Chapter X. MSR-FUJI General Information, Technical Features, and Operating Characteristics

X.1. Introduction

The fuel self-sustaining small Molten-Salt Reactor (**MSR**): FUJI-series MSR concept evolved from the Molten-Salt Breeder Reactor (**MSBR**)[X-1,-2] based on "single-fluid molten fluoride fuel", which was developed in the Molten-Salt Reactor Program(**MSRP**), Oak Ridge National Laboratory (ORNL), USA among 1950-1976. The basic MSR technology was prepared by this program, especially by the operation of the Molten-Salt Reactor Experiment (**MSRE**) in 1965-69, and the conceptual design work of **MSBR**[X-2]**.**

MSBR was a Th-U cycle thermal breeder applying a continuous chemical processing (**ccp**) in situ and periodical core-graphite replacement (**cgr**) to improve breeding performance. Therefore, the **MSBR** development is not easy[X-3], and such complex facility is not suitable for global application.

The fission process is not effective for fissile breeding, and power reactors should be simpler, size-flexible and fissile-fuel self-sustaining. Such idealistic performance of power stations was almost realized by **FUJI** concept[X-4,-5] simplifying without the above **ccp** and **cgr**.

The fissile breeding should separately be performed not using fission process but proton-spallation or DT-fusion processes. During the 1980's the technical feasibility of Accelerator Molten-Salt Breeder (**AMSB**)[X-6,-7]was established based on a "single-fluid target/blanket concept" using the same molten-salts as **FUJI**, coupling with proton beam of about 1 GeV. Later on, DT-fusion might be an alternative neutron source applying inertial confinement technology or other technologies.[X-8]

THORIMS-NES [*<u>Thori</u>um <u>M</u>olten-<u>S</u>alt <u>N</u>uclear <u>E</u>nergy <u>S</u>ynergetic System*][X-9,-10,-11], which is composed of (A) simple fission Molten-Salt power stations(FUJI series), and (B) fissile-producing **AMSB** will be a most promising system for global utilization of Thorium Fission Energy realizing the molten-salt breeding fuel cycle. A rational practical international developmental program has been proposed based on Japan-USA-Russian trilateral cooperation. During 1980s we had several cooperations with not only ORNL (Drs. A.Weinberg, H.MacPherson, M.W.Rosenthal, J.Angel, U.Gat), Ebasco (Drs. L.Reicle, D.R.deBoisblanc) but also EdF (Drs.C.Bienvenu, A.Lecocq, M.Israel), Euratom (Dr.J.J.Geist), CERCA(Dr.R.Romano), Paris IV Univ.(Prof.M.Chemla), MacMaster Univ.(Prof.A.Harms), Kurchatov Inst.(Dr.V.M. Novikov), Inst.Theo.Exp.Physics(Dr.I.V.Chuvillo), PSI(Mr.F.Atchison), Tennessee Univ.(Prof.L.Dodds) etc..

The pilot-plant: miniFUJI[X-11](about 7MWe) is suggested to be constructed at Russian Federal Institute of Technical Physics (ITP), Snezhinsk, Russia[X-12]. A prototype FUJI-233U (about 100 -300MWe) would be the next logical step, which will become critical 12-15 years later. Improving the basic MSR technology among these projects, AMSB will be implemented 20-25 years later. In initial stage, FUJI will be operated contributing for excess Pu incineration (FUJI-Pu), or partially using 235U in denatured mode (FUJI-235U).

International Thorium Molten-Salt Institute (ITHMSI, Presi. Dr.K.Fukukawa) is arranging the work of the proposed project. In USA, Energy Frontiers International (Presi. Dr.J.Pleasant) and Vallecitos Research Associates(Presi. Dr.R.Moir) are cooperating closely with ITHMSI. In Russia, ITP(Direc. Academician E.Avrorin, Deputy Sci.Direc. Prof.V.Simonenco) is coworking with the group of Kurchatov Institute, Institute of Chemistry etc. The important cooperating and supporting persons in Japan are as followings: Dr.K.Kato, High Energy Accelerator Research Organization; R.Yoshioka, Toshiba Co.; Prof. Y.Shimazu, Hokkaido Univ.; Prof. K.Mitachi, Toyohashi Tech.Sci. Univ.; Dr. Dr.A.Furuhashi; Dr. H.Numata, Tokyo Institute of Tech.; Dr. K. Arakawa, Osaka Univ.; Toyo Tanso Co.(Presi.T.Konndo) etc.. In Turkey, Prof. L.B. Erbay, Osmangazi Univ., Eskisehir is studying on a conceptual design for power generation by a combined system of MSR with Free Piston Stirling Engine: FUJI-str (cf.A-X.4). Drs R.Moir and E.Teller, Lawrence Livermore National Laboratory (LLNL), are proposing a Underground version: FUJI-ug (cf. A-X.3.). In Brazil, Dr.C. W.Urban and Minas Gerais Federal Univ. are cooperating with us.

Concerning the R&D of AMSB [X-6], in initial K.Furukawa started with colleagues at Japan Atomic Energy research Institute (JAERI) and followed by the work with Institute for Theoretical and Experimental Physics(ITEP), USSR. And recently more coworking with Joint Institute for Power and Nuclear Research(JIPNR)–Sosny, Belarus (Drs. S. Chigrinov, A. Kievitskaya and K.Rutkovskaya), who are performing the neutronic calculations of AMSB [X-13].

X.2. NPP Application

The standard concept: FUJI-233U is substantially designed for Th-^{233}U fuel cycle. However, we need not wait the fissile supply from AMSB. In the initial 20~30years, we can use ^{235}U or ^{239}Pu with Th to co-exist with present U-Pu plants serving to burn the excess Pu[X-14].The electric generation system is illustrated in **Figures X-1 and X-4** . **Figure X-1** shows the bird-eye view of FUJI. **Figure X-2** shows the cutaway of reactor vessel, internal graphite moderator and control rod. The pilot-plant, named miniFUJI, is also shown to compare the core size. **Figure X-3** shows a vertical cross section of the primary fuel salt system. The general configuration of FUJI is shown in **Figure X-4**.

We have carried out several analyses for nuclear characteristics of FUJI series' cores, which have different kinds of fissile (^{233}U, ^{235}U and ^{239}Pu) and output powers. **Figures X-1 to X-4** give the common images or concepts for these reactors. The core is constituted by directly immersed hexagonal graphite rods only, which have a center hall and thin ditch on each flat side for fuel salt pass. The volume fraction of fuel salt is different in each radial zone of core, which is three zones in FUJI-233U. The standard fuel salt is a LiF-BeF_2-ThF_4-UF_4 and the melting point is about 730K. The fuel salt flows upward through the core and is heated up to about 970K. A centrifugal pump transfers the outlet fuel salt to a heat exchanger, where the heat is transferred to a secondary coolant salt of $NaBF_4$-NaF, which transports the heat to a super critical steam generation system and leads to an overall thermal efficiency of more than 44% [X-5].

The primary fuel-salt system is located in the gas-tight high temperature containment kept about 770K. In FUJI-233U, two sets of primary and secondary loop systems are planned. A drain tank located below the reactor room has a natural convection cooling system. For the heat rejection of the fission product decay heat, a reliable system of such as the heat pipe is equipped in the drain tank. The emergency dump tank is also located in the borated water pool for a buck up of drain tank. A large catch-pan is set up on the bottom of high temperature containment and in the event of a fuel salt spill, the salt is guided to the emergency tank and cooled to its freeze point by the outside water.

In the industrial countries, a steam condition of 25Mpa 870K (at turbine inlet) class ultra super-critical (USC) turbine system has been proven on the coal/gas-fired power plant technology and now the USC system of thermal efficiency more than 45% has been developed.

The standard outlet-temperature of MSR fuel-salt is 970K. But structural Ni alloy, Hastelloy N is applicable up to 1170K or more, and economical industrial-heat supply up to 930K or more than 1030K in future will not be difficult, including the applications for hydrogen production, cogeneration, desalination, and district heating (cf.**A-X.1**). The excellent feature of the molten fluoride salt as a heat transfer medium is as follows: a) low vapor pressure, b) high heat capacity, c) adequate viscosity and d) no chemical reaction with air, H_2O or other useful mediums. Therefore the compact and economical heat utility systems with FUJI can be designed.

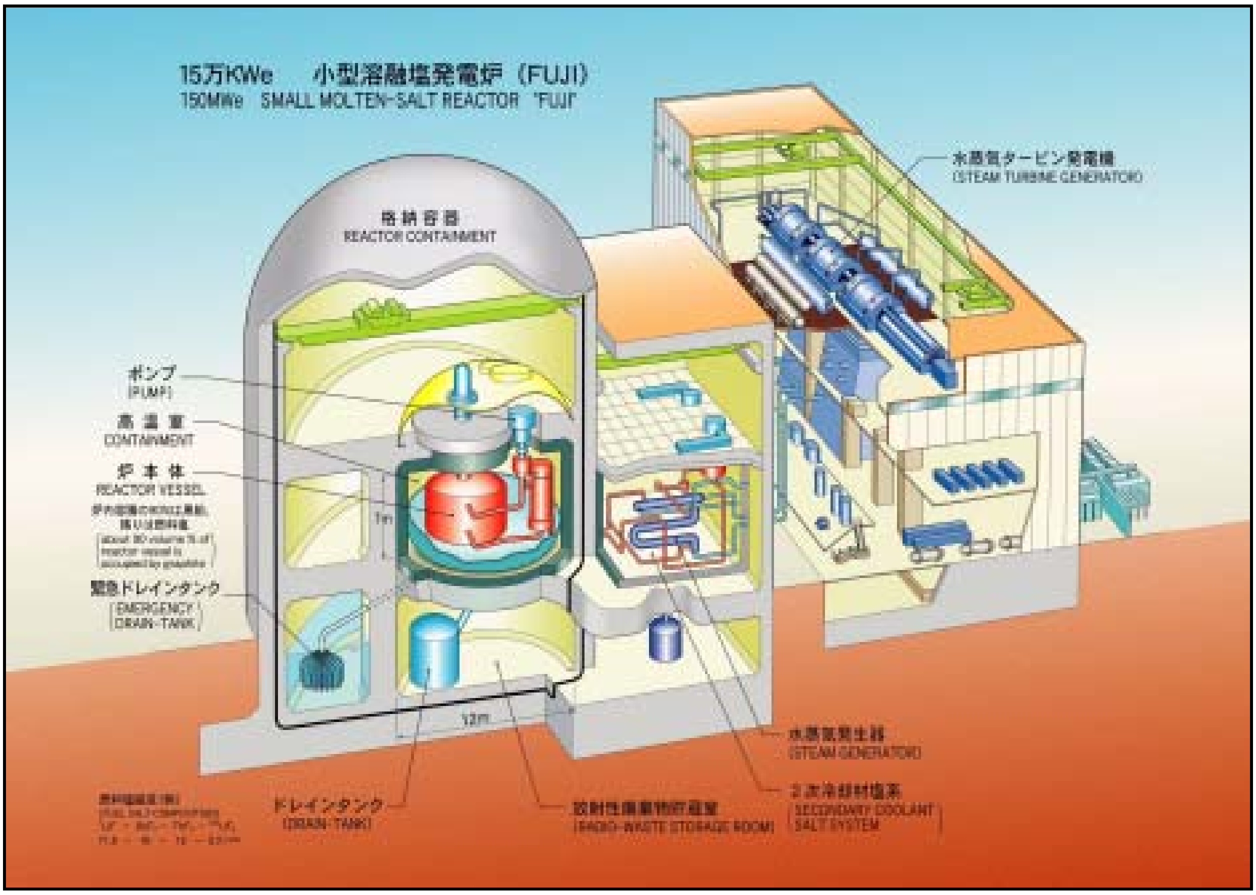

Figure. X-1.　Bird-eye View of FUJI

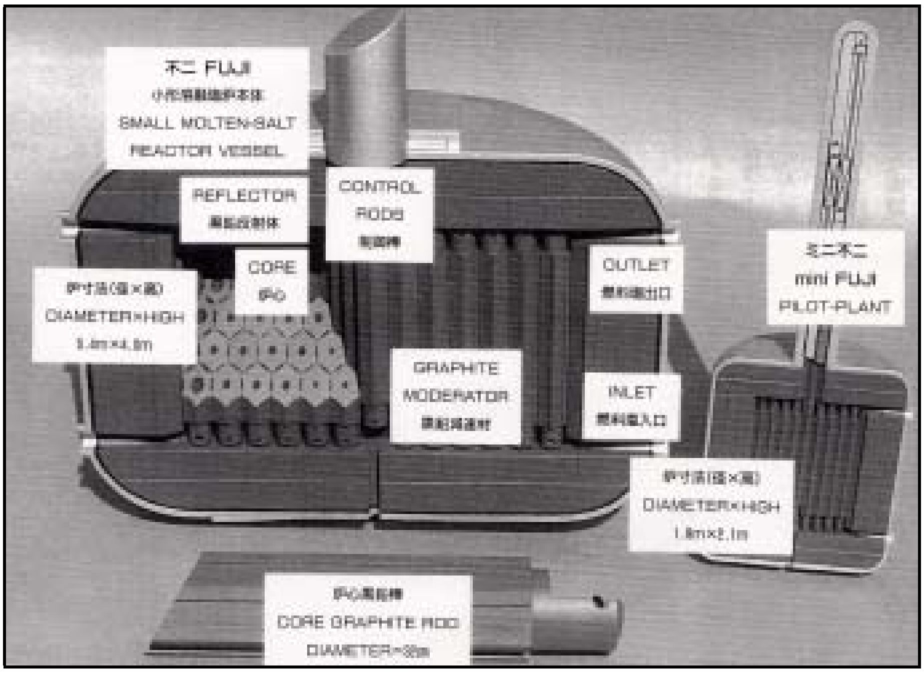

Figure. X-2.　Cut Model of FUJI Core (left), and mini-FUJI (right)

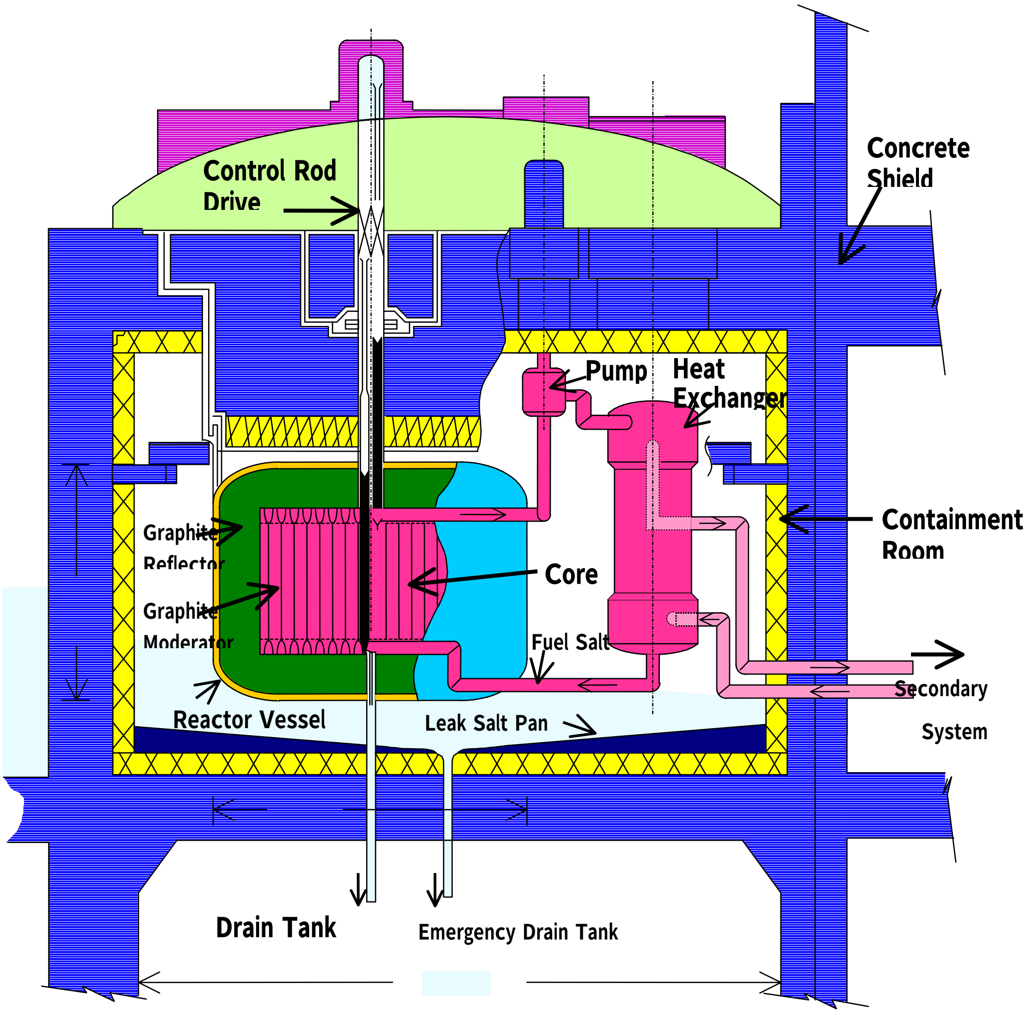

Figure. X-3. Vertical View of FUJI Primary Loop

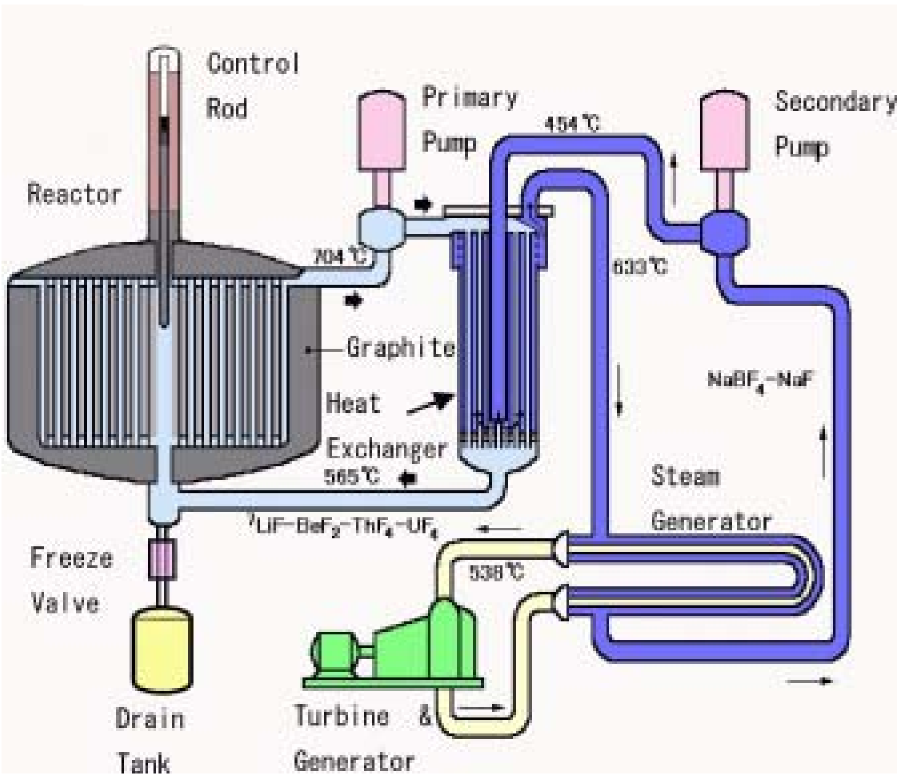

Figure. X-4. General Configuration of FUJI

X.3. Special features of the Nuclear Installation

BASIC CONCEPT

As described in X.1., our nuclear installation depends on the three principles [X-9,-10,-15], which are summarized as: (a) Thorium utilization [X-16], (b) Application of molten-salt fuel technology, and (c) Separation of fissile-producing breeder and fission power station. Therefore in summarizing the special features of FUJI-233U, it is inevitable to note briefly the special features of THORIMS-NES too.

[A] FUJI-233U [X-4,-5,-17,-18]

・**Application of Molten-fluoride Fuel Technology---** The molten salt 7LiF-BeF2 (Flibe) is a significant low thermal neutron cross-section medium and the best solvent of fissile and fertile materials. This liquid is multi-functional, useful not only as medium for nuclear reactions, but also as medium for heat transfer and chemical processing. Therefore, we can result a simple configuration to establish the complex and excellent performance in MSR including high safety and economy. The technological basis has been built up by ORNL since the sixties.

・**Simple reactor operation and maintenance, and few radio-wastes---** Fuel composition is slow in change and will be done make-up in long interval without any chemical processing. Maintenance work is few resulting a few amount of low-level radio-waste. Very small production of trans-U elements (TRU) will contribute to minimize a high-level radio-waste.

・**Flexibility on Application---** FUJI has the following flexibilities, which are required to deploy the fission reactors in the world :

(a) MSR can operate using any kind of fissile materials (^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu etc.) without any serious regards to their mutual compositions.

(b) Regarding fuel compositions, not only fissile but also several chemical components (including fission products (f.p.) and chemical impurities) in substantial amounts may flexibly be accommodated in fuel content, introducing no serious nuclear and physico-chemical problems.

(c) Initially, a small size MSR should be developed for easier R&D due to their small power size and cooperative (-with excellent load following ability-) coexistence with larger U-Pu-cycle power stations mainly operated for base load.

・**Size-Flexible and Near Breeder---** FUJI-233U is size-flexible and fuel-self- sustainable without continuous chemical processing, and is suitable to achieve economical small power stations owing to the simple structure of reactor-vessel without any big flange and no need of core-graphite replacement. Therefore, it will widely be applicable in any area of the world allowing large-scale nuclear energy utilization.

・**Effective use and incineration of Pu and TRU---** Large amount of Pu and TRU in dismantled weapons and in spent solid-fuels might be incinerated directly using as an initial start up fuel of FUJI-Pu. The conversion of spent-fuel to molten fluoride salt can economically proceed using an initial fluorination step of **FREGATE project** without any solid-fuel re-fabrication (cf. X.5) [X-20].

・**Excellent Safety---** MSR has many advantages on safety issue. The primary system of MSR has low pressure, no chemical activity and no radiation damage of fuel-salt. Inside of reactor vessel only has fuel-salt and graphite, which has high radiation-resistance, high thermal conductivity and high melting point of 4000K. All unstable radioactive materials such as Xe, Kr, T are always removed from the system and kept in the tightly-sealed decay containers. FUJI has a prompt negative temperature coefficient, which mitigates the nuclear accidents. The reactor can only reach just critical, when fuel salt is coupling with graphite moderator. Therefore, leaked fuel-salt will not induce any re-criticality accident. Essentially there is **no**

chance of “severe accidents”. This system will be substantially safe even under military attack or internal sabotage (cf. **X.6.3. to 6.5.**).

・**Substantial Economic Advantages---** FUJI-Series MSRs have following substantial economic advantages :

a) Lower capital cost of facilities based on the simple structure and safety features

b) Significantly lower electric generation cost based on the simpler fuel management, relating on lower fuel cycle, operation and maintenance costs.

c) Lower R&D cost : Technological bases have already been established by ORNL. There are substantially few R&D items only. Details are described in X.6.1.

d) Low fissile inventory : For the commission of many power stations rapidly accepting social requirement, low fuel inventory of FUJI is very useful and promising a rational breeding system.

[B] THORIMS-NES [X-9,-10,-11]

・**Requirement on New Nuclear Energy**-- We should not continue to use fossil fuel as a major energy source, even though we have to use more clean natural gas in the first half of this century. We should improve the issues of safety, radio-waste management, proliferation resistance and economy of the nuclear energy for implementaion of new huge-size industry, which growth rate should be ca.10 years in doubling time, and its peak output ca. 10 TWe (30 times bigger than the present) achieved by 2065 (cf. **Figure X-5**). Therefore, we need a new nuclear fission energy system as an interim between fossil and solar technologies for global survival in this century.

・**Opening of the Thorium ERA** --- After starting the operation of AMSBs, we will gradually achieve the Pu-free nuclear energy system.

・**Breeding System of Short Doubling Time**--- The breeding fission power reactor systems have too long doubling time (DT). Even the MSBR developed in ORNL expecting the DT of about 20 years, cannot satisfy this demand of less than 10years.　In the AMSB, 1GeV, 300mA proton accelerator produces about 400 Kg of ^{233}U even if there is no any fissile but fertile ^{232}Th in the initial target fuel salt. The initial ^{233}U inventory of FUJI-233U (200MWe) is about 800 Kg, so the AMSB can support the commission of one FUJI-233U every two years. Also, high gain type AMSB with Pu can produce more ^{233}U possibly resulting an effective DT of about 2~3 years and generate large thermal output power to make AMSB a self-sustained system in enough, which will be able to start up the sufficient number of SM-MSR to meet the steep growth of energy demand replacing the fossil energy.

・**Based on the Proven Technology**--- Most of the basic technologies in THORIMS-NES have been proved except the development of high current proton current accelerator in AMSB.

・ **Positive Heritage to The Next Century---** there are no need of large scale facilities such as the solid-fuel re-cycling, spent fuel interim storage etc. Instead of them, the AMSBs with batch chemical process facilities should be considered for the total cost estimation of the system. However, **economical nuclear transmutation** inside of the fuel-cycle [X-15,-24] will be performed by using the large amount of low-cost excess-neutrons (excess fissiles) including high energy neutrons from AMSB in the recession age (after about 2070) of Th-ERA as shown in **Figure X-5** , and AMSBs will be remained and perform another rolls in the field of nuclear incineration (transmutation), neutron science such as material development and of the medical use of proton irradiation, as the same purpose of SNS in USA or J-Park in JAPAN (those are now under construction). Therefore, the AMSBs will be the positive heritage for the next century.

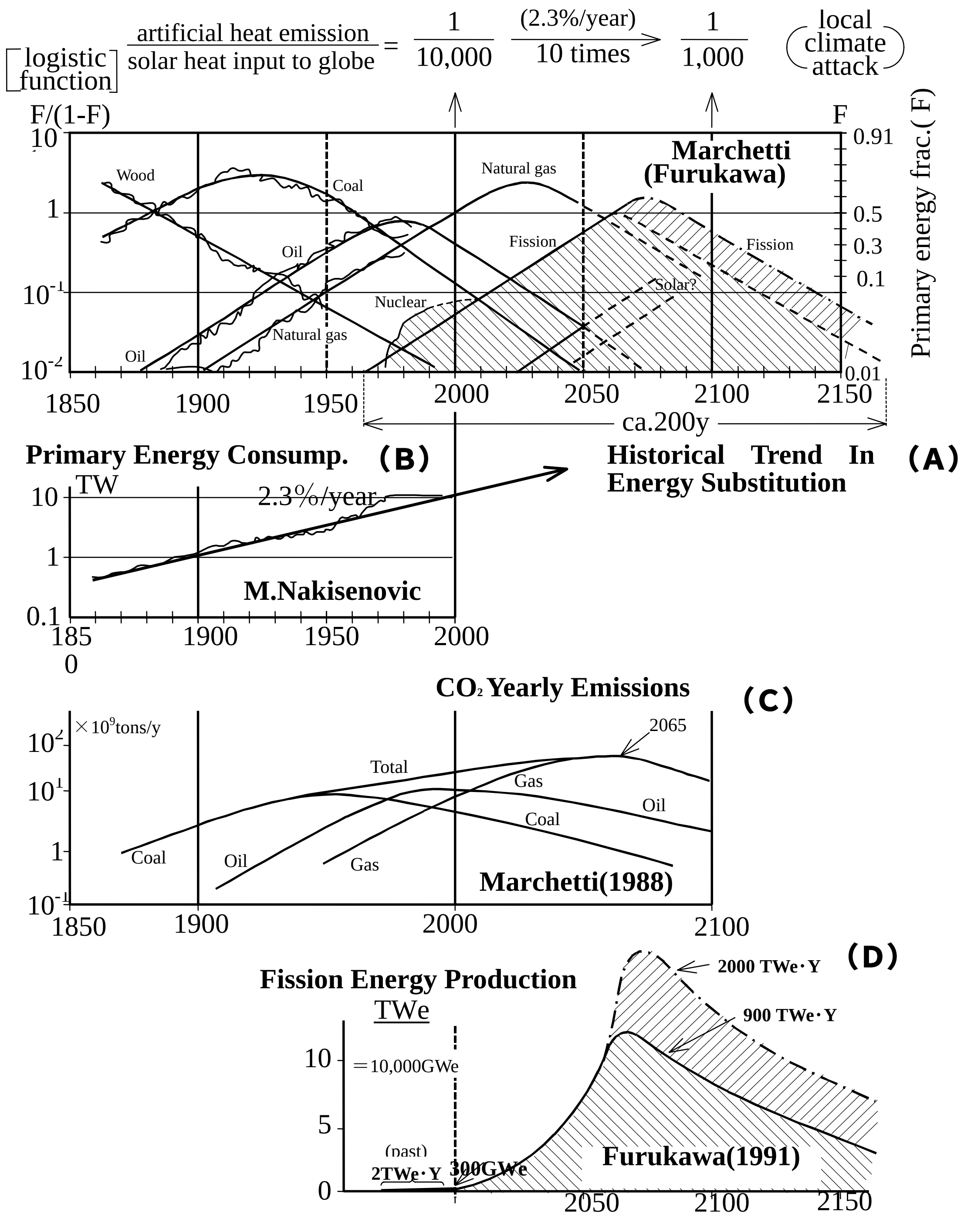

Figure. X-5 Global Future Energy Prediction.

[(A) is a further extension of Marchetti's estimate of historical trend in energy substitution; (B) yearly growth-rate of world primary energy consumption.
The predictions of (C) CO_2 yearly emission from fossil fuels, and (D) nuclear fission-energy production are estimated basing on the assumptions of (A) and (B).]

X.4. Summary table of major design and operating characteristics

The many versions of FUJI concept have been examined in the past. In here, the newest one is shown.

FUJI-233Um

Characteristics	Value
Rated Thermal Power	450 MWt
Rated Electric Power	200 MWe
Fuel	Molten Fluoride Salt: $LiF\text{-}BeF_2\text{-}ThF_4\text{-}UF_4$ Initial Composition: 71.75-16-12-0.25 mol%
Moderator	Graphite
Reactor Internal Structural Materials	Moderator & Reflector Brocks made of Graphite Design Lifetime: 30 years
Core	Core-I, Radius: 2.2 m, Graphite Fraction: 64 vol% Core-II, Outer Radius: 2.8 m, Graphite Fraction: 71 vol% Core-III, Outer Radius: 3.0 m, Graphite Fraction: 76 vol% Height: 2.1 m Power density in the core: 7.3 kW/*l*
Reactor Vessel	Material: Modified Hastelloy-N Comp.: Ni(base), Mo(11-13), Cr(6-8), Nb(1-2), Si(0-1) w% Inner Diameter: 6.84 m Height: 2.94 m Wall Thickness: 5.0 cm Design Lifetime: 30 years
Power Cycle Type	Steam Rankin Cycle Steam Conditions at Turbine Inlet: p=24 MPa, T=810 K
Number of Fuel-salt Cooling Circuits	2
Neutronics Characteristics	Fuel Conversion ratio (Initial): 0.97 Reactivity: Temperature Coeff. (Initial): -3×10^{-5}dk /K Void Coeff. (Initial): 0.07 %dk /%Void Maximum Axial (F_z) Peaking Factor in the core and blanket: 1.3 Maximum Horizontal (F_{xy}) Peaking Factor in the core : 1.2 Power flattening by radial 3 zones structure as shown in the column "Core".
Type of Reactivity Control, Reactor Protection	Graphite control rod for normal operation: (2 rods) Total control rod worth: 0.12%dk Emergency shut-down rod (B_4C particles in clad): (4 rods) Total control rod worth: 3.6%dk at one-rod stuck condition Alternate Shutdown by draining fuel-salt in the core.
Thermral-Hydraulic Characteristics	Fuel Salt at normal operation Volume in Reactor: 21.1 m^3 Total Volume: 26.4 m^3 Flow rate: 0.711 m^3/s Temperature at Inlet: 840 K Temperature at Outlet: 980 K Pressure: 0.5 MPa Graphite Moderator at normal operation Mass: 166 ton Temperature (max.): 1220 K

<table>
<tr><td rowspan="1">Fuel Cycle</td><td colspan="2">Operation Intervals of Fuel-salt
Fissile Feeding: 30 Effective Full Power Days
Adjustment of Fuel-salt Compositions: 150 EFPD
Chemical Processing of Fuel-salt: 2000 EFPD*
Fissile Inventories
Initial ^{233}U Inventory: 800 kg
Feeded ^{233}U in 30 years: 629 kg
(^{233}U+^{235}U) Inventory after 30 years: 1107 kg</td></tr>
<tr><td>Hydrogen Production</td><td colspan="2">120 tons H_2/day at 450MWt</td></tr>
<tr><td>Desalination</td><td colspan="2">28,000 m^3/day from MED at 450MWt w/electricity</td></tr>
<tr><td rowspan="6">Economics
(Power Generation Cost
(cent/kWh) at year 2000)</td><td>Capital cost</td><td>2.01</td></tr>
<tr><td>O&M cost</td><td>0.58</td></tr>
<tr><td>Fuel cycle cost</td><td>0.12</td></tr>
<tr><td>Waste disposal cost</td><td>0.10</td></tr>
<tr><td>Decommissioning cost</td><td>0.04</td></tr>
<tr><td>Total cost</td><td>2.85(cent/kWh)</td></tr>
</table>

* As one version, in 5.5 years interval a flesh fuel salt is charged instead of the old, which is processed in the Regional center.

Typical fast/thermal neutron flux distribution of FUJI-233Um is shown in Figure X-6.

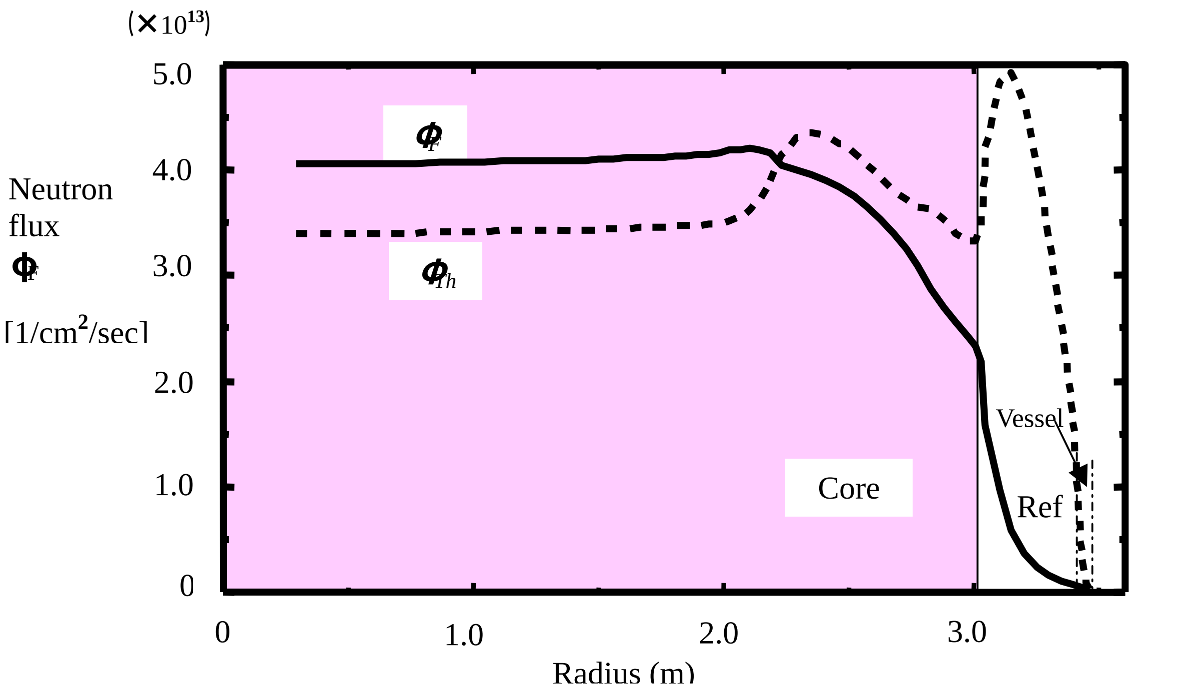

Figure. X-6 Fast neutron flux distribution of FUJI-233Um

X.5. Outline of Fuel Cycle Options [X-10,-16,-20]

As shown in **Figure X-7**, the spent fuel-salt after finishing the FUJI reactor life will be sent back to the Regional Center, which is distributed in the world and safeguarded in enough. The salt is processed in batch mode for removing ^{233}U by fluorination and some fission products (f.p.), which contents will be decided from the integral optimization on material compatibility, neutron economy, cost economy, etc.

The decontaminated diluent salt is done some make-up for keeping a suitable ThF4 content and charged to the storage tank of AMSB for keeping the constant ^{233}U content of target/blanket salt such as in 0.5 mol%. The salt taken out from its storage tank will be enriched by adding the above removed ^{233}U, and be sent as the fuel-salt of FUJI power station site.

The significant aspects of this fuel cycle are the followings:

・ **Single-liquid breeding fuel cycle without any solid species---** The total system is simply integrated by one phase of molten fluorides based on Flibe (^{7}LiF-BeF$_2$ binary salt) as solvent, even the composition is done some make-up in each component facilities.

・ **Working medium suitable for chemical processing---** This liquid is a stable ionic liquid with the following characteristics suitable for nuclear systems: 1) no radiation damage, 2) low vapor pressure, 3) chemical inert, 4) high heat capacity, 5) high solubility of several ions widely, and 6) low possibility in re-criticality owing to no hydrogen species.

・ **Physico-chemically well established medium---** The chemical behavior of this liquid can effectively be predicted by physico(electro)-chemical theorem due to the classical character depending on the electro-static force between ions in salt., and the basic theoretical chemical data is well established [X-21] by the ORNL-MSR program. It is practically very useful in R&D works, as it was already verified by ORNL using so small time, expense and personnel.

Spallation products (s.p.) behavior similar to f.p.--- The existing past f.p. technology can apply to s.p. treatment not inducing any serious problems, because the concentration of s.p. in AMSB salt is one order of magnitude less than f.p. in FUJI due to the bigger neutron generation by spallation than fission.

Several applicable process technologies--- Modified continuous chemical process for MSBR can apply, bur Batch Process is suitably applicable in the Regional Centers (**Figure X-7**). including fluorination removal of U, semi-noble and noble metal extraction by liquid Bi, some electrolysis, etc. Minor actinides and some f.p. can be circulated for incineration in salt.

Partial processing in FUJI and AMSB--- The estimated f.p. production and final concentrations in salt after full 15 years operation in FUJI is shown in **Table X-1**.

a) Group I elements are removed in operation. Kr, Xe are shifted to He cover gas phase due to no solubility. T is transferred to coolant salt through heat-exchanger metal wall, and caught as water species THO in cover gas, which was examined in ORNL expecting a leakage less than 1 curie /day of T.
b) Group II elements are stably dissolved in low content in salt, which influence will be negligible. This should be confirmed by the reactor operation in life.
c) Group III elements will be floated or segregated in the stagnant salt zone, although the total amount is not big. The behavior should be reconfirmed by the pilot-plant operation.

The mutual cooperation with the solid fuel reactor systems will be achieved as shown in the left side of **Figure X-7**. The spent solid-fuels will be treated by **FREGATE process** [X-19,-20], for example, which was developed by the cooperation of France-Soviet-Union-Czech applying F$_2$ gas flame reactor technology for fluorination, by which Pu-containing fluoride salts will be obtained without coming back to solid, and can directly use them for the fuel salt of FUJI economically. This is convenient for Pu incineration and for initial fissile fuel of FUJI not requiring a rapid investment for AMSB development, which can be performed more rationally.

The integrated system such as **Figure X-7** will be realized after more than 30 years gradually.

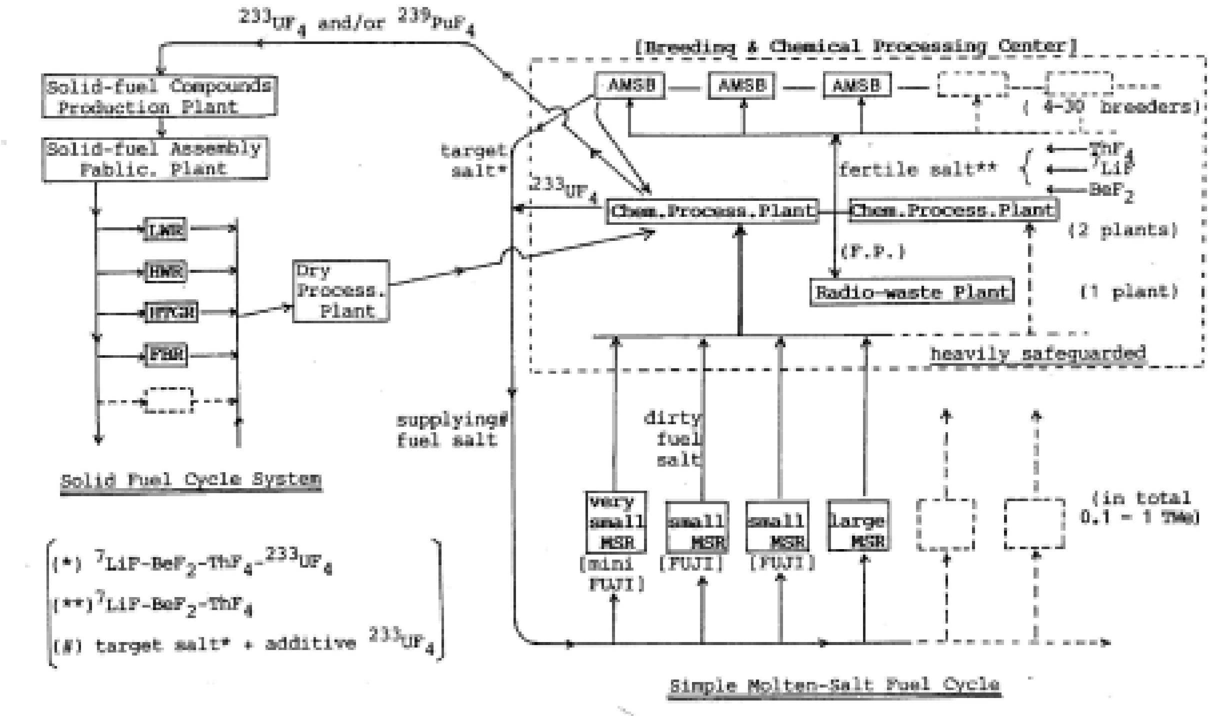

Figure. X-7 Thorium Molten-Salt Breeding Fuel-Cycle System
Regional Center accommodating AMSB, Chemical-Process & Radio-Waste Facility, and Molten-Salt Power Reactors are coupled by MS Fuel.
The connection with U-Pu Cycle System is shown in the transient stage.

Table X-1 Predicted Amount of Fission Products
Accumulated at the Life-end of “FUJI-II”
(in a/0(atomic %), m/0(mole 5) and kg)

	production from ^{233}U	amount dissolved in fuel salt	amount separated to gas phase
Group I			
Xe	27.6 a/o		312 Kg
Kr	6.5 a/o		45.9Kg
T			ca.0.1Kg
Group II			
I	2.6 a/o	27.6 Kg [0.032m/o]	
Br	0.42a/o	2.8 Kg [0.005m/o]	
Te	4.1 a/o	43.5 Kg [0.05 m/o]	
Cs	17.8 a/o	56 Kg [0.06 m/o]	144 Kg
Rb	7.2 a/o	0.5 Kg [0.001m/o]	51 Kg
Sr	11.8 a/o	28.1 Kg [0.047m/o]	60.5Kg
Ba	6.3 a/o	0.3 Kg [0.005m/o]	72 Kg
Ce	14.1 a/o	166 Kg [0.17 m/o]	
Nd	16.4 a/o	199 Kg [0.2 m/o]	
Y	5.9 a/o	1.5 ~ 7.5 Kg [0.003~0.013m/o]	42~37 Kg
Zr	30.0 a/o	232 Kg [0.37 m/o]	2~10 Kg(?)
Group III			
Mo	21.6 a/o	[deposite 175.9Kg]	2~10 Kg(?)
Se	0.9 a/o	6.1 Kg [0.01 m/o]	
Sn	0.3 a/o	3.0 Kg [0.004m/o]	

X.6.1. Economics and maintainability

One estimation on MSR economics has been reported, although data are a little old [X-2,-22]. Assuming the same power output of 1,000MWe plant of conventional Light Water Reactor (LWR) and Molten Salt Reactor (MSR), cost components such as Capital cost, Fuel Cycle Cost, Operation and Maintenance cost had been examined comparing each other.
(1) **Capital cost** of MSR is almost similar to LWR. There are many pro and cons between these two reactors. MSR has 1st/2nd/3rd loop such as FBR. On the other hand, the thermal efficiency is relatively 30% higher than PWR, and the core pressure is very low, and the safety system is simplified.
(2) **Fuel Cycle Cost (FCC)** is lower than LWR. This is because MSR requires quite small thorium and quite small ^{233}U (fissile) for plant lifetime, meanwhile LWR requires much larger natural uranium and large ^{235}U (fissile). Besides that, MSR is a liquid fuel, and does not need fuel fabrication process as LWR.
(3) **Operation and Maintenance (O&M) Cost** of MSR is almost similar or less than LWR in the literature, although MSR needs remote maintenance, because molten fuel salt of high radioactivity circulates outside the reactor vessel. Meanwhile, MSR can operate longer than LWR, and then MSR can save downtime.
(4) **Plant Capacity Factor** of MSR is higher than LWR, because MSR does not need fuel shuffling as LWRs.
When considering the above total components, it has been concluded that the economy of MSR is almost similar or better than LWR in the past works.

Recently, power generation cost for MSR and PWR was re-evaluated at the LLNL [X-23], using the original evaluation by the ORNL [X-2,-22]. The five cost component, which are Capital cost, O&M cost, Fuel cost, Waste disposal cost and Decommissioning cost, are considered. Assuming the capacity factor of MSR as 90%, and 80% for PWR, the results are shown in **Table X.-2**, and it shows that MSR is about 30% cheaper than PWR as a total power generation cost. In here, only FCC value is reexamined depending on the recent FUJI-233Um design, because LLNL result was based on de-natured MSR feeding ^{235}U. Also, PWR data is re-evaluated using recent data.

The FUJI-series reactor does not implement a continuous chemical processing facility and is removed a graphite-exchanging facility. Therefore, the total cost of FUJI will be improved more than the above result.

Table X.-2. Power Generation Cost (cent/kWh)

	MSR
Capital cost	2.01
O&M cost	0.58
Fuel cycle cost	0.12
Waste disposal cost	0.10
Decommissioning cost	0.04
Total cost (cent/kWh)	2.85

As for the maintainability of MSR, the MSR has excellent 4 years operation at MSRE, which fuel burn up amount corresponds to the one third (about 10 years) burn up of FUJI in full life. Based on this experience, some part of maintenance work is discussed in the literature [X-1].

The MSR has very simplified reactor internals and safety systems. This fact will make maintenance simpler. On the other hand, fuel salt, which has high radioactivity, circulates in the high-temperature containment room. Therefore, several equipments such as primary pumps or primary heat exchanger must be inspected by remote-maintenance equipments. But drive motors/mechanism of primary pumps and control rods are located outside the high-temperature containment room, in order to make maintenance easier. In this area, recent technology can be applied.

X.6.2. Provisions for Sustainability, Waste Management and Minimum Adverse Environmental Impact

As described in the previous section, our basic philosophy is as follows: To adopt the steep energy demand growth due to the population increase in the world and also to restrict the CO_2 emission, we have proposed the system of Th-233U cycle, named THORIMS-NES, with short doubling time [X-10,-15].

・ **Sufficient amount of resources ---** Th resource is non-localized, and geochemically three times more prevalent than U. Th resources have already been confirmed about 2 M tons and estimated about 4 M tons as shown in **Table X-3.** Th resources necessary for 1,000 Twe・y production globally required for the 21st century will be only about 2 M tons (about one third will be fissioned), which is comparative with about 1.5M tons U already extracted from the earth. Th will be obtained from the “heavy sand” of beach with relatively little pollution.

Table X-3. Estimated world thorium resources [X-25]

Countries	RRA	RSE	Total	%
Europe				
Greenland	54	32	86	
Norway	132	132	264	6,4
Turkey	380	500	880	21,4
Total	**566**	**724**	**1290**	**31,4**
America				
Brazil	606	700	1306	31,8
Canada	45	128	173	
United States	137	295	432	10,5
Total	**790**	**1125**	**1915**	**46,6**
Africa				
Egipt	15	280	295	7,2
Niger	?	?	29	
South Africa	18	?	115	
Total	**36**	**309**	**479**	**11,7**
Asia				
India	319		391	7,8
Total	**343**	**30**	**403**	**9,8**
World total	**1754**	**2188**	**4106**	**100**

RRA: Resources reasonable achievable

RSE : Resources supplementary estimated

Additionally, we will require 0.6 M tons of Li, 0.2 M tons of Be and 2 M tons of F for fuel-salt in the 21st century, but expecting the reduction in one order of magnitude by recycling technology. Substantially the used Ni alloy and part of the used graphite will be recycled or reused in THORIMS-NES [X-24].

Necessary water resources will be decreased by the high thermal efficiency, and land resources will also be smaller due to greater safety, system simplicity and compactness, and fewer radio-wastes including overall simpler infrastructure related to THORIMS-NES. In **Table X-4.** global energy/environment problems and achievable solutions by THORIMS-NES is shown.

・**Rational radio-waste management ---** Radio-waste management issue will be more rational due to the following reasons [X-24]:

(i) Practically **no TRU production** establishing their economical incineration--- The productions of Pu and Am+Cm in FUJI-^{233}U are 0.5kg and 0.3g in every 1 Gwe・y in average, respectively, i.e. very small amounts comparing with LWRs case producing about 230kg and 25kg, respectively.

(ii) Few chance of chemical processing, fuel preparation and maintenance works resulting a very **few low-level radio-waste production**.

(iii) Fuel-salt can accommodate fairly large amount of fission-products, which will be decayed and destroyed while circulating in the molten-salt fuel cycle.

(iv) **Economical nuclear transmutation** inside of the fuel-cycle of molten salt as a most suitable working medium might be performed using the low-cost excess-neutrons (from excess fissile materials) in the recession time period after about 2070 when nuclear energy is assumed to be declining and replaced by solar (cf. **Figure X-5.).** The **high-level radio-waste management** issue will become a “**hundreds years**” problem.

・**Fully remote handling system for maintenance of primary system ---** The maintenance of primary system will be fully performed by remote handling system. This technology will be fairly easily applied in our reactor systems due to the simplicity of configuration without heater and shielding. In an ordinary case, the operation of the remote handling system is made after the fuel salt is drained. Fuel salt does not wet the surface of Hastelloy N and graphite.

Table X-4. **Global Energy/Environment Problems and Achievable Solutions by THORIMS-NES [Thorium Molten-Salt Nuclear Energy Synergetics]**

	Technical Problems	Achievement	New Fission Technology
RESOURCE	*U* : localized, monopolized.	*Th* : non-localized, popular	^{232}**Th+n=>**^{233}**U**
ENVIRONMENT. ADAPTABILITY (FOSSIL FUEL)	Thermal pollution Acid rain Greenhouse effect	Low: high thermal effici. no NOx, SOx. no CO_2, CH_4.	
RADIO-WASTES	*Trans-U [Pu, Am, Cm]* Kr, Xe, T Low-level waste	*no production* (very few) Always isolated from reactor core. Minimized by few maintenance work.	***Th - 233U* CYCLE**
NUCLEAR -PROLIFE. & -TERRORISM	Military diversion Pu(weak prolife. resist.) Safeguard difficulty	No Pu-produc. *Pu-burnabe* ^{233}U(*high gamma of* ^{232}*U).* Easy safeguard	**MOLTEN FLUORIDE FUEL** Triple functions: nuclear reaction heat transfer chem.processing
SAFETY BASIC ENGINEERING	Chemical reactive Mechanical failure Excess nucl. reactivity *SOLID-FUEL ASSEM.* Configuration, operation, maintemance transport, reprocessing Core-melt, re-criticality	Chemical inert Low pressure, low flow. Very low, Fuel self-sustaining. *LIQUID FUEL* (fuel :no radi. damage) All simpler & fewer No severe accident	***SEPARATION* of *BREEDING* &*POWER* *GENERATION***
BREEDING FUEL-CYCLE	Completion difficulty Doubling time: too long	Simple M.-S. Fuel-cycle Short : 5 to 10yrs by *AMSB*	
SOCIAL - ADAPTABILITY of POWER- STATION ECONOMY	Siting difficulty Large power size Process-heat : not easy Safety, nucl.proliferation radio-waste issues	Easy : near to utility Smaller : size flexible Easy:industry-heat, district-heat Large improvements	

X.7. List of enabling technologies relevant to the Nuclear Installation with an innovative SMR and status of their development

During the 1960's ORNL significantly succeeded in construction/operation of the experimental reactor MSRE and the conceptual design study of 1 GWe MSBR. Depending on these results our development will be much sound and easier in implementation [X-10]. In here and the next X.8. we will mostly discuss on the FUJI project, which we are nominating **"F-plan"**(cf.**X.8.**).

The most important key engineering/technological problems are as follows:

Reactor physics [X-15,-22] **---** Since MSR is a typical thermal neutron reactor, there are no specific concerns on reactor physics. Actually, most of recent design codes will be able to handle MSR with enough accuracy, because reactivity change due to burn-up or power-change is small. In addition, the experimental reactor MSRE had excellent 4 years operation.

The criticality examination is much simpler than solid-fuel reactors not requiring the exact value by critical assembly examination. The final approach to criticality will be achieved by a slow addition of fissile salt through the storage tank, from which the salt is supplying to fuel pump bowl for recovering the salt overflowing from pump bowl.

Fuel chemistry [X-3,-17] **---** There are no serious problems except the examination of the detailed PuF_3 solubility data in the relation with other f.p. ions.

Structural alloy--- As shown in the **A-X.1.**, modified Hastelloy N suitable for MSR is chosen already even the endurance test should be performed in pilot-plant miniFUJI. And the data base on the high-temperature performance should be prepared as soon as possible after deciding the final specification. This kind of Hastelloy is similar to Stainless steels, and fabrication and weld-ability is comparatively easy as a low-brittle alloy.

Graphite--- As shown in the **A-X.1.**, the homogeneous Graphite suitable for MSR can be produced depending on the past developmental efforts. However, its irradiation test should be performed using the powerful irradiation-test reactor: MS-4, Demitrovgrad, Russia, for example. The further development improving the radiation dose limit is essential, because its improvement results straightly and significantly the smaller size of reactors, which means the lower cost of electricity.

High temperature containment technology--- This is a new technology not using in MSRE yet. However, as miniFUJI is a compact small reactor, a mock-up test will be performed easily.

Electric generation facility--- This is also a new facility, and miniFUJI should demonstrate its integrated technology including structural alloys. However, Hastelloy N is a Nickel alloy suitable for steam atmosphere, and the design will not be so difficult.

Several components and instruments--- MSR is a high-temperature molten material reactor similar to LMFBR. Therefore, the huge effort on the Na technology development in LMFBR will be usefully applied for getting safer and reliable designs without too high chemical activity and thermal conductivity in sodium. Especially it is real in the mechanical pumps and steam generator.

Chemical monitoring--- The change of chemical behavior of salts is very slow, but the further development of continuous in situ technologies will be recommended.

Remote maintenance--- The primary fuel salt system has a high radioactivity, and needs fully remote operation and maintenance. The recent significant progress of robotics will be a big advantage. It should be improved for application under high temperature and high radiation field.

X.8. Status of R&D and planned schedule [X-10,-15,-32]

In ORNL the operation of the experimental reactor MSRE (7.3MWth) had been successfully performed among 1965–1969 as a civilian project. The all of basic information is opened in public. But it was 33 years ago, and a real MSR technology should be recreated again.

Our basic program for developing THORIMS-NES is constructed the three plans:

F-plan: Fission reactors development including miniFUJI, FUJI in several versions.

D-plan: Dry-processing of spent fuel and target/blanket salts including not only molten salt but also solid fuels of ordinary reactors such as LWR, FBR, HWR etc. for getting molten fluoride fuel salt of FUJI. or target salt of AMSB.

A-plan: Accelerator Molten-salt Breeder development in several versions.

In here, mostly **F-plan** will be explained briefly, a little mentioning on D- and A-plan too.

The skeleton of our program is shown in **Figure X-8.** As shown in **A.** of this figure, after getting the project fund several test loops and component/instrument will be prepared and operated for education and training of project staffs. The decision of material specifications is important and will be started the high temperature tests including irradiation test. The purpose and aims will be known from the previous description including X.7.

As shown in **B.**, the design of miniFUJI will be finished 4 years later conservatively tracing on MSRE, and getting the help from trial mock-up work, by which remote technology is also examined for development. These facilities are useful in the real reactor operation. The construction of miniFUJI will be finished 6-7 years later. After charging salts and doing several preliminary tests miniFUJI will become critical 7 years later. No serious problem will exist in it owing to MSRE experience.

Getting several experiences for reconfirmation and modification of previous MSR data from miniFUJI operation, now detailed design work and several R&D for FUJI will be performed in the period 6-9 years later. Although the several kind of much innovative design works will proceed, some conservative design such as FUJI-233U of X.4. should be recommended as the first prototype power station, even the improved design optimization work is necessary to satisfy the flexibility on reactor operation, especially on the fuel design in the same core configuration. We are expecting its criticality 12-15 years later. This is the middle term program shown in **C.** of **Figure X-8.**

Since then several efforts for the real commercialization might be started gradually. In parallel, basing on the real technology obtained from the above projects, not only medium- or large-size FUJI but also AMSB should be developed. Several preliminary experiments will be finished using 1 GeV accelerator of several mA proton beam, and the AMSB development till 25years later might be quite promising.

Also in parallel, the following studies should be progressed:

* The development of D-plan--- At first, the treatment of spent solid fuel will be done simplifying the FREGATE project [X-19,-20].
* Study on social acceptance of THORIMS-NES concept in the world--- Depending the result of assessment several modification or several version fitting in each district should be done, including FUJI and AMSB themselves. The DT-fusion application also might be realized.
* Real designing and preparation of several Regional Centers in stepwise.

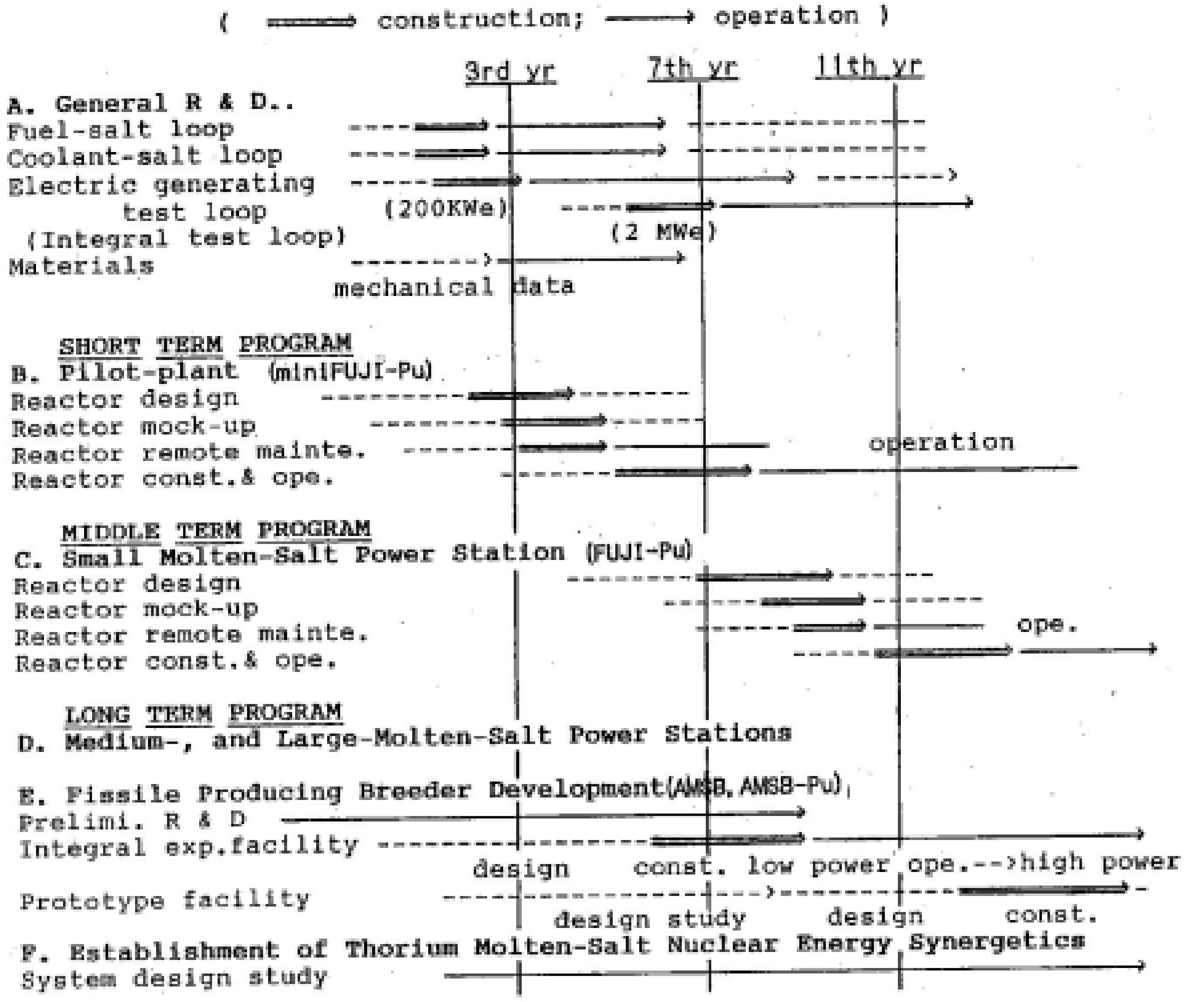

Fig. X-8 Developmental Schedule in THORIMS-NES

X.9. Justification of why a demonstration prototype or a significant amount of demonstrations will be needed, i.e., why this particular SMR could be rated as innovative

In 1960's ORNL significantly succeeded in construction/operation of the experimental reactor: MRSE. However, they could not get any approval on the next Prototype: MSTR in 1970's [X-3]. The reason was mostly in political such as "Breeder Moratorium", although US President J.Carter himself had recommended for Japan to choose MSBR than LMFBR. But the world promotion of non-proliferation issue was violent and urgent [X-30].

The basic MSR technology is most promising for real utilization of Th resouces, although the development of its continuous chemical processing such as MSBR was especially not a short-term job, which was eliminated in our strategy. Now we should do a new restart from the Experimental Reactor level, which is aiming a rapid but conservative (careful) approach: miniFUJI simulating the successful MSRE operated 34 years ago.

The miniFUJI will contribute for verifying the rationality of the integral MSR technology included an electric generation system, and for the education of MSR engineers. Afterward we can rapidly implement the industrialization of Th-MSR, in which the past result of Liquid-sodium cooled FBR development is very helpful as a same high temperature molten material reactors.

X.10. List of other similar or relevant SMRs for which the design activities are on-going

FUJI-233U is a simplified standard model, and will be able to modify in several direction effectively in future after establishing the basis of MSR technology by this.

Already many several versions are examined and reported in past including the excess Pu incineration (FUJI-Pu), or partially using 235U in denatured mode (FUJI-235U). And Drs Ralph Moir and Edward Teller, USA are proposing a Underground version: FUJI-ug (cf. **A-X.3.**). Prof.Berrin Erbay, Turkey are examining a combined system of MSR with Free Piston Stirling Engine: FUJI-str (cf.**A-X.4**). Higher outlet-temperature version will also be very useful in public for the industrial heat supply including the hydrogen generation, etc.

X.11. References

X-1. M.W.Rothental, P.N.Haubenreich, R.B.Briggs, "The Development Status of Molten-Salt Breeder Reactors", **ORNL-4812**(1972).

X-2. R.C. Robertson, ed., "Conceptual Design Study of a Single Fluid Molten Salt Breeder Reactor", **ORNL-4541**(1971).

X-3. L.E.McNeese & Staff og MSRP, "Program Plan for Development of Molten-salt Breeder Reactors",**ORNL-5018** (1974).

X-4. K.Furukawa, K.Minami,T.Oosawa,M.Ohta,N.Nakamura,K.Mitachi,Y.Kato,"Design Study of Small Molten-Salt Fission Power St. suitable for coupling with Accel.Molten-Salt Breeders", **Proc. 4th ICENES**,1986**"Emerging Nuclear Energy Systems"** World Sci. (1987) P.235-239.

X-5. K.Furukawa, K.Mitachi, Y.Kato, "Small Molten-Salt Reac. with Rational Thorium Fuel Cycle", **Nucl.Engineering & Design**, **136,** 157-165 (1992).

X-6. K.Furukawa, K.Tsukada & Y.Nakahara,"Single-Fluid-type Accelerator Molten-Salt Breeder Concept ",**J. Nucl. Sci. Tech.,18**, (1981) 79.

X-7. K. Furukawa & K. Tsukada: "Single-fluid-type Accelerator Molten-Salt Breeder (AMSB)", JAERI-M 83-050 (1983) **Japan Atomic Energy Research Inst.**

X-8. K. Furukawa,"Symbiotic Molten-Salt Systems coupled with Accelerator Molten Salt Breeder (AMSB) or Inertial-Confined Fusion Hybrid Molten-Salt Breeder (IHMSB) and their Comparison,[3rd:ICENES, Helsinki,1983] **Atomkernene. /Kerntech, 44 [1]** (1984) P.42-45.

X-9. K. Furukawa, Y, Kato, T. Ohmichi, H. Ohno:"The Combined System of Accelerator Molten-Salt Breeder(AMSB) and Molten-Salt Converter Reactor(MSCR)", [Japan-US Seminar on "Thorium Fuel Reactors" (Oct.1982,Nara)] "**Thorium Fuel Reactors",,** Atom. Ene.Soc. Japan (1985) P.271-281**.** [Russian Trans.:"**Atomnaja Texnika za Rubezom", 1983 [6]** (1983)]P.23-29

X-10. K.Furukawa, A.Lecocq, Y.Kato and K.Mitachi, "**Summary Report** : Thorium Molten-Salt Nucl. Energy Synergetics", **J.Nucl.Sci.Tech**., **27,** 1157-1178 (1990).

X-11. K.Furukawa, K.Minami, K.Mitachi, Y.Kato, "High-Safety and Economic Small Molten-salt Fission Power Stations and their Devel.Program---Th Molten-Salt Nuclear Energy Synergetics(THORIMS-NES)", "**Alternative Energy Sources VIII**", (Miami, 1987)Hemisphere Pub.,(1989), Vol.**2**, pp.2-22.

X-12. V.A.Simonenko, K.Furukawa, K.F.Grebyonkin & K.Mitachi : "Concept of the Molten-Salt Reactor Technology Implementation." **ICENES'96**: : [June, 1996, Obninsk, Russia], (1997) P.175-179, IPPE.

X-13. Y.Kato, K.Furukawa, K.Mitachi & S.E. Chigrinov: “Fuel Trajectory in Acceretator Molten-Salt Breeding Power Reactor System including Pu Burning”,"Emerging Nuclear Energy Systems ICENES'93", Ed.H.Yasuda, (1994) P.439-443, World Sci.Pub.; K.Furukawa, S.E.Chigrinov, Y.Kato & K.Mitachi: “Accelerator Molten-Salt Breeding Power Reactor useful for Pu-Burning and ^{233}U-Production” ibid. P.429-433.

X-14. K.Mitachi, K.Furukawa, M.Murayama & T.Suzuki: Nuclear Charatreristics of a Small Molten-Salt Power Reactor Fueled with Pu. "Emerging Nuclear Energy Systems ICENES'93", Ed.H.Yasuda, (1994) P.326-331, World Sci.Pub.

X-15. K.Furukawa, V.Simonenko, K.Mitachi, A.Furuhashi, R.Yoshioka, S.E.Chigrinov, A.Lecocq,, Y.Kato: : “Thorium Cycle Implementation through Pu－Incineration by Thorium Molten-Salt Nuclear Energy Synergetics”,**IAEA-TECDOC-1319** [**Thorium Fuel Utilization:Options and Trends.]** (Nov.,2002), p.123-137.

X-16. K.Furukawa, Uri Gat, J.R.Engel: “Supporting the Thorium Reactor”, **Nuclear News** / March 1994 (1994) p.21

X-17. K.Furukawa, N.Nakamura,K.Mitachi,Y.Kato & K.Komatsu: “Simplified Safe Small Molten-Salt Reactor--"FUJI"--for a Global Measure of Greenhouse Effects”,"Energy & Environmen. Progress-I",Ed.T.N.Veziroglu[9th Miami ICEE1989]]Vol.C, P.341-355(1991)Nova Sci.Pub.

X-18. OECD/IEA, /NEA & IAEA: [***final report*** of **THREE AGENCY STUDY, 1999]** “Innovative Nuclear Reactor Development, ----Opportunities for International Co-operation”, Oct., 2002.

X-19. V.Bachacek, I.Peka, V.Vesely: "Uranium branch for FBR Reprocessing by fluiride volatility methods", presented at ANS (1982?).

X-20. A.Lecocq, K.Furukawa,”Chemistry of Dry Process in the THORIMS-NES Concept”, Int. Sem. Molten-Salts in Nucl.Tech., June 19-23, 1995, Demitrovgrad, Russia.

X-21. C.F.Baes, “The Chemistry and Thermodynamics of Molten-salt Reactor Fuels”**, J.Nucl. Materials, 51**(1974), 149-162.

X-22. J.R.Engel, W.R.Grimes, H.F.Bauman, H.E.McCoy, J.F.Dearing, W.E.Rhoades, "Conceptual Design Charac. of Denatured Molten-Salt Reactor with Once-through Fueling", **ORNL/ TM-7207** (1980).

X.-23. R. W. Moir, "Cost of Electricity from Molten Salt Reactors", **Nuclear Technology, 138**, (APR.2002), P.93-95.

X-24. K.Furukawa, A.Lecocq, Y.Kato, & K.Mitachi,”Radiowaste Manag.in Global Appl.of Thorium Molten-Salt Nucl.Energy Synergetics with Accel.Breeders”,Specialist Meet." Accel.-driv.Transmut.Tech.for Radwaste and Other Appl.",(24-28 June, 1991, Saltsjobaden, Stockholm, SWEDEN)**LA-12205-C** Conf. SKN Rep.No.54; **UC-940(** 1991) P.686-697.

X-25. **IAEA-TECDOC-1155,** “Thorium based fuel options for the generation of electricity: Developments in the 1990s”(May, 2000).

X-26. Y. Shimazu, "Locked Rotor Accident Analysis in a Molten Salt Breeder Reactor" , **J.Nucl.Sci.Tech,15** (1978) 935.

X-27. ”Molten Salt Breeder Reactor", Special Research Committee Report, The Atomic Energy Society of Japan, (1981/4) (in Japanese).

X-28. Y. Shimazu, "Nuclear Safety Analysis of a Molten Salt Breeder Reactor", **J.Nucl.Sci.Tech, 15** (1978) 514.

X-29. Y. Shimazu, "Safety Aspects on Depressurization Accident of a Molten Salt Reactor", **ICAPP-03**, Cordoba (2003).

X-30. H.G.MacPherson, “The Molten Salt Adventure”, **Nuclear Sci.Tech., 90**(1985), 374-380.

X-31. K.Furukawa, Y.Kato and C.Urban: “Thorium molten Salt Nuclear Energy Synergetics System for the Global Energy Demand", **INAC-2002**, Brazil(2002), XIII ENFIR, R-14.

X-32. K.Furukawa, K.Mitachi, A.Lecocq, A.Furuhashi, S.E.Chigrinov: “Th Fuel-Cycle Deployment through Pu Incineration by THORIMS-NES”, **IAEA-TECDOC-916** [Proc.Tech.Comm. Meeting,”Adv.Fuels reduc.Actinide Generation”, Vienna, 21-23 Nov.1995](1996), p.115-128.

Appendix A-X. Design Description and Data for each Nuclear Installation with an Innovative SMR

A-X.1. Reactor Material Problems

Materials There are two principal materials in MSR : structural alloy Hastelloy N and moderator graphite, constituting the reactor vessel, tubing, reactor components and the moderator in the reactor core.

Structural alloy : Hastelloy N is served as the main container material whose components were Ni, Cr, Fe, Mo and the other alloying elements (see **Table A-X-1**). It had been developed for use with molten fluorides at less than 750 °C. To improve high temperature embitterment due to He (f.p.) bubble, two modified Hastelloy N had been developed where Mo, Si and B were reduced and Ti (1.5~2.0%) and Nb (2%) added (**Table A-X-1**). From thermodynamic analysis less noble Cr is most reactive element among those of the alloying constituents and the impurity. As expected from thermodynamics Cr depleted zone was observed on the surface exposed to MSRE fuel salt for 22000 hr at 650 °C [A-X-1]. The depth of the degraded zone did not propagate any more over 0.2 mil (1 mil = 25μm), which means that Cr as metal constituent attains an equilibrium with Cr component in fluoride melt. Apart from thermodynamic expectations, advanced corrosion tests simulating non-isothermal dynamic conditions had been performed by thermal and forced convection loop. **Figure A-X-1** shows the weight change of Hastelloy N and modified Hastelloy N for over 22,000 hr exposure to MSBR fuel salt in maximum temperature 704 °C and temperature difference 170 °C [A-X-2]. The corrosion specimens exposed in the hot legs resulted in weight loss and weight gain in the cold legs. The estimated corrosion rate of Hastelloy N was 0.02 mil/y and modified Hastelloy N exhibited higher corrosion resistance. These levels of corrosion rate are adequately acceptable to our design, although careful dehydration of salt and graphite is essential.

Standard Hastelloy N exposed to fuel salt under irradiation revealed the material embrittlement due to intergranular attack, where grain boundaries were degraded due to the existence of Te (F.P.). To improve Te attack **[A]** modified Hastelloy N had been developed where the addition of 1~2% Nb significantly reduces the susceptibility of Te intergranular attack (**Figure A-X-2**) [A-X-3].

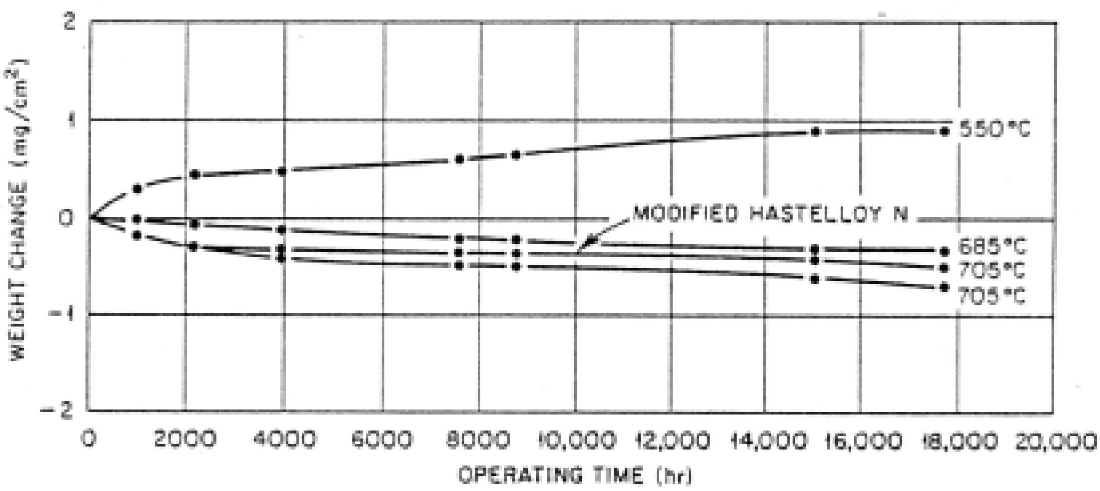

Figure A-X-1 Weight change vs. time of Hastelloy N specimens exposed to fuel salt in thermal-convection loop NLC-19A

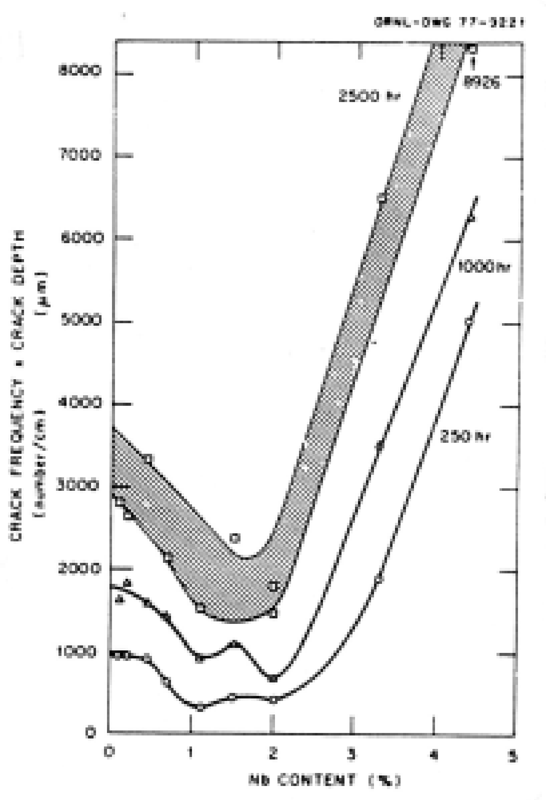

Figure A-X-2 Variations of severity of cracking with Nb content
Samples were exposed for indicated times to salt containing Cr_3Te_4 and Cr_5Te_6 at

Table A-X-1 Chemical compositions of Hastelloy N [X-3]

Element	Content (% by weight) *	
	Standard alloy	Favored modified Hastelloy N (Ti modified) − (Nb modified)
Nickel	base	base
Molybdenum	15 - 18	11 - 13
Chromium	6 - 8	6 - 8
Iron	5	0.1 * *
Manganese	1	0.15 - 0.25 * *
Silicon	1	0.1
Phosphorous	0.015	0.01
Sulfur	0.020	0.01
Boron	0.01	0.001
Titanium and Hafnium		2 --> 0 (1976)
Niobium		(0 to 2) --> (1 to 2) (1976)
Cobalt		low enough

* Single values are maximum amounts allowed. The actual concentrations of these elements in an alloy can be much lower.

** These values are not felt to be very important. Alloys are now being purchased with the small concentration specified, but the specification may be changed in the future to allow a higher concentration.

Table A-X-2 Chemical compositions of presented alloys, steels and standard Hastelloy N under investigation

Alloy or steel mark	Nb	W	Al	Si	P	Cu	Other elements
HN80MT	2.6	-	-	-	-	-	-
HN80MTY	-	-	1.03	0.01	0.006	-	-
12H18N10T	-	-	-	0.8	0.035	-	-
AP - 16	-	4.0 - 5.0		0.6	0.035	-	0.025 Ce
Hastelloy N	-	0.06	0.26	0.5	0.007	0.02	0.07 Cb

Alloy or steel mark	Ni	Fe	Cr	C	Mn	Ti	Mo
HN80MT	base	-	6.9	0.02	-	1.6	12.2
HN80MTY	base	0.06	5.5	0.03	0.01	0.83	12.3
12H18N10T	11 - 13	base	17 - 19	0.12	2.0	0.6 - 0.8	-
AP - 16	22 - 25	base	14 - 16	0.08	0.5 - 1	1.4 - 1.8	-
Hastelloy N	base	3.97	7.52	0.05	0.52	0.26	16.28

Graphite

Graphite is used for the moderator and reflector directly immersing in the fuel-salt. The basic requirements on the material were dictated by the research on MSBR performed at ORNL [X-1]. The material is required to be stable against the neutron irradiation, not to be penetrable by the fuel-bearing molten salt, and not to absorb Xe and Kr into itself.

Graphite has been employed as a nuclear material as long as there have been reactors. The extensive studies have been performed on the behavior of graphite by neutron irradiation, and numerous data on the material have been accumulated so far.

By the irradiation, point defects are formed and they agglomerate one another in each crystallite composing the material, which causes the remarkable growth of each crystallite in the c-axis direction and the little shrink in the other two directions [A-X-6]. Such change in the shape of each crystallite causes the distortion of the whole material. Life time of the material is determined by the failure criterion, and it is also decreased by the radiation-induced degradation of thermal conductivity. **Figure A-X-3** shows the volume changes for monolithic graphites, which were evaluated to be the best ones by the MSBR program at ORNL, irradiated with fast neutrons (> 50 keV) at 715 ^{o}C. The life times of these monolithic graphites lay in the 2 to 3 x 10^{22} n/cm^2, which means the time recovering to the original size after shrinkage [X-1].

The maximum dose with which graphite will be irradiated by fast neutrons higher than 50 keV (in the life for thirty years in MSR is estimated to be about 3 x 10^{23} n/cm^2 [A-X-4]. However, no materials which can withstand such a high dose have been developed yet. If, the aimed dose) is 2 to 3 x 10^{22} n/cm^2 in the best graphites produced among the MSBR R&D efforts of ORNL, EdF-CEA and former USSR . Therefore, although the graphite of MSBR of ORNL was designed assuming the replacement every three to four years, the graphite of FUJI will not be replaced in full life.

Thermal and radiation-induced stresses are no problem even for the relatively poorly behaved conventional type graphites. The problem of effective sealing graphite against salt penetration has been resolved by choosing the pole-diameter less than 1 micron m, considering the surface tension of fuel salt. It can be stated that graphite presents no serious problem for MSR, although the large size homogeneous graphite is not easy in fabrication.

However, a great need to improve the graphite still remains. If the life time (irradiation dose limit) is increased, the electric generation cost is lowered significantly. The irradiation characteristics and sealing characteristics should be improved by a well-qualified graphite manufacturer. In Japan, Toyo Tanso Co. (Pres. T. Kondo) is holding the top share of isotropic graphite in the world. He is an only manufacturer which developed and manufactured nuclear-reactor isotropic graphite IG-110 and supplied the reactor-core graphite for a currently operating high-temperature gas reactor HTTR at Japan Atomic Energy Research Institute and a reactor HTR-10 at China Tsinghua University. He promised to corporate with us. And, the basic research on the developed materials by the irradiation with energetic particles including carbon ions and high-energy electrons should be also performed in order to understand the mechanism of the damage more precisely and develop better materials.

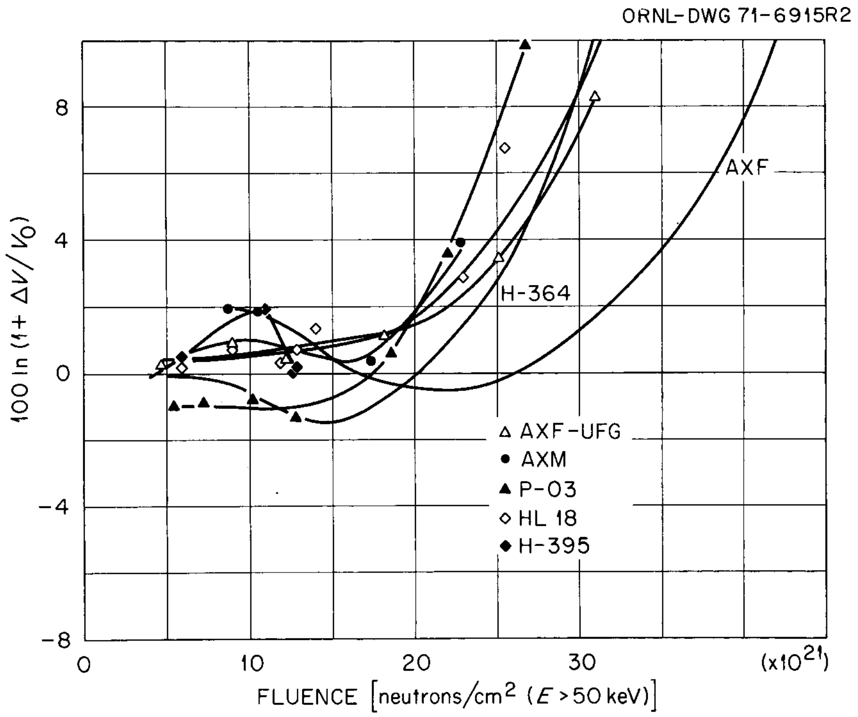

Figure A-X-3 Volume changes for monolithic graphites Irradiated at 715 deg.C (ORNL) [X-1]**.**

Reuse or recycle of materials: After ending the life of reactors, all reactor components are sent back to the regional centers for reuse or disposal. Low Cobalt Hastelloy N will mostly be able to remelt remotely in vacuum and recycle for the new components after cooling one year and grinding out the contaminated surface.

The graphite irradiated in low level will be able to reuse as reflector parts of the next reactors after grinding out the contaminated surface in 0.1 mm depth.

Monitoring : The development of monitoring techniques is necessary for ensuring sound and efficient reactor operation. Fortunately, the reactor system does not offer a continuous monitoring of major fuel constituents such as Li, Be, Th, F and U [A-X-4]. Therefore, there has been developed electrochemical in-line monitoring of the redox potential; only the U^{4+}/U^{3+} ratio, which responds to the corrosive atmosphere and the distribution of fission products and tritium in the reactor system. In-line monitoring of the U^{4+}/U^{3+} ratio in MSRE showed that the observed values were well agreed with ones obtained from thermodynamic and spectroscopic analyses, accompanied with a developed

Ni/NiF_2 reference electrode [A-X-7]. The U^{4+}/U^{3+} ratio in the fuel salt can be easily kept in the suitable region by varying dissolution time of Be into the melt.

With the aid of development in material tailoring technology, a few subjects as shown below will be solved efficiently :

* Modification of the alloy with rare earth, higher Cr and reduced Co contents
* Checking low enough tensile strength at the design (especially for the elbow part of tubing)
* Preparation of modified Hastelloy N data for ASTM standard and ASME coding
* Tensile test data, Ductility data, Creep test data, Toughness data on base and weld metal
* Required items along Guidelines for Required data on New Materials for Elevated Temperature Nuclear Construction
* Use of conventional cheep materials for the parts under less severe conditions

Additive 1 Te intergranular attack was prevented under the redox potential control**.** **Figure A-X-4** shows the U^{4+}/U^{3+} ratio, that is, the redox potential vs. the extent of cracking where there appeared little cracking at U^{4+}/U^{3+} ratio ≤ 60 and at the ratio above 80 cracking was extensive. The MSRE experiences suggested that Te possibly converts to innocuous telluride (e.g. "CrTe") by a reaction,

$$CrF_2 + Te + 2UF_3 \rightarrow 2UF_4 + CrTe,$$

where the equilibrium of the reaction is controlled by varying U^{4+}/U^{3+} ratio, that is, the redox potential by adding Be (reducing) or NiF_2 (oxidizing). Thus, Te causing the grain boundary degradation was decreased with decrease of U^{4+}/U^{3+} ratio by adding Be.

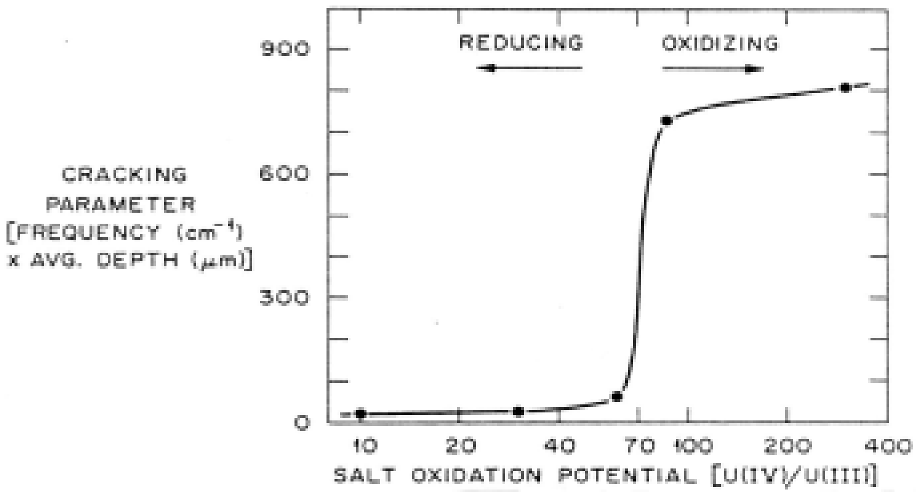

Figure A-X-4 Cracking behavior of Hastelloy N
exposed 260 h at 700 deg.C to MSBR fuel salt containing $CrTe_{1.266}$ [A-X-3]

A-X.2. Accelerator Molten-Salt Breeder (AMSB) System [X-6,-8,-10]

Accelerator Molten-Salt Breeder (AMSB) [X-6,-8,-10] is composed of three parts : (1) 1GeV-200~ 300mA Proton Accelerator, (2) single-fluid molten fluoride target / blanket system and (3) heat transfer and electric power recovery system. The schematic figure of its main part is shown in **Fig. A-X-5**. The size of target salt bath is 4.5 ~ 5m in diameter and 7m in depth.

The proton energy is selected as 1GeV for total system economy. Inside of the breeder vessel made of modified Hastelloy N is protected by graphite reflector. The target salt is introduced from the upper part forming a vortex of about 1m in depth. The proton beam is directly injected in off-centered position near vortex bottom, saving the emitted neutron leakage and improving the heat dissipation.

This target/blanket system is sub-critical, and essentially there is no influence by irradiation. The dynamic salt could manage the heat removal. The shuffling does not need at all. Except unknown engineering of a beam injection port even expecting to solve by gas-curtain technology, this simple configuration will be manageable in engineering, depending on the technical development for MSBR in ORNL.

The molten salt composition of standard spallation target is ^{7}LiF-BeF_2-ThF_4: 64-18-18 mol%, which is modifiable in several. The concentration of ^{233}U will be kept in 0.5 mol% adding diluent salt, which is prepared in chemical process plant after removing ^{233}U from spent fuel salt.. The target / blanket salts could be directly supplied as a fuel–salt for FUJI.

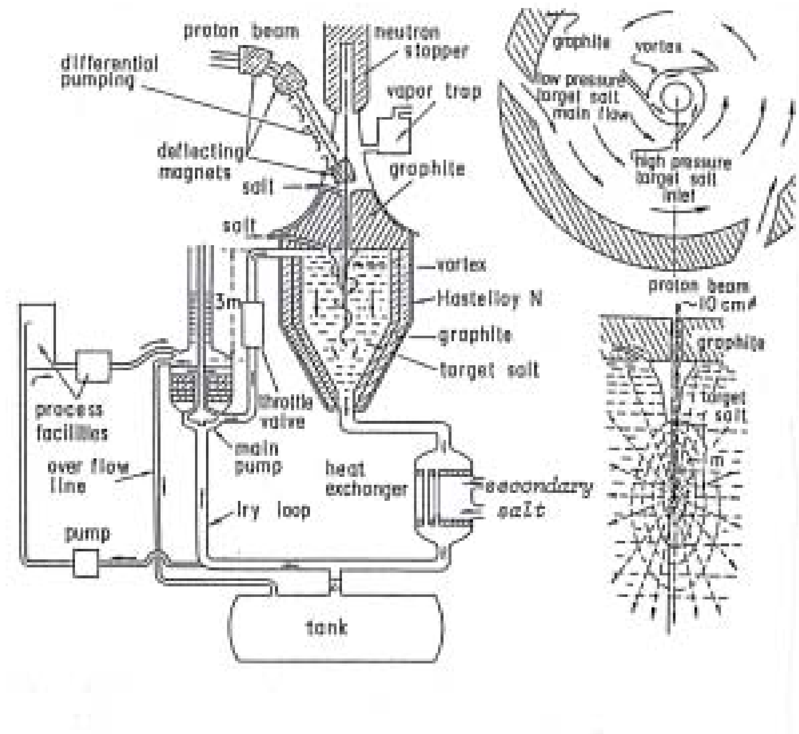

Figure A-X-5 Schematic figure of single-fluid molten-salt target / blanket system in Accelerator Molten-Salt Breeder (AMSB)

A model developed in JIPNR-Sosny NAS of Belarus describes the transport by Monte Carlo method permitting to better and more accurately represent the principal peculiarities of high energy particle interactions with matter. A characteristic feature of interaction of high-energy particles (about 103 MeV) with a dense medium is development of a nucleon-meson cascade where the main multiplying part is low-energy neutron component. Energy distribution of neutrons formed in the target is rather wide - from several eV up to hundreds of MeV. The main contribution to the neutron yields from the targets irradiated with high-energy protons is due to equilibrium stage of the spallation reactions which describes de-excitation of excited nuclei by means of evaporation of several light particles [X-6,-8,-10]

The yearly production of ^{233}U is about 400 to 450 kg, which is enough for commissioning one (fuel self-sustaining) FUJI power station in each year. This value can increase 50~100 % by adding PuF_3 for incineration, producing electricity enough for driving its accelerator in parallel [X-6,-8,-10]. After settling the operation of AMSB the $^{233}UF_4$ concentration would be increased till 0.3 to 0.5 mol%, which means the possibility of direct supply of target salt as a fuel of FUJI even necessary make-up of salt at reactor site.

Each “Breeding & Chemical Processing (Regional) Center” settled on about several-ten sites in the world will accommodate 4 to 10 AMSBs, two Chemical Processing Plants and one Radio-Waste Managing Plant. These Regional Centers are heavily safe-guarded.

A-X.3. Underground MSR: FUJI-ug

Drs. Ralph W.Moir and Edward Teller, Lawrence Livermore National Laboratory in a recent paper [A-X-11] recommend undergrounding the MSR plants such as FUJI, as shown in **Figure A-X-6**, to lessen the interest as a target for terrorists but not so deep underground (only 10 m) as to harm economics competitiveness. It recommends recycling unused fuel back into the plant to minimize transport of wastes and cut, by several orders of magnitude, the longterm wastes compared to present day nuclear power plants. In this paper, the long-lived wastes are considered as fuel to be further used rather than disposed of as in present day nuclear power plants. The non fuel-useable wastes would be stored at the same underground location for a long time while the power plant operates. Some components of the plant might be rebuilt many times during the plants lifetime. Finally, possibly as long as several hundred years or at any time, the stored wastes could be transported to a permanent disposal site. These strategies minimize the transportation of radioactive material. The use of carbon composite materials, not available in the early days of the molten salt reactor, could avoid a corrosion worry and allow higher operating temperatures ~1000 °C for higher efficiency power conversion and possibly hydrogen production.

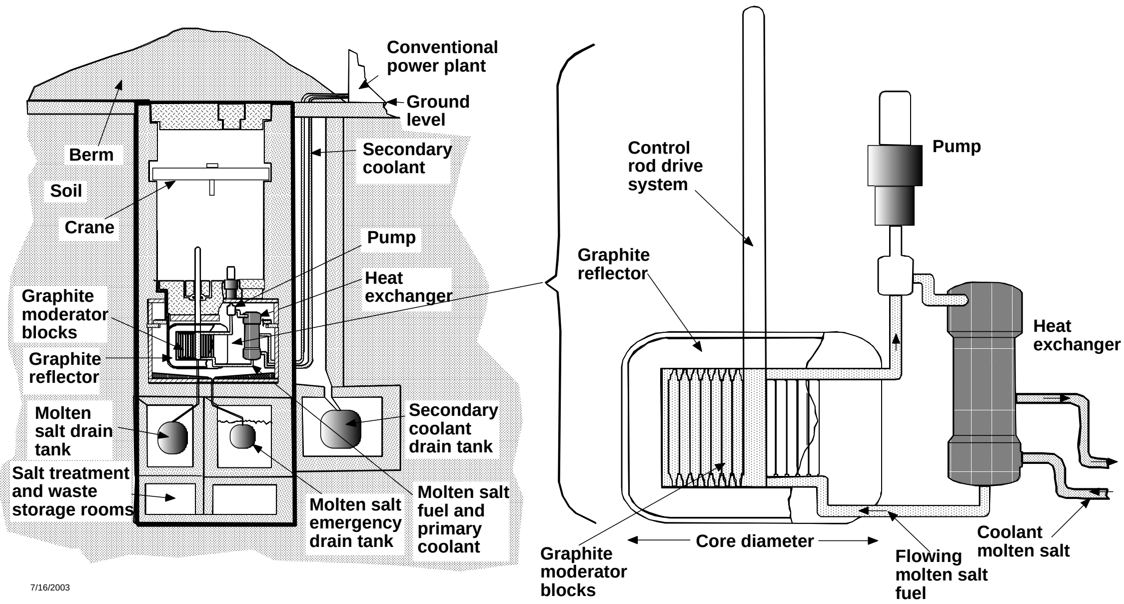

Figure A-X-6 The nuclear part of the molten salt power plant [See Furukawa, Mitachi and Kato, 1992 [X-4]] is illustrated below ground with the non-radioactive conventional part above ground; many rooms and components are not shown. New cores would be installed after each continuous operating period of possibly 30 years. Alternatively, the graphite in the cores could be replaced.

A-X.4. Stirling-engine MSR: FUJI-str

Professor L.Berrin Erbay, Osmangazi University, Turkey, proposed the steam-turbine of MSR-FUJI should be replaced with Stirling heat engine[A-X-12,-13]. The hot molten-salt (secondary coolant salt) transfers heat through a heat exchanger, which has constituted the heater (hot end head) of the Stirling engine.

The Stirling heat engine is considered as a free-piston type working at 950K. Stirling engines are known as the most efficient devices for converting heat into mechanical work or electric current as preferred in this study. They operate quietly, works on the principle of closed operating chambers, necessities long life designs with minimum maintenance and requires high temperature for high efficiencies. Further improvement of the Stirling engines is currently being undertaken for achieving less weight, more compactness, longer life, higher power level and efficiency.

Beside the expectations about small-scale nuclear power production by FUJI and more powerful Stirling engines in the future, the power levels of both sides in the present case are very different. The power inconsistency can be solved by coupling the coolant salt loop to a number of Stirling heat engines to supply sufficient heat removal from the nuclear reactor through the primary loop heat exchanger. The case is explained as a second alternative including the parallel connection of the Stirling heat engines which is shown in **Figure A-X-7**, schematically. Due to the parallel instrumentation, the effects of oscillation of the gas (herium) flow in the heater-head exchanger can be compensated and a steady non-oscillatory flow is obtained. A high mass flow rate is obtained by using parallel connection although a more complex installation system is necessary.

The system has particular problems to be studied in detail at least as follows:

1) The oscillatory nature of the flow in the heater-head aeffecting the flow of the coupling liquid in the secondary loop.
2) Heater-head design as a heat exchanger for the flows of a coolant-salt and a gas working in the engine.
3) The present power limit of Stirling heat engines.

Professor Erbay referred Flibe as a coolant salt, but it should be replaced with NaF-NaBF4 salt, which is lower melting point, economical and effective for collecting tritium.

In future, after improving to the higher outlet salt temperature, the MSR-FUJI-str will become more useful power station.

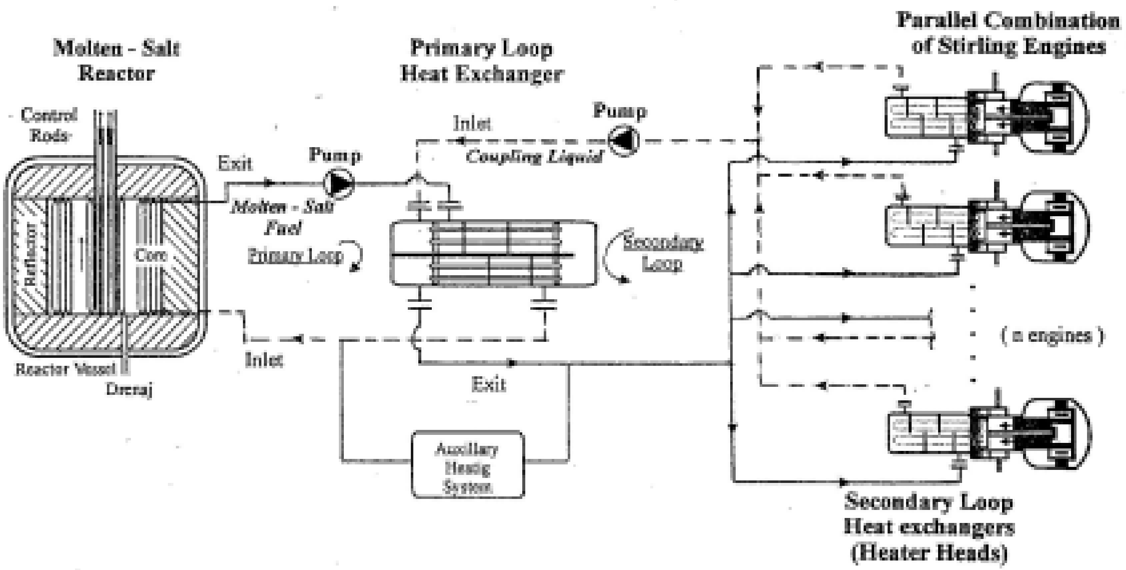

Figure A-X-7 Scheme of Present System with Parallel Sterling Engines

A-X.5. References

A-X-1 H.E.McCoy, Jr., "An Evaluation of the Molten Salt Reactor Experiment Hastelloy-N Surveillance Specimens- First Group", ORNL-TM-1997 (Nov.1967).

A-X-2 J.W.Koger, ORNL-4832 (1972) p.124.

A-X-3 J.R.Keiser,"Status of Tellurium-Hastelloy N Studies in Molten Fluoride Salts",ORNL/TM-6002(1977).

A-X-4 MSR Program Semiann. Progr. Rept. Aug. 1972, ORNL-4812, p.146.

A-X-5 V.Ignatyev, V.M.Novikov, A.I.Surenkov, V.I.Fedulov, IAE-5678/11(1993) Russian Research Center "Kurchatov Institute".

A-X-6 W. N. Reynolds, "Radiation Damage in Graphite", Chemistry and Physics of Carbon, 2 (1966), 121.

A-X-7 MSR Program Semiann. Progr. Rept. Feb. 28, 1969, ORNL-4396, p.200.

A-X-8 S.Chigrinov, K.Furukawa, V.Petlytskii, A.Kievitskaia, K.Rutkovskaia ″Calculation method of energy systems based on high energy particle and nuclei accelerator″, Proceedings of 7th ICENES, Sept. 1993. Makuhari, Japan. World Sci. Pub., Singapore (1994) P.159.

A-X-9 S.Chigrinov, A.Kievitskaia, K.Furukawa, A.Lecocq, V. Petlytski, K.Rutkovskaia ″Pu-239 burning-up, U-233 and thermal power producing in the THORIMS- NES concept″. Proceedings of GLOBAL'95, Versalies, France, Sept., 1995. V. 2. P. 1444 .

A-X-10 S.Chigrinov, A.Kievitskaia, V.Petlytski, K.Rutkovskaia, I.Rakhno, K.Furukawa, A.Lecocq ″Accelerator Molten Salt Breeder as Fissile Producing Component of THORIUMS-NES Concept for Energy Production and Transmutation of Plutonium″. In Proceedings of 8th ICENES, June, 1996. Obninsk, Russia. P. 564 .

A-X-11 R. Moir and E. Teller, "Thorium fueled underground power plant based on molten salt technology," submitted to ***Foreign Affairs*** (2004).

A-X-12 L.B.Erbay, "Examining the Power Generation by the Stirling Heat Engine combined with the Molten-Salt Reactor", Proceedings of ECOS'99, June 8-10, 1999, Tokyo, P.429-434.

A-X-13 L.B.Erbay, "A Discussion for the MSR-Stirling Power System", Proceedings of IECEC'01, July 29-August 2, 2001, Savannah, Georgia, USA, P.423-427.