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New Sustainable Secure Nuclear Industry Based on Thorium Molten-Salt Nuclear Energy Synergetics (THORIMS-NES)

Kazuo Furukawa[1], Eduardo D. Greaves[2], L. Berrin Erbay[3],
Miloslav Hron[4] and Yoshio Kato[5]
[1]Thorium Tech Solution Inc.,
[2]Universidad Simón Bolívar, Caracas,
[3]Eskişehir Osmangazi University,
[4]Nuclear research Institute Retz,
[5]International Thorium Molten-Salt Forum,
[1,5]Japan
[2]Venezuela
[3]Turkey
[4]Czech Rep.

1. Introduction

1.1 The energy problem

In the 21st century the stress due to environmental issues like the greenhouse effect, pollution, desertification, and local climate abnormality, as well as social issues like population explosion (100 M per year), poverty and starvation, may become intolerable, leading to large-scale disorder. However, it seems that there isn't a more effective measure for averting such disorder and solving human poverty as ensuring an adequate supply of clean and cheap energy. In principle, it is impossible to predict the future. Nevertheless, a hypothetical prediction based on reliable principles, can be quite useful. A future energy scenario based on the initial work of Marchetti [Marchetti 1985, 1987, 1988, 1992], and later modified by the members of the Thorium Molten-Salt Forum [Furukawa, 2006; Furukawa, et al. 2008] is shown in Figures 1-4. The growth rate of primary energy in the world is estimated at 2.3% yearly (see Figure 1).

In Figure 2 the historical/predicted fractional contribution F from prominent sources is shown as a function of time. In the figure the "logistic function" logarithm of $F/(1 - F)$ is plotted against the calendar year. The main sources of energy shown are wood in the past, coal, oil and natural gas at present and nuclear and solar for the future. For the solar energy two graphs are shown in view of the uncertainty in the introduction of this source for large-scale deployment. For nuclear energy two scenarios are shown, one with a total nuclear energy production measured in power times years of 900 TWe year and the other with 2000 TWe year (see Figure 3).

In the past 30 years the market share of usages of all main sources of energy (coal, oil, natural gas and fission) has been surprisingly constant as can be seen from Figure 2. This

logistic function analysis suggests that political or financial influences on the energy market have been stronger than market mechanisms resulting in non-rationality.
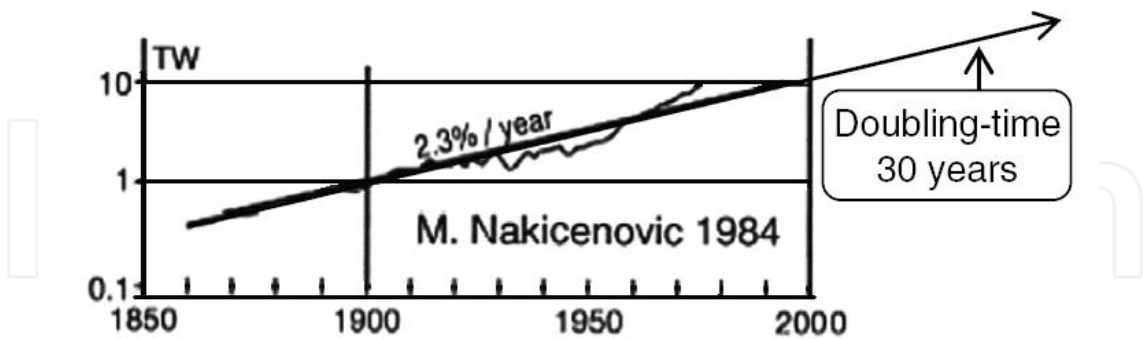

Fig. 1. Primary world energy consumption in the last 150 years.

In principle any variant of a solar-based technology could provide clean energy as it will not cause global warming or man-made abnormal weather patterns. But solar energy is low in energy density, irregular in output and it is still uneconomical and impractical for large industrial scale power plants. Even with a concentrated effort, the first industrial scale solar energy plant may only come on line after a few decades and large-scale deployment to meet projected demand would take more than 50 years after that (see Figure 2).
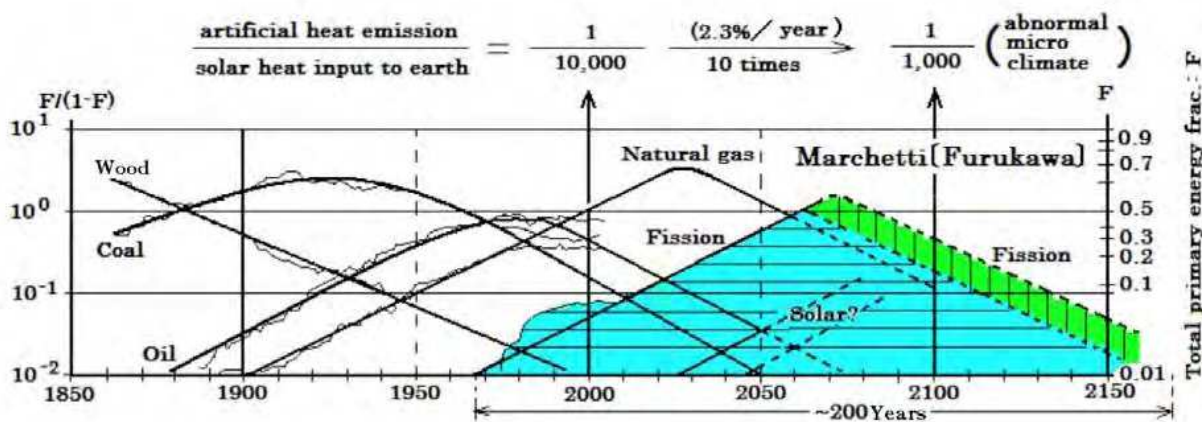

Fig. 2. Historical and predicted trends in energy usage by source in a 300 year time span.

CO_2 production and its attributed effect on global energy warming are expected to be a continuing problem. Even if a rapid increase in nuclear fission power generation takes place as shown in the shaded areas of Figures 2 and 3, the decrease of CO_2 will not be sufficient as shown in Figure 4

Other efforts such as local solar energy, eolic or wind power and hydroelectric power use etc., and in particular energy saving, are also necessary. The message of Figure 4 is that a great reduction in the use of energy will be required. Depending on the population explosion and energy demand increase, the artificial heat emission will increase tenfold in the next hundred years, based on the assumption of a 2.3 % yearly growth of world primary energy consumption as shown in Figure 1. A global warming, especially a local micro-climate abnormality will not be tolerable, and it will encourage shifting to the solar energy

era. In the long term there is no option but to curtail population growth and to reach a plateau of zero growth. Fred Hoyle [Hoyle, 1977], has argued that at current rate of energy production growth, within long times for human perception (a couple of thousand years) but very short times in geological terms, the amount of predicted energy generation on the surface of planet Earth will match the energy production on the surface of the Sun! The inescapable implication is that growth has to be curtailed until a state of equilibrium is attained with no increase in energy production.
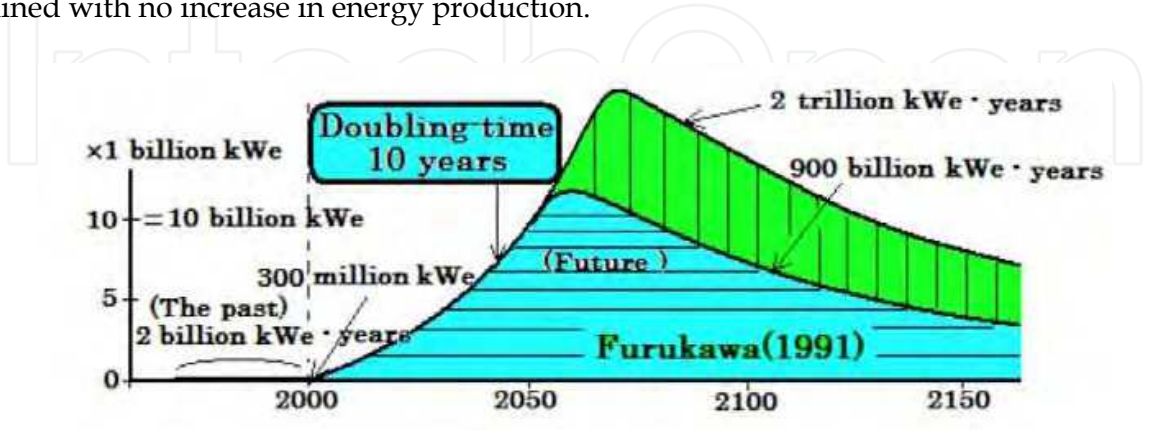

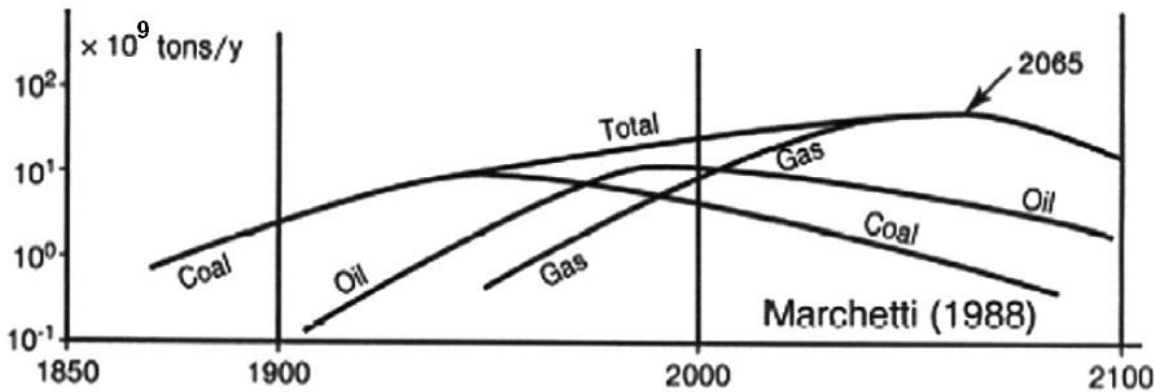Fig. 3. Fission Energy Production

Fig. 4. Historical and predicted CO_2 yearly emissions. A twofold increase from the present, 50 billion tons/year is expected by 2065.

A recent technological development with influence in the future energy panorama is the introduction of electric energy for massive surface transport. It is the result of the recent developments of new batteries with greatly improved (energy /weight) ratio which are revolutionizing land transport. As fully electric vehicles supplant the internal combustion engine vehicle, the reliance on fossil fuels, petrol and gas, CO_2 emitting fuels, will slowly decrease. This will create an additional, increasing demand on electric utility generation which will have to supply the energy load of land transport that currently is provided by gasoline, diesel and gas.

In a final analysis, at the present time, there is no other technological choice but to rely on nuclear energy. Hence a revolution in the global energy strategy is called for by increasing the investment for fission-energy systems so that we return to a rising fission use while the market share of other energy sources falls as shown by the curves of other energy sources in Figure 2.

1.2 The nuclear energy problem

The current structure of the nuclear industry is inadequate to the challenges that the human society requires in the next hundred years. In spite of 60 years of development the present nuclear industry presents a number of shortcomings that require a profound reassessment for a sustainable and secure nuclear industry. In spite of the notable safety record of the nuclear power electric generation industry, compared to almost all other forms of electric generation, there is the not unjustified fear that present nuclear power technology is not safe. The safety issue is of paramount importance for society to accept nuclear power. Nuclear radioactive waste and nuclear weapons proliferation issues of current technology are also high in the agenda for the rejection of current technology by society. If the foregoing problems could be solved then the economic factors take prevalence in the selection of the energy supply system for ensuring public acceptance.

The root cause of the problems outlined lie in the extreme complexity of the solid fuel reactor, which itself is the result of the form in which the technology developed. Nuclear energy was born in the war effort of the 1940's. The huge research and development investment in nuclear science to develop the atomic bomb, before and after the war in 1945, was applied to produce and develop the nuclear reactor. In the US and elsewhere governments financed nuclear power plants for naval use and nuclear facilities designed to produce the required materials ^{235}U and ^{239}Pu that allowed the construction of weapons. However, this effort was not confined to the war era or its immediate aftermath. The cold war that prevailed from 1945 to the fall of the Berlin wall in 1989, continued to dictate the science and technology that was developed. Hence concepts such as economy, simplicity safety and non-proliferation characters of the nuclear technology that was developed, were the least important factors taken into consideration in the building of the industry that we have now. The technological fallout from military applications is what mostly constitutes the present nuclear industry.

Thus the compact boiling water reactor used in warships, a technological development fully paid for by governments, became, after scaling up, the current BWR for civilian use.

Thus the nuclear fuel cycle that was able to breed extra fissionable ^{239}Pu for bomb production was chosen to supplant, with advantage, the production of ^{235}U which needs natural uranium mining and costly enrichment.

Thus the PUREX hydrometallurgical process was developed in order to be able to extract, from spent nuclear fuel, the pure plutonium for weapons manufacture and to obtain as useful by-product uranium with the remaining concentration of ^{235}U that could be used in other nuclear reactors. The nuclear energy industry that resulted from this, military biased, development has the following shortcomings:

1. The employment of discrete solid fuel elements containing either uranium enriched in ^{235}U or plutonium as metal oxides (MOX) or as metal alloys clad in special zirconium alloy. They have to be built to extreme quality standards so as to withstand, during its short service life, high mechanical stress factors in the form of high temperature, thermal shock, high pressure, and extreme gamma and neutron radiation doses.

2. The employment of very high pressures in a large reactor space for the containment of the pressurised water moderator and cooling media with a pressure flange to allow periodic opening for service and fuel elements change. This dictates a reactor vessel which has to be built to extremely demanding high standards in order to provide safety against a catastrophic failure; a technology which few enterprises worldwide can supply.

3. The wasteful and inefficient use of reactor fuel elements whose contained energy usage is in the range of only 5%, and which, in the case of reprocessing, requires destroying into scrap metal, chemical acid dissolution, refining and metallurgical reconstruction into new fuel elements.

4. The production of considerable radioactive nuclear waste which although being very small in size and weight in comparison with waste from burning fossil fuels, it is of orders of magnitude of higher toxicity and, the actinide fraction of it, of extremely long lifetimes.

5. The requirement of scarce sources of natural uranium. The known amounts of this source is a matter of debate, on whether they are sufficient for providing energy for future generations.

6. The production of large amounts of plutonium, of the order of 230 kg/year for each 1000 MWe power plant. This becomes a nuclear proliferation nightmare if deployed globally, particularly in view of the present century's phenomenon of uncontrolled terrorism.

7. The inability of current technology to satisfy the world's energy demand due to its long doubling time, of the order of 20-30 years, caused by the high complexity of the technology.

8. In view of these shortcomings, the future development of nuclear energy requires a profound and fundamental reassessment if it is to supply worldwide, plentiful energy and support a clean lasting human society.

In this paper we propose and describe a thorium molten salt nuclear energy system (THORIMS-NES) which is a complete concept designed to overcome most of the stated shortcomings by the employment of several important factors: The use of thorium instead of uranium as the fertile element, the eventual use of ^{233}U as the fissile element instead of ^{235}U or ^{239}Pu, the use of a liquid fuel instead of solid fuel elements and a stepwise chronology of introduction and development of items of technology. This system has the virtue of simplicity and will result in an affordable, sustainable, secure, clean and safe source of the required huge sized nuclear power industry and therefore will be acceptable to society so that humanity may look with optimism to a future of progress with plentiful energy for many generations.

2. New nuclear system THORIMS-NES

As stated above, the Thorium Molten Salt Nuclear Energy System (THORIMS-NES) is a complete fuel cycle concept which departs from current or presently employed fuel cycles. It proposes a power reactor which is radically different from current practice in the sense that: (A) – It uses a liquid fuel instead of solid fuel elements, (B) – It uses thorium instead of uranium as the as the fertile element to breed the fissile isotope ^{233}U. (C) - It separates the nuclear power production from the nuclear fuel breeding by proposing a simple thorium molten salt reactor (Th-MSR) devoted exclusively for energy generation by burning initially ^{235}U or ^{239}Pu and eventually ^{233}U. (D) – It proposes an Accelerator Molten Salt Breeder (AMSB) devoted exclusively to the production of fissile ^{233}U and (E) - It will incorporate fuel reprocessing in Regional Centers. It is a "Symbiotic" system with each function optimized by its simplicity.

The THORIMS-NES concept includes a planned timetable beginning, in the first stage, with the construction of the miniFUJI, a 10 MWe small power reactor whose purpose is to recover the know-how of the Oak Ridge National Laboratory (ORNL) obtained in the period 1964-1969 during which the molten salt reactor experiment (MSRE) took place [Rosenthal et al., 1970]. The miniFUJI is a demonstration reactor that may be developed in a short time

estimated at 7 years. The second stage is the building of the FUJI reactor. This is a 150 MWe thorium molten salt reactor planned to go online in 14 years and to be deployed worldwide as a affordable, simple, safe and reliable power reactor burning either ^{235}U or ^{239}Pu with the purpose of using up fuel derived from dismantling nuclear weapons from spent fuel reprocessing. The third and last stage estimated some 25 years in the future is the establishment of regional Breeding and Chemical Processing Centers with production of ^{233}U by thorium spallation in AMSB to supplant the use of uranium or plutonium and enter into the thorium nuclear power stage.

In the following section the properties of the various, present-day fuel cycles are summarized in order to point out how the THORIMS-NES concept is able to deal with the shortcomings and problems of current nuclear power technology for the sake of a sustainable and secure tomorrow.
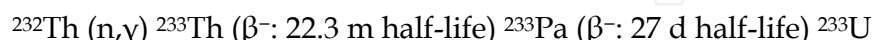
2.1 Review of nuclear fuel cycles

Table 1 contains a classification of Nuclear Fuel Cycles. The table is a modification of the classification introduced by W. H. Hannum, et al. 2005. It contains the following fuel cycles: 1.-Once-through route, 2.- Plutonium recycling in thermal reactor, 3.- Full recycling in fast reactor, to which we introduce a forth class of fuel cycle: 4.- Full recycling in molten salt reactor. For each fuel cycle there is a text about the various items which characterize it allowing a comparison of the virtues and undesirable qualities and a clear idea of the differences and advantages that the proposed THORIMS-NES affords.

2.2 Why thorium?

Thorium-based reactor fuels have a number of advantages over uranium–based fuels.

Th is geochemically three times more abundant in the Earth than U. Resources of about 2 M tons have been confirmed with estimated amounts of about 4 M tons [IAEA, 2000]. The amount of Th necessary for production of 1,000 TWe per year required for this century, as shown in Figure 3, is estimated at only about 2 M tons, which compares with more than 1.5M tons of U already extracted from the earth. Large resources exist as heavy components of "beach sands" which can be mined with little pollution.

Natural Th has only one isotope, ^{232}Th, of 100% abundance except for about 10ppm ^{230}Th. (An isotope which is fairly rich in Th from U-ores). Hence in the production of a fuel no "enrichment" of the fuel is required. Chemically refined thorium is added directly to the molten salt as discussed below. ^{232}Th in the reactor fuel is converted to the fissile ^{233}U by the reaction:

$$^{232}\text{Th } (n,\gamma) \ ^{233}\text{Th } (\beta^-: 22.3 \text{ m half-life}) \ ^{233}\text{Pa } (\beta^-: 27 \text{ d half-life}) \ ^{233}\text{U}$$

Fissile ^{233}U is suitable for thermal reactors with the advantage that with fertile ^{232}Th it can largely eliminate the production of long lived trans-uranium elements (TRU, or actinides) including Pu isotopes. These elements have exceedingly long half lives of the order of 10 000 years or more. Actinide production in a thorium-fueled reactor is estimated to be 2 or 3 order of magnitude smaller than that in a uranium-fueled reactor. This is due to the lighter nature of ^{232}Th against ^{238}U. The negligible production of plutonium makes the thorium-fueled reactor a nuclear weapons proliferation-resistant technology. Plutonium is the ideal isotope for the manufacture of atomic bombs due to the weak accompanying radioactivity.

Name	Once-through Route	Plutonium recycling in Thermal reactor	Full recycling in Fast reactor	Full recycling in Molten salt reactor
Fuel Process	Solid Fuel is burned in thermal reactor. Fuel is not reprocessed in general. (Occurs in the U. S.)	Solid fuel is burnt in thermal reactor. Metallic or oxide fuel chopped. Hydrometallurgy reprocessing recovers ^{235}U, ^{238}U and ^{239}Pu with **Purex** Process	Solid fuel is burnt in advanced fast breeder reactor. Metallic or oxide fuel chopped. Reprocessing done by Pyrometallurgy (Electrorefining) or Purex process recovers ^{235}U, ^{238}U and ^{239}Pu	Liquid fuel is burnt in Molten salt reactor. Radioactive gasses removed on line. Reprocessing done by Pyrometallurgy in molten salt. Recovers ^{233}U, ^{235}U, ^{238}U and ^{239}Pu. Fissile ^{233}U produced by Accelerator Molten Salt breeder.
Energy use	Uses about 5 % of energy in thermal reactor fuel and less than 1 % of energy in uranium.	Uses about 5 % of energy in thermal reactor fuel in each fuel charge. Overall recovery depends on number of fuel recharges	Uses up to 15 % of energy in reactor fuel in each fuel charge. Theoretically high burn up of energy in fuel. Depends on number of fuel recharges	Continuous burning of fuel without recharge. Minor fuel chemistry adjustment. Theoretically 100% of energy in fuel
Nature of fuel	Solid cladded fuel elements of metal oxide or metal. Uranium enriched in ^{235}U. Burns generated ^{239}Pu.	Solid cladded fuel elements of metal oxide or metal. Uranium enriched in ^{235}U. Burns generated ^{239}Pu.	Solid fuel elements. Metallic alloy of U-Pu-Zr. ^{238}U enriched in ^{235}U, ^{239}Pu with Zircaloy cladding. Burns generated ^{239}Pu, ^{235}U and some fissionable actinides	Liquid molten salt fluoride LiF-BeF$_2$ (Flibe) with dissolved fertile ^{232}Th and ^{235}U, ^{233}U or ^{239}Pu fissile. Blanket is ThF$_4$ producing ^{233}U.
Nature of cooling	Pressurized or boiling Water	Pressurized or boiling Water	Liquid sodium or lead metal	Liquid molten salt fluoride LiF-BeF$_2$ (Flibe)
Doubling time	-	—	Greater than 20 years [1]	7 - 10 years

[1] [Furukawa K. & Erbay L. B., 2010]

Table 1. Nuclear fuel cycle classification.

|			
Nature of Fuel container	High pressure vessel	High pressure vessel	Low pressure vessel (low carbon steel)	Low pressure vessel (Hastealloy N steel)
Safety assessment. Major accident	High pressure vessel explosion danger. Fuel element meltdown on loss of coolant circulation. Control-rods danger.	High pressure vessel explosion danger. Fuel element meltdown on loss of coolant circulation. Control-rods danger.	No high pressure vessel explosion danger. Reduced meltdown danger. Control-rods danger.	No high pressure vessel explosion danger. No meltdown danger. No control-rod danger.
Plutonium fate				
Weapons proliferation assessment	Increasing inventories of Pu in used fuel. Excess weapons grade Pu	Increasing inventories of Pu and available for economic trade.	Continuous generation of Pu. Possibly on site dry-reprocessing and burn up in reactor	Burn up of Pu from weapons and spent fuel. (Decreasing Pu inventories) Eventually possible total Pu burn up and elimination.
Energy cost assessment	High pressure vessel capital cost. High fuel fabrication cost. No fuel reprocessing cost. High transport and fuel repository cost.	High pressure vessel capital cost. High fuel fabrication cost. Fuel reprocessing cost. Reduced transport and fuel repository cost.	High pressure vessel capital cost. High fuel fabrication cost. Fuel reprocessing cost. Reduced transport and fuel repository cost.	No high pressure vessel capital cost. Nill fuel fabrication cost. Single molten salt media for power reactor, for fuel reprocessing and for ^{233}U breeding in AMSB.

[1] [Furukawa K. & Erbay L. B., 2010]

Table 1. Nuclear fuel cycle classification (continued).

[233]U can also be used to make nuclear weapons, if it were possible to get pure materials. But this is a difficult matter, as shown below. The military issue is a very confidential matter, but as was conclusively explained by Sutcliffe, a specialist of Lawrence Livermore National Laboratory (LLNL): "[235]U is most easily made into a weapon; Pu is next most easily made into a weapon; [233]U is hardest and least desirable for weapons." [Sutcliffe W., 1994]. "No nuclear weapons that have ever been built use fissile [233]U" [Sorensen K., 2010]. The reason for this fact is that [233]U fuel is accompanied by very strong gamma activity requiring sophisticated remote handling or a liquid-fuel technology for easy handling. The gamma activity is due to production of the [232]U isotope which takes place in a thorium-fueled system by the following five reactions [ORNL-5132, 1976; Ganesan, et al., 2002].

$$^{232}Th(n, 2n \& \gamma, n)\ ^{231}Th\ (\beta^-)\ ^{231}Pa(n, \gamma)\ ^{232}Pa\ (\beta^-)\ ^{232}U$$

$$^{230}Th(n, \gamma)\ ^{231}Th\ (\beta^-)\ ^{231}Pa(n, \gamma)\ ^{232}Pa\ (\beta^-)\ ^{232}U$$

$$^{233}U(n, 2n \& \gamma, n)\ ^{232}U$$
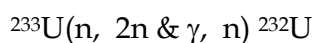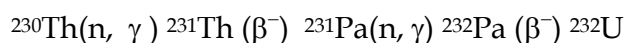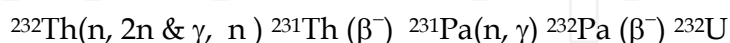
These reactions occur with the initial inventory of [232]Th and [233]U present and [230]Th, even though only traces are present.

[233]U is formed in situ with burn-up and thus [232]U is also formed by a small amount through the breeding reaction from [232]Th following the above:

$$^{232}Th(n, \gamma)\ ^{233}Th\ (\beta^-)\ ^{233}Pa\ (\beta^-)\ ^{233}U(n, 2n)\ ^{232}U$$
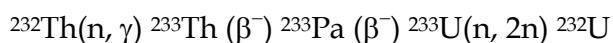
The production of [232]U is greater in a fast neutron spectrum because of the threshold nature of the (n, 2n) reactions. In other words, the production of [232]U would be higher in a fast reactor in comparison to the production in a thermal reactor. Other possibilities exist as discussed by Ganesan, et al., 2002.

The strong gamma activities associated with [232]U are such that detection of any diverted [233]U is easy providing increased security and non-proliferation. [Moniz and Neff, 1978; Ganesan, et al., 2002]. Transport of significant amounts of [233]U with more than 10 ppm level of [232]U require remote handling operations and constitutes a high radiological hazard that requires lead or concrete shielding. This property is such as to make impractical any form of diversion for illegal purposes. Note that it is the daughter products [212]Bi (1.8 MeV gamma) and [208]Tl (2.6 MeV gamma) isotopes that are very strong gamma emitters and not [232]U itself. These daughter products are formed after five successive alpha decays.

2.3 Why liquid fuel?

At the first nuclear reactor seminar that took place at Chicago University during World War II, in collaboration with some Nobel-prized scientists, Dr. Eugene Wigner [Weinberg A.M., 1997] argued: What is the nuclear power generator, primarily? Quite simply, it is a "Chemical Engineering Device", since it means "equipment for utilizing the nuclear chemical reaction energy". Wigner also predicted and recommended that in case of "Chemical Engineering Devices", a "fluid" concept would be most desirable as reaction media for the nuclear fuel, and advocated that an ideal nuclear power reactor would probably be "the molten-fluoride salt fuel reactor". This concept was later developed by Oak Ridge National Laboratory (ORNL), USA through the Molten-Salt Reactor Program (MSRP) during 1957-1976 [Rosenthal M.W, et al., 1972; Engel J. R. et al., 1980] under the able

guidance of his successor Dr. Alvin Weinberg. In the course of this program a Molten Salt Reactor (MSR) operated at ORNL during the four years between 1964 and 1969.

The operation was successful; it ran its course without any accident or incident and the program was fully documented. This extensive and invaluable literature is freely available in the WEB site established by Kirk Sorensen in 2010 [Sorensen K., 2010]. The operation of a power reactor with a liquid fuel as opposed to the well established practice of using solid fuel elements has a large number of advantages. These advantages are most apparent with the liquid media that was developed during the MSRP: A eutectic mixture of lithium fluoride and beryllium fluoride called FLIBE, with fertile thorium and fissile uranium or plutonium dissolved in the fluoride molten salt. (^{7}LiF-BeF$_2$-ThF$_4$-UF$_4$; 73,78 -16 – 10 - 0,22 mol %) This fluid serves a triple function: 1.- as fuel element, 2.- as heat transfer medium, 3.- as fuel processing medium. Each of these functions will be described in the following.

2.3.1 As liquid fuel element

In a molten salt reactor the fissionable isotopes, the fertile isotopes and the products of the nuclear reactor operation: both, fission products and heavy elements resulting from neutron capture reactions, reside as ionic elements dissolved in the molten salt. The liquid is forced to circulate in such a fashion that when it enters the reaction chamber, the presence of graphite moderator material creates conditions for nuclear reaction criticality. The fuel generates heat as the fission reaction proceeds. The heated liquid fuel exits the reaction chamber and the criticality of the fuel ceases while it circulates through the pump, heat exchangers and other devices before returning to the reaction chamber.

Under reactor operation the fuel is subject to an extremely intense field of α, β and γ radiation as well as a very high neutron flux which produces damage in the reactor fuel elements. Radiation damage is well known [Olander D.R., 1976; Weber H.W., et al., 1986]. It affects the crystal structures, produces point defects and dislocations in the solids and grain boundaries, swelling due to fission gases, pore migration and fuel restructuring. Solid fuel elements are heterogeneous materials and assemblies. There are possible interactions between components and different behavior of the constituents. Extensive studies have to be done both experimentally as well as modeling of the structural behavior of fuel elements and assemblies, for radiation damage assessment, whenever a change of components is proposed. This radiation damage determines a very short life for solid fuel elements such that safety determines an obligatory exchange when only 5%, to at most 10%, of the useful energy has been burned.

On the other hand, a molten liquid fuel is free from structural radiation damage. An ionic liquid can be considered a randomly organized dynamic aggregate of ions that has no fixed structure. Any effects on the atomic level produced by radiation such as atomic displacements due to nuclear fission or reactions are inconsequential and in no way alter the basic properties or structure of the liquid. This property determines that there is **no need for fuel element replacement** during the life of the reactor. The chemistry of the liquid fuel may be monitored and may be adjusted by a very simple addition of components in an external section outside of the reactor vessel. It is easy to add additional fuel salt containing fissile ^{233}U, ^{235}U or ^{239}Pu   in order to maintain an optimum fuel composition or likewise to remove some deleterious component as we discuss in the following.

An important advantage of a liquid fuel relates to radioactive gasses produced by the fission process. Radioactive gasses such as ^{133}Xe and ^{135}Xe in solid fuels are entrapped in the crystal

structure and produce fuel swelling. They also act as neutron poisons due to the huge cross section for thermal neutrons of ^{135}Xe of 2.6×10^6 barns [Stacey W. M., 2007]. The accumulation in the fuel represents a potential danger in the case of accidental release to the atmosphere. The presence of ^{135}Xe in the fuel requires additional reactivity in order to compensate for its neutron absorption properties. ^{135}Xe reactor poisoning played a major role in the Chernobyl disaster. [Pfeffer J. I. & Nir S., 2000]. A further deleterious effect of the entrapped radioactive gasses in solid nuclear fuels is the inability of a solid fuelled reactor to significantly decrease the power level. Reducing reactor power alters the equilibrium condition between ^{135}Xe production as a decay product of ^{135}I and its "burn up" as a result of the neutron capture reaction. A large reduction of power produces a buildup of ^{135}Xe to an extent capable of shutting down the reactor. Further problems are Xenon-135 oscillations due to the interdependence of ^{135}Xe buildup and the neutron flux which can lead to periodic power fluctuations [Iodine pit, 2011].

In a molten salt fuel reactor fission product gasses diffuse and are uniformly distributed in the fuel preventing these oscillations. Moreover: "Fission product Kr and Xe are virtually insoluble in the (Molten salt) fuel and can be removed, if the moderator graphite is sufficiently impermeable, by simple equilibration with an inert gas (helium)" [Grimes W. R. 1969]. Hence simple injection of an inert carrier gas such as He can continuously remove fission product poison gasses. The gasses are collected in active charcoal and can be stored and allowed to decay before final disposal. The poison gas removal and the possibility of fuel replenishment or retrieval imply that a molten salt reactor can operate at a low excess reactivity or "sub criticality" by leakage. These properties significantly reduce the possibility of any severe accidents. Furthermore, if poison gasses are removed, then the reactor power can be reduced or increased at will allowing it to follow the load demand without the limitation that ^{135}Xe buildup imposes on solid fuelled reactors.

The molten salt reactor shares with liquid-metal cooled reactors some advantages: The first is that the reactor vessel may operate at low pressure. The container housing the liquid metal or molten salt only requires resisting just the necessary pressure to ensure fuel circulation. Pressure range contemplated in a MSR is about 0.5 MPascal (4.93 Atm or 72,5 PSI) which contrasts to pressures in the range of 15 MPascal (148 Atm or 2180 PSI) as are used in PWR. Hence no large pressure sealing flange is required. This constitutes a significant safety and cost advantage. The possibility of catastrophic reactor vessel failure completely disappears in a liquid fuel reactor.

The second advantage shared by the molten salt reactor and liquid-metal cooled reactors is the feasibility of high temperature operation which is several hundred degrees higher than any water cooled reactor. This implies significantly higher thermal efficiency for electrical energy production as well as the possibility of using the high temperature for hydrogen production. Development of less expensive methods of production of bulk hydrogen is relevant to the establishment of a hydrogen economy which is being currently considered. [Häussinger P., et al., 2002].

A molten salt fuelled reactor has the property that increasing operating temperature in the reactor vessel produces a volumetric expansion of the liquid. This has the consequence of a corresponding exit from the reactor of an amount of liquid reactant fuel. This produces a decrease in the overall reactivity and hence an inherent automatic mechanism of power reduction. This negative reactivity coefficient with temperature is universally recognized as a most desirable safety feature for power reactors.

Among engineering circles a very popular dictum is: "Simple is beautiful". Perhaps the most attractive feature of a fluid fuel reactor is the beauty of its **simplicity.** This connects tightly with ECONOMY in general. Economy in capital costs, economy in fuel manufacturing costs and economy in operational costs. Economy is closely related to the possibility of nuclear energy deployment in lesser developed or underdeveloped countries. Bringing nuclear power to an economy level to make it competitive with coal fired power plants is the most powerful mechanism to replace fossil fuel utilization and meet greenhouse gas emission standards required by international agreements.

2.3.2 FLIBE as the fluid fuel medium

ORNL made a choice of a fuel-salt based on the ^{7}LiF-BeF$_2$ (FLIBE) solvent [Rosenthal M.W., et al., 1972; Engel J. R., et al., 1980; Yoyuuenn & Zousyyokuro. 1981; Furukawa, K. et al., 2005], on the basis of its very low thermal neutron cross-section, but also on the structural-chemical properties which make it very similar to MgO-SiO$_2$, which is a main component of the earth mantle and has a deep correlation with silicate slag useful in the metal refining furnace. Furthermore, it has very promising properties as a chemical processing medium [Furukawa K. & Ohno H., 1978].(see below).

A comprehensive data-book of FLIBE has been prepared [Furukawa K.& Ohno H., 1980]. The important thermo-physical properties of molten FLIBE are shown in Table 2 and are compared with other technologically important molten-salts and liquid Na. This solvent salt has significant and useful characteristics. It dissolves fertile ThF$_4$ and fissile ^{233}UF$_4$ (and/or ^{239}PuF$_3$) salts as shown in Table 3. Its flexibility is significant for the selection of fuel-salt composition. Suitable combination sets of fuel-salt compositions for obtaining a melting point (MP) lower than 773K [500°C] (lower zone in BeF$_2$) are:

^{7}LiF	73 -	73 -	74 -	74 -	72 -	69 mole %
BeF$_2$	19 -	18 -	16 -	15 -	16 -	16 mole %
ThF$_4$	8 -	9 -	10 -	11 -	12 -	15 mole %
^{233}UF$_4$			0.2 ---- 0.4 mole %,			

and for obtaining a melting point lower than 798K [525°C] are:

^{7}LiF	73----53	74---58	74---63	74---64 Mole %
BeF$_2$	20---40	16---32	13---24	12---21 mole %
ThF$_4$	7	10	13	15 mole %
^{233}UF$_4$		0.2 ---- 0.4 mole %.		

It is necessary to predict the viscosity coefficient of the fuel salt. This is easily obtained with the known semi-empirical method using the huge experimental data available [Cantor S. 1968] and applying the mutual replacement ability of UF$_4$ and ThF$_4$ due to the similar ionic

size. This is also supported by the good similarity of phase diagrams in LiF-BeF$_2$-UF$_4$ and LiF-BeF$_2$-ThF$_4$.

a.   (i) In UF$_4$ (+ThF$_4$) < 30 mole %, viscosity will decrease until 15 mole % BeF$_2$.

       (ii) In UF$_4$ (+ThF$_4$) > 30 mole %, viscosity will increase by an increase of BeF$_2$.

b.   UF$_4$ (+ThF$_4$) < 25 mole %, and 10 mole % < BeF$_2$ < 30 mole %, which is the most interesting region in MSR.

2.3.3 As heat transfer medium

The important thermo-physical properties of molten FLIBE and the characteristics of FLIBE-based fuel salt seen in Table 2 and 3, respectively, indicate that this solvent salt has significant and acceptable characteristics as a working fluid and coolant. Such excellent characteristics are based on (1) low pressure coolant, (2) highest heat capacity due to the main constituent ions being the smallest possible, (3) low viscosity fluid, and (4) suitable Prandtl number of 10-20 in the fuel-salt. Especially the parameter for heat-transfer per unit pump power has the highest value for FLIBE among the others included in the Table 2.

	NaNO$_3$-KNO$_3$-NaNO$_2$ [HTS]	NaBF$_4$-NaF	Li$_2$CO$_3$-Na$_2$CO$_3$-K$_2$CO$_3$	LiF-NaF-KF [FLINAK]	LiF-BeF$_2$ [FLIBE]	Na
Chemical composition. (mol%)	7 44 49	92 8	41 36 23	46.5 11.5 42	66 34	-
Melting point (K) [°C]	415 [142]	657[384]	672 [399]	727 [454]	732[458.9]	371[98]
Volumetric heat capacity C (J/m^{3}K) x10^{-6}	2.79	2.82	3.49	4.01	4.80	1.05
Density d (Kg/m^3) x 10^3	1.79	1.87	2.02	2.17	2.05	0.83
Thermal conductivity. h (W/mK)	0.59	0.35	0.55	1.2	1.00	66.
Kinematic viscosity. k (m^2/s)x10^6	2.26	0.8	11.8	4.2	7.44	0.29
heat-transfer capability (C/k)$^{0.4}$ h$^{0.6}$ x 10^{-3}	49.7	55.7	25.2	69.4	52.8	1.264
heat-transfer per unit pump -power C^3/d x 10^{-15}	12.1	12.1	21	29.4	53.8	1.39

The viscosity coefficients are about (6-7) x 10-3 Pa s and do not depend strongly on composition. More detailed physical properties data was summarized by Cantor (1968), although it should be supplemented by several reports of ORNL. The phase diagrams of binary and ternary fluoride systems are comprehensively collected by Thoma (1975) and Janz G. J., et al., 1978.

Table 2. Thermo-Physical Properties of Molten-Salts and Sodium (At about 770K [500°C]).

This parameter represents the power requirement for pumping the fluid through a channel, which is related to the design of the pump for a molten salt as working fluid and coolant in the power cycle. Since there will be no phase change during the process, it will enable the design of an effective and simple coolant loop. In the power - heat generation system the main design studies will depend on the thermal aspects of the heat generating fluid flow

system in the core and heat exchanger group (see Figure 5). There are no doubts about the FLIBE-based fuel salt being a suitable heat transfer medium.

Fuel-salt FLIBE: ^7LiF – BeF_2 – ThF_4 –($^{233}UF_4$ –$^{239}PuF_3$) Weak nuclear chemical reaction of solvent* Ideal ionic liquid with stable ions. Large heat-capacity & high fluidity.*,** Low melting-point (480-530ºC). **,*** Chemically inert, low aqueous solubility.** Compatible with Hastelloy-N (Ni-Mo-Cr) alloy. High (flexible) solubility for multiple ions.*** Solubility for nuclear fission/spallation products. Easy prediction of physico-chemical behavior.*** Very limited radioactivity release in accident. Easy reactor operation.	[mole %] [72-74]–[15-18]– [13-9] – [0.2-0.8] Very small thermal neutron cross-section Immune to radiation damage* Transparent, ambient pressure liquid** *** Single phase fluid: High Boiling point about 1400°C, no need to pressurize the system. Good compatibility with structural materials.**,*** Compatible with graphite (no wetting) Solubility for nuclear capture reaction materials (actinides). No-solubility for Xe/Kr/T *,*** If released it solidifies as a stable glass*** With trapping of radioactivity and no dissemination. Easy on maintenance/repair/dismantling
★ Triple-functional medium for * NUCLER-ENGINEERING, ** HEAT-TRANSFER MEDIUM *** CHEMICAL-ENGINEERING purposes	

Table 3. Summary of Characteristics of FLIBE-Based Fuel Salt

2.3.4 As fuel processing medium
Most present day spent fuel reprocessing is by a hydrometallurgical procedure called PUREX (**P**lutonium and **U**ranium **R**ecovery by **EX**traction). This is the most developed and widely used process in the industry at present. A number of variations of this basic process (UREX, TRUEX, DIAMEX, SANEX, UNEX) have been developed all of them being variations of the organic solvent extraction from aqueous solutions which result from acid dissolution of spent fuel. Alternative procedures that do not use water or organic liquids are high temperature processes called by the generic terms "Pyroprocessing or Dryprocessing" In this case solvents are molten salts (e.g. LiCl+KCl or LiF+CaF2) and molten metals (e.g. cadmium, bismuth, magnesium) rather than water and organic compounds.
In the THORIMS-NES concept the reprocessing media is FLIBE, the same media that is used as a molten salt fluid fuel. Although less developed than hydrometallurgical methods these high temperature procedures have a number of advantages. Among them 1.- They do not use solvents containing hydrogen and carbon, which are neutron moderators creating risk of criticality accidents, 2.- They are more compact than aqueous methods, 3.- They can separate many or almost all of the elements contained in spent fuel: remaining fertile or fissile uranium and plutonium, fission products and transuranic actinides, 4.- Simplicity of the separating equipment, 5.- No radiation damage is expected on the processing media, the liquid molten salt.
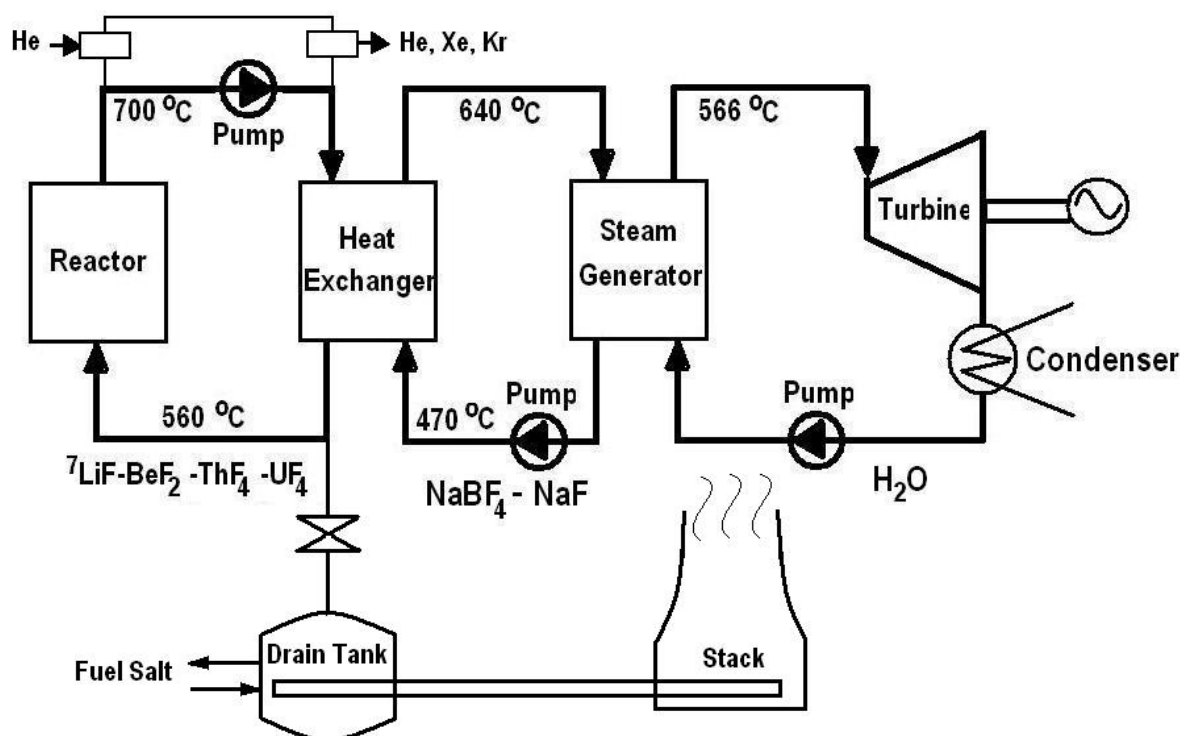
Fig. 5. Block diagram showing the three heat exchange loops in the FUJI series reactors

Separation of components in a molten salt can be achieved by a number of chemical processes such as: 1.- Electro deposition in a liquid or solid electrode [Kennedy J. W. 1950; Yang H., et al., 2010]; 2.- Absorption into a liquid metal cathode (cadmium or bismuth) [Delpech S. et al., 2008]; 3.- By the production of volatile compounds which can be separated by fractional distillation [ORNL-4577, 1971] or 4.- By selective precipitation of oxides [Rosenthal M.W et al., (1972)]. Pyroprocessing is the ideal procedure for processing fuel from a molten salt reactor. All the materials to be processed, separated or recovered are already in a suitable molten salt medium which can be used for the recovery process.

The Oak Ridge National Laboratory (ORNL) Molten Salt Reactor Program (MSRP) included on-line fuel processing of the fuel salt. Additionally, an intense R & D for chemical processing of spent fuel salts was done by ORNL [Whatley M.E (1970)]. One of the most significant advantages in THORIMS-NES is the elimination of the continuous chemical processing of the molten salt fluid during reactor operation. This deviation from the ORNL-MSRP is performed in order to achieve the greatest simplification of the reactor. The chemical processing of spent reactor fuel is considered in the THORIMS-NES concept as a separate operation from reactor operation. Fuel processing at the reactor during power operation is limited to removal of gaseous products.

Spent fuel reprocessing is to be performed under strict supervision at regional centers where chemical treatment of spent fuel takes place, breeding of fissile ^{233}U is carried out by an Accelerator Molten Salt Breeder (AMSB) and new molten salt fuel is produced. (see **4.2** below)

2.4 Construction materials: Hastelloy N and graphite
The FUJI reactor vessel and all components in contact with the molten salt as well as the AMSB are constructed exclusively by a structural Ni alloy, Hastelloy N, and graphite.

2.4.1 Structural alloy

Structural alloy Hastelloy N (Ni- 15~18% Mo- 6~8% Cr- 5% Fe) [Rosenthal M.W et al., 1972; Engel J.R. et al., 1980; Zousyokuro Y., 1981; Haynes International, Inc., 2002] is used as a main container material. It is composed of Ni, Cr, Fe, Mo and other minor alloying elements. To improve high temperature embrittlement due to He of fission products (f.p.), two modified Hastelloy N have been developed where Mo, Si and B were reduced and Ti (1.5~2.0%) and Nb (2%) added.

Less noble Cr is the most reactive among the alloy constituents. A Cr depleted zone was observed on the surface exposed to MSRE fuel salt after 22.000 hr at 650°C (923K) [McCoy H. E. Jr., 1967], but the depth of the degraded zone did not propagate above 0.2 mil (= 5Wm). Advanced corrosion tests simulating non-isothermal dynamic conditions had been performed by thermal and forced convection test loops. The weight change of standard and modified Hastelloy N after over 22.000 hr exposure to MSBR fuel salt at maximum 704°C (977K), and temperature difference 170°C [Koger J.W., (1972)] have been measured. The corrosion specimens 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 corrosion levels are acceptable to the reactor design, although careful dehydration of salt and graphite is essential.

Standard Hastelloy N exposed to fuel salt under irradiation revealed material embrittlement due to inter-granular attack, where grain boundaries were degraded due to the existence of Te (f.p.) although not penetrating deeply. To solve this problem the following action was taken: [A] Hastelloy N was modified adding 1 to 2% Nb significantly reducing the Te attack. [B] The redox potential, that is the U4+/U3+ ratio, control is essential for preventing Te attack [Keiser J. R., 1977]. The extent of cracking appeared very weak at a U4+/U3+ ratio ≤ 60, and extensive at the ratio > 80. The experience of the Molten-Salt Reactor Experiment (MSRE) at ORNL suggested that Te possibly converts to innocuous telluride (e.g. "CrTe") by a reaction: CrF_2 + Te + $2UF_3$ → $2UF_4$ + CrTe, where the equilibrium of the reaction is controlled by varying the U^{4+}/U^{3+} ratio, that is, the redox potential by adding Be (reducing) or NiF_2 (oxidizing). The potential should be kept within the region of stable Te compound (U^{4+}/U^{3+} <60) and beyond that of U-carbide deposition on graphite (U^{4+}/U^{3+} > 6). The Te problem will be solved by applying both measures [A] and [B]. Alternatively, Russian work developed candidate materials for the MSR. Under similar test conditions the alloy showed a maximum corrosion rate ≈ 6Wm/y [Ignatyev V. et al., 1993], and no trace of Te attack in the promising material. The development of monitoring techniques is necessary for ensuring sound/efficient reactor operation.

The reactor system does not need continuous monitoring of major fuel constituents such as Li, Be, Th, F and U [Rosenthal M.W. et al., 1972], because the chemical composition drift is very slow. Electrochemical in-line monitoring of the redox potential has been developed: the U^{4+}/U^{3+} ratio, which responds to the corrosive atmosphere and the distribution of f.p. and tritium in the reactor system. In-line monitoring of the U^{4+}/U^{3+} ratio in MSRE showed that the observations agreed well with the result of thermodynamic and spectroscopic analyses, accompanied with a Ni/NiF_2 reference electrode [Rosenthal M.W. ,et al., 1972] The U^{4+}/U^{3+} ratio can be easily kept in the suitable region by varying dissolution of Be into the salt. 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 metals) should be carried out as the first step of the miniFUJI project.

2.4.2 Graphite

Graphite is used as neutron moderator and reflector with the Th and fuel molten salt directly immersed in it. The basic requirements were dictated by the research in the MSBR at ORNL. [Rosenthal M.W et al., 1972 ; Engel J.R et al., 1980 ;Zousyokuro Y., 1981]. The graphite used should be stable under neutron irradiation, not penetrable by the fuel salt, and immune to absorption of Xe and Kr.

Under irradiation, point defects are formed and they tend to be agglomerated with each other in crystallites causing growth in the c-axis direction and a small shrinkage in the other two directions [Reynolds W. N., 1966] resulting in material distortion. The lifetime is determined by the failure criterion, and by the degradation of thermal conductivity. The volume change for monolithic graphite has been concluded to be the best criteria of quality after irradiating with fast neutrons (> 50 keV) at 715˚C (988K). Such results obtained at ORNL were supported by R&D of EdF-CEA and the former USSR.

Although the core graphite in the MSBR should have to be replaced every 4 years, the graphite in FUJI does not need be replaced in its full life. The effective seal of graphite against fuel salt penetration is resolved by choosing a pore-diameter less than 1 µm due to the surface tension of the salt. It means that graphite presents no serious problem for the MSR, although large size homogeneous graphite is not easy to fabricate.

If the irradiation dose limit of graphite can be increased, the electric generation cost of FUJI will be improved significantly. Toyo Tanso Co. (Japan) holds the top share of isotropic graphite in the world, having supplied reactor grade IG-110, for the High-Temperature Gas Reactor (HTGR) at the Japan Atomic Energy Research Institute (JAERI) and HTR-10 at China Tsinghua University. They promised to cooperate with the FUJI development, in the basic research. Irradiation with energetic particles, including carbon ions and high-energy electrons will be performed to understand more precisely the damage mechanism and to develop better materials.

2.5 Separation of power generation and fissile production process

At the early days of nuclear power it was realized that the operation of a nuclear reactor produced additional fissile material via the capture reaction on "fertile" isotopes: mainly ^{238}U or ^{232}Th. This realization opened the door to the dream of producing fuel in the same amount or even exceeding the amount of burnt fuel and opening a practically inexhaustible source of energy. This possibility gave birth to the concept of breeder reactors used in nuclear power plants to produce nuclear power and more fissile nuclear fuel than it consumes. Breeder reactors have been built and operated in the USA, UK, France, Russia, India and Japan. A breeding ratio substantially larger than 1 can only be obtained in a fast neutron spectrum. Hence water as cooling media is precluded and a substantial amount of experience has been obtained in fast breeder reactors cooled by liquid metal, either liquid sodium or liquid lead.

After considerable effort in breeder reactor development the "Fission Breeding Power Stations" has become a sophisticated, huge-size complex system. However, still insufficient in its breeding capacity and with a doubling-time longer than 30 years in the Fast Breeder Reactor (FBR) and 20 years even in the Molten-salt Breeder Reactor (MSBR) proposed by ORNL. [ORNL WASH-1222, 1972]. In order to overcome the breeding limitation of the power reactor and to ensure the required design simplicity, the THORIMS-NES system separates into distinct operations the power producing function and the fuel breeding function. THORIMS-NES is composed of simple power generation stations: Molten Salt

Reactors (MSR), named FUJI-series (See **3** FUJI Reactor, **4.1** miniFUJI Reactor;) and fissile producing stations Accelerator Molten Salt Breeder (AMSB). (See **4.2** AMSB**).** These two, power and breeding stations are complemented by batch-type process-plants (See **4.4** Regional Center) establishing a **Symbiotic** Th Breeding Fuel-cycle System. This separation recognizes that the fission reactors are neutron-poor and are best for producing energy and that fission power stations themselves should be simpler and size-flexible. They should be extremely reliable power producing units for continuous uninterrupted operation deployed in the neighborhood of population centers.

This separation also recognizes that the fuel breeding, neutron-rich fissile producer power plant does not need continuous operation. It may operate producing, in a batch-wise manner, the required fuel. This separation will be essential for establishing a huge-size breeding fuel cycle, growing with a doubling time of 10 years. The THORIMS-NES system also recognizes that at the present time there are considerable weapons grade ^{235}U and ^{239}Pu inventories resulting from the Cold War era. Also there are at the present time ^{235}U enrichment facilities as well as facilities under construction or operation which are issues of international concern. The weapons grade fuel is being used for power plant fuel manufacture [Megatons to Megawatts, 2010] and the MSR is an ideal platform for its burn up. This resource is not estimated to last very long. Another source of fissile material is the considerable amounts of spent nuclear fuel at repositories containing remaining ^{235}U and ^{239}Pu which can be accessed by reprocessing and recovering operations as discussed below.

The U.S. and Germany have abandoned reprocessing, and the plants in the UK, France, Russia, Japan, China and Pakistan can process only a fraction of the spent nuclear fuels accumulating all over the world. The reprocessing could be performed by a fluorination molten-salt chemical-processing Method. [Uhlir Jan., 2011]. France laid the foundation for this method. The former Soviet Union, with the cooperation of France and the former Czechoslovakia, nearly completed it by about 1988. [Novy, I., et al., 1989; Furukawa K., 2001]. In this method, the spent oxide fuels are pulverized and made to react instantaneously in fluorine gas (Called a "flame reaction" because of the high-temperature combustion). The fluorination flame reactor technology used as the basis of this method has been put into practical use on a large scale at Pierrelatte in southern France to produce uranium hexafluoride gas for uranium enrichment from uranium oxide. The former Soviet Union called it the "FREGATE" project. By eliminating the last process of solid nuclear fuel production from the original FREGATE process and aiming only at supplying molten-salt nuclear fuel, the resulting modified process becomes surprisingly simple. Further, the solid spent nuclear fuel around the world can be processed economically. The best and most economical solution is to utilize the FREGATE process all over the world to prepare plutonium-containing molten-salt to be burnt in a thorium molten-salt reactor. If FLIBE is added to the molten-salt resulting from the modified FREGATE process allowing the fluorides of various fission products to contaminate the salt, then a molten-salt nuclear fuel containing plutonium can be easily prepared. Plutonium and other nuclear wastes retained in the salt gradually disappear owing to neutron absorption reactions and radioactive decay while the molten-salt reactor is operated.

These resources depend on the economics of reprocessing against the production of fresh fuel from mining and enrichment operations. The existence of these two sources of nuclear fuel implies that, within the THORIMS-NES system, the deployment of the AMSBs (See **4.2** AMSB) and associated process-plants is only required some 25 years in the future.

3. FUJI reactor

3.1 FUJI reactor description

The FUJI-series power reactors are modeled on the successful molten salt reactor program (MSRP) carried out at ORNL [ORNL reports 2010]. However important differences are incorporated: FUJI Power reactors should be simpler, size-flexible and fissile-fuel self-sustaining, which allows a most simple/stable operation and requiring minimum maintenance work. Such idealistic performance was almost realized by the FUJI concept, eliminating the continuous chemical processing in situ and periodical core-graphite replacement, both of which were needed for the Molten Salt Breeder Reactor (MSBR). [Furukawa K., et al., 1985; 1989; 1990]. Figure 6 shows a vertical cross section of the primary fuel salt system of FUJI. A standard conceptual design of FUJI [Furukawa K., et al., 1987; 1992] is 350 MW thermal and 160 MW electric. The reactor-vessel is cylindrical 5.4 m diameter and 4.0 m high, inside of which it is filled only by graphite (93.9 vol.%) and fuel-salt as shown in Figure 6. The reactor-vessel is weld-sealed in the factory and does not need opening during its entire life. The core is constituted by liquid fuel directly immersed inside central hexagonal graphite rods surrounded by a graphite neutron reflector. Graphite inventory is 161 tons and spatially arranged to get a best performance attaining an initial conversion-ratio of 1.002.
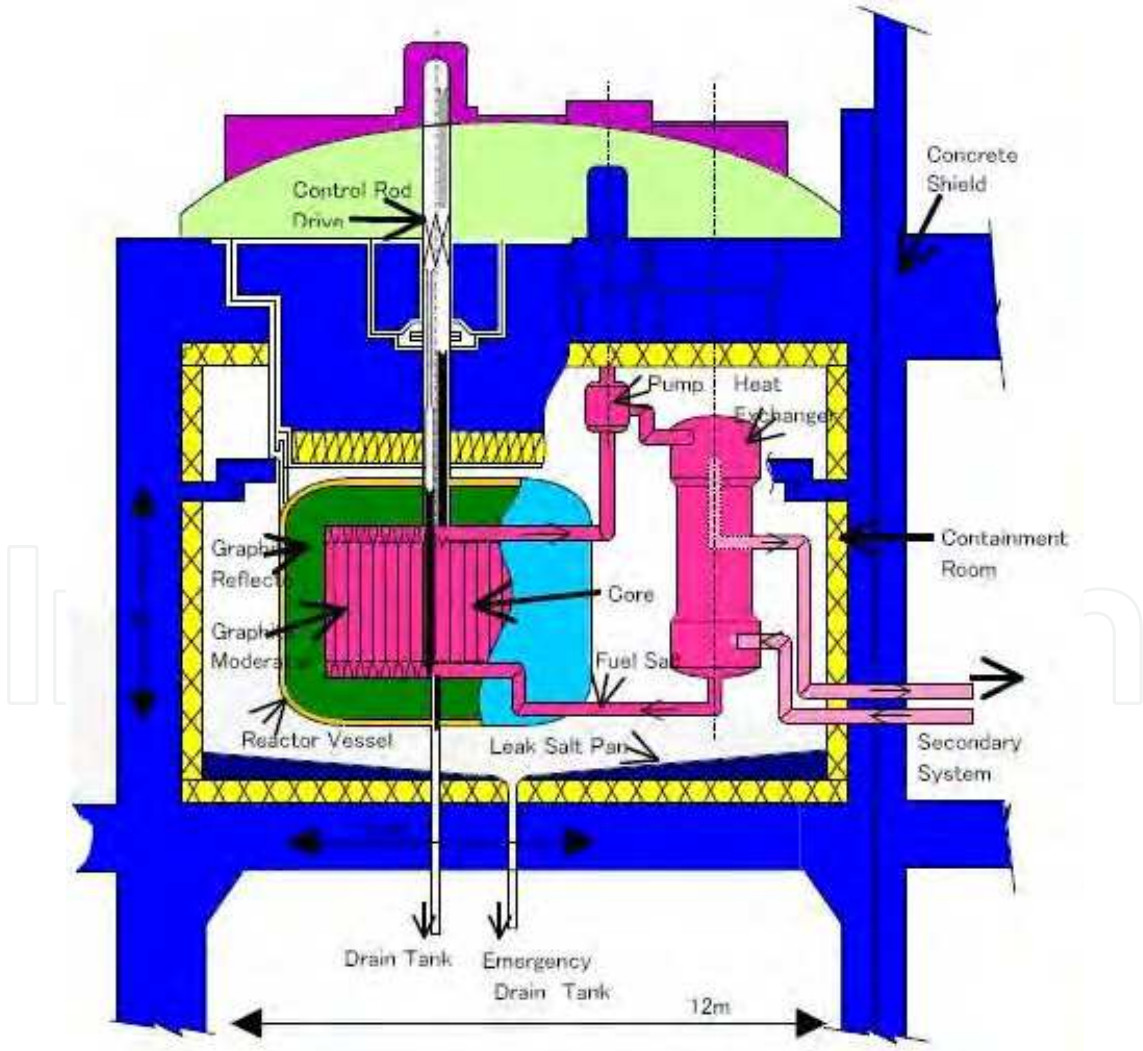
Fig. 6. Cross section of the primary system of Molten-Salt Power Reactor (FUJI)

The standard fuel salt of FUJI is ^{7}LiF-BeF$_2$-ThF$_4$-UF$_4$ (69.78-18-12-0.22 mol%). (See **2.3.2)** The total volume of fuel salt is 13.7 m^3 flowing upward at a rate of 33.2 m^3/min. The inner diameter of the main fuel piping is 25 cm. The structural Ni alloy, Hastelloy N (See **2.4.1**) is appropriate for up to 1170K or more. Hence it can operate as industrial-heat supply up to 930K, and in the future 1030K will be feasible. As such it allows for hydrogen production, as well as cogeneration, desalination, and district heating. Centrifugal pumps transfer the outlet fuel salt to heat exchangers, where the heat is transferred to a secondary coolant salt of NaBF$_4$-NaF, which transports the heat to a super-critical steam generator for electric generation, resulting in a thermal efficiency of more than 44%.

Several analyses have been carried out to establish nuclear characteristics of FUJI series cores which use several kind of fissile materials (^{233}U, ^{235}U and ^{239}Pu) with denominations such as FUJI-233U or FUJI-Pu [Mitachi K., et al., 1994; Mitachi K. & Furukawa K. 1995] and several output powers. A full schematic image of the FUJI molten salt reactor is shown in Figure 7. It includes the reactor containment building, primary heat transfer and cooling circuit, secondary cooling salt circuit, supercritical steam generator, turbines and electrical generators.
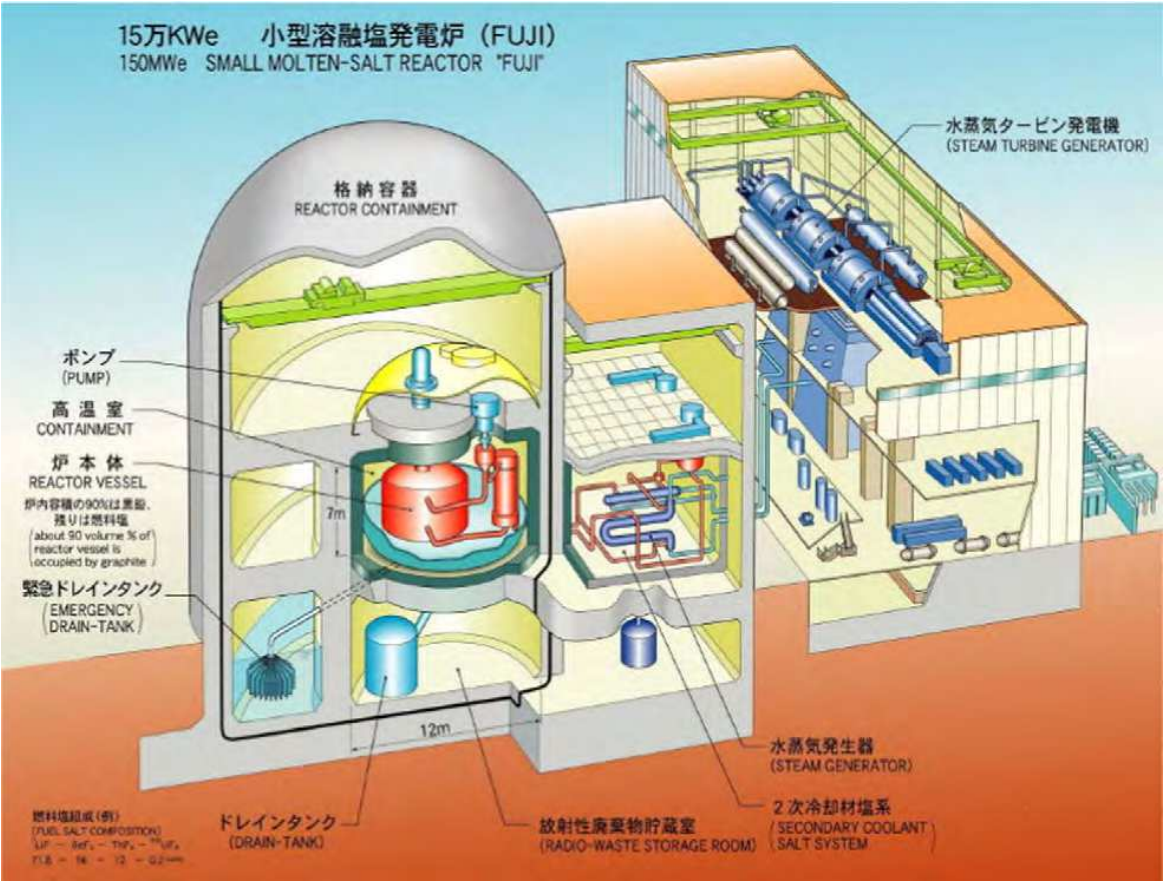
Fig. 7. Full view of FUJI molten salt reactor.

The reactor design has a three level containment security: The reactor core is contained in a primary Hastelloy N vessel which is inside the high-temperature containment. The tertiary level is the reactor containment building. The design is extremely safe as the fuel is only critical inside the core. In the unlikely event of reactor fuel salt leakage the molten salt will be caught by a spill pan and flow into a drain tank preventing any release of radioactive

material. Full power failure results in a freeze-valve melting and releasing all the fuel to a passively-cooled emergency drain tank surrounded by borated water.

3.2 Coolant salt

In the experimental reactor MSRE at ORNL, ortho-FLIBE ($^7Li_2BeF_4$, MP: 732K, see Table 2) was used as a coolant salt. In spite of its higher melting point, it has a high price, and was replaced by sodium-fluoroborate (92 mole % $NaBF_4$ - 8 mole% NaF, MP: 653K, see Table 2). The detailed physico-chemical data have been summarized [Cantor S., 1968; Ohno H. and Furukawa K., 1972]. This molten salt has a satisfactory good compatibility with Hastelloy N, and its contact with the fuel-salt would not induce any serious result. It has a slight higher vapour pressure over BeF_3, but only 174 Torr (23.2 kPa) at 873K. This salt has about 30 ppm hydrogen, which is useful for the collection of tritium penetrating from the fuel salt. Collected tritium (T) is extracted as water vapor (HTO) in the He cover gas, and the environment release will be less than the regulation limit. The T management has been successful in the MSR system.

3.3 Reactor chemistry

Reactor Fuel and the Behavior of Fission Products (f.p.)

FUJI will consume 400 g/day of ^{233}U including 350g/day by fission, meaning that 1920 kg (8300 moles) of ^{233}U are converted to f.p. in its full life. [Furukawa K., et al., 1989; 1990; Rosenthal M. W., et al 1972 ; Engel J.R., et al., 1980]. This behaviour will be explained by dividing the f.p. in the following three groups. Their crude amounts in full life operation were estimated as shown in Table 4.

Group I: Noble-gas f.p. and tritium (Kr, Xe and T). Kr and Xe have practically no solubility, and more than 99 % can be easily separated to cover-gas, reducing their huge neutron absorption to achieve a fuel self-sustaining condition. 210 g in total of T will be produced at the rate of about 380 Ci (14TBq)/day, decaying with the half-life of 12.36 y. From the experimental examination of ORNL [Engel J. R., et al., 1980; Houtzeel A. & Dyer F. F., 1972] the environmental release of T in FUJI will be less than 1 Ci (37GBq)/day (See **3.2**)

Group II: Soluble/stable fluorides (Rb, Cs, Sr, Ba, Y, the Lanthanides, Zr, Br, I, etc.) Almost all isotopes of Rb, Cs, Sr, Y, Ba have noble-gas precursors, which will be isolated to gas-phase of off-gas lines, and finally deposited in beds, except the short half-life isotopes: 32s ^{90}Kr (final daughter: Sr), 3.1m ^{89}Kr (Y), 3.8m ^{137}Xe (Cs), 14.1m ^{138}Xe (Ba), and some parts of their daughters will remain in the fuel salt. These fluorides are stably dissolved in fuel salt owing to the low concentrations (See Table 4), and will not have any severe problems in reactors. I and Br will be soluble as I- and Br- ions in the reducing condition containing an appreciable UF_3.

Group III : Noble and Semi-noble f.p. (Ge, As, Nb, Mo, Ru, Rh, Pd, Ag, Cd, Sn, Sb, etc,) They will exist in the elemental state in the reductive fuel salts. In the MSRE, the deposit of such elements mostly occurred as an accumulation of finely divided well-mixed materials rather than "plating" [Houtzeel A. & Dyer F. F., 1972]. Some part will escape as aerosol to off-gas phase.

The total amount of trivalent fluorides in fuel salt including U^{+3}, the lanthanides and Y^{+3} should not exceed 1.3 mole% of the solubility limit. In FUJI it will be safely about 0.63 mole % or less in the final stage. FUJI can soundly and simply be operated without any chemical processing except fission gases and T separation for its full life. The detailed examination of f.p. behavior in MSRE was reported by ORNL [Houtzeel A. & Dyer F. F., 1972; Rosenthal M.W, et al., 1972].

		production from ^{233}U	amount dissolved in fuel salt	separated to gas phase
I	Xe	27.6 a/o		312.0 Kg
	Kr	6.5 a/o		45.9 Kg
	T			0.1 Kg
II	I	2.6 a/o	27.6 Kg [0.032m/o]	
	Br	0.42 a/o	2.8 Kg [0.005m/o]	
	Te	4.1 a/o	43.5 Kg [0.050m/o]	
	Cs	17.8 a/o	56.0 Kg [0.060m/o]	144.0 Kg
	Rb	7.2 a/o	0.5 Kg [0.001m/o]	51.0 Kg
	Sr	11.8 a/o	28.1 Kg [0.047m/o]	60.5 Kg
	Ba	6.3 a/o	0.3 Kg [0.005m/o]	72.0 Kg
	Ce	14.1 a/o	166.0Kg [0.170m/o]	
	Nd	16.4 a/o	199.0Kg [0.200m/o]	
	Y	5.9 a/o	1.5-7.5Kg [0.003-0.013m/o]	42-37 Kg
	Zr	30.0 a/o	232.0Kg [0.370m/o]	2-10 Kg(?)
III	Mo	21.6 a/o	[deposit 175.9Kg]	2-10 Kg(?)
	Se	0.9 a/o	6.1 Kg [0.010m/o]	
	Sn	0.3 a/o	3.0 Kg [0.004m/o]	

Table 4. Predicted amount of fission products accumulated at the life-end of "FUJI-II" (in (atomic %), (mole %) and kg)

3.4 The reactor moderator

The reactor moderator is graphite (See Table 5) which fills a very large portion of the reactor vessel. [Furukawa K., et al., 1987] As shown in Figure 8, it is mostly built up of perforated graphite vertical rods of hexagonal shape. As shown in the figure, the rods have a varying size perforation. The core is divided in various sections: The central section houses the control rods. They are surrounded by Zone I of hexagonal graphite rods with the smaller diameter perforations. Zone II occupies the largest area and contains intermediate size perforations. In the outer, blanket zone of the core, the perforations are larger. The core is surrounded on sides, top and bottom by a graphite reflector.

The radial moderator-fuel variation has been designed in order to ensure the flattest neutron flux distribution in the core. A graphite density of 1,8 g/cm^3 was assumed in calculations. The graphite moderator is not expected to be replaced in the life of the reactor. To prevent damage, a neutron irradiation limit of 3×10^{22} nvt is selected (<50 keV). [Op cit.]. Therefore the maximum core flux should be less than 6×10^{13} n cm^{-2}s^{-1} in a 30 year life with 60% load as a local power station. High quality graphite with high irradiation resistance and small pore size (< 1 µm) is used.
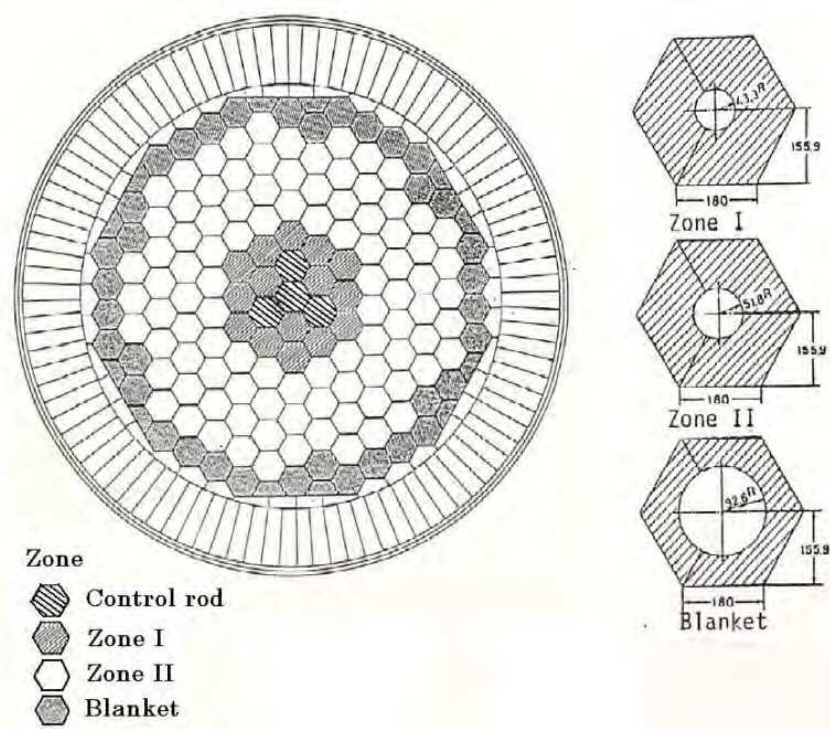
Fig. 8. Geometry of the FUJI reactor graphite moderator. Molten fuel flows through the circular channels in the graphite.

3.5 The reactor control system

The MSR FUJI has very high safety and reliability owing to the very small excess reactivity, due to the fuel-salt loss and other several advantages, and has essentially no possibility of a severe accident. [IAEA, 2007].

The MSR uses control rods which are made of graphite for the control of power. It employs shut-down control rods using B_4C, which are always withdrawn when the reactor is in operation. When inserting the graphite control rod into the core, this graphite functions as a moderator which promotes the fission reaction, contrary to the scram control rods.

Graphite rods will be drawn by floating power in a fail-safe mode.

3.6 The Reactor fuel chemistry control

The chemical valence shift by the fission of ^{233}U fuel salt is quite small. However, the development of a monitoring technique is necessary to ensure sound and efficient operation of the FUJI. Fortunately, the reactor system does not require continuous monitoring of the major fuel constituents such as Li, Be, Th, F and U [ORNL-4812, 1972]. Therefore, an electrochemical on-line monitoring of the redox potential has been developed; it only depends on the U^{4+}/U^{3+} ratio, which responds to the corrosive atmosphere and to the distribution of fission products and tritium in the reactor system. On-line monitoring of the U^{4+}/U^{3+} ratio in the MSRE has proven the results previously obtained in thermodynamic and spectroscopic analyses. These have shown that observations agreed with those from thermodynamic and spectroscopic analyses, in the presence of a Ni/NiF_2 reference electrode [ORNL-4396, 1969]. The U^{4+}/U^{3+} ratio can easily be kept within the suitable region by varying the time during which beryllium (Be) is dissolved in the melt.

	Experimental Reactor ORNL (Operated 1965-1969) MSRE	Pilot plant miniFUJI	Standard power station FUJI-II (Fuel self sustaining)
Heat capacity (MWTh)	7,3	16,7	350,0
Electric power (netMWe)	-	7,0	155
Thermal efficiency (%)	-	42[b]	44,3[b]
Reactor vessel size (m) (diameter x height)	1,45 x 2,2	1,8 x 2,1	5,5 x 4,1
Core: Max diameter (m)	1,14	0,6	1,4; 3,4[a]
Graphite frac. (Vol%)	77,5	90	93,90[a]
Blanket: Thickness (cm)	-	20	35
Graphite frac. (Vol%)	-	70	65
Reflector: Thickness (cm)	7?	40	68
High temp. Containment (m) Diameter x height	(5,8 x 7,2)	3,7 x 3,2	12,0 x 8,0
Core/blanket Power density Average-Peak (KWth/l)	2,9-6,6	16,4-24,9	9,5-17,5
Neutron flux: (n/cm²s) Maximum thermal	$0,5 \times 10^{14}$	$0,58 \times 10^{14}$	$2,4 \times 10^{14}$
Max graphite (>50 keV)	$0,3 \times 10^{14}$	$0,75 \times 10^{14}$	$0,8 \times 10^{14}$
Fuel conversion ratio	-	0,58[c]	1,002[c]
^{233}U inventory (kg)	32,0	27,0	370,0
[Per 1 GWe] (Ton)	[-]	[6,4]	[2,4]
^{232}Th (Ton)	-	0,65	20,1
Fuel salt:^{233}UF$_4$ (mol %)	0,14[c]	0,47[d]	0,22[d]
Total volume (m³)	2,1	0,45	13,7
[Reactor](m³)	[0,54]	[0,30]	[9,7]
Flow rate (m³/Min)	4,5	1,59	33,2
Temperature (°C)	632-654	560-700b	585-725b
Main piping (inn. dia.cm)	15,0	8,0	25,0
Graphite inventory (ton)	3,7	8,8	161,6

a The second core-zone.
b 2% up by "Ultra-ultra supercritical steam-turbine cycle [593 oC, 317 Atm]" at a fuel temperature of 725°C.
c In the stationary state after 500 days from start up.
d 7LiF - BeF2 - ThF4 - 233UF4 = (72-x) – 16 – 12 – x (mol%)
e 7LiF - BeF2 - ZrF4 - UF4 = 64,5 - 30,2 - 5,2 - 0,154 (mol%) (91% 233U)

Table 5. Main design parameters of three molten salt reactors: the MSRE at ORNL, the miniFUJI and the FUJI Power reactor. [Furukawa K., et al., 1992]

4. Technological description of THORIMS-NES

THORIMS-NES depends on the following three principles: [Furukawa K, et al., 1990; 2005] (I) Thorium utilization (See **2.2**). (II) Application of molten-fluoride fuel technology (See **2.3**). (III) Separation of fissile-producing breeder's process plants: Accelerator Molten-Salt Breeder (AMSB) from power generating fission-reactors: utility facilities Molten-Salt Reactor (MSR). This separation will be essential for the global establishment of breeding-cycle applicable over the world. This separation is dictated by the need for a doubling time of the fission industry to be 10 years or less as mentioned in section **2.**

The THORIMS-NES concept is composed of: simple power stations Molten Salt Reactors (MSR) named FUJI-series, fissile-producers Accelerator Molten salt Breeders (AMSB), and batch-type process plants establishing a symbiotic Th breeding fuel-cycle system. The THORIMS-NES concept also includes a detailed timetable for development [Furukawa 2008]. It includes a Short Term Program estimated in 7 years during which the prototype miniFUJI reactor would be built, a Middle Term Program, estimated in 14 years from the initial start of the miniFUJI, during which the small molten salt power station FUJI would be built, and a Long Term Program, estimated 25 years from the start of the miniFUJI, during which the AMSB and regional fuel processing center would be constructed in order to complete the Symbiotic Thorium Breeding fuel cycle capable of satisfying the huge electric energy demand (see Figure 1). In the following sections the main features of each of these programs will be described.

4.1 miniFUJI reactor

The development program begins with the Pilot-plant: miniFUJI followed with the FUJI power reactor. The development of FUJI-series MSR is based on the reliable R&D at ORNL [Rosenthal M.W et al., 1972; Engel J.R et al., 1980; Zousyokuro Y., 1981]. Almost all reactor engineering problems were clarified and/or solved. Their experimental reactor 7 MWth MSRE successfully operated 17.655 hrs without any accident (1965-69). The operation time of molten-salt loops was 26.076 hrs corresponding to just 3 years. However, it was about 40 years ago, and such technology experience should be rebuilt by the operation of the pilot-plant (miniFUJI) [Furukawa K., et al., 1987; 1989; 1990].

The fuel salt does not need any irradiation test, and the development of the structural steel Hastelloy N was essentially finished and is currently on the phase of performing an endurance test of modified Hastelloy N. The time and expenditure required for the development of FUJI would be short and small. The "miniFUJI" will differ from the ORNL-MSRE, in that it will generate electricity of 7 MWe. The design contemplates a reactor-vessel of 1.8m in diameter and 2.1m in hight [Harms A.A. & Heindler M., 1982] and with piping of 8 cm in diameter. Figure 9 shows a cross section of the miniFUJI. This facility could be commissioned in about 7 years. The demonstration reactor "FUJI" which will follow will be started with the detailed design work in parallel. [Furukawa, et al., 1983; Furukawa, et al., 1985] Table 5 provides comparative information of the main design parameters for the experimental MSRE operated at ORNL, the miniFUJI and the FUJI power reactor. [Furukawa, et al., 1992]

4.2 Production of [233]U in accelerator molten-salt breeder (AMSB)

During the 1980s, the technical feasibility of an accelerator-based nuclear fuel breeding facility AMSB [Baes C. F., 1969; 1974; McCoy Jr. H. E., 1967], was established based on a

''single-fluid target/blanket concept'' using the same kind of molten salts as FUJI, except with a higher ThF_4 content to establish an idealistic single-phase molten-fluoride fuel cycle. AMSB is composed of three parts: (1) a 1 GeV and 200–300 mA proton accelerator, (2) single-fluid molten-fluoride target/blanket system and (3) heat transfer and electric power recovery system. A diagram of the system is shown in Figure 10. The size of target/blanket salt bath is 4.5 m in diameter and 7 m in depth. The Hastelloy N vessel is protected by a graphite reflector.

The salt is introduced at the top forming a vortex of about 1 m in depth. The proton beam is injected in an off-centered position hitting near the bottom of the vortex to minimize the neutron leakage and to improve the generated heat dissipation. This target/blanket molten-salt system is sub-critical and is not affected by radiation unlike similar systems based on solid targets. This makes heat removal easy, and does not need target shuffling.
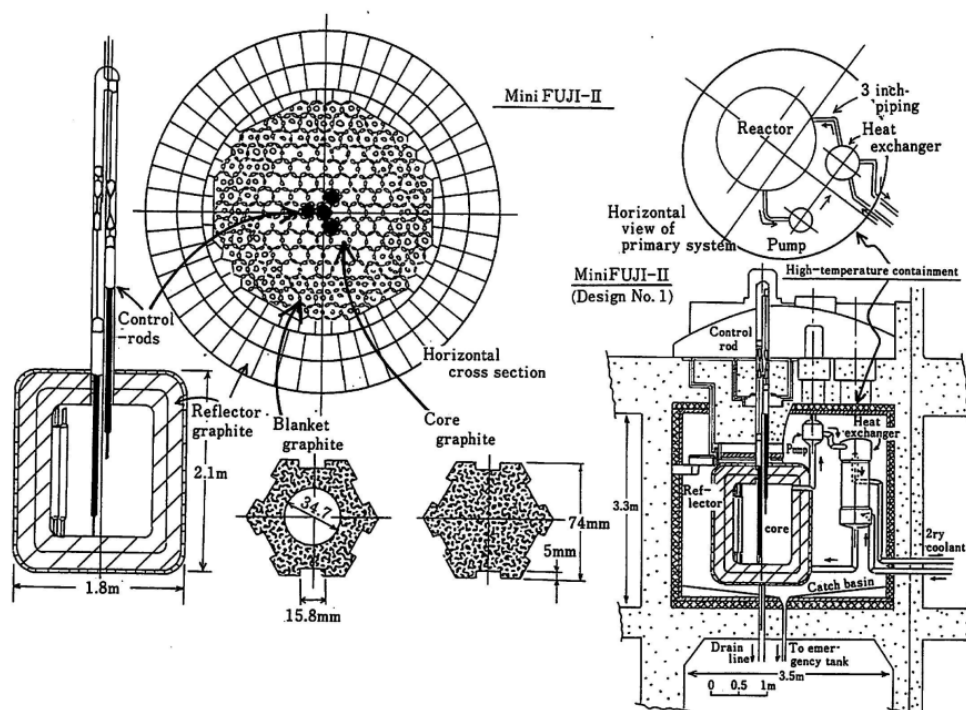
Fig. 9. Cross section of the miniFUJI reactor vessel and secondary containment housing the primary cooling circuit, control rod and the path for gravity fuel drainage.

The design of the beam injection port will be aided by improved gas-curtain technology. Engineering of this simple configuration, based on the MSR technology, will be manageable. The high proton current accelerator will utilize multi-beam funneling. The spallation neutrons transmute Th to ^{233}U and also cause fission in the target. The following two items need to be considered. (i) Suppression of the fission of produced ^{233}U, (ii) Utilization of the fission energy in the target/blanket salt for electric-power feedback for the operation of the AMSB. A heat output of about 1400MWth is required to achieve the power for the accelerator proton beam of 1 GeV, 300 mA. The above two requirements will be satisfied by adding Pu to the target salt composition; for example: $LiF–BeF_2– ThF_4–^{233}UF_4–^{239}PuF_3$: 64–18–17.15–0.3–0.55 mol%. The role of the Pu component is the same as FUJI-Pu, that is, burning itself and increasing the net production rate of ^{233}U.
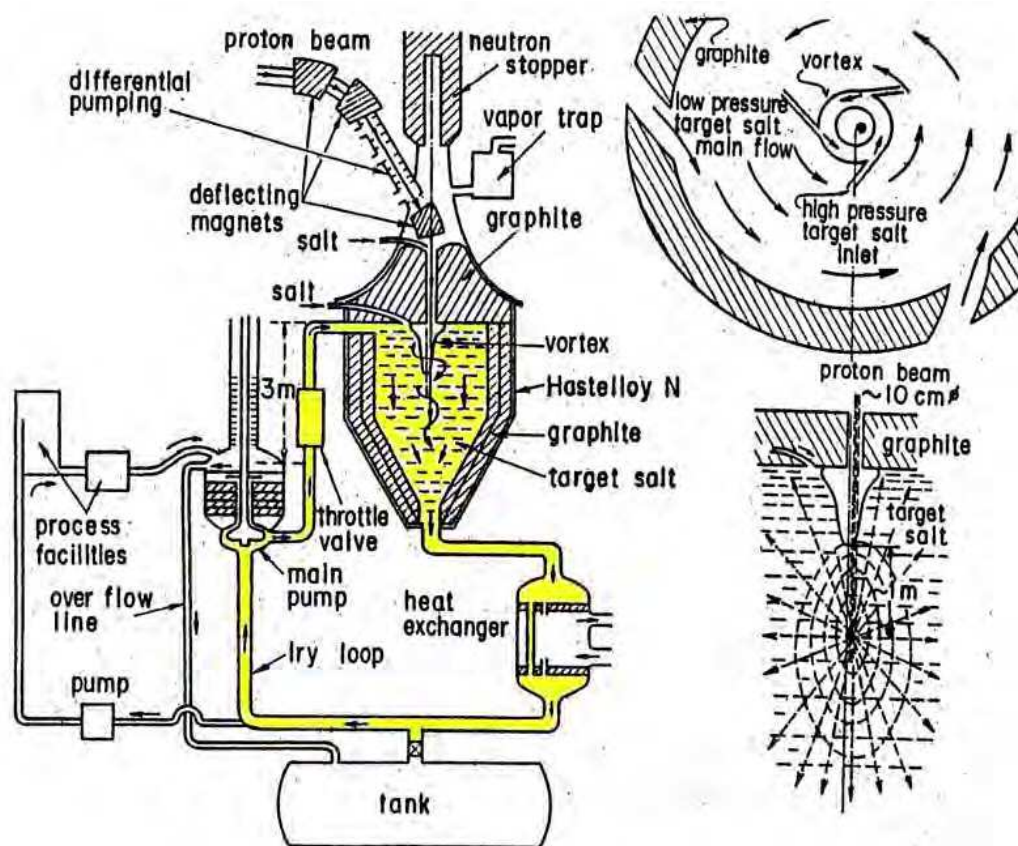
Fig. 10. Schematic diagram of single-fluid molten-salt target/blanket system in accelerator molten-salt breeder (AMSB).

The annual net production rate of ^{233}U is about 700 kg/y under the following beam and target conditions: [Koger J.W., 1972] Proton beam: (1 GeV, 300 mA), Target/blanket size: 4.5 m in diameter, 6 m in depth; Initial fissile/fertile inventory: (^{233}U: 2.240 kg), (^{239}Pu: 4.200 kg), (^{232}Th: 28.000 kg). This performance can be improved by in situ chemical processing.

4.3 Regional center chemical processing and fissile production

As mentioned previously the THORIMS-NES program contemplates three consecutive time terms leading to a global deployment opening of the Thorium Era in nuclear power utilization. The long term program, some 25 years after the initiation with the miniFUJI reactor (See **4.1**), implies the building of medium and large size molten salt FUJI power stations (See **3.1**), the establishment of fissile production: the Accelerator Molten Salt Breeder (See **4.2**) (AMSB, AMSB-Pu), as well as the establishment of molten salt fuel processing units to supply the power stations.

These last two operations are highly sensitive about the issue of weapons proliferation as it implies the handling of tonnage quantities of sensitive ^{233}U, ^{235}U and ^{239}Pu fissile material as well as highly radioactive fission products and extremely long lived actinides for treatment or waste disposal. Hence these facilities should be built in specially planned, 20–30 bases or ''Regional Centers'', heavily safeguarded under international supervision throughout the world. [Furukawa K., et al., 2008] Figure 11 shows in a block diagram the units, relationships and functions that the regional center would house.

As shown, after finishing the FUJI reactor life, the spent fuel-salt will be sent back to the "Regional Centers". The salt is processed in batch mode for removal of [233]U, other remaining fissile and some fission products (f.p.). Which elements are removed will be decided on the basis of optimization, material compatibility, neutron economy, cost economy, etc. [Harms A.A. & Heindler M., 1982]. The decontaminated diluents salt is adjusted to make-up to a suitable ThF_4 content and charged to AMSB to keep constantly a 0.5 mole% in [233]U content for the target/blanket salt. After starting the operation of AMSBs the thorium breeding cycle system will gradually be achieved. In the AMSBs (See **4.2**), 1GeV/300mA proton accelerators produces about 400 kg/year of [233]U.

The initial [233]U inventory of FUJI-233U (160MWe) is about 800 kg, so each of the AMSB can support the commissioning of one FUJI-233U every two years. Also, high gain type AMSB with Pu can produce more [233]U possibly resulting in an effective doubling-time of about 2~3 years and generate sufficient thermal output power to make the AMSB a self-sustained system, which will be able to start up sufficient numbers of FUJI reactors to meet the steep growth of energy demand.
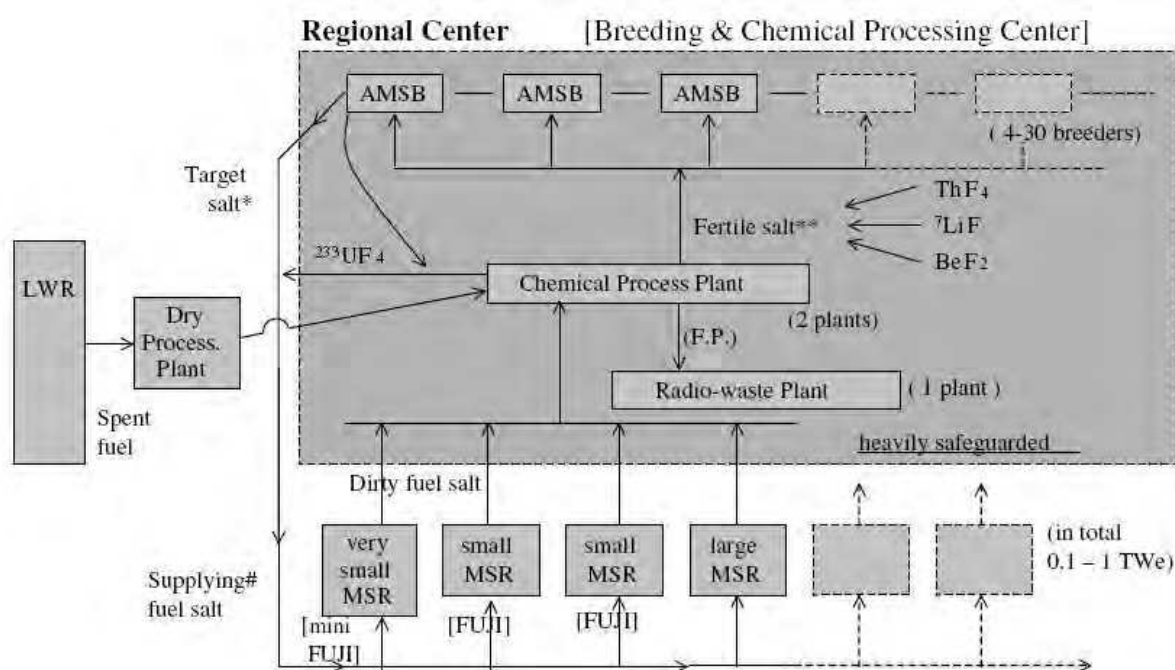
Fig. 11. Breeding & Chemical Processing Regional Center. Accommodating the AMSBs, chemical-processing plant and radio-waste plant. External input and output units are: Molten salt power reactors coupled by molten salt. The connection with the U–Pu cycle systems is shown via external or internal fuel processing plants.

The significant aspects of this fuel cycle are the followings: The total system is simply integrated by a single phase of molten fluorides based on $^7LiF\text{-}BeF_2$ solvent, after doing some make-up in each component facilities. Each "Breeding & Chemical Processing Regional Center" (See Figure 11) will accommodate 4 to 10 AMSBs, two Chemical Processing Plants and one Radio-Waste Managing Plant. It is contemplated that on the long term several tens of centers will be built in different locations in the world.

5. Technical advantages of FUJI

5.1 Safety

FUJI and the AMSB are significantly safe reactors, and there is no possibility of "Severe accidents". [Furukawa K., et al., 2002] The most important safety properties are due to the following:

1. The system pressure is lower than 0.5 MPa (72 psi, 5.1 kg/cm²)
2. The fuel salt is chemically inert, is not reactive with air or water, and suffers no radiation damage.
3. The boiling point of the fuel salt is 1673K, much higher than the operation temperature 973K.
4. The fuel will only become just critical when it coexists with the graphite moderator. Therefore, leaked fuel salt will not be able to introduce re-criticality.
5. FUJI has a large negative prompt temperature reactivity coefficient. The temperature-coefficient of graphite is slightly positive, but controllable due to the slow temperature increase due on its large heat capacity.
6. The delayed-neutron yield of ^{233}U fission is smaller than that of ^{235}U, but it is controllable due to longer neutron lifetime, and large negative prompt temperature coefficient.
7. The fuel composition can be adjusted at any time if necessary. The excess reactivity is very small and the control-rod worth is also small. A large reactivity shift by control-rods is not required due to the very small excess reactivity in a MSR.
8. Gaseous fission products such as Xe, Kr and T are permanently removed from the fuel salt, and their leakage in accidents are minimized.
9. Provision for loss of coolant circulation: In an event of a catastrophic accident in which total power failure prevents power delivery to the fuel pumps, so that fuel circulation and residual heat removal from the reactor is interrupted, such as the accident at the Fukushima I power plant in Japan due to the earthquake and tsunami on March 11, 2011 [Fukushima, 2011] the reactor automatically shuts down without operator intervention. In FUJI reactors the fuel is already in the molten state and quite fluid. If fuel circulation stops, the temperature inside if the reactor would increase. The Freeze Valve will automatically melt open by lack of valve-cooling due to electric-power loss. The molten fuel salt will drain to the Emergency Drain Tank (see Figure 5) covered by a big borated water pool, which will be cooled by a NaK Heat-Pipe. Re-criticality cannot occur in such an event.
10. The triple confinement of radioactivity is ensured in FUJI like in the other solid-fuel reactors. The reliability of each barrier in FUJI is much higher depending not only on the above (1), (2), (3), (8) and (9), but also on the soundness of the first barrier, which is composed of a simpler reactor-vessel, piping and intermediate heat-exchanger that have very thick tube-walls working at low pressure and without any serious irradiation and thermal-stresses. The event of a "severe accident" will never occur in FUJI even assuming full stoppages of primary and secondary pumps resulting in a reactor scram.

Even in the case of "severe reactor destruction" by military attack or sabotage, FUJI might be the safer reactor compared with any other ordinary Solid Fuel Reactors owing to the following: 1.- Only a weak release of gaseous radioactivity due to the continuous fission-gas removal. 2.- No core melt down, and no re-criticality due to the separation of graphite and fuel, which might be drained automatically and/or leaked. Leaked fuel-salt is solidified as a

stable glass confining radioactivity, and does not produce any troublesome aerosol (See **5.4**). 3.- Graphite (MP: 4000K) with large heat-capacity and thermal conductivity is not easily fired by virtue of inexistence of high temperature heat-sources. Even if a fire started, simple suffocation extinction will be enough, because graphite is not wetted by the fuel salt.

5.2 Nuclear proliferation and terrorism resistance

The present ^{238}U-Pu fuel cycle is difficult to protect, not only from nuclear proliferation but also from nuclear terrorism [Semenov B.A. and Oi N., 1993]. For example, an article in the US newspaper New York Times, 11th August, 2004 , discussed that an "American Hirosima" could take place by a 10-kiloton weapon at New York killing half a million people. Such a weapon could supposedly be stolen from Russia and transported undetected. This is because concealment of ^{239}Pu or ^{235}U is easy due to the very weak gamma activity.

On the other hand a weapon made from the Th - ^{233}U fuel cycle is very difficult to conceal due to the very strong gamma activity of ^{232}U associated with fissile ^{233}U as explained below. (see also **2.2**). Deputy-Director General, IAEA, Semenov and Oi (1993) stated that "Another way is to revisit the thorium/uranium cycle that is free of the stigma associated with plutonium." ^{233}U in fuel salt inevitably contains small amount of ^{232}U (about 500 ppm) and its daughter nuclides, produce very high radiation dose rates due to the high-energy gamma ray of 2.6 MeV from the daughter ^{208}Tl.

Even if fuel salt is withdrawn from the core, Pa separation should be done in a short time, because the decay of ^{233}Pa producing pure ^{233}U is fast, with a half-life of 27 days. The separation is very difficult, because spent fuel salt has very high radiation, and about 50 tons of fuel salts are necessary for getting 1 SQ (Significant Quantity for a weapon): 8 kg of pure ^{233}U from ^{233}Pa.The required 50 tons of salt is more than the fuel inventory of FUJI. The theft of 1 SQ : 8 kg of ordinary (dirty) ^{233}U is also very difficult due to the very low fissile concentration (about 1 wt%) and the reactor would stop. It would cause a lethal dose (about 1 Sv/h at 50 cm distance) of ^{208}Tl to the possible criminals. To shield this substance for handling, lead of about 20 cm in thickness is necessary, resulting impossible to transfer and fabricate the nuclear explosives, and very easy to detect by the high gamma radiation. FUJI produces negligible Trans-U elements (TRU: Np, Pu, Am and Cm) by virtue of the lower nuclear mass of fertile ^{232}Th, six units smaller than ^{238}U. This fact will greatly contribute to the non-proliferation and an effective incineration of TRU by the Th-U cycle system. In general the weapon usability of ^{233}U is classified information. But the specialist of proliferation issues of Lawrence Livermore National Laboratory (LLNL), W.G. Sutcliffe advised that "^{235}U is most easily made into a weapon, Pu is next easily made into a weapon, and ^{233}U is hardest and least desirable for a weapon", and "any of the above can be made into 'any old-type weapon' but a dedicated arsenal builder would prefer ^{235}U or Pu rather than ^{233}U. Therefore ^{233}U is less "weaponizable." [Sutcliffe, W., 1994]

5.3 Radioactive waste management

The difficult radioactive waste management becomes significantly easier owing to negligible TRU production in THORIMS-NES [Furukawa K., et al., 1991; David S., et al., 2007]. The production of Pu and Am + Cm in FUJI are 0.5 kg and 0.3 g Am + Cm for every 1 GWe y, respectively, i.e. very small compared to about 230 kg Pu and 25 kg Am + Cm in Light Water Reactors. Furthermore, avoiding solid fuel fabrication results in a lower production of low level radioactive waste and the need for maintenance.

In the long term after about year 2070, the recession age of the Thorium Era as shown in Figure 1, some very long-lived radioactive wastes should be transmuted to stable isotopes using the neutron capture reaction. Such work is not easy within the ordinary fuel-cycle. However in THORIMS-NES huge amounts of low-cost neutrons could be easily obtained, because excess fissile fuels could be burned out generating neutrons for this purpose. The circulating molten-fluoride system is the "best medium" for transmutation work due to the high solubility of several kinds of ions, good reaction-heat reservoir and no radiation damage. This is the case for both circulating systems: FUJI, a thermal neutron system, and AMSB, a fast neutron system. Considering this scheme, managing the radioactive waste becomes not a "Million Years" issue but rather a "Hundred Year" problem. This argument is shown in Figure 12 where the radio toxicity of nuclear waste is presented as a function of storage time. [Adapted from David S., et al., 2007]

On the issue of management of non-nuclear materials at the end of reactor life: All components would be sent back to the regional centers for reuse or disposal. (see **2.5**) Hastelloy N would mostly be re-melted and recycled and graphite, that has received low irradiation levels, would be reused after grinding the surface to 0.1 mm depth thus minimizing production of low level radioactive waste.

5.4 Operation and maintenance

Reactor systems in THORIMS-NES are quite safe (See **5.1**), and their operation and maintenance simple and easy. Some additional evidence of this are: 1.- The reactor vessel is not opened in its life, and the control rods are very few (or no rods might be feasible). The reactor vessel is a simple welded tank without any big flange or fuel-handling machine. 2.- The primary circuit system is confined inside a high-temperature containment at about 810K, which results in a simpler reactor configuration without any heater, insulator, neutron shield or instrumentation with the exception of some flow-meters and surface-level meters, etc. Therefore, fully remote maintenance, inspection and repairing would be easier, without any worker's exposure. The recent advances of robot technology are a great benefit. 3.- In the unlikely case of leakage of fuel salt, the bottom of the high-temperature containment is a spill-pan, whereby leaked salt is guided to the drain-tank. (See Figure 5) If necessary, the salt can be washed out by clean carrier salt (^{7}LiF-BeF$_2$). 4.- A MSR has excellent load-following ability by virtue of the liquid fuel and the removal of poison gas ^{135}Xe. The output power can be controlled by the fuel salt flow rate due to its negative temperature coefficient of reactivity not using the control rod. In the event of a lowering of temperature the salt freezes as a water-insoluble glassy matter stably sticking on the walls. The reactor building will be negligibly contaminated not as in an "Aqueous Homogeneous Reactors "[Lane J.A., et al., 1958], where radioactive aerosol will be dissipated after fuels dry up.

As MSR (FUJI-233U or FUJI-Pu) are self-controllable, the operational work is simple comprising the following main items: (a) off-gas management, (b) redox electrochemical potential control of the fuel salt, and (c) semi-periodical additions of Th, Pu and/or U in the eutectic-salt forms: [Th] ^{7}LiF-29 mole % ThF$_4$ (MP: 838 K), [Pu] ^{7}LiF-19.5 mole % PuF$_3$ (MP: 1016 K), and [U] ^{7}LiF-27 mole % UF$_4$ (MP: 763 K). Th is added once per year holding the variation to less than 1 wt %. The chemical balance due to the fission reaction of ^{233}UF$_4$ will introduce some excess free F atoms, which is controlled by Be addition. But in FUJI-Pu the main fission depends on ^{239}PuF$_3$, and neutrality will nearly be held. By the above additions of fuel components, ^{7}LiF content will increase, which will be corrected by BeF$_2$ addition. Component additions are achieved by the use of the above eutectic salts.
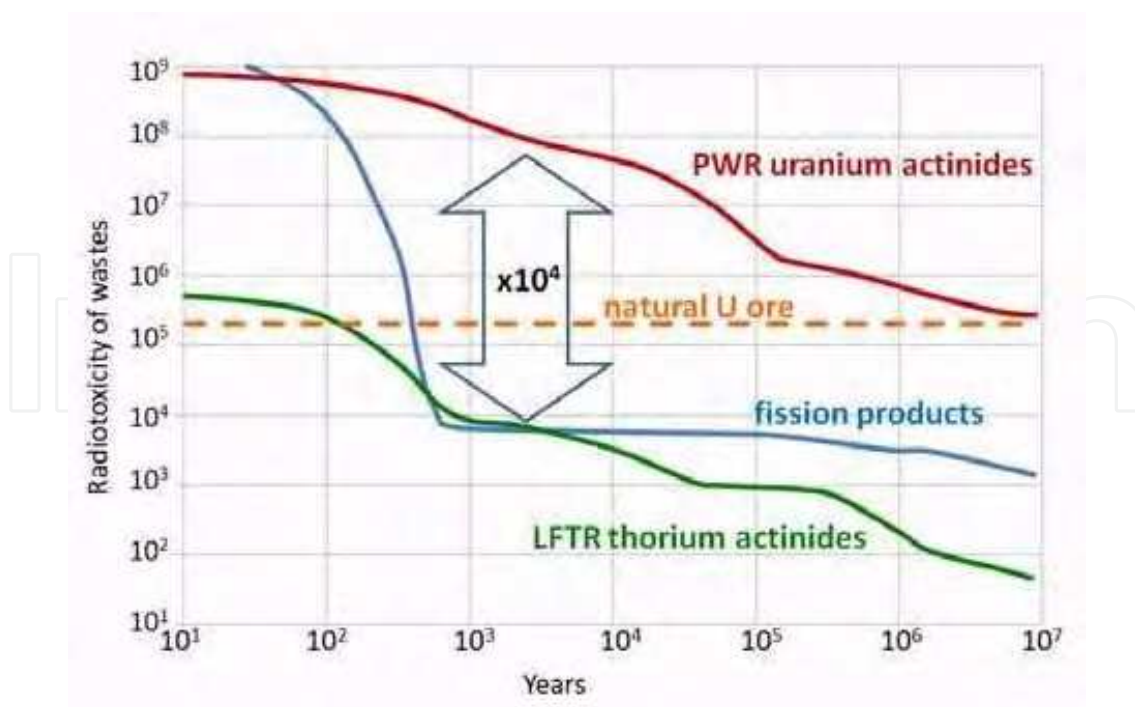
Fig. 12. Comparison of the radio toxicity of nuclear waste produced by a uranium-based PWR to that produced by a Liquid Fueled Thorium Reactor (LFTR) as a function of years of decay time. [Adapted from David S., et al., 2007].

6. Conclusion

One of the most promising philosophical and technical strategies for the world survival in this century has been presented. Although many more detailed design and optimization studies are needed and should proceed with international cooperation, we have to start from the very simple pilot-plant, miniFUJI, to rapidly demonstrate the rational technological integrity of THORIMS-NES and to make the initial step into the Thorium era.

We hope that our work will be valuable as a reply to the sincere wish of David E. Lilienthal [Lilienthal D. E., 1980] a most significant American/Human of the 20th century, given on the final sentence of his last book "Atomic Energy: A New Start": "What I have reflected upon and written about is not merely a new source of electrical energy, nor energy as an economic statistic. My theme has been our contemporary equivalent of the greatest of all moral and cultural concerns—fairness among men and the endless search for a pathway to peace."

For such purpose, "I have proposed that we make a new start toward a safer peaceful atom, using a technology that will not, as the present technology does, produce bomb material in the process of creating the peaceful atom." And he recommended to us that "We need to back away from our present nuclear state in order to find a better way, a route less hazardous to human health and to the peace of the world and its very survival."

One of the authors (K.F.) deeply benefited from the strong support of Bernal [Brown A., 2005] in his early scientific work on inorganic liquid structure chemistry as a base of this work. Bernal was also one of the scientists who were most concerned to achieve a "World without War" [Bernal J. D., 1958], and was the first to use the phrase "weapons of mass

destruction". On his birthday towards the end of his life he wrote: "I am sure that you share my hope that in the not too distant future science may come to be used for the benefit of all mankind"

7. Acknowledgements

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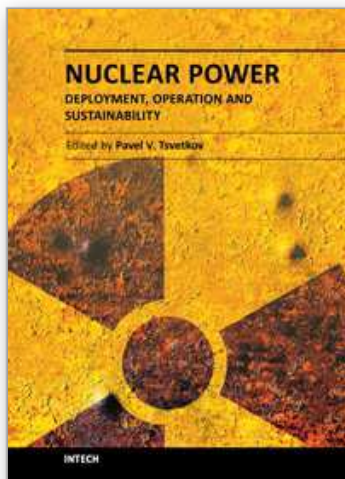
Nuclear Power - Deployment, Operation and Sustainability

Edited by Dr. Pavel Tsvetkov

We are fortunate to live in incredibly exciting and incredibly challenging time. Energy demands due to economic growth and increasing population must be satisfied in a sustainable manner assuring inherent safety, efficiency and no or minimized environmental impact. These considerations are among the reasons that lead to serious interest in deploying nuclear power as a sustainable energy source. At the same time, catastrophic earthquake and tsunami events in Japan resulted in the nuclear accident that forced us to rethink our approach to nuclear safety, design requirements and facilitated growing interests in advanced nuclear energy systems. This book is one in a series of books on nuclear power published by InTech. It consists of six major sections housing twenty chapters on topics from the key subject areas pertinent to successful development, deployment and operation of nuclear power systems worldwide. The book targets everyone as its potential readership groups - students, researchers and practitioners - who are interested to learn about nuclear power.

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