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Conceptual design of Indian molten salt breeder reactor

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Abstract. The third stage of Indian nuclear power programme envisages the use of thorium as the fertile material with ^{233}U, which would be obtained from the operation of Pu/Th-based fast reactors in the later part of the second stage. Thorium-based reactors have been designed in many configurations, from light water-cooled designs to high-temperature liquid metal-cooled options. Another option, which holds promise, is the molten salt-fuelled reactor, which can be configured to give significant breeding ratios. A crucial part for achieving reasonable breeding in such reactors is the need to reprocess the salt continuously, either online or in batch mode. India has recently started carrying out fundamental studies so as to arrive at a conceptual design of Indian molten salt breeder reactor (IMSBR). Presently, various design options and possibilities are being studied from the point of view of reactor physics and thermal hydraulic design. In parallel, fundamental studies on natural circulation and corrosion behaviour of various molten salts have also been initiated.

Keywords. Thorium; molten salt; breeder; molten salt breeder reactor.

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1. Introduction

India's unique, sequential three-stage nuclear power programme is aimed at optimum utilization of India's nuclear resource profiles of modest uranium and abundant thorium. The three stages are: (i) natural uranium-fuelled pressurized heavy water reactors (PHWRs), (ii) plutonium-fuelled fast breeder reactors (FBRs), including metallic fuelled ones and (iii) ^{233}U-fuelled systems. The spent fuel of one stage is used as a resource for the subsequent stages based on a closed fuel cycle. When the spent fuel is reprocessed to produce fuel for the next stage, it multiplies the energy potential of the fuel many-fold and greatly reduces the quantity of waste. It is thus an optimum solution for meeting the energy needs of a large country like India in a sustainable manner, securing its energy freedom in the long term. Schematic of Indian three-stage nuclear power programme is shown in figure 1.

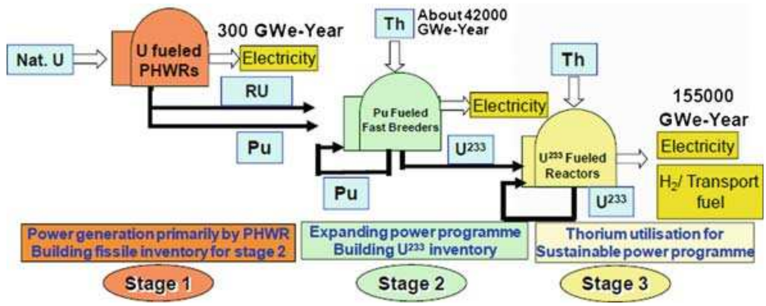

Figure 1. Three stages of Indian nuclear power programme.

The first stage, with many PHWRs in operation and many under construction and planning stages, has reached a state of commercial maturity. The first commercial FBR of second stage is at the advanced stage of construction. Technologies for the third-stage systems, based on thorium-based fuel, are under various stages of development. Under this programme, technologies for all aspects related to thorium fuel cycle have been developed on a laboratory scale. A thorium fuel-based 300 MWe AHWR is in the project implementation mode and its construction is likely to begin within the next few years. The design is based on Indian experiences and studies. As a long-term programme, and to effectively utilize thorium for the third stage of its nuclear power programme, India is working on the development of technologies for the molten salt breeder reactor (MSBR), which can breed fissile material effectively, and hence would provide a long-term sustainable solution. The large-scale deployment of thorium-based reactors is likely to take place in the second half of this century, and all technologies would be mastered before that and prototype reactors would be deployed much before the large-scale deployment stage to demonstrate the technologies involved and design optimization.

2. Relevance of molten salt breeder reactors

As mentioned earlier, efficient utilization of domestic thorium resources is one of the primary aims of the third stage of the Indian nuclear programme. In this respect, use of thorium in solid- or liquid-fuelled reactors can be considered. To maintain or expand the scale of the programme in the third stage in a sustainable manner, it is essential that the reactor systems (i.e., the reactor and its associated fuel cycle facilities) should have a breeding ratio of at least unity, after considering all the material losses. Hence, the reactor system in the third stage should be a breeder with a breeding ratio at least slightly more than unity. While it is possible to breed ^{233}U from thorium in reactors of conventional design (i.e. with solid fuel rod bundles), thorium if used in conventional reactors, suffers from one disadvantage. ^{233}Pa is an intermediate isotope in the chain of nuclear reactions (figure 2), which results in the conversion of ^{232}Th to ^{233}U. The conversion of ^{233}Pa to ^{233}U is possible by β decay. However, in a reactor core ^{233}Pa is always subjected to the presence of fission neutrons, which if absorbed, neither leads to the formation of ^{233}U, nor to fission, and hence is a parasitic loss. This is unavoidable in solid-fuelled reactors, because the fuel pins need to be exposed to certain burn-ups before being extracted for reprocessing. This can be avoided if the fuel is in fluid form, so that ^{233}Pa is separated

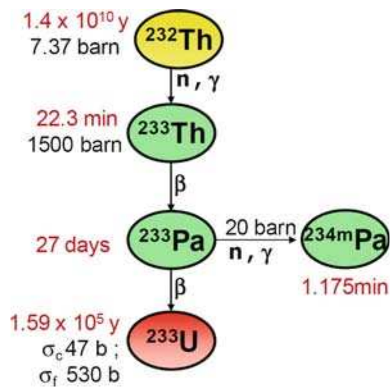

Figure 2. Conversion chain of ^{232}Th to ^{233}U.

out as soon as it is produced, and allowed to decay to ^{233}U out of core. A feasible way of doing this is the MSBR.

The fuel in MSBR is in the form of a continuously circulating molten salt, which contains the fissile and the fertile elements as its constituents. The critical configuration of the salt is achieved only when it enters the core. When the salt is in other parts of the primary circuit, it is not critical and hence does not undergo fission. Thus, if the salt is dumped into smaller dump tanks, subcriticality and thus safe reactor shutdown is automatically achieved. A crucial part for achieving reasonable breeding in such reactors is the need to reprocess the salt continuously. Depending on the design, reprocessing may be done either in an online mode, or in an offline batch mode. In an online mode, the reprocessing plant is located with the reactor and continuously reprocesses the fuel salt. On the other hand, in an offline mode, the reprocessing is done in a separate plant located nearby, with some portion of the fuel salt removed daily and the reprocessed salt added to the primary circuit. This constitutes a major technological challenge for these types of reactors.

3. Classification of MSBRs

Different options for MSBRs have been studied world over. All the concepts make use of a fuel salt and a fertile salt. The fuel salt has the fissionable nuclei in salt form along with other salt components. Similarly, the fertile salt has the fertile nuclei in salt form along with other salt components. In both cases, the salts are circulated by pumps/natural circulation through a cooling circuit. Depending on the choice, the salt may contain both fuel and fertile materials, or, these may be part of two different salts, circulating in separate loops.

3.1 *Single-fluid MSBR* [1]

In this concept (figure 3), the primary salt, containing fluorides of ^{233}U (as UF_4) and thorium (ThF_4), is circulated through the core by means of a pump and passed through intermediate heat exchangers (IHXs). The IHXs, pump and the core are located inside the containment; while an intermediate loop carries the heat from the primary circuit to the power generating circuit (e.g., supercritical steam cycle or supercritical CO_2-based Brayton

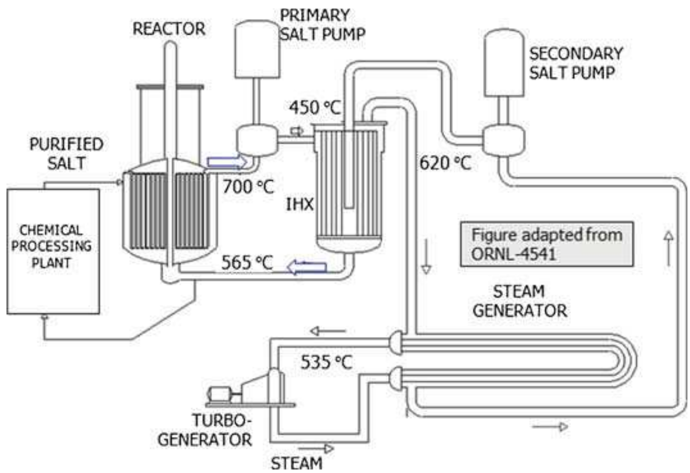

Figure 3. Schematic of the single-fluid type MSBR [1].

cycle). A bleed from the salt is taken to an online reprocessing plant, where most of the fission products are removed. The protactinium is removed, allowed to convert to ^{233}U and reintroduced to the primary loop. The entire salt volume is also subjected to bubbling by helium to remove the gaseous fission products (Xe, Kr) and also volatile noble metals. This concept is defined as a single fluid type. The aircraft reactor experiment (ARE) and molten salt reactor experiment (MSRE) of ORNL operated between 1960s and 1970s were reactors of this type, but without the online reprocessing plant. The MSBR concept developed in ORNL was a similar reactor concept.

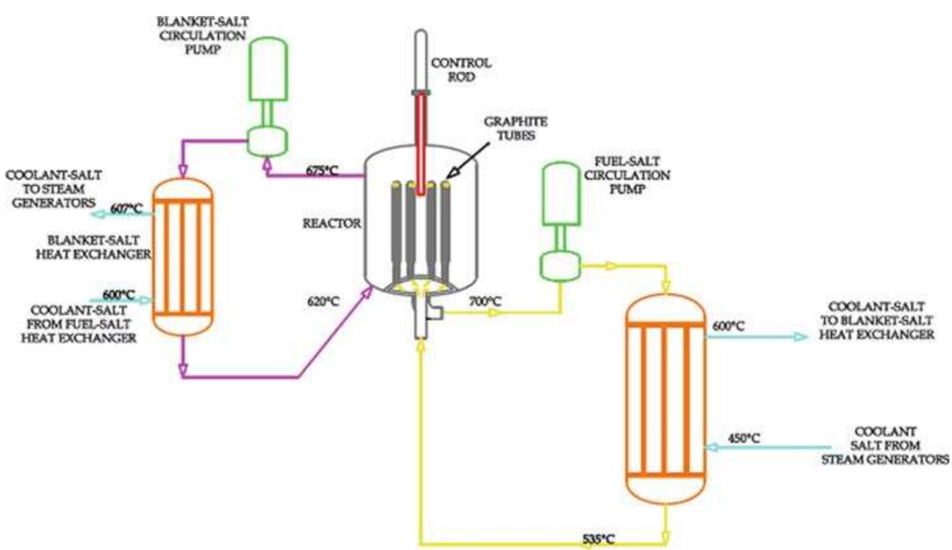

Figure 4. Schematic of the two-fluid type MSBR [2].

Table 1. Comparison of MSBRs with different neutron energy spectrum [3,4].

Thermal	Epithermal	Fast
Minimum fissile inventory ($<$1–1.5 t for 1 GWe)	Higher fissile inventory ($\sim$1.5–2 t for 1 GWe)	Maximum fissile inventory ($\sim$3.3 t for 1.3 GWe)
1. Limited graphite lifetime		
(a) four full power years (fpy) – for a power density of 22 kW/l	Very limited graphite lifetime ($<$4 y)	No graphite needed
(b) 15 fpy for a power density of 7.2 kW/l		
2. Needs 670 t of graphite/GWe		
Minimum breeding ratio (BR) $\sim$1.03	BR 1.03–1.1 possible	BR $\sim$ 1.1 possible
Needs continuous online reprocessing	Increased reprocessing time leads to poorer BR	Reprocessing could be done in batch mode
Possible to design with negative temperature coefficient of reactivity	Negative temperature coefficient of reactivity	Higher negative temperature coefficient of reactivity
1. Best fissile material economy		1. Less attractive fissile material economy
2. Requires online reprocessing and graphite reprocessing technology		2. Decoupling of reprocessing plant possible
		3. No graphite waste

3.2 *Two-fluid MSBR* [2]

Another concept of MSBR is shown in figure 4. Here the fuel salt, i.e., the salt containing UF_4 is distinct from the fertile salt, i.e., salt containing ThF_4. The fuel salt is circulated through the core via channels transparent to neutrons (e.g., graphite). The fertile salt circulates outside these channels. About 97% of the energy is generated inside the fuel salt, and the rest in the fertile salt. IHXs are provided for both the fuel and fertile salts. The coolant salt (in the intermediate loop) first passes through the IHX for the fuel salt (i.e., where the majority of the heat is generated) and then it is passed through the IHX of the fertile salt. Both the fuel and fertile salts are processed through online reprocessing systems. This concept is called the two-fluid type.

Both the single-fluid and the two-fluid types of MSBRs can be operated over a wide range of neutron energies, thus leading to thermal, epithermal and fast concepts. A comparison between them is shown in table 1 [3,4].

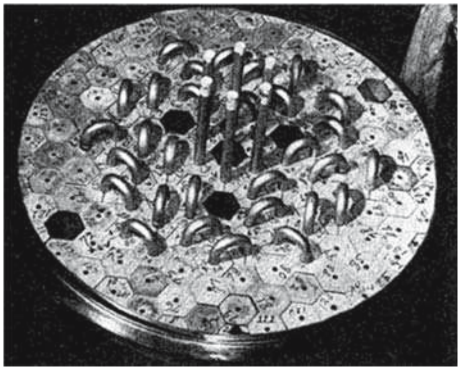

Figure 5. Core of ARE [5].

4. International experience and current efforts on MSBR

Few reactor systems were operated by ORNL in the past. New reactor systems are being designed in many countries. These are described briefly in the following paragraphs.

4.1 *Aircraft reactor experiment (ARE)* [5]

It was a small reactor built at Oak Ridge to investigate the use of molten fluoride fuels for aircraft propulsion reactors and particularly to study the nuclear stability of the circulating fuel system. The fuel salt for ARE (figure 5) was a mixture of NaF, ZrF_4 and UF_4, the moderator was BeO, and Inconel was used for piping. In 1954, the ARE was operated successfully for nine days at steady-state outlet temperatures ranging up to 860°C and at powers up to 2.5 MWt. No mechanical or chemical problems were encountered, and the reactor was found to be stable and self-regulating.

4.2 *Molten-salt reactor experiment (MSRE)* [6]

As ARE had been operated only for a short period, another reactor experiment was needed to investigate some of the technology for power reactors. The design of MSRE began in

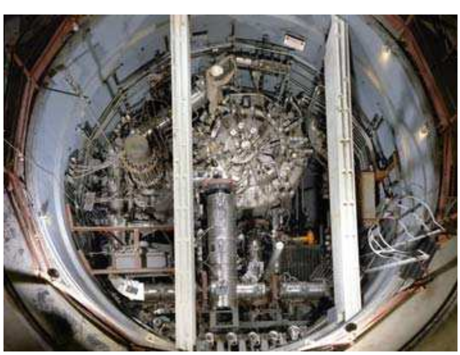

Figure 6. Core of MSRE [6].

1960. A single-fluid reactor was selected that resembled a converter in its engineering features, but the fuel salt did not contain thorium and thus was similar to the fuel salt for a two-fluid breeder. The MSRE (figure 6) fuel salt was a mixture of uranium, lithium-7, beryllium and zirconium fluorides. Graphite was chosen as the moderator material. All other parts of the system that were in contact with salt were made from the nickel-based alloy, Hastelloy-N, which was specially developed in the aircraft programme for use with molten fluorides. The maximum power was 8 MWt, and the heat was planned to be released into the atmosphere. Construction of the MSRE began in 1962, and the reactor was first made critical in 1965. Sustained operation at full power began in December 1966. Successful completion of a six-month run in March 1968 accomplished the first phase of operation during which the initial objectives were achieved. The molten fluoride fuel was used for many months at temperatures around 650°C without corrosive attack on the metal and graphite parts of the system. The reactor equipment operated reliably and the radioactive liquids and gases were contained safely. The fuel was completely stable and xenon was removed rapidly from the salt. When necessary, radioactive equipment was repaired or replaced in reasonable time and without overexposing maintenance personnel. The second phase of MSRE operation began in August 1968, when a small processing facility attached to the reactor was used to remove the original uranium by treating the fuel salt with fluorine gas. A charge of ^{233}U fuel was added to the same carrier salt, and on October 2, 1968 the MSRE was made critical on ^{233}U. Six days later the power was increased to 100 kW initiating the operation of the first reactor with ^{233}U fuel. During this time it was discovered that lithium fluoride and beryllium fluoride in a fuel salt can be separated from rare earths by vacuum distillation at temperatures near 1000°C. This was a significant discovery, as it provided an inexpensive, on-site method for recovering these valuable materials. As a consequence, the study effort looking at future reactors focussed on a two-fluid breeder in which the fuel salt would be fluorinated to recover uranium and distilled to separate the carrier salt from fission products. The blanket salt would be processed by fluorination alone, because few fission products would be generated in the blanket if the uranium concentration was kept low. The major disadvantage of this two-fluid system was that the graphite had to serve as a piping material in the core where it was exposed to very high neutron fluxes. At about the same time, the problems associated with long graphite exposure became evident; a chemical processing development occurred that greatly improved the prospect for a single-fluid breeder. To obtain good breeding performance in a single-fluid reactor, ^{233}Pa (27.4 day half-life) must be kept outside the core until it decays to ^{233}U. The processing development that showed promise of accomplishing this was a laboratory demonstration of the chemical steps in a liquid–liquid extraction process for removing protactinium and uranium from molten fluoride salts. The MSRE operated for 5 years. The salt was loaded in 1964 and nuclear operation ended in December 1969, and all the objectives of the experiment were achieved during this period. The MSRE ran successfully for 9000 h on both ^{233}U and ^{235}U. The reactor was shut down on schedule in 1969 as funds were needed for designing and building a bigger prototype MSR called the MSBR.

4.3 *Molten salt breeder reactor (MSBR)* [7]

The next stage planned at ORNL was to construct a MSBR for breeding ^{233}U from thorium. In 1972, ORNL proposed a major development programme for the construction and operation of a demonstration reactor called the molten salt breeder experiment. The programme was estimated to cost a total of \$350 million over a period of 11 years. However, another expensive development programme for LMFBR was already planned and therefore it was difficult to initiate another expensive development programme so the ORNL proposal was rejected. In January 1973, ORNL was directed to terminate the MSR development work. The programme was reinstated a year later, and in 1974 ORNL submitted a more elaborate proposal with suitably inflated costs calling for about \$720 million to be spent over an 11-year-period. This last proposal was also rejected, and in 1976 ORNL was again ordered to shut down the MSR programme for budgetary reasons. MSR work continued at a low budget. The ORNL experts came up with likely solutions to most of the problems raised at the MSRE. Others throughout the world worked on salt chemistry problems and developed many other designs, including some chloride salt fast breeder MSRs.

4.4 *Current efforts for MSRs*

Lately, there is a renewed interest worldwide for MSRs. In 2001, the MSR was chosen by the Generation IV forum as one of the six Gen-IV concepts. Another variant of this reactor was the molten salt-cooled reactors. These reactors have solid fuel in the form of TRISO (TRi-ISOtropic) particles (similar to the high-temperature gas-cooled reactors), but rather than pumping high-pressure helium coolant through, low-pressure liquid salt is either circulated naturally or pumped. The aim is to obtain high-temperature and high-burn-up benefits of the gas reactors without the risk of depressurization accidents. Besides India, these reactor concepts are currently being studied by the USA and many other countries (including China). There is ongoing work in MSRs around the world. The Europeans are working on MOSART [8], the Japanese have FUJI [9] and the Americans have focussed on the FHR [10]. Most recently, in 2011, the Chinese Academy of Science started a project in Shanghai to use substantial resources in developing MSR technologies.

5. Development programme in India for MSRs

An active programme to establish the chemical feasibility of a LiF–BeF_2–PuF_3–ThF_4-based MSBR was conducted in the beginning of 1970 in BARC. Lately, designs based on molten salts as coolant as well as designs based on fluid fuel fluoride salts are being worked out in BARC. The following paragraphs briefly describe these efforts:

5.1 *Previous work done in BARC on molten salt studies* [11–15]

The studies carried out during the 1970s in BARC included, among other things, preparation of pure ThF_4 and LiF_4 including the development of equipment, solubility of PuF_3 in LiF–BeF_2–ThF_4 mixtures, thermodynamics of U–Bi alloys and determination of vapour pressure of materials of interest in MSBR. The Radiochemistry Division of BARC collaborated with ORNL for investigating the solubility of PuF_3 in LiF–BeF_2–ThF_4. The following facilities were set up at the Old Radiological Labs, BARC:

(1) A two-module SS glove box with recirculatory argon atmosphere having inert gas purification system: Molecular sieves were used for the removal of moisture, uranium-

turnings heated to 750°C was used for the removal of oxygen from argon and titanium sponge heated to 900°C was used for the removal of oxygen from hydrogen.

(2) Facility for purification of salts by hydrofluorination: As the salt contained beryllium fluoride; it was set up inside a glovebox. Continuous monitoring of the atmosphere was carried out by Health Physics Group of BARC. The beryllium concentration either in the atmosphere or on the working surfaces did not exceed or reach the maximum tolerance limit.
(3) Hydrogen fluoride manifold: It was constructed mainly of nickel tubes, with silver-brazed joints and teflon gaskets. The hydrogen fluoride waste was released to a fume hood after passing through towers packed with sodium fluoride.
(4) Facility for hydrofluorination of CeO_2 and PuO_2 to CeF_3 and PuF_3, respectively.
(5) Fabrication of Hastelloy N or C, nickel vessels.
(6) Design and fabrication of a molten salt sampler.
(7) Hydrogen annealing furnace to anneal sampler tubes.
(8) Fabrication of shielding materials for gamma and neutrons around the solubility apparatus.

The results of the investigation have been widely published, and appreciated at ORNL.

5.2 *Major design goals of Indian molten salt breeder reactor (IMSBR)*

Efficient utilization of thorium resources is essential in the third stage of Indian nuclear power programme. Results published in [16] suggest that considering an out-of-core time of three years, the cycle fissile inventory required for setting up one GWe FBR with metallic fuel consisting of ^{233}U and thorium is much higher compared to the ^{233}U inventory required for the MSBR, while the breeding ratio for both are comparable. Thus, MSBR seems to be the better option for the third stage from the point of view of fuel economy. Following are the design goals, which have been proposed for the IMSBR:

(1) Electrical power of the reactor: 850 MWe.
(2) Reactor design is based on ^{233}U/ Th fuel cycle.
(3) The design should avoid use of graphite/carbon-based moderators; hence, the design should be preferably based on fast/epithermal neutron energy spectrum.
(4) Breeding ratio should be such that a self-sustaining operation of the reactor is achieved after accounting for losses in reprocessing and associated fuel handling.
(5) High-efficiency power conversion system ($\eta \geq 45\%$).
    (a) Supercritical carbon dioxide-based Brayton cycle is the preferred option.
    (b) Ultrasupercritical steam cycle is an alternative option.
(6) Aim to have air-cooled condenser. Design to incorporate long-term cooling even in the case of prolonged station black-out (SBO).
(7) Safety comparable or better than the currently designed advanced reactors.
(8) Provision for passive dumping into passively cooled criticality safe dump tanks under postulated accident conditions.
(9) Strong negative temperature and void coefficients.
(10) The removal of decay heat by natural circulation.
(11) The reactor core outlet temperature $\geq$800°C.
(12) Low pressure system with low salt velocities to avoid issues of erosion.

(13) Minimization of initial requirement of fissile material inventory.
(14) Decoupling the reactor from the reprocessing plant leading to simplified reprocessing: Aim to achieve batch mode reprocessing and minimization of the amount of salt to be reprocessed each year.
(15) Reduced chemical toxicity, hence avoiding salts based on beryllium.
(16) Economics: Use of salts based on easy availability. Preferable to avoid lithium-based salts.
(17) Metallic structural material: High creep strength alloys compatible to selected salts. Preferable to have 100 years life. This will require development of special alloys for use at high temperatures and joining techniques to minimize corrosion issues, especially those prevalent in the heat-affected zone.
(18) Accessibility of all major internal components for in-service inspection and on-site repair.
(19) A future goal of fuel cycle could be the incineration of minor actinides.
(20) A future goal could also be hydrogen production or preferably a co-generation system, where a part of the thermal energy is utilized for hydrogen and a part for electricity generation. Desalination to obtain potable water from the waste heat of thermodynamic cycle is also an important goal.

Table 2. Thermophysical properties of the fuel salt as a function of its temperature [17].

Quantity		Units	Expression T (K)	Validity range (K)
Specific heat	C_p	J/kg/K	$-1111 + 2.78 \times T$	867–1073 [17,18]
Thermal conductivity	K	W/m/K	$0.928 + 8.40 \times 10^{-5} \times T$	891–1020 [18]
Density	ρ	kg/m^3	$4983.56 - 0.882 \times T$	893–1123 [18]
Dynamic viscosity	μ	Pa·s	$\rho \times 5.55 \times 10^{-8} \times e^{3689/T}$	898–1119 [18]

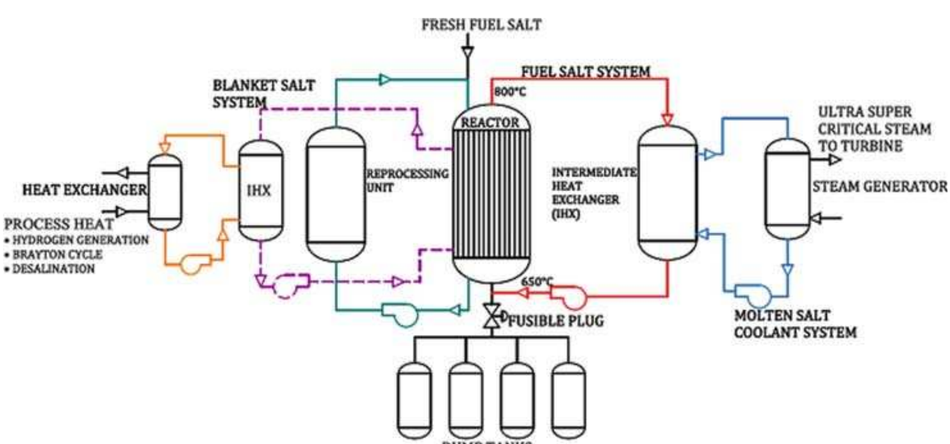

Figure 7. Preliminary lay-out of the proposed loop-type Indian MSBR.

Based on these goals, two versions of IMSBR are currently being designed. The first design is based on the loop-type concept and the second one on the pool-type concept. The blanket salt selected for the reactor is LiF–ThF_4, with appropriate amount of ^{233}U added as UF_4 as per the reactor physics requirement. The themophysical properties of this salt are given in table 2. After successfully completing their thermal hydraulics designs, reactor physics designs for both are in progress. In parallel, many R&D areas have been identified where significant technology developments are needed. These are shown in subsequent paragraphs. A typical schematic of loop-type IMSBR is shown in figure 7. To demonstrate the technologies, a small power prototype MSBR would be constructed and operated.

6. Major R&D areas for Indian MSBR [19]

The MSBR technology is quite different from the currently existing reactor systems in India. Hence, design and development-related activities are being initiated. A partial list which has been identified for the initial development, includes the following:

(1) Fuel salt studies

- (a) Preparation, characterization and purification of salts.
- (b) Corrosion studies and corrosion inhibition (redox control).
- (c) Determination of the behaviour of fission products.
- (d) Studies on thermal hydraulics of molten salts.
- (e) Development of techniques for the online electrochemical and spectrophotometric analyses of salts.

(2) Reactor physics

- (a) Development of closely-coupled neutron transport and CFD codes with capability to account for online reprocessing.
- (b) Parametric analysis by considering various reprocessing rates.
- (c) Decay loads.
- (d) Shielding requirements.
- (e) Optimizing lithium enrichment requirement.

(3) Reprocessing scheme

- (a) Studies and development of online reprocessing system.
- (b) Xe and Kr removal by helium bubbles.
- (c) Fluorination studies.
- (d) Reductive extraction.
- (e) Oxide precipitation.
- (f) Electroseparation.
- (g) Salt reconstitution.

(4) Structural materials

- (a) Material compatibility studies for the core and structural materials.
- (b) Development of materials compatible with fuel salts (e.g., Ni–W–Cr, etc.).

(c) Metal forming and joining.
(d) Carbon–carbon composites and their coatings.

(5) Component development and power cycle
(a) Development of pumps, valves, flow metre, off-gas system, dump tanks, IHX, steam generator, etc.
(b) Supercritical CO_2-based Brayton cycle/supercritical or ultrasupercritical steam cycle.
(c) Development of instrumentation and sensors.

(6) Evolution of safety philosophy

Developments in many of these areas have already been initiated. Some of them are listed in the following paragraphs:

(1) Molten salt natural circulation studies: Computational analytical (figure 8) studies as well as experimental studies (figure 9) have been carried out to evaluate natural circulation capabilities of molten salts. Molten salt natural circulation loop (MSNCL) [20] has been set up for this purpose and experiments are being conducted with FLiNaK salt. Figure 8 compares the steady-state natural circulation mass flow rate at different powers for various salts and compared with water, which shows that except molten fuel and blanket salts, all other salts have better natural circulation capability than water. Figure 9 compares the steady-state correlation for natural circulation derived in [21] with the experimental data. It was observed that the correlation overpredicts the experimental results by 28%. Although adequate thermal insulation was provided on all the piping components, the heat losses from them could not be prevented. Hence, the deviation of correlation from the experimental results may be attributed to heat losses.

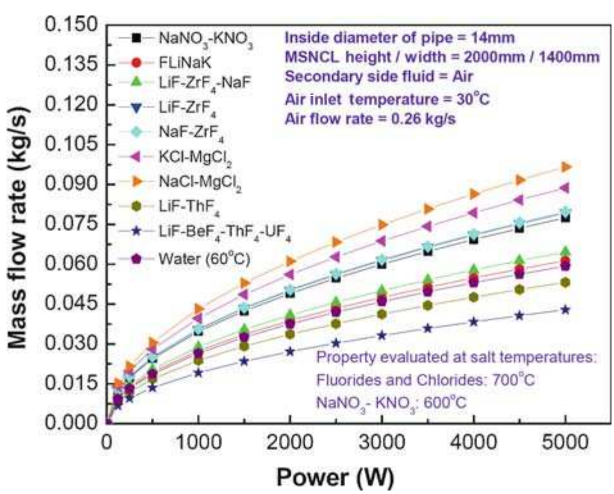

Figure 8. Comparison of natural circulation capabilities of various molten salts with water.

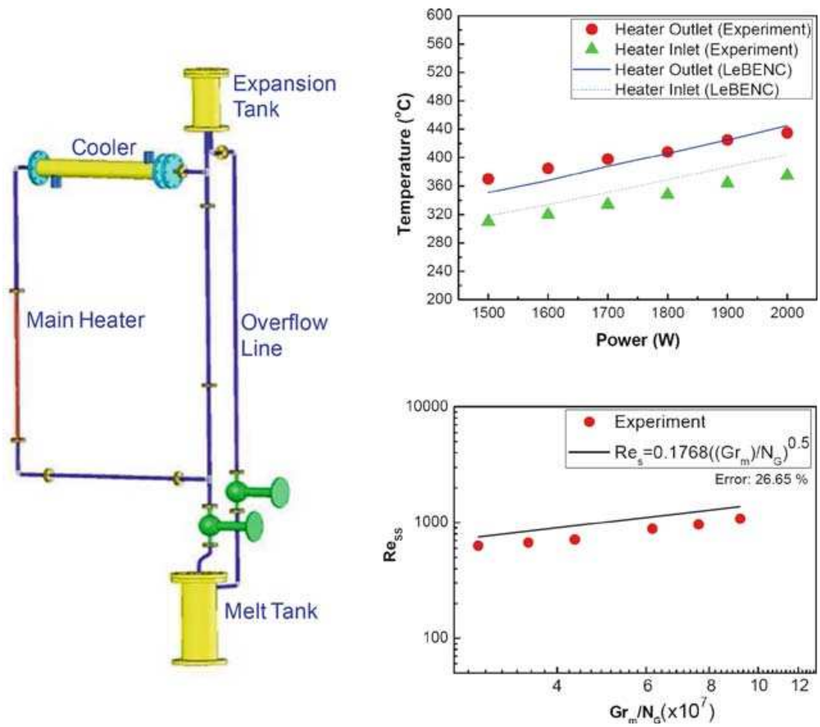

Figure 9. Schematic of MSNCL and comparison of measured and predicted results.

Additionally, theoretical analysis for steady-state cases has also been carried out for different powers of main heater using in-house developed natural circulation code, LBENC (lead bismuth eutectic natural circulation), originally developed for modelling natural circulation in liquid metals. The results (figure 9) show the comparison of main heater inlet and outlet temperatures at different powers calculated by LBENC with that of the experimental data and it is evident that the

Figure 10. Schematic of MOSCOT.

Table 3. Corrosion rates of different materials in 'mpy' under static condition (molten salt used was a eutectic mixture of LiF–NaF–KF).

Temp. (°C)	Inconel			Incoloy
	600	617	625	800
550	6.2	10.0	–	17.4
600	12.2	18.2	6.1	30.1
650	25.9	22.7	5.0	31.2
700	25.4	33.9	71.3	45.4
750	15.8	97.9	127.6	33.2

theoretical predictions of LBENC are in good agreement with the experimental data.

(2) Molten salt corrosion test facility (MOSCOT) [20] has been set up (figure 10) and corrosion experiments have been conducted on many structural materials (table 3).

7. Inherent safety features of MSBR

MSBR has attractive features, which make it inherently safe. Some of the major ones are listed here:

(1) Reduced excess reactivity – fissile salt is added as required.
(2) No need for xenon override margins. Xe and Kr are continuously removed, no release of gaseous fission products during postulated accidents.
(3) Reactors have full passive safety. Under accident conditions the fuel is automatically drained into passively cooled and criticality safe storage tanks.
(4) Low pressure system – reduced stress on the vessel and the piping.
(5) Leakage leads to dispersion and solidification.
(6) No containment pressurization.
(7) Probability of severe accidents leading to large-scale radioactivity release are reduced.
(8) No burn-up limits dictating fuel integrity.
(9) Passive heat transfer devices to prevent salt from overheating.
(10) Negative fuel salt reactivity coefficient – density of salt decreases with increase in temperature.
(11) 'Fuel melt down' and hydrogen issues are eliminated.
(12) Reduced decay heat and source terms – FP and TRU are removed continuously.
(13) No violent interaction of molten fluorides with water or air.
(14) Transparent liquid – allows optical windows for in-service inspection.

8. Summary

MSBR provides significant advantages and is an attractive option to achieve a self-sustainable ^{233}U/Th cycle for long-term energy security. Initial studies for MSBR development have been initiated. In parallel, conceptual designs of the Indian MSBR are being carried out. Experience in dealing with molten salts of nuclear materials exists in various divisions in BARC e.g. Uranium Extraction Division (UED), Chemistry Division (CD), Chemical Technology Division (ChTD), Product Development Division (PDD), Process Development Division (PSDD). Analytical and experimental studies have been initiated in many divisions of BARC. It is aimed to arrive at MSBR design satisfying the objectives of advanced reactors such as enhanced safety, sustainability and economic competitiveness.

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