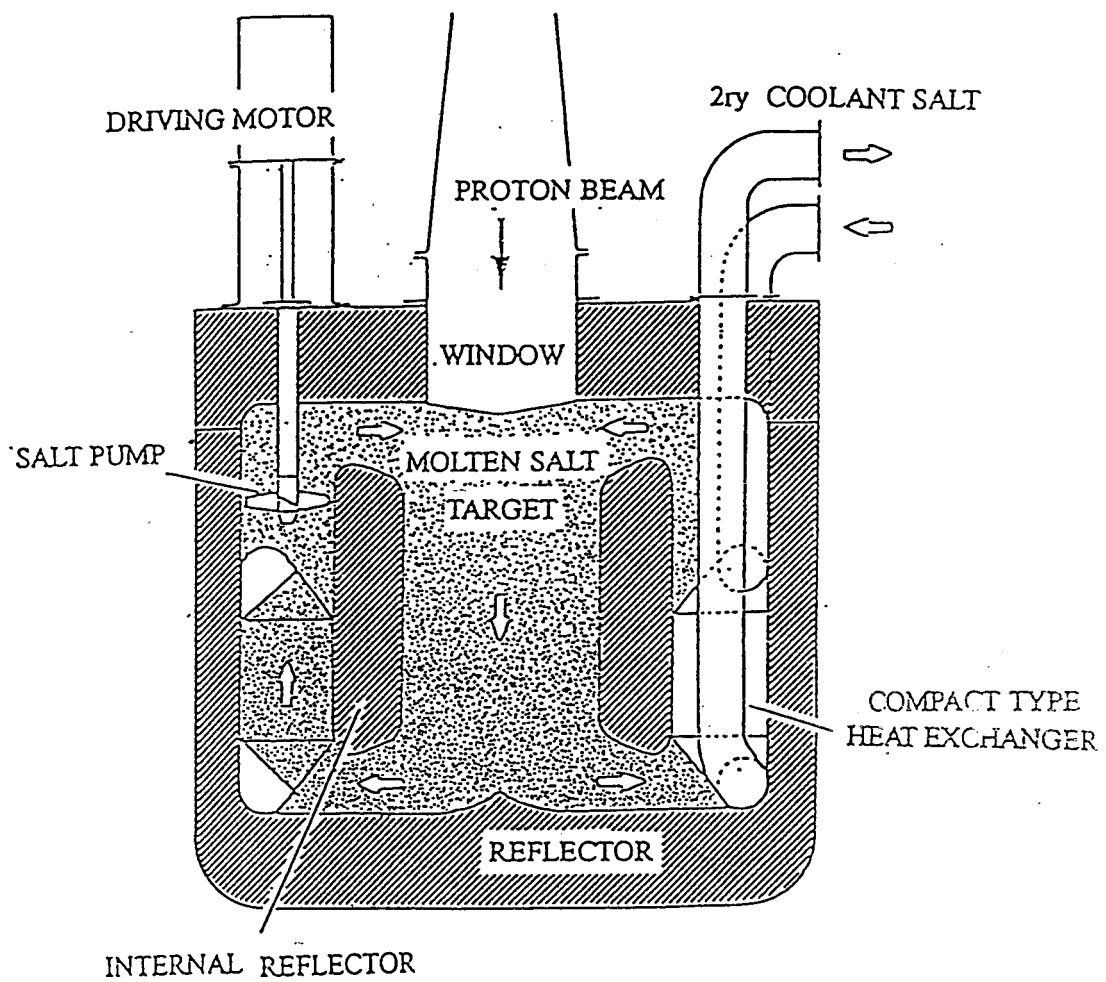


Fig. 19 Concept of the molten salt target system

**Table. 4 Design parameters of target system and results of the calculation for MA transmutation**

Target	Salt
	Composition : $63\text{NaCl}-36(\text{Pu}, \text{MA})\text{Cl}_3$ (Pu : MA = 15: 85) (MA : Np, Am, Cm)
	Volume: 2.6 m <sup>3</sup>
	Weight : 9 ton
	TRU inventory : 5 ton
	Effective Multiplication Factor $k_{\text{eff}} = 0.92$
	Beam Power : 1.5 GeV, 25 mA
	Thermal Output : 800 MW
	Transmutation Rate of MA : 250 kg/y

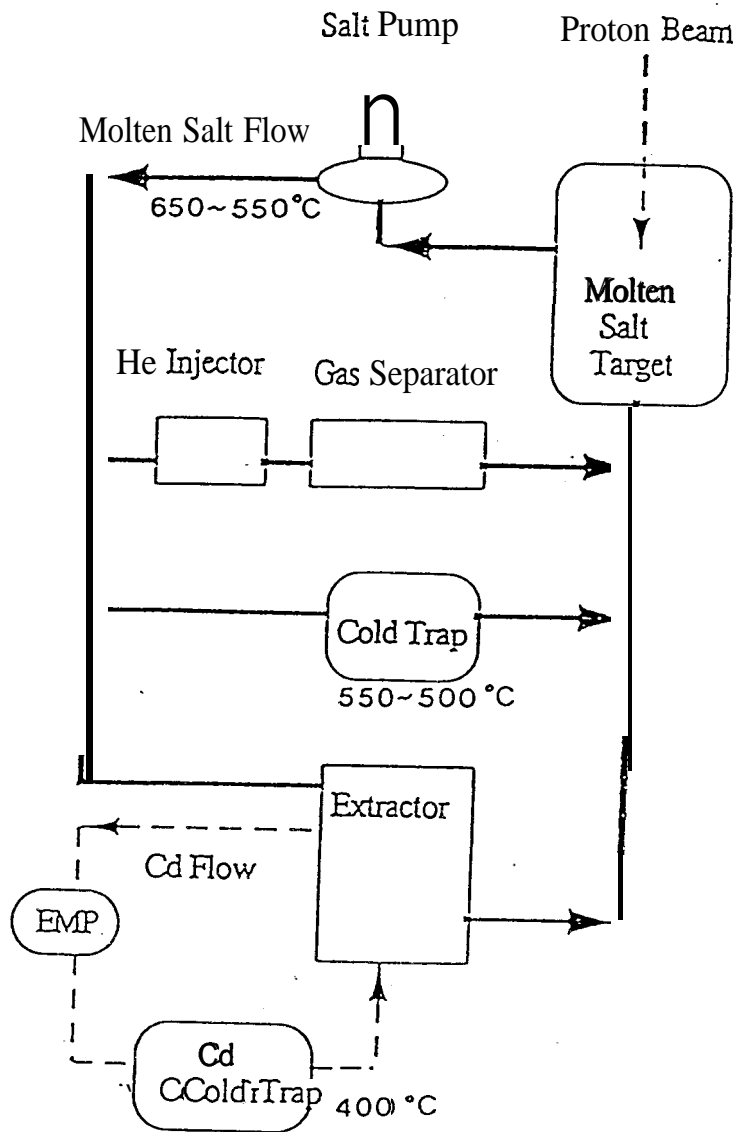


Fig. 20 Concept of the on-line separation system

Development for Basic Accelerator Technology

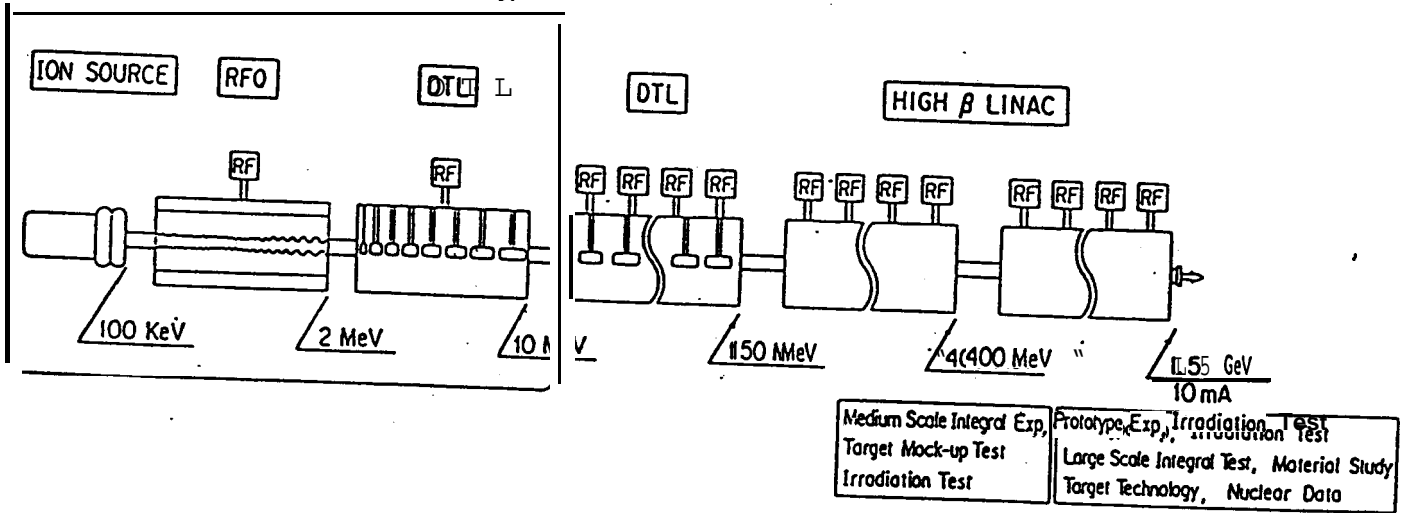


Fig. 21' Concept of Engineering Test Accelerator (ETA)

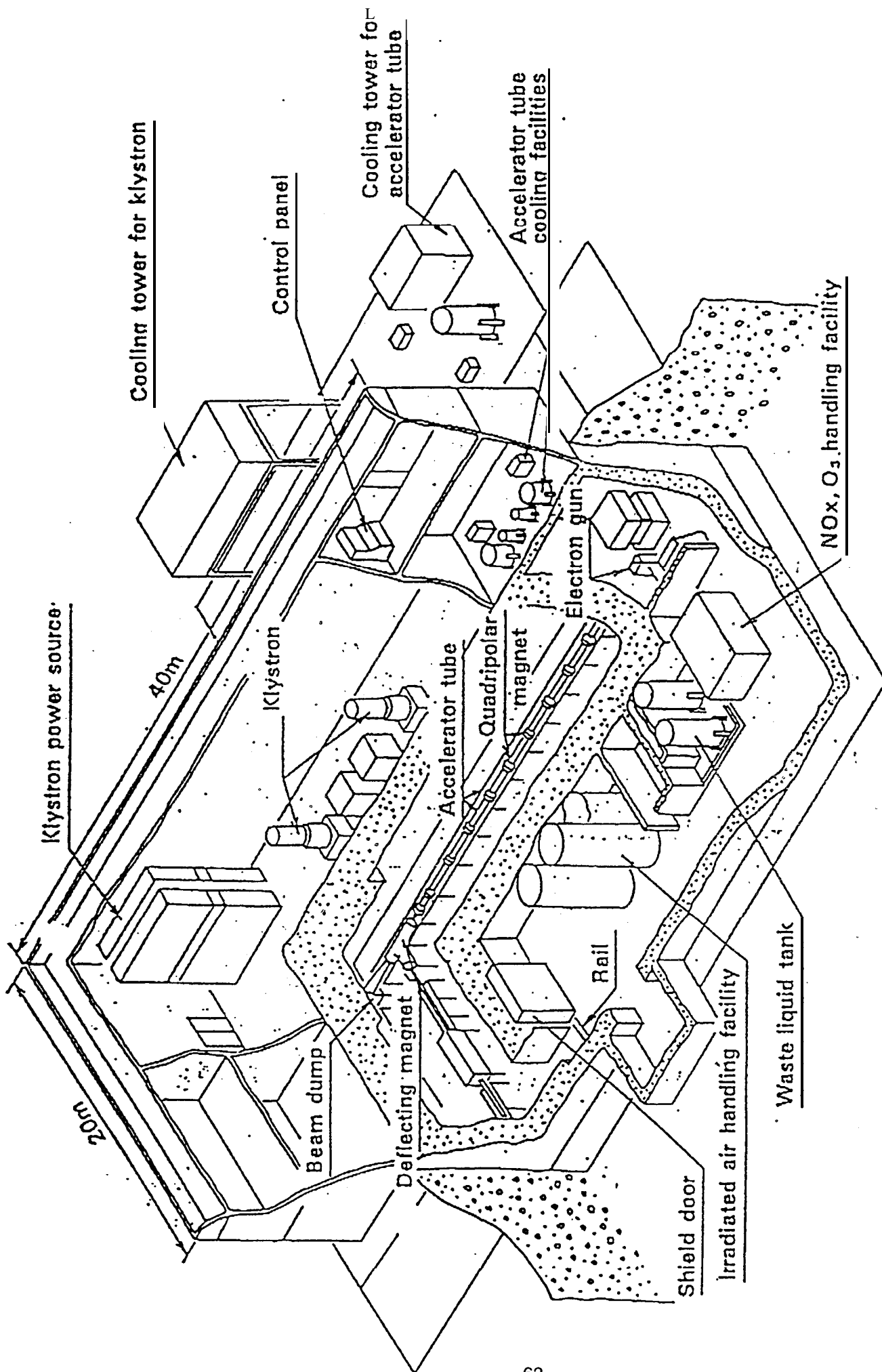


Fig. 22

HIGH CURRENT ELECTROSTATIC LINAC UNDER DEVELOPMENT AT PNC