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PROGRAM OF NUCLEAR CRITICALITY SAFETY EXPERIMENT AT JAERI

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Iwao KOBAYASHI, Shoichi TACHIMORI,
Isao TAKESHITA, Takenori SUZAKI
and Nobuaki OHNISHI

日本原子力研究所
Japan Atomic Energy Research Institute

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Department of Nuclear Safety Research
Tokai Research Establishment, JAERI

(Received November 9, 1983)

JAERI is promoting the nuclear criticality safety research program, in which a new facility for criticality safety experiments (Criticality Safety Experimental Facility : CSEF) is to be built for the experiments with solution fuel. One of the experimental researches is to measure, collect and evaluate the experimental data needed for evaluation of criticality safety of the nuclear fuel cycle facilities. Another research area is a study of the phenomena themselves which are incidental to postulated critical accidents. Investigation of the scale and characteristics of the influences caused by the accident is also included in this research.

The result of the conceptual design of CSEF is summarized in this report.

Keywords: Critical Safety Experimental Facility, Research Program, Solution Fuel, Uranium, Plutonium, Conceptual Design, Fuel Cycle.

Presentation at the International Seminar on Criticality Studies Programs and Needs, September 19-22, 1983 Dijon, France

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臨界安全性実験計画

日本原子力研究所東海研究所安全工学部
小林岩夫・館盛勝一・竹下　功
須崎武則・大西信秋

（１９８３年１１月９日受理）

臨界安全性実験施設（ＣＳＥＦ）を新たに建設し，主として溶液状の核燃料物質に関する臨界安全研究を行う計画を日本原子力研究所として進めている。第一の研究目的は，核燃料サイクル施設の臨界安全性評価に必要な実験データを測定，収集及び評価することであり，第二の目的は臨界事故事象の究明ならびに事故に起因した諸現象の把握とその対策である。

本報告書にはＣＳＥＦについて実施された概念設計結果の概要が述べられている。

＊　この報告書は１９８３年9月１９日から２２日に，フランスのデイジョン市において開催された「臨界実験の必要性と実験計画に関する国際セミナー」において発表されたものである。

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1. Introduction

JAERI is promoting the nuclear criticality safety research program, in which a new facility for criticality safety experiments (Criticality Safety Experimental Facility : CSEF) is to be built for the experiments with solution fuel. In 1981, general survey was made to find out areas where the data are insufficient for safety evaluation of future nuclear fuel facilities in Japan. Preliminary design and conceptual design of the CSEF was made in 1981 and 1982.

At present (1983), the review of the conceptual design is in progress. This report summarizes the CSEF program.

2. Criticality Experiments

The first priority is given to the experiments for safety evaluation in reprocessing plants of LWR spent fuels. Main experimental subjects are followings.

2.1 Experiments on Solution Fuel

1) Fundamental critical experiments on low-enriched uranyl nitrate solutions, and mixed solutions of plutonium nitrate and uranyl nitrate. Enrichment and isotopic composition of plutonium are widely varied.
2) Geometrical effects on reactivity in storage system such as annular and slab tanks with fully reflected condition.
3) Reflector effects on reactivity in storage system. Experiments without reflector are included.
4) Effect of fixed neutron poison inside and outside of solution fuels.

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2.2 Transient Experiments

Transient experiment data with low-enriched uranyl-plutonium nitrate solution are not found and little knowledge exists on physical and chemical phenomena during criticality accident.

The investigation items to be considered in the CSEF program are as follows:

1) Transient characteristics of criticality accident for parameters such as fissile material, insertion rate of reactivity, total reactivity inserted, and initial neutron flux level.
2) Mechanism and total amounts of energy releases.
3) Spatial distribution of radiation and total fission product release.

(Table 3)

3. Study of Process Chemistry Associated with Criticality Safety

Prevention of nuclear criticality during handling fissile materials is essential for the chemical facilities of fuel cycle. During operation of the facilities, flow rate and quanity of each chemicals must be accurately measured so as to control profiles of actinide concentration in each apparatus and all equipment must be regularly maintained to prevent mal-function.

For a base of good design and a safe operation of the processes, it is necessary to fully comprehend chemical behavior of actinides at the various conditions of the process covering sequences caused by mis-operation and/or mal-function of equipments.

It is one of the objectives of this study to investigate transient phenomena which lead to potential criticality in the processes, i.e. dissolution, solvent extraction, concentration, mixing and precipitation of the fuel materials, by laboratory experiments putting emphasis on an accumulation of the sub-

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stances at some part of apparatuses.

Mathematical model which simulates conditions of process upsets (transient state) and normal operation (steady state) provides benefit in assessing the safety of the plant and in preventing the occurence of criticality accidents. The other objective is to develop and varify the models by experiments for the elementary processes, i.e. dissolution, solvent extraction and concentration.

Scale up of the processing plant would be possibly achieved by virtue of neutron poison, soluble and/or fixed, of which application is valid only when the stability and reliability of the system involving poison are confirmed. Study of feasibility of applying neutron poisons to the fuel handing facility would be also carried out.

4. Critical Assemblies

Two types of critical assemblies are to be installed in the CSEF.

4.1 Tank Type Solution Assembly

This is a general-purpose assembly fueled with 4-10% enriched uranyl nitrate solution, and uranyl-plutonium (0-100% enriched) nitrate solution with or without soluble poison. Maximum fuel concentration is 1000 gU/l and 450 gPu/l. core tank and safety/control systems are interchangeable depending upon required core configuration. Maximum tank capacity is 1,100 l in case of uranyl nitrate fuel core. Tank geometries are cylinder, slab and annular cylinder. In case of cylinder core, such heterogeneous configuration can be composed as fuel rods array immersed in nitrate solution fuel which simulates dissolving process in reprocessing plant. Reactivity is controlled basically by fuel solution level or control blade. Excess reactivity in this assembly is less than 0.8 $ and maximum reactivity addition rate is 0.02 % k/k/sec.

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Minimum critical level is restricted to be 30 cm high because of safety reactivity control.

(Fig. 1, 2, Table 4)

4.2 Transient Critical Assembly

This assembly is a tool to investigate phenomena caused by criticality accidents on solution system. Initially, 6% or 10% enriched uranyl nitrate solution fuel is used for fast and slow burst experiment. Uranium concentration will be limited to 300 - 700 g/l. Maximum solution temperature is allowed up to 120°C during transient. Control will be provided by varying the solution level and maximum reactivity insertion rate is 3 $/100msec. Operating limit of the burst is 10^{18} fissions, which was decided considering reasonable thickness of reactor room sheilding (2 m concrete), core tank design pressure (5 kg/cm^2g), and environmental radiation safety. Core tank is a cylinder of 3 m high and max. 1 m in diameter. Water can be supplied into the annular area surrounding the inner core tank. Several instruments such as solution level detector, visual system for solution surface, thermocouples and neutron detectors are to be installed in the core tank. Volatile fission products are led to gaseous waste treatment line through vented gas hold up system. Experiment is terminated by fuel solution dump or by safety blade insertion. Radiation beam is led to adjacent reactor room, when this assembly is used for radiation shielding experiments and radiation detector tests during transient.

(Fig. 3, 4, Table 5)

5. Nuclear Fuel Supply System

The CSEF has a nuclear fuel supply system which consists of pre-treatment, adjustment, purification, conversion, solvent recovery and off-gas treatment processes. Uranyl nitrate solution and uranyl-plutonium nitrate solution with various fissile concentrations are prepared in the system and supplied to critical

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assemblies. Chemical processes and capacity of each process are described below.

(Table 6, Fig. 5)

5.1 Pre-Treatment Process

Received fuel powder (UO_2, PuO_2, etc) is blended, pressed and sintered to make pellets suitable for dissolution. The pellets are dissolved with boiling nitric acid, and resulting solution is filtered to remove insoluble solids and sampled for accounting. Capacity of the process is 5 kgU/day or 2.5 kgPu+U/day.

5.2 Adjustment Process

In accordance with the requirements of critical experiments, nitrate solutions are concentrated, diluted, mixed or poisoned to provide feeds to critical assemblies. Feeds to the purification process and to storage tanks are also adjusted in this process. Capacity of the process is 10 kgU/day or 5 kgPu+U/day.

(Fig. 6)

5.3 Purification Process

Nitrate solution is purified, when necessary, by solvent extraction with TBP which removes soluble nuclear poisons, fission products and ^{241}Am. Separation of uranium and plutonium is performed also in this process. Capacity of the process is 10 kgU/day or 1 kgPu/day.

(Fig. 7)

5.4 Conversion Process

After the completion of the program, uranyl nitrate and plutonium nitrate are converted by micro wave heating to UO_3

or mixed oxide (UO_3-PuO_2) powder suitable for transportation or long-term storage. The oxides may be reduced to UO_2 or UO_2-PuO_2, if required. Capacity of the process is 10 kgU/day or 5 kgPu+U/day.

5.5 Solvent Recovery Process

Used solvent is washed periodically with alkali and acid, and recycled to the purification process. Capacity of the process is 50 l/day.

5.6 Off-gas Treatment Process

Process off-gas and vessel vent gas are treated to remove NO_x, and led to waste treatment system.

Besides the above processes, there are several supporting systems in the CSEF such as chemical make-up system, analytical laboratory, waste treatment system, etc.

6. Schedule of the Program

After the licensing review by government and construction of the facility, experiments are to be initiated in 1989.

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Table 1 PARAMETERS IN MIXED NITRATE SOLUTION OF URANIUM AND PLUTONIUM EXPERIMENTS

EXPERIMENT SERIES	Pu ENRICHMENT %	^{235}U ENRICHMENT %	^{240}Pu %	(Pu+U)CONCENTRATION g(U+Pu)/l
I. Pu ENRICHMENT	5~50, 100	Nat, 4	20	15~500
II. SHAPE, REFLECTOR	15~50, 100	"	20	"
III. ABSORBER EFFECT	30,50,100	"	20	100~500
IV. ^{240}Pu EFFECT	"	"	5,20,25	15~500
V. INTERACTION	"	"	20	100~300
(VI. ORGANIC SOLVENT)	-	-	-	-

REFLECTOR : WATER,IRON, CONCRETE AND BARE
CORE TANK : CYLINDER (20~60cm$^{\phi}$), SLAB, ANNULAR CYLINDER
SOLUBULE ABSORBER : Max. 20 gGd/l

Table 2 PARAMETERS IN URANYL NITRATE SOLUTION EXPERIMENTS

EXPERIMENT SERIES	^{235}U ENRICHMENT %	CONCENTRATION (g/l)	OTHERS
I. BASIC	6, 10, 4	50~1000	CYLINDER DIA 40~100cm
II. ABSORBER EFFECT	6, 10	400~600	Max.0.5gGd/l
III. HETEROGENEOUS EFFECT	6, 10	0~500	
(IV. INTERACTION)	6, 10	50~1000	
(V. ORGANIC SOLVENT)	-	-	30%TBP-DODECANE, KEROSENE

REFLECTOR : WATER, IRON, CONCRETE, BARE
CORE TANK : CYLINDER, SLAB, ANNULAR

Table 3 PARAMETERS IN TRANSIENT EXPERIMENTS

PARAMETERS	RANGES OF PARAMETERS
FUEL SOLUTION	URANYL NITRATE
^{235}U	10%, 6%
U CONCENTRATION	300~700 gU/l
REACTIVITY INSERTION	TRANSIENT ROD, FUEL SOLUTION LEVEL (STEP AND RAMP)
AMOUNT OF REACTIVITY	Max. 3$
TOTAL FISSIONS	Max. 1×10^{18} FISSIONS
REFLECTOR (SHIELDING)	WATER, BARE, CONCRETE, IRON

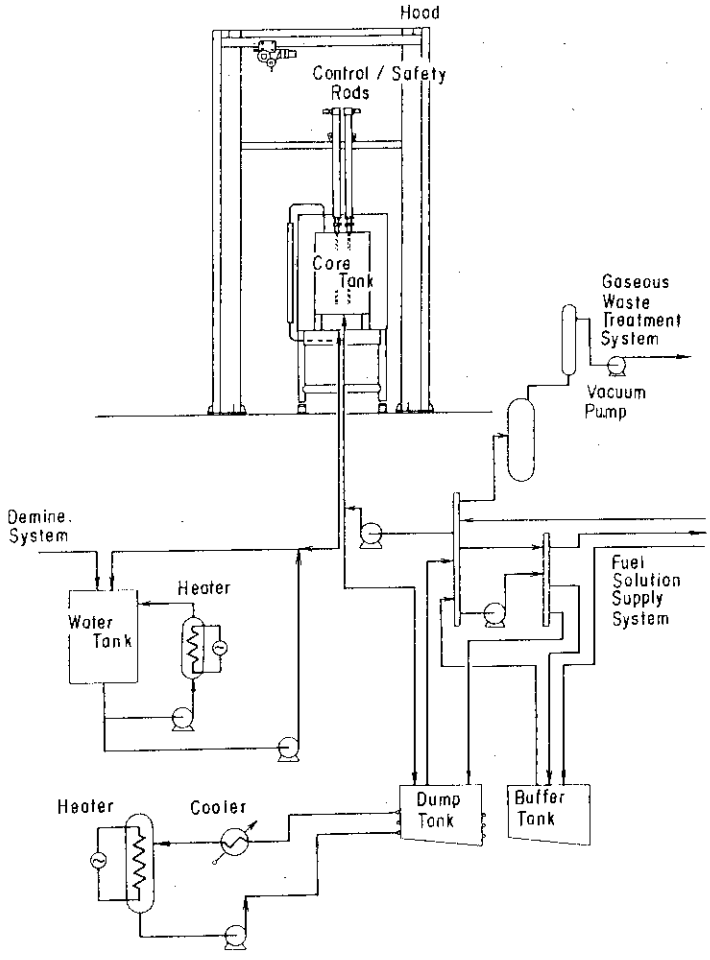

Fig. 1 Tank Type Solution Critical Assembly System

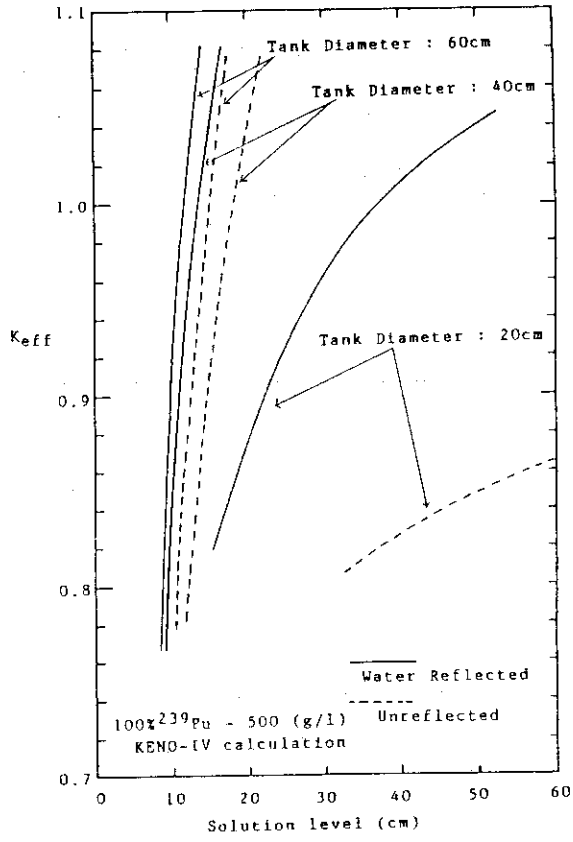

Fig. 2 Calculation Results of Effective Multiplication Factor of Pultonium Nitrate Solution (500g/l, 100%^{239}Pu) with Different Solution Level

Table 4 MAJOR SPECIFICATIONS OF TANK TYPE SOLUTION CRITICAL ASSEMBLY

CORE CONFIGURATIONS	CAPABILITIES
CORE TANK:CYLINDER(Max.1 m^{ϕ} x 1.4 m^{H}) SLAB, SPHERE, ANNULAR CYLINDER FUEL :URANYL NITRATE ENRICHMENT 4~10% CONCENTRATION Max.1000 gU/l VOLUME Max.1100 l URANYL PLUTONIUM NITRATE Pu ENRICHMENT 0~100% CONCENTRATION Max.450 gPu/l VOLUME Max.300 l POISON :SOLUBLE, FIXED POISON REFLECTOR:WATER, CONCRETE, SUS	POWER :Max.200W INTEGRATED POWER:Max.5 kwh/year CORE PRESSURE :One atm CORE TEMPERATURE:Max.90°C REACTIVITY CONTROL :CONTROL BLADE, FUEL SOLUTION LEVEL REACTIVITY ADDITION RATE :Max.0.02%k/k/sec EXCESS REACTIVITY :Max.0.8$ SHUTDOWN METHOD :SAFETY BLADE INSERTION FUEL SOLUTION DRAINING

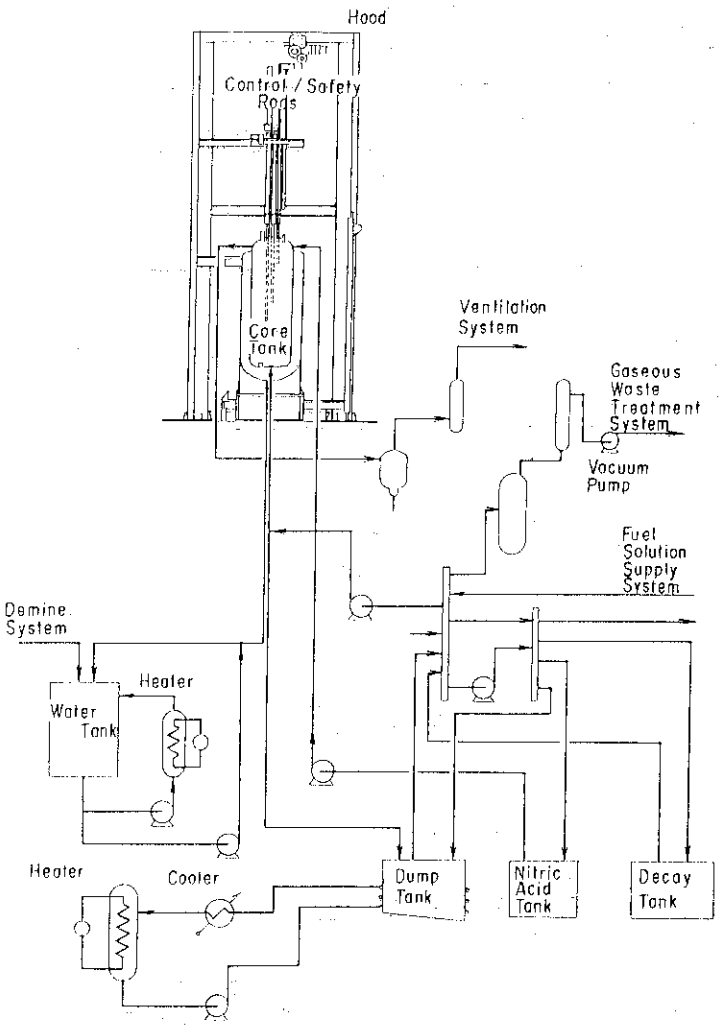

Fig. 3 Transient Critical Assembly System

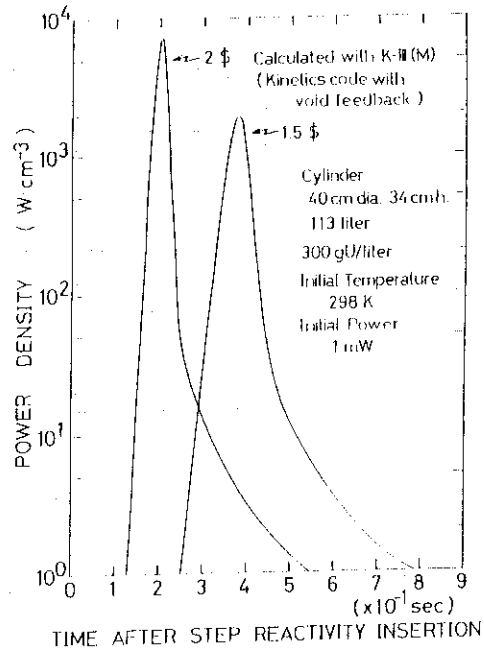

Fig.4 Calculation Results of Power Transients with K-III

Table 5 MAJOR SPECIFICATIONS OF TRANSIENT CRITICAL ASSEMBLY

CORE CONFIGURATION	CAPABILITIES
CORE TANK:CYLINDER(Max.1 m^{ϕ} x 3 m^{H}) URANIUM NITRATE ENRICHMENT 10%, 6% CONCENTRATION Max. 700 gU/l VOLUME Max. 1700 l REFLECTOR:BARE, WATER	POWER(BURST) :Max.1x10^{18} FISSIONS EQ. INTEGRATED POWER:Max.0.23 MWh/year CORE PRESSURE :Max.5 kg/cm^2G CORE TEMPERATURE:Max.120°C REACTIVITY INSERTION RATE :Max.3$/100msec REACTIVITY INSERTION METHOD :TRANSIENT ROD, FUEL SOLUTION SUPPLY SHUTDOWN METHOD :SAFETY ROD, FUEL SOLUTION SUPPLY

Table 6 SOLUTION FUEL SPECIFICATION

FUEL	ENRICHMENT	INVENTORY	ACIDITY	CONCENTRATION
URANYL NITRATE	10%	200 kgU	0~6 Mol/l	1000 gU/l
	6%	800 kgU		
	4%	100 kgU		
URANYL PLUTONIUM NITRATE	U:(nat)	150 kgU	1~3 Mol/l	450 gPu/l
	10% ^{240}Pu/Pu	20 kgPu		
	20% ^{240}Pu/Pu	60 kgPu		
	25% ^{240}Pu/Pu	20 kgPu		

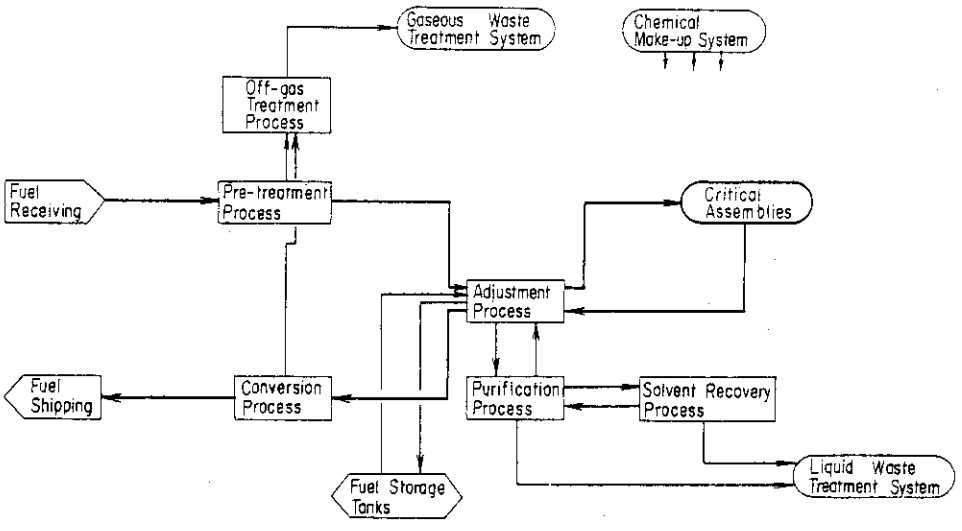

Fig. 5 Nuclear Fuel Supply System

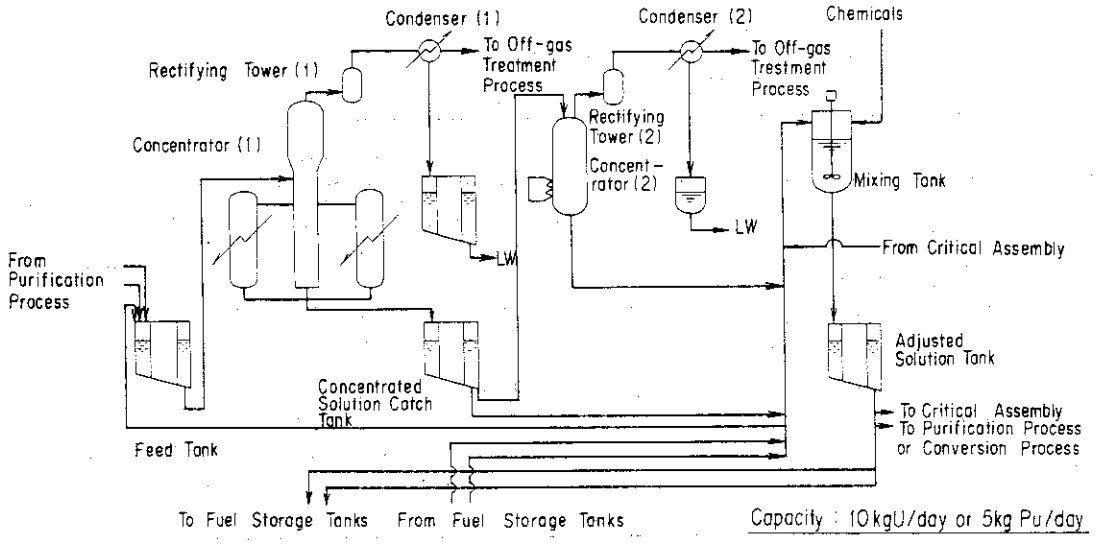

Fig. 6 Adjustment Process of Nuclear Fuel Supply System

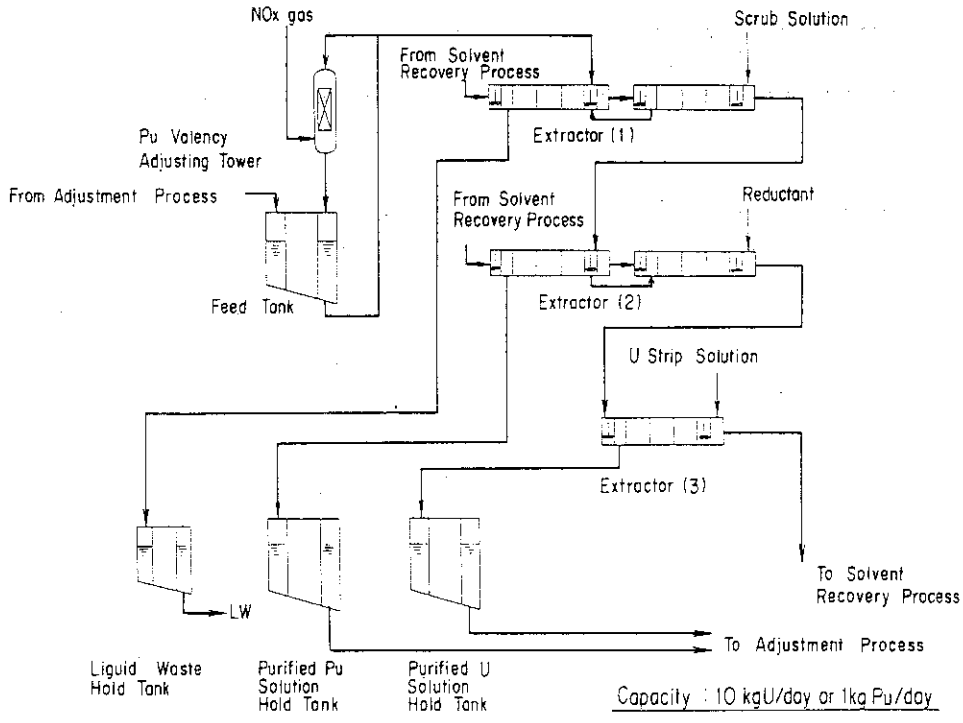

Fig. 7 Purification Process of Nuclear Fuel Supply System

Table 7 CSEF PROGRAM LONG TERM SCHEDULE

As of Sept. 1983

1982	1983	1984	1985	1986	1987	1988	1989
CONCEPTUAL DESIGN	DETAILED DESIGN						
		LICENSING REVIEW OF THE FACILITY		LICENSING FOR CONSTRUCTION		INSPECTION	
				U, Pu PROCUREMENT			
							INITIATION OF EXPERIMENT