NEW REACTOR CONCEPTS FOR NEW GENERATION OF NUCLEAR POWER PLANTS: AN OVERVIEW

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Invited paper

Abstract – *The outlook for energy demand underscores the need to increase the share of nuclear energy production. Achieving the vision of sustainable growth of nuclear energy will requires development of both advanced nuclear fuel cycles and next generation reactor technologies and advanced reprocessing and fuel treatment technologies. To achieve this vision, the U.S. Department of Energy (DOE) has adopted a new strategy, the Global Nuclear Energy Partnership (GNEP), which integrates earlier programs: the Generation IV Nuclear Energy Systems Initiative (Generation IV), the Nuclear Hydrogen Initiative (NHI), and the Advanced Fuel Cycle Initiative (AFCI) with proliferation-resistant spent fuel reprocessing to minimize nuclear waste. Generation IV furthers this vision beyond previous energy systems, such as Generation III+, through incremental improvements in economic competitiveness, sustainability, development of passively safe systems, and breakthrough methods to reduce the routes of nuclear proliferation. This paper summarizes the main characteristics of the six most promising nuclear energy systems identified by the Generation IV Roadmap and reviews some of Generation IV system designs for small-scale proliferation-resistant reactors being developed by University of California at Berkeley.*

1. INTRODUCTION

The uranium resources would be depleted within a few decades without the deployment of fast breeder reactors in symbiosis with the existing and new reactor types. The current LWRs (identified by the DOE as Generation II reactors) use active means to assure that the consequences of the accident remain within specified acceptable limits. More advanced Generation III reactors, such as EPR or ABWR, still employ active safety systems, while the newest, such as Generation III+ (AP-1000 and ESBWR), relay on passive safety features to the maximum extent possible.

Concerns over energy resource availability, climate change, air quality, energy independence and security suggest an important role for nuclear power in future energy supplies. At the beginning of this year, President Bush announced a new initiative for Global Nuclear Energy Partnership (GNEP) with a goal to ″develop worldwide consensus on enabling expanded use of economical, carbon-free nuclear energy to meet growing electricity demand″ [1]. The initiative will promote expansion of nuclear energy in the U.S. and in the World, through: minimizing nuclear waste through the closed fuel cycles including proliferation resistant recycling of spent fuel and use of advanced fast reactors as actinide burners. In order to provide affordable energy to developing countries, GNEP envisions the nations with secure, advanced nuclear capabilities providing ″nuclear fuel services″ to other countries. These services might include: providing to developing countries small-scale nuclear reactors that are cost-effective, secure, and proliferation resistant, providing fresh fuel and recovery of spent fuel. While the current Generation II and III nuclear power plant designs provide an economically, technically, and publicly acceptable electricity supply in many markets, further advances in nuclear energy system design can broaden the opportunities for the use of nuclear energy. In addition to developing small-scale nuclear reactors, the longer term development envisions the next-generation nuclear energy systems known as "Generation IV" [2], that could become an abundant, reliable, affordable, clean and secure source of both electricity and hydrogen.

2. FINAL GENERATION IV SYSTEM SELECTED

GNEP will develop the next generation of nuclear energy systems with closed fuel cycle capable of providing clean, affordable energy by:

- developing and demonstrating advanced nuclear energy systems that meet future needs for safe, sustainable, environmentally responsible, economical, proliferation-resistant, and physically secure energy,
- developing and demonstrating technologies that enable the transition to a stable, long-term, environmentally, economically, and politically acceptable advanced fuel cycle, including proliferation-resistant recycling.

Generation IV supports this mission through the development of innovative, next-generation reactor technologies. Within Generation IV, the Next Generation Nuclear Plant (NGNP) project includes developing of advanced high-temperature, gas-cooled reactor technology to demonstrate the capability of this technology to power the economic production of hydrogen and electricity. This initiative will develop hydrogen production technologies that are shown to be compatible with nuclear energy systems through scaled demonstrations. A commercial-scale hydrogen demonstration plant could be coupled with a Generation IV demonstration facility by the middle of the next decade.

Achieving the vision of sustainable growth of nuclear energy will also require transition from the current once through fuel cycle to an advanced fuel cycle that recycles nuclear materials. AFCI will develop fuel systems for Generation IV reactors and create enabling fuel cycle technologies (i.e., fuel, cladding, separations, fuel fabrication, waste forms, and disposal technology) to significantly reduce the disposal of long-lived, highly radiotoxic transuranic isotopes while reclaiming spent fuel's valuable energy.

The emphasis on fast reactors reflects their excellent potential to make significant gains in reducing the volume and radiotoxicity, and increasing the manageability of spent nuclear fuel. Fast reactors also hold the potential for extending the useful energy yield of the world's finite uranium supply many-fold for long-term sustainable nuclear energy. The principal issues in the development of a next-generation fast-spectrum reactor are its economic competitiveness and management of the overall risks to workers and the public from the deployment of a closed fuel cycle. The most promising fast-spectrum Generation IV systems are the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), and the Sodium-cooled Fast Reactor (SFR).

The Generation IV Roadmap identified six most promising systems, four of which are mentioned above. The additional two are the Supercritical Water-cooled Reactor (SCWR) and the Molten Salt Reactor (MSR). The SCWR employs water above the critical temperature and pressure that affords a considerable increase in thermal efficiency as well as major simplifications and savings in the balance of plant. The MSR employs a circulating liquid fuel mixture that offers considerable flexibility for recycling actinides and may provide a favorable alternative to accelerator-driven systems for actinide destruction.

2.1. Very-High Temperature Reactor

The VHTR design will be a graphite-moderated thermal neutron spectrum reactor that will produce electricity and hydrogen in a highly efficient manner. The VHTR core could be either a prismatic graphite block type core or a pebble bed core cooled by helium. Use of a liquid-salt coolant is also being evaluated. The VHTR will use very high-burnup, low-enriched uranium, TRISO-coated fuel, and will have a projected plant design service life of 60 years. The process heat for the hydrogen production, and possibly the electricity production, will be transferred through an intermediate heat exchanger (IHX). The VHTR concept is considered the nearest-term reactor design that has the capability to efficiently produce hydrogen. The plant size, reactor thermal power and core configuration will be designed to assure passive decay heat removal without fuel damage during any hypothetical accident. Figure 1 provides a sample schematic of one possible design for the VHTR.

A target schedule for the development and construction of the VHTR includes starting preconceptual designs in 2006, completing preliminary design in 2009, completing major R&D activities by 2012, and starting construction in 2012. VHTR operations are scheduled to begin in 2017.

The DOE laboratories, led by the Idaho National Laboratory (INL), will perform R&D that will be critical to the success of the VHTR, primarily in the areas of:

- high-temperature gas reactor fuels behavior;
- high-temperature materials qualification;
- design methods development and validation;
- Hydrogen production technologies; and
- energy conversion.

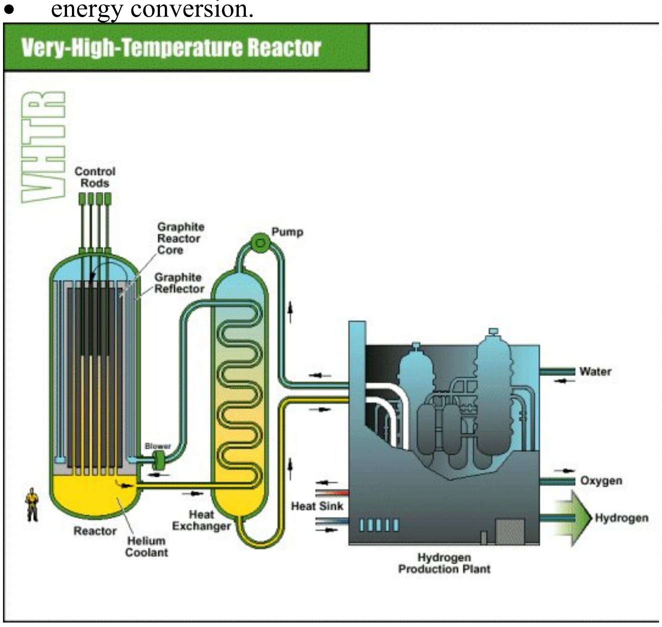

Fig. 1. *Conceptual VHTR system*

The VHTR design has not yet been selected; consequently, the R&D process is focused on scenarios and phenomena previously identified as important by the advanced gas-cooled reactor community. The calculation and experimental needs, and consequently the required R&D, will be focused in several distinct areas:

- basic differential and integral nuclear cross-section data measurement and evaluation, including mathematically rigorous sensitivity studies of the effects of uncertainties in the differential nuclear data on key integral reactor properties;
- reactor assembly cross-section preparation;
- discrete ordinates transport; nodal diffusion; reactor kinetics; thermal-hydraulics; fuel behavior; and fission product transport.

Development and qualification of TRISO-coated, low-enriched uranium fuel is a key R&D activity. This activity includes work on improving the kernel fabrication, coating, and compacting technologies; irradiation and accident testing of fuel specimens; and fuel performance and fission product transport modeling. The primary goal of these activities is to successfully demonstrate that TRISO-coated fuel can be fabricated to withstand the high temperatures, burnup, and power density requirements of a prismatic block type VHTR with an acceptable failure fraction.

Fuel performance modeling will address the structural, thermal, and chemical processes that can lead to coated-particle failures. The release of fission products from the fuel particle will also be modeled, including the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties.

The Materials R&D Program will focus on testing and qualification of the key materials commonly used in VHTRs and will address the materials needs for the NGNP reactor, intermediate heat exchanger, and associated balance of plant.

Significant quantities of graphite have been used in nuclear reactors, and the general effects of neutron irradiation on graphite are reasonably well understood. However, models relating structure at the micro and macro level to irradiation behavior are not well developed.

The irradiation damage and property changes of VHTR materials must be measured. This program is directed at the development of C/C and SiC/SiC composites for use in selected very high temperature/very-high neutron fluence applications, such as control rod cladding and guide tubes where metallic alloy are not feasible.

2.2. Supercritical Water Reactor

Supercritical water-cooled reactors are promising advanced nuclear energy systems because of their high thermal efficiency (about 45% vs. about 33% efficiency for current LWRs) and considerable plant simplification. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct, once-through coolant cycle. Critical parameters for the water are 374°C and 22.1 MPa. Operation above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. Thus, the need for recirculation and jet pumps, pressurizer, steam generators, and steam separators and dryers in current LWRs is eliminated. The main mission of the SCWR is generation of low-cost electricity. It is built upon two proven technologies: LWRs, which are the most commonly deployed power generating reactors in the world, and supercritical fossil-fired boilers, a large number of which are also in use around the world.

Fig. 2. *Conceptual SCWR system*

The work during 2005–2013 will be focused on identification of the most promising design. The R&D program needs to address current basic knowledge gaps in areas such as thermal-hydraulic phenomena expected during normal operation and accidents, system performance under a variety of conditions, and analytical methods needed for safety and system performance assessment. The necessary experiments will be conducted, databases will be developed, and analytical models and codes will be assessed and improved where necessary.

The actual R&D on materials will need to focus on the following key areas:

- oxidation, corrosion, and stress corrosion cracking;
- radiolysis and water chemistry;
- strength, embrittlement, and creep resistance; and
- dimensional and microstructural stability.

For any SCWR core design, materials for reactor internals and fuel cladding will need to be evaluated and identified. Zirconium-based alloys, so pervasive in conventional water-cooled reactors, will not be a viable material for most of the proposed SCWR core designs without some sort of thermal and/or corrosion-resistant barrier. Based on the available data for other alloy classes, there is currently no single alloy that has been studied enough to unequivocally ensure its viability in an SCWR

2.3. Gas-Cooled Fast Reactor

The gas-cooled fast reactor (GFR; see Figure 3) was chosen as one of the Generation IV nuclear reactor systems to be developed based on its excellent potential (1) for sustainability through reduction of the volume and radiotoxicity of both its own fuel and other spent nuclear fuel and (2) for extending/utilizing uranium resources orders of magnitude beyond what the current open fuel cycle can. The main characteristics of the reference GFR are a self-generating core (i.e., CR = 1) with a fast neutron spectrum, robust refractory fuel, high operating temperature, direct energy conversion with a gas turbine, and full actinide recycling (possibly with an integrated, on-site fuel reprocessing facility).

The reference GFR system features a fast-spectrum, helium-cooled reactor and closed fuel cycle. This was chosen as the reference design due to its close relationship with the VHTR, and thus its ability to utilize as much VHTR material and balance-of-plant technology as possible. Like thermal-spectrum helium-cooled reactors the high outlet temperature of the helium coolant makes it possible to deliver electricity, hydrogen, or process heat with high conversion efficiency.

The specific GFR research objectives include:

- system design and safety research, which includes conceptual studies of a reference GFR system, analyses of the safety approach and the development of computational tools for these studies;
- materials research, which includes the identification and/or development of materials that can withstand the high temperatures and high fluence that will be encountered within the core region; and
- fuel and fuel-cycle research, which will identify and fabricate those fuels that will perform well under extreme temperature and radiation conditions.

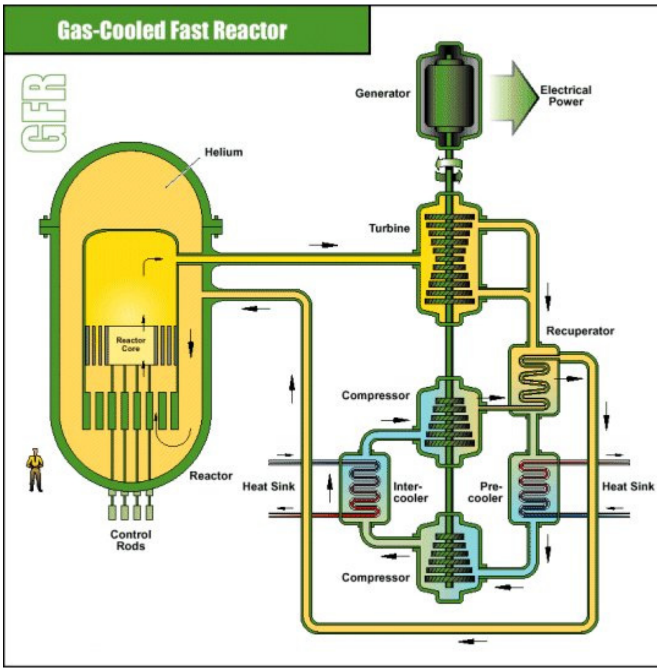

Fig. 3. *Conceptual GFR system*

The major activities within the System Design include design and evaluation of passive and active safety systems for decay heat removal, system control and transient analysis, design and construction of experiments for thermal-hydraulic/safety tests and coolant chemistry control, code development/adaptation for neutronic and thermal-hydraulic analysis; and feasibility studies of a direct Brayton cycle (including component testing) and development of the turbomachinery for helium and CO_2 systems.

2.4. Lead-Cooled Fast Reactor

The Lead-Cooled Fast Reactor (LFR) is proposed to advance all of the Generation IV goals of non-proliferation, sustainability, safety and reliability, and economics. Two key technical aspects of the envisioned LFR that offer the prospect for achieving these goals are the use of lead (Pb) coolant and a long-life, cartridge-core architecture in a small, modular system. The Pb coolant is a poor absorber of fast neutrons and enables the traditional sustainability and fuel-cycle benefits of a liquid metal-cooled fast spectrum core to be realised. Lead does not interact vigorously with air, water/steam, or carbon dioxide, thus eliminating concerns about exothermic reactions. It has a high boiling temperature (1740 °C) such that the prospect of boiling or flashing of the ambient pressure coolant is realistically eliminated.

The incorporation of inherent thermo-structural feedback imparts walk-away passive safety, while the use of a sealed cartridge core with a 20-year or longer cycle time between refueling imparts strong proliferation resistance. The R&D plan addresses viability issues associated with the LFR leading to the Generation IV fast reactor selection in 2010 and a follow-on decision in 2014 to proceed with design and construction of the LFR demonstration plant.

R&D tasks for System Design and Evaluation will address the areas of core neutronics, system thermal hydraulics, passive safety evaluation, containment and building structures, in-service inspection, and assessing cost impacts. R&D tasks associated with this work include further optimization of the core configuration, establishing a startup/shutdown rod and control rod strategy, and calculating reactivity feedback coefficients. System thermal hydraulic studies are essential to establishing the parameters for potential natural circulation cooling in the primary system, identifying any safety issues to be addressed in subsequent design, and establishing parameters for ensuring passively safe response. Viability of the long-life core and passive safety under all upset conditions (including seismic events that might unacceptably reconfigure a core) requires materials that can withstand stresses at high temperature and, for some components, contact with liquid lead.

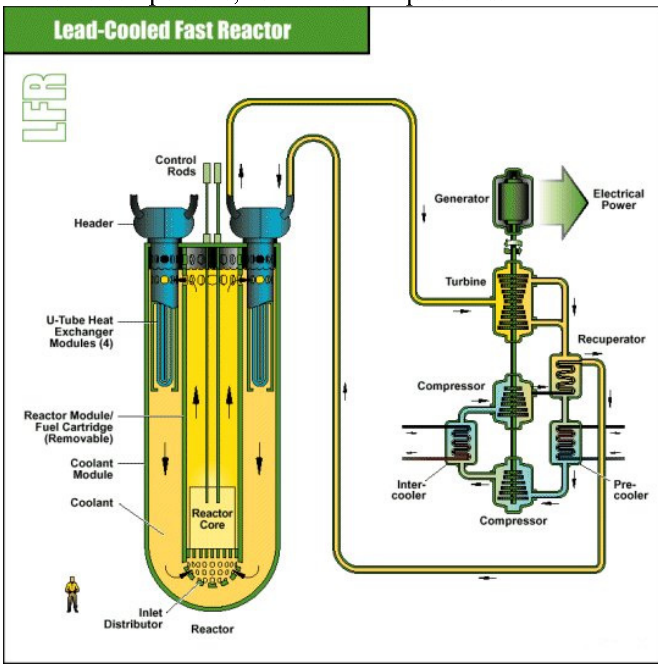

Fig. 4. *Conceptual LFR system*

Achieving long core life, passive safety, and reliable operation will require robust and predictable fuel performance. Nitride fuel has many properties and characteristics that render it well suited for LFR application; however, there is very little data on nitride fuel performance.

Use of an S-CO_2 Brayton cycle for energy conversion (rather than a steam Rankine cycle) offers the prospect of acceptable efficiencies with lower Pb coolant outlet temperatures, which reduces the challenges for materials.

R&D tasks associated with viability of long core lifetime, passive safety include identifying candidate Si-enhanced ferritic-martenistic steels, testing the compatibility of candidate materials with heavy liquid metal coolants, demonstrating control of corrosion to ensure adequate thickness of cladding and structural elements at operating temperatures over long core and reactor lifetimes.

2.5. Sodium-Cooled Fast Reactor

The sodium-cooled liquid metal reactor system features a fast-spectrum reactor and closed fuel recycle system. The primary mission for the SFR is the management of high-level wastes and, in particular, management of plutonium and other actinides. With innovations to reduce capital cost, the mission can extend to electricity production, given the proven capability of sodium reactors to utilize almost all of the energy in the natural uranium. A variety of plant size options is available for the SFR, ranging from modular systems of a few hundred MWe to large reactors of about 1500 MWe. The primary coolant system in a SFR can either be arranged in a pool layout (a common approach, where all primary system components are housed in a single vessel, see Figure 5) or in a compact loop layout. For both options, there is a relatively large thermal inertia of the primary coolant. A large margin to coolant boiling is achieved by design and is an important safety feature of these systems. Another major safety feature is that the primary system operates at essentially atmospheric pressure. A secondary sodium system acts as a buffer between the radioactive sodium in the primary system and the energy conversion system in the power plant.

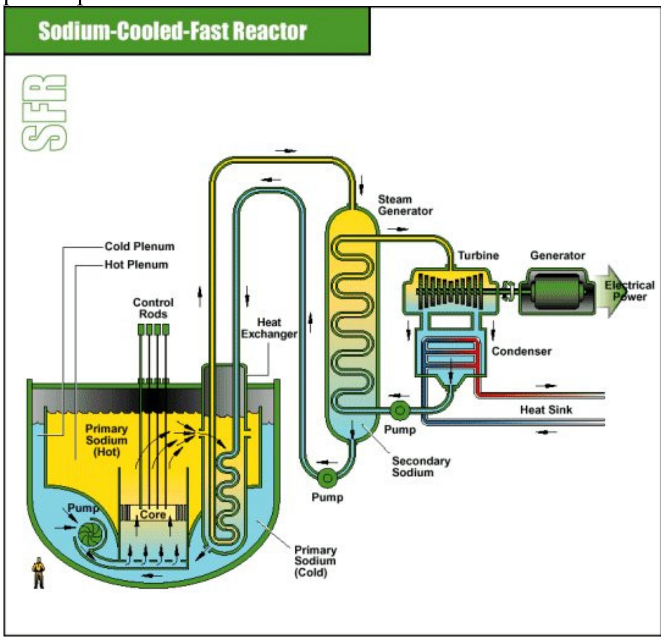

Fig. 5. *Conceptual SFR system*

The performance targets affecting the SFR development, include completion of the preconceptual reference design by 2007 and the initial phase of materials research and reactor design by 2010 to facilitate selecting the preferred fast spectrum system by the end of 2010.

R&D is needed to demonstrate the design and safety characteristics, especially with fuels containing transurancis, and to optimise the design with innovative approaches to meet the objectives of the specific Generation IV missions, primarily waste management.

2.6. Molten Salt Reactor

Molten Salt Reactors (MSR; see Figure 6) are liquid-fueled reactors that can be used for burning of actinides and production of electricity, hydrogen, and fissile fuels. Fissile, fertile, and fission products are dissolved in a high-temperature, molten-fluoride salt with a very high boiling point (1400 °C). The near-atmospheric-pressure molten-fuel salt flows through the reactor core that contains graphite moderator. In the core, fission occurs within the flowing fuel salt that is heated to ~700ºC, which then flows into a primary heat exchanger where the heat is transferred to a secondary molten-salt coolant. The fuel salt then flows back to the reactor core. The clean molten salt in the secondary heat transport system transfers the heat from the primary heat exchanger to a high-temperature Brayton cycle that converts the heat to electricity. The Brayton cycle may use either nitrogen or helium as a working gas.

In the 1950s and 1960s, two experimental MSRs built at the Oak Ridge National Laboratory (ORNL) established the basic technology for the MSR. In addition, there are overlaps between the MSR and the technologies being developed for the NGNP, which would provide the basis for an Advanced Molten Salt Reactor (AMSR) with major improvements in economics and reductions in R&D requirements.

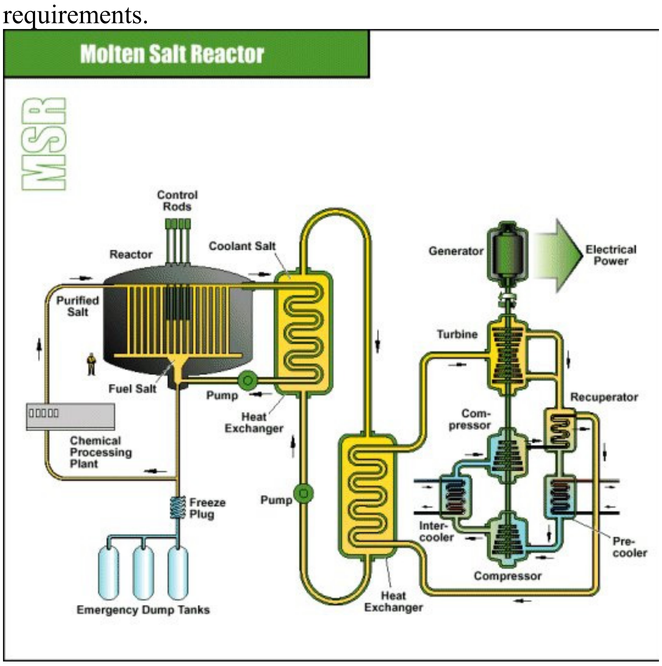

Fig. 5. *Molten Salt Reactor with Brayton Power Cycle*

The overall systems timeline is to determine viability by 2014. Because the basic technology of the MSR has been demonstrated, viability is defined as sufficient information to make a credible determination on the commercial viability of a MSR that meets the defined design goals.

Because of ongoing programs, major advances in development and understanding of MSRs are expected to occur within the next decade with a modest investment of resources. This should enable the program to develop a credible understanding of the economics, capabilities to perform alternative missions (i.e., burning of actinides and production of electricity, hydrogen, and fissile fuel), and issues associated with a modern MSR, thus providing the basis for a decision on whether to initiate a large-scale developmental program with the goal of deployment.

3. NEW DESIGN AT WESTINGHOUSE

The International Reactor Innovative and Secure (IRIS) is not a Generation IV design, since it could become available for deployment decades ahead of the 2020 to 2030 time frame projected for the six Generation IV system [3].

IRIS, however, has been the first to formulate and implement the philosophy that next generation systems should leverage their design and operational characteristics to prevent accidents to the highest extent possible.

Although still at the preliminary stage, the IRIS nuclear power plant design has moved rapidly from idea to viable commercial entry. IRIS is a very innovative reactor design with many attractive new features, especially in the safety area, but at the same time its technology is grounded on well-proven and universally familiar water reactors experience. The main characteristic of the IRIS is integral configuration. All the main primary system components (core with reflector/shield, pressurizer, reactor coolant pumps, steam generators, and control rod drive mechanisms) are located inside the reactor pressure vessel. It is offered in configurations of single or multiple modules, each having a power rating of 1000 MWt (about 335 MWe).

The integral arrangement eliminates all the pressure vessels outside the reactor vessel (RV), as well as the large connecting loop piping between them, resulting in a compact, more economical configuration and in the physical elimination of the LOCAs. It is larger than a traditional RV, and has an inside diameter of 6.2 m and an overall height of 21.3 m, including the closure head (Figure 7).

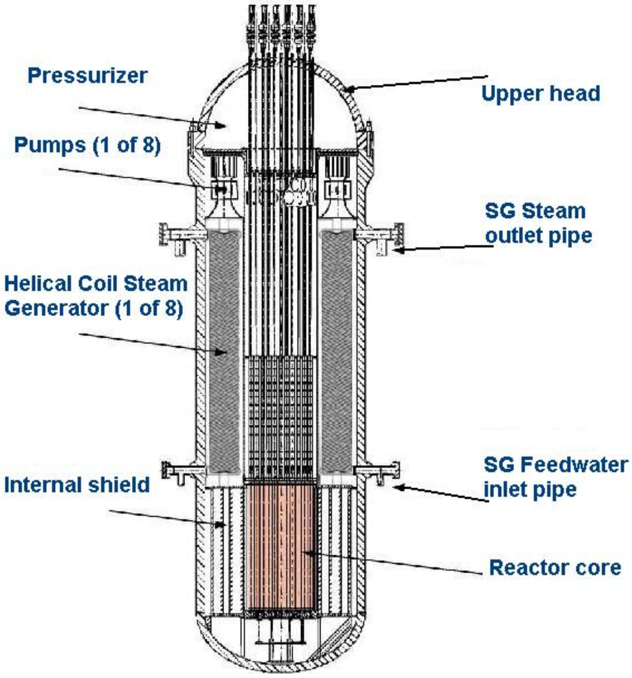

Fig. 7. *IRIS integral layout*

The IRIS steam generators (SG) are of a once through, helical-coil tube bundle design, with the primary fluid outside the tubes. Eight steam generator modules are located in the annular space between the core barrel (outside diameter 2.85 m) and the reactor vessel (inside diameter 6.2 m). The enveloping outer diameter of the tube bundle is 1.64 m. Each SG has 656 tubes, and the tubes and headers are designed for the full external reactor coolant system (RCS) pressure. A unique aspect of the IRIS SG design is that the high pressure primary coolant flows on the outside of the tubes. Thus, the IRIS SG tubes are in compression, and therefore, tensile stress corrosion cracking which has been responsible for more than 70 percent of all the SG tube failures is automatically eliminated. The IRIS pumps are of the spool type, which has been used in marine applications and designed for chemical plant applications requiring high flow rates.

The integral configuration is ideal for locating the control rod drive mechanisms (CRDM) inside the vessel, in the region above the core and surrounded by the steam generators. Their advantages are in safety and operation.

The IRIS pressurizer is integrated into the upper head of the reactor vessel. By utilizing the upper head region of the reactor vessel, the IRIS pressurizer provides a very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel. The IRIS pressurizer has a total volume of about 71 m^3, which includes a steam volume of about 49 m3 . This steam volume is about 1.6 times bigger than the pressurizer steam space for a large PWR, while IRIS has about 1/3 the core power. Because of this large steam volume-to-power, IRIS does not need a pressurizer spray function to prevent the pressurizer safety valves from lifting for any design basis heatup transients.

Adoption of an integral configuration has a very positive impact on the reactor's overall intrinsic safety, well beyond the obvious elimination of the large LOCAs. This has allowed IRIS to implement an extremely effective "safety-by-design" approach. Of the eight class IV accidents that must be considered in PWRs, only one remains unaffected in IRIS (Design basis fuel handling accidents). All the others are either eliminated outright or are downgraded to a lower classification.

4. NEW DESIGN AT UC BERKELEY

The Department of Nuclear Engineering at University of California at Berkeley (UCB), in collaboration with Westinghouse and Lawrence Livermore National Laboratory (LLNL) designed the Encapsulated Nuclear Heat Source (ENHS) [4]. The ENHS is a novel 125 MWth fast spectrum reactor concept that was selected by the 1999 DOE NERI program as a candidate Generation-IV reactor. It uses Pb-Bi or other liquid-metal coolant and is intended to be factory manufactured.

A schematic description of the reference design concept of the ENHS reactor is depicted in Figure 8. The nuclear steam supply system (NSSS) consists of one ENHS module and eight small steam generators. There is no mechanical connection between the module and the steam generators. Both primary and secondary coolants flow by natural circulation. The primary coolant that is heated in the core flows up the riser, turns over into the Intermediate Heat Exchanger (IHX) and flows back into the coolant plenum underneath the core. The secondary coolant flows from the pool outside of the vessel into the bottom of the IHX and exits back to the pool near the top of the IHX. Relative to circular tube IHX the rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller friction losses due to elimination of grid spacers. The mean temperature difference between the primary and secondary coolants is <323 K. Cover gas from the top of the module is injected into the coolant just above the core. The cover gas bubbles reduce the effective density of coolant in the riser, thus increasing the head for coolant circulation. The circulator would be located above the reactor pool, outside of the module vessel.

Unique features of the ENHS include at least 20 years of operation without refueling; factory fuelled, weld-sealed - no fuel handling in the host country; no pumps, pipes and valves; excess reactivity does not exceed β_{eff} ; fully passive removal of the decay heat; very small probability of core damaging accidents; autonomous operation and capability of load-following over a wide range; very long plant life. The ENHS reactor is designed to meet the requirements of Generation IV reactors including sustainable energy supply, low waste, high level of proliferation resistance, high level of safety and reliability, acceptable risk to capital and, hopefully, also competitive cost of electricity.

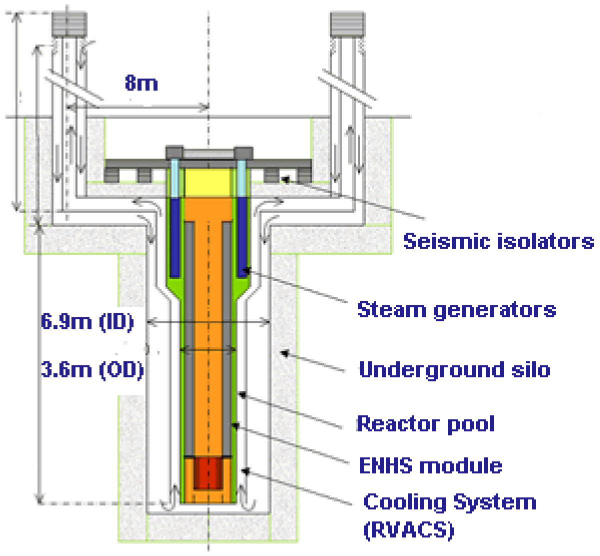

Fig. 8. *ENHS integral layout*

5. ADVANCED METHODS DEVELOPMENT AT UC BERKELEY FOR ADVANCED REACTOR DESIGNS

5.1 IRIS benchmark calculations

A new methodology SAS2H/KENO-V.a [5] for fuel depletion analysis of entire cores of the IRIS reactor, founded on the application of SCALE-4.4a [6] sequences was developed. This methodology combines a 3D Monte Carlo full core calculation of node power distribution and a 1D Wigner-Seitz equivalent cell transport method for independent depletion calculation of each of the nodes.

In order to assess the feasibility/practicality of the SAS2H/KENO-Va methodology for full core modeling, we compared it with the well-established Westinghouse ALPHA/PHOENIX/ANC deterministic code system [7]. In analyzing the IRIS core benchmark, it was not always possible/practical to use exactly identical assumptions. Therefore, some differences in the results are expected. We should emphasize that the objective of this comparison was not to investigate and resolve these small differences, but to establish that with reasonably similar assumptions we can obtain reasonably close results. The reasons for the differences include the following: (1) non-standard use of the Westinghouse codes, (2) slightly different reflector representation, (3) different treatment of the Doppler feedback, (4) burnup and change in fuel isotopics is followed on a different axial mesh, (5) statistical noise in Monte Carlo results, combined with a possibility that the solution is still not completely converged to fundamental mode, and (7) different cross section libraries, including B-V and B-VI based data. It should also be noted that the IRIS benchmark core does not necessarily represent an actual (acceptable) core configuration.

Figure 9 compares k_{eff} evolution during the IRIS core fuel burnup. The difference is acceptable, almost constant over the whole range, and primarily due to different assumptions. Figure 10 compares power peaking factors as a function of burnup; the behaviour is very similar. In summary, a very good agreement of the two code system results has been observed.

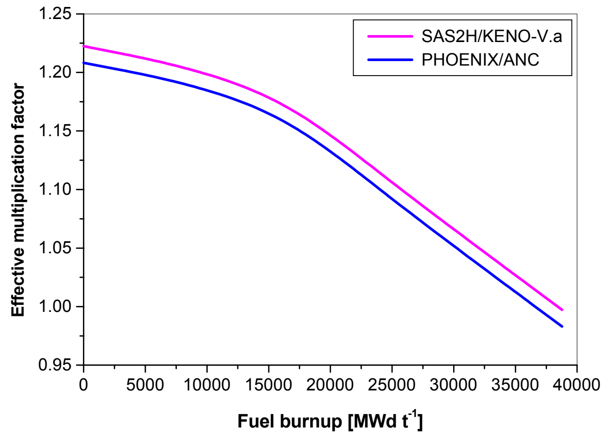

Fig. 9. k_{eff} *evolution with the IRIS fuel burnup*

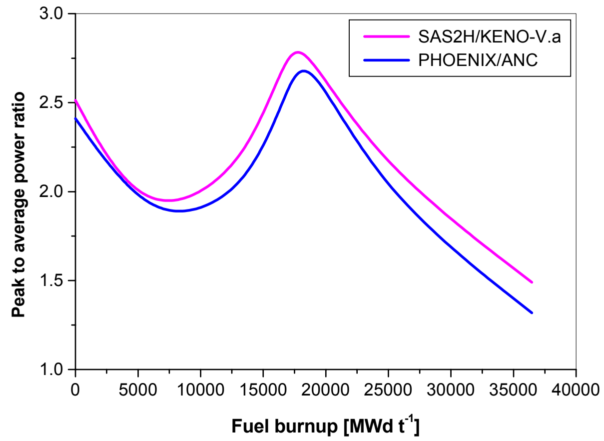

Fig. 10. *Evolution of IRIS core peak to average power ratio*

5.2. ENHS benchmark calculations

Two computational procedures for the analysis of the ENHS benchmark core were selected. The first procedure is based on the application of the MCNP -4C [8] and ORIGEN2.1 [9] codes, interfaced by the MOCUP driver [10]. The second procedure KWO2 [11], is based on the coupling of the KENO -V.a and ORIGEN2.1 codes.