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CURRENT LIQUID METAL COOLED FAST REACTOR CONCEPTS: USE OF THE DRY REPROCESS FUEL

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Abstract

Recent Liquid Metal Cooled Fast Reactor (LFR) concepts are reviewed for investigating the potential usability of the Dry Reprocess Fuel (DRF). The LFRs have been categorized into two different types: the sodium cooled and the lead cooled systems. In each category, overall design and engineering concepts are collected which includes those of S-PRISM, AFR300, STAR, ENHS and more. Specially, the nuclear fuel types which can be used in these LFRs, have been summarized and their thermal, physical and neutronic characteristics are tabulated. This study does not suggest the best-matching LFR for the DRF, but shows good possibility that the DRF fuel can be used in future LFRs.

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I. Introduction

Recent evolution of the Liquid Metal Cooled Fast Reactor (MFR) design and engineering concepts is closely linked to the reprocessing technology development which is being directed to meeting two important objectives: increasing the proliferation resistance of the fuel cycle, and reducing the long-term radiotoxicity. These objectives are consistent with the Gen IV goals, that is, "dirty fuel" and "clean waste" can be produced as the result.

The reprocessing technology can be categorized into two different types which are hydrometallurgical ("aqueous") and pyrochemical ("dry") processes. Both of these can be applied to the separation of the long-lived radionuclides. The well-known hydrometallurgical process is PUREX in which U, Pu and eventually Np can be separated. On the other hand, in the pyrochemical processes, refining is carried out in molten salt media and the resulted fuel is less pure. The major advantage of the pyrochemical technique for reprocessing of advanced fuel, in comparison with PUREX which is well developed commercialized process, are greater compactness of the equipment and the possibility to form an integrated system between irradiation reprocessing and refabrication facility, thus reducing considerably transport of nuclear materials. At this moment, it is quite obvious that PUREX cannot be securely deployed worldwidely due to its Pu separation capability.

The Dry Reprocess Fuel (DRF) technology being developed in KAERI can be a significant mean to control the spent PWR fuel disposition if a proper fuel cycle strategy is adopted in Korea. The DRF technology does not require any aquatic additives. The DRF is simple and secure. The DRF is nuclear profileration-resistent since no extensive separation effort is supposed to be involved. These are significant advantages of the DRF technology within the contemporary notion of the peaceful nuclear energy utilization of world-leading countries. It should be noted, however, that such a nature of the DRF technology may dwarf different technical developments.

This report is not written for who seeks extensive comparative study of MFRs. Rather, some of significant liquid MFR concepts have been collected as in Table 1. Among these, a few have been selected for illustrating the MFR concepts and eventually for a potential use of the dry reprocessed fuel. The materials in Chapters II and III are distilled from papers presented in different technical meetings [1,2,3,4].

II. Sodium-Cooled Fast Breeder Reactors (SFR)

II.1 S-PRISM

S-PRISM is an advanced Fast Reactor plant design that utilizes compact modular pool-type reactors. GE developed and assessed the technical viablility and economic potential of S-PRISM based on the previous DOE sponsored Advanced Liquid Metal Reactor (ALMR) program,

Key features included in the reference S-PRISM design include:

-Passive decay heat removal

-Passive accommodation of ATWS Events

-Passive reactor cavity cooling

-Automated safety grade actions limited to containment isolation, reactor scram, and steam side isolation and blow-down

The reference commercial S-PRISM plant utilizes six modules arranged in three identical 760 MWe(net) power blocks for an overall plant net electrical rating of 2280 MWe. The pool type reactor modules contain the complete primary system including the core, pumps, and intermediate heat exchangers that transfer heat to twin secondary sodium loops that are connected to a single helical coil steam generator. The two modular steam supply systems in each power block jointly supply steam to one of three 825 MWe (gross) turbine-generator units.

The reference S-PRISM core is based on metal fuel; however, an alternative core using oxide fuel can also be utilized since both cores are designed to fit the same layout including the number and location of the control assemblies. The small number of fuel assemblies in each S-PRISM reactor module reduces the refueling time and simplifies in-vessel spent fuel handling and storage. In addition, it allows the radial power profile to be flattened by shuffling the internal blankets from the center of the core to the periphery of the core as their fissile enrichment increases. The reference core is optimized for minimum fuel cycle cost and thus has a breeding mission of fissile breakeven. Axial blankets can be added to this basic core to provide a breeding ratio greater than 1.3 within the axial envelope of the core.

The metal core uses 138 driver fuel assemblies. There are 49 internal blanket assemblies distributed throughout the driver core to enhance internal breeding. Surrounding the driver core are 48 radial blanket positions. Outward of the blankets are two rings of steel-filled reflector assemblies and a ring of boron carbide shield assemblies.

There are 9 primary control assemblies and three secondary assemblies within the driver core. SIx Gas Expansion Modules (GEMs) are located at the perimeter of the core to provide enhanced negative reactivity feedback during a loss of flow event.
The fuel assemblies remain in each location for the full in-core residence time of six years. At the end of each two year long operating cycle one third of the core is replaced.

The high internal conversion ratio achievable with metal fuel in S-PRISM reduces cycle burnup swing to near zero. It should be noted that the allowable MOX fuel burnup (180 MWd/kgHM) is greater than metal (150 MWd/kgHM). Other plant performance characteristics of S-PRISM are summarized in Table 2.

II.2 AFR-300, Advanced Fast Reactor (300 MWe)

The 300 MWe advanced fast reactor concept, AFR-300, is a system that consists of a fast-spectrum nuclear reactor that uses metallic fuel and liquid metal (sodium) cooling, coupled with the technology for high-temperature electrochemical recycling, and with processes for preparing wastes for disposition. The concept is based on the major successful design features of the EBR II reactor and its fuel cycle facilities, and on decades of experience with fast reactors, adapted to priorities also recognized in the Generation IV Technology Goals that have evolved markedly from the early days of nuclear power.

The AFR-300 concept has four significant, distinguishing benefits in terms of the basic goals for generation IV (sustainability, safety and reliability, and economics): efficient use of the natural and man-made resources, inherent safety characteristics, reduced burdens of nuclear waste, and unique proliferation resistance. These fundamental characteristics offer benefits in economics and environmental protection. Furthermore, the fuel cycle never involves separated plutonium, simplifying and enhancing the non-proliferation aspects of the AFR-300 concept.

A key feature of the AFR-300 fuel cycle concept is that the fuel is metallic, which brings pronounced benefits in improved inherent safety and lower processing costs. Metallic fuel was in fact the original choice in the early development of liquid-metal reactors and was successfully developed and used as the driver fuel in EBR II. Metallic fuel development has led to designs with superior-tolerance to heavy irradiation, thus allowing for very high burnup.

Another key feature of the AFR-300 concept is pyroprocessing, a fuel processing method that utilizes high temperatures with molten salt and molten-metal solvents. Pyroprocessing is advantageous for metallic fuels because the product is a highly radioactive uranium-plutonium metal alloy containing minor actinides and rare earth elements that is suitable for immediate fabrication into new fuel pins. Separated plutonium is never present in the fuel cycle, a significant factor in the superior non-proliferation aspects of the AFR-300. The most important step in the pyroprocessing of the fuel is electrorefining, which recovers the valuable fuel constituents, uranium and plutonium and removes most of the fission products. In the electrorefining operation, uranium and plutonium are selectively transported from the anode to the cathode of an electrolytic cell, leaving impurity elements (fission products) either in the anode compartment of in the molten-salt electrolyte. An important aspect of the process is that the minor actinide elements (americium, neptunium, curium) always accompany plutonium through the entire process.

A summary of the principal characteristics of the AFR-300 plant is provided in Table 3. The basic characteristics of the AFR-300 are retained from the basic successful design features of the EBR II; in particular, the passive safety design features demonstrated in EBR II are retained. The reactor core has large margins between the operating and physical safety limits, low pressure primary loop, high thermal conductivity metal fuel, and favorable reactivity feedbacks characteristic of a fast reactor system. The use of liquid metal coolant allows a compact core design with high power density. The AFR-300 core can be configured for a wide range of conversion factors; this allows flexibility in the management of fissile materials based on future nuclear energy supply and demand.

The basic reactor design is based on a primary circuit fully contained in a sodium-filled double-walled tank. The entire primary system and sodium pool operates at nearly atmospheric pressure. The reactor is contained in a vessel inside the primary tank, and the major primary components are connected by piping. The piping is designed with the expectation that the sodium will leak from the connections to the primary pumps and past the control rod drive penetrations in the reactor vessel cover without being lost to the system. Heat is transferred to a secondary sodium circuit through intermediate heat exchangers immersed in the primary tank sodium pool. The hot sodium from the core is piped directly to the intermediate heat exchangers, resulting in a nearly isothermal cool pool of sodium in the primary tank, ensuring limited thermal stress on this structure as well as the components inside it. Natural circulation is established in case of failure of the primary pumps. It should also be noted that the secondary circuit contains no valves.

II.3 4S-LMR

The 4S (Super, Small, and Simple)-LMR incorporates the following design policies:

- Security through elimination of on-site refueling
- Simple design and construction with a transportable reactor assembly with internals
- Simple operation and maintenance with no control rods and drive mechanisms, and
- Inherent and passive safety features

The original 4S system was designed by CRIEPI and Toshiba (1988-1992) with a core life of 10 years. The current design effort by Toshiba shows a potential to extend its core life up to 30 years with proliferation resistant plant system technologies.

The core is designed to be operated for 30 years without refueling. A rotating mechanism of the shield plug for refueling is eliminated. The primary pump is an in-sodium annular electromagnetic pump (EMP) rather than centrifugal pumps. There is a simple primary heat transport system with annular designed internals (IHX, Flow pass without piping). A sealed concept if the primary cover gas system is utilized. The containment has a simple shape with a top dome and a guard vessel. Decay heat removal systems are passive: the Primary Reactor Auxiliary Cooling System (PRACS) and the Reactor Vessel Auxiliary Cooling system (RVACS) are driven by natural circulation and natural draft. Residual heat rejection from the secondary system room is accomplished using heat sink equipment of the appropriate heat capacity when a loss of off-site power occurs. Emergency power for the reactor auxiliary heat rejection systems or HVAC systems is eliminated by the above features.

The control system relies on a reflector rather than control rods. Reactivity is controlled by vertical movement of the annular reflector system during the plant startup, operation, and shutdown. Also, there is a back-up safe shutdown rod comprised of a neutron absorber assembly at the top of the core center. Operation starts from heating the system up to 350。C by the primary EMP. Then the reflector begins to be lifted up by a hydraulic system to a critical position, after a neutron absorber is withdrawn. The turbine generator will be connected when power is raised to 20% of rated power. After the rated power is achieved, regular operation is attained by moving the reflector upward at a limited speed of approximately 1 mm/month to make up the reactivity decrease from burn up. The fuel can be U-Zr metallic fuel with 130 MWD/kg of burnup. The 4S has at least two design options for its fuel and performance to meet the Generation IV goals. Their selection depends on the time for deployment which is dependent on market demands. The first is a reference design option-with Uranium fuel enriched to no more than 20%, and a lower conversion ratio approximately 0.5-for the earlier deployment in the 2010's market, especially as envisioned for the developing countries of Asia. The second option uses U-Pu-Zr fuel with a higher conversion ratio (~1.0), for better utilization of uranium suitable to the Generation IV goal.

III. Lead-Cooled Fast Breeder Reactors (LFR)

III.1 Secure Transportable Autonomous Reactor Liquid Metal (STAR-LM)

The Secure Transportable Autonomous Reactor Liquid Metal (STAR-LM) is a 300 to 400 MWt class modular, factory-fabricated, overland transportable, passively safe reactor concept that takes advantage of the intrinsic benefits of a fast neutron spectrum core utilizing high thermal conductivity nitride fuel and natural circulation heat transport using lead-bismuth eutectic (LBE) heavy liquid metal coolant. STAR-LM has the potential of meeting all of the Generation IV goals. Key elements of the STAR-LM concept are shown in Figure 1.

A reactor module that is functionally a flow-thru fuel cartridge that contains the integral 15-year ultra long life core, inlet distributor plate, steel shielding/reflector structures, and the reactor vessel. The module is a welded ferritic steel (e.g. HT9) structure that affords no access to any core materials. It is inserted into the larger coolant module vessel and is removable for replacement at the end of the 15-year core life.

Modular steam generators are inserted inside the coolant module vessel in the annular region between the reactor and coolant vessels. Coolant flows upward through the core and downward through the inside of the vertical steam generator tubes driven by natural circulation alone. Feedwater is delivered through a vertical pipe to the bottom of each steam generator. Superheated steam is produced on the shell side. The feedwater temperature provides a margin above the LBE freezing temperature.

The coolant module consists of the coolant vessel, coolant vessel liner, structure for positioning the reactor and steam generator modules, an upper closure head with penetrations for inserting and sealing the reactor and steam generator modules, and piping for conditioning the coolant via control of oxygen level and removal of corrosion products/impurities.

The coolant module vessel is contained in a guard vessel which also functions as the containment boundary inside the reactor silo. The annular space between the two vessels contains removable steel venetian conductors that occupy ~ 80 percent of the annular volume and provide a good thermal conduction path between the vessels.

Redundant passive heat removal is provided by cooling the outside of the guard vessel with air driven by natural circulation in the event that all of the modular steam generators are unavailable for heat removal.

The containment boundary is the guard vessel plus the overlying interior of the head access region.

A nuclear island containing the modular reactor system, steam and feedwater piping, RECS, and containment volume is supported by seismic resistor pads for seismic isolation where required by site conditions.

There is no refueling of shuffling of fuel during the lifetime of the core. Transuranic nitride is currently selected as the fuel material based upon its compatibility with LBE and ferritic steel such as HT9 as well as its high thermal conductivity. The heavy metal consists of depleted uranium enriched with 8 percent plutonium. The mean and peak discharge burnups are 72 and 121 Megawatt days per Kilogram, respectively. The low burnups reflect the low core power density needed for a 15 year lifetime. The reactor could also be fueled with depleted uranium enriched with U^{235}, although this would incur greater reactivity losses during the fuel lifetime. The required U^{235} enrichment would also be greater.

III.2 Encapsulated Nuclear Heat Source (ENHS)

The ENHS is an innovative Pb or Pb-Bi cooled 125 MWth fast spectrum reactor module. The ENHS is completely factory fabricated, assembled, fuelled and seal welded, and can be transported to the power plant site with the fuel embedded in solid Pb-Bi. No on-site refueling and fueling hardware is required. It operates for 20 years of full power without refueling. ENHS maintains nearly constant fissile fuel contents and multiplication factor; hence, needs very small excess reactivity built-in and has a very simple reactivity control system. There are no mechanical connections between the ENHS module and the energy conversion plant, making it possible to replace the module at the end of its core life. At end of life the ENHS module serves as a spent fuel storage cask and, later, as a spent-fuel shipping-cask. That is, the fuel is locked inside the ENHS when in the host country. The fission-generated heat is transferred from the primary-to the secondary coolant through a reactor vessel wall of a novel design; a novel IHX is integrated within the vessel wall. There are no pumps, valves or pipes in the primary and secondary coolant loops. The coolants flow by natural circulation resulting in passive load following capability and autonomous control. The compensation for reactivity changes and for power variations is done via temperature feedback. The ENHS is simple to operate and to maintain; the required staff is small. There are no special decay heat removal systems other then the RVACS. The natural circulation combined with the large heat capacity of the primary and secondary coolants make the ENHS reactor of utmost passive safety. Postulated severe accidents such as LOCA and LOFA are eliminated by design. The large safety margins permit demonstration of tolerance to postulated accidents such as insertion of all available reactivity. There will be no need for an emergency planning zone beyond the site boundary of an ENHS power plant. A number of ENHS modules can be installed in a single pool of secondary coolant making a highly modular power plant of up to several hundred MWe in capacity.

A description of one of several possible embodiments of a reactor concept having a single ENHS module is depicted in Figure 1 and Table 4. The nuclear island consists of nine components - one ENHS module and eight steam generators (SGs) and the secondary pool vessel. The ENHS module and SGs are completely factory fabricated and transported to the power plant site ready to be installed; they are supported within a reinforced concrete cavity. There is no mechanical connection between the ENHS module and the SGs. This makes it possible to replace the ENHS module while the pool vessel and balance-of-plant remain in place for possibly 60 years; all components undergoing radiation damage are replaced along with the fuel every 20 years. The SGs can be inspected and maintained from the top of the nuclear island while the plant continues to operate. Although Pb-Bi is the preferred primary coolant, Pb or even Na can serve as coolants as well.

Both primary and secondary coolants of the ENHS feature 100% natural circulation. The primary coolant that is heated in the core flows up the riser, turns over into the Intermediate Heat Exchanger (IHX) and flows back into the coolant inlet plenum underneath the core. The IHX is integrated between the inner-, and the outer-structural walls. The secondary coolant flows from the pool outside of the module into the bottom of the IHX and exits black to the pool near the top of the IHX. Heat is conducted from the primary to the secondary coolants through the walls of the IHX channels. The IHX consists of rectangular channels that are connected at their top and bottom to a rube sheet. The rectangular channel walls provide the barrier between the primary and the secondary coolants whereas the inner and outer walls provide the structural support. More conventional IHX made of circular tubes could be used as well. Relative to circular tube IHX the rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller friction losses due to elimination of grid spacers.

The ENHS module will be manufactured and fuelled in the factory and shipped to the site as a seal welded unit with solidified Pb-Bi (or Pb) filling the vessel up to the upper level of the fuel rods. On site the ENHS module is installed into the pool hot Pb-Bi while it is filled with primary coolant. The hot primary Pb-Bi, along with the hot Pb-Bi in the pool, will melt the solid Pb-Bi that was installed at the factory.

Following 20 years of operation the ENHS module is removed and replaced with a new module. The used module is stored on site while it cools and the Pb-Bi in and below the core freezes. The used module is then shipped to a recycling/waste-disposal facility.

The shutdown assembly includes an electromagnetic latch that does not engage until a primary coolant temperature of 350。C is achieved. At this point the shutdown assembly can be withdrawn and held out until permanent shutdown is required. Normal operational shutdowns can be accomplished, if needed, with the reflectors.

The path for decay-heat removal is from the primary coolant through the IHX walls to the secondary coolant and from there either to the steam generators or, through the pool vessel walls, to a passive reactor vessel air cooling system (RVACS). The ENHS is characterized by a very large surface area per MWth of decay heat removal both from the primary to the secondary coolant, and from the secondary coolant to the RVACS. In addition, the ENHS is characterized by a large thermal inertia due to the large inventory of the primary and secondary Pb and the very large margin to coolant boiling. Both features, combined with a low power density, make the ENHS highly passively safe. There is no need for any special decay heat removal system other than the RVACS. Multiple ENHS module power plants, having a capacity of 100 to 600 MWe, can be developed and attractive for all industrial countries.

III.3 BREST

The BREST reactor and fuel cycle system concept has been under study in Russia for roughly the last decade. Argonne National Laboratory performed a technical assessment of the BREST concept in 2000 as a prelude to a proposed U. S. Russian bilateral study on improving proliferation resistance of civil nuclear power.

The BREST reactor is a lead-cooled, fast spectrum reactor designed to operate in a mode of slight net breeding, and thus capture the resource utilization benefits of breeders, and simultaneously utilize the large stocks of plutonium that are accumulating from operation of thermal reactors as well as from release of excess plutonium from weapons programs.

The BREST system two different sizes of reactors, a 300MWe version considered a mid-size reactor amenable to use as a test or demonstration plant, and a 1200MWe version that is viewed as the best candidate for large-scale deployment. Neither design incorporates a Western-style reactor containment, consistent with prior Russian liquid metal cooled reactor design practice.

The fuel for BREST reactors is mixed nitride (U-Pu)N, which is compatible with the lead coolant and with the chromium ferrite-martinsite steel used as fuel element cladding.

BREST reactors have the pool-type primary system arrangement, wherein the reactor core and other major primary-coolant system components (pumps, steam generators) are contained in a single, contiguous vessel.

The safety approach adopted in BREST aims to preclude severe accidents that may result in fuel failure and release of radioactivity by exploiting natural phenomena and intrinsic characteristics. These characteristics include low system pressure, large heat capacity, natural circulation flows, negative temperature coefficient of reactivity, chemically inert materials, and low excess reactivity. The goal of this natural safety approach is to make the reactor plant immune to human errors or to failure of equipment or engineered safety systems. All potential accidents, aside from massive external forces (e. g. impact of an asteroid or nuclear attack), are thus considered within the design basis. Proponents claim that this approach avoids reliance on probabilistic arguments and analyses for substantiating reactor safety and is therefore labeled deterministic safety .

An important BREST feature is a core design with a core breeding ratio slightly greater than unity (CBR~1.05). With CBR slightly exceeding one, the fissile mass increases slightly over an operating cycle and compensates for the reactivity loss associated with buildup of fission products and change in fuel isotopic composition. As a result, the reactivity change over an operating cycle due to fuel depletion is essentially zero. This greatly reduces the excess reactivity requirement and the potential for, and consequences of, reactivity insertion accidents.

The mixed nitride fuel used in the BREST design has high thermal conductivity and low stored energy and therefore has small reactivity effects associated with fuel temperature change. These characteristics make it possible to minimize excess reactivity still further, and to facilitate passive accommodation of loss of flow sequences.

The choice of lead coolant was made in an attempt to achieve a higher level of intrinsic safety, and improved economics. Until now, sodium was universally adopted as the coolant for fast-spectrum power reactors, owing to its superior heat transport characteristics and low pump-power requirements in the tight-lattice, high pressure-drop reactor core designs that were characteristics of reactors aimed at high breeding ratio and low doubling time.

For submarine propulsion, Russian designers adopted a lead-bismuth eutectic coolant to achieve a compact, high-performance core. The coolant selection was motivated mainly by the Pb/Bi inertness with air, and with the steam/water working fluid. It resulted in a major simplification of the traditional sodium-cooled heat transport layout by elimination of the intermediate heat transport loop. This simplification was adopted in BREST.

IV. Summary and Conclusion

It has been shown that the fuel can be used in MFRs as one in the forms of the oxide, the metal and the nitride. The major characteristics of each fuel type are summarized in Table 5. It should be reminded that each item in the leftmost column of the Table 5 does not necessarily have a uniform importance in the fuel selection.

Regardless of the reactor type, the thermo-chemical stability of the fuel material under the high temperature and pressure condition is very important, and the oxide fuel is the best in this regard. It is believed, therefore, that the proposed oxide fuel (one of most important outcomes of the DRF technology) would easily find the best-matching MFR which is not found yet.

It has been found from this study that the fuel type evaluation is quite difficult at the stage of the reactor design since the reactor is merely a fuel burner, that is, the reactor should be so designed to be capable of burning the selected fuel economically and safely. Nevertheless, seeking fuel type which is strategically proper in the energy planning and the waste disposal should be important.

This study has revealed the relationships between different fuel types and current MFR concepts.

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Table 1 GEN-IV Liquid Metal Reactor Designs and Concepts

Reactor	Coolant	Fuel	Country	Note
BREST	Pb-Bi	(Pu-U)N	Russia	BR=1.05
BN-800	Na	UO_2	Russia	Large
SVBR-75/100	Pb-Bi	Oxide/Metal	Russia	Small
S-PRISM	Na	Oxide/Metal	U.S.A.	G.E.
PBWR	Pb-Bi	Oxide/Metal	U.S.A.	MIT/INEEL
ENHS	Pb-Bi	Pu-MA	U.S.A.	UC, Berkeley
AFR-300	Na	U-Zr or U-TRU-Zr	U.S.A.	ANL
STAR-LM	Pb-Bi	Depleted U /w 8% Pu	U.S.A.	ANL
LMFR-ULLC	Na	Metal	Korea	KAERI

Table 2. S-PRISM Plant Performance Characteristics

Overall Plant	
- Net Electrical Output	2280 MWe
- Net Station Efficiency	38%
- Number of Power Blocks	Three
- Number of Reactor Modules: (Per power block/plant)	Two/six
- Plant Capacity Factor (Nominal, Expected)	85% 93%
- Number Control Consoles in CR	3
Power Block	
- Number of Reactors	Two
- Net Electrical Output	760 MWe
- Steam Generator	Two
- Steam Generator Type	Helical Coil
- Steam Cycle	Superheat
- Turbine Type	3600 rpm TC-4F
- Turbine Throttle Conditions	468℃/177kg/㎠
- Main Steam Flow	996 kg/sec
- Feedwater Temperature	215℃
Reactor Module	
- Thermal Power (Core)	1,000 MWt
- Primary Sodium Inlet/Outlet Temperature	371/510℃
- Primary Sodium Flow Rate	5,666kg/sec
- Intermediate Sodium Inlet/Outle Temperature	321/469℃
- Intermediate Sodium Flow Rate	4,493kg/sec
- Number of IHX Units	2
- Number of Primary Pumps	4
- Number of Primary Control Rods	9
- Number of Secondary Control Rods	3

Table 3. 300MWe AFR-300 System Main Design Characteristics

Reactor plant output, MWe	300
Termal effciency (typical)%	38
Plant design lifetime, years	60
Primary Coolant	Sodium
Intermediate circuit coolant	Sodium
Core Diameter, m	2.5
Core Active (Heated) Zone Height. m	1.0
Fuel Material	Binary U-Zr or ternary U-TRU-Zr
Fuel utilization: average discharge BU (at%)	20
Reactor plant LLW	No emissions; no sodium feed during plant lifetime
Gap Bond Material	Sodium
Mean Temperature Rise Across Core, ℃	150
Core Outlet Temperature, ℃	510
Core Inlet Temperature, ℃	360
Primary tank pool temperature, ℃	360
Primary loops	3
Decay heat removal	Passive; Natural circulation
Decay heat removal system	DRACS
Fuel handling	Direct pull of fuel assemblies; out-of-tank fuel transfers during reactor operation
Economic advantages	Fuel utilization, fuel cycle facility collocation; long lifetime (no corrosion); simple design, low number of safety systems, passive systems (few components, low testing and mainernance requirements); low waste (in both cycle and plant operations); short refueling outages; no pressure systems in primary

Table 4. Design and Performance Characteristics of ENHS Reference Designs

Design parameter	ENHS1	ENHS2
Primary Pb coolant circulation	100% natural	With lift-pump
Average linear heat-rate (W/cm)	60	60
Average discharge BU (MWd/tHM)	52,000	52,000
Core life (effective full power years)	20	20
BU reactivity swing	<1$	<1$
Maximum excess reactivity	<1$	<1$
Core height (m)	1.25	1.50
Core diameter (m)	1.98	1.87
Fuel rod diameter(cm)	1.0	1.0
Clad thickness (cm)	0.1	0.1
Lattice (hexagonal) pitch (cm)	1.45	1.50
Overall module height (m)	19.6	10.1
Outer module diameter (m)	3.24	3.35
Number of rectangular channels in IHX	135	245
Inner dimensions of channel (cm x cm)	40 x 2.5	50 x 1.0
IHX channel length (m)	13	6
Weight of fueled module for shipment (t)	360	300
Coolant core inlet/outlet trmperature (℃)	400/564	400/543
Primary-to-secondary mean △T (℃)	49.1	47.3
Number of steam generators per ENHS	8	8
Steam generator module diameter (m)	1.0	1.0
Active length of SG tubes (m)	7.5	7.5

Table 5. Different Fuel Types for MFRs

	Oxide Fuel	Metal Fuel	Nitride Fuel
Neutron spectrum hardening	No	Yes	No
Engineering experience	High	High	Low
F.P. Holding Capability	High	Medium	n/a
Fissile Density	Low	High	High
Typical Form	UO_2/PuO_2	U-Pu-Zr	U-Pu-N
Inherent Safety	Good	Excellent	Good
Thermal endurance	High	Low	Medium
Burnup	High	Medium	Low
Erosive endurance/ Rx Internal Material Compatibility	High	Medium	High
Compatible Fuel Cycle	Flexible	Proliferation-resistent emote recycling	Pyrometallugical
Melting temp.	High	High (with Zr)	High
Handling/fabrication	Good	Good	Good
Thermal Conductivity	Low	High	High
Chemical stability	High	Medium	Low
Possible MFR (example)	BN600, BN350	AFR300, S-PRISM	SP-100, PBWR

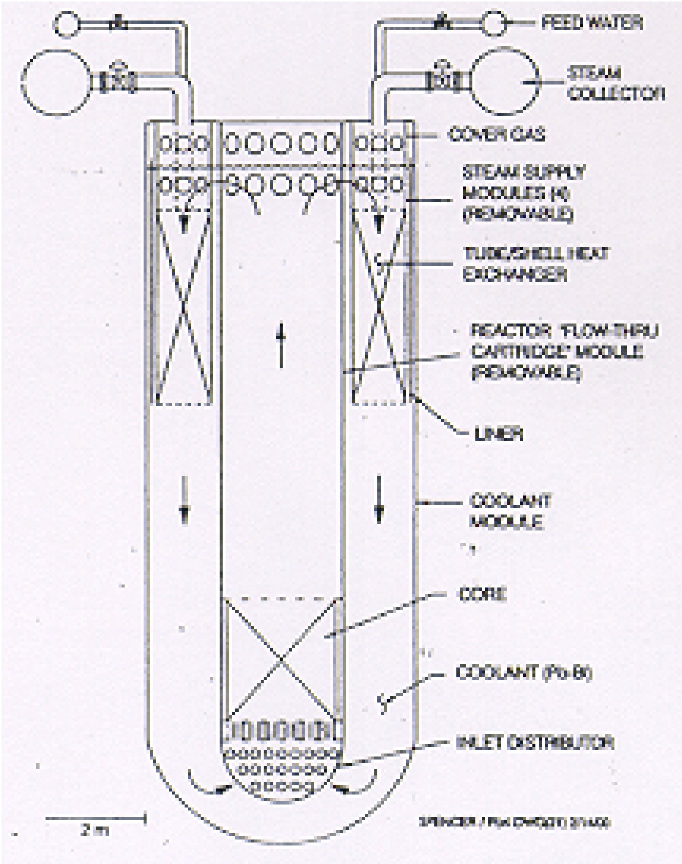

Figure 1. STAR-LM Design Concept

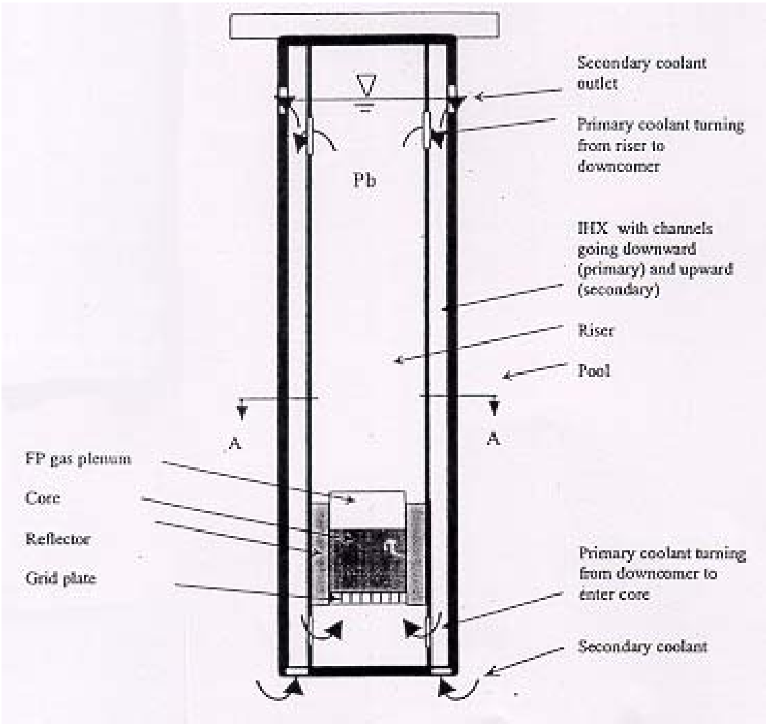

Figure 2. ENHS Design Concept

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Abstract (15-20 Lines)

Recent Liquid Metal Cooled Fast Reactor (LFR) concepts are reviewed for investigating the potential usability of the Dry Reprocess Fuel (DRF). The LFRs have been categorized into two different types: the sodium cooled and the lead cooled systems. In each category, overall design and engineering concepts are collected which includes those of S-PRISM, AFR300, STAR, ENHS and more. Specially, the nuclear fuel types which can be used in these LFRs, have been summarized and their thermal, physical and neutronic characteristics are tabulated. This study does not suggest the best-matching LFR for the DRF, but shows good possibility that the DRF fuel can be used in future LFRs.

Subject Keywords (About 10 words)	fuel, reactor, dry reprocess, liquid metal, sodium, lead, neutron

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초록 (15-20줄내외)

최근의 액금로의 개발방향은 핵연료 재처리 기술과의 관련이 깊다. 현재 세계의 핵연료 재처리 기술은 핵비확산성 및 장주기 방사성 물질의 감소라는 목적을 가지고 있는데 이는 Gen-IV에서 제시하는 목표와 부합한다.
본 연구에서는 건식재처리 핵연료의 액금로에의 활용성을 위하여 최근에 개발되고 있는 주요한 액금로 개념을 정리하였다. 산화물, 금속 및 질화물 핵연료의 물리화학, 중성자물리적 특성을 상호 비교하였다. 건식재가공 핵연료는 산화물 핵연료로서 고온에서 열화학적으로 매우 안정적인 특성을 장점으로하여 미래의 액금로에 활용성이 높을 것으로 기대된다.

주제명키워드 (10단어내외)	핵연료, 액금로, 건식재가공, 소듐, 납