RESEARCH NEEDS FOR GENERATION IV NUCLEAR ENERGY SYSTEMS

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ABSTRACT

A review of the research and development (R&D) needs for Generation IV nuclear energy systems is presented. The review is introduced by explaining the roadmap that was developed to first determine the technology goals, and then to select the nuclear energy systems that would qualify for future consideration as Generation IV concepts. The R&D plan, also developed as part of the roadmap, is discussed with the emphasis on those issues that might be of most interest to the PHYSOR 2002 audience. These include: reactor physics issues, including neutronic design; and thermal hydraulics, fuels, and reactor safety, as these latter topics frequently are closely coupled to reactor physics.

1. INTRODUCTION

1.1 BACKGROUND

The development of new reactor concepts for use beyond 2030 is being pursued enthusiastically by the nuclear community around the world. In the U.S. this effort has been spearheaded by the Department of Energy (DOE) and its Generation IV (Gen IV) Roadmap activities. The roadmap, begun in January 2000, is a planning tool being developed with participants from the U.S. and nine other countries. It begins with technology goals and then considers a wide variety of nuclear energy system concepts. Several are selected as being the most promising and the research and development (R&D) requirements for those concepts are then developed. The documentation of the roadmap is to be completed by the end of 2002. The future development of new nuclear energy concepts in the U.S., as well as in other countries, is expected to be guided by the roadmap.

1.2 TECHNOLOGY GOALS

The technology goals for Gen IV concepts are intended to stretch the envelope of current technologies. They are in four areas: sustainability, safety and reliability, economics, and proliferation resistance and physical protection. The eight goals in these areas are listed in Table 1 and explained below.

Sustainability is the ability to meet the needs of present generations while enhancing and not jeopardizing the ability of future generations to meet society's needs indefinitely into the future. There is a growing desire in society for the production of energy in accordance with sustainability principles. Sustainability requires the conservation of resources, protection of the environment, preservation of the ability of future generations to meet their own needs, and the avoidance of placing unjustified burdens upon them. Existing and future nuclear power plants meet current and increasingly stringent clean air objectives, since their energy is produced without combustion processes. The two sustainability goals encompass the interrelated needs of improved waste management, minimal environmental impacts, effective fuel utilization, and the development of new energy products that can expand nuclear energy's benefits beyond electrical generation.

Safety and reliability are essential priorities in the development and operation of nuclear energy systems. During normal operation or anticipated transients, nuclear energy systems must preserve their safety margins, mitigate accidents, and keep accidents from deteriorating into severe accidents. At the same time, competitiveness requires a very high level of reliability and performance. There has been a definite trend over the years to improve the safety and reliability of nuclear power plants, reduce the frequency and degree of off-site radioactive releases, and reduce the possibility of significant core damage. Gen IV systems have goals to achieve the highest levels of safety and reliability and to better protect workers, public health, and the environment through further improvements. The three safety and reliability goals continue the past trend and seek simplified designs that are safe and further reduce the potential for severe accidents.

Economic competitiveness is a requirement of the marketplace and is essential for Gen IV nuclear energy systems. In today's environment, nuclear power plants are primarily baseload units that were purchased and operated by regulated public and private utilities. A transition is taking place worldwide from regulated to deregulated energy markets, which will increase the number of independent power producers and merchant power plant owner/operators. Future nuclear energy systems should accommodate a range of plant ownership options and anticipate a wider array of potential roles and options for deploying nuclear power plants, including load following and smaller units. While it is anticipated that Gen IV nuclear energy systems will primarily produce electricity, they will also help meet anticipated future needs for a broader range of energy products beyond electricity. For example, hydrogen, process heat, district heating, and potable water will likely be needed to keep up with increasing worldwide demands and long-term changes in energy use. Gen IV systems have goals to ensure that they are economically attractive while meeting changing energy needs.

Proliferation resistance and physical protection are also essential priorities in the expanding role of nuclear energy systems. Nuclear energy systems for electrical generation have always been deployed with means to prevent nuclear weapons proliferation. This goal applies to all inventories of fissile materials in the system involved in mining, enrichment, conversion, fabrication, power production, recycling, and waste disposal. In addition, existing nuclear plants are highly secure and designed to withstand external events such as earthquakes, floods, tornadoes and fires. Their many protective features considerably reduce the impact of external or internal threats through the redundancy, diversity, and independence of safety systems. This goal points out the need to increase public confidence in the security of nuclear energy facilities in light of the September 11th terrorist attacks. Advanced systems need to be designed from the start with improved physical protection against acts of terrorism, to a level commensurate with the protection of other critical systems and infrastructure.

1.3 NUCLEAR ENERGY SYSTEM CONCEPTS

The reactor concepts considered were categorized according to whether they were cooled by water, liquid metal, or gas, or whether they were considered "non-classical." In each of these four categories a technical working group (TWG) considered a variety of concepts, each at different levels of maturity. This was made manageable by grouping individual concepts into concept sets. For example, the water-cooled TWG considered 38 different concepts and lumped them into 10 concept sets. Next, using 24 metrics developed to implement the eight technology goals, each TWG recommended a subset to advance to the next stage of consideration. For the water-cooled group, five concepts sets were advanced. Lastly, a committee, with representatives from the 10 countries participating in the roadmap, considered the attributes of all the concept sets that had been advanced by all TWGs and selected six that would be considered as "Gen IV." These are:

- Supercritical Water-Cooled Reactor System (SCWR)
- Very High Temperature Reactor System (VHTR)
- Gas-Cooled Fast Reactor System (GFR)
- Sodium Liquid Metal-Cooled Reactor (Na LMR)
- Lead Alloy-Cooled Reactor System (Pb Alloy)
- Molten Salt Reactor System (MSR)

In addition to these concepts for Gen IV, the roadmap activity has led to the recommendation of near-term concepts for further consideration. These are reactors that could be deployed as early as 2010. These reactors are not part of the longer-term Gen IV program and will not be discussed herein.

2. RESEARCH AND DEVELOPMENT NEEDS OF INTEREST TO THE PHYSOR 2002 COMMUNITY

2.1 CROSS CUTTING ACTIVITIES

Specific R&D needs for the six Gen IV nuclear energy systems were developed by the TWGs. The portions of these needs that are of most interest to PHYSOR 2002 are discussed below. These are reactor physics issues, including nuclear design, thermal-hydraulics and fuels (because they are so closely coupled to neutronics), and reactor safety as it relates to these disciplines. These subjects are a small (albeit important) portion of the overall Gen IV research plans which include consideration of materials for fuels and coolant systems, energy conversion systems, operation and maintenance, construction, etc. In addition to the R&D plan development by the TWGs, there are crosscut groups that will also recommend R&D that is relevant. The two groups whose plans might overlap with the needs discussed below are the Risk and Safety Crosscut Group and the Fuel Cycle Crosscut Group. For example, it is the responsibility of the former group to consider static and transient analysis, design basis analysis, and instrumentation and control, among other areas. The relevant R&D from these crosscut groups has not been reviewed for this paper.

One cross cutting area that does not fall into any specific crosscut group is nuclear data. These data are needed for reactor physics and neutronic design, and safety analysis as well as for fuel cycle issues and materials research (to assess radiation damage). In general, the needs of reactor designers are not expressed in terms of nuclear data but in terms of required accuracy of design parameters obtained through reactor physics calculations. The use of nuclear data with large or unknown uncertainties will result in the need for increased design margins that can negatively impact performance and economic competitiveness of a particular reactor design. Although a careful assessment of data needs for Gen IV reactors requires knowledge of specific designs, some generalities apply.

For Gen IV applications, deficiencies in the current U.S. library, ENDF/B-VI, must be addressed. Comprehensive data covariance files (which are often not available for many materials) must be created, as they are particularly important for obtaining reasonable estimates of uncertainties. Data needs are expected to come from the use of innovative fuels, coolants, moderators, and absorbers. For example, neutron capture, elastic and inelastic cross section data with improved accuracy are expected to be needed for elements used as matrix/diluents of composite fuels such as Zr, Mg, Y, Ti, and W, and for new coolants such as Pb and Pb-Bi. Extensive benchmark testing is required to assure the overall quality of the ENDF library.

2.2 SUPERCRITICAL WATER REACTOR SYSTEM (SCWR) R&D SCOPE[a]

2.2.1 SCWR Description

SCWRs are a class of high temperature, high pressure water-cooled reactors that operate above the thermodynamic critical point of water (374°C, 22.1 MPa). They hold the potential for significant advantages compared to current generation light water reactors (LWRs). A major advantage is the significant increase in thermal efficiency. Estimated efficiencies for SCWRs are 45% compared to 33-35% for LWRs.

A lower coolant mass flow rate per unit core thermal power results from the higher enthalpy content of the coolant. This leads to a reduction in the size of the reactor coolant pumps, piping, and associated equipment, and a reduction in the pumping power. In addition, the elimination of steam dryers, steam separators, re-circulation pumps, as well as steam generators means the SCWR will be a simpler plant with fewer major components.

A lower coolant mass inventory results from the once-through coolant path in the reactor vessel and the lower coolant density. This opens the possibility of smaller containment buildings. Another advantage is the elimination of boiling crisis (i.e., departure from nucleate boiling or dry out) due to the lack of a second phase, thereby avoiding discontinuous heat transfer regimes within the core.

The Japanese supercritical light water reactor (SCLWR) with a thermal spectrum has been the subject of the most development work in the last 10 to 15 years and is a reference concept. The SCLWR reactor vessel is somewhat similar in design to an ABWR. High-pressure (25.0 MPa) coolant enters the vessel at 280°C. The inlet flow splits, partly to a downcomer and partly to a plenum at the top of the core to flow downward through the core in special water rods. This strategy is employed to provide moderation in the core. The coolant is heated to 508°C and delivered to a power conversion cycle that blends LWR and supercritical fossil plant technology: high-, intermediate-, and low-pressure turbines are employed with two reheat cycles.

The SCWR can also be designed to operate as a fast reactor with full actinide recycle. The difference between thermal and fast versions is primarily the amount of moderator material in the SCWR core. The fast spectrum reactors use no additional moderator material, while the thermal spectrum reactors need additional moderator material in the core.

2.2.2 Technology Gaps for the SCWR

The important supercritical water technology gaps are in the areas of SCWR materials and structures, as well as safety. The SCWR safety research program is expected to be organized around the following topics:

- Reduced uncertainty in SCW transport properties.
- Further development of appropriate fuel cladding to coolant heat transfer correlations under a range of fuel rod geometries.
- Critical flow measurements, as well as models and correlations.
- Measurement of integral loss-of-coolant accident (LOCA) thermal-hydraulic phenomena and related computer code validation.
- Fuel rod cladding failure mechanisms including ballooning during LOCAs.

[a] Based on work done by the members of TWG-1, Water-Cooled Concepts

- Design optimization studies including investigations to establish the reliability of passive safety systems.
- Power-flow stability assessments
- Instrumentation and control issues especially during transitional operating modes (e.g., startup and shutdown)

The purpose of making additional basic thermal-hydraulic property measurements at and near the pseudo-critical temperatures would be to improve the accuracy of the international steam-water property tables.

The fuel cladding to SCW heat transfer research would consist of a variety of out-of-pile experiments starting with tubes and progressing to small and then relative large bundles of fuel rods. The bundle tests would include some variations in geometry (fuel rod diameter and pitch, bundle length, channel boxes, etc.), axial power profiles, coolant velocity, pressure, grid spacer design, etc. The larger bundle tests will require megawatts of power and the ability to design electrically heated test rods with appropriate power shapes that can withstand the superheated steam conditions.

The integral SCWR LOCA thermal-hydraulic experiments would be similar to the Semiscale experiments previously conducted in the U.S. to investigate LOCA phenomena for current LWRs. A full-height system and a large enough core diameter for appropriate scaling would be needed. The scaling laws would have to be validated to be assured they could be applied to SCWR conditions.

SCWR design optimization is also important. All of the known accident scenarios must be carefully evaluated (large and small break LOCAs, reactivity initiated accidents, loss of flow, main steam isolation valve closure, over-cooling events, anticipated transients without scram, high and low pressure boil off, etc.) to assure compliance with reactor protective criteria. There may be safety features (e.g., very high pressure accumulators) that require special designs. Core flooding with subcooled water may prove to be highly demanding on safety system design.

The objective of the power-flow stability R&D is a better understanding of instability phenomena in SCWRs, the identification of the important variables affecting these phenomena, and ultimately the generation of maps identifying the stable operating conditions of the different SCWRs designs. Consistent with the U.S. Nuclear Regulatory Commission (NRC) approach to BWR licensing, the licensing of SCWRs will probably require, at a minimum, demonstration of the ability to predict the onset of instabilities. This can be done by means of a frequency-domain linear analysis. Prediction of the actual magnitude of the unstable oscillations beyond onset, although scientifically interesting and relevant to beyond-design-basis accidents, will likely not be required for licensing.

Both analytical and experimental studies need to be carried out for the conditions expected during the different operational modes and accidents. The analytical studies can obviously be more extensive and cover both works in the frequency domain as well as direct simulations. These studies can consider the effect of important variables such as axial and radial power profile, moderator density and fuel temperature reactivity feedback, fuel rod thermal characteristics, coolant channel hydraulic characteristics, heat transfer phenomena, core boundary conditions (including the effect of direct or indirect cycles), etc. Mitigating effects like orificing, insertion of control rods, and fuel modifications to obtain appropriate thermal and/or neutronic response time constants can also be assessed using analytical simulations. Computer models that are used in this analysis must be validated using results from experiments. Specifically, the heat-transfer, pressure-drop, and critical-flow correlations and models used in LWR codes must be upgraded or replaced with models developed for prototypical SCWR conditions.

It is envisioned that stability experiments will be conducted at a multi-purpose SCW thermal-hydraulic facility planned for the safety experimentation discussed above. The test section will be designed to accommodate a single bundle as well as multiple bundles. This will enable studying in-phase and out-of-phase density-wave oscillations. Moreover, the facility will provide a natural circulation flow path for the coolant to study buoyancy loop instabilities.

The analytical work would include making sure that the codes are validated, doing the appropriate analyses, and predicting experiments for different reactor designs. This program would be closely linked to other research activities validating the models used in neutronic and thermal-hydraulic computer codes and applying the codes to other issues such as transient/accident analysis. Note too that this validation would require a database of reactor physics measurements with prototypical core layouts.

2.3 VERY HIGH TEMPERATURE REACTOR SYSTEM (VHTR) R&D SCOPE[b]

2.3.1 VHTR Description

Very High Temperature Reactors are those concepts that have average coolant outlet temperatures above 900°C or operational fuel temperatures above 1250°C. These concepts provide the potential for increased energy conversion efficiency and for high-temperature process heat applications, such as coal gasification or thermochemical hydrogen production. While all the gas-cooled reactor concepts considered have sufficiently high temperature to support process heat applications such as desalination or cogenerative processes, as well as some thermochemical processes of interest to alternative fuel production, the VHTRs higher temperatures open a broader and more efficient range.

The reference concept has a block type core and is based on the Gas Turbine – Modular Helium Reactor (GT-MHR) connected to a steam reformer/steam generator unit in the primary circuit. It can also be equipped with an intermediate heat exchanger (IHX), as is the case in the High Temperature Engineering Test Reactor (HTTR), to broaden the application spectrum. Pebble bed concepts are also applicable options. The VHTRs excel in achieving safety goals for Gen IV, may be excellent in economics for their hydrogen mission, and offer comparable sustainability to current reactors.

2.3.2 VHTR Technology Gaps

R&D issues for the VHTR are complementary to those for pebble bed and prismatic gas-cooled reactors by extension of temperatures, efficiencies and applications beyond dedicated electricity generation and should be performed in parallel for making best use of synergies between these R&D activities. Much of the research will center on high temperature materials and the power conversion or energy product end of the system.

The present HTR fuel has been developed about 20 years ago for a large HTR. The fuel was only optimized for low fission product release under operational conditions. The TRISO particle was a result from former direct cycle plants to allow for easy maintenance of the gas turbine within the primary circuit. Later on, it was discovered that this type of fuel is capable of fission product retention up to more than 1600°C where chemical degradation of the SiC layer starts. This feature gave rise to the invention of modular HTR (MHR) with inherent limitation of maximum accident temperatures below this limit. On the other hand, the TRISO particle was not yet optimized for application in MHR and for keeping these features under higher burn-up.

The main targets and motivations for VHTR fuel development are:

[b] Based on work done by members of TWG-2, Gas-Cooled Reactor Systems.

- Rise of operational coolant temperatures from 850 up to 950-1000°C
- Rise of tolerable accident temperature limits beyond 1600°C
- Increase in maximum burn-up from 80 GWd/t up to 150-200 GWd/t
- Higher power density above $6MW/m^3$

The pebble-bed reactor is considered as one of the most promising inherently safe reactors with reasonable cost. Its power distribution remains constant during burnup in the equilibrium state, and the excess reactivity remains constant. Therefore, it does not require the control mechanism for the power and coolant-flow distribution and the reactivity during burnup. However, it requires an on-power fuel charge and discharge mechanism. The use of movable fuel makes the analysis of these features more difficult than for a stationary fuel. The block type graphite moderated high temperature gas cooled reactor is stationary and does not require the complex modeling needed for movable fuel. Nevertheless, its power shape and excess reactivity change with burnup still need to be tracked. In the high temperature gas-cooled reactor of prismatic type, development of burnable poison coated particles or using minor actinides instead for better neutron economy and reactivity control is needed.

Symbiotic fuel cycles of LWR and HTR can reach much better waste minimization performance because of the specific features of HTR core physics and the epithermal neutron spectrum. One of the benefits of the HTR is to accommodate a wide variety of mixtures of fissile and fertile materials without any significant modification of the core design as demonstrated in the AVR test reactor in Germany. This flexibility is due to a decoupling between the parameters of cooling geometry, and those that characterize neutronic optimization (i.e., moderation ratio or heavy nuclide concentration, self-shielding effects, etc.). In fact, it is possible to modify the packing fraction of coated particles in the fuel within the graphite matrix without changing the dimensions of the fuel elements. Other physical reasons favor the much better adaptability of HTRs with regard to the fuel cycle in comparison with reactors using moderators in the liquid form, such as LWRs or LMRs. An illustration of this is the fact that in an LWR the void coefficient limits the plutonium content of MOX fuels; something that is not a constraint for the HTR.

It is to be noted also that an HTR core has a better neutron economy than an LWR because there is much less parasitic capture in the moderator (capture cross section of graphite is 100 times less than that of water). The much larger thermalisation length leads to an epithermal neutron spectrum with considerable fast flux contributions. Finally, HTR fuels are able to reach very high burn-ups, which are far beyond the possibilities offered by other thermal reactors. All these capabilities permit essentially complete plutonium fission in a single burnup and minimizes the proliferation risk.

2.4 GAS-COOLED FAST REACTOR SYSTEM (GFR) R&D SCOPE[c]

2.4.1 GFR Description

GFR concepts offer a closed fuel cycle through high conversion or breeding of fissile materials. A breeding capability around unity may be of interest if the GFR is used in a synergistic fuel cycle with LWRs. GFRs using a direct Brayton cycle have the potential to combine the advantages of high sustainability and economic competitiveness, while making nuclear energy benefit from the most efficient conversion technology available.

A reference concept is a 600 MWth/288 MWe, helium cooled reactor system operating with an outlet temperature of about 850EC, and using a direct Brayton cycle gas turbine. The thermal efficiency is

[c] Based on work done by members of TWG-2, Gas-Cooled Reactor Systems.

estimated to approach 48%. There are several fuel design options including both the prismatic (with fuel particles or composite fuels) and fuel pins (with actinide compound/solid solution).

For loss-of-fluid accidents (LOFAs), GFRs are designed with natural convection and heat exchange, with the heat exchanger mounted at the top of the pressure vessel. For LOCAs, GFR designs facilitate long-term passive decay heat removal by conductive and radiative heat transfer across the core and the pressure vessel; pressurized gas injection and natural convection at a back-up pressure of 5 to 15 bar (depending on the gas) assured by the containment of the primary system.

2.4.2 GFR Technology Gaps

The main characteristics of the GFR are high operating temperature, fast neutron spectrum, robust refractory fuel, direct conversion with a gas turbine and integrated on-site fuel cycle. Its development approach is to rely, as much as possible, on technologies already used for the Prismatic Fuel Modular Reactor (PMR) concept, but with significant extrapolations to reach the objectives stated above. The R&D plan includes fuel, fuel cycle considerations including reprocessing, reactor system structure materials, and safety.

Current PMR designs rely on the use of specific coated particle fuels encased in large hexagonal graphite blocks that provide for neutron moderation, heat transfer, and thermal inertia. These technologies are not adapted to the fast neutron spectrum of the GFR. Innovations are required to adapt the concept of fuel particle and the core layout (without large graphite blocks) to plutonium and other transuranic-bearing fuels, to allow operation with a fast neutron spectrum and to resist high levels of fast neutron fluence. Technology breakthroughs are needed to develop innovative fuel forms which preserve the most desirable properties of standard gas reactor fuel particles including withstanding temperatures up to 1600°C with an excellent confinement of the fission products, while accommodating an increased heavy nuclei content and withstanding fast neutron irradiation. Fuel development efforts must be conducted in concert with reactor design efforts so that a fuel that meets core design requirements and a core that operates within fuel limits is developed.

Fuel forms and configurations to be considered and developed in this plan are governed by neutronic and thermal hydraulic factors which are determined by the goal of high sustainability for the GFR concept, and by other Gen IV criteria such as economics, safety, and non-proliferation. In order to achieve fast neutron spectra needed for high sustainability, a high heavy atom fuel density is needed; low absorption-low moderation material should be used in the core, and the fuel burnup should be high. There should be an ability to incorporate the minor actinides into the fuel.

The fast-hardened spectrum design of the GFR offers an opportunity for enhanced reactivity feedback through core expansion, which, together with a refractory fuel, would offer promising prospects of surviving anticipated transients without scram without severe core damage. Nevertheless, specific R&D should be devoted to innovative and possibly passive shutdown systems.

2.5 SODIUM LIQUID METAL-COOLED REACTOR SYSTEM (Na LMR) R&D SCOPE[d]

2.5.1 Na LMR Description

The Na LMR can have fuel that is either metal-alloy or oxide. The former is expected to be applicable for reactors of intermediate size (150-500 Mwe) whereas the latter is expected to be more applicable in a higher power range (500-1500 MWe). The metal fuel is a uranium, plutonium, minor actinide, and zirconium alloy supported by a fuel cycle based on the pyrometallurgical or "pyro" process. This concept targets improvements in economics, passive safety, and fuel cycle performance compared to traditional sodium-cooled reactors. The oxide fuel variant uses uranium and plutonium (MOX) and is supported by a fuel cycle based on advanced aqueous reprocessing. This concept targets improved economics, safety, and component performance in comparison to traditional sodium-cooled reactors.

The target missions of both types of reactors are electricity production and actinide management (waste consumption and, if required, excess fissile creation). Sodium temperature at the exit of these reactors is expected to be ~550ºC resulting in improved thermodynamic efficiency of about 40% for electrical power generation.

2.5.2 Na LMR Technology Gaps

Although there has already been extensive research on sodium-cooled reactors, there is still considerable work that is needed with regard to reactor and plant design, especially with materials, as well as with the fuel cycle. The fuel cycle strategy may be based on "dirty fuel/clean waste" which means that all transuranics are recovered in a commixed stream, and recycled to the reactor for total consumption by fissioning in the fast neutron flux. Although reactor physics impacts with the fuel cycle, most technology gaps here depend on fabrication technology, reprocessing chemistry, and other technologies.

Additional reactor physics work is needed to capture innovations that can lead to significant cost reduction and to help make the safety case for the concept. The positive void reactivity effect is a major safety issue that will require additional innovations in neutronic design and analysis. Another safety issue is the need to complete the database for axial expansion of metal fuel in overpower transients. This involves experimental work and would lead to data that could be used to validate code predictions for design-basis, and beyond design basis, accidents. Other safety issues are the verification of reactivity feedback for radial expansion and assembly bowing, and transient analysis for overpower and undercooling events.

2.6 LEAD ALLOY-COOLED REACTOR SYSTEM (Pb Alloy) R&D SCOPE [d]

2.6.1 Pb Alloy Description

The Pb Alloy system features a closed fuel cycle for efficient conversion of fertile uranium and management of minor actinides. A full actinide recycle fuel cycle with central or regional fuel cycle facilities is envisioned. The system uses a lead or lead/bismuth eutectic liquid metal-cooled fast reactor. The fuel is metal or nitride-based, containing fertile uranium and transuranics. The most advanced of these concepts is the Pb/Bi battery, which employs a small size core with a very long (10-30) year core life. The reactor module is designed to be factory fabricated and then transported to the plant site. The reactor is cooled by natural convection and sized between 120–400 MWt with a reactor outlet coolant temperature of 540°C, possibly ranging up to 750°C, depending upon the materials R&D success. The

[d] Based on work done by members o TWG-3, Liquid Metal Reactors.

system is specifically designed for distributed generation of electricity and other energy products, including hydrogen and potable water.

The Pb Alloy system has a number of options, including the specific choice of coolant and fuel, and a range of sizes that need further examination.

2.6.2 Pb Alloy Technology Gaps

Work on these reactors has been ongoing in other countries but there is much to be done compared with the sodium-cooled concepts. This involves not only reactor and plant design, but also the innovative use of new materials. In the area of reactor physics there is a need to fill in gaps in nuclear data, design passive reactivity control, and provide measurements compatible with core design including those that consider reactivity mechanisms important in accident analysis. In addition, the need to design for a very long core life will be challenging for the Pb/Bi battery design where no fuel shuffling or replacement occurs for the 10-30 year core lifetime.

2.7 MOLTEN SALT REACTOR SYSTEM (MSR) R&D SCOPE[e]

2.7.1 MSR Description

The Liquid Core Reactor Concept Set includes reactors that produce fission power in a circulating molten salt or molten metal fuel. Based on past experience the molten salt reactors (MSRs) are usually fueled with uranium or thorium fluorides dissolved in a mixture of molten lithium and beryllium fluorides, with Na and Zr fluorides as primary alternatives. Because the molten salt reactor operates at near-atmospheric pressures, and because of its stability in ambient environment there is no need for a very high-pressure (>2–10 bar) primary vessel. MSRs feature unique flexibility in fuel cycle design with almost all fuel cycle operations performed at the reactor site. The key features that give MSRs advantages over current LWRs are the following:

- The molten salt core combines the functions of fuel and coolant into one.
- The circulating molten salt from the core directly transports energy to an intermediate heat exchanger.
- Molten fluoride salts are chemically stable at high temperatures and have a very low vapor pressure.
- They have the capability for thermal breeding, or actinide burning, or once through cycle.
- Fission products and actinides are soluble in the molten salt.
- Online feeding, processing and fission product removal is possible.
- Passive cooling features and failsafe drainage can be utilized.
- High temperature operation with high efficiency and the potential for thermochemical hydrogen production is possible.

There are four major fuel cycle options, which can be used with different goals in mind: (1) maximum breeding (breeding ratio ~1.07), (2) denatured Th-^{233}U breeder with minimum inventory of weapons-usable material, (3) denatured “once-through” transuranic (TRU) burner (Pu + minor actinides (MA)) fuel cycle with minimum chemical processing, and (4) actinide (TRU = Pu + MA) burning with multiple recycling . The reactor can use ^{238}U or ^{232}Th, as a fertile fuel. Because of the thermal or epithermal spectrum of the fluoride molten salt reactor, only ^{232}Th can be used as the primary fertile material if the reactor is operated as a breeder reactor. All of the MSRs can be started using low-enriched uranium or other fissile materials. The range of operating temperatures of MSRs is above the melting point of

[e] Based on work done by members of TWG-4, Non-Classical Concepts.

eutectic fluorine salts (~ 500°C) and below the chemical compatibility temperature of nickel-based alloys (~ 800°C).

2.7.2 MSR Technology Gaps

The molten salt breeder reactor (MSBR) program which ended in the 1960s resolved many technical issues but also identified several technical issues that are considered essential to further establish the viability of MSR systems for commercial power generation: molten salt chemistry, solubility of actinides and lanthanides in the fuel, compatibility of irradiated molten salt fuel with structural materials and graphite, and metal clustering in heat exchangers.

Those issues having to do with reactor physics and safety seem minor relative to these. However, for concept development, fuel development and new cross section data and qualification to enable selection of molten salt composition for the target MSR type (burner, converter/transmuter) are still important. Despite the successes of MSBR, recent neutronics calculations performed with the MSBR salt showed some discrepancies with previous evaluations of Doppler and salt dilatation reactivity effects, raising questions about the value of the temperature reactivity coefficient of this salt. These discrepancies are partly because of the use in the new calculations of cross-sections libraries issued in more recent evaluations (JEF2.2 and ENDF/B-VI). All this means that new data measurements and qualification are needed, in order to reduce the uncertainties weighing at the present time on thorium cycle isotopes data (mainly ^{232}Th and ^{233}U), but also on the salt components data (particularly on their scattering cross-sections).

There are also challenges involved in modelling a reactor with moving fuel. There is the need to have appropriate average and bounding conditions as well as a need to be able to factor movement into analysis of slow transients. Keeping track of the moving fuel is also necessary for designing the fuelling and reprocessing that takes place during operation.

The current regulatory structure is designed for solid fuel reactors. Changes in design may be required to meet the intent of current regulations. Work is required with regulators to define equivalence in safety for MSRs that feature vastly different design characteristics than light water reactors. Previous work has provided much of the information required to demonstrate MSR safety, showing that all the safety issues can be resolved with existing technology. Nevertheless, a comprehensive safety analysis equivalent to the one applied to other more classical reactor concepts remains to be done. Consequently, some additional technology development is required to demonstrate thoroughly the safety of these reactor systems.

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Table 1. Technology Goals for Gen IV Nuclear Energy Systems

SUSTAINABILITY	
Fuel utilization	Gen IV nuclear energy systems including fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.
Waste minimization	Gen IV systems will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.
SAFETY AND RELIABILITY	
Excellence	Gen IV nuclear energy systems operations will excel in safety and reliability.
Core Damage	Gen IV systems will have a very low likelihood and degree of reactor core damage.
Emergency response	Gen IV systems will eliminate the need for offsite emergency response.
ECONOMICS	
Life cycle cost	Gen IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.
Risk to capital	Gen IV nuclear energy systems will have a level of financial risk comparable to other energy projects.
PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION	
Nonproliferation/physical protection	Gen IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.