

# ADVANCED SUPERCRITICAL WATER AND MOLTEN SALT REACTORS

**R. Schenkel<sup>(1)</sup>, T. Schulenberg<sup>(2)</sup>, C. Renault<sup>(3)</sup> and D. Haas<sup>(4)</sup>**

(1) Roland Schenkel – European Commission Joint Research Centre (*roland.schenkel@ec.europa.eu*)

(2) Thomas Schulenberg – Forschungszentrum Karlsruhe (*thomas.schulenberg@iket.fzk.de*)

(3) Claude Renault – Commissariat à l'énergie atomique (*clauder.renault@cea.fr*)

(4) Didier Haas – European Commission Joint Research Centre (*didier.haas@ec.europa.eu*)

## I. INTRODUCTION

Out of the six energy systems covered under GIF (Generation IV International Forum), 3 concern purely fast neutron reactors (cooled with Sodium, Lead or Gas), and the fourth one is the thermal neutron very high temperature reactor. The two remaining systems, which will be described hereafter, have quite different characteristics from the former four. When they are mastered technically, they both might prove meeting quite well the main requirements of Generation IV systems.

The super-critical water coolant enables a thermal efficiency about one-third higher than current light-water reactors, as well as simplification in the balance of plant. The balance of plant is considerably simplified because the coolant does not change phase in the reactor and is directly coupled to the energy conversion equipment. The reference system is 1 500 MWe with an operating pressure of 25 MPa, and a reactor outlet temperature of 500°C or more, possibly ranging up to 625°C. The fuel is uranium dioxide, MOX or possibly thorium dioxide. Passive safety features shall be incorporated similar to those of simplified boiling water reactors.

As the system uses existing light water reactor technology, there is already extensive worldwide experience in constructing and

operating this sort of reactor. A SCWR design could be developed with a fast neutron spectrum. Using fast neutrons with higher kinetic energies would enable the system to produce at least as much fissile material as it consumes (thereby fulfilling the sustainability goal as set out in the Generation IV roadmap). This concept's tendency to have a positive void reactivity coefficient together with the potential for design basis loss-of-coolant accidents are likely to make this difficult to develop. The other major challenges for the SCWR are to develop a viable core design, accurately estimate the heat transfer coefficient and develop materials for the fuel and core structure that will be sufficiently corrosion-resistant to withstand SCWR conditions.

In the MSR system, the fuel is dissolved in a fluoride salt liquid mixture also playing the role of primary coolant. In the original design developed by ORNL in the 60-70's, the molten salt fuel flows through graphite core channels, producing an epithermal spectrum. The heat generated in the molten salt is transferred to a secondary coolant system through an intermediate heat exchanger, and then through a tertiary heat exchanger to the power conversion system. The reference plant has a power level of up to 1 000 MWe. The system has a coolant outlet temperature of 700°C, possibly ranging up to 800°C, affording improved thermal efficiency.

The interest is focused today on fast neutron MSR concepts for breeding and/or minor actinide burning, without graphite in the core (see section IV).

The MSR's liquid fuel allows addition of actinides such as plutonium and avoids the need for fuel fabrication. Actinides - and most fission products - form fluorides in the liquid coolant.

The main benefits of the MSR system are that it offers an integrated fuel cycle, embodying a burner/breeder reactor concept whilst taking advantage of the excellent heat transfer properties and very low vapour pressure of molten salt. These properties imply that the building housing a MSR could be smaller than for other reactor concepts under development and that the thermal power output would be higher. A number of other promising applications for molten salts beyond the MSR itself have been identified. These use a variety of salt compositions that vary according to the envisioned application.

These include: liquid fuel, primary or secondary coolant, and pyrochemistry solvent. Molten salts might also be used as a substitute for primary or secondary circuit working fluids in the SFR and VHTR. The molten salt chemistry and handling, with the resulting corrosion of reactor components, along with the development of materials and the fuel cycle, are the main challenges for the development of this system.

## II. STATUS OF PARTICIPATION IN SCWR AND MSR SYSTEMS

As can be seen from the Table 1 hereunder, the System Arrangement (SA) for the SCWR has been signed by Canada, Japan and France. No Project Arrangement (PA) has been signed yet, but three partners (Canada, EURATOM and Japan) are provisional participants in the four Projects of this system. France is provisional participant in the SCWR Materials and Chemistry (M&C) Project. The Republic of Korea is observer in three Projects. The situation of MSR is such that no System Arrangement was signed yet, but provisional participants are from EURATOM, France and USA.

## III. STATUS OF THE SUPERCRITICAL WATER REACTOR SYSTEM (SCWR)

### III.A. Main characteristics of the system

The Super-Critical Water Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point of water (374°C, 22.1 MPa). Two design options – pressure vessel and pressure tube design – are considered for the SCWR. Technologies and thus most of the R&D needs to assess the technical feasibility, like materials, water chemistry, fuel, heat transfer, and safety systems are common to both designs, which provides valuable collaboration opportunities for countries and organizations working out either design option.

The main advantage of the SCWR is improved economics because of the higher steam enthalpy, increasing the thermal efficiency while decreasing the steam mass flow rate, and the potential for plant simplification. Improvements in the areas of sustainability, proliferation resistance and physical protection are also possible and are being pursued by considering several options for design using thermal as well as fast spectra, including the use of advanced fuel cycles.

### III. B. Status of cooperation

In 2008, efforts focused on finalizing the Thermal-Hydraulics and Safety and the Materials and Chemistry Project Arrangements. For the System Integration and Assessment project, a provisional project was created and worked in 2008 on drafting the technical part of the PA. The project on Fuel Qualification was recently created with the objective of testing the SCWR fuel in a suitable research reactor under prototypical super-critical water conditions.

While waiting for the signature of PAs, signatories of the SA are sharing results from R&D through informal exchanges and project meetings.

Table 1: Signed arrangements and informal cooperation within GIF (Dec. 2008)

	CAN	EUR	FRA	JPN	PRC	ROK	RSA	RUF	CHE	USA
<b>VHTR SA</b>	X	X	X	X	X	X			X	X
VHTR HP PA	X	X	X	X		X			O	X
VHTR FFC PA	O	X	X	X		X				X
VHTR CMVB Project		P	P	P		P	P			P
VHTR MAT Project	P	P	P	P		P	P		P	P
<b>SFR SA</b>		X	X	X	O	X		O		X
SFR AF PA		X	X	X		X				X
SFR GACID PA			X	X						X
SFR CDBOP PA			X	X		X				X
SFR SO Project			P	P		P				P
SFR SIA Project		P	P	P		P				P
<b>SCWR SA</b>	X	X		X						
SCWR M&C Project	P	P	P	P		O				
SCWR TH & S Project	P	P		P		O				
SCWR SIA Project	P	P		P		O				
SCWR FQ Project	P	P		P						
<b>GFR SA</b>		X	X	X					X	
GFR FCMFC Project		P	P	P					O	
GFR CD & S Project		P	P						P	
<b>LFTR System</b>		P		P						P
<b>MSR System</b>		P	P							P

X = Signatory

P = Provisional participant

O = Observer

### Acronyms of Projects

HP	Hydrogen Production	CDBOP	Component Design and Balance-Of-Plant
FFC	Fuel and Fuel Cycle	SO	Safety and Operation
CMVB	Computational Methods Validation and Benchmarking	SIA	System Integration and Assessment
MAT	Materials	M&C	Materials and Chemistry
AF	Advanced Fuel	TH & S	Thermal-Hydraulics and Safety
GACID	Global Actinide Cycle International Demonstration	FQ	Fuel Qualification
		FCMFC	Fuel, Core Materials and Fuel Cycle
		CD & S	Component Design and Safety

Since 2007, research organizations from China were showing increasing interest to join the SCWR projects. Currently, a consortium of 8 Chinese partner organizations is working on a larger R&D program on design and technologies of SCWR to assess its future potential.

### III. C. R&D Objectives

Regarding system design, the objective is to pursue pre-conceptual design studies for several concepts in order to investigate their respective potentials.

In the field of materials and chemistry, the main objective is to select key fuel cladding and structural materials for the pressure tube and pressure vessel designs. The work includes the definition of a reference water chemistry, based on materials compatibility and radiolysis behavior at supercritical conditions.

In the field of thermal-hydraulics and safety, significant gaps exist in the heat transfer database and the assessment of safety systems for the SCWR. Data needed for thermal-hydraulics and safety analysis at prototypical SCWR conditions will be produced as part of the TH & S project.

### III. D. Main activities and outcomes

The 4<sup>th</sup> International Symposium on Supercritical Water Cooled Reactors has been held from March 8-11, 2009, in Heidelberg, Germany, summarizing the latest status of worldwide R&D activities in this field. More than 100 participants and observers from GIF member states were listening to around 80 presentations given on core and system design, materials and chemistry, thermal hydraulics, safety systems and overall assessment of the SCWR. Proceedings may be downloaded from [www.hplwr.eu](http://www.hplwr.eu). The following chapter is illustrating some highlights of this symposium. In the field of system integration and assessment, the main activities were the development of pre-conceptual SCWR designs, including core design with thermal or fast neutron spectrum, pressure tube and pressure vessel design, as well as first plant layout.

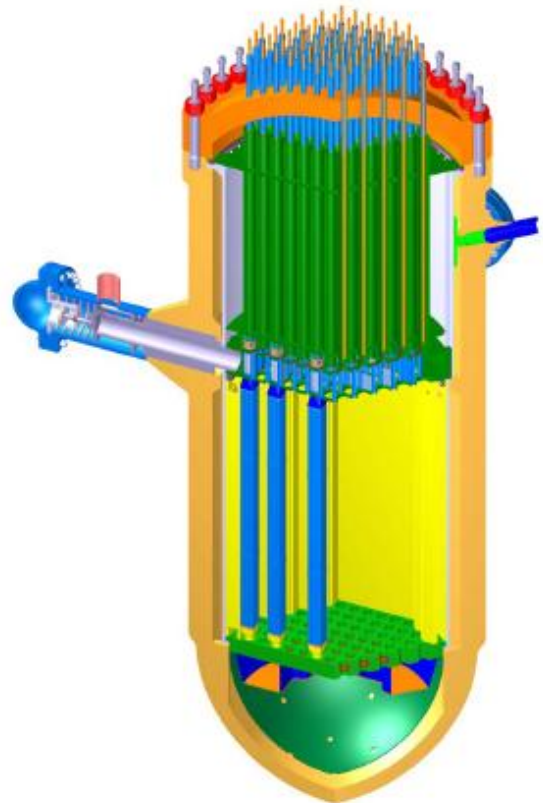


Figure 1. Design of the High Performance Light Water Reactor, Schulenberg and Starflinger [1]

European organizations were presenting their latest design concept of the High Performance Light Water Reactor, Figure 1. It features a pressure vessel type reactor with a thermal core which is heating up the coolant in three steps to 500°C average core outlet temperature, and includes mixing chambers above and underneath the core to minimize peak cladding temperatures. A steam cycle has been designed using state of the art high pressure, intermediate pressure and low pressure turbines, as well as seven pre-heater stages in the feed water line to optimize efficiency. Control of power, pressure and mass flow has been modelled with a system code to operate the reactor at 25 MPa constant core inlet pressure. Coupled neutronic and thermal-hydraulic analyses of the core demonstrate that the envisaged power profile is feasible, which differs significantly from the core design of conventional light water reactors by different power density levels in different core regions. Burn-up analyses have been performed to

estimate redistribution of the power profile during a burn up cycle and to determine refuelling intervals. A first layout of the containment and its safety systems is based on the design of latest boiling water reactors. It includes a pressure suppression pool, 4 core flooding pools, depressurization systems, and a passive high pressure residual heat removal system, which need to be analyzed next for a number of postulated accident scenarios. Phase 2 of the HPLWR project started in 2006 and will run until 2010.

Canada has been focusing on the general layout and thermodynamic cycle options for pressure tube reactors. Main objectives for developing and utilizing SCWRs are an increase of gross thermal efficiency of current nuclear power plants from 33-35% to approximately 45–50%, decrease of the capital and operational costs and, in doing so, decrease of electrical energy costs, and co-generation of hydrogen through thermo-chemical cycles, as outlined by Naidin *et al.* [2] To decrease significantly the development costs of a SCWR, to increase its reliability, and to achieve similar high thermal efficiencies as the advanced fossil steam cycles, it should be determined whether the SCWR power plant can be designed with a steam-cycle arrangement which closely matches that of latest supercritical fossil power plants. A two loop system with supercritical water in the primary loop and a steam generator for a secondary loop has been assessed for comparison. First coupled neutronics / thermal-hydraulics analyses of a fuel channel for a pressure tube SCWR with 625°C outlet temperature have been presented, indicating the need for further core optimization to meet material limits.

Japan is pursuing two pressure vessel designs (thermal and fast spectrum), as summarized by Ishiwatari *et al.* [3] The fast reactor is expected to be designed with a similar plant system as the thermal reactor. The fast reactor will produce a higher power density than the thermal reactor because less moderator is needed, and thus more thermal power can be produced using the same reactor pressure vessel size, which will further reduce the unit capital costs. With the scope of developing an economical fast reactor system, the Japanese

research project of the “Super Fast Reactor” started in December 2005 and will run until March 2010. The University of Tokyo, Kyushu University, JAEA and TEPCO are contributing to it. The purpose of the concept development is to pursue the advantage of high power density of fast reactors over thermal reactors to achieve economic competitiveness of fast reactors for its deployment without waiting for exhausting uranium resources. The design goal is not breeding but maximizing the power density and utilizing plutonium from the LWR spent fuel. The reference fuel rod and core have been designed. Solid moderator (ZrH) in the blanket assembly enables the Super Fast Reactor to have negative void reactivity without adopting a flat core shape. 3D neutronic/thermal-hydraulic coupled calculations have been used for the core design. Sub-channel analyses have been performed for all the fuel assemblies to calculate the maximum cladding surface temperature.

The Republic of Korea continued further assessment of a conceptual SCWR design. It features a 1 400 MWe reactor core with a solid moderator, ZrH<sub>2</sub>, showing reasonable results although a further refinement is definitely needed. The idea of a solid moderator has been introduced since it was intended to simplify the coolant passage in a reactor upper dome. The shape of the solid moderator is basically a cross type but alternative versions are being studied in parallel. As shown in Figure 2, the fuel assembly has a 21x21 fuel rods array with a pitch of 1.15 cm, and the fuel assembly pitch is 25.15 cm, including a 1 cm gap between the fuel assemblies. The fuel assembly is composed of 300 fuel rods, 25 cruciform-type solid moderator pins, and 16 single solid moderator pins. The pellet diameter and the outer diameter of the cladding are 0.82 cm and 0.95 cm, respectively. The clad material is a nickel-based alloy, which is highly resistant to a stress corrosion cracking (SCC) at a supercritical water condition.

In China, some preliminary reactor core concepts have been worked out, among them a novel concept with mixed neutron spectrum. The core concept, sketched in Figure 3, combines the merits of both thermal and fast spectrum as far as possible. The basic idea is to divide the reactor

core into two zones with different neutron spectrum. In the outer zone, the neutron energy spectrum is similar to that of a thermal reactor. In this zone, the fuel assembly has a square arrangement but with downward, co-current flow of coolant and moderator water.

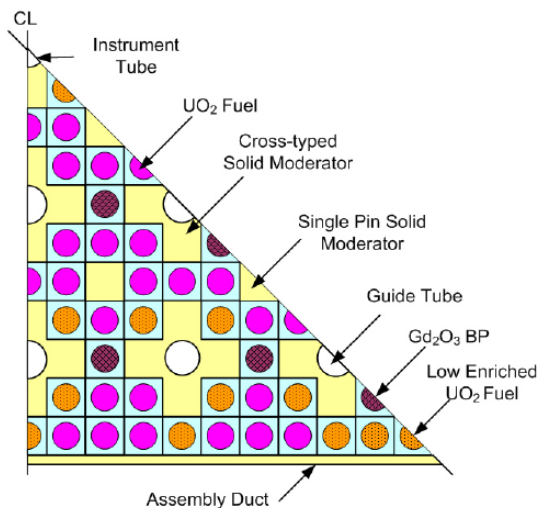


Figure 2. Fuel assembly design of a 1400 MWe core, Bae *et al.* [4]

Regarding materials and chemistry, progress was made in the areas of corrosion and stress corrosion cracking (SCC) testing, coatings tests, radiolysis, and modelling. Corrosion and SCC tests are being carried out at temperatures up to 650°C and pressures of about 25 MPa to evaluate the suitability of existing materials for the SCWR. In Japan two kinds of alloys have been developed with low swelling and high corrosion resistance; one was a SUS310S base alloy containing small amounts of Zr, the other one was SUS310S with fine grain microstructure.

Also stress corrosion cracking susceptibilities of selected austenitic stainless steels (316NG, 1.4970, 347H and an experimental creep resistant steel BGA4) and a high chromium Oxide Dispersion Strengthened alloy (PM2000) were studied in super-critical water.

Work on coatings involves the use of corrosion-resistant coatings on materials which exhibit good mechanical properties but have poor corrosion characteristics as a back-up option if existing materials are not suitable at supercritical conditions. The preparation of several ceramics

samples for preliminary evaluation in a static autoclave was pursued. In addition, Cr-coated samples, using advanced physical vapour deposition technique, were successfully tested and showed negligible corrosion.

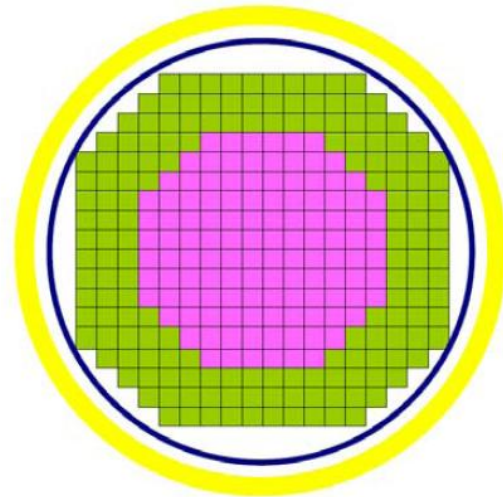


Figure 3. Scheme of the mixed SCWR core, Cheng [5]

Fundamental work, including experimental test and simulation, continued on the effect of radiation on supercritical water in a large range of temperature and pressure. Experimental techniques involved the use of a picoseconds pulse radiolysis method while molecular dynamics and Monte Carlo simulations were used to study radiolytic reactions. Manufacturing and assembly of the in-pile radiolysis and water chemistry loop at the Rez Research Center in the Czech Republic has been completed and the loop is being commissioned prior to the installation in the research reactor.

Other related activities included the evaluation of mechanical properties of several irradiated materials. High-temperature strength and creep strength, void swelling, helium embrittlement and phase stability have been evaluated by means of pressurized tube tests. The results of these tests have revealed that the creep deformation is dominated by thermal effects rather than irradiation effects at 700°C.

Regarding research on thermal-hydraulics, more heat transfer tests were conducted at supercritical conditions using water and modelling fluids (Freon and CO<sub>2</sub>). In addition, computational fluid dynamics (CFD) simulations were completed and compared with experiments. As an example, we show in Figure 4 the CFD simulation of flow inside a fuel assembly with wire wrapped fuel rods, as predicted by Kiss, *et al.* [6]

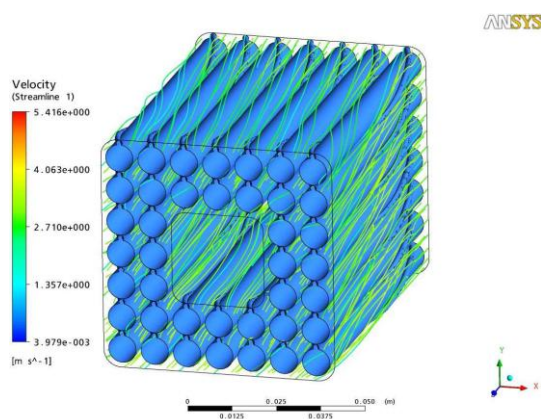


Figure 4. Streamlines of flow through a SCWR fuel assembly with wire wrapped fuel rods, Kiss *et al.* [6]

The physics of heat transfer deterioration in a supercritical water flow with low mass flux through a tube with high heat flux was studied using CFD. If the boundary layer is well resolved, and if physical properties of supercritical water are included properly in the analysis, the numerical simulation can model the observed phenomena with reasonable accuracy. A numerical study of turbulence enhancement by ribs on the heated wall indicates potential to avoid the deterioration of heat transfer. Efforts are under way to perform tests in water using annuli and a technique to scan the surface temperature of the test section (rather than using fixed thermocouples at specified locations). If successful, this technique will make it possible to obtain much better coverage in heat transfer tests and will be valuable for investigating the occurrence of deteriorated heat transfer (or the avoidance of deteriorated heat transfer in bundles or in enhanced surfaces). Initial tests resulted in failure of the test section due to improper electrical insulation and overheating of components that were not cooled by design. The

test section will be repaired using better insulation materials and testing will resume following the repair.

## IV. STATUS OF THE MOLTEN SALT REACTOR SYSTEM (MSR)

### IV. A. Main characteristics of the system

In a Molten Salt Reactor (MSR), the fuel is dissolved in a fluoride salt coolant. The technology was partly developed in the 1950's and 1960's in USA (ORNL). Compared with solid-fuelled reactors, MSR systems have lower fissile inventories, the absence of radiation damage that can limit fuel burn up, the possibility of continuous fission-product removal, the avoidance of the expense of fabricating fuel elements, the possibility of adding makeup fuel as needed, which precludes the need for providing excess reactivity, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics may enable MSRs to have potentially unique capabilities and competitive economics for actinide burning and extending fuel resources.

With changing goals for advanced reactors and new technologies, there is currently a renewed interest in MSRs. The new technologies include: Brayton power cycles (rather than steam cycles) that eliminate many of the historical challenges in building MSRs; and the conceptual development of several fast-spectrum MSRs that have large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors.

The challenges linked with the use of this concept are the materials corrosion, circuit contamination, maintenance at high temperature and confinement. They will be addressed in the R&D work plan.

### IV. B. Status of cooperation

The decision for setting up a provisional System Steering Committee (SCC) for the MSR was taken by the GIF Policy Group in May 2004. The participating members are EURATOM, France and the United States. Other countries

have been represented systematically (the Russian Federation) or occasionally (Japan) as observers in the meetings of the provisional SSC. Russia has played an important role at identifying R&D issues based on long-lasting programs initiated in the 1970s.

Beyond the GIF framework, the MSR provisional SSC has significantly contributed to enhance and harmonize international collaboration. A European network on MSR R&D has been active from 2001 until today [7]. The major contribution of EURATOM to MSR R&D within GIF has been the ALISIA (Assessment of LIquid Salts for Innovative Applications) project which was part of its 6<sup>th</sup> Framework Programme. A continuation study is proposed as a contribution to the 7<sup>th</sup> Framework Programme.

Partners of the MSR provisional SSC are involved also in the EURATOM-funded ISTC-3749 project, started in 2009 with official support from France, Germany, the Czech Republic, the United States, Canada and the IAEA.

Presently, formal MSR SA signature is not foreseen, but rather the settlement of a Memorandum of Understanding, which should encourage the parties interested to pursue their active collaboration, without having to be engaged into binding legal commitments.

#### *IV. C. R&D objectives*

The renewal and diversification of interests in molten salts have led the MSR provisional SSC to shift the R&D orientations and objectives initially promoted in the original Generation IV Roadmap issued in 2002, in order to encompass in a consistent body the different applications envisioned today for fuel and coolant salts.

Two baseline concepts are considered which have large commonalities in basic R&D areas, particularly for liquid salt technology and materials behavior (mechanical integrity, corrosion):

- The Molten Salt Fast-neutron Reactor (MSFR) is a long-term alternative to solid-

fuelled fast neutron reactors offering very negative feedback coefficients and simplified fuel cycle. [8] Its potential has been assessed but specific technological challenges must be addressed and the safety approach has to be established.

- The AHTR [9] is a high temperature reactor with better compactness than the VHTR and passive safety potential for medium to very high unit power (> 2400 MWth).

In addition, the opportunities offered by liquid salts for intermediate heat transport in other systems (SFR, LFR, VHTR) are investigated.

Liquid-salt chemistry plays a major role in the viability demonstration, with such essential R&D issues as: the physicochemical behavior of coolant and fuel salts, including fission products and tritium; the compatibility of salts with structural materials for fuel and coolant circuits, as well as fuel-processing material development; the on-site fuel processing; the maintenance, instrumentation and control of liquid-salt chemistry (redox, purification, homogeneity); and safety aspects, including interaction of liquid salts with sodium, water, and air.

The factorization into projects will emphasize cross-cutting R&D areas. A major commonality is the understanding and mastering of fuel and coolant salt technologies, including development of structural materials, fuel and coolant clean-up, measurement of physical properties, chemical and analytical R&D.

#### *IV.D. Milestones*

The MSR research plan describes the R&D program to establish the viability of the Molten Salt Reactor by 2018 and to optimize its design features as well as operating parameters by 2025. As such, it is intended to cover the needs of the viability and performance phases of the development plan described in the Technology Roadmap for the Generation IV Systems. The MSR research plan also accounts for a defined approach to establishing system

baseline(s) and accomplishing system integration as needed.

The MSR provisional SSC has re-evaluated the milestones mentioned in the GIF Technology Roadmap owing to the peculiar and more innovative position of MSR among other Generation IV systems. This led to identify a scoping and screening phase (up to 2011), prior to the viability and performance phases, 2012-2017 and 2018-2025 respectively. The main milestones for the demonstration phase (final design, construction and operation of prototypes) have also been discussed, envisioning a MSR prototype after 2035. For the AHTR, the schedule is more compact, with a prototype planned to be in operation by 2031.

#### IV. E Main activities and outcomes

Significant progress has been achieved in 2008 in critical areas of MSR-AHTR R&D. In brief, the essential facts are the following: salt selection for different applications is stabilized, the needs of complementary data have been clarified. [10, 11]

- A strongly improved (versus MSBR) fuel salt clean-up scheme has been developed. [8, 12]
- Criticality tests are being performed for the assessment of MSR and AHTR fuel and core behavior. [13]

The detailed description of these topics is made in a complementary presentation at this symposium. [15]

## V. CONCLUSION

For both systems, SCWR and MSR, extensive R&D work is being carried out, in view of the great promises if a successful development can be achieved. Indeed, both systems face big challenges due to the technical difficulties associated to the reactor system on the one hand, and to the fuel cycle, for what concerns the MSR. The international support exists and System Agreements are signed by three partners (Canada, Japan, EURATOM) for the SCWR (Project Arrangements are in preparation), whereas MSR is at an earlier status, with confirmed interest from France, EURATOM and USA.

## References

1. Schulenberg, T., J. Starflinger, "European research project on the HPLWR", 4<sup>th</sup> Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
2. Naidin, M., I. Pioro, U. Zirn, S. Mokry, G. Naterer, "Supercritical water-cooled NPPs with co-generation of hydrogen: general layout and thermodynamic cycle options", 4<sup>th</sup> Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
3. Ishiwatari, Y., Y. Oka, K. Yamada, "Japanese R&D project on pressure vessel type SCWR", 4<sup>th</sup> Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
4. Bae, Y.Y., H.Y. Kim, J.H. Kwon, S.M. Bae, Lee Kwangho, Y.B. Kim, S.Y. Hong, "Update on the SCWR research in Korea", 4<sup>th</sup> Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
5. Cheng, X., "R&D activities on SCWR in China", 4<sup>th</sup> Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
6. Kiss, A., E. Laurien, A. Aszódi, Yu Zhu, "Improved numerical simulation of a HPLWR fuel assembly with wrapped wire spacers", 4<sup>th</sup> Intern. Symp. on SCWR, Heidelberg, Germany, March 8-11, 2009.
7. Renault, C., *et al.*, "The Molten Salt Reactor (MSR) – R&D Status and Perspectives in Europe", FISA 2009, Prague, June 22-24, 2009.
8. Delpech, S., E. Merle-Lucotte, D. Heuer, M. Allibert, V. Ghetta, C. Le Brun, L. Mathieu, G. Picard, "Reactor physics and reprocessing scheme for innovative molten salt reactor system", *J. of Fluorine Chemistry*, 130, Issue 1, 11-17 (2009).

9. Forsberg, C.W., P.F. Peterson, and R.A. Kochendarfer, “Design Options for the Advanced High-Temperature Reactor”, Proc. 2008 International Congress on Advances in Nuclear Power Plants (ICAPP’08), Anaheim, CA USA, June 8-12, 2008.
10. Benes, O., C. Cabet, S. Delpech, P. Hosnedl, V. Ignatiev, R. Konings, D. Lecarpentier, O. Matal, E. Merle-Lucotte, C. Renault, J. Uhlir, “Review Report on Liquid Salts for Various Applications, Deliverable D50, Assessment of Liquid Salts for Innovative Applications”, ALISIA project, Euratom 7<sup>th</sup> FWP, 2008.
11. Zherebtsov, A., V. Ignatiev, *et al.*, “Experimental Study of Molten Salt Technology for Safe, Low-Waste and Proliferation Resistant Treatment of Radioactive Waste and Plutonium in Accelerator Driven and Critical Systems”, ISTC-1606 Project, Final Report, International Scientific Centre, Moscow, 2008.
12. Delpech, S., *et al.*, “Actinides/lanthanides separation for the Thorium Molten Salt Reactor fuel treatment”, ATALANTE 2008, Montpellier, France (2008).
13. Hron, M., M. Mikisek, “Design Reactor Physical Program in the frame of the MSR-SPHINX Transmuter Concept Development”, Proc. 2008 International Congress on Advances in Nuclear Power Plants (ICAPP ‘08), Anaheim, CA USA, June 8-12, 2008.
14. Renault, C., M. Hron, R. Konings, D-E. Holcomb, “The Molten Salt Reactor in Generation IV: Overview and Perspective”, GIF Symposium, Paris, September 9-10, 2009.