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Revision 2

Assessment of the Technical Maturity of Generation IV Concepts for Test or Demonstration Reactor Applications

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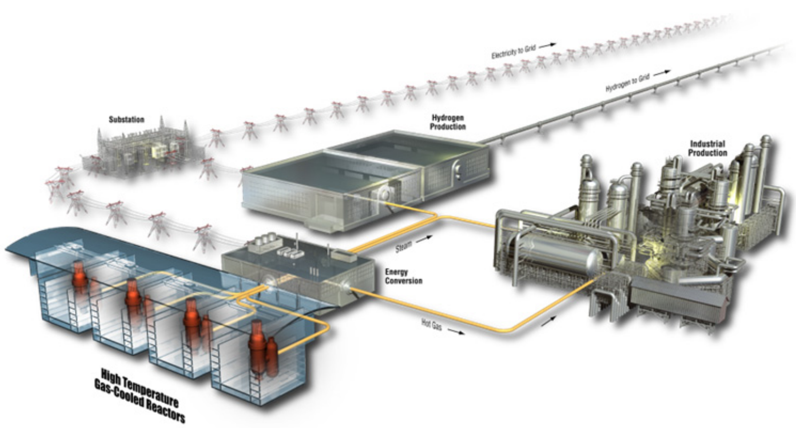

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SUMMARY

The United States Department of Energy (DOE) commissioned a study to assess the suitability of different advanced reactor concepts to support the primary mission of materials irradiation (i.e., a test reactor) and/or to demonstrate an advanced power plant/fuel cycle concept (demonstration reactor). As part of the study, an assessment of the technical maturity of the individual concepts was undertaken to see which, if any have sufficient maturity for either test or demonstration reactor missions, and can support deployment of the test and/or demonstration reactor. A working group was appointed to perform the maturity assessment using a technical readiness scale adopted by DOE.

Of the six Generation IV concepts that have been the subject of past or current research and development in the United States, a sodium fast reactor, built upon the technologies demonstrated in the Experimental Breeder Reactor-II and the Fast Flux Test Facility, is considered mature enough to support preliminary design and licensing activity within the next 10 years, assuming funding levels appropriate for those activities. Likewise, such activities could commence on a test reactor built upon the graphite moderated high temperature reactor technology platform provided that the current tristructural isotropic (TRISO) fuel and material qualification programs are completed as planned and that a few other subsystems are adequately developed and qualified.

Based upon the TRLs evaluated for the major plant systems, a graphite-moderated high temperature reactor similar to that proposed by AREVA (steam cycle high temperature gas-cooled reactor) and the sodium fast reactor similar to that proposed by GE-Hitachi (power reactor innovative small module), are deemed to support preliminary design and licensing activities for a demonstration reactor within the next 10 years assuming funding levels appropriate for those activities. A concerted research and development effort for either a molten salt reactor or supercritical water reactor focused on the issues identified in this assessment could lead to preliminary design and licensing in the next 10–20 years. Significant research and development (>20 years) is needed for lead-cooled and gas-cooled fast reactor technology before these concepts are ready for preliminary design.

The evaluations of relative maturity of the different concepts expressed in this document should not be misconstrued as recommendations for more or less funding of particular concepts. DOE maintains a role in exploring relatively immature but potentially disruptive energy technologies that may not be commercially viable for decades. However, the conclusions in this document can be considered an indicator of the time and effort needed to raise the maturity of a concept such that it can be deployed on a production scale.

ACKNOWLEDGEMENTS AND DISCLAIMER

This assessment leaned heavily upon prior technical assessments conducted on different systems submitted to the DOE Technical Review Panel (Phillip Finck, Chair). The authors also wish to thank the following contributors who provided overall guidance or information on specific concepts: Robert Hill (ANL), David Petti (INL), Chris Grandy (ANL), David Holcomb (ORNL), Lewis Lommers (AREVA), James Kinsey (INL), and W. Bart Roe. This report represents the consensus views of the authors based upon the information obtained from these sources and recent publications of the Generation IV International Forum. These are the views of the authors and do not represent those of DOE, its national laboratories, the Generation IV International Forum, or vendors associated with the concepts described herein.

CONTENTS

TABLES

ACRONYMS

AFR	advanced fast reactor
AGR	advanced gas reactor
ANL	Argonne National Laboratory
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor II
EM^2	Energy Multiplier Module
FHR	fluoride-cooled high temperature reactor
FLIBE	lithium/beryllium fluoride eutectic
GA	General Atomics
GFR	gas-cooled fast reactor
GIF	Generation IV International Forum
HTR	high temperature reactor
LFR	lead-cooled fast reactor
LWR	light water reactor
MHTGR	modular high temperature gas-cooled reactor
MSBR	molten salt breeder reactor
MSR	molten salt reactor
MSRE	molten salt reactor experiment
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PCS	power conversion system
PRISM	power reactor innovative small module
R&D	research and development
SC-HTGR	steam cycle high temperature gas-cooled reactor
SCWR	supercritical water reactor
SFR	sodium-cooled fast reactor
SNM	special nuclear material
TRISO	tristructural isotropic
TRL	Technology Readiness Level
VHTR	very high temperature reactor
WG	working group

Assessment of the Technical Maturity of Generation IV Concepts for Test or Demonstration Reactor Applications

1. INTRODUCTION

The United States Department of Energy (DOE) commissioned a study to assess the suitability of different advanced reactor concepts to support materials irradiations (i.e., a test reactor) or to demonstrate an advanced power plant/fuel cycle concept (demonstration reactor). As part of the study, an assessment of the technical maturity of the individual concepts was undertaken to determine how soon each could be deployed as a test or demonstration reactor given a concerted development effort. A working group (WG) composed of the authors of this document performed the maturity assessment using the Technology Readiness Levels (TRLs) as defined in DOE's Technology Readiness Guide.[1] The charter for this assessment is given in Appendix A.

One representative design was selected for assessment from of each of the six Generation-IV reactor types: gas-cooled fast reactor (GFR), lead-cooled fast reactor (LFR), molten salt reactor (MSR), supercritical water-cooled reactor (SCWR), sodium-cooled fast reactor (SFR), and very high temperature reactor (VHTR). Background information was obtained from previous detailed evaluations such as the Generation IV Roadmap, Advanced Reactor Technologies Technical Review Panel reports, the Next Generation Nuclear Plant Project (NGNP) technical reports, and other technical references as well as private consultations with concept proponents and subject matter experts. Outside of Generation IV activity in which the U.S. is a party, non-U.S. experience or data sources were generally *not* factored into the evaluations as one cannot assume that this data is easily available or of sufficient quality to be used for development, demonstration, and ultimately licensing and deployment of a U.S. facility. The WG established the scope of the assessment (which systems and subsystems needed to be considered), adapted a specific technology readiness scale, and scored each system through discussions designed to achieve internal consistency across concepts.

In general, the WG sought to determine which of the reactor options have sufficient maturity to serve either the test or demonstration reactor missions. The purpose of a test reactor is to irradiate test specimens, often within a specific range of thermochemical conditions, while that of a demonstration reactor is to demonstrate certain performance and safety attributes in the integrated system. A higher level of maturity in certain systems and subsystems may be required of a test reactor to meet its mission objectives. The maturity is quantified in terms of TRLs, which can be assigned to the overall concept as well as to major systems and subsystems of the concept. More specifically, for each advanced reactor concept, the following were specified as key questions and assumptions for the assessment:

1. Where is the current technology readiness? What are the key technology hurdles that must be overcome for deployment? Can these hurdles be addressed in a test or demonstration reactor?
2. How soon will the technology be mature enough to be used as the base technology for a test reactor (with a primary mission of irradiation services)?
3. How soon will the technology mature enough to be considered for a demonstration reactor? If so, what technology features would be demonstrated? If not, what feasibility issues need to be resolved before demonstration?

This report documents the process followed and the results of that assessment.

2. APPROACH

The evaluation process consisted of the following steps:

1. Choose the reference advanced reactor design (or design family) for each concept to be evaluated
2. Adopt a technology readiness scale appropriate to nuclear reactor systems
3. Identify the major systems and subsystems to which TRLs are to be assigned
4. Review the technical background for each reference design and assign TRLs to each subsystem based upon this information
5. "Roll up" the subsystem scores into an overall TRL for each major plant system and into an overall TRL for the design.

Detailed design information about a given concept may not exist or may not be easily acquired and condensed in the amount of time allowed to complete the study. Much of the knowledge acquired for systems that were first investigated 40 years ago may only exist in a "tribal" sense and thus would have to be regenerated for licensing purposes. There is sufficient ambiguity in the definitions of terms such as lab scale, pilot scale, engineering scale, and prototypical to defy easy mapping into one of the TRL definitions. Experiments and facilities associated with one concept may not easily map onto those of another concept, thus confounding "apples to apples" comparison. Overall, the reference information was only used as a basis for the assessment and comparison, but all values of the TRLs were independently assigned by the experts in the WG. The reference designs for each chosen reactor concept are described in the next section.

A technical readiness scale was developed by DOE and is described in Reference 1. It was deemed most appropriate by the WG for the nuclear systems under evaluation in this study. Other TRL scales, specifically those developed by the Department of Defense and the National Aeronautics and Space Administration, were considered. The use of the DOE scale also leads to results that are consistent with those of the findings of the Advanced Reactor Concepts Technical Review Panel.[2] This panel was convened on behalf of DOE to identify the research and development (R&D) needs of viable advanced reactor concepts and to help guide DOE investments therein.

However, it should be understood that the effort and time needed to raise the technical maturity of different subsystems may vary significantly. Thus, different designs with the same overall TRL may require very different development pathways and timelines.

The major systems and subsystems constructed by the WG are shown in Table 1. A TRL was assigned to each subsystem (if applicable). The overall system TRL was derived from the subsystem values this is explained further below.

When a subsystem TRL was not readily apparent from the literature or common knowledge, a WG member would share his knowledge of the subsystem in question and propose a TRL. The subsequent discussion among the WG generally yielded acceptance of the value or one close to it. There is a certain amount of uncertainty in this process in that further investigation or discussion among experts with different backgrounds could result in a different TRL. However, for this study importance was placed upon achieving consistency of scores for a given subsystem across the reactor concepts such that the relative maturity could be ascertained with confidence. The systems and subsystems are listed in Table 1 with essential subsystems shown in bold.

Table 1. Systems and subsystems evaluated. Essential systems and subsystems are shown in bold.

System	Subsystem
Nuclear Heat Supply	**Fuel element** **Reactor internals** **Reactivity control** **Reactor enclosure (pressure vessel, core barrel, and supports)** Operations/inspection/maintenance Core instrumentation
Heat Transport	**Coolant chemistry control/purification** **Primary heat transport** **Intermediate heat exchanger** Pumps/valves/piping Auxiliary cooling **Residual heat removal**
Power Conversion	**Turbine** **Compressor/recuperator (Brayton)** **Steam generator (Rankine)** **Pumps/valves/piping** **Process heat plant**
Balance of Plant	Fuel handling and interim storage Waste heat rejection Instrumentation and control Radioactive waste management
Safety	**Inherent safety features** **Active safety system**
Licensing	Safety design criteria and regulations Licensing experience Safety and analysis tools
Fuel Cycle	Recycled fuel fabrication technology Used fuel separation technology
Safeguards	Proliferation resistance—intrinsic design features and special nuclear material (SNM) accountability Physical protection – intrinsic design features and plant security

In terms of overall technical maturity, some of the systems were weighted more heavily than others because of the essential nature of the technology and to the effort and resources needed to get them to a deployable state. The balance-of-plant, licensing, and safeguards systems are important parts of any nuclear plant but, for the most part, the concepts do not differ significantly from a technical viewpoint. The power conversion system is an important differentiator insofar as the reference concept departs from using a traditional steam cycle. The fuel cycle system is important only for those reference concepts that require reprocessing to achieve high fuel utilization and waste reduction performance goals. By contrast, the nuclear heat supply and heat transport subsystems are critical to the performance of any concept. In particular, the robustness of the fuel-clad-coolant combination of subsystems drives the overall safety and performance in the sense that vulnerabilities in this ensemble can be mitigated only with extraordinary performance by other subsystems, if at all.

Therefore, while the TRL of a specific system is derived from all of the subsystems supporting it, the *overall technical readiness of a reference design was derived from a select set of key systems and subsystems*.

The "roll up" of the subsystem TRLs into a single value for each major system also had an element of expert judgment. For example, if all but one of the subsystems within a given system is assigned a TRL of 5, but one subsystem is scored at TRL of 4, the WG members would discuss the importance of that subsystem to the overall system. In other words, does the safety and performance of the plant require further R&D of this subsystem (important) or is it simply a matter of engineering the necessary component to a desired specification (not important)? The overall system TRL is then assigned the minimum value of the important subsystems.

Finally, the non-U.S. experience with particular technologies was largely discounted. The licensing and quality assurance model in the U.S. is such that credit cannot be taken for design data generated and operating experience acquired in other countries. One can point to these experiences as an indicator of the viability of a technology but deployment in the U.S. would still require licensing to U.S. standards.

3. DESCRIPTION OF EVALUATED CONCEPTS

The WG was charged with evaluating one of each of the six Generation IV reactor concepts as this family spans the range of power reactors being considered for near- or medium-term deployment. However, each Generation IV family may include a number of specific designs or unique design features. For this study, specific designs were chosen based upon the extent of U.S. public and private interest and investment. If the U.S. government or a U.S.-based vendor is not investing a specific concept, one of the reference designs adopted by the Generation IV International Forum (GIF) was chosen based upon past U.S. involvement in its conception. For reactor families of low overall technical maturity, the choice of specific design is less important as the subsystems have not been developed sufficiently to be distinguishable.

The specific designs chosen are described briefly here. The extent of design detail and technology pedigree differs greatly among the concepts and this is reflected in the following descriptions. In all cases, it was assumed that no power conversion system (PCS) would be driven by a test reactor and that all demonstration reactors would drive a Rankine (steam cycle) PCS unless otherwise specified. The exceptions are: the advanced fast reactor (AFR)-100 sodium-cooled fast reactor (for which a supercritical CO_2 Brayton-cycle system is part of the basic design) the General Atomics (GA) Energy Multiplier Module (EM^2), which is designed to drive a helium Brayton cycle, and the fluoride-cooled high-temperature reactor for which an open air Brayton cycle is proposed.

3.1 Gas-Cooled Fast Reactor

GFR is not the subject of study or technology development by DOE nor has any particular design been evaluated by the Nuclear Regulatory Commission (NRC) as part of a licensing application. One private vendor, GA, has developed a conceptual GFR design, the EM^2.

GFR was first proposed and studied in France in the 1990s. Some R&D continues under the GIF umbrella. The original GFR reference concept, the 600 MWt ALLEGRO,[3] was eventually replaced with a 2400 MWt version as calculations indicated that the 600 MWt reactor was unable meet the breakeven breeding requirement. The U.S.-designed 500 MWt EM^2 was selected as the reference GFR for the purposes of this technical evaluation, but all GFRs share the major attributes of, and challenges confronting, the $EM.^2$

EM^2 system is a high-temperature, helium-cooled, fast-spectrum reactor supporting a closed fuel cycle. It combines the advantages of fast-spectrum systems for long-term sustainability of uranium resources and waste minimization (through multiple fuel reprocessings and fission of long-lived actinides) with those of high-temperature systems (high thermal cycle efficiency and industrial use of the generated heat, similar to VHTR). As with the VHTR, the coolant is chemically inert, neutronically transparent, and remains in the gas phase.

The reactor would use uranium carbide (UC) fuel with 6.5% average enrichment and has a 30-year refueling cycle. The design has one loop and utilizes two shutdown systems, control drums, and separate shutdown rods. It would use the PCS for normal decay heat removal from the reactor vessel with the passive direct auxiliary cooling system (DRACS). The fuel would be a vented porous uranium carbide clad in silicon carbide. Conventional light water reactor (LWR) pressure vessel steel (SA508/533) would be used for the pressure vessel and the modular construction below grade. The reactor would use helium (850°C at the outlet) to drive a Brayton PCS rather than a steam cycle. While it could probably drive a steam turbine as well, for this evaluation the gas turbine PCS is assumed. The features of, and basic technical challenges faced by, the ALLEGRO and EM^2 are comparable.

GFR: 500-MWt Energy Multiplier Module

3.2 Lead-Cooled Fast Reactor

The LFR is not currently the subject of study or technology development by the U.S. DOE nor has any particular design been evaluated by the NRC as part of a licensing application. A small, transportable battery LFR has been developed at Lawrence Livermore National Laboratory—the Small, Sealed, Transportable, Autonomous Reactor. Russia has operated LFRs for submarine propulsion. One private vendor, Gen4 Energy, has developed a conceptual small modular GFR design—the Gen4 Energy Reactor-which was used as the reference design for this assessment but all proposed LFRs share the important attributes of, and challenges facing, this particular design.

The 25-MWe, Gen4 Energy Reactor[2] would use a lead-bismuth eutectic (LBE) as its coolant and would drive a Rankine conversion cycle with a reactor outlet temperature of 500°C. The reactor would use uranium nitride (UN) fuel with 19.8 % enrichment and to achieve a 10-year refueling cycle. The design has one primary loop and one secondary loop and utilizes two independent shutdown systems. The design utilizes passive natural circulation for decay heat removal from the reactor vessel with water as the ultimate heat sink. Specific design features include containing the reactor in a sealed cartridge to avoid onsite refueling, a primary shutdown system with inner and outer B4C control rods, and a secondary shutdown system having a central cavity into which a single B4C control may be inserted. Special benefits of the design include passive decay heat removal from the reactor vessel with a water jacket and the ability to operate in remote locations.

As with the SFR, the lead (or lead-bismuth) fast reactor uses a molten metal coolant with excellent heat transfer properties. The high boiling point of lead allows operation at near-atmospheric pressures. Unlike sodium, neither lead nor lead-bismuth react chemically with water. However, lead corrodes metallic structures, and significant research must be undertaken in qualifying alloys and in chemistry control. Lead also has a high-melting point that puts the system at risk of freezing if the temperature is not actively maintained. The lead-bismuth eutectic melts at a lower temperature, but neutron capture by the bismuth results in radioactive polonium production, a significant radiological hazard.

LFR: 25-MWe GenIV Energy Reactor

3.3 Molten Salt Reactor

MSRs were first proposed and developed shortly after World War II when a 2.5-MWt proof-of-principle test reactor (Aircraft Reactor Experiment) was developed and operated for 100 hours at high temperature (860°C) in 1954. At Oak Ridge National Laboratory, the 8-MWt Molten Salt Reactor Experiment (MSRE) was operated from 1965 to 1969, with over 13,000 fuel power operation hours, including an 8,000-hour continuous period of operation. This experiment demonstrated the basic technology of molten salt reactor with dissolved and recycled fuel (both U-235 and U-233).

One type of MSR, the Fluoride-cooled High-temperature Reactor (FHR) is currently the subject of a DOE Cooperative Research and Development Agreement (CRADA) with China (analysis and experimental work) as well as two Integrated Research Programs (IRPs) with a number of leading U.S. universities. The FHR is cooled (and moderated to some extent) by salt, moderated by graphite, and uses the coated particle fuel form of the HTR. FLIBE (lithium/beryllium fluoride eutectic) is the primary salt candidate but others are being considered. The choice of fuel geometry may be either block (prismatic) or pebble. The high outlet temperature of the FHR would support efficient Brayton cycle PCS and an open-air gas turbine system, using off-the-shelf components, is part of the power plant design.

The high boiling point of molten salts enables operation at near atmospheric pressures, greatly diminishing a source term release vector. The high-heat capacity and thermal conductivity (compared to helium) allows high-power densities to be attained in the fuel. Increased core temperatures lead to a decrease in salt density that reduces moderation and, along with Doppler feedback in the fuel, yields strong negative temperature feedback.

A major subfamily of the MSR is the liquid-fueled MSR (LFMSR), of which the MSRE and molten salt breeder reactor noted above are examples. Fresh fuel, actinides, and fission products are dissolved in the salt and circulated through the primary heat transfer loop. Depending on the coolant and core structural materials, the liquid-fueled MSR may have a fast neutron spectrum. Online fuel processing supports continuous operation and limits fission product buildup such that high conversion or even breeding can be achieved, particularly in the fast spectrum versions. A liquid-fueled MSR design has been proposed by a number of private U.S. and non-U.S. companies (e.g., Transatomic Power, Terrestrial Energy), but currently the U.S. government supports no research into liquid-fueled MSR.

The FHR was chosen for this TRL evaluation with background data obtained from a previous technical review.[4] Because of the considerable interest among private North American companies, the readiness of the liquid-fueled MSR was also evaluated.

MSR: FHR (base case) and Liquid-Fueled MSR

3.4 Sodium-Cooled Fast Reactor

The SFR has been under development by the U.S. government almost since the inception of nuclear electricity production in the 1950s. Experimental and demonstration facilities have been built and operated starting in the early 1960s with the Experimental Breeder Reactor II (EBR-II) in Idaho and the Enrico Fermi power plant in Michigan, both of which generated electricity. The Fast Flux Test Facility is a 400-MWt SFR in Washington State that was used for materials irradiations. Both EBR-II and Fast Flux Test Facility were shut down in the 1990s.

EBR-II used a metal fuel cladded in stainless steel that was resistant to radiation damage and has a high thermal conductivity (the Fast Flux Test Facility used a mixed oxide fuel with a higher melting point, but metallic fuels were irradiated in Fast Flux Test Facility). Recycling of the fuel was achieved in an electro-metallurgical process developed at Argonne National Laboratory (ANL). Fission heat from EBR-II was transferred to a steam generator via an intermediated heat exchange system. The sodium loops were driven by electromagnetic pumps.

Passive safety was demonstrated in EBR-II in 1986 in a series of experiments in which the electricity supply to the plant was disconnected, thereby disabling the emergency shutdown system and the primary coolant pumps. The subsequent temperature increase led to expansion of the core and subcriticality via excess neutron leakage. Decay heat was removed through natural heat transfer mechanisms and the plant shut itself down safely.

The basic technology of EBR-2 was adopted by General Electric in its design of the PRISM reactor. The 471-MWt PRISM/Mod-A design was submitted for a pre-application review by the NRC,[5] during which a number of issues were identified as requiring further development and demonstration. Higher power rate designs were also developed (840 MWt – PRISM/Mod-B and 1000 MWt-SPRISM). The PRISM submitted to the DOE Technical Review Panel[2] would feature a higher efficiency supercritical water PCS, but the more mature steam cycle of the earlier variant was assumed for this assessment. DOE-sponsored SFR development continues at ANL, focusing on the design of the next generation of SFR. The AFR-100[6] is a 100 MWe small-modular SFR adopting advanced new fuel design and driving a supercritical CO_2 PCS for higher efficiency and lower risk of sodium-water interaction.

Given active DOE support for the AFR-100 and the experience base of PRISM, TRLs were evaluated for both concepts.

SFR: AFR-100 and PRISM (Mod-B)

3.5 Supercritical Water-cooled Reactor

SCWR is not currently the subject of study by the U.S. DOE or vendor nor has any particular design been evaluated by the NRC. DOE participated to a limited extent in SCWR research through the GIF.[7]

Superheated coolant emerging from the core can be sent directly to a turbine without separation and drying and returned to the core as feedwater. Furthermore, reactor coolant pumps are not required (just feedwater pumps), and thus the primary loop is much simpler compared to a BWR. The SCWR operates at higher temperatures than today's LWRs and can potentially achieve higher efficiencies (about 45%).

The SCWR reference design supported by the U.S. in the early years of GIF[8] featured a thermal spectrum and direct-cycle PCS for simplicity. Water at a supercritical temperature (about 510°C) and pressure (25 MPa) served as both coolant and moderator. Variations on the SCWR concept include a heavy water-moderated version (Canada) and a fast spectrum version (Europe and Japan). All designs under investigation by GIF member countries are at the pre-conceptual stage and face the challenges stated above.

The SCWR uses water as a coolant and thus can draw upon 50 years of operational experience and research infrastructure with conventional LWRs and fossil plants that use supercritical water-based power conversion. The path to deployment may thus be relatively straightforward and can rely on existing R&D infrastructure.

SCWR: Generic thermal spectrum, light-water moderated SCWR

3.6 Very High Temperature Reactor

The graphite-moderated, helium-cooled High Temperature Reactor (HTR) has been the subject of U.S. government and industry R&D efforts for decades. A subset of the HTR, the VHTR, is so-called because the coolant outlet temperature exceeds 850°C. DOE continues the R&D of the VHTR that began under the Advanced Gas Reactor (AGR) program, continued under the NGNP program, and is now funded through the Advanced Reactor Technologies office. Even though these temperatures have been achieved in engineering scale reactors such as the German Arbeitsgemeinschaft Versuchsreaktor and the Japanese High Temperature Engineering Test Reactor, only HTRs with lower outlet temperatures (675°C–800°C) are being proposed for commercial deployment at this time.

The 625 MWt AREVA Steam Cycle High Temperature Gas-Cooled Reactor (SC-HTGR) has been selected by the NGNP Industry Alliance for this role. The NRC conducted a review of key licensing issues for the NGNP.[9] The reference design burns fuel once to high burnup (no recycle) to drive a Rankine PCS. The fuel is that being qualified under the AGR program (final irradiation in 2020) and consists of low-enriched uranium (15.5% enriched) in the form of coated particles. New graphite grades are being similarly qualified within approximately the same timeframe. The SC-HTGR is designed to reject decay heat to the atmosphere via a passive reactor cavity cooling system.

The Fort St. Vrain commercial power reactor was designed and built by GA. Although not designed to reject decay heat passively, this commercial power reactor used an earlier version of tristructural isotropic (TRISO) fuel (highly enriched uranium and thorium) and demonstrated the basic physics and system technology likely to be deployed in the AREVA SC-HTGR. GA also designed the 350 MWt Modular High Temperature Gas-Cooled Reactor (MHTGR) and submitted the design to the NRC for a pre-application safety evaluation[10] in the 1990s. The SC-HTGR shares the major features of the MHTGR, including passive decay heat removal.

VHTR: AREVA SC-HTGR

4. RESULTS OF THE EVALUATION

TRLs obtained for each reference design are summarized in this section. For some systems and subsystems, values could be applied broadly as the technology is somewhat independent of the nuclear heat supply system that for the most part distinguishes the concepts. These generic evaluations include the following:

1. For demonstration reactors, a Rankine (steam) cycle power conversion was assumed unless a different cycle is explicitly included in the design description. It was generally assumed that steam cycle PCS components currently available “off the shelf” for the nuclear and fossil industries are adequate for advanced reactor applications. Minor differences in power conversion subsystems are based upon independent evaluations obtained from cited documents. Power conversion was not deemed essential to a test reactor mission.
2. Safety design criteria for advanced reactors are being developed by the DOE and the NRC is currently reviewing the criteria. A two-tiered set of criteria has been proposed: a general set of criteria applicable to all advanced systems and lower detailed set, which is specific to a reactor concept. Technology-specific criteria have been developed for the SFR and VHTR and are under review. International efforts to develop safety design criteria are acknowledged. Likewise, pre-application reviews by the NRC of the proposed adaptations of the current LWR-based regulations and regulatory requirements for certain advanced (non-LWR) reactor designs are credited toward the licensing TRL. However, although that process has been exercised, it remains a source of significant uncertainty for certain designs, since the underlying NRC policy issues that would support this adaptation process are largely unresolved. Any application and evaluation effort undertaken in the near future will likely take considerable time to complete and will set a precedent for future applications. Safety and analysis tools were evaluated from the perspective of the licensee, not that of the regulator. The state of the NRC’s analysis tools are considered in the Licensing Experience subsystem.
3. Fuel cycle systems (recycled fuel fabrication and separations technology) TRLs were evaluated for only the fast spectrum concepts and for the liquid-fueled MSR, and then only for the demonstration reactors. The primary mission of a test reactor was assumed to be that of material irradiation, and whether or not the fuel would be recycled is not essential to this purpose. Fuel recycle is considered an essential technology to be demonstrated in fast spectrum concepts. Uranium extraction/plutonium and uranium recovery by extraction systems have been licensed and used for LWR fuel while electrometallurgical refinement (pyroprocessing) has been demonstrated at an engineering scale.
4. Safeguards (nonproliferation and plant protection technologies) are considered to be at a comparably low state compared to LWRs. In the Generation IV program, preliminary studies were performed for all six concepts by their developers (as they existed in 2011) in cooperation with the Proliferation Resistance and Physical Protection Working Group.[11] It was evident from these studies that each of the concepts is in the early stages of their development of proliferation resistance and physical protection implementation. As the six concepts mature, and as they incorporate safeguards-by-design into their program, they will achieve the needed robustness to meet institutional requirements. (The LWR-based SCWR is obviously most advanced in this regard since it can readily adopt LWR safeguards technology.)

The numerical scores for the different reactors are tabulated in the following table. The scale, using the readiness levels in Appendix A, ranges from 9 for technologies with operational experience down to 1 where the technologies basic principles have been observed and reported. Explanations of overall system results are then provided.

Table 2. Technology Readiness Levels for each system and subsystem for reactor deployment (key subsystems are shaded).

	GFR	LFR	SFR		VHTR	SCWR	MSR	
	EM[2]	Gen4	AFR-100	PRISM	SC-HTGR		FHR	LF-MSR
Nuclear Heat Supply	2	3	3	5	5	3	3	3
Fuel Element (fuel, cladding, assembly)	2	3	3	5	6	3	6	5
Reactor Internals	3	3	3	6	6	3	6	5
Reactivity Control	4	3	6	6	6	3	4	4
Reactor Enclosure	4	3	3	5	5	3	3	3
Operations/Inspection/Maintenance	4	3	3	5	5	5	3	5
Core instrumentation	3	3	3	5,3	6,3	3	3	5,3
Heat Transport	3	3	4	4	5,3	5	4	3
Coolant Chemistry Control/Purification	6	3	6	6	6	5	4	3
Primary Heat Transport System (hot duct)	6	3	6	6	6	5	4	4
Intermediate Heat Exchanger (if applicable)	NA/3	3	3	6	NA/3	NA	4	4
Pumps/Valves/Piping	5	3	4	4	5	5	4	4
Auxiliary cooling	6	3	NA	NA	6	5	4	4
Residual Heat Removal	3	4	5	5	5	5	4	4
Power Conversion	3	7	4	7	6	7	6	6
Turbine	3	7	4	7	7	7	7	7
Compressor/Recuperator (Brayton)	3	NA	4	NA	NA	NA	NA	NA
Reheater/Superheater/Condenser (Rankine)	NA	7	4	7	7	7	7	7
Steam generator	3	7	4	7	7	7	7	7
Pumps/Valves/Piping	3	7	4	7	6	7	6	7
Process heat plant (e.g., H_2)	NA/3	NA	NA	NA	NA/3	NA	NA/3	NA
Balance of Plant	6	6	4	4	6	7	4	4
Fuel handling and Interim Storage	6	6	4	4	6	7	6	4
Waste heat rejection	7	6	6	6	7	7	6	6
Instrumentation and Control	7	6	6	6	6	7	4	6
Radioactive waste management	6	6	6	6	6	7	6	6
Safety	2	3	6	6	6	3	3	3
Inherent (passive) safety features	3	3	3	6	6	3	4	5
Active safety system	2	3	3	6	6	3	3	3
Licensing	1	3	3	3	3	1	2	2
Safety Design Criteria and Regulations	3	3	3	3	3	3	3	3
Licensing Experience	1	1	3	3	3	1	2	2
Safety and Analysis tools	3	3	5	5	4	3	3	3
Fuel Cycle	6	6	6	6	NA	NA	NA	5
Recycled fuel fabrication technology	3	3	6	6				5
Used fuel separation technology	3	3	6	6				5
Safeguards	3	3	3	3	3	7	3	3
Proliferation resistance – intrinsic design features (e.g., SNM accountability)	3	3	3	3	3	7	3	3
Plant Protection – intrinsic design features	3	3	3	3	3	7	3	3

The overall TRL for the reference concepts was obtained by taking the minimum value of the TRLs of the key subsystems shown in shaded cells in Table 2. The "roll-up" is shown in Table 3.

Table 3. Technology Readiness Level of the overall design.

	GFR	LFR	SFR		VHTR	SCWR	MSR	
	EM^2	Gen4	AFR-100	PRISM	SC-HTGR		FHR	LF-MSR
Overall Technology Readiness Level	2	3	3	5	5	3	3	3

The justification for the TRLs of the individual reference designs is detailed in the following subsections.

4.1 Gas-Cooled Fast Reactor

Except for residual heat removal, the heat transport, power conversion, and other non-nuclear systems of EM^2 were considered to be similar to the HTR. The TRLs for the GFR and VHTR (see Section 4.6) are identical except for the nuclear heat supply and safety systems. TRL scores were determined mainly from the information given in *Generation IV International Forum Annual Report 2014*.[12] No gas-cooled fast spectrum reactor has been built.

Gas is a poor heat transfer medium and, without the thermal inertia of the VHTR's graphite moderator, rapid heat-up of the core would be expected following loss of forced cooling. As a fast spectrum reactor, the power density is characteristically high such that the VHTR phenomena of a benign "conduction cool-down" is not feasible and powerful decay heat removal systems must be considered. Also, the gas-coolant density is too low to achieve enough natural convection to cool the core, and the power requirements for the blower are important at low pressure. Lastly, additional consideration will need to be given to the effects of the fast neutron dose on the reactor pressure vessel in the absence of core moderation (the graphite moderator provides protection for VHTR systems). However, the GFR would avoid a number of the issues of the SFR such as chemical compatibility (due to its inert gas coolant), high positive void coefficient (smaller than other concepts but still positive), and reduction in outlet temperature to avoid coolant boiling (by having single phase gas coolant). Therefore, the GFR allows high-temperature operation without the corrosion and coolant radiotoxicity problems associated with heavy liquid metal reactors.

The main challenge is the development of the carbide or other ceramic fuel form that can withstand the high temperatures and power densities of a gas-cooled core, particularly during an upset condition. The nitride and oxide fuels under consideration among GIF countries are at a similarly low development state. The pressure vessel and core internals will need to withstand a drop in forced cooling without the benefit of the thermal and radiation buffer afforded by the graphite in an HTR. The overall nuclear heat supply system was therefore assigned a TRL of 2.

The other approach to preventing severe core damage is through the use of a decay heat removal system that can extract core heat at a sufficient rate to keep fuel temperatures below acceptable limits. This almost surely would be an active safety system as opposed to the passive vessel cooling systems proposed for smaller SFRs and HTRs. Such a system has not yet been designed for GFR duty. The overall safety system score was assigned a TRL of 2.

Mainly because of the challenges of fuel qualification and decay heat removal during an upset condition, the GFR concept is not likely to be ready for preliminary design and licensing activities within the next 20 years.

4.2 Lead-Cooled Fast Reactor

The absence of an energetic reaction with water comparable to the sodium case is an attractive feature of the LFR, but corrosion of metallic components by lead,[13] particularly at higher temperatures (>500°C), is considered a significant technical challenge. The current corrosion suppression strategy relies on carefully managing oxygen concentration in the lead, but the effectiveness of this approach in large pool-type LFRs is unproven. This drives the lower TRL (~3) for most of the subsystems with the nuclear heat supply and heat transport systems. Like the other metal-cooled systems, in-service inspection of reactor internals remains a challenge, the magnitude of which depends upon heretofore unspecified requirements by the regulator. It is assumed that any residual heat removal system and accident behavior would be similar to that of the SFR, but no such system has been tested in an LFR environment.

The Gen IV Energy design requires significant development and qualifications of fuel, materials, and components. Specific gaps exist with regard to the development and testing of the nitride fuel and HT-9 cladding, LBE components (pumps, heat exchangers, etc.), validation of LBE natural circulation models and correlations, and associated corrosion of the alloys used.

The full range of beyond design basis accident scenarios has not been formulated. No LFR has been licensed for operation in the U.S. The Russian experience with this reactor type is significant but not altogether positive, and it cannot be assumed that this can be applied toward a U.S. license application.

Mainly because of the unresolved corrosion challenges,[14] the LFR concept is not likely to be ready for preliminary design and licensing activities within the next 20 years.

4.3 Molten Salt Reactor

The current design path for the FHR uses the TRISO particle and graphite being qualified for the VHTR so the TRLs for the fuel element and reactor internals (graphite) match those of the VHTR.[4] (Interestingly, the FHR designers plan to take no credit for the retentive properties of the TRISO fuel in the FHR safety case.) Uncertainty in the long-term corrosion effects of FLIBE on core metallic components greatly limits the TRL of the reactivity control, instrumentation, and reactor enclosure (vessel). One of the strategies is to develop C/C or SiC/SiC composites for instrumentation sleeves and other barriers between the metals and the salt and to help withstand the higher temperatures. Qualification of metals in the MSR environment needs to be conducted. As with the other concepts, the maturity of the instrumentation will depend upon modern regulatory requirements.

The open-air Brayton cycle PCS would use commercial gas turbine components, but these have yet to be coupled to a reactor.

Likewise, the heat transport system TRL is also constrained by the lack of data on long-term corrosion in FLIBE and other salts. Furthermore, lithium-bearing salts generate considerable amounts of tritium under irradiation that permeates structural materials and poses a radiological hazard. Both tritium and FLIBE have been studied under the fusion program, but the issues remain unresolved to date. Mitigation and control of corrosion in some metallic components were demonstrated in the MSRE (Oak Ridge National Laboratory, 1956–1976)[15] along with salt purification, but qualification of all the necessary metals and components is a long-term effort for any salt that is chosen. Like lead, the salt temperature must be maintained (>400°C) to prevent freezing.

The passive safety characteristics of the MSR were demonstrated in the MSRE. This should be considered test laboratory scale or perhaps engineering scale although it was never connected to PCS. The full range of design basis accident scenarios (e.g., failure of the heat exchanger) has not been established so the need for active safety systems cannot be ruled out. The MSRE was operated at one time in the U.S., but a concerted R&D program will be required to re-capture the knowledge and to fill in missing gaps before work on an actual design can commence.

The MSRE was a liquid-fueled MSR, which in many respects successfully demonstrated online fuel reprocessing in a thermal reactor spectrum. This also confers particular challenges with regard to nuclear material safeguards.

Mainly because of the unresolved corrosion and tritium challenges, the solid-fueled MSR (FHR) will may be ready for preliminary design and licensing activities in the 10 to 20 year timeframe. At least another decade of R&D would be required for a Liquid-fueled MSR.

4.4 Sodium-Cooled Fast Reactor

SFRs have been successfully tested as materials irradiation facilities in the U.S. (EBR-2, FFTF) and elsewhere. The only major outstanding issue to achieve a design of TRL 5 or higher is that of fuel qualification and licensing. Experimental or demonstration SFRs currently operate in Russia, India, and China. Japan, France, and the United Kingdom also operated such facilities at one time. Except for EBR-2 and the Indian Prototype Fast Breeder Reactor under construction, these reactors use an oxide fuel form.

A demonstration reactor built upon the EBR-II platform is relatively mature. The metallic fuel forms U-Zr and U-Zr-Pu were extensively tested in that reactor, but the U-TRU-Zr fuel testing was not completed, which would limit its ability to demonstrate a closed fuel cycle in the near to mid-term. An important gap to fill prior to licensing is the characterization of the source term from metallic fuel under normal and accident conditions. Adequate fuel fabrication capability may exist at INL to support test reactor operation, but a production-level capability needs to be developed to support demo or commercial deployment.

The General Electric Hitachi PRISM reactor design, using, for the most part, technologies demonstrated in EBR-II, was the subject of an NRC pre-application Safety Evaluation review.[5] PRISM/Mod-B is a much larger core than EBR-II (840 MWt vs. 62 MWt), and thus issues may emerge in scaling to commercial size, particularly with regard to the size of the vessel and support structures. In-Service Inspection, depending upon NRC requirements, may require new detection techniques suitable to the opaque coolant.

The Direct Reactor Auxiliary Cooling System utilized in EBR-II was effective for a reactor of that size. A larger reactor may require a different technology, such as a Reactor Vessel Auxiliary Cooling System, that would require qualification for NRC licensing. Experiments with a scaled Reactor Cavity Cooling System (RVACS) cooling system are underway at ANL. The fuel handling system in EBR-II will require further development for use in PRISM as will the electromagnetic coolant pumps.

Electrometallurgical reprocessing of spent SFR fuel was demonstrated at ANL and has achieved a relatively high, though not yet commercial-scale, maturity.

The 100-MWe small-modular AFR-100 under development at ANL builds upon EBR-II technology (in particular, the sodium-bonded metallic fuel) but will also employ features that will require considerable technical development. The fission-gas vented fuel is being investigated and may have a significant impact upon worker dose in a demonstration reactor, perhaps less so in a test reactor. AFR-100 would use a DRACS decay heat rejection system. The other major advance is a supercritical CO_2 PCS. An electrically heated system has been tested at a laboratory scale, but the coupling to the sodium loop via an intermediate heat exchanger would still require demonstration to support licensing, particularly given some observed reactions between sodium and lead. For these reasons the TRL is considered low.

SFR: Preliminary design and licensing activities for an SFR built upon the EBR-II platform may be initiated within 10 years if final fuel qualification and source term characterization can be conducted in conjunction with demonstration reactor operation. Production-scale fuel fabrication must also be achieved to support a demonstration concept (PRISM/Mod-B). Continued R&D of key AFR-100 technologies are needed before such activities can begin.

4.5 Supercritical Water-Cooled Reactor

Although the Supercritical Water Reactor[3] relies on existing water reactor technology, especially for the fuel and geometry, there are technical challenges to be overcome.[3] The high pressure requires thick-walled pipes and components. The higher temperature leads to greater corrosion and degradation of structural materials than observed in LWR systems. Indeed, supercritical fluid systems are considered promising for leaching metals from metal ores with a minimal amount of waste. The same attributes also lead to extensive corrosion in certain alloys, particularly zircaloy, a problem that is compounded by radiolysis. SCWR technology can rely on the considerable data available on supercritical water behavior in fossil fuel plants but little of that data is applicable to water reactor fuel assemblies. Experiments with test loops are underway among GIF SCWR members, but much more data is needed to support a licensing effort. The major area of research in SWCR systems is materials characterization and qualification.

The heat transfer and other thermal-hydraulic properties of water near its critical point are not well characterized and thus extensive testing in tube bundles is necessary to provide data for safety analysis codes. Non-uniformities in local power and coolant mass flow rates give rise to hotspots exacerbated by the larger enthalpy rise in the core. Large density variations in water near the critical point could lead to significant local fluctuations in neutron flux and cladding temperature.

As no SCWR has been designed as a test reactor, the major attributes of a SCWR demonstration plant were assumed for evaluating the maturity of a hypothetical SCWR irradiation facility.

The need to qualify new cladding and structural material and to characterize and control neutronic and thermal fluid behavior in SCWR fuel assemblies prevents near-term deployment of this reactor concept as either a test or demonstration reactor. If adequately supported, the SCWR may be ready for preliminary design work in the 10 to 20-year timeframe.

4.6 Very High Temperature Reactor

The HTR, with outlet temperatures limited to less than 800°C, is suitable for near-term deployment as a demonstration reactor. Graphite-moderated, helium-cooled HTRs have been the subject of U.S. government and industry investment on and off since the 1960s. The unique coated particle fuel embedded in a graphite matrix was found early on to provide superior (melt-down proof) fuel form even as the coolant and core are driven to temperatures significantly higher than other reactor types. Demonstration plants put electricity on the grid (Peach Bottom -1 and Fort St. Vrain) despite experiencing engineering difficulties not uncommon to new technologies. In Fort St. Vrain, many of the basic technologies of HTRs were demonstrated.

As with the PRISM reactor, the U.S. government collaborated with industry to develop a small, modular version of the HTR in the 1980s. The GA MHTGR was subjected to pre-application safety review[10] by the NRC. (Fuel development by the federal government continued under the New Production Reactor Program, which had as its mission the production of tritium for the weapons program.) An important difference between the MHTGR and the Fort St. Vrain plant was the lower core power density enabling the ability to reject decay heat passively to the environment even in the most severe loss of coolant accident. This is achieved by limiting the core power (about 600 MWt) and building a tall core with a relatively small diameter, thus providing a short conduction path from the core to the vessel. All HTRs designed since then have adopted this inherent safety feature. The detailed technical status of the MHTGR and its pebble bed counterpart developed in Germany, the HTR-Modul, are described in Gouger's 2014 report.[16] The HTR-Modul design was submitted to the German regulator in the late 1980s, but none was ever built. A two-unit pebble-bed modular HTR power plant based upon the German design is under construction in China.

After Fort St. Vrain, TRISO fuel made in the U.S. exhibited a failure fraction and release rates unacceptable by today's regulatory standards. DOE has funded a significant effort to improve the fuel fabrication process and improve the retentive capabilities of the TRISO particle. Laboratory- and engineering-scale irradiation and heat-up testing under the AGR program have demonstrated fuel performance that exceeded designer requirements. Radiological source terms for the VHTR under operating and accident conditions have been characterized and bounded based upon data from the AGR tests completed thus far. Furthermore, a production-scale fabrication capability with a fuel vendor (Babcock and Wilcox) was developed as part of the AGR program. Similarly, the new grades of graphite being qualified under the Advanced Graphite Creep program will support a design certification. The AREVA SC-HTGR and any test reactor built upon the HTR platform would need to use this fuel to support a near-term deployment schedule.

Metallic components exposed to core conditions may be subjected to failure 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 and seals, however, may fail in the event of the most severe loss-of-forced-cooling events, with subsequent depressurization of the core. While this is not expected to cause significant fuel failure, circulating radiological inventory would be released and expensive core repairs would be necessary. Qualification of new alloys or even the use of carbon or SiC composites for the guide tubes may be needed. The control elements in a smaller reactor (<100 MWt) are not anticipated to reach failure temperatures, but this has yet to be confirmed. Additionally, the need for a containment vessel (as opposed to a confinement vessel) is an unresolved issue with the regulator. For these reason, the reactor enclosure subsystem for the demonstration plant was assigned a TRL of 5.

Under contract form the NGNP program, AREVA conducted a technical readiness evaluation of their original NGNP design (Antares). It has the same core as the SC-HTGR, but it would drive a Brayton cycle PCS. The TRLs listed for the SC-HTGR were influenced by, or taken from, the AREVA report.[17]

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. The technical readiness of the VHTR is comparable to the SCWR in this regard.

VHTR: HTR fuel and materials are mature enough to support preliminary design and licensing of either a test or demonstration reactor within the next five to ten years provided that coolant outlet temperatures are limited to about 800°C. In the near term, a demo reactor would drive a steam cycle PCS, however, as helium gas turbine components are at a relatively low technical readiness and so cannot form the basis of a demonstration reactor in the short term.

5. SUMMARY AND CONCLUSIONS

The technical readiness of six advanced reactor concepts was evaluated and quantified using TRLs as defined by the DOE. A primary objective of this study was to provide a self-consistent evaluation of the maturity of different designs. A common set of systems and subsystems were evaluated and discussed by members of a WG possessing complementary knowledge of the various systems. Values were adopted, derived, or deduced from a number of design descriptions and technical summaries.

The TRL for a plant system was derived from all of the subsystem TRLs of which it is comprised. Certain key subsystems (e.g., fuel/cladding and coolant chemistry) were deemed critical to the technical viability of the plant. The TRLs of these selected subsystems were used to derive the overall technical readiness of the reference design.

A steam cycle PCS was assumed for each demonstration reactor concept unless a different PCS is explicitly being developed for it. Fuel cycle technologies were included in fast spectrum system evaluations.

General (technology neutral) Safety Design Criteria have been being developed by the DOE in cooperation with the NRC and are under review. Complementary, concept-specific design criteria have been developed for the SFR and VHTR and are also under review. Licensing experience for all systems are all low as none of the advanced reactor concept considered have been licensed by the NRC. For the SFR (PRISM) and the VHTR (MHTGR; very similar to the SC-HTGR), pre-application safety reviews were completed by the NRC.

The sodium-cooled fast reactor (built upon the EBR-II platform) and the VHTR with outlet temperature under 800°C) are considered mature enough to support preliminary design and licensing as either test or demonstration reactors within a decade, although the licensing infrastructure for all advanced reactors is still a work in progress. Advanced fuel or power conversion technologies for these systems would significantly delay deployment.

Fuel cycle TRLs were assigned only to fast spectrum demonstration reactor concepts (and the liquid-fueled MSR). For the SFR with metal fuel, the electrorefining process developed at ANL is assumed. For the other fast concepts, a commercial aqueous process (PUREX or a UREX-variant) is assumed.

R&D on the SCWR could lead to preliminary design and licensing activity in the ten-to-twenty year timeframe if certain control and water-based corrosion issues could be resolved. Similarly, based on U.S. experience of MSR technology (solid and liquid fueled), design and licensing of an MSR could begin in this timeframe following a concerted effort particularly in relation to the material corrosion and tritium issues.

The GFR and LFR, have less of an experience base in the U.S. and require long-term (>20 years development and qualification efforts to overcome acknowledged and considerable technical barriers, mostly involving material corrosion by non-traditional coolants and, in the case of the GFR, a significant decay heat removal challenge.

6. REFERENCES

1 Technology Readiness Assessment Guide, DOE G 413.3-4A, U.S. Department of Energy, September 2011.

2 DOE, *Advanced Reactor Concepts Technical Review Panel Report—Evaluation and Recommendations for Future R&D on Eight Advanced Reactor Concepts*, Department of Energy, November 2012.

3 OECD Nuclear Energy Agency, "Technology Roadmap Update for Generation IV Nuclear Energy Systems," January 2014.

4 Holcomb, D. E., et al., "Small, Modular Advanced High-Temperature Reactor—Carbonate Thermochemical Cycle," ORNL/M-2014/88, Oak Ridge National Laboratory, March 2014.

5 Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor, NUREG-1368, Nuclear Regulatory Commission, 1994.

6 Grandy, C. et al, Advanced Fast Reactor – 100 (AFR-100) Report for the Technical Review Panel, ANL-ARC-288, Argonne National Laboratory, June 2014.

7 Buongiorno, J. et al, *Progress Report for the FY-03 Generation IV R&D Activities for the Development of the SCWR in the U.S.*, INEEL/EXT-03-01210, September 2003.

8 DOE, *A Technology Roadmap for Generation IV Nuclear Energy Systems*, Issued by the U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, 2002.

9 Tracy, G., "Next Generation Nuclear Plant—Assessment of Key Licensing Issues," *Letter to John E. Kelly*, Nuclear Regulatory Commission, July 2014.

10 NUREG-1338, *Draft Preapplication Safety Evaluation Report for the Modular High Temperature Reactor*, Nuclear Regulatory Commission, 1989.

11 Proliferation Resistance and Physical Protection of the Six Generation IV Nuclear Energy Systems, Prepared jointly by The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group and the System Steering Committees of the Generation IV International Forum, GIF/PRPPWG/2011/002, July 15, 2011; https://www.gen-4.org/gif/jcms/c_40414/proliferation-resistance-and-physical-protection-of-the-six-generation-iv-nuclear-energy-systems.

12 OECD Nuclear Energy Agency, *Generation IV International Forum Annual Report 2014*, 2015.

13 IAEA, *Liquid Metal Coolants for Fast Reactors Cooled by Sodium Lead, and Lead-Bismuth Eutectic*, International Atomic Energy Agency Nuclear Energy Series No. NP-T-1.6, 2012.

14 Tarantino, M. et al, "Lead-Cooled Fast Reactor (LFR) Development Gaps," Proceedings of the 2012 Technical Meeting to Identify Innovative Fast Neutron Systems Development Gaps, International Atomic Energy Agency, February 2012.

15 MacPherson, H. G., "Development of Materials and Systems for Molten Salt-Reactor Concept," *Reactor Technology*, Vol. 15, No. 2, 1972, pp. 136–155.

16 Gougar, H., "Baseline Concept Description of a Small Modular High Temperature Reactor," INL/EXT-14-31541 Rev. 1, Idaho National Laboratory, May 2014.

17 AREVA, *NGNP Technology Readiness Levels for Conventional Steam Cycle Configuration*, TDR-3001463-000, March 2009.

Appendix A

Charter for Technology Assessment Working Group

Purpose:

This working group (WG) will assess the technology readiness of advanced reactor technology options. The goal is to provide specific recommendations on which technology options have sufficient maturity for either test or demonstration reactor missions.

Approach:

The starting point will be the six Generation-IV reactor types: gas-cooled fast reactor (GFR), lead-cooled fast reactor (LFR), molten salt reactor (MSR), supercritical water-cooled reactor (SCWR), sodium-cooled fast reactor (SFR), and very high temperature reactor (VHTR). The technology readiness of each system will be assessed. This process will rely on previous detailed evaluations (e.g., Generation-IV Roadmap and other references noted below) of technology status for each option. The WG will establish the scope of this Assessment (which systems and features need to be considered), clarify the specific technology readiness scale definition, and assure a consistent evaluation between the options. The WG should examine the readiness of the major system and subsystems for each concept to establish an overall readiness assessment. As needed, the group will consult with reactor concept proponents and technology subject matter experts to inform and update the technology assessment.

The outcome of the study will be focused on three key questions for each concept:

1. Where is the current technology readiness? What are the key technology hurdles that must be overcome for deployment? Can these hurdles be addressed in a test or demonstration reactor?
2. Is the technology mature enough (TRL at least 6) to be used as the base technology for a test reactor (with a primary mission of irradiation services)
3. Is the technology mature enough (TRL 4–8) to be considered for a demonstration reactor? If so, what technology features would be demonstrated? If not, what feasibility issues need to be resolved before demonstration?

Members:

Hans Gougar (INL), chair

Tom Sowinski (DOE)

Taek Kyum Kim (ANL)

Andrew Worrall (ORNL)

Robert Bari (BNL)

Schedule:

To assure timely input to the overall advanced test/demonstration reactor planning study, the group should complete its work in a 60 day period (by July 31, 2016).

The product will be a draft report provided to the Planning Study leadership team, including the specific assessment and recommendations for each Generation-IV option.

Appendix B

Technology Readiness Levels used in this Assessment

Department of Energy (DOE) Guide 413.3-4A was developed to assist individuals and teams involved in conducting Technology Readiness Assessments and developing Technology Maturation Plans for the DOE capital acquisition assets subject to DOE O 413.3B, “Program and Project Management for the Acquisition of Capital Assets.”

The complete guide can be obtained from DOE and, as of the date of issue of this document, downloaded from the DOE website at https://www.directives.doe.gov/directives-documents/400-series/0413.3-EGuide-04a.

The following table was extracted from the guide.

Relative Level of Technology Development	TRL	TRL Definition	Description
System Operations	TRL 9	Actual system operated over the full range of expected mission conditions.	The technology is in its final form and operated under the full range of operating mission conditions. Examples include using the actual system with the full range of wastes in hot operations.
System Commissioning	TRL 8	Actual system completed and qualified through test and demonstration.	The technology has been proven to work in its final form and under expected conditions. In almost all cases, this Technology Readiness Level (TRL) represents the end of true system development. Examples include developmental testing and evaluation of the system with actual waste in hot commissioning. Supporting information includes operational procedures that are virtually complete. An Operational Readiness Review has been successfully completed prior to the start of hot testing.
	TRL 7	Full-scale, similar (prototypical) system demonstrated in relevant environment	This represents a major step up from TRL 6, requiring demonstration of an actual system prototype in a relevant environment. Examples include testing a full-scale prototype in the field with a range of simulants in cold commissioning. Supporting information includes results from the full-scale testing and analysis of the differences between the test environment, and analysis of what the experimental results mean for the eventual operating system/environment. Final design is virtually complete.

Relative Level of Technology Development	TRL	TRL Definition	Description
Technology Demonstration	TRL 6	Engineering/ pilot-scale, similar (prototypical), system validation in relevant environment	Engineering-scale models or prototypes are tested in a relevant environment. This represents a major step up in a technology's demonstrated readiness. Examples include testing an engineering scale prototypical system with a range of simulants. Supporting information includes results from the engineering scale testing and analysis of the differences between the engineering scale, prototypical system/environment, and analysis of what the experimental results mean for the eventual operating system/environment. TRL 6 begins true engineering development of the technology as an operational system. The major difference between TRL 5 and 6 is the step up from laboratory scale to engineering scale and the determination of scaling factors that will enable design of the operating system. The prototype should be capable of performing all the functions that will be required of the operational system. The operating environment for the testing should closely represent the actual operating environment.
Technology Development	TRL 5	Laboratory scale, similar system validation in relevant environment	The basic technological components are integrated so that the system configuration is similar to (matches) the final application in almost all respects. Examples include testing a high-fidelity, laboratory-scale system in a simulated environment with a range of simulants[a] and actual waste[b]. Supporting information includes results from the laboratory scale testing, analysis of the differences between the laboratory and eventual operating system/environment, and analysis of what the experimental results mean for the eventual operating system/environment. The major difference between TRL 4 and 5 is the increase in the fidelity of the system and environment to the actual application. The system tested is almost prototypical.
	TRL 4	Component and/or system validation in laboratory environment	The basic technological components are integrated to establish that the pieces will work together. This is relatively "low fidelity" compared with the eventual system. Examples include integration of ad hoc hardware in a laboratory and testing with a range of simulants and small scale tests on actual waste. Supporting information includes the results of the integrated experiments and estimates of how the experimental components and experimental test results differ from the expected system performance goals. TRL 4–6 represent the bridge from scientific research to engineering. TRL 4 is the first step in determining whether the individual components will work together as a system. The laboratory system will probably be a mix of on hand equipment and a few special purpose components that may require special handling, calibration, or alignment to get them to function.

Relative Level of Technology Development	TRL	TRL Definition	Description
Research to Prove Feasibility	TRL 3	Analytical and experimental critical function and/or characteristic proof of concept	Active research and development is initiated. This includes analytical studies and laboratory-scale studies to physically validate the analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative tested with simulants. Supporting information includes results of laboratory tests performed to measure parameters of interest and comparison to analytical predictions for critical subsystems. At TRL 3, the work has moved beyond the paper phase to experimental work that verifies that the concept works as expected on simulants. Components of the technology are validated, but there is no attempt to integrate the components into a complete system. Modeling and simulation may be used to complement physical experiments.
	TRL 2	Technology concept and/or application formulated	Once basic principles are observed, practical applications can be invented. Applications are speculative, and there may be no proof or detailed analysis to support the assumptions. Examples are still limited to analytic studies. Supporting information includes publications or other references that outline the application being considered and that provide analysis to support the concept. The step up from TRL 1 to TRL 2 moves the ideas from pure to applied research. Most of the work is analytical or paper studies with the emphasis on understanding the science better. Experimental work is designed to corroborate the basic scientific observations made during TRL 1 work.
Basic Technology Research	TRL 1	Basic principles observed and reported	This is the lowest level of technology readiness. Scientific research begins to be translated into applied research and development. Examples might include paper studies of a technology's basic properties or experimental work that consists mainly of observations of the physical world. Supporting information includes published research or other references that identify the principles that underlie the technology.

a. Simulants should match relevant chemical and physical properties.
b. Testing with as wide a range of actual waste as practicable and consistent with waste availability, safety, as low as reasonably achievable, cost, and project risk is highly desirable