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CRITICAL EXPERIMENTS FACILITY AND CRITICALITY SAFETY PROGRAMS AT JAERI

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Critical Experiments Facility and Criticality Safety Programs at JAERI

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The nuclear criticality safety is becoming a key point in Japan in the safety considerations for nuclear installations outside reactors such as spent fuel reprocessing facilities, plutonium fuel fabrication facilities, large scale hot alboratories, and so on. Especially a large scale spent fuel reprocessing facility is being designed and would be constructed in near future, therefore extensive experimental studies are needed for compilation of our own technical standards and also for verification of safety in a potential criticality accident to obtain public acceptance.

Japan Atomic Energy Research Institute is proceeding a construction program of a new criticality safety experimental facility where criticality data can be obtained for such solution fuels as mainly handled in a reprocessing facility and also chemical process experiments can be performed to investigate abnormal phenomena, e.g. plutonium behavior in solvent extraction process by using pulsed colums. In FY 1985 detail design of the facility will be completed and licensing review by the government would start in FY 1986. Experiments would start in FY 1990. Research subjects and main specifications of the facility are described.

Keywords: Critical Experimental Facility, Nuclear Installations, Criticality Safety Program, Solution Fuel, Reprocessing, Specification

原研の臨界安全性実験計画と臨界実験装置

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（１９８５年9月１２日受理）

再処理工場，プルトニウム燃料加工施設，大型ホットラボ施設などの原子炉以外の核燃料施設における臨界安全性は，わが国において重要な課題となりつつある。特に第二再処理工場の建設が計画されていることもあり，わが国独自の実験データにもとづいた技術基準の作成や，仮想される臨界事故時における安全性の評価が必要とされている。

原研では，軽水臨界実験装置ＴＣＡを用いて，中性子相互干渉効果や高未臨界度の測定などに関する実験的研究と，ＫＥＮＯコードの検証作業を実施している。一方，ＴＣＡでは取扱うことのできない，溶液状の燃料など，再処理工場において主に取扱われる燃料についての臨界実験と，抽出など溶液燃料の取扱い工程において問題となる異常事象について化学的実験を行うため，新しく燃料サイクル安全工学研究施設（ＮＵＣＥＦ）を建設することを計画している。NUCEFの詳細設計は1985年度に終了し，１９８６年度には国の安全審査を受ける予定である。計画通り進めば１９９０年から実験が開始される。ＮＵＣＥＦにおける臨界実験計画と，臨界実験施設の概要を記載した。

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The nuclear criticality safety is becoming a key point in Japan in the safety considerations for nuclear installations outside reactors such as spent fuel reprocessing facilities, plutonium fuel fabrication facilities, large scale hot laboratories, and so on. Especially a large scale spent fuel reprocessing facility is being designed and would be constructed in near future, therefore extensive experimental studies are needed for compilation of our own technical standards and also for verification of safety in a potential criticality accident to obtain public acceptance.

At the Japan Atomic Energy Research Institute (JAERI), experimental studies for measurements of neutron interaction effects and highly subcritical reactivities is performed by using the TCA facility (a light-water moderated critical assembly) and improvements and verifications of calculation code systems are also conducted. In addition to the above efforts, JAERI is proceeding a construction program of a new criticality safety experimental facility (CSEF) where criticality data can be obtained for such solution fuels as mainly handled in a reprocessing facility and also chemical process experiments can be performed to investigate abnormal phenomena, e.g. plutonium behavior in solvent extraction process by using pulsed columns. In FY 1985 detail design of the CSEF will be completed and licensing review by the government would start in FY 1986. Experiments would start in FY 1990. Research subjects and main specifications of the CSEF are described in this article.

2. Research Subjects

Criticality control related to the reprocessing of spent fuels in Japan was based primarily on, so called, safe geometry for unlimited fuel concentration. The control method has an advantage that the possibility of criticality accident can be almost neglected, however, it is clearly not applicable to large-scale plants. In design of such plants, it is required to increase sizes of vessels or fuel amounts to be treated, by adopting advanced criticality control methods in which changes in fuel composition will be limitted within the realistic range. Counterplans against potential criticality accidents will

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The purposes of CSEF experiments are focused on the following two items.

a) Development of advanced criticality control methods for reprocessing facilities

This requires organic combination between process chemistry and reactor physics. The former concerns a wide area covering basic chemical problems, process safety analyses, development of advanced monitoring methods, and so on. In the latter, the criticality safety margin will be verified through systematic critical and subcritical experiments with fuel systems modeled on process equipments in reasonable limiting conditions, and transient behaviors of fuel systems and fission products in supercritical states will be demonstrated through a series of transient experiments with realistically reasonable reactivity addition conditions.

b) Development of chemical process and nuclear criticality safety analysis codes

Simulation models of elementary chemical processes such as dissolution, solvent extraction and concentration should be developed with first priority to clarify the conditions of abnormal conditions. Criticality data are also required to ascertain the bias of computational results. Statistical analyses and simulation models of criticality accidents remain to be investigated in near future. The effective criticality control method will be accomplished by a accumulating such efforts in both of hardware and software.

For these purposes, the experimental subjects are devided into three areas. The immediate subjects in each area are summarized as follows.

2.1 Critical experiments

From the view point of criticality safety, we have investigated reprocessing plants and found the areas where the advanced criticality control would be applicable. As an example, several possibilities of advanced criticality control would be considered as shown in Table 1 for the dissolution process. However, it requires further experimental data on criticality and demonstration of safety margin with possible advancement of process control methods.

The criticality safety evaluation of out-of-reactor facilities is characterized by the extremely wide variety of system conditions. It is very important to evaluate the accuracy of computer code calculations applying to many kinds of nuclear fuel system. An effort continued to know the bias values which should be taken into account for the results calculated by JACS code system developed at JAERI. Table 2 shows an attempt to classify the bias values according to the reactor-physical properties of examined nuclear fuel systems. The code system is proven to possess the similar performance to those of other countries, as illustrated in Fig. 1. As shown in Table 2, further experimental data are required to minimize the bias values especially on the systems being sensitive to neutron leakage and non-thermal fissions. Being considered above requirements, experimental program using a solution critical assembly in the CSEF has fundamental features as follows.

a) Uranyl (low-enriched) - plutonium mixture :

Nitrate solution fuels of uranyl (low-enriched) - plutonium mixture are mainly examined to contribute to the problems in LWR and FBR spent fuels reprocessing . These fuels are effective to carry out critical experiments with wide variations in neutron spectrum and leakage rate.

b) Mock up process abnormal conditions :

To mock up abnormal conditions of the chemical processes critical and subcritical experiments are performed on the fuel systems where parameters on fuel solution and geometry are varied as widely as possible and reasonable.

c) Kinetic and reactivity parameters :

Neutron life time, delayed neutron fractions, reactivity coefficients of temperature and void are the parameters to be measured in critical experiments to analize the results of transient experiments explained in the next section.

In order to pursue the safety analysis on the solution critical assembly, the experimental systems were categorized into several groups as shown in Table 3. The experiments with systems belonging to "phase II" will be started later than those in "phase I".

2.2 Transient experiments

Transient experiments will be performed to investigate the transient phenomena of potential criticality accidents by using low-enriched uranium solution fuel which is mainly used in reprocessing plants. These experimental results contribute the safety design and assessment of the plants. The investigation items of this experiment are as follows:

a) Transient characteristics of criticality accident for parameters such as fissile material, insertion rate of reactivity, total reactivity inserted, initial neutron density, etc.

b) Mechanism and total amounts of energy releases.

c) Spatial distribution of radiation and total fission products release.

The reactivity insertion to the transient critical assembly is made by

a) rapid withdrawal of a transient control rod from the core,
b) slow withdrawal of a control rod, and
c) addition of fuel solution to the core tank.

In the experiments, such parameters are measured as number of fissions, temperature, pressure, radiation doses, amounts of FP's and so on. Transient phenomena are terminated by inserting safety rods or draining the fuel solution.

Experimental parameters and limitations are as follows:

a) Form of fuel : Uranyle nitrate solution
b) Enrichment of U : 10 wt% and 6 wt%
c) Concentration of fuel : Max. 700 g/l
d) Acid molarity ; Max. 6 mol/l
e) Core tank : Cylindrical tanks with 50 cm dia. and 80 cm dia.
f) Reflector : Bare (No-reflector) or water etc.
g) Inserted reactivity : Max. 3 $
h) Initial temperature : Max. 80 ℃
i) Initial neutron density : 0 - 1 KW eq.

Table 4. shows experimental systems and parameters. Three experimental systems are considered as a future plan which are called phase II experiments. Parameters in these experiments are not described here.

2.3 Chemical Process Experiments

One of the objectives of critical experiment programs is to investigate abnormal phenomena of chemical processes with nuclear fuels which are caused by mis-operation and/or mal-function of equipments and lead to potential criticality accidents. The experiment is carried out in laboratory scale putting emphasis on an accumulation of plutonium at some part of apparatuses.

For the solvent extraction process, for example, an experimental system with pulsed columns is installed in the CSEF and behavior of plutonium in the column at various conditions are observed. The results are applied to develop and verify the process calculational model of pulsed column for the purex process. The model is then used to discuss criticality safety of pulsed column system in fuel reprocessing plant.

The other objective is to develop a process control system for criticality safety of fuel reprocessing plant. The system is composed of highly reliable in-line monitors sensitive to fissile materials, a device for subcriticality measurement, an alarm system, and a computor system.

In an advanced criticality safety design, neutron absorbing materials will be widely utilized in soluble or solid states of which stability and reliability will be required to be confirmed any time. Then, the compatibility of the neutron absorber with nuclear fuel materials and it's reliability in the life of equipments would be examined also in this experimental programs.

3. Experiment Facility

Critical assemblies and equipments for chemical process experiments such as pulsed column system are the major tools for this research program. Nuclear fuel supply system and the other supporting systems are also included in the same building. Uranyl nitrate solution and uranyl-plutonium nitrate solution with various fissile concentrations are prepared and supplied to the above experimental systems.

3.1 Tank type solution critical 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 dissolution 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 $\%\Delta k/k$/sec. Minimum critical level is restricted to be 30 cm high because of safety reactivity control. Major specifications of this system and conceptual system flow diagram are shown in Table 5 and Fig. 2.

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3.2 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 experiments. Uranium concentration will be limited to 300 - 700 g/l. Maximum solution temperature is allowed up to 120°C during transient. Reactivity control will be made by a transient control rod withdrawal and solution fuel level increase. 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 shielding (2 m concrete), core tank design pressure (5 kg/cm^{2}G), 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 soltuion 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. Major specifications of transient critical assembly and conceptual system flow diagram are shown in Table 6 and Fig. 3.

3.3 Fuel storage system

This storage system is mainly used to store the uranium and plutonium nitrate solutions adjusted in the nuclear fuel supply system, and rod type fuels of PWR and FBR type. The uranium and plutonium oxide powder which are converted to solution fuels are also stored. The design conditions for each type fuels are summarized in Table 7.

For the criticality safety of the storage system, the dimension of a single unit such as a slab tank and an annular tank is limited to satisfy the safety criteria. In addition, the neutron interactions between multiple units in a cell are evaluated by using validated calculational codes such as Monte carlo code KENO-IV, and one dimensional transport code ANISN.

A typical calculational result is presented in Fig. 4, where the neutron multiplication factor of the storage racks which contain PWR fuel rods is shown as a function of water content ratio in the inner region.

3.4 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 assemblies. The total system is shown in Fig. 5

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After the licensing review by government and construction of the facility, experiments are to be initiated in 1990.
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Table 1 Requirements for advanced criticality control (Ex.: Dissolution of LWR fuels)

Present model	Alternatives	Reactivity effects	Requirements for advanced process control	Requirements for criticality data
• Initial enrich.	• Taking into consideration of 1. Burnup	$\sim$ 1% $\Delta k/k$ / GWd/t	• Non-destructive burnup monitoring	• Effective enrich. of burnup (U+Pu) fuel
• Optimum moderation of fuel rods in water (Fuel volume ratio: $\sim$ 25%)	2. Realistic fuel volume ratio ($\sim$50%)	$\sim$10% $\Delta k/k$	• Subcriticality monitoring	• Reactivity effects of FP and nitric acid
• Full reflection	3. Nitric acid	$\sim$ 1% $\Delta k/k$ /N		
• Geometrical safety margin (x 0.85)	• Isolation from reflector	$\sim$ 5% $\Delta k/k$	• Diagnostic techniques	• Systematic data of fuel rods + fuel solution + neutron poison
	• Utilization of neutron poison 1. Soluble poison	$\sim$20% $\Delta k/k$ / gGd/l	(• Continuous dissolution)	
(Ex.) Dissolver vessel of $\sim$22cm dia. for 4%-enr. LWR fuel	2. Fixed poison	$\sim$60cm dia. with Hf basket (West Germany)		

Table 2 Bias values of the criticality safety evaluation system JACS

Fission ratio F_1: Fast, F_2: Epithermal, F_3: Thermal	$\frac{\text{Leakage}}{\text{Absorption}}$	No. of cases	*) k_{limit}	Bias ($1 - k_{limit}$)
$F_3 \geqq 0.7$	$L/A < 0.7$	475	0.94	0.06
〃	$\geqq 0.7$	26	0.88	0.12
$0.4 < F_3 < 0.7$, $F_2 \geqq 5F_1$	< 0.7	15	0.90	0.10
〃 , 〃	$\geqq 0.7$	54	0.92	0.08
〃 , $F_2 < 5F_1$	< 0.7	34	0.87	0.13
$F_3 \leqq 0.4$, 〃	〃	81	0.94	0.06
〃 , 〃	$\geqq 0.7$	17	0.94	0.06

*) Upper limit of calculated effective neutron multiplication factor certifing the questioned system to be subcritical (σ = 0.005).

Table 3 Experimental cores and test parameters for critical experiments

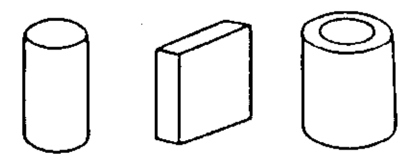
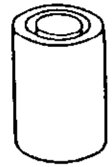
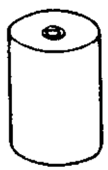
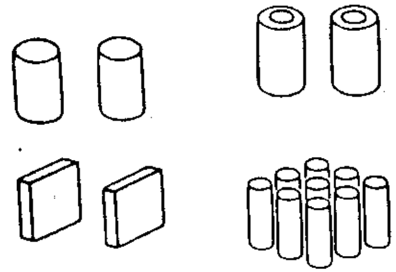
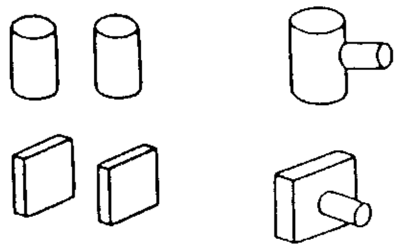
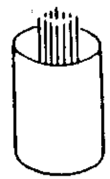
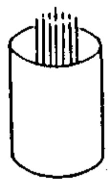
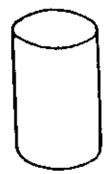
Phase I	Core geometry	Phase II	Core geometry
(1) Basic core system 1) Enrichment 2) Concentration 3) Vessel geometry 4) Reflector 5) Pu content 6) Temperature 7) Neutron poison		(5) Two Region System 1) Inner region Organic solution Nitrate solution(Pu) Precipitate Unhomogeneous state 2) Outer region Nitrate solution	
(2) Capusule core system 1) Inner region Organic solution Nitrate solution (Uranyl.) Precipitate 2) Outer region Nitrate solution		(6) Interaction system (II) 1) Different solution 2) Vessel geometry 3) Reflector 4) Array configulation of multiple units 5) Isolation material	
(3) Interaction system 1) Vessel geometry 2) Reflector 3) Distance between two units 4) Isolation material 5) Pipe sumilation		(7) Heterogeneous system (II) 1) Fuel rod(FBR) 2) U/Pu concentration 3) Lattice pitch 4) Soluble poison 5) Temperature	
(4) Heterogeneous system 1) Fuel rod (PWR,BWR) 2) U concentration 3) Lattice pitch 4) Soluble poison 5) Temperature		8) Organic solution system 1) Enrichment 2) Concentration 3) Pu content 4) TBP content 5) Reflector 6) Neutron poison	

Table 4 Experimental systems and parameters for transient experiments

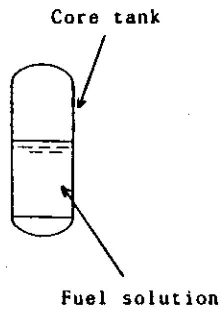
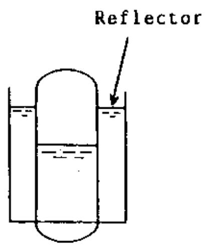
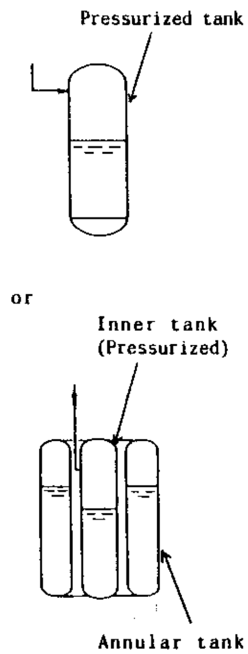
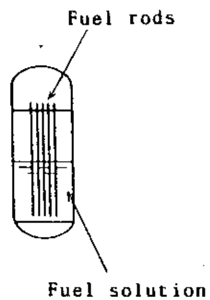
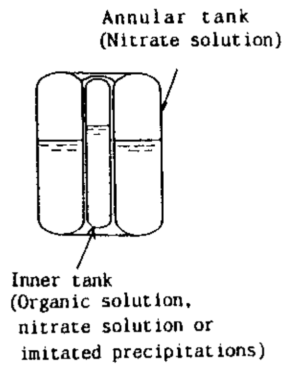
Type of systems	(1-a) Bare-U system	(1-b) Reflector-U system	(2) Pressurized system	(3) Dissolver system	(4) Inhomogeneous system
Experiment phase	Phase I	Phase I	Phase II	Phase II	Phase II
Parameters and items to be measured	1) Fuel conditions 2) Rate and amount of reactivity insertion 3) Initial neutron density 4) Initial temperature 5) Tank diameter 6) Radiation dose 7) Amounts of FPs	1) - 7) in (1-a) and 8) Reflector	1) - 8) in (1-b) and 9) Initial pressure	1) - 8) in (1-b) and 9) Fuel rods arrengement	1) - 8) in (1-b) and 9) Contents of inner tank
Core model					

Table 5 MAJOR SPECIFICATIONS OF TANK TYPE SOLUTION CRITICAL ASSEMBLY

CORE CONFIGURATIONS	CAPABILITIES
CORE TANK:CYLINDER(Max. 1mϕ x 1.5mH) SLAB. 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 .etc.	POWER :Max. 200 W INTEGRATED POWER:Max. 3 kWh/year CORE PRESSURE : 1 atm CORE TEMPERATURE:Max. 90 °C REACTIVITY CONTROL :CONTROL BLADE, FUEL SOLUTION LEVEL REACTIVITY ADDITION RATE :Max. 0.02 % Δk/k/sec EXCESS REACTIVITY :Max. 0.8 \$ SHUTDOWN METHOD :SAFETY BLADE INSERTION, FUEL SOLUTION DUMP

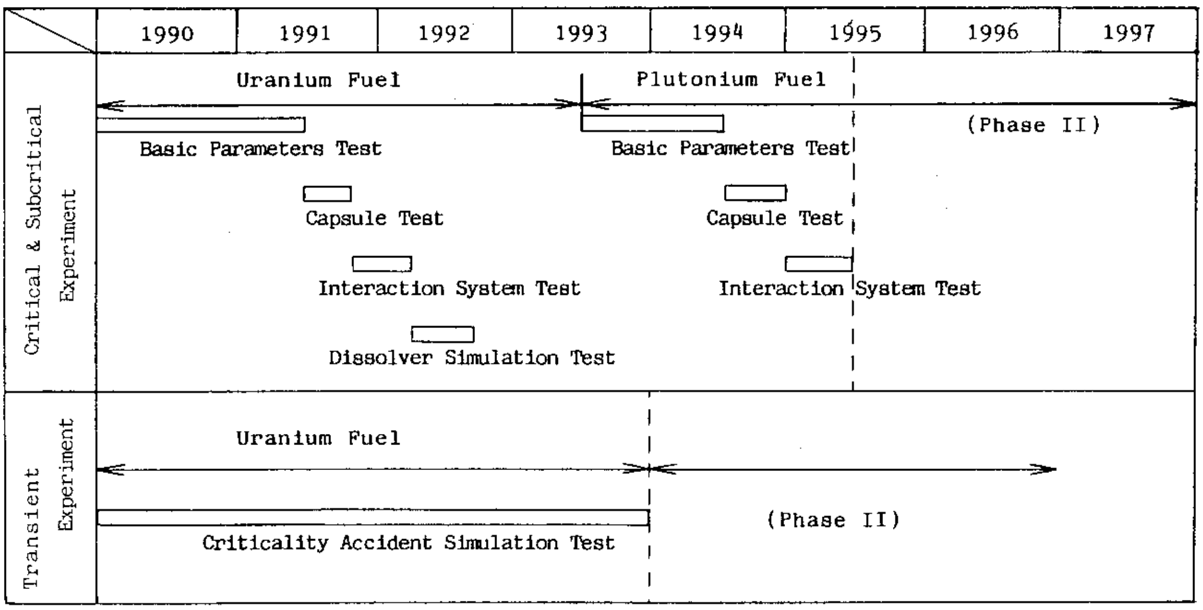
Table 6 MAJOR SPECIFICATIONS OF TRANSIENT CRITICAL ASSEMBLY

CORE CONFIGURATIONS	CAPABILITIES
CORE TANK:CYLINDER(Max. 0.8mϕ x 3 mH) FUEL :URANYL NITRATE ENRICHMENT 10 %, 6 % CONCENTRATION Max. 700 gU/l VOLUME Max. 700 l REFLECTOR:BARE, WATER, etc.	POWER (BURST) :Max. 1 x 10^{18} fiss. INTEGRATED POWER :Max. 0.23 MWh/year CORE PRESSURE :Max. 5 kg/cm G CORE TEMPERATURE :Max. 120 °C REACTIVITY INSERTION RATE :Max. 3 \$/100msec REACTIVITY INSERTION METHOD :TRANSIENT ROD, FUEL SOLUTION SUPPLY SHUT DOWN METHOD :SAFETY RODS, FUEL SOLUTION DUMP

Table 7 DESIGNING CONDITIONS OF NUCLEAR FUEL STORAGE SYSTEM

SYSTEM	FORM OF FUEL	FORM OF STORAGE
ROD TYPE FUEL TEMPORARY STORAGE	PWR ROD FUEL ENRICHMENT of U-235 4-6 wt% LENGTH:1470 mm O.D. : 9.5 mm	CONTAINER BOX
U-POWDER FUEL STORAGE	POWDER or PELLET of UO_2 or UO_3 ENRICHMENT OF U-235 Nat.- 5 wt% 6 - 10 wt%	 DRUM CAN 10 GALLON CAN
Pu-POWDER FUEL STORAGE	POWDER of PuO_2 or PuO_2-UO_2 4-5 CANS of POWDER/CONTAINER	TRANSPORT/STORAGE CONTAINER in the CONCRETE PIT
U-SOLUTION FUEL STORAGE	URANYL NITRATE SOLUTION ENRICHMENT of U-235 4,6 and 10 wt% CONC. of FUEL 400 gU/l	SLAB TANK STORAGE
Pu-SOLUTION FUEL STORAGE	PLUTONIUM NITRATE SOLUTION ENRICHMENT of Pu-240 about 5,20,25 wt% CONC. of FUEL 250 gPu/l URANYL NITRATE SOLUTION CONC. of FUEL 400 gU/l	ANNULAR TANK STORAGE

Table 8 Experiment Schedule



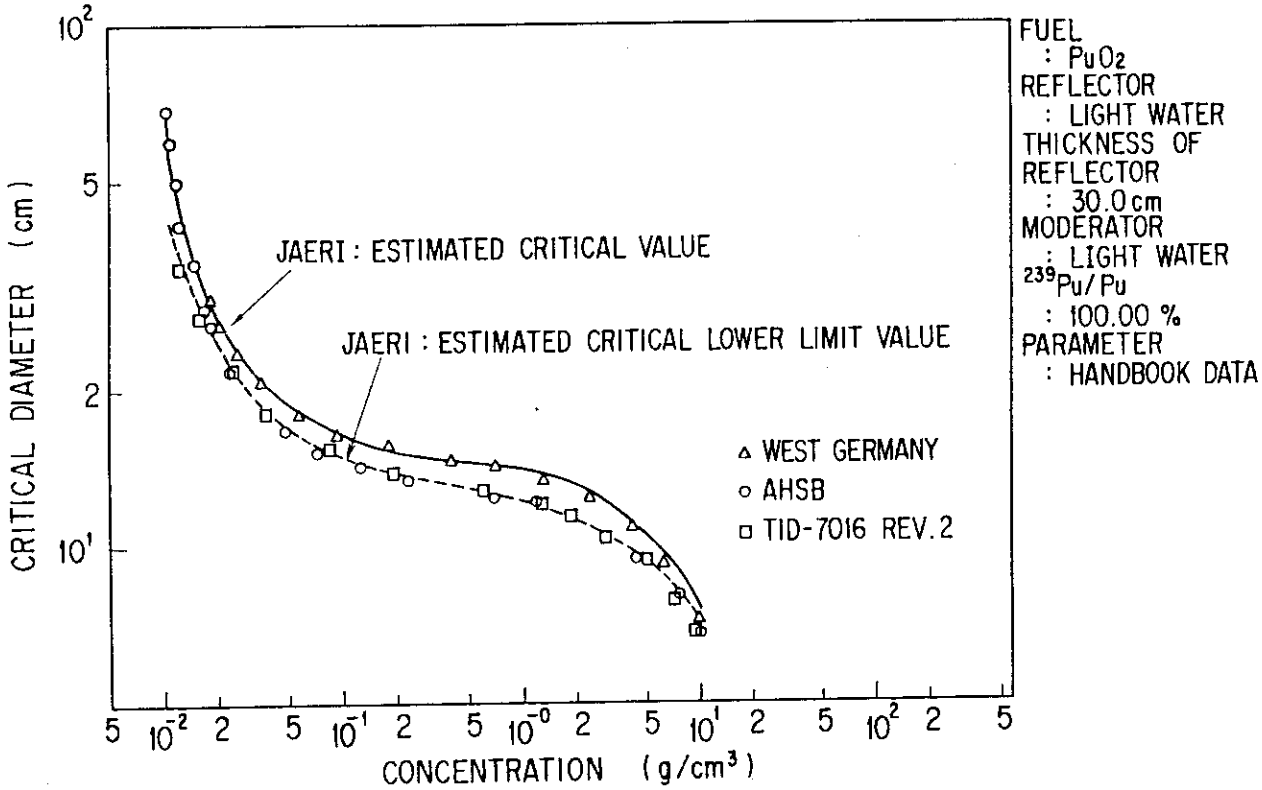


Fig. 1 Comparison of estimated critical dimensions of infinitely long homogeneous $PuO_2 - H_2O$ solution cylinder with those reported by foreign countries

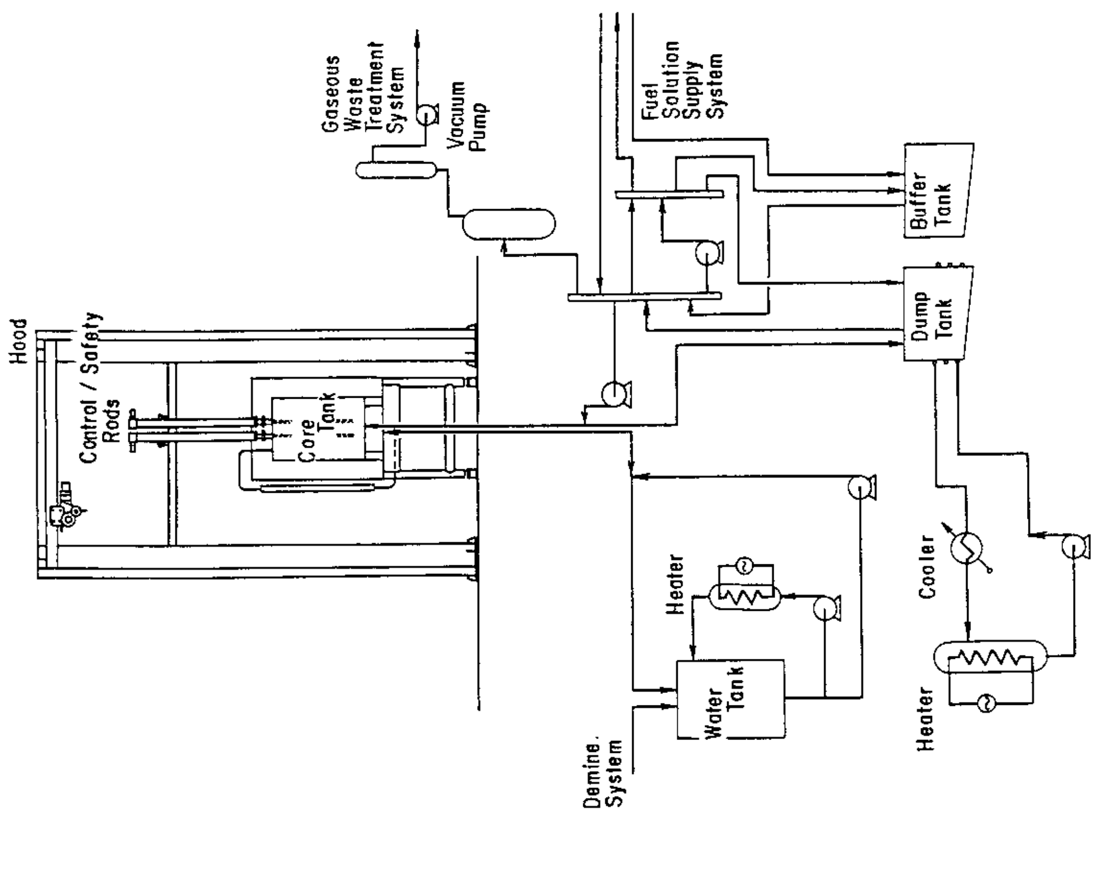


Fig. 2 TANK TYPE SOLUTION CRITICAL ASSEMBLY SYSTEM

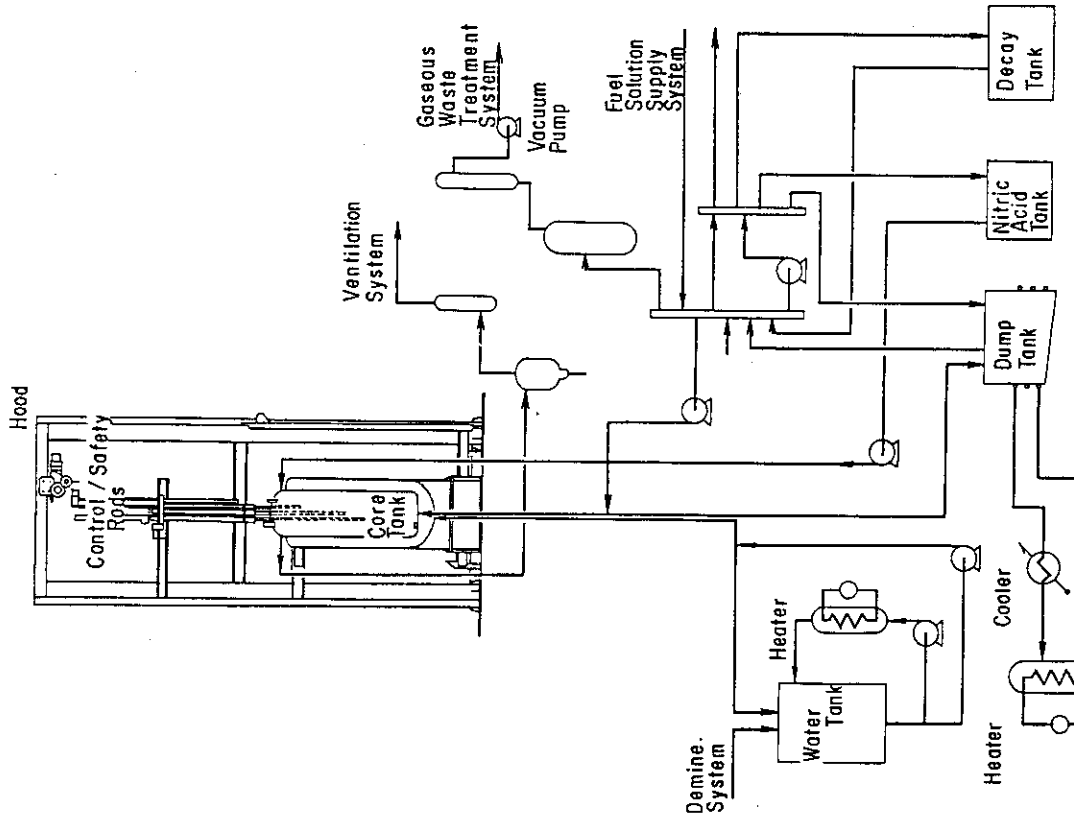


Fig. 3 TRANSIENT CRITICAL ASSEMBLY SYSTEM

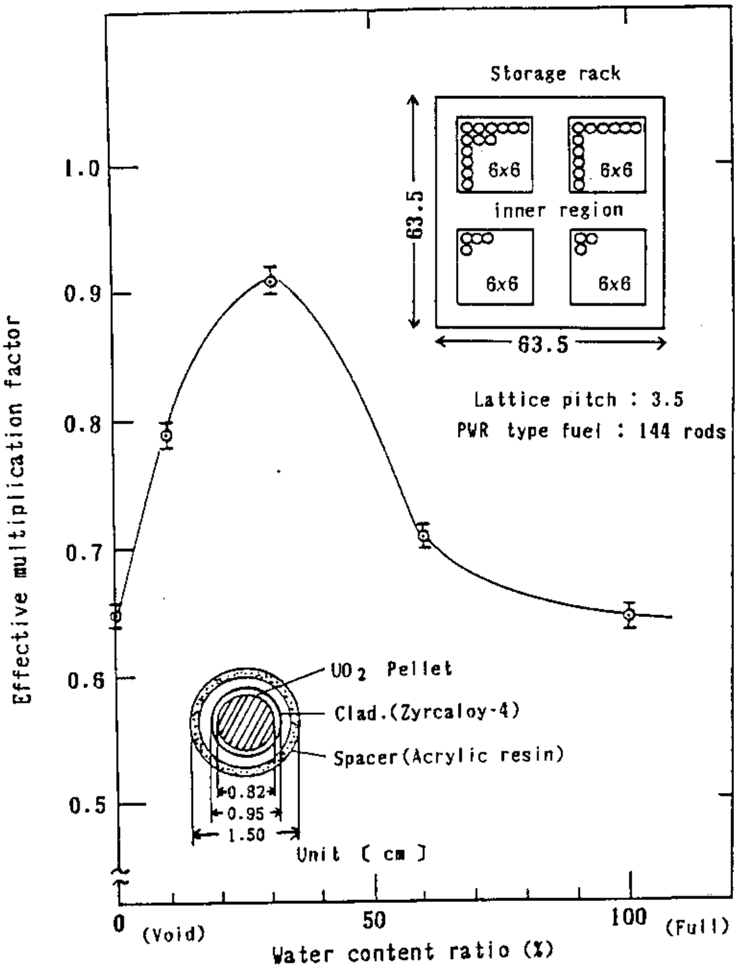


Fig. 4 Dependance of multiplication factor on the water content ratio (Storage system for PWR type fuel rods)

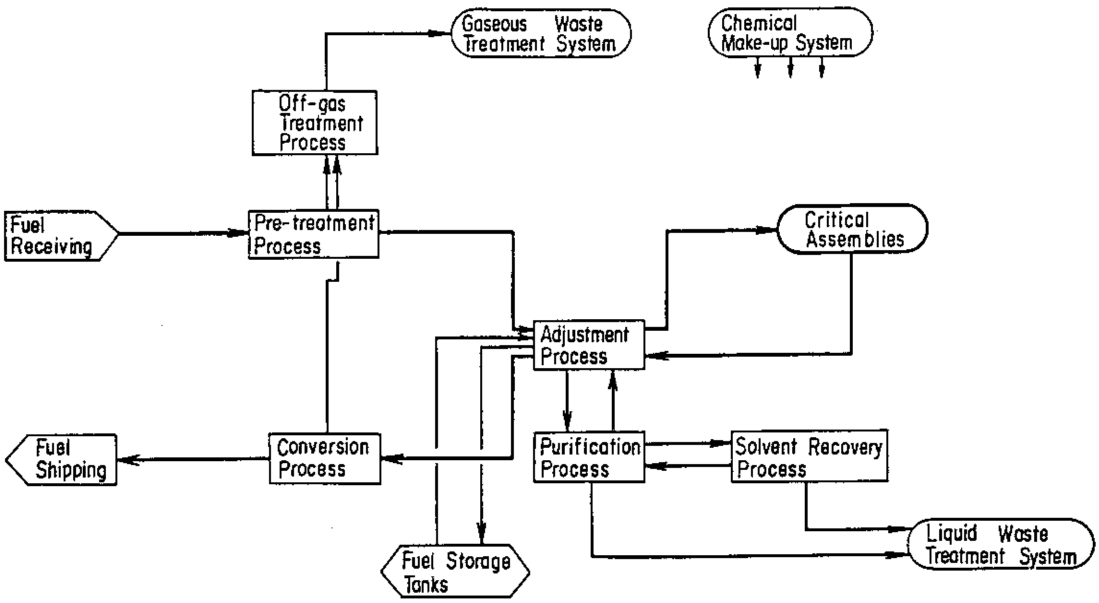


Fig.5 NUCLEAR FUEL SUPPLY SYSTEM