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High Temperature Reactor Research and Development Roadmap, Draft for Public Comment

Hans D. Gougar

October 2017

Idaho National Laboratory

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Under DOE Idaho Operations Office
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High-Temperature Gas-Cooled Reactor Research and Development Roadmap

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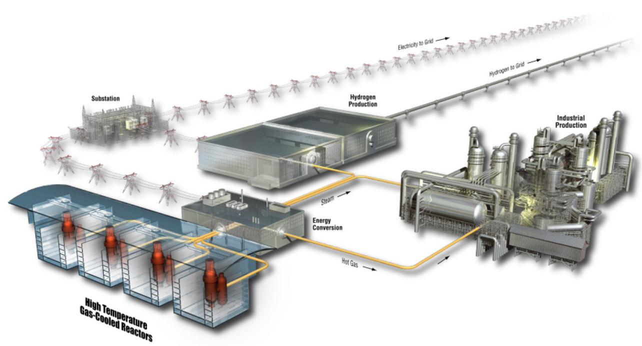
Advanced Reactor Technologies
Development Office

October 2017



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Revision 3

Diane V. Croson, Deputy Director INL ART Technology Development Office	10/5/17 Date
Hans D. Gougar, Director INL ART Technology Development Office	10/4/2017 Date
Michelle T. Sharp INL Quality Engineer	10/5/17 Date

EXECUTIVE SUMMARY

American leadership in science and technology is critical to achieving national priorities for national security, economic growth, and job creation. New nuclear energy sources are expected to serve an important role in achieving a secure energy supply in the coming decades by providing clean, continuously available power for an expanding economy. Advanced reactor concepts that feature higher temperatures, lower pressures, and enhanced passive safety are expected to contribute a greater share of United States power generation. Innovative advanced reactor concepts offer significant potential benefits, including possible lower costs, enhanced safety and security, greater resource utilization, and easier operation, as well as a supply of high-paying jobs. In order to promote a healthy advanced reactor pipeline in the United States, the Department of Energy conducts early stage research and development (R&D) on advanced reactor technologies and supports work on generic topics that can apply to various advanced reactor concepts, including fast reactors, gas-cooled reactors, and molten-salt-cooled reactors.

The Department of Energy commissioned the development of technology roadmaps for advanced non-light-water reactor concepts. The roadmaps show the R&D needed to support demonstration of advanced reactor concepts, for either performance demonstration or commercial demonstration depending on concept maturity, by 2035. The starting point for the roadmaps is the technical readiness assessment performed as part of an advanced test and demonstration reactor study released in 2016 [1] and subsequent development was based on a review of technical reports and vendor literature summarizing the technical maturity of each concept and the outstanding R&D needs. Tasks for specific systems were highlighted on the basis of time and resources needed to complete the tasks and the importance of the system to the performance of the reactor concept.

The roadmaps are generic (i.e., not specific to a particular vendor's design) but vendor design information may have been used as representative of the concept family. In the case of the graphite-moderated, high-temperature gas-cooled reactor, a single roadmap is constructed for both a near-term high-temperature reactor with an outlet temperature limited to 750°C and an advanced version with an outlet temperature as high as 950°C by 2035.

The R&D for the lower-temperature high-temperature gas-cooled reactor includes fuel and graphite qualification, core simulation methods development and validation, and development and testing of a number of structure components. Development of the higher-temperature version would require all these plus an additional material qualification effort conducted in parallel with the R&D tasks of the lower-temperature version. The qualification efforts for the lower- and higher-temperature concepts are described in this report.

The selection of tasks and associated timelines relies heavily on the technology roadmaps produced for the Next Generation Nuclear Plant Project and on recent input from the High-Temperature Reactor Technology Working Group.

ACKNOWLEDGEMENTS

The author wishes to thank Lew Lommers of AREVA and members of the High-Temperature Reactor Technology Working Group for providing input to this report. This report was prepared by selected experts within the United States Department of Energy complex who are familiar with the technology. It draws heavily on the information obtained from openly available technical reports generated during the Next Generation Nuclear Plant Project.

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ACRONYMS

AGC	advanced graphite creep
ANL	Argonne National Laboratory
ART	Advanced Reactor Technologies
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinschaft Versuchs Reaktor (Germany)
CTF	Component Test Facility
DOE	Department of Energy
GA	General Atomics
GFR	gas-cooled fast reactor
GT-MHR	gas turbine-modular helium reactor
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor
HTR-PM	high-temperature reactor-pebble molecule
HTR TWG	High-Temperature Reactor Technology Working Group
HTTF	High Temperature Test Facility
HTTR	High-Temperature Engineering Test Reactor (Japan Atomic Energy Agency)
IHX	intermediate heat exchanger
INL	Idaho National Laboratory
LWR	light-water reactor
MHTGR	modular high-temperature gas-cooled reactor
MSR	molten-salt-cooled reactor
N/A	not applicable
NEUP	Nuclear Energy University Program
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PBR	pebble-bed reactor
PCS	power conversion system
PIE	post-irradiation examination
R&D	research and development
RCCS	reactor cavity cooling system
RTDP	regulatory technology development plan
SC-HTGR	steam-cycle high-temperature gas-cooled reactor (AREVA)
SFR	sodium gas-cooled fast reactor

SiC	silicon carbide
THTR	Thorium HochTemperatur Reaktor (Germany)
TRISO	tristructural isotropic
TRL	technology readiness level
U.S.	United States
VHTR	very-high-temperature reactor

High-Temperature Gas-Cooled Reactor Research and Development Roadmap

Draft for Public Comment

1. INTRODUCTION

Among advanced reactors, the graphite-moderated, high-temperature gas-cooled reactor (HTGR) possesses a relatively high technical maturity, with in-pile testing and engineering demonstrations dating back to the 1960s and commercial demonstrations in the 1980s. Further deployment never materialized, however, and research and development (R&D) slowed considerably before a resurgence began in the late 1990s. A major design shift occurred around this time; reactor vendors developed small modular HTGR (MHTGR) concepts that relied completely on passive systems to safely remove decay heat even in the most severe loss-of-coolant events. All current designs share this attribute.

Around the turn of the 21st century, interest returned in the form of a number of government-funded efforts around the world, along with the creation of a handful of private vendors. Under the United States (U.S.) Government's Next Generation Nuclear Plant (NGNP) Project and similar programs in other countries, efforts began to reevaluate and qualify fuels, materials, and design and safety analysis methods for HTGR operation at the very high temperatures that could support efficient hydrogen production.

Industrial interest has since focused on the MHTGR with the lower (<750°C) outlet temperature, because these designs could be deployed by the 2035 timeframe, while still capturing a large portion of the process heat market. Some research continues to be conducted on the materials that can withstand the higher temperatures, but the main focus in the United States remains the qualification of tristructural isotropic (TRISO) coated particle fuel and available grades of nuclear graphite that would be used in both HTGRs and very-high-temperature reactors (VHTRs).

This report describes the R&D needed to support commercial demonstration, by 2035, of the MHTGR, either prismatic or pebble bed, with outlet temperatures limited to 750°C. Additional R&D needed to support long-term operation with higher temperatures (<950°C) is also summarized. Both pebble-bed and prismatic-core configurations are assessed, but the differences are small enough to be represented on a single roadmap.

2. CONCEPT DESCRIPTIONS

Variations on the HTGR theme emerged early in the development of the concept, but all possess the same basic features and performance attributes. Coated ceramic-fuel particles are embedded in a graphitic matrix and formed into fuel elements that can withstand high burnup, fluence, and temperatures greater than 1600°C. The graphite matrix, fuel-element structure, and reflectors provide a tremendous thermal buffer, highly conductive medium, and barrier to fission-product release. Helium is a neutronically transparent and chemically inert coolant that cannot undergo a phase change. Along with the low power density, these features preclude catastrophic fuel failure and fission-product release, even in the event of a total loss-of-coolant flow and pressure. The high outlet temperatures enable efficient energy production with either steam or gas turbine power conversion.

Some of the more distinguishing features can best be highlighted by comparing the HTGRs to light-water reactors (LWRs) and gas-cooled fast reactors (GFRs).

The *advantages* of HTGRs in comparison to LWRs are:

- Chemically inert coolant. Helium does not corrode metals or react with graphite or water, essentially eliminating corrosion-induced degradation of components or reactivity insertions.
- Negligible activation of coolant. Helium also does not absorb neutrons, although fission products released from the fuel elements may be entrained in the primary coolant flow.
- Gas coolants. These coolants cannot change phase in the core, eliminating sudden drops in heat-transfer properties due to voiding.
- Large thermal buffer offered by the graphite. Transient conditions play out over hours or even days, providing a very long grace period for accident mitigation.
- Passive decay heat removal via conduction and radiation, eliminating the requirement for active emergency core-cooling systems

The *disadvantages* of the HTGRs relative to the LWRs are:

- Large core volume for the relatively low thermal power output (large pressure vessel).
- Susceptibility to water ingress after a steam generator or shutdown cooler tube rupture. Water in the core can pick up fission products, which would be released through pressure-relief valves.
- High-temperature coolant that exceeds the temperature limits of many metal alloys used in the power conversion system (PCS).
- Susceptibility to air ingress in the event of a leak in the primary coolant boundary. Oxygen can react with graphite, degrading the surface integrity and creating combustible gases.

In contrast to metal-cooled reactors, the optical transparency of helium facilitates fuel shuffling and maintenance (advantage). Compared to the GFR, the thermal spectrum in an HTGR leads to the buildup of plutonium and minor actinides that present a waste-disposal challenge (disadvantage). The graphite matrix, which composes the bulk of the core mass, results in large volumes of spent fuel per unit of energy generated, unless the graphite is separated from the TRISO fuel particles before disposal. Surface contamination in the graphite becomes radioactive, however, and would also need to be separated during the graphite recycling process.

2.1 Prismatic and Pebble-Bed Core Configurations

Although they share the attributes described above, two design variants of the MHTGR have been developed. Outside of the active core, they are quite similar. The core geometry differs significantly between the two and thus both are described here. General Atomics (GA) began development of the prismatic (block) MHTGR in the 1960s and, with support from the U.S. Government, pursued variations of the concept in support of different missions. Two HTGRs designed by GA—the 842-MWt Fort St. Vrain (Figure 1) and 115-MWt Peach Bottom 1 (Figure 2)—delivered power to the U.S. grid. One prismatic reactor is in operation today, the 30-MWt High-Temperature Engineering Test Reactor (HTTR), built by the Japan Atomic Energy Agency and located at the Oarai Research Laboratory. In the 1980s, GA developed the MHTGR and commenced preliminary licensing activities with the Nuclear Regulatory Commission (NRC). The MHTGR serves as the reference prismatic design for a number of code-to-code benchmark evaluations.

Figure 1. Operating deck above the Fort St. Vrain core.

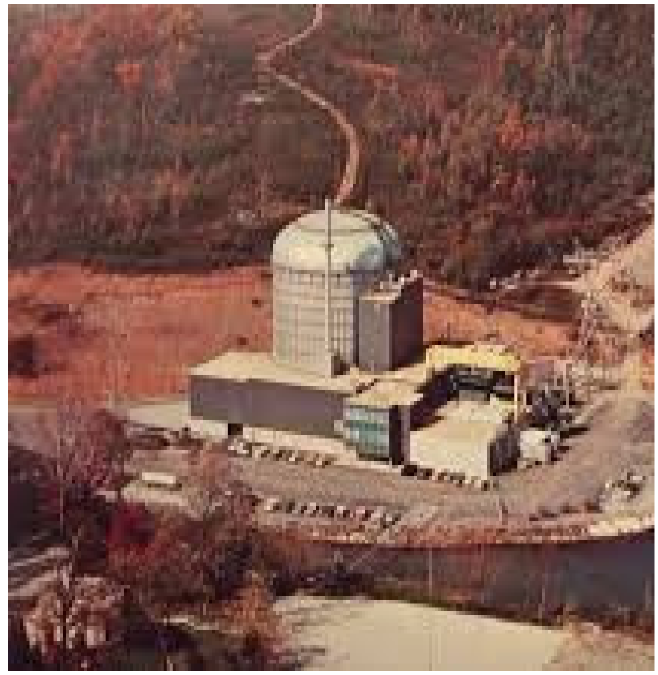

Figure 2. Peach Bottom 1 HTGR (1966–1974).

The other class of HTGRs is the pebble-bed reactor (PBR) pioneered in Germany, following roughly the same development trajectory as the prismatic core. The 46-MWt Arbeitsgemeinschaft Versuchs Reaktor (AVR) (Figure 4) operated for 20 years, primarily as a demonstration system and a testbed for pebble fuel, but it also delivered power to the grid. It was succeeded by the Thorium HochTemperatur Reaktor (THTR) (Figure 4), which operated only for a few years, but also generated electricity. One PBR is in operation today, the 10-MWt high-temperature reactor (HTR)-10 built by China's Institute for Nuclear Energy Technology. In the 1980s, Interatom, a German industrial consortium, commenced licensing activities (in Germany) on the HTR module, a 200-MWt modular PBR design that featured online fueling. China adapted this design for the 250-MWt HTR-pebble molecule (HTR-PM) and is in the process of building a two-unit plant that is scheduled to commence operation in 2018.

Figure 3. AVR (Germany, 1967–1988).

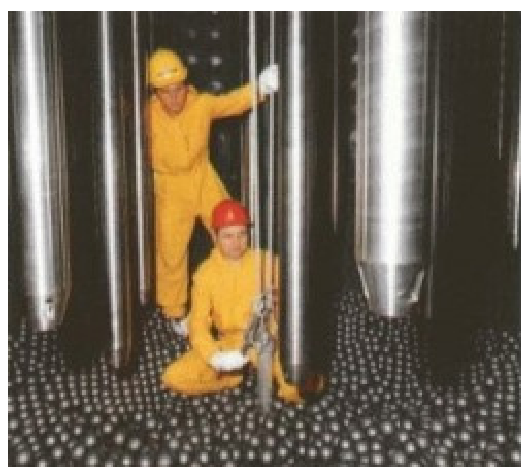

Figure 4. Workers on top of the THTR core.

Both HTGR variants employ TRISO fuel particles are bonded in a graphite matrix to form either a cylindrical 'compact' or a spherical pebble (Figure 3). TRISO particles consist of various layers acting in concert to provide a containment structure that limits radioactive fission-product release. They include a fuel kernel surrounded by porous carbon, inner pyrolitic carbon, silicon carbide, and outer pyrolytic carbon layers. The buffer layer allows for limited kernel migration and provides some retention of gas compounds. The silicon carbide layer ensures the structural integrity of the particle under constant pressure and also helps retain metallic fission products. For the prismatic reactor, compacts are inserted into hexagonal graphite blocks to form the prismatic fuel elements. For the PBR, the pebbles are made up of TRISO fuel particles surrounded by 5-mm layer of graphitic matrix material that forms a protective shell around the inner fueled zone (Figure 3).

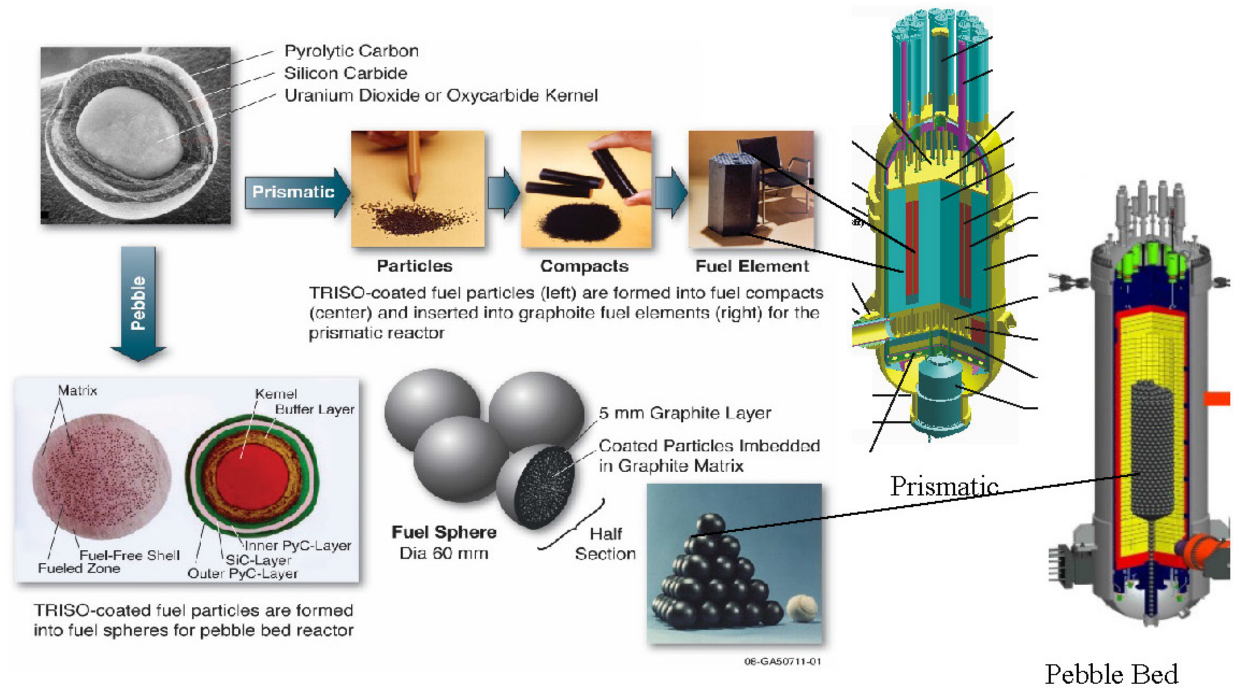

Figure 3. TRISO fuel as loaded into a prismatic or PBR.

In the prismatic reactor, the core consists of hexagonal graphite fuel elements containing blind holes for fuel compacts and full-length channels for helium coolant flow. The fuel elements are stacked to form columns that rest on support structures. Rings of block columns form either a cylindrical or annular active core surrounded by reflector blocks composed of graphite. Control rods for power shaping and load following penetrate holes in the radial reflector. Additional shutdown rods may penetrate holes in the fuel columns. The core is defueled and refueled with a block-handling machine that is lowered through the vessel head through one of the control-rod drive penetrations.

In the PBR, the core consists of a packed bed of (randomly loaded) pebbles forming a cylinder or annulus. Control rods are limited to holes in the reflector blocks. Fresh pebbles are dropped into the core cavity and (in most designs) circulate through the core during operation. After traveling through a discharge chute, the burnup of each pebble is measured. If a pebble has not achieved its burnup limit, it is routed back to the top of the vessel using a pneumatic transfer system; otherwise, the pebble is routed to a spent-fuel canister. In principle, this allows continuous operation punctuated by maintenance outages.

In both concepts, helium gas at pressure (4 to 7 MPa) is circulated downward through the core and sent on to a steam generator. The high-temperature coolant can be used to generate steam for industrial processes and high-efficiency electrical generators.

In the event of a blower trip, loss of load, or break in the primary coolant boundary, temperature reactivity feedback quickly stops the fission reaction. If limited in core power (<650 MWt for the prismatic and <400 MWt for the pebble bed with appropriate core configuration), neither would require active decay heat removal to ensure fuel integrity. The TRISO-coated particle fuel, large graphite mass, low power density, and natural heat removal mechanisms are sufficient to maintain fuel integrity under all postulated scenarios. This eliminates the need for expensive active decay heat removal systems.

More detailed descriptions of the prismatic MHTGR and pebble-bed HTR-module can be found in [2].

2.2 HTGR and VHTR

The term ‘high-temperature gas-cooled reactor (HTGR)’ was coined during the early development of the concept and covers the entire family of thermal reactors moderated by graphite and cooled by helium. Outside of the United States, the term ‘high-temperature reactor (HTR)’ is more commonly used. That term will be largely avoided here to avoid confusion with the molten-salt-cooled reactor (MSR) that operates over a similar temperature range. The term ‘very-high-temperature reactor (VHTR)’ refers to an HTGR that attains reactor outlet temperatures greater than 750°C. The VHTR was selected as one of the six Generation IV concepts [3]. Even though these temperatures have been attained for limited intervals in the engineering-scale reactors AVR and HTTR, structural alloys have not been qualified in the United States for extended use at the temperatures. The HTGRs being proposed currently for commercial deployment are designed to produce lower outlet temperatures (650 to 800°C).

2.3 Technological Maturity of the HTGR and VHTR

The HTGR has been the subject of government and industry R&D efforts for decades. Power reactor development of the HTGR was pursued by companies in the United States and Germany, with plants licensed by state regulatory agencies and operated for commercial use. While much of the early development of LWRs was conducted by the U.S. Navy in support of its ship propulsion program, from the beginning HTGRs were largely a commercial power venture. These early HTGR power plants thus experienced technical challenges typical of first-of-a-kind technology. Although the reliability of these plants did not match the performance of the LWRs operating at the time, they did demonstrate that the HTGR could evolve into an industrial energy source.

The Peach Bottom Atomic Power Station was ordered by the Philadelphia Electric Company and operated as a prototype from 1966 to 1974 with a rated power of 115 MWt. Designed and built by GA, Peach Bottom 1 converted 40 MW of its 115 MW of thermal power into electricity and supplied the grid with an 88% capacity factor.

Fort St. Vrain was also designed by GA. Although too large to operate without active decay heat removal systems, this commercial power reactor used an earlier version of TRISO fuel (containing highly enriched uranium and thorium) and demonstrated the basic physics and system technology likely to be deployed in today’s MHTGRs. GA also designed the 350-MWt MHTGR and submitted the design to the NRC for a pre-application safety evaluation [4] in the 1990s. Further efforts by GA led to the gas turbine-modular helium reactor (GT-MHR), that would have featured a Brayton PCS, coolant outlet temperature of 850°C, and 9Cr-1Mo pressure vessel for higher-temperature service. The GT-MHR effort spurred international collaboration on HTGR/VHTR R&D.

AREVA drew upon GT-MHR features, including the 850°C outlet temperature, in the design of the ANTARES plant. For the NGNP Project, AREVA modified the ANTARES design to yield a higher (900°C) outlet temperature to support hydrogen production [5]. More recently, however, AREVA scaled back the outlet temperature in the development of the 625-MWt steam-cycle high-temperature gas-cooled reactor (SC-HTGR), again adapting many features of the MHTGR and GT-MHR. The NRC conducted a review of key licensing issues as part of the NGNP Project. The reference design burns fuel once to high burnup (no recycle) to drive a Rankine power cycle. Likewise, X-energy is developing a 200-MWt pebble-bed HTGR with design features similar to the HTR-PM. Both would exhibit the safety and performance features of the MHTGR, while incorporating proven steam-based power conversion technology.

Outside of the United States, the governments of South Africa, Japan, Korea, and China have invested significantly in HTGRs over the past 25 years. Although the South African effort (the pebble-bed modular reactor) effectively ended in 2010, China, Korea, and Japan still invest in the basic technology. Moreover, Japan and China operate engineering-scale reactors, as mentioned above.

Even though the technology has been demonstrated on a commercial scale, materials and fuels used in the first-generation HTGRs are no longer available and would nonetheless need to be re-qualified to meet today's performance and safety standards. Since 2003, the U.S. Department of Energy (DOE) has supported HTGR and VHTR development through:

- TRISO fuel fabrication, irradiation, and accident testing in cooperation with Oak Ridge National Laboratory and BWX Technologies, Inc. (fuel vendor)
- Characterization, irradiation, and qualification of new grades of graphite
- Testing and American Society of Mechanical Engineers (ASME) codification of Alloys 617 and 800H for structural metals exposed to the high-temperature helium
- Design and safety analysis methods and validation, including the operation of integral test facilities at Oregon State University and Argonne National Laboratory (ANL) and numerous separate and mixed effects experiments at Idaho National Laboratory (INL) and universities.

Under the NGNP Project that ran from 2005 through 2011 with the Congressionally authorized goal of demonstrating a VHTR by 2021 [6], three vendors conducted design and engineering analyses, defined design data needs, and evaluated the commercial applications and viability of the VHTR. This work included assessments of technological maturity for two prismatic designs (AREVA and GA) and one pebble-bed design (Westinghouse/pebble-bed modular reactor). The maturity levels of the systems and subsystems were largely similar among the three vendors, reflecting the significant commonalities among the designs and development history.

In 2015 as part of a study examining needs and options for non-LWR demonstration reactors and irradiation test reactors, a team of DOE laboratory technical staff performed technological maturity assessments for the Generation IV advanced reactor concepts. As available, design information from vendors pursuing one of these concepts helped to assess the maturity of the family. For each reference design, the technological maturity of each subsystem and the encompassing systems were evaluated based on available vendor design information and recent a technology assessment performed by the vendors or by DOE. Technology readiness was quantified using the scale established by DOE [7] and summarized in Appendix A.

The overall maturity of each concept was defined as the minimum of the technology readiness levels (TRLs) of a set of key subsystems that are required for a concept to achieve its performance goals. A technology with a TRL of 3 or less is considered to be in the exploratory R&D phase; between TRLs 4 and 6, the technology is in the engineering development phase; and above TRLs 6, the technology is in the commercial deployment phase. Of the concepts reviewed in the study, the sodium-cooled fast reactor (SFR) built on the Experimental Breeder Reactor-II platform and the HTR with outlet temperatures limited to 750°C were both assessed to have a TRL of 5.

Table 1 lists the TRLs assigned by the assessment team to the AREVA SC-HTGR and a VHTR design based on the modified ANTARES with 900°C outlet temperature. The cells in the TRL value columns that are shaded indicate the key systems and subsystems to be developed fully in order for a design to achieve its performance objectives. At the time the technology assessment was performed, the AREVA SC-HTGR was the only vendor design for which sufficient information was available to perform an assessment. It should be noted that these TRLs represent the consensus opinion of the authors of the technology assessment and may not match those reported by the vendor. Nonetheless, the assessment was greatly informed by data provided by the vendors in their reports [8,9].

Table 1. TRLs for each system and subsystem of the HTGR and VHTR (key subsystems are shaded).

System/Subsystem	**HTGR SC-HTGR**	**VHTR**
Nuclear heat supply	5	5
Fuel element (fuel, cladding, assembly)	6	6
Reactor internals	6	6
Reactivity control	6	6
Reactor enclosure	5	5
Operations/inspection/maintenance	5	5
Core instrumentation	6	3[a]
Heat transport	5	3
Coolant chemistry control/purification	7[b]	7[b]
Primary heat transport system (hot duct)	6	6
Intermediate heat exchanger (IHX) (if applicable)	Not applicable (N/A)	3
Pumps/valves/piping	5	5
Auxiliary cooling	6	6
Residual heat removal	5	5
Power conversion	6	6
Turbine	7	5
Compressor/recuperator (Brayton)	N/A	5[b]
Reheater/superheater/condenser (Rankine)	7	7
Steam generator	7	7
Pumps/valves/piping	6	6
Process heat plant (e.g., H_2)	N/A	3
Balance of plant	6	6
Fuel handling and interim storage	6	6
Waste heat rejection	7	7
Instrumentation and control	6	6
Radioactive waste management	6	6
Safety	6	6
Inherent (passive) safety features	6	6
Active safety system	6	6
Licensing	3	3

Table 1. (continued).

System/Subsystem	HTGR SC-HTGR	VHTR
Safety design criteria and regulations	3	3
Licensing experience	3	3
Safety and analysis tools	4	4
Fuel cycle[c]	N/A	N/A
Recycled fuel fabrication technology		
Used fuel separation technology		
Safeguards	3	3
Proliferation resistance—intrinsic design features (e.g., spent nuclear fuel accountability)	3	3
Plant protection—intrinsic design features	3	3

a. Core instrumentation that can withstand full power conditions in the core is still under development. Previous and current HTGRs operated are without it so it is not considered a key subsystem limiting the maturity of the overall system.

b. Revised upward since the publication of [1] based upon recent input from vendors.

c. Included for consistency with the other concepts studied. Fuel is not intended to be recycled in current HTGR designs but recycling is not categorically precluded by the technology.

As can be inferred from Table 1, most of the systems and subsystems of the HTGR will be used in the VHTR. The VHTR may drive a hybrid PCS, making some combination of a gas turbine and steam for process heat and electricity. Nonetheless, the lack of qualified high-temperature alloys is the main difference for the maturity of the HTGR and VHTR. This is discussed further in Section 3.

3. R&D NEEDS

3.1 Common R&D Needs of Advanced Reactors

General information about the technical maturity and development needs of different non-LWR advanced reactors are identified in a roadmap [10] and a technical review of eight advanced concepts performed by DOE in 2012 [11]. The technology maturity assessment, conducted as part of the advanced demonstration and test reactor options study [1], identified key subsystems that must be matured to achieve performance and safety goals. This assessment, however, is only a statement of the readiness of each concept and its likely subsystems, not a plan for its development.

This roadmap report goes a step beyond technology assessment with identification of the sequencing and rough schedule for maturation of the subsystem technologies. Any roadmap must include the impact of development on those subsystems that are vital for near-term deployment. Other systems and subsystems must be developed and/or adopted in order to provide nominal operational readiness and longer-term performance goals. In this process, key technology items and long-lead R&D needs are identified.

Most advanced reactor concepts share common features such as high performance fuels and materials, passive decay heat removal systems, improved efficiency power conversion, and advanced instrumentation and controls. However, the technology options employed vary for each specific concept. Based on previous evaluations noted above, some HTGR R&D needs shared by other systems include:

- TRISO fuel and graphite (VHTR, solid-fueled MSR)
- Core instrumentation (VHTR, solid or liquid-fueled MSR)
- High-temperature structural alloys and joining techniques (VHTR, GFR)

- Gas turbine (Brayton) cycle (VHTR, GFR, MSR)
- Supercritical carbon-dioxide PCS (advanced SFR [AFR-100], MSR, LFR, VHTR)
- Reactor vessel cooling (VHTR, SFR, GFR).

General information about the technical maturity and development needs of different advanced reactor was obtained from the roadmap [10] and a technical review of eight advanced concepts performed by DOE in 2012 [11].

3.2 HTGR R&D Needs

The prismatic reactor and PBR concepts are in a comparable state of development. For coolant outlet temperatures limited to 750°C, existing alloys are adequate for the metallic components in the primary loop. Except for certain metallic structures in the primary loop, the R&D needs of the lower-temperature HTGR and VHTR are largely the same.

Based upon the technology readiness and the effort and resources needed to complete key development tasks, the R&D needs and priorities are described in the following bullets. Specific items were gauged as low, medium, or high priority as judged by members of the HTR Technology Working Group (HTR TWG), composed of representatives of HTGR vendors actively developing concepts. Much of the following is derived from a memo sent by the HTR TWG to DOE in January 2017 [12]:

- ***Fuel Element (TRISO fuel qualification).*** A major technical requirement for a new reactor concept is qualification of the fuel, the result of which is a data set that would support licensing and operation of an HTGR [13]. The fuel qualification program serves to develop and qualify fuel manufacturing processes as the foundation for commercial-scale, coated-particle fuel manufacturing in the United States. It is comprised of the following major elements: fuel fabrication, fuels and materials irradiation, post-irradiation examination (PIE) and safety testing, fuel performance modeling, and fission-product transport and source-term determination.

  The purpose of these program elements is to develop and qualify a TRISO-coated-particle fuel fabrication process, while providing fuel for a series of irradiation experiments and subsequent PIE and safety testing to obtain an understanding of fuel performance under various reactor conditions. These data will assist in the development and validation of fuel-performance models and provide fission-product transport and source-term determination data.

  The Advanced Gas Reactor Fuel Program currently sponsored by DOE is centered on seven different irradiation experiments, with some of the experiments combined into the same experiment capsule for cost and schedule efficiency. Important elements of the program include fuel kernel fabrication and coating, compacting (cylindrical and pebble), irradiation, and PIE, and heating (accident) tests. These are important not only for fuel performance but also for characterizing the source term for accidental releases of radioactivity to the environment.

  The procedures and specifications for manufacturing TRISO particles will be documented in a limited-scope topical report to be submitted to the NRC.

- ***Reactor Internals.*** Qualified nuclear-grade graphite is no longer available due to depletion of feedstock used in its manufacture. The purpose of graphite creep (advanced graphite creep [AGC]) qualification is to obtain irradiation performance data on new nuclear graphite grades at different temperatures, compressive loads, and fluences to support design of an HTGR [14]. The program currently sponsored by DOE is centered on six capsule irradiations in INL's Advanced Test Reactor, that are designated as AGC-1 through AGC-6, followed by PIE of the graphite specimens.

  To achieve the 2035 deployment target, the HTR TWG has recommended that graphite R&D be performed in two phases: preliminary and detailed. The R&D needs for graphite materials are considered “high priority” by HTR vendors as it is necessary to the licensing of an HTGR and it required extensive irradiation and PIE capabilities only available at national laboratories.

Development of ASME and American Society for Testing and Materials codes and standards for graphite is essential for timely application of graphite for HTGR technology.

Qualification of TRISO fuel and codification of graphite in ASME and American Society for Testing and Materials require long lead times and are therefore essential to deployment by 2035. Other essential R&D requires fewer resources and can be conducted by DOE laboratories directly or by a vendor given sufficient financial support. Cost and schedule estimates were provided by the HTR TWG but are not included here.

The following items are needed for deployment but, individually, present less of a cost and schedule challenge compared to the fuel and graphite qualification.

- ***Reactivity Control (rods, fixed absorbers, etc.).*** Control rods (using boron carbide) used in the first-generation HTGRs are adequate for today’s designs. The control-rod guide tubes located in the upper plenum and the absorber jackets within the control-rod assemblies may be subjected to very high coolant temperatures in the event that forced cooling is lost. High-temperature alloys or composites (described in the next subsection) may be candidates for these structures. Such advanced components are not essential for deployment, but they would reduce investment risk and downtime following certain rare events.
- ***Fuel-Handling System.*** Currently, the fuel server portion of the prismatic-block HTGR fuel-handling system requires additional development as the core geometry of modern prismatic HTGRs differs from the earlier demonstration plants. The remainder of the fuel-handling system components, including the fuel elevator, adaptor plate, and fuel-handling machine, would be similar to those demonstrated at the Fort St. Vrain reactor. In addition, the Japanese HTTR utilized a similar set of components. These parts would not require further research. Due to its “low priority” [12], the fuel server system will be designed during the design program. Testing of the fuel server system, beyond initial component testing, will be incorporated into the fuel-handling system development and testing program (regardless of who sponsors it).

  The pebble-bed HTGR uses an online refueling system, a key component of which is a burnup measurement device that determines the degree of fuel burnup in each fuel element (pebble) as it exits the reactor. This measurement determines whether the pebble is returned to the core or sent to a spent pebble storage vessel. A prototype burnup measurement system has to be developed for any PBR with online fuel recirculation.
- ***Reactor Enclosure (pressure vessel, core barrel, etc.).*** Existing materials are deemed suitable for the pressure vessel and core barrel structures in HTGRs operating less than 750°C; however, there may still be licensing and codification issues remaining with SA508/533 steel related to long-term creep behavior up to 500,000 hours of service time at elevated temperatures.

  Certain metallic components exposed to core conditions may be subjected to damage during accident sequences. If coolant temperatures are limited as mentioned above, SA508/533 (the steel alloy used in LWRs) is adequate for the pressure vessel. Metallic control-rod drive tubes, however, may be damaged in the event of the most severe loss-of-forced-cooling events. While this is not expected to lead to fuel damage or vessel failure, it would necessitate extended shutdown for repairs. These vulnerabilities may be designed away but new materials may offer a better performance options. Qualification of new alloys, or carbon or silicon carbide (SiC) composites for the guide tubes are discussed in later sections.
- ***Coolant Chemistry Control and Purification.*** The components of the helium-purification system and the shutdown cooling system have been evaluated, and no R&D needs have been identified due

to similar subsystems currently in use, or were used, in various other helium-cooled reactors. Qualification of the helium-purification charcoal can be performed during the commissioning phase.

- ***Primary Heat Transport Loop Structures, Components (pumps/valves/piping), and IHXs.*** Existing alloys are suitable for the HTGRs with outlet temperatures limited to 750°C. This includes SA508/533, which is used for pressure vessels. However, there is a general understanding that material R&D is essential for VHTRs with gas turbines and IHXs that would operate for extended periods at higher temperatures. IHXs are not used in current HTGR designs.

  Circulators up to 4 MWe have already operated in HTGRs. A test program would be dedicated to component qualification during the commissioning phase rather than as an R&D task. Planned tests ("low priority") include air tests of the impeller (at scale 0.2 to 0.4), helium tests of magnetic and catcher bearings, tests of the circulator shutoff valve, and full-scale integrated tests.

- ***Residual Heat-Removal System.*** Most HTGR designs employ a reactor cavity cooling system (RCCS). The uninsulated reactor vessel radiates excess heat to an array of air- or water-cooled panels connected to a heat exchanger for ultimate rejection to the atmosphere. Such a system has not been demonstrated in the United States on a power plant, but the HTTR in Japan employs one that is termed the vessel cooling system. The basic physics and components of the system are fairly common and well-understood, but, depending on the system configuration, the complex fluid behavior observed in some experiments can affect structural design loads. Proper design and sizing of the system will require a demonstrated understanding of key heat-transfer and fluid-flow parameters for the vessel wall, panel surfaces, riser channels, and system piping.

  Large-scale demonstration of the capability of the RCCS to remove reactor decay heat is under way at the Natural Circulation Shutdown Heat Removal Facility at ANL. In 2016, ANL completed a series of tests on a half-scale, air-cooled RCCS. The hardware is now being reconfigured with water-cooled panels. These tests provide valuable validation data for safety analysis models. The HTR TWG gives "high priority" to the completion of these tests.

- ***Instrumentation.*** An HTGR demonstration project will be the test bed for testing and validating HTGR technology, and specific instrumentation might be required for operation at high temperature. The details of this instrumentation (in particular the operating conditions) will be a function of the type of the HTGR, actual tests and experiments envisioned, and surveillance strategy.

  For neutron flux detectors, some R&D and qualification efforts may be desirable to select detector technology and to verify adequate sensitivity and durability. For temperature measurements, the standard thermocouples used in nuclear plants today are capable of measuring operating temperatures up to 1200ºC. Monitoring accident conditions may require the use of Pt-Rh thermocouples for operation at higher temperatures. These types of thermocouples are not currently used in nuclear environments and limited data about their reliability in nuclear environments exists. R&D will be needed to qualify Pt-Rh thermocouples for use in the HTGR, particularly if measurement of temperatures within the core is desired. Existing instruments for measuring flux or local power can be inserted into control-rod channels while the reactor operates at low power (and temperature). A high-temperature fission chamber is under development at Oak Ridge National Laboratory.

  Instrumentation cannot be inserted into a pebble-bed core. Peak coolant temperatures can be obtained by circulating graphite pebbles containing meltwires. This is useful for estimating bypass flow and, along with flux profiles measured in the reflector, fuel temperatures. For the most part, however, in-core fuel and coolant conditions must be inferred from data taken outside the core. These tests will likely be performed during the initial operation of the demonstration reactor.

- ***Component Testing.*** A large (10 MW) helium test loop with integral system testing, as well as component and separate-effects testing capabilities, is required for prototype tests of components, particularly for the VHTR. This can be performed within a single component test facility or at separate smaller facilities. Additional funding would be needed for actual hardware tests (including a smaller 1-MW test facility). A component test facility was a major item under the NGNP Project [15] as such a facility is needed to complete testing of systems, structures, and components of any deployment effort. The Component Test Facility (CTF) planned under NGNP would have been a 25- to 50-MWt facility operating under prototypical conditions (up to 950°C). The CTF would provide:

- Qualification and testing of large-scale components in a high-temperature, high-pressure (7 to 25 MPa) environment, such as the IHX, ducting and insulation, mixing chambers, steam generator, high-temperature valves, specific application high-temperature instrumentation, and helium
- Helium circulators, reactor internals testing, chemistry control systems for helium coolant with associated contaminants and impurities, and steady-state and transient analysis of coupled systems and components
- Design code development verification and validation collaboration
- Materials development and qualification
- Manufacturer and supplier evaluation and development.

Requirements of a CTF would be modest for a lower-temperature HTGR, but would be significant for a VHTR. If a CTF were to be constructed to support near-term HTGR deployment, enough margin and consideration should be given to including margin so that it can be used for higher-temperature component testing later on in support of the VHTR and GFR.

• ***Design and Safety Analysis Methods.*** Codes and models were developed for the first generation of HTGRs, but underwent little maintenance or development after the 1980s. Those employed for LWR analysis lack some of the physics and features needed to accurately simulate HTGRs under all anticipated scenarios, particularly with respect to accurate treatment of the time-scale of transients. Some of the missing features include cross-section generation accounting for multiple (at least three) layers of heterogeneity in the fuel and core, fuel-element shuffling (three-dimensional block for prismatic, online pebble movement and recirculation for the pebble bed), bypass flow of helium between blocks, and radiant heat transfer between pebbles and blocks. The first-generation codes accounted for these phenomena at the engineering scale, but the high-fidelity techniques available today were unavailable then. Design and safety analyses were therefore characterized by large uncertainties and simplifying assumptions in order to facilitate simulation on the relatively limited computational resources available at the time. Because modern computational tools are adapted to provide more sophisticated consideration of many of these assumptions, the differences in the fundamental analysis framework required for HTGR analysis compared to LWR analysis is an important consideration.

The few first-generation HTGR codes and models available today if used, would need to be subjected to extensive and expensive verification and validation to be accepted by the regulator. More modern tools are being developed at the national laboratories and by vendors, but need further development to treat the features of the HTGR. They will also require verification and validation but that task may be easier given the modern architectures upon which they are being built. A limited amount of code development for HTGRs has been conducted at INL in fuel performance (PARFUME), core analysis (PHISICS-RELAP5 and PEBBED), and pebble dynamics (PEBBLES). These codes and models are considered 'proof-of-principle' or TRL ~3 and were developed to address the unique physics of HTGRs. They are not suitable to support vendor design and licensing activities without additional work.

Models of the HTGR using the system codes are being validated using data from the High Temperature Test Facility (HTTF) at Oregon State University and other experiments, as needed. System codes such as RELAP5-3D run models that are considered low-resolution and low-fidelity codes, but are suitable for many accident scenarios, as they capture the predominate physics involved. Nonetheless, studies at INL have shown that RELAP5-3D models may under-predict coolant and fuel temperatures because of the assumptions and simplifications employed in building and running them. In general, system and computational fluid dynamics codes designed for LWR analysis will have to be evaluated on their abilities to simulate all of the design basis events and operating modes of a given HTGR. It is likely that they will need to be re-optimized for this application.

High-fidelity multi-physics tools are the subject of significant DOE-funded R&D. These tools can assist in the design of plants and provide power insights into the behavior of HTGR systems, structures, and components. Many of these tools, such as BISON, NEK5000, MAMMOTH, SCALE, PRONGHORN, RELAP7, and others in the Nuclear Energy Advanced Modeling and Simulation Toolkit, have been applied on a very limited basis to HTGR problems but show great potential. Computational fluid dynamics, in particular, can be used to investigate natural circulation flow under accident conditions and to investigate the extent of hot streaking of the coolant emerging from the core and flowing through the hot duct.

Fission-product transport codes such as MELCOR, are likely suitable for HTGR applications, but some validation data (such as isotherms in graphite) are needed. The Advanced Reactor Technologies (ART) HTGR Program plans to assess current capabilities against needs.

Data from integral and separate-effects experiments are needed to validate neutronic, thermal-fluid, fuel-performance, and fission-product-transport behavior during operational and accident scenarios. Past critical experiments for HTGRs can provide validation data for physics codes such that a new critical facility may not be necessary. A number of separate-effects experiments have been conducted at universities and the national laboratories. A quarter-scale integral experiment, the HTTF at Oregon State University, is just beginning tests of the fluid behavior of an HTGR during and after depressurization. If its capabilities and features are fully exploited, the HTTF can provide much of the data needed for HTGR thermal-fluid model validation. Additional experiments would be required depending on the specific designs and scenarios required to be analyzed for a license application.

- ***Design and Analysis Methods Validation.***

  Post-Break Cavity and Vessel Flow: In the event of a break in the primary coolant boundary, the core will depressurize at a rate dependent upon the size of the break. After pressure equilibrium is reached, cavity gas (air/helium mixture) may enter the core and cause some oxidation of the graphite near the inlet plenum. A combination of integral and separate-effects tests is needed to characterize the extent to which air can enter the core. The HTTF at Oregon State University was commissioned by the NRC to study this scenario. Testing is now supported by the DOE ART Program. A number of complementary separate- and mixed-effects tests are being performed by universities under DOE sponsorship through the Nuclear Energy University Program (NEUP) and by the NGNP Alliance. Data from the combined set of separate effects and integral effects are needed to support design and licensing.

  Water Ingress: In the event of a steam-generator or shutdown-cooler-tube rupture, water can enter the primary loop from the secondary loop. In the event of water ingress, isolation valves in the secondary loop close to limit water ingress, dump valves will open to drain remaining water in the steam generator, and the reactor will shut down. Although large-scale water ingress events occurred at both Fort St. Vrain and AVR with no damage to the fuel, those plants were shut down for extended periods for dryout and repairs. Also, the release of fluid from the primary is a significant contributor to radiological releases from the plant. Mixed- and separate-effects tests would be useful for understanding plant and component behavior in these scenarios. Simple assumptions may be sufficient for bounding safety analyses, but this would have to be confirmed. Some work is being performed at universities under NEUP sponsorship and should continue.

Other Experiments: The full validation matrix investigates certain phenomena that may affect material performance in an HTGR. Explored phenomena included bypass coolant flow between blocks, hot jet behavior in the lower plenum, core heat transfer, and gravity-driven exchange flow in the primary duct after a break. A few experiments conducted at INL under NGNP have been supplemented with a number of similar experiments at universities under NEUP. The outstanding tests remaining to be performed are design-specific and are to be determined in cooperation with the vendors.

3.3 VHTR R&D Needs

If operation at higher temperatures (>850°C) is required (e.g., for a VHTR), new alloys or composites will need to be developed and qualified, in addition to all of the R&D items described above. The ART Program is currently sponsoring a limited R&D program in high-temperature materials [9].

- ***High-Temperature Alloys.*** The primary candidate for the VHTR pressure vessel is modified 9Cr1Mo (modified Grade 91) steel. This alloy has already been used in fossil-fueled power plants and has been the subject of past and current fast reactor R&D programs, including the DOE ART Program.

  Major issues with modified 9Cr-1Mo are availability and welding. If the vessel is to be fabricated from stacked forged rings, one must weld the rings circumferentially, a difficult task. In the United States, one can weld a vessel from rolled semi-circular sections using longitudinal welds. There are very few forge shops that can forge rings of the size necessary for a large gas-cooled reactor vessel. Japan Steel Works may be the only one currently capable of such large forgings, but they have little experience with 9Cr-1Mo melting or forging and they may decline to perform this work.

  Grade 91 steel is challenging, because it needs a very specific high-temperature solution heat treatment, quench, and then tempering heat treatment. While it has been used in fossil steam generators, it has been difficult to get vendors that are not used to this steel to actually carry out the proper heat treatment. Welding heavy sections, in particular, is challenging, as it degrades the local heat treatment and material properties. For example, this welding process may lead to Type 4 cracking in the heat-affected zone adjacent to the weld metal, a type of failure that occurs at times well short of the predicted creep life. The solution to this problem is to either re-heat treat the steel (not recommended by vendors) or operate in a lower-temperature range to avoid creep. A significant characterization and testing effort would need to be conducted to determine if modified 9Cr-1Mo would be suitable for higher-temperature HTGR vessels.

  The remaining data needed for this material are the mechanical properties of heavy-section products (base and weld metal), effects of aging and radiation, corrosion in helium environment, weldability risk, emissivity, negligible creep conditions, and creep fatigue. A specific test program on representative plates and forgings (including welded joints) will be required for component qualification. Modified 9Cr-1Mo is covered by the ASME *Boiler and Pressure Vessel Code* [16] up to 371°C in Subsection NB and beyond 371°C in Subsection NH. This subsection does not currently cover heavy section products and needs to be updated to cover specific aspects of modified 9Cr-1Mo. Actions have already been launched in the context of the DOE/ASME Generation IV material project to provide a basis for code development.

  For metal structures operating in cold helium and outside of the radiation field, Alloy 800H is a prime candidate [17].

- ***IHX.*** The R&D inputs are based on two base IHX concepts (tubular IHX and plate IHX for either power conversion or direct process heat interface) and a third compact IHX concept.

  INL is conducting tests on Alloys 617 and 800H to extend their applications to higher temperatures (up to 950°C for 617 and 850°C for 800H) in ASME III Subsection NH of the *Boiler and Pressure Vessel Code*. This will allow the use of these alloys in IHXs. Additional experiments at higher temperatures may be needed for the VHTR.

  Development of the IHX or a custom-built steam generator requires specialized testing facilities (heated test loops) to be designed and built.

- ***Composites.*** No nuclear components or structures made of composites were used for the past HTGRs or for other reactor concepts. The use of composites is driven by their high resistance to high or very high temperatures. An R&D program should be launched to explore the possible use of such materials inside the primary circuit, because they promise better performance and durability. Thermal insulation using composite materials will be needed to provide thermal protection of metallic components that would otherwise be subjected to helium at very high temperatures.

  The R&D needs for applied composite materials (carbon/carbon or carbon/SiC composites) emphasize qualification of material properties such as:

  - Thermal physical properties (thermal conductivity [K]), coefficient of thermal expansion, and heat capacity
  - Mechanical properties, including multiaxial strength and fracture properties
  - Fatigue properties
  - Behavior in an oxidizing atmosphere
  - Oxidation effects on properties.

  In addition, for thermal insulation, ceramic materials qualification should yield thermal-physical properties and behavior under oxidation.

  Additional tests for control-rod ceramic materials include irradiation effects on properties, including irradiation-induced dimensional change and irradiation-induced creep and tribology.

  No control rods made of composites were used for past HTGRs, or for other reactor concepts. Other composites, such as carbon/SiC, are also envisioned. An R&D program should be launched to explore the possibility of employing such composites for the control rods. SiC/SiC composites are not considered mature enough to meet the near-term HTGR deployment. Composite material usage would be considered for the next generation of the HTGR, namely the VHTR. The R&D needs for ceramics are considered a “medium priority” by the HTR TWG.

- ***Hot Gas Ducts.*** The reference design for the primary and secondary hot gas duct is the V-shaped metallic concept. This design appears to be compatible with the core expected outlet temperature, subject to demonstrating that no significant hot streaks occur. There is also a ceramic insulated concept, which will be retained as a fallback option.

  Hot gas duct qualification should be performed in three steps: (1) elementary tests to characterize the fiber conditions, assembly techniques, spacers, etc.; (2) sub-scale mockup tests, about 1 MWt in helium if possible, to validate fiber specification and ceramic spacer specification; and (3) full-scale mockup tests, around 10 MWt. In the first stages of the design, tests should cover both the metallic and ceramic concepts. The HTR TWG gives “high priority” to the completion of these tests.

- ***PCS.*** The VHTR power cycle includes a combination of Rankine and Brayton cycle components. The major components of the Rankine cycle including cycle controls and ducting, heat recovery system, steam cycle, and generator and electrical equipment. Many decades of use in fossil-fired systems have taken these to a very high TRL. No R&D needs have been identified for the Rankine cycle components. However, Brayton cycle turbomachinery has not been employed to this extent and will require development, particularly if used in direct-cycle HTGRs.

  Metal corrosion will occur when exposed to hot cycle gases such as nitrogen or carbon dioxide. The nitriding effect tends to embrittle metallic parts. This could lead to failures of turbine blades and pressure boundaries, such as boiler tubes and gas shells. The need to experimentally determine the degree of nitriding that occurs in potential PCS materials, and to quantify the effects of temperature on nitriding, has been identified. This R&D need is not only for turbomachinery, but also for IHX (tube) and the Brayton-cycle gas duct. In addition, R&D is also needed for compressor blade performance in order to ensure high efficiency, mitigating the risk of lower than expected PCS efficiency.

4. LICENSING

The information generated in an advanced reactor research effort is used to support the safety basis of a reactor design and thus is necessary to successfully license a nuclear plant, regardless of the licensing pathway chosen by the vendor [1]. Consequently, test plans and conclusions that support a technology safety case and demonstrate regulatory compliance should consider those requirements while protocols are planned and performed. Properly informed planning helps ensure technology research activities adequately address later licensing needs. The advanced reactor technology regulatory technology development plan (RTDP) [18] links major research activities in advanced non-LWR technologies, as sponsored by the DOE Office of Energy, to key regulatory requirements and licensing challenges likely to affect deployments in the domestic commercial energy market. The expected outcome is a new framework for the licensing of advanced reactors. The RTDP currently focuses on two technology types likely to undergo NRC safety review within in the next 20 years (i.e., the MHTGR and SFR), both of which have received the bulk of ART R&D funding in the past decade.

Establishing linkage between reactor research and licensing is complex and requires interaction and coordination with the design community, NRC staff, and researchers working to bring conceptual system designs to maturity. The RTDP was created to aid that linkage and further NRC's Advanced Reactor Policy Statement of 2008 (restated in NRC's 2012 report to congress on advanced reactor licensing [19]). This statement encourages reactor research in new safety and security features, or proposals for simplified, inherent, and passive means to accomplish a safety or security function. That information is then to be presented to NRC staff to help assure adequate confirmatory testing, provide for collection of sufficient data to validate computer codes, and show system interaction effects are acceptable.

5. WASTE MANAGEMENT AND SAFEGUARDS

All HTRs currently proposed would operate on a 'once-through' fuel burning policy. Moreover, the coated-particle fuel form would require an additional deconsolidation process to remove the graphite matrix and expose the kernel material for reprocessing using existing techniques. These attributes make HTGR fuel even less attractive than LWR fuel as a medium for producing weapons material. Furthermore, spent HTGR fuel possesses a higher fraction of non-fissile actinides; thus making spent fuel diversion even less attractive.

One potential issue confronts the PBR. The small fuel element size and online fueling feature of most current designs allows a diversion pathway by which fuel elements passing once through the core can be collected and removed rather than being re-injected into the core to achieve their target burnup level. These diverted elements may contain fissile Pu-239 in a relatively pure form (although not a pure as what can be obtained in a sodium fast reactor breeder blanket designed for the purpose.) With only 7 to 10 g of heavy metal per sphere, a large number of such pebbles would need to be diverted to achieve a ‘significant quantity.’ Diversion at this scale would be detectable in a rational safeguards regime. It would also have a significant negative impact on fuel economy as these reactors are designed to operate with very little excess reactivity that could be exploited to mask weapons material production. Application of ‘safeguards by design’ principles have been proposed to manage these characteristics of the pebble-bed HTGR [20].

Prismatic HTGRs ‘batch-load’ their fuel in large blocks making such automated diversion of fuel much more difficult (comparable to the LWR). Application of ‘safeguards by design’ principles have been proposed to manage specific characteristics of the prismatic HTGR [21].

Deployment of large numbers of HTGRs would stress the existing waste management and safeguards infrastructure, as it would for any advanced reactor concept, albeit in different ways. HTGR fuel is not intended to be recycled (although it is technically feasible) thus avoiding the risk of diversion of special nuclear material in a separations facility. Because fuel elements consist mostly of graphite, HTGRs generate much *higher volumes* of spent fuel per unit of energy than both LWRs and fast reactor concepts. On the other hand, the decay heat generation rate, which limits repository capacity, is comparably lower. Deconsolidation (and recycling) of the graphite matrix would significantly reduce the spent fuel volume but the remaining spent fuel particles would possess a comparably higher volumetric decay heat rate.

The policy and infrastructural challenges of large-scale deployment of HTGRs is beyond the scope of this report.

6. SCHEDULE

Figure 4 shows the high-level schedule for completing the identified research, development, design, licensing, and construction tasks. The HTR TWG provided much of the information on the material and component qualification tasks.

		2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032-2035
FOAK Design, Licensing, Construction																
Licensing (Application and Review)								Pre-Application			Construction Permit			Operating License		
Design							Conceptual		Preliminary		Final					
Procurement																
Construction & Startup testing																
Generic HTR Research, Development, Qualification																
Fuel and Structural Materials																
	TRISO Particle Fuel Qualification	AGR-5/6/7 Irradiation and PIE														
	Fuel Qualification Proof Tests & PIE (design specific)															
	Graphite Qualification	AGC-4 and AGC-5 Irradiation an				AGC-6 and PIE										
	Additional Codification of SA508															
Component Testing																
	Ducting and Insulation															
	Steam generator															
	Isolation valves															
Methods																
	Validation of Neutronic & Thermal Fluids	HTTF and RCCS Integral Tests, SET (NEUP)														
	Code/Model Development & Verificatior	Core analysis methods			Model & Code Verification/Qualification											
Instrumentation																
	Pebble Burnup Measurement System															
	HT Flux Detector (fission chamber)															
	HT Thermocouple															
Fuel Handling System																
Shutdown Cooling System																
Materials & Components for Very High Temperature and Fast Reactor designs (VHTR & GFR)																
	Alloy 617 (<950°C) and 800H (<850 °C)															
	Modified 9Cr1Mo Qualification															
	Composites Research and Qualification															
	IHX	Compact HX			Tubular IHX			Plate IHX								
	High Temperature Valves															
	Power Conversion (Brayton)	Helium, Air Brayton, or sCO2														
	Hot Gas Duct															
Licensing Framework		Generic AR Licensing			HTGR Specific Licensing											

R&D and Licensing Framework

Design, Licensing, Construction (Vendor)

Figure 4. High-level schedule supporting 2035 commercial demonstration of an HTGR.

7. SUMMARY

An R&D roadmap has been constructed to support commercial demonstration of an HTGR by 2035. The starting point for the roadmap is the technical readiness assessment performed as part of an advanced test and demonstration reactor study released in 2016 [1]. As the HTGR is a relatively mature technology, most of the R&D consists of development and qualification of fuels, materials, and components, and validation of modeling methods, rather than basic material selection and research.

The highest-priority items are the completion of the TRISO fuel and graphite testing and qualification efforts that have been funded by DOE for the past decade. These two items are the ‘long poles in the tent’ because of the length and complexity of the required irradiation experiments and PIE. The special infrastructure needed to do this work (a high-flux-materials test reactor and hot-cell facilities) is available only within the DOE complex.

Design methods and validation of codes and models are also high-priority items, but the required infrastructure is less of a hurdle, as model development and experiments can be conducted at universities, if properly coordinated and integrated. Two large integral experiments have been built with DOE funding and are producing results. A number of important separate- and mixed-effects experiments are being conducted at universities and in collaboration with other domestic and international organizations.

Component testing will be needed to complete testing and qualification of heat exchangers, circulators, and other equipment needed for a first-of-a-kind power plant.

Research into new materials is not required to support demonstration of an HTGR with an outlet temperature limited to 750°C. For VHTRs with outlet temperatures greater than this value, metallic alloys and composites that can withstand the higher temperatures will need to be qualified and inserted into the ASME code. Likewise, components to be made of the alloys (heat exchangers, steam generators, etc.) will need to be designed and tested. Deployment of a VHTR by 2035 will require these additional tasks to be performed in parallel with the other HTGR R&D activities. Alloys 617, 800H, and modified 9Cr-1Mo are currently undergoing testing at laboratories in the DOE complex.

The sharing of costs for the R&D between industry and the U.S. Government is not addressed in this report.

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Appendix A

Summary of DOE Technology Readiness Levels (TSLs)

Appendix A

Summary of DOE Technology Readiness Levels (TRLs)

Phase	TRL	Attribute
Basic research and development	1	Basic principles observed and reported
	2	Technology concept and/or application formulated
	3	Analytical and experimental critical function and/or characteristic proof of concept
Engineering-scale development and demonstration	4	Component and/or system validation in laboratory environment
	5	Laboratory scale—similar system validation in relevant environment
	6	Engineering/pilot-scale—prototypical system validation in relevant environment
Commercial demonstration and deployment	7	Full-scale, prototypical system demonstrated in relevant environment
	8	Actual system completed and qualified through test and demonstration
	9	Actual system operated over the full range of expected conditions