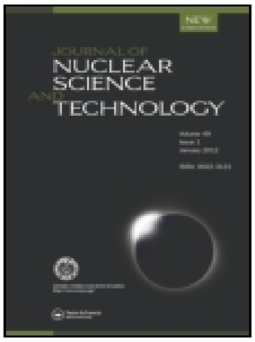

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SUMMARY REPORT

Thorium Molten-Salt Nuclear Energy Synergetics

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In the next century, the "fission breeder" concept will not be practical to solve the global energy problems, including environmental and North-South problems. As a new measure, a simple rational Th molten salt breeding fuel cycle system, named "Thorium Molten-Salt Nuclear Energy Synergetics [THORIMS-NES]", which composed of simple power stations and fissile producers, is proposed. This is effective to establish the essential improvement in issues of resources, safety, power-size flexibility, anti-nuclear proliferation and terrorism, radiowaste, economy, *etc.* securing the simple operation, maintenance, chemical processing, and rational breeding fuel cycle. As examples, 155 MWe fuel self-sustaining power station "FUJI-II", 7 MWe pilot-plant "miniFUJI-II", 1 GeV-300 mA proton Accelerator Molten-Salt Breeder "AMSB", and their combined fuel cycle system are explained.

KEYWORDS: ***energy demand, thorium, molten salts, fluorides, fission reactors, nuclear power plants, fissile producers, accelerator breeder, breeding fuel cycle, safety, research programs***

I. INTRODUCTION

The next (21st) century will be a transient period from fossil fuel era to solar era through nuclear energy. The present fossil fuel technologies induce, not only severe global environmental problems, but also regional difficulties connected with the effective and economical energy supply, which is necessary to overcome poverty and desertification, *i.e.* North-South problem.

However, the future nuclear energy technology as a measure of global problems in the medium and long terms such as 2020~2070 will not be established merely by some extensions or improvements of present technologies, due not only to recent several troubles of nuclear facilities, but also to the huge increase in amount of nuclear energy capacity such as 100 times more than the present, which means a need of 100 times safer technology[(1)].

This might require a technological revolution or a new philosophy on the following issues keeping deep connections among each other: (a) nuclear resources, (b) breeding fuel-cycle, (c) safety, (d) radio-wastes and their incineration, (e) nuclear proliferation and terrorism and (f) public and institutional acceptances related not only with the aboves, but also with the technological simplicity, flexibility and economy in the global applications.

A "nuclear energy system" should be a

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"nuclear chemical engineering facility", and essentially a "**chemical plant**". Following on this sense, more rational nuclear energy system should be developed for the global energy measure of the next century, fully re-examining the scientific and engineering efforts devoted in this century.

II. Energy Problems in Next Century

1. Global Energy Demand[2]

The growth rate of energy consumption in the world was about 2.3% per year in the past one hundred years. It means about 10 times increase after one hundred years. If this growth rate is also applied in the next one hundred years, the total energy demand surrounding the end of the next century will amount to about 115 TWth (th=thermal), which is about 9×10^{-2}% of the total solar energy input to the globe. The latter value is almost the limiting value of 0.1%, which is estimated by some specialists considering the influence on the world climate. Thinking of the local abnormal weather, about 100 TWth seems to be a sound maximum heat emission to the atmosphere.

At the end of the next century, there is a consensus that the world population may be saturated at the value between 10 and 15 billion. This means about 10 kWth or 7 kWth per capita on an average corresponding to the 2 times (10 billion) or 3 times (15 billion) increase in the world population, respectively. This level is still less than that of the present USA, which is about 10 kWth per capita.

Therefore, at least, this energy demand should be satisfied in the next century of the world. This is the target of this report. An anticipating global energy demand in the next century is summarized in connection with the prediction of the population in **Table 1**[2], which includes an average energy consumption per capita, too.

Table 1 World energy demands in next century

	population	energy* [TWth]	(nuclear)*3 [GWe]	energy per capita [KWth/cap]
1988	5.1 B	9.	(200)	1.8
2000	6. B	11.8	(300)	2.0
2035	8. B	26.	(2,700)	3.3
2050	10. B	37.	(6,000)	3.7
2065	12. B	52.	(13,000)	4.3
[2100	10~15 B	115. (=0.09% of solar)*2	(30,000)	12~7.6]

*) assuming the average growth rate of 2.3 % per year.
*2)assuming less than 0.1 % of total solar input energy to earth, which would be a limit allowable in climate and environmental influence.
*3)assuming from the prediction by Matchetti et al., IIASA [2].

If a sound industrial trend in energy supply is held in the next century, it will conform to the logistic prediction made by Marchetti *et al.* as shown in **Fig. 1**[3], where the amount of primary energy is plotted as fraction "F" of the total energy market. The dotted curve of nuclear (fission) energy is added by the authors. The total electric capacity of the nuclear fission energy will become about 300 GWe in 2000, about 2,700 GWe in 2035, and about 13,000 GWe in 2065 as shown in Table 1.

2. Nuclear Proliferation and Terrorism

For the improvement of this issue the Pu fuel cycle strategy should be criticized, and Th cycle would be recommended due to the high γ-activity of polluted fissile ^{233}U, which is effective for the safeguard and detection[4].

3. Environmental Problem

This significant nuclear energy growth rate presented here seems to be necessary not only for securing the healthy, cultural and economical life, but also for reserving the sound environmental conditions in the world by the diminution of pollution including the "Greenhouse Effect"[5], although the solar energy utilization could become more impor-

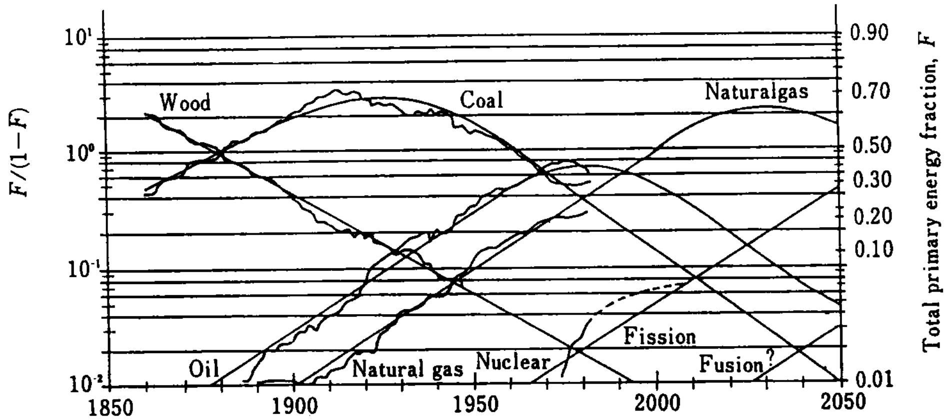

Fig. 1 Historical trend in energy substitution

tant toward the end of the next century because of the limitation on heat emission (*cf.* the item (*2) in Table 1).

4. Radio-waste

In this issue pure fusion technologies will have some advantages. However, it should be recognized as a long term program, because the energy gain even in the DT-fusion reaction is about one tenth of fission reaction which need not any essential energy input. In fission energy program, the following points should be considered for the radio-waste minimization:

(i) Minimization of nuclear chemical productions such as fission and spallation products, and trans-uranium elements: A higher thermal efficiency in electric generation will contribute to minimize F.P. production. Thorium fuel cycle will have a big advantage to minimize or eliminate trans-uranium elements.

(ii) Mimimization of spent fuel materials: All irradiated fuel material should be fully used as far as possible and not be easily sent to graveyards. They should be fully burned out establishing a complete breeding fuel cycle (*cf.* Chaps. III & VIII).

(iii) Minimization of fuel reprocessing steps and of reactor maintenance works.

5. Doubt on "Fission Breeder" Concepts

It seems that the establishment of fission breeders such as LMFBR is a world's consensus as one of the better solutions of energy and environmental problems. However, this concept might yield not only economical difficulty[6] but also several practical misfits related to public requirements[2]. Numerical examination is shown in APPENDIX.

(a) The first problem is a doubt whether the reactor could achieve the necessary growth rate as shown in Table 1, due to its **low breeding power**. This has been examined applying the following scenario acceptable as a sound example: all reactors until the year 2019 are thermal LWR of 1 GWe producing 0.25 t/yr of Pu, and all renewed and added reactors after 2020 are 1 GWe LMFBR, which need 5 t of Pu for initial inventory. This concludes the necessity of a very short doubling time such as 6∼8 years in fissile breeding, and fully prepared reprocessing plants. These seem to be impossible.

(b) Power stations should be simple facilities, **flexible in power size**, responding the natural demand of world utilities. It would not be acceptable that all stations become **sophisticated large high-performance** LMFBR.

(c) The integrated Pu inventory necessary for FBR in minimum would be 11,000 and 28,250 t at 2036 and 2049, respectively. **Handling and transportation of such a huge amount Pu** would be fatally troublesome. (In practice, the shortage of Pu will appear at 2037 even using FBR of 20 years in doubling time.)

Molten-Salt Breeder Reactors (MSBR) such as designed by Oak Ridge National Lab. (ORNL), USA[7] also will not be acceptable by the items of (a) & (b) due to the long doubling time of about 20 years.

6. Establishment of More Practical Breeding Fuel-cycle

The fission energy technology holds an unique position due to its nature of "energy rich" and "neutron poor". The latter will require helps from the other nuclear technologies such as nuclear spallation, D-T fusion, *etc.*, to get enough fissile materials to establish real "breeding fuel cycle" (*cf.* Chap. Ⅷ).

7. Basic and Engineering Safety

The significant improvement such as 100 times better than the probability of the present level of 10^{-4} severe accident per reactor-year operation will be required to avoid total refusal by public[1]. The following points are essential: simplicity in operation and maintenance, flexibility in power size, approach to civilian life area, and essential improvement in economy.

This might be established only by the essentially new safe reactor concept composed of chemical inert fluid-fuel, which has no radiation-damage, negligible excess nuclear-reactivity (nearly no control-rods), and low power density in value without harming its economy. [High Temperature Gas-cooled Reactor or Pebble-Bed Reactor which are more acceptable in safety issue than the other solid fuel reactors would not be suitable for establishing a breeding fuel cycle, too.]

Ⅲ. New Philosophy: THORIMS-NES

Thorium Molten-Salt Nuclear Energy Synergetics

Now, to solve the several energy problems shown in Chap. Ⅱ and **Fig. 2**, a new philosophy should be proposed to the public. The

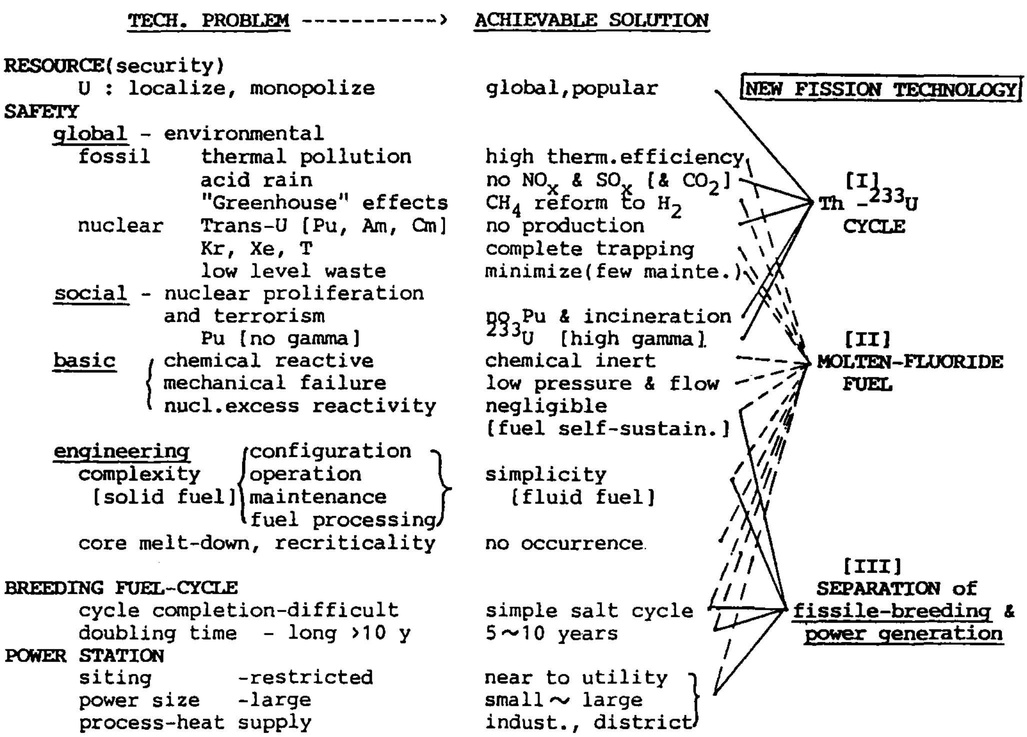

Fig. 2 Global energy problems and achievable solutions by New FISSION TECHNOLOGY [THORIMS-NES]

new proposal depends on the following three principles:

[I] **Thorium utilization**

Natural Th has only one isotope: ^{232}Th, which can be converted to the fissile ^{233}U in the similar manner as ^{239}Pu converted from ^{238}U. Thorium is more abundant and non-localized resource than U.

[II] **Application of molten-fluoride fuel technology**

The molten salt ^{7}LiF-BeF_2 (**Flibe**—named by ORNL) is the significantly low thermal-neutron capture cross section material (*cf.* **Table 2**) and the best solvent of fissile and fertile materials. This functions as one of the best liquids useful for nuclear fuel, blanket or target keeping coolant function in parallel.

Table 2 Natural elements and practically separable isotopes having tiny thermal neutron absorption cross-sections

		(natural abundance)	cross-section m barn	(natural element)
1.	$_{8}O$		0.19	
2.	$^{2}_{1}H$ [D]	(0.0148%)	0.519	($_{1}H$ 332.6)
3.	$_{6}C$		3.53	
4.	$^{11}_{5}B$	(80.0%)	5.5	($_{5}B$ 767,000)
5.	$_{2}He$		6.9	
6.	$_{4}Be$		7.6	
7.	$_{9}F$		9.6	
8.	$_{83}Bi$		33.8	
9.	$_{10}Ne$		39.	
10.	$^{7}_{3}Li$	(92.5%)	45.4	($_{3}Li$ 70,500)
15.	$_{40}Zr$		185.	
17.	$_{1}H$		332.6	
16.	$_{13}Al$		231.	
67.	$_{17}Cl$		33500.	

[III] **Separation of fissile producing breeders (process plants—MSB) and power producing reactors (utility facilities—MSR)**

It will be essential for the global establishment of breeding fuel cycle flexibly applicable in any area.

This philosophy was named as "**Thorium Molten-Salt Nuclear Energy Synergetics**" [**THORIMS-NES**][(2)(8)(9)]. Its main benefits will be briefly understood from the summary in Fig. 2.

IV. SIGNIFICANT ASPECTS OF MOLTEM-SALT REACTOR TECHNOLOGY

1. Molten Fluoride Fuel Concept[(7)(8)]

Some brief historical presentation of significant studies on MSR concept are given in **Fig. 3**. MSR is classified to fast, epithermal and thermal reactors according to the kinds of neutron spectrum utilized. Fast or epithermal reactors not having any solid moderators might have more attractive due to the simple structure and high performance. However, it seems that no one has succeeded to present any practical designs yet. Fluoride salt fueled thermal MSR would be the most promising reactor concept.

The "**Flibe** (**LiF-BeF$_2$**) base molten-salt fuel" concept has already been established its sound technological basis by the brilliant long effort of ORNL, throughout 1947∼1976. It depends on the successful operation of **MSRE** (Molten-Salt Reactor Experiment, 7.5 MWth) in 1965∼1969, and on their intensive R&D works on **MSBR** (1963∼1976)[(7)]. The excellent final summary on ORNL works could be found in *ORNL/TM*-7207[(10)]. The short explanation will be given below.

However, the practical deployment of **MSBR** is not easy because of the following difficulties: (a) necessity of core-graphite exchange in every 4 years, (b) development of continuous chemical process *in situ*, and (c) necessity of improving doubling-time from the present estimation of 20 years, and (d) flexibility in power size (*cf.* Sec. II-5).

Now, the principle [III] should be accepted.

2. Flibe-base Molten-Salts

The molten ^{7}LiF-BeF_2-$^{232}ThF_4$-$^{233}UF_4$ system will be one of the better and idealistic nuclear materials as a carrier of Nuclear Energy Synergism due to the following characteristics:

(1) A triple-functional medium working as
  - **A. nuclear reaction medium**—fuel, blanket and target
  - **B. heat transfer medium**—coolant
  - **C. chemical processing medium.**

(2) No radiation damage as an idealistic

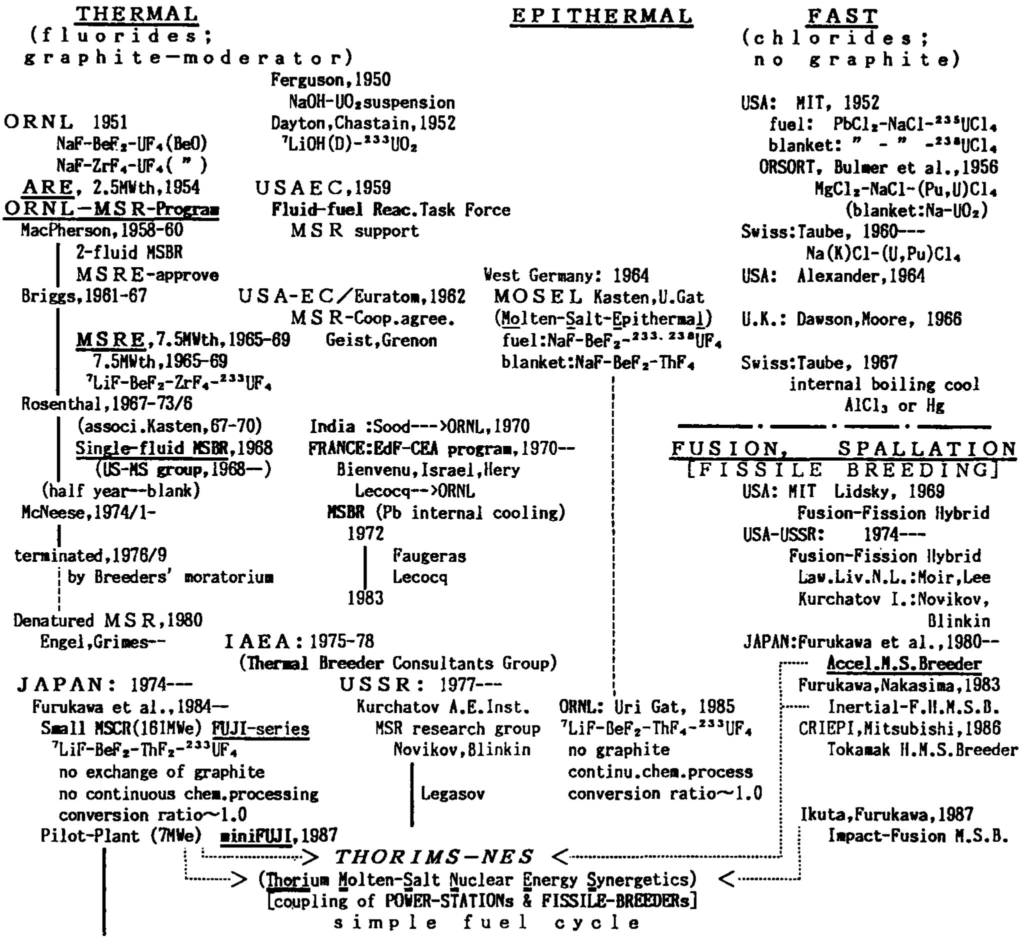

Fig. 3 Historical presentation of Molten-Salt Reactor concepts

"ionic liquid".

(3) High safety and economy assurance promised by chemically inert, low vapor pressure, moderate viscosity and thermal conductivity, and high heat capacity in fairly high working temperature (770～1,070 K).

Sodium technology has three severe disadvantages such as (i) high chemical reactivity, (ii) high thermal shock, and (iii) oxidized vapor condensation in gas spaces, different from molten salt technology.

Molten-Salt technology is much easier than water (high pressure, corrosive and reactive with Zircaloy) and Na technologies.

3. Graphite

Graphite is the best material for neutron moderator and reflector in MSR (*cf.* Table 2), and is really compatible with molten fluorides in bare state. This is a great benefit in designing MSR. The satisfactory materials have been developed by the efforts of USA and France[(11)].

4. Containment Materials

Fortunately, an easy manufacturable, weldable and high temperature resistant (till *ca.* 1,100 K) alloy compatible with molten fluorides has been developed basically according on the proposal of Dr. H. Inouye, ORNL. This alloy, Hastelloy N (Ni-(15～18)Mo-(6～8)Cr in %), has been modified by adding about 1% Nb for the protection of Te attack on surface. This was experimentally supported by Kurchatov Institute[(12)].

This alloy needs not to be introduced in reactor-core region suffering severe irradiation and thermal shock. (MSR-core is occupied by graphite and fuel salt only.) The thinnest material will be one or 2/3 in. diameter tube in the heat exchanger. Therefore, the reactor design would be very simple and easy. In practice, the corrosion of Hastelloy N by introduced contaminants such as air and moisture was surprisingly small and negligible even at the small experimental reactor MSRE. Commercial power stations would be much safer against corrosion.

5. Confinement of Fuel Salt and Fission Products

The triple confinement of fuel-salt is soundly established by (i) reactor vessel, (ii) high-temperature containment, in which the atmospheric temperature is kept in about 500°C (*cf.* Photo. 2), and (iii) reactor container, in contrast with the solid-fuel reactors, which are also depending on the triple confinement. However, in the latter the troublesome thin clad-tubes are placed in the high neutron flux, high flow velocity and high thermal shock region of core.

The **fission product elements** in fuel salt can be classified to the 4 groups concerning with their chemical behaviors;

Group (1): Rare gas elements……(practically no solubility in salt)
Group (2): Stable salt elements such as rare earths, Zr, Ba, Sr, Cs, Br, I……(no chemical problems)
Group (3): Noble (insoluble) metal elements such as Mo, Nb……
Group (4): Unstable salt element such as Te, O, H, D, T…….

Group (1) elements are removed semi-automatically from salt. This is great advantages for preventing their harmful neutron-absorption and their release to environment in accidents.

Group (2) elements are trapped definitely as ionic species in salt, and do not present any trouble.

Group (3) elements behave as (i) undissolved floating materials (shifting to cover gas system), and (ii) plate-out materials on graphite, or (iii) on metal surfaces. Their mass ratios were 50 : 10 : 40% in MSRE. Some part will be filtrated out. The plate-out materials on graphite will be removed for reuse by grinding 0.5 mm in depth to diminish the quantity of radio-waste.

Group (4) elements, Te might induce a shallow surface brittleness on Hastelloy N. However, it is effectively protected by controlling electro-chemical redox-potential of fuel salt, and modifying the alloy composition by the addition of about 1% Nb.

Tritium will be effectively transferred to coolant salt NaF-$NaBF_4$ (8~92 mol%) through the heat-exchanger tube-wall exchanging with H of water content (200 ppm) in coolant salt, and will be recovered in He cover-gas phase effectively minimizing T release to environment. MSR system only has been solved the problem of tritium management among all proven reactor systems.

V. POWER PRODUCING REACTORS—MSR

1. Design of Simplified Small Molten-Salt Power Station "FUJI-II"

Depending on the rational MSR technology explained above, the developmental program of ORNL was mostly concentrated on the large breeding reactors. Therefore, the design study of Small Molten-Salt Power Station, named as **FUJI-series**, has been proceeded by Furukawa and his group[9][13][14].

The main design principle is (a) single-fluid, multi-region type, graphite moderated core, (b) no need of core-graphite exchange keeping low power density, (c) no continuous chemical processing, except the simple removal of fission-gases (Kr, Xe) and tritium, and (d) high conversion ratio and low fissile inventory. The purpose is the establishment of nuclear power stations more simplified in the maintenance and operation modes and more flexible in size than MSBR.

Finally, as a standard design, the fuel self-sustaining compact MSR, 350 MWth FUJI-II was proposed[13][14]. Its important parameters and conceptual model figures are shown in **Table 3** and **Photo. 1**, respectively. (About 90 % in reactor vessels is occupied by moderator

Table 3 Important characteristics of Small Molten-Salt Reactors

	standard power station (fuel self-sustaining) <u>F U J I-II</u>	pilot-plant (super-compact) <u>miniFUJI-II</u>	experimental reactor(ORNL) (operated in 1965~69) <u>M S R E</u>
heat capacity (MWth)	350.	16.7	7.3
electric power(netMWe)	155.	7.	--
thermal efficiency(%)	44.3*	42.*	--
reactor vessel size(m)			
(diameter x high)	5.5x 4.1	1.8x 2.1	1.45x 2.2
core: max. diameter(m)	1.4, 3.4+	0.6	1.14
graphite frac.(vol%)	93, 90+	90	77.5
blanket: thickness (cm)	35	20	---
graphite frac.(vol%)	65	70	---
reflector:thickness(cm)	68	40	7 ?
high temp.containment(m)			
(diameter x high)	12. x 8.	3.7x 3.2	(5.8 x 7.2)
core/blanket power density			
average--peak(KWth/l)	9.5--17.5	16.4--24.9	2.9--6.6
neutron flux: ($n/cm^2 sec$)			
maximum thermal	2.4 $\times 10^{14}$	0.58 $\times 10^{14}$	0.5 $\times 10^{14}$
max.graphite(>50KeV)	0.8 $\times 10^{14}$	0.75 $\times 10^{14}$	0.3 $\times 10^{14}$
fuel: conversion ratio	1.002**	0.58**	---
^{233}U inventory(kg)	370.	27.	32.
[per 1GWe](ton)	[2.4]	[6.4]	[--]
^{232}Th (ton)	20.1	0.65	---
fuel salt: $^{233}UF_4$(mol%)	0.22*1	0.47*1	0.14*2
total volume (m^3)	13.7	0.45	2.1
[reactor](m^3)	[9.7]	[0.30]	[0.54]
flow rate (m^3/min)	33.2	1.59	4.5
temperature (°C)	585-->725*	560-->700*	632-->654
main piping(inn.dia.cm)	25.	8.	15.
graphite inventory(ton)	161.6	8.8	3.7

\+ the second core-zone.
* 2% up by "ultra-ultra supercritical steam-turbine cycle[593°C,317 atm]" at the fuel temperature: 725°C.
** in the stationary state after 500 days from start up.
*1 $^7LiF - BeF_2 - ThF_4 - {}^{233}UF_4$ = (72-x) - 16 - 12 - x [mol%]
*2 $^7LiF - BeF_2 - ZrF_4 - UF_4$ = 64.5 -30.2- 5.2-0.154[mol%](91% ^{233}U)

& reflector graphite.)

This fuel self-sustaining character, meaning no need of fissile supply except Th-supply of 400 g/d, is strictly effective for (i) minimization of excess-reactivity, (ii) simplification of operation and maintenance, (iii) minimization of fuel transportation and radiowaste, too. This contributes to the improvements of nuclear-proliferation and -terrorism resistances, and of coupling performance with fissile producing breeders (MSB), which is needed only for initial fissile supply.

2. Reactor Chemistry

Its <u>reactor chemical behaviors</u> will be surprisingly simple and manageable[(10)(15)]. The behavior of fission products accumulated in the fuel salt of "FUJI-II" at the final stage was analyzed in detail; the final amounts are $CsF=0.06$ mol%, $SrF_2=0.05$ mol%, $BaF_2=0.005$ mol%, $CeF_3=0.17$ mol% and $ZrF_4=0.37$ mol%, for example. The total amount of trivalent fluorides such as rare earth' will be fairly large, but still soundly soluble in fuel salt. In conclusion, the fission products will need not any

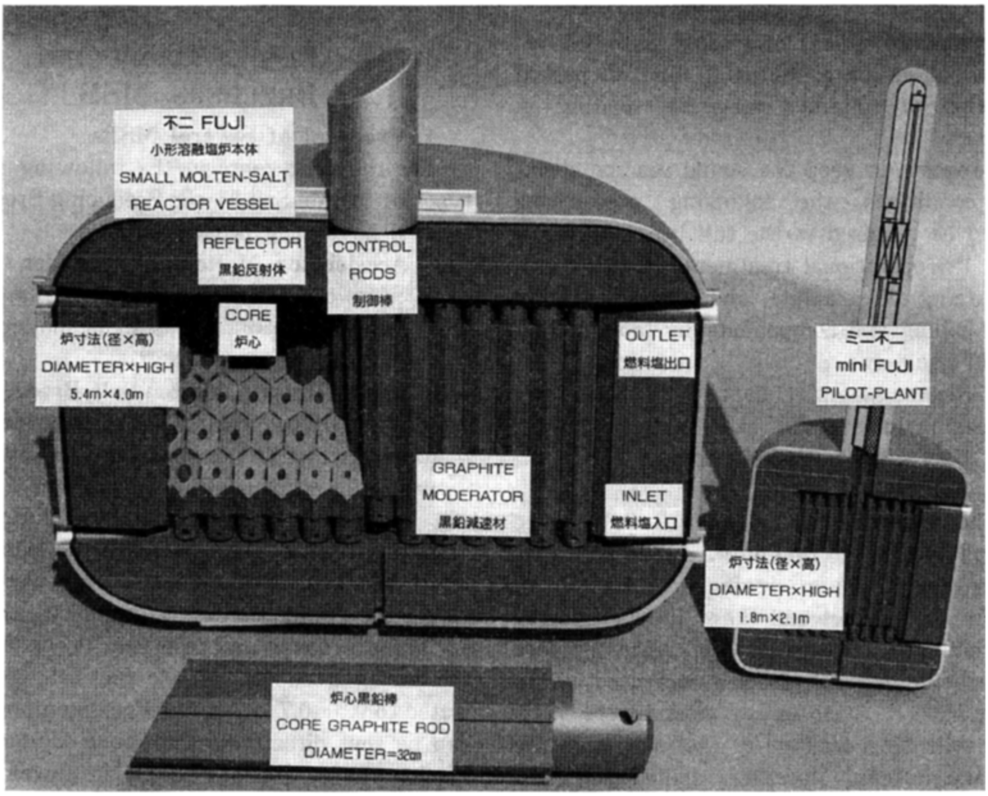

Photo. 1 Vertical cross-sectional view of reactor-vessel "FUJI-II" and "miniFUJI-II" in same scale-sized models

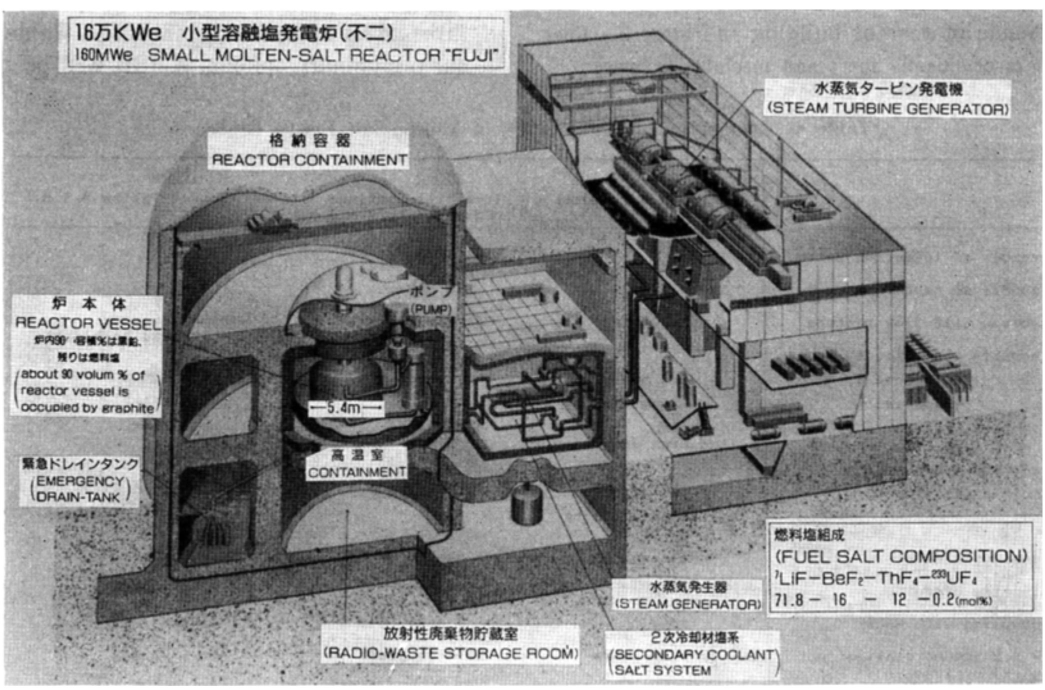

Photo. 2 General view of Small Molten-Salt power station ["FUJI-II"]

separation except (1) originally designed removals of fission gases and T, and (2) partial filtration of solidified materials floating in fuel salt.

However, to keep the sound reactor chemical conditions, the following precautions should be required in the full life:

(i) Pre-operational treatments such as outgassing of graphite, and surface-cleaning of piping and components by molten LiF-BeF_2 salt

(ii) Redox potential control of fuel salt keeping its ratio UF_3/UF_4 in 0.02～0.17

(iii) Fuel salt composition control.

3. Safety

The general view of "FUJI-II" Power Station is shown in **Photo. 2.** Its excellent safety features are significant (*cf.* Fig. 2). Entire stop of fuel salt flow in the reactor will not result in any damage, since the core graphite temperature is only increased up to about 1,300 K. One of the most severe accident will be a leakage of fuel salt from the primary system. However, spilled salt will be caught on the catch basin and automatically guided to the emergency tank, which is dipped in a big water pool as shown at the left side of reactor building in Photo. 2. (Fuel salt is chemically inert and insoluble in water.)

VI. Fissile Producing Breeders—MSB

1. Proposal of Several MSBs

For the next century, the following three types of Molten-Salt Fissile Producing Breeders (MSB's) have been proposed by us:

A) **Accelerator Molten-Salt Breeder (AMSB)**: by the neutrons generated from spallation reaction of Th nuclei in molten salt with 1 GeV protons[17)~(20)],

B) **Impact Fusion Molten-Salt Breeder (IFMSB)**: by the application of new ideas of axially-symmetric mass-driver and shaped-projectile accomodating DT-pellet[21)],

C) **Inertial-confined Fusion Hybrid Molten-Salt Breeder (IHMSB)**: by the adoption of the first wall of molten-salt waterfall for the elimination of radiation damage[20)].

These systems are designed to guarantee the straight supply of the fuel salt concentrated to 0.5～0.7 mol% in $^{233}UF_4$ content, which can be sent directly to the above Molten-Salt Fission Power Stations fueled in lower concentration (0.2～0.3 mol%). They should be developed until 2010～2030 of the next century.

A brief comparison of general performances of three **MSB** systems are shown in **Table 4.** In our preliminary opinion, **AMSB** will be the

Table 4 Comparison of three types of Molten-Salt Fissile Breeders

	AMSB (Accele.M.S.Breeder)	IHMSB (Inertial-confined Fusion Hybrid M.S.B.)	IFMSB (Impact-Fusion M.S.B.)
number of injection ports	1	5 ～ 10	1
generated neutron energy	500 ～ 1 MeV	14 MeV～	14 MeV～
molten salt flow pattern	simple vortex	complex spiral	simple vortex (He-gas bubbling)
neutron shielding thickness			
molten-salt	2.2 meter	1 meter	1.5 meter
graphite	1 meter	1 meter	1 meter
reacter physics	known (inc.high ene. comp.)	not clear	not clear
reactor chemistry			
Tritium	10^3 Ci/d	10^7 Ci/d	10^7 Ci/d
	sallation products (complex but few)		Th debris (deposit)
energy balance	not easy	sound	good
fissile breeding	weak	medium	strong
coupling with MSCR	easy	easy	easy
R & D (commercialization)	15 ～ 25 years	25 ～ 35 years	20 ～ 30 years
IN CONCLUSION	sound (low breeding)	expensive(difficult)	simple(high breeding)

most sound and promising concept. However, **IFMSB** would be able to become a "dark horse".

In this report, only AMSB will be explained in a little more detail.

VII. ACCELERATOR MOLTEN-SALT BREEDER (AMSB)

1. Technical Problems in Accelerator Breeders (or Spallators)

The physical principle of spallators has been clarified by the successful Material Test Accelerator Project (1949~54) at Lawrence Radiation Lab.[22], and by the intense basic researches in Chalk River Nucl. Labs. (1952~84)[22][23], Brookhaven Nat. Lab.[24], ORNL[25], Kurchatov Inst. of Atomic Energy[26], *etc.* (*cf.* Ref. (22)), and also in the development of Molten-Salt Technology by ORNL[7][10].

The spallation reaction of heavy nuclei with 0.8~2 GeV proton will be one of the most effective methods for production of neutron, which is useful for fissile material production (*cf.* **Fig. 4**).

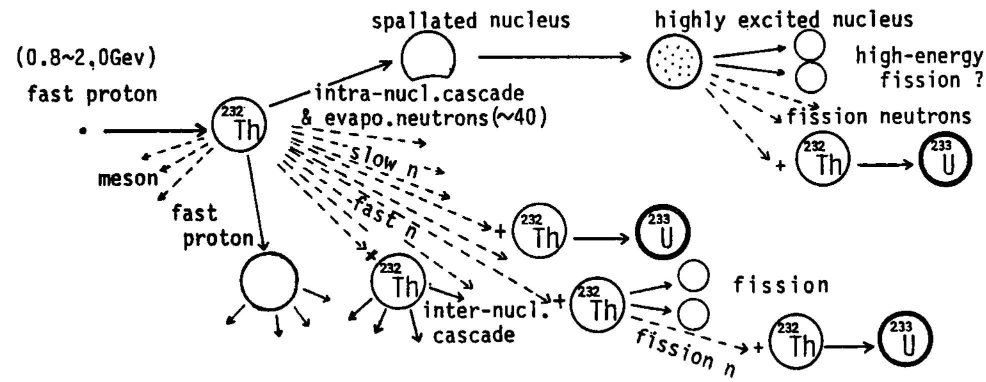

Fig. 4 Schematic figure of spallation nuclear reaction by high energy proton in case of Th thick target

The technical problems in accelerator breeders[22] are the developments of high-current proton accelerator and target/blanket system. In the design there are problems such as:

(a) radiation damage, (b) heat removal, (c) irradiation material shuffling, and (d) spallation chemistry of target/blanket.

Almost all would become simpler by the application of the single-fluid type M. S. target /blanket concept[17]~[20]. The reasons are the followings:

(1) Th (or U) is chosen as a target nucleus, and single-phase molten fluoride system including ThF_4 (or $^{238}UF_4$) will serve as chemically inert target and blanket medium.

(2) Molten-Salt target/blanket system will promise simple and safe configuration, no radiation damage, easy heat removal, automatic material shuffling and easy chemical processing.

(3) The elimination of beam window will be possible due to the low vapor pressure of salts.

(4) The degradation of neutron yield than that in the liquid metal target will be recovered by elimination of window and structural materials inside the target/ blanket phase and by the inclusion of Be nuclei due to the BeF_2 component addition.

Often liquid Bi, Pb or their alloys are chosen as target materials. However, they have several disadvantages as follows: (i) container materials: applicable W, Mo graphite or carbon-steel are all not suitable for sound construction of nuclear facilities; (ii) high density and high pumping power; (iii) complex chemical behaviors of spallation-products such as Po vapor problem, formation of insoluble metallic or non-metallic compounds, *etc.* will produce several troubles; (iv) the choice of "liquid" will not be extended to fertile blanket zone, where now it is necessary again to consider (a) cladding or barrier wall, and its radiation damage and neutron-loss by them, (b) shuffling and exchange, (c) heat removal, (d) chemical processing, *etc.*.

2. Design Principle of AMSB

AMSB is composed of three parts: an 1 GeV–300 mA proton accelerator (LINAC *etc.*), a molten fluoride target/blanket system and a heat transfer and electric power recovery system. The schematic figure of its main parts is shown in **Fig. 5**. The size of target salt bath is 4.5∼5 m in diameter and 7 m in depth. To keep them smaller in size, comparatively slow proton of 1 GeV was chosen and the current was specified mainly from the thermal output. Inside of the breeder vessel made of Hastelloy N (Ni-Mo-Cr alloy) is covered by thick graphite blocks immersed in salt. The target salt is introduced from the upper part forming a vortex of about 1 m in depth of salt. The proton beam is directly injected in off-centered position near the vortex bottom, reducing the neutron leakage and improving the heat dissipation.

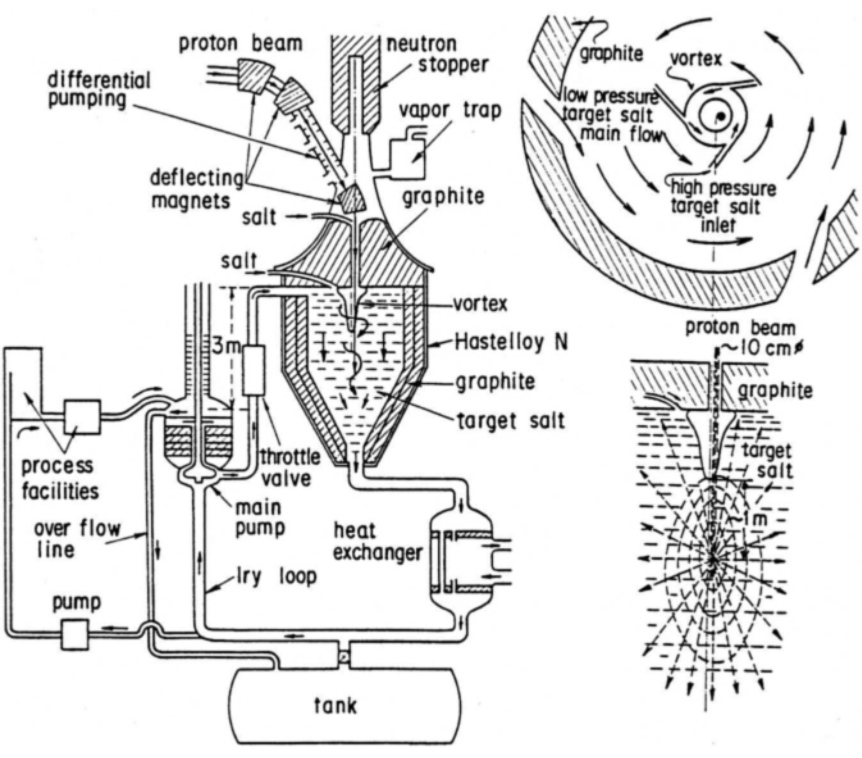

Fig. 5 Schematic figure of molten-salt target/blanket system of Accelerator Molten-Salt Breeder (AMSB)

This target/blanket system is subcritical, and not influenced by radiation damage. The heat removal could be managed by the dynamic salt, which is dilute in heavy isotopes. The shuffling is automatic. Except unknown engineering abont a beam injection port, this simple configuration will be manageable in engineering, depending on MSR technology in general.

3. Selection of Molten-Salt Compositions

Considering the phase diagrams of tertiary salts including U and Th fluorides, several candidate salt compositions are proposed. Now, the composition of

$^{7}LiF\text{-}BeF_2\text{-}ThF_4$ (64-18-18 mol%)

was chosen as a **standard target/blanket salt**, which is not far from the standard fuel salt composition of Table 3 for the MSR.

The neutron yield will be improved by the addition of fissile nuclei to target/blanket system. By the calculation of Nakahara it was verified that the replacement of 3 a/o of Th atoms in fluoride salts into fissile atoms may induce about 50% increase in neutron yield. Therefore, neutron yield of standard type AMSB will be increased by about 45% in fissile material production by the replacement of about 0.5 mol% of $^{233}UF_4$ into ThF_4 (corresponding to 2.8 a/o in Th), that is

$^{7}LiF\text{-}BeF_2\text{-}ThF_4\text{-}^{233}UF_4$ (64-18-17.5-0.5 mol%).

This type of AMSB using fissile added salts was named as the **high-gain type AMSB**[18], which is effective for the improvement of electric power recovery by an increased heat generation. Their predicted performances are shown in **Table 5**.

Table 5 Predicted reactor performances of standard type and high-gain type AMSB

<table>
<tr><th></th><th>standard type AMSB</th><th>high-gain type AMSB</th></tr>
<tr><td>Proton beam</td><td>1GeV 300mA</td><td>1GeV 300mA</td></tr>
<tr><td>target salt</td><td>$^{7}LiF\text{-}BeF_2\text{-}ThF_4$
64-18-18 m/o</td><td>$^{7}LiF\text{-}BeF_2\text{-}ThF_4\text{-}^{233}UF_4$
64-18-17.5-0.5 m/o</td></tr>
<tr><td>melting point (T_m)</td><td colspan="2">540°C</td></tr>
<tr><td>density at T_m+100°C</td><td colspan="2">$2.7 g/cm^3$</td></tr>
<tr><td>viscosity coefficient</td><td colspan="2">12-14 c poise (600°C)
6- 7 c poise (700°C)</td></tr>
<tr><td>salt temperature</td><td colspan="2">inlet 580°C outlet 680°C</td></tr>
<tr><td>salt volume</td><td colspan="2">$90m^3$</td></tr>
<tr><td>weight (Th)</td><td colspan="2">243ton (126ton)</td></tr>
<tr><td>neutron production per proton</td><td>25~40</td><td>36~58</td></tr>
<tr><td>^{233}U production</td><td>0.57~0.92 ton/y</td><td>0.82~1.33 ton/y</td></tr>
<tr><td>fissile inventory
(doubling time)</td><td>0~0.5ton
(0~1 y)</td><td>3.4ton
(2~4 y)</td></tr>
<tr><td>spallation products
(fission prod.
spallation prod)</td><td>~91Kg/y
~46Kg/y
~46Kg/y</td><td>~220Kg/y
~174Kg/y
~46Kg/y</td></tr>
<tr><td>thermal output
electric output</td><td>1000~1500MWth
430~650MWe</td><td>1400~2100MWth
600~900MWe</td></tr>
<tr><td>(linac consump.)</td><td>(600~700MWe)</td><td>(600~700MWe)</td></tr>
</table>

4. Nuclear- and Reactor-Chemical Aspects of AMSB

The mass distribution of spallation products is shown in **Fig. 6**. The absolute amount of spallation products will be smaller than the fission products in fission reactors. The radio-waste concentration ratios in molten salts between AMSBs and MSBR of ORNL are only 1/14 for standard-AMSB, and 1/6 even for high-gain AMSB.

The behavior of produced elements was examined in the preliminary but considerable level[18] in the similar manner as Sec. IV-5. It is understood that almost all the products would be manageable except oxygen, which may be gettered by Ti-metal hot trap.

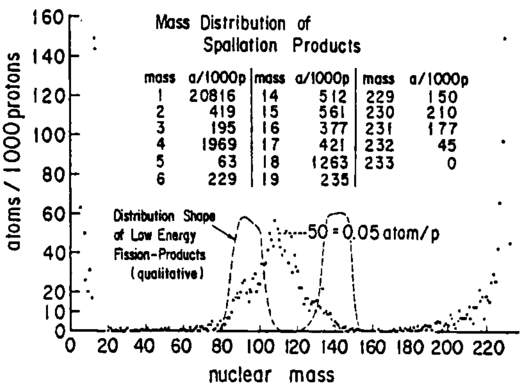

Fig. 6 Mass distributions of spallation- and fission-products in target/blanket system of Accelerator Molten-Salt Breeder

VIII. ESTABLISHMENT OF Th-^{233}U FUEL CYCLE COUPLING WITH FISSILE PRODUCING BREEDERS

As shown in **Fig. 7**, each **"Breeding & Chemical Processing (Regional) Center"** settled in 10~20 sites in the world will accommodate 4~10 Molten-Salt Fissile Producing Breeders (MSB), two Chemical Processing Plants and one Radio-waste Managing Plant[9]. These Regional Centers should and might be heavily safeguarded. They are very simply connected with the MSR power stations by the molten salt fuel cycle as mentioned in the above section. Other types of fission reactors such as the present U-Pu cycle solid-fuel reactors could also be supported by the fissile materials (^{233}U or ^{239}Pu) produced by MSB, if necessary or in the transient stage. Spent solid fuel will be treated by the Dry Processing Plant applying molten fluoride technology, which is now being developed in Dimitrovgrad, USSR[27].

The target/blanket salts of high ^{233}U concentration such as 0.5 mol% prepard in MSBs could be directly used as a fuel salt for MSR constituting the **simplest Molten-Salt fuel cycle.**

The excess or dirty fuel salts (not so much) from MSRs will be sent back to the Chemical Processing Plant of Regional Center, where the components will be easily isolated into (A) fissile $^{233}UF_6$, (B) radio-waste (F.P.) and (C) fertile salt (^{7}LiF-BeF_2-ThF_4). The component (A) can be added to the supplying fuel salts after reduction, and (C) should be used for the preservation of increasing $^{233}UF_4$ concentration in MSB target/blanket salts, if necessary, by adding some constituents.

In the matured ^{232}Th-^{233}U fuel cycle system constituted by some MSBs and small or large MSRs, the total electric power capacity of fission reactors supported by a given amount of fissile materials is proportional to

$$S=\eta_e/(1-CR), \qquad (1)$$

where η_e and CR are electric conversion efficiency and fissile conversion ratio, respectively[9]. As shown in **Table 6**, the effect of CR is significant, whose improvement should be encouraged.

Table 6 Relative capacity of fission reactors supported by given amount of fissile materials

		LWR	HWR	HTGR	MSR		
electric power	(MWe)	1100	745	1160	7	155	155
therm.efficiency	η_e	0.33	0.29	0.39	0.42	0.44	0.44
conversion-ratio	CR	0.55	0.80	0.65	0.58	0.95	1.0
$S = \eta_e/(1-CR)$		0.73	1.45	1.11	1.0	8.8	∞

For establishing an excellent molten salt fuel cycle, MSR should have higher η_e and CR. Already a semi-ideal MSR design work was performed even small in size as explained in Chap. V. FUJI-II has the fuel self-sustaining characteristics, which means practically CR=1.0 [*in situ*] resulting $S=\infty$ (*cf.* Table 6). The amounts of initial inventory and addition

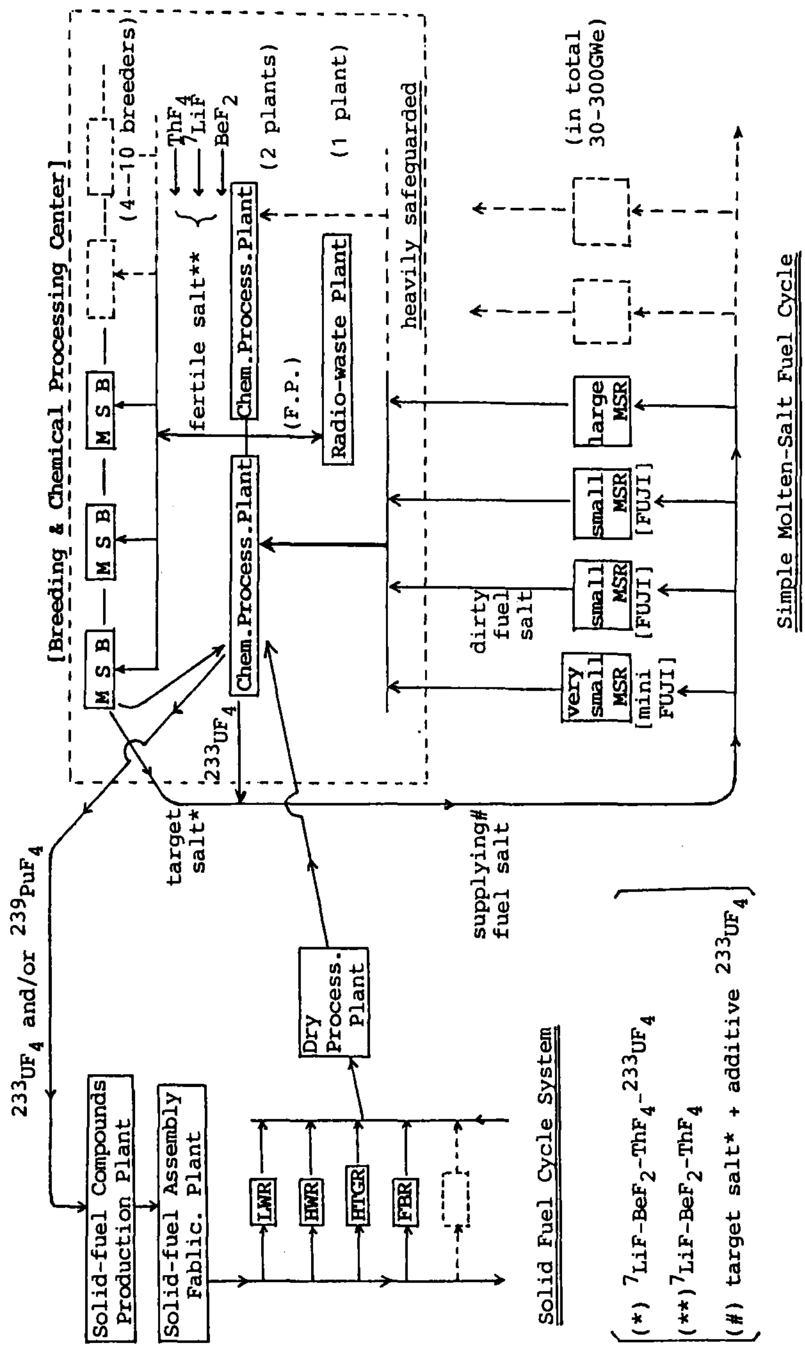

Fig. 7 Simple molten salt fuel cycle in Thorium Molten-Salt Nuclear Energy Synergetics [THORIMS-NES] composed of MSBs and MSRs

in transient stage are also excellently small. Its simple structure, easy operation and maintenance are enhancing its safety and economy. Now, the active R & D of MSB should be encouraged more for the next century.

IX. TECHNOLOGICAL FEASIBILITY IN THORIMS-NES

1. Advantages

In conclusion, integrating with the above design studies, the following excellent results in the THORIMS-NES will be obtained as shown in Fig. 2. The most significant ones are:

(1) Practical utilization of the rich and popular resource thorium, liberating the use of the localized and nearly monopolized uranium,

(2) Global (environmental) safety aspects: (i) low thermal pollution by high thermal efficiency (43~46%), (ii) no production of Trans-uranium (Pu, Am, Cm) elements, (iii) minimization of radio-waste,

(3) Social safety aspects (nuclear proliferation and terrorism): (i) practically no Pu and easy incineration of Trans-U elements, (ii) easy safeguard by high γ-activity of ^{233}U fuel, (iii) scarce fissile fuel transportation,

(4) Basic (technological) safety aspects: (i) chemical inertness, (ii) low pressure, (iii) negligible excess nuclear reactivity,

(5) Engineering safety aspects: (i) simplicity in configuration, operation, maintenance and fuel processing, (ii) no core melt-down and recriticality,

(6) Breeding fuel-cycle: (i) easy completion of simple salt cycle, (ii) effective doubling time of 5~10 years in fissile production,

(7) Power station: (i) no restriction in siting, (ii) flexible power size, (iii) process heat supply till *ca.* 1,070 K.

2. Technological Handicaps

From the above explanations the rational features of molten fluoride fuel concept will be clear. However, in general, there are several doubts on it. Some conjectures on their reasons will be tried in the followings, although almost all of them seem to be "aberrations", that is, no essential reasons.

(i) Non-popularity of fluid-fuel reactor concepts: Till now all developed reactors except MSR were solid-fuel reactors. However, nuclear reactors are "nuclear chemical-reaction facilities, which should be "chemical plants". Solid-fuel reactors do not look like them. It seems not impossible that the solid-fuel concept would be "aberrations" or "cul-de-sac (blind lane)", in considering the difficulty of their fuel-cycle system establishment even spending 40 years or more.

(ii) Failure of the other fluid-fuel reactors: This is another problem apart from MSR. There are a lot of unsuccessful solid-fuel reactor types. However, in analogy with the other fluid-fuel reactors, many people are imagining that MSR also might have a difficulty in its container materials. This was exaggerated by the discovery of Te-attack phenomena on Hastelloy N after dismantling of MSRE on 1970. This was solved nicely during the final R & D stage (1972~76) in ORNL[28][10]. USSR research group of Kurchatov Inst. reconfirmed it by getting better results[12].

However, nobody seems to refer on it beyond 1970. (Almost all textbooks of nuclear reactors seem not to be preferable to make a new part of fluid-fuel concept due to the essential difference of design philosophy from the solid-fuel concept.)

(iii) No existence of fissile isotopes in Th resource: This is the reason that the first touch for fission-energy technology was from the U-Pu cycle, clearly relating with Nuclear Armament technology. However, for the matured fission-energy era, this might not be essential or rational as explained in Chaps. II & IV. After the choice of breeding fuel cycle concept, Th is equivalent with natural U and excellent in thermal reactor performance.

(iv) Only few people and few money on MSR-R & D: MSR-R & D has been fully performed by only one team in ORNL, except few works by France (CEA-EdF), India (Bhabha Centre), USSR (Kurchatov Inst.), Japan (JAERI, Tokai Univ., *etc.*) and so on. The number of personel in ORNL group was about 230 in maximum, and the sum of budget seems to be about \$130 M among 30 years (1947~76) including the construction of two experimental reactors (ARE and MSRE). However, the reports on MSR are huge and excellent. They look like threatening LMFBR, for which several hundred times bigger money has been spent already.

It should be understood that MSR program needs not any big money and people. The reason is its significant "simplicity" in scientific and technological bases, which mean few soft- and hard-wares (few components and instrumentations).

(v) Historical unhappiness in the Seventieth: The success of MSRE operation and MSBR design study was significant in the age of 1968~70. Therefore, many countries or groups such as MS Group of USA, France, EC, India, Japan, *etc.* were aiming to work with ORNL at that stage (*cf.* Fig. 3). US-Congress cut budget once on 1971 by the non interest of major plant-makers due to the plant simplicity (no expectation to get a profit from that construction) and their enjoyment of LMFBR jobs at that time, although it was recovered soon; and finally on 1976, MSR was terminated by the "Breeder Moratorium", *i.e.* President Carter's policy not depending on the technological reasons[29].

In conclusion, non-popularity of MSR would be gradually solved by the understanding of its necessity of a new rational nuclear energy, which is already starting. However, its real demonstration by a pilot-plant such as miniFUJI-II is essential.

3. Technological Problems

(1) Reactor Physics

MSR has a sufficiently thermalized neutron spectrum, in which neutronics is simple except the delayed neutron problem, and a small correction of critical mass is easy by the control of salt composition. The recent intense nuclear data study[30] will be useful. The neutronics and heat generation of a spallator (AMSB) should be more improved theoretically. However, the essential check should be done experimentally.

(2) Reactor Chemistry

The technology of chemical impurity monitoring[31] should be reconfirmed or developed. The data base of modified Hastelloy N should be established in high temperature.

However, the most important study is the tracing of long-term behavior (*cf.* Secs. IV-5, VII-4) of fission and spallation products by the experimental reactors (miniFUJI-II and AMSBE).

(3) Reactor Engineering

Construction or operation of the first experimental facilities will not have any severe difficulties. However, among and after the long endurance test operation the several engineering developmental efforts should be devoted for system simplification and technological maturing.

X. DEVELOPMENTAL PROGRAM OF THORIMS-NES

1. R & D Schedule

The preliminary developmental schedule of THORIMS-NES is shown in **Fig.** 8. The basic technology of molten fluoride fueled reactors has been prepared by the intense R & D study of ORNL in 1947~76. After confirming them, the first step will be the development of Small Molten-Salt Reactors as a flexible-sized simple power station, which needs not compete with proven Large Solid-fueled Power Reactors.

2. Pilot-plant "miniFUJI-II" of MSR

After about 3 years' works for general R & D and reactor design, the construction of pilot-plant named **"miniFUJI-II"**[14] could be started. Its reactor performances are shown in Table 3, comparing with FUJI-II and MSRE successfully operated in ORNL. Its conceptual model is shown in Photo. 1. This reactor is nearly the same in size as MSRE and the diameter of the main piping is about 8 cm, which is a half of that of MSRE. Therefore, its construction is essentially proven work.

The operation is easy in general and only including the 99% removal of fission-gases insoluble in salts, and supplying about 7.5 g/d (=2.7 kg/yr) of Th and 5.8 g/d (=2.1 kg/yr) of ^{233}U. The total demand of ^{233}U is only about 30 kg in initial stage followed by 2.1 kg each

```
            ( =====> construction; -----> operation )

                                     1992         1996         2000
A. General R & D..                    |            |            |
Fuel-salt loop            --------====>-|---------> -|------------|--
Coolant-salt loop         --------====>-|---------> -|------------|--
Electric generating       ----------====|>-----------|-----> -----|--->
      test loop              (200KWe)   |    ----====|>-----------|---->
 (Integral test loop)                   |      (2 MWe)
Materials                 --------->----|---------->
                       mechanical data

   SHORT TERM PROGRAM
B. Pilot-plant (miniFUJI-II)
Reactor design          -----------==|===>-------
Reactor mock-up            ----------|=====>-----|
Reactor remote mainte.  -------------|=====>-----|----     operation
Reactor const.& ope.                -|-----------|==>---------|----------

   MIDDLE TERM PROGRAM
C. Small Molten-Salt Power Station (FUJI-II)
Reactor design                       |    -------|=======>----|--
Reactor mock-up                      |         --|--======>---|-----
Reactor remote mainte.               |          -|-------===>-|-----  ope.
Reactor const.& ope.                 |           |  -----=----|======>-------->

   LONG TERM PROGRAM
D. Medium-, and Large-Molten-Salt Power Stations
                                     |           |
E. Fissile Producing Breeder Development(AMSB,IFMSB,or IHMSB)
Prelimi. R & D  ---------------------|-----------|------>     |
Integral exp.facility ---------------|-----------===|====>-----|-------------->
                                   design    const. low power ope.-->high power
Prototype facility AMSBE    ---------|-----------|---->-------|---=========>-
                                     | design study   design      const.
F. Establishment of Thorium Molten-Salt Nuclear Energy Synergetics
System design study         ---------|-----------|------------|--------------->
```

Fig. 8 Time schedule of R & D program of Thorium Molten-Salt Nuclear Energy Synergetic [THORIMS-NES]

year. The main purpose of this pilot-plant operation is an endurance test of the integral facility.

3. Significant Feature in R & D of MSR

In the development of small MSR, the following aspects depending on the simple and rational nature of MSR technology in principle should be recognized as summarized here:

[A] No needs of irradiation test of fuel salts, due to no radiation damage,

[B] Simple chemistry—high theoretical predictability of physico-chemical behaviors of fuel-salt,

[C] Simplicity in design-principle and configuration: (i) no core-graphite exchange —no opening of the reactor vessel; (ii) fuel self-sustainable (no additive supply of fissile materials), reactor self-controllable and load-following characters—nearly no control-rod; and (iii) no need of sophisticated continuous chemical process facility.

[D] Practical approach to commercialization from the small and economical reactors in the first step, as a most rational procedure for maturing the new MSR technology, due to (i) low capital cost in smaller ones, (ii) straight application of excellent results of the experimental reactor "**MSRE**", which was successfully operated throughout 4 years in ORNL, and (iii) few additional R & D items,

[E] Wide applicability of **Liquid-Na Reactor-Components** development results, with the advantages of MSR technology on the chemical inertness, low thermal shock, and no oxidized vapor condensation.

The MSRs including above mentioned small MSR might keep an excellent performance as effective partners of Fissile-producing MSBs in order to establish the practical **THORIMS-NES** all over the world including the developing countries and isolated areas.

4. R & D for AMSB

In parallel, MSBs, especially **AMSB** will be developed by the rational procedure increasing the proton current in step-wise during 20 years, aiming the wide application from about 2020.

The first experimental research works on the spallation products and heat generation have to be recommended including the proton irradiation analysis of solidified salt targets in big size such as 50~80 cm in diameter and 150~250 cm in length. After these examinations our preliminary design features could be very much improved due to their big flexibility. As the technological basis of AMSB is clear except the proton injection-port, it will be able to develop within 20 years by the stepwise improvement of test facilities (*cf.* Chap. X). The economy of AMSB depends on the recovery of electricity.

Already 1 GeV-1 mA cw (continuous wave) class proton LINAC technology and basic molten fluoride reactor technology have been developed. The new items necessary for engineering developments of a small integral experimental facility **AMSBE** (Accelerator Molten-Salt Breeder Experiment) will be quite few, because we can start using pure molten Flibe (LiF-BeF_2) of low melting point and low vapor pressure and using a low intensity 1 GeV pulsed proton beam initially. As a next stage nuclear material contents and beam intensity can be stepwise increased in the same facility, in the rational and economical mood. The general idea of R & D schedule including MSR is shown in Fig. 8.

The R & D of the spallation breeder AMSB would be surprisingly low in cost ($4~8 B) and short in time (15~25 years) in comparison with the other similar breeding system developmental projects, promising the wide global application in 2020~2070[22][23][18][19].

XI. CONCLUSION

We have tackled one of the most promising approaches for establishing the simple and practical **Thorium resource utilization program**, which might be effective to solve the principal energy problems, concerning resource, safety, nuclear proliferation and terrorism, greenhouse effect, power size and fuel cycle economy, for the next century.

The first step will be the development of Small Molten-Salt Reactors (MSR). Its basic technology has been established by ORNL already, and, therefore, its R & D cost would be surprisingly low. The next is the development of Molten-Salt Fissile Breeders (MSB). Our proposed MSR is an effective partner of MSB for establishing the simplest and economical Thorium molten-salt breeding fuel cycle system named **THORIMS-NES** all over the world, including the developing countries and isolated areas, for solving poverty, *i.e.* North-South problem. This would be one of the most practical replies to the Lilienthal's appeal of "A NEW START" in Nuclear Energy by encouraging "a revival of positive and affirmative fighting spirit" of scientist and engineers[32].

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[APPENDIX]

Nuclear Material Balance in Next Century

If the fission energy industry is to be soundly established in the next century, there is need to establish and prepare the logistics of growth such as shown in Fig. 1. The following discussion is based on the inclination of this "fission" line. The complete study will be done on this line accounting for the following resonable simple scenario:

(i) The total mean electric power equivalent produced by fission is 300 GW(e) in 2000 and afterwards follows the line of Fig. 1 (*cf.* Table 1),

(ii) An average annual energy growth is 2.36 % same as the value in the past 100 yr,

(iii) All reactors until the year 2019 are thermal LWR (Light Water Reactors) of 1 GWe and 30 yr life, and they produce 0.25 t Pu annually,

(iv) All fission breeders are 1 GW(e) LMFBR (30 yr life), which need 5 t Pu initial inventory, breeding an excess of 0.22 t Pu annually (corresponding to a doubling time (T_d) of about 20 yr), and

(v) All renewed and added reactors are FBR after 2020.

The total demand of natural uranium for LWR is also examined assuming 200 and 160

Table A1 Fraction "F" of fission in Fig. 1, total mean electric power equivalent produced by fission, and account of supply and consumption of Pu in next century in simplified scenario

time, year*	2000	2019	2036	2049	2059	2066
global energy growth	1.0	1.56	2.31	3.14	3.96	4.66
F of fission power	5%	10%	20%	30%	40%	50%
total electric power eq. [GW(e)]	300	936	2,770	5,650	9,500	13,980
total number of LWR#(effective)	300 ---->	936 ------>	570 ------->	0		
total number of FBR#(effective)		0 --->	2,200--->	5,650--->	9,500--->	13,980
Nat.U for LWR[ton x 10^4]						
(no recycle)	80	235	260	74		
[total]		[315]	[575]	[649]		
(recycle)	64	188	204	59		
[total]		[252]	[456]	[515]		
Pu from LWR [ton]	1,000	2,935	3,250	1,850		
[total]		[3,935]	[7,185]			
Pu inventory in FBR# [ton]			-11,000	-17,250	-19,250	-22,400
Pu from FBR [ton]			4,050	11,230	16,670	18,080
total account of Pu [ton]	1,000	3,935	235	-3,935	-6,515	-10,835

(*) calculations were performed summing up to respective year.
(#) assumed as 1 GW(e) systems of load factor 100%.

t/yr per reactor in no-U-recycling and U-recycling, respectively.

Several feature of the above fairly optimistic scenario are shown in **Table 1A**. As can be easily seen, in the period around the 2036, or a little later, this scenario results in a severe shortage of fissile materials. This results from the assumed doubling time, T_d, for the LMFBR (assumed as 20 yr) which is much lower than the requirement resulting from Fig. 1 and an annual growth, which corresponds to a T_d of 12 or 15 yr in the time range near 2020 or 2030, respectively[2].

This need can not simply be covered by an reduction of T_d of the LMFBR or changes of the other assumptions of the scenario due to the several reasons. At least, reprocessing of spent fuels of LWR would not be performed for all reactor fuels and there will be a fairly big time lag. This applies also to the LMFBR. The depression in the imaginal load factor of 100% will severely bring to the worse direction in material balance. The other reasons will be explained in Sec. II-**5**.