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Hongbin Zhang  
Haihua Zhao  
Vincent Mousseau  
Ronaldo Szilard

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# Design Considerations for Economically Competitive Sodium Cooled Fast Reactors

Hongbin Zhang, Haihua Zhao, Vincent Mousseau and Ronaldo Szilard  
Idaho National Laboratory, P.O. Box 1625, Idaho Falls, ID 83415-3870, USA  
Tel: +1 (208)526-9511, Fax: +1 (208)526-0528, Email: [Hongbin.Zhang@inl.gov](mailto:Hongbin.Zhang@inl.gov)

**Abstract** – *The technological viability of sodium cooled fast reactors (SFR) has been established by various experimental and prototype (demonstration) reactors such as EBR-II, FFTF, Phénix, JOYO, BN-600 etc. However, the economic competitiveness of SFR has not been proven yet. The perceived high cost premium of SFRs over LWRs has been the primary impediment to the commercial expansion of SFR technologies. In this paper, cost reduction options are discussed for advanced SFR designs. These include a hybrid loop-pool design to optimize the primary system, multiple reheat and intercooling helium Brayton cycle for the power conversion system and the potential for suppression of intermediate heat transport system. The design options for the fully passive decay heat removal systems are also thoroughly examined. These include direct reactor auxiliary cooling system (DRACS), reactor vessel auxiliary cooling system (RVACS) and the newly proposed pool reactor auxiliary cooling system (PRACS) in the context of the hybrid loop-pool design.*

## I. INTRODUCTION

The sodium cooled fast reactor (SFR) has been studied since the early period of nuclear energy development more than 50 years ago. As a matter of fact the first nuclear reactor to generate electricity in the world is a fast reactor – Experimental Breeder Reactor I (EBR-I). It was cooled by sodium-potassium (NaK) and was started in 1951 at Idaho National Laboratory. The original primary objective of developing fast reactor was to breed plutonium fuel to maximize uranium resource utilization. SFR has been chosen as one of the six concepts for Generation IV (Gen-IV) reactors. Recently SFRs have been proposed to be used as actinide burners to close the nuclear fuel cycle to reduce nuclear waste management burdens per unit of energy production [1]. So far there had been 20 fast reactors gone into operation. Four of them are still operating (Phénix, JOYO, BN-600, FBTR) [2]. Currently, there are three SFRs under construction (CEFR, FPBR and BN-800). With the exception of SuperPhénix, all other SFRs are experimental reactors or prototype (demonstration) reactors. The technology viability of SFR has been demonstrated by these experimental and demonstration reactors. However, the economic competitiveness of SFR has not been proven yet. The perceived higher cost premium of SFR compared with LWRs impeded its commercial deployment [3]. The higher cost comes from the higher plant construction overnight

cost as well as from the low capacity factors. The operating experiences from the demonstration scale (prototype) SFRs such as Dounreay prototype fast reactor (PFR), BN-350, BN-600, Phénix and MONJU and one commercial scale SFR – SuperPhénix have shown fairly low capacity factors. The best record is around 75% while LWRs in the U.S. are above 90%.

The higher capital cost and low capacity factor of SFR resulted from the unique challenges associated with liquid sodium. Sodium belongs to the alkali metal family. It is inexpensive with excellent thermal conductivity and is compatible with metallic materials. It does not corrode structure materials such as stainless steel and is compatible with the metal fuel. Unlike the water coolant which must be pressurized at 60-150 times normal atmospheric pressure in a typical commercial LWR, sodium can operate at near normal atmospheric pressure because of its high boiling point (Table I). However, the thermophysical and chemical properties of sodium create unique challenges for use as coolants for fast reactors: (1). Sodium is highly reactive with water, air, CO<sub>2</sub> and concrete etc. Extra prevention, detection and mitigation measures have to be taken to ensure the safety of SFRs. All the extra measures contribute to the higher capital cost of constructing a SFR. For example, to mitigate the consequences of sodium water reactions, SFRs normally have three heat transport systems – the primary, the intermediate and the power conversion

system compared one or two heat transport systems for LWRs. BWR uses direct cycle while PWR has two cycles. (2). Liquid sodium is opaque. This presents difficulty to perform in-service inspection and repair. If debris accidentally got into the core, it is difficult to find them. (3). The volumetric heat capacity,  $\rho c_p$ , of sodium is relatively

low as shown in Table I [4]. It is only one fourth of that of water. SFRs normally require more number of heat transport loops compared to LWRs of the same power level. (4). Positive sodium void reactivity coefficient. To avoid sodium boiling, reactor temperature has to stay well below the sodium boiling point.

TABLE I  
 Comparison of sodium coolant thermophysical properties with alternative coolants and materials (Reference 4)  
 (approximate values at 700°C and 1 atm, except 290°C for water)

	Sodium	Lead	Helium (7.5 MPa)	Water (7.5 MPa)	Graphite
Melting Point, °C	97.8	328	—	0	—
Boiling Point, °C	883	1,750	—	290	—
Density ( $\rho$ ), kg/m <sup>3</sup>	790	10,540	3.8	732	1,700
Specific volume heat capacity ( $\rho c_p$ ), kJ/m <sup>3</sup> C	1,000	1,700	20	4,040	3,230
Thermal Conductivity (k), W/m <sup>2</sup> C	62.	16.	0.29	0.56	200.

The difficulties experienced with operating Superphénix and prototype SFRs and the higher cost of SFRs indicate that the economics and safety of SFRs need much further improvement. To attract large-scale investment in SFR technologies, it is imperative for SFRs to achieve economic competitiveness as compared to advanced LWRs. Therefore, the SFR technology needs an overhaul and one major activity in the SFR technology development is to reduce its capital cost sufficiently to be commercially deployable [4]. The demonstration of the economic competitiveness of SFRs requires thorough investigation of all components of the power generation cost, i.e. plant capital cost, fuel cycle cost, operation and maintenance costs, etc. We focus more on the plant capital cost aspect in this paper.

The next generation sodium cooled fast reactor will need to meet three design goals: (1) Improvement of safety, the risks of sodium fire and sodium-water reaction must be eliminated or minimized. (2). Improvement of economic competitiveness by design simplification and improvement of thermal efficiency to attract large-scale investment to commercialize SFR technology and (3). Improvement of in-service inspection and repair to improve the plant availability.

Novel technologies must be developed as cost reduction options that will reduce plant size and construction schedule. Advances in R&D for advanced SFR fuel and structural materials will provide key opportunities to improve SFR economics. However, major breakthroughs in advanced fuel and materials technology

will be long term. Meanwhile, even with available fuels and materials, other new opportunities are emerging to further improve SFR economics [5]. These include: 1). a newly proposed hybrid loop-pool reactor design to optimize the primary system of SFRs to improve plants' economics, safety, and reliability and 2). a multiple reheat and intercooling helium Brayton cycle to improve plants' thermal efficiency and reduce safety related overnight and operation costs [5]. These two innovations will be briefly discussed in the Section II and readers are referred to reference [5] for detailed discussions.

Passive design provides the means to simplify the design and is a basic requirement to achieve economic competitiveness. It relies on natural processes to hold power production in balance with heat removal. Passive safety system can extinguish neutron reactions under abnormal conditions and to remove decay heat to the environment without operator intervention. Auxiliary decay heat removal system is required independent of the non-safety grade equipment and components in the balance-of-plant. This principle was pioneered in EBR-II and has been refined and improved in the later designs such as PRISM, ALMR within the context of the Integral Fast Reactor (IFR) program [2]. In these designs, auxiliary decay heat removal systems were added and decay heat removal is driven by natural convection and is always in operation. Direct Reactor Auxiliary Cooling System (DRACS) was used for EBR-II and Reactor Vessel Auxiliary Cooling System (RVACS) was used for PRISM and ALMR. Passive safety was demonstrated with two landmark tests carried out in EBR-II in 1986. The first test

was a loss of flow without scram from 100% power, and the second was a loss of heat sink without scram from 100% power. Both tests demonstrated that natural processes such as negative reactivity feedback and natural convection of the primary sodium coolant were able to shut down EBR-II and maintain coolability without activation of the active safety system.

Three design options exist for decay heat removal system. DRACS and RVACS are the existing decay heat removal systems. RVACS is integrated with the reactor vessel and is a simpler system than DRACS. However RVACS's decay heat removal capability is limited by the reactor vessel size and thus has the scalability issue. DRACS is a slow response system that can remove decay heat from the hot or cold pool to the environment. However it does not necessarily ensure adequate sodium coolant flow in the core during transients such as loss of flow. The newly proposed Pool Reactor Auxiliary Cooling System (PRACS) in the context of the hybrid loop-pool design ensures adequate flow in the core during transients and the decay heat can be effectively transferred from the core to the cold pool and in turn be rejected to the ultimate heat sink by DRACS.

This paper reviews the design options for advanced SFRs as well as the decay heat removal. Section II provides summarized discussion on the design options of the primary system, power conversion system and intermediate heat transport system. Section III thoroughly examines the decay heat removal options and Section IV provides the option for an advanced SFR design.

## II. REACTOR DESIGN OPTIONS

### II.1. Primary System

The existing SFRs have two types of designs – loop type and pool type. In the loop type designs, such as JOYO [6], MONJU [7], and JSFR [8,9] in Japan and FFTF [2] in the USA, the primary coolant is circulated through IHXs and pumps external to the reactor tank. The major advantages of the loop design include compactness and easy in-service inspection and maintenance. The disadvantages are higher possibility of sodium leakage and less thermal inertia than a pool design [9]. In pool type designs such as EBR-II, PRISM, ALMR and S-PRISM (USA), BN-600 (Russia), Phénix and Superphénix (France), PFBR (India) and EFR (EU) [2], the reactor core, primary pumps, IHXs and direct reactor auxiliary cooling system (DRACS, if used) heat exchangers (DHX) all are immersed in a pool of sodium coolant within the reactor vessel, making a loss of primary coolant accident extremely unlikely. However, the pool type design makes the primary system large and the in-service inspection difficult.

In a typical pool SFR design, the hot sodium at core outlet temperature in the hot pool is separated from the cold sodium at the core inlet temperature in the cold pool by a single integrated structure called a redan. In order to accommodate different and often contradictory design requirements under normal and transient/accident operating conditions, design restrictions have to be imposed. Consequently, a conventional pool type design presents difficulties to simultaneously obtain optimal economical and safety benefits. For example, in order to achieve cost reduction, the primary system components such as IHXs and primary pumps should be designed as compact as allowable. However, during loss of forced flow cooling (LOFC) transients, small flow resistance in the IHX is essential to establish adequate natural circulation flow to remove heat from the reactor core to the cold pool. Due to this requirement, only traditional tube and shell IHXs with low flow resistance can be used. More compact heat exchangers with higher flow resistance are difficult to be used as IHXs. Therefore, it is difficult to further reduce the size of the sodium tank and consequently the size of the containment for pool type SFRs without compromising the safety of the reactor. A feasible solution is to add an independent passive safety system such that the functions of the primary circuit for normal power operations and the passive safety system for abnormal operating conditions are decoupled. The newly proposed hybrid loop-pool design [5, 10] provides an opportunity for such an improvement.

The hybrid loop-pool design is a loop-in-a-tank concept with a closed primary loop design immersed in a separate pool of cold sodium and an independent passive safety system added. The design takes advantage of the easy in-service inspection and compactness of loop designs and the inherent safety of pool designs. Primary loops are formed by connecting hot sodium at reactor outlet plenum (hot pool), IHX, primary pumps and reactor inlet plenum with pipes. The primary loops are immersed in the cold pool (buffer pool), which provides an extra safety barrier and large thermal inertia. During accidents, the modular Pool Reactor Auxiliary Cooling System (PRACS) transfers heat from the reactor core to the cold pool. Fig. 1 illustrates the schematic comparison of the conventional pool design and the new hybrid loop-pool design configurations. Under normal operations for the hybrid loop-pool design, the primary loops operate in forced circulation cooling driven by primary pumps that could be located either in the reactor hot leg or in the cold leg, depending on the pressure loss through the IHX. The primary pumps take suction from the hot pool at near atmospheric pressure and drive the hot sodium through IHXs back to the reactor core inlet plenum at the bottom of the reactor. The IHXs could be either traditional tube-shell heat exchangers or modular, compact heat exchangers such as printed circuit heat exchangers where the coolants flow in millimeter sized semi-circular

flow channels. Compact heat exchangers have much higher power density (5 to 10 times higher) and are much smaller than the tube-shell type heat exchangers. A small bypass with reactor inlet temperature flows upward through the PRACS. This bypass flow adds a small amount of heat to the cold pool, depending on the temperature difference between the reactor inlet and buffer pool. This added heat is mainly removed by the DRACS to the environment. DRACS is a natural circulation system with a set of modular DHX immersed in the sodium buffer pool. The primary systems and the cold pool are thermally coupled by the PRACS, which is composed of PRACS heat exchangers (PHX), fluidic diodes and connecting pipes. Fluidic diodes are simple, passive devices that provide large flow resistance in one direction and small flow resistance in reverse direction. A fluidic diode generates an irreversible loss of kinetic energy by creating a strong vortex flow in one direction, while flow in the opposite direction does not have this effect. Therefore, the fluidic diodes restrict upward leakage flow through PRACS during forced circulation and provide low resistance during buoyancy-driven natural circulation flow in the reverse (downward) direction. Fluidic diodes have also been used in Japan's advanced loop type SFR design [8, 9]. PHX modules use conventional tube bundles to reduce flow resistance and are in baffles to enhance natural circulation as illustrated in Fig. 1.

Under the abnormal operating conditions such as LOFC transient with or without scram, reduced heat transfer in the reactor core causes the core temperatures to rise. Natural circulation establishes quickly and flow reversal happens through the PRACS loops. If the intermediate loop heat removal is continued, natural circulation continues to remove decay heat through the IHX modules. If the secondary heat sink is lost, decay heat removal to the buffer tank mainly occurs through the PHX modules. Heat rejection from the buffer sodium tank to the environment occurs dominantly through DRACS.

Adding PRACS will not incur much extra cost. It is estimated that the PRACS will cost less than 1% of the primary loops. However, the potential cost savings with this design by utilizing more compact IHXs and primary pumps will more than compensate for the extra cost of PRACS. Additionally, the modular design of PRACS allows it to be easily scaled up or down according to the reactor power output level. A similar design was used for the liquid salt cooled advanced high temperature reactor (AHTR) system developed by UC Berkeley [11]. The most recent concept design for fast spectrum molten salt reactors also employed similar configuration [12].

This hybrid loop-pool design fully decouples reactor inlet temperature and the cold pool temperature, (as does

the primary heat transfer system and the passive safety system) and physically decouples the hot pool sodium and cold pool (buffer pool) fluid. The decoupling provides much more design flexibility to optimize the design. For example, the cold pool temperature can be set at a lower value than that for a conventional pool design, which consequently increases the thermal inertia. Both core inlet and outlet temperatures could be increased which yields higher thermal efficiency for electricity generation. PRACS may provide better natural circulation ability to keep the peak cladding temperature staying below its limit during loss of forced circulation transients. Other benefits include cost reduction and easier in-service inspection and maintenance. Readers are referred to reference [5] for detailed discussions. Section III also has more discussions on the benefits of the PRACS.

It should be pointed out that with the hybrid loop-pool design, forced and natural circulation operations more closely resemble that in the advanced loop design such as JSFR [8,9] rather than a conventional pool-type sodium cooled fast reactor.

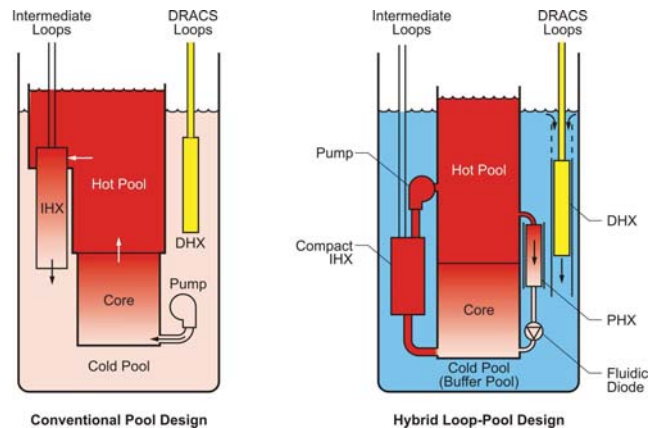


Fig. 1. Comparison of conventional pool design and innovative hybrid loop-pool design for SFR under loss of forced flow condition.

## II.2. Power Conversion System (PCS)

One of the biggest problems that inhibited the commercial deployment of SFR is the sodium water reaction. This is resulted from the steam Rankine cycle that has been employed as the only power conversion system. Steam generators are subject to high pressure difference, high operation temperature and large heat transfer area, the potential for leakage is high. Extra safety systems have to be provided to detect, mitigate and rapidly terminate a steam generator leak to protect against the potential investment loss and to protect the primary coolant boundary at the IHX. For example, in the General Electric's S-PRISM steam generator design [13], three sub-

systems are added to assure that sodium leaks in the steam generator are safely accommodated with a minimum damage to the steam generator and intermediate heat transport systems. These are: an advanced integrated leak detection system (ILDS), a fast acting steam side isolation and blow down system and a sodium water reaction pressure relief system (SWRPRS). Sodium-water reactions have happened in the steam generators of all the prototype (demonstration) SFRs. Phénix had five incidences of sodium-water reaction between 1982 and 2003, BN-600 experienced 12 leaks in steam generators. BN-350 had a sodium water reaction incident in 1975 from a steam generator leak that led to a sodium fire lasted for two hours. PFR experienced 37 leaks in steam generator units in the period 1974 to 1984 [2]. Therefore a critical component of the SFR technology development is to eliminate the sodium water reaction. Double walled tubes steam generators provide the option to minimize sodium water reaction. The double walled tubes steam generator was originally used in EBR-II and successfully operated for three decades. The JSFR design also adopts this concept. However two reasons make this concept less desirable. The first one is that even though the double walled tubes steam generator worked well for a small test reactor like EBR-II, there is no guarantee that double walled tubes will be leak-proof for large power steam generators for commercial scale reactors. The heat transfer surface area for large power steam generators is much larger than that for an experimental reactor like EBR-II. The probability of sodium leak increases in proportion to the heat transfer surface area. The sodium leak detection and mitigation measures of sodium water reaction are still necessary. The second reason is that the double walled tubes are expensive to fabricate, which increases the capital cost of the SFRs.

Due to the safety concerns over steam cycles, SFR researchers have been exploring alternative power conversion cycles. Supercritical CO<sub>2</sub> recompression Brayton cycles [14,15] were recently proposed as the power conversion system for the latest ABTR design [1]. SCO<sub>2</sub> Brayton cycle has high cycle thermal efficiency using the gas turbine. As a cost reduction option, the system can be made compact. However, just like water, CO<sub>2</sub> is also incompatible with sodium. Recent studies showed that CO<sub>2</sub> reacts with sodium violently and the energetic reaction generates higher temperature than sodium-water reaction [16,17]. If large amount of sodium reacts with CO<sub>2</sub>, the released energy may threaten the integrity of the primary loop. Due to the severe nature of the Na-CO<sub>2</sub> reaction, it can be anticipated that expensive and complex safety systems will be required to detect and mitigate the Na-CO<sub>2</sub> reaction, which would significantly increase the capital cost. Consequently, the compatibility issue of CO<sub>2</sub> and sodium makes SCO<sub>2</sub> cycle a less desirable choice.

A basic requirement for an advanced SFR design is to choose a power conversion system that has chemically compatible fluid with sodium. Advanced closed gas Brayton cycles with the inert gas helium provide an opportunity to address this issue. Helium Brayton cycles are chosen for the power conversion cycle in Gen-IV High Temperature Gas Cooled Reactor (HTGR) designs such as the General Atomics Gas Turbine – Modular Helium Reactor (GT-MHR) [18] and Pebble Bed Modular Reactor (PBMR) [19]. The net thermal efficiency is quite high with the high turbine inlet temperature for HTGR. However for the temperature range of SFR, the conventional closed helium Brayton cycle has significantly lower efficiency than steam cycles and SCO<sub>2</sub> cycle. ABTR design selected SCO<sub>2</sub> cycle over closed helium cycle because of its lower thermal efficiency [1]. The thermal efficiency of the Brayton cycle can be improved through reheat and intercooling. A recent study extended the multiple reheat helium Brayton cycle design to SFR [20]. With the multiple reheat stages, the average heat absorbing temperature is close to the highest heat source temperature. With multiple cooling stages, the average heat rejection temperature is close to the heat sink temperature. Thus high thermal efficiency is obtained.

Compared to water and SCO<sub>2</sub>, helium leakage into intermediate sodium loop has no safety implication for the plant and has very little effect on plant availability. The multiple reheat helium Brayton system operates at much lower pressure (10 MPa versus 20 MPa for SCO<sub>2</sub> cycle), which means a much smaller pressure difference across heaters, and helium is inert gas and does not corrode structure materials while both water and CO<sub>2</sub> are corrosive at high temperature. Therefore, the possibility of heat exchanger wall breaks should be much lower for helium. If any wall break does occur, the break area will not increase due to absence of heating from Na-H<sub>2</sub>O or Na-CO<sub>2</sub> reactions from steam Rankine cycle or SCO<sub>2</sub> Brayton cycle. The leaked helium can be recovered in the expansion tank of the intermediate heat transfer loop. The plant can continue to operate unless the break area becomes too large.

Due to higher power density and elimination of large and bulky equipment items like condensers which operate under sub-atmosphere pressure in steam cycles, the Brayton cycle PCS tends to be much smaller and more compact than steam cycles [21]. The capital and operating cost of heat transport is consequently reduced, so are staffing and skill requirements. Zhao and Peterson [20] have shown that the thermal efficiency of the multiple reheat and intercooling helium Brayton cycle is about the same as that of SCO<sub>2</sub> cycle. Detailed tradeoff studies of the multiple reheat and intercooling helium Brayton cycle can be found

in references [5] and [20]. The companion paper by Zhao [25] has additional studies on this subject.

### II.3. Intermediate Heat Transport System (IHTS)

SFRs have three transport systems while LWRs have one or two. The IHTS is a major contributor to the higher cost of SFRs. Based on EFR's cost estimate, the IHTS accounts for about 10% of the overall plant cost [22]. In addition, IHTS is also one of the most vulnerable points in the system in terms of sodium leakage. Many sodium leak incidents have happened for the prototype (demonstration) SFRs. For example, a sodium leak in the IHTS of MONJU reactor forced it to be shutdown for 13 years. To improve the safety of SFR, sodium leak issue at IHTS has to be addressed. Researchers have been investigating alternative non-reactive fluid as the IHTS coolant. Saez and Rodriguez [23] have compared the thermophysical and thermal properties of a number of coolants: sodium, lithium, tin, bismuth, lead, lead-bismuth, lead-lithium, gallium, indium, potassium and sodium-potassium. They concluded that gallium's thermophysical properties satisfy the requirements as an IHTS coolant. However gallium is chemically corrosive to other metals at high temperature (above 400 °C). It does not appear to be a practical alternative fluid. Liquid salt has been considered as other primary candidate. However, due to high melting point of liquid salts no ideal candidate has been identified.

Suppressing the IHTS has also been a major activity to achieve cost reduction [3]. However leak-proof steam generator as IHX is required if steam Rankine cycle or  $\text{SCO}_2$  Brayton cycle is used. A number of activities have been carried out to improve the performance reliability of the steam generators. These include double-walled tubes or various geometric configurations to separate sodium from water [3].

However, double-walled tubes steam generators are expensive and not leak proof. The possibility of sodium water reaction is greatly reduced with double walled tubes, but not eliminated. Other geometric variants of steam generator designs to separate sodium from water would reduce the heat transfer from sodium to water and hence reduces the steam generator efficiency. All these measures are not practical to eliminate the IHTS if water or  $\text{SCO}_2$  are used as PCS coolant.

With the inert gas helium as power conversion fluid and the highly reliable compact IHX such as compact printed circuit heat exchangers and, it has the potential to realize the suppression of IHTS. Reactor safety analysis has to be performed to quantify the risk with a reactivity

insertion from helium passing through the core from a hypothetical helium gas leakage events in the IHX.

Economic trade-off studies will also need to be carried out. Replacing the sodium to sodium IHX with sodium to helium IHX will increase the size the IHX. The pipes that carry helium into IHX will also be much bigger than IHTS sodium pipes. Thus the savings from elimination of IHTS could be compromised by the extra cost incurred by larger IHX and reactor vessel if a pool type design is assumed.

Eliminating IHTS is probably better suited for advanced loop type design. Researchers have explored potentials to eliminate the intermediate heat transport system. Mito et. al. [24] proposed an advanced loop type SFR design using supercritical carbon dioxide ( $\text{SCO}_2$ ) gas turbine to suppress the intermediate heat transport system.  $\text{CO}_2$  gas is heated by the sodium-to- $\text{CO}_2$  heat exchanger. As a cost reduction option, the compact printed circuit heat exchanger (PCHE) was proposed as the sodium-to- $\text{CO}_2$  heat exchanger. However as was discussed in section II, the sodium- $\text{CO}_2$  reaction issue will inhibit the deployment of such system. On the tube rupture events,  $\text{CO}_2$  gas discharges through the leak hole of ruptured tubes and reacts with the primary sodium so the core integrity may be damaged. The products from sodium- $\text{CO}_2$  reaction include CO and  $\text{Na}_2\text{CO}_3$ . When CO enters the reactor core, it induces positive reactivity.  $\text{Na}_2\text{CO}_3$  creates the passage blockage issue which may cause local overheating. In addition, the high temperature sodium- $\text{CO}_2$  reaction jet by a tube rupture may cause the neighboring tubes to rupture, thus causing highly undesirable cascading effect of tube rupture events.

The multiple reheat helium Brayton cycle discussed in the previous subsection provides a practical means to realize the suppression IHTS. Fig. 2 shows a schematic view of an advanced loop design from SFR using multiple reheat helium Brayton cycle to suppress the IHTS. The companion paper [25] provides the details of this design.

### III. DECAY HEAT REMOVAL OPTIONS

Under normal shutdown conditions, decay heat is transferred to the environment through the intermediate heat transport system and the power conversion system. These systems are not safety grade equipment. The heat transfer is controlled to maintain the desired temperature value of the sodium in the vessel. In the highly unlikely event that after shutdown the heat removal from the primary system through secondary systems fails, the heat is retained in the primary system. Separate dedicated safety qualified decay heat removal systems bypassing the intermediate heat transport system and power conversion

system are required to meet the safety requirement. Due to the very high power density of SFRs, decay heat removal is required at all times and has to be absolutely reliable. Hence, passive decay heat removal system is essential to meet the reliability and safety requirement with reduced cost. The passive decay heat removal system relies on natural circulation of coolant in the system and the air. The

heat removal capacity of the passive system is directly proportional to the pool sodium temperature variation. The system heat load can be reliably discharged into the environment without either an operator action or any special provisions.

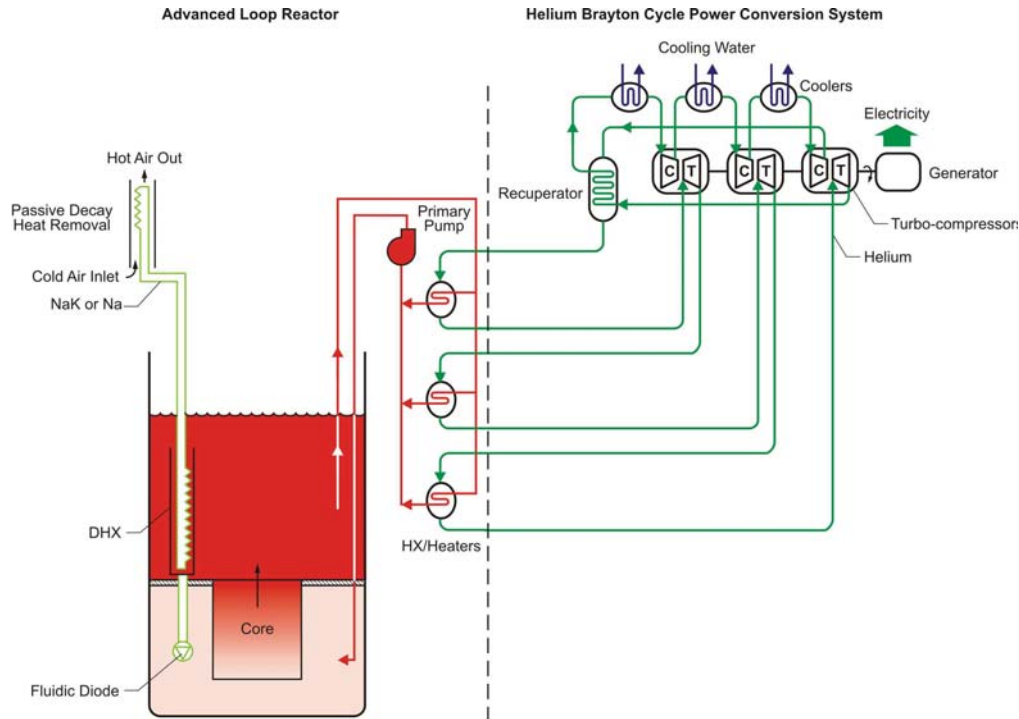


Fig. 2. Advanced loop type SFR design with the multiple reheat helium Brayton cycle to suppress the IHTS

Three auxiliary cooling system designs are available. These are: (1) the direct reactor auxiliary cooling system (DRACS) which was used in the Experimental Breeder Reactor-II (EBR-II) and many later designs, (2). the reactor vessel auxiliary cooling system (RVACS) proposed for the General Electric's PRISM and S-PRISM designs, and (3) the newly proposed pool reactor auxiliary cooling system (PRACS) system to supplement the DRACS in the context of hybrid loop-pool design described in section II.

Forsberg did a comprehensive study of the advantages and disadvantage of these approaches for liquid salt cooled advanced high-temperature reactor [26]. Similar comparative studies are carried out in this section for sodium cooled fast reactors.

### III.1. Direct Reactor Auxiliary Cooling System (DRACS)

The fully passive direct reactor auxiliary cooling system was originally utilized for the EBR-II reactor [27]. EBR-II has two DRACS loops. The sodium-potassium (NaK) eutectic alloy was used as the coolant. Each DRACS loop consists of a natural circulation heat-transport loop that moves heat from a sodium to sodium-potassium heat exchanger (DHX) immersed in the primary tank to a sodium-potassium-to-air heat exchanger (AHX) with the ultimate heat sink, the environment. Such an arrangement is preferred because it is independent of the intermediate heat transport system. The AHX is placed in an air stack outside the reactor containment building. DRACS operated continuously and minimum heat loss was maintained during normal operation by dampers in the air stack that restricted natural circulation of air through the stack. The damper was held closed by an electrically energized magnet and a minimum flow of NaK occurred in the DRACS loop. Upon the loss of electrical power, the damper opened and the air flow through the stack increased and consequently the heat removal rate increased.



The main advantage of DRACS is its modularity and scalability. The general design concept has been accepted elsewhere in the world and widely used in the existing pool type SFR designs, such as the Prototype Fast Reactor (PFR) Dounreay, EFR, KALIMER-600, Indian 500 MWe Prototype Fast Breeder Reactor (PFBR), and ABTR design etc. This design has also been adopted in the Japanese advanced loop design JSFR.

Various design variants exist for the DRACS for different SFR designs, especially in terms of DHX configuration. These different configurations have different consequent impact on the natural circulation flow.

EFR [22] and Indian 500 MWe pool-type Prototype Fast Breeder Reactor PFBR [28] place DHX in the hot pool. The cold sodium coming out of the DHX mixes with the hot sodium from the reactor core and forms cold stratification at the bottom of the hot pool. This cold stratification thus yields negative buoyancy force which works against the natural circulation from the hot pool through IHX, primary pumps to the reactor inlet. The negative buoyancy force reduces the positive buoyancy driving head and lowers the mass flow rate into the core. The decay heat removal capability is degraded. During normal operations, the cold sodium coming out of the DHX reduces the hot pool sodium temperature flowing into the IHX. Hence the thermal efficiency is slightly reduced. This type of configuration results the system being over designed.

ABTR [1] design configures DHX in the cold pool. Since the DRACS is always in the stand-by mode with this design, it prevents the coolant from freezing in cold weather and provides positive starting when needed. Having DHX in the cold pool only provides long term decay heat removal capability. Since the clad temperature normally reaches its peak value during a short period of time following the initiation of a transient, the reactor relies on the large thermal inertia of the hot pool and cold pool sodium and the natural circulation capability from the reactor core, hot pool, IHX, primary pumps and reactor inlet to remove the heat generated in the reactor to the cold pool and to keep the peak clad temperature below its safety criterion.

KALIMER-600 [29] configures the DHX at a position higher than the cold pool free surface during normal operations and thus it is not directly contacted with the sodium. During transients when the primary pumps are shutdown and the normal heat transport is inoperable, the cold pool level rises and the hot pool sodium expands due to higher temperature. The hot sodium overflows into the cold pool through DHX. Natural circulation establishes between the hot pool and cold pool through DHX. The

DHX heat removal capability is elevated due to the rapid increase of the convection heat transfer rate. Unlike the designs with DHX dipped in the hot pool or cold pool in which the DHX is always in stand-by mode, this design subjects the DRACS to thermal shock issue due to the superior heat conductivity of sodium. In addition, the system may not start up as smoothly as expected.

JSFR configures DHX in the hot plenum and is connected with the cold (lower) plenum by in-vessel pipes. Hot sodium flowing into the hot plenum from the core is cooled by the DHX and the cold sodium flows down to the cold plenum by natural convection force through the in-vessel pipes. Fluidic diodes are used at the penetration between the hot plenum and the cold plenum to restrict leakage flow from the cold plenum to the hot plenum during normal operations [30]. Similar to the PRACS design proposed for the hybrid loop-pool design, fluidic diode is used to restrict the leakage flow from the cold plenum to the hot plenum during normal operations and enhance natural circulation from the hot plenum to the cold plenum during abnormal conditions. During transients, the hot sodium in the hot plenum flows into DHX and the cold sodium exiting DHX directly feeds into the cold plenum. The advantage of this design is that not only the DHX removes heat from the hot plenum, it also enhances the natural circulation flow rate into the reactor during abnormal conditions.

Some experiments have been conducted to study the impact of different configurations of the sodium to sodium heat exchangers on the characteristics of natural circulation for pool type fast reactors design [31] under steady state condition. Three configurations have been investigated, placing the heat exchanger in the hot pool, in the cold pool and between the hot pool and the cold pool. The authors concluded placing the heat exchanger between the hot pool and the cold pool is the most desirable option.

### *III.2. Reactor Vessel Auxiliary Cooling System (RVACS)*

RVACS was originally developed for General Electric's PRISM, ALMR and S-PRISM design. Fig. 3 shows a schematic view of RVACS which is drawn based on Fig. 5 in reference [32]. With RVACS, the decay heat is moved from the reactor core to the cold pool by natural circulation of sodium coolant. The heat is then conducted through the reactor vessel wall and transferred across an argon gap by radiation to the containment vessel. The heat is further conducted through the containment vessel and then removed from outside of the containment vessel by natural circulation of upwardly flowing atmospheric air around the vessel. The rate of heat removal is controlled primarily by the radiation heat transfer through the argon gas from the reactor vessel to the guard vessel. Because

RVACS is always in operation a small amount of parasitic heat loss exists during normal operation. During transients when RVACS is required to remove decay heat, as the temperature of reactor sodium and reactor vessel rises, the radiation heat transfer across the argon gap to the containment increases since the radiant heat transfer varies proportional to the absolute temperature to the fourth power. Consequently, as the containment vessel temperature rises, the heat transfer from the containment vessel to the environment increases.

RVACS has been widely used in the small modular reactor designs, such as Toshiba's 4S (Super-Safe, Small and Simple) sodium cooled fast reactor design [33]. Similar design concept has also been applied to the reactor cavity cooling of high temperature gas cooled GT-MHR design [18].

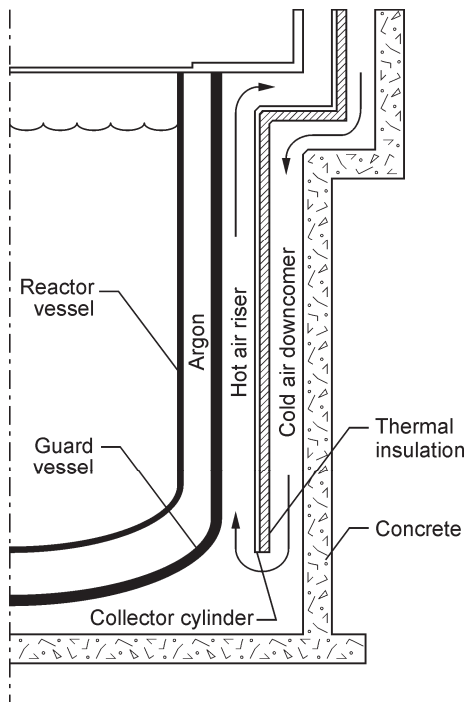


Fig. 3. Schematic view of RVACS system.

RVACS has the advantage of the simplicity of the system. However several disadvantages are associated with RVACS for SFR applications. RVACS relies on the reactor vessel surface area to transfer decay heat. Consequently, the reactor power output is limited by the reactor vessel size and the maximum allowable vessel temperature under accident conditions. It is difficult to scale up RVACS to a commercial scale monolithic plant design to take advantage of the economy of scale. In addition, the reactor vessel has two conflicting functional requirements -- containment of the reactor system and transfer of decay heat under accident conditions. To maximize vessel integrity, vessel temperatures should be minimized. However, for removal

of decay heat, the vessel should operate at high temperatures under certain accident conditions [26].

### III.3 Pool Reactor Auxiliary Cooling System (PRACS) to Enhance the DRACS

The pool reactor auxiliary cooling system (PRACS) is a new decay heat removal system that was proposed for the hybrid loop-pool design described in section II. In the hybrid loop-pool design, the cold pool has been buffered from the hot pool. The primary sodium does not mix with the buffer sodium in the pool. The hot sodium from the reactor flows to the IHX, dumps its heat to the intermediate heat transport loops and returns to the reactor core driven primary pumps. During normal operation the buffer sodium is at the same or lower temperature than the coldest primary sodium, that is, less than the core inlet temperature.

Decay-heat removal from the primary system to the pool upon loss of a circulation pump can be enhanced by the PRACS.

Analogous to the benefits Forsberg [26] discussed for the AHTR, PRACS has several potential advantages for SFR designs:

**Enhanced natural circulation:** With the elevation differences and temperature differences between the reactor inlet and the hot pool, the hydro-static head of PRACS is always in the "stand-by" mode even during normal operations. PRACS has very small inventory of primary sodium coolant that can respond rapidly to changes in reactor operating conditions such as temperatures and flow rate. The buoyancy force will overcome the pumping power at the later stage of the primary pump coastdown during a loss of forced circulation cooling transient and drive the flow to reverse in PRACS. The natural circulation within PRACS establishes before the primary pump coastdown fully stops and thus provides a smooth transition from pump coastdown to fully established natural circulation.

**Inherent safe reactivity control:** As a two-sodium-system with the closed primary loops immersed in a buffer tank design, the primary sodium inventory has been reduced significantly. Hence the primary system can respond rapidly to changes in core temperatures and a large buffer sodium inventory that responds slowly to temperature changes in the reactor core. If heat removal by the intermediate heat exchangers stops and the reactor is not shut down, the primary coolant can heat up rapidly and ensure rapid shutdown of the reactor under a wide variety of conditions because of negative reactivity feedback. The primary sodium assists in ensuring reactor

reactivity control while the high heat capacity buffer sodium is available for longer-term decay heat removal.

**Increased thermal inertia:** The buffer sodium can be set at a much lower temperature than the reactor core or primary coolant and thus can absorb very large amounts of decay heat relative to conventional pool design in which almost all the sodium is at a single temperature.

**Primary-system integrity:** The temperature-limited safety components in the primary system are the reactor vessel and piping. In a pool reactor, the outside of the reactor vessel and piping is bathed in cooler buffer sodium. The lower-temperature buffer sodium and the excellent heat transfer provided by a cool liquid provide a method to limit primary metal component temperatures and thus provide a high level of assurance of primary-system protection from excessive temperatures.

**Optimal DHX Configuration:** The buffer sodium temperature distribution is thermally stratified [34] with the higher temperature sodium in the upper portion of the tank. The thermally stratified phenomenon is more pronounced under abnormal conditions. DRACS heat exchangers

(DHX) can be located in the hotter zone of the buffer pool, and baffling can be used so the hottest sodium in the buffer pool enters the DRACS heat exchangers. By maximizing the buffer sodium temperatures flowing into DRACS heat exchangers (DHX) enables the maximal decay-heat removal by DRACS. Consequently, the buffer pool temperatures are minimized and mixing in the buffer sodium can be enhanced to reduce thermal stratification.

However PRACS is a new system approach that presents design uncertainties. It also adds some complexity to the system.

#### IV. FUTURE ADVANCED SFR DESIGN OPTIONS

Fig. 4 shows the schematic of an advanced SFR design concept. The hybrid loop-pool design will be employed for the primary system. The multiple reheat and intercooling helium Brayton cycles will be employed for the power conversion system. The intermediate heat transport system may be suppressed depending on safety analysis results and economic trade-off studies. The PRACS and DRACS are proposed as the decay heat removal option.

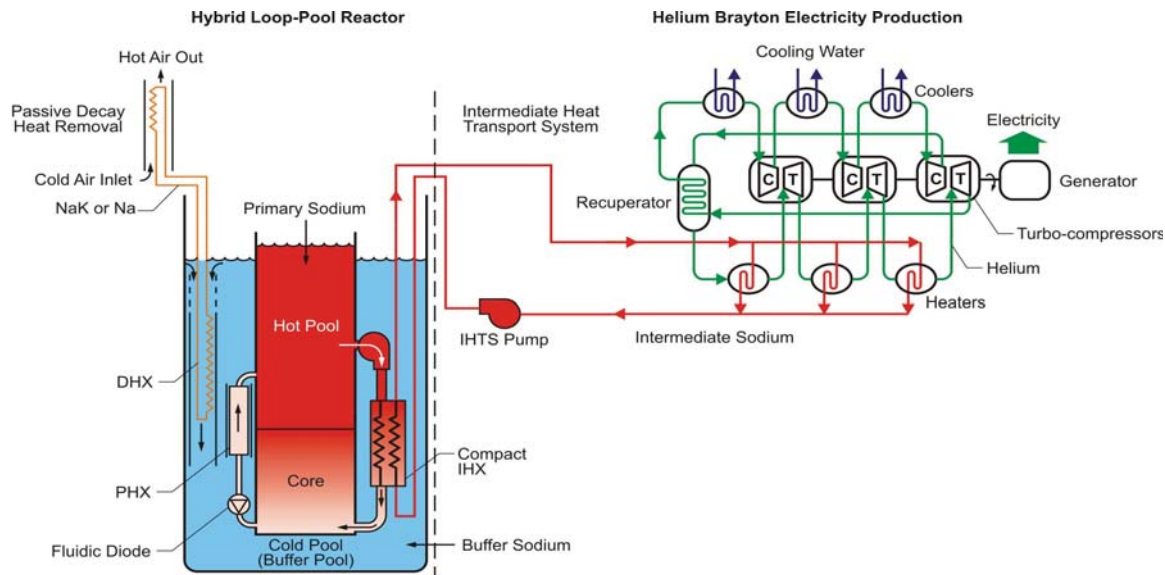


Fig. 4. Schematic of an Advanced SFR design

#### V. CONCLUSIONS AND FUTURE WORK

This paper presents some considerations for advanced SFR design aiming at achieving the economic competitiveness of SFRs. Future work includes developing a reference reactor design with these innovations incorporated to demonstrate cost reduction with sufficient

safety margins using materials input method [21]. Demonstration of sufficient cost reduction will lead to wide spread interest to the “renaissance” of sodium cooled fast reactor technology and the large scale commercial investment in SFR power plants.

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## ACRONYMS

SFR – sodium cooled fast reactor  
PRACS – pool reactor auxiliary cooling system  
PHX – PRACS heat exchangers  
DRACS – direct reactor auxiliary cooling system  
RVACS – reactor vessel auxiliary cooling system  
DHX – DRACS heat exchangers  
IHX – intermediate heat exchangers  
ABTR – advanced burner test reactor  
AHTR – advanced high temperature reactor

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