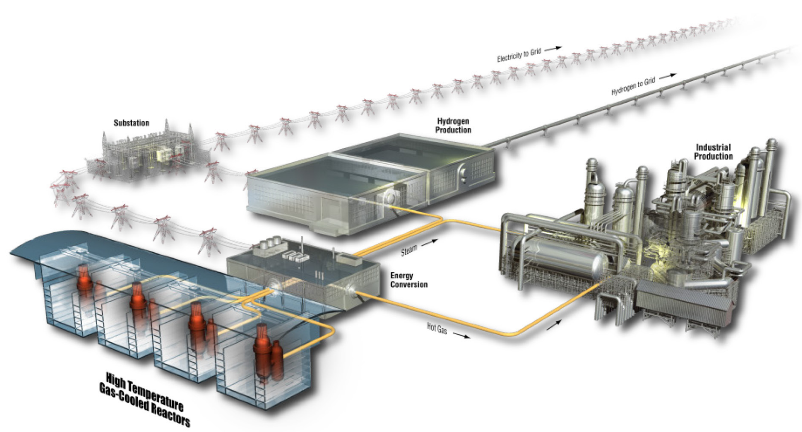
INL/EXT-17-41800
Revision 6

Gas-Cooled Fast Reactor Research and Development Roadmap

Draft for Public Comment

May 2018



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http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

INL ART Program

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Revision 6

May 2018

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EXECUTIVE SUMMARY

Nuclear power provides clean, reliable energy, contributing about 20% of the electricity generated in the United States (U.S.). It supplies approximately 60% of our non-greenhouse-gas emitting power, making it our nation's single largest contributor of carbon-free electricity. This vital component of the U.S. energy portfolio avoids hundreds of millions of tons of carbon dioxide emissions each year and supports hundreds of thousands of U.S. jobs, and yet is facing unprecedented challenges. Complex market factors, falling alternative generation costs and lower electricity demand forecasts have made operating nuclear power plants uneconomical in some parts of the country. The industry is confronting premature shut downs, a lack of new plants in the pipeline, profound market challenges and intense financing requirements. However, none of these challenges reflect a reduced need for this reliable and clean source of baseload power. Our nation's nuclear sector urgently needs to adjust to these challenges to ensure continued availability of this vital national energy resource.

The U.S. Department of Energy (DOE) and its national laboratories are aggressively working to revive, revitalize, and expand U.S. nuclear energy capacity. We are advancing nuclear energy technologies through targeted early-stage investments to ensure a strong domestic industry now and into the future. By leveraging public-private partnerships and the national laboratory system, we are developing an advanced nuclear infrastructure, encouraging a resilient supply chain, and promoting a strong nuclear pipeline. A key element of this effort is to support the recent and rapid expansion in innovative advanced reactor development already being led by the U.S. nuclear industry. Advanced reactors, particularly non-light water reactor concepts, offer the potential for significant improvements to safety, economics and environmental performance, to help sustain and expand the availability of nuclear power as a clean, reliable and secure power source for our nation.

As part of this strategy, the DOE commissioned the development of technology roadmaps for advanced non-light-water reactor concepts. 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]. The roadmaps show the Research and Development (R&D) needed to support demonstration of advanced reactor concepts, for either engineering 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. The roadmaps were developed 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 intended to be generic (i.e., not specific to a particular vendor's design); however, features of the General Atomics Energy Multiplier Module were included to provide specific examples for illuminating needed R&D tasks to be conducted in support of demonstration in the near future.

This roadmap does not discuss projected market performance or economic attractiveness for any potential concept, nor does it provide guidance on crafting regulatory positions or following policy. While these are important factors, they are largely uncoupled from the technology development requirements.

This report highlights the technical challenges of the gas-cooled fast reactorconcepts and the R&D tasks required to support a reactor demonstration under the following assumptions prescribed by the DOE:

1. Roadmaps will be developed based on technology assessments completed for the ATDR study.

2. Roadmaps will capture timelines for advancing technology readiness states to higher levels and highlight critical path activities and milestones against which progress toward demonstration can be measured.

3. Available roadmaps and technical program plans will be cited and updated.

4. Roadmaps will be developed for concepts with significant U.S. industrial interest or experience without reference to vendor-specific design attributes. Nonproprietary design features and components that have been tested or deployed on an engineering or commercial scale can be reflected in shorter development pathways.

5. Roadmaps will assume that the demonstrator reactors will use balance-of-plant systems and ancillary technologies with the highest level of technological maturity that allows performance objectives to be demonstrated.

6. Timetables will be constructed to show deployment between 2030 and 2035 as an update to past similar studies, and to show pathways for acceleration to meet projected U.S. energy needs.

This report provides the roadmap for engineering demonstration of a gas-cooled fast reactor with an outlet temperature around 850°C. The selection of tasks and associated timelines rely heavily on the Generation IV International Forum Technology Roadmaps [2,3] and more recent information provided by the gas-cooled fast reactor developer, General Atomics.

ACKNOWLEDGEMENTS

The author wishes to thank Dr. Hangbok Choi and his colleagues at General Atomics 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 and the United States Department of Energy Technical Review Panel that evaluated a number of advanced reactor concepts.

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ACRONYMS

ART	Advanced Reactor Technologies
ASME	American Society of Mechanical Engineers
Be_2C	beryllium carbide
DHR	decay heat removal
DOE	Department of Energy
EM^2	Energy Multiplier Module (General Atomics)
GA	General Atomics
GBR	gas breeder reactor
GCFR	gas-cooled fast reactor
GFR	gas-cooled fast reactor
HTGR	high-temperature gas-cooled reactor
LWR	light-water reactor
MSR	molten salt reactor
N/A	not applicable
NRC	Nuclear Regulatory Commission
PCS	power conversion system
R&D	research and development
RTDP	regulatory technology development plan
SFR	sodium-cooled fast reactor
SiC	silicon carbide
TiN	Titanium Nitride
TRL	technology readiness level
U.S.	United States
VHTR	very high-temperature reactor

Gas-Cooled Fast Reactor Research and Development Roadmap

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1. INTRODUCTION

Among the advanced reactors being considered by vendors and governments, the gas-cooled fast reactor (GFR) possesses a relatively low technical maturity. No GFRs have operated even at a strictly experimental level, however, a number of design studies and some laboratory-scale testing of materials and fuels continue at modest levels in Europe and the United States. The GFR promises high fuel utilization of a fast-spectrum reactor with a high outlet temperature that supports high-efficiency electricity generation and process heat for industrial applications.

This report describes the essential research and development (R&D) needed to support engineering demonstration of a GFR by 2035. The major features and attributes of this reactor concept are shared among all pre-conceptual designs proposed to date, thus the roadmap proposed in this report can be considered applicable for the family. Nonetheless, features of the General Atomics (GA) Energy Multiplier Module (EM^2) were used as specific examples to help illuminate technical readiness and R&D needs.

2. CONCEPT DESCRIPTIONS

2.1 General Features

The typical GFR design features an unmoderated, helium-cooled core with a ceramic fuel that operates at high temperatures (i.e., 1450°C in the GA design) and cladding temperatures as high as 1000°C. The fast-neutron spectrum supports burning of uranium, thorium, or plutonium with minimal buildup of minor actinides. Very high fuel utilization can be achieved if the fuel is recycled. Through the combination of a fast-neutron spectrum, lack of a fertile blanket, and full recycling of actinides, GFRs could minimize production of long-lived radioactive waste isotopes, while burning available fissile and fertile materials (including depleted uranium from enrichment plants). The GFR reference concept described below assumes an integrated, onsite spent fuel treatment and refabrication plant. However, this roadmap outlines the R&D needed to demonstrate only the reactor itself; fuel reprocessing technology needs are not addressed.

As with thermal-spectrum, helium-cooled or molten-salt-cooled reactors (i.e., the high-temperature gas reactors [HTGRs]), the high outlet temperature of the helium coolant makes it possible to deliver electricity, hydrogen, or process heat with high thermodynamic efficiency. Most GFR designs would drive a direct-cycle (i.e., helium) Brayton power conversion system (PCS) for electricity and would generate process heat for industrial applications. GFRs would drive either a gas turbine (i.e., Brayton) cycle drive or a steam (i.e., Rankine) cycle with or without an intermediate heat exchanger.

Some of the more distinguishing features can be best highlighted by comparing the GFR to metal-cooled fast reactors and gas-cooled thermal reactors [4].

GFRs would have the following advantages when compared to metal-cooled fast reactors:

- Chemical compatibility with water, obviating the need of an intermediate coolant loop, and generally good chemical compatibility with structural materials
- Negligible activation of coolant
- Optically transparent, simplifying fuel shuffling operations and inspection

- Gas coolants cannot change phase in the core, reducing the potential of reactivity swings under accidental conditions
- Reduction of the positive void effect typically associated with sodium
- Gas coolants generally allow a harder neutron spectrum, which increases the breeding potential of the reactor.

Compared to the graphite-moderated HTGR, the fast-spectrum GFR can support high fuel utilization and even breeding. Along with elimination of the graphite moderator, the waste volume of GFR spent fuel would be a small fraction of that generated by an HTGR using a once-through fuel 'cycle.'

The disadvantages of the GFR relative to the metal-cooled concepts are as follows:

- Higher pumping power required to cool the core.
- The need to maintain high pressure in the system; typically around 7 MPa for helium and around 25 MPa for supercritical carbon dioxide; to support sufficient cooling of the fuel.
- Depending on fuel geometry, gas cooling often requires artificial roughening of the metallic cladding to maintain an acceptable cladding temperature, resulting in an increased pressure drop over the core and a higher requirement for pumping power. Silicon carbide (SiC) cladding does not require roughening.
- High coolant flow velocity can lead to significant vibrations of the fuel pins. Grid supports may prevent these vibrations.
- The high power density, relative to the light-water reactor (LWR) and HTGR assumed in some designs, and lack of thermal capacity of the coolant require a reliable and fast response from a decay heat removal (DHR) system with considerable pumping power. EM^2 would have a lower power density (58 W/cm^3) that would place lower demands on the DHR system.

Similarly, the lack of graphite means the GFR has no thermal buffer to absorb the energy of a transient. In the event of a loss of forced cooling such as a blower trip, the core temperatures would quickly rise to failure temperatures without a robust (and probably active) DHR system. This places additional emphasis on the need to identify and qualify a temperature-tolerant fuel form.

2.2 Design History and Variations

An overview of different GFR conceptual designs and development programs is provided in Reference [4]) and the Generation IV International Forum Technology Roadmap [2] and 2014 update [3].

GFR concepts were first proposed in the 1960s. They are all characterized by a ceramic fuel form, temperature-resistant alloys or refractory metals, and powerful DHR systems. Variations on the GFR theme emerged early in development of the concept, mainly in fuel geometry, thermal power, and operating temperatures; however, none of these variations pointed to a significant change in performance characteristics.

One of the earliest concepts for a GFR was developed by GA (Figure 1). Their gas-cooled fast reactor (GCFR) would have produced 835 MW of thermal power with mixed oxide fuel arranged in pins and a coolant outlet temperature of 550°C [4]. The experimental program supporting GCFR development included critical reactor investigations under GCFR-PROTEUS (1972 through 1979) that were aimed at validating data sets and calculational methods for the design of fast breeder reactors cooled with gas [5,6,7,8].

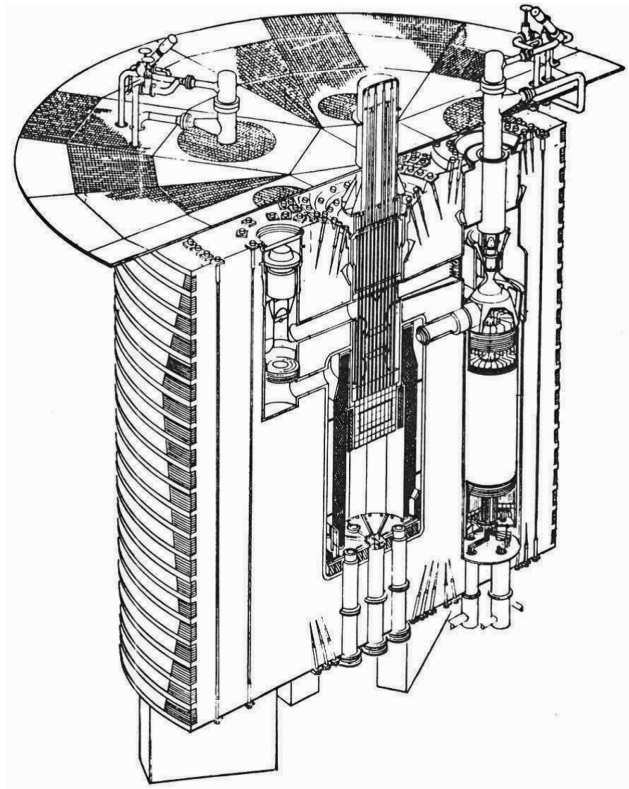

Figure 1. Early GA GCFR concept.

In the late 1960s, Germany proposed various concepts with pin-type fuel in stainless-steel cladding, a pressurized concrete pressure vessel, and driving a steam cycle. Three gas breeder reactor (GBR) concepts had higher power ratings (greater than 3000 MW) and either pin-type fuel (GBR-1), a coated-particle fuel formed into either annular fuel elements (GBR-2), or a packed bed of fuel spheres through which helium flowed radially (GBR-3). All were meant to breed fuel, typically with conversion ratios of about 1.4. The need to minimize neutron leakage to support breeding forced the cores to be relatively large and of high power, further exacerbating the ability to remove decay heat. In addition, difficulty in fabricating structural components often led to lower-temperature designs that use more traditional fast reactor cladding and structural alloys.

A United Kingdom program that ran into the 1970s produced a design using technologies developed for their sodium fast reactor program and the carbon-dioxide coolant used in their early advanced gas-cooled reactors.

Similarly, Japan developed a 2,400-MWt GFR concept as part of their fast reactor development program. This core would have used a nitride-based kernel surrounded by multiple layers made of Titanium Nitride (TiN) and other compounds. Structural materials would have been made with SiC. The design effort terminated in the late 1990s.

Interest in GFR reawakened globally with the Generation IV Initiative. Europe and Japan cooperated on development of a reference design for a 600-MWt GFR [3]. Breeding as an objective was mostly abandoned, but high conversion (i.e., a conversion ratio of about 1) was still sought to meet fuel utilization and waste minimization goals. The initially small Generation IV reference design (i.e., 600 MWt) did not appear to meet fuel utilization targets; therefore, it was superseded by the 2400-MWt design illustrated in Figure 2 [9]. The 600-MWt design is still considered an option for a gas-cooled small modular reactor, if high conversion is not required.

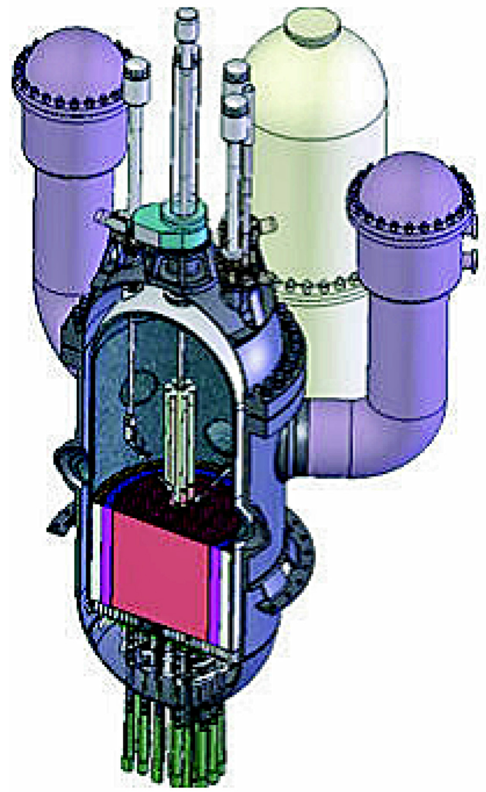

Figure 2. 2400-MWt Generation IV reference concept.

The direct power conversion cycle chosen as a reference in the original roadmap is no longer considered the only option. It was originally assumed the HTGR community would develop this technology in projects such as the pebble-bed modular reactor in South Africa and the gas-turbine modular helium reactor in the United States and Russia.

The PCS for referencing the Generation IV concept is the combined (i.e., gas-steam) cycle proposed by AREVA for their ANTARES graphite-moderated very high-temperature reactor (VHTR) [10].

Recently, GA embarked on a new GFR development project featuring their EM^2 [11]. This conceptual design features a novel PCS and a direct helium Brayton cycle (i.e., 850°C outlet), with the heat that has been rejected through the pre-cooler used to drive an organic Rankine cycle (Figure 3).

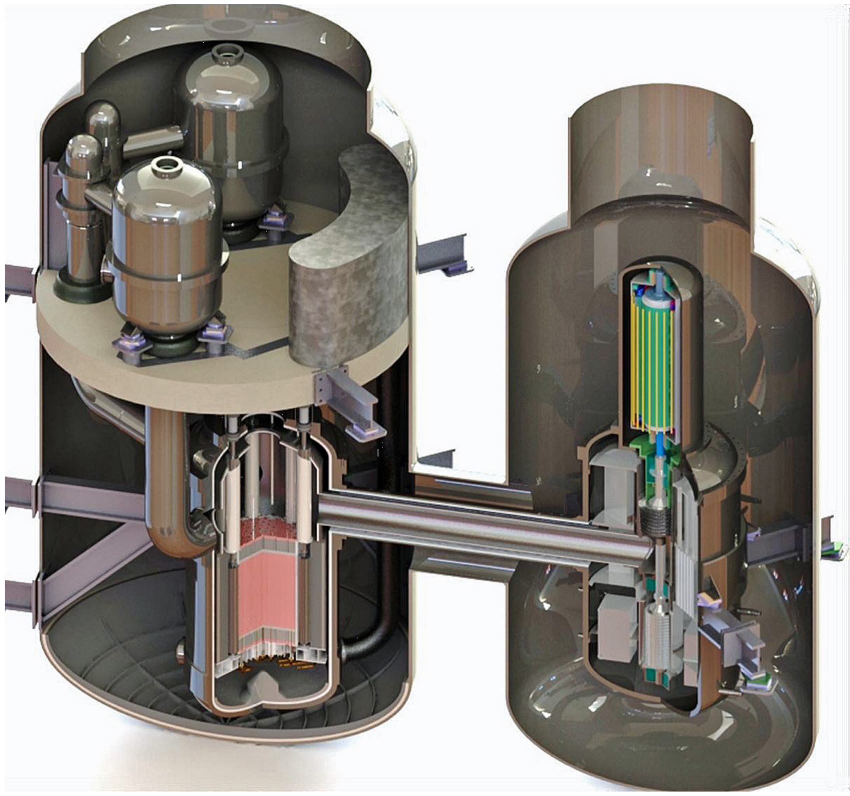

Figure 3. GA EM^2.

The EM^2 core would consist of fuel assemblies comprised of 91 annular fuel rods fabricated from uranium carbide particles sintered into pellets (Figure 4, right). The porosity of the pellets allows the particles to swell and the fission gases to vent. The fuel rods would be clad in a SiC composite material (SiC-SiC).

The core is surrounded by a beryllium carbide (Be_2C) inner reflector and graphite outer reflector to reduce fluence to the pressure vessel (Figure 4, left).

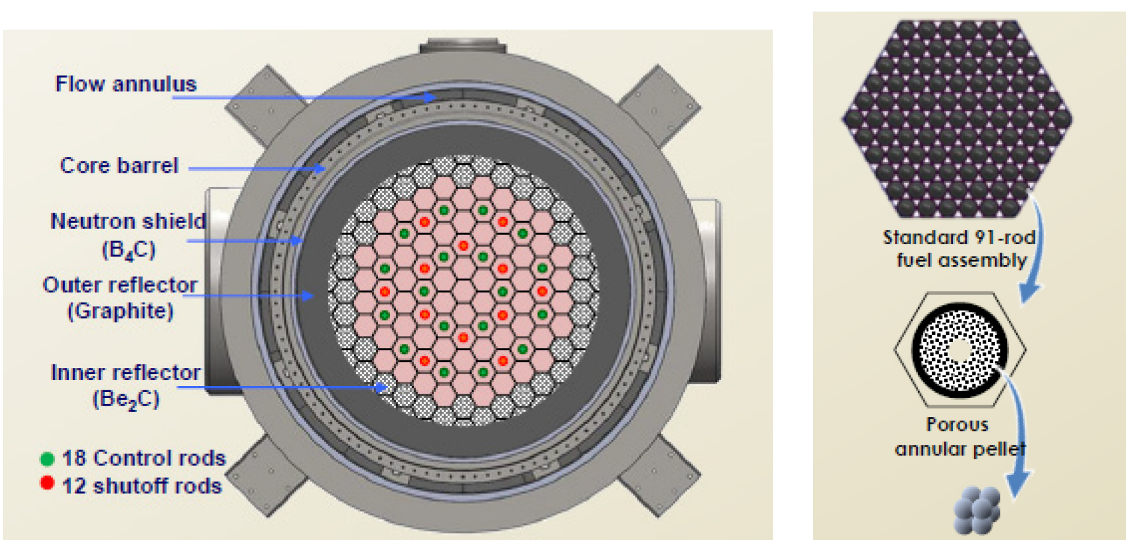

Figure 4. GA EM^2 core and fuel design.

2.3 Technological Maturity

In 2014, as part of a study examining the needs and options for non-LWR demonstration reactors and irradiation test reactors, technological maturity assessments were performed for the Generation IV advanced reactor concepts by a team of U.S. Department of Energy (DOE) national laboratory technical staff. As available, design information from vendors pursuing a concept helped to assess the maturity of the family. For each concept, the technological maturity of each subsystem and the encompassing systems were evaluated based on available vendor design information and recent technology assessments performed by the vendors or by DOE. Technology readiness was quantified using the scale established by DOE [12] and summarized in Appendix A. The overall maturity of the concept was defined as the minimum technology readiness level (TRL) for a set of key subsystems required for a concept to achieve its performance goals. Using this process, the GFR was assessed to be the least technically mature of all of the concepts considered, with an overall TRL of 2.

Table 1 lists the TRLs assigned by the assessment team to the GA EM^2 systems and subsystems. Major technologies used in all GFRs proposed to date are largely common in a way that these TRL values can be considered representative of the concept. The shaded cells in the TRL value columns indicate the key systems and subsystems needing to be developed fully in order for a design to achieve its performance objectives. It should be noted that these TRLs represent the consensus opinion of the authors for the technology assessment and may not match those reported by the vendor.

Table 1. TRLs for each system and subsystem of the GA EM^2 GFR with a combined cycle PCS (key subsystems are shaded).

System/Subsystem	GFR EM^2
Nuclear heat supply	2
Fuel element (fuel, cladding, assembly)	2
Reactor internals	3
Reactivity control	6[a]
Reactor enclosure	4
Operations/inspection/maintenance	4
Core instrumentation	3
Heat transport	3
Coolant chemistry control/purification	7 [a]
Primary heat transport system (hot duct)	6
Intermediate heat exchanger (if applicable)	Not applicable (N/A) 3
Pumps/valves/piping	5
Auxiliary cooling	6
Residual heat removal	3
Power conversion	3
Turbine	5
Compressor/recuperator (Brayton)	5[a]
Reheater/superheater/condenser (Rankine)	N/A
Steam generator	7[a]

Table 1. (continued).

System/Subsystem	GFR EM[2]
Pumps/valves/piping	6[a]
Process heat plant (e.g., H_2)	N/A 3
Balance of plant	6
Fuel handling and interim storage	6
Waste heat rejection	7
Instrumentation and control	7
Radioactive waste management	6
Safety	2
Inherent (passive) safety features	3
Active safety system	2
Licensing	1
Safety design criteria and regulations	3
Licensing experience	1
Safety and analysis tools	3
Fuel cycle	3
Recycled fuel fabrication technology	3
Used fuel separation technology	3
Safeguards	3
Proliferation resistance—intrinsic design features (e.g., special nuclear material accountability)	3
Plant protection—intrinsic design features	3
a. Updated since the release of [1].	

Most GFR concepts, including the original 600-MWt Generation IV reference design, would use a gas-turbine PCS operating in the 250 to 850°C range. The technology of such a system is to a large extent the same as that proposed for the VHTR, and thus a reasonable assessment of the maturity can be obtained from the maturity assessment conducted for the Next Generation Nuclear Plant Program [13]. For a GFR driving a steam cycle, as with the current Generation IV reference design, Reference [14] describes the state of high-temperature nuclear steam systems. For a gas turbine system driven by very high-temperature helium, Reference [15] describes the state of Brayton cycle technology.

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 [2], its 2014 update [3], 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 [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 DHR 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:

- Core instrumentation (HTGR, solid or liquid-fueled molten salt reactor [MSR])
- High-temperature structural alloys and joining techniques (HTGR)
- Gas-turbine (Brayton) cycle (HTGR, MSR)
- Supercritical carbon-dioxide PCS (advanced sodium-cooled fast reactor [SFR], MSR, lead-cooled fast reactor, HTGR)
- Reactor vessel cooling (HTGR, SFR).

As described above, the GFR PCS is largely the same as that of the HTGR; therefore, both systems would benefit from R&D performed on power conversion components and materials. However, the GFR core is subject to operating conditions that are quite different from both the HTGR and the metal-cooled fast reactors. Therefore, for the GFR, development and qualification of fuel and in-core structural materials are most significant in terms of time, expense, and capabilities of facilities. In particular, the need for in-pile testing (preferably in a fast spectrum) of fuels and materials is the main item on the critical path to deployment of the GFR. Another essential item is design and testing of DHR systems. The following subsection describes these and other R&D needs in more detail.

3.2 Gas-Cooled Fast Reactor R&D Needs

3.2.1 Fuel and Cladding

The biggest task, by far, that needs to be performed before a GFR can be deployed is developing and qualifying fuel and the cladding that encapsulates it. The core can be designed so that fuel operates at a temperature comparable to that of an HTGR (i.e., about 1000°C); however, in some designs, it can be higher (i.e., about 1300 to 1500°C) for the EM^2 and 2,400-MWt Generation IV reference design [16]. More challenging is that the temperatures of the fuel and cladding must withstand accident conditions that can be as high as 2000°C. Furthermore, these materials are subjected to much higher damage rates due to the fast flux levels to which they are subjected. The estimate for the 600-MWt Generation IV GFR damage rate is 60 displacements per atom at a burnup of 5% fissions per initial metal atom [9]. This is far higher than what is attained for thermal spectrum reactors; even for high flux test reactors that are currently the only option for testing fuels and materials.

Carbide and nitride fuels are considered for GFRs because they possess the high melting points of oxide fuels but have substantially higher thermal conductivities. This heat can be transported out of the fuel more easily during an accident scenario, reducing the peak temperatures likely to be attained. Table 2 shows the properties of different fuels proposed for fast reactors [17].

Table 2. Properties of fast reactor fuel types.

Parameter	Metal U-20Pu-10Zr	Oxide UO_2-20PuO2	Nitride UN-20PuN	Carbide UC-20PuC
Heavy metal density (g/cm^3)	4.1	9.3	13.1	12.4
Melting temperature (K)	1350	3000	3035	2575
Thermal conductivity (W/cmK)	0.16	0.023	0.26	0.20
Fuel-cladding solidus (K)	650	1675	1400	1390
Coefficient of thermal expansion (K^{-1})	17E-6	12E-6	10E-6	12E-6

Nitride and carbide fuels have not been developed and qualified for reactor use. An extensive testing and qualification effort is therefore required to support their use in a reactor demonstration. This is a four-phase process illustrated in Table 3 and adapted from Reference [18].

Table 3. Phases of nuclear fuel and material testing.

Selection of Potential Candidates	Laboratory-Scale: Concept, Definition, and Feasibility	Improvement and Scale-Up	Qualification and Demonstration
- Early scoping studies to measure basic properties - Establish performance criteria - Identify options and parameters for testing	- Investigate the range of fuel types - Down-select and establish a reference design - Fill in knowledge gaps	- Best results of laboratory-scale tests - Scale-up of fabrication process to pilot or engineering scale - Process optimization - Performance demonstration - Validation of the fuel fabrication process	- Test large quantities of rods, plates, or particles - Production-scale fabrication process - Statistical demonstration of performance under anticipated and bounding reactor conditions - Validation of performance models

Up to 8 years	5 to 8 years	8 to 15 years

Some experience with carbide fuel is being acquired in the Indian sodium fast breeder test reactor, which uses a mixed carbide driver fuel. It has reached a burnup of 155 GWd/t with a relatively high linear power [19,20]. Because the fast breeder test reactor is a sodium-cooled reactor, the fuel burnup performance is not directly transferrable to the GFR carbide fuel.

Early activities in a phase can overlap the later activities of the previous phase, so the entire process, if adequately funded, can take between 15 and 20 years. This is consistent with fuel development and qualification in the Advanced Gas Reactor Tristructural Isotropic Fuel Program currently underway [21]. It began in 2003 with a primary candidate (UCO kernels) and a secondary candidate (UO_2 kernels), both in a tristructural isotropic coating design. At historical funding levels activities will be completed around 2023.

Fuels (i.e., nitride or carbide) for GFRs have been proposed. Preliminary irradiation testing has been conducted; therefore, fuel development can be considered to be in the second phase (i.e., laboratory-scale) of testing. However, the difficulty with completing the program in a timely manner is the lack of a fast-spectrum materials irradiation facility. There are very few fast-spectrum test reactors operating in the world today and none operating in the United States. The Advanced Test Reactor and High Flux Isotope Reactor boast some of the highest fluxes of any materials test reactors, but mainly in the thermal energy range. Practically speaking, this means that fuels tested in the Advanced Test Reactor and High Flux Isotope Reactor can achieve high burnup, but low accumulated damage rates (i.e., less than 10 displacements per atom/year). Eight to ten years of continuous irradiation would be needed to achieve the desired irradiation performance data.

Cladding must meet rigorous in-core service specifications for length, diameter, surface roughness, apparent ductility, level of leak tightness (including the potential need of a metallic liner on the clad), compatibility with helium coolant (plus impurities), and anticipated irradiation conditions (spectrum, temperature, chemistry). Needs include fabrication capacities and material characterization under normal and accident conditions for fresh and irradiated fuel. Target criteria are as follows:

- Clad temperature of 1000°C during normal operation
- No fission product release for a clad temperature of 1600°C for a few hours
- Maintaining the core-cooling capability up to a clad temperature of 2000°C.

Under the Generation IV International Forum Program, a consortium of four countries (i.e., the Czech Republic, Hungary, Poland, and the Slovak Republic) plan to build an experimental GFR (ALLEGRO) to serve as both a demonstration of GFR systems and as a fuel testing platform. ALLEGRO would be the first fast-spectrum gas-cooled reactor to be constructed and would also serve as a test bed for developing and qualifying the high-temperature, high-power density fuel and cladding that is required for a commercial-scale high-temperature GFR. This fuel qualification would be carried out at full scale at representative temperatures, coolant conditions, and with correct neutron spectrum and flux.

Different cladding concepts, including the SiC-SiC cladding proposed by GA for the EM^2, would also require a lengthy testing and qualification process using a fast-spectrum test reactor. This cladding concept is being studied under contract with Westinghouse as part of the DOE Accident-Tolerant Fuels Program. As a schedule reference, the graphite characterization and qualification program currently being conducted by the DOE Advanced Reactor Technologies (ART) Program for HTGR started in 2005 and will be completed by 2026.

Any fast-spectrum test reactor would supply the fast fluence levels needed to qualify GFR fuels and materials. In a sodium-cooled fast test reactor such as the (now inoperable) Fast Flux Test Facility, test trains could be configured to test nitride and carbide fuels under near prototypical conditions. With a well-funded government effort to build a fast-spectrum materials test reactor, this capability could be available around 2030 to start a GFR fuel and material qualification campaign. This would support a full GFR demonstration system coming online 10 to 15 years later and thus would not meet the target deployment date of 2035. This date may be achievable with a combined engineering demonstration and materials irradiation campaign, such as envisioned for the European ALLEGRO program.

Another option may be to choose a viable fuel/cladding combination and immediately begin pilot-scale testing in a high (thermal) flux test reactor. If sufficient performance data could be generated, prototype fuel elements could be used to fuel the initial core of an engineering demonstration reactor to gain the additional data to fully qualify it for use in a commercial demo.

3.2.2 Components and Systems

3.2.2.1 Out-of-Pile Experimental Facilities for Qualification of the Main Systems. In terms of neutronics and zero-power reactor needs, existing calculational tools and nuclear data libraries have to be validated for GFR designs. The wide range of validation studies on SFRs must be complemented by specific experiments that incorporate the unique aspects of gas-cooled designs, including slightly different spectral conditions, innovative materials and various ceramic materials (UC, PuC, SiC, ZrC, Zr_3Si_2), and unique abnormal conditions (i.e., depressurization and steam ingress).

For core thermal hydraulics, air and then helium tests on subassembly mock-ups are necessary to assess heat transfer and pressure drop uncertainties of the specific GFR technology selected. A large-scale demonstration of the passive DHR system will be required (air and then helium tests). Testing of this system and other transient behavior must be built into the licensing process.

Development and qualification of components for heat transfer out of the core and power conversion are similar to those identified for HTGR, but with some additional requirements, specifically with respect to the following:

- Thermal barriers—During normal operation, GFR metallic structures are protected from the hot (850°C) helium flow by thermal barriers. These thermal barriers must continue to be effective during transients, typically up to 1250°C for 1 hour, withstand helium velocities of about 60 m/s, and depressurization rates in the range of 2 MPa/s. GFR-specific solutions must be developed and qualified in the relevant facilities.
- Valves and check-valves—Safety demonstration of GFRs relies on continuous core cooling by gas circulation, either through normal loops or dedicated DHR loops for which it is necessary to isolate the main loops and open the DHR loops with a high degree of reliability. Valves and check-valves are critical components of GFRs. Qualification tests of candidate technologies for these components are needed and must be performed using a dedicated helium loop.
- Instrumentation—Development of instrumentation that can survive under GFR conditions is one of the main challenges of GFRs. In particular, the main safety issue concerns temperature measurement at the core outlet; this measurement is taken in order to detect hot spots on the fuel cladding or fuel assembly plugging. The primary development objective is reduction of measurement uncertainties and development of innovative measurement methods using, if possible, helium transparency. This challenge is shared by HTGRs, but at a higher fluence rate. An instrumentation R&D program includes core temperature measurements, monitoring of structural temperatures, and optical viewing during the fuel handling and maintenance phases.
- DHR systems—The need to ensure robust DHR without external power input, even under depressurized conditions, is regarded as a requirement. Previous concepts used electrically (battery) driven blowers at low pressure. Although a given plant design may not use diesel power units that would need protection from potential flooding, the integrity of the electrical infrastructure following an extreme event must be maintained. Self-powered systems are being explored that do not require external power supplies; however, these systems are still only conceptual in nature.

The Natural Convection Shutdown Heat Removal Test Facility at Argonne National Laboratory was constructed to perform ex-core heat removal studies. In 2016, Argonne National Laboratory completed a series of tests on a one-half scale, air-cooled reactor cavity cooling system. The hardware is now being reconfigured with water-cooled panels. These tests provide valuable validation data for safety analysis models. Originally designed to support PRISM (General Electric) sodium-cooled reactor development and licensing, the Natural Convection Shutdown Heat Removal Test Facility could be re-configured to study GFR ex-vessel cooling.

3.2.3 Materials

A candidate for the GFR pressure vessel is modified 9Cr-1Mo (modified Grade 91) steel. This alloy is being considered for the pressure vessels of other advanced reactors, including VHTRs. Modified 9Cr-1Mo has already been used in conventional power plants and is also supported by significant R&D test results from 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, which is 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 neither melting nor forging experience with modified 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 working with this steel to 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 GFR 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. It has been estimated that the qualification of modified 9Cr-1Mo will take approximately 72 months due to the need for procuring a large forging with a long lead procurement time. (Additional time may be needed to complete the American Society of Mechanical Engineers [ASME] balloting and approval process.) Modified 9Cr-1Mo is covered by the ASME (ASME Boiler and Pressure Vessel code 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.

3.2.4 Modeling and Simulation

Since the early days of the GFR concept, existing tools have been used to design and evaluate concepts. Because no GFR has been built, validation of these tools has not been possible. The basic neutronics of a fresh core can be captured with modern tools such as Monte Carlo N-Particle and SERPENT. Estimates of core neutronic parameters at high burnup can be made, but uncertainties in resonance parameters and other nuclear data inject considerable uncertainty that will only be reduced with operation of a plant. In the GA design, neutronic coupling between the core and dual reflector materials may be complex and require validation using a critical experiment.

Thermal-fluidic analysis of GFRs under operating and accident conditions is challenged by the lack of data on gas cooling of a ceramic core. A matrix of separate effects and integral tests will need to be constructed and executed to generate validation data. The High Temperature Test Facility at Oregon State University is now generating data for validating models of prismatic HTGRs during depressurization events. This facility may be useful for GFR code validation, if the core internals can be re-configured to yield the thermophysical properties of a GFR.

Given the anticipated development time of GFR fuel and cladding, there is time to develop multi-physics models and codes such as those being developed under the Nuclear Energy Advanced Modeling and Simulation and other DOE programs. Although the challenge of validating these high-fidelity models is significant, these models may capture the complex physics of GFR transients and provide key insights into fuel and system design.

4. LICENSING

Data and information resulting from an advanced reactor research effort are often key parts of the technical development effort needed 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 ART Program regulatory technology development plan (RTDP) [22] links major research activities in advanced non-LWR technologies 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. Until recently, the ART RTDP currently focused on two technology types likely to undergo Nuclear Regulatory Commission (NRC) safety review within in the next 20 years (i.e., the modular HTGR and the SFR). This effort would need to be expanded to include the GFR.

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 ART 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) [23]. This statement encourages reactor research in new safety and security features or proposals for simplified, inherent, and passive means for accomplishing a safety or security function. This 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. To support deployment of GFRs, the ART RTDP should be expanded to include this reactor concept, focusing on the GA EM^2 because it is the only U.S.-based GFR design currently under development.

5. WASTE MANAGEMENT AND SAFEGUARDS

The fast-neutron spectrum in a gas-cooled reactor with no moderator enables the core to burn fuel more 'cleanly' than a graphite-moderated HTGR. As with other fast reactor concepts, the fission-to-capture ratio of uranium or thorium is much higher in a GFR compared to reactors operating with a thermal spectrum. This enable it to burn the fuel with much less buildup of the plutonium and minor actinides that pose a repository challenge. No actinides are generated in the inert (non-fertile) reflector in the EM^2. All conversion of fertile material to fissile fuel takes place in the active core.

Neutron leakage from the core requires that it be fueled initially with a starter core containing a high fissile content (12% enriched uranium in the case of the EM^2). The fertile material is then converted to fissile fuel that can then sustain the reaction in subsequent cycles. This fertile material may include depleted or natural uranium, spent LWR fuel, or thorium thus extending available fuel supplies into the indefinite future.

This level of fuel utilization, however, requires (in existing design concepts) fuel reprocessing. Separation of fission products from fissile and fertile fuel elements in such a facility poses a safeguards challenge that would need to be addressed through a combination of technical and policy advances.

For these reasons, deployment of large numbers of GFRs (or any reactor relying upon spent fuel separations to achieve high utilization targets) would require changes to the existing waste management and safeguards infrastructure. The benefits of a significant reduction in actinide volumes and increased fuel supplies must be weighed against the increased risk of diversion posed by the separations process.

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

6. SCHEDULE

Figure 5 shows the high-level schedule for completing the identified R&D tasks. This schedule assumes that the ex-core materials and components developed for the VHTR would be used in the GFR.

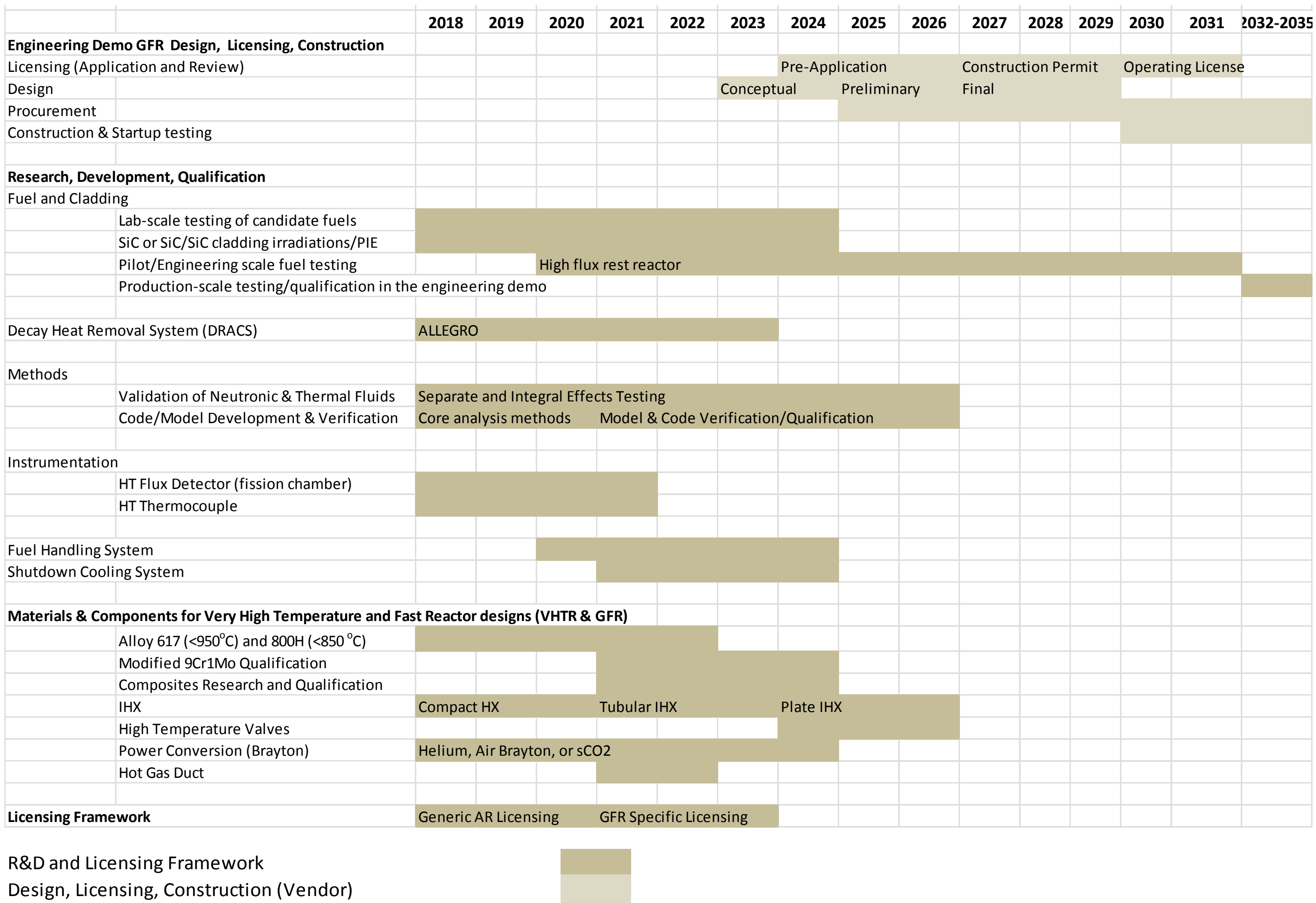

Figure 5. Approximate schedule for supporting GFR engineering demonstration.

7. SUMMARY

An R&D roadmap has been constructed to support an engineering demonstration of a GFR 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]. Because the GFR is a relatively immature technology with challenging requirements for fuel/cladding performance and DHR, most of the R&D must focus on the development and qualification of the ceramic fuel form (carbide or nitride) and the SiC cladding. Strong emphasis must be placed on development and testing of a reliable and powerful DHR system. The first item is the 'long pole in the tent' because of the length and complexity of the required irradiation experiments and post-irradiation examination (for fuel and cladding).

The special infrastructure needed to do this work (i.e., high-flux materials test reactor and hot cell facilities) is available only within the DOE complex. Even then, the available materials test reactors produce a relatively low fast flux needed for reproducing damage in fuel and cladding candidates. Basic testing of these materials must commence soon in order for adequate fluences to be attained in candidate materials.

A matrix of separate, mixed, and integral effects experiments must be designed and carried out, ideally with a number of university and international partners. These will provide data needed to validate as-yet developed codes and models.

As with the VHTR PCS, the operating parameters of which are very similar to the GFR, a component test facility will be needed to complete testing and qualification of valves, circulators, and other equipment needed for a first-of-a kind power plant. Research into new materials is required to support demonstration of a GFR. Materials being developed for the VHTR (with outlet temperatures greater than 800°C), metallic alloys, and composites that can withstand the higher temperatures will need to be qualified and incorporated into the ASME code. Likewise, components made of alloys (e.g., heat exchangers and steam generators) will need to be designed and tested. Alloys 617, 800H, and modified 9Cr-1Mo are currently undergoing testing at laboratories in the DOE complex. Supporting deployment of a GFR by 2035 will require many tasks to be performed in parallel.

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 (TRLs)

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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