

# Maximizing Temperatures of Delivered Heat from the Advanced High-Temperature Reactor

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](mailto:forsbergcw@ornl.gov)

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

Paul Pickard  
Sandia National Laboratories  
P.O. Box 5800; Albuquerque, NM 87185  
Tel: (505) 845-3046  
E-mail: [pspicka@sandia.gov](mailto:pspicka@sandia.gov)

File Name: ICAPP.2004.Hightemp.Paper  
Review Comments: February 15, 2004  
Final Paper: March 15, 2004

Paper 4151  
ICAPP '04, Topical Area 2, Session #2.021  
2004 International Congress on Advances in Nuclear Power Plants (ICAPP '04)  
Embedded International Topical Meeting  
2004 American Nuclear Society Annual Meeting  
Pittsburgh, PA  
June 13–17, 2004

The submitted manuscript has been authored by a contractor of the U.S. Government under contract DE-AC05-00OR22725. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

---

\*Oak Ridge National Laboratory, managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract DE-AC05-00OR22725.

## Maximizing Temperatures of Delivered Heat from the Advanced High-Temperature Reactor

C. W. Forsberg  
Oak Ridge National Laboratory  
P.O. Box 2008  
Oak Ridge, TN 37831  
Tel: (865) 574-6783  
[forsbergcw@ornl.gov](mailto:forsbergcw@ornl.gov)

P. F. Peterson  
University of California  
4153 Etcheverry  
Berkeley, CA 94720-1730  
Tel: (510) 643-7749  
[peterson@nuc.berkeley.edu](mailto:peterson@nuc.berkeley.edu)

Paul Pickard  
Sandia National Laboratories  
P.O. Box 5800  
Albuquerque, NM 87185  
Tel: (505) 845-3046  
[pspicka@sandia.gov](mailto:pspicka@sandia.gov)

**Abstract**—A new high-temperature reactor concept is being developed for hydrogen ( $H_2$ ) and electricity production: the Advanced High-Temperature Reactor (AHTR). The goal is to develop a large economic reactor with passive safety systems that delivers high-temperature heat with the coolant exit temperature as high as 1000°C. The high temperatures enable the production of  $H_2$  using heat and water by efficient thermochemical cycles. The safety is to be equivalent to that of a modular high-temperature gas-cooled reactor (MHTGR). The AHTR fuel is a graphite-matrix coated-particle fuel, the type used in MHTGRs. The coolant is a molten fluoride salt with a boiling point near 1400°C. Because of this low-pressure liquid coolant, the types of passive safety systems proposed for liquid-metal reactors (such as the General Electric S-PRISM) can be used. The use of a low-pressure liquid coolant, rather than high-pressure helium, may reduce the materials and engineering challenges for very high-temperature reactors. Electricity is produced using a multi-reheat helium or nitrogen Brayton cycle. A preliminary preconceptual design of a 2400 MW(t) reactor has been developed with an output of 1300 MW(e) or an equivalent amount of  $H_2$ .

### I. INTRODUCTION

A new type of high-temperature reactor is proposed: the Advanced High-Temperature Reactor (AHTR). This reactor uses a new combination of existing technologies: (1) high-temperature, low-pressure molten-fluoride-salt reactor coolants, (2) coated-particle graphite-matrix fuel developed for high-temperature gas-cooled reactors, (3) passive safety systems from the proposed modular gas-cooled and liquid-metal-cooled reactors, and (4) a high-efficiency Brayton power cycle for electricity production.

The production of hydrogen ( $H_2$ ) by thermochemical processes and the efficient production of electricity require very high temperatures. Hydrogen production may require that heat be provided to chemical reagents within the thermochemical processes at  $\sim 850^\circ\text{C}$ . Coolant reactor exit temperatures must significantly exceed 850°C to provide the temperature drops across the intermediate heat transfer loop from the reactor to the

$H_2$  production plant. Because of the interest in the production of  $H_2$ , work is underway to develop reactors with coolant exit temperatures of 1000°C. High temperatures are a major materials and engineering challenge.

Historically, helium has been proposed as the coolant of choice for very high-temperature reactors. We are examining an alternative option; use of a molten fluoride salt as the coolant, but with the same fuel types. The superior heat transfer and transport characteristics of liquids compared to gases enable delivery of high-temperature heat at a given temperature with lower reactor fuel and component temperatures. This provides a major incentive to consider liquid-cooled high-temperature reactors. There are also other benefits. This paper describes the AHTR, the potential benefits of liquid cooling when there is a requirement to maximize the temperature of the delivered heat, and the developmental challenges.

## II. AHTR DESCRIPTION

The AHTR<sup>1-2</sup> is a high-temperature reactor (Fig. 1, Table 1) that uses coated-particle graphite-matrix fuels and a molten-fluoride-salt coolant. 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. The optically transparent molten-salt coolant is a mixture of fluoride salts with freezing points near 400°C and atmospheric boiling points of ~ 1400°C. The reactor operates at atmospheric pressure. At operating conditions, the molten-salt heat-transfer properties are similar to those of water. Heat is transferred through an intermediate heat-transfer loop to a multi-reheat nitrogen or helium Brayton cycle power conversion system for the production of electricity or to a thermochemical plant that converts water and high-temperature heat to hydrogen (H<sub>2</sub>) and oxygen.

The AHTR facility layout (Fig. 2) is similar to the S-PRISM sodium-cooled fast reactor designed by General Electric. Both reactors operate at low-pressure and high-temperature; thus, they have similar design constraints. The 9.2-m diameter vessel is the same size as that used by the S-PRISM. The S-PRISM sodium-cooled fast reactor has a thermal power output of 1000 MW(t), with an electric power output of 380 MW(e). The same size reactor vessel with the same type of passive decay-heat-cooling system, a similar size nuclear island, and similar system configuration potentially can contain a 2400-MW(t) AHTR with an electrical output of 1300 MW(e). The larger power output in a similar size system is a consequence of several factors.

03-239R

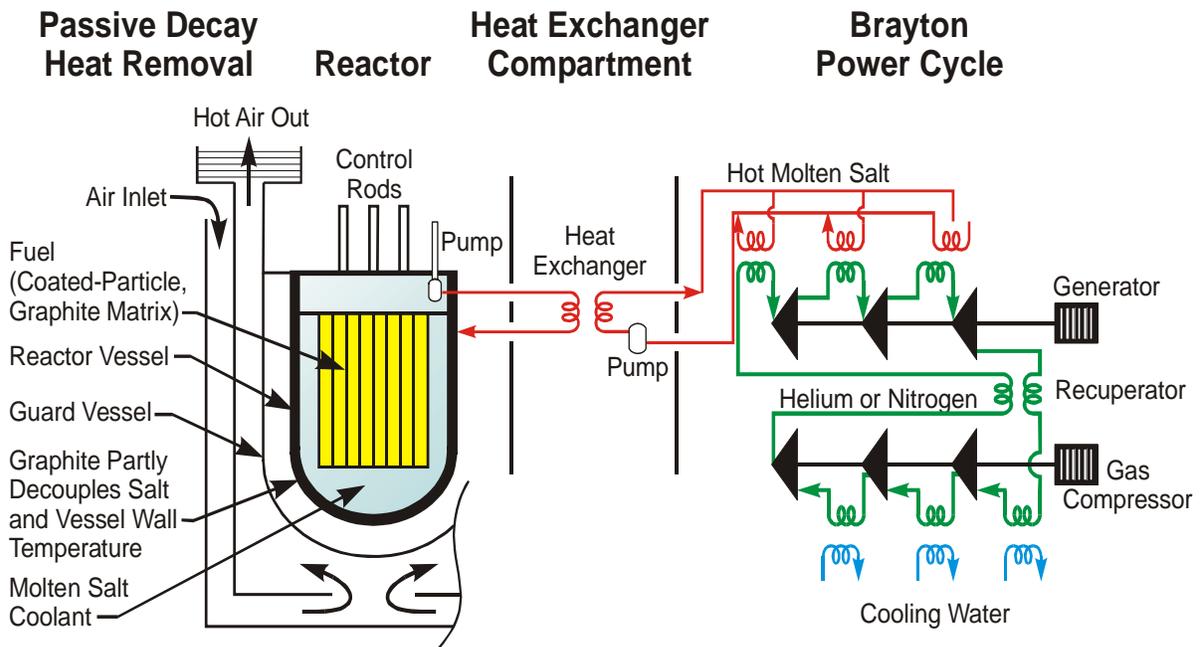


Fig. 1. Schematic of the AHTR for electricity production.

**Table 1. AHTR Preconceptual Design Parameters**

<b>Power level</b>	<b>2400 MW(t)</b>	<b>Electricity</b>	<b>1300 MW(e)</b>
Core inlet/outlet temperature	900EC/1000EC	Power cycle	3-stage multi-reheat Brayton
Coolant (several options)	$2^7\text{LiF-BeF}_2$ (NaF-ZrF <sub>4</sub> )	Power cycle working fluid	Nitrogen (helium longer-term option)
Coolant Mass flow rate	12, 070 kg/s (20% core bypass)	Core inlet pressure outlet pressure	0.230 MPa 0.101 MPa
Vol. Flow rate	5.54 m <sup>3</sup> /s	Pressure drop	0.129 MPa
Channel diam.	0.95 cm	Shape	Annular
Fraction (core)	6.56%	Diameter	7.8 m
Velocity	2.32 m/s (7.6 ft/s)	Height	7.9 m
Fuel Kernel	Uranium carbide/oxide	Fuel annulus	2.3 m
Enrichment	10.36 wt % <sup>235</sup> U	Pumping power	716 kW
Form	Prismatic	Power density	8.3 W/cm <sup>3</sup>
Block. Diam.	0.36 m (across flats)	Reflector (outer)	138 columns
Block height	0.79 m	Reflector (inner)	55 columns
Columns	324	Vessel diameter	9.2 m
Mean temp.	1050EC	Vessel height	19.5 m
Peak temp.	1168EC		

03-261

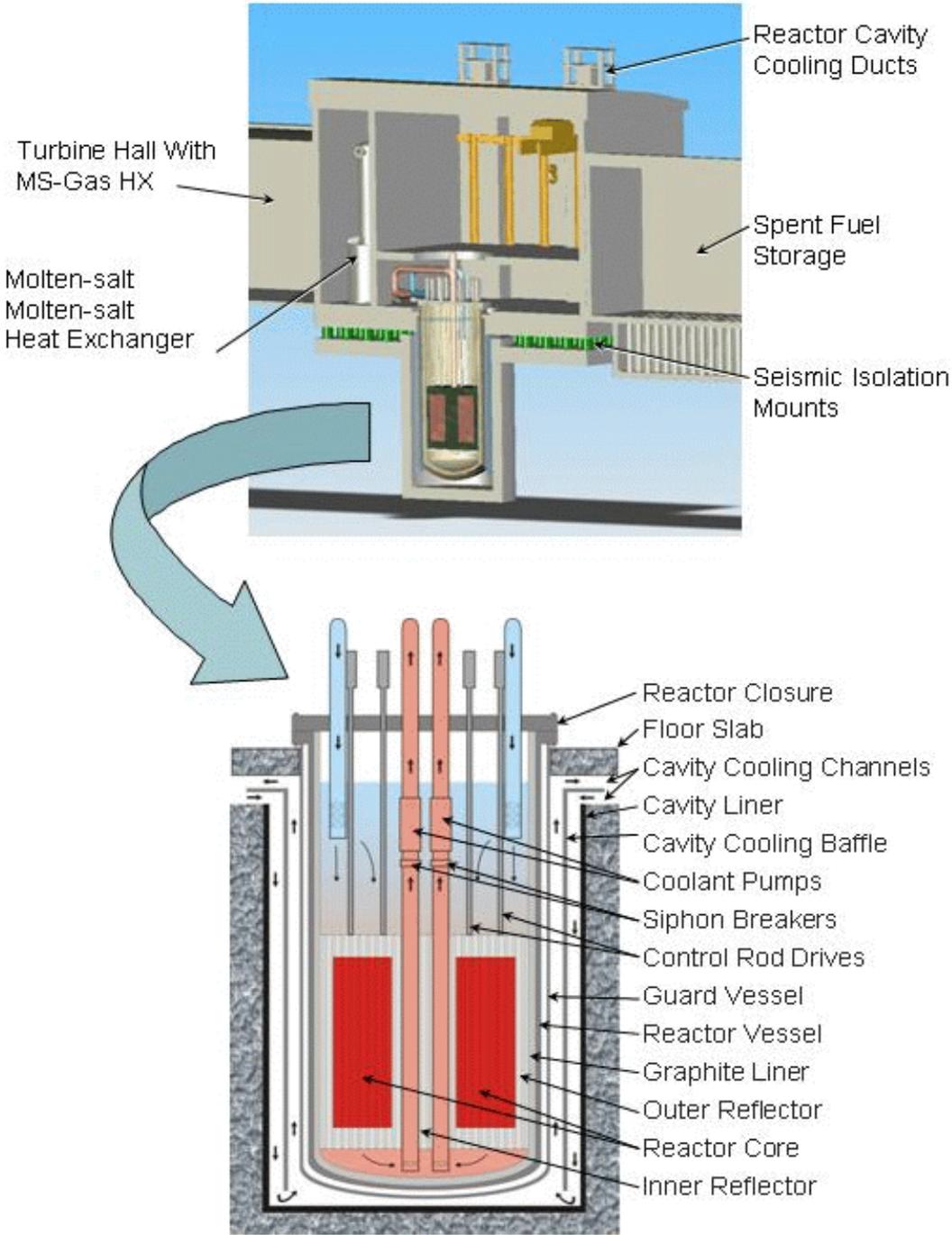


Fig. 2. Schematic of the AHTR nuclear island and vessel.

- *Layout.* Unlike the S-PRISM vessel which contains the core, spent nuclear fuel (SNF) storage, and heat exchangers to transfer heat from the primary sodium system to the intermediate sodium system, the AHTR vessel contains only the reactor core. This provides sufficient space for a 2400 MW(t) reactor with a power density ( $8.3 \text{ W/cm}^3$ ) about equal to gas-cooled reactors with coated-particle fuels. There is a separate SNF storage area. The molten salt to molten salt intermediate heat exchangers (IHX) are also located outside the reactor vessel but inside the containment. Because the salts have low corrosion rates and do not react with each other, it may be possible to utilize a plate-type IHX design similar to that for helium-to-helium gas-cycle recuperators with a power density an order of magnitude greater than a tube-based S-PRISM. The AHTR heat exchangers between the clean secondary molten salt and the gas in the Brayton cycle are located on the turbine floor next to the turbines. This is a requirement for the design of an efficient closed Brayton cycle. The gas friction losses strongly impact Brayton-cycle efficiency, whereas the energy cost of pumping liquids is low; consequently, gas flow path lengths must be minimized.
- *Temperature.* The higher temperature allows for higher decay heat removal in the same size vessel (below) and results in higher efficiency in converting thermal energy to electricity.
- *Physical properties.* Molten salts have about four times the volumetric heat capacity of sodium. As a consequence, the pumps, valves, and piping are smaller although the heat being transferred from the reactor core to the power conversion equipment is greater. The greater density creates larger vessel hydrostatic loads, requiring the AHTR vessel to be somewhat thicker than the S-PRISM vessel. The lower thermal conductivity of the salts (similar to water) reduces vessel thermal stresses and potential thermal shocks due to liquid flow and temperature differentials.

Like that of the MHTGR, the AHTR includes a graphite blanket system that separates the vessel from the reactor core so that the fuel and coolant can operate at higher temperatures than the vessel. In the current design, the AHTR, like the MHTGR, has an annular core with coolant flowing downward through the core. The molten salt coolant flows upward

through the nonfuel graphite section in the middle of the reactor. The molten salt pumps and their intakes are located above the reactor core; thus, the reactor cannot lose its coolant except by vessel failure.

The reactor core physics are generally similar to those for the MHTGR because the molten salt coolant has a low neutron-absorption cross section. Reactor power is limited by a negative temperature coefficient, control rods, and other emergency shutdown systems.

The AHTR uses passive reactor vessel auxiliary cooling (RVAC) systems<sup>2</sup> similar to that developed for the General Electric sodium-cooled S-PRISM for decay heat removal. The reactor and decay heat cooling system are located in a below-grade silo. In this pool reactor, RVAC system decay heat is (1) transferred to the reactor vessel graphite reflector by natural circulation of the molten 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. 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. The design allows transfer of the heat by efficient liquid natural convection from the center of the reactor core (hot-spot location) to near the vessel wall. The vessel layout also allows the addition of supplemental direct reactor auxiliary cooling (DRAC) heat exchangers, similar to those used in the EBR-II, to augment decay heat removal by the RVAC system.

For electricity production, a recuperated gas (nitrogen or helium) Brayton cycle (Fig. 1) is used with three stages of reheating and three stages of intercooling. The gas pressure is reduced through three turbines in series, with reheating of the gas to its maximum temperature with hot molten salt before it reaches each turbine. The projected efficiency is 55%. The major advantage of the nitrogen Brayton cycle is that the turbomachinery is commercially available—it is similar to those used by electric utilities in natural-gas, combined-cycle plants. The major difference is that the low pressure in a closed Brayton cycle (0.5 to 1 MPa, depending upon design) is significantly above atmospheric pressure;

consequently, the equipment is smaller per unit of output than an open Brayton cycle. In the longer-term, helium<sup>3</sup> may be the preferred gas for the Brayton cycle because it reduces the gas pressure losses and size of the molten-salt-to-gas heat exchangers.

For H<sub>2</sub> production, the intermediate loop delivers the high-temperature heat to the thermochemical H<sub>2</sub> production plant<sup>4,5</sup>. In a thermochemical plant, high-temperature heat plus water yield H<sub>2</sub> and oxygen. All other chemicals are fully recycled in the facility.

### III. CAPABILITIES OF MOLTEN-SALT COOLING

The physical characteristics of molten fluoride salt coolants allow the design of reactors with coolant exit temperatures from ~ 700°C to in excess of 1200°C. However, the development of an economic very high-temperature reactor is a very challenging task. The use of a low-pressure, high-temperature molten-salt coolant, rather than helium, may significantly reduce this challenge by several mechanisms. As a basis of comparison, Table 2 shows the characteristics of three reactors each designed with a coolant exit temperature of 1000°C: the AHTR, a helium-cooled reactor with prismatic fuel<sup>6</sup>, and a helium-cooled reactor with pebble-bed fuel<sup>6</sup>.

#### III.A. Reactor Size and Economics

Molten salt coolants<sup>2</sup> enable construction of large high-temperature reactors with passive safety systems and potentially superior economics. All of the above reactors have conceptually similar passive safety systems: (1) heat is transferred from the center of the reactor core to the vessel surface and (2) a silo cooling system removes the heat from the vessel surface to the atmosphere. Additionally, the AHTR can augment this decay heat removal capability using a DRAC system. All of the reactors have the same basic fuel that can retain fission products for some time at temperatures as high as 1650°C. This provides ~ 1600°C to drive decay heat in an accident condition from the hottest point in the reactor to the environment without destruction of the fuel. The difference between the reactors is the coolant.

- *Helium.* Decay heat in a depressurization accident is transferred from the center of the core

to the vessel wall by conduction through fuel and reactor components. (Convective cooling by helium is very inefficient.) This requires a temperature drop of almost a 1000°C to remove the decay heat from a 600 MW(t) reactor. The remainder of the available temperature drop is required to move heat from the reactor vessel to the environment.

- *Molten salt.* Decay heat in an accident is transferred from the center of the core to the vessel wall by natural circulation of the molten salt. Very large quantities of decay heat can be transferred with a temperature drop of only a few tens of degrees C. A much greater temperature drop is available between the vessel wall and the environment to transfer heat by the silo cooling system. If vessel temperatures are allowed to reach 750 to 800°C, the decay heat from a 2400 MW(t) reactor can be dumped to the atmosphere by passive means. This is viable in an emergency with a low-pressure reactor vessel.

#### III.B. Lower Peak Fuel Temperature

Temperature limits on the fuel are a major constraint for high-temperature reactors. As shown in Table 2, with a somewhat higher core power density, the peak fuel temperature for a molten-salt-cooled reactor is 50 to 100°C cooler than an equivalent helium-cooled reactor. This is a direct consequence of the better heat transfer between a liquid molten salt and the fuel versus helium and the fuel. The peak reactor fuel temperatures of the AHTR will be lower than those in a gas-cooled reactor, or the reactor can be designed with a higher power density for any given peak fuel temperature.

#### III.C. Heat Transport

Molten salts have superior heat transport capabilities relative to other fluids. Table 3 shows the number of 1-m diameter pipes required to move 1000 MW(t) of heat, assuming a 100°C rise in the coolant temperature. Half a pipe is required if the coolant is a molten salt; 12.3 pipes are required if the coolant is helium. The different physical properties imply that pipes, valves, and pumps within the reactor station will be much smaller for molten-salt-cooled reactors than for helium-cooled reactors.

**Table 2. Characteristics of High-Temperature Reactors**

<b>Property</b>	<b>AHTR</b>	<b>He/Pebble Bed</b>	<b>He/Prismatic</b>
Power (MW(t))	2400	600	600
Electricity (MW(e))	1300	300	300
Coolant	Molten salt	Helium	Helium
Core parameters			
Pressure (MPa).	0.23	7.1	7.1
Active core height (m)	7.9	9	7.93
Temperature inlet (EC)	900	600	491
Temperature outlet (EC)	1000	1000	1000
Power density (W/cm <sup>3</sup> )	8.3	5.5	6.5
Fuel			
Peak temperature (EC)	1168	1236	1276
Geometry	Prismatic	Pebble bed	Prismatic
Coolant flow rate (kg/s)	12,070	288	226
Vessel diameter	9.2	7.02	7.66
Power cycle			
Type	Indirect	Direct	Direct
Fluid	Nitrogen	Helium	Helium

**Table 3. Heat Transport Properties of Different Coolants under Reactor Conditions**

	<b>Water</b>	<b>Sodium</b>	<b>Helium</b>	<b>Molten salt</b>
Pressure, MPa	15.5	0.69	7.07	0.69
Outlet temp, EC	320	545	1000	1000
Velocity, m/s (ft/s)	6 (20)	6 (20)	75 (250)	6 (20)
Number of pipes to transport 1000 MW(t) heat with 100EC rise	0.6	2.0	12.3	0.5

If thermochemical H<sub>2</sub> is to be produced, the heat must be transported from the reactor to the chemical plant with minimum loss of heat. Regulatory and safety considerations may require separation of the facilities by many hundreds of meters. In the 1970s, Westinghouse<sup>7</sup> developed a conceptual plant design for the hybrid thermochemical cycle using a 3345 MW(t) very high-temperature reactor to produce 10.1 million m<sup>3</sup> of H<sub>2</sub> per day. The physical size of the chemical facilities requires transport of heat over hundreds of meters even if there is close siting of the nuclear and chemical plant. These factors strongly favor a molten-salt heat transfer loop between any reactor and a H<sub>2</sub> production plant. The AHTR is most compatible with such an intermediate heat transfer loop.

#### III.D. Lower Peak Coolant Temperatures for Hydrogen Production

There are two important reactor coolant temperatures: the peak temperature and the average temperature. The peak temperature determines the requirements for fuels and materials, while the average temperature is a measure of the useful energy that the reactor can deliver for electricity or H<sub>2</sub> production. Because of their much higher volumetric heat capacities, liquid coolants have low pumping power costs in comparison with gas coolants. Consequently, as shown in Fig. 3, liquid-cooled reactors (water-cooled pressurized water reactors, sodium-cooled liquid metal fast breeder reactors, and the molten-salt-cooled AHTR) deliver most of their heat at near-constant temperatures while gas-cooled reactors (carbon-dioxide-cooled advanced gas reactor and helium-cooled MHTGR) deliver their heat over a wide range of temperatures due to pumping power limitations. This has several implications.

- *Hydrogen production.* For H<sub>2</sub> production, a molten salt coolant may allow the peak reactor coolant temperature to be 50 to 100EC cooler than for the equivalent helium-cooled reactor (Fig. 3). The thermochemical production of H<sub>2</sub> requires delivery of substantial quantities of heat between 800 and 850EC. After considering temperature drops in the IHX loop, the peak molten-salt coolant temperatures will be between 850 and 950EC. However, with helium coolants, the peak helium temperature must be higher to deliver a significant fraction of the reactor heat at high temperatures. This is because helium delivers its heat over a temperature range of several hundred degrees centigrade. There is also the additional factor that the temperature

drops across heat exchangers are less in molten salt systems than in helium systems.

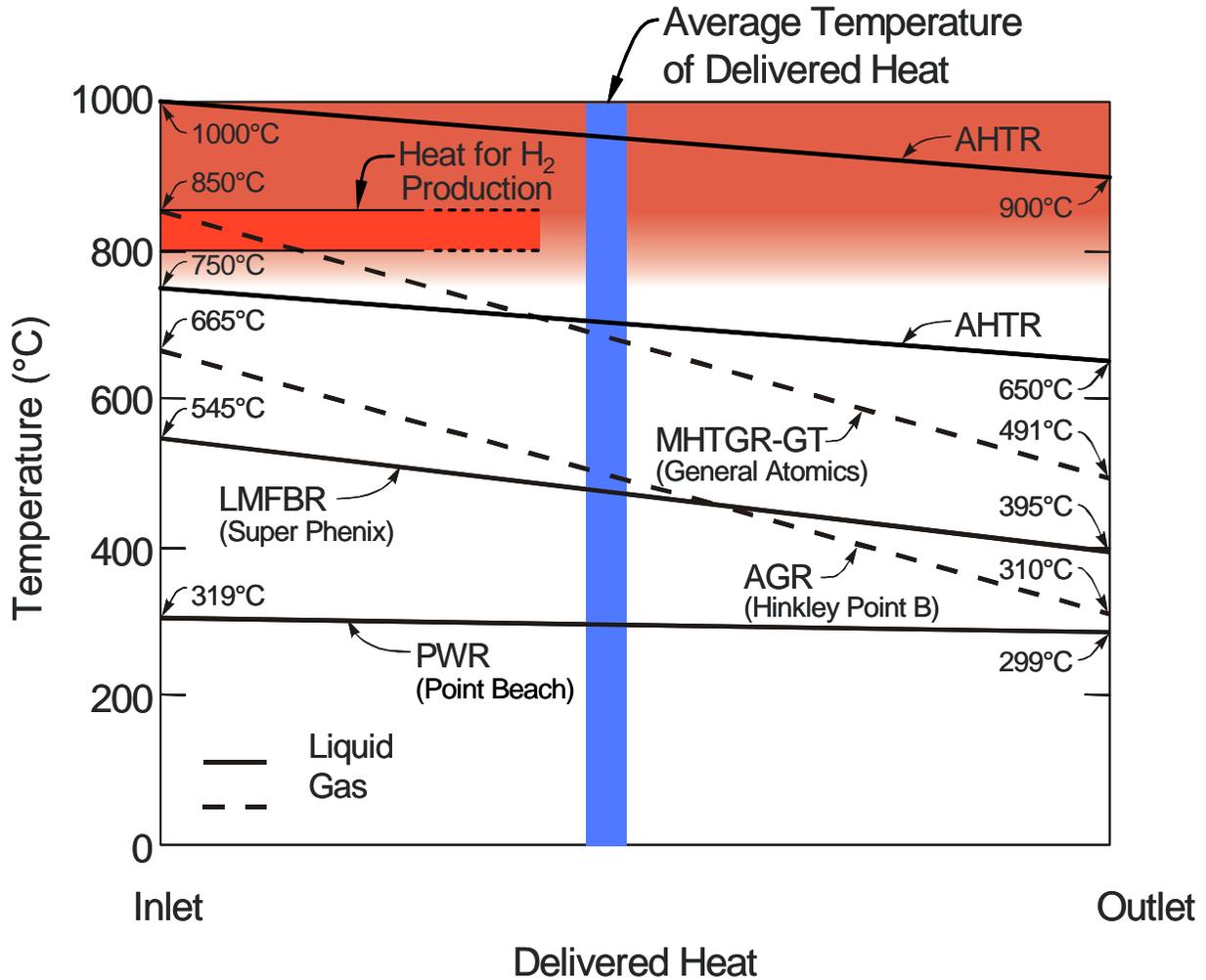
- *Electricity production.* The delivery of the heat at a near-constant temperature allows the AHTR to use more efficient multi-reheat power cycles. As a consequence, the AHTR, with an indirect Brayton power cycle<sup>3</sup>, will be more efficient than a gas-cooled reactor with a direct Brayton power cycle with the same coolant exit temperatures. In a gas-cooled reactor with a 1000EC exit temperature, half or more the heat is delivered below 800EC (Table 2).

## IV. MATERIALS

For high-temperature operations, materials and fuels are critical technologies and the primary challenge. The lower peak fuel temperatures and potentially lower component temperatures are an advantage for molten salt coolants compared to helium coolants.

Molten fluoride salts are compatible with graphite fuels. There is a century of large-scale experience in the use of fluoride molten salts. Aluminum is made by electrolysis of a mixture of bauxite and sodium aluminum fluoride salts at - 1000EC in large graphite baths. The molten salt reactor programs demonstrated that molten salts are compatible with graphite in experiments to 1400EC. In a molten salt reactor, the fuel is dissolved in the coolant whereas the AHTR uses a clean salt and a solid fuel<sup>1</sup>.

The primary challenges are the materials of construction for components and especially heat exchangers. There are four issues for very high-temperature service: strength over time, long-term creep, corrosion resistance to air, and corrosion resistance to molten fluoride salts. Nickel alloys such as Hastelloy-N have been qualified for service to 750EC, but no materials have been qualified to higher temperatures. In these systems, the constituents of the fluoride salt are thermodynamically stable with respect to fluorides of the alloying constituents of the materials of construction the molten salts. Like sodium, the molten salt is noble with respect to the materials of construction. A number of metals and carbon-carbon composites<sup>8</sup> have been identified for use at much higher temperatures.



**Fig. 3. Temperature of delivered heat from different reactors.**

A large database exists on interactions between molten salts and materials. The initial developments of fluoride molten salt nuclear technology were for the nuclear aircraft propulsion program and the molten salt breeder reactor program. Today, molten salt coolants are being developed for fusion reactors and heat-transfer applications. In general, the corrosion rates<sup>9</sup> of nickel alloys are low if fluoride salts are maintained under chemically reducing conditions. Maintaining highly reducing chemical conditions is not possible in molten salt reactors with dissolved uranium in the coolant because highly reducing conditions would precipitate much of the

uranium; however, it is a viable approach for the AHTR where the molten salt is a clean coolant. Like light-water reactors, coolant chemistry control is critical. The available data suggests that with appropriate chemistry control, fluoride salt corrosion will not be the limiting factor in materials. However, significant experimental work is required before there will be high confidence in this conclusion. This implies that the primary materials challenges will be developing practical materials with appropriate high-temperature mechanical properties and resistance to air oxidation—the same challenges as exist for very high-temperature helium-cooled reactors.

The other materials issue is the choice of molten fluoride salt. Several options exist, including salts containing  ${}^7\text{Li}$ , Be, Na, Rb, and Zr fluorides. This involves complex tradeoffs between neutronics, cost, freezing points, boiling points, and other factors.

## V. CONCLUSIONS

The AHTR is a new reactor concept. The unique characteristic of the reactor is its combination of a very high-temperature fuel (graphite-matrix coated-particle fuel) with a low-pressure, very high-temperature molten-salt coolant. Combining these two technologies may enable the construction of large very high-temperature reactors with high efficiency and passive safety systems. Preliminary scoping studies have been completed; however, many uncertainties remain. The next step is to develop a more detailed preconceptual design to (1) understand the complex technical, safety, and economic tradeoffs; (2) provide a credible cost estimate for an  $n^{\text{th}}$ -of-a-kind plant; and (3) develop a detailed research and development plan that defines the issues that must be addressed.

## REFERENCES

1. 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**, pp. 289–302 (December 2003).
2. C. W. FORSBERG and P. F. Peterson, "Making Core Melt Accidents Impossible in a Large 2400-MW(t) Reactor," *Global 2003, Embedded Topical Within 2003 American Nuclear Society Winter Meeting, November 16–20, 2003, New Orleans, Louisiana* (2003).
3. P. F. PETERSON, "Multiple-Reheat Brayton Cycles for Nuclear Power Conversion with Molten Coolants" *Nucl. Technol.* **144**, pp. 279–288 (December 2003).
4. L. C. BROWN, G. E. Besenbruch, R. D. Lentsch, K. R. Schultz, J. F. Funk, P. S. Pickard, A. C. Marshall, and S. K. Showalter, *High Efficiency Generation of Hydrogen Fuels Using Nuclear Power, Final Technical Report for the Period August 1, 1999 through September 30, 2002*, GA-A24285, General Atomics Corporation, La Jolla, California, June 2003.
5. J. E. GOOSSEN, E. J. Lahoda, R. A. Matzie, and J. P. Mazzocchi, "Improvements in the Westinghouse Process for Hydrogen Production," *Global 2003, Embedded Topical Within 2003 American Nuclear Society Winter Meeting, November 16–20, 2003, New Orleans, Louisiana* (2003).
6. P. E. MAC DONALD, et al., *NGNP Preliminary Point Design—Results of the Initial Neutronics and Thermal-Hydraulics Assessments*, INEEL/EXT-03-00870, Rev. 1, Idaho National Engineering and Environmental Laboratory (2003).
7. G. H. FARBMAN, *The Conceptual Design of an Integrated Nuclear Hydrogen Production Plant Using the Sulfur Cycle Water Decomposition System*, NASA-CR-134976, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania (April 1976).
8. P. F. PETERSON, C. W. Forsberg, and P. S. Pickard, "Advanced CsiC Composites for High-Temperature Nuclear Heat Transport with Helium, Molten Salts, and Sulfur-Iodine Thermochemical Hydrogen Process Fluids," *Proc. OECD/NEA Second Information Exchange Meeting on Nuclear Production of Hydrogen*, October 2–3, 2003, Argonne National Laboratory, Argonne, Illinois, Nuclear Energy Agency, Paris (2003).
9. 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).