The Longer-Term Option: The Advanced High-Temperature Reactor

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ABSTRACT

The Advanced High-Temperature Reactor (AHTR) is a new reactor concept that combines four existing technologies in a new way: (1) coated-particle graphite-matrix nuclear fuels (traditionally used for helium-cooled reactors), (2) Brayton power cycles, (3) passive safety systems and plant designs from liquid-metal-cooled fast reactors, and (4) low-pressure liquid-salt coolants with boiling points far above the maximum coolant temperature. The new combination of technologies enables the design of a large [2400- to 4000-MW(t)] high-temperature reactor, with reactor-coolant exit temperatures between 700 and 1000ºC (depending upon goals) and passive safety systems for economic production of electricity or hydrogen. The capital costs for the AHTR [2400 MW(t)] have been estimated to be 49 to 61% per kilowatt (electric) relative to modular gas-cooled [600 MW(t)] and modular liquid-metal-cooled reactors [1000 MW(t)], assuming a single AHTR and multiple modular units with the same total electrical output. Because of the similar fuel, core design, and power cycles, about 70% of the required research for the AHTR is shared with that for other high-temperature gas-cooled reactors. However, while several high-temperature helium-cooled reactors have been built, no salt-cooled high-temperature reactors have yet been constructed. Consequently, there is a much better understanding of high-temperature gas-cooled reactors and such reactors are closer to commercial deployment. The AHTR, as a new reactor concept, is earlier in development, has greater uncertainties, and thus a different path forward to a commercial machine. It is a longer-term option relative to near-term gas-cooled reactor options.

1. INTRODUCTION

The Advanced High-Temperature Reactor (AHTR) is a new reactor concept with three design goals: (1) high reactor-coolant exit temperatures (700 to 1000°C) to enable the efficient production of hydrogen by thermochemical cycles and the efficient production of electricity, (2) passive safety systems for public acceptance and reduced costs, and (3) competitive economics. To achieve competitive economics, the AHTR is a large reactor with passive safety systems and high temperatures. The high temperatures minimize the size of the safety systems and the power conversion equipment per kilowatt (electric) output. This paper also provides a description of the reactor; the liquid-salt coolant; the power conversion systems; the economics; the required research and development (R&D); and a path forward.

2. AHTR DESCRIPTION

The AHTR (Fig. 1, Table 1) uses coated-particle graphite-matrix fuels and a liquid-fluoride-salt coolant (Forsberg et al., 2003; Ingersoll et al., 2004; Forsberg, 2004). The fuel is the same type that is used in modular high-temperature gas-cooled reactors (MHTGRs), with fuel-failure temperatures in excess of 1600°C. Inert gases (helium, etc.) and fluoride salts are the only coolants that have been demonstrated to be compatible with high-temperature graphite-matrix fuel. The optically transparent liquid salt coolant is a mixture of fluoride salts with freezing points near 400°C and atmospheric boiling points of ~1400°C. Several different salts are being evaluated as the primary coolant, including lithium-beryllium, sodium-zirconium, and sodium-rubidium-zirconium fluoride salts. 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 (H_2) production facility to produce H_2 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 bottoming steam 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 nonnuclear Brayton power cycle and turbine hall, 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.

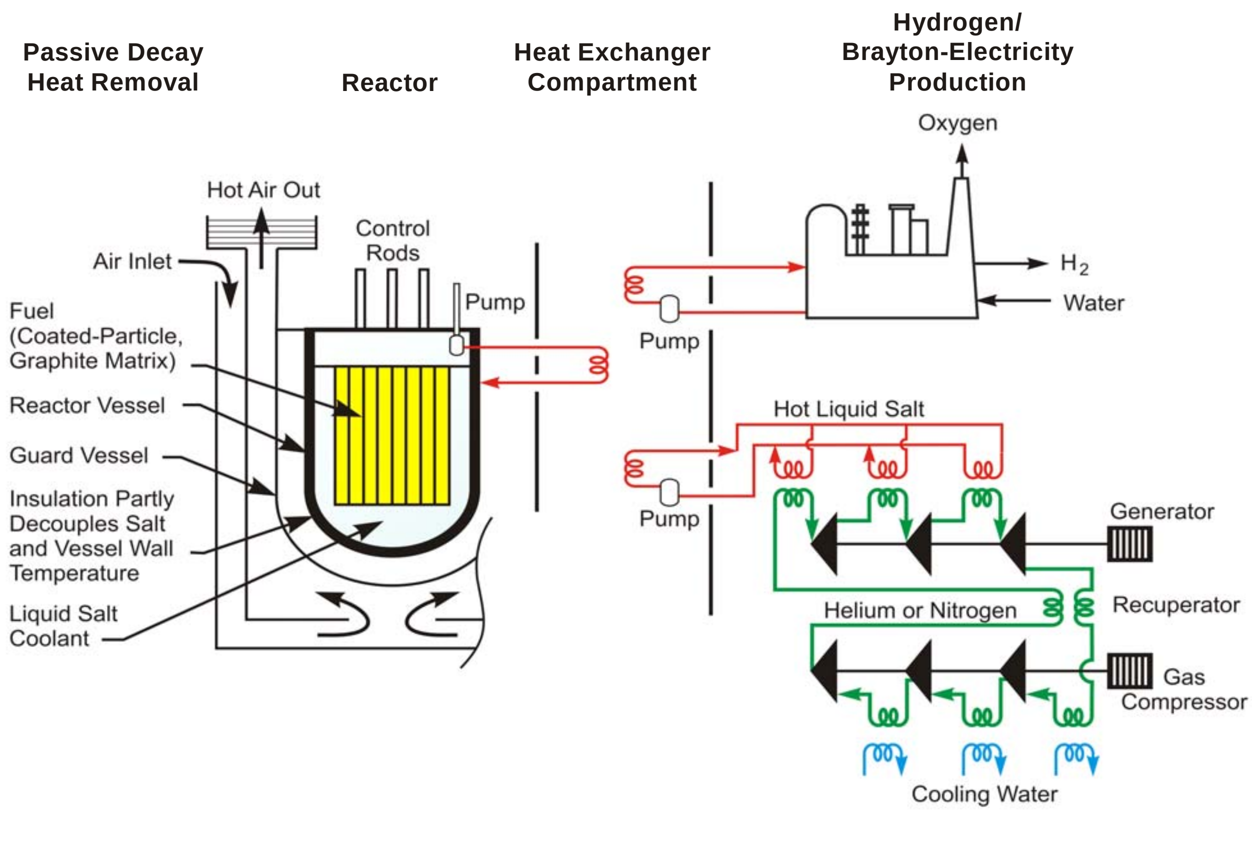

Fig. 1. Schematic of the AHTR for electricity production.

Table 1. AHTR preconceptual design parameters

Power level	2400 MW(t)	Power cycle	3-stage multi-reheat Brayton
Core inlet/outlet temperature (options)	900°C/1000°C 700°C/800°C 670°C/705°C	Electricity (output at different peak coolant temperatures)	1357 MW(e) at 1000ºC 1235 MW(e) at 800ºC 1151 MW(e) at 705ºC
Coolant (several options)	$2^7LiF\text{-}BeF_2$ ($NaF\text{-}ZrF_4$)	Power cycle working fluid	Nitrogen (helium longer-term option)
Fuel		Vessel	
Kernel	Uranium carbide/oxide	Diameter	9.2 m
Enrichment	10.36 wt % ^{235}U	Height	19.5 m
Form	Prismatic	Reactor core	
		Shape	Annular
Block diam	0.36 m (across flats)	Diameter	7.8 m
Block height	0.79 m	Height	7.9 m
Columns	324	Fuel annulus	2.3 m
Decay heat system	Air cooled	Power density	8.3 W/cm^3
Volumetric flow rate	5.54 m^3/s	Reflector (outer)	138 fuel columns
Coolant velocity	2.32 m/s	Reflector (inner)	55 fuel columns

The baseline AHTR facility layout (Fig. 2) that was developed is similar to that for the S-PRISM sodium-cooled fast reactor designed by General Electric. Both reactors operate at low coolant pressure and high temperature; thus, they have similar design constraints. The 9.2-m-diam vessel is the same size as that used by the S-PRISM. The vessel size determines the core size, which, in turn, determines the power output. In the initial studies, it was assumed that the fuel and power density (8.3 W/cm^3) were similar to those of the MHTGR. This is a conservative assumption because higher power densities are possible with liquid coolants.

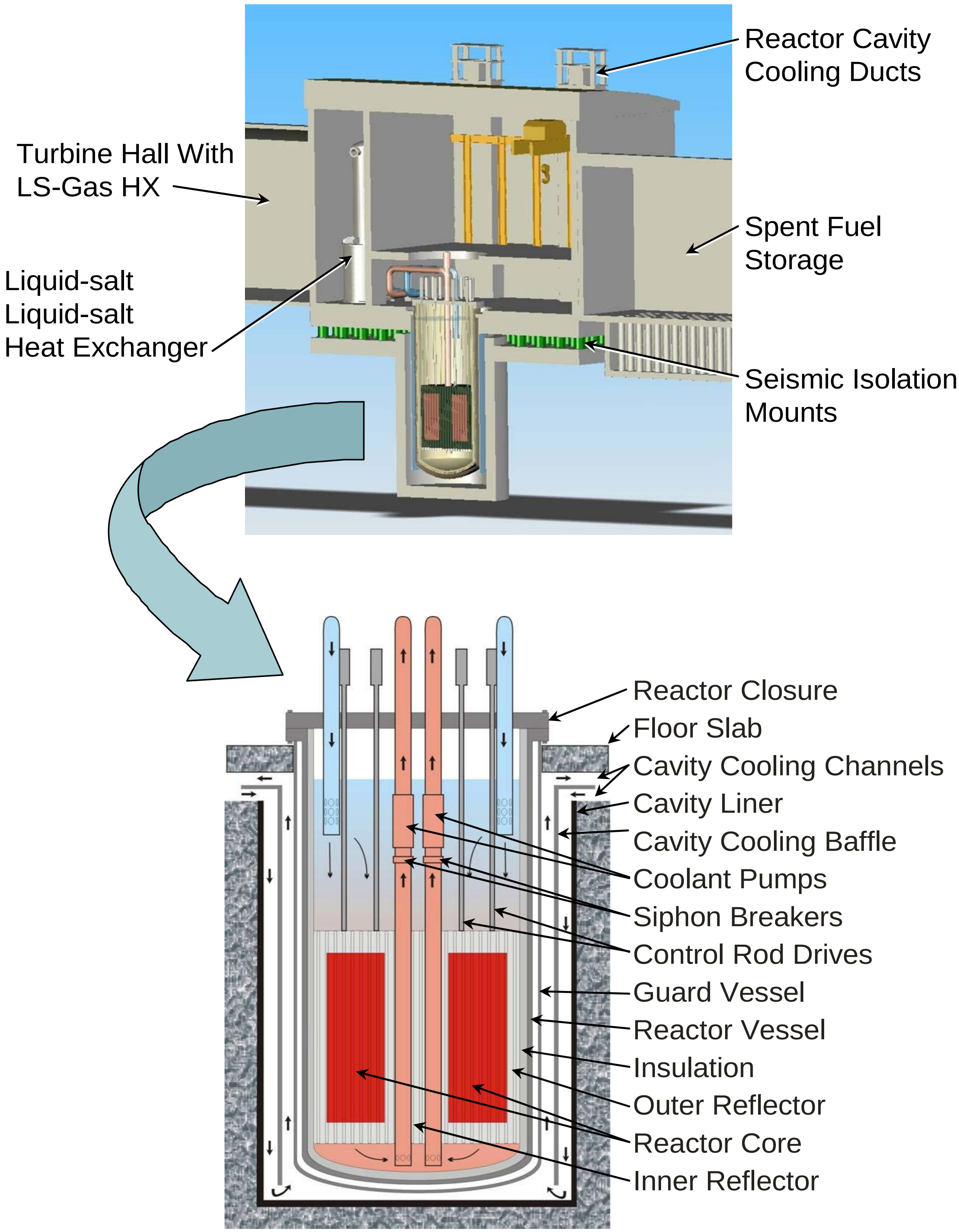

Fig. 2. Schematic of the AHTR nuclear island and vessel.

Three peak coolant temperatures have been evaluated: 705, 800, and 1000ºC, for the AHTR─Low Temperature (AHTR-LT), the AHTR─Intermediate Temperature (AHTR-IT), and the AHTR─High Temperature (AHTR-HT), respectively. If the reactor power output is 2400 MW(t), the respective electricity production capacities are 1151, 1235, and 1357 MW(e). The AHTR-LT uses existing code-qualified materials, the AHTR-IT uses existing materials that have not been fully tested, and the AHTR-HT uses advanced materials. The AHTR-LT has a metallic blanket system that separates and insulates the reactor vessel from the reactor core so that the fuel and coolant can operate at higher temperatures than the vessel. The AHTR-HT and AHTR-IT have similar systems using a graphite blanket system. The insulation ensures long vessel life (minimizing long-term creep) and minimizes heat losses during normal operations.

In the current preconceptual design, the AHTR has an annular core through which the coolant flows downward. The liquid-salt coolant flows upward through the nonfuel graphite section in the middle of the reactor. Alternative core designs are being investigated. The coolant pumps and their intakes are located above the reactor core with appropriate siphon breakers for external intermediate heat exchangers; thus, the reactor cannot lose its coolant except by failure of the primary vessel. Options for internal intermediate heat exchangers are being investigated. The guard vessel is sized so that even if the primary vessel fails, the core remains covered with salt.

When a reactor shuts down, radioactive decay heat continues to be generated in the reactor core at a rate that decreases over time. If this heat is not removed, the reactor will ultimately overheat and the core will be damaged, such as occurred during the Three Mile Island accident. The reference AHTR design (Fig. 1) uses passive reactor vessel auxiliary cooling (RVAC) systems similar to that 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 AHTR, RVAC system decay heat is (1) transferred from the reactor core to the reactor vessel graphite reflector by natural circulation of the liquid salts, (2) conducted through the graphite reflector and reactor vessel wall, (3) transferred across an argon gap by radiation to a guard vessel, (4) conducted through the guard vessel, and then (5) 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. Preliminary analysis suggests that under accident conditions such as a loss-of-forced-cooling accident, natural circulation flow of liquid salt up the hot fuel channels in the core and down the edge of the core rapidly results in a nearly isothermal core with about a 50°C temperature difference between the top and bottom plenums. For a typical simulation of the reactor with an average coolant exit temperature of 1000°C, the calculated peak fuel temperature in such an accident is ~1160°C, which will occur at ~30 h after loss of pumped coolant flow with a peak reactor vessel temperature of ~750°C at ~45 h. The average core temperature in this accident rises to approximately the same temperature as the hottest fuel during normal operations.

Control of heat transfer through the reactor vessel thermal blanket will be a key design issue for the AHTR because leak paths through the blanket could allow natural-circulation flows to cause excessive heat transport and localized overheating of the reactor vessel. Design and testing of the thermal blanket system will therefore have high priority. Options to simplify the vessel thermal blanket system include the use of a natural-circulation-driven Direct Reactor Auxiliary Cooling (DRAC) system to augment or replace the heat removal function of the RVAC system, similar to the approach demonstrated in the Experimental Breeder Reactor at Idaho Nuclear Laboratory.

In terms of passive decay-heat-removal systems, a major difference is noted between the *liquid-cooled* AHTR and *gas-cooled* reactors. The AHTR can be built in very large sizes [>2400 MW(t)], while the maximum size of a gas-cooled reactor with passive-decay-heat removal systems is limited to ~600 MW(t). The controlling factor in decay heat removal is the ability to transport this heat from the center of the reactor core to the vessel wall (RVAC) or to a heat exchanger in the reactor vessel (DRAC). The AHTR uses a liquid coolant, where natural circulation can move very large quantities of decay heat from the hottest fuel to the vessel wall with a small coolant temperature difference (~50°C). Unfortunately, in a gas-cooled reactor under accident conditions when the reactor is depressurized, the natural circulation of gases is not efficient in transporting heat from the fuel in the center of the reactor to the reactor vessel. The heat must be conducted through the reactor fuel to the vessel wall. This inefficient heat transport process limits the power of the reactor to ~600 MW(t) to ensure that the fuel in the hottest location in the reactor core does not overheat and fail under accident conditions.

Because the AHTR uses the same fuel form and moderator, the performance of the reactor core physics and fuel cycle options is generally similar to that of high-temperature helium-cooled reactors. Reactor power is limited by a strong negative temperature coefficient, control rods, and other emergency shutdown systems.

Several liquid fluoride salts with generally similar properties are being evaluated to determine the optimum coolant salt [Ingersoll et al., 2004]. Such evaluations involve trade-offs in neutronics, cost, operations, and other parameters. The initial baseline AHTR design used the same salt ($2^7LiF\text{-}BeF_2$) that was used in the Molten Salt Reactor Experiment (MSRE). The MSRE (*Nucl. Appl. Technol*., 1970) was a molten salt test reactor in which the fuel was dissolved in the liquid (molten) salt—in contrast to the AHTR, in which solid fuel is used along with a clean liquid-salt coolant.

The chemistry of this salt is well understood. Several other salts, such as sodium-zirconium fluoride and other alkali-zirconium fluoride salts, have operational and cost advantages. However, there are other considerations. With the traditional prismatic gas-cooled-reactor core designs, the lithium-beryllium salts have negative void coefficients while some of these other salts have small positive void coefficients. The preliminary transient analysis indicates that the very large negative temperature reactivity coefficient limits the temperature rise in the core to a few tens of degrees Celsius upon large-scale coolant voiding. The long neutron lifetime results in transients of several tens of seconds. This type of mild behavior under accident conditions is very different from that of other reactors with positive void coefficients. Conceptual core design studies with more heterogeneous core designs are under way to determine whether negative coolant void coefficients can be obtained for a wide variety of fluoride salts. As a point of reference, earlier versions of CANDU (heavy-water-moderated) and Hanford-type (water-cooled, graphite-moderated) reactors had positive void coefficients, while the Advanced CANDU Reactor and the Hanford-N Reactor have negative void coefficients.

3. LIQUID SALTS

A significant experience base exists for only three high-temperature liquids: molten iron, molten glass, and liquid fluoride salts. Since the 1890s, essentially all aluminum has been produced by the Hall electrolytic process. In the Hall process, aluminum oxide is dissolved in a mixture of sodium and aluminum liquid fluoride salts (cryolite: $3NaF\text{-}AlF_3$) at ~1000ºC in a graphite-lined bath. Massive graphite electrodes provide the electricity that converts aluminum oxides to aluminum metal.

In the 1950s, at the beginning of the Cold War, the United States launched a large program to develop a nuclear-powered aircraft (Fraas and Savolainen, 1956). MSRs were to provide the heat source, with the heat transferred to a jet engine via an intermediate heat-transport loop. In the MSR, which is a liquid-fuel reactor, the uranium is dissolved in the molten salt coolant. MSRs were chosen for this application to minimize aircraft weight. The nuclear aircraft program was ultimately cancelled because of the peacetime risks of aircraft crashes and the high shielding weight required to protect the crew. In the 1960s and 1970s, the MSR was investigated as a thermal-neutron breeder reactor (*Nucl. Appl. Technol*., 1970). The program was ultimately cancelled when the United States decided to concentrate on development of a single breeder reactor concept, the sodium-cooled fast reactor.

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 fluoride salt. This was followed in 1965 by the Molten Salt Breeder Reactor (MSBR) Experiment, 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, the AHTR uses solid fuel and a clean liquid salt as a coolant (i.e., a coolant with no dissolved fissile materials or fission products). However, these earlier MSR programs demonstrated two critical technologies for the AHTR:

- *Graphite.* In an MSR, molten fuel salt flows by bare graphite blocks in the reactor core, which provide the moderator to slow neutrons. In test loops and reactor experiments, graphite was demonstrated to be compatible with flowing fluoride salts. Only helium and fluoride salts have been demonstrated to be compatible with high-temperature graphite-matrix fuels.

- *Metals of construction*. The MSR programs qualified Hastelloy-N to 750°C as a material of construction. Equally important, it was observed in test loops that clean purified liquid salts were relatively noncorrosive. On the other hand, major development programs were required to develop materials of construction for the molten fuel salt that contained uranium and all of the fission products in high concentrations. Furthermore, it was recognized that with clean salts the coolant chemistry could be adjusted to be chemically reducing and that such chemistry resulted in very low corrosion rates. In the MSR, this was not an option because highly reducing conditions would precipitate some of the dissolved uranium from solution. For the AHTR, the chemistry of the liquid-salt coolant will be controlled. These results are not surprising. It is well known that the corrosion rates in all reactors are strongly dependent upon control of the impurities in the coolant.

Several alternative liquid-salt mixtures can be used (Forsberg, 2004). All are chosen to have low nuclear cross sections and melting points between 350 and 500°C. The optimum salt involves trade-offs between core design, freeze temperatures, and other parameters. If low-cost methods to separate boron isotopes could be developed, potentially useful lower-melting-point fluoride salts would be available.

4. ELECTRICITY AND HYDROGEN PRODUCTION

For electricity production, recuperated gas (nitrogen or helium) Brayton cycles (Fig. 1) are proposed with two to three stages of reheating and three to eight stages of intercooling (Peterson et al., 2003; Peterson et al., 2004). Brayton power cycles are the only efficient power cycles that have been developed that match the temperatures of high-temperature reactors. The multi-reheat Brayton cycle is somewhat similar to the multi-reheat Rankine steam cycle used in many coal-fired power stations. The gas pressure is reduced through multiple turbines in series, with reheating of the gas to its maximum temperature with hot liquid salt before it reaches each turbine. The gas then flows through a recuperator and is compressed in multiple stages with interstage cooling. Both nitrogen and helium Brayton cycles are being considered.

The AHTR is a high-temperature liquid-cooled reactor, a fact that significantly increases the electrical efficiency (4 to 8%) relative to that of gas-cooled reactors with the same exit coolant temperatures. Gas-cooled reactor systems have high pumping costs relative to those of liquid-cooled systems. Because gas cooling has high pressure losses, practical designs of gas-cooled reactors [such as the General Atomics helium-cooled MHTGR with a direct gas-turbine (GT) cycle and the British carbon-dioxide-cooled Advanced Gas Reactor (AGR)] have large temperature increases across the reactor core and deliver their heat to the power cycle over a large temperature range (Fig. 3). Typical temperature increases across the core are 350°C. For example, while the proposed General Atomics MHTGR has an exit temperature of 850°C, the average temperature of delivered heat is only 670°C and the lowest temperature of delivered heat is 491°C. In contrast, liquid-cooled reactors such as the French sodium-cooled Super Phenix liquid-metal fast breeder reactor (LMFBR) and pressurized-water reactors (PWRs) have low pumping costs and are designed to deliver their heat from the reactor core to the power cycle over a small temperature range, 20 to 150°C. The same is true for the AHTR. Efficiency is roughly dependent upon the average temperature delivered to the power cycle, not the peak temperature; thus an AHTR with a peak coolant exit temperature of 750ºC delivers heat at a higher average temperature to the power cycle than an MHTGR with a gas-coolant exit temperature of 850ºC. Liquid cooling results in higher power-plant efficiencies at lower peak reactor-coolant temperatures.

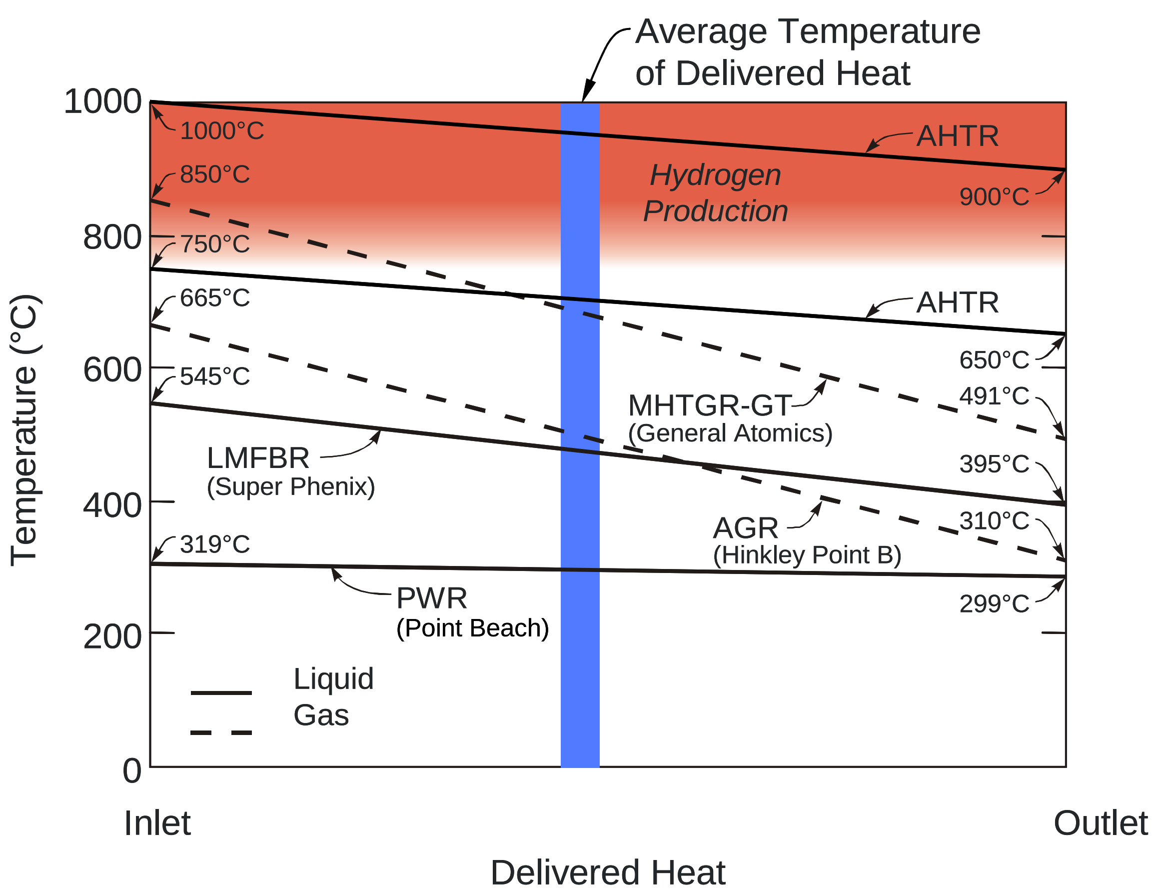

Fig. 3. Temperature of delivered heat for different reactors.

The leading technologies for the low-cost production of hydrogen using nuclear energy are high-temperature thermochemical cycles (Nuclear Energy Agency, 2003), in which, through a series of chemical reactions, high-temperature heat and water yield hydrogen and oxygen. The primary technical challenge is that heat must be provided, depending upon the process, at temperatures between 700 and 850°C. The reactor temperatures must be significantly higher to transfer heat from the reactor fuel, through an intermediate heat-transfer system, and then to the thermochemical hydrogen plant. The required reactor temperatures are at the limits of conventional engineering materials. As a liquid-cooled reactor, the AHTR (Forsberg et al., 2004) has peak reactor-coolant temperatures 100 to 200°C lower than those of gas-cooled reactors for heat delivered at the same temperatures to a thermochemical hydrogen plant. This is because liquids are better coolants than gases and have smaller temperature drops (1) from fuel to coolant and (2) across the two heat exchangers in the intermediate heat-transport loop. As a consequence, the high-temperature materials requirements to deliver high-temperature heat may be less for the AHTR than for gas-cooled reactors. However, the AHTR is a new reactor concept; thus, significantly larger design uncertainties remain.

5. ECONOMICS

Preliminary overnight capital costs (Table 2) of a 2400-MW(t) AHTR for several exit temperatures were determined relative to other higher-temperature reactor concepts [i.e., the modular S-PRISM and the GT–modular helium reactor (GT-MHR)] based on the relative size of systems and quantities of materials. *This approach provides relative, but not absolute, costs.* The initial analysis indicates capital costs per kilowatt electric 53 to 61% of those of the GT-MHR. There are several reasons for these capital cost savings relative to those for helium-cooled modular reactors.

- *Plant size*. The AHTR [2400 MW(t)] is four times larger than the GT-MHR [600 MW(t)] with the same safety basis because the liquid coolant allows the design of very large reactors with passive decay-heat-removal systems. There are large economics of scale.

- *Higher efficiency*. The AHTR, because it is a liquid-cooled reactor, has higher efficiency than the GT-MHR for the same peak reactor temperatures because more of the heat is delivered to the power cycle at high temperatures (Fig. 3).

- *Reduced containment requirements*. The liquid-salt coolant is a low-pressure system versus the high pressures associated with helium-cooled reactors. The liquid salt prevents air ingress and access to the fuel under severe accident conditions. Fission products, such as cesium and iodine, dissolve in liquid salts; thus, the liquid salt provides an additional barrier to prevent fission product release. These characteristics reduce containment requirements.

- *Reduced equipment sizes*. Volumetric heat capacities of liquid salts are somewhat greater than those for water, several times larger than those for sodium, and much larger than those for helium. Liquid salts operate at low pressures. This reduces the size of pipes, valves, pumps, and heat exchangers per unit of energy transferred (Table 3).

Any advanced reactor must compete with light-water reactors (LWRs). Preliminary studies have compared the quantities of materials required to build various advanced reactors (Peterson and Zhao, 2004). As shown in Fig. 4, the initial estimates of AHTR materials requirements are about half those for the newest LWR reactor concepts. *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*.

Table 2. Comparison of estimated overnight capital cost [2002$/kW(e)] of the AHTR-IT and AHTR-HT, as a percentage of the costs of the S-PRISM and GT-MHR [with multi-module output of 1145 MW(e)][a]

		S-PRISM $1681/kW(e)	GT-MHR $1528/kW(e)
AHTR-IT	$930/kW(e)	55%	61%
AHTR-HT	$816/kW(e)	49%	53%

[a]The General Electric S-PRISM consists of four reactor modules, each producing 1000 MW(t) and 380 MW(e). The peak sodium temperature is 510ºC. The General Atomics GT-MHR consists of four reactor modules, each producing 600 MW(t) and 285 MW(e). The peak helium temperature is 850ºC.

Table 3. Relative capabilities of different coolants to transport 1000 MW(t) of heat with a 100ºC rise in coolant temperature

	Water	**Sodium**	**Helium**	**Liquid salt**
Pressure, MPa	15.5	0.69	7.07	0.69
Outlet temp, ºC	320	545	1000	1000
Velocity, m/s (ft/s)	6 (20)	6 (20)	75 (250)	6 (20)
Number of 1-m-diam pipes required to transport heat	0.6	2.0	12.3	0.5

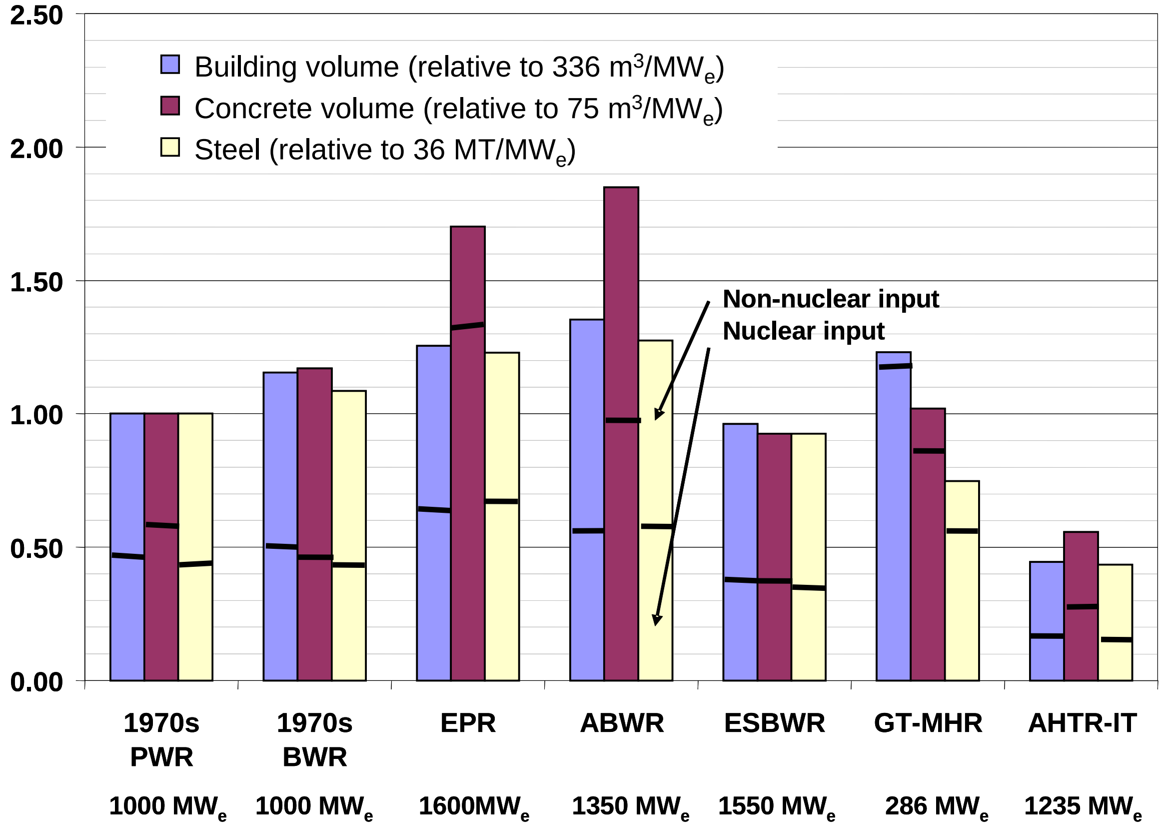

Fig. 3. Relative quantities of materials per unit power output to construct various types and generations of reactors.

(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.)

6. RESEARCH AND DEVELOPMENT REQUIREMENTS

About 70% of the R&D required for the AHTR (Ingersoll et al., 2004) is shared with that for high-temperature helium-cooled reactors. This includes fuel development, materials development, and Brayton power cycles. Many other areas are shared with the facility design considerations for liquid-metal reactors. The required R&D is strongly sensitive to the peak coolant temperatures. Near 700°C, existing materials and fuels may be used. At 1000°C, major material development programs are required. Other areas that require significant R&D include (1) the reactor vessel insulation system, to allow the vessel temperature to be lower than the reactor core temperature; (2) optimization of core design; (3) refueling and maintenance operations in the reactor vessel at 350 to 500°C (the higher refueling temperature is required to maintain the salt as a liquid); (4) selection of the preferred liquid fluoride salt; and (5) analysis of transient operations. The close coupling of fast reactor and AHTR facility designs implies that advances in either reactor facility concept assists in the development of the other. As a new reactor concept, the AHTR is early in its development, with significant technical uncertainties remaining.

Associated with the Next Generation Nuclear Plant (NGNP) is the DOE Nuclear Hydrogen Initiative (NHI), which has the goal of demonstrating nuclear hydrogen production. This requires the transport of large quantities of high-temperature heat from the high-temperature reactor to the chemical plant. Liquid fluoride salts are the leading candidates for this heat transport system because of the excellent heat transport properties of these salts (Table 3) relative to those of helium and other alternatives. There is massive overlap between the R&D requirements for the NHI and the liquid-salt components of the AHTR, including materials, pumps, heat exchangers, and salt cleanup systems.

7. PATH FORWARD

The helium-cooled NGNP and the AHTR are high-temperature reactors that share the same technological foundations: fuels, Brayton cycles, and materials. The NHI and AHTR share liquid-salt technology. Consequently, there is extensive overlap in the technology and the R&D requirements. Common reactor challenges include extending the graphite-matrix coated-particle fuels and materials of construction to the higher temperatures demanded for efficient Brayton power cycles and hydrogen production. For both concepts, closed Brayton cycles must be developed to couple to these reactors. However, there is a significant difference in terms of reactor experience.

- *High-Temperature helium-cooled reactors*. High-temperature helium-cooled reactors have been under development for 40 years. Several lower-temperature helium reactors with steam power cycles have been built in the United States─Peach Bottom [1967–74: 725ºC; 115 MW(t)] and Fort St. Vrain [1976–89: 775ºC; 842 MW(t)]. Helium-cooled reactors have been built elsewhere─Dragon [1963–76: 750ºC: 20 MW(t)]; AVR [1967–88: 950ºC: 49 MW(t)] and the Thorium High-Temperature Reactor (THTR) [1986–89: 750ºC; 750 MW(t)]. Two new high-temperature test reactors have been recently built: High Temperature Test Reactor (HTTR) in Japan [950ºC; 30 MW(t)] and the High-Temperature Reactor (HTR-10) in China [700ºC; 10 MW(t)]. The helium-cooled NGNP is using this experience base to create a 600-MW(t) higher-temperature reactor that is coupled to the higher-temperature efficient Brayton power cycles or is capable of being coupled to a hydrogen production plant. Without this experience, the option of constructing a large helium-cooled NGNP would not exist.

- *AHTR*. No liquid-salt-cooled high-temperature solid-fuel reactor has been built; however, there is limited experience with MSRs. The reasons that no liquid-salt-cooled reactor have yet been built are fundamental. For the last 80 years the utility industry has been well served by steam turbines than convert heat to electricity. However, because of corrosion issues, the practical upper temperature limit of large steam turbines is ~550ºC. This temperature is not that far from the freezing points of a liquid salt and is sufficiently close (particularly during transient and startup operations) that special design features must be incorporated for steam cycles coupled to liquid-salt systems. Salt-cooled reactors are intrinsically high-temperature (>700ºC) reactors. Until the development of large, efficient high-temperature Brayton power cycles, no method to existed to efficiently convert high-temperature heat to electricity. Thus, there was no incentive to develop an AHTR for electricity production. The earlier molten salt programs (fuel dissolved in the salt) had the goals of (1) aircraft propulsion, which required very high temperatures; and (2) fuel breeding, for which a special steam cycle was developed to couple to the reactor.

These fundamental differences imply that the next big step for the helium-cooled NGNP is a full-scale [600 MW(t)] power reactor with higher temperatures than those for previous helium-cooled reactors and the use of a Brayton power cycle. The follow-on for the helium-cooled NGNP is a commercial reactor. In contrast, after further development of the concept, the AHTR will require a pilot plant to demonstrate and validate the technology. Such a pilot plant would be used to demonstrate high-temperature refueling, passive decay-heat-removal systems, reactor core behavior, insulation and decay-heat removal systems, and other unique aspects of salt-cooled reactors. Based on historical experience with other reactors, the likely size of such a pilot plant will be between 25 and 100 MW(t). The small scale also allows relatively easy and low-cost replacement of major components to test alternative designs and technologies for heat exchangers, pumps, and other components. The pilot plant would be followed by progressively larger reactors until a full-scale reactor is deployed. The development schedules for the two reactors also differ.

The helium-cooled NGNP and the AHTR form a naturally complementary program for the development of high-temperature reactors. The helium-cooled NGNP is the near-term precommercial option. The AHTR is the longer-term option that offers potentially better economics. The AHTR is also the backup option if unforeseen challenges are identified with the helium-cooled NGNP. Massive overlap exists in the R&D requirements; thus, relatively little additional R&D funding is required to explore the potential of the AHTR. The AHTR provides strong additional incentives for public and private investments in high-temperature reactors because the base technology leads to many options.

Liquid-salt cooling is a generic technology that can be applied to a wide variety of other applications. The development of a pilot plant may attract interest from outside the high-temperature-reactor community. Nuclear reactor types can be classified by power output and the peak temperatures of their coolants (Fig. 4). 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. Liquid fluoride salts can be used for multiple nuclear applications.

- *Liquid-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 2): low pressure, low pumping costs, and small equipment sizes (very high volumetric heat capacity). Lower-temperature liquid-salt systems (600ºC) are now used in the chemical industry and in solar power towers. This provides industrial experience to apply for operations at higher temperature. In turn, the development of higher-temperature liquid-salt heat-transport systems for nuclear applications may have applications in other industries.
- *Advanced High-Temperature Reactor (AHTR)—this paper.*
- *Liquid-salt-cooled fast reactor (LSFR).* LSFRs are similar to sodium-cooled fast reactors except that liquid salts replace sodium as the coolant (Hong and Greenspan 2004; Forsberg et al., 2005). There is interest in several countries in these advanced reactors because of the potential for lower capital costs. The potential lower capital costs are a consequence of (1) higher plant efficiency by operating at higher temperatures with the new Brayton power cycle technology, (2) avoidance of sodium-water interactions, and (3) a transparent coolant that may simplify refueling and in-vessel inspection. The LSFR represents a longer-term option for salt-cooled reactors because of the need to develop and qualify high-temperature metals for in-core components (structure and claddings).
- *Molten salt reactor (MSR).*
- *Fusion reactors.* Liquid salts are major candidates for cooling inertial and magnetic fusion energy systems (Moir et al., 1994).

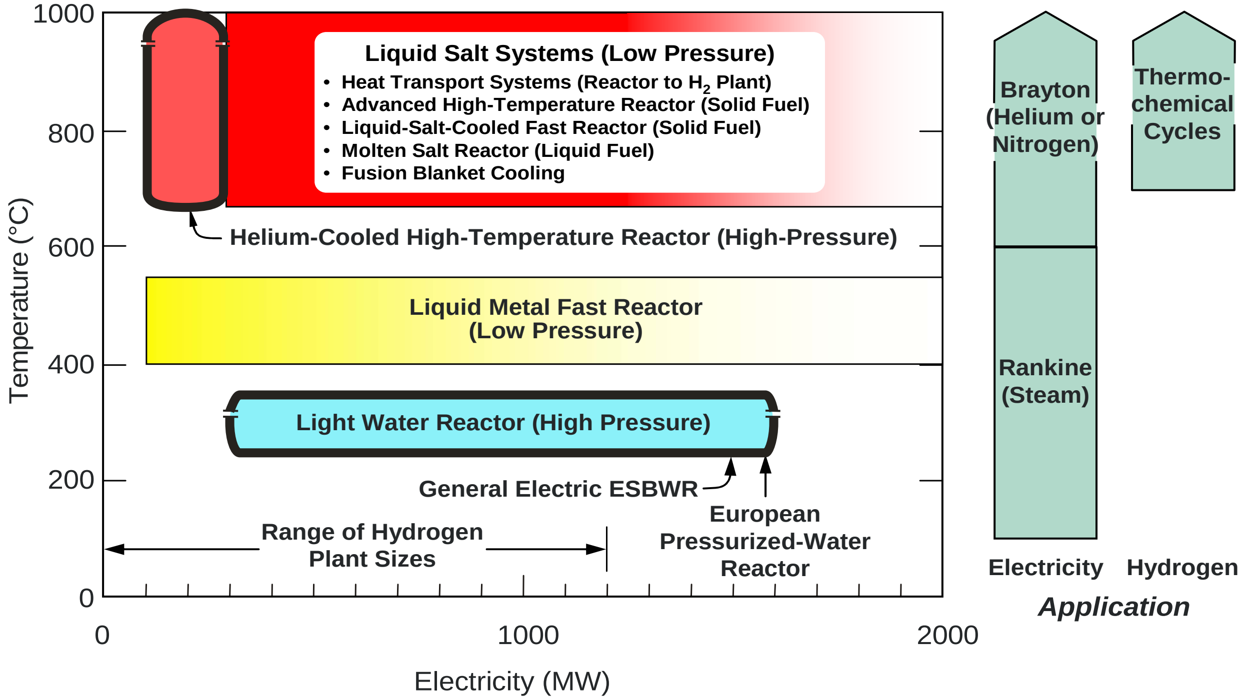

Fig. 4. Reactor type vs temperature and power output.

In each of these applications, there is an increased interest in liquid-salt cooling because efficient Brayton power cycle technology now exists that can efficiently couple to the higher temperatures. For each of these other applications, the development of an AHTR pilot plant would demonstrate key technologies. Thus, there is potentially a large community with incentives to support such a pilot plant.

8. CONCLUSION

The AHTR is a new reactor concept for the efficient production of electricity and hydrogen. Initial analyses indicate the potential for the AHTR to have excellent economics as a large passively safe high-temperature reactor. The characteristics of the AHTR follow from the unique high-temperature capabilities of coated-particle graphite-matrix high-temperature fuel and low-pressure, high-temperature liquid-fluoride-salt coolants. The reactor technology is based primarily on the technologies being developed for the high-temperature gas-cooled reactor (fuel and power cycle) and the liquid-metal reactor (facility design). Because the AHTR is a new reactor concept, there are larger technical uncertainties relative to those for helium-cooled reactors. Although the development pathway of the AHTR is complementary to that of the helium-cooled NGNP, significant differences exist.

REFERENCES

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

Forsberg, C. W. (2004), "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.

Forsberg, C. W., Peterson, P. F., and Ott, L. (2004), "The Advanced High-Temperature Reactor (AHTR) for Producing Hydrogen to Manufacture Liquid Fuels," *Proc. 2004 Americas Nuclear Energy Symposium, Miami Beach, Florida, October 3−6, 2004,* American Nuclear Society, La Grange Park, Illinois.

Forsberg, C. W., Peterson, P. F., and Williams, D. F. (2005), "Practical Aspects of Liquid-Salt-Cooled Fast-Neutron Reactors," *Proc. 2005 Int. Conf. on Advances in Nuclear Power Plants, Seoul, Korea, May 15–19, 2005,* American Nuclear Society, La Grange Park, Illinois.

Fraas, A. P., and Savolainen, A. W. (1956), *Design Report on the Aircraft Reactor Test*, ORNL-2095, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Hong, S. G. and Greenspan E. (2004), "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*.

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

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

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

Nuclear Energy Agency (2003), *Proc. OECD/NEA Second Information Exchange Meeting on Nuclear Production of Hydrogen*, Paris, France.

Peterson, P. F., Zhao, H., and Fukuda, G. (2003), *Comparison of Molten Salt and High-Pressure Helium for the NGNP Intermediate Heat Transfer Fluid*, UCBTH-03-004 (Rev. B), University of California at Berkeley, Berkeley, California, December 5.

Peterson, P. F., Zhao, H., Ballinger, R., et al. (2004), *Next Generation Nuclear Plant Power Conversion Study: Technology Options Assessment*, U.S. Department of Energy, Washington., D.C.

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