Practical Aspects of Liquid-Salt-Cooled Fast-Neutron Reactors

Charles W. Forsberg
Oak Ridge National Laboratory[*]
P.O. Box 2008
Oak Ridge, TN 37831-6165
Tel: (865) 574-6783
Fax: (865) 574-0382
E-mail: forsbergcw@ornl.gov

Per F. Peterson
University of California, Berkeley
4153 Etcheverry, Berkeley, CA 94720-1730
Tel: (510) 643-7749
E-mail: peterson@nuc.berkeley.edu

David F. Williams
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6170
Tel: (865) 574-5769
E-mail: williamsdf2@ornl.gov

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Charles W. Forsberg
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6165
forsbergcw@ornl.gov

Per F. Peterson
U. of California, Berkeley
4153 Etcheverry
Berkeley, CA 94720-1730
peterson@nuc.berkeley.edu

David F. Williams
Oak Ridge National Laboratory
P. O. Box 2008
Oak Ridge, TN 37831-6170
williamsdf2@ornl.gov

Abstract — *For a century, the only utility-scale technology to convert heat to electricity has been the steam turbine, which has an upper practical temperature limit of about 550°C. The last decade has seen the successful deployment of higher-temperature utility Brayton power systems that enable efficient conversion of higher-temperature heat to electricity. This technology creates an incentive to consider higher-temperature reactors for electricity production. Sodium-cooled fast reactors (SFRs) are capable of producing nuclear fuel and destroying long-lived wastes; however, their capital cost is greater than that of light-water reactors (LWRs). One approach to improve the economics of fast reactors is to develop more efficient higher-temperature fast reactors that can couple to the new Brayton power cycle technology. However, sodium cannot be used as a high-temperature coolant because of its low boiling point. One solution is to replace the sodium coolant with a liquid fluoride salt. Such salts have boiling points above 1200ºC. This paper describes preliminary investigations into the practicality of a 700 to 800°C liquid-salt-cooled fast reactor (LSFR) and examines plant design, salt selection, materials, and economics. The analysis indicates that the capital costs for LSFRs may be significantly lower than those for SFRs and lower than those for LWRs. However, significant technical challenges remain in LSFR development.*

I. INTRODUCTION

The last decade has seen the successful deployment of large, efficient, low-cost high-temperature Brayton power systems—typically using natural gas as a fuel. This advance has major implications for nuclear energy. Many high-temperature reactors have been developed; however, the economics have been unfavorable. One reason is the use of the steam turbine, which has been limited to a practical peak temperature of ~550ºC. The economic incentive to produce high-temperature heat is limited if the heat cannot be efficiently converted to electricity. The development of higher-temperature Brayton cycles (with or without bottoming steam cycles) now provides the technology to efficiently convert high-temperature heat to electricity.

With the development of Brayton power technology, it is appropriate to ask if a high-temperature fast reactor can be developed that can be coupled to high-temperature Brayton power cycles. Fast reactors have the advantages of producing more fuel than they consume (breed) and having the capacity destroy long-lived radionuclides. This paper examines the practicality of a liquid-salt-cooled fast reactor (LSFR). Because of the lower boiling point of sodium (883ºC), the temperatures of traditional sodium-cooled fast reactors (SFRs) cannot be increased and thus cannot be coupled to the new higher-temperature Brayton power cycles.

Nuclear reactor types can be classified by power output and the peak temperatures of their coolants (Fig. 1). Light-water reactors (LWRs), such as the General Electric Economic Simplified Boiling Water Reactor (ESBWR), are low-temperature, high-pressure reactors. Traditional fast reactors cooled with liquid sodium operate at medium temperatures and low pressures. Two options exist for high-temperature reactor coolants: (1) high-pressure gases and (2) low-pressure liquids with boiling points above the peak coolant temperatures.

Helium is the traditional high-temperature, high-pressure gas coolant. For most applications, helium-cooled reactors are modular reactors with relatively small power output. For higher-temperature nuclear applications, liquid fluoride salts with low nuclear cross sections offer potential advantages as a high-temperature, low-pressure liquid coolant. Liquid salts were originally developed in the 1950s and 1960s for the Aircraft Nuclear Propulsion Program and the Molten Salt Breeder Reactor (MSBR) Program. These programs operated various salt test loops for several hundred-thousand hours and produced over 1000 technical reports. (The chemical industry uses a variety of liquid salts in heat transport systems but, at somewhat lower temperatures.) Liquid fluoride salts can be used for multiple nuclear applications.

- *Salt heat-transport systems*. Liquid salts are being considered for transport of heat from high temperature nuclear reactors to hydrogen production plants. The salts have several advantages relative to other coolants (Table I): low pressure, low pumping costs, and small equipment sizes (very high volumetric heat capacity).

- *Advanced High-Temperature Reactor (AHTR).* The AHTR (Fig. 2) combines four existing technologies in a new way[1]: (1) coated-particle graphite-matrix nuclear fuels (traditionally used for helium-cooled reactors and compatible with liquid fluoride salts), (2) Brayton power cycles, (3) passive safety systems and plant designs from liquid-metal-cooled fast reactors, and (4) low-pressure liquid-salt coolants. The heat from the reactor core is carried by the clean salt to an intermediate heat exchanger. The intermediate liquid-salt heat-transport loop transfers the heat to a Brayton power cycle. Because the AHTR integrates existing technologies and the demonstrated compatibility of graphite fuels with liquid salts, it is the near-term option for liquid-salt-cooled reactors.

- *LSFR.* LSFRs (described in this paper) combine the AHTR plant design (Fig. 2) with traditional metal-clad-fuel fast-reactor cores.

- *Molten salt reactor (MSR).* The MSR is a liquid-fuel reactor in which uranium, fission products, and actinides are dissolved in a molten fluoride salt. Historically, the term *molten salt* has been associated with MSRs, in which the fuel is dissolved in a liquid salt. To avoid possible confusion, we use the term *liquid salt* when referring to a clean salt (such as that proposed as the coolant for the LSFR and AHTR) and the term *molten salt* or *fuel salt* when referring to a salt that contains uranium and fission products in high concentrations and is also used as a coolant. In the MSR, the fuel salt flows through a graphite reactor core, which acts as a moderator, to an intermediate heat exchanger and back to the reactor core. This reactor concept was partly developed in the 1970s, and two test reactors were successfully built.[2]

- *Fusion reactors.* Liquid salts are major candidates for cooling inertial and magnetic fusion energy systems.[3]

This body of research provides the basis for consideration of an LSFR.

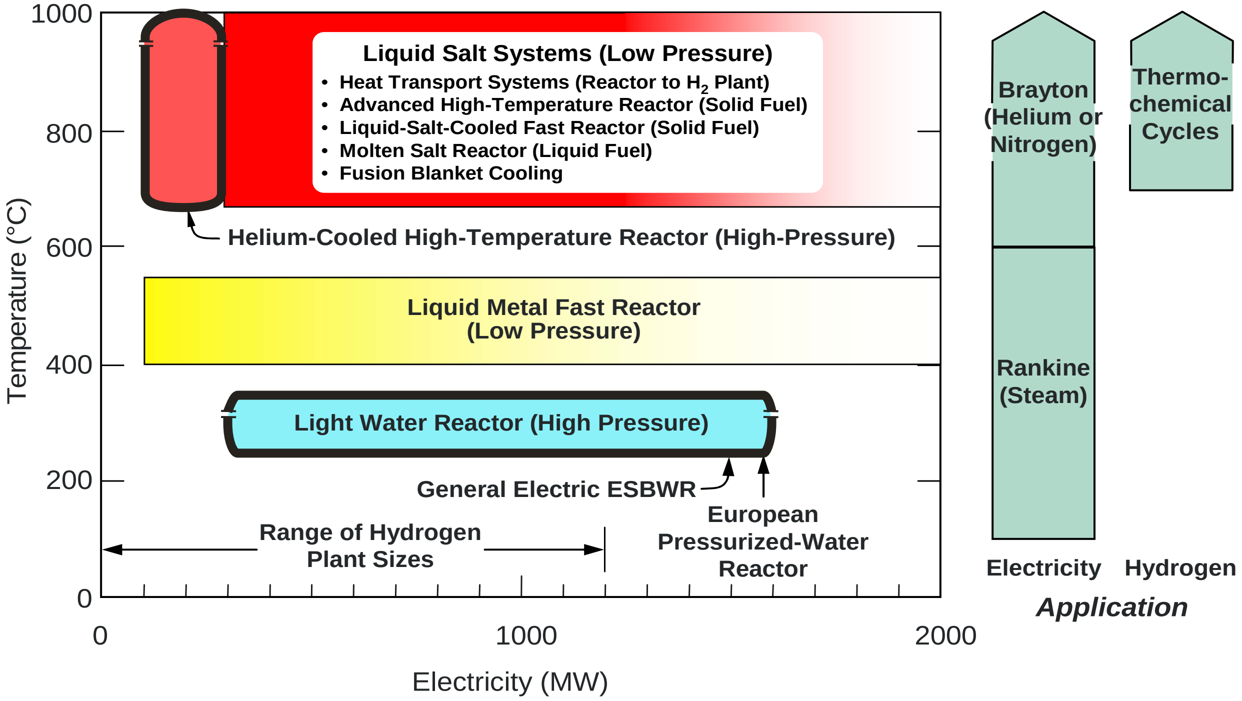

Fig. 1. Reactor type vs temperature and power output.

Table I. Characteristics of reactor coolants[a]

Coolant	T_{melt} (°C)	T_{boil} (°C)	ρ (kg/m^3)	C_p (kJ/kg °C)	ρC_p (kJ/m^3 °C)	K (W/m °C)	$v \cdot 10^6$ (m^2/s)	N_ρ (mol/cm^3)
Li_2BeF_4 (Flibe)	459	1,430	1,940	2.34	4,540	1.0	2.9	0.137
0.58NaF-0.42ZrF_4	500	1,290	3,140	1.17	3,670	~1	0.53	0.108
Sodium	97.8	883	790	1.27	1,000	62	0.25	0.034
Lead	328	1,750	10,540	0.16	1,700	16	0.13	0.051
Helium (7.5 MPa)			3.8	5.2	20	0.29	11.0	0.001
Water (7.5 MPa)	0	100	732	5.5	4,040	0.56	0.13	0.122

[a] Nomenclature: ρ is density; C_p is specific heat; k is thermal conductivity; v is viscosity; and N_ρ is number density.

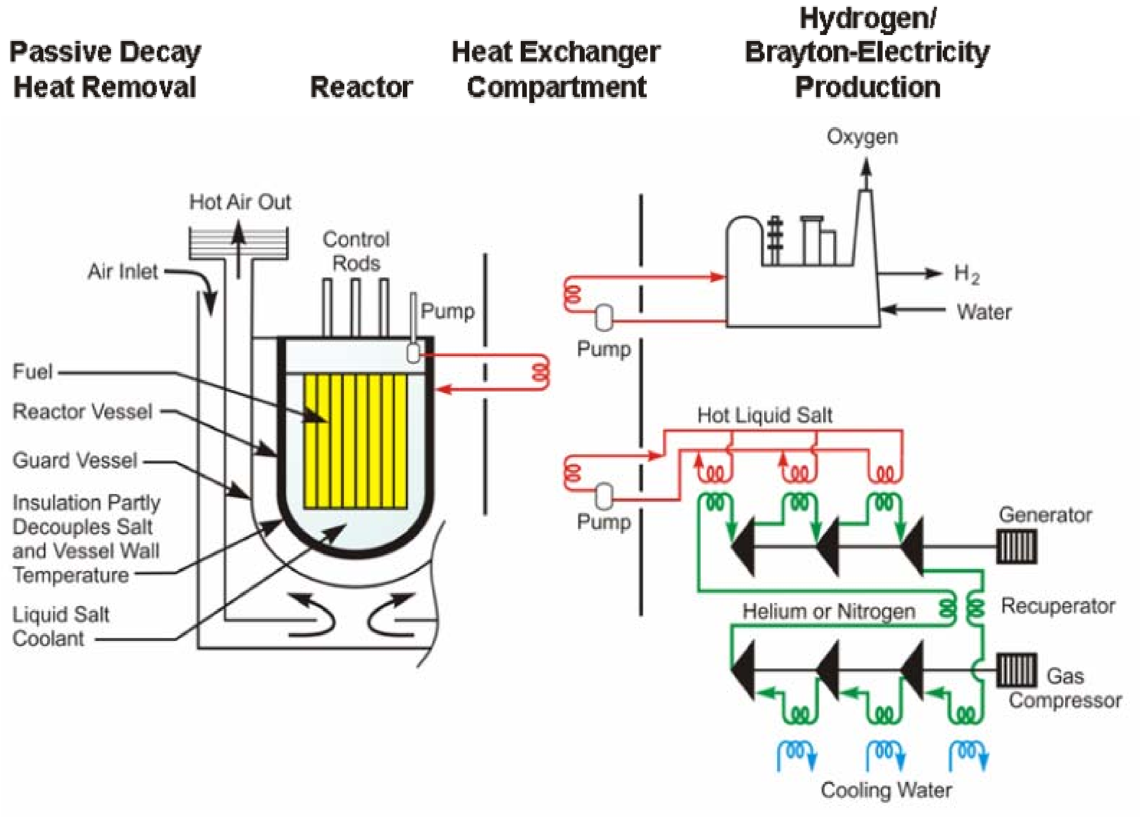

Fig. 2. Schematic of Advanced High-Temperature Reactor and salt-cooled fast reactor.

II. LSFR FACILITY DESIGN AND ECONOMICS

SFRs have many attractive features, such as the capability to produce fuel and destroy long-lived radioactive wastes. However, the projected capital costs per kilowatt (electric) are greater than those of LWRs because of the more complex design. A new fast reactor coolant is of interest only if there is a reasonable potential for significantly better economics than those of SFRs. Economics depends primarily upon the facility design.

A conceptual facility design for the LSFR was developed based on the General Electric S-PRISM 1000-MW(t) [380-MW(e)] SFR. The facility design (Fig. 2) is similar to that for the S-PRISM except for the changes required when (1) the sodium coolant is changed to a salt coolant and (2) the operating temperatures are increased. Major changes include using a Brayton power cycle, which replaces the lower-temperature steam cycle; moving the intermediate heat exchangers outside the reactor vessel; moving spent nuclear fuel (SNF) storage from inside the vessel to a separate SNF storage facility; and making changes in the materials of construction. The reactor vessel size remains unchanged. The consequences of these changes, as will be explained, are to increase the thermal power output to ~2400 MW(t) and the electrical power output to ~1200 MW(e) in the same size facility. The S-PRISM was used as a starting point for analysis because it was the result of the last major SFR development program in the United States. Many other SFR plant designs could be used as a starting point for the development of a LSFR.

The LSFR facility design is based on previous work on the AHTR.[4-6] Because the LSFR is a low-pressure high-temperature liquid-salt-cooled fast reactor, its facility design will be very similar to that of a low-pressure, high-temperature liquid-salt-cooled AHTR. In effect, the facility designs for a LSFR and the AHTR are almost identical and are both based on the S-PRISM. The primary differences are in the reactor core.

II.A. Facility Design

The reactor core is a variant of traditional metal-clad fast reactor cores, except that the fuel must operate at higher temperatures in a liquid-salt environment. The optically transparent liquid salt coolant is a mixture of fluoride salts with freezing points near 400°C and an atmospheric boiling point of ~1400°C. Several different salts are being evaluated as the primary coolant. The reactor operates at near-atmospheric pressure. At operating conditions, the liquid-salt heat-transfer properties are similar to those of water. Heat is transferred from the reactor core by the primary liquid-salt coolant to an intermediate heat-transfer loop. The intermediate heat-transfer loop uses a secondary liquid-salt coolant to move the heat to a thermochemical hydrogen production facility to produce hydrogen or to a turbine hall to produce electricity. If electricity is produced, a multi-reheat nitrogen or helium Brayton power cycle (with or without a steam bottoming cycle) is used. The intermediate loop serves several functions: (1) it isolates the high-pressure Brayton cycle from the low-pressure reactor, (2) it isolates the nuclear island and activated salt coolant from the non-nuclear Brayton power cycle, and (3) it allows the design of an efficient Brayton cycle with the salt-gas heat exchangers located next to the Brayton cycle turbines inside the turbine hall.

The LSFR facility layout (Fig. 2) is similar to that of the S-PRISM SFR designed by General Electric. The 9.2-m-diam vessel is the same size as that used by the S-PRISM. Three peak coolant temperatures were evaluated: 705, 800, and 1000ºC. If the reactor power output is 2400 MW(t), the respective electricity production capacities are 1151, 1235, and 1357 MW(e). Metallurgical limits of near-term materials will limit peak coolant temperatures to somewhere between 700 and 800ºC.

The coolant pumps and their intakes are located above the reactor core with appropriate siphon breakers; thus, the reactor cannot lose its coolant except by failure of the primary vessel. The guard vessel is sized so that even if the primary vessel fails, the core remains covered with salt. With advanced heat exchangers, it may be possible to include the primary salt to intermediate salt heat exchangers inside the reactor vessel—like the S-PRISM.

When a reactor shuts down, radioactive decay heat continues to be generated in the reactor core at a rate that decreases over time. The LSFR (Fig. 2) uses passive reactor vessel auxiliary cooling (RVAC) systems developed for decay heat removal in the General Electric sodium-cooled S-PRISM. The reactor and decay-heat-cooling system are located in a below-grade silo. In the LSFR, RVAC system decay heat is (1) transferred from the reactor core to the reactor vessel by natural circulation of the liquid salts, (2) transferred across an argon gap by radiation to a guard vessel, (3) conducted through the guard vessel, and then (4) removed from outside of the guard vessel by natural circulation of ambient air. The rate of heat removal is controlled primarily by the radiative heat transfer through the argon gas from the reactor vessel to the guard vessel. Radiative heat transfer increases by the temperature to the fourth power (T^4); thus, a small rise in the reactor vessel temperature (as would occur upon the loss of normal decay-heat-removal systems) greatly increases heat transfer out of the system. For an accident involving loss of decay heat removal, the reactor vessel can dump the decay heat from a 2400-MW(t) core with a peak reactor vessel temperature of ~750°C.

In the S-PRISM, the reactor vessel size was limited by practical fabrication and design considerations to a diameter of 9.2 m. To avoid the potential of boiling sodium, the sodium-coolant temperature limits were about ~550ºC. Given the fixed vessel size and the upper temperature limit of the sodium, the decay heat from a 1000-MW(t) reactor core could be removed. Liquid salts have boiling points over 1200ºC, which allows the reactor to operate at higher temperatures. This enables the same size reactor vessel and decay-heat-cooling system to reject the decay heat from a 2400-MW(t) core. The cost for this gain in decay heat removal is the requirement for better materials of vessel construction to allow vessel temperatures to 750ºC.

II.B. Capital Cost Estimates

Capital cost estimates were made for a 2400-MW(t) AHTR relative to those for the four-module General Electric S-PRISM SFR. Because the LSFR and the AHTR have essentially the same facility design, those cost estimates[6] were used to provide an initial capital cost estimate for an LSFR. The primary differences between the LSFR and S-PRISM are (1) a single reactor with an output of 2400 MW(t) [1235 MW(e)], versus four modules each with outputs of 1000 MW(t) [380 MW(e)]; (2) separate storage of the SNF, versus storing SNF in the reactor vessel; and

(3) higher operating temperatures with the use of the higher-efficiency Brayton power cycles. The S-PRISM stores SNF in the reactor vessel. To provide space for the larger LSFR core, a separate SNF storage system is required.

Preliminary overnight capital costs[6] for several exit temperatures were determined relative to the S-PRISM SFR. For a salt-cooled reactor with an exit coolant temperature of 800ºC, the capital cost [$930/kW(e)] was 55% of that of the S-PRISM per kilowatt (electric). There are several reasons for these major capital cost savings.

- *Higher efficiency*. The higher temperature implies higher efficiency (51 vs 38%). This results in lower costs per kilowatt (electric) because of the smaller power conversion equipment, cooling systems to reject heat from the power cycle, and decay-heat-removal systems.

- *Passive decay heat removal*. The salt-cooled and sodium-cooled reactors use the same decay-heat-removal systems. Heat rejection is dependent upon the vessel size and peak vessel temperature. The reactor vessel size is the same for both concepts. However, a salt-cooled reactor operates at higher temperatures. By allowing the peak vessel temperature in an accident to increase from 550 to 750ºC, the same size vessel used in a 1000-MW(t) reactor can reject the decay heat from a 2400-MW(t) reactor. Although the nuclear island size does not change, the power output is much larger. The added cost is the requirement to use higher-temperature materials for the vessel and core components.

- *Reduced containment requirements*. The liquid-salt coolant coupled to a Brayton power cycle prevents any sodium–water interactions. Liquid salts do not react with air and only slowly react with water. This eliminates the highly energetic chemical accidents associated with sodium safety systems.

- *Reduced equipment sizes*. Volumetric heat capacities (Table I) for liquid salts are several times larger than those for sodium and double those of lead. This reduces the size of pipes, valves, and heat exchangers per unit of energy transferred and implies smaller equipment for a 2400-MW(t) liquid-salt-cooled reactor than for a 1000-MW(t) sodium-cooled reactor.

- *Transparent coolant*. Unlike liquid metals, liquid salts are transparent. This simplifies refueling, maintenance, and inspection of the primary system.

Any advanced reactor must compete with LWRs. Preliminary studies have compared the quantities of materials required to build various advanced reactors.[7] As shown in Fig. 3, the initial estimates of AHTR materials requirements are about half those for the newest LWR reactor concepts. An LSFR would be expected to have similar materials requirements. *There are large uncertainties in these estimates because of the early state of technological development. The potentially favorable economics are a consequence of coupling a large passively-safe high-temperature reactor with a low-pressure high-volumetric-heat-capacity coolant to an efficient high-temperature Brayton power cycle.*

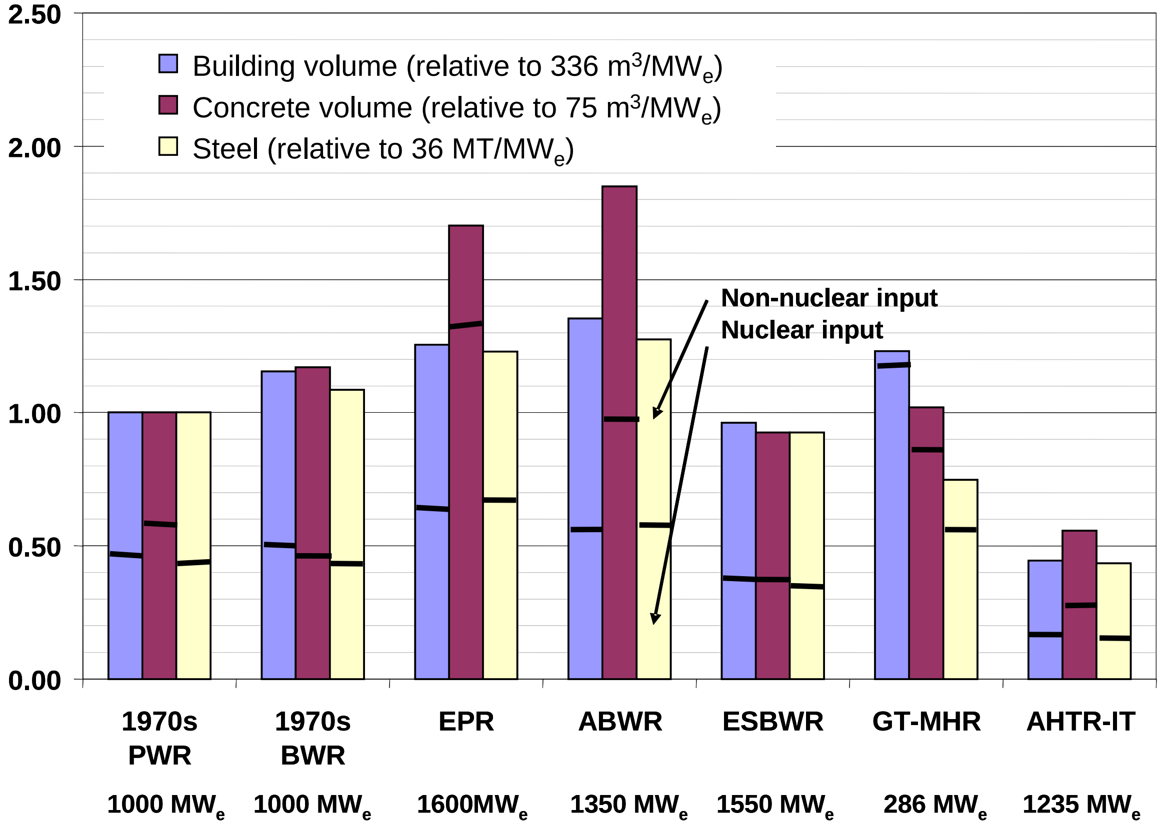

Fig. 3. Relative quantities of materials per unit power output to construct various types and generations of reactors. (PWR: pressurized-water reactor; BWR: boiling-water reactor; EPR: Framatome European PWR; ABWR: GE Advanced BWR [in licensing]; GT-MHR: General Atomics Gas-turbine Modular Helium Reactor [proposed]; and AHTR: Advanced High-Temperature Reactor)

Table II. Physical properties of selected molten salts

Salt (mol %)	Form. wt (g/mol)	Melt. pt. (°C)	Density (g/cm^3), T(°C)	700°C Heat capacity (cal/g-°C)	Viscosity (cP), T(K)	Thermal cond. (W/cm-°C)
Alkali-Fluorides (IA) : Nontoxic						
LiF-NaF-KF (46.5-11.5-42)	41.2	454	2.53–7.3^{E}-4*T	0.45	0.04exp(4170/T)	0.006-0.01
LiF-RbF (43-57)	70.7	475	3.30–6.9^{E}-4*T	0.284 (est)	0.021exp(4678/T)	~ 0.06
Alkali + Alkaline Earth Fluorides (IA + IIA)						
LiF-BeF$_2$ (66-34)	33.1	458	2.28–4.884E-4*T	0.57	0.116exp(3755/T)	0.011
NaF-BeF$_2$ (57-43)	44.1	360	2.27–3.7E-4*T	0.52	0.034exp(5164/T)	~ 0.01
Alkali + ZrF$_4$: Nontoxic and Low Tritium Yield						
NaF-ZrF$_4$ (50-50)	104.6	510	3.79 –9.3E-4*T	0.28	0.071exp(4168/T)	~0.01 (est)
NaF-KF-ZrF$_4$ (10-48-42)	102.3	385	3.45–8.9E-4*T (est.)	0.26 (est.)	0.061exp(3171/T) (est.)	~0.01

III. TECHNOLOGICAL CONSIDERATIONS FOR AN LSFR

The primary difference between an LSFR and an SFR is the use of a liquid-salt coolant. Table I compares the properties of various reactor coolants, including the two molten salts that were used in MSRs (see below). Unlike other coolants (water, sodium, or helium), fluoride liquid and molten salts are a family of coolants with similar general properties. The choice of a specific molten salt for a specific application is determined by functional requirements and costs. Many salts have been examined. Table II lists several leading candidates for various nuclear applications and their key physical properties. The remainder of this section discusses the choices of salts and the constraints that limit the choice of salt.

III.A. Existing Knowledge Base

In the 1950s, at the beginning of the Cold War, the United States launched a large program to develop a nuclear-powered aircraft. MSRs were to provide the heat source, with the heat transferred to a jet engine via an intermediate heat-transport loop. In the 1960s and 1970s, the MSR was investigated as a thermal-neutron breeder reactor.[2] These billion-dollar programs developed the technology base for use of liquid salts in nuclear systems. Two experimental reactors were built and successfully operated. The Aircraft Reactor Experiment, the first MSR, was a 2.5-MW(t) reactor that was operated in 1954 at a peak temperature of 860ºC and used a sodium-zirconium-uranium fluoride salt. This was followed in 1965 by the Molten Salt Reactor Experiment (MSRE), an 8-MW(t) reactor that used a lithium-beryllium fluoride salt and demonstrated most of the key technologies for a power reactor. In addition, test loops with liquid salts were operated for hundreds of thousands of hours, materials of construction were code qualified to 750ºC, and a detailed conceptual design of a 1000-MW(e) MSBR was developed. Over 1000 technical reports were produced.

Unlike the MSR, an LSFR uses solid fuel and a clean liquid salt as a coolant (i.e., a coolant with no dissolved fissile materials or fission products). This has two major implications.

- *Salt selection*. For the MSR, a major constraint was the requirement for high solubility of fissile materials and fission products in the salt. For the LSFR, this requirement does not exist. A wider choice of fluoride salts can be considered.

- *Materials selection*. The corrosive characteristics of liquid salts are dependent upon the major salt constituents and the impurities in the salt. The coolant salt can be prepared and maintained so the impurities do not control the corrosion response. It is expected that coolant salts can be used at significantly higher temperatures than were established in the MSR program because of the different corrosion characteristics of a clean liquid salt versus a molten salt with uranium, actinide, and fission product fluorides. A wider range of material options also exist. The presence of uranium dissolved in the salt was always found to accelerate corrosion, and there exist additional methods to prevent corrosion when uranium is not present in the salt.

III.B. Salt-Metal Constraints

Fluoride salts are fluxing agents that rapidly dissolve protective layers of oxides. After carbon, borides and nitrides appear to be the most compatible nonmetallic materials. To avoid corrosion, liquid-salt coolants must be chosen that are thermodynamically stable relative to the materials of construction of the reactor; that is, the materials of construction are chemically noble relative to the salts. This limits the choice to highly thermodynamically-stable salts. Table III shows candidate fluorides suitable for a liquid salt and their thermodynamic free energies of formation. The general rule to ensure that the materials of construction are compatible (noble) with respect to the salt is that the difference in the Gibbs free energy of formation between the salt and the container material should be >20 kcal/(mol. ºC). The corrosion strategy is the same as that used in SFRs, where the materials of construction are noble relative to metallic sodium.

Table III. Thermodynamic stability of components of molten salts and structural materials

Constituent	Free energy of formation at 1000ºK (kcal/mol-F)	Cation thermal capture cross section (barns)
	Majority Constituents (>99.9 mol %)	
^{7}LiF	-125	0.033 (^{7}Li)
MgF_2	-113	0.063
NaF	-112	0.52
RbF	-112	0.70
KF	-109	2.1
BeF_2	-104	0.01
ZrF_4	-94	0.18
$^{11}BF_3$	~-95	0.05 (^{11}B)
$A1F_3$	-90	0.23
F	N	0.01
	Structural Metal Constituents (Trace)	
CrF_2	-75.2	3.0
FeF_2	-66.5	2.5
NiF_2	-55.3	4.5
MoF_6	-50.9	2.5

In high-temperature systems, there are also temperature-induced mechanisms for corrosion that are dependent upon large temperature differences in the heat transport system. The classic example is the transport of carbon in the form of various oxides in gas-cooled systems, resulting in depletion of carbon in one part of the system and the deposition of carbon in another part. Parallel mechanisms have been seen in liquid-salt test loops. In this system, the primary concern has been the selective chemical transport of chromium (a constituent of many alloys) from hot to cold locations in the reactor system by uranium and other multi-valence species due to a redox equilibrium that is temperature dependent:

$$2UF_4 + Cr = 2UF_3 + CrF_2$$

For most alloys of interest, chromium is the least thermodynamically stable element among the materials of construction. This and other corrosion mechanisms were studied and resulted in the development of an optimized alloy, Hastelloy-N, as a material of construction. This alloy offers very good corrosion resistance in MSR systems.

Avoiding corrosion in an MSR with metallic components is significantly more challenging than avoiding corrosion in clean-salt-coolant applications (heat transport loops, AHTR, LSFR, and fusion). In an MSR, the dissolved uranium and other such species in the fuel salt results in the presence of additional corrosion mechanisms that can limit the useful service temperature of an alloy. In clean-salt applications, these types of corrosion mechanisms can be reduced or eliminated by (1) using purified salts that do not contain chemical species that can transport chromium and other alloy constituents or (2) operating under chemically reducing conditions. Under chemically reducing conditions, chromium fluoride has an extremely low solubility, which limits chromium transport. In the MSRE, almost no corrosion was observed in the intermediate heat-transport loop with its clean salt operating at 650ºC. Higher-temperature (815ºC) test loops[8] indicate low corrosion rates with liquid fluoride salts when these conditions are met. The practical engineering conclusion is that the development and qualification of metallic materials for higher-temperature clean-liquid-salt systems (>750°C) will be significantly easier than for a higher-temperature MSR with its fuel salt. Although the mechanisms of corrosion are quite different, liquid-salt systems require careful control of coolant chemistry just as conventional LWRs do.

III.C. Salt Properties and Selection

Liquid fluoride salts do not react with helium or nitrogen but will react slowly with water. Liquid salts have been injected into water with no violent reactions. Unlike water, fluoride salts do not undergo radiolysis in radiation fields when in a liquid state and no fluorine will be generated. No radiolysis was detected in flowing loops of molten salt operated in intense radiation fields of the Materials Testing Reactor. In irradiated solid salts at low temperatures, radiolysis can occur. The toxicity of the molten fluoride coolant depends upon the specific salt and varies from the fluoride salts used in toothpaste for prevention of tooth decay to toxic materials. Some salts produce tritium under radiation (lithium and beryllium). Other salts such as sodium produce short-lived gamma-emitting radionuclides. The choice of coolant impacts the need for tritium control systems and radiation shielding in the primary system.

The neutron-absorption cross sections of any liquid salt for reactor applications must be low to avoid excessive parasitic capture of neutrons. While this is less of a restriction for a fast reactor than for a thermal reactor, it is still important. For thermal and intermediate neutron-spectrum reactors, this probably eliminates chloride salts with their higher nuclear cross sections, even if the high-cross-section ^{35}Cl is removed. Only fluoride salts are practical candidates. A wide variety of salt constituents have low cross sections; however, the realistic candidates are also restricted by the requirements of thermodynamic stability to ensure viable materials of construction for the container. Table II shows several classes of salt coolant options. If a lithium or boron salt is used, isotopically separated lithium or boron is required to have a salt with a low absorption cross section.

The chemical and nuclear criteria define the allowable elements for a liquid salt coolant. Physical property requirements are used to define the candidate liquid salts for specific applications. The requirements include (1) a good coolant, (2) low coolant freezing points, and (3) application-specific requirements. In all cases, binary or more complex fluoride salt mixtures are preferred because the melting points of fluoride salt mixtures are much lower than those for single-component salts. With some three-component mixtures such as ^{7}LiF-BeF$_2$-NaF, and potentially with four-component mixtures, it is possible to reduce melting points to ~300EC.

IV. LIQUID-SALT COOLANT

Detailed studies will be required to determine the optimum salt compositions. This involves complex trade-offs with considerations of neutronics, thermal hydraulics (including performance under natural circulation), freezing point (ease of refueling), vapor pressures, and costs. For a fast reactor, it is desirable to avoid low-Z materials that can degrade the neutron spectrum. Based on a preliminary examination of the options, the leading candidates include the alkali-zirconium fluoride mixtures such as NaF-KF-Zr$_4$, which has a melting point of 385°C. Other candidates with somewhat different nuclear properties include NaF (6.2%)-RbF (45.8%)-ZrF$_4$ (48%), with a melting point of 380°C; NaF (8%)-RbF (50%)-ZrF$_4$ (42%), with a melting point of 400°C and a significantly lower vapor pressure; and RbF (52%)-ZrF$_4$ (48%) and a melting point of 390°C. Other candidates include the better-understood (but perhaps not as attractive) salts such as the two higher-melting-point salts used in the two molten salt test reactors (Table II) and ^{7}LiF-NaF-KF, with its melting point of 454°C.

V. METALS OF CONSTRUCTION

The MSR programs have code-qualified Hastelloy-N to 750°C as a material of construction for liquid salt operations. However, there are no metallic components in an MSR core. The MSR core consists of graphite moderator blocks with a fuel salt flowing through the reactor core. Graphite is fully compatible with fluoride salts. While Hastelloy-N, or another nickel-based alloy, is clearly suitable for heat exchangers and other equipment, it may or may not be suitable for LSFR in-core components (structure and fuel cladding) which will be subject to higher temperatures and receive large fast-neutron doses in the reactor core. The metal in-core components are likely to be the primary technical challenge for an LSFR with the triple requirements for higher-temperature service, resistance to neutron radiation damage, and corrosion resistance to liquid salts. The practical metal systems are based on iron, nickel, or molybdenum. The use of binary metallic materials (either clad or coated) may be desirable for some applications (including the reactor vessel) in order to confer appropriate strength and corrosion resistance.

V.A. Iron-Based Alloys

To overcome the historic temperature limits of iron-based systems, there has been significant developmental work on oxide-dispersion-system (ODS) iron alloys for fast reactors.[9] These alloys contain rare-earth oxides such as yttrium oxide that enable iron alloys to maintain strength up to 80% of their melting point versus 50% for traditional alloys. The alloys have good radiation resistance. The primary LSFR question associated with these and other iron alloys is their long-term corrosion resistance in liquid salts. The limited corrosion testing of iron-based alloys in clean liquid fluoride salts indicates the potential for corrosion-resistant iron-based systems. However, more corrosion testing will be required before there will be high confidence in such an approach.

V.B. Nickel-Based Alloys

The traditional high-temperature alloys are nickel based, including the traditional superalloys. Candidate alloys for high-temperature service in liquid salts (but not as incore components) have been evaluated as part of the AHTR research and development (R&D) activities[6]. Some of these alloys are known to have excellent chemical compatibility with liquid salts; however, there is mixed experience with the irradiation performance of nickel alloys. British[10-11] experience with PE-16, a nickel alloy (17% Cr, 43% Ni, 3% Mo, 2.5% Ti+Al, remainder Fe) in the Prototype Fast Reactor was very good, but at lower temperatures than required for an economic LSFR. [Alloy compositions are reported in weight % whereas salt compositions are in mole %.] At the same time, many nickel alloys have shown two weaknesses in high radiation fields. The potential of nickel alloys at the higher temperatures for use in a LSFR is not well understood.

- *Strength*. The strength of many nickel alloys is a consequence of nickel-silicon precipitates. In irradiation fields, these precipitates can dissolve, with the silicon migrating to the grain boundaries and causing the alloy to weaken. For these alloys, it may be feasible to overcome this difficulty by the development of ODS nickel alloys. However, only very limited work has been done on these systems. For nonnuclear applications, the nickel superalloys are easier to manufacture and thus became the preferred class of high-temperature nickel alloys. In contrast, there were no equivalent classes of iron superalloys. As a result, ODS iron alloys were developed for higher-temperature applications.

- *Helium*. The second constraint is the production of helium in nickel alloys ($^{58}\mathrm{Ni}(n,\gamma)\rightarrow{}^{59}\mathrm{Ni}(n,\alpha)\rightarrow{}^{56}\mathrm{Fe}$), which at higher temperatures migrates to the grain boundaries and weakens the alloy. The rate of helium production is strongly dependent upon the thermal neutron flux. Such alloys have limited lifetimes in thermal neutron reactors but may be viable in fast reactors. Fusion energy programs are examining various alloy additives to trap helium in the metal matrix and thus avoid this failure mechanism.

V.C. Molybdenum-Based Alloys

Molybdenum alloys are compatible with liquid salts, have good thermophysical properties, and have been successfully tested as cermet fuels.[12] Molybdenum has a very high melting point (2600ºC), a high thermal conductivity, and a moderate thermal neutron cross section (2.65 barns). However, isotopically separated molybdenum[12] with its very low nuclear cross section is an option. There are significant challenges with molybdenum alloys: (1) such alloys are difficult to weld, (2) the fracture toughness is somewhat low with concerns about radiation embrittlement, and (3) high temperature oxidizing conditions must be avoided because MoO_3 has a melting point of 795ºC. The potential oxidation should not be a significant concern for an LSFR because the salt (like sodium) will be subjected to chemically reducing conditions. The fracture toughness is primary a concern at lower temperatures. Radiation damage is temperature dependent and is minimized by operating at higher temperatures in the range of 700 to 1000ºC. Molybdenum-based alloys may ultimately allow the construction of a very high temperature LSFR and is a class of materials where higher temperatures improve material properties.

VI. REACTOR PHYSICS/CORE DESIGN

Fast reactors have been designed with a wide variety of coolants including helium, supercritical water, lithium, sodium, and lead. Limited neutronic studies of salt-cooled fast reactors have been completed[13] for small [125-MW(t)] reactor cores. The studies show the neutronic viability of such a reactor core and have examined several salts including NaF (57%)-BeF_2 (43%), ^{7}LiF (66%)-BeF_2 (34%), and ^{7}LiF (46.5%)-NaF (11.5%)-KF (42%). Several different clad materials were considered including SS304, HT-9, NM-316SS, and Hastelloy-N. The challenge is to have a large reactor core that meets all the economic and technical requirements. The following observations are made.

- *Temperature*. The primary technical challenge is likely to be fuel cladding with the higher temperature being the major challenge. Economics demand coolant exit temperatures above 700°C to enable the use of efficient Brayton power cycles; however, the economic and efficiency incentives decrease with increasing temperature. For peak coolant temperatures of 705, 800, and 1000ºC, the respective thermal-to-electric efficiencies are 48.0, 51.5, and 56.5%. This suggests a goal of peak coolant temperatures between 700 and 800°C for first-generation LSFRs.

- *Neutronics*. A requirement for a fast reactor is to avoid thermalization of the neutron spectrum. This encourages avoidance of low atomic weight elements such as lithium. The atomic weight of fluorine (19) is similar to that of sodium (23). The effect of salt constituents on coolant void worth is another important aspect that will have to be considered in the salt-selection process.

- *Neutronic-hydraulic coupling*. The atom densities of typical liquid salts are about 3 times that of sodium; thus, one might expect added neutron scattering and a more thermalized neutron spectrum. However, the volumetric heat capacity of typical salts is about 3 times that of sodium. This implies that only one-third the volume of liquid salt needs to flow through the core for equivalent removal of heat and the need for only-one third of the volume of coolant in the core. However, other heat transfer constraints and physical requirements for minimum coolant channel diameters exist. The limited studies to date indicate a somewhat softer spectrum for an LSFR relative that for an SFR.[13]

Multiple core design studies will be required before LSFR core design options for large fast reactors are well understood.

VIII. RESEARCH AND DEVELOPMENT NEEDS

The development of a new coolant for fast reactors is a major undertaking. However, relatively modest resources will be required to provide a reasonable assessment of the viability of an LSFR and the development challenges:

- *Incore materials*. An evaluation is required of potential incore material (structure and clad) options with out-of-reactor corrosion testing of the candidate materials. If the tests confirm that some of the existing reasonably well understood structural and clad options do not corrode in flowing liquid-salt systems with high temperature gradients, the largest challenge for a LSFR will have been addressed.

- *Salt evaluations*. Liquid salts were developed for the MSR; however, the requirements for a liquid salt in an LSFR are different than those of an MSR. Different fluoride salts are likely to be preferred. Assessments of alternative salt coolants are required, with better measurements of the physical properties of the leading coolant candidates.

- *Core design*. Detailed core designs are required to understand the potential issues and aid in selection of the preferred liquid salt. This must include assessments of different missions such as low-cost electricity (once-through fuel cycle), fuel production, and actinide burning—similar to recent assessments of lead-cooled fast reactors.[14]

VIII. CONCLUSIONS

The development of large, efficient, industrial Brayton cycles is potentially the enabling technology for more-efficient, lower-cost high-temperature (>700°C) reactors. Liquid salts are a coolant option for high-temperature liquid-cooled reactors. Preliminary economic assessments indicate the potential that an LSFR will have significantly lower capital costs than SFRs and lower costs than LWRs; thus, the LSFR should be carefully examined as a class of fast reactor option. No fundamental feasibility issues have been identified; however, many unknowns still exist.

REFERENCES

1. C. W. FORSBERG, "The Advanced High-Temperature Reactor: High-Temperature Fuel, Liquid-Salt Coolant, and Liquid-Metal Plant," *1st International Conference on Innovative Nuclear Energy Systems for Sustainable Development of the World (COE INES-1)*, Tokyo Institute of Technology, Tokyo, Japan (October 31–November 4, 2004), in press: *Progress in Nuclear Energy* (2004).

2. *Nucl. Appl. Technol.*, **8**(2) (Entire issue) (1970).

3. R. W. MOIR, et al., "HYLIFE-II: A Molten-Salt Inertial Fusion Energy Power Plant Design—Final Report," *Fusion Energy*, **25**, 5–25 (1994).

4. C. W. FORSBERG, P. S. Pickard, and P. F. Peterson, "Molten-Salt-Cooled Advanced High-Temperature Reactor for Production of Hydrogen and Electricity," *Nucl. Technol.* **144**, 289−302 (2003).

5. C. W. FORSBERG, "Reactors with Molten Salts: Options and Missions," *Proc. Frederic Joliot & Otto Han Summer School on Nuclear Reactors: Physics, Fuels, and Systems, Cadarache, France, August 25–September 3, 2004* (2004).

6. D. T. Ingersoll, et al., *Status of Preconceptual Design of the Advanced High-Temperature Reactor*, ORNL/TM-2004/104, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2004).

7. P. F. PETERSON and J. Zhao, "Material Input for Advanced Brayton Cycle Power Conversion Systems," *Trans. 2004 American Nuclear Society Winter Meeting, November 14–18, 2004, Washington, D.C.* (2004).

8. D. F. WILLIAMS, D. F. Wilson, L. M. Toth, J. Caja, and J. K. Keiser, "Research on Molten Fluorides as High-Temperature Heat Transfer Agents," *Global 2003, Embedded Topical Within 2003 American Nuclear Society Winter Meeting, November 16–20, 2003, New Orleans, Louisiana* (2003).

9. Nuclear Energy Research Advisory Committee, *Generation IV Roadmap: Description of Candidate Liquid-Metal-Cooled Reactor Systems,* GIF-017-00, U.S. Department of Energy (December 2002).

10. C. BROWN, R. J. Lilley, and G. C. Crittenden, "PFR's Contribution to Fuel Development: The Nuclear Engineer, 35(4); 122–128 (1994).

11. Y. KONOBEEV and G. A. Birzhevoi, "Prospects for Using High-Nickel Alloys in Power Reactors with Supercritical-Pressure Water," *Atomic Energy*, 96(5); 365–373 (2004).

12. K. BAKKER, F. C. Klaassen, R. P. C. Schram, A. Hogenbrick, R. K. Meulekmp, A. Bos, H. Rakhorst, and C. A. Mol, "The Use of Molybdenum-Based Ceramic-Metal (CerMet) Fuel for the Actinide Management in LWRs," *Nucl. Technol.* **146**, 325–331 (June 2004).

13. S. G. HONG and E. Greenspan, "Molten-Salt-Cooled Encapsulated Nuclear Heat Source (ENHS)-Like Reactors," *1st International Conference on Innovative Nuclear Energy Systems for Sustainable Development of the World (COE INES-1)*, Tokyo Institute of Technology, Tokyo, Japan (October 31–November 4, 2004), in press: *Progress in Nuclear Energy* (2004).

14. N. E. TODREAS, P. E. MacDonald, P. Hejzlar, J. Buongiorno, and E. P. Loewen, "Medium-Power Lead-Alloy Reactors: Missions for This Reactor Technology," *Nucl. Technol.*, **147**, 305–320 (September 2004).