# The Advanced High-Temperature Reactor: High-Temperature Fuel, Molten Salt Coolant, and Liquid-Metal-Reactor Plant

**Charles 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

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

To be Published In
*Progress in Nuclear Energy*
Elsevier Ltd.; Great Britain

Session: 2: Innovative Reactors: High-Temperature Reactors
Manuscript date: October 7, 2004
File: AHTR: Japan.AHTR.Nov2004.Paper



---

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

# THE ADVANCED HIGH-TEMPERATURE REACTOR: HIGH-TEMPERATURE FUEL, MOLTEN SALT COOLANT, AND LIQUID-METAL-REACTOR PLANT

CHARLES FORSBERG

Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN, 37831-6165
United States of America

## ABSTRACT

The Advanced High-Temperature Reactor 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 molten-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 1000EC (depending upon goals) and passive safety systems for economic production of electricity or hydrogen. The AHTR [2400-MW(t)] capital costs 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 is shared with that for high-temperature gas-cooled reactors.

## KEYWORDS

High-temperature reactor; Molten salt coolant; Hydrogen

## 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 power conversion equipment per kilowatt (electric) output. The reactor concept is being developed in the United States as a joint effort by Oak Ridge National Laboratory, Sandia National Laboratories, and the University of California at Berkeley. This paper provides a description of the reactor, the molten salt coolant, the power conversion systems, the economics, and the required research and development (R&D).

## 2. AHTR DESCRIPTION

The AHTR (Fig. 1, Table 1) uses coated-particle graphite-matrix fuels and a molten-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. 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. Several different salts are being evaluated as the primary coolant, including lithium-beryllium and sodium-zirconium fluoride salts. The reactor operates at near-atmospheric pressure. At operating conditions, the molten-salt heat-transfer properties are similar to those of water. Heat is transferred from the reactor core by the primary molten-salt coolant to an intermediate heat-transfer loop. The intermediate heat-transfer loop uses a secondary molten-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 is used.

The baseline AHTR facility layout (Fig. 2) that was developed is similar to 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 baseline studies, it was assumed that the fuel and power density (8.3 W/cm$^3$) were essentially identical to those of the MHTGR. This is a conservative assumption because higher power densities are possible with liquid coolants.

Three peak coolant temperatures were 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.

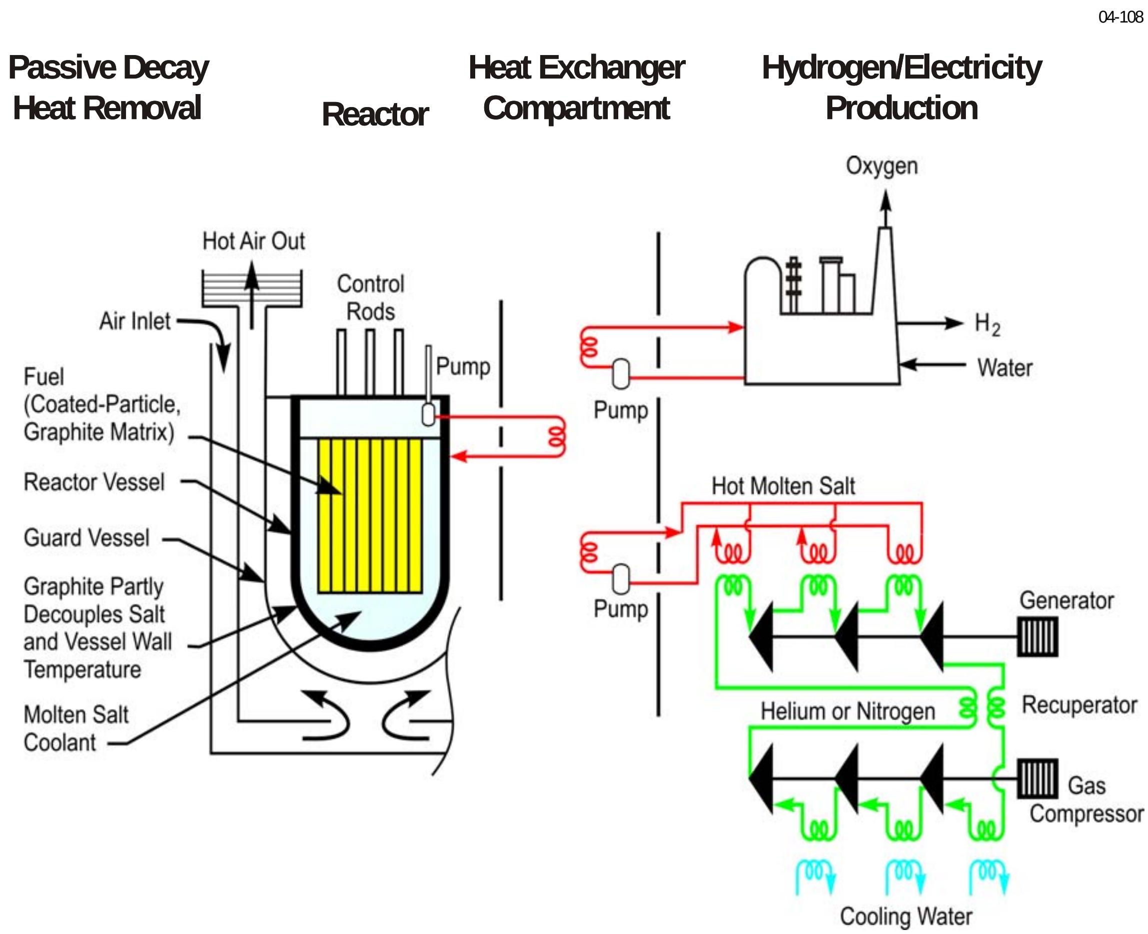


**Fig. 1.** Schematic of the AHTR for electricity production.

In the current preconceptual design, the AHTR has an annular core through which the coolant flows downward. The molten salt coolant flows upward through the nonfuel graphite section in the middle of the reactor. 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.

**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<br>700°C/800°C<br>670°C/705°C | Electricity (output at different peak coolant temperatures) | 1357 MW(e) at 1000ºC<br>1235 MW(e) at 800ºC<br>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 |

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 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.

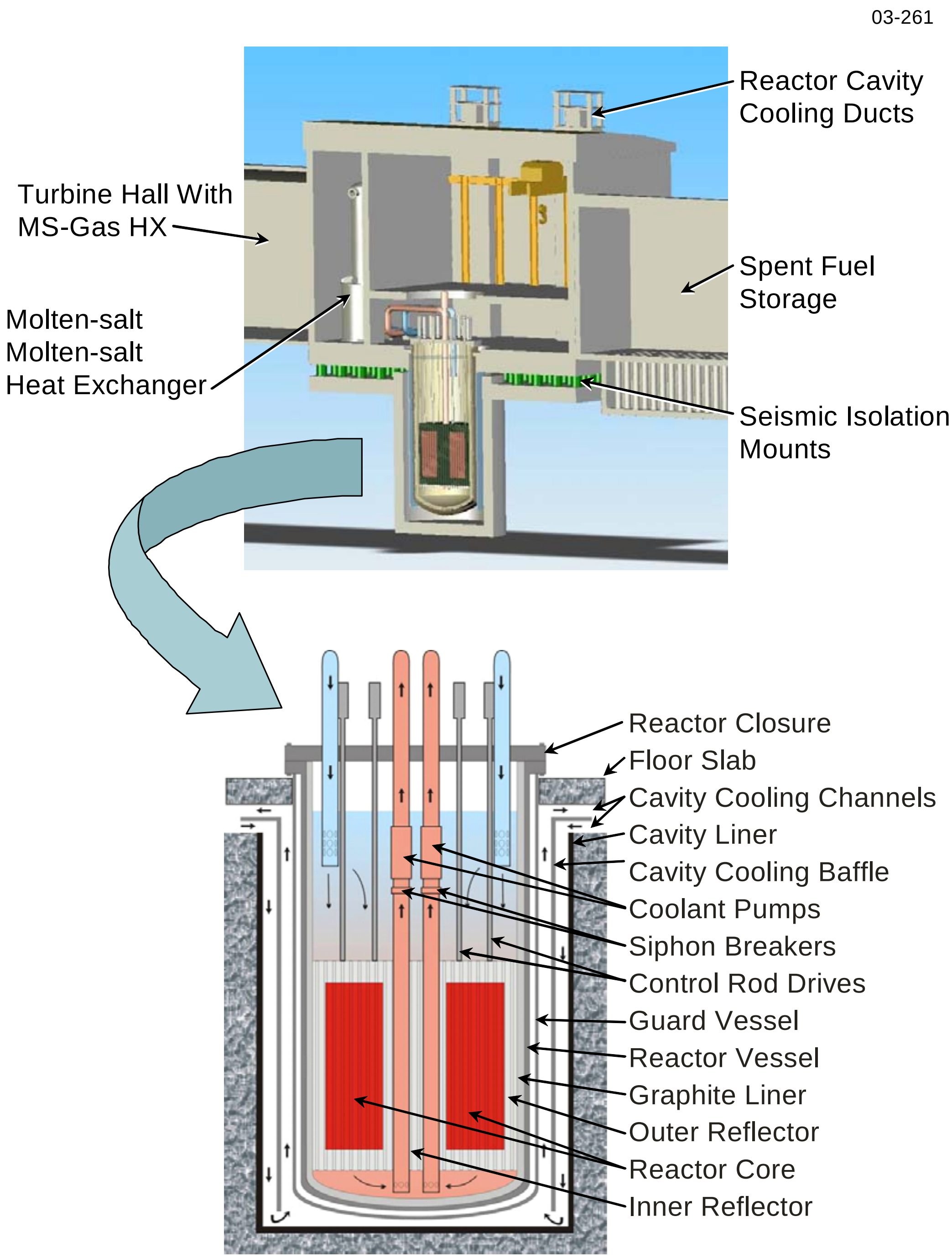


**Fig. 2.** Schematic of the AHTR nuclear island and vessel.

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. Under accident conditions such as a loss-of-forced-cooling accident, natural circulation flow of molten 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 hours after loss of pumped coolant flow with a peak reactor vessel temperature of ~750°C at ~45 hours. The average core temperature in this accident rises to approximately the same temperature as the hottest fuel during normal operations.

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 or to a heat exchanger in the reactor vessel. 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 size 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 basic fuel type and the molten salt coolant has a low neutron-absorption cross section, the reactor core physics and fuel cycle options are generally similar to those for helium-cooled high-temperature reactors. Reactor power is limited by a negative temperature coefficient, control rods, and other emergency shutdown systems.

Several molten 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 ($2^7LiF-BeF_2$) 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 molten salt—in contrast to the AHTR, in which solid fuel is used along with a clean molten salt coolant.

This salt is well understood and has a negative coolant void coefficient. Several other salts, such as sodium-zirconium fluoride and other alkali-zirconium fluoride salts, have operational and cost advantages. However, with the core designs of traditional gas-cooled-reactors, these other salts have small positive void coefficients. While undesirable, a small positive void coefficient may be acceptable because of the small predicted consequences of a voiding accident. 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 void 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. Earlier versions of CANDU (heavy-water-moderated) and Hanford-type (water-cooled, graphite-moderated) reactors had positive void coefficients. The Advanced CANDU Reactor and Hanford-N Reactor have negative void coefficients. The same core design strategies used in these reactors may allow the use of a wide variety of salts in the AHTR with negative void coefficients.

## 3. MOLTEN SALTS

A significant experience base exists for only three high-temperature liquids: molten iron, molten glass, and molten 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 molten 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). Molten salt reactors (MSRs) were to provide the heat source, with the heat transferred to a jet engine via an intermediate heat-transport loop. In a MSR, the uranium is dissolved in the molten salt coolant. It is a liquid-fuel reactor. 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 molten 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 molten 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 a 1000 technical reports were produced. Unlike the MSR, the AHTR uses solid fuel and a clean molten 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 molten fluoride salts. Only helium and molten 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 molten salts were relatively noncorrosive. One 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 molten 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 molten salt mixtures can be used (Forsberg, 2004a). All have low nuclear cross sections and melting points between 350 and 500°C. The optimum salt involves tradeoffs 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. Boron-11 has a very low nuclear cross section (0.05 barns).

## 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 molten 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 liquid-cooled high-temperature 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.

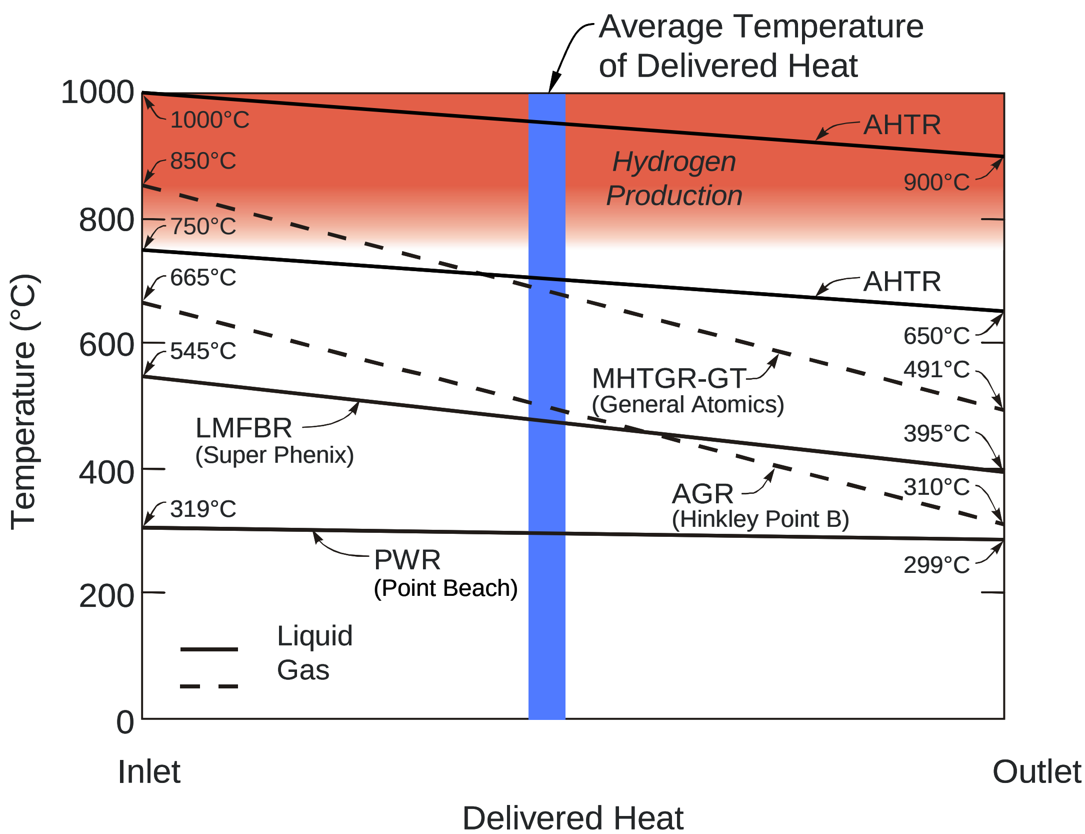


**Fig. 3.** Temperature of delivered heat for different reactors.

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 significantly 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 S-PRISM and the gas turbine−modular helium reactor (GT-MHR)] based on the relative size of systems and quantities of materials. *This approach provides relative, but not absolute, costs.* Only the construction of multiple reactors can provide reliable absolute costs. The lower capital costs are a consequence of several factors: economics of scale [a 2400-MW(t) reactor vs four 600- or 1000-MW(t) reactors], passive safety in a large reactor system, and higher thermal efficiency (see Sect. 4.3). Preliminary assessments indicate the potential to raise power outputs to ~4000 MW(t).

**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<br>$1681/kW(e) | GT-MHR<br>$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.

Because the facility designs of the AHTR and fast reactors are similar, an evaluation is under way to understand the costs of the AHTR relative to those of large sodium-cooled fast reactors such as the Japanese 1500-MW(e) reactor (Ichimiya *et al.*, 2003). Several factors indicate the potential for significantly lower costs per kilowatt hour (electrical).

- *Higher efficiency*. The higher temperature implies higher efficiency (~50% vs 42%). 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 smaller decay-heat-removal systems.
- *Passive decay heat removal*. The higher AHTR temperatures, combined with the high-temperature fuel, enable the development of passive safety systems for large reactors. Passive safety systems have the potential for lower costs.
- *Reduced containment requirements*. The molten salt coolant avoids the potential for steam−sodium interactions, absorbs radionuclides that escape the fuel, and eliminates highly energetic accidents, all of which lower containment requirements.
- *Reduced equipment sizes*. Volumetric heat capacities for molten salts are several times larger than those for sodium. This reduces the size of pipes, valves, and heat exchangers per unit of energy transferred.
- *Transparent coolant*. Unlike liquid metals, molten salts are transparent. This simplifies maintenance and inspection of the primary system with significant cost advantages.

## 6. RESEARCH AND DEVELOPMENT REQUIREMENTS

About 70% of the R&D required for the AHTR is shared with that for helium-cooled high-temperature 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, (2) optimization of core design, (3) refueling and maintenance operations in the reactor vessel at 350 to 500°C, and (4) selection of the preferred molten fluoride salt. 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.

## 7. CONCLUSION

The AHTR is a new reactor concept for the efficient production of electricity and hydrogen. Initial analyses indicates its potential for 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 molten-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, major technical uncertainties remain relative to helium-cooled and sodium-cooled advanced reactor systems.

## 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.

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

Ichimiya M., Mizuno T., and Konomura M. (2003), A Promising Sodium-Cooled Fast Reactor Concept and Its R&D Plan, Proc. Global 2003, American Nuclear Society, La Grange Park, Illinois.

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.

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

Peterson P. F., Zhao H., and Fukuda G. (Dec. 5, 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.

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.

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