Revision 0
June 2004

# The Very High Temperature Reactor: A Technical Summary

Prepared for

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# 1 Introduction

## 1.1 Background

In 2000, the United States Department of Energy formed the Generation IV International Forum (GIF) to advance nuclear energy in order to fulfill future energy needs. The GIF has categorized the goals for future nuclear power into four areas, which are referred to throughout this report.

– **Sustainability**: Sustainability is the ability to meet the energy needs of the present generation while enhancing the ability to meet the energy needs of future generations indefinitely. Sustainability goals focus on waste management and resource utilization. The sustainability of GENIV systems also includes extending nuclear power into other energy areas, such as transportation, by using nuclear process heat to manufacture other energy products, such as hydrogen.

– **Economic Competitiveness**: Economic goals consider competitive costs and financial risks. Economic goals focus on reducing operating and capital costs through increased efficiency, design simplification, advances in fabrication and construction techniques, and possible standardization and modularization.

– **Safety and Reliability**: Safety and reliability goals include safe and reliable operation, improved accident management and mitigation, investment protection, and reduced off-site response. The focus for GENIV systems is on the use of inherent safety features and designs.

– **Proliferation Resistance and Physical Protection**: Proliferation resistance and physical protection goals consider methods for controlling and securing nuclear material and nuclear facilities against unintentional and intentional actions.

Through the efforts of ten countries, the GIF released a roadmap (cf. Ref. 2) outlining the research and development necessary for six of the most promising future reactor designs. In the upcoming years, each member state of the GIF will focus their efforts on the reactor design(s) which best fulfills their future energy needs.

To outline their efforts, the U.S. has developed an implementation strategy for advancing the Generation IV Roadmap (Ref. 13). This strategy consists of two priorities for the U.S. Generation IV Program:

– Develop a Next Generation Nuclear Plant (NGNP) to achieve economically competitive energy products, including electricity and hydrogen, in the mid-term

- Develop a fast reactor to achieve significant advances in proliferation resistance and sustainability in the long term

This document focuses on the Next Generation Nuclear Plant (NGNP), which will be constructed at the Idaho National Engineering and Environmental Laboratory (INEEL) by 2015. Although still in the early conceptual phase, the NGNP is expected to be based on the Very High Temperature Reactor (VHTR), one of the six proposed Generation IV concepts. The feasibility of the VHTR has been demonstrated in past gas-cooled reactors (see Appendix B). The NGNP will instead be a pilot facility for commercial deployment and licensing of future VHTR units. The ultimate goals for commercial VHTR units are:

- Generate electric power at a cost of less than 1.5 cents/kW-hr
- Produce hydrogen at a cost of less than $1.50/gallon-gasoline equivalent
- Cost between $500-$1000/kW to construct

The VHTR is a suitable candidate for the NGNP due to the high efficiency electrical generation and hydrogen production provided by its high operating temperatures. The VHTR is a helium-cooled, graphite-moderated, thermal neutron spectrum reactor with a coolant outlet temperature of 1000 °C or above. It will be a mid-size reactor with a thermal power of about 600 – 800 MWth. The final reactor power and core configuration will be designed to guarantee passive decay heat removal during accidents in order to preclude radioactive release. The reactor will utilize coated fuel particles in a once-through low-enriched uranium fuel cycle capable of very high burnup. The motivation for basing the NGNP on the VHTR concept stems from the high outlet temperatures. The high outlet temperature of the VHTR affords high efficiency (> 50%) electrical generation and the use of nuclear power for potential process heat applications, specifically carbon-free hydrogen production using nuclear heat.

## 1.2 PURPOSE

This report provides a technical description of the VHTR and a summary of the design and development challenges facing the VHTR. Section 2 provides an overview of the general VHTR design characteristics for the NGNP, planned to be built at the INEEL. Section 3 discusses the technical issues which must be resolved for the NGNP and for commercial VHTRs. Next, Section 4 contains the VHTR R&D timelines of the GIF and INEEL. The appendices provide additional background and reference information. Appendix A lists the acronyms employed throughout the report. Appendix B contains a brief survey of existing gas-cooled reactor designs, and Appendix C briefly describes the other GENIV advanced reactor concepts chosen by the GIF.

# 2

# The Very High Temperature Reactor

The term Very High Temperature Reactor (VHTR) loosely covers any reactor design with a coolant outlet temperature of 1000 °C or above. The term typically refers to the next step in the evolutionary development of high-temperature gas-cooled reactors (HTGRs). The Next Generation Nuclear Plant (NGNP) refers specifically to the advanced reactor system which will be constructed at the INEEL. Although a final decision has not yet been made, the NGNP is expected to be based on the GENIV VHTR concept.

## 2.1 DESIGN BASIS

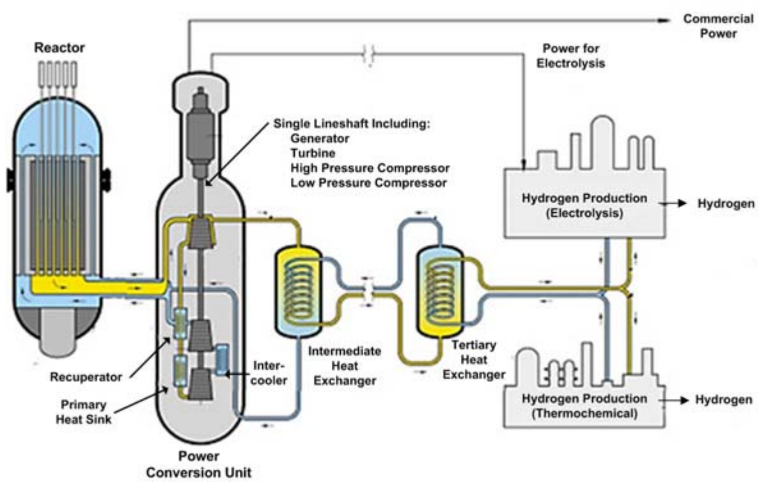


**Figure 2.1-1. The NGNP Reactor Layout (Ref. 18)**

Gas-cooled reactors (GCRs) have been investigated since the early days of nuclear power. The early gas reactors were used commercially in the United Kingdom, but were overshadowed elsewhere by LWRs and other designs. International interest in gas-cooled reactor technology focused on development rather than deployment. This led to the construction of a number of high temperature gas cooled reactor (HTGR) prototype and demonstration plants in Britain, Germany, and the U.S. The focus of these plants was on evolutionary increases in coolant temperature and plant efficiency. An overview of the development of HTGRs is provided in Appendix B, "Brief Survey of Gas-Cooled Reactor Designs."

Recently, interest in gas-cooled reactors has been renewed, and a number of HTGR designs have been developed for near-term deployment (i.e., Generation III+ reactors). These include the Pebble Bed Modular Reactor (PBMR) project, led by Eskom, and the Gas Turbine Modular Helium Reactor (GT-MHR) project, led by General Atomics. The VHTR concept proposed by the GIF is an evolutionary extension beyond the near term HTGR designs. Consequently, a large portion of the VHTR preliminary design is based directly on the existing near term designs.

Figure 2.1-1 depicts the reference VHTR conceptual design proposed by the INEEL for the NGNP project. The basic features of the VHTR are similar to past HTGRs. Namely, the VHTR is a helium-cooled, graphite-moderated reactor with a ceramic core and TRISO coated fuel particles. The VHTR extends current HTGR technology to increase the coolant outlet temperature to 1000 °C or above from past temperatures of about 850 °C. The increase in temperature allows more efficient electrical generation and better thermal conditions for process heat applications.

## 2.2 REACTOR

The VHTR is still in the early conceptual design phase, and a specific reactor design has not yet been developed. As a result, the current VHTR core description is based largely on its Generation III+ predecessors. The GT-MHR by General Atomics is the basis for the prismatic VHTR; the PBMR by PBMR (Pty.), Ltd., is the basis for the pebble-bed VHTR. This section will address the general structure of the two designs.

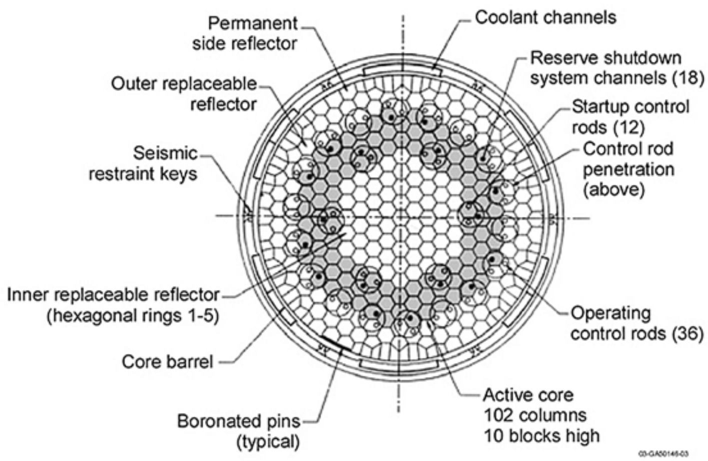


**Figure 2.2-1. The GT-MHR Reactor Core (Prismatic VHTR Reference Core) (Ref. 15)**

The VHTR will use an annular core configuration. In a prismatic core (Figure 2.2-1), hexagonal moderator and fuel blocks are arranged to form an inner graphite reflector (rings 1 – 5), a center active fuel core (rings 6 – 8), and an outer replaceable graphite reflector (rings 9 – 10). The active core is approximately 26 ft in height and consists of 102 fuel columns, each of which is a stack of ten fuel blocks (1,020 fuel blocks total). In addition to the replaceable graphite components, the prismatic core also includes a permanent side graphite reflector, vessel coolant channels, and the core barrel. Helium enters the reactor core and flows up through the vessel coolant channels before flowing downward through the integral coolant channels in the fuel assemblies. This exposes the core barrel to the cooler inlet helium, rather than the hotter outlet helium, thereby reducing the operating temperature of the barrel material.

The arrangement of the pebble bed VHTR core is similar to the prismatic core. The prismatic fuel blocks in the active annular core region are replaced by mobile fuel pebbles, each approximately the size of a tennis ball. These pebbles continuously circulate downward through the core driven only by gravity. The pebbles are removed from the bottom of the core, and their total burn-up is measured. Active pebbles are returned to the top of the core, while spent pebbles are diverted to storage. Similar to the prismatic core, the inner and outer reflectors are constructed from static moderator blocks. This simplifies the fuel handling process, but the reflectors will require replacement at least once throughout the life cycle of the plant. The pebble-bed core requires additional fuel handling systems, which increase costs, but can be refueled while still online. In addition, the pebble-bed core requires additional analysis to predict and verify the pebble dynamics. Both reactor designs use control rods for reactivity control and shutdown, although the pebble-bed design also uses small absorber pebbles which are inserted into the core for emergency shutdown.

As the VHTR design progresses, the fuel loading and core geometry will be optimized to provide the coolant temperatures, inherent safety, and capability for high burnup that are the goals of future Generation IV reactor systems. Examples of parameters which may be adjusted are the dimensional parameters of the inner reflector, active core, and outer reflector and fuel parameters such as the enrichment and packing fraction. Additional vessel cooling channels may also be required in order to maintain the core barrel temperatures within acceptable material limits.

Despite differences between the fuel forms, the prismatic and pebble-bed VHTR reactors share similar safety characteristics. The two designs use a mostly ceramic core which has a very high thermal capacity and can withstand extremely high temperatures under accident conditions. This is a major part of the inherent safety of the VHTR, as the core itself can dissipate a large amount of decay heat before the fuel thermally degrades. On the other hand, the non-ceramic components in the core such as the control rod sheaths and the core barrel typically suffer from problems with high-temperatures, which are discussed further in Section 3.2. The low volume fraction of fissile material within the fuel results in a low core power density. The moderator, coolant, and fuel provide a strong overall negative temperature coefficient of reactivity. These features give the VHTR reactor a large amount of thermal stability and reactivity control, which provide inherent safety under accident conditions.

Due to the TRISO fuel (see Section 2.4), the pressure vessel is not required to be as leak-tight or robust as traditional LWR vessels. In a departure from the near-term reactors used as a design basis, the VHTR may use a modular pre-stressed cast iron pressure vessel (PCIV) to reduce construction time and costs, pending further development and validation. Typical steel vessels have size limitations for manufacture and transport, which can be overcome with a modular approach. Cast iron materials may also have better high-temperature characteristics than typical pressure vessel steels, although the high-temperature properties of candidate materials are mostly unknown and require further research. The PCIV is a modular pressure vessel which will be shipped to the site in pre-fabricated segments. The segments are assembled and pre-stressed on site using axial and circumferential tendons and a bolted inner liner for leak tightness. This pressure vessel design eliminates the possibility of a sudden catastrophic rupture, and the superimposed compressive stresses limit the progression of large cracks.

## 2.3 POWER CYCLE

The VHTR will employ a direct Brayton cycle for the generation of electricity. For the production of hydrogen, the VHTR will use an indirect cycle with an intermediate heat exchanger (IHX).

First generation gas-cooled reactors included steam generators to accommodate an indirect Rankine cycle for power generation. At the time, the closed Brayton cycle had not been extensively developed, since the majority of gas turbine applications used an open cycle. For nuclear applications, a closed cycle is required in order to retain the process gas for radiological reasons. Recently, the closed Brayton cycle has undergone significant development in the aerospace industry, which has demonstrated its high efficiency. In addition, the direct Brayton cycle is much simpler than an indirect Rankine cycle, which leads to a number of safety and cost benefits. Therefore, the VHTR will use a direct Brayton cycle in order to maximize the safety, simplicity, and economy of its electrical generation.

For process heat applications like hydrogen production, the VHTR will use an indirect cycle with an intermediate heat exchanger (IHX) to supply heat to the process application. An indirect cycle isolates the process heat loop from the nuclear reactor, which allows the process heat systems to be designed and built to non-nuclear standards. In addition, the thermal conditions required for process heat applications can vary significantly from those within the reactor core. These variations include temperature, pressure, and the frequency and/or magnitude of thermal transients. Therefore, an indirect cycle is needed in order to provide a thermal interface between the reactor and the chosen process heat application.

## 2.4 FUEL AND THE FUEL CYCLE

The VHTR will build upon the fuel developed for past HTGRs, the triple-isotropic coated fuel particle (TRISO CFP). These particles are dispersed within a graphite matrix to form fuel elements, the final form of which will be either prismatic or spherical (i.e., pebbles). The final selection will be made pending completion of the conceptual designs for each. While this arrangement results in a very flexible fuel design which can accommodate fast neutron conditions, the fuel in the VHTR will be used in a once-through fuel cycle with a thermal neutron spectrum.

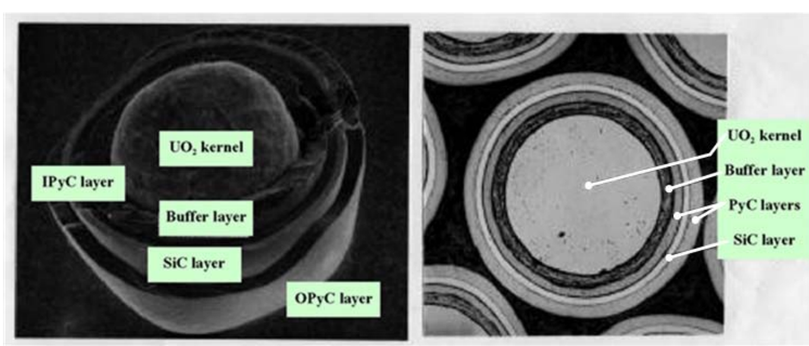


**Figure 2.4-1. TRISO (Triple Isotropic) Coated Fuel Particle (Ref. 11)**

TRISO fuel consists of a low-enriched uranium oxycarbide (about 15% UCO) fuel kernel surrounded by three layers of pyrolytic carbon (PyC) which protect an additional ceramic layer (see Figure 2.4-1). Fission products formed from uranium oxycarbide contain no free oxygen, which could otherwise aggravate chemical degradation of the ceramic layer. The ceramic layer, composed of silicon carbide (SiC), acts as a miniaturized pressure vessel that completely retains all fission products. The layer begins to lose its integrity above approximately 1600 °C, which represents the limiting fuel temperature under accident conditions. Two inner layers of PyC protect the SiC layer against chemical attack from fission gases and against mechanical stress due to irradiation swelling of the fuel kernel. An outer layer of PyC protects the SiC layer from mechanical failure during handling and operation. These particles are mixed with graphite powder and binders before being shaped and molded into the final fuel element.

The form of the final fuel assembly has taken two different forms in past designs. German HTGRs, such as the AVR and THTR, used spherical fuel assemblies approximately 5 – 6 cm in diameter. These have been traditionally referred to as "pebbles." In HTGRs such as Peach Bottom and FSV, U.S. designers used the "prismatic block" fuel assembly in which the CFPs are formed into cylindrical fuel compacts before being inserted into hexagonal graphite fuel elements. Modern pebble designs include the Chinese HTR-10 test reactor and the South African PBMR Generation III+ reactor. Modern prismatic designs include the Japanese HTTR test reactor and the General Atomics GT-MHR Generation III+ reactor.

Both fuel configurations have their advantages. The pebble designs typically include an automated pneumatic fuel handling system which allows more flexible refueling options but at greater expense. This greater expense may be offset by the potential for continuous online refueling and decreased down time. The block design consists of integral coolant channels. In the pebble design, the helium coolant flows between the interstices of the pebbles. The integral coolant channels allow better core cooling, which in turn allows greater power density and total core power with block fuel. The final choice of fuel element will be made following the point designs of each.

The TRISO fuel design results in a very flexible fuel arrangement by essentially decoupling the cooling geometry and neutronic optimization of the fuel. The fuel assembly shape, core configuration, number of coolant channels, and packing fraction of fuel particles can all be adjusted independently for different power levels, outlet temperatures, and fuel cycles. For example, initial optimization studies have illustrated that greater packing fractions may extend the overall cycle length due to additional self-shielding within the fuel (Ref. 15).

The fuel flexibility can also accommodate other fuel cycles, such as a closed fuel cycle with a fast neutron spectrum. Over the next 50 years, the once-through open fuel cycle is considered the most economical and proliferation-resistant choice for commercial reactors (Ref. 9). In addition, the current SiC layer in TRISO fuel particles has increased susceptibility to fission product release under the fast neutron conditions needed for a closed fuel cycle. The closed fuel cycle also requires further development prior to commercialization and would delay other design efforts. For these reasons, the VHTR is being designed for a once-through fuel cycle with a thermal neutron spectrum.

## 2.5 BALANCE OF PLANT

### *2.5.1 Primary Heat Transfer Loop*

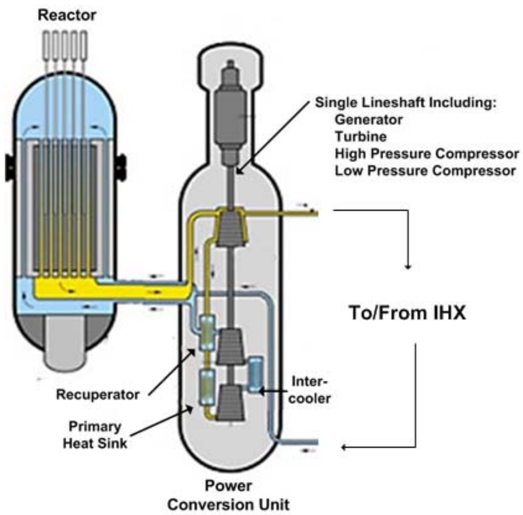


**Figure 2.5-1. NGNP Primary Heat Transfer Loop**

The VHTR will have a primary heat transfer loop which will be used to generate electricity using a closed Brayton cycle. Cool helium flows into the bottom of the core through the outer annulus of the cross duct between the reactor and power conversion unit. The helium then flows upward along the inner surface of the core barrel, through the vessel coolant channels within the core. The coolant then is heated as it flows back down through the active core. The hot helium flows out of the reactor through the inner pipe of the cross duct and into the power conversion unit. After passing through the turbine, a portion of the helium will be diverted to the IHX to heat the secondary loop, before returning back through the remainder of the power conversion unit.

The proposed configuration for the power conversion unit (PCU) is based on the configuration used in the near-term GT-MHR design. In order to reduce onsite construction time and cost, the power conversion unit (PCU) will be factory-fabricated in either modular units or as one complete self-contained module. The PCU will contain all the necessary turbomachinery for electrical generation. The turbine, generator, and compressors will be installed on a single lineshaft in a vertical orientation. This provides a smaller footprint, which is advantageous since both the reactor and the PCU will be situated below-grade, and also minimizes the length of the cross duct between the reactor and PCU. The single lineshaft reduces the number of required bearings and overall complexity of the turbomachinery, but introduces weight and alignment issues. The near-term PBMR reactor offers an alternate configuration with the turbomachinery on multiple horizontal shafts. The final configuration choice for the PCU requires a better understanding of the advantages and disadvantages of each design.

### 2.5.2 Secondary Heat Transfer Loop

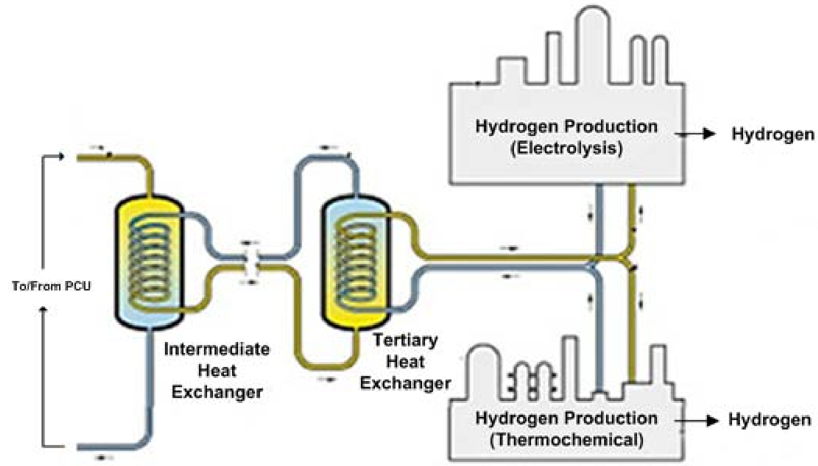


**Figure 2.5-2. NGNP Secondary Heat Transfer Loop**

The VHTR includes a secondary heat transfer loop to provide process heat for non-electrical energy products, of which hydrogen is the primary candidate. The NGNP will also have a tertiary heat transfer loop, as shown in Figure 2.5-2. This simple loop adds an additional heat exchanger to further isolate the reactor and hydrogen loops and mitigate contamination and thermal transients. The NGNP will evaluate the cost-benefits of this additional loop, which may be eliminated in future VHTR generations (or the final NGNP design) for simplification and cost-reductions.

Helium, or another suitable heat transfer medium such as molten salt, enters the IHX where it is heated by helium from the primary loop. Due to the low thermal transfer of helium gas, the IHX requires a very large surface area for heat transfer. This would require an extremely large and uneconomical tube and shell heat exchanger. Instead, the VHTR will use a compact plate and fin or a "printed circuit" heat exchanger, pending qualification of a viable design.

The remainder of the secondary loop will depend on the specific process heat application. Although the long-term vision for the VHTR involves many potential process heat applications, the NGNP will demonstrate nuclear hydrogen production. The NGNP will include two hydrogen production loops in order to investigate and develop different hydrogen production systems. Hydrogen production using nuclear process heat is discussed in Section 2.6. Other potential process heat applications are discussed in Section 2.7.

### 2.5.3 Reactor Cavity Cooling System

A reactor cavity cooling system (RCCS) is needed in order to remove decay heat from the core during accident events. A design goal of the VHTR is to use its intrinsic safety features to make an active RCCS unnecessary. A passive heat removal system will be used to limit core and fuel temperatures in order to prevent structural damage or radioactive release. Passive heat removal methods do not require operator intervention or external power and therefore are more reliable. Figure 2.5-3 illustrates the passive RCCS of the near-term GT-MHR. The primary decay heat removal methods are radiation and conduction to the reactor cavity structure and to the earth. The RCCS contains cooling panels and an intake/exhaust duct to allow naturally convecting air to remove additional heat. The VHTR will employ a similar passive system, if such a system can be shown to maintain vessel and fuel temperatures within acceptable limits.

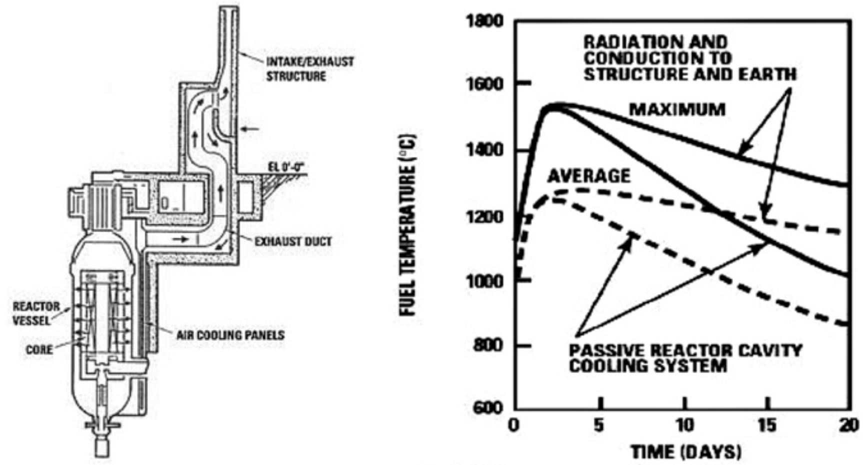


**Figure 2.5-3. GT-MHR Passive RCCS and Peak Accident Core Temperatures (Ref. 19)**

One issue in finalizing the VHTR design is to find the optimal balance between economics and safety. That is, the cost-benefits of higher power densities and a smaller reactor size must be balanced with the RCCS capability and vessel temperature capabilities to maintain post-accident temperatures below acceptable limits. Therefore, the amount of decay heat which the RCCS is capable of removing becomes a key element in the design. Similar to the vessel material temperature limit (see Section 3.2.3), improving the capability of the RCCS will improve the cost-safety envelope of the VHTR.

### *2.5.4 Other BOP Equipment*

Helium gas was chosen as the coolant for the VHTR because of its chemical and radiological inertness. The helium coolant will not corrode components or equipment. However, the high temperatures of the VHTR aggravate chemical attack from impurities in the helium. This issue is discussed further in Section 3.2.2, below. To prevent this, the VHTR will require systems to monitor and purify the helium coolant. These systems will most likely not be safety-related, but will still be important in preventing long-term structural damage and failure.

Past HTGR units, specifically Fort St. Vrain, used traditional bearings in the turbomachinery. As a result, FSV experienced extensive downtime resulting from contamination problems due to lubrication ingress. The VHTR will use magnetic bearings, a relatively recent development, in order to prevent this. However, magnetic bearings are active components which require power to create the electromagnetic field. Therefore, the VHTR will require additional catcher bearings, which must be designed for potential drops and coast-downs.

## 2.6 HYDROGEN PRODUCTION USING NUCLEAR PROCESS HEAT

The major impetus in the U.S. for the VHTR is its potential for hydrogen production. Hydrogen represents a key element in the future U.S. energy policy for reduced carbon emissions and increased energy independence. The long-term vision of the GIF (Ref. 2) is to allow a 600 MWth VHTR dedicated to hydrogen production to produce over 2 million cubic meters of hydrogen per day (details such as assumed efficiency, etc. were not provided). This is the energy equivalent of over 160,000 gallons of gasoline per day.

Three categories of methods are typically considered for generating hydrogen. These are steam reformation of methane, electrolysis, and thermochemical cycles. Of these, the U.S. will investigate conventional electrolysis, high-temperature steam electrolysis, and a variety of thermochemical cycles for use in the NGNP hydrogen production plant, due to their expected roles in the future hydrogen economy.

### *2.6.1 Steam Reformation*

Steam reforming of methane is the current process of choice for large-scale hydrogen production, and may have a role as an early centralized production method. However, methane is already a high quality fuel, and steam reformation is a mature process which does not require further development. Furthermore, the process results in carbon emissions and diverts natural gas from residential use. Therefore, steam reformation is unable to fulfill the future U.S. energy needs since it is not sustainable in the long-term. This method may fulfill large-scale production needs during the early periods of the hydrogen economy, but will not be considered for the NGNP hydrogen production plant since it will be phased out in the long-term and is already well-developed.

### *2.6.2 Electrolysis*

Conventional water electrolysis is a well-established process and is the traditional benchmark for other hydrogen production processes. The overall production process has low efficiency due to the typical inefficiencies absorbed with electrical generation. However, electrical energy is easily transported. This makes conventional electrolysis ideal for distributed hydrogen production, which will be most profitable in the initial stages of the hydrogen economy. Before significant demand for hydrogen exists, distributed production will avoid hydrogen transportation costs, which can be significant with currently available storage technology (Ref. 16). Therefore, conventional electrolysis will play a key role in enabling early distributed production in the future hydrogen economy.

Steam electrolysis uses thermal energy to produce high-temperature steam prior to electrolysis. This displaces a portion of the required electrical energy with thermal energy, which improves the overall efficiency. This process requires a separate high temperature heat source which is disadvantageous for distributed production. However, steam electrolysis will play an important role as an early, efficient, and emission-free (when coupled with nuclear power) hydrogen production method. Both conventional electrolysis and high-temperature steam electrolysis will employ modular scaling to allow increased production capacity. In addition, each has potential applications in peak-shaving by utilizing electricity from the grid during off-peak times. Both methods are undergoing further development to increase their performance.

Both conventional electrolysis and high-temperature steam electrolysis are being considered for application in the NGNP, since they will be key elements of the future hydrogen economy and can benefit from the VHTR. The cost of electrolysis is highly dependent on the cost of electricity. In this area, the VHTR will provide modest benefits to both conventional and steam electrolysis since its high-temperatures and direct Brayton cycle result in highly efficient (> 50%), and therefore low-cost, electrical generation. The VHTR can also be the source of high-temperature process heat for steam electrolysis, providing emission-free thermal energy which would otherwise have come from fossil fuels.

### *2.6.3 Thermochemical Cycles*

The direct pyrolysis of water requires temperatures greater than 4000 °C. Thermochemical cycles are able to decompose water into hydrogen and oxygen at significantly lower temperatures. This is achieved by using chemical reactions to initially dissociate water and then splitting the intermediate compounds with either heat or electricity. Thermochemical cycles have been widely investigated in the past, and a large number of different cycles exist today. However, most require significant development for commercial application. These processes are expected to provide centralized, large-scale, efficient, and emission-free hydrogen production when coupled with high-temperature nuclear reactors.

The NGNP program is currently investigating a number of thermochemical cycles which belong to two general families. The highest priority is given to the sulfur-based family consisting of the sulfur-iodine cycle, the hybrid sulfur cycle, and the sulfur-bromine cycle. Lower priority is given to the calcium-bromine family of cycles, which will not be discussed here.

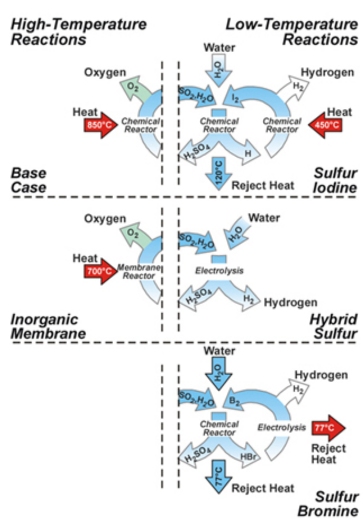


**Figure 2.6-1. Sulfur-Based Thermochemical Cycles (Ref. 14)**

The sulfur-based cycles are illustrated in Figure 2.6-1. The sulfur-iodine cycle is generally presented as the ideal long-term goal for hydrogen production since it is the most efficient production method. It is an all-liquids-and-gases cycle with three thermochemical steps. The sulfur-iodine cycle is the most efficient of the sulfur-based cycles and has been demonstrated at the laboratory-scale at JAERI. However, the separation techniques employed in the process require further development for commercial scale-up and system design. The hybrid sulfur cycle is also an all-liquid-and-gases cycle, but with one thermochemical step and one electrolytic step. With only two steps using only two sulfur compounds, it is the simplest thermochemical process. The last sulfur-based cycle, the sulfur-bromine cycle, is being considered as a contingency since this process is more complicated and less efficient than the hybrid sulfur cycle.

## 2.7 OTHER PROCESS HEAT APPLICATIONS

The impetus for higher operating temperatures consists of higher thermal efficiency and the eventual expansion of nuclear energy beyond electrical generation. The latter motive manifests itself in the desire to expand the potential for nuclear process heat (NPH) in industrial applications, beyond hydrogen production in the near-term. Figure 2.7-1 shows the required temperatures for industrial applications and the coolant temperatures of various reactor designs. Note that the upper limit of 1500 °C is a very optimistic goal for the VHTR which will not be realized until later generations. NPH applications range from desalination and district heating on the low-end of the temperature scale to iron and glass manufacture on the high-end. Research conducted during the Nuclear Steelmaking System (NSS) Project in Japan during the 1970s showed that most of the high temperature processes could be modified to use temperatures around 1000 °C. One of the long-term goals of the VHTR includes possible deployment at industrial park sites. This would allow the VHTR to replace fossil fuels in the long-term as a source of high-temperature process heat for energy-intensive industrial processes in order to reduce carbon emissions.

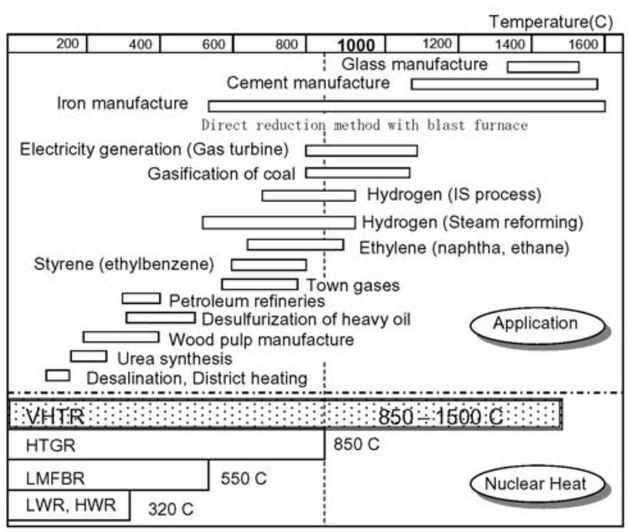


**Figure 2.7-1. Nuclear Process Heat Applications and Required Temperatures (Ref. 4)**

## 2.8 SAFETY AND PROLIFERATION RESISTANCE

Past and near-term HTGRs have illustrated the high level of safety inherent to gas-cooled reactors. The graphitic core structure, helium coolant, and coated fuel particles allow the VHTR to withstand accident temperatures without structural damage or fission product release. This provides a significant amount of inherent safety which eliminates the need for active, and expensive, safety systems such as those in current LWRs. Also, HTGRs have inherently better proliferation resistance compared to current LWRs due to their dilute fuel form and difficulty in reprocessing. This section discusses the safety and proliferation resistance inherent to all gas-cooled reactor designs, since the evolutionary changes to the VHTR do not provide any additional inherent safety or proliferation resistance. To the contrary, the higher temperatures and hydrogen production plant present unique safety challenges which must be overcome to maintain the inherent safety of HTGRs; these are discussed further in Section 3.

The graphite core of HTGRs has a high thermal conductivity, which aids in preventing hotspots from forming within the core. The high thermal capacity of graphite combined with the low core power provides a relatively long delay in the thermal response during loss of coolant accidents or reactivity insertions. The maximum fuel temperature is not expected to occur for several days following a loss of coolant (Ref. 2), providing significant time for operators to take action. Figure 2.8-1 shows the peak core temperatures during a loss of coolant computer analysis for different VHTR core layouts. In each case, the peak temperatures are not reached for approximately 2 – 3 days. In addition, the helium coolant, graphite moderator, and TRISO fuel combine to give the core a strong negative temperature coefficient of reactivity. This provides power and temperature attenuation during accidents, since the fission reaction rate (i.e., the rate of heat generation) slows as the core temperature increases.

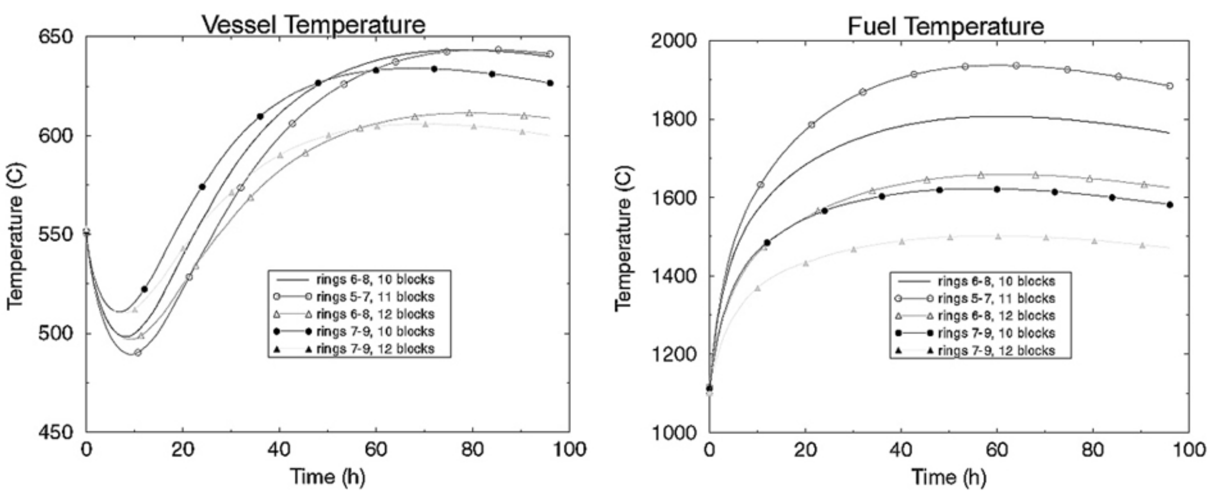


**Figure 2.8-1. VHTR Peak Temperatures During Accident (Ref. 23)**

Initial safety demonstration tests are being performed at the HTTR in Japan (Ref. 12). Reactivity insertion and partial coolant flow reduction tests have already been completed, and they confirm the core safety aspects mentioned above. During the reactivity insertion, the reactor power slowly increased. Once insertion was completed, the increasing fuel and moderator temperatures immediately began to slow the fission rate and reduce reactor power. During this test, the reactor power peaked but was attenuated within an hour to a small net increase. The test illustrated that the reactor power during a reactivity insertion event can be regulated solely by the negative reactivity feedback of the core, without requiring the use of active reactor power control systems. A similar effect was seen when test personnel reduced coolant flow. The initial reduction in coolant flow reduces the amount of heat transfer from the core. The resultant increase in core temperature causes a reduction in the rate of heat generation due to negative reactivity feedback. Thus, both the reactivity insertion and the partial coolant flow reduction events illustrate the intrinsic safety and stability of HTGR reactors.

The self-attenuating reactivity coefficient of HTGRs results in eventual power stabilization during accidents and transients. Therefore, the major goal of accident mitigation and prevention becomes the limitation of maximum core and fuel temperatures. The ceramic and non-ceramic core components will be designed to have sufficient high-temperature capabilities to preclude structural damage. Thus, the limiting case becomes the degradation of the fuel coatings at high temperatures and subsequent release of fission products. This is the motivation for limiting the maximum fuel temperature during accidents below the degradation temperature of the fuel (i.e., 1600 °C for SiC TRISO fuel).

The TRISO coated fuel particles represent another intrinsic safety feature of HTGRs. As mentioned before, each fuel particle is essentially its own pressure vessel able to retain fission products. This results in very little radioactive release and plate-out during operation, as has been shown by past HTGR prototype and demonstration plants. For example, personnel exposure at FSV was exceptionally low, approximately 1 person-rem/year (Ref. 8, p. 59). In addition, the carbide pressure vessel retains fission products even after the operational lifetime of the fuel is over. Therefore, coated fuel particles represent an ideal final waste form, if they can be separated from the large amounts of extraneous low-level graphite waste. As a result, TRISO fuel may require less overpacking than traditional LWR fuel, reducing the total amount of repository space required.

Another intrinsic safety feature of HTGRs is the helium coolant. Helium is chemically and neutronically inert. This precludes safety complications which can arise due to irradiation of the coolant or corrosion of component materials. Contrary to water-cooled reactors, helium does not undergo a phase change at or above reactor operating temperatures. This simplifies the mechanical design and operation of the reactor, thereby improving the safety. On the other hand, helium does not have the same biological shielding effect as water. This results in higher radiation exposure in and around the core than traditional LWRs.

Gas-cooled reactors are able to retain fission products effectively and are designed to prevent radioactive release without operator intervention or active safety systems. Therefore, no external accident management should have to be undertaken outside the plant fence. That is, HTGRs do not require any offsite emergency response. Also, HTGRs do not require as leak-tight a containment building as LWRs, which could reduce capital costs. These advantages have significant economic benefits, but raise a number of safety concerns from opponents of nuclear power who are hesitant to rely on the inherent safety of advanced reactors. Ultimately, these features may allow the VHTR to be built at industrial sites in areas with dense population, in order to support process heat applications and reduce carbon emissions.

HTGRs also have intrinsic design features for proliferation resistance. The VHTR has a relatively low power density and overall power output compared to contemporary monolithic water-cooled reactors. These features result in initially low fissile inventories and a highly dilute fuel form. In addition, TRISO fuel is difficult to reprocess. Each coated fuel particle has a diameter of approximately 650 to 850 microns. A full fuel load for the VHTR will contain approximately 10 billion coated particles which must be separated from the graphite pebbles or blocks in which they are dispersed. Reprocessing is further complicated since the protective pyrolytic carbon and silicon carbide layers must also be removed in order to gain access to the fissile material. Although these features do not completely prohibit reprocessing, they greatly complicate the process when compared to traditional LWR fuel reprocessing.

# 3 Technical Challenges

The technology of the VHTR is based largely on former HTGR plants, which provide an extensive knowledge base. The VHTR will also benefit from the similarities of other advanced HTGR designs. Two gas-cooled reactor designs are being developed for near-term deployment in the 2010 time frame, the Gas Turbine Modular Helium Reactor (GT-MHR) by General Atomics and the Pebble Bed Modular Reactor (PBMR) by Eskom. These Generation III+ reactors have coolant outlet temperatures around 850 – 950 °C, and their development and operation will help to resolve many of the issues facing the VHTR. Nevertheless, the VHTR will require R&D to increase the operating temperature to 1000 °C and beyond, to develop the interface and systems for nuclear process heat applications, and to qualify the system for commercial licensing.

## 3.1 Fuel

TRISO coated fuel particles have been used extensively in past HTGR prototype and demonstration reactors and already have a well-developed knowledge base. Consequently, the majority of fuel development work is focused on the modeling and testing required to demonstrate safe operation and to support fuel licensing for commercial operation. Although not required in the near-term, several developmental activities are envisioned for increasing the safety, economics, and sustainability of TRISO fuel in the long-term.

The main goal of fuel development work for the NGNP is to develop an understanding of the fabrication process, fuel properties, irradiation performance, and the release and transport of fission products. Research in the manufacture of TRISO fuel will focus on further developments in the fabrication process and quality control methods. Quality control is an important issue due to the large number of coated fuel particles necessary for the VHTR fuel assemblies. A full fuel load will contain on the order of 10 billion coated particles, each approximately 650 to 850 microns in diameter. Activities are also necessary to identify temperature and irradiation effects on fuel properties and to model possible failure mechanisms. Also, fission product transport and release due to particle failure must be understood and modeled in order to support licensing. These tasks are aimed at developing and validating the computer codes and models required for design and licensing activities, in order to demonstrate a thorough understanding of the in-service behavior of the fuel.

The current HTGR fuel, SiC TRISO coated fuel particles, is acceptable for use in the NGNP, even with its higher coolant temperature. Preliminary analyses for the NGNP have shown that the VHTR target outlet temperature (1000 °C) can be achieved through thermal-hydraulic optimization of the core, without increasing the maximum fuel temperature during operation and accident conditions (Ref. 14). Furthermore, the traditional 1600 °C design limit for SiC fuel provides significant margin since noticeable failure does not occur until approximately 1900 –

2000 °C (Figure 3.1-1). However, the economics, safety, and sustainability of the TRISO fuel can be improved in the long-term.

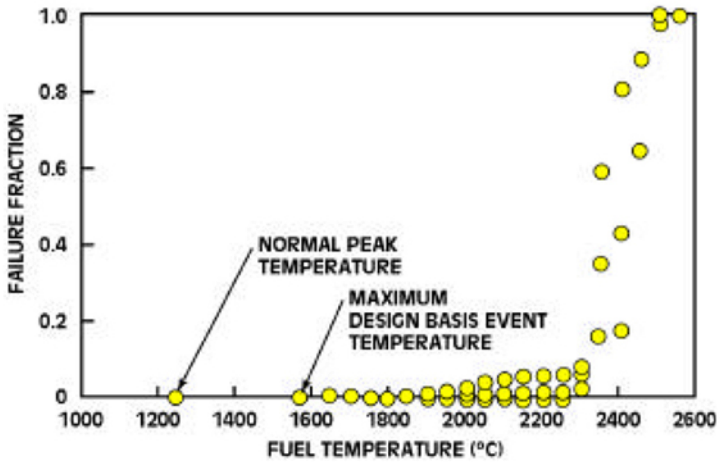


**Figure 3.1-1. SiC Coated Fuel Particle Temperature Capability (Ref. 19)**

SiC fuel does not begin to degrade until higher temperatures, but 1600 °C is traditionally used as the limiting fuel temperature during accidents. Preliminary studies show that zirconium carbide (ZrC) may provide a greater temperature margin than SiC, raising the fuel design temperature to approximately 1800 °C. This would provide increased safety margins and perhaps even allow higher temperatures, which would improve the overall efficiency of the VHTR. ZrC may also be a viable coating for use in gas-cooled reactors with a closed fuel cycle, such as the Gas-Cooled Fast Reactor. However, ZrC requires significantly more fabrication development and validation testing than current SiC fuel.

An automated fabrication process may ultimately be developed to lower costs and improve the quality of the fuel. Developmental studies are also aimed at extending the achievable fuel burnup, which would improve the sustainability and operating costs of the VHTR. Ultimately the VHTR will provide a maximum fuel burnup of 150 – 200 GWd/MTHM (Ref. 2), which is approximately 3 – 4 times that of current LWR fuel (Ref. 21). Research is also needed for the back-end of the fuel cycle, namely the disposal of spent VHTR fuel. The fuel particles themselves, the high-level waste, represent an ideal waste form due to the SiC "pressure vessel" which retains all fission products. This may reduce the amount of required shielding and the overall required repository space. However, these particles are dispersed throughout a much larger volume of low-level graphite waste. This requires the development of separation and disposal processes for the graphite low-level waste, which may prove to be too complex or costly to be effective. Incineration may present an effective disposal method for the graphite, but the full ramifications have not yet been investigated.

## 3.2 MATERIALS

The bulk of research for the VHTR is in the area of material development and qualification for the reactor, fuel, and components. Near-term designs have contributed significantly to the development and analysis of gas-cooled reactors, and their operation will be a significant contributor to the state of the art. However, the VHTR will require materials to operate under higher temperatures and neutron fluxes than have been experienced in prior nuclear service, including near-term designs. This issue is further complicated by the fact that the VHTR design is not completely finalized; therefore, the exact material requirements may change somewhat as the design progresses. Instead, development of a material property database for candidate materials will be initiated early in the project, in order to support an iterative process of refining component and material requirements. This section first describes the properties database which will be established to support VHTR design activities. Next, a brief description is given of the differences in the VHTR service conditions and their material effects. Finally, specific components of the VHTR are discussed for which materials issues still need to be resolved.

### *3.2.1 Material Properties Database*

The higher temperatures and neutron fluxes of the VHTR will result in a rather complex interaction between radiation damage, diffusion phenomena, and direct chemical reactions. The resulting changes to the microstructure of materials affect a number of material properties, which are listed below. Part of the material research for the VHTR will focus on developing a better understanding of the relationship between microstructure changes and material properties through testing and modeling. The complexity of the issue will ultimately require significant empirical testing. The results of this testing will be used to begin development of a design database for the material properties and high-temperature behavior under the expected VHTR service conditions, in order to support licensing.

The VHTR service conditions are beyond the temperatures and irradiation levels of contemporary LWRs. A number of high-temperature candidate materials have been developed over the past 60 years for gas turbines in the aerospace industry and for past HTGR demonstration and near-term reactors. However, a significant amount of further effort is required to validate and qualify these materials for commercial nuclear service. After surveying the existing data for candidate materials, researchers will perform additional testing to quantify necessary material properties for the most promising materials. For each selected material, the design database must contain sufficient data to demonstrate that the material will perform its design function over the relevant temperatures, irradiation conditions, and component lifetime. Examples of material properties which must be documented are:

- Thermal Expansion
- Irradiation Growth
- Grain Boundary Growth
- Void Swelling
- Thermal Creep
- Irradiation Creep
- Irradiation Embrittlement
- Stress Relaxation
- Strength
- Ductility
- Toughness
- Creep Rupture Strength
- Fatigue Cracking Resistance
- Helium Embrittlement
- Corrosion Resistance
- Oxidation Resistance
- Contamination Effects

### 3.2.2 VHTR Service Conditions

Materials will be subjected to higher temperatures and irradiation in the VHTR than in past designs. Larger amounts of irradiation will cause more displacements and transmutations than in current nuclear service conditions. Higher temperatures will increase the atomic diffusion rates and chemical reaction rates at the surface of and within the VHTR materials, which will affect the microstructure and material properties. Significant empirical testing is required to understand the interplay between the many material effects. For example, the damage caused by higher radiation levels could be annealed away by self-diffusion within the material due to the higher temperatures. This section briefly discusses the types and magnitude of differences in the VHTR service conditions, compared to past nuclear service conditions.

Radiated neutrons contain significantly more kinetic energy when compared to thermally excited atoms (MeV vs. fractions of eV). Due to the high energies involved, neutron irradiation causes direct microstructural damage in the form of displacements and transmutations. In order to improve its sustainability and fuel utilization, the VHTR will use fuel with about 3 – 4 times the burnup of contemporary LWR fuel (Ref. 21). The higher fuel burnup will subject the core materials to larger amounts of radiation damage over the core lifetime. Therefore, VHTR materials will need to be tested and qualified for irradiation levels significantly beyond those qualified for current LWRs.

The atoms in the VHTR will have increased thermal energies due to the higher temperatures compared to past reactors. The higher energies will increase chemical reaction rates (i.e., corrosion) and atomic diffusion rates at the surface and within the material which will alter the material properties. Both of these rates increase exponentially with temperature and can be described by the Arrhenius equation, which has the general form:

$$X = X_o \exp\left(\frac{-Q}{RT}\right)$$

where:
X = Reaction Rate ($s^{-1}$) or Diffusion Coefficient ($cm^2/s$)
$X_0$ = Pre-exponential Factor ($s^{-1}$ or $cm^2/s$)
Q = Activation Energy for Reaction or Diffusion (kJ/mol)
R = Ideal Gas Constant (8.314 J/mol K)
T = Temperature (K)

When the temperature increases from $T_C$ to $T_H$, the reaction rates and diffusion rates increase as follows:

$$\frac{X_H}{X_C} = \exp\left(\frac{E}{R}\left(\frac{1}{T_C} - \frac{1}{T_H}\right)\right) = \left(\exp\left(\frac{E}{R}\right)\right)^{\frac{1}{T_C}-\frac{1}{T_H}}$$

This relationship illustrates an important challenge facing the VHTR. Not only do higher temperatures aggravate chemical and diffusion effects in general, but the amount of aggravation (i.e., $X_H/X_C$) depends on the required activation energy. In other words, slower processes (i.e., with higher activation energies) will be aggravated the most by an increase in temperature. Phenomena which were negligible at low temperatures, and therefore were not thoroughly investigated in past reactors, may become significant in the VHTR. In addition, long-term chemical processes such as corrosion will become more like direct and (relatively) instantaneous chemical reactions. Figure 3.2-1 illustrates the amount of increase between the VHTR and FSV (770 °C) and between the VHTR and GT-MHR (850 °C), as a function of activation energy.

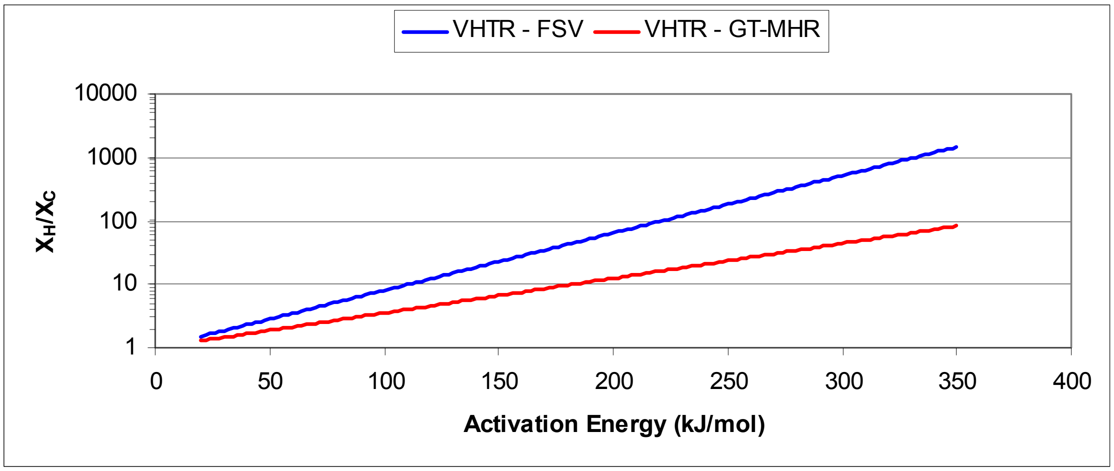


**Figure 3.2-1. Increased Reaction/Diffusion Rates for the VHTR**

The oxidation of graphite can be used as an example. Graphite oxidation is controlled by the rate at which carbon monoxide forms from the reaction of oxygen and graphite and the diffusion rate at which oxygen can permeate the material (Ref. 22). At low temperatures, oxygen diffuses into the structure faster than it can react with the graphite. This rather uniform attack causes macroscopic property changes without overall dimensional changes. At high temperatures, the graphite completely oxidizes at the surface to form gaseous carbon monoxide before oxygen can diffuse into lower layers. In this case, dimensional changes occur before macroscopic property changes. As a result, the dimensional changes resulting from graphite oxidation are no longer negligible at the temperatures planned for the VHTR.

In addition, the oxidation rate in the VHTR will increase significantly, compared to that experienced in past HTGRs. The VHTR target coolant temperature is 1000 °C, an increase of only 22% (when using absolute K) from that of FSV (770 °C). The activation energy for graphite oxidation ranges between 80 and 375 kJ/mol (Ref. 22). Using an average activation energy of 225 kJ/mol, the graphite oxidation rate in the VHTR will be approximately 100 times more than that experienced at FSV:

$$\frac{X_{VHTR}}{X_{FSV}} = \exp\left(\frac{E}{R}\left(\frac{1}{T_{FSV}} - \frac{1}{T_{VHTR}}\right)\right) = \exp\left(\frac{225000}{8.314}\left(\frac{1}{1043} - \frac{1}{1273}\right)\right) = 109$$

### 3.2.3 Component Material Challenges

A suitable metallic material must be found for the reactor pressure vessel (RPV) in order to balance the trade-off between the cost and safety of the VHTR. Although the coolant outlet temperature is 1000 °C, the vessel material can be maintained at a lower temperature by only exposing the vessel wall to the cooler inlet helium. Currently, acceptable material candidates are available; however, peak accident temperatures must be maintained below the acceptable values for these materials to prevent structural damage. The temperature can be limited by altering the vessel size (i.e., surface area), power density, or overall power rating in order to reduce the net decay heat flux and resulting peak vessel temperature. However, these changes have economic trade-offs that result from increased capital and/or operating costs. Therefore, the temperature capability of the vessel material is a key parameter for improving the combined economics and safety of the VHTR. Current prospects include 9% Cr steels, which provide high creep strength up to 650 °C. However, the cost-safety envelope can be pushed further if higher-temperature materials can be developed and qualified.

The higher operating temperatures of the VHTR require further material development and qualification for reactor structural components and cooling system components. These components include control rod sheaths, core restraints, hot gas ducts, and isolation valves. Up to temperatures of 1100 °C, iron or nickel oxide dispersion strengthened alloys may provide suitable high-temperature behavior, but are relatively expensive. Refractory based metals such as molybdenum and tungsten may be suitable above 1100 °C, although these metals exhibit extremely poor oxidation resistance. Superplastic ceramic materials may also be considered, although they will also require the development of suitable joining methods. Inside the core, where high neutron fluxes may prohibit the use of metallic materials, carbon-carbon fiber composites are being considered for control rod sheaths. Like other VHTR materials, these advanced materials will require significant testing to document their mechanical, thermal, and irradiation properties and behaviors.

Development is also needed for the graphite used in the core internals. Graphite is prone to chemical attack from air or water ingress into the core, the potential of which still needs to be evaluated for the VHTR. Methods, such as fuel block coatings, need to be developed to increase the chemical resistance of graphite. In addition, graphite is a semi-isotropic material, whose composition and behavior varies depending on the origin of its constituent coke particles and its method of manufacture. Current data on the behavior of graphite is semi-empirical and limited to certain coke stocks, some of which are no longer available. This semi-empirical process will most likely continue for future VHTRs, but will require significant qualification each time a new coke stock or manufacturing process is employed. Instead, an analytical model would allow a better prediction of the irradiation behavior of graphite, in order to increase the reliability and life-time of core graphite for future commercial VHTRs. Initial progress in this regard includes the development of an analytical finite element model of the graphitic microstructure, able to approximate bulk property changes (Ref. 7).

It should also be noted that development and qualification of new material coatings for the VHTR may also be required. These coatings would be used essentially anywhere they are needed, if a suitable material for the component in question could not be developed. As temperature increases, combining the structural strength of a material with environmental resistance becomes increasingly difficult. Corrosion-resistance and thermal insulation will be the major aim of coating development. As mentioned elsewhere, the IHX may also use coatings for a tritium barrier to prevent contaminating the product hydrogen. However, the adhesion strength, failure modes, and lifetime of new coatings must all be quantified before the coatings can be qualified for nuclear applications.

## 3.3 BALANCE OF PLANT

### *3.3.1 Turbomachinery*

Although significant progress has been made in the gas turbine industry during recent years, the constraints and requirements for nuclear applications require further development. Typical gas turbine experience is limited to large machines with low operating pressures and low pressure ratios and to small machines with high operating pressures and high pressure ratios. The VHTR requires an intermediate design, a large turbine with high operating pressure and low pressure ratio (Ref. 3). The turbine will also require approximately twice the power rating of currently available commercial turbines. In addition, magnetic bearings are required in order to prevent contamination from lubrication ingress. Optimal control methods, such as vane adjustment or inventory control, must also be developed to allow load-following and transient operation. These issues are also being addressed for the near-term GEN III+ designs, the GT-MHR and PBMR.

Additional BOP challenges unique to the VHTR are mostly material issues. As with most of the VHTR components, the turbine and other turbomachinery will be required to perform under higher operating temperatures than in both current applications and in future GEN III+ applications. Current single crystal turbine blades are adequate for traditional HTGR temperatures (Ref. 3), but may require blade-cooling in the VHTR, which reduces efficiency and increases the costs per kilowatt. Ultimately, the VHTR temperatures may require development of ceramic materials for the turbine.

### *3.3.2 Intermediate Heat Exchanger*

The VHTR is being designed to use a direct cycle for electrical generation, but will require an indirect cycle with an intermediate heat exchanger (IHX) for hydrogen production. The IHX isolates the nuclear reactor from the process heat application, allowing the system to be built to non-nuclear standards. The IHX also establishes additional physical barriers between the reactor and the process heat loop. Additional barriers will help reduce contamination from activated radionuclides such as cobalt, which may be abundant in some high-temperature alloys. However, there are still several issues to be resolved before an adequate IHX can be designed.

Due to the low heat transfer properties of helium gas, the IHX will require an extremely large surface area for heat transfer. This would require traditional tube and shell heat exchangers which are too large to be economical. Advanced compact heat exchangers are commercially available which use either the plate-fin or “printed circuit” design. However, they do not completely comply with current ASME nuclear codes due to the fabrication methods currently used (Ref. 10). Regardless of the chosen design, the IHX will be a critical component and will become the ASME Section III – Section VIII pressure boundary. Due to the high temperatures of the VHTR, the IHX must limit thermally induced stresses, thermal cycling, and pressure cycling to acceptable values. Also, the IHX will have large temperature gradients which could cause significant thermal expansion, compared to that encountered in traditional heat exchangers. Furthermore, compact heat exchanger designs will have more precise components and less geometrical tolerance than traditional designs, which will make the IHX less tolerant to thermal expansion. These issues may be resolved through careful design and/or through development of advanced high-temperature materials.

### *3.3.3 Helium Purification & Magnetic Bearings*

The purity of the helium working fluid needs to be maintained in order to prevent high temperature corrosion from impurities. As discussed in Section 3.2, corrosion in the VHTR will more closely resemble direct, and relatively instantaneous, chemical reactions than the long-term corrosive processes in traditional LWRs. This places significant importance on maintaining the purity of the helium coolant in order to prevent long-term component damage. In order to support licensing, the corrosive effects of impurities must be further quantified and acceptable operational limits must be established. In addition, the VHTR must be designed to limit the ingress of impurities from a number of sources.

For example, the power conversion unit includes a water-cooled heat sink which could potentially leak and contaminate the system. Routine maintenance could introduce a myriad of potential contaminates, and careful maintenance procedures will have to be developed to preclude such contamination. One source of impurities in early HTGR plants was the bearing lubrication of the turbomachines, such as the water-ingress issues at FSV due to the water-lubricated helium circulators. To prevent this, the VHTR will use magnetic bearings in its turbomachinery. Further design development is needed to finalize the rotordynamics and control of the magnetic bearings. Their operation during loss of power events must also be understood, and sufficient countermeasures must be provided.

## 3.4 PROCESS HEAT APPLICATIONS

A number of technical challenges face the use of nuclear process heat for industrial applications. These include issues with safety, reliability, product quality, and the reactor-process interface. The NGNP will focus on hydrogen production using nuclear process heat, although other process heat applications are envisioned in the long-term. The challenges facing nuclear hydrogen production relate either directly to the hydrogen production system or to its interface with the VHTR. Similar process and interface development will also be required for other process heat applications in the long-term, but the current focus is on the shorter-term goal of demonstrating the viability of nuclear hydrogen production.

### *3.4.1 Hydrogen Production Issues*

A number of different hydrogen production methods exist, but most are not well-developed. The DOE is sponsoring a significant amount of research to develop and refine numerous hydrogen production processes through its Office of Energy Efficiency and Renewable Energy (EERE).

The NGNP program will work closely with the EERE in order to identify and design the optimal production processes for the VHTR hydrogen plant. As mentioned earlier, the NGNP will investigate both high-temperature electrolysis and thermochemical cycles for hydrogen production.

In general, electrolysis is a mature technology. Therefore, the main developmental challenge is to make the process more economic and efficient. The efficiency of the actual electrolytic process is not expected to improve significantly with further development. Therefore, development will focus on optimizing the high-temperature steam electrolysis process for commercial production. This process has the potential for modest efficiency improvements in the overall process, since it replaces a portion of the required electrical energy with thermal energy. This process will require further development in order to optimize the process for the thermal conditions of the VHTR, once those are better defined. Other developmental efforts for steam electrolysis will focus on improving the design and manufacture of the electrolyzer modules to reduce capital costs and on evaluating the cost effectiveness of modular scaling for large-scale production.

The NGNP program will evaluate a number of thermochemical cycles for hydrogen production, of which the most promising is the iodine-sulfur (IS) process because of its projected high efficiencies. However, the IS process is rather complex and requires significant development before it can be commercially viable. JAERI has successfully demonstrated the IS cycle at the laboratory scale. The hydriodic and sulfuric acids involved in the process required the entire system to be constructed from glass. This is undesirable for large-scale commercial applications, and alternate materials will have to be developed. In addition, developmental efforts will focus on membrane technology for the efficient separation of the hydriodic and sulfuric acids, which both occur in the liquid state (Ref. 14). Other issues to be resolved include reducing the recycle rates and inventories of the chemicals involved, since these chemicals are expensive and somewhat toxic.

### *3.4.2 VHTR Interface Issues*

In addition to the challenges facing the production processes themselves, a number of process control and safety-related issues arise when coupled to a nuclear reactor. These include matching the thermal behavior of the two loops, maintaining the reactor core pressure boundary, and the prevention of explosions and contamination. A number of these issues have already been addressed at JAERI (Ref. 5) in preparation for interfacing a steam reforming hydrogen production system to the HTTR (Ref. 17).

#### Thermal Load Absorber

The VHTR may require a thermal load absorber in order to compensate for differences in the thermal behavior of the reactor and process heat loops. The reactor power has a slow thermal response due to its large heat capacity and negative temperature coefficients. On the other hand, the chemical reactor power in a hydrogen production system responds quickly to thermal variations caused by changing feed rates or reaction temperatures. Therefore, some method of trimming the reactor power is required to match the power consumed by the hydrogen production system. This may not be necessary for the NGNP, since its demonstration hydrogen production system will use only approximately 10% of the total reactor power. Acceptable limits for thermal variations and suitable thermal load absorbers will need to be determined for future VHTR units, which may be completely dedicated to hydrogen production.

Initial computer simulations at JAERI have identified one possible way to accomplish this objective (Ref. 17). A steam generator is placed downstream of the chemical reactor, which was a steam reformer in this simulation. The steam generation rate then varies as needed in order to attenuate thermal disturbances originating from the chemical reactor. The outlet temperature of the steam generator remains relatively constant throughout, thereby mitigating the propagation of thermal transients to the reactor loop.

**High-Temperature Isolation Valve**

The IHX of the VHTR isolates the reactor system from the process heat application and is the pressure boundary between the two loops. An isolation valve is needed to reduce the design pressure of the IHX and to maintain the pressure boundary in case of a rupture. For the VHTR, development is required to design an isolation valve for high-temperature service. In this regard, JAERI has already completed initial design and testing (Ref. 5). Design efforts were focused on mitigating the effects of thermal expansion and developing a new coating to reduce seat friction. However, the valve design still requires demonstration of successful long-term operation and qualification for nuclear service.

**Combustion and Explosion Issues**

The use of nuclear process heat may require process facilities to be located on-site or in close proximity to the reactor. This will require strict leak-tightness constraints for the hydrogen plant components, as well as designing for potential combustible explosions and their effects. Initial countermeasures being proposed for the VHTR include below-grade siting of the reactor and the use of cut-off valves and coaxial piping in order to retain combustible gases. However, additional countermeasures need to be addressed and developed. Regardless of the countermeasures involved, the explosive nature of the hydrogen production plant will present significant obstacles for the safety analysis and licensing of the VHTR.

A remotely located hydrogen production plant may mitigate a number of the safety issues involved with co-location. The German PNP (Prototype Nuclear Process Heat Project, 1970s – 1980s) sought to develop a method to remotely utilize nuclear process heat. The chosen process used a reversible reaction of carbon monoxide and hydrogen to produce methane. The endothermic reaction was performed at the nuclear heat source, and the products transported through pipelines to their destination. The reverse exothermic reaction released the stored nuclear heat for process applications. However, the viability of a similar process for achieving the temperatures required for hydrogen production is uncertain.

**Contamination Issues**

The final hydrogen product must be kept free of contamination, particularly from radioisotopes formed within the core. The IHX creates additional physical barriers that reduce contamination from activated radionuclides and fission products. This is not the case for tritium, a radioisotope of hydrogen. Tritium can be formed in the reactor core from impurities in the helium coolant and can easily diffuse through metallic barriers at high temperatures. As such, it is possible for this radioisotope to diffuse from within the core through the IHX and contaminate the product hydrogen.

This issue requires more investigation, but preliminary studies have been conducted at JAERI to measure the permeability of reaction tubes in a steam reformer (Ref. 5). The test was conducted with hydrogen and deuterium; however, the results should be applicable to tritium as well, since the diffusion rate of tritium is bounded by the diffusion rate of the less-massive and smaller deuterium atoms. The larger amount of hydrogen causes most of the interstitial sites within the metal surface to be occupied by hydrogen atoms. These hydrogen atoms inhibit most of the deuterium atoms from diffusing through the surface. The results indicate that the amount of permeated tritium may be kept within acceptable limits without requiring additional countermeasures. These amounts must be further quantified, and prevention of tritium diffusion may ultimately require the development of special-purpose coatings for heat exchangers.

## 3.5 Modeling, Analysis, and Licensing Issues

The VHTR also requires significant effort to resolve the licensing issues necessary to demonstrate safe and reliable commercial operation. The licensing of advanced nuclear reactors does not have the benefit of thousands of reactor-years operating experience, which existing LWRs have. This creates an immense burden to demonstrate safety and reliability through testing and analysis. As mentioned elsewhere, the VHTR will require development of analytical models and computer codes. These will include models and codes for fuel failure, fission product release, material behavior, and overall reactor system operation. In order to support licensing, these models and codes will require significant amounts of testing for verification and validation, in order to compensate for the lack of operational experience.

However, licensing of the NGNP is expected to greatly facilitate future VHTR licensing. Most likely, the NGNP will be granted a separate construction and operating license through 10CFR50 (Ref. 14). It is expected that the NGNP will be able to resolve a number of the open issues in demonstrating the safety and reliability of a gas-cooled reactor coupled to a hydrogen production plant, as well as verifying the operation of the passive safety systems of the VHTR. Once the NGNP is operable, a number of safety demonstration tests are planned in order to support design certification of future VHTR units under 10CFR52 for a combined construction and operating license.

# 4

# R&D Programs

The VHTR will require considerable research and development before deployment as a commercial reactor in order to resolve the technical issues discussed above. The PBMR and GT-MHR designs, scheduled for near term deployment by 2010, will contribute significantly to the development of the VHTR. The HTTR (JAERI), HTR-10 (INET), and the ATR (INEEL) provide excellent test facilities for high-temperature material and fuel research. The VHTR is scheduled to be one of the first GEN IV reactors, ready for commercial deployment near 2020.

## 4.1 THE GIF VHTR ROADMAP

The GIF has constructed an initial roadmap of the necessary R&D required to allow commercial deployment of the VHTR (see Ref. 2 for more details), which will become better defined as the project progresses. As shown in Figure 4.1-1, development will consist of three research stages:

- Viability Phase – Determine the feasibility of the VHTR concept
- Performance Phase – Resolve performance issues and optimize the final design
- Demonstration Phase – Construct and operate a demonstration reactor

The GIF Roadmap provides initial timelines and cost estimates for the required VHTR R&D, in order to support commercial deployment by 2020. As shown in Figure 4.1-1, fuels and materials and the balance of plant will require the most R&D, in order to resolve the technical challenges posed by the higher operating temperatures and hydrogen production plant, respectively. The GIF Roadmap is only an initial high-level outline of the required R&D, which is meant to provide the foundation for national GENIV R&D plans. Each GIF member state will pursue concepts of their own choosing and develop more detailed R&D plans. The U.S. has developed its own implementation strategy, discussed below.

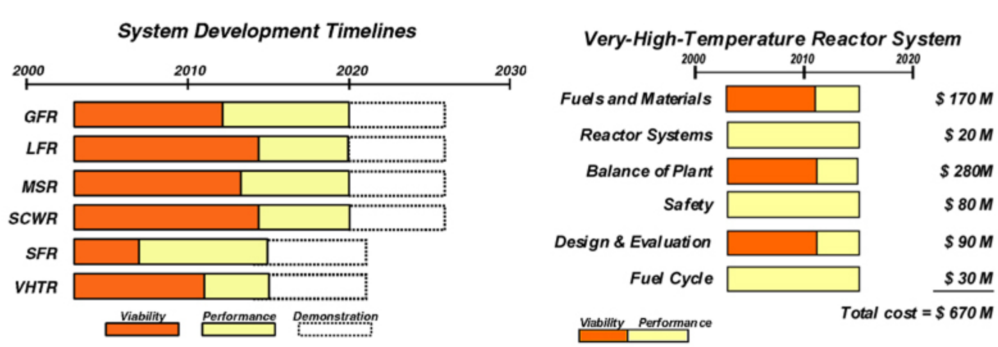


**Figure 4.1-1. The GIF VHTR R&D Timeline (Ref. 2)**

## 4.2 THE U.S. NGNP ROADMAP

To support the GIF Generation IV Roadmap, the U.S. has developed its own implementation strategy (Ref. 13) to construct the NGNP at the INEEL, based on the GENIV VHTR concept. The preliminary project schedule is shown in Figure 4.2-1 and will become better defined as the project progresses. The NGNP Project consists of the overall facility design and construction and four major supporting activities to resolve the technical challenges described above. The supporting activities are listed in Figure 4.2-1. The main objectives of the NGNP program are:

- Demonstrate a full-scale prototype NGNP by the year 2015
- Demonstrate high-temperature Brayton cycle electric power production at full scale
- Demonstrate nuclear-assisted production of hydrogen (with about 10% of the heat)
- Demonstrate the exceptional safety of the VHTR concept through testing
- Obtain an NRC License (via 10CFR50) to construct and operate the NGNP in order to provide a basis for future performance-based and risk-informed licensing
- Support the development, testing, and prototyping of the infrastructure necessary for the future hydrogen economy

To accomplish these objectives within the GIF GENIV time-frame, the NGNP requires a very aggressive schedule. Furthermore, the U.S. NGNP Roadmap has accelerated the design activities of the GIF in order to finish construction of a demonstration reactor by 2015, when the GIF plans to begin construction. The accelerated schedule is necessary to support the U.S. hydrogen economy roadmap, to provide sufficient time for addressing U.S. licensing issues prior to commercial deployment, and to re-establish the U.S. as a leader in nuclear technology. Operation of the NGNP will then demonstrate the safe and reliable operation of the VHTR design, in order to support commercial deployment by 2020 or sooner.

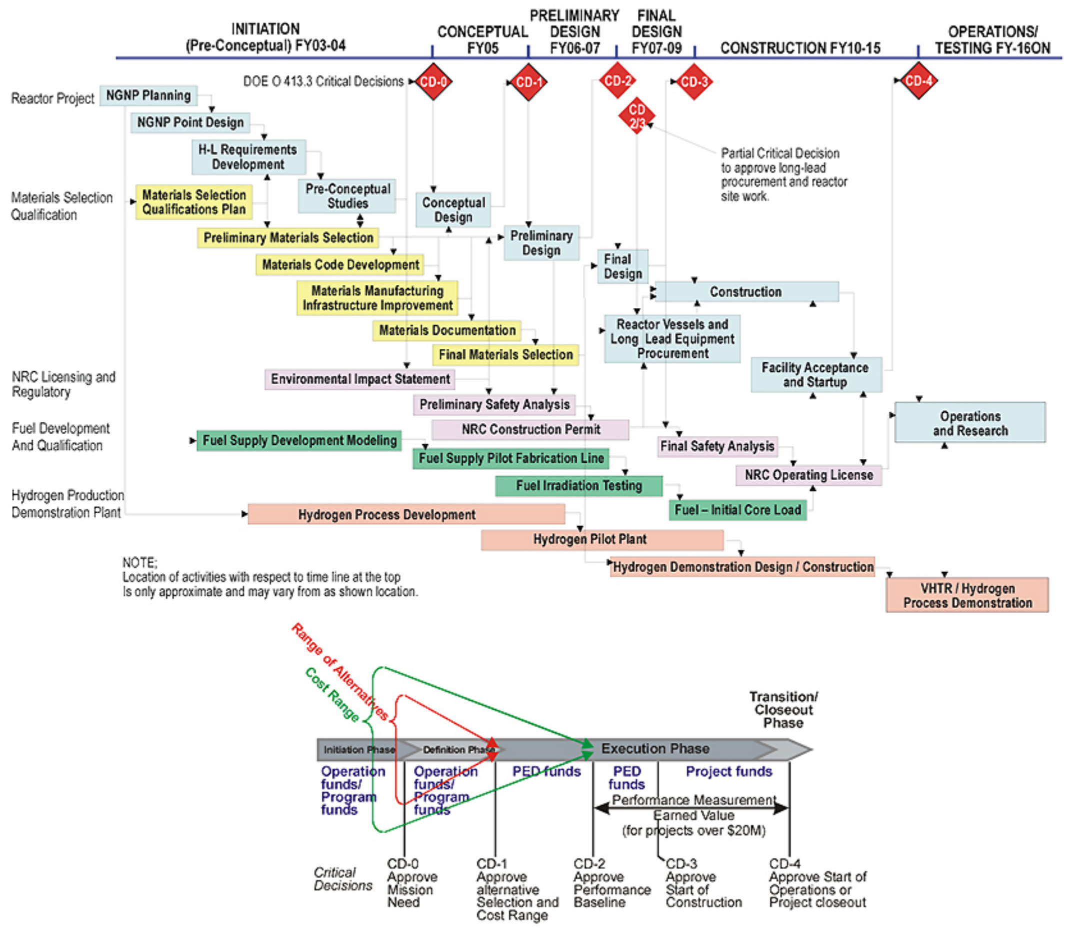


**Figure 4.2-1. The U.S. NGNP Development Roadmap (Ref. 14)**

# 5 References

1. Brey, H. L. “The Evolution and Future Development of the High Temperature Gas Cooled Reactor.” International Conference on Global Environment and Advanced Nuclear Power Plants (Sept. 2003). Paper 1194.

2. Generation IV International Forum. GIF-002-00. A Technology Roadmap for Generation IV Nuclear Energy Systems. December 2002.

3. Generation IV International Forum. GIF-004-00. Generation IV Roadmap: R&D Scope Report for Gas-Cooled Reactor Systems. December 2002.

4. Generation IV International Forum. GIF-016-00. Generation IV Roadmap: Description of Candidate Gas-cooled Reactor Systems Report. “Appendix C: Very-High-Temperature Reactor (VHTR) Systems Summary Report.” December, 2002.

5. Inagaki, Yoshiyuki, et al. “Research and Development Program on HTTR Hydrogen Production System.” International Conference on Global Environment and Advanced Nuclear Power Plants (Sept. 2003). Paper 1062.

6. International Atomic Energy Agency. IAEA-TECDOC-1198. Current Status and Future Development of Modular High Temperature Gas Cooled Reactor Technology. February 2001.

7. Marsden, Barry J., Siu Lun Fok, and Graham Hall. “High Temperature Gas-cooled Reactor Core Design Future Material Consideration.” International Conference on Global Environment and Advanced Nuclear Power Plants (Sept. 2003). Paper 1222.

8. Melese, Gilbert, and Robert Katz. Thermal and Flow Design of Helium-Cooled Reactors. The American Nuclear Society, 1984.

9. MIT Nuclear Energy Study Advisory Committee. “The Future of Nuclear Power: An Interdisciplinary MIT Study.” Massachusetts Institute of Technology, 2003.

10. Nuclear Energy Research Initiative Annual Report. Project 00-105. Balance of Plant System Analysis and Component Design of Turbo-Machinery for High Temperature Gas Reactor Systems. Submitted by Massachusetts Institute of Technology, September 2001.

11. Sawa, Kazuhiro, et al. “The High Temperature Gas Cooled Reactor Fuel.” International Conference on Global Environment and Advanced Nuclear Power Plants (Sept. 2003). Paper 1221.

12. Tachibana, Yukio, et al. “Safety Demonstration Tests using High Temperature Engineering Test Reactor (HTTR).” International Conference on Global Environment and Advanced Nuclear Power Plants (Sept. 2003). Paper 1095.

13. United States Department of Energy: Office of Nuclear Energy, Science and Technology. The U.S. Generation IV Implementation Strategy. September 2003.

14. Southworth, F. H., MacDonald, P. E., et al. “The Next Generation Nuclear Plant (NGNP) Project.” Proceedings of GLOBAL 2003: Atoms for Prosperity. November 2003, New Orleans, pp. 276 – 287.

15. Sterbentz, J. W., Sant, R. L. “Neutronics Point-Design of a Prismatic Very High Temperature Reactor.” Proceedings of Global 2003: Atoms for Prosperity. November 2003, New Orleans, pp. 300 – 311.

16. Miller, A. I., Duffey, R. B. “Hydrogen and High Temperature Reactors: a Source Map.” Proceedings of Global 2003: Atoms for Prosperity. November 2003, New Orleans, pp. 1514 – 1519.

17. Nishihara, Tetsuo, and Inagaki, Yoshiyuki. “Development of Control Technology for the HTGR Hydrogen Production System.” Proceedings of Global 2003: Atoms for Prosperity. November 2003, New Orleans, pp. 320 – 324.

18. LaBar, Malcolm. “General Atomics HTHX Planned Activities.” High Temperature Heat Exchanger (HTHX) Kickoff Meeting. University of Nevada, Las Vegas, October 2003. <http://nstg.nevada.edu/heatpresentations/Microsoft%20PowerPoint%20-%20LaBar.pdf>

19. LaBar, Malcolm. “The Gas Turbine – Modular Helium Reactor: A Promising Option for Near Term Deployment.” April 2002. <http://www.ga.com/gtmhr/images/ANS.pdf>

20. Ion, S., Nicholls, D., Matzie, R., and Matzner, D. “Pebble Bed Modular Reactor: The First Generation IV Reactor To Be Constructed.” World Nuclear Association Annual Symposium, September 2003. London. <http://world-nuclear.org/sym/2003/matzie.htm>

21. Güldner, R., and Burtak, F. “Contribution of Advanced Fuel Technologies to Improved Nuclear Power Plant Operation.” The Uranium Institute 24th Annual Symposium, September 1999. London. <http://www.world-nuclear.org/sym/1999/pdfs/guldner.pdf>

22. Snead, L., and Burchell, T. “Oxidation of High-Quality Graphite for IFE.” 3rd Laser IFE Program Workshop, November 2001. Pleasanton, CA.

23. Bayless, P. “VHTR Thermal-Hydraulic Scoping Analyses Using RELAP5-3D/ATHENA.” Proceedings from Global 2003: Atoms for Prosperity. November 2003, New Orleans, pp. 312 – 319.

24. Forsberg, Charles. Oak Ridge National Laboratory. “Advanced High-Temperature Reactor Website.” <http://www.ornl.gov/sci/nuclear_science_technology/ahtr.htm>

# A

# Acronyms and Abbreviations

**Table A-1.** Acronyms and Abbreviations

| Acronym/Abbreviation | Meaning |
|---|---|
| ATR | Advanced Test Reactor (INEEL) |
| AVR | Arbeitsgemeinshaft Versuchsreaktor (Germany) |
| CFP | Coated Fuel Particles |
| FSV | Fort St. Vrain (U.S.) |
| GENIV | Generation IV Nuclear Reactor |
| GENIII+ | Generation III+ (Near-Term Deployment) Reactor |
| GCR | Gas-Cooled Reactor |
| GFR | Gas-Cooled Fast Reactor |
| GIF | Generation IV International Forum |
| GT-MHR | Gas Turbine Modular Helium Reactor (General Atomics) |
| HTGR | High Temperature Gas-Cooled Reactor |
| HTR-10 | High Temperature Gas-Cooled Reactor Test Module (China) |
| HTTR | High Temperature Test Reactor (Japan) |
| IHX | Intermediate Heat Exchanger |
| INEEL | Idaho National Engineering and Environmental Laboratory (U.S.) |
| INET | Institute of Nuclear Energy Technology (China) |
| IS | Iodine-Sulfur Thermochemical Process |
| JAERI | Japan Atomic Energy Research Institute |
| LFR | Lead-Cooled Fast Reactor |
| LWR | Light Water Reactor |
| MSR | Molten Salt Reactor |
| NGNP | Next Generation Nuclear Plant (INEEL) |
| NPH | Nuclear Process Heat |
| NSS | Nuclear Steelmaking System Project (Japan) |
| PBMR | Pebble Bed Modular Reactor (PBMR Ltd. / Eskom) |
| PCIV | Pre-Stressed Cast Iron Vessel |
| PCU | Power Conversion Unit |
| RCCS | Reactor Cavity Cooling System |
| SCWR | Super-Critical Water-Cooled Reactor |
| SFR | Sodium-Cooled Fast Reactor |
| THTR | Thorium High Temperature Reactor (Germany) |
| TRISO | Triple Coated Isotropic Ceramic Fuel Particle |
| VHTR | Very High Temperature Reactor |

# B

# Brief Survey of Gas-Cooled Reactor Designs

Gas-cooled reactor designs are not a new concept. Rather, they have been investigated since the beginning days of nuclear power. This section provides a brief outline of the history of gas-cooled reactors in order to provide an understanding of the existing knowledge-base for the VHTR. The coolant outlet temperatures of the VHTR predecessors are illustrated below in Figure 4.2-1. Note that the maximum VHTR temperature of 1500 °C listed in Figure 4.2-1 is an optimistic long-term target temperature, which will not be realized until later generations.

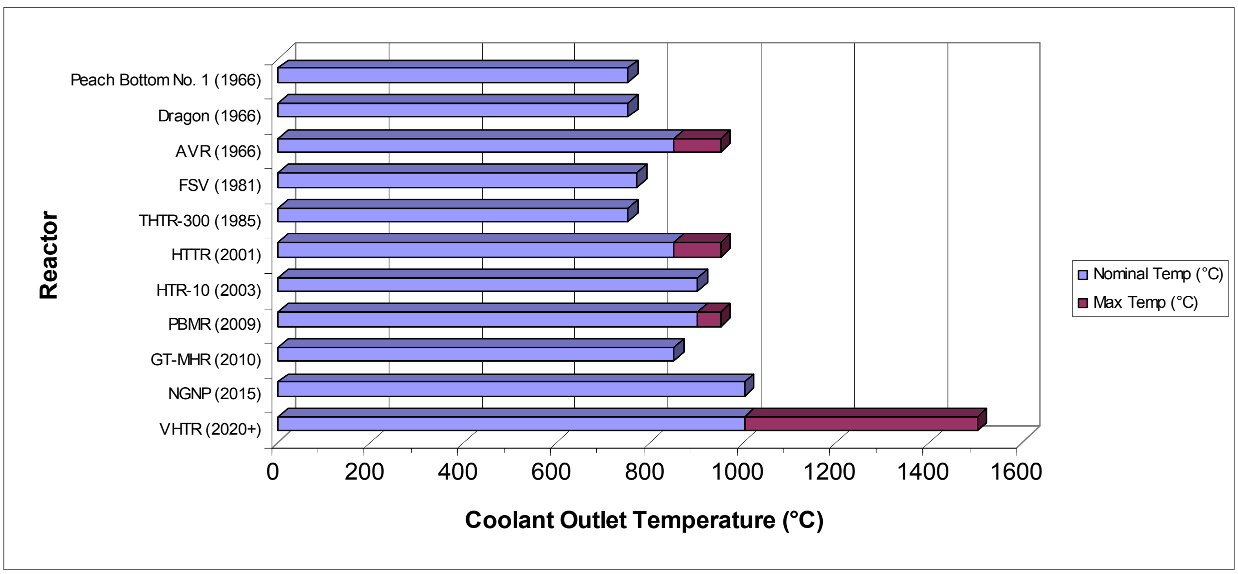


**Figure 4.2-1. HTGR Coolant Outlet Temperatures**

## 1. EARLY GAS-COOLED REACTORS

The history of gas-cooled reactors (GCRs) begins with the X-10 reactor in Oak Ridge, Tennessee, which was constructed for the Manhattan Project and attained criticality in November, 1943. The X-10 was a graphite-moderated and air-cooled reactor capable of generating 3.5 MWth. However, gas-cooled reactors typically were not used for commercial electricity generation except in the United Kingdom. The U.K. has since built a number of carbon-dioxide cooled gas reactors which are still in commercial operational today. The success of these reactors prompted the motivation for high temperature gas-cooled reactors (HTGRs) with increased coolant outlet temperatures for greater efficiency.

## 2. PROTOTYPE HTGRS

Research and development for HTGRs began in the 1950s to improve upon the performance of GCRs. HTGRs are typically characterized by an all ceramic core, graphite moderator, and helium coolant. The first HTGR prototype plants were the Dragon Reactor in the United Kingdom, the Arbeitsgemeinshaft Versuchsreaktor (AVR) in Germany, and Peach Bottom Unit 1 in the United States.

The first HTGR prototype was the Dragon reactor. The project began in 1959, and the reactor reached its full-power of 20 MWth by 1966. The objective of the Dragon reactor was to demonstrate the feasibility of the HTGR concept. The reactor used a steel pressure vessel, graphite fuel elements with coated fuel particles, and helium coolant. The helium coolant was circulated through the reactor with an inlet and outlet temperature of 350 °C and 750 °C, respectively. The Dragon prototype operated for approximately 10 years before the project was terminated. During this time, the Dragon reactor demonstrated the viability of the HTGR concept and provided significant test data on coated fuel performance and material irradiation behavior under high-purity helium conditions and elevated temperatures.

Construction of the AVR HTGR in Germany also began in 1959, and initial criticality was achieved in 1966. The 46 MWth AVR was able to generate 15 MWe using helium coolant at an outlet temperature of 850 °C. In 1974, the coolant outlet temperature was increased to 950 °C. The AVR operated for 21 years before the project was cancelled. The AVR used a steel containment vessel and coated fuel particles contained in graphite spheres (pebbles) 6 cm in diameter. The AVR was the main development tool for the pebble bed concept as well as the first source of performance data for coated fuel particles.

The first prototype HTGR in the U.S. was Peach Bottom Unit 1. This 40 MWe plant achieved first criticality in 1966. Peach Bottom used fuel compacts composed of coated fuel particles dispersed within large hexagonal graphite elements. Peach Bottom operated until 1974 when it was shut down for decommissioning. The reactor project was able to confirm the core physics, to verify the design analysis methods, and to provide a performance database for future prismatic HTGR development.

## 3. DEMONSTRATION HTGRS

Following the successful prototype HTGR plants, larger HTGR demonstration plants were constructed to further study the commercial viability of the HTGR concept. In the United States, an 842 MWth HTGR demonstration plant was constructed at Fort St. Vrain (FSV). FSV began generating electricity using a standard indirect steam cycle in 1976, and reached full power in 1981. Operation of FSV was plagued by low reliability due to mechanical problems, mainly lubrication and water ingress. However, the plant was able to successfully demonstrate the commercial viability of triple isotropic coated fuel particles (TRISO CFPs) in prismatic block form. In Germany, the Thorium High Temperature Reactor (THTR-300) was built to generate 296 MWe. It was connected to the electrical grid in 1985 and operated until 1989, when it was shut down due to a lack of funding. Nevertheless, the THTR-300 was able to demonstrate the safety characteristics, control response, and fission product retention of HTGRs.

# 4. HTGR Test Facilities

Recently, interest in the HTGR has increased due to its potential for cogeneration of electricity and high temperature process heat for industrial applications. This has resulted in the construction of two test reactors. JAERI began construction of the High Temperature Engineering Test Reactor (HTTR) in 1991. First criticality was obtained in 1998, and full power was obtained in 2001 at an outlet temperature of 850 °C. INET completed construction of the 10-MWth HTR-10 reactor in Beijing in 2000. Initial criticality was achieved the same year, and full power operation began in 2003 with an outlet temperature of 900 °C. It should be noted that the Advanced Test Reactor (ATR) at the INEEL in the U.S. will be used for material and fuel tests for the NGNP reactor. However, the ATR is a generic test facility which was not built specifically for HTGR test activities.

The HTTR and HTR-10 represent the current state of the art in HTGR design, with the HTTR using prismatic block fuel and the HTR-10 using pebble fuel. These reactors will verify and demonstrate the safety features and operational characteristics of HTGRs. They will be used for development and testing necessary for future HTGR improvements, such as establishing design databases for the high-temperature properties and irradiation behavior of advanced materials. These reactors will also provide an opportunity for innovative research and for further development of process heat applications. The HTTR and HTR-10 will be instrumental in performing a significant amount of VHTR development.

# 5. Near-Term HTGRs

## *5.1. The GT-MHR*

The Gas Turbine – Modular Helium Reactor (GT-MHR) is a near-term HTGR developed in a joint United States (General Atomics) and Russian Federation program. The GT-MHR was developed primarily to burn surplus weapons plutonium, but General Atomics is pursuing plans for near-term commercial deployment with uranium fuel. An initial prototype is scheduled to be built in Russia and operational by 2010, with U.S. units available for operation soon after. Reference 6 provides a more thorough, albeit preliminary, description of the GT-MHR design, while Reference 19 gives a more recent description of the characteristics and schedule for the GT-MHR.

The GT-MHR is a 600 MWth helium-cooled and graphite-moderated reactor which uses TRISO coated fuel particles, similar to other HTGR designs. The annular core and block fuel are based on the FSV reactor design. Furthermore, the GT-MHR is the starting point for the prismatic VHTR conceptual design. As such, much of the current VHTR design description is taken directly from that of the GT-MHR. The GT-MHR design will be modified to provide increased outlet temperatures (850 °C for GT-MHR, 1000 °C for VHTR) and to interface with a hydrogen production system, while fulfilling the goals of future Generation IV nuclear power plants.

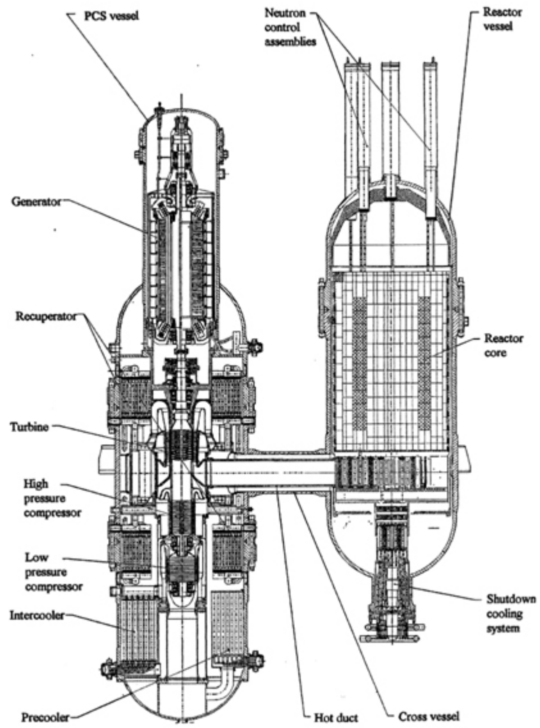


**Figure 4.2-2. The GT-MHR by General Atomics (Ref. 6)**

### *5.2. The PBMR*

The Pebble Bed Modular Reactor is a near-term HTGR which is being developed in an international effort by PBMR (Pty.), Ltd. This corporation is led by Eskom and originally included British Nuclear Fuels and Exelon. The PBMR project has overcome a number of political and economical roadblocks, including Exelon's withdrawal from the project. The French energy conglomerate Areva, which includes Framatome-ANP, has recently entered the project to take the place of Exelon. A demonstration PBMR is scheduled to be built at the Koeberg nuclear site in South Africa, with commercial operation scheduled for 2009. Reference 6 provides a thorough description of the preliminary PBMR design. Reference 20 gives a more recent description, which includes a comparison of the PBMR to the VHTR design goals.

The PBMR is a small modular gas-cooled reactor which will generate approximately 165 $MW_e$. The reactor is largely based on the German pebble-bed reactors such as the AVR and THTR. The PBMR modules will be factory produced, with up to 10 modules installed at a single site in order to take advantage of common plant facilities. The PBMR is the design basis for the pebble-bed VHTR, but is expected to require greater modification than the GT-MHR in order to meet the VHTR design goals. For example, the current PBMR design has a thermal power rating around 300 MWth, about one half of the VHTR goal of 600 MWth. Also, the PBMR uses a three-shaft design for the turbomachinery, instead of the single-shaft design of the GT-MHR. Although this simplifies the rotordynamics and maintenance, it reduces the efficiency and complicates the control system. Similar to the GT-MHR, the PBMR also requires an increase in operating temperature (from 900 °C to 1000 °C) and an interfaced hydrogen production system in order to meet the VHTR design objectives.

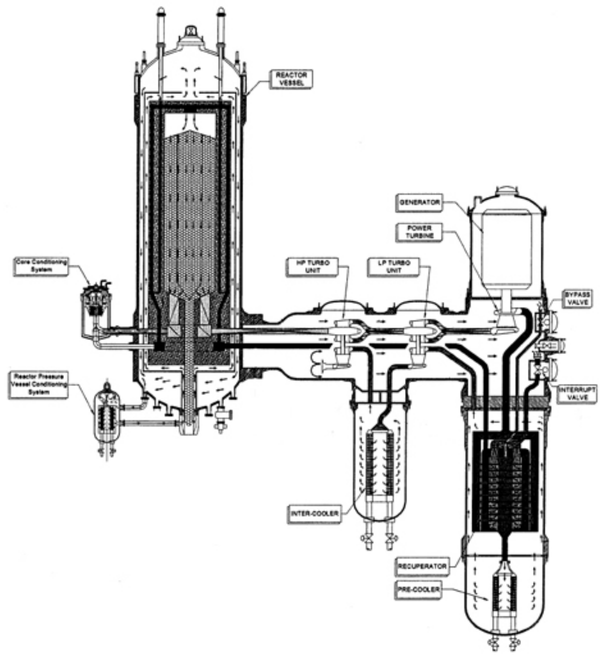


**Figure 4.2-3. The Pebble Bed Modular Reactor (PBMR) by PBMR (Pty.), Ltd. (Ref. 6)**

## 6. BEYOND THE VHTR

Although the VHTR is still in the preliminary design stage, design activities have already commenced for the long-term successors of the VHTR. These long-term designs aim to improve the sustainability and economy of the VHTR.

As mentioned elsewhere, the U.S. Generation IV Implementation Strategy contains two objectives. The long-term objective is to develop a gas-cooled fast reactor (GFR), another GENIV concept. The GFR is essentially a VHTR, but with a closed fuel cycle for long-term sustainability. Closing the fuel cycle raises a number of additional technical challenges which must be resolved in addition to those for the VHTR.

In addition to the GFR, another long-term VHTR design has already been started. Oak Ridge National Laboratory, Sandia National Laboratories, and the University of California-Berkeley are developing the Advanced High Temperature Reactor (AHTR). The AHTR is essentially a combination of the VHTR and MSR (Molten Salt Reactor) GENIV concepts. This reactor will use the TRISO fuel of the VHTR but will use the molten salt coolant of the MSR for greater heat transfer. The coolant outlet temperatures will remain the same, around 1000 °C. However, the molten salt coolant will allow monolithic reactor designs with power ratings near 2400 MWth, about 4x the power of the helium-cooled VHTR, for greater economics of scale as shown in Figure 4.2-4.

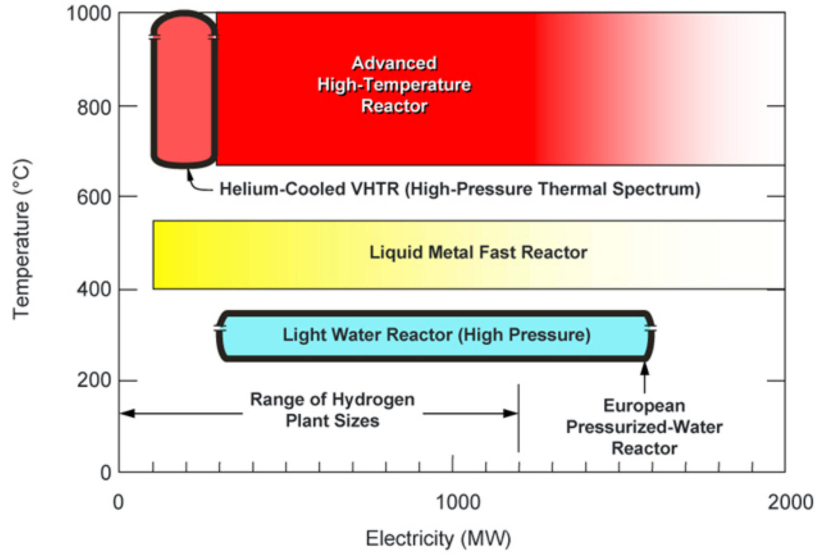


**Figure 4.2-4. The High-Temperature Monolithic AHTR (Ref. 24)**

# C

# Brief Description of Other Generation IV Reactors

The GIF roadmap for the development of advanced nuclear reactors contains five other reactor designs besides the VHTR. All six designs are illustrated below in Figure 4.2-5. These designs are meant to fulfill the long-term requirements of nuclear energy in the areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. After establishing a roadmap of the required R&D for these systems in 2002, each of the ten member countries is now allowed to focus their efforts on the reactor designs most suitable for their long-term nuclear needs. The U.S. has chosen to pursue the VHTR, particularly because of its potential for hydrogen production. The other five GEN IV reactor concepts are described briefly, below.

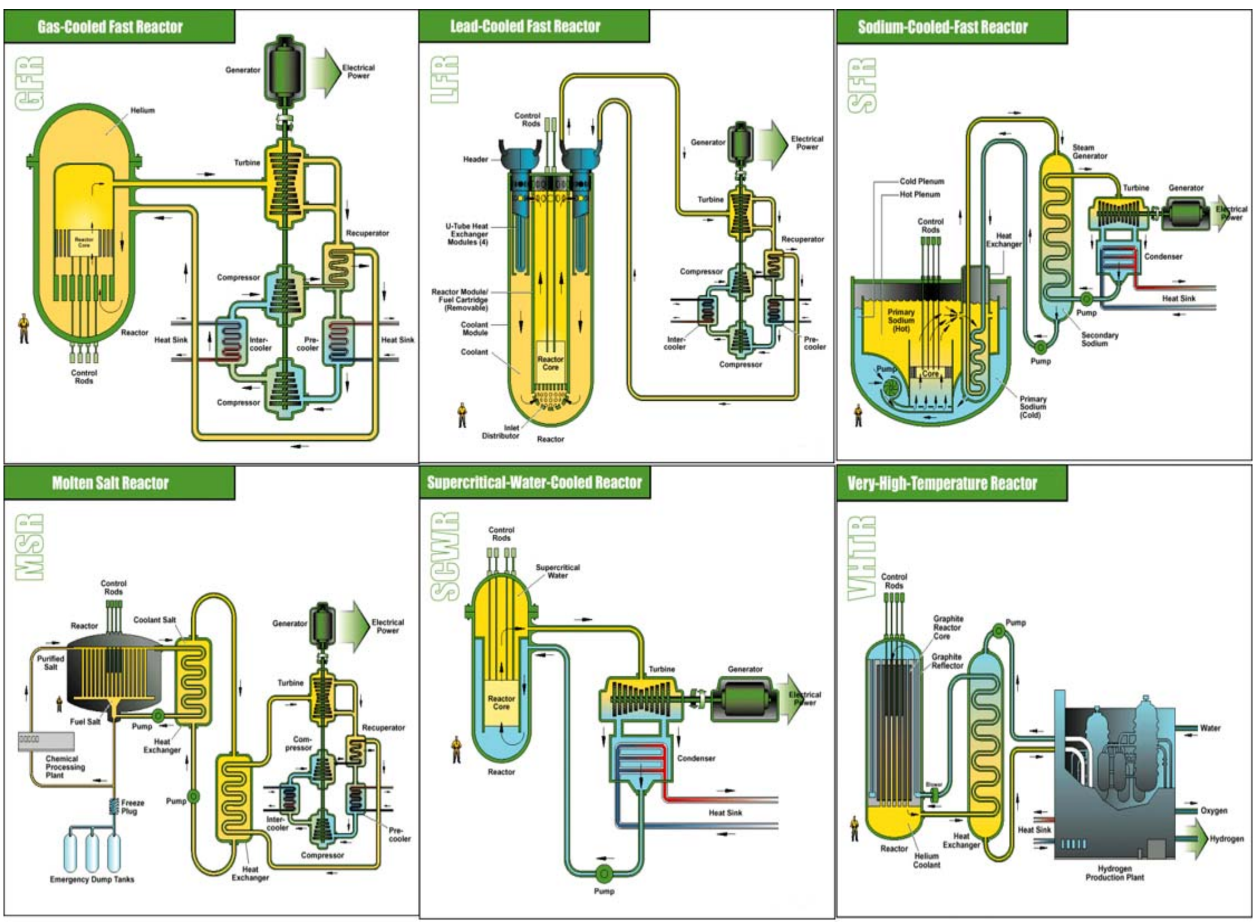


**Figure 4.2-5. Generation IV Reactor Concepts (Ref. 2)**

## 1. THE GAS-COOLED FAST REACTOR (GFR)

The Gas-Cooled Fast Reactor System (GFR) is essentially an HTGR, similar to the VHTR and near-term GEN III+ gas reactors. This 600 MWth/288 MWe reactor is helium-cooled and uses a direct Brayton cycle, with outlet temperatures of 850 °C. However, the GFR features a fast-neutron spectrum and a closed fuel cycle. Its primary mission is actinide management, and it is ranked high in sustainability. The GFR can be located on site with open cycle reactors. It is primarily envisioned for electrical production, but its temperatures may be able to support hydrogen production as well.

## 2. THE LEAD-COOLED FAST REACTOR (LFR)

The Lead-Cooled Fast Reactor (LFR) also uses a fast-neutron spectrum and a closed fuel cycle for actinide management, similar to the GFR. The LFR uses a lead or lead/bismuth liquid-metal to cool the reactor. A range of sizes is envisioned, from 1200 MWe monolithic plants to small 50 – 150 MWe *battery* plants, with outlet temperatures near 550 °C. The small *battery* plants are designed to be factory-fabricated and then transported to the site. The very long core life (10 – 30 years) allows continuous and reliable operation, with a minimal number of operator personnel. It is ranked high in sustainability, due to its closed fuel cycle, and high in proliferation resistance and physical protection, due to its factory-fabricated long-lifetime core. It is being designed primarily for distributed electricity generation and actinide management, with a possibility for low-temperature hydrogen production.

## 3. THE MOLTEN SALT REACTOR (MSR)

The Molten Salt Reactor (MSR) features a closed fuel cycle with an epithermal to thermal neutron spectrum. In the MSR, the fuel and coolant are the same. A circulating liquid mixture of sodium, zirconium, and uranium fluorides produces a thermal spectrum when in the graphite core. The coolant-fuel mixture also removes heat from the core and is the working fluid in the power conversion system. There is no downtime for refueling, since the fuel can be fed directly into the circulating liquid. The reference system has a power rating of 1000 MWe and an outlet temperature of 700 °C. The MSR is ranked high in sustainability due to its closed fuel cycle and excellent waste burndown. Its primary goal is electricity production and waste management.

## 4. THE SODIUM-COOLED FAST REACTOR (SFR)

The Sodium-Cooled Fast Reactor (SFR) is also envisioned for actinide management with its fast-neutron spectrum and closed fuel cycle using mixed uranium/plutonium oxide fuel. The SFR may have a small or large power rating, ranging from 150 – 1500 MWe, with an outlet temperature of 550 °C. The larger SFR designs will be supported by a central fuel processing facility. The SFR provides the greatest sustainability of all the GEN IV reactor designs. Primarily envisioned for electricity production and actinide management, the SFR is also the nearest term GEN IV reactor, with plans for deployment by 2015.

# 5. THE SUPERCRITICAL WATER-COOLED REACTOR (SCWR)

The Supercritical Water-Cooled Reactor (SCWR) will use either an open or closed fuel cycle. The water coolant is maintained at high temperature and high pressure to operate above the thermodynamic critical point of water to prevent a phase change. The SCWR has the largest power rating of the GEN IV reactors, 1700 MWe, with an outlet temperature of 550 °C. The lack of coolant phase change and high temperatures provides an efficiency near 44%, giving the system a high ranking in economics. The monolithic SCWR will be used primarily for electricity production, with the possibility of actinide management if a closed fuel cycle design is developed.